OYSTER CREEK

INDIVIDUAL PLANT EXAMINATION

FOR

EXTERNAL EVENTS

(IPEEE)

Originator:	Kenneth R. Canavan Risk Analysis Engineer	12/1/95 Date
Concurred by:	Louis C. Lanese Manager, Risk Analysis	12/1/95 Date
Approved by:	G. R. Bond Director, Nuclear Analysis & Fuel	12/1/95 Date

9601110208 951229
PDR ADOCK 05000219
P PDR

OYSTER CREEK
INDIVIDUAL PLANT EXAMINATION
for
EXTERNAL EVENTS (IPEEE)

GPU Nuclear Corporation

Louis Lanese - Manager, Special Projects

Kenneth Canavan - Lead Engineer

SEISMIC ANALYSIS	FIRE ANALYSIS	OTHER EXTERNAL EVENTS
GPU Nuclear Corporation	***GPU Nuclear Corporation***	***GPU Nuclear Corporation***
Kenneth Canavan	Kenneth Canavan	Rasool Baradaran
Charles Adams (i)	Timothy Trettel (i)	David Distel (i)
Rene Lopez** (i)	Raymond Daley (i)	Kenneth Canavan (i)
David Miller	Fred Barbieri (i)	Louis Lanese (i)
Swadesh Ramdeen* (i)	Louis Lanese (i)	
Ravi Panicker** (i)	James Manoleas	
Enrique Tang**		
Ken Whitmore* (i)		
Consultants	***Consultants***	***Consultants***
Mitchel Waller (Delta Prime, Inc.)	Mitchel Waller (Delta Prime, Inc.)	
Thomas Kipp (EQE International)		
David Nakaki (EQE International)		
L. W. Tiong (EQE International)		

* - Seismic Capacity Evaluations Only.

** - Seismic Relay Evaluation Only.

(i) - Participant in the independent review.

SECTION 1

EXECUTIVE SUMMARY

1.0 EXECUTIVE SUMMARY

1.1 Objectives

This report is a response to Generic Letter 88-20, Supplement 4 (Reference 1) which requested each licensee to "perform an Individual Plant Examination of External Events to identify vulnerabilities, if any, to severe accidents and report the results together with any Licensee determined improvements and corrective actions to the Commission." In our initial response (Reference 2) to GL 88-20 Supplement 4, we described our plan for completing the Oyster Creek Individual Plant Examination for External Events (IPEEE) which is to use probabilistic risk assessment (PRA) techniques to address the seismic and fire analyses and to apply the NRC staff's recommended screening approach for high winds and other events where possible.

This report section provides a summary of the methods, results and conclusions of the Oyster Creek IPEEE.

1.2 Background

In December 1991, GPU Nuclear Corporation submitted the Oyster Creek Individual Plant Examination (IPE) to the NRC. The Oyster Creek IPE is a Level 1 and a Level 2 PRA as defined by Reference 4. The study was completed by GPU Nuclear with the primary contractor PLG, Incorporated providing the initial approaches as well as guidance and assistance in the use of the PC software package RISKMAN.

The IPEEE (this report) addresses external events including seismic, internal fires, high winds and tornadoes, external floods, transportation and nearby facility accidents. The IPEEE analysis was performed by GPU Nuclear with limited assistance from Mitchell Waller (Delta Prime, Incorporated) on external event analysis methods and approaches and EQE International in the form of site response spectra, structure and component fragility analysis.

1.3 Plant Familiarization

As part of the IPEEE study, GPU Nuclear and its contractors (Delta Prime, Inc. and EQE) performed an extensive review of plant documentation and numerous plant walkdowns to assure correct modeling of Oyster Creek. The plant documentation included the FSAR, layout drawings, and operating and emergency procedures. In addition, the IPEEE team reviewed the USI A-46 safe shutdown equipment list. Walkdowns were performed to verify as-built configuration in support of the fire analysis, the seismic analysis, the seismic/fire interaction and the seismic qualification of equipment issues (USI A-46). The GPU Nuclear inhouse staff was involved in all phases of the IPEEE study.

1.4 Overall Methodology

The general methodology for examining external event risk is Probabilistic Risk Assessment (PRA) as defined by NUREG/CR-2300 (Reference 4). The impact of external events hazards on plant systems was assessed by special methods and later was incorporated with internal events into the plant model. The general approach for external events is as follows:

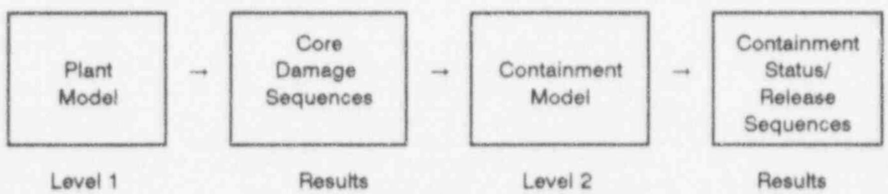
> **Hazard Analysis** - Defining the hazard levels as initiating events (e.g. seismic acceleration or fire), and estimating the frequency of their occurrence.
>
> **Fragility Analysis/Equipment Failures** - predicting the damage to plant equipment and structures that might occur given a hazard level and, assessing the failure frequency of mitigating systems which are designed to prevent the initiating event from proceeding to core damage.
>
> **Plant Response Model/Core Damage Assessment** - Evaluating the impact on the plant from the combination of damaged equipment as a result of the initiator and estimating the frequencies of these damage states.

Screening Analysis - Using the screening process recommended in Section 5.0 of NUREG-1407, any external event shown to have a frequency of $1x10^{-5}$ per year with conditional core damage frequency of less than 0.1 was screened out.

The general steps are applied differently for each hazard, as discussed in the detailed sections of the IPEEE report.

Plant Model

The Oyster Creek IPE for the internal events consists of a Level 1 and Level 2 as shown in the following figure.



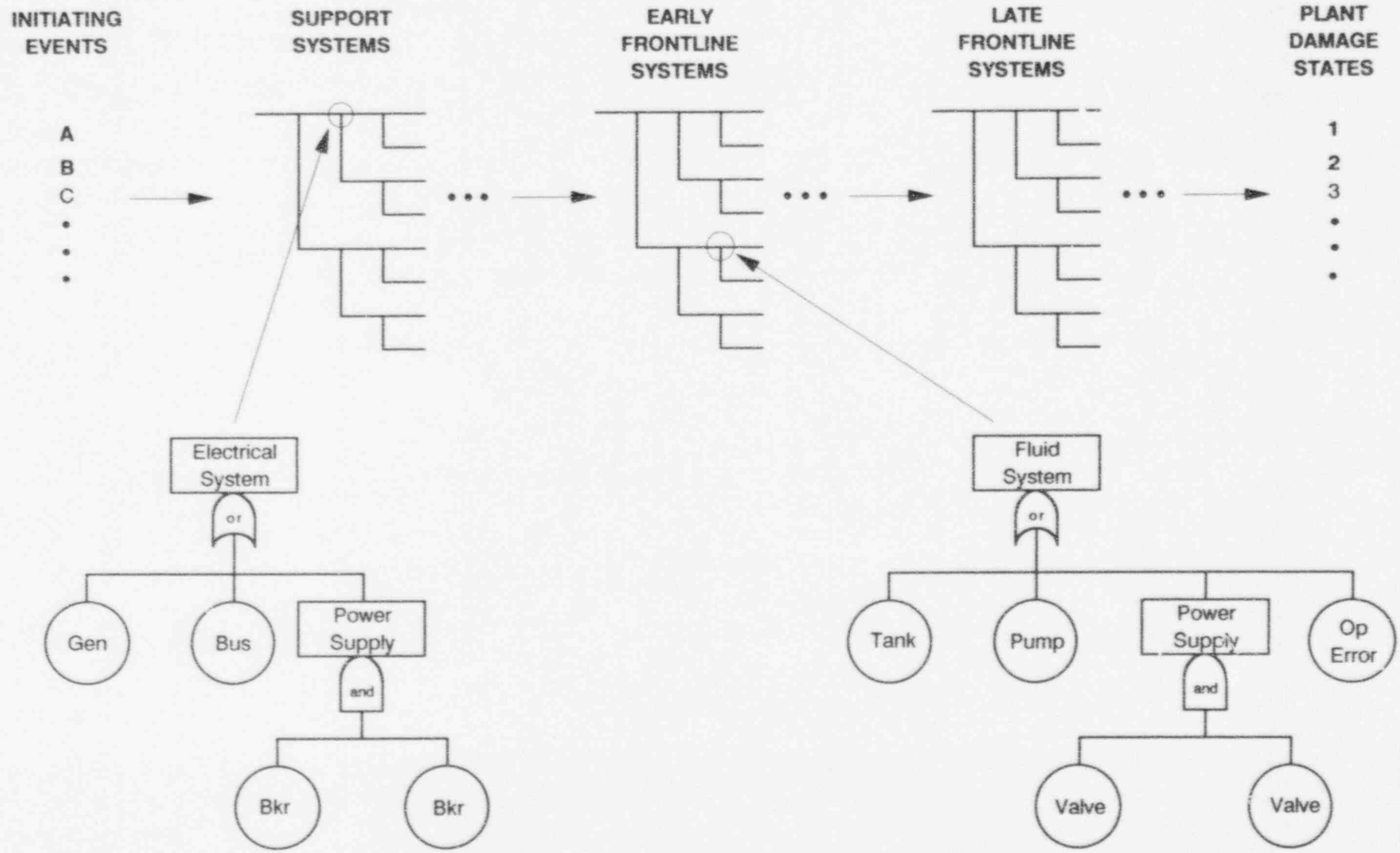
The Level 1 PRA examines a set of initiating events (LOCAs and transients) and the possible plant response to estimate the likelihood of core damage. The Level 2 examines core melt progression phenomena and its effect on the containment structures and estimates the likelihood of releases of fission products to the environment. The Level 1 PRA logic model structure consists of three major pieces which include:

1 - A plant model (event trees) that defines which systems must fail for a given initiating event to lead to core damage

2 - Systems models (fault trees) which define what combinations of component failures must occur to cause failure of each system in the event trees.

3 - A database containing the frequency of initiating events and the failure rates of the system components.

FIGURE 1
PLANT MODEL EVENT TREES AND SYSTEM FAULT TREES



In addition, human action models are used to estimate the likelihood of failure of key operator actions. These estimates are stored in the database and used in the system models (fault tree) in a manner similar to component failure rates.

Each of the above major pieces are described in detail in the Level 1 PRA.

The IPEEE (this report) also assesses the containment performance following external events. The external initiators are treated with the same method as internal events which can be described as "Event tree linking approach". The approach links the major three pieces as shown in Figure 1. The PLG computer code RISKMAN (Reference 8) combined and quantified all major pieces of the logic model. RISKMAN contains five major program modules which analyze data, systems, external events, event tree, and important sequences. Results from each module are incorporated into the overall plant model to create a powerful database useful in examining event scenarios, plant systems and individual components, and the effects of changes in the module on overall risk.

1.5 Summary of Major Findings

The results of this IPEEE show that there are no vulnerabilities to severe accident risk from external events. Internal fires are the most significant contributors to external event core damage followed by seismic events and high winds. In the case of the fire events, the cable spreading room and "A" 480 VAC switchgear room fire (unscreened fire events) produce the risk. Risks due to external floods and nearby facility and transportation accidents are not significant.

The following sections summarize the results of the Oyster Creek IPEEE.

1.5.1 Core Damage Frequency Results.

The calculated mean core damage frequency due to external events is 1.23×10^{-5} per year. The contribution of the major external initiators to the total core damage (CDF) is shown in the following table.

Initiating Events Core Damage Frequencies

External Event Description	Core Damage Frequency (CDF) (per year)	Percent of External Events CDF
**Fires	7.7E-6	62.6%
Seismic Events	3.6E-6	29.3%
High Winds	9.9E-7	8.1%
External Floods	Screened	N/A
Transportation and Nearby Facility Accidents	Screened	N/A
TOTAL EXTERNAL EVENTS	1.23E-5	100%
** CDF due to fires is conservative since human actions to put out small fires were not explicitly credited in the analysis of fires.		

As shown above, fire events contribute 63% to the total risk of all external initiators, followed by seismic (29%) and others (high winds) (8%). Table 1-1 contains a more detailed list of the external initiators risk.

Fire Analysis Results

The fire analysis is dominated by unscreened fires events in the Cable Spreading Room and the "A" 480 VAC Switchgear Room. In both of these fire areas, the dominant fire scenarios are the result of the assumed spread of the fire to engulf the entire area following failure of the suppression system. Equipment failed by the fire as well as independent equipment failure results in core damage. Detailed results are presented in Section 4 of this report.

Seismic Analysis Results

The seismic risk is dominated by seismic events with ground acceleration of 0.54g which is approximately three times the safe shutdown earthquake (SSE). This single acceleration range accounts for approximately 44% of the total seismic risk. A 0.13g seismic event has the highest probability of occurrence, but accounts for only 4% of the seismic CDF because most systems are expected to withstand the shock during these relatively low ground acceleration. Ground accelerations of 0.36g (two times SSE) and 0.72g (four times SSE) contribute approximately equal percentages (26%) to total seismic core damage. This coincidental result is due to a combination of a lower frequency of occurrence of the higher ground acceleration and the increasing seismic failure rates as ground motion increases.

The dominant seismic sequences involve failure of the following components:

- Offsite Power
- Diesel Generator Building (and Diesel Generators)
- Isolation Condenser and Shell Side Makeup Sources
- Nearsite Combustion Turbines

Section 3 of the IPEEE report provides a detailed description of all seismic initiators, important systems and plant damage states.

Other External Events

In the case of other external events, only the high winds analysis produced quantitative results. An upper bound value of 9.9×10^{-7} per year or 8.1% of the total external event core damage frequency was estimated. All other external events (external floods and nearby facility and transportation accidents) were screened. Details on the Other External Events analysis and results are presented in Section 5 of this report.

1.5.2 Containment Performance Results.

External events, impact on containment performance has been examined from the following perspectives:

Containment Structure - The containment was evaluated in the seismic analysis for seismic capacity. The containment structure at Oyster Creek consists of the Reactor Building which is comprised of four main sub-structures: the main structure, the drywell, the biological shield wall and the reactor pressure vessel. All reactor building structure fragilities exceeded 1.0g capacity and were screened from further consideration.

Containment Isolation - The seismic impact on containment isolation was evaluated by examining piping/supports and isolation signals. Piping, isolation valves and support for containment penetrations have median fragilities greater than 1.0g. Containment penetrations have isolation valves which fail closed on loss of instrument air (a non-seismic system) or loss of power. In addition, the A-46 program has addressed isolation of the major release containment pathways. Manual isolation of the containment prior to core damage is modeled. Oyster Creek seismic results indicate a low seismic risk of containment isolation failure.

Containment response - External events were judged to have no impact on the containment phenomenological response model. An external event initiated core damage event is modeled the same as an internal event with regard to containment response. External events can impact active systems that affect containment response. That impact is handled the same as any other sequence in the model.

In conclusion, none of the external events analyzed significantly impact the performance of the containment.

1.5.3 Conclusion

This report concludes that no vulnerabilities with regard to severe accident risk exist at Oyster Creek.

The term vulnerability is defined as any core damage sequence that exceeds $1x10^{-4}$ per reactor year, or any containment bypass sequence or large early containment failure sequence that exceeds $1x10^{-6}$ per reactor year.

Section 7 of the IPEEE report describes the potential plant improvements which are identified as a result of IPEEE. These improvements are being evaluated and pursued as part of the ongoing risk management program.

Table 1-1
External Events Initiators -
Contribution to Total External Event Core Damage Frequency

External Events	Percent of Total Core Damage Frequency
Fire Events	**62.6%**
"A" 480 VAC Switchgear Room Fire	41.5%
Cable Spreading Room Fire	21.1%
Seismic Events Total	**29.3%**
0.13 g ground acceleration	1.0%
0.36 g ground acceleration	7.8%
0.54 g ground acceleration	12.9%
0.72 g ground acceleration	7.6%
Other External Events	**8.1%**
High Winds	8.1%
External Floods	Screened
Nearby Facility and Transportation Accidents	Screened

1.6 REFERENCES

1. USNRC Generic Letter 88-20, Supplement No. 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)," June 28, 1992.

2. GPU Nuclear Letter No. C321-91-2323, "Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination for External Events (IPEEE)," December 17, 1991.

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

4. American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedure Guide: A Guide to the Performance of Probabilistic Risk Assessment for Nuclear Power Plants vs Nuclear Regulatory Commission," NUREG/CR-2300, April 1993.

5. GPU Nuclear, Oyster Creek Individual Plant Examination Submittal, June 1992.

6. GPU Nuclear, Oyster Creek Probabilistic Risk Assessment (Level 1), Volumes 1 through 6, November 1991.

7. GPU Nuclear, Oyster Creek Probabilistic Risk Assessment (Level 2), Volume 1, June 1992.

8. Riskman Release 4-5.2, "PRA Workstation Software," by PLG, Incorporated.

SECTION 2

EXAMINATION DESCRIPTION

2.1 Introduction

The Oyster Creek Individual Plant Examination for the External Events (IPEEE) was performed by using the NUREG-1407 format. The IPEEE includes a Level I Seismic Probabilistic Risk Assessment (with an evaluation of containment performance), a Fire Analysis (with evaluation of containment performance) and treatment of other external events. Each of these analyses are described in additional detail in their respective report sections as well as summarized below.

Seismic Analysis

The examination of seismic events at Oyster Creek used the state-of-the-art probabilistic risk analysis (PRA) methods. A seismic PRA was performed using the Oyster Creek Level I and Level II PRAs as a framework for the combination of independent failures from the Level I PRA and seismic system and component failures in the form of fragilities. The methods and results of this evaluation are presented in detail in Section 3.0 of this report.

Fire Analysis

The method used to examine the fire risk is a modified EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. This method also used the Oyster Creek Level I and II PRAs as a basis to examine independent failures as well as those caused by fire events. The methods and results of this evaluation are presented in Section 4 of this report.

Other External Events

The method used to address high winds, floods and transportation and nearby facility accident is consistent with the screening process in NUREG-1407. The methods, events analyzed and the results are presented in Section 5 of this report.

In addition, plant walkdowns have been performed by GPUN staff and its contractors in support of the various analyses to assure correct modeling of the as-built plant.

The above methods were used in accordance with the recommendations of Generic Letter 88-20 Supplement 4 and NUREG-1407. This report summarizes the IPEEE process and describes the results and conclusions.

2.2 Conformance with Generic Letter and Supporting Material

Generic Letter 88-20, Supplement 4 identified five purposes for the utilities required to perform the IPEEE. GPUN has satisfied these purposes as follows:

1. **Develop an Appreciation of Severe Accident Behavior -**
   GPUN personnel have been heavily involved in all phases of the original PRA. Most of the IPEEE was conducted by GPUN in-house risk analysts. Delta Prime Inc, as a contractor provided input to both the seismic and fire analysis. The acknowledgement section of this report lists the participants involved in the evaluation of each external event, and indicates significant involvement of GPUN personnel.

2. **Understand the Most Likely Severe Accident Sequences that Could Occur Under Full Power Operating Conditions -**
   The Oyster Creek IPEEE is a detailed study which was conducted with a purpose to identify and understand the most likely severe accident sequences which are initiated by external events. The IPEEE was modeled and quantified using current plant data and the state-of-the art risk analysis tools and methods. The manner in which the study was conducted has provided insights into the systems/ components/operator behavior during external events.

3. **Gain a Qualitative Understanding of the Overall Likelihood of Core Damage and Radioactive Material Release -**
   The use of the RISKMAN Computer Code for Level I and II PRAs allowed for a clear understanding of the risk of core damage and release category end state. In addition to a ranked list of sequences, RISKMAN ranked the core damage risk for each initiating event as well as the risk importance of each split fraction down to the component and operator failure levels. This permitted a qualitative and quantitative understanding of the risk contributors in terms of initiating events, accident sequences, components and operator actions importance measures.

4. **Reduce, if necessary, the Overall Probabilities of Core Damage and Releases.**

   The IPEEE did not identify any significant vulnerabilities. The risk enhancements described in Section 7 of this report can potentially be effective at reducing the risk.

5. **Assure the technical adequacy and the reasonableness of the results**

   The IPEEE was reviewed by an independent in-house review group consisting of managers and senior engineers. The reviews were conducted to assure the study was performed accurately using proper documentation and data.

2.3 General Methodology

The general methodology for examining the external event risk is Probabilistic Risk Assessment (PRA) as defined by NUREG/CR-2300. In the case of the seismic and fire analyses the impact of the external events hazards on plant systems was assessed by special methods (seismic fragilities and fire event impacts) and these impacts were combined with the independent component and system failures modeled in the internal events Probabilistic Risk Assessment (PRA). The general approach for the external events is as follows:

Hazard Analysis - Defining the hazard levels as initiating events (seismic acceleration or fire event) and estimating the frequency of their occurrence.

Fragility Analysis/Equipment Failures - predicting the damage to plant equipment and structures that may occur given the hazard and, assessing the independent failure frequency of mitigating systems which are designed to prevent the initiating event from proceeding to core damage.

Plant Response Model/Core Damage Assessment - Evaluating the impact on the plant from the combination of damaged equipment as a result of the initiator and the independent failures and estimating the frequencies of these damage states.

Screening Analysis - Using the screening process recommended in Section 5.0 of NUREG-1407, any external event shown to have a frequency of 1E-5/yr with conditional core damage frequency of less than 0.1 was screened out.

Each external event was evaluated by a specific method consistent with the recommendation of NUREG-1407 as follows:

Seismic Events - Assessment of the seismic events used the existing Level I PRA as a starting point for the seismic IPEEE. It incorporated hazard sensitivity (i.e. using EPRI and LNLL acceleration curves), updated plant walkdown and effect of relay chatter. Section 3 of this report describes the seismic analysis methods.

Internal Fires - Assessment of the internal Fires Risk used EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology for an area screening and fire damage evaluation. Areas with fire initiating event or core damage sequence frequency of less than 10^{-6} were screened out. All others were assessed using the PRA method. Section 4 of this report describes the Internal Fires method in more detail. Section 4.6.2 explains how the impact of internal fires are integrated into the Level 1 plant model.

Other External Hazards. The assessment of other external hazards (e.g., external flooding, high winds, nearby facility or transportation accidents) is performed using screening approaches which are consistent with those presented in NUREG-1407.

In the case of the seismic and fire analyses, the latest version of the RISKMAN computer code was used. The RISKMAN computer code has the capability of quantifying the impacts of the independent failures events identified in the Level 1 PRA model as well as those due to the external hazard. The risk or plant model uses the Rule Based Methodology where blocks of logic can be input to RISKMAN in the form of descriptive logic rules. RISKMAN uses the rules, links and evaluates the frequencies of all scenarios from the initiating events to the end states. Section 7.1.1, "Rules Methodology" of the Oyster Creek Probabilistic Risk Assessment (Level 1) describes the Rule-Based Methodology in more detail.

2.4 Information Assembly

Oyster Creek IPEEE includes, directly or by reference, the current plant data used in the examination which include:

The Oyster Creek Probabilistic Risk Assessment (Level I). This study provides the first cut event sequence development, human action analysis, and system analyses fault trees as well as the component failure and maintenance data base.

Final Updated Safety Analysis Report (FSAR). The FSAR was used in the development of the plant model with emphasis on the plant response to design basis accidents. Also, the FSAR was used to determine the original list of systems to be modeled.

Operation Plant Manual (OPM). The OPM provides details on system design, operation and control and was used to help verify system dependencies and certain details of system design and operation.

Emergency Operating Procedures (EOPs). Emergency Operating Procedures were used in the development of the plant model and operator action analysis.

System Surveillance, Abnormal and Operating Procedures were used in the development of the individual system analyses as well as for the collection of system demands in the data analysis task. Abnormal and operating procedures were used in the human action analysis task.

Piping and Instrument Diagrams (P&IDs) and Electrical Diagrams were used in the system analyses as well as in determination of system dependencies.

Transient Assessment Reports (TARs) (actual data on plant response to transients) were used in the development of the plant model as well as the data analysis task (trip data).

Oyster Creek Fire Hazard Analysis Report was used in development of the fire analysis.

Many other information sources were reviewed a list of references is available following each major report section of the Oyster Creek IPEEE. As well as the above information, plant walkdowns were performed to assure the IPEEE represents the as-built as operated configuration of Oyster Creek. The IPEEE team contains mostly GPUN engineers which include seismic, fire, and other system engineers. The activities of this team were performed and/or directed by the GPU Risk Analysis Section.

SECTION 3

SEISMIC ANALYSIS

OYSTER CREEK SEISMIC ANALYSIS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF FIGURES

3. SEISMIC ANALYSIS

3.0 Methodology Selection (PRA)

In the case of Oyster Creek seismic IPEEE the response is accomplished by performing a seismic Probabilistic Risk Assessment or Seismic PRA. The overall approach in performing a seismic PRA consists of a seven step process:

Step 1: **Determination of site specific seismicity characteristics.** This step involves the development of the frequencies of occurrence and magnitude of seismic events for the site. Site structure analysis is also performed. The resulting frequencies and magnitudes of seismic events are the initiating events for the SPRA. Site structure responses are input into step 5 where the capacity of components which impact risk are calculated. This step is presented in Section 3.1.1.

Step 2: **Identification of those components important to plant safety**, including equipment, structures and procedures. The level 1 PRA developed for Oyster Creek is utilized to determine those components which impact risk. Other studies such as the A-46 Safe Shutdown Equipment List (SSEL) as well as a qualitative review of systems, structures and components which may impact the probability of core damage due to seismic initiators are used to ensure that the list of components which impact risk is comprehensive. This step is discussed in detail in report Section 3.1.2.

Step 3: **An initial plant walkdown** of the identified systems and components, is performed, and any plant seismic system interactions and unique plant features which may impact risk are identified. This step is discussed in Section 3.1.2.

Step 4: **Develop plant logic models.** The plant logic models are developed using the level 1 Oyster Creek 1 PRA with the addition of the failure rates of components due to seismically initiated events. The level 1 model is modified to ensure that the independent as well as seismic failures are accounted for in the logic model. This step is detailed in report Section 3.1.6.

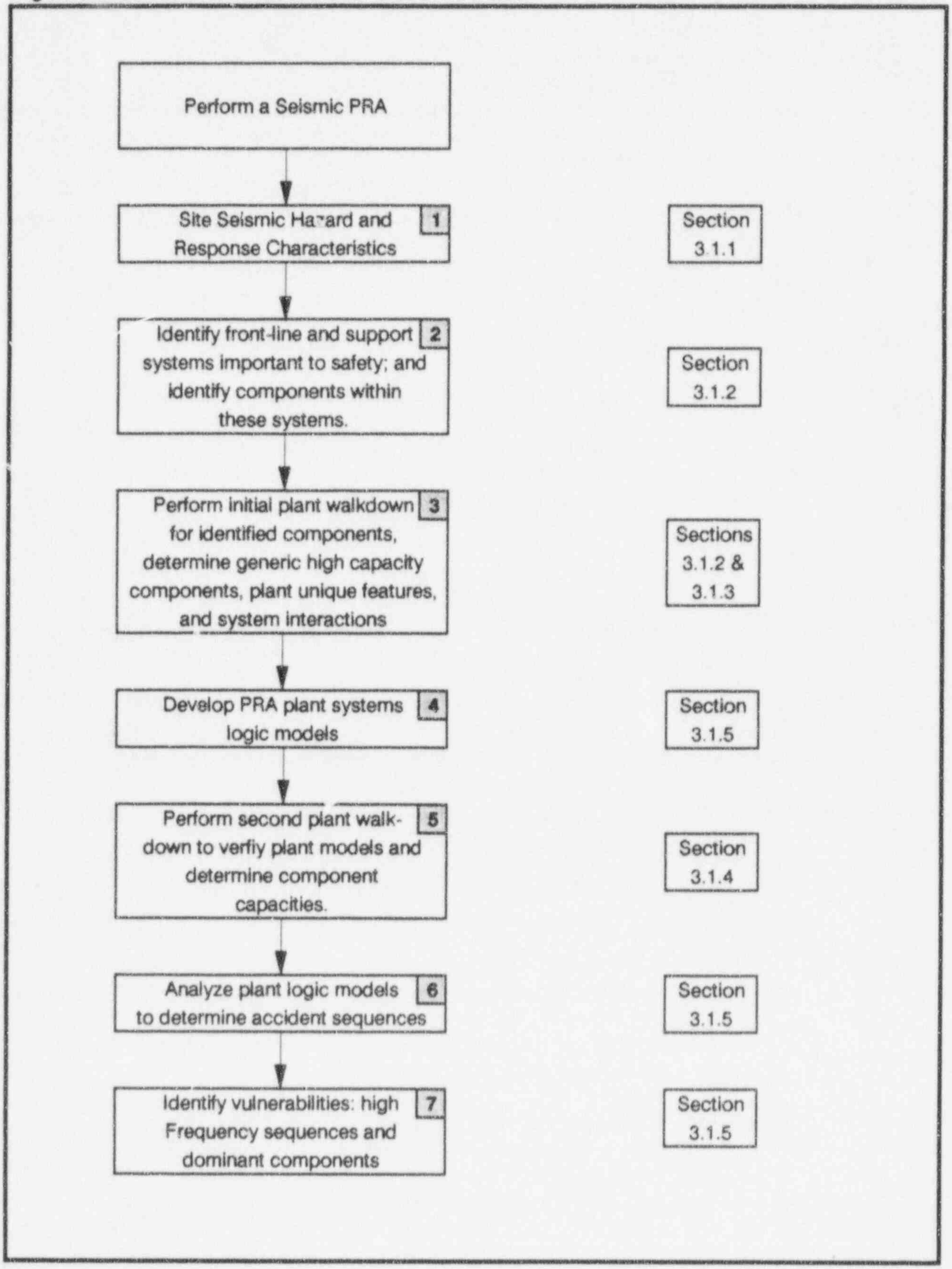
Step 5: **A second plant walkdown** is performed to verify plant models and to collect data to determine component capacities. This step is documented in report Section 3.1.5.

Step 6: **Analyze the plant models** to determine seismic initiated accident sequences and their frequency. This step involves the assembly and quantification of the plant logic models as well as the reporting and analyzing of the results. This step is detailed in report Section 3.1.6.

Step 7: **Identify plant seismic vulnerabilities.** This step defines any site specific vulnerabilities which are discovered as a result of the performance of the study.

This process is illustrated in Figure 3-1 and discussed in the subsequent report sections.

Figure 3-1 Seismic PRA Procedure for a Seismic IPEEE



3.1 Seismic PRA

The seven steps used in the development of the Oyster Creek Seismic PRA, discussed in the introduction and illustrated in Figure 3-1 are discussed in the subsequent report sections.

3.1.1 Hazard and Fragility Analysis

Step one in the seismic PRA procedure for a seismic IPEEE is the development of site specific seismic hazard and response characteristics. The following sections describe the methods used to develop the seismic hazard curve values and component fragilities for the Oyster Creek site and plant, respectively.

3.1.1.1 Hazard Analysis

The EPRI site seismic hazard study defines the initiating event frequencies used in this study for peak ground acceleration intervals corresponding to discrete ground accelerations of up to 800 cm/sec^2 (0.82g force). For point estimate computations, mean frequencies were determined from the NP-6395-D seismic hazard estimates and the representative interval frequency was calculated. These hazard curve point estimate values were then used as initiating event input to the seismic PRA model.

Table 3-1 shows the mean values from the NP-6395-D curves which were used as the Oyster Creek initiating event frequencies for the PRA model. The representative ground acceleration for each seismic initiating event is based (approximately) on the safe shutdown earthquake (SSE) ground acceleration of 0.18g force. A summary of the results from that study are presented in Section 3.1.5.3.

Table 3-1
Seismic Initiating Events

Initiating Event	Ground Acceleration Range	Initiating Event Frequency
SEIS1 (0.13g) (~1 x SSE)	0.007g - 0.26g	7.44E-03
SEIS2 (0.36g) (~2 x SSE)	0.26g - 0.46g	2.73E-05
SEIS3 (0.54g) (~3 x SSE)	0.46g - 0.62g	2.72E-06
SEIS4 (0.72G) (~4 x SSE)	0.62g - 0.82g	9.83E-07

The frequencies in Table 3-1 were derived from "Probabilistic Seismic Hazard Evaluations at Nuclear Power Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," EPRI Report NP-6359-D, April 1989).

As a sensitivity case, the revised Lawrence Livermore National Laboratory (LLNL) seismic hazard estimates, as listed in NUREG-1488, are used as input to the PRA model as initiating event frequencies.

The seismic IPEEE results including those using the LLNL seismic hazard curves are discussed in report Section 3.1.5.

3.1.1.2 Fragility Analysis

A seismic fragility, or failure vulnerability, analysis was conducted by EQE International. The plant components that were chosen for seismic fragility analysis were selected as a result of the listing of equipment required to attain a successful endstate in the Oyster Creek PRA and the USI A-46 Safe Shutdown Equipment List (SSEL). The list of components chosen was supported by a seismic walkdown performed to gather equipment specific information.

The fragility results for the equipment and structures found to be important to the core damage frequency and containment isolation and core cooling at Oyster Creek are included in the tables of this report.

3.1.1.3 Definition of Failure

For purposes of this study, seismic Category I structures are considered to have failed when inelastic deformations of the structure under seismic load could potentially interfere with the operability of safety related equipment attached to the structure. These limits on inelastic energy absorption capacity (i.e. ductility limits) are estimated to correspond to the onset of significant structural damage, and do not necessarily correspond to structural collapse. For many potential modes of failure, this is believed to represent a conservative bound on the level of inelastic structural deformation which might interfere with the operability of components housed within the structure. It is important to note that considerably greater margins of safety against structural collapse are believed to exist for the Oyster Creek structures. Structures which are susceptible to sliding are considered to have failed when sufficient sliding deformation has occurred to fail piping or electrical duct banks or to cause sufficient damage resulting from structure to structure impact to interfere with equipment operation.

Piping, electrical, mechanical and electro-mechanical components that are required to mitigate the effects of a plant trip due to earthquake are considered to fail due to seismic considerations when they can no longer perform their designated functions. Also, ruptures of pressure boundaries are considered as failures. In most cases, however, the equipment will lose its ability to function at lower ground accelerations before pressure boundaries fail because the pressure boundaries for equipment such as pumps and valves are usually very conservatively designed. It should be noted that seismic failure has been added to the independent hardware failures that have already been considered in the Level 1 plant model.

3.1.1.4 Fragility Curve Formulation

Seismically induced failure data are generally unavailable for specific nuclear plant components or structures. Thus, fragility curves which plot the peak ground acceleration at which the component is expected to fail must be developed primarily from analysis and engineering judgement, supported by limited test data.

Earthquakes that result in the same peak ground acceleration at the plant site can have different energy contents and durations. These factors vary randomly and affect the fragility of structures and components. While the median acceleration capacity can be determined for structures and components, it is still necessary to assign a random variability to the capacity due to the differences in earthquake characteristics. In addition, the strengths and response characteristics of the structures and components are not known exactly, so the uncertainty associated with these values also needs to be expressed.

The fragility descriptions are based on a logarithmic distribution because the data tend to fit a logarithmic distribution and due to the central limit theorem, which states that the products of data will tend to describe a logarithmic distribution, even if the data itself is not logarithmic. However, the fragility data do not fit very well in the tails of the distribution, below failure fractions of about five percent. At these levels, the fragility curves are considered to be conservative. For example, conventional components, such as pipes and conduits routinely withstand static vertical loads of 0.1g force without failing. Also, small dynamic loads resulting from cranes, forklifts and other component handling equipment regularly occur in the plant without causing structural damage. For low acceleration levels, below 0.06g, it is therefore considered unrealistic to assume that well engineered structures will have even a small chance of failure.

It is therefore assumed that below some acceleration threshold, there is virtually no chance of failure due to seismic considerations. Material strength and damping, for example, do not have infinitely low and high values, but instead have some lower and upper thresholds. Further, extensive studies have been conducted to develop response spectra from available earthquake records and, while dispersion exists about the median values, spectra with essentially zero or infinite response do not occur. For these, as well as other variables contributing to the seismic fragility of a given structure or component, it is apparent that some lower and upper cutoffs on the tails of the dispersion exist. Since the overall fragility curves are based on a combination of these variables, it is expected that a lower threshold exists below which no failures will occur. This is supported by past earthquake experience. Documentation from the Diablo Canyon Long Term Seismic Program provides guidance in determination of the lower level cutoff for component fragilities. In the following sections, the specifics of the fragility analysis, as they apply to the Oyster Creek plant, are described.

3.1.1.5 Structures and Equipment Fragilities

The key fragility parameters resulting from the fragility analysis of structures conducted by EQE Engineering Consultants are tabulated in Table 3-2 below. Walkdowns of structures at the Oyster Creek facility were performed separately from those performed in support of the component facility analysis. Walkdown team members are provided in report Section 3.1.2.1.

Table 3-2 Seismic Fragility of Oyster Creek Civil Structures				
Structure (Failure Mode)	**Median Acceleration Capacity ($\hat{A}_g$)**	**Randomness Variability (β_R)**	**Uncertainty Variability (β_U)**	**HCLPF**
Reactor Building (Wall)	2.96g	0.29	0.25	1.21g
Reactor Building (Column Anchorage)	1.00g	0.38	0.27	0.34g
Turbine Building (Diaphragm Shear)	0.88g	0.20	0.26	0.41g
Turbine Building (Column Anchor Bolt)	0.74g	0.35	0.24	0.28
Turbine Pedestal	2.17g	0.24	0.28	0.92g
Intake Structure	0.82g	0.18	0.26	0.40g
Emergency DG Building	1.18g	0.37	0.22	0.45g
EDG Building (with liquefaction)	0.69g	0.14	0.28	0.35g
Fire Pond Pump House	1.21g	0.25	0.44	0.39g
Combustion Turbine Fuel Tank	0.66g	0.37	0.39	0.19g

Notes: HCLPF denotes the High Confidence of Low Probability of Failure, which corresponds to the lower 5% tail of the 5% uncertainty variability curve.

The values shown above were taken from Seismic Fragilities of Civil Structures at Oyster Creek Nuclear Generating Station (EQE International, October 1994).

Fragility results for mechanical and electrical equipment whose failure can result in the initiation of a sequence or in the degradation of the plant response to such accidents are tabulated in Table 3-5.

The seismic hazard curve results indicate that the upper bound ground acceleration at the Oyster Creek site is 1.0g. Further, the annual exceedance frequency for higher ground accelerations is extremely small. As a step in reducing the number of structures and equipment that have to be considered in the plant seismic model, those that have median ground acceleration capacities greater than 1.0g are generally excluded. The remaining components are shown in Table 3-6, which is a summary of plant components having lower median ground acceleration capacities.

Table 3-6 lists the components that were evaluated for seismic fragility, the component's median acceleration capacity and includes the random and uncertainty variables, B_r and B_u, respectively.

The seismic fragility parameters shown in Table 3-6 are then used to generate plant specific seismic failure probabilities at each of the ground accelerations associated with the four seismic initiating events (approximately the safe shutdown earthquake, 2 x SSE, 3 x SSE and 4 x SSE). These failure probabilities are then listed in the seismic master frequency file, which is shown in Table 3-8.

3.1.1.6 Earthquake Induced Soil Liquefaction

The potential for earthquake induced soil liquefaction and ground failures was evaluated for the Oyster Creek site by Geomatrix Consultants, as documented in their report, "Assessments of Potential for Liquefaction and Permanent Ground Displacements at Designated Facilities, Oyster Creek Nuclear Generating Station (December, 1994)". The likelihood of soil liquefaction for varying water table conditions was evaluated and expressed in terms of probabilities of occurrence conditional on the occurrence of a given ground acceleration. Should soil liquefaction occur, foundation bearing strength failures are not expected, but ground settlements on the order of 0.5 to 1.0 inch could be expected.

The Reactor Building is founded on the Lower Cohansey formation, which consists of compacted sands with a geological age of more than 10 million years. Due to the compaction of these sands (equivalent strength of much higher than 30 blows/foot), significant liquefaction below the Reactor Building is unlikely to occur for any level of earthquake induced ground shaking. The assessment also notes that the "geological age of the Cohansey sands (10,000,000+ years old) would also indicate a low susceptibility to liquefaction, as such relatively old deposits have not been known to liquefy during historic earthquakes."

For the Turbine Building, the assessment found that liquefaction of the fill beneath portions of the building would require a peak ground acceleration in excess of 0.70g.

The Emergency Diesel Generator building is founded on fill that was placed to construct the slope between the building and the discharge canal. For this construction, soil liquefaction can be expected to occur at a peak ground acceleration of 0.40g. Therefore, soil liquefaction of below the Emergency Diesel Generator Building was separately considered. The effect of this phenomena is to increase the likelihood of failure of the diesel generator building itself, as noted in Table 3-2, in addition to increasing the failure likelihood of the diesel generators themselves, as noted in Table 3-6. Incorporation of this failure is described in Section 3.1.5, top event DU.

Finally, the fire protection piping may be vulnerable to permanent ground movements where it crosses the discharge canal. Soil liquefaction can be expected to occur at a ground acceleration of 0.40g, however the discharge canal banks will serve to support the piping and significant displacement is not expected below 1.0g. Due to the extent of piping that is susceptible to this disruption, fire protection piping failure is conservatively assumed to occur following liquefaction at the 1.0g peak ground acceleration. The incorporation of this failure mode is described in Section 3.1.5 under top event FX.

3.1.2 Review of Plant Information and Walkdown

In the development of the Oyster Creek seismic PRA many sources of information were reviewed. These include the Level 1 OCPRA, the A-46 Safe Shutdown Equipment List (SSEL) and design basis information. As well as reviews of the available plant information site walkdowns were performed and are discussed in the sub-sections below.

3.1.2.1 Site Walkdowns

Site walkdowns were performed during various stages of the project. The first walkdowns performed were for site familiarization purposes. Other walkdowns included Structural Fragility, A-46 and seismic equipment capacity (components modeled in the seismic PRA) walkdowns. The Structural and Seismic Capacity walkdowns are documented in report Section 3.1.3.

3.1.2.2 Identification of Components for Seismic Capacity Evaluation

The purpose of this step is to identify the components required to prevent or mitigate a severe core damage accident and achieve and maintain safe shutdown. The Oyster Creek Level 1 PRA (November 1991), the A-46 Component List were utilized to provide the initial list of components which may potentially be important in the mitigation of seismic events. The A-46 Component List was reviewed to ensure that the list was comprehensive and included those components not explicitly modeled in the Oyster Creek PRA (that is, subsumed into initiating event calculation, etc.) are included in the seismic capacity evaluation

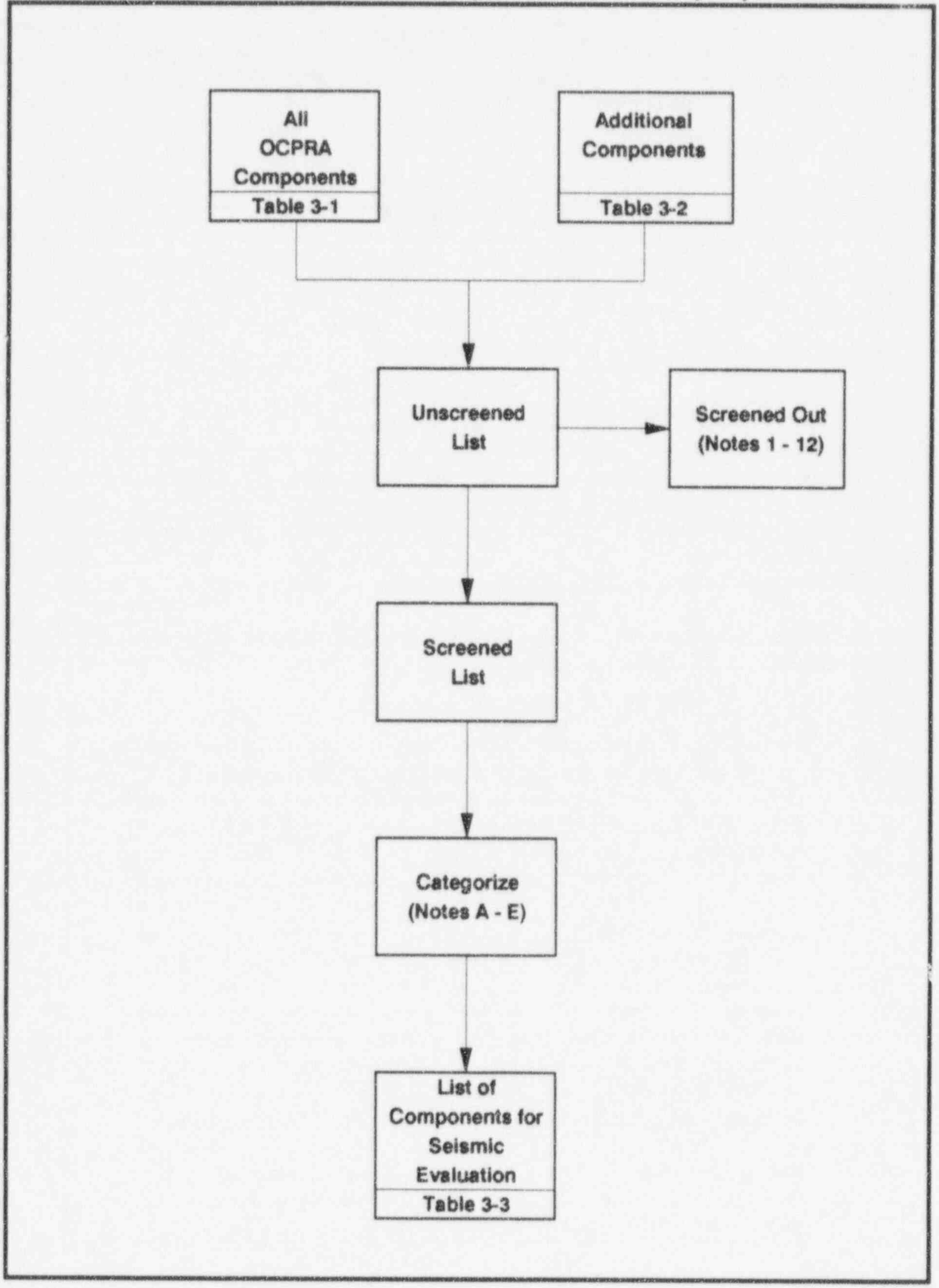
The process of determining those components for which seismic capability evaluation will be performed is a four step process as follows:

- **Identify Risk Significant Components.** This step involves the identification of all components modeled in the current Oyster Creek Level 1 PRA and review of the A-46 component list (Safe Shutdown Equipment List) to determine any components which are risk significant. This list is presented as Table 3-3 (Oyster Creek level 1 PRA components) and Table 3-4 (Additional Components).

- **Screen Out Generic High Capacity Components.** In this step both lists developed above are screened based on generic seismic capacities. Those components which are considered rugged are screened out. Notes 1 through 13, described below, provide the basis for this screening. Walkdowns of all components, including those which have been screened out, are performed to verify seismic ruggedness. The remaining components appear on Table 3-5 titled "List of Components for Seismic Capability Evaluation".

- **Categorize the List of Components for Seismic Capacity Evaluation** to facilitate the determination of component capacities and verify completeness.

Figure 3-2 Process for the Identification of Components for Seismic Capacity Evaluation



- **Perform Initial Walkdown** to verify screening criteria and generic high capacity components, plant unique features, as well as potential system interactions.

The list of components for seismic capacity evaluation is utilized as input in the determination of equipment fragilities and further screening analysis. Seismic capacity evaluation is presented in Section 3.1.4. The process diagram for the development of the list of components for seismic capacity evaluation is presented in Figure 3-2 and described in detail below.

First, the components modeled in the Oyster Creek Level 1 PRA are collected and presented in Table 3-3. In addition to the Level 1 PRA components, the A-46 Safe Shutdown Equipment List (SSEL) and the OCPRA initiating events, Systems and Special Analyses were reviewed to ensure completeness, and additional components are presented in Table 3-4. Following the identification of all components modeled in the PRA and A-46, the process of screening out of components based on generic seismic ruggedness is begun. The components which do not screen out based on generic seismic capacity are presented in Table 3-5 titled "Components for Seismic Capacity Evaluation" and analyzed in detail in Section 3.1.5.

On the attached tables the column titled seismic note is used to refer to the descriptions presented below. Notes 1 through 12 describe the different screening criteria on which components are excluded from seismic capacity evaluation. These notes address the component type and the failure mode modeled in the original studies. The column "Category" is included on Tables 3-3 and 3-4 to further subdivide the list into equipment type.

Note 1 - Inadvertent Valve Transfer is used to screen those valves which are model as inadvertent valve transfers. In the case of manual valves this includes all basic events which model transfer open and transfer closed failure modes. For powered valves (air operated, motor operated or solenoid valves), only those valves which fail in a functional position on loss of supporting systems are screened out.

For example, air operated valves (AOVs) for which the initial, actuated and loss of support positions are the identical are screened. In this case, the valve is not required to be active for success and the loss of support would also not require the valve to change position (that is, the valve is in its fail safe position).

In the case of motor operated valves (MOVs) the transfer failures which are screened are those in which the actuated and initial states are the same. This indicates that the valve is not required to be active for success.

Lastly, in the case of solenoid valves (SOVs) the transfer failures which are screened using note 1 are those in which the actuated, loss of support and initial states are identical. This indicates that the valve is not required to be active for success and with a loss of support will not change state (that is, the valve is already in a fail safe position).

Note 2 - Check Valve Failures. Check valves are considered to be seismically rugged (due to the nature of check valve operation) and are screened from further consideration.

Note 3 - Failure of passive components resulting in loss of system integrity. This category only includes the fire protection piping itself. It should be noted that failure due to soil liquefaction is separately considered (see Section 3.1.1.6).

Note 4 - Failure of passive components which result in system blockage. Failure which are the result of heat exchanger, filter blockage, demineralizer and stainer blockage are screened out.

Note 5 - Failure of inactive electronic components. This class of components include push button switches and temperature elements. With no active parts these components are considered seismically rugged and screened out.

Note 6 - Other Sensors. This class of components include flow transmitters, level and pressure switches, and level and pressure transmitters. These components are considered seismically rugged and inadvertent actuation is the most probable seismic failure mode. Inadvertent actuation results in initiation of safety systems and therefore; system success. These components are screened out.

Note 7 - Fail safe components. This category of components includes only bistables, various DC power circuit breakers and turbine stop and control valves required to trip either the reactor or turbine. A seismic event is assumed to result in a turbine trip and therefore stop and control valves are assumed successful. In the case of RPS bistable and DC circuit breakers the most probable failure mode of these fail safe electronic components is to the actuated position. The actuated position of RPS results in a reactor trip. These components are screened.

Note 8 - Relays and control circuitry, including nuclear instrumentation. Control circuits (solid state) are assumed to be seismically rugged. Relays are screened from further consideration, pending the outcome of A-46 relay evaluation.

Note 9 - Fuses. Fuses are screened based on a high degree of seismic ruggedness. It should also be noted that, in general, fuses are located within the panels and buses for which seismic capacities are calculated and are therefore bounded by the seismic capacity of the panel or bus. This note is not used.

Note 10 - Circuit breakers. Circuit breakers are contained within switchgear for which seismic capacities are calculated and are bounded by the seismic capacity of the switchgear which are analyzed. Therefore circuit breakers as individual components are screened out.

Note 11 - Duplicate entries. These items appear more than once in the component database since they are utilized in more than one top event or split fraction. The development of separate fragilities is not necessary.

Note 12 - Piece/Part Components. Items identified in the component list with this seismic note are those which are pieces or parts of a larger component for which a seismic capacity is determined. Since the seismic capacity evaluation is based on the weakest operational piece/part or anchorage of a component these items are evaluated with the larger component and screened out in this list. To ensure completeness the "category" column indicates the larger component designator in which the item is considered for seismic evaluation.

Note 13 - Operator Action. The impact of seismic failure is subsumed within the hardware failure for these items.

Initial walkdowns of the site are performed to verify the screening criteria and (generic high capacity components), plant unique features as well as potential system interactions. The remaining items (components which have not been screened out) are divided into five groups (seismic notes A through E) to assist in the seismic capacity determination and to provide a reviewable format to ensure completeness. The final list of components for seismic capacity evaluations are presented on Table 3-5. The column titled seismic note refers to the following:

Note A - Active components. The equipment in this category consists of major rotating components including compressors, fans and pumps.

Note B - Components which potentially threaten system integrity. These components include the inadvertent relief valve operation and tank rupture.

Note C - Active components that threaten system flow. These components include flow control valve failures and active (required to change position for system success) motor, air and solenoid operated valves.

Note D - Components evaluated with greater than 1.0g. This category includes those components which are judged to be generically rugged. A verification of the seismic ruggedness of these components was performed in this analysis to ensure screening out of these components is justified.

Note E - Other components evaluated for seismic fragility. This category is used in Table 3-5 and applies to components that were included from Table 3-4 that are not associated with individual top events.

Table 3-4, titled "Additional Components" is the result of reviews of the A-46 Safe Shutdown Equipment List (SSEL) and OCPRA initiating event and System and Special Analyses. This list is included to ensure completeness of the final list of components for seismic evaluation by capturing components which may have been included in initiating events calculations, etc. Components which appear on this list are screened from further consideration by the notes above (seismic notes 1 through 13). The components which are not screened out with notes 1 through 13 are assigned seismic notes A through E as described above. These items are added to Table 3-5 (Components for Seismic Capacity Evaluation).

The components for seismic capacity evaluation (Table 3-5) is the result of all unscreened components from Tables 3-3 and 3-4 and is used as input to the development of seismic capacities (fragilities) which are presented in Section 3.1.5. As expected the list of components for seismic capacity evaluation is comprised mainly of major rotating and active equipment. Issues of seismically induced fires (see Fire Analysis, Section 4) and floods as well as plant structure capacities are addressed separately in report Sections 3.1.2 through 3.1.4.

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
1	PI	AV616T	Air-Operated Valve V-26-16	AOV XFER	1
2	PI	AV618T	Air-Operated Valve V-26-18	AOV XFER	1
3	PI	AV701T	Air-Operated Valve V-27-1	AOV XFER	1
4	PI	AV702T	Air-Operated Valve V-27-2	AOV XFER	1
5	PI	AV703T	Air-Operated Valve V-27-3	AOV XFER	1
6	PI	AV704T	Air-Operated Valve V-27-4	AOV XFER	1
7	PI	AV847T	Air-Operated Valve V-28-47	AOV XFER	1
8	SG	AV819T	Common Inlet Valve V-28-19	AOV XFER	1
9	SG	DA049T	Train Inlet Damper DM-28-0049	DAMPER	1
10	SG	DA050T	Train Inlet Damper DM-28-0050	DAMPER	1
11	BI	HV194T	V-19-4 Tank Isolation Valve	HOV XFER	1
12	BI	HV195T	V-19-5 Pump B Suction Valve	HOV XFER	1
13	BI	HV196T	V-19-6 Pump A Suction Valve	HOV XFER	1
14	BI	HV197T	V-19-7 Pump A Discharge Valve	HOV XFER	1
15	BI	HV198T	V-19-8 Pump B Discharge Valve	HOV XFER	1
16	BI	HV919T	V-19-19 Injection Path Inboard Isolation	HOV XFER	1
17	BI	HV925T	V-19-25 Injection Path Outboard Isolation	HOV XFER	1
18	CD	CV029T	V-15-29 Manual Valve	HOV XFER	1
19	CD	HV005T	V-15-5 Manual Valve	HOV XFER	1
20	CD	HV006T	V-15-6 Manual Valve	HOV XFER	1
21	CD	HV026T	V-15-26 Manual Valve	HOV XFER	1
22	CD	HV030T	V-15-30 Manual Valve	HOV XFER	1
23	CP	HV037T	Pump manual isolation valve V-2-37	HOV XFER	1
24	CP	HV161T	Manual isolation valve V-2-161	HOV XFER	1
25	CP	HV226T	Manual valve V-2-26	HOV XFER	1
26	CP	HV227T	Pump manual isolation valve V-2-27	HOV XFER	1
27	CP	HV228T	Pump manual isolation valve V-2-28	HOV XFER	1
28	CP	HV229T	Condenser Discharge Valve F-2-29	HOV XFER	1
29	CP	HV230T	Condenser Discharge Valve F-2-30	HOV XFER	1
30	CP	HV231T	Condenser Discharge Valve F-2-31	HOV XFER	1
31	CP	HV232T	Condenser Discharge Valve F-2-32	HOV XFER	1
32	CP	HV233T	Condenser Discharge Valve F-2-33	HOV XFER	1
33	CP	HV234T	Condenser Discharge Valve F-2-34	HOV XFER	1
34	CP	HV236T	Manual valve V-2-236	HOV XFER	1
35	CP	HV235T	Injection line manual isol V-2-35	HOV XFER	1
36	CP	HV236T	Injection line manual isol V-2-36	HOV XFER	1
37	CP	HV237T	Manual valve V-2-237	HOV XFER	1
38	CP	HV238T	Pump manual isolation valve V-2-38	HOV XFER	1
39	CP	HV239T	Pump manual isolation valve V-2-39	HOV XFER	1
40	CP	HV240T	Manual valve V-2-40	HOV XFER	1
41	CP	HV241T	Manual valve V-2-41	HOV XFER	1
42	CP	HV254T	Manual valve V-2-54	HOV XFER	1
43	CP	HV255T	Manual valve V-2-55	HOV XFER	1
44	CP	HV256T	Manual valve V-2-56	HOV XFER	1
45	CP	HV290T	Manual isolation valve V-2-90	HOV XFER	1
46	CP	HV484T	Condensate demin. manual isol V-2-484	HOV XFER	1

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
47	CP	HV488	Condensate demin. manual isol V-2-488	HOV XFER	1
48	CP	HV489T	Condensate demin. manual isol V-2-489	HOV XFER	1
49	CP	HV490T	Condensate demin. manual isol V-2-490	HOV XFER	1
50	CP	HV492T	Condensate demin. manual isol V-2-492	HOV XFER	1
51	CP	HV493T	Condensate demin. manual isol V-2-493	HOV XFER	1
52	CP	HV566T	TBCCW isolation valve V-5-66	HOV XFER	1
53	CP	HV567T	TBCCW isolation valve V-5-67	HOV XFER	1
54	CP	HV571T	TBCCW isolation valve V-5-71	HOV XFER	1
55	CP	HV572T	TBCCW isolation valve V-5-72	HOV XFER	1
56	CP	HV576T	TBCCW isolation valve V-5-76	HOV XFER	1
57	CP	HV577T	TBCCW isolation valve V-5-77	HOV XFER	1
58	CP	HV702T	Condensate demin. manual isol V-2-702	HOV XFER	1
59	CP	HV703T	Condensate demin. manual isol V-2-703	HOV XFER	1
60	CP	HV704T	Condensate demin. manual isol V-2-704	HOV XFER	1
61	CP	HV705T	Condensate demin. manual isol V-2-705	HOV XFER	1
62	CP	HV706T	Condensate demin. manual isol V-2-706	HOV XFER	1
63	CP	HV921T	Condensate demin. manual isol V-2-491	HOV XFER	1
64	CS	HV017T	Manual Valve V-20-17	HOV XFER	1
65	CS	HV023T	Manual Valve V-20-23	HOV XFER	1
66	CS	HV082T	Manual Valve V-20-82	HOV XFER	1
67	CS	HV083T	Manual Valve V-20-83	HOV XFER	1
68	CS	HV090T	Manual Valve V-20-90	HOV XFER	1
69	CS	HV091T	Manual Valve V-20-91	HOV XFER	1
70	CT	HV004T	Manual Valve V-11-4	HOV XFER	1
71	CT	HV008T	Butterfly Valve V-11-8	HOV XFER	1
72	CT	HV009T	Butterfly Valve V-11-9	HOV XFER	1
73	CT	HV010T	Isolation Valve V-11-10	HOV XFER	1
74	CT	HV011T	Isolation Valve V-11-11	HOV XFER	1
75	CT	HV015T	Butterfly Valve V-11-15	HOV XFER	1
76	CT	HV041T	Manual Valve V-11-41	HOV XFER	1
77	CT	HV160T	Manual Valve V-11-160	HOV XFER	1
78	EC	HV003T	Manual Valve V-39-3	HOV XFER	1
79	EC	HV017T	Manual Valve V-39-17	HOV XFER	1
80	EC	HV049T	Manual Valve V-36-49	HOV XFER	1
81	EC	HV392T	Manual Valve V-39-2	HOV XFER	1
82	ED	HV392T	Manual Valve V-39-2	HOV XFER	1
83	ED	HVVVVT	Manual Valve V-39-VVV	HOV XFER	1
84	ED	HVWWWT	Manual Valve V-39-WWW	HOV XFER	1
85	ED	HVZZZT	Manual Valve V-36-ZZ	HOV XFER	1
86	FP	HV334T	Manual Suction Valve V-9-334	HOV XFER	1
87	FP	HV440T	Manual Isolation Valve V-9-440	HOV XFER	1
88	FP	HV904T	Manual Discharge Valve V-9-4	HOV XFER	1
89	FP	HV905T	Manual Discharge Valve V-9-5	HOV XFER	1
90	FP	HV907T	Manual Isolation Valve V-9-7	HOV XFER	1
91	FP	HV908T	Manual Isolation Valve V-9-8	HOV XFER	1
92	FP	HV913T	Manual Isolation Valve V-9-13	HOV XFER	1

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
93	FP	HV918T	Manual Isolation Valve V-9-18	HOV XFER	1
94	FP	HV933T	Manual Isolation Valve V-9-33	HOV XFER	1
95	FW	AV564T	Temperature control valve AOV-5-64	HOV XFER	1
96	FW	AV569T	Temperature control valve AOV-5-69	HOV XFER	1
97	FW	AV574T	Temperature control valve AOV-5-74	HOV XFER	1
98	FW	HV563T	TBCCW manual isol V-5-63	HOV XFER	1
99	FW	HV565T	TBCCW manual isol V-5-65	HOV XFER	1
100	FW	HV568T	TBCCW manual isol V-5-68	HOV XFER	1
101	FW	HV570T	TBCCW manual isol V-5-70	HOV XFER	1
102	FW	HV573T	TBCCW manual isol V-5-73	HOV XFER	1
103	FW	HV575T	TBCCW manual isol V-5-75	HOV XFER	1
104	IA	HV005T	Air Receiver Inlet Manual Valve V-6S-5	HOV XFER	1
105	IA	HV006T	Air Receiver Inlet Manual Valve V-6S-6	HOV XFER	1
106	IA	HV028T	Manual Valve V-9-28	HOV XFER	1
107	IA	HV028T	Manual Valve V-9-28	HOV XFER	1
108	IA	HV048T	Manual Valve V-9-48	HOV XFER	1
109	IA	HV048T	Manual Valve V-9-48	HOV XFER	1
110	IA	HV053T	TBCCW Inlet Manual Valve V-5-53	HOV XFER	1
111	IA	HV055T	TBCCW Outlet Manual Valve V-5-55	HOV XFER	1
112	IA	HV058T	TBCCW Inlet Manual Valve V-5-58	HOV XFER	1
113	IA	HV060T	TBCCW Outlet Manual Valve V-5-60	HOV XFER	1
114	IA	HV074T	Manual Valve V-9-74	HOV XFER	1
115	IA	HV074T	Manual Valve V-9-74	HOV XFER	1
116	IA	HV075T	Manual Valve V-9-75	HOV XFER	1
117	IA	HV076T	Manual Valve V-9-76	HOV XFER	1
118	IA	HV177T	Manual Valve V-6-3177	HOV XFER	1
119	IA	HV179T	Manual Valve V-6-3179	HOV XFER	1
120	IA	HV180T	Manual Valve V-6-3180	HOV XFER	1
121	IA	HV193T	Prefilter 1-1 Isolation Manual Valve V-6-193	HOV XFER	1
122	IA	HV194T	Prefilter 1-1 Isolation Manual Valve V-6-194	HOV XFER	1
123	IA	HV205T	Air Dryer Inlet Manual Valve V-6-205	HOV XFER	1
124	IA	HV207T	Air Dryer A-B Inlet Isolation Valve V-6-207	HOV XFER	1
125	IA	HV222T	Air Dryer A-B Outlet Isolation Valve V-6-22	HOV XFER	1
126	IA	HV240T	Air Dryer Outlet Manual Valve V-6-240	HOV XFER	1
127	IA	HV243T	Postfilter 1-1 Isolation Valve V-6-243	HOV XFER	1
128	IA	HV244T	Postfilter 1-1 Isolation Valve V-6-244	HOV XFER	1
129	IA	HV314T	Air Receiver Outlet Manual Valve V-6S-314	HOV XFER	1
130	IA	HV315T	Air Receiver Outlet Manual Valve V-6S-315	HOV XFER	1
131	MU	HV044T	Manual Valve V-11-44	HOV XFER	1
132	MU	HV049T	Manual Valve V-11-49	HOV XFER	1
133	MU	HV063T	Manual Valve V-11-463	HOV XFER	1
134	ST	HV001T	Butterfly valve V-11-1	HOV XFER	1
135	ST	HV002T	Manual Valve V-11-2	HOV XFER	1
136	ST	HV161T	Manual Valve V-11-161	HOV XFER	1
137	SW	HV033T	Pump Discharge Isolation Valve V-3-3	HOV XFER	1
138	SW	HV034T	Pump Discharge Isolation Valve V-3-34	HOV XFER	1

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
139	SW	HV058T	Circ. Water-TBCCW Heat Exchgr Isol.V-3-58	HOV XFER	1
140	SW	HV059T	Common Hdr.-TBCCW Heat Exchgr V-3-59	HOV XFER	1
141	TB	CV010T	Pump Discharge Iso. Valve V-5-10	HOV XFER	1
142	TB	CV011T	Pump Discharge Iso. Valve V-5-11	HOV XFER	1
143	TB	HV009T	Heat Exchanger Bypass Valve V-5-9	HOV XFER	1
144	TB	HV012T	Pump Discharge Iso. Valve V-5-12	HOV XFER	1
145	TB	HV013T	Pump Suction Iso. Valve V-5-13	HOV XFER	1
146	TB	HV014T	Pump Suction Iso. Valve V-5-14	HOV XFER	1
147	TB	HV015T	Pump Suction Iso. Valve V-5-15	HOV XFER	1
148	TB	HV058T	Circ. Water to Heat Exchanger Iso.V-3-58	HOV XFER	1
149	TB	HV058T	Circ. Water to Heat Exchanger Iso. V-3-58	HOV XFER	1
150	TB	HV059T	Serv. Water to Heat Exchanger Iso.V-3-59	HOV XFER	1
151	TB	HVHXAT	Heat Exchanger(C-5-004)Inlet Iso. V-5-8	HOV XFER	1
152	TB	HVHXAT	Heat Exchanger(C-5-004)Outlet Iso. V-5-7	HOV XFER	1
153	TB	HVHXAT	HX (C-5-004)Cool.Water Inlet Iso.V-3-75	HOV XFER	1
154	TB	HVHXAT	HX (C-5-004)Cool.Water Outlet V-3-77	HOV XFER	1
155	TB	HVHXBT	Heat Exchanger(C-5-003)Inlet Iso. V-5-6	HOV XFER	1
156	TB	HVHXBT	Heat Exchngr(C-5-003)Outlet Iso. Val V-5-5	HOV XFER	1
157	TB	HVHXBT	HX (C-5-003)Cool.Water Inlet Iso.V-3-74	HOV XFER	1
158	TB	HVHXBT	HX (C-5-003)Cool Water Outlet Iso. V-3-76	HOV XFER	1
159	BI	MV162T	V-16-2 RWCU Aux Cleanup Pump Isolation	MOV XFER	1
160	BI	MV164T	V-16-2 RWCU Aux Cleanup Pump Isolation	MOV XFER	1
161	CC	MV003T	Pump 51D Suction Valve V-21-3	MOV XFER	1
162	CC	MV005T	Drywell Header Inlet V-21-5	MOV XFER	1
163	CC	MV007T	Pump 51B(1-2) Suction Valve V-21-7	MOV XFER	1
164	CC	MV009T	Pump 51A(1-1) Suction Valve V-21-9	MOV XFER	1
165	CC	MV011T	Drywell Header Inlet V-21-11	MOV XFER	1
166	CC	MV013T	Dynamic Test (Torus) Return	MOV XFER	1
167	CC	MV087T	HX ESW Outlet Isol V-21-87	MOV XFER	1
168	CC	MVO01T	Pump 51C Suction Valve V-21-1	MOV XFER	1
169	CC	MVO88T	HX ESW Outlet Isol V-21-88	MOV XFER	1
170	CP	MV021T	SJAE Isolation MOV-2-1	MOV XFER	1
171	CP	MV022T	SJAE isolation valve MOV-2-2	MOV XFER	1
172	CP	MV023T	SJAE isolation valve MOV-2-3	MOV XFER	1
173	CP	MV024T	SJAE isolation valve MOV-2-4	MOV XFER	1
174	CP	MV025T	SJAE isolation valve MOV-2-5	MOV XFER	1
175	CP	MV026T	SJAE isolation valve MOV-2-6	MOV XFER	1
176	CP	MV027T	FW train A isol MOV-2-7	MOV XFER	1
177	CP	MV028T	FW train B isol MOV-2-8	MOV XFER	1
178	CP	MV029T	FW train C isol MOV-2-9	MOV XFER	1
179	CP	MV210T	FW train A isol MOV-2-10	MOV XFER	1
180	CP	MV211T	FW train B isol MOV-2-11	MOV XFER	1
181	CP	MV212T	FW train C isol MOV-2-12	MOV XFER	1
182	CS	HV026T	Recirc. Torus Valve V-20-26	MOV XFER	1
183	CS	HV027T	Recirc. to Torus Valve V-20-27	MOV XFER	1
184	CS	MV003T	Pump Suction MOV V-20-3	MOV XFER	1

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
185	CS	MV004T	Pump Suction MOV V-20-4	MOV XFER	1
186	CS	MV012T	Discharge MOV V-20-12	MOV XFER	1
187	CS	MV018T	Discharge MOV V-20-18	MOV XFER	1
188	CS	MV032T	Pump Suction MOV V-20-32	MOV XFER	1
189	CS	MV033T	Pump Suction MOV V-20-33	MOV XFER	1
190	CW	MV311T	Discharge Valve MOV V-3-11	MOV XFER	1
191	CW	MVO10T	Discharge Valve MOV V-3-10	MOV XFER	1
192	CW	MVO38T	Discharge Valve MOV V-3-8	MOV XFER	1
193	CW	MVO39T	Discharge Valve MOV V-3-9	MOV XFER	1
194	FW	MV027T	FW train A isol MOV-2-7	MOV XFER	1
195	FW	MV028T	FW train B isol MOV-2-8	MOV XFER	1
196	FW	MV029T	FW train C isol MOV-2-9	MOV XFER	1
197	FW	MV210T	FW train A isol MOV-2-10	MOV XFER	1
198	FW	MV211T	FW train B isol MOV-2-11	MOV XFER	1
199	FW	MV212T	FW train C isol MOV-2-12	MOV XFER	1
200	IC	MV030T	Steam Isolation valve MOV-14-30	MOV XFER	1
201	IC	MV031T	Steam Isolation valve MOV-14-31	MOV XFER	1
202	IC	MV032T	Steam Isolation valve MOV-14-32	MOV XFER	1
203	IC	MV033T	Steam Isolation valve MOV-14-33	MOV XFER	1
204	IC	MV034T	Condensate Isolation valve MOV-14-34	MOV XFER	1
205	IC	MV035T	Condensate Isolation valve MOV-14-35	MOV XFER	1
206	IC	MV036T	Condensate Isolation valve MOV-14-36	MOV XFER	1
207	IC	MV037T	Condensate Isolation valve MOV-14-37	MOV XFER	1
208	IA	SV211T	Air Dryer Four-Way Valve V-6-211	SOV XFER	1
209	PI	SV838T	Solenoid Valve V-38-38	SOV XFER	1
210	BI	CV916D	V-19-16 Common Line Check Valve	CHECK VLV	2
211	BI	CV916T	V-19-16 Common Line Check Valve	CHECK VLV	2
212	BI	CV920D	V-19-20 Common Line Check Valve	CHECK VLV	2
213	BI	CV920T	V-19-20 Common Line Check Valve	CHECK VLV	2
214	BI	CV937D	V-19-37 Pump A Discharge Check Valve	CHECK VLV	2
215	BI	CV937T	V-19-37 Pump A Discharge Check Valve	CHECK VLV	2
216	BI	CV938D	V-19-38 Pump B Discharge Check Valve	CHECK VLV	2
217	BI	CV938T	V-19-38 Pump B Discharge Check Valve	CHECK VLV	2
218	CC	CV002D	Discharge Valve V-21-2	CHECK VLV	2
219	CC	CV002D	Pump Discharge Valve V-21-2	CHECK VLV	2
220	CC	CV002L	Pump Discharge CV V-21-2	CHECK VLV	2
221	CC	CV004D	Pump Discharge CV V-21-4	CHECK VLV	2
222	CC	CV004D	Pump Discharge CV V-21-4	CHECK VLV	2
223	CC	CV004L	Pump Discharge Valve V-21-4	CHECK VLV	2
224	CC	CV008D	Pump Discharge Valve V-21-8	CHECK VLV	2
225	CC	CV008D	Pump Discharge Valve V-21-10	CHECK VLV	2
226	CC	CV008D	Pump Discharge Valve V-21-8	CHECK VLV	2
227	CC	CV008L	Pump Discharge CV V-21-8	CHECK VLV	2
228	CC	CV010D	Pump Discharge CV V-21-10	CHECK VLV	2
229	CC	CV010L	Pump Discharge CV V-21-10	CHECK VLV	2
230	CC	CV065D	ESW Pump D Discharge CV V-3-65	CHECK VLV	2

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
231	CC	CV065D	Discharge Check Valve V-3-65	CHECK VLV	2
232	CC	CV065L	Discharge Check Valve V-3-65	CHECK VLV	2
233	CC	CV066D	Discharge Check Valve V-3-66	CHECK VLV	2
234	CC	CV066D	Discharge Check Valve V-3-66	CHECK VLV	2
235	CC	CV067D	ESW Pump 52B Disch. CV V-3-67	CHECK VLV	2
236	CC	CV067L	ESW Pump Discharge CV V-3-67	CHECK VLV	2
237	CC	CV068D	ESW Pump 52A Disch. CV V-3-68	CHECK VLV	2
238	CC	CV068L	ESW Pump Discharge CV V-3-68	CHECK VLV	2
239	CC	CVO66D	ESW Pump Discharge CV V-3-66	CHECK VLV	2
240	CC	CVO68D	ESW Pump Discharge CV	CHECK VLV	2
241	CC	PM52B	ESW Pump Discharge CV	CHECK VLV	2
242	CD	CV007C	V-15-7 Stop Check Valve	CHECK VLV	2
243	CD	CV007D	V-15-7 Stop Check Valve	CHECK VLV	2
244	CD	CV007L	V-15-7 Stop Check Valve	CHECK VLV	2
245	CD	CV007T	V-15-7 Stop Check Valve	CHECK VLV	2
246	CD	CV010D	V-15-10 Stop Check Valve	CHECK VLV	2
247	CD	CV010L	V-15-10 Stop Check Valve	CHECK VLV	2
248	CD	CV010T	V-15-10 Stop Check Valve	CHECK VLV	2
249	CD	CV027T	V-15-27 Check Valve	CHECK VLV	2
250	CD	HV028T	V-15-28 Check Valve	CHECK VLV	2
251	CP	CV272T	Injection line check valve V-2-72	CHECK VLV	2
252	CP	CV273T	Discharge check valve V-2-78	CHECK VLV	2
253	CP	CV273T	Injection line check valve V-2-73	CHECK VLV	2
254	CP	CV274T	Injection line check valve V-2-74	CHECK VLV	2
255	CP	CV275R	Discharge check valve V-2-75	CHECK VLV	2
256	CP	CV275T	Discharge check valve V-2-75	CHECK VLV	2
257	CP	CV276R	Discharge check valve V-2-76	CHECK VLV	2
258	CP	CV276T	Discharge check valve V-2-76	CHECK VLV	2
259	CP	CV277R	Discharge check valve V-2-77	CHECK VLV	2
260	CP	CV277T	Discharge check valve V-2-77	CHECK VLV	2
261	CP	CV279T	Discharge check valve V-2-79	CHECK VLV	2
262	CP	CV280T	Discharge check valve V-2-80	CHECK VLV	2
263	CP	CVV271T	Injection line check valve V-2-71	CHECK VLV	2
264	CS	CV008D	Pump Discharge CV V-20-8	CHECK VLV	2
265	CS	CV008P	Pump Discharge CV V-20-8	CHECK VLV	2
266	CS	CV009D	Pump Discharge CV V-20-9	CHECK VLV	2
267	CS	CV009P	Pump Discharge CV V-20-9	CHECK VLV	2
268	CS	CV016D	Pump Discharge CV V-20-16	CHECK VLV	2
269	CS	CV016P	Pump Discharge CV V-20-16	CHECK VLV	2
270	CS	CV022D	Pump Discharge CV V-20-22	CHECK VLV	2
271	CS	CV022P	Pump Discharge CV V-20-22	CHECK VLV	2
272	CS	CV022R	Pump Discharge CV V-20-22	CHECK VLV	2
273	CS	CV030D	Check Valve V-20-30	CHECK VLV	2
274	CS	CV031D	Check Valve V-20-31	CHECK VLV	2
275	CS	CV050L	Vacuum Break (Sys I) CV V-20-50	CHECK VLV	2
276	CS	CV051D	Check Valve V-20-51	CHECK VLV	2

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
277	CS	CV052D	Discharge Check Valve V-20-52	CHECK VLV	2
278	CS	CV052P	Discharge Check Valve V-20-52	CHECK VLV	2
279	CS	CV053D	Booster Discharge CV V-20-53	CHECK VLV	2
280	CS	CV053L	Booster Discharge CV V-20-53	CHECK VLV	2
281	CS	CV053P	Booster Discharge CV V-20-53	CHECK VLV	2
282	CS	CV054D	Booster Discharge CV V-20-54	CHECK VLV	2
283	CS	CV054P	Booster Discharge CV V-20-54	CHECK VLV	2
284	CS	CV055D	Booster Discharge CV V-20-55	CHECK VLV	2
285	CS	CV055L	Booster Discharge CV V-20-55	CHECK VLV	2
286	CS	CV055P	Booster Discharge CV V-20-55	CHECK VLV	2
287	CS	CV060D	Check Valve V-20-60	CHECK VLV	2
288	CS	CV060T	Check Valve V-20-60	CHECK VLV	2
289	CS	CV061D	Check Valve V-20-61	CHECK VLV	2
290	CS	CV061T	Check Valve V-20-61	CHECK VLV	2
291	CS	CV088D	Check Valve V-20-88	CHECK VLV	2
292	CS	CV088T	Check Valve V-20-88	CHECK VLV	2
293	CS	CV089D	Check Valve V-20-89	CHECK VLV	2
294	CS	CV089T	Check Valve V-20-89	CHECK VLV	2
295	CS	CV150D	Testable CV NZ02A (V-20-150)	CHECK VLV	2
296	CS	CV150P	Testable CV NZ02A (V-20-150)	CHECK VLV	2
297	CS	CV151D	Testable CV NZ02B (V-20-151)	CHECK VLV	2
298	CS	CV151P	Testable CV NZ02B (V-20-151)	CHECK VLV	2
299	CS	CV152D	Testable CV NZ02C (V-20-152)	CHECK VLV	2
300	CS	CV152P	Testable CV NZ02C (V-20-152)	CHECK VLV	2
301	CS	CV153D	Testable CV NZ02D (V-20-153)	CHECK VLV	2
302	CS	CV153P	Testable CV NZ02D (V-20-153)	CHECK VLV	2
303	CT	CV012L	Check Valve V-11-12	CHECK VLV	2
304	CT	CV012P	Check Valve V-11-12	CHECK VLV	2
305	CT	CV013D	Check Valve V-11-13	CHECK VLV	2
306	CT	CV013P	Check Valve V-11-13	CHECK VLV	2
307	CT	CV042P	Check Valve V-11-42	CHECK VLV	2
308	EC	CV01AD	Check Valve 1A	CHECK VLV	2
309	EC	CV01BD	Check Valve 1B	CHECK VLV	2
310	EC	CV391D	Check Valve V-39-1	CHECK VLV	2
311	EC	CVXXXD	Check Valve XXX	CHECK VLV	2
312	ED	CV02AD	Check Valve 2A	CHECK VLV	2
313	ED	CV02BD	Check Valve 2B	CHECK VLV	2
314	ED	CV391D	Check Valve V-39-1	CHECK VLV	2
315	ED	CVYYYD	Check Valve YYY	CHECK VLV	2
316	FP	CV336D	Check Valve V-9-336	CHECK VLV	2
317	FP	CV336P	Check Valve V-9-336	CHECK VLV	2
318	FP	CV901D	Check Valve V-9-1	CHECK VLV	2
319	FP	CV901D	Check Valve V-9-1	CHECK VLV	2
320	FP	CV901P	Check Valve V-9-1	CHECK VLV	2
321	FP	CV902C	Check Valve V-9-2	CHECK VLV	2
322	FP	CV902D	Check Valve V-9-2	CHECK VLV	2

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
323	FP	CV902P	Check Valve V-9-2	CHECK VLV	2
324	FW	CV278R	Check valve V-2-78	CHECK VLV	2
325	FW	CV278T	Discharge check valve V-2-78	CHECK VLV	2
326	FW	CV279R	Check valve V-2-79	CHECK VLV	2
327	FW	CV279T	Discharge check valve V-2-79	CHECK VLV	2
328	FW	CV280R	Check valve V-2-80	CHECK VLV	2
329	FW	CV280T	Discharge check valve V-2-80	CHECK VLV	2
330	IA	CV078D	Check Valve V-6-3078	CHECK VLV	2
331	IA	CV079D	Check Valve V-6-3079	CHECK VLV	2
332	MU	CV033D	Check Valve V-11-33	CHECK VLV	2
333	MU	CV033P	Check Valve V-11-33	CHECK VLV	2
334	MU	CV035D	Check Valve V-11-35	CHECK VLV	2
335	MU	CV035P	Check Valve V-11-35	CHECK VLV	2
336	MU	CV042L	Check Valve V-11-42	CHECK VLV	2
337	PI	CV515L	Check Valve V-26-15	CHECK VLV	2
338	PI	CV617L	Check Valve V-26-17	CHECK VLV	2
339	SW	CV062C	Pump Discharge Check Value V-3-62	CHECK VLV	2
340	SW	CV062D	Pump Discharge Check Valve V-3-62	CHECK VLV	2
341	SW	CV062L	Pump Discharge Check Value V-3-62	CHECK VLV	2
342	SW	CV062P	Pump Discharge Check Valve V-3-62	CHECK VLV	2
343	SW	CV063D	Pump Discharge Check Valve V-3-63	CHECK VLV	2
344	SW	CV063L	Pump Discharge Check Valve V-3-63	CHECK VLV	2
345	SW	CV063P	Pump Discharge Check Valve V-3-63	CHECK VLV	2
346	TB	CV016D	Pump Discharge Check Valve V-5-16	CHECK VLV	2
347	TB	CV016L	Pump Discharge Check Valve V-5-16	CHECK VLV	2
348	TB	CV016T	Pump Discharge Check Valve V-5-16	CHECK VLV	2
349	TB	CV016T	Pump Discharge Check Valve V-5-16	CHECK VLV	2
350	TB	CV017D	Pump Discharge Check Valve V-5-17	CHECK VLV	2
351	TB	CV017L	Pump Discharge Check Valve V-5-17	CHECK VLV	2
352	TB	CV017T	Pump Discharge Check Valve V-5-17	CHECK VLV	2
353	TB	CV017T	Pump Discharge Check Valve V-5-17	CHECK VLV	2
354	TB	CV018D	Pump Discharge Check Valve V-5-18	CHECK VLV	2
355	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	2
356	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	2
357	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	2
358	FP	PP001B	Piping Integrity	PIPE	3
359	CP	FLDMAP	Condensate demineralizer A	DEMIN	4
360	CP	FLDMBP	Condensate demineralizer B	DEMIN	4
361	CP	FLDMCP	Condensate demineralizer C	DEMIN	4
362	CP	FLDMDP	Condensate demineralizer D	DEMIN	4
363	CP	FLDMEP	Condensate demineralizer E	DEMIN	4
364	CP	FLDMFP	Condensate demineralizer F	DEMIN	4
365	EC	FLF1AP	Fuel Oil Filter 1A	FILTER	4
366	EC	FLF1BP	Fuel Oil Filter 1B	FILTER	4
367	ED	FLF2AP	Fuel Oil Filter 2A	FILTER	4
368	ED	FLF2BP	Fuel Oil Filter 2B	FILTER	4

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
369	IA	FL01AP	Prefilter 1-1	FILTER	4
370	IA	FL01BP	Postfilter 1-1	FILTER	4
371	IA	FL01CP	Air Intake Filter	FILTER	4
372	IA	FL01DP	Air Intake Filter	FILTER	4
373	SG	FL007P	Train HEPA Filter F-1-7	FILTER	4
374	SG	FL008P	Train Charcoal Filter F-1-8	FILTER	4
375	SG	FL009P	Train Postfilter F-1-9	FILTER	4
376	SG	FL010P	Train HEPA Filter F-1-10	FILTER	4
377	SG	FL011P	Train Charcoal Filter F-1-11	FILTER	4
378	SG	FL012P	Train Postfilter F-1-12	FILTER	4
379	SG	FL029P	Train Prefilter F-1-29	FILTER	4
380	SG	FL030P	Train Prefilter F-1-30	FILTER	4
381	CP	HX0A1P	LP heater A-1	HX PLUG	4
382	CP	HX0A2P	LP heater A-2	HX PLUG	4
383	CP	HX0B1P	LP heater B-1	HX PLUG	4
384	CP	HX0B2P	LP heater B-2	HX PLUG	4
385	CP	HX0C1P	LP heater C-1	HX PLUG	4
386	CP	HX0C2P	LP heater C-2	HX PLUG	4
387	CP	HXDCAP	Drain cooler A	HX PLUG	4
388	CP	HXDCBP	Drain cooler B	HX PLUG	4
389	CP	HXDCCP	Drain cooler C	HX PLUG	4
390	CP	HXHPAP	HP heater A	HX PLUG	4
391	CP	HXHPBP	HP heater B	HX PLUG	4
392	CP	HXHPCP	HP heater C	HX PLUG	4
393	CP	HXICAP	SJAE Inter condenser A	HX PLUG	4
394	CP	HXICBP	SJAE Inter condenser B	HX PLUG	4
395	CP	HXICCP	SJAE Inter condenser C	HX PLUG	4
396	CP	HXSSEP	Steam seal exhauster	HX PLUG	4
397	FW	HX0A1P	LP heater A-1	HX PLUG	4
398	FW	HX0A2P	LP heater A-2	HX PLUG	4
399	FW	HXHPAP	HP heater A	HX PLUG	4
400	CD	ST51AT	NC51A Y-Strainer	STRAINER	4
401	CD	ST51BT	NC51B Y-Strainer	STRAINER	4
402	BI	HTSLCR	Tank Discharge Piping Heat Tracing	HT TRACE	5
403	TT	SWNLSC	No Load Switch	SWITCH	5
404	AD	SL18AD	Mech Switch RE18A (Triple low)	PRESS SW	6
405	AD	SL18BD	Mech Switch RE18C (Triple low)	PRESS SW	6
406	AD	SL18CD	Mech Switch RE18C (Triple low)	PRESS SW	6
407	AD	SL18DD	Mech Switch RE18C (Triple low)	PRESS SW	6
408	AD	SP40AD	Differential Pressure Switch RV40A	PRESS SW	6
409	AD	SP40BD	Differential Pressure Switch RV40B	PRESS SW	6
410	AD	SP40CD	Differential Pressure Switch RV40C	PRESS SW	6
411	AD	SP40DD	Differential Pressure Switch RV40D	PRESS SW	6
412	BI	FIL06D	FS 1L06 Flow Sensor To Isolate RWCU	PRESS SW	6
413	CC	SP15AD	Drywell Pressure Switch IP-15A	PRESS SW	6
414	CC	SP15BD	Drywell Pressure Switch IP-15B	PRESS SW	6

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
415	CC	SP15CD	Drywell Pressure Switch IP-15C	PRESS SW	6
416	CC	SP15DD	Drywell Pressure Switch IP-15D	PRESS SW	6
417	CC	SP17AD	Low Flow Pressure Switch PS17A	PRESS SW	6
418	CC	SP17BD	Low Flow Pressure Switch PS17B	PRESS SW	6
419	CF	SP15AD	Drywell Pressure Switch IP-15A	PRESS SW	6
420	CF	SP15BD	Drywell Pressure Switch IP-15B	PRESS SW	6
421	CF	SP15CD	Drywell Pressure Switch IP-15C	PRESS SW	6
422	CF	SP15DD	Drywell Pressure Switch IP-15D	PRESS SW	6
423	CS	SP29AD	PS RV29A-Sys.I Booster Pumps	PRESS SW	6
424	CS	SP29BD	PS RV29B-Sys.II Booster Pumps	PRESS SW	6
425	CS	SP29CD	PS RV29C-Sys.I Booster Pumps	PRESS SW	6
426	CS	SP29DD	PS RV29D-Sys.II Booster Pumps	PRESS SW	6
427	CS	SP40AD	Pressure Switch RV40A	PRESS SW	6
428	CS	SP40BD	Pressure Switch RV40B	PRESS SW	6
429	CS	SP40CD	Pressure Switch RV40C	PRESS SW	6
430	CS	SP40DD	Pressure Switch RV40D	PRESS SW	6
431	DP	SP46AD	Drywell Pressure Switch RV46A	PRESS SW	6
432	DP	SP46BD	Drywell Pressure Switch RV46B	PRESS SW	6
433	DP	SP46CD	Drywell Pressure Switch RV46C	PRESS SW	6
434	DP	SP46DD	Drywell Pressure Switch RV46D	PRESS SW	6
435	IC	PSCA1I	IC A isolation condensate line PS 1	PRESS SW	6
436	IC	PSCA2I	IC A isolation condensate line PS 1	PRESS SW	6
437	IC	PSCB1I	IC B isolation condensate line PS 1	PRESS SW	6
438	IC	PSCB2I	IC B isolation condensate line PS 1	PRESS SW	6
439	IC	PSSA1I	IC A isolation steamline pressure switch 1	PRESS SW	6
440	IC	PSSA2I	IC A isolation steamline pressure switch 2	PRESS SW	6
441	IC	PSSB1I	IC B isolation steamline pressure switch 1	PRESS SW	6
442	IC	PSSB2I	IC B isolation steamline pressure switch 2	PRESS SW	6
443	ME	SP23AD	Main Steam Low Pressure Switch RE-23A	PRESS SW	6
444	ME	SP23BD	Main Steam Low Pressure Switch RE-23B	PRESS SW	6
445	ME	SP23CD	Main Steam Low Pressure Switch RE-23C	PRESS SW	6
446	ME	SP23DD	Main Steam Low Pressure Switch RE-23D	PRESS SW	6
447	MI	SP1A1D	IC Condensate Line DP Switch 11A1	PRESS SW	6
448	MI	SP1A2D	IC Condensate Line DP Switch 11A2	PRESS SW	6
449	MI	SP1B1D	IC Condensate Line DP Switch 11B1	PRESS SW	6
450	MI	SP1B2D	IC Condensate Line DP Switch 11B2	PRESS SW	6
451	MI	SP5A1D	IC Steam Line DP Switch 5A1	PRESS SW	6
452	MI	SP5A2D	IC Steam Line DP Switch 5A2	PRESS SW	6
453	MI	SP5B1D	IC Steam Line DP Switch 5B1	PRESS SW	6
454	MI	SP5B2D	IC Steam Line DP Switch 5B2	PRESS SW	6
455	PR	SP15AD	Reactor Pressure Switch RE15A	PRESS SW	6
456	PR	SP15BD	Reactor Pressure Switch RE15B	PRESS SW	6
457	PR	SP15CD	Reactor Pressure Switch RE15C	PRESS SW	6
458	PR	SP15DD	Reactor Pressure Switch RE15D	PRESS SW	6
459	RL	LS02AD	Reactor Water Level Switch RE02A	PRESS SW	6
460	RL	LS02BD	Reactor Water Level Switch RE02B	PRESS SW	6

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
461	RL	LS02CD	Reactor Water Level Switch RE02C	PRESS SW	6
462	RL	LS02DD	Reactor Water Level Switch RE02D	PRESS SW	6
463	SG	SP255D	Train Flow Switch FIS-28-255	PRESS SW	6
464	TT	EPR01D	Electrical Pressure Regulator	PRESS SW	6
465	VO	SP23AD	Reactor Pressure Switch PS-IDA83A	PRESS SW	6
466	VO	SP23BD	Reactor Pressure Switch PS-IDA83B	PRESS SW	6
467	VO	SP23CD	Reactor Pressure Switch PS-IDA83C	PRESS SW	6
468	VO	SP23DD	Reactor Pressure Switch PS-IDA83D	PRESS SW	6
469	VO	SP23ED	Reactor Pressure Switch PS-IDA83E	PRESS SW	6
470	VR	SP83AD	Reactor Pressure Switch PS-IDA83A	PRESS SW	6
471	VR	SP83BD	Reactor Pressure Switch PS-IDA83B	PRESS SW	6
472	VR	SP83CD	Reactor Pressure Switch PS-IDA83C	PRESS SW	6
473	VR	SP83DD	Reactor Pressure Switch PS-IDA83D	PRESS SW	6
474	VR	SP83ED	Reactor Pressure Switch PS-IDA83E	PRESS SW	6
475	RS	LG018D	Trip Unit PT-622-1018	RESS TRAN	6
476	RS	LG019D	Trip Unit PT-622-1019	RESS TRAN	6
477	RS	PX018R	Reactor Pressure Transmitter PT-622-1018	RESS TRAN	6
478	RS	PX019R	Reactor Pressure Transmitter PT-622-1019	RESS TRAN	6
479	RS	AVSC12	Scram Valves(137 Outlet Valves CV127s)	FAIL SAFE	7
480	BT	TVBBVD	Trip Oil Dump Valve	FAIL TRIPPED	7
481	TT	CV001D	Turbine Control Valve CV-1	SUCCESS	7
482	TT	CV001T	Turbine Control Valve CV-1	SUCCESS	7
483	TT	CV002D	Turbine Control Valve CV-2	SUCCESS	7
484	TT	CV002T	Turbine Control Valve CV-2	SUCCESS	7
485	TT	CV003D	Turbine Control Valve CV-3	SUCCESS	7
486	TT	CV003T	Turbine Control Valve CV-3	SUCCESS	7
487	TT	CV004D	Turbine Control Valve CV-4	SUCCESS	7
488	TT	CV004T	Turbine Control Valve CV-4	SUCCESS	7
489	TT	EPR01T	Electrical Pressure Regulator	SUCCESS	7
490	TT	RLMT1D	Main Trip Solenoid No. 1	SUCCESS	7
491	TT	RLMT3D	Main Trip Solenoid #3	SUCCESS	7
492	TT	SV001D	Turbine Stop Valve SV-1	SUCCESS	7
493	TT	SV001T	Turbine Stop Valve SV-1	SUCCESS	7
494	TT	SV002D	Turbine Stop Valve SV-2	SUCCESS	7
495	TT	SV002T	Turbine Stop Valve SV-2	SUCCESS	7
496	TT	SV003D	Turbine Stop Valve SV-3	SUCCESS	7
497	TT	SV003T	Turbine Stop Valve SV-3	SUCCESS	7
498	TT	SV004D	Turbine Stop Valve SV-4	SUCCESS	7
499	TT	SV004T	Turbine Stop Valve SV-4	SUCCESS	7
500	TT	VTEDYD	Emergency Trip Oil Dump Valve	SUCCESS	7
501	BV	EPR01D	Electrical Pressure Regulator	CONTROL	8
502	BV	EPR01T	Electrical Pressure Regulator	CONTROL	8
503	RF	LGLLSD	Low level setdown	CONTROL	8
504	RF	SMFWAR	FRV A controller	CONTROL	8
505	RF	SMFWBR	FRV B controller	CONTROL	8
506	RF	SMFWCR	FRV C controller	CONTROL	8

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
507	RF	SMMASR	Master RPV level controller	CONTROL	8
508	AD	RL01AD	Relay 16K201A	RELAY	8
509	AD	RL01BD	Relay 16K201B	RELAY	8
510	AD	RL02AD	Timer Relay 16K202A	RELAY	8
511	AD	RL02BD	Timer Relay 16K202B	RELAY	8
512	AD	RL14AD	Relay 16K114A	RELAY	8
513	AD	RL14BD	Relay 16K114B	RELAY	8
514	AD	RL14CD	Relay 16K114C	RELAY	8
515	AD	RL14DD	Relay 16K114D	RELAY	8
516	AD	RL20AD	Timer Relay 16K220A	RELAY	8
517	AD	RL20BD	Timer Relay 16K220B	RELAY	8
518	AD	RL24AD	Relay 16K214A	RELAY	8
519	AD	RL24BD	Relay 16K214B	RELAY	8
520	AD	RL25AD	Relay 16K215A	RELAY	8
521	AD	RL25BD	Relay 16K215B	RELAY	8
522	AD	RL26AD	Switching Relay 16K216A	RELAY	8
523	AD	RL26CD	Switching Relay 16K216C	RELAY	8
524	AD	RL26ED	Switching Relay 16K216E	RELAY	8
525	AD	RL27AD	Relay 16K217A	RELAY	8
526	AD	RL27BD	Relay 16K217B	RELAY	8
527	AD	RL27CD	Relay 16K217C	RELAY	8
528	AD	RL27DD	Relay 16K217D	RELAY	8
529	AD	RLAX1D	Relay 16K115AX1	RELAY	8
530	AD	RLAX2D	Relay 16K115AX2	RELAY	8
531	AD	RLAX3D	Initiation Relay 16K115AX3	RELAY	8
532	AD	RLBX1D	Relay 16K115BX1	RELAY	8
533	AD	RLBX2D	Relay 16K115BX2	RELAY	8
534	AD	RLBX3D	Initiation Relay 16K115BX3	RELAY	8
535	AD	RLCX1D	Relay 16K115CX1	RELAY	8
536	AD	RLCX2D	Relay 16K115CX2	RELAY	8
537	AD	RLCX3D	Initiation Relay 16K115CX3	RELAY	8
538	AD	RLDX1D	Relay 16K115DX1	RELAY	8
539	AD	RLDX2D	Relay 16K115DX2	RELAY	8
540	AD	RLDX3D	Initiation Relay 16K115DX3	RELAY	8
541	AD	RLT4BD	Timer Relay 16K224B	RELAY	8
542	AD	RLT4CD	Timer Relay 16K224C	RELAY	8
543	AD	RLT4ED	Timer Relay 16K224E	RELAY	8
544	AD	RLT5BD	Timer Relay 16K225B	RELAY	8
545	AD	RLT5CD	Timer Relay 16K225C	RELAY	8
546	AD	RLT5ED	Timer Relay 16K225E	RELAY	8
547	BI	RL3K7D	Relay 3K7 in RWCU Isolation Logic	RELAY	8
548	BT	RLMT2D	Condenser Vacuum Trip No. 2	RELAY	8
549	CC	RL27AD	Low Flow Relay 16K27A	RELAY	8
550	CC	RL27BD	Low Flow Relay 16K27B	RELAY	8
551	CC	RLK2AD	Relay 16K2A	RELAY	8
552	CC	RLK2BD	Relay 16K2B	RELAY	8

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
553	CC	RLK4AD	Time Delay Relay 16K4A	RELAY	8
554	CC	RLK4BD	Time Delay Relay 16K4B	RELAY	8
555	CC	RLK6AD	Relay 16K6A	RELAY	8
556	CC	RLK6BD	Relay 16K6B	RELAY	8
557	CC	RLK8AD	Relay 16K8A	RELAY	8
558	CC	RLK8BD	Relay 16K8B	RELAY	8
559	CC	RLTK1D	Relay TK1	RELAY	8
660	CF	RLTK3D	Relay TK3	RELAY	8
661	CF	RL25AD	Relay 16K25A	RELAY	8
662	CF	RL25BD	Relay 16K25B	RELAY	8
663	CF	RL26AD	Relay 16K26A	RELAY	8
664	CF	RL26BD	Relay 16K26B	RELAY	8
665	CF	RLK2AD	Pump Start Relay 16K2A	RELAY	8
666	CF	RLK2BD	Pump Start Relay 16K2B	RELAY	8
667	CS	RL01AD	Pump Actuation Relay 101A	RELAY	8
668	CS	RL01BD	Pump Actuation Relay 101B	RELAY	8
669	CS	RL01CD	Pump Actuation Relay 101C	RELAY	8
670	CS	RL01DD	Pump Actuation Relay 101D	RELAY	8
671	CS	RL02AD	Booster Pump Actuation Relay 102A	RELAY	8
672	CS	RL02BD	Booster Pump Actuation Relay 102B	RELAY	8
673	CS	RL02CD	Booster Pump Actuation Relay 102C	RELAY	8
674	CS	RL02DD	Booster Pump Actuation Relay 102D	RELAY	8
675	CS	RL03AD	Pump Auxiliary Relay 103A	RELAY	8
676	CS	RL03BD	Pump Auxiliary Relay 103B	RELAY	8
677	CS	RL03CD	Pump Auxiliary Relay 103C	RELAY	8
678	CS	RL03DD	Pump Auxiliary Relay 103D	RELAY	8
679	CS	RL04AD	Pump Time Delay Relay 104A	RELAY	8
680	CS	RL04BD	Pump Time Delay Relay 104B	RELAY	8
681	CS	RL04CD	Pump Time Delay Relay 104C	RELAY	8
682	CS	RL04DD	Pump Time Delay Relay 104D	RELAY	8
683	CS	RL05AD	Relay 105A	RELAY	8
684	CS	RL05BD	Relay 105B	RELAY	8
685	CS	RL05CD	Relay 105C	RELAY	8
686	CS	RL05DD	Relay 105D	RELAY	8
687	CS	RL09AD	Booster Pump Auxiliary Relay 109A	RELAY	8
688	CS	RL09BD	Booster Pump Auxiliary Relay 109B	RELAY	8
689	CS	RL09CD	Booster Pump Auxiliary Relay 109C	RELAY	8
690	CS	RL09DD	Booster Pump Auxiliary Relay 109D	RELAY	8
691	CS	RL11AD	Relay 111A	RELAY	8
692	CS	RL11BD	Relay 111B	RELAY	8
693	CS	RL11CD	Relay 111C	RELAY	8
694	CS	RL11DD	Relay 111D	RELAY	8
695	CS	RL12AD	Booster Time Delay Relay 112A	RELAY	8
696	CS	RL12BD	Booster Time Delay Relay 112B	RELAY	8
697	CS	RL12CD	Booster Time Delay Relay 112C	RELAY	8
698	CS	RL12DD	Booster Time Delay Relay 112D	RELAY	8

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
699	CS	RL14AD	Relay 114A	RELAY	8
700	CS	RL14BD	Relay 114B	RELAY	8
701	CS	RL14CD	Relay 114C	RELAY	8
702	CS	RL14DD	Relay 114D	RELAY	8
703	DP	RL15AD	Relay 115A	RELAY	8
704	DP	RL15BD	Relay 115B	RELAY	8
705	DP	RL15CD	Relay 115C	RELAY	8
706	DP	RL15DD	Relay 115D	RELAY	8
707	DP	RL5AXD	Relay 115AX	RELAY	8
708	DP	RL5BXD	Relay 115BX	RELAY	8
709	DP	RL5CXD	Relay 115CX	RELAY	8
710	DP	RL5DXD	Relay 115DX	RELAY	8
711	ME	RL117D	Main Steam Low Pressure Relay 1K117	RELAY	8
712	ME	RL118D	Main Steam Low Pressure Relay 1K118	RELAY	8
713	ME	RL173D	MSIV Actuation Relay 1K73	RELAY	8
714	ME	RL174D	MSIV Actuation Relay 1K74	RELAY	8
715	ME	RL217D	Main Steam Low Pressure Relay 2K117	RELAY	8
716	ME	RL218D	Main Steam Low Pressure Relay 2K118	RELAY	8
717	ME	RL273D	MSIV Actuation Relay 2K73	RELAY	8
718	ME	RL274D	MSIV Actuation Relay 2K74	RELAY	8
719	MI	RL03AD	Relay 6K3A	RELAY	8
720	MI	RL03BD	Relay 6K3B	RELAY	8
721	MI	RL04AD	Relay 6K4A	RELAY	8
722	MI	RL04BD	Relay 6K4B	RELAY	8
723	MI	RL05AD	Relay 6K5A	RELAY	8
724	MI	RL05BD	Relay 6K5B	RELAY	8
725	MI	RL06AD	Relay 6K6A	RELAY	8
726	MI	RL06BD	Relay 6K6B	RELAY	8
727	MI	RL07AD	Relay 6K7A	RELAY	8
728	MI	RL07BD	Relay 6K7B	RELAY	8
729	MI	RL08AD	Relay 6K8B	RELAY	8
730	MI	RL08BD	Relay 6K8B	RELAY	8
731	MS	RL173D	MSIV Actuation Relay 1K73	RELAY	8
732	MS	RL174D	MSIV Actuation Relay 1K74	RELAY	8
733	MS	RL273D	MSIV Actuation Relay 2K73	RELAY	8
734	MS	RL274D	MSIV Actuation Relay 2K74	RELAY	8
735	OV	RL102D	Control relay 6K102	RELAY	8
736	OV	RL103D	Control relay 6K103	RELAY	8
737	OV	RL108D	Control relay 6K108	RELAY	8
738	OV	RL109D	Control relay 6K109	RELAY	8
739	OV	RL659D	Mode switch relay 6K59	RELAY	8
740	OV	RL660D	Torus vent permissive relay 6K60	RELAY	8
741	OV	RL661D	Drywell vent permissive relay 6K61	RELAY	8
742	RI	RLK37D	Actuation Relay	RELAY	8
743	RL	RL10AD	Relay 16K110A	RELAY	8
744	RL	RL10BD	Relay 16K110B	RELAY	8

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
745	RL	RL10CD	Relay 16K110C	RELAY	8
746	RL	RL10DD	Relay 16K110D	RELAY	8
747	RP	RL0AAD	Relay 6K10AA	RELAY	8
748	RP	RL1AAD	Relay 6K11AA	RELAY	8
749	RP	RL1OAD	Relay 6K11A	RELAY	8
750	RP	RL1OAD	Relay 6K10A	RELAY	8
751	RP	RL1ZAD	Relay 6K12A	RELAY	8
752	RP	RL2AAD	Relay 6K12AA	RELAY	8
753	RP	RL9AAD	Relay 6K9AA	RELAY	8
754	RP	RLK9AD	Relay 6K9A	RELAY	8
755	RP	TC0AID	Trip Coil A-1	RELAY	8
756	RP	TC0BID	Trip Coil B-I	RELAY	8
757	RP	TC0CID	Trip Coil C-I	RELAY	8
758	RP	TC0DFD	Trip Coil D-I	RELAY	8
759	RP	TC0EID	Trip Coil E-I	RELAY	8
760	RP	TCAIID	Trip Coil A-II	RELAY	8
761	RP	TCBIID	Trip Coil B-II	RELAY	8
762	RP	TCCFFD	Trip Coil C-II	RELAY	8
763	RP	TCDIID	Trip Coil D-II	RELAY	8
764	RP	TCEIID	Trip Coil E-II	RELAY	8
765	RS	RL11AD	Reactor Trip Relay 1K21A	RELAY	8
766	RS	RL151D	Reactor Trip Relay 1K51	RELAY	8
767	RS	RL152D	Reactor Trip Relay 1K52	RELAY	8
768	RS	RL1ZAD	Reactor Trip Relay 2K21A	RELAY	8
769	RS	RL251D	Reactor Trip Relay 2K51	RELAY	8
770	RS	RL252D	Reactor Trip Relay 2K52	RELAY	8
771	VO	RL16AD	DC Transfer Relay K216A	RELAY	8
772	VO	RL16BD	DC Transfer Relay K216B	RELAY	8
773	VO	RL16CD	DC Transfer Relay K216C	RELAY	8
774	VO	RL16DD	DC Transfer Relay K216D	RELAY	8
775	VO	RL16ED	DC Transfer Relay K216E	RELAY	8
776	CF	CBCSAO	Pump 51A(1-1) Supply Breaker	CKT BKR	10
777	CF	CBCSCO	Pump 51C(1-3) Supply Breaker	CKT BKR	10
778	DB	CB001T	DC Circuit Breaker to DC Bus B	CKT BKR	10
779	DB	CB002T	DC Circuit Breaker to MCC DC-1	CKT BKR	10
780	DB	CB003T	DC Circuit Breaker to AC Switchgear	CKT BKR	10
781	DB	CB004T	DC Circuit Breaker to Panel DC-D	CKT BKR	10
782	DB	CB005T	DC Ckt Bkr to Inst. Bus 3 Rotary Inverter	CKT BKR	10
783	DC	CB010T	DC Circuit Breaker to DC Bus C	CKT BKR	10
784	DC	CB011T	DC Circuit Breaker to MCC DC-2	CKT BKR	10
785	DC	CB012T	DC Circuit Breaker to AC Switchgear	CKT BKR	10
786	DC	CB013T	DC Circuit Breaker to Panel DC-F	CKT BKR	10
787	DC	CB014T	DC Circuit Breaker to Computer Supply	CKT BKR	10
788	EA,EB	CB01AO	Breaker 1A	CKT BKR	10
789	EA,EB	CB01BO	Breaker 1B	CKT BKR	10
790	EA,EB	CBS1AC	Breaker S1A	CKT BKR	10

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
791	EA,EB	CBS1AT	Breaker S1A	CKT BKR	10
792	EA,EB	CBS1BC	Breaker S1B	CKT BKR	10
793	EA,EB	CBS1BT	Breaker S1B	CKT BKR	10
794	EC	CB01CO	Breaker 1C	CKT BKR	10
795	EC	CB1A1D	Breaker 1A1M	CKT BKR	10
796	EC	CB1A1T	Breaker 1A1M	CKT BKR	10
797	EC	CB1A2T	Breaker 1A2M	CKT BKR	10
798	EC	CBA2PT	Breaker 1A2P to Bus 1A2	CKT BKR	10
799	EC	CBDG1C	Breaker EDG1	CKT BKR	10
800	ED	CB01DO	Breaker 1D from 4,160VAC Bus 1B	CKT BKR	10
801	ED	CBB1MT	Breaker 1B2M from Transformer to Bus 1B1	CKT BKR	10
802	ED	CBB1MT	Breaker 1B1M from Transformer to Bus 1B1	CKT BKR	10
803	ED	CBB2MT	Breaker 1B2M from Transformer to Bus 1B2	CKT BKR	10
804	ED	CBB2PT	Breaker 1B2P to Bus 1B2	CKT BKR	10
805	ED	CBDG2C	EDG Output Breaker EDG2	CKT BKR	10
806	EE	CB01CO	Breaker 1C	CKT BKR	10
807	EE	CB01DO	Breaker 1D from 4,160VAC Bus 1B	CKT BKR	10
808	EE	CBDG1C	Breaker EDG1	CKT BKR	10
809	EE	CBDG2C	EDG Output Breaker EDG2	CKT BKR	10
810	FP	CB1E1T	480V AC Feeder Breaker to Bus 1E1	CKT BKR	10
811	FP	CBE17T	480V AC Feeder Breaker to 1E17	CKT BKR	10
812	RF	CBFWAO	FWP A supply breaker	CKT BKR	10
813	RF	CBFWBO	FWP B supply breaker	CKT BKR	10
814	RF	CBFWCO	FWP C supply breaker	CKT BKR	10
815	RP	BKMGAO	M/G Set Supply Breaker A	CKT BKR	10
816	RP	BKMGBO	M/G Set Supply Breaker B	CKT BKR	10
817	RP	BKMGCO	M/G Set Supply Breaker C	CKT BKR	10
818	RP	BKMGDO	M/G Set Supply Breaker D	CKT BKR	10
819	RP	BKMGEO	M/G Set Supply Breaker E	CKT BKR	10
820	VO	CB069T	DC Circuit Breaker CB62-69	CKT BKR	10
821	VO	CB079T	DC Circuit Breaker CB62-79	CKT BKR	10
822	VO	CB162T	DC Circuit Breaker CB62-162	CKT BKR	10
823	VO	CB163T	DC Circuit Breaker CB62-163	CKT BKR	10
824	XB	CB006T	AC Circuit Breaker from VMCC 1B2	CKT BKR	10
825	XB	CB007T	DC Circuit Breaker to DC Bus B	CKT BKR	10
826	XB	CB008T	DC Circuit Breaker to DC Bus B	CKT BKR	10
827	XB	CB009T	AC Circuit Breaker from 1B2	CKT BKR	10
828	XC	CB015T	AC Circuit Breaker from VMCC 1A2	CKT BKR	10
829	XC	CB016T	DC Circuit Breaker to DC Bus C	CKT BKR	10
830	XC	CB017T	DC Circuit Breaker to DC Bus C	CKT BKR	10
831	XC	CB018T	AC Circuit Breaker from 1A2	CKT BKR	10
832	BI	PM91AS	P-19-001A SLC Pump A	DUPL	11
833	BI	PM91BS	P-19-001B SLC Pump B	DUPL	11
834	BI	MV161T	V-16-1 RWCU Inboard Isolation	DUPL	11
835	BV	BV30AD	Turbine Bypass Valves V-1-130	DUPL	11
836	BV	BV30AD	Turbine Bypass Valve V-1-130A	DUPL	11

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
837	BV	BV30BD	Turbine Bypass Valve V-1-130B	DUPL	11
838	BV	BV30CD	Turbine Bypass Valve V-1-130C	DUPL	11
839	BV	BV30DD	Turbine Bypass Valve V-1-130D	DUPL	11
840	BV	BV30ED	Turbine Bypass Valve V-1-130E	DUPL	11
841	BV	BV30FD	Turbine Bypass Valve V-1-130F	DUPL	11
842	BV	BV30GD	Turbine Bypass Valve V-1-130G	DUPL	11
843	BV	BV30HD	Turbine Bypass Valve V-1-130H	DUPL	11
844	BV	BV30ID	Turbine Bypass Valve V-1-130I	DUPL	11
845	CC	HX1AB	Heat Exchanger H-21-001B	DUPL	11
846	CC	HX1AB	Heat Exchanger H-21-001B	DUPL	11
847	CC	HXCCAP	Heat Exchanger H-21-001A	DUPL	11
848	CC	HXCCAS	Heat Exchanger H-21-001A	DUPL	11
849	CC	HXCCCP	Heat Exchanger H-21-001C	DUPL	11
850	CC	HXCCCS	Heat Exchanger H-21-001C	DUPL	11
851	CC	HXCCDP	Heat Exchanger H-21-001D	DUPL	11
852	CC	HXCCDS	Heat Exchanger H-21-001D	DUPL	11
853	CC	MV011D	Drywell Header Inlet V-21-11	DUPL	11
854	CC	PM51A	Pump 51A(1-1)	DUPL	11
855	CC	PM51B	Pump 51B(1-2)	DUPL	11
856	CC	PM51C	Pump 51C(1-3)	DUPL	11
857	CC	PM51D	Pump 51D(1-4)	DUPL	11
858	CC	PM52A	ESW Pump 52A (P-3-003A)	DUPL	11
859	CC	PM52B	ESW Pump 52B (P-3-003B)	DUPL	11
860	CC	PM52C	ESW Pump 52C (P-3-003C)	DUPL	11
861	CC	PM52D	ESW Pump 52D (P-3-003D)	DUPL	11
862	CC	MV005T	Drywell Header Inlet V-21-5	DUPL	11
863	CC	MV013T	Dynamic Test (Torus) Return	DUPL	11
864	CD	PM08AS	NC08A CRD Pump	DUPL	11
865	CD	PM08BR	NC08B CRD Pump	DUPL	11
866	CF	MV211D	Drywell Header Inlet V-21-11	DUPL	11
867	CF	MV211D	Drywell Header Inlet V-21-11	DUPL	11
868	C[illegible]	HXACAP	SJAE after condenser A	DUPL	11
869	CP	HXACBP	SJAE after condenser B	DUPL	11
870	CP	HXACCP	SJAE after condenser C	DUPL	11
871	CS	PM01AS	Core Spray Pump NZ01A	DUPL	11
872	CS	PM01BS	Core Spray Pump NZ01B	DUPL	11
873	CS	PM01CS	Pump NZ01C	DUPL	11
874	CS	PM01DS	Pump NZ01D	DUPL	11
875	CS	PM03AS	Booster Pump NZ03A	DUPL	11
876	CS	PM03BS	Booster Pump NZ03B	DUPL	11
877	CS	PM03CS	Booster Pump NZ03C	DUPL	11
878	CS	PM03DS	Booster Pump NZ03D	DUPL	11
879	CS	MV015T	Parallel Isol. MOV V-20-15	DUPL	11
880	CS	MV021T	Parallel Isol. MOV V-20-21	DUPL	11
881	CS	MV040T	Parallel Isol. MOV V-20-40	DUPL	11
882	CS	MV041T	Parallel Isol. MOV V-20-41	DUPL	11

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
883	CT	PM002S	Condensate Transfer Pump 1-2	DUPL	11
884	DB	BTDCBR	125V DC Battery B	DUPL	11
885	DC	BTDCCR	125V DC Battery C	DUPL	11
886	EC	DG001S	Emergency Diesel Generator #1	DUPL	11
887	EC	FN008S	Alternate Exhaust Fan FN-56-008	DUPL	11
888	EC	PM01AS	Fuel Oil Transfer Pump 1A	DUPL	11
889	EC	PM01BS	Fuel Oil Transfer Pump 1B	DUPL	11
890	EC	XXEAFF	4160V AC Bus 1A Failed	DUPL	11
891	ED	DG002S	Emergency Diesel Generator #2	DUPL	11
892	ED	PM02AS	Fuel Oil Transfer Pump 2A	DUPL	11
893	ED	PM02BS	Fuel Oil Transfer Pump 2B	DUPL	11
894	ED	XXEBFF	4160V AC Bus 1B Failed	DUPL	11
895	EE	BTDB1D	Diesel Generator Battery #1	DUPL	11
896	EE	BTDG2D	Diesel Generator Battery #2	DUPL	11
897	EE	DG001R	Emergency Diesel Generator #1	DUPL	11
898	EE	DG001S	Emergency Diesel Generator #1	DUPL	11
899	EE	DG002R	Emergency Diesel Generator #2	DUPL	11
900	EE	DG002S	Emergency Diesel Generator #2	DUPL	11
901	EE	XXEAFF	4160V AC Bus 1A Failed	DUPL	11
902	EE	XXEBFF	4160V AC Bus 1B Failed	DUPL	11
903	FP	PD001S	Fire Pump 1	DUPL	11
904	FP	PD002S	Fire Pump 2	DUPL	11
905	FP	PM003S	Redundant Fire Pump	DUPL	11
906	FP	XXLOWS	Diesel Fire Water Pump Supply	DUPL	11
907	FW	AV732T	Feed regulating valve AOV-2-732	DUPL	11
908	FW	AV733T	Feed regulating valve AOV-2-733	DUPL	11
909	FW	AV734T	Feed regulating valve AOV-2-734	DUPL	11
910	FW	HX0B1P	LP heater B-1	DUPL	11
911	FW	HX0B2P	LP heater B-2	DUPL	11
912	FW	HX0C1P	LP heater C-1	DUPL	11
913	FW	HX0C2P	LP heater C-2	DUPL	11
914	FW	HXDCAP	Drain cooler A	DUPL	11
915	FW	HXDCBP	Drain cooler B	DUPL	11
916	FW	HXDCCP	Drain cooler C	DUPL	11
917	FW	HXHPBP	HP heater B	DUPL	11
918	FW	HXHPCP	HP heater C	DUPL	11
919	IA	CP011S	Air Compressor 1-1	DUPL	11
920	IA	CP012S	Air Compressor 1-2	DUPL	11
921	IA	SV054T	TBCCW Inlet Solenoid Valve V-5-54	DUPL	11
922	IA	SV059T	TBCCW Inlet Solenoid Valve V-5-59	DUPL	11
923	ME	NV03AT	Inboard MSIV NS3A	DUPL	11
924	ME	NV03BT	Inboard MSIV NS3B	DUPL	11
925	ME	NV04AT	Outboard MSIV NS4A	DUPL	11
926	ME	NV04BT	Outboard MSIV NS4B	DUPL	11
927	MI	MV034D	Condensate Return Iso. Valve MOV-14-34	DUPL	11
928	MI	MV035D	Condensate Isolation Valve MOV-14-35	DUPL	11

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
929	MI	MV032T	Steam Isolation Valve MOV-14-32	DUPL	11
930	MI	MV033T	Steam Isolation Valve MOV-14-33	DUPL	11
931	MI	MV034T	Condensate Return Iso. Valve MOV-14-34	DUPL	11
932	MI	MV035T	Condensate Isolation Valve MOV-14-35	DUPL	11
933	MI	MV036T	Condensate Return Iso. Valve MOV-14-36	DUPL	11
934	MI	MV037T	Condensate Isolation Valve MOV-14-37	DUPL	11
935	MS	NV03AD	Inboard MSIV NS3A	DUPL	11
936	MS	NV03AT	Inboard MSIV NS3A	DUPL	11
937	MS	NV03BD	Inboard MSIV NS3B	DUPL	11
938	MS	NV03BT	Inboard MSIV NS3B	DUPL	11
939	MS	NV04AD	Outboard MSIV NS4A	DUPL	11
940	MS	NV04AT	Outboard MSIV NS4A	DUPL	11
941	MS	NV04BD	Outboard MSIV NS4B	DUPL	11
942	MS	NV04BT	Outboard MSIV NS4B	DUPL	11
943	MU	AV034T	Air-Operated Valve V-11-34	DUPL	11
944	MU	AV036T	Air-Operated Valve V-11-36	DUPL	11
945	OV	SV595D	Solenoid valve V-6-595	DUPL	11
946	PI	AV202T	Air-Operated Valve V-22-2	DUPL	11
947	PI	AV313T	Air-Operated Valve V-23-13	DUPL	11
948	PI	AV314T	Air-Operated Valve V-23-14	DUPL	11
949	PI	AV315T	Air-Operated Valve V-23-15	DUPL	11
950	PI	AV316T	Air-Operated Valve V-23-16	DUPL	11
951	PI	AV317T	Air-Operated Valve V-23-17	DUPL	11
952	PI	AV318T	Air-Operated Valve V-23-18	DUPL	11
953	PI	AV319T	Air-Operated Valve V-23-19	DUPL	11
954	PI	AV320T	Air-Operated Valve V-23-20	DUPL	11
955	PI	AV321T	Air-Operated Valve V-23-21	DUPL	11
956	PI	AV322T	Air-Operated Valve V-23-22	DUPL	11
957	PI	AV701T	Air-Operated Valve V-22-1	DUPL	11
958	PI	AV778T	Air-Operated Valve V-22-28	DUPL	11
959	PI	AV779T	Air-Operated Valve V-22-29	DUPL	11
960	PI	AV817T	Air-Operated Valve V-28-17	DUPL	11
961	PI	AV818T	Air-Operated Valve V-28-18	DUPL	11
962	PI	SV809T	Solenoid Valve V-38-9	DUPL	11
963	PI	SV810T	Solenoid Valve V-38-10	DUPL	11
964	PI	SV816T	Solenoid Valve V-38-16	DUPL	11
965	PI	SV817T	Solenoid Valve V-38-17	DUPL	11
966	PI	SV837T	Solenoid Valve V-38-37	DUPL	11
967	PI	SV839T	Solenoid Valve V-38-39	DUPL	11
968	PI	SV840T	Solenoid Valve V-38-40	DUPL	11
969	PI	SV841T	Solenoid Valve V-38-41	DUPL	11
970	PI	SV843T	Solenoid Valve V-38-43	DUPL	11
971	PI	SV8445	Solenoid Valve V-38-44	DUPL	11
972	PI	SV846T	Solenoid Valve V-38-46	DUPL	11
973	RF	AV732C	Feed regulating valve AOV-2-732 (A)	DUPL	11
974	RF	AV733C	Feed regulating valve AOV-2-733 (B)	DUPL	11

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
975	RF	AV734C	Feed regulating valve AOV-2-734 (C)	DUPL	11
976	RI	AV003T	RB General Area Supply Valve V-28-3	DUPL	11
977	RI	AV004T	RB General Area Supply Valve V-28-4	DUPL	11
978	RI	AV005T	RB General Area Supply Valve V-28-5	DUPL	11
979	RI	AV006T	RB General Area Supply Valve V-28-6	DUPL	11
980	RI	AV007T	RB General Area Supply Valve V-28-7	DUPL	11
981	RI	AV008T	RB General Area Supply Valve V-28-8	DUPL	11
982	RI	AV009T	RB General Area Supply Valve V-28-9	DUPL	11
983	RI	AV010T	RB General Area Supply Valve V-28-10	DUPL	11
984	RI	AV011T	CRD Rebuild Area&Gen Area Sup V-28-11	DUPL	11
985	RI	AV012T	CRD Rebuild Area&Gen Area Sup V-28-12	DUPL	11
986	RI	AV013T	RB General Area Supply Valve V-28-13	DUPL	11
987	RI	AV014T	RB General Area Supply Valve V-28-14	DUPL	11
988	RI	AV015T	RB General Area Supply Valve V-28-15	DUPL	11
989	RI	AV016T	RB General Area Supply Valve V-28-16	DUPL	11
990	RI	AV021T	RB General Area Exhaust Valve V-28-21	DUPL	11
991	RI	AV022T	RB General Area Exhaust Valve V-28-22	DUPL	11
992	RI	AV036T	RB General Area Supply Valve V-28-36	DUPL	11
993	RI	AV037T	RB General Area Supply Valve V-28-37	DUPL	11
994	SG	AV823T	Train Inlet Valve V-28-23	DUPL	11
995	SG	AV826T	Train Outlet Valve V-28-26	DUPL	11
996	SG	AV827T	Train Inlet Valve V-28-27	DUPL	11
997	SG	AV830T	Train Outlet Valve V-28-30	DUPL	11
998	SG	FNTNAS	Train A Fan	DUPL	11
999	SG	FNTNBS	Train B Fan	DUPL	11
1000	SR	SV28DR	Safety Valve (NR-28D) V-1-160	DUPL	11
1001	SR	SV28ER	Safety Valve (NR-28E) V-1-161	DUPL	11
1002	SR	SV28FR	Safety Valve (NR-28F) V-1-162	DUPL	11
1003	SR	SV28GR	Safety Valve (NR-28G) V-1-163	DUPL	11
1004	SR	SV28HR	Safety Valve (NR-28H) V-1-164	DUPL	11
1005	SR	SV28JR	Safety Valve (NR-28J) V-1-165	DUPL	11
1006	SR	SV28KR	Safety Valve (NR-28K) V-1-166	DUPL	11
1007	SR	SV28LR	Safety Valve (NR-28L) V-1-167	DUPL	11
1008	SR	SV28MR	Safety Valve (NR-28M) V-1-168	DUPL	11
1009	SW	PM01AS	Pump P-3-001A	DUPL	11
1010	SW	PM01BS	Pump P-3-001B	DUPL	11
1011	TB	HX003P	Heat Exchanger(C-5-003)	DUPL	11
1012	TB	HX004P	Heat Exchanger C-5-004	DUPL	11
1013	TB	PM003S	Pump P-5-003	DUPL	11
1014	TB	PM003S	Pump P-5-004	DUPL	11
1015	TB	PM003S	Pump P-5-005	DUPL	11
1016	VO	EMRVAD	EMRV (NR108A) V-1-173	DUPL	11
1017	VO	EMRVBD	EMRV (NR108B) V-1-174	DUPL	11
1018	VO	EMRVCD	EMRV (NR108C) V-1-175	DUPL	11
1019	VO	EMRVDD	EMRV (NR108D) V-1-176	DUPL	11
1020	VO	EMRVED	EMRV (NR108E) V-1-177	DUPL	11

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
1021	VR	EMRVAR	EMRV (NR108A) V-1-173	DUPL	11
1022	VR	EMRVBR	EMRV (NR108B) V-1-174	DUPL	11
1023	VR	EMRVCR	EMRV (NR108C) V-1-175	DUPL	11
1024	VR	EMRVDR	EMRV (NR108D) V-1-176	DUPL	11
1025	VR	EMRVER	EMRV (NR108E) V-1-177	DUPL	11
1026	IA	RV301T	Air Compressor 1-1 Relief Valve V-6S-301	AIR COMP 1-1	12
1027	IA	RV081T	Air Compressor 1-2 Relief Valve V-6S-81	AIR COMP 1-2	12
1028	IA	RV298T	Air Compressor 1-3 Relief Valve V-6S-298	AIR COMP 1-3	12
1029	IA	RV008T	Air Receiver 1-1 Relief Valve V-6S-8	AIR REC 1-1	12
1030	IA	RV317T	Air Receiver 1-1 Relief Valve V-6S-317	AIR REC 1-1	12
1031	IA	RV009T	Air Receiver 1-2 Relief Valve V-6S-9	AIR REC 1-2	12
1032	IA	RV318T	Air Receiver 1-2 Relief Valve V-6S-318	AIR REC 1-2	12
1033	IA	RV319T	Air Receiver 1-3 Relief Valve V-6S-319	AIR REC 1-3	12
1034	CD	RV045T	V-15-45 Relief Valve	CRD PUMP A	12
1035	CD	RV046T	V-15-46 Relief Valve	CRD PUMP B	12
1036	CS	RV024P	Relief Valve V-20-24	CS PUMP C	12
1037	CS	RV025P	Relief Valve V-20-25	CS PUMP D	12
1038	FP	RV957T	Relief Valve V-9-57	FP-1	12
1039	FP	RV967T	Relief Valve V-9-67	FP-2	12
1040	BI	RV942T	V-19-42 Pump A Discharge Relief Valve	SLC PUMP A	12
1039	BI	RV943T	V-19-43 Pump B Discharge Relief Valve	SLC PUMP B	12
1040	EC	BTDB1D	Diesel Generator Battery #1	DIESEL	12
1041	ED	BTDG2D	Diesel Generator Battery #2	DIESEL	12
1042	AD	ZHEAD3	Operator Actuates ADS	OP ACT	13
1043	AD	ZHEAD3	Operator Actuates ADS	OP ACT	13
1044	AD	ZHEAD4	Operator Actuates ADS	OP ACT	13
1045	BI	ZHEBI1	Operator actuates liquid poison	OP ACT	13
1046	BI	ZHEBI2	Operator actuates liquid poison	OP ACT	13
1047	BI	ZHEBI3	Operator actuates liquid poison	OP ACT	13
1048	BI	ZHEBI4	Operator actuates liquid poison	OP ACT	13
1049	CC	ZHECC3	Operator Starts Cool Torus Cooling	OP ACT	13
1050	CC	ZHECC4	Operator Starts Pump 51B(1-2)	OP ACT	13
1051	CC	ZHECC4	Operator Starts Pump 51D	OP ACT	13
1052	CC	ZHECC4	Operator Starts Pump 52D	OP ACT	13
1053	CC	ZHECC4	Operator Starts Pump 52B	OP ACT	13
1054	CC	ZHECC5	Alignment to spray containment	OP ACT	13
1055	CC	ZHECC6	Actuate Spray during ATWS	OP ACT	13
1056	CD	ZHECD1	Operator Starts Injection	OP ACT	13
1057	CD	ZHECD4	Operator Starts Injection	OP ACT	13
1058	CF	ZHECF1	Operator closes drywell supply	OP ACT	13
1059	CF	ZHECF1	Operator closes drywell supply	OP ACT	13
1060	CS	ZHECS4	Operator manually starts core spray	OP ACT	13
1061	CS	ZHECS5	Operator aligns fire protection to CS	OP ACT	13
1062	FP	OP001F	Operator Action	OP ACT	13
1063	IA	OPIA3F	Operator Recovers from LOSP	OP ACT	13
1064	IA	OPIA4F	Operator Recovers from Loss of TBCCW	OP ACT	13

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
1065	IA	OPIA4F	Operator Action	OP ACT	13
1066	IC	ZHEIC4	Op. actuates IC after logic failure	OP ACT	13
1067	ME	ZHEME2	Operator Closes MSIVs After RL Failure	OP ACT	13
1068	MI	ZHEMI2	Operator Prevents IC Actuation	OP ACT	13
1069	MS	ZHEME2	Operator Closes MSIVs After RL Failure	OP ACT	13
1070	MU	ZHEMU1	Operator aligns system	OP ACT	13
1071	MU	ZHEMU2	Operator aligns system	OP ACT	13
1072	OV	ZHEOV1	Operator vents - no core damage	OP ACT	13
1073	OV	ZHEOV2	Operator vents with core damage	OP ACT	13
1074	PI	ZHEP12	Operator isolates containment after logic fail	OP ACT	13
1075	RF	ZHE0F1	Operator recovers level control	OP ACT	13
1076	RF	ZHERF1	Operator controls level	OP ACT	13
1077	RI	ZHERI3	Operator Actuates RB Iso. on Relay Failure	OP ACT	13
1078	RP	ZHERP2	Oper trips recirc pumps & actuates BI	OP ACT	13
1079	RS	ZHERS3	Manual Scram intitiated after TT failure	OP ACT	13
1080	RS	ZHERS4	Manual Scram intitiated after logic failure	OP ACT	13
1081	SW	ZHESW1	Manual pump start by operator	OP ACT	13
1082	TB	ZHETB4	SW Aligned to Hx on loss of Circ. Water	OP ACT	13
1083	TB	ZHETB5	Pumps Restart. If Loss of Offsite Power	OP ACT	13
1084	TT	ZHETT3	Operator Fails To Trip Turbine	OP ACT	13
1085	XB	OP001F	Operator aligns standby charger	OP ACT	13
1086	XC	OP001F	Operator aligns standby charger	OP ACT	13
1	IA	CP011R	Air Compressor 1-1	COMPRESS	A
2	IA	CP012R	Air Compressor 1-2	COMPRESS	A
3	EC	DG001R	Emergency Diesel Generator #1	DIESEL	A
4	ED	DG002R	Emergency Diesel Generator #2	DIESEL	A
5	EC	FN004R	Supply Fan FN-56-004	FAN	A
6	EC	FN007R	Exhaust Fan FN-56-007	FAN	A
7	EC	FN008R	Alternate Exhaust Fan FN-56-008	FAN	A
8	ED	FNE21R	Exhaust Fan EF 1-21	FAN	A
9	ED	FNS21R	Supply Fan SF 1-21	FAN	A
10	SG	FNTNAR	Train A Fan	FAN	A
11	SG	FNTNBR	Train B Fan	FAN	A
12	XB	FNE20R	A/B Battery Room Exhaust Fan	FAN	A
13	XB	FNS20R	A/B Battery Room Supply Fan	FAN	A
14	BI	PM91AR	P-19-001A SLC Pump A	PUMP	A
15	BI	PM91BR	P-19-001B SLC Pump B	PUMP	A
16	CC	PM51A	Pump 51A(1-1)	PUMP	A
17	CC	PM51B	Pump 51B(1-2)	PUMP	A
18	CC	PM51C	Pump 51C(1-3)	PUMP	A
19	CC	PM51D	Pump 51D(1-4)	PUMP	A
20	CC	PM52A	ESW Pump 52A (P-3-003A)	PUMP	A
21	CC	PM52B	ESW Pump 52B (P-3-003B)	PUMP	A
22	CC	PM52C	ESW Pump 52C (P-3-003C)	PUMP	A
23	CC	PM52D	ESW Pump 52D (P-3-003D)	PUMP	A
24	CD	PM08AR	NC08A CRD Pump	PUMP	A

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
25	CD	PM08BS	NC08B CRD Pump	PUMP	A
26	CP	PMCPAR	Condensate pump 1-A	PUMP	A
27	CP	PMCPBR	Condensate pump 1-B	PUMP	A
28	CP	PMCPCR	Condensate pump 1-C	PUMP	A
29	CS	PM01AR	Core Spray Pump NZ01A	PUMP	A
30	CS	PM01BR	Core Spray Pump NZ01B	PUMP	A
31	CS	PM01CR	Pump NZ01C	PUMP	A
32	CS	PM01DR	Pump NZ01D	PUMP	A
33	CS	PM03AR	Booster Pump NZ03A	PUMP	A
34	CS	PM03BR	Booster Pump NZ03B	PUMP	A
35	CS	PM03CR	Booster Pump NZ03C	PUMP	A
36	CS	PM03DR	Booster Pump NZ03D	PUMP	A
37	CT	PM001R	Condensate Transfer Pump 1-1	PUMP	A
38	CT	PM002R	Condensate Transfer Pump 1-2	PUMP	A
39	CW	PMO2AR	Circulating Water Pump P-3-002A	PUMP	A
40	CW	PMO2BR	Circulating Water Pump P-3-002B	PUMP	A
41	CW	PMO2CR	Circulating Water Pump P-3-002C	PUMP	A
42	CW	PMO2CR	Circulating Water Pump P-3-002D	PUMP	A
43	EC	PM01AR	Fuel Oil Transfer Pump 1A	PUMP	A
44	EC	PM01BR	Fuel Oil Transfer Pump 1B	PUMP	A
45	ED	PM02AR	Fuel Oil Transfer Pump 2A	PUMP	A
46	ED	PM02BR	Fuel Oil Transfer Pump 2B	PUMP	A
47	FP	PD001R	Fire Pump 1	PUMP	A
48	FP	PD002R	Fire Pump 2	PUMP	A
49	FP	PM003R	Redundant Fire Pump	PUMP	A
50	FW	PMFWAR	Feedwater pump A	PUMP	A
51	FW	PMFWBR	Feedwater pump B	PUMP	A
52	FW	PMFWCR	Feedwater pump C	PUMP	A
53	SW	PM01AR	Pump P-3-001A	PUMP	A
54	SW	PM01BR	Pump P-3-001B	PUMP	A
55	TB	PM003R	Pump P-5-003	PUMP	A
56	TB	PM003R	Pump P-5-004	PUMP	A
57	TB	PM003R	Pump P-5-005	PUMP	A
58	AD	PV18AD	ADS Valve NR108A	EMRV	B
59	AD	PV18BD	ADS Valve NR108B	EMRV	B
60	AD	PV18CD	ADS Valve NR108C	EMRV	B
61	AD	PV18DD	ADS Valve NR108D	EMRV	B
62	AD	PV18ED	ADS Valve NR108E	EMRV	B
63	CC	HX1AB	Heat Exchanger H-21-001B	HEAT EX	B
64	CC	HXCCAB	Heat Exchanger H-21-001A	HEAT EX	B
65	CC	HXCCCR	Heat Exchanger H-21-001C	HEAT EX	B
66	CC	HXCCDR	Heat Exchanger H-21-001D	HEAT EX	B
67	TB	HX003B	Heat Exchanger(C-5-003)	HEAT EX	B
68	TB	HX004P	Heat Exchanger C-5-004	HEAT EX	B
69	FP	XXLOWS	Diesel Pump Water Supply	POND	B
70	SO	SV28DD	Safety Valve (NR-28D) V-1-160	SAFETY	B

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
71	SO	SV28ED	Safety Valve (NR-28E) V-1-161	SAFETY	B
72	SO	SV28FD	Safety Valve (NR-28F) V-1-162	SAFETY	B
73	SO	SV28GD	Safety Valve (NR-28G) V-1-163	SAFETY	B
74	SO	SV28HD	Safety Valve (NR-28H) V-1-164	SAFETY	B
75	SO	SV28JD	Safety Valve (NR-28J) V-1-165	SAFETY	B
76	SO	SV28KD	Safety Valve (NR-28K) V-1-166	SAFETY	B
77	SO	SV28LD	Safety Valve (NR-28L) V-1-167	SAFETY	B
78	SO	SV28MD	Safety Valve (NR-28M) V-1-168	SAFETY	B
79	BI	TKSLCL	T-19-001 Liquid Poison Tank	TANK	B
80	EC	TK001A	Fuel Oil Storage Tank	TANK	B
81	ED	TK001B	Fuel Oil Storage Tank	TANK	B
82	FP	TK001B	Redundant Fire Water Pump Supply	TANK	B
83	IA	TK001B	Air Receiver 1-1	TANK	B
84	IA	TK002B	Air Receiver 1-2	TANK	B
85	IA	TK003B	Air Receiver 1-3	TANK	B
86	IC	IC00AP	Isolation Condenser "A"	TANK	B
87	IC	IC00BP	Isolation Condenser "B"	TANK	B
88	ST	TK001B	CST T-11-001	TANK	B
89	IA	ADOABP	Air Dryer A-B	AIR DRIER	C
90	CP	AV216T	Control valve V-2-16	AOV	C
91	CP	AV235T	Control valve V-2-235	AOV	C
92	CP	AV732T	Feed regulating valve AOV-2-732	AOV	C
93	CP	AV733T	Feed regulating valve AOV-2-733	AOV	C
94	CP	AV734T	Feed regulating valve AOV-2-734	AOV	C
95	CS	AV092D	Min.Flow Valve V-20-92	AOV	C
96	CS	AV092T	Min.Flow Valve V-20-92	AOV	C
97	CS	AV093D	Min. Flow Valve V-20-93	AOV	C
98	CS	AV093T	Min. Flow Valve V-20-93	AOV	C
99	CS	AV094D	Min. Flow Valve V-20-94	AOV	C
100	CS	AV094T	Min. Flow Valve V-20-94	AOV	C
101	MU	AV034D	Air-Operated Valve V-11-34	AOV	C
102	MU	AV036D	Air-Operated Valve V-11-36	AOV	C
103	OV	AV271D	Air operated valve V-27-1	AOV	C
104	OV	AV272D	Air operated valve V-27-2	AOV	C
105	OV	AV817D	Air operated valve V-28-16	AOV	C
106	OV	AV818D	Air operated valve V-28-15	AOV	C
107	PI	AV202D	Air-Operated Valve V-22-2	AOV	C
108	PI	AV313D	Air-Operated Valve V-23-13	AOV	C
109	PI	AV314D	Air-Operated Valve V-23-14	AOV	C
110	PI	AV315D	Air-Operated Valve V-23-15	AOV	C
111	PI	AV316D	Air-Operated Valve V-23-16	AOV	C
112	PI	AV317D	Air-Operated Valve V-23-17	AOV	C
113	PI	AV318D	Air-Operated Valve V-23-18	AOV	C
114	PI	AV319D	Air-Operated Valve V-23-19	AOV	C
115	PI	AV320D	Air-Operated Valve V-23-20	AOV	C
116	PI	AV321D	Air-Operated Valve V-23-21	AOV	C

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
117	PI	AV322D	Air-Operated Valve V-23-22	AOV	C
118	PI	AV701L	Air-Operated Valve V-22-1	AOV	C
119	PI	AV778D	Air-Operated Valve V-22-28	AOV	C
120	PI	AV779D	Air-Operated Valve V-22-29	AOV	C
121	PI	AV817D	Air-Operated Valve V-28-17	AOV	C
122	PI	AV818D	Air-Operated Valve V-28-18	AOV	C
123	RI	AV003D	RB General Area Supply Valve V-28-3	AOV	C
124	RI	AV004D	RB General Area Supply Valve V-28-4	AOV	C
125	RI	AV005D	RB General Area Supply Valve V-28-5	AOV	C
126	RI	AV006D	RB General Area Supply Valve V-28-6	AOV	C
127	RI	AV007D	RB General Area Supply Valve V-28-7	AOV	C
128	RI	AV008D	RB General Area Supply Valve V-28-8	AOV	C
129	RI	AV009D	RB General Area Supply Valve V-28-9	AOV	C
130	RI	AV010D	RB General Area Supply Valve V-28-10	AOV	C
131	RI	AV011D	CRD Rebuild Area&Gen Area Sup V-28-11	AOV	C
132	RI	AV012D	CRD Rebuild Area&Gen Area Sup V-28-12	AOV	C
133	RI	AV013D	RB General Area Supply Valve V-28-13	AOV	C
134	RI	AV014D	RB General Area Supply Valve V-28-14	AOV	C
135	RI	AV015D	RB General Area Supply Valve V-28-15	AOV	C
136	RI	AV016D	RB General Area Supply Valve V-28-16	AOV	C
137	RI	AV021D	RB General Area Exhaust Valve V-28-21	AOV	C
138	RI	AV022D	RB General Area Exhaust Valve V-28-22	AOV	C
139	RI	AV036D	RB General Area Supply Valve V-28-36	AOV	C
140	RI	AV037D	RB General Area Supply Valve V-28-37	AOV	C
141	SG	AV823D	Train Inlet Valve V-28-23	AOV	C
142	SG	AV826D	Train Outlet Valve V-28-26	AOV	C
143	SG	AV827D	Train Inlet Valve V-28-27	AOV	C
144	SG	AV830D	Train Outlet Valve V-28-30	AOV	C
145	ST	AV016T	Air operated valve V-2-16	AOV	C
146	ST	AV235T	Air operated valve V-2-235	AOV	C
147	BI	MV161D	V-16-1 RWCU Inboard Isolation	MOV	C
148	BI	MV164D	V-16-14 RWCU Outboard Isolation	MOV	C
149	CC	MV005D	Drywell Header Inlet V-21-5	MOV	C
150	CC	MV011D	Drywell Header Inlet V-21-11	MOV	C
151	CC	MV013D	Dynamic Test (Torus) Return V-21-13	MOV	C
152	CS	MV015D	Parallel Isol. MOV V-20-15	MOV	C
153	CC	MV017D	Dynamic Test (Torus) Return V-21-17	MOV	C
154	CS	MV021D	Parallel Isol. MOV V-20-21	MOV	C
155	CS	MV040D	Parallel Isol. MOV V-20-40	MOV	C
156	CS	MV041D	Parallel Isol.MOV V-20-41	MOV	C
157	IC/MI	MV034D	Condensate Isolation valve MOV-14-34	MOV	C
158	IC/MI	MV035D	Condensate Isolation valve MOV-14-35	MOV	C
159	MI	MV030D	Steam Supply Isolation Valve MOV-14-30	MOV	C
160	MI	MV030T	Steam Supply Isolation Valve MOV-14-30	MOV	C
161	MI	MV031D	Steam Supply Isolation Valve MOV-14-31	MOV	C
162	MI	MV031T	Steam Supply Isolation Valve MOV-14-31	MOV	C

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
163	MI	MV032D	Steam Isolation Valve MOV-14-32	MOV	C
164	MI	MV033D	Steam Isolation Valve MOV-14-33	MOV	C
165	MI	MV036D	Condensate Return Iso. Valve MOV-14-36	MOV	C
166	MI	MV037D	Condensate Isolation Valve MOV-14-37	MOV	C
167	ME	NV03AD	Inboard MSIV NS3A	MSIV	C
168	ME	NV03BD	Inboard MSIV NS3B	MSIV	C
169	ME	NV04AD	Outboard MSIV NS4A	MSIV	C
170	ME	NV04BD	Outboard MSIV NS4B	MSIV	C
171	IA	SV054D	TBCCW Inlet Solenoid Valve V-5-54	SOV	C
172	IA	SV059D	TBCCW Inlet Solenoid Valve V-5-59	SOV	C
173	OV	SV591D	Solenoid valve V-6-591	SOV	C
174	OV	SV592D	Solenoid valve V-6-592	SOV	C
175	OV	SV596D	Solenoid valve V-6-596	SOV	C
176	PI	SV809D	Solenoid Valve V-38-9	SOV	C
177	PI	SV810D	Solenoid Valve V-38-10	SOV	C
178	PI	SV816D	Solenoid Valve V-38-16	SOV	C
179	PI	SV817D	Solenoid Valve V-38-17	SOV	C
180	PI	SV837D	Solenoid Valve V-38-37	SOV	C
181	PI	SV838D	Solenoid Valve V-38-38	SOV	C
182	PI	SV839D	Solenoid Valve V-38-39	SOV	C
183	PI	SV840D	Solenoid Valve V-38-40	SOV	C
184	PI	SV841D	Solenoid Valve V-38-41	SOV	C
185	PI	SV843D	Solenoid Valve V-38-43	SOV	C
186	PI	SV844D	Solenoid Valve V-38-44	SOV	C
187	PI	SV846D	Solenoid Valve V-38-46	SOV	C
188	RS	VS117X	Scram Pilot Solenoid Valves I (137 SO117s)	SOV	C
189	RS	VS118X	Scram Pilot Solenoid Valves II (137 SO118s)	SOV	C
190	RS	VS450D	Backup Scram Valve I (V-6-450)	SOV	C
191	RS	VS451D	Backup Scram Valve II (V-6-451)	SOV	C
192	RS	VSAB2D	ARI Block Valve #2	SOV	C
193	RS	VSAR1D	ARI Main Header Vent Valve #1	SOV	C
194	RS	VSAB1D	ARI Block Valve #1	SOV	C
195	BI	EV944D	V-19-44 Squib Valve B	SQUIB	C
196	BI	EV945D	V-19-45 Squib Valve A	SQUIB	C
197	BT	BV30AD	Turbine Bypass Valve V-1-130A	TBV	C
198	BT	BV30BD	Turbine Bypass Valve V-1-130B	TBV	C
199	BT	BV30CD	Turbine Bypass Valve V-1-130C	TBV	C
200	BT	BV30DD	Turbine Bypass Valve V-1-130D	TBV	C
201	BT	BV30ED	Turbine Bypass Valve V-1-130E	TBV	C
202	BT	BV30FD	Turbine Bypass Valve V-1-130F	TBV	C
203	BT	BV30GD	Turbine Bypass Valve V-1-130G	TBV	C
204	BT	BV30HD	Turbine Bypass Valve V-1-130H	TBV	C
205	BT	BV30ID	Turbine Bypass Valve V-1-130I	TBV	C
206	DB	BTDCBD	125V DC Battery B	BATTERY	D
207	DC	BTDCCD	125V DC Battery C	BATTERY	D
208	DB	BSDCBR	125V DC Bus B	BUS	D

Table 3-3 OYSTER CREEK PRA COMPONENT LIST (Sorted by Seismic Note)

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
209	DC	BSDCCR	125V DC Bus C	BUS	D
210	EA,EB	BS01AR	4,160V AC Bus 1A	BUS	D
211	EA,EB	BS01BR	4,160V AC Bus 1B	BUS	D
212	EC	BS01CR	4,160V AC Bus 1C	BUS	D
213	EC	BS1A1R	480V AC Bus 1A1	BUS	D
214	EC	BS1A2R	480V AC Bus 1A2	BUS	D
215	ED	BS01DR	4,160V DC Bus 1D	BUS	D
216	ED	BS1B1R	480V AC Bus 1B1	BUS	D
217	ED	BS1B2R	480V AC Bus 1B2	BUS	D
218	FP	BS1E1R	480V AC Bus 1E1	BUS	D
219	XB	BCDABR	Standby Static Charger A/B	CHARGER	D
220	XB	MGDCBR	Battery Charger B Motor Generator Set	CHARGER	D
221	XC	BCDC1R	Battery Charger C1	CHARGER	D
222	XC	BCDC2R	Standby Static Charger C2	CHARGER	D
223	RS	CRDR12	Control Rod Drives (137)	CNTRL ROD	D
224	EA,EB	XR0SAR	Startup Transformer SA	XFORMR	D
225	EA,EB	XR0SBR	Startup Transformer SB	XFORMR	D
226	EC	XR1A1R	4,160V/480V Transformer to Bus 1A1	XFORMR	D
227	EC	XR1A2R	4,160V/480V Transformer to Bus 1A2	XFORMR	D
228	ED	XR1B1R	4160V 480V Transformer to Bus 1B1	XFORMR	D
229	ED	XR1B2R	4160V 480V Transformer to Bus 1B2	XFORMR	D
230	FP	XR480R	34.5 kV 480V AC Transformer	XFORMR	D

Seismic Category:

AOV XFER - Air Operated Valve with transfers closed failure mode
HOV XFER - Manual Valve with transfers closed failure mode
MAN VALVE - Manual Valve modeled in human action
MOV XFER - Motor Operated Valve with transfers close failure mode
CHECK VLV - Check Valve
HEAT EX - Heat Exchanger
PIPE - Pipe integrity
DEMIN - Demineralizers
FILTER - Filters
STRAINER - Strainers
SWITCH - Hand Switches
FLOW TRANS - Flow Transmitter
LEV SW - Level Switch
LEV TRANS - Level Transmitter
PRESS SW - Pressure Switch
PRESS TRANS - Pressure Transmitter
FAIL TRIPPED - Component fails in the tripped or actuated position
SUCCESSFUL - Successful turbine trip assumed as a result of seismic event.
CONTROL - Solid state control circuitry
INSTRUMENT - Instrumentation
CKT BKR - Circuit Breaker or Fuse
DUPL - Duplicate Component (dual failure modes modeled in the PRA)
CONTROL ROD - Control rod drive failure

Table 3-4 Additional Components for the Oyster Creek Seismic Analysis

System Name	Component	Data Source	Seismic Note
Offsite Power	Offsite Grid	2	
	ceramic insulators	2	
AC Power Recovery	Combustion turbines	3	
	CT Fuel Oil Tank	3	
	CT Gas Supply Lines	3	
	CT crosstie/stepdown transformer to OC buses	3	
Emergency Power	Fuel oil day tank	2	Note B
	Diesel generator control/breaker panel	2	
DC Power	DC Panel D	2	
	DC Panel F	2	Note D
ESF Actuation	Sensors	2	Note 6
	ES relay cabinets	2	Note 8
Service Water	Service water pumps	2	Note A
TBCCW	Surge tank integrity	1	
Main Feedwater	Demineralizer blockage	1	Note 4
	Heater string failure	1	Note 4
Emergency Power	Diesel generator battery for start and control	4	
	Diesel generator control/breaker panel	2	
RPS	Scram discharge volume integrity	1	
	Control rod piston supply/exhaust piping integrity	1	
Service Water	Supply piping integrity	1	
TBCCW	System piping integrity	1	
Main Steam	Main steam lines	2	
Core Spray	Suppression pool suction	1	
Containment Spray	Suppression pool suction	1	
	ESW supply lines	1	
Main Condenser	Main condenser integrity	1	
	Circulating water supply and return tunnels	1	
ADS	EMRV discharge lines	1	
Primary Containment Isolation	Reactor building to torus vacuum breakers	1	Note 2
	Torus-to-drywell vacuum breakers	1	Note 2
	Containment purge/vent lines	1	
	Shutdown cooling lines	1	
	Reactor water cleanup lines	1	
	Drywell-to-torus vent lines	1	
	RBCCW supply and return lines	1	
SGTS	Discharge stack	1	
	Discharge line to stack	1	
Fire Protection	Fire pond supply (including dike)	1	
	Fire Pump Fuel Supply	1	
	Underground Piping	1	Note B

Table 3-4 Additional Components for the Oyster Creek Seismic Analysis

System Name	Component	Data Source	Seismic Note
Condensate Transfer	Condensate Transfer Building	1	
	Condensate transfer supply line integrity	1	
Inst. Air	System piping failure	1	
CRD	CRD lines to control rod drives	1	
Containment Vent	Vent line integrity	1	
	Discharge stack integrity	1	
Intake	Screen Wash Pumps	3	
	Traveling screens	3	
Reactor Coolant System	Reactor pressure vessel	2	
	Recirculation pumps	2	
	RPV internals	2	

Notes

1. Seismic notes are as defined for Table 3.1-3

2. Shaded items are in addition to those found in the OCPRA component lists

3. Data sources 1. Review of OCPRA system analysis assumptions and modeling.
   2. Review of Three Mile Island Unit 1 seismic analysis.
   3. Review of OCPRA special analyses (Appendix B).
   4. Review of Oyster Creek Operations Plant Manual (OPM)

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
1	IA	CP011R	Air Compressor 1-1	COMPRESS	A
2	IA	CP012R	Air Compressor 1-2	COMPRESS	A
3	EC	DG001R	Emergency Diesel Generator #1	DIESEL	A
4	EC	NONE	EDG control/circuit breaker panel	DIESEL	A
5	ED	DG002R	Emergency Diesel Generator #2	DIESEL	A
6	ED	NONE	EDG control/circuit breaker panel	DIESEL	A
7	LP	AC PWR	Combustion turbines	DIESEL/CT	A
8	EC	FN004R	Supply Fan FN-56-004	FAN	A
9	EC	FN007R	Exhaust Fan FN-56-007	FAN	A
10	EC	FN008R	Alternate Exhaust Fan FN-56-008	FAN	A
11	ED	FNE21R	Exhaust Fan EF 1-21	FAN	A
12	ED	FNS21R	Supply Fan SF 1-21	FAN	A
13	SG	FNTNAR	Train A Fan	FAN	A
14	SG	FNTNBR	Train B Fan	FAN	A
15	XB	FNE20R	A/B Battery Room Exhaust Fan	FAN	A
16	XB	FNS20R	A/B Battery Room Supply Fan	FAN	A
17	BI	PM91AR	P-19-001A SLC Pump A	PUMP	A
18	BI	PM91BR	P-19-001B SLC Pump B	PUMP	A
19	CC	PM51A	Pump 51A(1-1)	PUMP	A
20	CC	PM51B	Pump 51B(1-2)	PUMP	A
21	CC	PM51C	Pump 51C(1-3)	PUMP	A
22	CC	PM51D	Pump 51D(1-4)	PUMP	A
23	CC	PM52A	ESW Pump 52A (P-3-003A)	PUMP	A
24	CC	PM52B	ESW Pump 52B (P-3-003B)	PUMP	A
25	CC	PM52C	ESW Pump 52C (P-3-003C)	PUMP	A
26	CC	PM52D	ESW Pump 52D (P-3-003D)	PUMP	A
27	CD	PM08AR	NC08A CRD Pump	PUMP	A
28	CD	PM08BS	NC08B CRD Pump	PUMP	A
29	CP	PMCPAR	Condensate pump 1-A	PUMP	A
30	CP	PMCPBR	Condensate pump 1-B	PUMP	A
31	CP	PMCPCR	Condensate pump 1-C	PUMP	A
32	CS	PM01AR	Core Spray Pump NZ01A	PUMP	A
33	CS	PM01BR	Core Spray Pump NZ01B	PUMP	A
34	CS	PM01CR	Pump NZ01C	PUMP	A
35	CS	PM01DR	Pump NZ01D	PUMP	A
36	CS	PM03AR	Booster Pump NZ03A	PUMP	A
37	CS	PM03BR	Booster Pump NZ03B	PUMP	A
38	CS	PM03CR	Booster Pump NZ03C	PUMP	A
39	CS	PM03DR	Booster Pump NZ03D	PUMP	A
40	CT	PM001R	Condensate Transfer Pump 1-1	PUMP	A
41	CT	PM002R	Condensate Transfer Pump 1-2	PUMP	A
42	CW	PMO2AR	Circulating Water Pump P-3-002A	PUMP	A
43	CW	PMO2BR	Circulating Water Pump P-3-002B	PUMP	A
44	CW	PMO2CR	Circulating Water Pump P-3-002C	PUMP	A
45	CW	PMO2CR	Circulating Water Pump P-3-002D	PUMP	A
46	EC	PM01AR	Fuel Oil Transfer Pump 1A	PUMP	A

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
47	EC	PM01BR	Fuel Oil Transfer Pump 1B	PUMP	A
48	ED	PM02AR	Fuel Oil Transfer Pump 2A	PUMP	A
49	ED	PM02BR	Fuel Oil Transfer Pump 2B	PUMP	A
50	FP	PD001R	Fire Pump 1	PUMP	A
51	FP	PD002R	Fire Pump 2	PUMP	A
52	FP	PM003R	Redundant Fire Pump	PUMP	A
53	FW	PMFWAR	Feedwater pump A	PUMP	A
54	FW	PMFWBR	Feedwater pump B	PUMP	A
55	FW	PMFWCR	Feedwater pump C	PUMP	A
56	SW	PM01AR	Pump P-3-001A	PUMP	A
57	SW	PM01BR	Pump P-3-001B	PUMP	A
58	TB	PM003R	Pump P-5-003	PUMP	A
59	TB	PM003R	Pump P-5-004	PUMP	A
60	TB	PM003R	Pump P-5-005	PUMP	A
61	AD	PV18AD	ADS Valve NR108A	EMRV	B
62	AD	PV18BD	ADS Valve NR108B	EMRV	B
63	AD	PV18CD	ADS Valve NR108C	EMRV	B
64	AD	PV18DD	ADS Valve NR108D	EMRV	B
65	AD	PV18ED	ADS Valve NR108E	EMRV	B
66	CC	HX1AB	Heat Exchanger H-21-001B	HEAT EX	B
67	CC	HXCCAB	Heat Exchanger H-21-001A	HEAT EX	B
68	CC	HXCCCR	Heat Exchanger H-21-001C	HEAT EX	B
69	CC	HXCCDR	Heat Exchanger H-21-001D	HEAT EX	B
70	TB	HX003B	Heat Exchanger(C-5-003)	HEAT EX	B
71	TB	HX004P	Heat Exchanger C-5-004	HEAT EX	B
72	FP	XXLOWS	Diesel Pump Water Supply	POND	B
73	SO	SV28DD	Safety Valve (NR-28D) V-1-160	SAFETY	B
74	SO	SV28ED	Safety Valve (NR-28E) V-1-161	SAFETY	B
75	SO	SV28FD	Safety Valve (NR-28F) V-1-162	SAFETY	B
76	SO	SV28GD	Safety Valve (NR-28G) V-1-163	SAFETY	B
77	SO	SV28HD	Safety Valve (NR-28H) V-1-164	SAFETY	B
78	SO	SV28JD	Safety Valve (NR-28J) V-1-165	SAFETY	B
79	SO	SV28KD	Safety Valve (NR-28K) V-1-166	SAFETY	B
80	SO	SV28LD	Safety Valve (NR-28L) V-1-167	SAFETY	B
81	SO	SV28MD	Safety Valve (NR-28M) V-1-168	SAFETY	B
82	BI	TKSLCL	T-19-001 Liquid Poison Tank	TANK	B
83	EC/ED	TK001B	Fuel Oil Storage Tank	TANK	B
84	FP	TK001B	Redundant Fire Water Pump Supply	TANK	B
85	IA	TK001B	Air Receiver 1-1	TANK	B
86	IA	TK002B	Air Receiver 1-2	TANK	B
87	IA	TK003B	Air Receiver 1-3	TANK	B
88	IC	IC00AP	Isolation Condenser "A"	TANK	B
89	IC	IC00BP	Isolation Condenser "B"	TANK	B
90	ST	TK001B	CST T-11-001	TANK	B
91	TB	TBCCW	Surge tank integrity	TANK	B

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
92	IA	ADOABP	Air Dryer A-B	AIR DRIER	C
93	CP	AV216T	Control valve V-2-16	AOV	C
94	CP	AV235T	Control valve V-2-235	AOV	C
95	CP	AV732T	Feed regulating valve AOV-2-732	AOV	C
96	CP	AV733T	Feed regulating valve AOV-2-733	AOV	C
97	CP	AV734T	Feed regulating valve AOV-2-734	AOV	C
98	CS	AV092D	Min.Flow Valve V-20-92	AOV	C
99	CS	AV093D	Min. Flow Valve V-20-93	AOV	C
100	CS	AV094D	Min. Flow Valve V-20-94	AOV	C
101	CS	AV095D	Min. Flow Valve V-20-95	AOV	C
102	MU	AV034D	Air-Operated Valve V-11-34	AOV	C
103	MU	AV036D	Air-Operated Valve V-11-36	AOV	C
104	OV	AV271D	Air operated valve V-27-1	AOV	C
105	OV	AV272D	Air operated valve V-27-2	AOV	C
106	OV	AV817D	Air operated valve V-28-16	AOV	C
107	OV	AV818D	Air operated valve V-28-15	AOV	C
108	PI	AV202D	Air-Operated Valve V-22-2	AOV	C
109	PI	AV313D	Air-Operated Valve V-23-13	AOV	C
110	PI	AV314D	Air-Operated Valve V-23-14	AOV	C
111	PI	AV315D	Air-Operated Valve V-23-15	AOV	C
112	PI	AV316D	Air-Operated Valve V-23-16	AOV	C
113	PI	AV317D	Air-Operated Valve V-23-17	AOV	C
114	PI	AV318D	Air-Operated Valve V-23-18	AOV	C
115	PI	AV319D	Air-Operated Valve V-23-19	AOV	C
116	PI	AV320D	Air-Operated Valve V-23-20	AOV	C
117	PI	AV321D	Air-Operated Valve V-23-21	AOV	C
118	PI	AV322D	Air-Operated Valve V-23-22	AOV	C
119	PI	AV701L	Air-Operated Valve V-22-1	AOV	C
120	PI	AV778D	Air-Operated Valve V-22-28	AOV	C
121	PI	AV779D	Air-Operated Valve V-22-29	AOV	C
122	PI	AV817D	Air-Operated Valve V-28-17	AOV	C
123	PI	AV818D	Air-Operated Valve V-28-18	AOV	C
124	RI	AV003D	RB General Area Supply Valve V-28-3	AOV	C
125	RI	AV004D	RB General Area Supply Valve V-28-4	AOV	C
126	RI	AV005D	RB General Area Supply Valve V-28-5	AOV	C
127	RI	AV006D	RB General Area Supply Valve V-28-6	AOV	C
128	RI	AV007D	RB General Area Supply Valve V-28-7	AOV	C
129	RI	AV008D	RB General Area Supply Valve V-28-8	AOV	C
130	RI	AV009D	RB General Area Supply Valve V-28-9	AOV	C
131	RI	AV010D	RB General Area Supply Valve V-28-10	AOV	C
132	RI	AV011D	CRD Rebuild Area&Gen Area Sup V-28-11	AOV	C
133	RI	AV012D	CRD Rebuild Area&Gen Area Sup V-28-12	AOV	C
134	RI	AV013D	RB General Area Supply Valve V-28-13	AOV	C
135	RI	AV014D	RB General Area Supply Valve V-28-14	AOV	C
136	RI	AV021D	RB General Area Exhaust Valve V-28-21	AOV	C
137	RI	AV022D	RB General Area Exhaust Valve V-28-22	AOV	C

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
138	RI	AV036D	RB General Area Supply Valve V-28-36	AOV	C
139	RI	AV037D	RB General Area Supply Valve V-28-37	AOV	C
140	SG	AV823D	Train Inlet Valve V-28-23	AOV	C
141	SG	AV826D	Train Outlet Valve V-28-26	AOV	C
142	SG	AV827D	Train Inlet Valve V-28-27	AOV	C
143	SG	AV830D	Train Outlet Valve V-28-30	AOV	C
144	ST	AV016T	Air operated valve V-2-16	AOV	C
145	ST	AV235T	Air operated valve V-2-235	AOV	C
146	BI	MV161D	V-16-1 RWCU Inboard Isolation	MOV	C
147	BI	MV164D	V-16-14 RWCU Outboard Isolation	MOV	C
148	CC	MV005D	Drywell Header Inlet V-21-5	MOV	C
149	CC	MV011D	Drywell Header Inlet V-21-11	MOV	C
150	CC	MV013D	Dynamic Test (Torus) Return V-21-13	MOV	C
151	CS	MV015D	Dynamic Test (Torus) Return V-21-15	MOV	C
152	CS	MV015D	Parallel Isol. MOV V-20-15	MOV	C
153	CS	MV021D	Parallel Isol. MOV V-20-21	MOV	C
154	CS	MV040D	Parallel Isol. MOV V-20-40	MOV	C
155	CS	MV041D	Parallel Isol.MOV V-20-41	MOV	C
156	IC/MI	MV034D	Condensate Isolation valve MOV-14-34	MOV	C
157	IC/MI	MV035D	Condensate Isolation valve MOV-14-35	MOV	C
158	MI	MV030D	Steam Supply Isolation Valve MOV-14-30	MOV	C
159	MI	MV031D	Steam Supply Isolation Valve MOV-14-31	MOV	C
160	MI	MV032D	Steam Isolation Valve MOV-14-32	MOV	C
161	MI	MV033D	Steam Isolation Valve MOV-14-33	MOV	C
162	MI	MV036D	Condensate Return Iso. Valve MOV-14-36	MOV	C
163	MI	MV037D	Condensate Isolation Valve MOV-14-37	MOV	C
164	ME	NV03AD	Inboard MSIV NS3A	MSIV	C
165	ME	NV03BD	Inboard MSIV NS3B	MSIV	C
166	ME	NV04AD	Outboard MSIV NS4A	MSIV	C
167	ME	NV04BD	Outboard MSIV NS4B	MSIV	C
168	IA	SV054D	TBCCW Inlet Solenoid Valve V-5-54	SOV	C
169	IA	SV059D	TBCCW Inlet Solenoid Valve V-5-59	SOV	C
170	OV	SV591D	Solenoid valve V-6-591	SOV	C
171	OV	SV592D	Solenoid valve V-6-592	SOV	C
172	OV	SV596D	Solenoid valve V-6-596	SOV	C
173	PI	SV809D	Solenoid Valve V-38-9	SOV	C
174	PI	SV810D	Solenoid Valve V-38-10	SOV	C
175	PI	SV816D	Solenoid Valve V-38-16	SOV	C
176	PI	SV817D	Solenoid Valve V-38-17	SOV	C
177	PI	SV837D	Solenoid Valve V-38-37	SOV	C
178	PI	SV838D	Solenoid Valve V-38-38	SOV	C
179	PI	SV839D	Solenoid Valve V-38-39	SOV	C
180	PI	SV840D	Solenoid Valve V-38-40	SOV	C
181	PI	SV841D	Solenoid Valve V-38-41	SOV	C
182	PI	SV843D	Solenoid Valve V-38-43	SOV	C
183	PI	SV844D	Solenoid Valve V-38-44	SOV	C

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
184	PI	SV846D	Solenoid Valve V-38-46	SOV	C
185	RS	VS117X	Scram Pilot Solenoid Valves I (137 SO117s)	SOV	C
186	RS	VS118X	Scram Pilot Solenoid Valves II (137 SO118s)	SOV	C
187	RS	VS450D	Backup Scram Valve I (V-6-450)	SOV	C
188	RS	VS451D	Backup Scram Valve II (V-6-451)	SOV	C
189	RS	VSAB2D	ARI Block Valve #2	SOV	C
190	RS	VSAR1D	ARI Main Header Vent Valve #1	SOV	C
191	RS	VSAB1D	ARI Block Valve #1	SOV	C
192	BI	EV944D	V-19-44 Squib Valve B	SQUIB	C
193	BI	EV945D	V-19-45 Squib Valve A	SQUIB	C
194	BT	BV30AD	Turbine Bypass Valve V-1-130A	TBV	C
195	BT	BV30BD	Turbine Bypass Valve V-1-130B	TBV	C
196	BT	BV30CD	Turbine Bypass Valve V-1-130C	TBV	C
197	BT	BV30DD	Turbine Bypass Valve V-1-130D	TBV	C
198	BT	BV30ED	Turbine Bypass Valve V-1-130E	TBV	C
199	BT	BV30FD	Turbine Bypass Valve V-1-130F	TBV	C
200	BT	BV30GD	Turbine Bypass Valve V-1-130G	TBV	C
201	BT	BV30HD	Turbine Bypass Valve V-1-130H	TBV	C
202	BT	BV30ID	Turbine Bypass Valve V-1-130I	TBV	C
203	DB	BTDCBD	125V DC Battery B	BATTERY	D
204	DC	BTDCCD	125V DC Battery C	BATTERY	D
205	DB	BSDCBR	125V DC Bus B	BUS	D
206	DC	BSDCCR	125V DC Bus C	BUS	D
207	EA,EB	BS01AR	4,160V AC Bus 1A	BUS	D
208	EA,EB	BS01BR	4,160V AC Bus 1B	BUS	D
209	EC	BS01CR	4,160V AC Bus 1C	BUS	D
210	EC	BS1A1R	480V AC Bus 1A1	BUS	D
211	EC	BS1A2R	480V AC Bus 1A2	BUS	D
212	ED	BS01DR	4,160V DC Bus 1D	BUS	D
213	ED	BS1B1R	480V AC Bus 1B1	BUS	D
214	ED	BS1B2R	480V AC Bus 1B2	BUS	D
215	FP	BS1E1R	480V AC Bus 1E1	BUS	D
216	XB	BCDABR	Standby Static Charger A/B	CHARGER	D
217	XB	MGDCBR	Battery Charger B Motor Generator Set	CHARGER	D
218	XC	BCDC1R	Battery Charger C1	CHARGER	D
219	XC	BCDC2R	Standby Static Charger C2	CHARGER	D
220	RS	CRDR12	Control Rod Drives (137)	CNTRL ROD	D
221	LP	Recovery	CT crosstie/stepdown transformer	XFORMER	D
222	OP	AC Power	Offsite power (ceramic insulators, etc.)	XFORMER	D
223	EA,EB	XR0SAR	Startup Transformer SA	XFORMR	D
224	EA,EB	XR0SBR	Startup Transformer SB	XFORMR	D
225	EC	XR1A1R	4,160V/480V Transformer to Bus 1A1	XFORMR	D
226	EC	XR1A2R	4,160V/480V Transformer to Bus 1A2	XFORMR	D
227	ED	XR1B1R	4160V 480V Transformer to Bus 1B1	XFORMR	D
228	ED	XR1B2R	4160V 480V Transformer to Bus 1B2	XFORMR	D
229	FP	XR480R	34.5 kV 480V AC Transformer	XFORMR	D

Table 3-5 COMPONENTS FOR SEISMIC CAPACITY EVALUATION

Item No.	Top Event	I.D.	Component Description	Category	Seismic Note
230		RPV	Reactor pressure vessel	RPV	E
231		RPV	Recirculation pumps	RPV	E
232		RPV	RPV internals	RPV	E

Seismic Category:

AOV - Air Operated Valve
AOV XFER - Air Operated Valve with transfers closed failure mode
HOV XFER - Manual Valve with transfers closed failure mode
MAN VALVE - Manual Valve modeled in human action
MOV XFER - Motor Operated Valve with transfers close failure mode
MOV - Motor Operated Valve
SOV - Solenoid operated valve
CHECK VLV - Check Valve
HEAT EX - Heat Exchanger
PIPE - Pipe integrity
DEMIN - Demineralizers
FILTER - Filters
STRAINER - Strainers
SWITCH - Hand Switches
FLOW TRANS - Flow Transmitter
LEV SW - Level Switch
LEV TRANS - Level Transmitter
PRESS SW - Pressure Switch
PRESS TRANS - Pressure Transmitter
FAIL TRIPPED - Component fails in the tripped or actuated position
SUCCESSFUL - Successful turbine trip assumed as a result of seismic event.
CONTROL - Solid state control circuitry
INSTRUMENT - Instrumentation
CKT BKR - Circuit Breaker or Fuse
DUPL - Duplicate Component (dual failure modes modeled in the PRA)
CNTRL ROD - Control rod drive failure

3.1.3 Analysis of Plant Systems and Structure Response

In the seismic probabilistic risk assessment (SPRA) the seismic hazard, and the seismic capacity of plant structures and equipment and the probability of independent random failure are combined to produce a probability distribution of seismic induced core damage. Each of the above elements are discussed in various report sections.

In report Section 3.1.1, Hazard Analysis, the seismic hazards are discussed. Report Section 3.1.4 discusses the component fragilities and failure modes. In later report sections (3.1.5) the probability of independent plant components and equipment failures are discussed. In this report section the capacity of selected civil structures and soil properties are discussed.

Seismic Fragilities of Civil Structures

This report section discusses the methodology and presents the results for the seismic fragility evaluation of selected civil structures to be included in the SPRA. This summary of the structural fragility evaluation is taken in large part from the "Seismic Fragilities of Civil Structures at Oyster Creek Nuclear Generating Station", October 1994 which was produced by EQE International.

The seismic fragilities represent the probabilistic definitions of seismic capacity for civil structures and are reported in terms of the peak free field acceleration of the reference earthquake. The civil structures and components included in the evaluation documented here are:

- Reactor Building
- Turbine Building
  - Building
  - Turbine Pedestal
  - Outdoor HVAC Platform (on the north side)
- Intake Structure
- Emergency Diesel Generator Building
- EDG Duct Bank to Turbine Building
- Fire Pond Pump House
  - Pump House
  - Fire Pond Dam
  - Fire Protection Underground Piping
- Circulating Water Intake
  - Intake Tunnel
  - Discharge Tunnel
  - Discharge Outfall Structure

- Combustion Turbines
  - Combustion Turbines
  - Fuel Oil Tank
  - Gas Supply Piping

- Condensate Transfer Building

- Ventilation Stack

In the above, the seismic fragility for a single system composed of several components is based on the most vulnerable or weak link whose capacity controls the functionality of the system. For some structures, the seismic capacity is judged to be high and beyond the range of interest based on a review of the postulated seismic loading and credible failure modes.

A walkdown was conducted on January 11, 1994, to review the above structures and components. The walkdown team consisted of one engineer each from EQE and Geomatrix, as well as two engineers from GPU Nuclear. The walkdown focused on reviewing the general layout of the structures and identifying important features of the structures and the adjacent areas significant to the postulated earthquake ground motion.

Potential soil related seismic failures (liquefaction, seismic induced settlements, and slope deformation) were examined as part of the Seismic IPEEE. A summary of the results of the Oyster Creek Soil Failure Analysis is provided in the following report section. Soil related issues were found to be significant for several of the above structures and components. As a result, fragilities were evaluated for failure modes associated with soil liquefaction induced deformations, where applicable, in addition to those associated with direct seismic induced (inertial) effects.

The Oyster Creek site is characterized as a soil site in "Probabilistic Seismic Hazard Evaluations at Nuclear Power Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue", EPRI NP-6359-D and "Seismic Characterization of 69 Nuclear Power Plants Sites East of the Rocky Mountains", NUREG/CR-5250. The seismic hazard is described in terms of annual exceedance curves for the peak ground acceleration. In addition to the hazard curves, free field response spectra are provided at the 15th, 50th and 85th percentile for return periods ranging from 1000 years to 100,000 years. The fragilities described in this report section are anchored to the mean peak ground acceleration. This acceleration is the average of the peak accelerations from two horizontal components. For this evaluation, the spectral shape associated with the 50th percentile, 10,000 year return period response spectrum in EPRI NP-6359-D is taken to be median centered for all fragilities.

For most of the structures considered here, the fragilities are based upon the latest probabilistic seismic response analyses, "Probabilistic Seismic Response Analyses of the Oyster Creek Nuclear Generating Station in Support of the IPEEE", EQE Report No. 50124-R-003, July 1994 (Reference 3-1). In this report the soil-structure interaction analyses are performed for the Reactor Building, the Turbine Building and Turbine Pedestal, the Intake Structure, and the Emergency Diesel Generator Building which includes multiple time history analyses at peak free field ground acceleration levels of 1SSE (0.184g), 2SSE (0.368g) and 3SSE (0.552g). Soil stiffness and damping properties consistent with these ground motions are used in the analyses.

In the multiple time history analyses, the soil stiffness and damping properties as well as the structure stiffness and damping properties are varied. The selection of the values of these properties was based on the Latin Hypercube sampling techniques. Median structure responses and the total variability associated with the earthquake ground motion, the soil stiffness and damping, and the structure stiffness and damping are evaluated. Possible non-linear structure effects such as concrete cracking, buckling of compression members, etc., at higher acceleration levels are not included and hence, all of the response analyses represent elastic structure response. The approach adopted in this fragility evaluation is to determine the median factor of safety and its statistical variability which exists for the 3SSE earthquake in order to estimate the expected response at failure. Although inelastic energy dissipation is included in determining the factor of safety, no nonlinear analyses of the structures is conducted, and all evaluations are based on elastic analysis and load distributions.

The fragility results can be used together with the estimated annual frequency of occurrence of various ground motion levels to determined the frequency of seismic induced failure for each of the safety related structures in the plant. In the total probabilistic risk assessment, these conditional structure failure frequencies, together with similar equipment fragilities are used with independent failure probabilities to determine the core damage frequency.

The controlling failure modes for each of the Oyster Creek structures included in the IPEEE have been evaluated. The median load demands and variabilities corresponding to the reference earthquake motion, 3SSE are evaluated the reference 1 analysis. In some cases, the structure capacities are found to be high and thus, a statement of high capacity is made without a complete fragility evaluation. The resulting fragilities for the structures are discussed in the following paragraphs. Where applicable, fragilities are report for both the seismic inertia related failure modes and earthquake induced soil liquefaction or deformation failure modes.

3.1.3.1 Reactor Building

The Oyster Creek Reactor Building consists of four main sub-structures: the main structure, the drywell, the biological reactor shield wall (BSW) and the reactor pressure vessel (RPV). The main structure is a rectangular reinforced concrete structure up to the refueling floor at elevation 119'. Above the refueling floor, the structure consists of a steel frame with insulated metal siding. The drywell containment vessel is an axisymmetic steel shell surrounded by a heavy concrete shield wall which follows the contour of the vessel from the foundation of the drywell to the refueling floor. The interface between the reactor building, drywell, BSW and RPV includes the RPV stabilizer, the Star Truss, the Drywell lugs, and the radial support beams of two steel platforms at elevations 46' and 23' 4.5".

Failure modes investigated for the reinforced concrete portion of this building include shear and overturing failure of the perimeter and the drywell walls, and diaphragm shear failure. Additional failure modes investigated include shear and overturing of the concrete pedestal, failure of the BSW anchorage, failure of the Star Truss at the top of the BSW, failure of the steel roof trusses, and failure of the steel superstructure column anchorage. The controlling mode of failure for the reinforced concrete is overturing of the North exterior wall between elevations 23'6" and 51'3". The shear and flexural capacities of this segment of the wall are calculated in Reference 3-2.

As previously noted, the critical section for flexure is located at the construction joint at El. 23'-6" where the flexural tension capacity is limited to the contribution of only the inner curtain of vertical reinforcing as the outer curtain dowel bars do not have an adequate lap splice to develop their strength. Also, at this elevation, the thickness of the wall decreases and the concrete design strength decreases. The median acceleration capacity is estimated to be approximately 2.96g. Median factors of safety and variabilities are listed on Table 3-2 following this report section. Failure of this wall is expected to lead to damage to safety-related equipment inside the RB. The controlling failure mode for the steel superstructure is the anchorage of the corner columns. The anchorage consists of 4-1 1/2" diameter J-bolts and is controlled by pull-out. The median ground acceleration capacity is estimated to be 1.0g. The median factors of safety and variabilities are shown on Table 3-2, following this report section. Failure of the column anchorage may potentially lead to the collapse of the crane supported by the steel superstructure. An evaluation of the crane itself was not performed in this study.

As noted in Reference 3-6, no significant soil liquefaction or settlement which could affect this structure is expected due to the earthquake ground motion.

3.1.3.2 Turbine Building

Several structural systems of the Turbine Building (TB) are considered in the fragility evaluation. The main structure of the Turbine building includes the lower portion which consists of concrete shear walls with concrete beams and slabs. The reinforced concrete portion is located from the foundation mat to the turbine main operating floor at El. 46'6". The upper portion is a steel superstructure above El. 46'6", which consists of steel columns supporting long span roof trusses running in the East-West direction and steel braced frames in the North-South direction.

The turbine pedestal is a heavy reinforced concrete frame and shear wall structure which supports the turbine and other equipment. The turbine pedestal is supported on a common foundation mat with the main structure.

Also included in the scope of work for the Turbine building is the HVAC platform located on the north side of the building, just outside the control room. The capacity of each of these systems are evaluated separately.

The capacity of the concrete portion of the main Turbine building structure is governed by shear failure of the floor diaphragm at El. 46'6" due to EW seismic loads. The critical location is near the south-west corner of the building, where a floor slab opening reduces the effective area for shear transfer. Other potential failure modes that were examined include shear and flexure of the main shear walls and the failure of diaphragms at elevations 23'6", and 63'6". The median acceleration capacity is estimated to be approximately 0.88g. Median factors of safety and variabilities are listed in Table 3-2.

The capacity of the steel portion of the structure is governed by an anchorage failure of the columns on Line F which carry EW direction frame loads and NS direction braced frame loads. The median capacity acceleration was found to be approximately 0.74g. Failure of the anchorage is expected to result in a loss of stability of the steel frames, leading to a potential partial collapse of the column or a collapse of the crane. This could result in impacts with the floor slab at El. 46'6", which could affect safety related equipment or potentially induce relay chatter in electrical equipment in the control room.

The lateral force resistance of the turbine pedestal is provided by reinforced concrete beams, columns, and shear walls. In the NS direction, the beams and columns form a frame to resist the seismic loads. In the EW direction, three shear walls, which are located approximately at the center of the pedestal, and four one-bay frames resist the seismic loads. Failure modes investigated for the pedestal include shear and flexural failure of the shear walls as well as shear and flexural failure of the beams and columns. The shear capacities of the beams and columns exceed the flexural capacities such that the governing failure mode corresponds to the development of a plastic hinge mechanism of the EW frame at column line 3. The median ground acceleration capacity is estimated to be 2.17g.

The outdoor HVAC platform, located at El. 41', consists of steel framing with steel grating which supports three fans. The south end of the platform is anchored to the north concrete wall of the Turbine building, while the north end of the platform is supported by steel columns. Steel angle diagonal braces provide the seismic diaphragm for the platform to carry load back to the Turbine building wall. A review of the diaphragm brace and the platform anchorage indicates that the ground acceleration capacity of the platform exceeds that of the main structure's concrete diaphragm at El. 46'6". Therefore, the platform does not control.

From Reference 3-6, the free-field PGA required to cause liquefaction of the fill beneath the Turbine building exceeds 0.70g (considered a conservative bound). This acceleration corresponds to a high confidence value. Significant settlements are unlikely without liquefaction. This soil-related failure is not considered controlling.

3.1.3.3 Intake Structure

The Intake Structure (IS) is a partially embedded reinforced concrete structure, partly filled with water. The main lateral force resisting system consists of concrete slabs in the water intake tunnel at El. 2'6", at the operating floor at El. 6'0", and at the roof, the 2.5' to 3.5' thick concrete walls in the EW direction, and a 3.5' thick wall in the NS direction.

Failure modes investigated include shear and overturning failure of the perimeter walls, and shear failure of floor diaphragms. The capacity of the IS is governed by a shear failure of the diaphragm at El. 6' for NS loads. At this elevation, the diaphragm is perforated with numerous openings. The median acceleration capacity is estimated to be approximately 0.82g. Median factors of safety and variabilities are listed in Table 3-2 following this report section. Failure of this floor slab is expected to lead to damage to safety-related equipment located on the floor.

From Reference 3-6, significant liquefaction or settlements are unlikely beneath the Intake structure for any ground motion.

3.1.3.4 Emergency Diesel Generator Building

The Emergency Diesel Generator Building (EDGB) is a reinforced concrete shear wall structure, supported by a concrete foundation mat. While the walls and the foundation slab are monolithic, the roof consists of a series of precast panels which are bolted to the walls on two sides, but not to one another. The precast roof panels are bolted to the long NS direction walls. No shear transfer continuity is provided between the panels to span the EW seismic loads to the EW direction walls. This then forces the EW seismic roof inertia loads near the center of the building to be resisted by the out-of-plane bending of the long NS walls, with the walls acting as cantilevers fixed at the foundation. Significant failure modes are identified for both seismic inertia loading and for seismic induced ground deformation.

The seismic inertia induced failure modes investigated for this building include shear and overturning failure of the perimeter walls, shear failure of the roof panels, anchorage failure of the roof panels, and sliding of the building. The controlling mode of failure is out-of-plane bending of the N-S walls. The median acceleration capacity is estimated to be approximately 1.18g. Median factors of safety and variabilities are in Table 3-2. Failure of these walls is expected to lead to damage to safety-related equipment inside the EDGB.

Potential soil related failures are identified in Reference 3-6. The soil related failure of the Diesel Generator building is expected to be controlled by the sheet pile system located at the toe of the slope at the discharge canal. The extent of fill beneath the DG building is not known exactly. However, it appears possible that sufficient fill exists under the DG building that failure of the sheet pile restraint system could at least result in failure of the day tank piping.

In addition to the sheet piles, fender piles are located at 10 foot centers and the tie back system consists of a 1-1/2 inch diameter and a 2-1/2 inch diameter rod, each with turnbuckles, at 10 foot centers. Seismic loads acting on the wall for the liquefied and non-liquefied conditions are given in Reference 3-6. Liquefaction of the fill is expected at a free-field PGA of about 0.40g. Below is shown the soil-controlled seismic fragility for the DG building with the fill in the liquefied condition.

$$\hat{A} = 0.69 \text{ g}$$

$$\beta_R = 0.14$$

$$\beta_U = 0.28$$

$$\text{HCLPF} = 0.35 \text{ g}$$

The seismic capacity for the non-liquefied condition is above the range of interest.

In the preceding discussion, the structure failure mode associated with the out-of-plane bending of the NS walls is based on the assumption that liquefaction does not occur. However, the soil deformation failure mode is based on the occurrence of liquefaction. Since the liquefaction is a probabilistic event, different outcomes or failure modes are possible depending on whether liquefaction occurs. In developing the risk model for the seismic failure, a conditional model is required for the Diesel Generator building. This conditional model must include the probability that liquefaction occurs, then include the failure mode given that liquefaction does or does not occur. Specifically, the occurrence of liquefaction is defined by a fragility with a median peak ground acceleration capacity and variabilities. The failure mode of the EDGB is conditional on the occurrence of liquefaction. Given that liquefaction does not occur, the governing failure mode of the EDGB is the out-of-plane bending of the NS walls. On the other hand, given that liquefaction occurs, the failure mode is controlled by the soil deformations caused by the failure of the sheet pile system at the discharge canal.

Based on the results provided in Reference 3-6, the median PGA at which liquefaction is expected to occur is 0.40g and the randomness variability, β_R, is 0.14.

Bus Duct Bank From the Diesel Generator Building to the Turbine Building

The bus duct from the DG building to the Turbine building consists of two reinforced concrete duct banks. Each bank is 24 inches deep with 4 inch diameter cavities. The first bank is 12 inches wide with three cavities and the second is 18 inches wide with six cavities. Both are reinforced with #4 bars spaced at 8 inch centers top and bottom with an additional #4 bar at mid-depth at each face. No. 3 bar ties spaced at 18 inches on center are specified. The reinforcing steel is Grade 40. The concrete is assumed to have a 28 day design strength of 3000 psi.

Significant soil settlements are estimated in Reference 3-6 for the soil backfill adjacent to the Turbine building. The area where such settlements are postulated extends about 50 to 60 feet from the face of the Turbine building. Uncertainty in the compacted properties of the backfill leads to an estimated median settlement of 3 inches with 5% and 95% confidence bounds of 1.5 to 8.3 inches.

Assuming a worst case condition of 42 inches of soil cover above the duct bank and assuming none of the cover is dislodged by the ground shaking, it was found that the duct bank can be expected to form plastic hinges as a result of the most severe ground settlement estimated in Reference 3-6. However, the required displacement ductility is less than 3 for the $N_1 = 10$ soil condition 95th percentile settlement of 8.3 inches. Park and Paulay (see Reference 3-2, no. 27) indicate that much larger ductility ratios can be expected prior to failure for the above concrete and steel strengths. Since the shear capacity of the duct bank is significantly greater than the flexural capacity, it is concluded that even if significant soil settlement should occur in the region of the Diesel Generator - Turbine building, failure of the electrical cables within the duct bank is unlikely.

3.1.3.5 Fire Protection Water Pond Pump Area

Fire Pond Pump House

The Fire Pond Pump House is a fully embedded concrete box-type structure, partly filled with water. The main lateral force resisting system consists of a concrete slab at grade, concrete shear walls in the N-S direction, and a 1' shear wall in the E-W direction. A steel frame supporting a monorail and several other pieces of equipment are anchored to the slab at grade. A sheet metal structure encloses the equipment at grade.

The controlling failure mode of this structure is sliding of the building. Sliding initiates when the seismic base shear and the soil active forces overcome the sliding resistance forces. However, sliding-induced failure does not occur until sufficient displacement is developed to cause significant structural damage and/or damage to safety-related equipment. Failure of the pump house is expected to correspond to a sliding displacement of approximately 2 inches. For this displacement, damage to the fire water piping leaving the pump house is expected. The pump house was not entered and the pipe connections were not examined. Therefore, the median displacement to failure is based on judgment. The median acceleration capacity is approximately 1.21g, with the onset of sliding starting at around 0.36g. Median factors of safety and variabilities are listed in Table 3-2 following this report section.

Based on Reference 3-6, significant liquefaction or settlements are not expected for any ground motion.

Fire Water Buried Piping

The seismic capacity of the buried fire water piping between the fire pump house and the plant fire protection lines is expected to be controlled by possible ground movement where it crosses the dike separating the intake and discharge canals. It is assumed (Reference 3-6) that fill was placed above the reinforced concrete anchor wall and tiebacks. This fill could be expected to liquefy at about a peak ground acceleration of 0.40 g.

The sheet pile at the dike is restrained by 3-1/2 inch diameter tie rods with turnbuckles on 9 foot centers anchored in a reinforced concrete anchor wall. The wall is in turn supported by vertical and batter piles. Lateral hydrodynamic and soil loads for both the liquefied and non-liquefied soil conditions as a function of PGA are provided in Reference 3-6. Failure of the sheet pile is assumed to lead to sufficient ground displacement to fail the fire piping.

The seismic capacity of the sheet pile is controlled by the 3-1/2 inch ties. For the liquefied soil condition, this failure is expected at a median PGA of over 1.5 g with a High Confidence of Low Probability of Failure (HCLPF) of over 1.0 g as shown below.

$$\hat{A} = 1.57 \text{ g}$$

$$\beta_R = 0.14$$

$$\beta_U = 0.10$$

$$\text{HCLPF} = 1.06 \text{ g}$$

The seismic capacity for the non-liquefied soil condition is much higher and will not control.

Fire Pond Dam

Based on Reference 3-6, significant liquefaction, lateral movements, or settlements are not expected for any ground motion. The dam is therefore judged to have capacity beyond the range of interest.

3.1.3.6 Circulating Water Intake and Discharge Tunnels and Outfall Discharge Structure

The Circulating Water Intake and Discharge Tunnels are buried reinforced concrete rectangular tunnels. The tunnels run in the EW direction and are founded at El. - 14'. The intake tunnel is located on top of the discharge tunnel. Expansion joints are provided in the tunnels to allow longitudinal movement. The intake tunnel runs from the east side of the Intake structure to the Turbine building. The discharge tunnel runs from the Turbine building toward the discharge canal. The discharge structure is also reinforced concrete and extends from the discharge tunnel at an angle, opening to the discharge canal.

Adequate longitudinal flexibility is provided in the expansion joints to withstand the ground shaking effects. Also, based on the results in Reference 3-6, significant soil liquefaction or settlements are not expected beneath these structures for any ground shaking. Therefore, the seismic fragility of the intake and discharge tunnels and the discharge structure is judged to be beyond the range of interest.

3.1.3.7 Combustion Turbines

Based on a review of drawings, Reference 3-6, and the walkdown observations, the most vulnerable components of the combustion turbine system are the fuel oil tank and the gas supply piping. Significant soil settlements beneath the gas turbine building and the fuel oil tank are not expected at any level of ground motion.

Combustion Turbine Fuel Oil Tank

The Combustion Turbine Fuel Oil tank is a 1 million gallon capacity unanchored cylindrical steel tank, with a radius of 30 ft and a height of 48 ft. The shell is fabricated from ASTM A283, Grade C steel, with the wall thickness varying from 0.344 inches at the base to 0.25 inches at the top. The tank has a conical roof. The bottom plate is 0.25 inches thick. The tank is supported on a reinforced concrete ring wall and the bottom plate is supported by a sand cushion over a 9 inch thick concrete slab.

To evaluate the median response, the deterministic procedures (recommended in Reference 3-2, Reference 33) are used with median centered input parameters. The tank capacity is controlled by shell buckling due to base moment. This failure mode dominates over other potential failure modes, such as sliding or shell hoop tension failure, because this tank is unanchored and thus has relatively low resistance against seismic-induced base uplift. The fragility parameters for the fuel oil tank are as follows:

$$\hat{A} = 0.66\ g$$

$$\beta_R = 0.37$$

$$\beta_U = 0.39$$

$$HCLPF = 0.19g$$

Gas Supply Piping

Soil related failure of the combustion turbine system is expected to be controlled either by horizontal or vertical soil displacement of the soil supporting the gas supply piping. The gas supply pipe is located at the top of the slope on the west side of the intake canal. A minimum of 3 feet of cover is specified. Thus, the pipe may be partially surrounded by black peat or the fine to medium sand of the roadbed.

The pipe consists primarily of 16-inch schedule 30 (std.) pipe with some 10-inch schedule 40 pipe in the vicinity of the combustion turbine building and 16-inch schedule 80 pipe supported by the bridge. The gas pipe supported by the bridge is supported from the main bridge girder. The supply end is connected to the New Jersey Natural Gas Co. metering station by a 16 inch 600 lb. bolted flange. The greatest soil displacement as a function of ground acceleration is expected at Trench no. 4 and 6B. Both of these locations are in the 16 inch schedule 30 pipe portion of the line, and the seismic fragility is based on soil displacements for these locations.

Based on a weighted average of the modulus of subgrade reaction for the sand and the peat, an offset displacement of about 4 inches (10 cm.) is required to develop a fully plastic moment in the pipe. Since the pipe is ductile and can be expected to withstand additional displacement, the 4 inch displacement can be considered a conservative estimate of failure displacement. For the worst case trenches (nos. 4 and 6B), 4 inches displacement corresponds to a PGA of about 0.75g.

A check was also made on the bending moment capacity of the 600 lb. flange. Some slight leakage past the gasket seal may occur at bending moments less than the pipe fully plastic bending moment. Assuming the same soil condition at the flange as at Trench no. 4, this would indicate some slight leakage could occur at PGA's less than 0.75g. This minor leakage would not be expected to result in loss of the combustion turbines unless a source of ignition were present near the leak. In order to reach the elastic limit of the flange bolts and hence possibly result in a significant leak, PGA's well in excess of 1g with the maximum ground offset near the flange are required.

The fragility of the combustion turbine gas supply piping is conservatively estimated to occur once the fully plastic bending moment in the pipe occurs:

$$\hat{A} = 0.75\text{ g}$$

$$\beta_R = 0.14$$

$$\beta_U = 0.21$$

$$\text{HCLPF} = 0.42\text{ g}$$

No evaluation of the New Jersey Natural Gas Co. transmission system was conducted in the current program. A decision was made to assume a failure of the New Jersey Natural Gas Co. transmission system following an earthquake of any magnitude. Therefore, the gas supply is assumed lost for all seismic events and combustion turbines are restricted to fuel oil use only.

3.1.3.8 Condensate Transfer Building

The Condensate Transfer Building consists of a reinforced concrete slab on grade with reinforced concrete piers which run downward from the slab and are supported on the circulating water intake tunnel. Above grade, the building consists of steel members with a sheet metal exterior. Since the steel structure is relatively light, the capacity for seismic inertia loads is expected to be high. Significant liquefaction or settlements are not likely for this structure. Therefore, the seismic fragility of the Condensate Transfer building is judged to be beyond the range of interest.

During plant walkdowns, it was noted that the Condensate Storage Tank (CST) is likely the most vulnerable component of this system. However, the condensate storage tank is not included in the scope of the structural seismic capacity evaluation. A component fragility is developed in report Section 3.1.4, "Evaluation of Component Fragilities and Failure Modes".

3.1.3.9 Ventilation Stack

The Ventilation Stack is a 394 ft. tall reinforced concrete chimney. The stack has a circular cross section which tapers from an outside diameter of 31'-8 3/4" at the base (top of foundation) to 9'-6" at the top. The wall thickness varies from 18" at the base to 6" at the top. The stack is founded on an octagonal spread footing which is 45' across and 7' thick. The stack is partially embedded with the top of the foundation located at elevation -3', which is 26' below the plant grade elevation.

A detailed fragility evaluation was not conducted for the stack. Since the objective of the review of the stack is to assess whether there is the potential for failure leading to collapse or partial collapse and impact with the surrounding structures, a screening approach is used to verify that the stack has a sufficiently high HCLPF capacity. This is accomplished using the seismic margins approach outlined in "A Method for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)", EPRI NP-6041-SL, Rev 1, August 1991.

A simple stick model was generated for the stack, assuming a fixed base. Seismic loading was based on response spectrum with the median NUREG/CR-0098 shape anchored to a peak ground acceleration of 0.3 g. The shear and moment capacities were found to be adequate for the resulting load demands.

Reference 3-6 notes that no significant liquefaction or settlements are expected beneath the stack foundation. It is also noted that the soil bearing capacity is very high such that soil bearing failures are not expected. Therefore, the Ventilation Stack is judged to be non-controlling.

Table 3-2, Seismic Fragility of Oyster Creek Civil Structures, presented below provides a summary of the capacities of the various Oyster Creek structures. The first column of Table 3-2 provides the structure as well as the controlling failure mode in parentheses. The remainder of the columns provide the capacity in median acceleration ($\hat{A}$), the randomness variability (β_R), the uncertainty variability (β_U) and the High Confidence Low Probability of Failure (HCLPF). This table appears in Section 3.1 of this report and is repeated here for convenience.

Table 3-2 Seismic Fragility of Oyster Creek Civil Structures				
Structure (Failure Mode)	**Median Acceleration Capacity ($\hat{A}_g$)**	**Randomness Variability (β_R)**	**Uncertainty Variability (β_U)**	**HCLPF**
Reactor Building (Wall)	2.96g	0.29	0.25	1.21g
Reactor Building (Column Anchorage)	1.00g	0.38	0.27	0.34g
Turbine Building (Diaphragm Shear)	0.88g	0.20	0.26	0.41g
Turbine Building (Column Anchor Bolt)	0.74g	0.35	0.24	0.28
Turbine Pedestal	2.17g	0.24	0.28	0.92g
Intake Structure	0.82g	0.18	0.26	0.40g
Emergency DG Building	1.18g	0.37	0.22	0.45g
EDG Building (with liquefaction)	0.69g	0.14	0.28	0.35g
Fire Pond Pump House	1.21g	0.25	0.44	0.39g
Combustion Turbine Fuel Tank	0.66g	0.37	0.39	0.19g

Notes: HCLPF denotes the High Confidence of Low Probability of Failure, which corresponds to the lower 5% tail of the 5% uncertainty variability curve.

The values shown above were taken from Seismic Fragilities of Civil Structures at Oyster Creek Nuclear Generating Station (EQE International, October 1994), Reference 3-2.

The civil structure fragilities provided in Table 3-2 represent structural deformation and permanent displacement. These fragilities do not indicate building or structural collapse. The incorporation of the civil structure fragilities in the Oyster Creek seismic risk model uses judgement in the assignment of these fragilities. In many cases, component or equipment fragilities have much lower capacities than the structure which they occupy. In these cases the structure fragility may not be incorporated into the risk model since it would not contribute significantly to the results. This is discussed in additional detail in report Section 3.1.5, Analysis of Plant Systems and Sequences as well as the following report Section 3.1.4, Evaluation of Component Fragilities and Failure Modes.

3.1.4 Evaluation of Component Fragilities and Failure Modes

The walkdowns performed in support of the evaluation of component fragilities and failure modes took place during the 15R refueling outage. The walkdowns were performed for components on Table 3-5, supporting relays, A-46 equipment and masonry block walls. The fragilities analysis was performed by EQE Consulting and are presented in the report sub-sections below.

3.1.4.1 Evaluation of Component Fragilities

Each of the components in Table 3-5 was walkdown as part of the IPEEE component fragility walkdowns. The walkdown team consisted of a seismic capacity expert (EQE Consultants) and two GPUN engineers; one Civil/Structural engineer and the other a Risk Analysis engineer.

The components and equipment identified on Table 3-5 was supplemented with the A-46 Safe Shutdown Equipment List (SSEL) to ensure all those components which may be significant risk contributor in the seismic risk analysis are included. Information collected in the walkdowns as well as plant design information and A-46 walkdown analysis and information is used to evaluate whether a component is seismically rugged or requires a fragility calculation.

Substantial changes to the Table 3-5 component list resulted from the seismic capacity walkdowns. In addition to significant additions from the A-46 list (instrumentation, power supplies and relays), significant deletions were the result of the determination of low capacity systems and components. Many non-safety systems such as feedwater, instrument air, TBCCW, etc. were removed from the list based on low capacity. These items were guaranteed failed in the risk model for all seismically initiated events (see Section 3.1.5.1).

The list of components with their associated fragility values is presented in Table 3-6. Table 3-6 contains columns for the component description, plant designator, OCPRA top event, category, assigned fragility, and fragility parameters (A_g, B_R and B_U). The components in this table are then "mapped" to the Level 1 PRA top events using the process described in Section 3.1.6.1.

3.1.4.2 Evaluation of Essential Relay Fragilities

The Seismic Qualification Utility Group (SQUG) require that a relay evaluation be performed on the electrical control circuits for the equipment on the Safe Shutdown Equipment List (SSEL). The purpose of the evaluation is to determine if the plant safe shutdown systems or equipment could be adversely affected by relay malfunction in the event of a Safe Shutdown Earthquake (SSE). For relays which seismic malfunction is unacceptable, the seismic adequacy is evaluated.

Overview of Relay Evaluation Methodology

The SQUG (A-46) program developed a methodology to evaluate and document the results of the relay evaluation. The IPEEE relay evaluation followed the same methodology as recommended in NUREG-1407. This methodology is described in two documents: Section 6 of the "Generic Implementation Procedure", Revision 2; and EPRI NP-7148-SL titled "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality". The methodology is performed in three steps. The first is the Relay Circuit Evaluation, the second the Relay Seismic Adequacy Evaluation and the third is the Relay Walkdown.

Relay Circuit Evaluation

The safe shutdown equipment list (SSEL) identifies the minimum plant systems and equipment needed to maintain the plant in a safe shutdown condition during and following an SSE. The Relay Circuit Evaluation identifies the minimum required relays to support operation of these systems and equipment. A "Relay Review List" is developed which contains all the components on the composite Safe Shutdown Equipment List (SSEL) identified with "Y" in the relay review field, indicating that a relay review is required.

The associated circuits are identified and evaluated to determine if relay malfunction (relay chatter) is acceptable. Relays whose malfunction is acceptable (relays whose chatter does not impact system function) need not be seismically rugged. Relays whose chatter is determined to be unacceptable are termed "essential relays". In the case of essential relays, seismic adequacy must be determined. The relay circuit evaluation was performed by a GPU Nuclear Engineer familiar with the design and operation of Oyster Creek.

Relay Seismic Adequacy Evaluation

The seismic adequacy of essential relays can be determined by comparing the relays seismic capacity to the seismic demand imposed upon the relay. The seismic capacity of the relay can be established using the Generic Equipment Ruggedness Spectra (GERS), Earthquake Experience Data or plant specific or relay specific seismic test data. If the seismic adequacy of an essential relay cannot be determined, then the relay is identified as requiring corrective action. This corrective action can include seismic testing of the relay type or replacement of the existing relay with a relay of known acceptable ruggedness.

Relay Walkdown

Relay walkdowns were conducted by seismic capacity engineers. The relay walkdowns were performed to achieve four objectives:

(1) Obtain required information for use in the calculation of in-cabinet/ enclosure amplification.

(2) Verify the adequacy of the anchorage of cabinets and enclosures which contain essential relays.

(3) Check relay mountings, orientation, potential interactions and attached cable slack.

(4) Check relay models and locations.

All SQUG cabinets and enclosures containing SQUG essential relays have been walked down.

Results of the A-46/IPEEE Relay Evaluation

All essential relays in the A-46 program will have their seismic adequacy determined. For those relays which do not meet the requirements (i.e., seismic demand exceeds capacity) or whose seismic adequacy cannot be determined will be replaced with relays whose capacity exceeds the seismic demand.

In the case of the IPEEE, which is a beyond design basis study, all essential relays are examined for potential relay chatter related issues. Table 3-7, Electrical Panel Fragilities Based on Relay Function, is presented at the end of this report section. The lowest functional fragility provided for each Electrical Panel is used as the seismic capacity of the panel. Only those electrical panels whose seismic capacity, as defined by relay function, is less than 1.0g are included in the risk model. These are:

- DG-1 Diesel Generator Control Cabinet (Top Event DW)
- DG-2 Diesel Generator Control Cabinet (Top Event DW)
- Panel ER-18A (Containment Spray Logic) (Top Event CX)
- Panel ER-18B (Containment Spray Logic) (Top Event CX)
- Panel ER-8A (Core Spray Logic) (Top Event CZ)
- Panel ER-8B (Core Spray Logic) (Top Event CZ)

These relay failures (relay chatter) are assumed to fail the systems with which they are associated. No recovery is modeled. The A-46 and IPEEE relay lists are identical with the exception that A-46 does not require a relay evaluation of containment isolation function. Many reviews are conducted on the basis of regulatory guide 1.97 which reviews the availability of position indication for containment isolation but not the automatic function. In addition to the availability of position indication A-46 relay list does include the following systems:

- Reactor Water Cleanup System Isolation (RWCU)
- Main Steam Isolation Valves (MSIVs)
- Scram Discharge Volume Isolation (SDV)

These three systems comprise the major containment isolation pathways and are the only significant contributors (of non-LOCA contributors) to the large early release fraction in the base Level 1 and 2 PRAs. Based on the above, the seismic risk model reflects the manual closure of containment isolation valves (assumes relay chatter) at top event PI. This treatment is considered conservative since containment isolation valves fail closed on loss of power or instrument air and instrument air failure is assumed for all seismic initiators due to the low capacity of the air system.

3.1.4.3 Evaluation of Masonry Block Walls

This report sub-section summarizes the seismic evaluation of block walls which was performed in support this study (Reference 3-7). A brief description of the evaluation methodology and the results are provided.

Evaluation Methodology

The masonry block walls to be considered in the seismic IPEEE evaluation include walls which either support or are in the proximity of IPEEE equipment. Based on a review of the components on the IPEEE equipment list, it was concluded that only the masonry block walls that affect safety related equipment need to be considered in the IPEEE evaluation. The few non-safety related components on the IPEEE equipment list are not affected by masonry walls. Therefore, the safety related masonry walls were identified based on the results of the I&E 80-11 program. The walls considered here include the 24 walls evaluated as part of the I&E 80-11 program, with the addition of Wall No. 2. Wall No. 2 was removed from the I&E 80-11 scope because a unistrut bracing system was added which prevented the wall from collapsing into the control room panels. This wall was included in the IPEEE evaluation because the capacity of the unistrut members at beyond design basis seismic loading could not be assessed on judgment alone. Wall No. 15 was removed from further review since the I&E 80-11 modifications included the removal of the portion of the wall affecting safety related equipment and replacing it with drywall construction. The remaining portion of Wall No. 15 could collapse without impacting any safety related equipment.

Block wall walkdowns were not conducted. Rather than attempting to identify the specific IPEEE equipment and components affected by the masonry walls, all of the walls were assumed to affect IPEEE components. The evaluation approach focused on screening out seismically rugged walls. The screening process identified walls whose seismic capacities are sufficiently high such that they do not significantly contribute to the seismic risk. As a result, the screened out walls do not require detailed fragility analyses.

The screening approach involved selecting a set of bounding walls, representing the most vulnerable masonry walls. The bounding walls were selected based on a review of the parameters most significant to the wall seismic capacity. These parameters included: wall location, construction (running bond versus stack bond), wall thickness, maximum horizontal and vertical clear spans, reinforcement, design basis factor of safety, and structural modifications. Conservative seismic evaluations were performed for the bounding walls and were used as benchmarks. That is, if the bounding walls could all be screened out, then the other less vulnerable walls are also concluded to be screened out.

Conservative seismic capacity evaluations for the bounding walls were performed following the Seismic Margin Assessment (SMA) methodology described in EPRI NP-6041-SL, Rev. 1. The SMA approach is based on calculating High-Confidence-of-Low-Probability-of-Failure (HCLPF) capacities. The SMA screening was based on a review level earthquake with a ground response spectrum having a peak 5% damped spectral acceleration of 0.89g. Median-centered in-structure response spectra consistent with the review level earthquake were obtained by scaling the conservative design basis spectra developed for the USI A-46 program. The spectra scaling followed the guidance of EPRI NP-6041-SL, Rev.1. Conservative deterministic capacities were evaluated for the bounding walls. For reinforced walls, the recommendations of Appendix R of EPRI NP-6041-SL were used to calculate the out-of-plane capacity. For unreinforced walls, conservative masonry stress capacities were based on the allowable stresses of ACI 530-92 increased by the appropriate factors from Appendix A of Section 3.8.4 of the USNRC Standard Review Plan (NUREG-0800).

Walls that satisfied the SMA acceptance criteria were concluded to have HCLPF capacities which exceed the review level earthquake and, thus, could be screened from further fragility evaluation and from the PRA. Walls that could not be screened were subject to fragility evaluation.

Bounding Walls

The bounding walls were selected primarily on the basis of the lowest design basis margins, the highest span-to-thickness ratios, and locations higher in the structures. The bounding walls selected for evaluation were:

- Wall No. 2: Reinforced, removed from I&E 80-11 with the addition of a unistrut restraint.
- Wall No. 8: Unreinforced, stack bond wall with the lowest design basis margin.
- Wall No. 20: Unreinforced, stack bond wall with a low design basis margin.
- Wall No. 26: Unreinforced wall, no modifications, low design basis margin.
- Wall No. 29: Reinforced wall, lowest design basis margin of reinforced walls.
- Wall No. 53: Unreinforced wall with door opening.

Evaluation of Masonry Wall Results

The results of the evaluations of the bounding walls are summarized below:

Wall No. 2

This reinforced wall is controlled by one-way flexure spanning in the vertical direction. The flexural yield capacity, as calculated by the recommendations of Appendix R of EPRI NP-6041-SL, exceeded the review level earthquake SMA demand response. This was without consideration of the unistrut restraint system. The wall was screened out.

Wall No. 8

This unreinforced wall is controlled by one-way flexure spanning in the vertical direction. The wall structural modifications reduced the maximum vertical span to 7.79'. The wall satisfied the flexural capacity criterion for flexural tension normal to the bedjoint. The wall was screened out.

Wall No. 20

Similar to wall 8, this unreinforced wall is controlled by one-way flexure spanning in the vertical direction. The wall structural modifications reduced the maximum vertical span to 7.67'. The wall satisfied the flexural capacity criterion for flexural tension normal to the bedjoint and was screened out.

Wall No. 26

This unreinforced wall resists loads in two-way action. The critical stress corresponded to flexural tension normal to the bedjoint. The wall satisfied the

SMA acceptance criteria and was screened out.

Wall No. 29

This reinforced wall resists loads in two-way action, with the horizontal span controlling. The horizontal reinforcing consists of extra-heavy Dur-O-Wall joint reinforcing placed at every course. The horizontal wall reinforcing provides adequate flexural capacity to resist the SMA screening level earthquake. The wall was screened out.

Wall No. 53

This unreinforced wall carries loads in two-way action. The critical stress corresponded to flexural tension normal to the bedjoint at approximately midheight of the wall at the door opening. This wall did not satisfy the SMA acceptance criteria. Therefore, the seismic fragility of the wall was evaluated in which the capacity of the wall was based on initial cracking. This was judged to be conservative, since some out-of-plane displacement beyond the initial cracking is required to fail any attached equipment or conduit or to cause collapse of the wall. For the fragility evaluation, the wall seismic demand was based on the IPEEE probabilistic response analyses, which were anchored to the EPRI uniform hazard spectrum. The median peak ground acceleration capacity, associated logarithmic standard deviations, the High Confidence of Low Probability of Failure (HCLPF) acceleration capacity are:

$$\hat{A} = 0.77g$$

$$\beta_R = 0.28$$

$$\beta_U = 0.36$$

$$\text{HCLPF} = 0.77g \exp(-1.65 * (0.28 + 0.36)) = 0.27g$$

In the equipment seismic fragility evaluation, components having median acceleration capacities of 1.0g and HCLPF capacities of 0.30g were screened out on the basis of having capacities sufficiently high such that they do not significantly contribute to the risk. Although the HCLPF of wall No. 53 is extremely close to the screening level HCLPF, Wall No. 53 was not screened out.

In summary, all of the bounding walls selected for evaluation were screened out with the exception of Wall No. 53. Five of the walls met the SMA acceptance criteria. A fragility evaluation was performed for one wall. This wall was not screened out as its HCLPF capacity was slightly lower than the fragility screening level HCLPF capacity. Since all of the bounding walls with the exception of Wall No. 53 were screened out, it was concluded that all of the safety related masonry walls are screened out of the seismic PRA with this the singular exception of Wall No. 53. See Section 3.1.5, Top Event DX.

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
NZ-04A	C.S. SYS I FILL PUMP	RB	-19-6	Yes	6				
NZ-04B	C.S. SYS II FILL PUMP	RB	-19-6	Yes	6				
P-20-001A	C.S. SYS I PUMP (NZ01-A)	RB	-19-6	No	4	0.82	0.40	0.35	0.24
P-20-001B	C.S. SYS II PUMP (NZ01-B)	RB	-19-6	No	4	0.82	0.40	0.35	0.24
P-20-001C	C.S. SYS I PUMP (NZ01-C)	RB	-19-6	No	4	0.82	0.40	0.35	0.24
P-20-001D	C.S. SYS II PUMP (NZ01-D)	RB	-19-6	No	4	0.82	0.40	0.35	0.24
PIT-RV-0003A	C.S. SYS I PMP PRESS XMTR	RB	-19-6	Yes	6				
PIT-RV-0003B	C.S. SYS II PMP PRESS XMTR	RB	-19-6	Yes	6				
V-20-0004	C.S. SYS II 1B SUCT ISO VLV	RB	-19-6	Yes	6				
V-20-0033	C.S. SYS II PMP 1D SUCT ISO VLV	RB	-19-6	Yes	6				
PS-RV0029A	PS RV29A-Sys.I Booster Pumps	RB	-19-6	Yes	6				
PS-RV0029B	PS RV29B-Sys.II Booster Pumps	RB	-19-6	Yes	6				
PS-RV0029C	PS RV29C-Sys.I Booster Pumps	RB	-19-6	Yes	6				
PS-RV0029D	PS RV29D-Sys.II Booster Pumps	RB	-19-6	Yes	6				
V-20-0003	Pump Suction MOV V-20-3	RB	-19-6	Yes	6				
V-20-0032	Pump Suction MOV V-20-32	RB	-19-6	Yes	6				
P-21-001A	CNTMNT SPRY PMP (51A)	RB	-19-6	No	4	1.48	0.40	0.35	0.43
P-21-001B	CNTMNT SPRY PMP (51B)	RB	-19-6	No	4	1.48	0.40	0.35	0.43
P-21-001C	CNTMNT SPRY PMP (51C)	RB	-19-6	No	4	1.48	0.40	0.35	0.43
P-21-001D	CNTMNT SPRY PMP (51D)	RB	-19-6	No	4	1.48	0.40	0.35	0.43
V-21-0001(LOCAL)	KL LOCAL KEYLOCK SW	RB	-19-6	Yes	6				
V-21-0003(LOCAL)	KL LOCAL KEYLOCK SW	RB	-19-6	Yes	6				
V-21-0007(LOCAL)	KL LOCAL KEYLOCK SW	RB	-19-6	Yes	6				
V-21-0009(LOCAL)	KL LOCAL KEYLOCK SW	RB	-19-6	Yes	6				
V-21-0001	Pump 51C Suction Valve V-21-1	RB	-19-6	Yes	6				
V-21-0003	Pump 51D Suction Valve V-21-3	RB	-19-6	Yes	6				
V-21-0007	Pump 51B(1-2) Suction Valve V 21-7	RB	-19-6	Yes	6				
V-21-0009	Pump 51A(1-1) Suction Valve V-21-9	RB	-19-6	Yes	6				
V-23-0021	DW N2 RELIEF VENT VLV	RB	-19-6	Yes	6				
V-23-0022	DW N2 RELIEF VENT VLV	RB	-19-6	Yes	6				
LT-0037	TORUS WIDE RANGE LVL XMTR	RB	-19-6	Yes	6				
LT-0038	TORUS WIDE RANGE LVL XMTR	RB	-19-6	Yes	6				
TE-664-0030A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0030B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				
TE-664-0031A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0031B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				
TE-664-0032A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0032B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				
TE-664-0033A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0033B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
TE-664-0034A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0034B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				
TE-664-0035A	TORUS WATER TEMP CH. A	RB	-19-6	Yes	6				
TE-664-0035B	TORUS WATER TEMP CH. B	RB	-19-6	Yes	6				
V-27-0001	RB-DW VENT ISO BFLY VLV (NC)	RB	-19-6	Yes	6				
V-27-0002	RB-DW VENT ISO BFLY VLV (NC)	RB	-19-6	Yes	6				
V-28-0017	TORUS VENT EXHAUST BFLY VLV	RB	-19-6	Yes	6				
V-28-0018	TORUS VENT EXHAUST BFLY VLV	RB	-19-6	Yes	6				
V-28-0047	V-28-0017 BYPASS VLV	RB	-19-6	Yes	6				
V-6-0591	PILOT SOLENOID AIR SUPPLY FOR V-27-0001	RB	-19-6	Yes	6				
V-6-0592	PILOT SOLENOID AIR SUPPLY VALVE FOR V-27-0002	RB	-19-6	Yes	6				
V-6-0596	PILOT SOLENOID FOR V-28-0018	RB	-19-6	Yes	6				
P-15-001A	CONTROL ROD DRIVE PUMP (NC08A)	RB	-1-11	Yes	2				
P-15-001B	CONTROL ROD DRIVE PUMP (NC08B)	RB	-1-11	Yes	2				
V-15-0045	V-15-45 Relief Valve	RB	-1-11	Yes	6				
V-15-0046	V-15-46 Relief Valve	RB	-1-11	Yes	6				
P-20 002B	C.S. SYS II BOOSTER PMP (NZ03-B)	RB	23-6	Yes	2				
P-20-002D	C.S. SYS II BOOSTER PMP (NZ03-D)	RB	23-6	Yes	2				
V-20-0093	C.S. SYS II MIN FLOW BLOCK VLV (FO)	RB	23-6	Yes	6				
V-20-0095	C.S. SYS II MIN FLOW BLOCK VLV (FO)	RB	23-6	Yes	6				
V-20-0024	C.S. SYS II PRESS RELIEF VLV	RB	23-6	Yes	6				
DPS-RV0040B	Differential Pressure Switch RV40B	RB	23-6	Yes	6				
DPS-RV0040D	Differential Pressure Switch RV40D	RB	23-6	Yes	6				
SDIV(N)	SCRAM DISCH INST VOL (N)	RB	23-6	Yes	6				
SDIV(S)	SCRAM DISCH INST VOL (S)	RB	23-6	Yes	6				
SO-305-0117	SCRAM AIR PILOT SOL VLV	RB	23-6	Yes	6				
SO-305-0118	SCRAM AIR PILOT SOL VLV	RB	23-6	Yes	6				
SO-305-0120	FLOW CONTROL WITHDRAWL SOL	RB	23-6	Yes	6				
SO-305-0121	FLOW CONTROL INSERT SOL	RB	23-6	Yes	6				
SO-305-0122	FLOW CONTROL WITHDRAWL SOL	RB	23-6	Yes	6				
SO-305-0123	FLOW CONTROL INSERT SOL	RB	23-6	Yes	6				
V-15-0119	N. SCRAM DISC HDR VENT VLV	RB	23-6	Yes	6				
V-15-0120	S. SCRAM DISC HDR VENT VLV	RB	23-6	Yes	6				
V-15-0121	S. SCRAM DISC VOL DRN VLV	RB	23-6	Yes	6				
V-15-0133	N. SCRAM DISC VOL DRN VLV	RB	23-6	Yes	6				
V-15-0134	S. SCRAM DISC VOL DRN VLV	RB	23-6	Yes	6				
V-15-0135	N. SCRAM DISC VOL DRN VLV	RB	23-6	Yes	6				
V-15-0136	N. SCRAM DISC VOL HDR VENT VLV	RB	23-6	Yes	6				
V-15-0137	S. SCRAM DISC VOL HDR VENT VLV	RB	23-6	Yes	6				
HCU-305-02-19	HYDRAULIC CONTROL UNIT (ALL 137 UNITS)	RB	23-6	Yes	2				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
FT-IP0003A	CNTMNT SPRY FLOW XMTR	RB	23-6	Yes	6				
FT-IP0003B	CNTMNT SPRY FLOW XMTR	RB	23-6	Yes	6				
H-21-001A	CNTMNT SPRAY HEAT EXCHANGER	RB	23-6	Yes	2				
H-21-001B	CNTMNT SPRAY HEAT EXCHANGER	RB	23-6	Yes	2				
H-21-001C	CNTMNT SPRAY HEAT EXCHANGER	RB	23-6	Yes	2				
H-21-001D	CNTMNT SPRAY HEAT EXCHANGER	RB	23-6	Yes	2				
V-21-0005	CNTMNT SPRY DISCH TO SPRY HDR II MOV	RB	23-6	Yes	6				
V-21-0005(LOCAL)	KL LOCAL KEYLOCK SW	RB	23-6	Yes	6				
V-21-0013	CNTMNT SPRY DYNAMIC TEST VALVE (SYSTEM II)	RB	23-6	Yes	3				
V-21-0013(LOCAL)	KL LOCAL KEYLOCK SW	RB	23-6	Yes	6				
V-21-0015	CNTMNT SPRY TO TORUS SPRY MOV	RB	23-6	Yes	6				
V-21-0015(LOCAL)	KL LOCAL KEYLOCK SW	RB	23-6	Yes	6				
V-21-0017	CNTMNT SPRY DYNAMIC TEST VALVE (SYSTEM I)	RB	23-6	Yes	6				
V-21-0017(LOCAL)	KL LOCAL KEYLOCK SW	RB	23-6	Yes	6				
V-21-0018	CNTMNT SPRY TO TORUS SPRY MOV	RB	23-6	Yes	6				
V-21-0018(LOCAL)	KL LOCAL KEYLOCK SW	RB	23-6	Yes	6				
V-21-0021	CNTMNT SPRY HX-A PRESS RELIEF VLV	RB	23-6	Yes	6				
V-21-0022	CNTMNT SPRY HX-B PRESS RELIEF VLV	RB	23-6	Yes	6				
V-21-0023	CNTMNT SPRY HX-C PRESS RELIEF VLV	RB	23-6	Yes	6				
V-21-0024	CNTMNT SPRY HX-D PRESS RELIEF VLV	RB	23-6	Yes	6				
V-23-0015	N2 PURGE INLET PRESS CONT BFLY VLV	RB	23-6	Yes	6				
V-23-0016	N2 PURGE INLET PRESS CONT BFLY VLV	RB	23-6	Yes	6				
V-23-0019	N2 MAKEUP PURGE INLET CONT VLV	RB	23-6	Yes	6				
V-23-0020	N2 MAKEUP PURGE INLET CONT VLV	RB	23-6	Yes	6				
V-26-0016	TORUS TO RB VACUUM RELIEF	RB	23-6	Yes	2				
V-26-0018	TORUS TO RB VACUUM RELIEF	RB	23-6	Yes	2				
V-38-0009	DW O2 ANLZR,CAPGRMS & PASS DRYWELL ISO VLV	RB	23-6	Yes	6				
V-1-0009	MS LN "A" OUTBRD ISO VALVE-NS04A	RB	23-6	Yes	11				
V-1-0010	MS LN "B" OUTBRD ISO VALVE-NS04B	RB	23-6	Yes	11				
RK-411-001	MSIV SOLENOID AIR VALVE MOUNTING RACK	RB	23-6	Yes	2				
V-3-0082	C.S. HX-A PRESS RELIEF VLV	RB	23-6	Yes	6				
V-3-0083	C.S. HX-B PRESS RELIEF VLV	RB	23-6	Yes	6				
V-3-0084	C.S. HX-C PRESS RELIEF VLV	RB	23-6	Yes	6				
V-3-0085	C.S. HX-D PRESS RELIEF VLV	RB	23-6	Yes	6				
V-5-0167	RBCCW DW OUTLET ISO	RB	23-6	Yes	6				
V-5-0147	RBCCW DW INLET ISO	RB	23-6	Yes	6				
LSP-1A2	USS 1A2 PMP/BRKR CONT PNL	RB	23-6	Yes	2				
LSP-1AB2	V-37-0054, V-17-0019 & 54 LOCAL CONT PNL	RB	23-6	Yes	2				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
PNL ER-56	125 VDC PNL F RELAY PNL	RB	23-6	Yes	2				
PNL ER-57	CIP NO. 3 RELAY PANEL	RB	23-6	Yes	6				
PNL ER-58	INSTR PNL 4 RELAY PNL	RB	23-6	Yes	6				
PNL ER-59	INSTR PNL IP-4A PWR LOST RLY PNL	RB	23-6	Yes	6				
PNL ER-62	VITAL AC PNL RLY PNL	RB	23-6	Yes	2				
PNL ER-71	MCC DC-1 PNL	RB	23-6	Yes	6				
ER-642-079	ESAS ACTU AUX RLY PNL	RB	23-6	Yes	6				
PNL ER18A	CORE SPRAY RLY LOGIC PNL	RB	23-6	Yes	2				
PNL ER18B	CORE SPRAY RLY LOGIC PNL	RB	23-6	Yes	2				
PNL ER8A	CNTNMNT SPRY SYS RLY LOGIC PNL	RB	23-6	No	3				
PNL ER8B	CNTNMNT SPRY SYS RLY LOGIC PNL	RB	23-6	No	3				
V-6-3169	ARI SYSTEM SCRAM PILOT AIR EXHAUST VALVE	RB	23-6	Yes	6				
V-6-3170	ARI SYSTEM SCRAM PILOT AIR EXHAUST VALVE	RB	23-6	Yes	6				
V-6-3237	ALTERNATE ROD INJECTION SYSTEM VENT VALVE	RB	23-6	Yes	6				
V-6-3171	ARI SYSTEM SCRAM PILOT AIR EXHAUST	RB	23-6	Yes	6				
V-6-3176	ARI SYSTEM SCRAM PILOT AIR EXHAUST VALVE	RB	23-6	Yes	6				
V-38-0010	DW O2 ANLZR,CAPGRMS & PASS DRYWELL ISO VLV	RB	23-6	Yes	6				
V-38-0016	CAPGRMS SAMPLE RETURN TO TORUS ISOLATION VLV	RB	23-6	Yes	6				
V-38-0017	CAPGRMS SAMPLE RETURN TO TORUS ISOLATION VLV	RB	23-6	Yes	6				
1A2-460V-MCC	MCC 1A2 460V FOR RX BLDG	RB	23-6	Yes	1, 3				
1A2-460V-USS	460V USS 1A2 FOR RX BLDG	RB	23-6	Yes	1, 2				
1A2-XFMR-USS	USS 1A2-460V XFMR	RB	23-6	Yes	2				
1A21-460V-MCC	MCC 1A21 460V FOR RX BLDG	RB	23-6	Yes	1, 3				
1A21A-460V-MCC	MCC 1A21A 460V FOR RX BLDG	RB	23-6	Yes	3				
1A21B-460V-MCC	MCC 1A21B 460V FOR RX BLDG	RB	23-6	Yes	3				
1A23-460V-MCC	MCC 1A23 460V FOR RX BLDG	RB	23-6	Yes	3				
1AB2-460V-MCC	MCC 1AB2 460V FOR RX BLDG	RB	23-6	Yes	2				
1B2-460V-MCC	MCC 1B2 460V FOR RX BLDG	RB	23-6	Yes	1, 3				
1B2-460V-USS	460V USS 1B2 FOR RX BLDG	RB	23-6	Yes	1, 2				
1B2-XFMR-USS	USS 1B2-460V XFMR	RB	23-6	Yes	1				
1B21-460V-MCC	MCC 1B21 460V FOR RX BLDG	RB	23-6	Yes	1, 2				
1B21A-460V-MCC	MCC 1B21A 460V FOR RX BLDG	RB	23-6	Yes	3				
1B21B-460V-MCC	MCC 1B21B 460V FOR RX BLDG	RB	23-6	Yes	3				
ER-732-092	USS 1A2-460V XFMR CLG FANS CONT	RB	23-6	Yes	2				
ER-732-093	USS 1B2-460V XFMR CLG FANS CONT	RB	23-6	Yes	6				
CIP-3	CONTINUOUS INST PNL NO. 3	RB	23-6	Yes	2				
IP-4	120V AC INST PNL 4	RB	23-6	Yes	2				
IP-4A	208/120V AC INST PNL 4A	RB	23-6	Yes	6				
IP-4B	120V AC INST PNL 4B	RB	23-6	Yes	6				
IP-4C	120V AC INST PNL 4C	RB	23-6	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
IT-3 SW	AUTO XFER SW FOR CIP NO. 3	RB	23-6	Yes	2				
IT-3 XFMR	XFMR FOR MCC 1A2-460V TO CIP-3	RB	23-6	Yes	2				
IT-4	AUTO XFER SW FOR IP NO. 4	RB	23-6	Yes	2				
IT-4A	460/120 XFMR FOR IP NO. 4	RB	23-6	Yes	6				
IT-4B	460/120 XFMR FOR IP NO. 4	RB	23-6	Yes	2				
VACP-1	120/208 VITAL AC PWR PNL	RB	23-6	Yes	2				
VACP-1 XFMR	120V AC VITAL PWR PNL XFMR	RB	23-6	Yes	3				
VACP-1 AUTO XFER SW	AUTO TRANSFER SW FOR VACP-1	RB	23-6	Yes	2				
VACP-1 DISCN SWITCH	DISCONNECT SW FOR VACP-1	RB	23-6	Yes	2				
DC-F DSTPL	125V DC POWER PANEL "F"	RB	23-6	Yes	2				
DC-1 125V DC	MCC DC-1 125V DC AUTO XFER SW	RB	23-6	Yes	6				
DC-1 125V MCC	125V DC ISO VLV'S MTR CONT CNTR	RB	23-6	Yes	3				
SW-1B2	MANUAL THROWOVER 125V DC SW TO USS-1B2	RB	23-6	Yes	6				
V-6-0395	INST AIR OUTBD DW ISO VLV	RB	23-6	Yes	6				
V-6-0448	S.D.V. VALVES-PILOT SOL VLV (NC54-A)	RB	23-6	Yes	6				
V-6-0449	S.D.V. VALVES-PILOT SOL VLV (NC54-B)	RB	23-6	Yes	6				
V-6-0450	INSTRUMENT AIR SUPPLY VALVE(NC56-A) FOR V-15-0134	RB	23-6	Yes	6				
V-6-0451	INSTRUMENT AIR SUPPLY VALVE(NC56-B) FOR V-15-0134	RB	23-6	Yes	6				
V-6-0990	V-26-0016 PILOT OPERATOR	RB	23-6	Yes	6				
V-6-0991	V-26-0018 PILOT OPERATOR	RB	23-6	Yes	6				
V-6-2005	S.D.V. DRAIN-PILOT SOL VLV	RB	23-6	Yes	6				
V-6-2001	S.D.V. DRAIN-PILOT SOL VLV	RB	23-6	Yes	6				
V-6-2000	S. SCRAM DISC VOL VENT VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2004	S. SCRAM DISC VOL VENT VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2002	S. SCRAM DISC VOL DRN VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2006	S. SCRAM DISC VOL DRN VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2680	MSIV4A PILOT DC SOL VLV	RB	23-6	Yes	6				
V-6-2681	MSIV4A PILOT AC SOL VLV	RB	23-6	Yes	6				
V-6-2684	MSIV4B PILOT DC SOL VLV	RB	23-6	Yes	6				
V-6-2685	MSIV4B PILOT AC SOL VLV	RB	23-6	Yes	6				
V-6-2916	N. SCRAM DISC VOL VENT VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2003	N. SCRAM DISC VOL VENT VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2917	N. SCRAM DISC VOL DRN VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-2918	N. SCRAM DISC VOL DRN VLV PILOT S.V.	RB	23-6	Yes	6				
V-6-0431	PILOT SOLENOID AIR SUPPLY VALVE FOR V-20-0095	RB	23-6	Yes	6				
V-6-0995	PILOT SOLENOID AIR SUPPLY VALVE FOR V-20-0093	RB	23-6	Yes	6				
MCL-823-008	FN-56-0004 LOCAL MTR CONT PNL	RB	23-6	Yes	2				
MCL-823-009	FN-56-0007 LOCAL MTR CONT PNL	RB	23-6	Yes	6				
PNL ER4	SRM & IRM DC CONT RLY PNL	RB	33-5	Yes	6				
ER-215-087	CU DEMIN AUX RLY PNL	RB	35-0	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
MCL-215-007	DC LOCAL MOTOR STARTER FOR V-16-0014	RB	35-0	Yes	6				
MCL-215-006	DC LOCAL MOTOR STARTER FOR V-16-0002	RB	35-0	Yes	6				
ROTARY INVERTER CONT PNL	LOCAL CONTROL PANEL-ROTARY INVERTER	RB	35-0	Yes	3				
PNL ER-63	125 VDC PWR PNL RLY PNL	RB	35-0	Yes	6				
ROTARY INVERTER	120 VAC SPLY FOR CIP NO. 3	RB	35-0	Yes	3				
MG SET B	MAIN BATT CHRGR FOR DISTR CNTR "B" 125V DC	RB	35-0	Yes	2				
STATIC CHARGER	STANDBY STATIC CHARGER FOR A/B BATT	RB	35-0	Yes	3				
BATTERY BANK B	DIVISION B 125V BATT BANK	RB	35-0	Yes	2				
DC-B-125V	125V DC DISTRIBUTION CNTR B	RB	35-0	Yes	1				
DC-D-125V DC	AUTO XFER SW FOR 125V DC PWR PNL "D"	RB	35-0	Yes	1				
DC-D-DSTPL	125V DC POWER PANEL "D"	RB	35-0	Yes					
V-22-0001	HP-T-001B RW ISO VLV	RB	38-0	Yes	6				
V-22-0002	HP-T-001B RW ISO VLV	RB	38-0	Yes	6				
V-22-0028	DW SUMP PMPS DISCH CONT VLV	RB	38-0	Yes	6				
V-22-0029	DW SUMP PMPS DISCH CONT VLV	RB	38-0	Yes	6				
ER-642-078	ESAS ACTU AUX RLY PNL	RB	46-6	Yes	6				
PNL ER73	RX HEAD VENT REMOTE CONT PNL	RB	51-3	Yes	6				
DPIS-IB05 A1	IC A isolation steamline pressure switch 1	RB	51-3	Yes	6				
DPIS-IB05 A2	IC A isolation steamline pressure switch 2	RB	51-3	Yes	6				
DPIS-IB05 B1	IC B isolation steamline pressure switch 1	RB	51-3	Yes	6				
DPIS-IB05 B2	IC B isolation steamline pressure switch 2	RB	51-3	Yes	6				
DPIS-IB11 A1	IC A isolation condensate line PS 1	RB	51-3	Yes	6				
DPIS-IB11 A2	IC A isolation condensate line PS 1	RB	51-3	Yes	6				
DPIS-IB11 B1	IC B isolation condensate line PS 1	RB	51-3	Yes	6				
DPIS-IB11 B2	IC B isolation condensate line PS 1	RB	51-3	Yes	6				
FIT-RV0026A	C.S. SYS I FLOW XMTR	RB	51-3	Yes	6				
FIT-RV0026B	C.S. SYS II FLOW XMTR	RB	75-3	Yes	6				
V-20-0026	RECIRCULATION TEST VALVE TO TORUS (SYSTEM II)	RB	75-3	Yes	6				
MCL-732-012	V-20-0026 LOCAL MTR CONTROL	RB	51-3	Yes	6				
V-20-0026(LOCAL)	LOCAL KL PB SW	RB	75-3	Yes	6				
P-20-002A	C.S. SYS I BOOSTER PMP (NZ03-A)	RB	51-3	Yes	2				
P-20-002C	C.S. SYS I BOOSTER PMP (NZ03-C)	RB	51-3	Yes	2				
V-20-0025	C.S. SYS I PRESS RELIEF VLV	RB	51-3	Yes	6				
V-20-0027	RECIRCULATION TEST VALVE TO TORUS (SYSTEM I)	RB	51-3	Yes	6				
MCL-732-011	V-20-0027 LOCAL MTR CONTROL	RB	51-3	Yes	6				
V-20-0027(LOCAL)	LOCAL KL PB SW	RB	51-3	Yes	6				
V-20-0015	C.S. SYS I PARALLEL ISO VLV	RB	51-3	Yes	6				
V-20-0040	C.S. SYS I PARALLEL ISO VLV	RB	51-3	Yes	6				
V-20-0092	C.S. SYS I MIN FLOW BLOCK VLV	RB	51-3	Yes	6				
V-20-0094	C.S. SYS I MIN FLOW BLOCK VLV	RB	51-3	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
DPS-RV0040A	Differential Pressure Switch RV40A	RB	51-3	Yes	6				
DPS-RV0040C	Differential Pressure Switch RV40C	RB	51-3	Yes	6				
V-20-0012	Discharge MOV V-20-12	RB	51-3	Yes	6				
V-16-0002	INLET ISO VLV TO RWCU PMP	RB	51-3	Yes	6				
V-16-0001	RWCU INLET ISO VLV (INBOARD)	DW	46-0	Yes	6				
V-16-0014	RWCU INLET ISO VLV (OUTBOARD)	RB	51-3	Yes	6				
V-16-0061	RWCU REGNERATIVE HX OUTLET VLV TO RPV	RB	51-3	Yes	6				
V-21-0011	CNTMNT SPRY DISCH TO SPRY HDR I MOV	RB	51-3	Yes	6				
V-21-0011(LOCAL)	KL LOCAL KEYLOCK SW	RB	51-3	Yes	6				
PT-53	DRYWELL PRESS CH. A (WIDE RANGE)	RB	51-3	Yes	6				
PT-54	DRYWELL PRESS CH. B (WIDE RANGE)	RB	51-3	Yes	6				
PS-RV0046A	Drywell Pressure Switch RV46A	RB	51-3	Yes	6				
PS-RV0046B	Drywell Pressure Switch RV46B	RB	51-3	Yes	6				
PS-RV0046C	Drywell Pressure Switch RV46C	RB	51-3	Yes	6				
PS-RV0046D	Drywell Pressure Switch RV46D	RB	51-3	Yes	6				
RK01	RPS INSTRUMENT RACK 1A	RB	51-3	Yes	2				
RK02	RSP INSTRUMENT RACK 1B	RB	51-3	Yes	2				
RK03	RPS INSTRUMENT RACK RECIRC PMP	RB	51-3	Yes	2				
TE-59-0002B	RPV WTR LVL (FUEL ZONE) CH.B	RB	51-3	Yes	6				
TE-57-0002A	RPV WTR LVL (FUEL ZONE) CH.A	RB	51-3	Yes	6				
TE-56-0001A	RPV WTR LVL (FUEL ZONE) CH.A	DW	51-3	Yes	6				
TE-58-0001B	RPV WTR LVL (FUEL ZONE) CH.B	DW	51-3	Yes	6				
DPT-6-IA0091B	RPV WTR LVL (FUEL ZONE) CH.B	RB	51-3	Yes	6				
DPT-7-IA0090B	RPV WTR LVL (FUEL ZONE) CH.B	RB	51-3	Yes	6				
PS-IA0083A	Reactor Pressure Switch PS-IDA83A	RB	51-3	Yes	6				
PS-IA0083B	Reactor Pressure Switch PS-IDA83B	RB	51-3	Yes	6				
PS-IA0083C	EMRV NR108C HIGH PRESS SW	RB	51-3	Yes	6				
PS-IA0083D	EMRV NR108D HIGH PRESS SW	RB	51-3	Yes	6				
PS-IA0083E	EMRV NR108E HIGH PRESS SW	RB	51-3	Yes	6				
LIS-RE0018A	Mech Switch RE18A (Triple low)	RB	51-3	Yes	6				
LIS-RE0018B	Mech Switch RE18C (Triple low)	RB	51-3	Yes	6				
LIS-RE0018C	Mech Switch RE18C (Triple low)	RB	51-3	Yes	6				
LIS-RE0018D	Mech Switch RE18C (Triple low)	RB	51-3	Yes	6				
LT-RE0002A	Reactor Water Level Switch RE02A	RB	51-3	Yes	6				
LT-RE0002B	Reactor Water Level Switch RE02B	RB	51-3	Yes	6				
LT-RE0002C	Reactor Water Level Switch RE02C	RB	51-3	Yes	6				
LT-RE0002D	Reactor Water Level Switch RE02D	RB	51-3	Yes	6				
PT-RE0015A	Reactor Pressure Switch RE15A	RB	51-3	Yes	6				
PT-RE0015B	Reactor Pressure Switch RE15B	RB	51-3	Yes	6				
PT-RE0015C	Reactor Pressure Switch RE15C	RB	51-3	Yes	6				
PT-RE0015D	Reactor Pressure Switch RE15D	RB	51-3	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
PT-56-IA0092B	RX FUEL ZONE PRESS XMTR	RB	51-3	Yes	6				
PT-55-IA0092A	RX FUEL ZONE PRESS XMTR	RB	51-3	Yes	6				
ER-642-112	CORE SPRAY AUX RLY PNL	RB	51-3	Yes	6				
ER-642-113	CORE SPRAY AUX RLY PNL	RB	51-3	Yes	6				
ER-642-114	CORE SPRAY AUX RLY PNL	RB	51-3	Yes	6				
ER-642-115	CORE SPRAY AUX RLY PNL	RB	51-3	Yes	6				
PT-642-0009A	NARROW RANGE DRYWELL PRESSURE XMITTER	RB	51-3	Yes	6				
PT-642-0009B	NARROW RANGE DRYWELL PRESSURE XMITTER	RB	51-3	Yes	6				
V-28-0015	REACTOR BLDG VENTILATION ISOLATION INLET VALVE	RB	51-3	Yes	6				
V-28-0016	REACTOR BLDG VENTILATION ISOLATION VALVE	RB	51-3	Yes	6				
V-6-0430	AIR FLOW CONT S.V. TO V-20-0094	RB	51-3	Yes	6				
V-6-0996	AIR FLOW CONT S.V. TO V-20-0092	RB	51-3	Yes	6				
V-14-0030	STM INLET VLV TO "A" EMER CONDSR	RB	75-3	Yes	6				
V-14-0031	STM INLET VLV TO "A" EMER CONDSR	RB	75-3	Yes	6				
V-14-0032	STM INLET VLV TO "B" EMER CONDSR	RB	75-3	Yes	6				
V-14-0033	STM INLET VLV TO "B" EMER CONDSR	RB	75-3	Yes	6				
V-14-0034	EMER CONDSR "B" CONDSTE RTN VLV	RB	75-3	Yes	6				
V-14-0035	EMER CONDSR "A" CONDSTE RTN VLV	RB	75-3	Yes	6				
V-20-0018	C.S. SYS II BOOSTER PUMP DISCH	RB	75-3	Yes	6				
V-20-0021	C.S. SYS II PARALELL ISO VLV	RB	75-3	Yes	6				
V-20-0041	C.S. SYS II PARALELL ISO VLV	RB	75-3	Yes	6				
V-23-0013	N2 PURGE INLET PRESS CONT BFLY VLV	RB	75-3	Yes	6				
V-23-0014	N2 PURGE INLET PRESS CONT BFLY VLV	RB	75-3	Yes	6				
V-23-0017	N2 MAKEUP PURGE INLET CONT VLV	RB	75-3	Yes	6				
V-23-0018	N2 MAKEUP PURGE INLET CONT VLV	RB	75-3	Yes	6				
V-38-0037	OUTSIDE CONTAINMENT H2/O2 DW ISOLATION VLV	RB	75-3	Yes	6				
V-38-0038	OUTSIDE CONTAINMENT H2/O2 INLET ISOLATION VLV	RB	75-3	Yes	6				
V-38-0039	H2/O2 SYS A EXHAUST CONT ISOLATION VALVE	RB	75-3	Yes	6				
V-38-0040	H2/O2 SYS A EXHAUST CONT ISOLATION VALVE	RB	75-3	Yes	6				
V-38-0041	H2/02 SYS B DW SAMPLE CONTAINMENT ISOLATION VLV	RB	75-3	Yes	6				
V-38-0043	H2/O2 SYS B DW SAMPLE CONTAINMENT ISLATION VLV	RB	75-3	Yes	6				
V-38-0044	H2/O2 SYS B EXHAUST CONTAINMENT ISOLATION VLV	RB	75-3	Yes	6				
V-38-0046	H2/O2 SYS B EXHAUST CONTAINMENT ISOLATION VLV	RB	75-3	Yes	6				
DC-2 125V MCC	RX BLDG 125V DC MCC	RB	75-3	Yes	2				
V-27-0003	DW PURGE ISO BFLY VLV	RB	75-3	Yes	6				
V-27-0004	DW PURGE ISO BFLY VLV	RB	75-3	Yes	6				
CD-14-001A	ISOLATION CONDENSER (NE01A)	RB	95-3	No	4	0.98	0.40	0.35	0.28
CD-14-001B	ISOLATION CONDENSER (NE01B)	RB	95-3	No	4	0.98	0.40	0.35	0.28
V-14-0001	ISO COND B HI POINT VENT VLV	RB	95-3	Yes	6				
V-14-0005	ISO COND A HI POINT VENT VLV	RB	95-3	Yes	6				
V-14-0019	ISO COND B HI POINT VENT VLV	RB	95-3	Yes	6				
V-14-0020	ISO COND A HI POINT VENT VLV	RB	95-3	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
P-19-001A	STANDBY LIQUID CONTROL INJECTION PUMP (NP-02A)	RB	95-3	Yes	2				
P-19-001B	STANDBY LIQUID CONTROL INJECTION PUMP (NP-02B)	RB	95-3	Yes	2				
V-19-0044	SQUIB VALVE FOR TRAIN A (EV-NP05A)	RB	95-3	Yes	6				
V-19-0045	SQUIB VALVE FOR TRAIN B (EV-NP05B)	RB	95-3	Yes	6				
T-19-001	LIQUID POISON TANK (NP-01)	RB	95-3	No	4	3.10	0.40	0.35	0.90
FS-IL0006	FS 1L06 Flow Sensor To Isolate RWCU	RB	95-3	Yes	6				
V-19-0042	V-19-42 Pump A Discharge Relief Valve	RB	95-3	Yes	6				
V-19-0043	V-19-43 Pump B Discharge Relief Valve	RB	95-3	Yes	6				
V-11-0034	SHELL SIDE CONDENSATE MAKEUP TO ISO COND NE01B	RB	95-3	Yes	6				
V-11-0036	SHELL SIDE CONDENSATE MAKEUP TO ISO COND NE01A	RB	95-3	Yes	6				
DM-56-0079	EF-1-20 VORTEX DAMPER	RB	ROOF	Yes	6				
DM-56-0080	BATT/M-G EXHST DAMPER	RB	ROOF	Yes	6				
DM-56-0081	BATT/M-G RM HVAC RECIRC DMPR	RB	ROOF	Yes	6				
DM-56-0082	SF-1-20 VORTEX DAMPER	RB	ROOF	Yes	6				
DM-56-0083	BATT/M-G RM OUTSIDE AIR SPLY DMPR	RB	ROOF	Yes	6				
EF-1-0020	BATT & M-G RM EXHST FAN (FN-56-016)	RB	ROOF	No	10	0.30	0.40	0.35	0.09
SF-1-0020	BATT & M-G RM SPLY FAN (FN-56-017)	RB	ROOF	No	10	0.30	0.40	0.35	0.09
EF-1-0021	SWGR RM "B" EXHST FAN (FN-56-010)	RB	ROOF	Yes	2				
SF-1-0021	SWGR ROOM "B" SPLY FAN (FN-56-009)	RB	ROOF	Yes	2				
DM-56-0020	SPLY AIR CONT DMPR TO SWGR RM "B" (DE-1)	RB	ROOF	Yes	6				
DM-56-0023	EXHST AIR CONT DMPR FOR SWGR RM "B" (DE-3)	RB	ROOF	Yes	6				
DM-56-0024	RECIRC DMPR FOR SWGR RM "B" (DE-2)	RB	ROOF	Yes	6				
V-1-0007	MS LN "A" INBRD ISO VALVE-NS03A	DW	23-6	Yes	11				
V-1-0008	MS LN "B" INBRD ISO VALVE-NS03B	DW	23-6	Yes	11				
V-6-3306	MSIV3A PILOT DC SOL VLV	DW	23-6	Yes	6				
V-6-3307	MSIV3A PILOT AC SOL VLV	DW	23-6	Yes	6				
V-6-3310	MSIV3B PILOT DC SOL VLV	DW	23-6	Yes	6				
V-6-3311	MSIV3B PILOT AC SOL VLV	DW	23-6	Yes	6				
V-14-0036	ISOLATION COND CONDENSATE RETURN ISO VLV	DW	46-0	Yes	6				
V-14-0037	ISOLATION COND CONDENSATE RETURN ISO VLV	DW	46-0	Yes	6				
V-20-0150	C.S. SYS I INBRD ISO TESTABLE CHK VLV	DW	46-0	Yes	2				
V-20-0152	C.S. SYS I INBRD ISO TESTABLE CHK VLV	DW	46-0	Yes	2				
V-17-0019	S.C.S. INLET ISO VLV	DW	46-0	Yes	6				
V-17-0054	S.C.S. OUTLET ISO VLV	DW	46-0	Yes	6				
V-1-0173	ELECTROMATIC RELIEF VALVE-NR108A	DW	46-0	Yes	6				
V-1-0174	ELECTROMATIC RELIEF VALVE-NR108B	DW	46-0	Yes	6				
V-1-0175	ELECTROMATIC RELIEF VALVE-NR108C	DW	46-0	Yes	6				
V-1-0176	ELECTROMATIC RELIEF VALVE-NR108D	DW	46-0	Yes	6				
V-1-0177	ELECTROMATIC RELIEF VALVE-NR108E	DW	46-0	Yes	6				
V-5-0166	RBCCW DW ISO	DW	46-0	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities										
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
V-1-0160	SAFETY RELIEF VALVE-NR28D	DW	46-2	Yes	6				
V-1-0161	SAFETY RELIEF VALVE-NR28E	DW	46-2	Yes	6				
V-1-0162	SAFETY RELIEF VALVE-NR28F	DW	46-2	Yes	6				
V-1-0163	SAFETY RELIEF VALVE-NR28G	DW	46-2	Yes	6				
V-1-0164	SAFETY RELIEF VALVE-NR28H	DW	46-2	Yes	6				
V-1-0165	SAFETY RELIEF VALVE-NR28J	DW	46-2	Yes	6				
V-1-0166	SAFETY RELIEF VALVE-NR28K	DW	46-2	Yes	6				
V-1-0167	SAFETY RELIEF VALVE-NR28L	DW	46-2	Yes	6				
V-1-0168	SAFETY RELIEF VALVE-NR28M	DW	46-2	Yes	6				
V-20-0151	C.S. SYS II INBRD ISO TESTABLE CHK VLV	DW	75-3	Yes	2				
V-20-0153	C.S. SYS II INBRD ISO TESTABLE CHK VLV	DW	75-3	Yes	2				
V-1-0130A	TURBINE BYPASS VALVE NO. 1	TB	23-6	Yes	6				
V-1-0130B	TURBINE BYPASS VALVE NO. 2	TB	23-6	Yes	6				
V-1-0130C	TURBINE BYPASS VALVE NO. 3	TB	23-6	Yes	6				
V-1-0130D	TURBINE BYPASS VALVE NO. 4	TB	23-6	Yes	6				
V-1-0130E	TURBINE BYPASS VALVE NO. 5	TB	23-6	Yes	6				
V-1-0130F	TURBINE BYPASS VALVE NO. 6	TB	23-6	Yes	6				
V-1-0130G	TURBINE BYPASS VALVE NO. 7	TB	23-6	Yes	6				
V-1-0130H	TURBINE BYPASS VALVE NO. 8	TB	23-6	Yes	6				
V-1-0130I	TURBINE BYPASS VALVE NO. 9	TB	23-6	Yes	6				
CV-0001	Turbine Control Valve CV-1	TB	23-6	Yes	6				
CV-0002	Turbine Control Valve CV-2	TB	23-6	Yes	6				
CV-0003	Turbine Control Valve CV-3	TB	23-6	Yes	6				
CV-0004	Turbine Control Valve CV-4	TB	23-6	Yes	6				
PS-RE0023A	Main Steam Low Pressure Switch RE-23A	TB	23-6	Yes	6				
PS-RE0023B	Main Steam Low Pressure Switch RE-23B	TB	23-6	Yes	6				
PS-RE0023C	Main Steam Low Pressure Switch RE-23C	TB	23-6	Yes	6				
PS-RE0023D	Main Steam Low Pressure Switch RE-23D	TB	23-6	Yes	6				
SV-0001	Turbine Stop Valve SV-1	TB	23-6	Yes	6				
SV-0002	Turbine Stop Valve SV-2	TB	23-6	Yes	6				
SV-0003	Turbine Stop Valve SV-3	TB	23-6	Yes	6				
SV-0004	Turbine Stop Valve SV-4	TB	23-6	Yes	6				
LSP-1D	USS-1B2 & 1B3 LOCAL BRKR CONT PNL	TB	23-6	Yes	2				
1A-4160V	4160V 1A BUS & SWGR	TB	23-6	Yes	1, 2				
1B-4160V	4160V 1B BUS & SWGR	TB	23-6	Yes	1, 2				
1C-4160V	4160V 1C BUS & EMERG SWGR	TB	23-6	Yes	2				
1D-4160V	4160V 1D BUS & EMERG SWGR	TB	23-6	Yes	1, 2				
BATTERY BANK C	DIVISION C 125V BATT BANK	TB	23-6	Yes	2				
BT CHG C1	BATT CHRGR FOR DISTR CNTR "C" 125V DC	TB	23-6	Yes	1, 2				
BT CHG C2	BATT CHRGR FOR DISTR CNTR "C" 125V DC	TB	23-6	Yes	2				
DC-C-125V	125V DC DISTRIBUTION CNTR C	TB	23-6	Yes	6				
SW-1D4160	MANUAL THROWOVER 125V DC SW TO 1D-4160	TB	23-6	Yes	6				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
FN-59-008	4160V SWGR VAULT 1C ROOF VENTILATOR	TB	23-6	Yes	6				
FN-59-019	4160V SWGR VAULT 1D ROOF VENTILATOR	TB	23-6	Yes	6				
DM-59-0006	CONT FOR AIR INTAKE FROM SWGR RM DMPR (D-2)	TB	23-6	Yes	6				
DM-59-0008	BATT RM "C" AIR INTAKE CONT DMPR (D-4)	TB	23-6	Yes	6				
DM-59-0009	AIR INLET CONT DMPR (D-5)	TB	23-6	Yes	6				
FN-59-0006	"C" BATT RM EXHST FAN 1-1	TB	23-6	Yes	2				
FN-59-0007	"C" BATT RM EXHST FAN 1-2	TB	23-6	Yes	2				
H&V PNL (BAT RM C)	BAT RM 'C' H&V PANEL	TB	23-6	Yes	1				
PNL ER2A	ROD POSITION RLY PNL	TB	36-0	Yes	6				
PNL ER2B	ROD POSITION RLY PNL	TB	36-0	Yes	6				
PNL ER-67	24 VDC PWR PNL A RLY PNL	TB	36-0	Yes	6				
FN-56-0004	SWGR RM "A" MAIN SPLY FAN	TB	41-0	No	10	0.50	0.40	0.35	0.15
FN-56-0007	SWGR RM "A" MAIN EXHST FAN	TB	41-0	No	10	0.50	0.40	0.35	0.15
FN-56-0008	SWGR RM "A" ALTERNATE EXHAUST FAN	YARD		Yes	6				
DM-56-0008	AIR INTAKE DMPR FOR "A" SWGR RM SPLY FN	TB	41-0	Yes	6				
DM-56-0009	SWGR RM "A" EXHST DMPR	TB	41-0	Yes	6				
DM-56-0010	SWGR RM "A" RECIRC CONTROL DMPR	TB	41-0	Yes	6				
DM-56-0018	ALTERNATE EXHAUST DAMPER FOR SWGR RM A	TB	41-0	Yes	6				
FS-IP0017A	Low Flow Pressure Switch FS17A	TB	46-6	Yes	6				
FS-IP0017B	Low Flow Pressure Switch FS17B	TB	46-6	Yes	6				
PNL 10F	MCR AREA & PROCESS RAD PNL	TB	46-6	Yes	1				
PNL 10R	MCR PROCESS INSTR EQUIP PNL	TB	46-6	Yes	1				
PNL 11F	MCR ISOLATION PANEL	TB	46-6	Yes	1				
PNL 11R	MCR GAS TREAT & VENT PNL	TB	46-6	Yes	1				
PNL 11XR	MCR TELEM/GEN PROT PNL	TB	46-6	Yes	1				
PNL 12R	MCR GEN/XFMR PROT PNL	TB	46-6	Yes	1				
PNL 12XR	MCR TURBINE/AUX SYS TEMP PNL	TB	46-6	Yes	1				
PNL 16R	MCR CONT AIR H2 & O2 ANALYZING PNL	TB	46-6	Yes	1				
PNL 17R	MCR WIDE RANGE RX LVL PNL	TB	46-6	Yes	1				
PNL 18R	MCR RX PROTECTION SYS PNL	TB	46-6	Yes	1				
PNL 19R	MCR RX PROTECTION SYS PNL	TB	46-6	Yes	1				
PNL 1F/2F	MCR RX & DRYWELL COOLING PNL	TB	46-6	Yes	1				
PNL 2R	MCR AREA & RAD MONITOR PNL	TB	46-6	Yes	1				
PNL 3F	MCR CLEANUP & RECIRC PNL	TB	46-6	Yes	1				
PNL 3R	MCR NEUTRON MONITOR PNL	TB	46-6	Yes	1				
PNL 4F	MCR REACTOR CONTROL ROD PNL	TB	46-6	Yes	1				
PNL 4R	MCR NEUTRON FLUX CALIBRATION PNL	TB	46-6	Yes	1				
PNL 5F/6F	MCR FEEDWATER & CONDENSATE PNL	TB	46-6	Yes	1				
PNL 5R	MCR NEUTRON MONITOR PNL	TB	46-6	Yes	1				
PNL 6R	MCR REACTOR PROTECTION CH.1 PNL	TB	46-6	Yes	1				
PNL 6XR	MCR PROTECTION SYS OPERATOR PNL	TB	46-6	Yes	1				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities									
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
PNL 7R	MCR RX PROT CH.2 PNL	TB	46-6	Yes	1				
PNL 7F	MCR TURBINE PANEL	TB	46-6	Yes	1				
PNL 8F/9F	DG CONTROL PANEL	TB	46-6	Yes	1				
PNL 9R	MCR FEEDWATER & RECIRC PNL	TB	46-6	Yes	1				
PNL 9XR	MCR ENVIRONS MONITOR PNL	TB	46-6	Yes	1				
DM-59-0005	BATT RM "C" AIR INTAKE CONT DMPR (D-1)	TB	46-6	Yes	6				
DM-59-0007	BATT RM "C" EXHAUST CONT DMPR (D-3)	TB	46-6	Yes	6				
SW-826-125	FN-826-008B POWER XFER SW	TB	63-6	Yes	2				
M-826-001A	CONT RM AC UNIT (FN-826-008A)	TB	63-6	Yes	6				
FN-826-008A	CONT RM HVAC SYS "A" SPLY FAN (FN-56-015 & SF-1-15)	TB	63-6	Yes	6				
FN-826-008B	CONT RM HVAC SPLY FAN SYSTEM "B"	TB	63-6	Yes	2				
DM-826-0042	SPLY ISO DMPR FOR CONT RM HVAC SYS "A"	TB	63-6	Yes	6				
DM-826-0043	AIR RTN ISO DMPR FOR CONT RM HVAC SYS "A"	TB	63-6	Yes	6				
DM-826-0044	CONT RM HVAC SYS "A" EXHST TO ATMOS DMPR (DA-4)	TB	63-6	Yes	6				
DM-826-0046	MAX. OUTSIDE AIR CONT RM DMPR (DA-2)	TB	63-6	Yes	6				
DM-826-0047	RECIRC AIR CONT RM DMPR (DA-3)	TB	63-6	Yes	6				
I-826-001A	REFRIDGERATION SYS FOR A/C UNIT M-828-001A	TB	63-9	Yes	6				
C-826-003A	C.R. A/C UNIT M-826-001A COOLING COIL (CC-1-16)	TB	63-9	Yes	6				
XMR-826-021	120/120V ISO XFORMER FOR TRACER 100 STATUS PNL	TB	74-0	Yes	6				
IPS-826-009	125V DC POWER SUPPLY FOR CR HVAC SYSTEM B	TB	74-0	Yes	6				
M-826-001B	CONT RM AC UNIT (FN-826-008B)	TB	74-0	Yes	2				
PNL ER-826-134	CNTRL RM HVAC SYS "B" STATUS PNL	TB	74-0	No	9				
M-30-001	OFFICE BUILDING CHILLED WATER UNIT	TB	74-0	Yes	6				
DM-826-0034	OUTSIDE AIR INLET CONT DMPR FOR CONT RM A/C UNIT	TB	ROOF	Yes	6				
DM-826-0035	RECIRC CONT DMPR FOR CONT HVAC SYS "B"	TB	ROOF	Yes	6				
DM-826-0037	MTR OPERATED EXHST FOR CONT RM HVAC SYS "B"	TB	ROOF	Yes	6				
DM-826-0039	MTR OP AIR SPLY DMPR FOR CONT RM HVAC SYS "B"	TB	ROOF	Yes	6				
DM-826-0040	RTRN AIR MTR OP ISO DMPR FOR CONT RM HVAC SYS "	TB	ROOF	Yes	6				
T-39-002	DSL GEN FUEL OIL STORAGE TANK	DGB	18-4	Yes	2				
LSP-DG2	LOCAL SHUTDWN PNL EDG-DG2	DGB	23-0	Yes	2				
DG-1-SWGR	DG-1 SWITCHGEAR BUS	DGB	23-0	Yes	1, 2				
DG-2-SWGR	DG-2 SWITCHGEAR BUS	DGB	23-0	Yes	1, 2				
M-39-001	EMER DSL ENGINE-GENERATOR UNIT #1	DGB	23-0	No	8	0.56/0.81	0.12/0.25	0.31/0.31	0.28/0.32
M-39-002	EMER DSL ENGINE-GENERATOR UNIT #2	DGB	23-0	No	8	0.56/0.81	0.12/0.25	0.31/0.31	0.28/0.32
P-39-013	Fuel Oil Transfer Pump 1A	DGB	23-0	Yes	6				
P-39-014	Fuel Oil Transfer Pump 1B	DGB	23-0	Yes	6				
P-39-015	Fuel Oil Transfer Pump 2A	DGB	23-0	Yes	6				
P-39-016	Fuel Oil Transfer Pump 2B	DGB	23-0	Yes	6				
T-39-003	Fuel Oil Storage Tank	DGB	23-0	Yes	2				
T-39-004	Fuel Oil Storage Tank	DGB	23-0	Yes	2				

Table 3-6: List of Components for Seismic Evaluation with Associated Fragilities										
EQUIP. ID. NO.	EQUIPMENT DESCRIPTION	BLDG	FL. EL.	SCREENED	SCREEN OR FRAGILITY BASIS	A_m	β_R	β_U	HCLPF
P-3-003A	EMER SVC WATER PMP 1-1	IS	6-0	No	4	1.18	0.40	0.35	0.34
P-3-003B	EMER SVC WATER PUMP 1-2	IS	6-0	No	4	1.18	0.40	0.35	0.34
P-3-003C	EMER SVC WATER PMP 1-3	IS	6-0	No	4	1.18	0.40	0.35	0.34
P-3-003D	EMER SVC WATER PUMP 1-4	IS	6-0	No	4	1.18	0.40	0.35	0.34
1B3-XFMR-USS	USS 1B3-460V XFORMER 4160V-480V/277V 3PH 60HX	IS	6-0	Yes	7				
SW-1B3	MANUAL THROWOVER 125V DC SW TO USS-1B3	IS	6-0	Yes	6				
V-2-0016	MAIN CONDENSER A,B,&C AUTO MAKEUP VALVE	CF	23-6	Yes	6				
V-2-0235	BYPASS OF CONDENSER AUTO MAKEUP VALVE V-2-0016	CF	23-6	Yes	6				
P-11-001	CONDENSATE TRANSFER PUMP (1-1)	CF	23-6	Yes	2				
P-11-002	CONDENSATE TRANSFER PUMP (1-2)	CF	23-6	Yes	2				
V-9-0057	Relief Valve V-9-57	FWP	12-0	Yes	6				
V-9-0067	Relief Valve V-9-67	FWP	12-0	Yes	6				
P-9-102A	DIESEL DRIVEN FIRE PUMP (1-1)	FWPH	12-0	Yes	2				
P-9-102B	DIESEL DRIVEN FIRE PUMP (1-2)	FWPH	12-0	Yes	2				
	Diesel Fire Pump Fuel Tank No. 1	FWPH		Yes	6				
	Diesel Fire Pump Batteries (No. 1)	FWPH		Yes	6				
	Diesel Fire Pump Fuel Tank No. 2	FWPH		Yes	6				
	Diesel Fire Pump Batteries (No. 2)	FWPH		Yes	6				
T-11-001	CONDENSATE STORAGE TANK	YARD		No	4	0.31	0.40	0.35	0.09
SA	STARTUP TRANSFORMER	YARD		No	5	0.53	0.28	0.23	0.23
SB	STARTUP TRANSFORMER	YARD		No	5	0.53	0.28	0.23	0.23
XMR-743-043	SBO TRANSFORMER (BANK #3) 13.8/4.16KV,30	YARD		No	5	0.53	0.28	0.23	0.23
CT1	Combustion Turbine No. 1			No	12	0.60	0.37	0.39	0.17
CT2	Combustion Turbine No. 2			No	12	0.60	0.37	0.39	0.17

Seismic Screening Notes:

1. Assume all interaction concerns raised in A46 program resolved
2. Review of SEWS package showed adequate margin to screen
3. Review of proposed anchorage modification showed adequate margin to screen
4. Specific fragility calculation performed for structural capacity of anchorage
5. Fragility based on incipient overturning
6. Generically rugged equipment
7. Deteriorated condition noted in A46. Assume restored to orginal condition
8. First number is for incipient sliding. Second number is for a 1" sliding displacement.
9. Not in SSEL
10. Based on as-is condition for vibration isolator mounted component; design modifications not available
11. MSIV operator weight is outside the screening rules, however judged to be rugged
12. Based on Fuel Oil Tanks

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{Hat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
DGB 23-6	DG-1 CONTR CABINET	AGASTAT 7022	1.08	0.28	0.34	0.39	0.46
		AGASTAT EGPI003	0.89	0.28	0.34	0.32	0.38
DGB 23-6	DG-1 DSL GEN CONT CAB	ALLIS CHALMER 5OINT	0.81	0.28	0.34	0.29	0.35
		CUTLER HAMMER A10BN0	1.22	0.28	0.34	0.44	0.52
		EMD 8253241	0.81	0.28	0.34	0.29	0.35
		EMD 8253244	0.81	0.28	0.34	0.29	0.35
		EMD 8263337	0.81	0.28	0.34	0.29	0.35
		EMD 8263578	0.81	0.28	0.34	0.29	0.35
		EMD 8268542	0.81	0.28	0.34	0.29	0.35
		EMD 8269705	0.81	0.28	0.34	0.29	0.35
		EMD 8284295	0.81	0.28	0.34	0.29	0.35
		EMD 8288736	0.81	0.28	0.34	0.29	0.35
		EMD 8357843	0.81	0.28	0.34	0.29	0.35
		EMD 8365353	1.26	0.28	0.34	0.45	0.54
		EMD 8370706	1.26	0.28	0.34	0.45	0.54
		EMD 8370794	2.43	0.28	0.34	0.88	1.04
		EMD 8371879	0.81	0.28	0.34	0.29	0.35
		EMD 8380774	0.81	0.28	0.34	0.29	0.35
		EMD 8398823	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-03 (EMD8370706)	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-05 (EMD8370706)	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-07 (EMD8371879)	0.81	0.28	0.34	0.29	0.35
		VAPOR 3650082-70	1.26	0.28	0.34	0.45	0.54
		VAPOR 36530082-01 (EMD8370794)	2.43	0.28	0.34	0.88	1.04
		WILMAR WUV-1-120-H	0.81	0.28	0.34	0.29	0.35
		WOODWARD 8872-685	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9903-337	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9905-002	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9905-007	1.26	0.28	0.34	0.45	0.54
DGB 23-6	DG-1 SWGR	GE 12HEA61	2.03	0.28	0.42	0.63	0.75
		GE 12IJD52A	1.78	0.28	0.42	0.55	0.66
		SWGR DG1	1.60	0.28	0.42	0.50	0.59
DGB 23-6	DG-2 CONTR CABINET	AGASTAT 7022	1.08	0.28	0.34	0.39	0.46
		AGASTAT EGPD003	0.89	0.28	0.34	0.32	0.38

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{Hat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
DGB 23-6	DG-2 DSL GEN CONT CAB	ALLIS CHALMER 5OINT	0.81	0.28	0.34	0.29	0.35
		CUTLER HAMMER A10BN0	1.22	0.28	0.34	0.44	0.52
		EMD 8253241	0.81	0.28	0.34	0.29	0.35
		EMD 8253244	0.81	0.28	0.34	0.29	0.35
		EMD 8263337	0.81	0.28	0.34	0.29	0.35
		EMD 8263578	0.81	0.28	0.34	0.29	0.35
DGB 23-6	DG-2 DSL GEN CONT CAB	EMD 8268542	0.81	0.28	0.34	0.29	0.35
		EMD 8269705	0.81	0.28	0.34	0.29	0.35
		EMD 8284295	0.81	0.28	0.34	0.29	0.35
		EMD 8288736	0.81	0.28	0.34	0.29	0.35
		EMD 8357843	0.81	0.28	0.34	0.29	0.35
		EMD 8365353	1.26	0.28	0.34	0.45	0.54
		EMD 8370706	1.26	0.28	0.34	0.45	0.54
		EMD 8370794	2.43	0.28	0.34	0.88	1.04
		EMD 8371879	0.81	0.28	0.34	0.29	0.35
		EMD 8380774	0.81	0.28	0.34	0.29	0.35
		EMD 8398823	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-03 (EMD8370706)	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-05 (EMD8370706)	1.26	0.28	0.34	0.45	0.54
		VAPOR 36330019-07 (EMD8371879)	0.81	0.28	0.34	0.29	0.35
		VAPOR 3650082-70	1.26	0.28	0.34	0.45	0.54
		VAPOR 36530082-01 (EMD8370794)	2.43	0.28	0.34	0.88	1.04
		WILMAR WUV-1-120-H	0.81	0.28	0.34	0.29	0.35
		WOODWARD 8872-685	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9903-337	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9905-002	1.26	0.28	0.34	0.45	0.54
		WOODWARD 9905-007	1.26	0.28	0.34	0.45	0.54
DGB 23-6	DG-2 SWGR	GE 12HEA61	2.03	0.28	0.42	0.63	0.75
		GE 12IJD52A	1.78	0.28	0.42	0.55	0.66
		SWGR DG2	1.60	0.28	0.42	0.50	0.59
IS 6-0	1A3-460V-USS	SWGR 1A3 (GE AKD-5)	2.49	0.31	0.42	0.74	0.91
IS 6-0	1B3-460V-USS	SWGR 1B3 (GE AKD-5)	2.49	0.31	0.42	0.74	0.91
RB -1-11		MERCOID DAW-43	1.96	0.26	0.25	0.85	1.16
RB 23-6	1A2-460V-MCC	MCC 1A2	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{Hat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
RB 23-6	1A2-460V-USS	GE CR2820B	6.59	0.29	0.50	1.80	2.79
		SWGR 1A2 (GE AKD-5)	5.22	0.29	0.50	1.42	2.21
RB 23-6	1A21-460V-MCC	MCC 1A21	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1A21A-460V-MCC	MCC 1A21A	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1A21B-460V-MCC	MCC 1A21B	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1A23-460V-MCC	MC SEAL-IN RLY	1.55	0.28	0.40	0.50	0.78
		MCC 1A23	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1AB2-460V-MCC	MCC 1AB2	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
		XFER SW 067183-042	1.55	0.28	0.40	0.50	0.78
		XFER SW 138972	1.55	0.28	0.40	0.50	0.78
		XFER SW 143517-001	1.55	0.28	0.40	0.50	0.78
		XFER SW 269758	1.55	0.28	0.40	0.50	0.78
		XFER SW X247A3310	1.55	0.28	0.40	0.50	0.78
RB 23-6	1B2-460V-MCC	GE CR120A	9.32	0.28	0.40	3.02	4.69
		MCC 1B2	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1B2-460V-USS	GE CR2820B	6.59	0.29	0.50	1.80	2.79
		SWGR 1B2 (GE AKD-5)	5.22	0.29	0.50	1.42	2.21
RB 23-6	1B21-460V-MCC	MC AUX RLY	1.55	0.28	0.40	0.50	0.78
		MC SEAL-IN RLY	1.55	0.28	0.40	0.50	0.78
		MCC 1B21	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1B21A-460V-MCC	MCC 1B21A	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	1B21B-460V-MCC	MCC 1B21B	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{Hat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
RB 23-6	DC-1 125V DC	ASCO 067180-084	3.11	0.28	0.40	1.01	1.56
		ASCO 105A2130	3.11	0.28	0.40	1.01	1.56
		ASCO 138972	3.11	0.28	0.40	1.01	1.56
		ASCO 143517-004	3.11	0.28	0.40	1.01	1.56
		ASCO 269758	3.11	0.28	0.40	1.01	1.56
RB 23-6	DC-1 125V MCC	MCC DC1	4.35	0.28	0.40	1.41	2.19
		MTR CONTACTOR	4.66	0.28	0.40	1.51	2.35
RB 23-6	IT-3 SW	ASCO 067183-020	2.64	0.28	0.43	0.82	1.27
		ASCO 138972	2.64	0.28	0.43	0.82	1.27
		ASCO 143517-001	2.64	0.28	0.43	0.82	1.27
		ASCO 269758	2.64	0.28	0.43	0.82	1.27
		ASCO 906172C	2.64	0.28	0.43	0.82	1.27
RB 23-6	IT-4	ASCO 067183-026	2.64	0.28	0.43	0.82	1.27
		ASCO 138972	2.64	0.28	0.43	0.82	1.27
		ASCO 143517-002	2.64	0.28	0.43	0.82	1.27
		ASCO 269758	2.64	0.28	0.43	0.82	1.27
		ASCO 906173C	2.64	0.28	0.43	0.82	1.27
RB 23-6	LSP-1A2	PB MDR 138-3/173-1	5.27	0.28	0.43	1.63	2.54
RB 23-6	MCL-823-008	CUTLER HAMMER D40RR30A	8.79	0.28	0.43	2.72	4.23
RB 23-6	PNL ER18A	GE 12HFA151A	2.64	0.28	0.43	0.82	1.27
		GE 12HFA151A	2.64	0.28	0.43	0.82	1.27
		GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
RB 23-6	PNL ER18B	GE 12HFA151A	2.64	0.28	0.43	0.82	1.27
		GE 12HFA151A	2.64	0.28	0.43	0.82	1.27
		GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
RB 23-6	PNL ER8A	GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
		GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
		GE 12HFA51A	5.27	0.28	0.43	1.63	2.54
RB 23-6	PNL ER8B	GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
		GE 12HFA51A	0.88	0.28	0.43	0.27	0.42
		GE 12HFA51A	5.27	0.28	0.43	1.63	2.54
RB 23-6	RSP	PB MDR 138-8	5.27	0.28	0.43	1.63	2.54
		PB MDR 173-1	5.27	0.28	0.43	1.63	2.54

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{Hat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
RB 23-6	VACP-1A	ASCO 067183-020	2.64	0.28	0.43	0.82	1.27
		ASCO 138972	2.64	0.28	0.43	0.82	1.27
		ASCO 143517-001	2.64	0.28	0.43	0.82	1.27
		ASCO 269758	2.64	0.28	0.43	0.82	1.27
		ASCO 906172C	2.64	0.28	0.43	0.82	1.27
RB 35-0	DC-D 125V DC	ASCO 067180-004	3.42	0.28	0.41	1.09	1.52
		ASCO 138972	3.42	0.28	0.41	1.09	1.52
		ASCO 143517-005	3.42	0.28	0.41	1.09	1.52
		ASCO 269758	3.42	0.28	0.41	1.09	1.52
		ASCO 906128C	3.42	0.28	0.41	1.09	1.52
RB 35-0	ROTARY INV CNTRL PNL	GE CR120A	10.25	0.28	0.41	3.27	4.55
		GE CR122A	3.42	0.28	0.41	1.09	1.52
		GE CR2820B	11.39	0.28	0.41	3.64	5.06
RB 51-3	ER-642-112,113	PB MDR 138-8	6.84	0.28	0.42	2.13	2.96
RB 51-3	ER-642-114,115	PB MDR 138-8	6.84	0.28	0.42	2.13	2.96
RB 51-3	PS-IA0083A,-B,-C,-D,-E	BARKSDALE B2S-M12SS	3.42	0.28	0.42	1.06	1.48
RB 51-3	PS-RV-0046A,-B,-C,-D	BARTON 580A-0	3.42	0.28	0.42	1.06	1.48
RB 75-3	DC-2 125V MCC	MCC DC2	6.31	0.28	0.44	1.91	3.00
		MTR CONTACTOR	6.76	0.28	0.44	2.05	3.21
TB 3-6	1A1-460V-USS	SWGR 1A1 (GE AKD-5)	3.38	0.30	0.50	0.90	1.40
TB 23-6	1A-4160V SWGR	AGASTAT E7014	3.79	0.30	0.50	1.01	1.56
		SWGR 1A (GE TYPE M-26)	3.00	0.30	0.50	0.80	1.24
TB 23-6	1C-4160V SWGR	AGASTAT 2414	3.79	0.30	0.50	1.01	1.56
		AGASTAT E7014	3.79	0.30	0.50	1.01	1.56
		AGASTAT EGPD002	1.25	0.30	0.50	0.33	0.52
		BROWN BOVERI 27N211B0175	5.68	0.30	0.50	1.51	2.34
		GE 12HEA61	3.79	0.30	0.50	1.01	1.56
TB 23-6	1C-4160V SWGR	GE 12HFA151A	1.14	0.30	0.50	0.30	0.47
		GE 12HFA51A (assumed NO)	2.27	0.30	0.50	0.60	0.94
		GE 12IAC53	2.65	0.30	0.50	0.70	1.09
		GE 12IAC55	1.14	0.30	0.50	0.30	0.47
		GE 12IAV53	2.38	0.30	0.50	0.63	0.98
		GE 12PJC11	1.89	0.30	0.50	0.50	0.78
		SWGR 1C (GE TYPE M-26)	3.00	0.30	0.50	0.80	1.24

Table 3-7 Electrical Panel Fragilities Based on Relay Function

Building Elevation	Mounting Panel	Relay Model	A_{flat} (g)	β_R	β_U	$HCLPF_{50}$ (g)	$HCLPF_{84}$ (g)
TB 23-6	1D-4160V SWGR	AGASTAT 2414	3.79	0.30	0.50	1.01	1.56
		AGASTAT EGPD002	1.25	0.30	0.50	0.33	0.52
		BROWN BOVERI 27N211B0175	5.68	0.30	0.50	1.51	2.34
		GE 12HEA61	3.79	0.30	0.50	1.01	1.56
		GE 12HFA151A	1.14	0.30	0.50	0.30	0.47
		GE 12IAC53	2.65	0.30	0.50	0.70	1.09
		GE 12IAV53	2.38	0.30	0.50	0.63	0.98
		GE 12PJC11	1.89	0.30	0.50	0.50	0.78
		GE 12HFA51A (assumed NO)	2.27	0.30	0.50	0.60	0.94
		SWGR 1D (GE TYPE M-26)	3.00	0.30	0.50	0.80	1.24
TB 23-6	A1-24V	BATTERY CHARGER	4.51	0.28	0.37	1.55	2.41
TB 23-6	A2-24V	BATTERY CHARGER	4.51	0.28	0.37	1.55	2.41
TB 23-6	PNL ER1	GE CR120A	4.54	0.30	0.43	1.37	2.13
TB 46-6	PNL 18R	FOXBORO 2AI-2V	6.77	0.27	0.54	1.78	3.48
TB 46-6	PNL 19R	FOXBORO 2AI-2V	6.77	0.27	0.54	1.78	3.48
TB 46-6	PNL 1F/2F	AGASTAT E7024	2.25	0.30	0.62	0.50	0.97
		ALLEN BRADLEY 700RTC	6.31	0.30	0.62	1.39	2.71
		GE CR120A	4.06	0.30	0.62	0.89	1.74
TB 46-6	PNL 6R	AGASTAT EGPI003	1.49	0.30	0.62	0.33	0.64
		GE 12HFA51A	2.70	0.30	0.62	0.59	1.16
		GE CR120A	4.06	0.30	0.62	0.89	1.74
		GE CR205	2.03	0.30	0.62	0.45	0.87
TB 46-6	PNL 6XR	GE CR120A	4.06	0.30	0.62	0.89	1.74
TB 46-6	PNL 7R	AGASTAT EGPI003	1.49	0.30	0.62	0.33	0.64
		GE 12HFA51A	2.70	0.30	0.62	0.59	1.16
		GE CR120A	4.06	0.30	0.62	0.89	1.74
		GE CR205	2.03	0.30	0.62	0.45	0.87

3.1.5 Analysis of Plant Systems and Sequences

This report section describes the process used to analyze Oyster Creek plant systems and accident sequences initiated by seismic events. In overview, the process of plant system analysis and accident sequence development is comprised of three tasks:

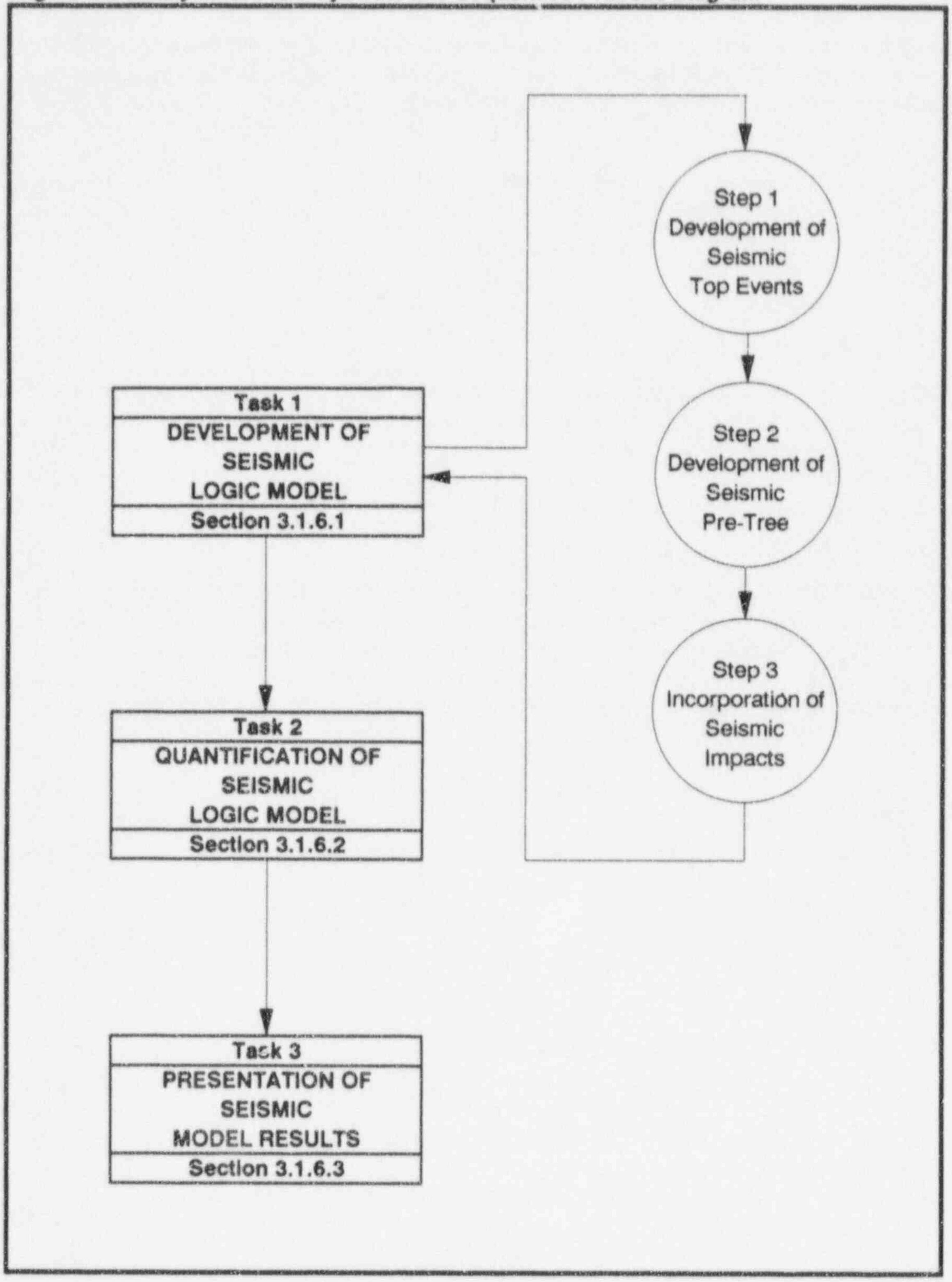
- Develop the Seismic Logic Model
- Quantification of the Seismic Logic Model
- Presentation of the Results of the Quantification

The seismic logic model provides a framework to combine the component failures due to seismic events with the existing Level 1 PRA model (independent component and systems failures) to produce and delineate the integrated plant response to seismically initiated events. The model is constructed such that both the failure of components and civil structures which are susceptible to seismic events (non-high capacity components) and independent failures can be combined.

> Task 1, Development of Seismic Logic Model, utilizes the Level 1 PRA general transient logic in combination with the seismic top events to create a new logic structure which is used in the development of the seismically initiated accident sequences. Civil Structure fragilities, component fragilities, relay fragilities (relay chatter) and masonry block wall fragilities developed in Section 3.1.3 and 3.1.4 are used in the creation of seismic top events and split fractions.
>
> The result of this task is the necessary logic to identify and quantify the accident sequences which result from seismically initiated events. This task is described in detail in Section 3.1.5.1.
>
> Task 2, Quantification of the Seismic Logic Model, assembles the logic created in Task 1 and produces the accident sequences. Section 3.1.5.2 describes this task in detail.
>
> Following the completion of Task 2, Task 3 assembles, sorts and presents the results of the quantification of the seismic logic from a variety of perspectives to allow interpretation and gleaning of insights. In addition, various sensitivity studies are performed including quantification of the Seismic Logic Model using the hazard curves developed by LLNL. The presentation of results is presented in detail in Section 3.1.5.3.

Figure 3-3 illustrates the process of the analysis of plant systems and sequences. The subsequent report sections describe each tasks in detail.

Figure 3-3 Analysis of Plant Systems and Sequences Process Diagram



3.1.5.1 Development of the Seismic Logic Model

The seismic logic model provides the framework for the combination of component failures due to seismic events (fragilities) with the independent failures modeled in the Level 1 PRA (general transient model) to delineate the accident sequences. Input for this task is taken from the evaluation of structure, component and block wall fragilities (Sections 3.1.3 and 3.1.4) and the Oyster Creek Level 1 PRA. The result of this task is a readily quantifable logic model. The development of the seismic logic model is composed of two steps:

- Development of Seismic Top Events and Split Fractions
- Incorporation of the Seismic Top Events and Split Fractions in the Level 1 PRA General Transient Linked Model

These steps are shown on Figure 3-3 as circles. Each step is described in subsequent sections.

Step 1 - Development of Seismic Top Events and Split Fractions. Each of the structure, component and block wall fragilities developed in Section 3.1.3 and 3.1.4 is "mapped" to the top event or function is impacted in the Oyster Creek Level 1 PRA as a result of its seismic failure. The term "mapped" refers to the development of a new seismic top event which represents the component's seismic failure rate. Several plant systems were noted (during seismic capacity walkdowns) as seismically fragile, these systems were mapped to the associated top event as guaranteed failure events and no fragility analysis was performed. For reference, those systems that are assigned to guaranteed failure are:

Circulating Water	Top Event CW	Support
Service Water	Top Event SW	Module
Turbine Building Closed Cooling Water	Top Event TB	
Instrument Air	Top Event IA	
Main Condensate	Top Event CP	General
Main Feedwater	Top Event FW	Transient
Standby Gas Treatment	Top Event SG	Long Term

Redundant fire pump was assumed to fail for all seismic events (top event FP).

For those systems that were not assigned to guaranteed failure, new top events were developed. These new top events contain only the seismic failures (based on fragility) of the components which impact the complement Level 1 top event. The new top event is named with the first letter of the Level 1 top event to which it is mapped, followed by the letters U, V, W, X, Y or Z in the second digit. The letters U, V, W, X, Y and Z are chosen since they do not interfere with the Level 1 PRA top event names and provide sufficient flexibility to allow for the naming of seismic top events.

For example, the seismic failure top event developed for OP, independent failure of offsite power, is OX, where top event OX represents the seismic failure of offsite power, its complement event.

In the case of essential 4160 VAC Bus 1C, the independent top event EC (4160 VAC Bus 1C) complement seismic top events are EW (essential 4160 VAC 1C with offsite power available) and EY (essential 4160 VAC 1C supplied by diesel generator). Two seismic top events are necessary for this independent top event since the boundary conditions within the independent event change to include diesel generators following the loss of offsite power.

In addition to the change in boundary conditions for essential AC power, seismic interaction through soil liquefaction can impact the availability of site diesel generators in the following ways:

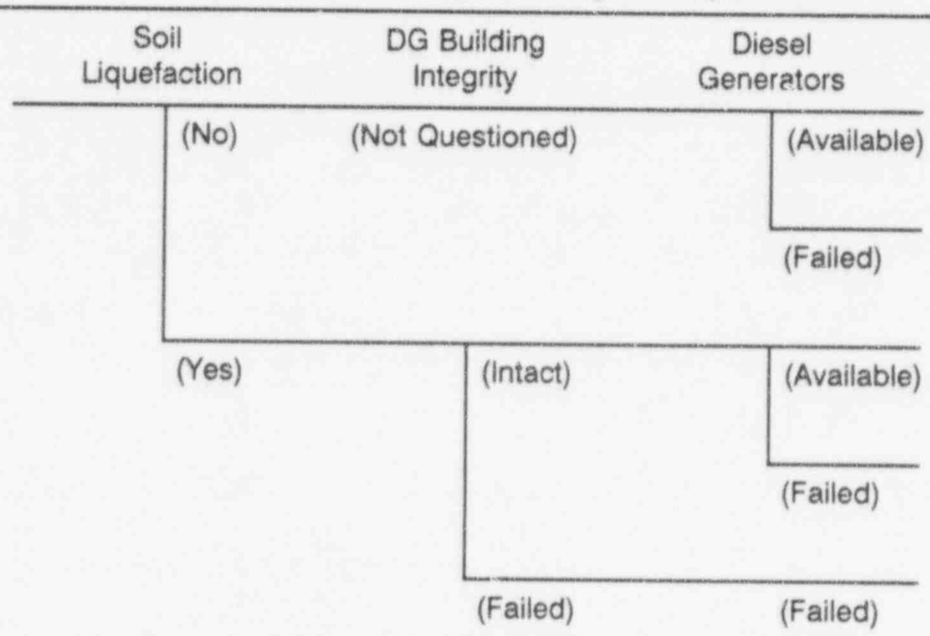
1. If soil liquefaction does not occur, diesel generator building integrity is not questioned (since the building capacity is beyond the screening criteria) and diesel generator seismic failure is evaluated.

2. If soil movement through soil liquefaction occurs, diesel generator building failure is much more likely (lower capacity). Building failure then fails both diesels.

3. If soil movement through soil liquefaction occurs and the diesel generator building remains intact, diesel generator failure is questioned.

This interaction is modeled by using three top events as shown below.

Figure 3-4
Modeling of Diesel Generator Building Soil Liquefaction



The scenarios shown in Figure 3-4 that indicate diesel generator availability is only questioned due to seismic causes. This is a simplification and the diesel generators failure is modeled due to both seismically initiated events as well as independent (normal hardware) causes (failure to start and run, etc.). This is accomplished using the Level 1 OCPRA top event EC and ED, as described in the Reference 3-21.

It should also be noted that, due to the nature of seismic failures, train independence is overridden by an assumed common-mode failure. In other words, diesel generator failure due to seismic concerns (failure of top event DW, above) is assumed to fail both diesel generators. No independence for this failure mode is assumed to exist. As noted above, success of top event DW then questions each diesel generator, as described in the Level 1 model documentation. The common mode of seismic failure is also assumed for the startup transformers in seismic top event EU, which then fails both top events EA and EB. This form of failure is also subsumed within individual systems. For example, seismic failure of any core spray pump at seismic top event CY is assumed to result in the failure of all similar pumps within the core spray system, resulting in the introduction of a single failure mode for the system due to the assumption of common mode seismic failure.

The split fractions that were developed for each seismic top event represent each of the four ground accelerations (initiating events) modeled in this study. These split fractions are named with the numbers 1, 2, 3 or 4 appearing in the third digit. For example, split fractions OX1, OX2, OX3 and OX4 are used to reflect the seismic failure of offsite power for each of the four ground accelerations analyzed (0.125g, 0.36g, 0.54g and 0.72g), respectively. In Figure 3-4, each of the top events questioned in a given scenario would have a corresponding higher failure rate for the higher ground accelerations.

One exception to the above method is the development of split fraction values for top event LY. Top event LY is not developed from fragility analysis as described above. Top event LY represents an adjustment to the independent recovery of offsite power failure event. The independent recovery of offsite power (top event LP) contains probabilities associated with the recovery of the offsite grid. Following a seismic event it is assumed that grid restoration is not possible. Top event LY models the probability of the independent failure to recover offsite power using the near site combustion turbines only. In combination with top event LX (seismic failure of recovery of offsite power), these events model the recovery of offsite power using the combustion turbines only.

Table 3-8 titled SEISMIC Master Frequency File, provides a brief description of the seismic top events developed each of the new seismic split fractions and descriptions and split fraction values developed for each. The application of each top event in the logic model structure and the seismic fragilities represented by the top event are described in subsequent paragraphs.

Step 2 - Incorporation of Seismic Top Events and Split Fractions. The role of the seismic top events in the seismic logic model is presented in Figure 3-5. Seismically initiated events enter the seismic logic model and question a seismic top event followed by its complement level 1 PRA independent failure top event. Impacts from the success and failure of seismic top event are propagated to the complement Level 1 top event thus preserving the original dependencies which are modeled in the Level 1 PRA general transient logic.

Table 3-8 Seismic Master Frequency File

T.E. Designator	Split Fraction	Frequency	Split Fraction Description	Components Fragilities
BX	BX1	2.88E-07	Standby Liquid Contol during SEIS1	FRAG09
	BX2	2.01E-05	Standby Liquid Contol during SEIS2	
	BX3	4.62E-04	Standby Liquid Contol during SEIS3	
	BX4	2.82E-03	Standby Liquid Contol during SEIS4	
CX	CX1	4.74E-05	Containment Spray/ESWduring SEIS1	FRAG15
	CX2	4.43E-02	Containment Spray/ESWduring SEIS2	FRAG16
	CX3	2.96E-01	Containment Spray/ESWduring SEIS3	FRAG33
	CX4	6.48E-01	Containment Spray/ESWduring SEIS4	FRAG35
CZ	CZ1	1.11E-04	Core Spray during SEIS1	FRAG17
	CZ2	7.20E-02	Core Spray during SEIS2	FRAG32
	CZ3	3.26E-01	Core Spray during SEIS3	
	CZ4	5.88E-01	Core Spray during SEIS4	
DU	DU1	2.61E-04	DG Building Liquefaction during SEIS1	FRAG02
	DU2	2.58E-01	DG Building Liquefaction during SEIS2	
	DU3	7.94E-01	DG Building Liquefaction during SEIS3	
	DU4	9.59E-01	DG Building Liquefaction during SEIS4	
DV	DV1	2.54E-05	DG Building after Liquefaction during SEIS1	FRAG03
	DV2	7.55E-02	DG Building after Liquefaction during SEIS2	FRAG07
	DV3	5.31E-01	DG Building after Liquefaction during SEIS3	
	DV4	8.77E-01	DG Building after Liquefaction during SEIS4	
EU	EU1	7.23E-05	Startup Transformers during SEIS1	FRAG20
	EU2	9.89E-02	Startup Transformers during SEIS2	
	EU3	4.83E-01	Startup Transformers during SEIS3	
	EU4	7.74E-01	Startup Transformers during SEIS4	
EW	EW1	8.38E-04	4160 VAC Bus 1C during SEIS1	FRAG05
	EW2	2.09E-01	4160 VAC Bus 1C during SEIS2	
	EW3	5.31E-01	4160 VAC Bus 1C during SEIS3	
	EW4	7.35E-01	4160 VAC Bus 1C during SEIS4	
FX	FX1	1.63E-05	Fire Protection System during SEIS1	FRAG00
	FX2	1.64E-02	Fire Protection System during SEIS2	
	FX3	1.05E-01	Fire Protection System during SEIS3	
	FX4	2.46E-01	Fire Protection System during SEIS4	
IY	IY1	2.65E-05	Isolation Condenser during SEIS1	FRAG18
	IY2	2.14E-02	Isolation Condenser during SEIS2	
	IY3	1.20E-01	Isolation Condenser during SEIS3	
	IY4	2.66E-01	Isolation Condenser during SEIS4	
LX	LX1	4.66E-04	Offsite Power Recovery during SEIS1	FRAG20
	LX2	2.16E-01	Offsite Power Recovery during SEIS2	FRAG25
	LX3	6.89E-01	Offsite Power Recovery during SEIS3	
	LX4	9.12E-01	Offsite Power Recovery during SEIS4	
OX	OX1	6.08E-03	Offsite Power during SEIS1	FRAG01
	OX2	5.44E-01	Offsite Power during SEIS2	
	OX3	8.36E-01	Offsite Power during SEIS3	
	OX4	9.34E-01	Offsite Power during SEIS4	

Table 3-8 Seismic Master Frequency File

T.E. Designator	Split Fraction	Frequency	Split Fraction Description	Components Fragilities
SX	SX1	4.91E-03	Condensate Storage Tank (CST) during SEIS1	FRAG10
	SX2	5.22E-01	Condensate Storage Tank (CST) during SEIS2	
	SX3	8.33E-01	Condensate Storage Tank (CST) during SEIS3	
	SX4	9.35E-01	Condensate Storage Tank (CST) during SEIS4	
XX	XX1	5.45E-03	Long Term 125 VDC Bus B during SEIS1	FRAG04
	XX2	5.46E-01	Long Term 125 VDC Bus B during SEIS2	
	XX3	8.48E-01	Long Term 125 VDC Bus B during SEIS3	
	XX4	9.43E-01	Long Term 125 VDC Bus B during SEIS4	
DW	DW1	2.14E-05	Diesel Generators during SEIS1	FRAG26
	DW2	3.94E-02	Diesel Generators during SEIS2	FRAG27
	DW3	3.03E-01	Diesel Generators during SEIS3	FRAG31
	DW4	6.45E-01	Diesel Generators during SEIS4	
DX	DX1	3.02E-05	Short Term 125 VDC B (Wall No 53) during SEIS1	FRAG34
	DX2	3.37E-02	Short Term 125 VDC B (Wall No 53) during SEIS2	
	DX3	2.00E-01	Short Term 125 VDC B (Wall No 53) during SEIS3	
	DX4	4.19E-01	Short Term 125 VDC B (Wall No 53) during SEIS4	

COMPONENT, RELAY ANS STRUCTURE FRAGILITY DESCRIPTIONS

Fragility	Description	A(g)	B(r)	B(u)
FRAG00	Generic Fragility for Others	1.00	0.40	0.32
FRAG01	Offsite Power (Ceramic Insulators)	0.30	0.25	0.50
FRAG02	DG Building Liquefaction	0.40	0.14	0.28
FRAG03	DB Building After Liquefaction	0.69	0.14	0.28
FRAG04	Battery Room Fans	0.30	0.40	0.35
FRAG05	Switchgear Room Fans	0.50	0.40	0.35
FRAG06	4160 VAC Switchgear	0.40	0.25	0.48
FRAG07	EDG After Liquefaction	0.56	0.12	0.31
FRAG09	SLC Tank	3.10	0.40	0.35
FRAG10	Condensate Storage Tank	0.31	0.40	0.35
FRAG15	Containment Spray Pump	1.48	0.40	0.35
FRAG16	Emergency Service Water Pump	1.18	0.40	0.35
FRAG17	Core Spray Pump	0.82	0.40	0.35
FRAG18	Isolation Condenser	0.98	0.40	0.35
FRAG20	Plant Transformer	0.53	0.28	0.23
FRAG25	Combustion Turbine	0.60	0.37	0.39
FRAG26	DG with No Liquefaction	0.81	0.25	0.31
FRAG27	DG Building with No Liquefaction	1.18	0.37	0.22
FRAG31	DG Control Cabinet - Relays	0.81	0.28	0.34
FRAG32	Relay Panel 18A and 18B	0.88	0.28	0.43
FRAG33	Relay Panel 8A and 8B	0.88	0.28	0.43
FRAG34	Masonry Wall No. 53	0.77	0.28	0.36
FRAG35	Intake Structure	0.82	0.18	0.26

The top events (system failures) modeled in the seismic logic model are complement events to those utilized in the Oyster Creek Level 1 general transient logic model. These top events have seismic fragilities (for non-high capacity components) modeled within the system or function. The top events modeled in the seismic logic model are questioned prior to their complement independent failures. These functions are listed below with their complement top events listed in brackets ():

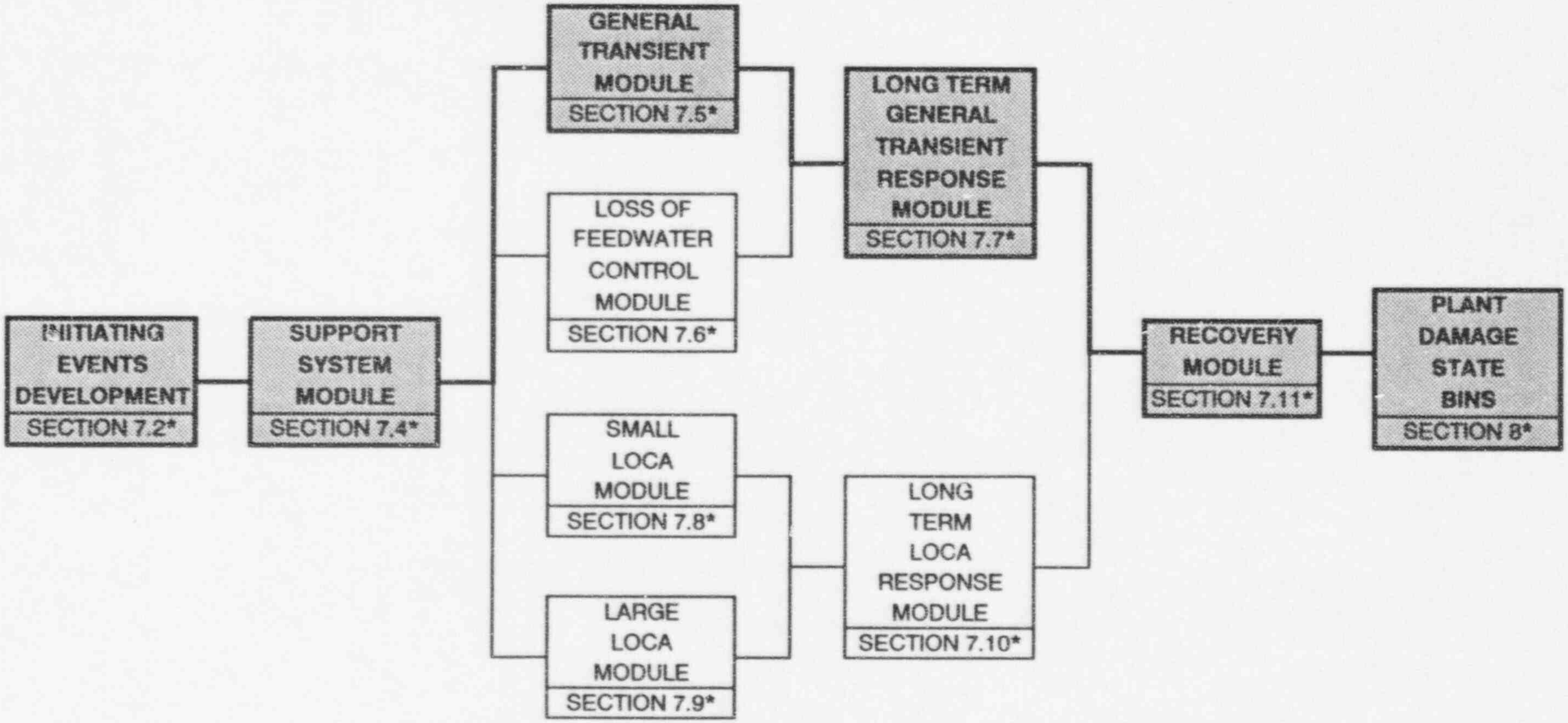
Support Module	•	Top Event DX	Masonry Block Wall No. 53 Failure (DB)
	•	Top Event OX	Offsite Power (OP)
	•	Top Event EU	Startup Transformers (EA/EB)
	•	Top Event DU	DG Building Soil Liquefaction (EC/ED)
	•	Top Event DV	DG Building Integrity Following Liquefaction (EC/ED)
	•	Top Event DW	Diesel Generator Availability (EC/ED)
	•	Top Event EW	Essential Power - 4160 VAC Bus 1C (EC)
	•	Top Event XX	Long Term 125 VDC Bus B (XB)
	•	Top Event FX	Fire Protection Pond (FP)
General Transient	•	Top Event SX	Condensate Storage Tank (ST)
Long Term General Transient	•	Top Event IY	Isolation Condenser (IC)
	•	Top Event BX	Standby Liquid Control (BI)
	•	Top Event LX	Offsite Power Recovery (LP)
	•	*Top Event LY*	*Independent Offsite Power Recovery*
	•	Top Event CX	Containment Spray/ESW (CC)
	•	Top Event CZ	Core Spray (CS)

The seismic model is a binary (branch everywhere) event tree structure, the same as that used in the Level 1 PRA with the addition of the seismic top events and split fractions. Also, as in the Level 1 PRA, successful branches are defined as success (or not necessary) when the top event or system function does not effect the outcome. The nomenclature, syntax and rules methodology are described in detail in the Level 1 PRA.

Except where noted below and in the treatment of recovery actions, the Level 1 plant model logic structure has been used intact. That is, plant system response, recovery and success requirements are as described in the Level 1 report. The recovery module rules for the seismic logic model have been modified to reflect guaranteed failure since the likelihood of success is significantly reduced following a seismically initiated event.

The seismic top events and split fractions developed in the step 1 are collected and the associated split fraction assignment rules are developed.For example, in the case of offsite power (Top Event OP), the original split fraction assignment rule "OP1 -(INIT=LOSP)" (i.e. split fraction OP1 is applied in all cases except for the loss of offsite power initiating event) is now preceded by the seismic top event OX (Seismic failure of offsite power) in the seismic support module.

Figure 3-5 Seismic Plant Model Arrangement



* - indicates the level 1 PRA report section where the details of the plant model module can be found.

The split fractions for the seismic failure of offsite power, OX1, OX2, OX3 and OX4 (one for each of the four ground accelerations) are assigned where the initiating event is equal to SEIS1, SEIS2, SEIS3 or SEIS4 corresponding to approximately 0.13g, 0.36g, 0.54g or 0.72g. The split fraction assignment rules for top event OX are:

OX1	INIT=SEIS1
OX2	INIT=SEIS2
OX3	INIT=SEIS3
OX4	1

The split fraction assignment rule for the independent failure of offsite power is then adjusted to reflect that independent failure is only applied when the corresponding seismic top event (OX) has been successful. That is, following the availability of offsite power following a seismic event (OX=S) the independent failure is questioned. Should offsite power fail during the seismic event, top event OX would be failed and therefore the split fraction OPF would be assigned. The adjusted rules for top event OP (offsite power) in the seismic plant model are therefore:

OP1	OX=S
OPF	1

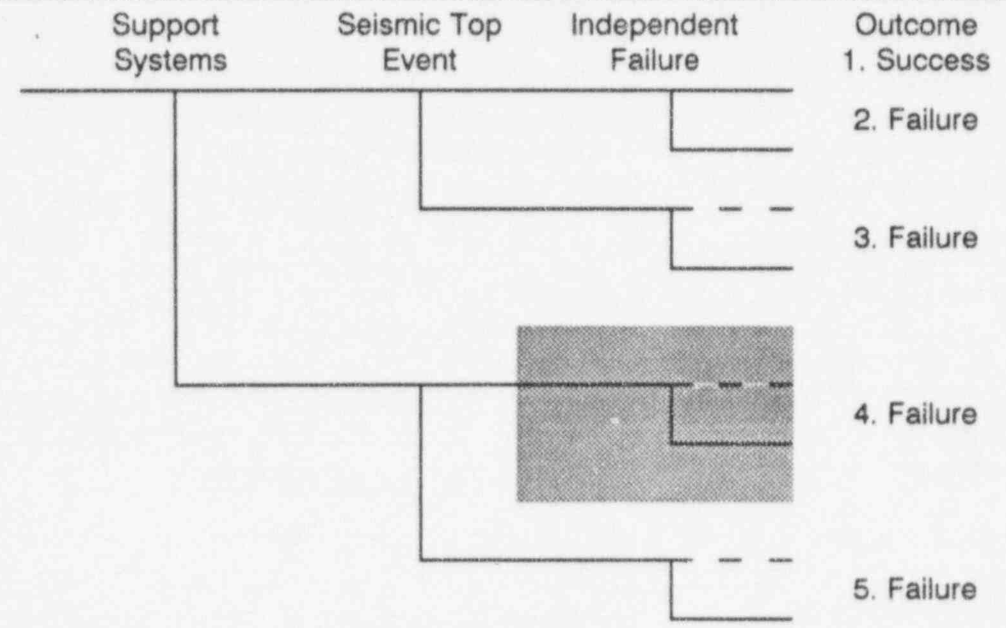
Also factored into the seismic logic model are any spatial impacts (new dependencies) which are a result of the seismic failure. Such is the case with top event DX which represents the seismic failure of masonry block wall number 53 which has impacts on top event DB (Short Term 125 VDC Distribution B). See the discussion under Seismic Support Module Split Fraction Rules.

Split fraction assignment rules are developed for each of the seismic split fractions generated in the previous task. In the offsite power case described above, the development of the split fraction assignment is relatively straightforward. However, in the case where the original independent failure top event has bypasses (i.e. "not necessary" - guaranteed successful) or is composed of complicated dependencies, the seismic top event split fraction assignment rule becomes more complex.

The reasons for this complexity is the desire to construct the model such that it is calculationally efficient and maintain the model sequences as clear, concise and reviewable as possible. That is, it is possible to allow each seismic top event to be conditional on the seismic initiating event only. In this situation, the seismic top event would be questioned regardless of the available supporting systems. The impact of the failure of the seismic top event would be factored into the independent top event, as in the offsite power case above. Since the independent top event requires both the success of the seismic top event as well as the required support systems, all the dependencies are correctly accounted for. Although this modeling approach simplifies the development of the seismic top event split fraction assignment rules, it unduly complicates the calculation and results.

Consider Figure 3-6 which illustrates the additional event tree branches which result from the unconditional questioning of the seismic top event (shaded area). In this simplified example, an additional branch is depicted. However, given the complexity of the possible combinations of

Figure 3-6
Simplified Event Tree Illustration



support systems, seismic events and independent failure events this seemingly minor reduction in event tree branches results in a significant reduction in the number of outcomes and therefore a corresponding reduction in the calculational intensity of the logic model.

Also, as can be seen on Figure 3-6, the unconditional assignment of seismic events results in an additional sequence. That is, outcome 4 is identical to outcome 5 with the exception of the status (success or failure) of the seismic top event. Since, in both scenarios 4 and 5, the support systems are failed which, in turn fails the independent event, the plant damage state assignments remain the same. However the single scenario of concern is now fractured into two components. The first component contains the success of the seismic top event (numerically equal to (1 - seismic split fraction value)). The second component contains the failure of the seismic top event (numerically equal to the (seismic split fraction value)). The sum of the two component scenarios is equivalent to bypass the seismic top event entirely. Therefore, the status of the seismic event does not contribute to the outcome (either probabilistically or by altering the scenario type (plant damage state)). This phenomena has been termed "artificial fracture" and is described in the plant model report section of the Level 1 OCPRA.

The scenario where support systems are failed, the seismic event is successful and the independent event failed (outcome 4) is confusing and can be eliminated from the calculation by including the required support systems as well as the seismic initiating event in the seismic top event split fraction assignment rule. In this case, the seismic top event would be guaranteed failed following the failure of the associated support systems (elimination of the shaded area) just as the independent event is guaranteed failed following the failure of the seismic event in outcome 3. This effectively eliminates outcome 4 from the event tree, reducing the calculational intensity of the model as well as the complexity of the results. These effects are not entirely eliminated from the model and the result is a more complex scenario list. However, results such as total core damage and initiator importance remain unaffected.

These top events and split fraction assignment rules are incorporated into the Level 1 PRA general transient logic to produce the SEISMIC Linked Model. Each of the seismic split fraction assignment rules are grouped by module and top event and discussed detail below. Tables 3-9 through 3-12 provide the actual split fraction assignment rules as they appear in the model.

Seismic Support Module Split Fraction Assignment Rules

Seismic Top Event DX is defined as the availability of short term 125 VDC power from DC Distribution Bus B. This seismic top event models the seismic failure of Masonry Block Wall Number 53 which can have impacts on the 125 VDC batteries and associated supporting equipment. Seismic top event DX is conditional on the initiating event only.

> Split fraction DX1, probability that Block Wall No. 53 fails and disables 125 VDC power following an acceleration of 0.13g is applied where the initiating event is SEIS1 (INIT=SEIS1).
>
> Split fraction DX2, probability that Block Wall No. 53 fails and disables 125 VDC power following an acceleration of 0.36g is applied where the initiating event is SEIS2 (INIT=SEIS2).
>
> Split fraction DX3, probability that Block Wall No. 53 fails and disables 125 VDC power following an acceleration of 0.54g is applied where the initiating event is SEIS3 (INIT=SEIS3).
>
> Split fraction DX4, probability that Block Wall No. 53 fails and disables 125 VDC power following an acceleration of 0.72g is applied in all other cases (1).

Split fraction assignment rules of the independent failure of 125 VDC Bus B (top event DB) are adjusted to reflect the guaranteed failure of top event DB following the seismic failure of 125 VDC Power (top event DX). That is, split fraction DB1 is applied only where top event DX has been successful (DX=S) and split fraction DXF is applied in all other cases.

Seismic Top Event OX is defined as the availability of Offsite Power following a seismic event. Event OX is conditional on the initiating event only. Top event OX models the seismic failure of the offsite grid.

> Split fraction OX1, probability of Offsite Power failure following a ground acceleration of 0.13g is applied where the initiating event is SEIS1 (INIT=SEIS1).
>
> Split fraction OX2, probability of Offsite Power failure following a ground acceleration of 0.36g is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction OX3, probability of Offsite Power failure following a ground acceleration of 0.54g is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction OX4, probability of Offsite Power failure following a ground acceleration of 0.72g is applied in all other cases (1).

Split fraction assignment rules of the independent failure of Offsite Power (top event OP) are adjusted to reflect the guaranteed failure of top event OP following the seismic failure of Offsite Power (top event OX). That is, split fraction OP1 is applied only where top event OX has been successful (OX=S) and split fraction OXF is applied in all other cases.

Seismic Top Event EU is defined as the availability of startup transformers following a seismic event. Since the failure of offsite power (top events OX or OP) results in the failure of independent top events EA and EB, the seismic failure of this support system (offsite power) is accounted for.

Split fraction EU1, probability of startup transformer failure following a ground acceleration of 0.13g is applied where offsite power is successful (OP=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction EU2, probability of startup transformer failure following a ground acceleration of 0.36g is applied where offsite power is successful (OP=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction EU3, probability of startup transformer failure following a ground acceleration of 0.54g is applied where offsite power is successful (OP=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction EU4, probability of startup transformer failure following a ground acceleration of 0.72g is applied where offsite power is successful (OP=S).

Split fraction EUF, guaranteed failure, is applied in all other cases (1).

The split fraction assignment rule for the independent failure of 4160 VAC Buses 1A (top event EA) and 1B (top event EB) have been adjusted to reflect the dependence on seismic failure of the startup transformers. Due to the nature of seismic failures and the similarity of the startup transformers, both supplies are assumed to fail on seismic failure. In other words, a common mode failure of both startup transformers is assumed to occur. Therefore, the assignment rule for split fraction EA1 becomes offsite power success (OP=S), 125 VDC Bus C success (DC=S) and startup transformer success (EU=S). The assignment rule for split fractions EB1 and EBA are similar, except that the support system dependency is on 125 VDC Bus B success (DB=S).

Seismic Top Event DU is defined as the likelihood of soil liquefaction in the backfilled area below the diesel generator building following a seismic event. Since diesel generator operation is only required following failure of at least one non-essential AC bus, top event DU is only questioned following failure of at least one non-essential AC bus (top event EA or EB). Therefore, Split fraction DUS, guaranteed success, is applied where non-essential 4160 VAC Buses 1A and 1B are successful (EA=S*EB=S). This split fraction assignment models the fact that diesel generator operation is not required when non-essential power is available.

Split fraction DU1, probability of DG building soil liquefaction following a ground acceleration of 0.13g, is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction DU2, probability of DG building soil liquefaction following a ground acceleration of 0.36g is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction DU3, probability of DG building soil liquefaction following a ground acceleration of 0.54g is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction DU4, probability of DG building soil liquefaction following a ground acceleration of 0.72g is applied in all other cases (1).

Following DG building soil liquefaction (failure of top event DU), diesel building integrity, including enhanced failure of the diesel generators, is questioned at top event DV. Otherwise, diesel generator operation is questioned at top event DW.

Seismic Top Event DV is defined as diesel generator building (DGB) integrity following soil liquefaction. This top event includes the enhanced failure rate for diesel generators following soil liquefaction, as well as questioning building integrity following diesel generator building liquefaction. If soil liquefaction does not occur at top event DU, top event DV is not questioned and diesel generator failure due to seismic failure (without Diesel Generator building liquefaction) will be questioned in top event DW. Therefore,

Split fraction DVS, guaranteed success, is applied where soil liquefaction did not occur (DU=S).

Split fraction DV1, probability of DGB failure following a ground acceleration of 0.13g is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction DV2, probability of DGB failure following a ground acceleration of 0.36g is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction DV3, probability of DGB failure following a ground acceleration of 0.54g is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction DV4, probability of DGB failure following a ground acceleration of 0.72g is applied in all other cases (1).

The split fraction assignment rule for the independent failure of 4160 VAC Buses 1C and 1D (top events EC and ED) is adjusted to reflect the dependence of these split fractions on diesel generator operation failure. That is, the failure of diesel generators at top event DV impacts the independent split fraction assignments.

Seismic Top Event DW is defined as the likelihood of diesel generator or diesel generator building (DGB) failure following a seismic event in which soil liquefaction does not occur. Since diesel generator operation is only required following failure of at least one non-essential AC bus, this top event is only questioned following failure of at least one of 4160 VAC buses 1A and 1B (top events EA and EB), as well as failure of top event DU, in which case soil liquefaction has occurred. Therefore,

Split fraction DWS, guaranteed success, is applied where non-essential 4160 VAC Buses 1A and 1B are successful (EA=S*EB=S). This split fraction assignment models the fact that diesel generator operation is not necessary when offsite supplies are available.

Split fraction DW1, probability of diesel generator failure following a ground acceleration of 0.13g is applied where soil liquefaction has not occurred (DU=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction DW2, probability of diesel generator failure following a ground acceleration of 0.36g is applied where soil liquefaction has not occurred (DU=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction DW3, probability of diesel generator failure following a ground acceleration of 0.54g is applied where soil liquefaction has not occurred (DU=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction DW4, probability of diesel generator failure following a ground acceleration of 0.72g is applied where soil liquefaction has not occurred (DU=S).

Split fraction DWF, guaranteed failure, is applied in all other cases (1).

The split fraction assignment rule for the independent failure of 4160 VAC Buses 1C and 1D (top events EC and ED) have been adjusted to reflect the dependence on seismic diesel generator failure. Therefore, the assignment rule for split fractions EC2, ED2, EDC and EDD include seismic diesel generator failure (DW=S).

Seismic Top Event EW is defined as the availability of Essential 4160 VAC Bus 1C following a seismic event. This top event models the seismic fragility of the 1C switchgear room fans. Independent top event EC split fractions require 125 VDC Bus C as a support system, so seismic top event EW also requires success of this system for success. Therefore,

Split fraction EW1, probability of 4160 VAC Bus 1C failure following a ground acceleration of 0.13g is applied where 125 VDC Bus C is successful (DC=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction EW2, probability of 4160 VAC Bus 1C failure following a ground acceleration of 0.36g is applied where 125 VDC Bus C is successful (DC=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction EW3, probability of 4160 VAC Bus 1C failure following a ground acceleration of 0.54g is applied where 125 VDC Bus C is successful (DC=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction EW4, probability of 4160 VAC Bus 1C failure following a ground acceleration of 0.72g is applied where 125 VDC Bus C is successful (DC=S).

Split fraction EWF, guaranteed failure, is applied in all other cases (1).

The split fraction assignment rule for the independent failure of 4160 VAC Bus 1C (top event EC) has been adjusted to reflect the dependence on seismic 4160 VAC Bus 1C failure. Therefore, the assignment rules require the success of seismic 4160 VAC Bus 1C (EW=S).

Seismic Top Event XX is defined as the unavailability of long term 125 VDC Bus B following a seismic event. This top event models the seismic failure of room ventilation.

Split fraction XX1, probability of long term 125 VDC Bus B failure following a 0.13g ground acceleration is applied where 4160 VAC Bus 1D is successful (ED=S), short term 125 VDC Bus B is successful (DB=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction XX2, probability of long term 125 VDC Bus B failure following a 0.36g ground acceleration is applied where 4160 VAC Bus 1D is successful (ED=S), short term 125 VDC Bus B is successful (DB=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction XX3, probability of long term 125 VDC Bus B failure following a 0.54g ground acceleration is applied where 4160 VAC Bus 1D is successful (ED=S) and short term 125 VDC Bus B is successful (DB=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction XX4, probability of long term 125 VDC Bus B failure following a 0.72g ground acceleration is applied where 4160 VAC Bus 1D is successful (ED=S) and short term 125 VDC Bus B is successful (DB=S).

Split fraction XXF, guaranteed failure is applied in all other cases (1).

The top event development for the independent failure of long term 125 VDC power (XB) is adjusted to model the seismic top event XX. That is, the assignment rule for split fraction XB1 becomes 4160 VAC Bus 1D success (ED=S), short term 125 VDC Bus B success (DB=S) and seismic 125 VDC Bus B success (XX=S).

Top Event CW (independent failure of circulating water) is set to guaranteed failure for all seismic initiating events.

Top Event SW (independent failure of service water) is set to guaranteed failure for all seismic initiating events.

Top Event TB (independent failure of Turbine Building Closed Cooling Water (TBCCW) is set to guaranteed failure for all seismic initiating events.

Seismic Top Event FX is defined as the likelihood of soil liquefaction near the fire pond dam or threatening fire protection piping due to displacement following a seismic event. Although, the fragilities calculated for the fire protection dam, house, pumps and underground piping screened for all acceleration levels indicate that the fire protection system can screen from consideration a screening fragility equal to 1.0g and 0.3 β_R is added to the model to account for the large uncertainties.

> Split fraction FX1, the probability of soil liquefaction following a 0.13g ground acceleration is applied where the initiating event is SEIS1 (INIT=SEIS1).
>
> Split fraction FX2, the probability of soil liquefaction following a 0.36g ground acceleration is applied where the initiating event is SEIS2 (INIT=SEIS2).
>
> Split fraction FX3, the probability of soil liquefaction following a 0.54g ground acceleration is applied where the initiating event is SEIS3 (INIT=SEIS3).
>
> Split fraction FX4, the probability of soil liquefaction following a 0.72g ground acceleration is applied in all other cases (1).

The complement independent failure top event for the FX seismic event is the independent failure of the fire protection system (top event FP). The split fraction assignment rules for top event FP have been adjusted to reflect their dependence on top event FX. This dependence is modeled as the requirement for the success of FX (FX=S) in the application of split fraction FP2. Following plant walkdowns, the redundant fire pump is modeled as guaranteed failure. Therefore, split fraction FP1 is not used. Should soil liquefaction occur, guaranteed failure (FPF) is assigned for independent top event FP.

Top Event IA (instrument air) is set to guaranteed failure for all seismic initiating events.

Seismic General Transient Split Fraction Assignment Rules

The following split fraction assignment rules provide the logic for the assignment of split fractions in the General Transient Logic.

Seismic Top Event BX is defined as the unavailability of the liquid poison injection tank following a seismic event. This top event models the seismic failure of the boron injection storage tank.

> Split fraction BXS, liquid poison not required, is applied where reactor trip (top event RS) was successful.

Split fraction BX1, the probability that the liquid poison injection tank fails following a 0.13g ground acceleration is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction BX2, the probability that the liquid poison injection tank fails following a 0.36g ground acceleration is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction BX3, the probability that the liquid poison tank fails following a 0.54g ground acceleration is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction BX4, the probability that the liquid poison tank fails following a 0.72g ground acceleration is applied in all other cases (1).

The split fraction assignment rules for independent failure of the liquid poison injection system (top event BI) are adjusted to reflect the dependence on the seismic top event BX. This is modeled by the insertion of the requirement for success of top event BX in the split fraction assignment of all BI split fractions, except the "not necessary" condition, when reactor trip (top event RS) is successful.

Seismic Top Event SX is defined as the unavailability of the condensate storage tank following a seismic event.

Split fraction SX1, the probability that the condensate storage tank fails following a 0.13g acceleration is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction SX2, the probability that the condensate storage tank fails following a 0.36g acceleration is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction SX3, the probability that the condensate storage tank fails following a 0.54g acceleration is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction SX4, the probability that the condensate storage tank fails following a 0.72g acceleration is applied in all other cases (1).

The split fraction assignment rules for independent failure of the condensate storage tank (top event ST) are adjusted to reflect the dependence on the seismic top event SX. This is modeled by the insertion of the requirement for success of top event SX in the split fraction assignment of ST1.

Top Event CP (main condensate) is set to guaranteed failure for all seismic initiating events.

Top Event FW (main feedwater) is set to guaranteed failure for all seismic initiating events.

Seismic Long Term General Transient Module Split Fraction Assignment Rules

Seismic Top Event IX is defined as the unavailability of the isolation condensers following a seismic event.

Split fraction IX1, the probability that the isolation condensers fail following a 0.13g acceleration is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction IX2, the probability that the isolation condensers fail following a 0.36g acceleration is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction IX3, the probability that the isolation condensers fail following a 0.54g acceleration is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction IX4, the probability that the isolation condensers fail following a 0.72g ground acceleration is applied in all other cases (1).

The split fraction assignment rules for independent failure of the isolation condensers (top event IC) are adjusted to reflect the dependence on the seismic top event IX. This is modeled by the insertion of the requirement for success of top event IX in the split fraction assignment for all IC split fractions.

Seismic Top Event LX is defined as the unavailability of offsite power recovery following a seismic event. This top event models the probability of failure of the combustion turbines, combustion turbine fuel oil tank and SBO transformer due to the seismic event. The natural gas supply is not considered in this evaluation.

Split fraction LXS, guaranteed success, is applied where offsite power recovery is not necessary. This split fraction uses the same logic as the guaranteed success term for the independent top event split fraction LP1 assignment rule.

Split fraction LX1, probability of offsite power recovery failure following a 0.13g ground acceleration is applied where a station blackout has occurred (SBO), the isolation condenser is successful (IC=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction LX2, probability of offsite power recovery failure following a 0.36g ground acceleration is applied where a station blackout has occurred (SBO), the isolation condenser is successful (IC=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction LX3, probability of offsite power recovery failure following a 0.54g ground acceleration is applied where a station blackout has occurred (SBO) and the isolation condenser is successful (IC=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction LX4, probability of offsite power recovery failure following a 0.72g ground acceleration is applied where a station blackout has occurred (SBO) and the isolation condenser is successful (IC=S).

Split fraction LXF, guaranteed failure, is applied in all other cases (1).

The split fraction assignment rules for the independent offsite power recovery failure (top event LY) are adjusted to reflect the dependence on the seismic failure of offsite power recovery (top event LX). This adjustment is the insertion of the term for success of the seismic offsite power recovery (LX=S). Should offsite power recovery fail seismically, then independent offsite power recovery is also failed.

Independent Top Event LY is defined as the independent unavailability of offsite power recovery following a seismic event. This top event replaces top event LP in the level 1 model. Top event LY does not model the potential for grid restoration following a seismic event (combustion turbines only). Grid restoration is not modeled following a seismic event since it is assumed that restoration may not be possible for an extended time period. The split fraction assignment for top event LY is the same as top event LP in the Level 1 model.

Seismic Top Event CX is defined as the unavailability of the containment spray/emergency service water (ESW) system following a seismic event. This top event models the seismic failure of the ESW pumps, containment spray pumps, the intake structure and relay chatter of containment spray actuation logic (not recovered).

Split fraction CXS, guaranteed success, is applied where the containment spray/ emergency service water system function is not necessary. This split fraction assignment rule is identical to the guaranteed success assignment rule of the independent failure event (top event CC).

Split fraction CX1, the probability of containment spray/ESW failure following a 0.13g acceleration is applied where the initiating event is SEIS1 (INIT=SEIS1).

Split fraction CX2, the probability of containment spray/ESW failure following a 0.36g acceleration is applied where the initiating event is SEIS2 (INIT=SEIS2).

Split fraction CX3, the probability of containment spray/ESW failure following a 0.54g acceleration is applied where the initiating event is SEIS3 (INIT=SEIS3).

Split fraction CX4, the probability of containment spray/ESW failure following a 0.72g ground acceleration is applied in all other cases (1).

The split fraction assignment rules for the independent failure of containment spray/ESW (top event CC) are adjusted to reflect the dependence on the seismic failure (top event CX). This adjustment is performed by the insertion of the term for the success of seismic containment spray (CX=S). Should containment spray fail due to the seismic event (i.e. at top event CX) the independent event CC is also failed.

Seismic Top Event CZ is defined as the unavailability of the core spray system following a seismic event. This top event reflects the seismic failure of the core spray pumps and failure (relay chatter) of the actuation and control logic (not recovered).

Split fraction CZS, guaranteed success, is applied where the core spray function is not necessary. This split fraction assignment rule is identical to the guaranteed success assignment rule of the independent failure (top event CS).

Split fraction CZ1, the probability of core spray failure following a 0.13g ground acceleration is applied where either 4160 VAC bus 1C or 1D is successful (EC=S+ED=S) and the initiating event is SEIS1 (INIT=SEIS1).

Split fraction CZ2, the probability of core spray failure following a 0.36g ground acceleration is applied where either 4160 VAC bus 1C or 1D is successful (EC=S+ED=S) and the initiating event is SEIS2 (INIT=SEIS2).

Split fraction CZ3, the probability of core spray failure following a 0.54g ground acceleration is applied where either 4160 VAC bus 1C or 1D is successful (EC=S+ED=S) and the initiating event is SEIS3 (INIT=SEIS3).

Split fraction CZ4, the probability of core spray failure following a 0.72g ground acceleration is applied where either 4160 VAC bus 1C or 1D is successful (EC=S+ED=S).

Split fraction CZF, guaranteed failure, is applied in all other cases (1).

The split fraction assignment rules for the independent failure of core spray (top event CS) are adjusted to reflect the dependence on the seismic failure (top event CZ). This adjustment is performed by the insertion of the term for the success of seismic core spray (CZ=S). Should core spray fail due to the seismic event (i.e. at top event CZ) the independent event CS is also failed.

Seismic Recovery Module Split Fraction Assignment Rules

Due to the lower likelihood of success of the Oyster Creek PRA recovery module top events; these events have been assigned guaranteed failure for all levels of seismic acceleration.

Top Event RC is set to guaranteed failure for all conditions.

Top Event RA is set to guaranteed failure for all conditions.

Top Event SD is set to guaranteed failure for all conditions where shutdown cooling is required.

The following tables provide the "Seismic Logic Model Split Fraction Assignment Rules". The tables are broken down by module and the "Seismic Top Events" are shown shaded.

Table 3-9 Seismic Split Fraction Assignment Rules (Support Module)

Split Fraction	Split Fraction Assignment Rule	Comments
DX1	INIT=SEIS1	Seismic failure of block wall No. 53 with impacts on short term 125 VDC Power at Top Event DB.
DX2	INIT=SEIS2	
DX3	INIT=SEIS3	
DX4	1	
DB1	DX=S	Independent failure of short term 125 VDC (Bus B) power.
DBF	1	
DC1	1	Independent short term 125 VDC power (Bus C)
OX1	INIT=SEIS1	Seismic offsite power - conditional on initiating event only.
OX2	INIT=SEIS2	
OX3	INIT=SEIS3	
OX4	1	
OP1	OX=S	Independent offsite power - conditional on seismic top event OX.
OPF	1	
EU1	OP=S*INIT=SEIS1	Seismic failure of startup transformers - conditional on offsite power and seismic initiating event.
EU2	OP=S*INIT=SEIS2	
EU3	OP=S*INIT=SEIS3	
EU4	OP=S	
EUF	1	
EA1	OP=S*DC=S*EU=S	Independent 4160 VAC Bus 1A - condtional on support systems (offsite and DC power) and seismic top event EU.
EAF	1	
EB1	EU=S*OP=S*DB=S*(EA=S+DC=F)	Independent 4160 VAC Bus 1B - conditional on support systems (offsite power and short term 125 VDC bus B) and seismic top event EU.
EBA	EU=S*OP=S*DB=S*EA=F	
EBF	1	
DUS	EA=S*EB=S	Diesel generators not necessary or DG building soil liquefaction - conditional on seismic initiating event only.
DU1	INIT=SEIS1	
DU2	INIT=SEIS2	
DU3	INIT=SEIS3	
DU4	1	

Table 3-9 Seismic Split Fraction Assignment Rules (Support Module)

Split Fraction	Split Fraction Assignment Rule	Comments
DVS	DU=S	DG operation not required or liquefaction did not occur.
DV1	INIT=SEIS1	DG building failure after soil liquefaction (includes enhanced
DV2	INIT=SEIS2	DG failure rate following liquefaction) - conditional on seismic
DV3	INIT=SEIS3	initiating event only.
DV4	1	
DWS	EA=S*EB=S	Diesel Generator operation not required
DW1	DV=S*INIT=SEIS1	Diesel Generator seismic failure given the diesel generator building
DW2	DV=S*INIT=SEIS2	soil did not undergo liquefaction. Conditional on DG building
DW3	DV=S*INIT=SEIS3	liquefaction and siesmic initiating event.
DW4	DV=S	
DWF	1	
EW1	DC=S*INIT=SEIS1	Seismic 4160 VAC Bus 1C (room fan fragility) - conditional
EW2	DC=S*INIT=SEIS2	on support system (short term 125 VDC Bus C) and seismic
EW3	DC=S*INIT=SEIS3	initiating event.
EW4	DC=S	
EWF	1	
EC1	EW=S**DC=S*EA=S	Independent 4160 VAC Bus 1C failure is conditional on:
EC2	EW=S*DC=S*DW=S	EC1 - EW and DC with offsite power (No DG required).
ECF	1	EC2 - EW and DC - no offsite power (DG required).
XC1	EC=S*DC=S	Independent long term 125 VDC power Bus C - conditional
XCF	1	on support systems only.
ED1	DB=S*EB=S*(EC=S+DC=F)	Independent 4160 VAC Bus 1D failure is conditional on:
ED2	DW=S*DB=S*EB=F*(EC=S+DC=F)	ED1 - DB with offsite power (No DG required).
EDA	DB=S*EA=S*EB=S*EC=F	ED2 - DW and DB - no offsite power (DG required).
EDB	DB=S*DC=S*EA=F*EB=S*EC=F	Split fractions EDA through EDD model the various
EDC	DW=S*DB=S*DC=S*EC=F*EA=S*EB=F	combinations of possible common cause failures between
EDD	DW=S*DB=S*DC=S*EC=F*EA=F*EB=F	4160 VAC buses and diesel generators.
EDF	1	

Table 3-9 Seismic Split Fraction Assignment Rules (Support Module)

Split Fraction	Split Fraction Assignment Rule	Comments
XX1 XX2 XX3 XX4 XXF	DB=S*ED=S*INIT=SEIS1 DB=S*ED=S*INIT=SEIS2 DB=S*ED=S*INIT=SEIS3 DB=S*ED=S 1	Seismic long term 125 VDC power Bus B - conditional on 4160 VAC Bus 1D, short term 125 VDC power (DB) success as well as seismic initiating event. Models room fan failure.
XB1 XBF	XX=S*DB=S*ED=S 1	Independent long term 125 VDC power Bus B - conditional on supports and seismic top event XX success.
CWF	1	Circulating water assumed to fail for all seismic events.
SWF	1	Service water assumed to fail for all seismic events.
TBF	1	TBCCW assumed to fail for all seismic events.
FX1 FX2 FX3 FX4	INIT=SEIS1 INIT=SEIS2 INIT=SEIS3 1	Soil liquefaction near fire protection pond and piping - conditional on seismic initiating event only.
FP2 FPF	FX=S 1	Fire protection - conditional on seismic top event FX only. Redundant fire pump is assumed to fail for all seismic events.
IAF	1	Instrument air is assumed to fail for all seismic events.
DP1 DP2 DPF	DB=S*DC=S DB=S+DC=S 1	Independent high drywell pressure acutation is conditonal on the support systems modeled in the level 1 study.
PR1 PR2 PRF	DB=S*DC=S DB=S+DC=S 1	Independent high RPV pressure signal actuation is conditional on the support systems modeled in the level 1.
RL1 RL2 RLF	DB=S*DC=S DB=S+DC=S 1	Independent reactor low level actuation is conditional on the support systems modeled in the level 1 PRA.

Note: The shaded areas indicate seismic top events which are incorporated into the level 1 PRA logic. In all other cases the comment field indicates all changes to the level 1 model (i.e. seismic impacts).

Table 3-10 Seismic Split Fraction Assignment Rules (General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
	RLC:=RF=S	
RSS	INIT=RT	Independent reactor scram failure is conditional on the level 1 support systems only.
RS1	DB=S*(-(INIT=LOIA))*(DP=S+PR=S+RL=S)	
RS5	DB=F*(-(INIT=LOIA))*(DP=S+PR=S+RL=S)	
RS4	(-(INIT=LOIA))	
RS2	1	
RSF	1	
TTS	INIT=TTRIP+INIT=CMSIV+INIT=EPRL	Independent turbine trip failure is conditional on the level 1 modeled dependencies.
TT4	-(INIT=LOFW+INIT=LOIA+INIT=LOCV+INIT=CMSIV+ INIT=TTRIP+INIT=LOTB)*RS=F	
TT1	-(INIT=LOFC)*(-(INIT=LOCV))*CW=S	
TT2	DB=S	
TTF	1	
SX1	INIT=SEIS1	Seismic condensate storage tank failure is conditional on the seismic initiating event only.
SX2	INIT=SEIS2	
SX3	INIT=SEIS3	
SX4	1	
ST1	SX=S	Independent condensate storage tank failure is conditional on the seismic failure (top event SX) only.
STF	1	
CN1	-(INIT=EPRL)*(-(INIT=LOCV))*(-(INIT=CMSIV))*TT=S*CW=S*IA=S	
CNF	1	
BTS	INIT=CMSIV+CN=S+INIT=EPRL	
BT1	1	
BTF	1	
BVS	INIT=EPRH	
BV1	CN=S*RS=S	
BV2	CN=S	
BVF	1	
RFS	INIT=LOFW+(EA=F*EB=F)+TT=F	
RF1	-(INIT=LOIA)	
RF2	1	
RFF	1	

Table 3-10 Seismic Split Fraction Assignment Rules (General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
OM1	TT=S	
OMF	1	
MES	INIT=CMSIV+OM=S	
ME1	OM=F*RF=S	
ME2	1	
MEF	1	
MIS	RF=S	
MI2	1	
MIF	1	
CPF	1	Main condensate is assumed to fail for all seismic events.
FWF	1	Main feedwater is assumed to fail for all seismic events.
MSS	ME=S*OM=F+INIT=CMSIV+RS=F*FW=F+FW=S*(-(INIT=PLOFW))	
MS1	RL=S*OM=S	
MS2	RL=S*ME=F	
MS3	1	
MSF	1	

Note: The shaded areas indicate seismic top events which are incorporated into the level 1 PRA logic. In all other cases the comment field indicates all changes to the level 1 model (i.e. seismic impacts).

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
	SATWS:=RS=F*RP=S*BI=S*(BV=S*FW=S*(VO=S+SO=S) +OL=S*SO=S*(FW=S*(CC=S+(IC=S*MU=S)) +IC=S*MU=S*VR=S*SR=S*CD=S))	Successful ATWS mitigation - requires RPT and SLC with either FW and reliefs, 2) operator mitigation with all relief and (or IC/MU) or 3) operator mitigation with IC/CC and relief
	SHSD:=RS=S*TT=S*ST=S*BV=S*RLC*OM=S*FW=S*VR=S*SR=S	Successful hot shutdown using feedwater and turbine bypass
	SIC:=RS=S*RLC*IC=S*MU=S*(VO=S+SO=S) *((VR=S*SR=S*VS=S)) *(TT=S*BV=S*CN=S+ME=S+MS=S)	Successful shutdown using isolation condenser heat removal
	SRV:=RS=S*CC=S*((FW=S*ME=S*MI=S+RLC*IC=S*(CD=S*VS=S)) *(VO=S+SO=S) +AD=S*ME=S*MI=S*(CP=S+CS=S)) +RS=S*(-RLC)*ME=S*MI=F*FP=S	Successful hot shutdown with relief valve heat removal to the drywell.
	SRVF:=RS=S*TT=S*CC=S*(FW=S*CS=S)*ME=S*MI=S* VO=F*SO=F*PI=S	Successful relief valve failure mitigation.
	SLOFC:=INIT=LOFC*RS=S*TT=S*CS=S*CC=S*AD=S	Successful loss of feedwater control mitigation.
	DWHR:=RS=S*((FW=S*(ME=S+BV=S*CN=S)*MI=S+RLC*IC=S*CD=S *VS=S*VR=S*(VO=S+SO=S) +AD=S*MI=S*ME=S*(CP=S+CS=S*LY=S*(CC=S+FS=S+OS=S)))	Drywell heat removal interim variable.
	RPAL:=PR=S+(RL=S*FW=F)	RPT/IC actuation logic signals
RPS RP1 RP2 RPF	EA=F*EB=F RPAL 1 1	Independent recirculation pump failure to trip.
VS2 VSF	1 1	Independent scram discharge volume failure to isolate.
IY1 IY2 IY3 IY4	INIT=SEIS1 INIT=SEIS2 INIT=SEIS3 1	Seismic failure of isolation condensers. This top event is conditional on the initiating event only.

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
IC1	IY=S*(DB=S*DC=S)*RP=S*RPAL*RS=S*RLC	Independent failure of the isolation condenser system. This event is conditional on seismic top event IY and the previously identified level 1 PRA dependencies.
IC2	IY=S*(DB=S+DC=S)*RP=S*RPAL*RS=S*RLC	
IC3	IY=S*(DB=S*DC=S)*RP=S*RPAL*RLC	
IC4	IY=S*(DB=S*DC=S)*RS=S*RLC	
ICF	1	
CTS	IC=F	Independent condensate transfer system failure.
CT1	EB=S*IC=S*ST=S	
CTF	1	
MUS	IC=F	Independent failure of isolation condenser makeup.
MU1	CT=S*IC=S*RS=S	
MU2	FP=S*IC=S	
MUF	1	
VOS	INIT=RT*TT=S*BV=S*OM=S*FW=S	Independent EMRV failure to open.
VO1	(DB=S+DC=S)*RS=S*(!INIT=RT+(INIT=LOFW+INIT=PLOFW+ +INIT=LOSP)*IC=S)	
VO2	DB=S+DC=S	
VOF	1	
	SON:=RS=S*VO=S*(-(INIT=CMSIV)*(-(INIT=LOIA)) *(-(INIT=TT)+INIT=TT*BV=S)*(-(INIT=EPRL)))	
SOS	SON	Independent safety valve failure to open.
SO1	RS=F*(VO=S*BV=S)+RS=S*(VO=S+BV=S)	
SO2	RS=F*(VO=S+BV=S)+RS=S	
SOF	1	
BXS	RS=S	Boron injection only required following reactor trip failure. Seismic failure is conditional on the initiating event only.
BX1	INIT=SEIS1	
BX2	INIT=SEIS2	
BX3	INIT=SEIS3	
BX4	1	

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
BIS	RS=S	Independent boron injection failure.
BI1	BX=S*(EC=S*ED=S)*RP=S*PR=S*BV=S	
BI2	BX=S*(EC=S+ED=S)*RP=S*PR=S*BV=S	
BI3	BX=S*(EC=S*ED=S)*RP=S*PR=S	
BI4	BX=S*(EC=S+ED=S)*RP=S*PR=S	
BI5	BX=S*(EC=S*ED=S)*RP=S	
BI6	BX=S*(EC=S+ED=S)*RP=S	
BIF	1	
SRS	SO=F+SON	Independent failure of safety valves to reseat.
SR1	SO=S*(RS=F*(VO=S*BV=S)+RS=S*(VO=S+BV=S))	
SR2	SO=S*(RS=S+RS=F*(VO=S+BV=S))	
SRF	1	
VRS	INIT=RT*TT=S*BV=S*OM=S*FW=S+D[illegible]F*DC=F	Independent failure of EMRVs failure to reseat.
VR1	VO=S*RS=S*(INIT=RT+(INIT=LOF[illegible]NIT=PLOFW+INIT=LOSP)*IC=S) *(-(INIT=IEMRV))	
VR2	-(INIT=IEMRV)	
VRF	1	
OLS	RS=S	ATWS operator actions.
OL1	RP=S*BI=S*FW=S*BV=S	
OL2	RP=S*BI=S*FW=S*VR=S*SR=S	
OL3	RP=S*BI=S*VR=S*SR=S	
OL4	RP=S*BI=S*FW=S	
OL5	RP=S*BI=S	
OLF	1	
	SBO:=OP=F*EC=F*ED=F	
LXS	(EC=S+ED=S)+SBO*IC=S*MU=S*VR=S*SR=S	Seismic loss of offsite power recovery failure. This event models the failure of the combustion turbines and associated operator actions following a seismic event. This event is conditional on the seismic initiating event as well as the identified level 1 PRA dependencies.
LX1	SBO*IC=S*INIT=SEIS1	
LX2	SBO*IC=S*INIT=SEIS2	
LX3	SBO*IC=S*INIT=SEIS3	
LX4	SBO*IC=S	
LXF	1	

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
LYS	(EC=S+ED=S)+SBO*IC=S*MU=S*VR=S*SR=S	Independent failure of offsite power recovery. This event
LY1	LX=S*SBO*IC=S*(VR=F+SR=F)	is used in place of top event LP in the level 1 PRA and
LY2	LX=S*SBO*IC=S*(MU=F+FP=F)	represents the independent failure to recover offsite power
LYF	1	following a seismic event (does not model grid restoration).
	LPS:=LY=S*VS=S	
LQS	-(INIT=LOSP)+EC=S*ED=S+EC=F*ED=F	Independent failure to cross-tie essential buses.
LQ1	INIT=LOSP*(EC=F*EW=S*EY=S*DC=S+ED=F*EX=S*EZ=S*DB=S)	
	*(EC=S+ED)	
LQF	1	
CD1	ST=S*EA=S*EC=S*ED=S	Independent failure of the control rod drive hydraulic system.
CD2	ST=S*EA=F*EB=F*(EC=S*ED=S)	
CD3	ST=S*EA=F*EB=F*(EC=S+ED=S+SBO*LPS*IC=S)	
CD4	ST=S*EB=S*ED=S	
CDF	1	
	LQRECOV:=INIT=LOSP*(LQ=S*VS=S)*(EC=F+ED=F)*(EC=S+ED=S)	
CXS	SHSD+SIC	Containment spray not questioned.
CX1	INIT=SEIS1	Seismic failure is conditional on the initiating event only.
CX2	INIT=SEIS2	
CX3	INIT=SEIS3	
CX4	1	
CCS	SHSD+SIC	Independent failure of the containment spray system.
CC3	CX=S*(-(INIT=LOIS))*(EC=S*ED=S+LQRECOV)*RS=S*IC=S*MU=F	
CC7	CX=S*(EC=S+ED=S)*RS=S*IC=S*MU=F	
CC4	CX=S*(-(INIT=LOIS))*(EC=S*ED=S+LQRECOV)*RS=S*IC=F	
CC8	CX=S*(EC=S+ED=S)*RS=S*IC=F	
CC5	CX=S*(-(INIT=LOIS))*(EC=S*ED=S+LQRECOV)*RS=S	
CC9	CX=S*(EC=S+ED=S+SBO*LPS)*RS=S	
CC6	CX=s*(-(INIT=LOIS))*(EC=S*ED=S+LQRECOV)	
CCF	1	
TCS	SHSD+SIC	Independent operator action to switch containment spray
TC1	CC=S	to torus cooling mode.
TCF	1	

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
FSS	SHSD+SIC	Independent failure of operators to align fire protection through core spray.
FS1	FP=S	
FSF	1	
OSS	FS=S	Independent failure of operators to align core spray suction to the condensate storage tank.
OS1	ST=S	
OSF	1	
OVS	SATWS+SHSD+SIC+SRV+CC=S	Independent failure of the containment vent.
OV1	DWHR*(XC=S+LY=S+(LQ=S*(-(EC=F*ED=F))))	
OV2	XC=S+(LY=S+(LQ=S*(-ED=F*EC=F)))	
OVF	1	
	ADN:=RS=S*FW=S*(VO=S+SO=S)*CC=S	
ADS	ADN+SHSD+SIC+SATWS	Independent failure of the automatic depressurization system.
AD1	(XB=S*XC=S+(DB=S*DC=S*VR=F))*DP=S*RL=S*FW=F*(-SON)	
AD2	(XB=S+XC=S+SBO*LPS+((DB=S+DC=S)*VR=F)) *DP=S*RL=S*FW=F*(-SON)	
AD7	DB=S*DC=S*RP=S*RPAL*RLC*RS=S*IC=S*VR=F*FW=F	
AD3	(XB=S+XC=S+((DB=S+DC=S)*VR=F))*(IC=F+VR=F+SR=F+VS=F) *FW=F	
AD4	(XB=S+XC=S+SBO*LPS+((DB=S+DC=S)*VR=F))*(MU=F*CD=F*FW=F)	
AD5	(XB=S+XC=S+SBO*LPS+((DB=S+DC=S)*VR=F))*CC=F	
ADF	1	
CZS	SHSD+SIC	Seismic failure of the core spray system. This event is conditional on the seismic initiating event and the previously idenifed level 1 PRA dependencies.
CZ1	(EC=S+ED=S)*INIT=SEIS1	
CZ2	(EC=S+ED=S)*INIT=SEIS2	
CZ3	(EC=S+ED=S)*INIT=SEIS3	
CZ4	(EC=S+ED=S)	
CZF	1	

Table 3-11 Seismic Split Fraction Assignment Rules (Long Term General Transient Module)

Split Fraction	Split Fraction Assignment Rule	Comments
CS1	CZ=S*(ED=S*EC=S)*(RL=S+DP=S)*VS=S	Independent failure of the core spray system. This event is conditional on seismic top event CZ and the previously identified level 1 PRA dependencies.
CS2	CZ=S*ED=S*(RL=S+DP=S)*VS=S	
CS3	CZ=S*EC=S*(RL=S+DP=S)*VS=S	
CS4	CZ=S*EC=S*ED=S*VS=S	
CS6	CZ=S*ED=S+EC=S+(SBO*LPS)	
CS5	CZ=S*(EC=F*ED=F)*FP=S*VS=S	
CS7	CZ=S*EC=S*ED=S	
CS8	CZ=s*(EC=S+ED=S)	
CSF	1	
PIS	SHSD+SIC+SATWS	Independent failure of the primary containment isolation. Operator action at split fraction PI2 required due to relay chatter issues.
PI2	VS=S	
PIF	1	
RIS	SHSD+SIC+SATWS	Independent failure of the reactor building isolation system.
RI1	RL=S+DP=S	
RI2	1	
RIF	1	
SGS	SHSD+SIC+SATWS	Independent failure of the standby gas treatment system. Guaranteed failed.
SGF	1	
SG1	EC=S*ED=S	
SG2	EC=S	
SG3	ED=S	
SGF	1	

Note: The shaded areas indicate seismic top events which are incorporated into the level 1 PRA logic. In all other cases the comment field indicates all changes to the level 1 model (i.e. seismic impacts).

Table 3-12 Seismic Split Fraction Assignment Rules (Recovery Module)

Split Fraction	Split Fraction Assignment Rule	Comments
RDS RD1 RDF	(DC=S+CC=S+OV=S) (-(DB=F*DC=F)) 1	Independent failure of the recovery of DC power.
RCS RCF	CC=S 1	Containment spray recovery not required. Otherwise, recovery failed for all seismic events.
RAS RAF	(IA=S+RC=S+OV=S) 1	Recovery of instrument air not necessary. Otherwise, recovery failed for all seismic events.
RVS RV1 RVF	OV=S*RC=S DWHR*((IA=F*RA=S*XB=S)+(DC=F*RD=S*IA=S)) 1	Independent failure of the recovery of the containment vent.
SDS SDF	(RV=S+RC=S) 1	Shutdown cooling not necessary. Otherwise, SDC assumed to fail for all seismic events.

Note: This table is provided for information on the level 1 PRA logic model. Seismic impacts have only been made to top events RA, RC and SD, as noted above.

3.1.5.2 Quantification of the Seismic Model

The quantification of the seismic logic model is accomplished through use of the computer code RISKMAN. The seismic top events are incorporated within the Level 1 plant model structure, as described above, with the impacts of each seismic top event then incorporated within the logic of the impacted top events. This logic structure is then quantified for the four seismic event initiators corresponding to the four ground accelerations modeled.

The quantification cutoff used to quantify this model was 10^{-12}. The results of the quantification of the seismic model are presented in the report section below.

3.1.5.3 Presentation of the Seismic Model Results

This report section summarizes the major results of the Oyster Creek seismic PRA by first providing the calculated total mean core damage frequency, and then presenting the results from various perspectives. These perspectives are:

- Initiating Event Importance
- System Importance
- Individual Sequence Importance
- Plant Damage State Importance

Each of the above perspectives offers insights into the response of the Oyster Creek Nuclear Generating Station following seismically initiated events.

The total core damage frequency from seismic initiating events is shown on Table 3-13 as 3.6×10^{-6}. Total unaccounted frequency from the summation of all scenarios that were truncated below the quantification cutoff of 10^{-12} is 3×10^{-8}, or equivalent to 0.8% of total core damage frequency.

Initiating Event Importance

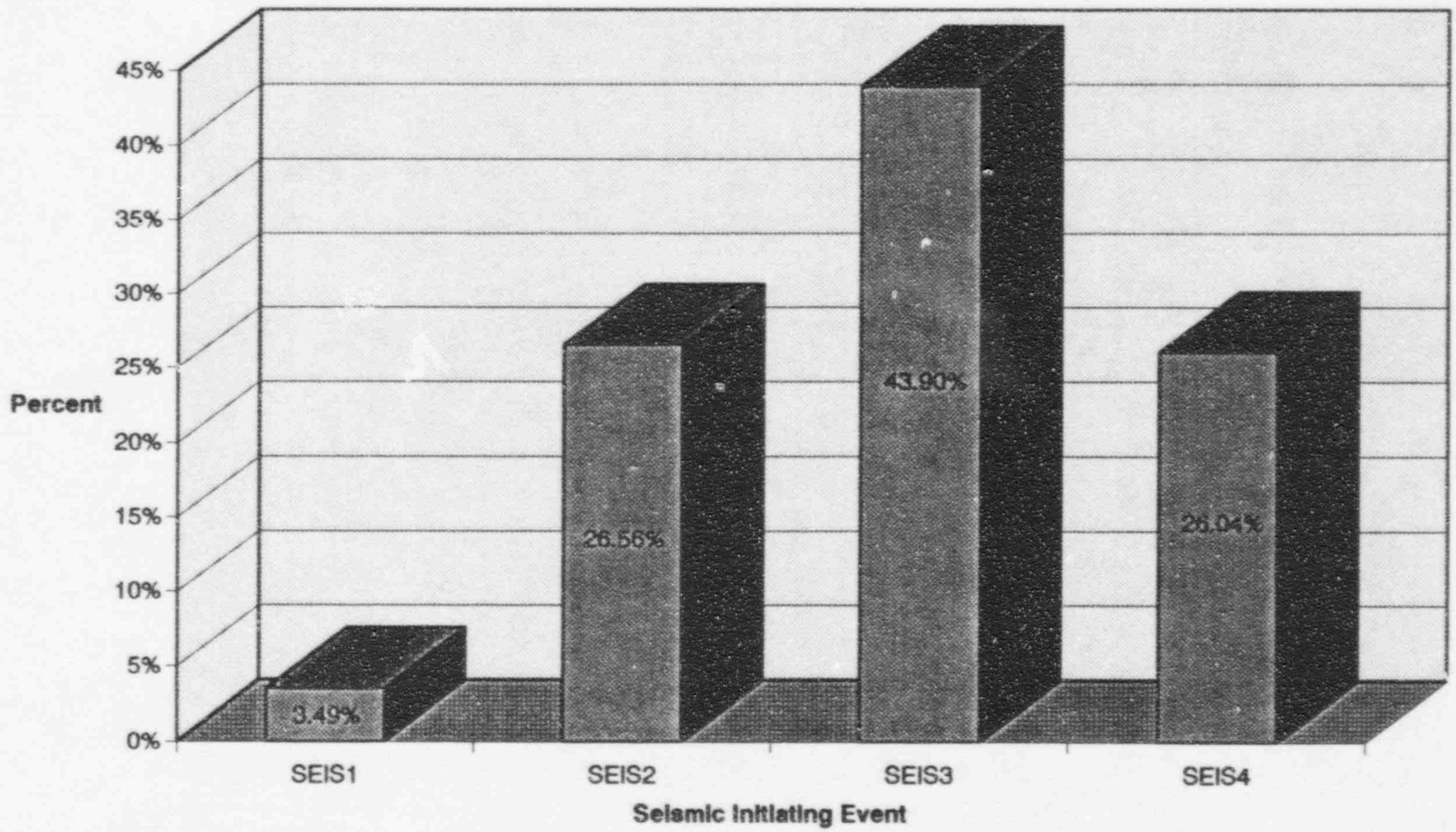
While the lowest ground acceleration initiating event (SEIS1) accounts for more than 99% of the total seismic initiating event frequency (see Table 3-13), this initiating event accounts for only 3.5% of total core damage frequency due to seismic events. This is primarily due to the low likelihood of system failure due to seismic shocks at low ground acceleration values, leading to a relatively low conditional core damage frequency. In other words, the plant design is relatively tolerant of ground force seismic events, up to and including the design basis safe shutdown earthquake.

The 0.36 g force seismic event (SEIS2) accounts for 26.5% of total seismic core damage frequency. The conditional core damage frequency for this initiating event is 0.04. In other words, this initiating event, which corresponds to a ground acceleration in the range of twice the safe shutdown earthquake (0.26 to 0.46g), is successfully mitigated in about 96% of the analyzed scenarios.

Table 3-13 Summary of Oyster Creek Seismic Results

Initiator Ground Accel. Initiator Frequency	SEIS1 0.13g 7.44E-03	SEIS2 0.36g 2.73E-05	SEIS3 0.54g 2.72E-06	SEIS4 0.72g 9.83E-07	TOTALS 0.05-0.81g 7.47E-03	Relative Contribution
Damage State						
MIA-x	6.03E-09	4.17E-07	2.07E-07	1.11E-08	6.41E-07	17.66%
MJA-x	2.67E-09	3.98E-10	3.57E-11	1.56E-12	3.11E-09	0.09%
MKC-x	8.82E-11	6.75E-09	3.35E-09	1.66E-10	1.04E-08	0.29%
MKH-x	1.79E-09	7.15E-12			1.79E-09	0.05%
M-TOTALS	***1.06E-08***	***4.24E-07***	***2.10E-07***	***1.12E-08***	***6.56E-07***	***18.08%***
NID-x	1.37E-11	1.23E-09	4.45E-10	1.18E-11	1.70E-09	0.05%
NIE-x	1.09E-10	3.78E-08	5.17E-08	4.84E-09	9.45E-08	2.60%
NIF-x	8.86E-12	8.82E-10	1.21E-09	1.01E-10	2.20E-09	0.06%
NIG-x		2.78E-09	2.42E-08	6.62E-09	3.36E-08	0.92%
NIH-x	2.70E-09	1.74E-07	1.01E-06	8.22E-07	2.01E-06	55.28%
NJA-x	2.13E-11	1.35E-11	2.87E-12		3.76E-11	0.00%
NJH-x	1.91E-11	1.31E-10	2.12E-10	2.06E-10	5.69E-10	0.02%
NKF-x		6.10E-10	8.37E-10	6.88E-11	1.52E-09	0.04%
NKH-x	8.74E-11	2.74E-09	1.68E-08	1.31E-08	3.27E-03	0.90%
N-TOTALS	***2.95E-09***	***2.20E-07***	***1.10E-06***	***8.47E-07***	***2.17E-06***	***59.88%***
OIA-x	8.19E-08	5.21E-08	2.16E-08	1.06E-09	1.57E-07	4.32%
OJA-x	8.10E-09	6.20E-11	2.67E-11	2.73E-12	8.19E-09	0.23%
OKC-x	1.33E-09	7.66E-10	3.13E-10	7.56E-12	2.42E-09	0.07%
O-TOTALS	***9.13E-08***	***5.30E-08***	***2.20E-08***	***1.07E-09***	***1.67E-07***	***4.61%***
PID-x	2.56E-10	3.10E-11	1.04E-12		2.88E-10	0.01%
PIE-x	2.60E-09	1.98E-07	1.55E-07	2.78E-08	3.83E-07	10.56%
PIF-x	1.17E-08	4.84E-09	3.60E-09	6.20E-10	2.07E-08	0.57%
PIG-x	3.75E-11	3.61E-09	7.26E-09	1.05E-09	1.20E-08	0.33%
PIH-x	6.89E-09	5.63E-08	8.84E-08	5.53E-08	2.07E-07	5.70%
PJA-x	2.26E-10	1.51E-11	7.84E-12	4.17E-12	2.53E-10	0.01%
PJH-x			3.98E-12	9.67E-12	1.37E-11	0.00%
PKF-x	2.23E-10	3.18E-09	2.46E-09	4.23E-10	6.29E-09	0.17%
PKH-x	1.04E-10	8.78E-10	1.45E-09	8.78E-10	3.31E-09	0.09%
P-TOTALS	***2.20E-08***	***2.67E-07***	***2.58E-07***	***8.61E-08***	***6.33E-07***	***17.44%***
TOTALS CDF	**1.27E-07**	**9.64E-07**	**1.59E-06**	**9.45E-07**	**3.63E-06**	**100.00%**
CDF Percentages	**3.49%**	**26.56%**	**43.90%**	**26.04%**		
Conditional CDF	**1.70E-05**	**0.04**	**0.59**	**0.96**		
Unaccounted	5.07E-09	1.47E-08	7.57E-09	2.19E-09	2.96E-08	
Percentages	0.14%	0.41%	0.21%	0.06%	0.81%	

Initiating Event Contribution to CDF



The 0.54 g force seismic event (SEIS3) accounts for 44% of seismic core damage frequency, primarily due to the low initiating event frequency. In other words, with a conditional core damage frequency of 0.59, more than half of the scenarios analyzed for this initiating event result in core damage, with only 41% of the scenarios resulting in successful mitigation.

The 0.72 g force seismic event (SEIS4) accounts for the remaining 26% of core damage frequency. The conditional core damage at the ground acceleration is 0.96. This indicates that 96% of the seismic frequency results at this acceleration results in core damage.

System Importance

Systems that support or back up the function of the Isolation Condensers dominate the seismic results, as shown in Table 3-14. These systems are primarily the condensate storage tank integrity and the fire protection system. In fact, of the top 25 scenarios shown in Table 3-15, which account for about half (49%) of all seismic core damage frequency, only scenario 7 does not contain seismic failure of both condensate transfer and fire protection water.

On of the reasons for the high contribution of the failure of isolation condenser support is the relatively fragile offsite power grid and on site 4160 VAC when compared with isolation condenser cooling. In addition, many of the seismic initiated sequences contain failure of 4160 VAC power in some form. As can be seen on Table 3-14, 6 of 10 of the dominant top events are associated with 4160 VAC.

Individual Sequence Importance

The 25 highest frequency scenarios account for 49% of total seismic core damage frequency. The top 10 of these scenarios are summarized below:

- Scenario 1 results in core damage following seismically induced failure of offsite power, diesel generator building failure following soil liquefaction, the diesel generator building fails (including the diesel generators), failure of fire protection and condensate storage tank integrity and seismic failure of the combustion turbines. This results in a site blackout with IC makeup through fire protection failed.

- Scenario 2 results is similar to scenario 1, except that seismic failures occur at a greater acceleration level.

- Scenario 3 results in core damage following seismically induced failure of 125 VDC bus B due to Block Wall No. 53 failure, failure of offsite power, diesel generator building liquefaction, failure of the diesel generator building (and diesels) in addition to failure of fire protection and the condensate storage tank. Following the failure to recover offsite power through the combustion turbines (seismically failed) core damage occurs.

Table 3-14 Top Event Importance (Sorted by Fussel-Vesely Importance)

No.	Top Event	Fussel-Vesely Importance	Top Event Description
1	FX	7.40E-01	Fire Protection System Underground Piping Liquefaction
2	SX	3.09E-01	Seismic Failure of the Condensate Storage Tank
3	EW	2.27E-01	Seismic Failure of the 4160 VAC 1C Bus
4	LX	2.15E-01	Seismic Failure of Offsite Power Recovery (Combustion Turbines)
5	XX	1.55E-01	Seismic Failure of Long Term 125 VDC Bus B
6	DU	8.82E-02	Liquefaction of the Diesel Generator Building
7	DV	8.31E-02	Failure of the Diesel Generator Building after Liquefaction
8	CX	7.93E-02	Seismic Failure of the Containment Spray/ESW
9	DW	6.08E-02	Seismic Failure of the Diesel Generators w/o Liquefaction
10	OX	5.79E-02	Seismic Failure of Offsite Power
NOTE:	Seismic top events contain the seismic fragilities which result in impacts (degradation or failure) of independent OCPRA top events. Often top events contain multiple fragilities.		

Table 3-15 Seismic Fragility Fussel-Vesely Importance

No.	Fragility	Top Event	Fussel Vesely	Seismic Failure Description
1	FRAG10	SX	3.09E-01	Condensate Storage Tank
2	FRAG05	EW	2.27E-01	Switchgear Room Fans
3	FRAG04	XX	1.55E-01	Battery Room Fans
4	FRAG02	DU	8.82E-02	Diesel Generator Building Liquefaction
5	FRAG20	LX	8.39E-02	Plant Transformers
6	FRAG01	OX	5.80E-02	Offsite Power (ceramic insulators)
7	FRAG25	LX	5.80E-02	Combustion Turbines
8	FRAG07	DV	5.13E-02	Diesel Generator after DG Building Liquefaction
9	FRAG33	CX	5.07E-02	Relay Panel 8A & 8B (Containment Spray - relay chatter)
10	FRAG18	IY	4.97E-02	Isolation Condenser

Table 3-16 Independent Top Event Importance Fussel-Vesely Importance

No.	Top Event	Fussel-Vesely	Independent Top Event Descriptions
1	VR	4.35E-02	EMRV Failure to Reclose
2	LY	3.29E-02	Failure of the Recovery of Offsite Power (combustion turbines only)
3	CC	2.87E-02	Containment Spray/Emergency Service Water
4	CS	2.72E-02	Core Spray System
5	EC	2.29E-02	4160 VAC Bus 1C (including diesel generator)
6	ED	7.37E-03	4160 VAC Bus 1D (including diesel generator)
7	AD	2.99E-03	Automatic Depressurization
8	OV	2.70E-03	Containment Vent
9	VS	2.41E-03	Scram Discharge Volume Isolation
10	MU	2.24E-03	Isolation Condenser Shell Side Makeup

- Scenario 4 results in core damage following seismically induced failure of the essential 4160 VAC Switchgear 1C and long term 125 VDC Bus B, in addition to the failure of fire protection and the condensate storage tank. Following the depletion of batteries ADS fails failing injection through core spray.

- Scenario 5 results in core damage following seismic failure of offsite power and failure of essential 4160 VAC bus 1C and long term 125 VDC bus B, in addition to failure of fire protection and the condensate storage tank. This scenario is similar to scenario 4 above, exception that this scenario also contains the seismic failure of offsite power.

- Scenario 6 is similar to scenario 2, above, except that the failure of the isolation condenser reduces the amount of time available for offsite power recovery to less than the time required and offsite power recovery is assumed failed.

- Scenario 7 is an independent failure scenario where the failure of an EMRV to reclose and the independent failure of the core spray system result in core damage. It should be noted that no seismic failures appear in the sequence and the sequence is initiated by SEIS1 the lowest ground acceleration.

- Scenario 8 is similar to scenario number 1 with the exception that the diesel generators do not failure due to the liquefaction of the diesel generator building (top event DV) as in scenario 1 but seismically at top event DW.

- Scenario 9 is the result of the failure of 125 VDC power due to block wall failure, the failure of offsite and onsite 4160 VAC sources (top event OX, DU and DV) as well as fire protection, condensate storage tank integrity and isolation condensers. The failure of isolation condensers results in the failure to recover offsite power.

- Scenario 10 is the result of the failure of 125 VDC power due to block wall failure, the failure of offsite and onsite 4160 VAC sources (top event OX, DU and DV) as well as fire protection, condensate storage tank integrity and the failure to recover offsite power at top event LX.

Of the top ten scenarios described 4 are the result of SEIS4, 3 are the result of SEIS3 and 2 are the result of SEIS2. The first and only scenario from a safe shutdown earthquake event (SEIS1) in the top ten is scenario 7. This scenario is similar to a loss of main condensate injection with failure of at least one EMRV to close and failure of core spray injection.

Plant Damage State Importance

The Oyster Creek seismic PRA uses the same plant damage state designators and assignment rules as used in the Level 1 PRA. A four character plant damage state naming scheme is used with the first two character providing the more significant information.

The first character indicates a combination of reactor pressure vessel status and drywell floor conditions at the time of vessel breach. The following is the general description of the first character of the plant damage state:

M - High RPV pressure and Wet drywell floor
N - High RPV pressure and Dry drywell floor
O - Low RPV pressure and Wet drywell floor
P - Low RPV pressure and Dry drywell floor

The second character of the plant damage state indicates the status of primary containment at the time of vessel breach. The following is a general description of the second character of the plant damage state assignment:

I - Primary containment INTACT
J - Primary containment BYPASSED
K - Primary containment FAILED EARLY
L - Primary containment FAILED LATE

The second and third characters of the plant damage state provide information on the status of core debris (third character) and the status of secondary containment (fourth character). Additional information is available on plant damage state assignment in the Level 1 OCPRA.

A large fraction of the seismic PRA core damage is a result of core damage at high pressure (78%). The remainder of the 22% is a result of low pressure scenarios in which ADS operation has been successful. RPV pressure high with the drywell floor dry dominate the results (60%) and are the result of total losses of DC power (as in the case of the Level 1 OCPRA) or the failure to depressurize. Other scenarios which can lead to this state include long term station blackout with station battery depletion. For comparison the Level 1 OCPRA core damage at high pressure contributed only 41% to the plant damage states assignment.

The results shown in Table 3-13 confirm that damage states with the containment intact dominate (>98%) the seismic model results. This is general due to the fact that the seismic analysis did not identify any potential for seismically induced LOCA events outside primary containment. The remaining 2% is contributed by reactor scram failures and bypass events (Top Event VS failure). Table 3-13 provides a complete list of plant damage states.

The table below provides a comparison of the second character of the seismic plant damage

OCPRA plant damage states. This table illustrates the larger fraction of containment intact endstate.

Plant Damage State Comparison

Character	Seismic PRA Second PDS Character	Level 1 OCPRA Second PDS Character
I	>98%	83%
J	0.3%	6%
K	1.6%	7%
H	0%	4%

In the seismic analysis the top five damage states (MIAx, NIHx, OIAx, PIEx and PIHx) contribute approximately 93% of the total core damage frequency. These damage states compare favorably (fractionally) to those in the Level 1 OCPRA. In that study the PIFx, NIFx and OIAx were the top three plant damage states; contributing 75%. The differences in third character are the result of an additional failure of containment spray (water to core debris) and additional failures of secondary containment features (guaranteed failed) in the seismic analysis. The similarities in plant damage states diverge since bypass events and reactor scram failures (ATWS) events begin to contribute in the Level 1 OCPRA plant damage states.

Table 3-17 Dominant Sequence Listing

No.	Initiating Event	Frequency	Sequence Percent	Cumulative Percent	Failed Events	Split Fraction Descriptions	Damage State
1	SEIS3	2.76E-07	7.61%	7.6%	OX3	Seismic Failure of Offsite Power	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LX3	Seismic Offsite Power Recovery Failure	
2	SEIS4	2.42E-07	6.66%	14.3%	OX4	Seismic Failure of Offsite Power	NIHX
					DU4	Diesel Generator Building Liquefaction	
					DV4	Diesel Generator Building Failure	
					FX4	Fire Protection Underground Piping Liquefaction	
					SX4	Condensate Storage Tank Failure	
					LX4	Seismic Offsite Power Recovery Failure	
3	SEIS4	1.71E-07	4.71%	19.0%	DX4	Short Term 125 VDC B (Wall No 53)	NIHX
					OX4	Seismic Failure of Offsite Power	
					DU4	Diesel Generator Building Liquefaction	
					DV4	Diesel Generator Building Failure	
					FX4	Fire Protection Underground Piping Liquefaction	
					SX4	Condensate Storage Tank Failure	
					LX4	Seismic Offsite Power Recovery Failure	
4	SEIS2	1.22E-07	3.37%	22.3%	EW2	4160 VAC Bus 1C	MIAX
					XX2	Long Term 125 VDC Bus B	
					FX2	Fire Protection Underground Piping Liquefaction	
					SX2	Condensate Storage Tank Failure	
5	SEIS2	1.08E-07	2.98%	25.3%	OX2	Seismic Failure of Offsite Power	MIAX
					EW2	4160 VAC Bus 1C	
					XX2	Long Term 125 VDC Bus B	
					FX2	Fire Protection Underground Piping Liquefaction	
					SX2	Condensate Storage Tank Failure	

Table 3-17 Dominant Sequence Listing

No.	Initiating Event	Frequency	Sequence Percent	Cumulative Percent	Failed Events	Split Fraction Descriptions	Damage State
6	SEIS4	9.61E-08	2.65%	28.0%	OX4	Seismic Failure of Offsite Power	NIHX
					DU4	Diesel Generator Building Liquefaction	
					DV4	Diesel Generator Building Failure	
					FX4	Fire Protection Underground Piping Liquefaction	
					SX4	Condensate Storage Tank Failure	
					IY4	Isolation Condenser	
7	SEIS1	7.43E-08	2.05%	30.0%	VR2	EMRVs Fail to Reclose	OIAW
					CS1	Core Sprays Failure	
8	SEIS3	7.39E-08	2.04%	32.1%	OX3	Seismic Failure of Offsite Power	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DW3	Diesel Generators	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LX3	Seismic Offsite Power Recovery Failure	
9	SEIS4	6.81E-08	1.88%	33.9%	DX4	Short Term 125 VDC B (Wall No 53)	NIHX
					OX4	Seismic Failure of Offsite Power	
					DU4	Diesel Generator Building Liquefaction	
					DV4	Diesel Generator Building Failure	
					FX4	Fire Protection Underground Piping Liquefaction	
					SX4	Condensate Storage Tank Failure	
					IY4	Isolation Condenser	
10	SEIS3	6.77E-08	1.87%	35.8%	DX3	Short Term 125 VDC B (Wall No 53)	NIHX
					OX3	Seismic Failure of Offsite Power	
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LX3	Seismic Offsite Power Recovery Failure	

Table 3-17 Dominant Sequence Listing

No.	Initiating Event	Frequency	Sequence Percent	Cumulative Percent	Failed Events	Split Fraction Descriptions	Damage State
11	SEIS3	5.48E-08	1.51%	37.3%	OX3	Seismic Failure of Offsite Power	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					LX3	Seismic Offsite Power Recovery Failure	
12	SEIS3	5.47E-08	1.51%	38.8%	OX3	Seismic Failure of Offsite Power	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					IY3	Isolation Condenser	
13	SEIS3	5.06E-08	1.40%	40.2%	OX3	Seismic Failure of Offsite Power	MIAX
					DU3	Diesel Generator Building Liquefaction	
					EW3	4160 VAC Bus 1C	
					XX3	Long Term 125 VDC Bus B	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
14	SEIS3	4.09E-08	1.13%	41.3%	OX3	Seismic Failure of Offsite Power	NIHX
					DW3	Diesel Generators	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LX3	Seismic Offsite Power Recovery Failure	
15	SEIS2	3.48E-08	0.96%	42.3%	OX2	Seismic Failure of Offsite Power	MIAX
					DU2	Diesel Generator Building Liquefaction	
					EW2	4160 VAC Bus 1C	
					XX2	Long Term 125 VDC Bus B	
					FX2	Fire Protection Underground Piping Liquefaction	
					SX2	Condensate Storage Tank Failure	

Table 3-17 Dominant Sequence Listing

No.	Initiating Event	Frequency	Sequence Percent	Cumulative Percent	Failed Events	Split Fraction Descriptions	Damage State
16	SEIS3	2.92E-08	0.80%	43.1%	EU3	Startup Transformers	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
17	SEIS3	2.80E-08	0.77%	43.9%	OX3	Seismic Failure of Offsite Power	MIAX
					EW3	4160 VAC Bus 1C	
					XX3	Long Term 125 VDC Bus B	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
18	SEIS3	2.45E-08	0.68%	44.5%	OX3	Seismic Failure of Offsite Power	MIAX
					DU3	Diesel Generator Building Liquefaction	
					EW3	4160 VAC Bus 1C	
					XX3	Long Term 125 VDC Bus B	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					CZ3	Core Spray Failure	
19	SEIS2	2.39E-08	0.66%	45.2%	OX2	Seismic Failure of Offsite Power	MIAX
					EC2	Independent Failure of 4160 VAC Bus 1C	
					XX2	Long Term 125 VDC Bus B	
					FX2	Fire Protection Underground Piping Liquefaction	
					SX2	Condensate Storage Tank Failure	
20	SEIS3	2.35E-08	0.65%	45.9%	OX3	Seismic Failure of Offsite Power	PIEX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					CX3	Containment Spray/ESW	

Table 3-17 Dominant Sequence Listing

No.	Initiating Event	Frequency	Sequence Percent	Cumulative Percent	Failed Events	Split Fraction Descriptions	Damage State
21	SEIS3	2.21E-08	0.61%	46.5%	DX3	Short Term 125 VDC B (Wall No 53)	NIHX
					OX3	Seismic Failure of Offsite Power	
					DU3	Diesel Generator Building Liquefaction	
					EW3	4160 VAC Bus 1C	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LX3	Seismic Offsite Power Recovery Failure	
22	SEIS4	2.19E-08	0.60%	47.1%	OX4	Seismic Failure of Offsite Power	NIHX
					DU4	Diesel Generator Building Liquefaction	
					DV4	Diesel Generator Building Failure	
					FX4	Fire Protection Underground Piping Liquefaction	
					SX4	Condensate Storage Tank Failure	
					LX4	Seismic Offsite Power Recovery Failure	
23	SEIS3	2.11E-08	0.58%	47.6%	EW3	4160 VAC Bus 1C	MIAX
					XX3	Long Term 125 VDC Bus B	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
24	SEIS2	2.03E-08	0.56%	48.2%	XX2	Long Term 125 VDC Bus B	PIEX
					FX2	Fire Protection Underground Piping Liquefaction	
					SX2	Condensate Storage Tank Failure	
					CX2	Containment Spray/ESW	
25	SEIS3	1.97E-08	0.54%	48.7%	OX3	Seismic Failure of Offsite Power	NIHX
					DU3	Diesel Generator Building Liquefaction	
					DV3	Diesel Generator Building Failure	
					FX3	Fire Protection Underground Piping Liquefaction	
					SX3	Condensate Storage Tank Failure	
					LY2	Independent Failure of Offsite Power Recovery	

3.1.5.4 Results Due to the NUREG-1488 Hazard Curves

The seismic model results shown above are due to the seismic hazard curves presented in EPRI report NP-6395-D. As a sensitivity study, the hazard curves presented in NUREG-1488 (Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains) were evaluated.

NUREG-1488 Seismic Event Frequencies		
Initiating Event	Earthquake Range	NUREG-1488 Hazard Estimates
SEIS1	0.05g - 0.31g	7.90E-04
SEIS2	0.31g - 0.46g	2.50E-05
SEIS3	0.46g - 0.77g	1.36E-05
SEIS4	0.77g - 1.01g	2.83E-06

The total core damage frequency from seismic initiating events using the NUREG-1488 hazard frequency curves was 6.36E-06, with a total unaccounted frequency (due to scenarios that were truncated below the quantification cutoff of 1E-12) of 4.24E-08. These results are summarized in Table 3-18.

Initiating Events

The EPRI and the NUREG-1488 hazard curves present different acceleration ranges and different occurrence intervals or frequencies. This presents complications when comparing the results of the two sets of hazard curves.

Consider the SEIS1 initiating event. The EPRI Hazard Curve SEIS1 initiator ranges from a low ground acceleration of 0.006g to a high ground acceleration level of 0.26g; whereas the NUREG-1488 Hazard Curve SEIS1 initiating event low ground acceleration is 0.05g to a high ground acceleration level of 0.31g. The inclusion of this additional acceleration was necessary since the hazard curves do not reflect the same ranges of acceleration. The EPRI hazard curves begin at lower acceleration levels and end at lower acceleration levels. It was therefore necessary to adjust the ranges of the initiating event ground accelerations to avoid addition of large amounts of additional frequency to either end of the range and therefore skew results. Without this adjustment all seismic risk would result from NUREG-1488 SEIS4 initiating event due to the extremely large range of ground motion modeled in the initiating event. As currently modeled each hazard curve is equally divided among seismic initiator for its entire range which is appropriate.

An additional outcome of the varied acceleration ranges and frequencies are perspectives and conclusions one might not expect. For example, using the EPRI Hazard Curves, SEIS1 initiating

Table 3-18 Summary of Seismic Results Using NUREG-1488 (LLNL Hazard Curves)

Initiator	SEIS1	SEIS2	SEIS3	SEIS4	TOTALS	
Ground Acceleration	0.181g	0.385g	0.615g	0.890g	0.05-101g	Relative
Initiator Frequency	7.90E-04	2.50E-05	1.36E-05	2.83E-06	8.31E-04	Contribution
Damage State						
MIA-x	2.32E-09	1.95E-07	2.12E-07	4.32E-10	4.10E-07	6.45%
MJA-x	4.24E-10	6.28E-10	4.65E-10	2.41E-11	1.54E-09	0.02%
MKC-x	2.18E-11	3.11E-09	3.42E-09		6.56E-09	0.10%
MKH-x	1.46E-10	7.81E-12			1.54E-10	0.00%
M-TOTALS	**2.91E-09**	**1.99E-07**	**2.16E-07**	**4.56E-10**	**4.19E-07**	**6.58%**
NID-x	2.20E-12	5.43E-10	4.10E-10		9.55E-10	0.02%
NIE-x	6.63E-11	2.52E-08	1.04E-07	7.66E-10	1.30E-07	2.04%
NIF-x		5.58E-10	2.44E-09	1.38E-11	3.01E-09	0.05%
NIG-x		2.51E-09	6.26E-08	2.39E-09	6.75E-08	1.06%
NIH-x	9.98E-10	1.53E-07	2.54E-06	1.67E-06	4.36E-06	68.57%
NJA-x	2.56E-12	3.67E-11	3.07E-11	1.57E-12	7.16E-11	0.00%
NJH-x	2.86E-11	3.39E-10	1.42E-09	6.73E-10	2.46E-09	0.04%
NKF-x		3.80E-10	1.69E-09	9.50E-12	2.08E-09	0.03%
NKH-x	8.82E-12	2.39E-09	4.24E-08	2.75E-08	7.23E-08	1.14%
NKHW	8.82E-12	1.71E-09	2.26E-08	1.08E-08	3.51E-08	0.55%
N-TOTALS	**1.12E-09**	**1.87E-07**	**2.77E-06**	**1.71E-06**	**4.68E-06**	**73.49%**
OIA-x	8.86E-09	5.57E-08	3.81E-08	4.86E-11	1.03E-07	1.61%
OJA-x	5.47E-10	4.29E-11	2.73E-11		6.17E-10	0.01%
OKC-x	1.27E-10	8.26E-10	5.68E-10		1.52E-09	0.02%
O-TOTALS	**9.53E-09**	**5.65E-08**	**3.87E-08**	**4.86E-11**	**1.05E-07**	**1.65%**
PID-x	2.25E-11	4.66E-11	6.97E-12		7.61E-11	0.00%
PIE-x	1.51E-09	3.83E-08	1.70E-07	9.95E-09	2.20E-07	3.46%
PIF-x	9.17E-10	1.04E-09	4.16E-09	2.21E-10	6.34E-09	0.10%
PIG-x	1.41E-11	5.00E-09	2.64E-08	3.71E-10	3.17E-08	0.50%
PIH-x	9.21E-09	1.33E-07	5.73E-07	1.71E-07	8.86E-07	13.92%
PJA-x	1.65E-11	2.36E-11	3.14E-10	1.85E-10	5.39E-10	0.01%
PJH-x		3.54E-12	9.55E-11	5.55E-11	1.55E-10	0.00%
PKF-x	2.31E-11	5.30E-10	2.71E-09	1.52E-10	3.41E-09	0.05%
PKH-x	1.37E-10	2.13E-09	9.62E-09	2.78E-09	1.47E-08	0.23%
P-TOTALS	**1.18E-08**	**1.80E-07**	**7.87E-07**	**1.85E-07**	**1.16E-06**	**18.28%**
TOTALS CDF	**2.54E-08**	**6.22E-07**	**3.82E-06**	**1.90E-06**	**6.36E-06**	**100.00%**
CDF Percentages	**0.40%**	**9.78%**	**59.96%**	**29.85%**		
Conditional CDF	**3.22E-05**	**0.025**	**0.281**	**0.671**		
Unaccounted	5.85E-09	1.38E-08	1.14E-08	1.14E-08	4.24E-08	
Percentages	0.09%	0.22%	0.18%	0.18%	0.67%	

event frequency is higher than SEIS1 initiating event frequency using the NUREG-1488 hazard curves. This is due to the fact that the EPRI hazard curves begins with an acceleration of 0.006g and the NUREG-1488 Hazard Curve data begin with an acceleration of 0.05g. Lower acceleration levels produce higher frequencies of occurrence and therefore higher initiating event frequencies.

In addition, the EPRI SEIS1 initiator produces core damage frequency of $1.3x10^{-7}$ per year whereas the NUREG-1488 SEIS1 initiator produces a core damage frequency of $2.5x10^{-8}$ per year. The higher core damage frequency contribution of the EPRI curves is true despite the fact that the SEIS1 initiating event using the NUREG-1488 data has a higher acceleration range than SEIS1 initiating event using the EPRI Hazard Curve data. Indicating that for low accelerations, initiating event frequency drives the results rather than seismic failures.

Initiating Event	EPRI Acceleration Ranges	NUREG-1488 Acceleration Ranges
SEIS1	0.006 - 0.26g	0.05g - 0.31
SEIS2	0.26g - 0.46g	0.31g - 0.46g
SEIS3	0.46g - 0.62g	0.46g - 0.77g
SEIS4	0.62g - 0.82g	0.77g - 1.01g

In the case of SEIS2 the difference in initiating event frequency is smaller since the differences in the range of accelerations between the NUREG-1488 and EPRI Hazard Curve data is also smaller. This is also reflected in the resulting core damage frequency estimates.

In both cases SEIS3 initiating event dominates the results. In the case of the SEIS3 initiating event the larger range of accelerations and different hazard curve data produce initiating event frequencies which differ by a factor of over seven with the NUREG-1488 frequency being the higher of the two frequencies. This intern produces core damage frequency estimates which are over a factor of two times greater.

In the case of SEIS4 initiating event the contribution to the total core damage frequency is much greater in the NUREG-1488 case. This is due to the larger range of acceleration modeled and is reflected in the both the larger initiating event and core damage frequencies. For the SEIS4 initiating event, the frequency for the NUREG-1488 case is approximately three times greater than the EPRI curves and produces core damage which is a factor of two greater.

System Importance

In addition to the complexities encountered when comparing the initiating events given the differences between the EPRI and the NUREG-1488 hazard curves, the differences in acceleration ranges for each initiator result in new NUREG-1488 specific split fraction values. A new master frequency file is provided in Table 3-18a which provides the seismic failure values used in the quantification of the NUREG-1488 (LLNL) hazard curves.

Table 3-18A NUREG-1488 Seismic Master Frequency File

T.E. Designator	Split Fraction	Frequency	Split Fraction Description	Components Fragilities
BX1	BX	4.39E-07	Standby Liquid Contol during SEIS1	FRAG09
BX2	BX	4.54E-05	Standby Liquid Contol during SEIS2	
BX3	BX	1.07E-03	Standby Liquid Contol during SEIS3	
BX4	BX	9.05E-03	Standby Liquid Contol during SEIS4	
CX1	CX	1.37E-03	Containment Spray/ESWduring SEIS1	FRAG15
CX2	CX	7.84E-02	Containment Spray/ESWduring SEIS2	FRAG16
CX3	CX	4.13E-01	Containment Spray/ESWduring SEIS3	FRAG33
CX4	CX	8.70E-01	Containment Spray/ESWduring SEIS4	FRAG35
CZ1	CZ	2.91E-03	Core Spray during SEIS1	FRAG17
CZ2	CZ	1.18E-01	Core Spray during SEIS2	FRAG32
CZ3	CZ	4.08E-01	Core Spray during SEIS3	
CZ4	CZ	7.68E-01	Core Spray during SEIS4	
DU1	DU	9.09E-03	DG Building Liquefaction during SEIS1	FRAG02
DU2	DU	4.07E-01	DG Building Liquefaction during SEIS2	
DU3	DU	8.47E-01	DG Building Liquefaction during SEIS3	
DU4	DU	9.92E-01	DG Building Liquefaction during SEIS4	
DV1	DV	1.29E-03	DG Building after Liquefaction during SEIS1	FRAG03
DV2	DV	1.46E-01	DG Building after Liquefaction during SEIS2	FRAG07
DV3	DV	6.57E-01	DG Building after Liquefaction during SEIS3	
DV4	DV	9.77E-01	DG Building after Liquefaction during SEIS4	
EU1	EU	2.71E-03	Startup Transformers during SEIS1	FRAG20
EU2	EU	1.72E-01	Startup Transformers during SEIS2	
EU3	EU	5.72E-01	Startup Transformers during SEIS3	
EU4	EU	9.10E-01	Startup Transformers during SEIS4	
EW1	EW	1.60E-02	4160 VAC Bus 1C during SEIS1	FRAG05
EW2	EW	2.90E-01	4160 VAC Bus 1C during SEIS2	
EW3	EW	5.93E-01	4160 VAC Bus 1C during SEIS3	
EW4	EW	8.49E-01	4160 VAC Bus 1C during SEIS4	
FX1	FX	4.96E-04	Fire Protection System during SEIS1	FRAG00
FX2	FX	2.90E-02	Fire Protection System during SEIS2	
FX3	FX	1.45E-01	Fire Protection System during SEIS3	
FX4	FX	3.94E-01	Fire Protection System during SEIS4	
IY1	IY	7.44E-04	Isolation Condenser during SEIS1	FRAG18
IY2	IY	3.65E-02	Isolation Condenser during SEIS2	
IY3	IY	1.62E-01	Isolation Condenser during SEIS3	
IY4	IY	4.13E-01	Isolation Condenser during SEIS4	
LX1	LX	1.09E-02	Offsite Power Recovery during SEIS1	FRAG20
LX2	LX	3.28E-01	Offsite Power Recovery during SEIS2	FRAG25
LX3	LX	7.70E-01	Offsite Power Recovery during SEIS3	
LX4	LX	9.78E-01	Offsite Power Recovery during SEIS4	
OX1	OX	8.60E-02	Offsite Power during SEIS1	FRAG01
OX2	OX	6.44E-01	Offsite Power during SEIS2	
OX3	OX	8.67E-01	Offsite Power during SEIS3	
OX4	OX	9.70E-01	Offsite Power during SEIS4	

Table 3-18A NUREG-1488 Seismic Master Frequency File

T.E. Designator	Split Fraction	Frequency	Split Fraction Description	Components Fragilities
SX1	SX	7.32E-02	Condensate Storage Tank (CST) during SEIS1	FRAG10
SX2	SX	6.28E-01	Condensate Storage Tank (CST) during SEIS2	
SX3	SX	8.65E-01	Condensate Storage Tank (CST) during SEIS3	
SX4	SX	9.73E-01	Condensate Storage Tank (CST) during SEIS4	
XX1	XX	7.99E-02	Long Term 125 VDC Bus B during SEIS1	FRAG04
XX2	XX	6.51E-01	Long Term 125 VDC Bus B during SEIS2	
XX3	XX	8.78E-01	Long Term 125 VDC Bus B during SEIS3	
XX4	XX	9.76E-01	Long Term 125 VDC Bus B during SEIS4	
DW1	DW	8.29E-04	Diesel Generators during SEIS1	FRAG26
DW2	DW	7.46E-02	Diesel Generators during SEIS2	FRAG27
DW3	DW	4.15E-01	Diesel Generators during SEIS3	FRAG31
DW4	DW	8.58E-01	Diesel Generators during SEIS4	
DX1	DX	9.85E-04	Short Term 125 VDC B (Wall No 53) during SEIS1	FRAG34
DX2	DX	5.94E-02	Short Term 125 VDC B (Wall No 53) during SEIS2	
DX3	DX	2.64E-01	Short Term 125 VDC B (Wall No 53) during SEIS3	
DX4	DX	6.04E-01	Short Term 125 VDC B (Wall No 53) during SEIS4	

COMPONENT, RELAY ANS STRUCTURE FRAGILITY DESCRIPTIONS

Fragility	Description	A(g)	B(r)	B(u)
FRAG00	Generic Fragility for Others	1.00	0.40	0.32
FRAG01	Offsite Power (Ceramic Insulators)	0.30	0.25	0.50
FRAG02	DG Building Liquefaction	0.40	0.14	0.28
FRAG03	DB Building After Liquefaction	0.69	0.14	0.28
FRAG04	Battery Room Fans	0.30	0.40	0.35
FRAG05	Switchgear Room Fans	0.50	0.40	0.35
FRAG06	4160 VAC Switchgear	0.40	0.25	0.48
FRAG07	EDG After Liquefaction	0.56	0.12	0.31
FRAG09	SLC Tank	3.10	0.40	0.35
FRAG10	Condensate Storage Tank	0.31	0.40	0.35
FRAG15	Containment Spray Pump	1.48	0.40	0.35
FRAG16	Emergency Service Water Pump	1.18	0.40	0.35
FRAG17	Core Spray Pump	0.82	0.40	0.35
FRAG18	Isolation Condenser	0.98	0.40	0.35
FRAG20	Plant Transformer	0.53	0.28	0.23
FRAG25	Combustion Turbine	0.60	0.37	0.39
FRAG26	DG with No Liquefaction	0.81	0.25	0.31
FRAG27	DG Building with No Liquefaction	1.18	0.37	0.22
FRAG31	DG Control Cabinet - Relays	0.81	0.28	0.34
FRAG32	Relay Panel 18A and 18B	0.88	0.28	0.43
FRAG33	Relay Panel 8A and 8B	0.88	0.28	0.43
FRAG34	Masonry Wall No. 53	0.77	0.28	0.36
FRAG35	Intake Structure	0.82	0.18	0.26

Table 3-19 System Importance Using NUREG-1488 Hazard Curves

No.	Top Event	Fussel-Vesely Importance	Top Event Description
1	IY	3.96E-01	Seismic Failure of Isolation Condenser System
2	FX	3.25E-01	Seismic Failure of Fire Protection Water System
3	EW	1.29E-01	Seismic Failure of 4160 VAC 1C Bus
4	LX	1.27E-01	Seismic Failure of Offsite Power Recovery
5	DU	1.13E-01	Diesel Generator Building Liquefaction
6	DV	1.05E-01	Diesel Generators after DG Building Liquefaction
7	VR	1.04E-01	Independent Failure of EMRVs to Reclose
8	SX	8.48E-02	Seismic Failure of the Condensate Storage Tank
9	OX	7.80E-02	Seismic Failure of Offsite Power
10	DW	6.23E-02	Seismic Failure of the Diesel Generators w/o Liquefaction

Table 3-20 Seismic Fragility Importance Using NUREG-1488 Hazard Curves

No.	Fragility	Top Event	Fussel-Vesely	Fragility Description
1	FRAG18	IY	3.96E-01	Isolation Condenser
2	FRAG00	FX	3.25E-01	Fire Protection Liquefaction (Generic Fragility)
3	FRAG05	EW	1.29E-01	Switchgear Room Fan
4	FRAG02	DU	1.13E-01	Diesel Generator Building Liquefaction
5	FRAG10	SX	8.48E-02	Condensate Storage Tank
6	FRAG01	OX	7.79E-02	Offsite Power (Ceramic Insulators)
7	FRAG04	XX	6.20E-02	Battery Room A/B Fans
8	FRAG07	DV	5.78E-02	Diesel Generators after DGB Liquefaction
9	FRAG20	LX	4.77E-02	Plant Transformers
10	FRAG25	LX	3.06E-02	Combustion Turbine

Table 3-21 Independent System Importance Using NUREG-1488 Hazard Curves

No.	Top Event	Fussel-Vesely Importance	Top Event Description
1	VR	1.04E-01	Independent Failure of EMRVs to Reclose
2	LY	2.22E-02	Independent Failure of Offsite Power Recovery
3	EC	8.16E-03	Independent Failure of 4160 VAC Bus 1C
4	MU	7.04E-03	Independent Failure of Isolation Condenser Makeup
5	ED	4.12E-03	Independent Failure of 4160 VAC Bus 1D
6	CC	2.55E-03	Independent Failure of Containment Spray/ESW
7	CS	1.75E-03	Independent Failure of Core Spray
8	FP	1.68E-03	Independent Failure of Fire Protection Water
9	IC	1.56E-03	Independent Failure of Isolation Condenser System
10	DC	2.71E-04	Independent Failure of 125 VDC Distribution C

Electrical power top event play an important role in the core damage frequency with six (6) of the top ten systems (ranked by Fussel-Vesely Importance) are associated with electrical power.

The Seismic Fragility importance remains essentially the same with the exception that at higher acceleration levels modeled in the NUREG-1488 Hazard Curves the isolation condenser and its support (condensate storage tank and fire protection injection) increase in importance.

With the increased acceleration levels resulting in additional challenges to the isolation condenser additional importance is placed on the 4160 VAC independent failure since this is the only other decay heat removal alternative.

Sequence Importance

A listing of the top ten sequences of the Seismic Model evaluated using the NUREG-1488 hazard curves is presented in Table 3-22. This listing of sequences is similar to those which are the result of the EPRI hazard estimates.

In Sequence 1, a seismic loss of offsite power (OX), liquefaction of the diesel generator building (DU) followed by failure of the diesels (DV) and a loss of isolation condenser system (IY) results in core damage. Offsite power recovery is failed due to the short term failure of the isolation condensers.

Sequence 2, is similar to sequence 1 with the exception offsite power recovery is failed at event due to combustion turbine failure at top event OX.

Sequence 3, is similar to Sequence 1 with the exception that it is initiated by SEIS4 and the additional failure of short term 125 VDC power occurs.

Sequence 4, is similar to Sequence 2 with the exception that it is the result of SEIS4 initiating event and the additional failures of short term 125 VDC power occurs and fire protection water resulting in a different damage state assignment.

Sequence 5, also a result of SEIS4 initiating event is the complement of sequence 3. That is, in this sequence 125 VDC power is successful. It should be noted that in sequences 3 and 5 the failure or success of 125 VDC power does not affect the fact that the sequence results in core damage or its damage state assignment. In this case as in many cases in the seismic model this is an "artificial fracture" of the sequence database. This does not affect the core damage frequency due to low truncation limits set. See OCPRA Level 1, Volume 2, Section 7.1 for a full description of artificial fracture.

Sequence 6, a seismic station blackout (OX, DU and DV) occurs as well as failure of the isolation condenser (and therefore failure of offsite power recovery) and failure of all injection (SX and FX).

Sequence 7, a seismic station blackout (OX, DU and DV) occurs and offsite power recovery is failed at top event LX. The failure of injection through CRD or fire protection (failure of top event SX and FX) results in the assignment of plant damage state NIHX).

Sequence 8, this sequence is the complement of sequence 1.

Sequence 9, this sequence is the complement of sequence 6 with short term 125 VDC power due to block wall failure (top event DX) successful.

Sequence 10, this sequence is the first sequence in the database which contains an independent failure. In this sequence a seismic loss of offsite power occurs with a stuck open EMRV and seismic failure of the combustion turbines.

All of the top ten sequences are the result of seismic loss of offsite power. The core damage is modeled as occurring following seismic station blackout (OX, DU and DV or DW) and either the isolation condenser fails (shortens the time available for recovery), the recovery fails at OX or OY. Top events associated with the condensate storage tank (SX) and fire protection (FX) provide information necessary for plant damage state assignment.

Plant Damage State Importance

As in the case with the EPRI hazard curves, the intact containment endstate (plant damage states xIxx) contributes approximately 98% of the total core damage frequency with the remaining 2% contributed by reactor scram failures (ATWS events, xKxx) and containment bypasses (plant damage states xJxx).

In addition, the fractions of core damage with the RPV at high and low pressure also compare favorabley with the EPRI hazard curve model. In the NUREG-1488 model, 80% of the core damage frequency occurs with RPV at high pressure (plant damage states Mxxx or Nxxx) and the remaining 20% with the RPV at low pressure (plant damage states Oxxx or Pxxx).

Table 3-22 List of Top Ten Sequences Using NUREG-1488 Hazard Curves

No.	Initiating Event	Sequence Frequency	End State	Failed Split Fractions	Failed Split Fraction Description
1	SEIS3	5.15E-07	NIHW	OX3	Seismic Failure of Offsite Power
				DU3	Diesel Generator Building Liquefaction
				DV3	Diesel Generators after DG Building Liquefaction
				SX3	Seismic Failure of the Condensate Storage Tank
				IY3	Seismic Failure of Isolation Condenser System
2	SEIS3	3.51E-07	NIHX	OX3	Seismic Failure of Offsite Power
				DU3	Diesel Generator Building Liquefaction
				DV3	Diesel Generators after DG Building Liquefaction
				FX3	Seismic Failure of Fire Protection Water System
				SX3	Seismic Failure of the Condensate Storage Tank
				LX3	Seismic Failure of Offsite Power Recovery
3	SEIS4	3.43E-07	NIHW	DX4	Short Term 125 VDC B (Wall No 53)
				OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				SX4	Seismic Failure of the Condensate Storage Tank
				IY4	Seismic Failure of Isolation Condenser System
4	SEIS4	3.11E-07	NIHX	DX4	Short Term 125 VDC B (Wall No 53)
				OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				FX4	Seismic Failure of Fire Protection Water System
				SX4	Seismic Failure of the Condensate Storage Tank
				LX4	Seismic Failure of Offsite Power Recovery
5	SEIS4	2.29E-07	NIHW	OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				SX4	Seismic Failure of the Condensate Storage Tank
				IY4	Seismic Failure of Isolation Condenser System
6	SEIS4	2.25E-07	NIHX	DX4	Short Term 125 VDC B (Wall No 53)
				OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				FX4	Seismic Failure of Fire Protection Water System
				SX4	Seismic Failure of the Condensate Storage Tank
				IY4	Seismic Failure of Isolation Condenser System
7	SEIS4	2.08E-07	NIHX	OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				FX4	Seismic Failure of Fire Protection Water System
				SX4	Seismic Failure of the Condensate Storage Tank
				LX4	Seismic Failure of Offsite Power Recovery

Table 3-22 List of Top Ten Sequences Using NUREG-1488 Hazard Curves

No.	Initiating Event	Sequence Frequency	End State	Failed Split Fractions	Failed Split Fraction Description
8	SEIS3	1.82E-07	NIHW	DX3	Short Term 125 VDC B (Wall No 53)
				OX3	Seismic Failure of Offsite Power
				DU3	Diesel Generator Building Liquefaction
				DV3	Diesel Generators after DG Building Liquefaction
				SX3	Seismic Failure of the Condensate Storage Tank
				IY3	Seismic Failure of Isolation Condenser System
9	SEIS4	1.50E-07	NIHX	OX4	Seismic Failure of Offsite Power
				DU4	Diesel Generator Building Liquefaction
				DV4	Diesel Generators after DG Building Liquefaction
				FX4	Seismic Failure of Fire Protection Water System
				SX4	Seismic Failure of the Condensate Storage Tank
				IY4	Seismic Failure of Isolation Condenser System
10	SEIS3	1.35E-07	PIHW	OX3	Seismic Failure of Offsite Power
				DU3	Diesel Generator Building Liquefaction
				DV3	Diesel Generators after DG Building Liquefaction
				SX3	Seismic Failure of the Condensate Storage Tank
				VR2	Independent Failure of EMRVs to Reclose
				LX3	Seismic Failure of Offsite Power Recovery

3.1.5.5 Oyster Creek Seismic Sensitivity Studies

This report section describes the seismic sensitivity studies performed to determine the relative contributions to core damage frequency from changes to the assumptions and model characteristics. Several sensitivity cases modeled and described below.

Case 1 - Removal of Relay Chatter

The "NOCHAT" was developed and removed assumptions regarding relay chatter which is modeled in the Oyster Creek Seismic Model. Specifically, the NOCHAT model removes the unrecoverable relay chatter of containment/ESW spray and core spray logic. The result is a core damage frequency of 1.26×10^{-6} per year. This represents a 65.3% decrease over the base case.

Case 2 - Removal of Block Wall Failure Impact

The seismic failure of block wall no. 53 which is actually the onset of cracking and wall displacement does not necessarily result in the failure of 125 VDC distribution center B. This impact is removed from the seismic model and the results recalculated in model "NOWALL". The result of the recalculation is a core damage frequency of 1.32×10^{-6} per year. This represents a 63.6% decrease over the base case core damage frequency.

Case 3 - Offsite Power Recovery Failure Given Isolation Condenser Failure

The base OCPRA modeled the success of offsite power recovery given the isolation condenser is successful only. This is due to the time require to align the offsite combustion turbines following a loss of all station power. This assumption is more valid when the event is a seismically initiated occurrence especially for the higher ground motion events. This model addresses the removal of this restraint since although less likely without isolation condenser success, offsite power recovery, may be possible. This is modeled in the "RECOVER" model.

The result of "RECOVER" model is a core damage frequency of 3.60×10^{-6} per year. This represents a decrease of less than 1% from the base case core damage frequency. The small decrease is the result of the fact that most of the base case scenarios in which the isolation condenser system has failed, the recovery of offsite power has also failed and therefore, removal of the dependance of offsite power recovery on isolation condenser success has little impact.

Case 4 - No Seismically Induced Ventilation Failure

The current seismic model reflects the failure of 4160 VAC Bus 1C and long term 125 VDC Bus B failure following failure of ventilation fans due to the seismic event. It is probable that systems (4160 VAC Switchgear) may not fail as a result of ventilation failure since ample time is available for eventual recovery of ventilation through other means. The "NOVENT" model reflects the removal of ventilation impacts from the seismic model.

The result of the NOVENT model is a core damage frequency of 1.08×10^{-6} per year. This represents a decrease of over 70% in total core damage frequency. This case shows the importance of the on-site 4160 VAC systems following a seismic event.

Case 5 - Adjustment of Outside Control Room Human Action Failure Rates

Human actions performed outside the control room following a seismic event may be hampered by accessibility problems and other performance shaping factors which may decrease the likelihood of success. This sensitivity case investigates the effects of increased failure rates of actions performed outside the control room following a seismic event.

Human actions included in the Level 1 OCPRA were review to determine those which occur outside the control room. Since in the seismic model may of the non-safety systems were failed following all seismic events (due to low capacity) many actions were already failed. All actions which occur outside the control room following a seismic event were increased failure rate by a factor of 10. These actions included (opening the CRD bypass line, alignment of fire protection to IC makeup, all offsite power recover actions (alignment of combustion turbines) and transfer of the core spray suction to condensate storage tank suction.) A total core damage frequency of 3.75×10^{-6} per year was calculated which is a 4.2% increase over the base seismic result.

Table 3-23 Summary of Seismic Sensitivity Studies

Case Name	CDF	Percent Difference	Case Description
Base	3.63E-06	n/a	Base seismic model using the EPRI Hazard Curves
NOCHAT	1.26E-06	- 65%	Removal of relay chatter (not recovered in base model)
NOWALL	1.32E-06	- 64%	Removal of Block Wall No. 53 Impacts on Short Term DC.
RECOVER	3.60E-06	- 1%	Removal of offsite power recovery dependence on isolation condenser success.
NOVENT	1.08E-6	- 70%	Removal of ventilation failure impacts on 4160 VAC 1C and 125 VDC Distribution B.
HA-SENS	3.75E-6	+4.2%	Increase in outside control room human action values.

3.1.5.6 Oyster Creek Seismic Vulnerabilities

As defined in the Oyster Creek IPE Submittal Report, Section 3.2, a vulnerability is defined as any core damage sequence which exceeds 1×10^{-4} per reactor year, or any containment bypass sequence or large early containment failure sequence that exceeds 1×10^{-6} per year. As a result of the Oyster Creek Seismic Analysis: No vulnerabilities have been found.

3.1.6 Analysis of Containment Performance

The effect of seismic events on the containment building has been evaluated from two perspectives: First is the drywell structural capacity and the second is the performance of the containment isolation function following a seismic event.

3.1.6.1 Containment Structure

The containment structure consists of the reactor building which is comprised of four main sub-structures: the main structure, the drywell, the biological reactor shield wall (BSW) and the reactor pressure vessel (RPV). The main structure is a rectangular reinforced concrete structure up to the refueling floor at elevation 119'. Above the refueling floor, the structure consists of a steel frame with insulated metal siding. The drywell containment vessel is an axisymmetic steel shell surrounded by a heavy concrete shield wall which follows the contour of the vessel from the foundation of the drywell to the refueling floor. The interface between the reactor building, drywell, BSW and RPV includes the RPV stabilizer, the Star Truss, the Drywell lugs, and the radial support beams of two steel platforms at elevations 46' and 23'4.5".

All reactor building structure fragilities exceeded 1.0g capacity and were screened from further consideration. Additional information is available in report Section 3.1.3.

3.1.6.2 Containment Isolation

All primary containment isolation valves which were modeled in the OCPRA Level 1 analysis were included in seismic capacity walkdowns. All primary containment isolation valves screen with estimated capacities greater than 1.0g. Containment isolation signals are generated as part of the reactor protection system logic and all relays in this logic screened with capacities (relay chatter) greater than 1.0g. However, the A-46 program did not include specific isolation valve actuation relays which may be subject to relay chatter. Therefore, the seismic model evaluated in this report used split fraction values from manual isolation only. No containment isolation failures were reported in the significant results for the Oyster Creek seismic analysis.

It should also be noted that containment isolation at Oyster Creek is a fail safe system in which instrument air and power is required to prevent isolation. Due to the low capacity of the instrument air system, the loss of station air is assumed for all levels of ground motion and would result in the guaranteed isolation of containment regardless of relay chatter.

OCPRA level 1 results were review to determine the types of containment bypass event which were significant in that study. From this review it was determined that plant damage states which indicated bypass of containment were dominated by failure of the reactor water cleanup system (RWCU), LOCAs outside of primary containment and ATWS events (reactor scram failure). Relays associated with the reactor cleanup system were reviewed and determined to have capacities beyond the range of interest (See Section 3.1.4.2). In addition, relays associated with the reactor scram system were also review and determined to have sufficient capacity. Walkdowns performed in support of the seismic capacity determinations did not identify any seismically induced LOCA concerns. On the basis of the above this issue is considered bounded by the use of manual containment isolation split fraction values.

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

This report section describes the treatment of USI A-45, GI-131 and other seismic safety issues.

USI A-45, Shutdown Decay Heat Removal Requirements

The decay heat removal capability for resolution of USI A-45 has previously been described and evaluated in Section 5.0 of the Oyster Creek IPE Submittal Report. This report section discusses the seismic capacity of the systems that makeup decay heat removal capability at Oyster Creek.

Decay Heat Removal Through the Main Condenser

This path is the normal path for decay heat removal and normal shutdown. This decay heat removal path is guaranteed failed during and following all seismic ground motions modeled.

Decay Heat Removal Through the Isolation Condenser

This decay heat removal path is utilized following reactor isolation transients. All seismic events with successful MSIV closure result in reactor isolation transients. The capacity of the isolation condenser is 0.98g and is included in the seismic model. In addition, support systems which provide makeup to the shell side of the condenser: Condensate Transfer, Condensate Storage Tank (CST) and the Fire Protection Water are also included in the seismic model. Condensate Transfer system capacity was greater than 1.0g and screened. The CST and the fire protection water system were included in the model with capacities of 0.31g and 1.0g, respectively.

Decay Heat Removal Through Containment Spray/Emergency Service Water

Should isolation condenser or their support fail core decay heat is discharged to the containment via the EMRVs or safety valves and removed to the ultimate heat sink (i.e., intake canal) via the containment spray and emergency service water system. The capacity of the containment spray system is controlled by the containment spray pumps which have a capacity of 1.48g. The emergency service water pumps have a capacity of 1.18g. Unrecovered relay chatter of the containment spray/ESW system is modeled with a capacity of 0.88g. The weakest link of this decay heat removal path, however, is the diaphragm failure of the intake structure itself at an average acceleration of 0.82g. RPV inventory is maintained using the core spray system which has a median capacity of 0.82g for the core spray pumps and a (non-recovered) relay chatter fragility of 0.88g for the core spray actuation logic.

Decay Heat Removal Through the Hardened Vent

This decay heat removal path utilizes the hardened vent path following failure of the containment spray/ESW system. Decay heat is rejected to the plant ventilation stack via hardened piping. The containment vent capacity was determined to be greater than 1.0g and was screened from consideration. RPV inventory is maintained using the core spray system which has a median capacity of 0.82g for the core spray pumps and a (non-recovered) relay chatter fragility of 0.88g.

The Oyster Creek Seismic PRA models failure as the various combinations of independent or seismic induced failure of decay heat removal or reactor vessel inventory. Section 8 of the Oyster Creek PRA provides details on the system requirements for success.

GI-131, Potential Seismic Interaction Involving Movable In-Core Flux Mapping System

The Oyster Creek Nuclear Generating Station does not utilize a movable in-core flux mapping system; therefore this issue is not applicable.

Other Seismic Safety Issues

- USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants". The USI A-46 issues for Oyster Creek is being addressed by a separate program. As a result this analysis does not address this issue. However, walkdowns for the A-46 safe shutdown equipment and IPEEE seismic analysis were co-ordinated to maximize the knowledge and insights gained from both efforts.

- USI A-17, "System Interactions in Nuclear Power Plants". The seismic analysis includes consideration of spatial interactions due to seismic events. This issue was addressed by the various walkdowns performed in support of the seismic analysis and the seismic/fire interaction walkdowns (Section 4.8). The conclusions from these walkdowns is that no significant adverse spatial interactions due to seismic events were identified. Thus, USI A-17, with regard to seismic risk is considered closed.

- NUREG/CR-5088, "Fire Risk Scoping Study" and GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment". These efforts have been included in the scope of the IPEEE and discussed in the Fire Analysis (Section 4.8). The IPEEE team performed a plant walkdown searching for potential seismic/fire vulnerabilities. Thus, the Fire Risk Scoping Study and GI-57 issues, with respect to seismic risk at Oyster Creek are considered closed.

- Eastern US Seismicity Issue - Per Generic Letter 88-20 Supplement 4, this is resolved by performing an IPEEE using the LLNL and EPRI hazard curves in evaluating the seismic risk. The conclusion from the study is that the hypotheses regarding the Charleston earthquake will fall within the family of seismicity curves developed for the Oyster Creek site. Both curves were used to evaluate the core damage frequency impact on Oyster Creek. This issue is considered closed for Oyster Creek.

3.3 References

3-1 EQE International, "Probabilistic Seismic Response Analysis of the Oyster Creek Nuclear Generating Station in Support of the IPEEE Program," Report Number 50124-R-003, July 1994.

3-2 EQE International, "Seismic Fragilities of Civil Structures at Oyster Creek Nuclear Generating Station," Report Number 990-2316, October 1994.

3-3 EQE International Memorandum, R.L. Tiong to K. Canavan, "Oyster Creek Fragilities - Updated Fragilities," April 27, 1995.

3-4 EQE International, Seismic Fire Interaction Walkdown Report, Report No. P:\42113-06\-OYCEQFIR.DOC, Final Report, August 8, 1995.

3-5 EPRI, "Probabilistic Seismic Hazard Evaluation at Nuclear Power Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," NP-6395-D, 1989.

3-6 Geomatrix Consultants, "Assessments of Potential for Liquefaction and Permanent Ground Displacements at Designated Facilities, Oyster Creek Nuclear Generating Station," Report Number 990-2323, December 1994.

3-7 EQE International Incorporated, "Transmittal of Results of the Evaluation of the Masonry Walls at Oyster Creek Nuclear Generating Station for the Seismic IPEEE", June 5, 1995.

3-8 EQE International Incorporated, "Development of Equipment Seismic Fragilities for OCNGS IPEEE", October 1991.

3-9 GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 0'-0" and 3'-6"," GPU Drawing Number 3E-151-02-001, Revision 3, December 4, 1989.

3-10 GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 6'-0", 0'-0" and 3'-6"," GPU Drawing Number 3E-151-02-002, Revision 4, July 26, 1991.

3-11 GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-151-02-003, Revision 4, July 26, 1991.

3-12 GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-151-02-004, Revision 4, August 1, 1991.

3-13 GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 46'-6"," GPU Drawing Number 3E-151-02-006, Revision 4, August 22, 1991.

3-14 GPU Nuclear, "General Arrangement - Reactor Building, Plan Floor Elevation -19'-6"," GPU Drawing Number 3E-153-02-001, Revision 3, December 4, 1989.

3-15 GPU Nuclear, "General Arrangement - Reactor Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-153-02-002, Revision 4, July 26, 1991.

3-16 GPU Nuclear, "General Arrangement - Reactor Building, Plan Floor Elevation 51'-3"," GPU Drawing Number 3E-153-02-003, Revision 4, July 26, 1991.

3-17 GPU Nuclear, "General Arrangement - Office Building, Plan Floor Elevation 23'-6" and 35'," GPU Drawing Number 3E-156-02-001, Revision 3, December 4, 1989.

3-18 GPU Nuclear, "General Arrangement - Office Building, Plan Floor Elevation 46'-6" and Roof Plan," GPU Drawing Number 3E-156-02-002, Revision 3, December 4, 1989.

3-19 GPU Nuclear, "Oyster Creek Nuclear Generating Station, Operation Plant Manual," Volumes 1 through 8, Serial Number 43, Various Dates.

3-20 GPU Nuclear, "Oyster Creek Individual Plant Examination Submittal Report," June 1992.

3-21 GPU Nuclear, "Oyster Creek Probabilisitc Risk Assessment (Level 1)," Volumes 1 through 6, November 1991.

3-22 GPU Nuclear, "Oyster Creek Probabilistic Risk Assessment (Level 2)," June 1992.

3-23 US NRC, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, 1994.

3-24 Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Final Report, June 1991.

3-25 Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54 (f)," Generic Letter No. 88-20, November 23, 1988.

3-26 Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Final Report, August 1989.

3-27 PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual I: Data Analysis, Volume 1, 1994.

3-28 PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual II: Systems Analysis, Volume 2, 1994.

3-29 PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual III: Event Trees, Volume 3, 1994.

3-30 PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual IV: Important Sequences, Volume 4, 1994.

SECTION 4

FIRE ANALYSIS

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

LIST OF TABLES

LIST OF FIGURES

4.0 Oyster Creek Fire Individual Plant Examination

The Oyster Creek Fire Individual Plant Examination report presents the methods and results of the fire analysis of the impacts of fire events at the Oyster Creek Nuclear Generating Station (OCNGS).

The study is performed in response to the Nuclear Regulatory Commission's (NRC) Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities". The analysis satisfies the requirement for the Internal Fire Analysis and presents the methods, calculations and results in the suggested NUREG-1407 format.

The analysis is performed using standard probabilistic methods similar to those used in the development of a Level 1 Probabilistic Risk Assessment with several notable exceptions.

- First, all accident sequences developed in this study are initiated by fire events which are internal to the plant.
- Second, a cutoff in the frequency of core damage is used to screen fire areas and hence the study is termed the Oyster Creek Individual Plant Examination or a scoping studying. One of the outcomes of using a screening approach is that the core damage frequency reported represents an upper bound since a more detailed evaluation would result in lower core damage contributions of individual fire areas. This approach to analyzing internally initiated fire events is a less resource intensive effort while still providing assurance that plant specific vulnerabilities, if any, are determined.
- Third, significant portions of the Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) methods are used in the study.

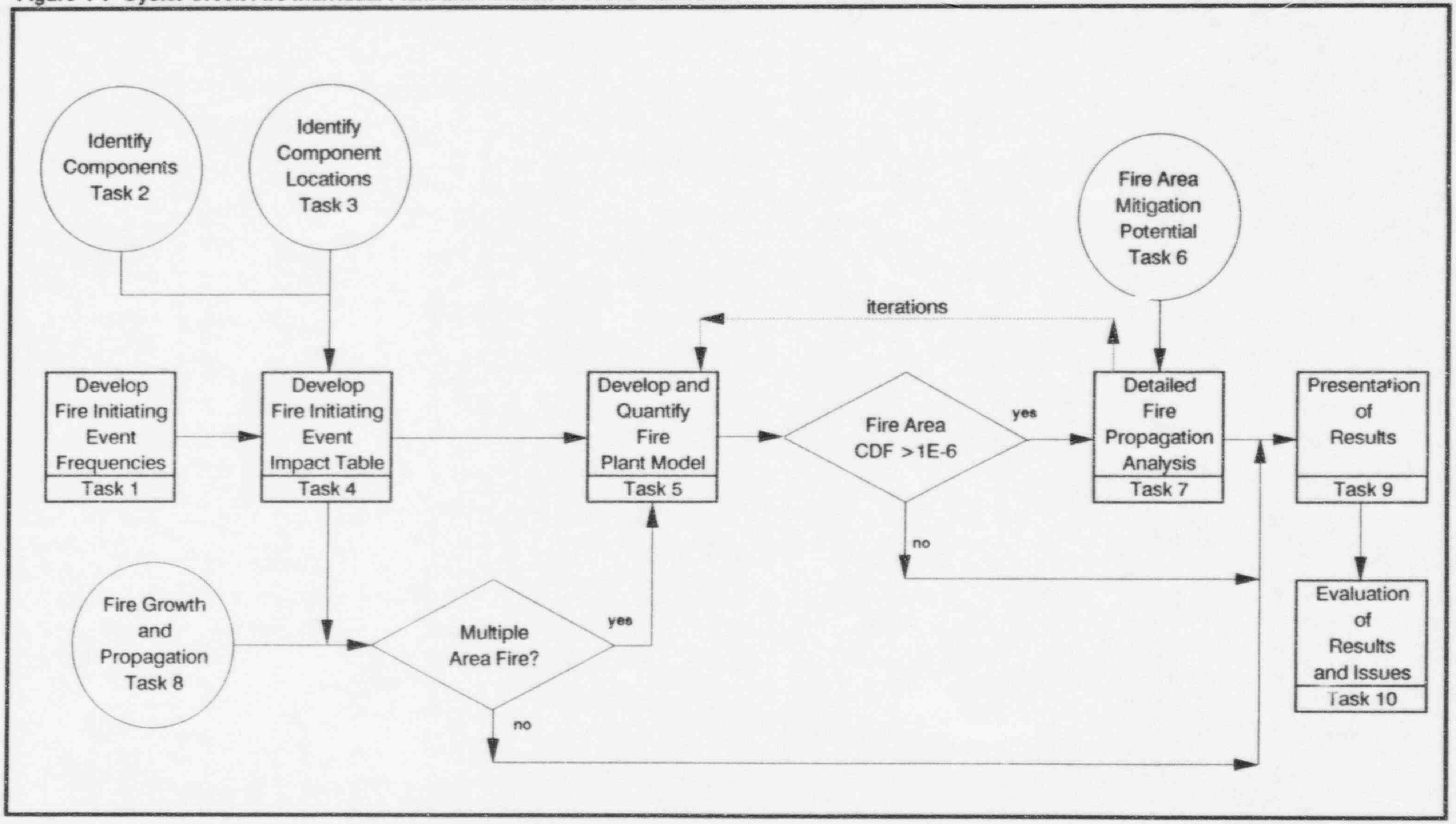
The study is comprised of ten tasks and results in the evaluation of the risk of internal fires at the Oyster Creek Nuclear Generating Station (OCNGS). The process is described in overview in the following paragraphs and illustrated in Figure 4-1. It should be noted that since the report is organized in the suggested NUREG-1407 format, multiple tasks are often documented in a single report section or sub-section. Each task or group of tasks as illustrated on Figure 4-1 is described as well as illustrated in the associated figure to the right.

Task 1 - Develop Fire Initiating Event Frequencies

This task identifies areas of the Oyster Creek Nuclear Generating Station in which the potential for fire initiation, growth and/or propagation can significantly impact plant operation from at power conditions.

The input to this task is from the Level 1 Oyster Creek Probabilistic Risk Assessment (OCPRA), the Fire Hazard Analysis Report, Oyster Creek Fire Mitigation Procedure and plant walkdowns.

Figure 4-1 Oyster Creek Fire Individual Plant Examination Process Illustration



In this task the Fire Hazard Analysis Report fire area and zone designations are used with the OCPRA, fire mitigation procedure and plant walkdowns to determine plant areas in which a fire event may perturb plant operation sufficiently to result in a demand for a reactor scram. The Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) methodology and database is then used to develop fire initiating event frequencies for each of the identified fire areas and zones (critical fire areas).

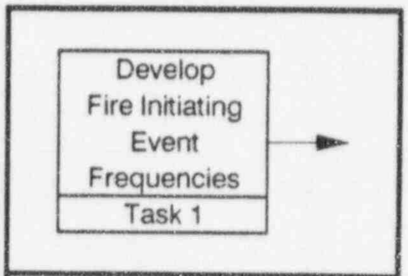


Figure 4-2
Develop Fire Initiating Events

Several fire areas are screened from further consideration based on the insignificant impact of the fire event. For example, the Site Emergency Building does not contain any plant equipment and is located some distance from the plant and, as such, a fire in this area is not expected to result in a demand for a plant trip or damage to plant equipment.

The list of fire areas and zones with their frequency of fire ignition serves as input to the development of Task 4 (Development of the Fire Initiating Event Impact Table) and Task 5 (Development and Quantification of the Plant Model). Details on this task are presented in report Section 4.1, Fire Hazard Analysis.

Task 2 - Identification of Risk Significant Components

In this task the Level 1 OCPRA, the Fire Hazard Analysis Report (FHAR) and plant walkdowns as well as the list of critical fire areas (from Task 1) are used to determine the potential risk significant components. These components are then screened on the basis of their susceptibility to fire events. The result is the list of risk significant components. Details on this task are presented in report Section 4.4, Evaluation of Component Fragilities and Failure Modes.

Task 3 - Identification of Risk Significant Component Locations

In this task, reviews of plant information and plant walkdowns are used to determine the location of the risk significant components and their supporting cables.

Supporting cables include any required electrical or other functional system support cables. Supporting cables also include the possibility of component failure due to "hot shorts" which cause the component to go to an active failure position. That is, supporting cables include those cables whose electrical hot short (i.e., energized) can result in a component changing state into an undesired state or position. For example, a normally closed valve changing to the open position due to a fire event which affects the cable in a remote location of the plant.

The result of this task is the Location of Risk Significant Components and Associated Cables Table, which is used in the Development of the Fire Initiating Event Impact Table (Task 4).

Several fire areas are screened from further consideration since they contain no risk significant components or supporting cables. These areas are screened from further consideration for their individual contribution to the core damage frequency however they are still considered for their potential to be involved in multiple area fires (Task 8).

The information developed in this task is used as input for the development of the Fire Initiating Event Impact Table (Task 4) and the Fire Growth and Propagation (Task 8). Details on Task 3 are presented in report Section 4.4, Evaluation of Component Fragilities and Failure Modes.

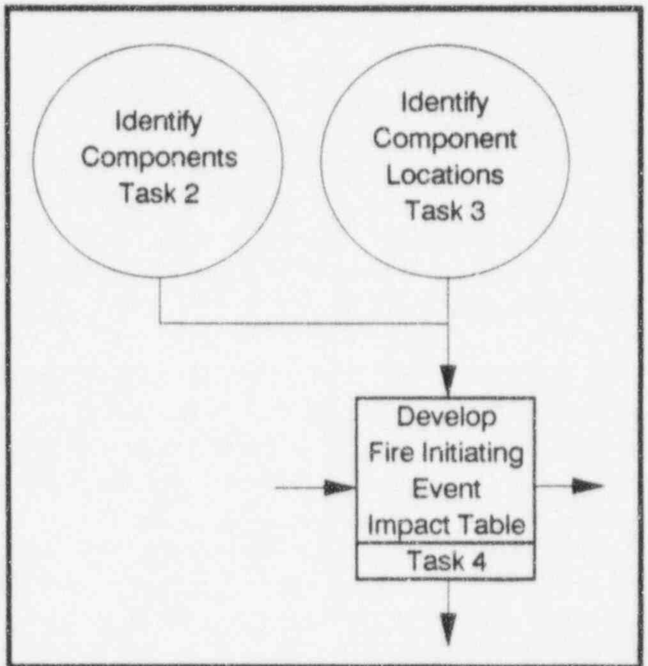


Figure 4-3
Identification of Risk Significant Components

Task 4 - Develop Fire Initiating Event Impact Table

In this task the Location of the Risk Significant Components and Associated Cables (Task 3) and the OCPRA are used to develop the Fire Initiating Event Impact Table.

Each critical fire area, as defined in Task 1, is considered an initiating event. Using the physical component locations and the locations of supporting cables a five event impact table can be developed. This impact table provides the affected components (an hence system functions) given an "all engulfing fire" within a fire area. The term "all engulfing fire" is used to describe the modeling of a fire which fails all components and cables in the area and does not account for detection, suppression or other area mitigative features. In addition, "hot short" impacts are included in the impact table. The Fire Initiating Event Impact Table is therefore the most conservative impacts which a fire event within a given fire area can cause.

The impact table is used as input into the Development and Quantification of the Fire Risk Model (Task 5). Details on Task 4 are presented in report Section 4.2, Review of Plant Information and Walkdowns.

Task 5 - Development and Quantification of the Plant Model

This task develops and documents the Oyster Creek Fire Risk Model. Actually three sub-tasks are performed in the development and quantification of the fire risk model and these sub-tasks are represented on Figure 4-4 as three separate paths of input and output. All three sub-tasks are documented in report Section 4.6, Analysis of Plant Systems, Sequences and Plant Response.

- The first input/output path develops the individual fire area upper bound core damage frequency estimations with input from Tasks 1 and 4 and is described in the "*Initial Estimate of Upper Bound Core Damage Frequency*" report sub-section.

- The second input/output path is represented as the iteration loop between the Detailed Fire Propagation Analysis (Task 7) and develops the refined core damage frequency estimates for those fire areas whose upper bound core damage frequency (UBCDF) was initially greater than $1x10^{-6}$. A single iterations is made which results in the calculation of the Revised Estimate of Upper Bound Core Damage Frequency. Any fire areas which are not screened (UBCDF less than $1x10^{-6}$) are analyzed in Task 7 and documented in the *"Detailed Evaluation of Core Damage Frequency"* report sub-section.

- The third input/output path develops the "multiple fire area" upper bound core damage frequency estimations.

Each input/output paths is discussed in detail below.

Initial Estimate of Upper Bound Core Damage Frequency (UBCDF)

The first input/output path used the Level 1 OCPRA, Fire Initiating Event Frequencies (Task 1) and the Initiating Event Impact Table (Task 4) to develop and quantify the fire risk model for the *"Initial Estimation of the Upper Bound Core Damage Frequency"* as a result of fire events within an individual fire area.

The impacts of a fire event (Task 4) together with the fire initiating event frequency (Task 1) are combined with the random failure probabilities of system functions modeled in the Level 1 OCPRA to produce the fire risk model. That is, the failures produced by the fire initiating event are added to the OCPRA plant model (the independent failures) to produce a risk model which calculates the core damage frequency due to fire events.

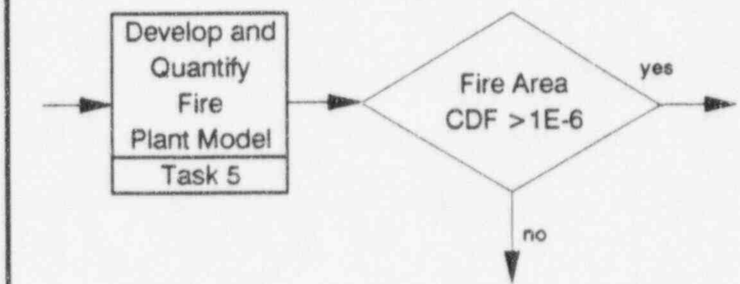


Figure 4-4a - Input/Output Path 1
Initial Estimation of Upper Bound Core Damage Frequency

Since the fire initiating event impact table represents the most conservative outcome of a fire in a given fire area (i.e., "all engulfing fire" and "hot shorts") and fire growth, propagation, detection, suppression or other fire area mitigative features are not modeled, the quantification of this fire risk model produces an upper bound core damage frequency for each fire event. Fire areas whose UBCDF is less than $1x10^{-6}$ per year are screened from further consideration. Fire areas whose total UBCDF contribution is greater than $1x10^{-6}$ per year require a Revised Estimate of Upper Bound Core Damage Frequency which is performed as part of the input/output path two, described below.

Revised Estimate of Upper Bound Core Damage Frequency

In the second input/output path the fire areas whose initial upper bound core damage frequency was greater that $1x10^{-6}$ per year are evaluated. Assumptions regarding the "all engulfing fire" and fire risk model simplifications are addressed and potentially relaxed to more accurately reflect the risk associated with a fire event in these particular fire areas.

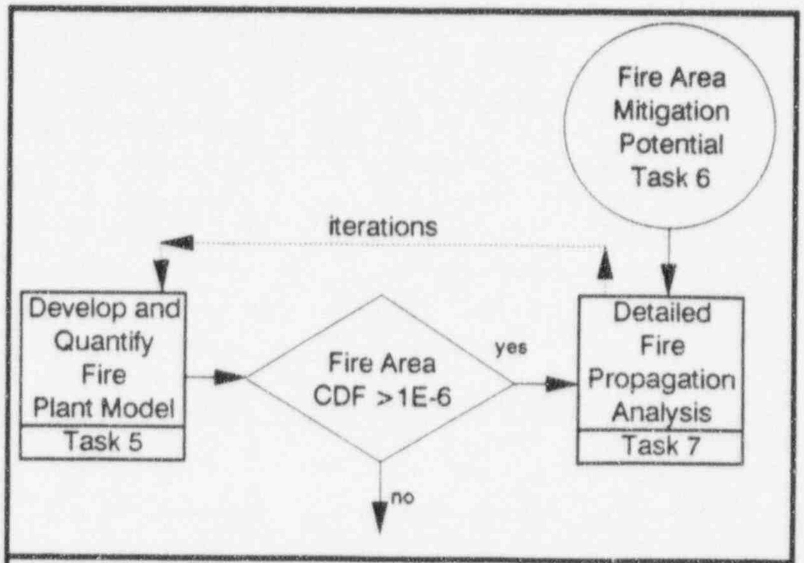


Figure 4-4b - Input/Output Path 2
Revised Estimation of Upper Bound Core Damage Frequency

Following the adjustment of the conservative assumptions the fire risk model is requantified. In the case where the total fire area UBCDF is less than $1x10^{-6}$ per year the fire area is screened from further consideration.

Where the total fire area UBCDF is greater than $1x10^{-6}$ per year the output is directed to Task 7, Detailed Evaluation of Fire Core Damage Frequency. This sub-task is documented in report Section 4.6.2, Revised Estimation of Upper Bound Core Damage Frequency.

Upper Bound Core Damage Frequency Estimation for Multiple Area Fires

The third input/output path develops and quantifies the fire risk model for multiple fire area events. Input is from the Fire Growth and Propagation Task (Task 8), the Development of the Fire Initiating Event Frequencies (Task 1) and the Fire Initiating Event Impact Table. For each multiple fire area event the frequency of the initiating event is calculated as the sum of the individual fire areas which comprise the event. The impacts of the newly defined initiators are also the sum of the impacts of the individual fire areas which comprise the multiple area fire. The impacts and frequencies are factored into the Level 1 OCPRA. The quantification of this fire risk model produces an estimation of the upper bound core damage frequency as a result of multiple fire area events. This

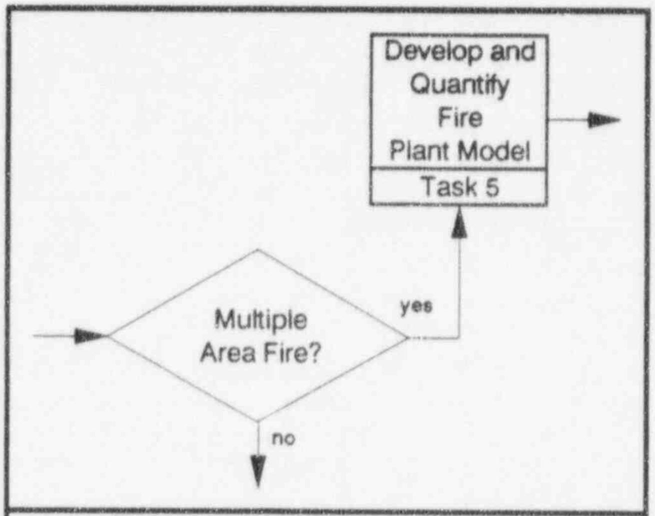


Figure 4-4c - Input/Output Path 3
Multiple Fire Area Quantification

input/output path is documented in report Section 4.3.

Task 6 - Critical Fire Area Mitigation Potential

This task documents the Oyster Creek Nuclear Generating Station's fire detection and suppression systems. Input to the task is from the Fire Hazard Analysis Report and the Fire Mitigation Procedure. The information developed in this task serves as input to the Detailed Fire Propagation Analysis (Task 7). Details on this task are contained in report Section 4.5, Fire Detection and Suppression.

Task 7 - Detailed Fire Propagation Analysis

Those fires areas whose upper bound core damage frequency is greater than $1x10^{-6}$ serve as input into the Detailed Fire Propagation Analysis. The Fire Area Mitigation information collected in Task 6 is used to adjust the conservative assumptions made in the risk model for these areas. The model is then re-quantified. The result of this task is revised risk model impacts and/or adjusted severe fire frequencies. Details on Task 8 are provided in report Section 4.6, Analysis of Plant Systems, Sequences and Plant Response.

Task 8 - Fire Growth and Propagation

This task investigates the potential for fire growth and propagation of fires beyond individual fire areas. Evaluations of fire growth and propagation within a fire area are addressed in the Detailed Fire Propagation Analysis (Task 7) which is presented in report Section 4.6. This task, Fire Growth and Propagation beyond individual fire areas is addressed qualitatively using the Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) assumptions regarding the effectiveness of fire barriers are applied.

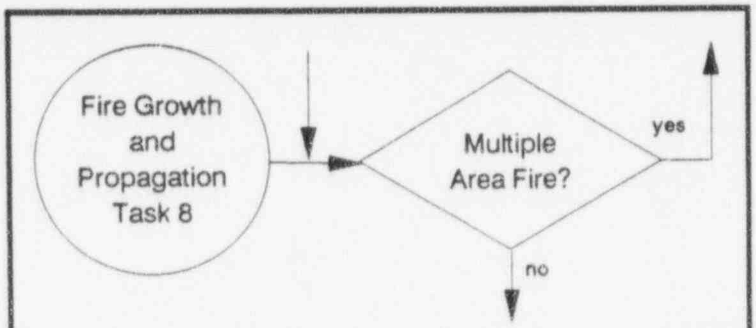


Figure 4-5
Multiple Area Fire Evaluation

The input to the task is from the Identification of Critical Fire Areas (Task 1) and the Fire Initiating Event Impact Table (Task 4). The result of this task is an evaluation of the potential "multiple area fires". In the case where a multiple area fire is assumed to occur, a new initiating event is developed. This initiating event is equal in frequency of occurrence to the sum of the frequency of fire initiation of the fire areas involved. The impacts of this new initiator is equal to the combined impacts of the fire areas involved. This new initiating event is input into the Development and Quantification of the Fire Risk Model (Task 5). Details on Task 8 are presented in report Section 4.3, Fire Growth and Propagation.

Task 9 - Presentation of Results

This task assembles, summarizes and presents the overall results of the Oyster Creek Fire Individual Plant Examination including a summary of containment performance. Details are presented in report Section 4.7, Presentation of Results.

Task 10 - Evaluation of the Results and Fire Issues

This task applies the results and lessons learned to the Sandia issues, A-45 and others. Details are presented in the following report sections:

- Section 4.7, Containment Failure Modes due to Fires
- Section 4.8, Treatment of Sandia Fire Risk Scoping Study Issues
- Section 4.9, USI A-45 and Requirements of NUREG-1407.

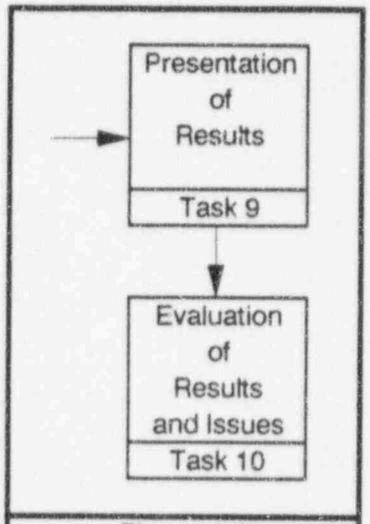


Figure 4-6
Results and Conclusions

Each of the sections of this report begins with a detailed description of the task including the input to the task, output of the task and the steps which are used in the analysis. Taken together, the introduction to each section provides the detailed methodology of the performance of the Oyster Creek Fire Individual Plant Examination.

4.1 Fire Hazard Analysis

This report section presents the first task in the development of the Oyster Creek Fire Individual Plant Examination. In this task, the Oyster Creek Nuclear Generating Station plant areas which contribute to the risk (core damage) due to plant fires from at power conditions are identified. For each of these plant areas, an evaluation of the frequency of fires is produced. That is, initiating events and associated frequencies are developed for the critical plant areas.

- Input to the task is taken from the Oyster Creek Fire Hazard Analysis Report (FHAR), which is used to define the fire initiating events. That is, each of the fire areas and zones described in the FHAR is considered to have the potential for a fire event to perturb normal power operation of Oyster Creek and result in a demand for a plant scram, i.e., a fire initiating event. Input is also taken from plant drawings. Walkdowns are used to determine the amount and type of equipment in each areas which is used in the calculation of the frequency of fires.

- The output o' this task is the list of fire initiating events and there associated frequencies of fire initiation.

The Electric Power Research Institute (EPRI) Fire Induced Vulnerability Evaluation (FIVE) methodology is used to develop the initiating event frequencies. The process as outlined in the FIVE Methodology consists of a two step process:

Step 1 - Generic Plant Location Fire Frequencies. The FIVE methodology uses generic plant location frequencies of fires obtained using a generic fire frequency database. This database consists of approximately 800 fires in the nuclear industry experience of over 1300 reactor years (114 plants). This database is divided into generic fire frequencies for typical structures/locations at nuclear power stations. This step consists of the comparison of Oyster Creek structures and locations to those contained in the FIVE database. The generic location fire frequency is then divided among the Oyster Creek specific fire areas contained in that location to produce a generic fire area frequencies. The calculation of generic fire area frequencies for Oyster Creek is described in Section 4.1.1.

Step 2 - Plant Specific Fire Area Frequencies. The generic fire area frequencies determined in the step above are supplemented with plant specific area information on both ignition sources and combustibles in the area. This results in the generation of a plant specific fire frequency for each plant area. The calculation of plant specific fire area frequencies is described in Section 4.1.2.

The sum of the generic fire area frequency and the plant specific fire frequency yields the total fire area frequency for each of the plant area. The summary of fire area frequency is described in Section 4.1.3.

For reference, the list of all plant fire areas and zones, as evaluated within this report, are shown below. The fire zone or area designations and area descriptions correspond to those used in the Oyster Creek Fire Hazards Analysis Report.

Table 4.1-1 Oyster Creek Plant Fire Areas and Zones	
Fire Area or Zone	Description of Fire Area or Zone
Office Building Areas	
OB-FZ-4	Cable Spreading Room - 36' El
OB-FZ-5	Main Control Room - 46' 6" El
OB-FZ-6A	"A" 480 VAC Switchgear Room
OB-FZ-6B	"B" 480 VAC Switchgear Room
OB-FZ-8A/B	MG Set Room and Mechanical Equipment Room (single area)
OB-FZ-8C	A/B Battery Room, Tunnel & Electric Tray Room (35' El)
OB-FA-9	Office Building
OB-FZ-10A	Monitoring and Change Room - 46' El
OB-FZ-10B	Chem Lab, Laundry and Instrument Shop - 35' El
OB-FZ-22A	New Cable Spreading Room (Mech Equipment Room) - 63' 6" El
OB-FZ-22B	North Cable Bridge Tunnel - 74' 6"
OB-FZ-22C	South Cable Bridge Tunnel
OB-FZ-22D	Mechanical Equipment Room - 74 Foot Elevation
Turbine Building Areas	
TB-FA-3A	4160 VAC Switchgear 1C Vault (TB Mezz Lvl) - 23' El
TB-FA-3B	4160 VAC Switchgear 1D Vault (TB Mezz Lvl) - 23' El
TB-FZ-11A	Turbine Operating Floor - 46' El
TB-FZ-11B	Lube Oil Storage, Pumping & Purification Areas - 0' & 27'
TB-FZ-11C	1A/1B 4160 VAC Switchgear Room, (TB Mezz Lvl) - 23' El
TB-FZ-11D	Basement Floor, South End
TB-FZ-11E	Condenser Bay, 3' 6" Elevation
TB-FZ-11F	Feedwater Pump Area 0' 6" and 3' 6" Elevations
TB-FZ-11G	Mezzanine Level SE Corner and Machine Shop - 23' 6" El
TB-FZ-11H	Demineralizer Tank and SJAE Area - 23' 6" El
TB-FA-26	Battery Room C
Reactor Building Areas	
RB-FZ-1A	Refuel Floor (119 Foot Elevation)
RB-FZ-1B	95 Foot Elevation
RB-FZ-1C	75 Foot Elevation
RB-FZ-1D	51 Foot Elevation
RB-FZ-1E	Main Floor (23 Foot Elevation)
RB-FZ-1F	Torus Area and Corner Rooms (-19 Foot Elevation)
RB-FZ-1G	Shutdown Cooling Area (38 Foot and 51 Foot Levels)
RB-FZ-1H	Trunnion Room (23' 6" Elevation)
RB-FA-2	Primary Containment (Drywell and Torus)

Table 4.1-1 Oyster Creek Plant Fire Areas and Zones	
Fire Area or Zone	Description of Fire Area or Zone
Other Plant Areas	
MT-FA-12	Main Transformer Yard and Condensate Storage Tank
DG-FA-15	No. 1 Diesel Generator Room
DG-FA-17	No. 2 Diesel Generator Room
FS-FA-16	Emergency Diesel Generator Fuel Storage Area
CW-FA-14	Circulating Water Intake Area
AB-FA-13	Auxiliary Boiler Building
FW-FA-18	Diesel Driven (Pond) Fire Pump Area
OR-FA-19	Old Radwaste Building
NR-FA-20	New Radwaste Building
OG-FA-21	Augmented Offgas Building
NW-FA-23	New Warehouse
MB-FA-24	Maintenance Building
PH-FA-25	Redundant Fire Pump Area
EB-FA-28	Site Emergency Building
LL-FA-29	Low Level Radwaste Storage Facility

The Oyster Creek FHAR provides a fire area or zone designator (XX-YY-00). The "XX" is a two digit alphabetic code representing the Oyster Creek structure or building. The "YY" part of the designation is a two digit alphabetic code which provides the indication of a "Fire Area" as indicated with an "FA" or a "Fire Zone" as indicated with a "FZ". The final two digit are numeric and represent a unique number of the plant area.

The most common Oyster Creek FHAR structure or building codes ("XX") used in this analysis are: OB - Office Building, RB - Reactor Building and TB - Turbine Building. The remainder of the building codes can be determined from the table above.

A fire area (XX-FA-00) is defined by the FHAR as an area bounded by construction which will contain a fire to that area without reliance on automatic or manual fire suppression activities. A fire zone (XX-FZ-00) is defined by the FHAR as a subdivision of a fire area in which the fire suppression systems are designed to combat a particular type of fire.

For most of this analysis the terms (fire area and fire zone) are transparent. That is, the initiating event definitions and frequencies, initiating event impacts and initial estimations of upper bound core damage are all developed the same regardless of fire area or zone designation.

Two areas of this analysis fire area and zone designations impact the treatment of an area. The Fire Growth and Propagation report section (Section 4.3) where potential for a fire event to spread beyond a single fire area or zone are investigated. The second is in the Detailed Evaluation of Core Damage Frequency report section (Section 4.6.3) where potential for fire event suppression and protected circuits are investigated. The treatment of fire areas and zones are discussed in additional detail in the individual report sections referred to above.

4.1.1 Calculation of Generic Fire Area Frequencies

The calculation of generic fire area frequencies is composed of the comparison of Oyster Creek locations to those in the EPRI FIVE fire database and the assignment of generic location frequencies to these Oyster Creek locations. A division of this frequency among the Oyster Creek plant areas that were assigned to that type of location results in the determination of generic Oyster Creek fire area frequencies.

Table 4.1-2 Generic Fire Location Frequencies

Generic Location	Fire Source	Fire Frequency	Location Total	Fire Area Frequency
Reactor Building	Cabinets Pumps	5.00E-02 2.50E-02	See Note 2	
Diesel Generator Rooms	Diesel Generators Cabinets	2.60E-02 2.40E-03	2.84E-02	1.42E-02
Switchgear rooms	Cabinets	1.50E-02	1.50E-02	2.14E-03
Battery rooms	Batteries	3.20E-03	3.20E-03	1.60E-03
Control Room	Cabinets	9.50E-03	9.50E-03	9.50E-03
Cable spreading room	Cabinets	3.20E-03	3.20E-03	8.00E-04
Intake structure	Cabinets Fire pumps Other	2.40E-03 4.00E-03 3.20E-03	9.60E-03	9.60E-03
Turbine building	Turbine Generator Exciter	4.00E-03	9.50E-03	9.50E-03
	Turbine Generator Hydrogen	5.50E-03		
	Turbine Generator Oil	1.30E-02	1.30E-02	6.50E-03
	Cabinets Other Pumps	1.30E-02 6.30E-03	1.93E-02	3.86E-03
	Feedwater Pumps	4.00E-03	4.00E-03	4.00E-03
	Boiler	1.60E-03	1.60E-03	1.60E-03
Radwaste area	Miscellaneous Sources	8.70E-03	8.70E-03	2.90E-03
Yard - propagating to TB Yard - LOSP Yard (other)	Transformers	4.00E-03 1.60E-03 1.50E-02	2.06E-02	2.06E-02

1. This table is taken from Table 1.2, Appendix 10.3, EPRI FIVE report.
2. Fire frequency for the reactor building fire areas is allocated by totalling the number of fire sources in a given area and dividing by the total number of similar sources. These calculations are shown in Table 4.1-4.

Table 4.1-2, Generic Fire Location frequencies, is taken from the EPRI FIVE report. This table summarizes the frequency of fire events (by plant location) contained in the FIVE database. These locations are assigned to Oyster Creek specific locations. The generic location frequency is then divided among the plant areas assigned to those types of locations. The columns on Table 4.1-2 present the generic location, the fire sources, the generic fire frequency and the generic Oyster Creek fire area frequency.

Table 4.1-3, Assignment Oyster Creek Fire Areas and the following paragraphs describe the plant areas that are assigned to each of these types of locations and the method for dividing the location frequency into fire area frequencies.

- **Reactor Building** area fire frequency is distributed by the ratio of the number of pumps and electrical cabinets in each area to the total pumps and electrical cabinets in the 12 similar reactor and office building fire areas, as shown in Table 4.1-4.

- **Diesel Generator Room** fire frequency is evenly split between fire areas DG-FA-15 (Diesel generator no. 1) and DG-FA-17 (Diesel generator no. 2). This results in a fire area frequency of 1.42×10^{-2} for each of these fire areas.

- **Switchgear Room** fire frequency is evenly divided among the following 7 fire areas below. The resultant fire area frequency is 2.14×10^{-3} for each of these fire areas.

OB-FZ-6A	480 VAC Switchgear Room A
OB-FZ-6B	480 VAC Switchgear Room B
OB-FZ-8A/B	MG Set Room and Mechanical Equipment Room (analyzed as a single area with fire frequency of 4.28×10^{-3})
TB-FA-3A	4160 VAC Switchgear 1C
TB-FA-3B	4160 VAC Switchgear 1D
TB-FA-11C	4160 VAC Switchgear 1A and 1B

- **Battery Room** fire frequency is evenly split between fire areas OB-FA-8C (A and B Battery Room) and TB-FA-26 (C Battery Room). This results in a fire area frequency of 1.60×10^{-3} for each of these fire areas.

- **Control Room** location fire frequency is allocated entirely to fire zone OB-FZ-5 (Control Room). This results in a fire area frequency of 9.50×10^{-3}.

- **Cable Spreading Room** location fire frequency is evenly distributed among the 4 fire areas listed below. This results in a fire area frequency of 8.00×10^{-4} for each of the fire areas listed below.

OB-FZ-4	Cable Spreading Room (36 Foot Elevation)
OB-FA-22A	New Cable Spreading Room (63' 6" Elevation)
OB-FA-22B	North Cable Bridge Tunnel
OB-FA-22C	South Cable Bridge Tunnel

- **Intake Structure** fire frequency is allocated entirely to fire area CW-FA-14 (Circulating Water Intake Area). This results in a fire area frequency of 9.60×10^{-3} for this fire area.

- **Turbine Building** fire frequency is assigned by contributor as described below.

  The frequency allocation for Turbine Generator Exciter and Turbine Generator Hydrogen (9.50×10^{-3}) is assigned to TB-FZ-11A (Turbine Operating Floor).

  The Turbine Generator Oil frequency allocation of 1.30×10^{-2} is divided between TB-FZ-11A (Turbine Operating Floor) and TB-FZ-11B (Lube Oil Storage, Pumping and Purification Areas). This contributes 6.50×10^{-3} to each area.

  The entire feedwater pump frequency allocation of 4.00×10^{-3} is assigned to TB-FZ-11F (Feedwater Pump Area).

  The entire boiler frequency allocation of 1.60×10^{-3} is assigned to AB-FA-13 (Auxiliary Boiler Area).

  Fire frequency for Cabinets and Other Pumps is then divided among the remaining areas. This results in a fire area frequency contribution of 3.86×10^{-3} for each of the fire areas below.

TB-FZ-11D	Basement Floor, South End
TB-FZ-11E	Condenser Bay
TB-FZ-11G	Mezzanine Level and Machine Shop
TB-FZ-11H	Demineralizer Tank and Steam Jet Air Ejector Area
OG-FA-21	Augmented Offgas Building

- **Radwaste Area** frequency is evenly divided among the following 3 fire areas. This results in a fire area frequency of 2.90×10^{-3} for each of these fire areas.

OR-FA-19	Old Radwaste Building
NR-FA-20	New Radwaste Building
LL-FA-29	Low Level Radwaste Storage Facility

- **Transformer Yard - propagating to TB, Yard - LOSP and Yard (other)** fire frequencies apply to fire area MT-FA-12 (Main Transformer and CST). This results in a fire area frequency of 2.06×10^{-2} for this fire area. The emergency diesel generator fuel oil storage area (FS-FA-16) is also included with this area but not assigned any generic fire area frequency. The primary source of combustibles for this fire area consists of an underground fuel oil tank.

A severe fire in the following plant areas is expected to result in a controlled plant shutdown with no damage to risk significant components or supporting cables. Therefore, no fire area frequency is developed for them:

RB-FA-2 (Drywell and Torus) is contained entirely within the reactor building (RB-FA-1) and is inerted with nitrogen during power operation. A severe fire in this plant area is not considered credible and therefore no fire event frequency is developed.

FW-FA-18 (Diesel Driven (Pond) Fire Pump Area) is located approximately 1/4 mile from the plant itself, on the other side of the intake canal. No adjacent buildings, components or other structures are present. Also, a redundant (electric motor driven) fire pump is installed to provide fire protection system backup, should the fire pond or diesel driven pond pumps be unavailable. This area is protected by an automatic suppression and detection system (due to the presence of diesel fuel). A severe fire in this plant area is expected to result in a controlled shutdown. Therefore, no fire event frequency is developed for this area.

NW-FA-23 (New Warehouse) is separated from other plant structures by at least 50 feet of open space. Automatic sprinkler protection is provided throughout the building. A severe fire in this plant area is expected to result in a controlled shutdown with no damage to risk significant components or cables.

MB-FA-24 (Maintenance Building) is separated from other plant structures by at least 30 feet of open space. Automatic wet pipe sprinklers are provided throughout the building. This structure contains no risk significant components or cables.

PH-FA-25 (Redundant Fire Pump Area) consists of the shack that houses the redundant fire pump and its associated switchgear and the tank, which is located near the shack. Neither of these structures is adjacent to any other plant components, buildings or structures. Also, the redundant fire pump is not normally aligned to provide fire main pressure. A severe fire in this plant area is expected to result in a controlled plant shutdown with no damage to any other risk significant components or cables and therefore no fire event frequency is developed for this area.

EB-FA-28 (Site Emergency Building) is well separated, across a site access road, from other plant structures. This structure contains no risk significant components or cables.

Table 4.1-3 Assignment of Oyster Creek Plant Fire Areas

Generic Location	Fire Area or Zone	Description
Reactor Building	RB-FZ-1A	Refuel Floor (119 Foot Elevation)
	RB-FZ-1B	95 Foot Elevation
	RB-FZ-1C	75 Foot Elevation
	RB-FZ-1D	51 Foot Elevation
	RB-FZ-1E	Main Floor (23 Foot Elevation)
	RB-FZ-1F	Torus Area and Corner Rooms (-19 Foot Elevation)
	RB-FZ-1G	Shutdown Cooling Area (38 Foot and 51 Foot Levels)
	RB-FZ-1H	Trunnion Room (23' 6" Elevation)
	OB-FA-9	Office Building
	OB-FZ-10A	Monitoring and Change Room - 46 Foot Elevation
	OB-FZ-10B	Chem Lab, Laundry, Instrument Shop - 35 Foot Elevation
	OB-FZ-22D	Mechanical Equipment Room - 74 Foot Elevation
Diesel Generator Rooms	DG-FA-15	No. 1 Diesel Generator Room
	DG-FA-17	No. 2 Diesel Generator Room
Switchgear Room	OB-FZ-6A	"A" 480 VAC Switchgear Room
	OB-FZ-6B	"B" 480 VAC Switchgear Room
	OB-FZ-8A/B	MG Set Room and Mechanical Equipment Room (analyzed as one area in this evaluation)
	TB-FA-3A	4160 VAC Switchgear 1C Vault (TB Mezzanine - 23' Elevation)
	TB-FA-3B	4160 VAC Switchgear 1D Vault (TB Mezzanine - 23' Elevation)
	TB-FZ-11C	4160 VAC Switchgear 1A and 1B
Battery Room	OB-FA-8C	A & B Battery Rm, Tunnel and Electric Tray Rm - 35' Elevation
	TB-FA-26	Battery Room South of 4160 VAC Switchgear
Control Room	OB-FZ-5	Control Room - 46'6" Elevation
Cable Spreading Rooms	OB-FZ-4	Cable Spreading Room - 36' Elevation
	OB-FA-22A	New Cable Spreading Room (Mech Equip Rm) - 63'6" Elevation
	OB-FA-22B	North Cable Bridge Tunnel - 74'6" Elevation
	OB-FA-22C	South Cable Bridge Tunnel
Intake Structure	CW-FA-14	Circulating Water Intake Area
Turbine Generator Exciter and Hydrogen	TB-FZ-11A	Turbine Operating Floor - 46' Elevation

Table 4.1-3 Assignment of Oyster Creek Plant Fire Areas

Generic Location	Fire Area or Zone	Description
Turbine Generator Oil	TB-FZ-11A TB-FZ-11B	Turbine Operating Floor - 46' Elevation Lube Oil Storage, Pumping and Purification - 0' & 27' Elevations
Feedwater Pump	TB-FZ-11F	Feedwater Pump Area, 0'6" and 3"6" Levels
Boiler	AB-FA-13	Auxiliary Boiler House
Turbine Building Cabinets and Pumps	TB-FZ-11D TB-FZ-11E TB-FZ-11G TB-FZ-11H OB-FA-21	Basement Floor - South End Condenser Bay - 3'6" Elevation Mezzanine Level SE Corner and Machine Shop - 23'6" Demin Tank and SJAE Area - 23'6" Elevation Augmented Offgas Building
Radwaste Area	OR-FA-19 NR-FA-20 LL-FA-29	Old Radwaste Area New Radwaste Area Low Level Radwaste Storage Facility
Transformer Yard (ALL)	MT-FA-12 FS-FA-16	Main Transformer and CST Emergency Diesel Generator Fuel Oil Storage Area
These fire areas do not cause a plant trip or damage risk significant equipment	RB-FA-2 FW-FA-18 NW-FA-23 MB-FA-24 PH-FA-25 EB-FA-28	Drywell and Torus Diesel Driven (Pond) Fire Pump Area New Warehouse Maintenance Building Redundant Fire Pump Area Site Emergency Building

In the case of the reactor building, generic fire frequency is divided among fire areas differently. Based on the guidance provide in the EPRI FIVE documentation, the generic fire location frequency is apportioned by the ratio of the number of pumps and cabinets (ignition sources) in a fire area divided by the total number of pumps and cabinets in the reactor building. The calculation of Reactor Building fire area frequency is shown in Table 4.1-4.

Area layout drawings were used to identify the pump and electrical cabinet fire sources within each area for allocation of Reactor Building fire frequency. The total fire area frequency for each of these areas is shown in Table 4.1-11.

The EPRI FIVE methodology does not specifically consider containment fires, since there is a small number of events and previous fire PRA efforts did not show containment fires as risk significant. For Oyster Creek containment fires are not addressed since the drywell and torus are inerted with Nitrogen during power operation.

Table 4.1-4 Calculation of Reactor Building Fire Area Frequency

Fire Area or Zone	Description of Area	No. of Cabinets	Frequency	No. of Pumps	Frequency	Total Frequency
RB-FZ-1A	119 Foot Elevation	3	5.77E-03	2	1.67E-03	7.44E-03
RB-FZ-1B	95 Foot Elevation	0	0	2	1.67E-03	1.67E-03
RB-FZ-1C	75 Foot Elevation	2	3.85E-03	6	5.00E-03	8.85E-03
RB-FZ-1D	51 Foot Elevation	9	1.73E-02	7	5.83E-03	2.31E-02
RB-FZ-1E	23 Foot Elevation	5	9.62E-03	4	3.33E-03	1.29E-02
RB-FZ-1F	-19 Foot Elevation	2	3.85E-03	4	3.33E-03	7.18E-03
RB-FZ-1G	38' and 51' Elevations	0	0	3	2.50E-03	2.50E-03
RB-FZ-1H	Trunnion Room	0	0	0	0	0
OB-FA-9	Office Building	0	0	0	0	0
OB-FZ-10A	Monitoring and Change Room - 46' Elevation	4	7.69E-03	0	0	7.69E-03
OB-FZ-10B	Chem Lab, Laundry, Inst Shop 35' Elevation	0	0	0	0	0
OB-FZ-22D	Mech Equip Room 74'	1	1.92E-03	2	1.67E-03	3.59E-03
Totals for the Reactor Building		26	5.00E-02	30	2.50E-02	7.50E-02

4.1.2 Plant Specific Fire Area Frequencies

In addition to the generic fire area frequencies calculated above, fire area frequency based on plant specific components is also calculated and presented in this report section. The EPRI FIVE methodology provides fire frequencies for specific plant components and combustible loading. These frequencies are divided among the similar equipment in the plant and result in additional frequency based on the type of components and combustible loading in the plant. The sources of additional fire event frequency are:

- Fire Protection Panels
- RPS Motor Generator Sets
- Non-qualified cable, Non-qualified and Qualified Junction [illegible]
- Transformers
- Battery Chargers
- Offgas/Hydrogen Recombiners
- Hydrogen Tanks
- Other Hydrogen Fires
- Gas Turbines
- Air Compressors
- Ventilation Subsystems
- Elevator Motors
- Laundry Dryers

- Transient Sources (welding, cutting and grinding) - calculated, based on the transient fire loading calculations provided in the FHAR, Section 8.0.

Each of the above components represent ignition or fuel sources and contribute additional fire frequency to the plant specific fire frequencies for the area of concern. However, the frequency is component (ignition or fuel source) specific. Therefore, the frequency of the plant-wide ignition and fuel sources is divided, using the ratio of number of ignition and fuel sources within a fire area divided by the total number of ignition and fuel sources in the entire plant. The paragraphs below detail the each of the ignition/fuel sources and their division among the Oyster Creek fire areas and zones.

Fire Protection Panels EPRI FIVE fire frequency is 2.40×10^{-3} for all fire protection panels in the plant. This frequency is evenly distributed among the 16 fire protection panels that are located in critical fire areas (see Operations Plant Manual table 19-2). Therefore, fire frequency is divided as follows.

Table 4.1-5 Fire Protection Panel Contribution to Fire Area Frequency

Fire Area	Fire Area Description	No. of Panels	Additional Fire Area Frequency
OB-FZ-4	Cable Spreading Room - 36' El	1	1.50E-04
OB-FZ-5	Control Room - 46'6" El	3	4.50E-04
OB-FA-9	Office Building	3	4.50E-04
OB-FZ-10A	Monitoring and Change Room - 46' El	1	1.50E-04
OB-FA-22A	New Cable Spreading Room (Mech Equip Room) - 63'6" El	1	1.50E-04
TB-FZ-11B	Lube Oil Storage, Pumping and Purification - 0' & 27' El	1	1.50E-04
TB-FZ-11C	4160 VAC Switchgear 1A and 1B	1	1.50E-04
RB-FZ-1D	Reactor Building - 51' El	1	1.50E-04
FS-FA-16	Diesel Generator Fuel Oil Storage	1	1.50E-04
AB-FA-13	Auxiliary Boiler House	1	1.50E-04
NR-FA-20	New Radwaste Building	1	1.50E-04
OG-FA-21	Augmented Offgas Building	1	1.50E-04
	Total Fire Protection Panels	16	2.40E-03

Fire frequency due to this source is calculated by dividing the total frequency (2.40×10^{-3}) by 16, resulting in a fire frequency of 1.50×10^{-4} for each panel.

RPS Motor Generator Sets are installed in the old cable spreading room (fire zone OB-FZ-4). The total FIVE fire frequency of 5.5×10^{-3} for motor generator sets is assigned to this area.

Non-qualified Cable, Non-qualified and Qualified Junction Boxes. EPRI FIVE fire frequency allocation is performed on the basis of total fire area BTU loading for instrument, power, control and general cables. Combustible loading is calculated for each of these types of cables in the fire hazards analysis report (FHAR) and appears in tabular form in Section 8.0 of that study. Since this information is given on the basis of fire loading per square foot, total area loading was calculated using the surface areas given in the text of the FHAR. Calculation of the plant specific fire area frequencies are shown in Table 4.1-6, below.

Table 4.1-6 Cable and Junction Box Contribution to Fire Area Frequency

Fire Area or Zone	Area (ft2) [A]	Combustible Loading due to Cable (BTU/ft2)					Total Cable BTU (000's) (=[A]x[B])	Total Fire Frequency
		Instrument	Power	Control	General	Total [B]		
OB-FZ-4	2,543	301	28	312	49,390	50,031	127,230	8.92E-04
OB-FZ-5	2,764	2,594	7,757	10,597	4,323	25,271	69,848	4.89E-04
OB-FZ-6A	1,157	0	54	0	46,303	46,357	53,635	3.76E-04
OB-FZ-6B	679	376	105	0	39,362	39,843	27,053	1.90E-04
OB-FZ-8A/B	2,607	0	4	0	5,841	5,845	15,237	1.07E-04
OB-FZ-8C	1,292	0	0	0	27,554	27,544	35,600	2.49E-04
OB-FA-9	5,739	547	4	0	1,294	1,845	10,589	7.42E-05
OB-FZ-10A	2,019	45	20	276	7,429	7,771	15,690	1.10E-04
OB-FZ-10B	1,517	67	0	0	0	67	120	7.15E-07
OB-FZ-22A	3,435	2,069	722	4,650	0	7,442	25,562	1.79E-04
OB-FZ-22B	1,084	765	1,573	789	0	3,127	3,389	2.37E-05
OB-FZ-22C	1,084	483	858	1,673	0	3,014	3,267	2.29E-05
OB-FZ-22D	173	0	0	0	0	0	0	0
TB-FA-3A	312	0	30	0	103,365	103,395	32,259	2.26E-04
TB-FA-3B	336	0	28	0	31,994	32,022	10,759	7.54E-05
TB-FZ-11A	20,787	7	0	0	0	7	148	1.04E-06
TB-FZ-11B	3,175	0	0	0	1,575	1,575	5,000	3.50E-05
TB-FZ-11C	2,666	0	3	0	0	3	9	6.31E-08
TB-FZ-11D	9,668	7	48	0	10,344	10,397	100,529	7.04E-04
TB-FZ-11E	26,427	0	2	0	6,433	6,435	170,062	1.19E-03
TB-FZ-11F	5,650	0	2	0	0	2	9	6.31E-08
TB-FZ-11G	5,987	7	5	0	0	12	71	4.98E-07
TB-FZ-11H	3,944	0	16	0	0	16	62	4.34E-07
TB-FA-26	140	0	0	0	2,143	2,143	300	2.10E-06
RB-FZ-1A	15,813	16	11	8	0	35	550	3.85E-06
RB-FZ-1B	10,380	50	139	400	0	589	6,113	4.28E-05
RB-FZ-1C	11,655	95	413	463	6,006	6,977	81,323	5.70E-04
RB-FZ-1D	9,100	62	957	89	17,582	18,690	170,079	1.19E-03
RB-FZ-1E	12,140	123	395	184	20,781	21,483	260,810	1.83E-03
RB-FZ-1F	13,690	36	0	0	0	36	494	3.46E-06
RB-FZ-1G	1,609	0	0	0	1,865	1,865	3,000	2.10E-05
RB-FZ-1H	530	856	0	0	0	856	454	3.18E-06

Table 4.1-6 Cable and Junction Box Contribution to Fire Area Frequency

Fire Area or Zone	Area (ft2) [A]	Combustible Loading due to Cable (BTU/ft2)					Total Cable BTU (000's) (=[A]x[B])	Total Fire Frequency
		Instrument	Power	Control	General	Total [B]		
MT-FA-12	N/A	0	0	0	0	0	0	0
DG-FA-15	1,269	0	0	0	158	158	200	1.40E-06
DG-FA-17	1,335	0	7	0	150	157	209	1.46E-06
FS-FA-16	281	0	0	0	0	0	0	0
CW-FA-14	N/A	N/A	N/A	N/A	N/A	N/A	10,211	7.16E-05
AB-FA-13	1,870	0	0	0	7	7	12	8.41E-08
OR-FA-19	11,713	0	60	0	0	60	698	4.89E-06
NR-FA-20	10,132	8	138	6	10,659	10,811	109,534	7.68E-04
OG-FA-21	3,972	0	38	0	1,385	1,423	5,650	3.69E-05
LL-FA-29		0	0	0	0	0	0	0
Total Fire Frequency for Cable and Junction Boxes								**9.50E-03**

Notes:
1. Area square footage is taken from FHAR area descriptions.
2. BTU loading is taken from FHAR Section 8.0. Values are rounded to integers.
3. Total BTU loading is rounded to the nearest 1,000 BTU.
4. No distiction is made between qualified, unqualified cable and junction boxes.

Transformer Fires. The EPRI FIVE fire frequency due to transformer fires (7.90×10^{-3}) is apportioned by the ratio of transformers in a fire area divided by the total number of transformers within the plant. Calculation of the fire area frequency due to plant transformers is presented in Table 4.1-7, below.

Table 4.1-7 Transformer Contribution to Fire Area Frequency

Fire Area	Transformers	Fire Frequency
OB-FZ-4	2	9.29E-04
OB-FZ-6A	1	4.65E-04
OB-FZ-6B	1	4.65E-04
OB-FZ-8A/B	1	4.65E-04
OB-FA-9	1	4.65E-04
OB-FZ-10A	1	4.65E-04
OB-FZ-22A	1	4.65E-04
TB-FZ-11A	4	1.86E-03
TB-FZ-11B	1	4.65E-04
TB-FZ-11D	2	9.29E-04
RB-FZ-1A	1	4.65E-04
OR-FA-19	1	4.65E-04
Total	17	7.90E-03

Battery Chargers are located in the A/B battery room (OB-FZ-8C, with two (2) - 125 VDC rotary units), the 1C switchgear room (TB-FA-3A, with two (2) - 125 VDC solid state units) and the old cable spreading room (OB-FZ-4, with four (4) - 24 VDC solid state chargers). Fire frequency due to this source is therefore calculated by dividing the total EPRI FIVE frequency of battery charger fires ($4.00x10^{-3}$) by four (4), resulting in a component fire frequency of $1.00x10^{-3}$ for OB-FZ-8C and TB-FA-3A and $2.00x10^{-3}$ for OB-FZ-4.

Offgas/Hydrogen Recombiner fires apply to the Augmented Offgas Building only. Therefore, the entire EPRI FIVE fire frequency of $8.6x10^{-2}$ for offgas/hydrogen recombiners is applied to fire area (OG-FA-21).

Hydrogen Tanks for the main generator are located outside the plant protected area and away from other plant components.

Other Hydrogen Fires include only those that are not associated either with site hydrogen tanks or with the main generator. Due to the potential presence of industrial gases in fire area OB-FZ-10B (Chem Lab, Laundry and Instrument Shop), this area was confirmed during walkdowns to contain no combustible gases, aside from a minor amount of propane.

The EPRI FIVE fire frequency for other hydrogen fires ($3.20x10^{-3}$) is therefore assigned to the feedwater Hydrogen injection system, located in fire area TB-FZ-11F.

Gas Turbine fires apply only to the combustion turbine peaking units installed across the intake channel, but still on the Oyster Creek site. Since these units are not located within the plant boundary, these fire sources are not separately considered in this analysis.

Air Compressor fire frequency is allocated among the main plant air compressors, which are located in the south end of the turbine building basement and in the new radwaste facility. Therefore, 3/4 of the EPRI FIVE fire frequency ($4.70x10^{-3}$) is applied to turbine building fire area TB-FZ-11D ($3.53x10^{-3}$) and $1.17x10^{-3}$ is allocated to NR-FA-20.

Ventilation Subsystem Fires are assigned based on the ratio of ventilation subsystems within a plant fire area to the total number of normally operating ventilation subsystems within the entire plant. Ventilation system location determination was made based on site area layout drawings and walkdowns. Calculation of this fire frequency for applicable areas is shown in Table 4.1-8.

Table 4.1-8 Ventilation Subsystem Fire Frequency Contribution

Fire Area	Vent Fans	Fire Frequency
OB-FZ-8A/B OB-FZ-22A	3 2	1.68E-03 1.12E-03
TB-FZ-3A TB-FZ-3B TB-FZ-11B	1 1 7	5.59E-04 5.59E-04 3.91E-03
RB-FZ-1C RB-FZ-1E	Not normally running	
NR-FA-20	3	1.68E-03
Total	17	9.50E-03

Elevator Motors contribute 6.30×10^{-3} to the component specific fire frequency. One third of this frequency or 2.10×10^{-3} is allocated to each of the Oyster Creek plant areas that contain elevator motors. These are:

Reactor Building (RB-FZ-1A)
Turbine Building (TB-FZ-11A)
Office Building (OB-FA-9).

Laundry Dryers are located only in the New Radwaste Building (NR-FA-20) and the Maintenance Building (MB-FA-24). Therefore, the EPRI FIVE fire frequency of 8.7×10^{-3} is equally applied to fire areas NR-FA-20 and MB-FA-24 (4.35×10^{-3} each).

Transient Sources, including cable fires from welding and fires resulting from cutting and welding are calculated, based on the transient fire loading calculations provided in the FHAR, Section 8.0, as shown in Table 4.1-9.

It is assumed that transient combustibles transported in safety containers do not contribute to the plant specific fire area frequency. The maximum allowable transient combustible without a safety container (equal to one pint of acetone) is listed in each fire area or zone in addition to any other transient combustible noted in the FHAR or plant walkdowns. This assumption is considered conservative since plant procedure (Control of Combustibles, 120.5, Revision 5) requires unattended combustibles to be removed from the area or stored in transient combustible lockers. Also, plant procedures (Cutting, Welding, Miscellaneous Hot Work Administrative Procedure, 120.1, Revision 13) require a fire watch be posted whenever welding activities are underway.

Table 4.1-9 Transient Combustible Contribution to Fire Area Frequency

Fire Area	Area (ft^2)	Transient Combustibles (BTU/ft^2)	Transient Combustibles (BTU)	Transient Fire Frequency	Welding Cable Fires	Cutting and Welding	Total
OB-FZ-4	2,543	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-5	2,764	141	391,000	3.15E-04	1.21E-04	7.38E-04	1.17E-03
OB-FZ-6A	1,157	366	423,000	3.40E-04	1.21E-04	7.38E-04	1.20E-03
OB-FZ-6B	679	623	423,000	3.40E-04	1.21E-04	7.38E-04	1.20E-03
OB-FZ-8A/B	2,607	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-8C	1,292	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FA-9	5,739	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-10A	2,019	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-10B	1,517	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-22A	3,435	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-22B	1,084	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-22C	1,084	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OB-FZ-22D	173	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FA-3A	312	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FA-3B	336	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11A	20,787	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11B	3,175	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11C	2,666	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11D	9,668	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11E	26,427	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11F	5,650	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11G	5,987	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FZ-11H	3,944	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
TB-FA-26	140	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1A	15,813	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1B	10,380	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1C	11,655	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1D	9,100	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1E	12,140	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1F	13,690	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1G	1,609	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
RB-FZ-1H	530	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
MT-FA-12		0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
DG-FA-15	1,269	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
DG-FA-17	1,335	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
FS-FA-16	281	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
CW-FA-14		0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
AB-FA-13	1,870	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OR-FA-19	11,713	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
NR-FA-20	10,312	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
OG-FA-21	3,972	0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
LL-FA-29		0	10,000	7.85E-06	1.21E-04	7.38E-04	8.67E-04
Transient Fire Area Frequency			1.66E+06	1.30E-03	5.10E-03	3.10E-02	3.74E-02

Table 4.1-10 Total Plant Specific Component Fire Area Frequency Contributions								
Fire Area	Description	Fire Panels	Cables (Note 1)	Transformer Fires	Ventilation Fires	Transient Combustibles	Misc. (Note 2)	Total
OB-FZ-4	Cable Spreading Room - 36 Foot Elevation	1.50E-04	8.92E-04	9.29E-04		8.67E-04	7.50E-03	1.03E-02
OB-FZ-5	Main Control Room - 46' 6" Elevation	4.50E-04	4.89E-04			1.17E-03		2.11E-03
OB-FZ-6A	"A" 480 VAC Switchgear Room		3.76E-04	4.65E-04		1.20E-03		2.04E-03
OB-FZ-6B	"B" 480 VAC Switchgear Room		1.90E-04	4.65E-04		1.20E-03		1.86E-03
OB-FZ-8A/B	MG Set Room and Mechanical Equipment Room		1.07E-04	4.65E-04	1.68E-03	8.67E-04		3.12E-03
OB-FZ-8C	A/B Battery Rm, Tunnel and Electric Tray Rm		2.49E-04			8.67E-04	1.00E-03	2.11E-03
OB-FA-9	Office Building	4.50E-04	7.42E-05	4.65E-04		8.67E-04	2.10E-03	3.96E-03
OB-FZ-10A	Monitoring and Change Room	1.50E-04	1.10E-04	4.65E-04		8.67E-04		1.59E-03
OB-FZ-10B	Chem Lab, Laundry and Instrument Shop		7.15E-07			8.67E-04		8.68E-04
OB-FZ-22A	New Cable Spreading (Mech Equip Rm)	1.50E-04	1.79E-04	4.65E-04	1.12E-03	8.67E-04		2.78E-03
OB-FZ-22B	North Cable Bridge Tunnel		2.37E-05			8.67E-04		8.91E-04
OB-FZ-22C	South Cable Bridge Tunnel		2.29E-05			8.67E-04		8.80E-04
OB-FZ-22D	Mechanical Equipment Room					8.67E-04		8.67E-04
TB-FA-3A	4160 VAC Switchgear 1C Vault		2.26E-04		5.59E-04	8.67E-04	1.00E-03	2.65E-03
TB-FA-3B	4160 VAC Switchgear 1D Vault		7.54E-05		5.59E-04	8.67E-04		1.50E-03
TB-FZ-11A	Turbine Operating Floor		1.04E-06	1.86E-03		8.67E-04	2.10E-03	4.83E-03
TB-FZ-11B	Lube Oil Storage, Pumping and Purif Areas	1.50E-04	3.50E-05	4.65E-04	3.91E-03	8.67E-04		5.42E-03
TB-FZ-11C	1A and 1B 4160 VAC Switchgear Room	1.50E-04	6.31E-08			8.67E-04		1.02E-03
TB-FZ-11D	Basement Floor, South End		7.04E-04	9.29E-04		8.67E-04	3.53E-03	6.02E-03
TB-FZ-11E	Condenser Bay		1.19E-03			8.67E-04		2.06E-03
TB-FZ-11F	Feedwater Pump Area		6.31E-08			8.67E-04	3.20E-03	4.07E-03
TB-FZ-11G	Mezzanine Level SE Corner and Machine Shop		4.98E-07			8.67E-04		8.67E-04
TB-FZ-11H	Demin Tank and Steam Jet Air Ejector Area		4.34E-07			8.67E-04		8.67E-04
TB-FA-26	Battery Room C		2.10E-06			8.67E-04		8.69E-04
RB-FZ-1A	Refuel Floor (119' Elevation)		6.17E-06	4.65E-04		8.67E-04	2.10E-03	3.43E-03
RB-FZ-1B	95' Elevation		6.86E-05			8.67E-04		9.36E-04
RB-FZ-1C	75' Elevation		5.83E-04			8.67E-04		1.45E-03
RB-FZ-1D	51' Elevation	1.50E-04	1.15E-03			8.67E-04		2.16E-03
RB-FZ-1E	Main Floor (23' Elevation)		1.74E-03			8.67E-04		2.60E-03
RB-FZ-1F	Torus Area and Corner Rooms (-19')		5.55E-06			8.67E-04		8.73E-04
RB-FZ-1G	Shutdown Cooling Area (38' and 51' Levels)		1.95E-05			8.67E-04		8.87E-04
RB-FZ-1H	Trunnion Room (23' 6" Elevation)		5.10E-06			8.67E-04		8.70E-04

Table 4.1-10 Total Plant Specific Component Fire Area Frequency Contributions								
Fire Area	Description	Fire Panels	Cables (Note 1)	Transformer Fires	Ventilation Fires	Transient Combustibles	Misc. (Note 2)	Total
MT-FA-12	Main Transformer and CST					8.67E-04		8.67E-04
DG-FA-15	No. 1 Diesel Generator Room		1.30E-06			8.67E-04		8.68E-04
DG-FA-17	No. 2 Diesel Generator Room		1.40E-06			8.67E-04		8.68E-04
FS-FA-16	Emergency Diesel Generator Fuel Oil Storage	1.50E-04				8.67E-04		1.02E-03
CW-FA-14	Circulating Water Intake Area		9.26E-05			8.67E-04		9.50E-04
AB-FA-13	Auxiliary Boiler House	1.50E-04	8.07E-08			8.67E-04		1.02E-03
OR-FA-19	Old Radwaste Area		7.83E-06	4.65E-04		8.67E-04		1.34E-03
NR-FA-20	New Radwaste Area	1.50E-04	7.20E-04		1.68E-03	8.67E-04	5.52E-03	8.93E-03
OG-FA-21	Augmented Offgas Building	1.50E-04	3.75E-05			8.67E-04	8.60E-02	8.70E-02
LL-FA-29	Low Level Radwaste Storage Facility					8.67E-04		8.67E-04
Total Plant Specific Fire Area Frequencies		**2.40E-03**	**9.50E-03**	**7.90E-03**	**9.50E-03**	**3.74E-02**	**1.15E-01**	**1.82E-01**
RB-FA-2	Drywell and Torus	**NOTE 3**						
FW-FA-18	Diesel Driven (Pond) Fire Pump Area							
PH-FA-25	Redundant Fire Pump Area							
NW-FA-23	New Warehouse							
MB-FA-24	Maintenance Building							
EB-FA-28	Site Emergency Building							

Note 1: The "Cables" column provides the contribution of non-qualified cable and qualified and non-qualified junction boxes.

Note 2: The "Misc." column provides the contribution of RPS motor generators sets, battery chargers, hydrogen fires (including offgas/hydrogen recombiner, hydrogen tank and other hydrogen fires), air compressors, elevator motors and laundry driers.

Note 3: Fires in the fire areas indicated by this note do not result in a plant trip or significant plant damage. Therefore, no fire frequency is developed.

4.1.3 Summary of Fire Area Frequency

The total Oyster Creek fire area frequency is obtained by summing the location specific fire frequency and the frequency associated with any plant-wide components located in the fire area. These calculations are shown in Table 4.1-11.

Table 4.1-11 Summary of Fire Area Ignition Frequency

Fire Area or Zone	Description	Generic Fire Frequency	Component Fire Frequency	Total Ignition Frequency
OB-FZ-4	Cable Spreading Room - 36 Foot Elevation	8.00E-04	1.03E-02	1.11E-02
OB-FZ-5	Main Control Room - 46' 6" Elevation	9.50E-03	2.11E-03	1.16E-02
OB-FZ-6A	"A" 480 VAC Switchgear Room	2.14E-03	2.04E-03	4.18E-03
OB-FZ-6B	"B" 480 VAC Switchgear Room	2.14E-03	1.86E-03	4.00E-03
OB-FZ-8A/B	MG Set Room and Mechanical Equipment Room	4.28E-03	3.12E-03	7.40E-03
OB-FZ-8C	A/B Battery Rm, Tunnel and Electric Tray Rm	1.60E-03	2.11E-03	3.71E-03
OB-FA-9	Office Building	0	3.96E-03	3.96E-03
OB-FZ-10A	Monitoring and Change Room	7.69E-03	1.59E-03	9.28E-03
OB-FZ-10B	Chem Lab, Laundry and Instrument Shop	0	8.68E-04	8.68E-04
OB-FZ-22A	New Cable Spreading (Mech Equip Rm)	8.00E-04	2.78E-03	3.58E-03
OB-FZ-22B	North Cable Bridge Tunnel	8.00E-04	8.91E-04	1.69E-03
OB-FZ-22C	South Cable Bridge Tunnel	8.00E-04	8.80E-04	1.69E-03
OB-FZ-22D	Mechanical Equipment Room	3.58E-03	8.67E-04	4.45E-03
TB-FA-3A	4160 VAC Switchgear 1C Vault	2.14E-03	2.65E-03	4.79E-03
TB-FA-3B	4160 VAC Switchgear 1D Vault	2.14E-03	1.50E-03	3.64E-03
TB-FZ-11A	Turbine Operating Floor	1.60E-02	4.83E-03	2.08E-02
TB-FZ-11B	Lube Oil Storage, Pumping and Purif Areas	6.50E-03	5.42E-03	1.19E-02
TB-FZ-11C	1A and 1B 4160 VAC Switchgear Room	2.14E-03	1.02E-03	3.28E-03
TB-FZ-11D	Basement Floor, South End	3.86E-03	6.02E-03	1.09E-02
TB-FZ-11E	Condenser Bay	3.86E-03	2.06E-03	5.92E-03
TB-FZ-11F	Feedwater Pump Area	4.00E-03	4.07E-03	8.07E-03
TB-FZ-11G	Mezzanine Level SE Corner and Machine Shop	3.86E-03	8.67E-04	4.73E-03
TB-FZ-11H	Demin Tank and Steam Jet Air Ejector Area	3.86E-03	8.67E-04	4.73E-03
TB-FA-26	Battery Room C	1.60E-03	8.69E-04	2.47E-03
RB-FZ-1A	Refuel Floor (119' Elev)	7.44E-03	3.43E-03	1.09E-02
RB-FZ-1B	95 Foot Elevation	1.67E-03	9.36E-04	2.61E-03
RB-FZ-1C	75 Foot Elevation	8.85E-03	1.45E-03	1.03E-02
RB-FZ-1D	51 Foot Elevation	2.31E-02	2.16E-03	2.53E-02
RB-FZ-1E	Main Floor (23' Elevation)	1.29E-02	2.60E-03	1.55E-02
RB-FZ-1F	Torus Area and Corner Rooms (-19' Elev)	7.18E-03	8.73E-04	8.05E-03
RB-FZ-1G	Shutdown Cooling Area (38' and 51' Elev)	2.50E-03	8.87E-04	3.39E-03
RB-FZ-1H	Trunnion Room (23' 6" Elev)	0	8.70E-04	8.70E-04

Table 4.1-11 Summary of Fire Area Ignition Frequency				
Fire Area or Zone	Description	Generic Fire Frequency	Component Fire Frequency	Total Ignition Frequency
MT-FA-12	Main Transformer and CST	2.06E-02	8.67E-04	2.14E-02
DG-FA-15	No. 1 Diesel Generator Room	1.42E-02	8.68E-04	1.50E-02
DG-FA-17	No. 2 Diesel Generator Room	1.42E-02	8.68E-04	1.50E-02
FS-FA-16	Emergency Diesel Generator Fuel Oil Storage	0	1.02E-03	1.02E-03
CW-FA-14	Circulating Water Intake Area	9.60E-03	9.50E-04	1.05E-02
AB-FA-13	Auxiliary Boiler House	1.60E-03	1.02E-03	2.62E-03
OR-FA-19	Old Radwaste Area	2.90E-03	1.34E-03	4.24E-03
NR-FA-20	New Radwaste Area	2.90E-03	8.93E-03	1.29E-03
OG-FA-21	Augmented Offgas Building	3.86E-03	8.70E-02	9.09E-02
LL-FA-29	Low Level Radwaste Storage Facility	2.90E-03	8.67E-04	3.77E-03
	TOTAL FIRE AREA FREQUENCY	2.21E-01	1.81E-01	4.02E-01
RB-FA-2	Drywell and Torus	**SCREENED FIRE AREAS**		
FW-FA-18	Diesel Driven (Pond) Fire Pump Area			
PH-FA-25	Redundant Fire Pump Area			
NW-FA-23	New Warehouse			
MB-FA-24	Maintenance Building			
EB-FA-28	Site Emergency Building			

4.2 Review of Plant Information and Plant Walkdowns

This report section presents and documents the review of plant information and plant walkdowns performed in support of the various tasks of this analysis.

Plant information gathering and walkdowns are continuous throughout the project and performed in support of all tasks. By utilizing the various plant references and by performing physical walkdowns of various plant areas, this step supports the various tasks shown in the Overview of Fire Individual Plant Examination Process, as illustrated on Figure 4-1.

Several information sources which were utilized significantly included the Oyster Creek Fire Hazard Analysis Report (FHAR), the Oyster Creek Probabilisitic Risk Assessment (Level 1 and 2) and site and building layout drawings. Many other sources of information were used in the development of the Oyster Creek Fire Individual Plant Examination and a more complete list is available in report Section 4.10, References.

In addition to the above mentioned sources, an interactive laserdisk walkdown system (Oyster Creek Surrogate Tour) was used to supplement physical walkdowns. This system contains digitized photographs of plant areas which can be viewed on a computer screen by selecting the areas of interest. This system was used for general familiarization with several fire areas, as well as a detailed review of the physical layout for the detailed screening analysis and was essential in reducing exposure in high radiation areas during the walkdown process.

In general, physical walkdowns of the various plant fire areas were used during several phases of the fire analysis process. These include:

- General Plant Familiarization walkdowns, which were performed to familiarize analysts with general plant layout and various plant systems.

- Component and Cable Location plant walkdowns, which were performed in support of the identification of component and cable locations (Tasks 3).

- Fire Growth and Propagation plant walkdowns, which were performed in conjunction with the component and cable location walkdowns. The information gathered during these walkdowns was used in support of the determination of potential multiple area fires. Specifically, fire area boundaries and mitigative features (including spatial separation of equipment and combustible loading within a fire area) were noted during the walkdown.

- Detailed Fire Propagation Analysis Walkdowns. Following the initial fire area screening analysis, which is described in detail in Sections 4.6.1 and 4.6.2, each of the areas that required further detailed analysis, as described in Section 4.6.3, were then reviewed by physical walkdown to confirm the validity of the analysis and that no additional fire hazards or concerns were present.

It should be noted that the walkdowns performed for this analysis are in addition to a number of other administrative procedures that direct plant management to confirm that unidentified fire hazards are not present. Specifically, these include:

- Walkdowns that are performed by plant personnel on a continual basis, as directed by the off shift management tour program. In essence, this process ensures that the accessible areas of the plant will be reviewed on an ongoing basis for safety hazards, including fire sources, as well as general plant housekeeping.

- The site fire protection engineer is required to physically walk down the various plant fire areas to confirm that unidentified fire sources do not exist in the plant and that plant fire barriers and detection systems are functioning normally.

- Plant procedures that require monthly surveillance to confirm availability and operability of equipment that may be required by the fire brigade.

These walkdowns are performed on a continuing basis and are in addition to those performed by the plant fire protection engineer during the maintenance of Appendix R documentation and various plant personnel during the review of plant design changes.

In addition to the general walkdowns discussed above, specific walkdowns were performed to support various tasks or to verify analysis assumptions and/or results. These walkdowns included the participation of Risk Analysis personnel (Ken Canavan), Risk Analysis Contractor personnel (Delta Prime, Inc., Mitch Waller) and a plant fire protection engineers (Raymond Daley and Timothy Trettel). In all seven (7) specific walkdowns lists were developed and these items were addressed during walkkdown activities.

1. Review areas that were screened from consideration in Report Section 4.1 (areas that would not result in a plant trip, even for a severe fire):

FW-FA-18	Pond pump area
PH-FA-25	Redundant fire pump area
NW-FA-23	New warehouse
MB-FA-24	Maintenance building
EB-FA-28	Site emergency building

Note: RB-FA-2 (Drywell and torus) was also removed, but is inerted

2. Review areas that were screened from consideration in Report Section 4.2 (areas that do not contain risk significant components or support cables):

RB-FZ-1A	Refuel floor
AB-FA-13	Auxiliary boiler building
OR-FA-19	Old radwaste building
NR-FA-20	New radwaste building
OG-FA-21	Augmented offgas building
OB-FZ-22D	Mechanical equipment room (74' elevation)
LL-FA-29	Low level radwaste storage facility

3. Review fire boundaries for the following areas, which were screened in Report Section 4.3 as fire rated or not adjacent to other plant areas:

MT-FA-12	Main transformer area
AB-FA-13	Auxiliary boiler
CW-FA-14	Intake area
DG-FA-15	DG 1 (adjacent to DG-2)
FS-FA-16	Fuel oil storage area
DG-FA-17	DG 2 (adjacent to DG-1)
OR-FA-19	Old radwaste building
NR-FA-20	New radwaste building
OG-FA-21	Augmented offgas building
LL-FA-29	Low level radwaste storage facility

4. Review fire zones for which fire growth across boundaries was screened, based on EPRI FIVE criteria 3, 4, 5 or 6 (i.e., concentrated combustibles, combustibles near zone barriers). The list below summarizes these plant fire areas and zones. For details on the application of the EPRI FIVE criteria on fire boundary screening, see report Section 4.3.

OB-FA-5 Control Room Complex — **Criteria 2 and 3**

OB-FZ-4	Cable Spreading Room - 36' El.	Criterion 6
OB-FZ-5	Control Room	Note 1
OB-FZ-22A	Upper CSR/Mech. Equip. 63'	Criterion 4
OB-FZ-22B	North Cable Bridge Tunnel 75' El.	Criterion 4
OB-FZ-22C	South Cable Bridge Tunnel 75' El.	Criterion 4
OB-FZ-22D	Mech. Equipment Room 74' El.	Criterion 4

Note 1: Automatic detection and suppression and continuously manned.

OB-FA-6 480V Switchgear Rooms — **Criteria 2 and 3**

OB-FZ-6A	A 480V Switchgear Room 23' El.	Criteria 2 and 6
OB-FZ-6B	B 480V Switchgear Room - 23' El.	Criteria 2 and 6

OB-FA-8 A/B Battery Room, MG Set Room — **Criteria 1, 2 and 3**

OB-FZ-8A	MG Set Room - 23' El.	Criterion 6
OB-FZ-8B	Mechanical Equipment Room	Criterion 6
OB-FZ-8C	A/B Batt Rm, Tunnel, Elec Tray Rm	Criteria 2, 3 and 6

OB-FA-10 Monitor and Change Areas **Criteria 2 and 3**

OB-FZ-10A	Monitor and Change Areas 46' El.	Criteria 4, 5 and 6
OB-FZ-10B	Chem Lab, PASS Rm, Inst Shop	Criteria 3 and 6

RB-FA-1 Reactor Building **Criteria 2 and 3**

RB-FZ-1A	119' El. Refuel Floor	Criteria 5 and 6
RB-FZ-1B	95 Foot Elevation	Criteria 4 and 5
RB-FZ-1C	75 Foot Elevation	Criteria 4 and 5
RB-FZ-1D	51 Foot Elevation	Criteria 4 and 5
RB-FZ-1E	23 Foot Elevation	Criteria 4 and 5
RB-FZ-1F	-19 Foot Elevation	Criteria 4 and 5
RB-FZ-1G	Shutdown Cooling Area	Criteria 4 and 5
RB-FZ-1H	Trunnion Room	Criterion 5

TB-FA-11 Turbine Building **Criterion 2**

TB-FZ-11A Turbine Operating Floor - 46' El. Criteria 4 and 6

Adjacent portions of fire zones TB-FZ-11B, E and G have automatic sprinkler systems to protect the boundary area. Automatic suppression systems are provided for the lube oil and hydrogen combustibles located in this area

TB-FZ-11B Turbine Lube Oil Area Criteria 4 and 6

Adjacent portions of fire zone TB-FZ-11E have an automatic sprinkler system to protect the boundary area. Also, automatic suppression systems are provided for the lube oil handling equipment and storage tanks.

TB-FZ-11C Swgr Room, West End of TB Mezzanine Criteria 4

Adjacent portions of fire zones TB-FZ-11E and G have automatic sprinkler systems.

TB-FZ-11D Basement Floor South End Criterion 6

Adjacent portions of fire zones TB-FZ-11E and G have automatic sprinkler systems. This area is protected by an automatic closed head area wide sprinkler system, in addition to an automatic sprinkler system to protect the Hydrogen seal oil unit.

TB-FZ-11E Condenser Bay Criteria 4, 5 and 6

Adjacent portions of fire zones TB-FZ-11A, B, D and G have automatic sprinkler systems. This area is protected by an automatic closed head area wide sprinkler system, except for the heater bay area. Criterion 5 - propagation to TB-FZ-11F, G and H.

TB-FZ-11F Feedwater Pump Area - 3' 6" Elevation Criteria 4 and 5

Adjacent portions of fire zone TB-FZ-11E and oil hazards in TB-FZ-11B have automatic sprinkler systems.

TB-FZ-11G SW Mezzanine Area/Machine Shop Criteria 4 and 6

A closed head sprinkler system protects the machine shop, office area and the area under the turbine.

TB-FZ-11H Condensate Demineralizer Area Criterion 4

5. Review areas with high combustible loading and suppression features for dominant combustibles:

CW-FA-14	500 gallons of oil and 2500 lb of plastic
MT-FA-12	Main transformers
OB-FZ-6A	Cable insulation 225 gallons of diesel fuel
OB-FZ-6B	Cable insulation 225 gallons of diesel fuel
TB-FZ-3A	32,000 feet of cable insulation
TB-FZ-11A	Lube oil
TB-FZ-11B	Lube oil

6. Review and walkdowns assumptions made for fire zones that were removed from further consideration during the revised evaluation. Walkdown area features.

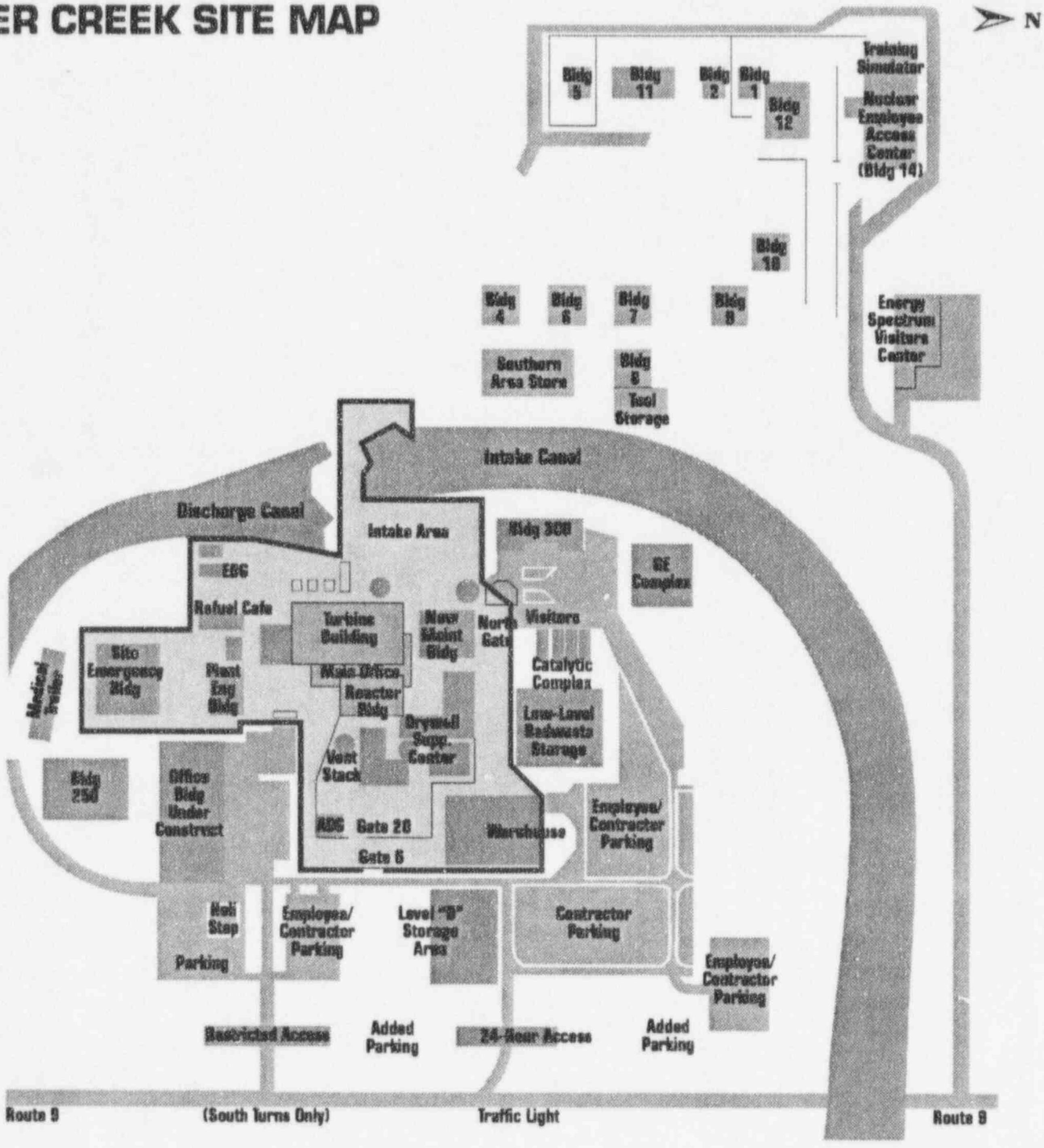
RB-FZ-1F	Verify separation between torus area and corner rooms precludes full area fire. Verify fixed sprays to protect grouped cables. Review area for intervening combustibles.
OB-FZ-10A	Recovery of switchgear room ventilation.
OB-FZ-22A	Recovery of switchgear room ventilation.
TB-FZ-11B	Recovery from loss of 480 VAC switchgear control.
TB-FZ-11C	Verify DC power protected by conduit. Initiator LOSP used.
TB-FZ-11E	Recovery from loss of 480 VAC switchgear control.
TB-FZ-11F	Modeled as LOFW. Verify AC cables contained in conduit.
TB-FZ-11H	Division 1 cables separated from general area by concrete wall.
MT-FA-12	Proximity of transformers to TB. Modeled as LOSP.

CW-FZ-14 Modeled recovery of 1 ESW pump.
A and B in trays D9 and D11, conduits 14-33 and 14-34.
C and D in trays D6 and D8, conduits 14-31 and 14-32.

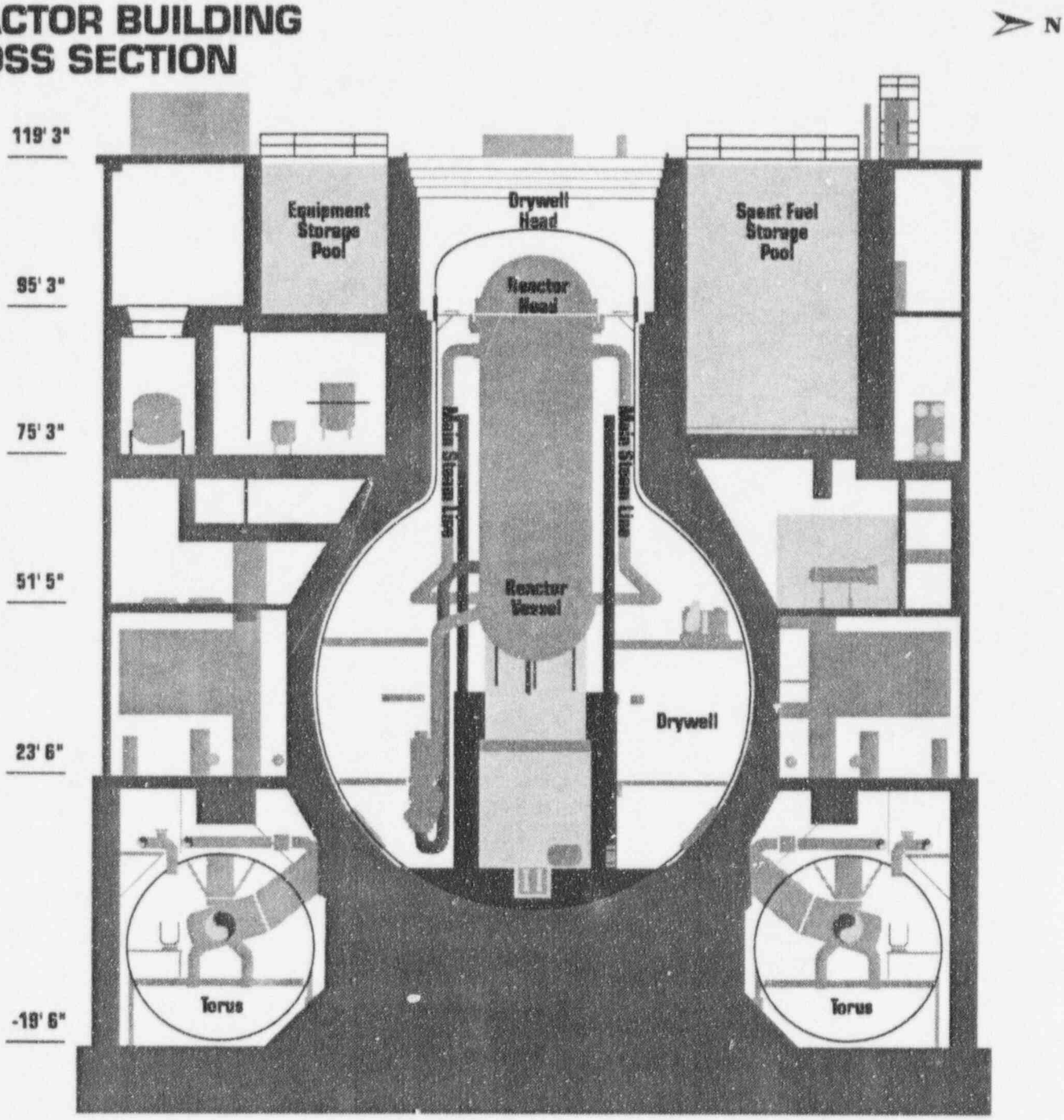
7. Review and walkdown assumptions made during detailed evaluation (Section 4.6)

RB-FZ-1D	Review paths from NW corner area. Verify fixed automatic water sprays protect grouped cables
RB-FZ-1E	Verify EMRV actuation cables protected by conduit. Review fire growth from north or south area.
OB-FZ-4	Verify isolation condenser and ADS/EMRV cables contained in conduit.
OB-FZ-5	Core spray 1 failure for case 1. LOFW for case 2. Fire suppression before engulfing area (no CR evacuation).
OB-FZ-6A	Equipment (USS 1A2) in first alley
OB-FZ-8C	Case 3 damages no risk-significant equipment
TB-FZ-26	Battery failure due to fire

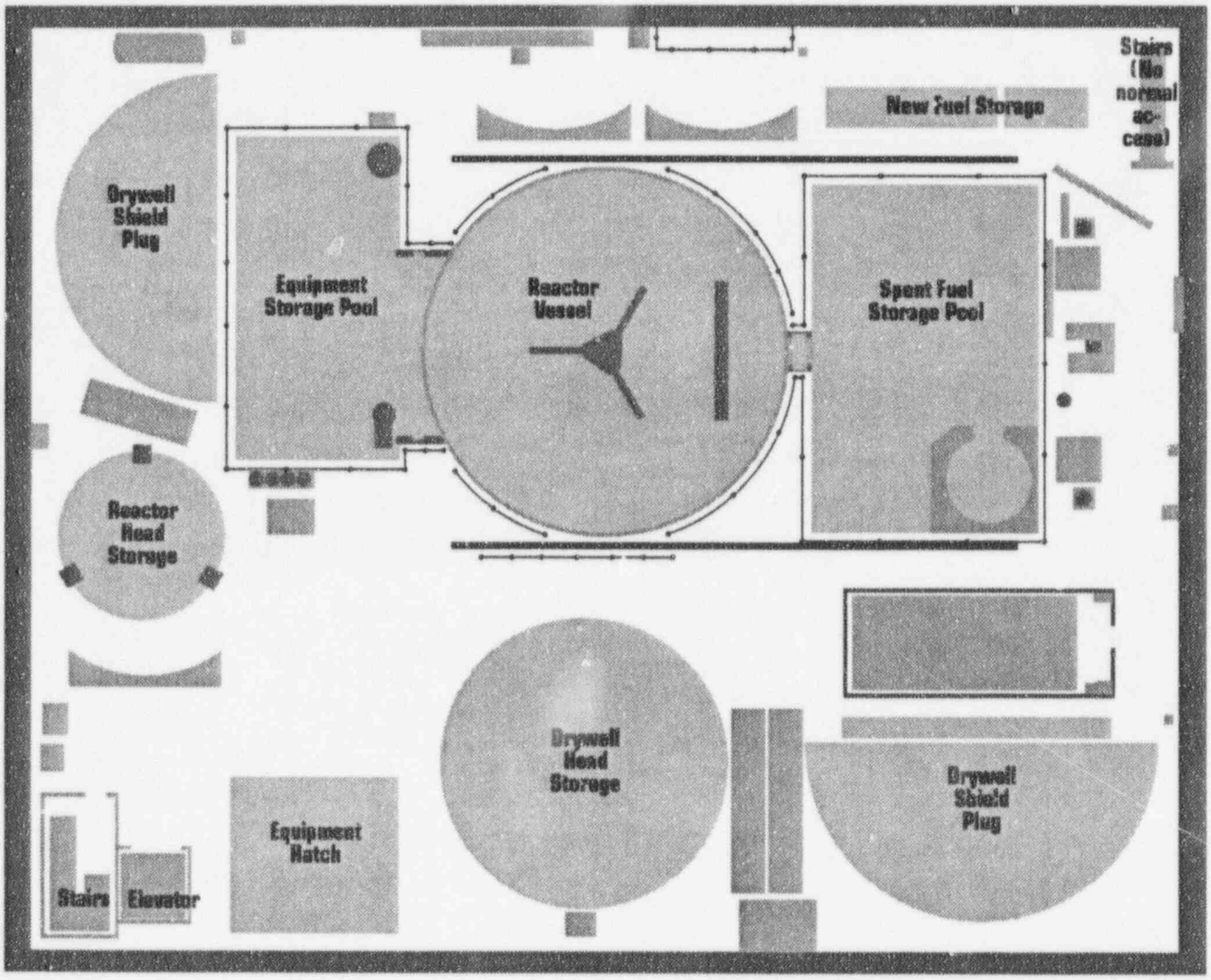
OYSTER CREEK SITE MAP



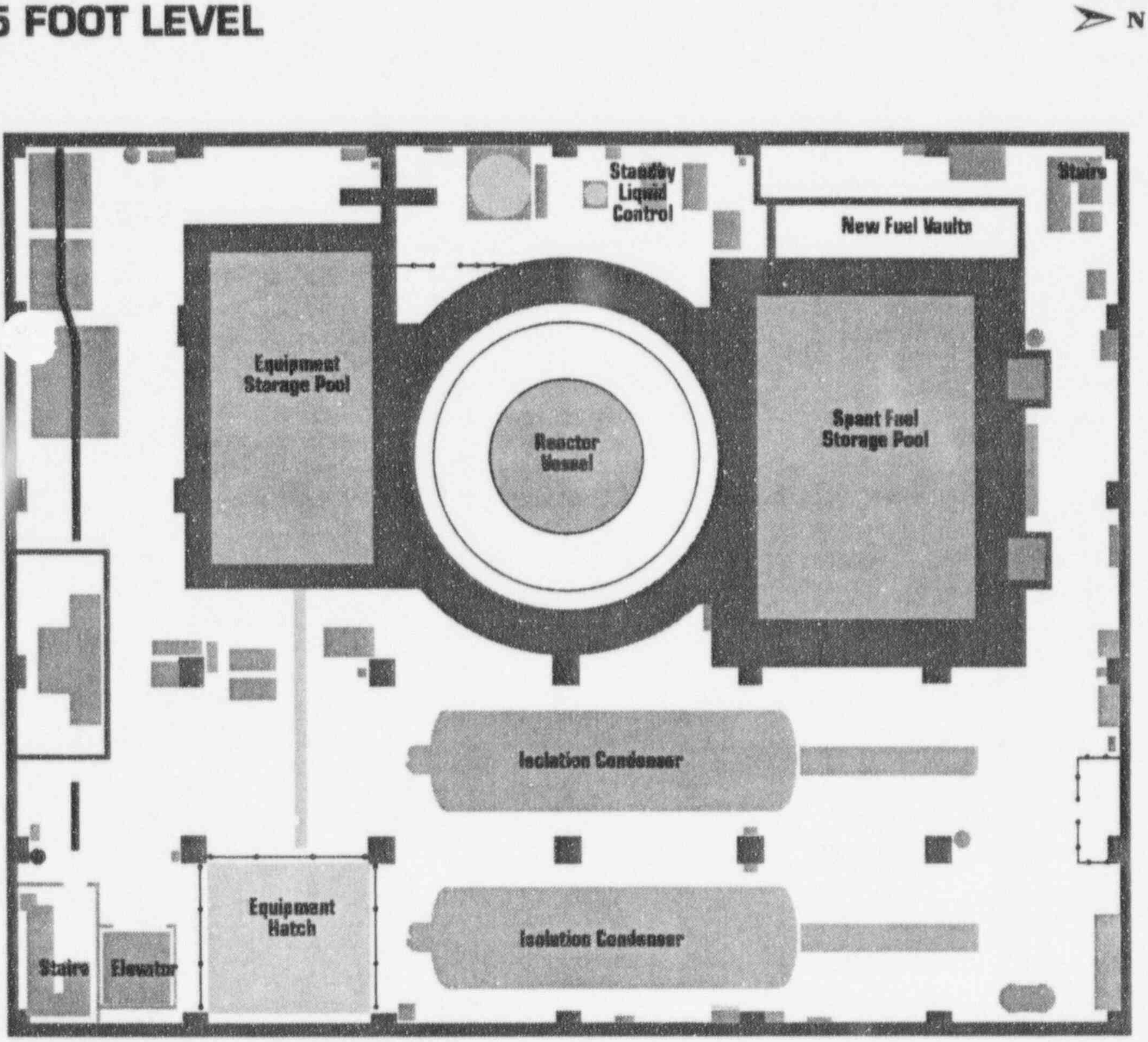
REACTOR BUILDING CROSS SECTION



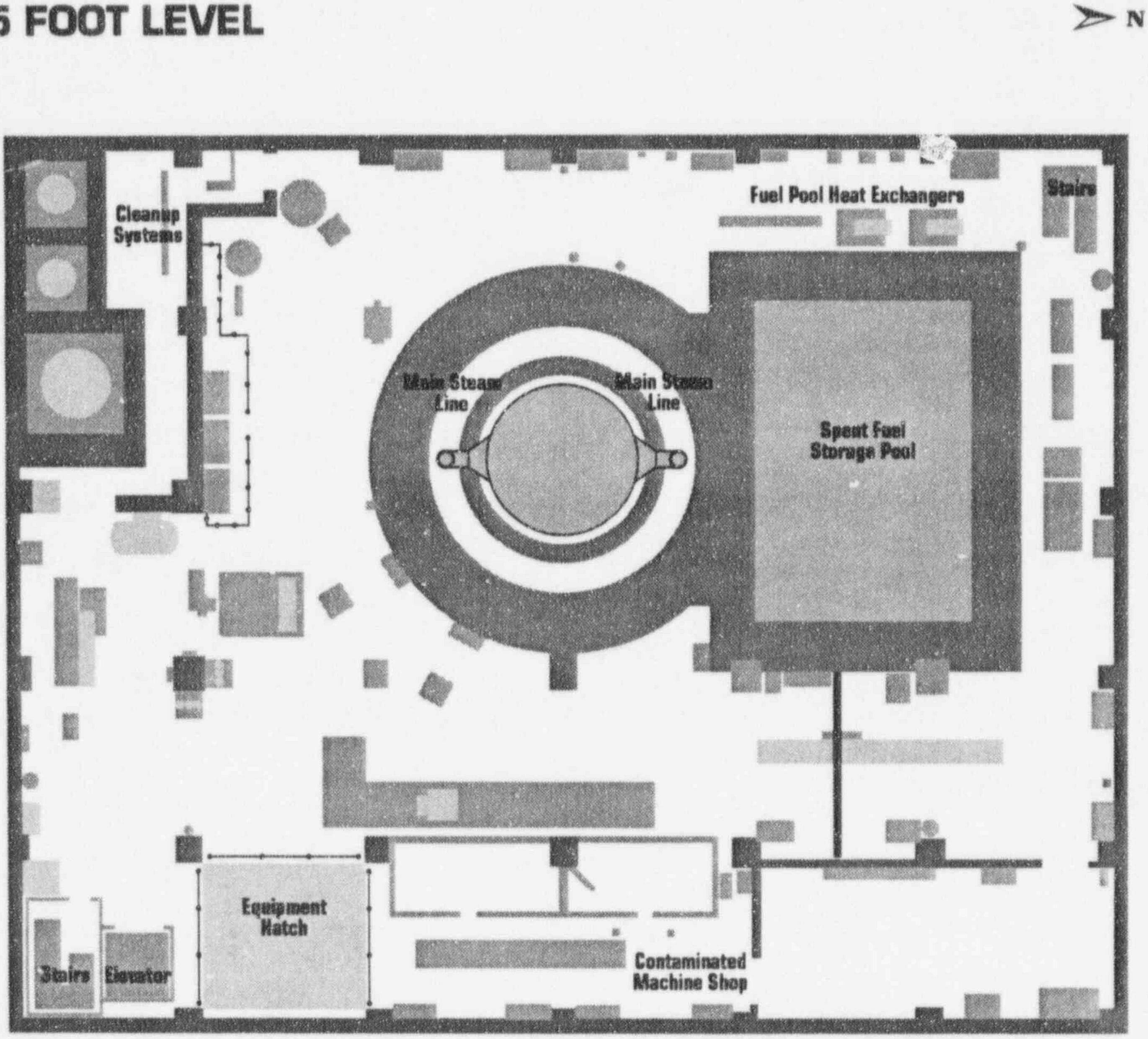
119 FOOT LEVEL



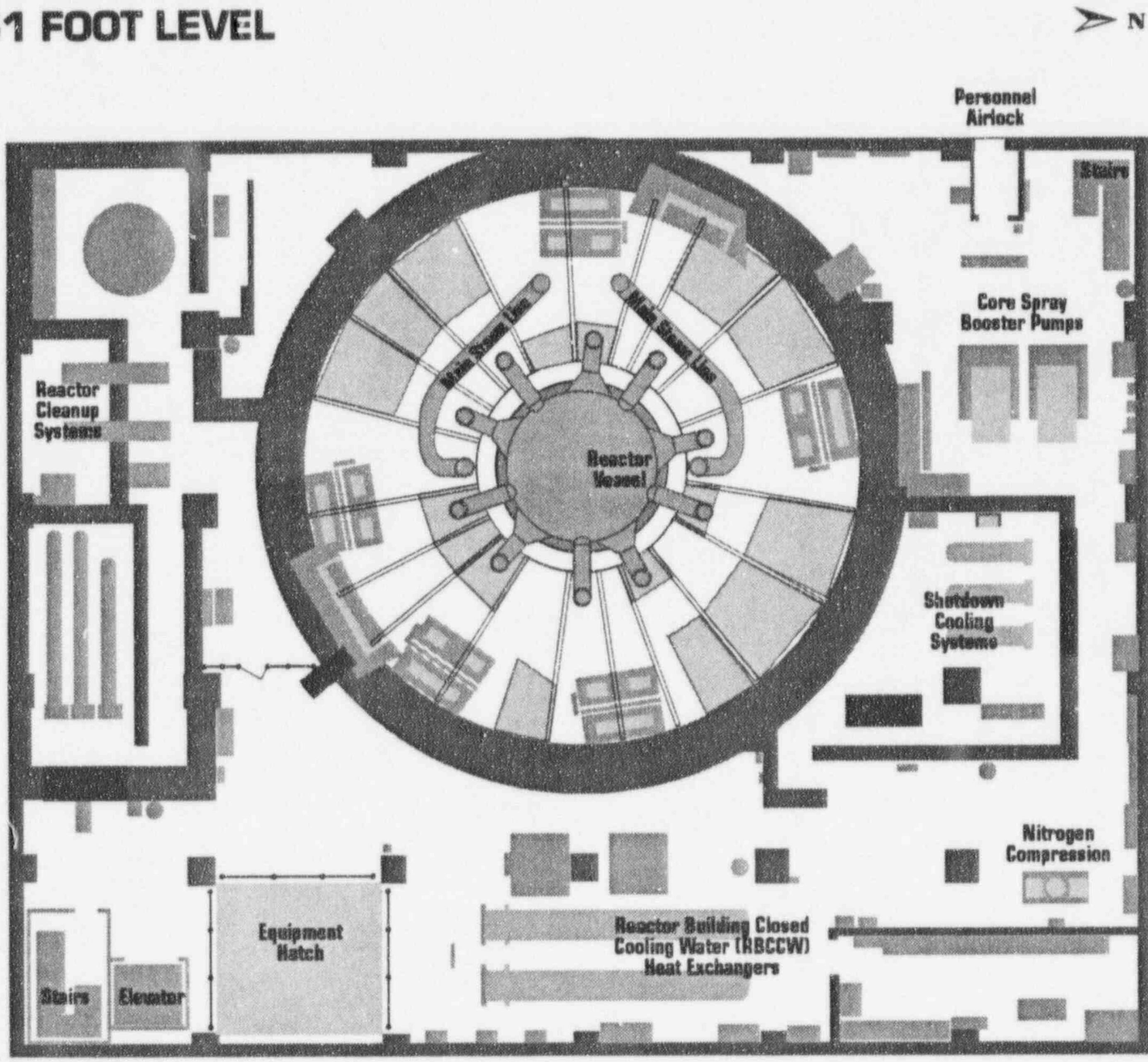
95 FOOT LEVEL



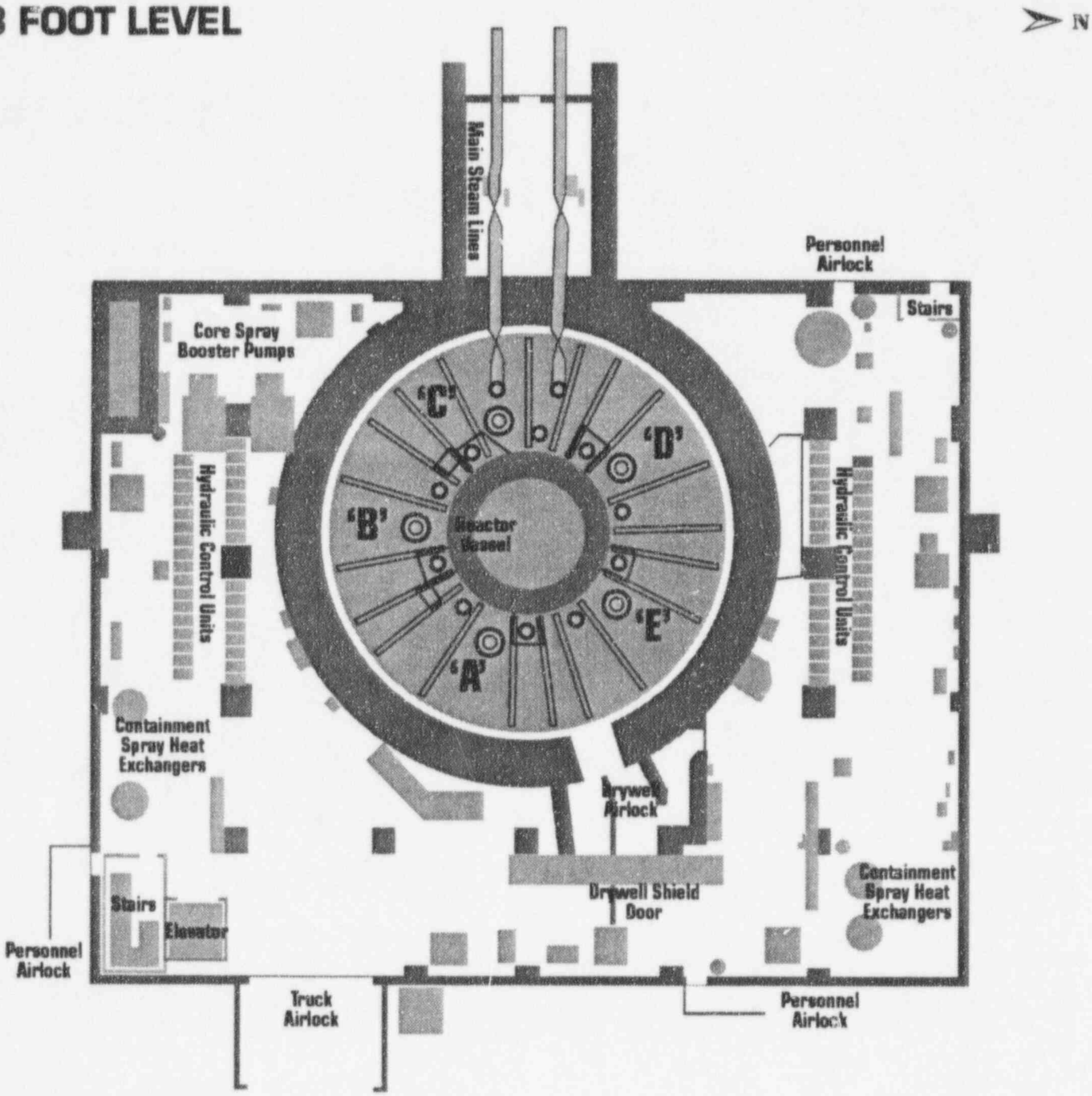
75 FOOT LEVEL



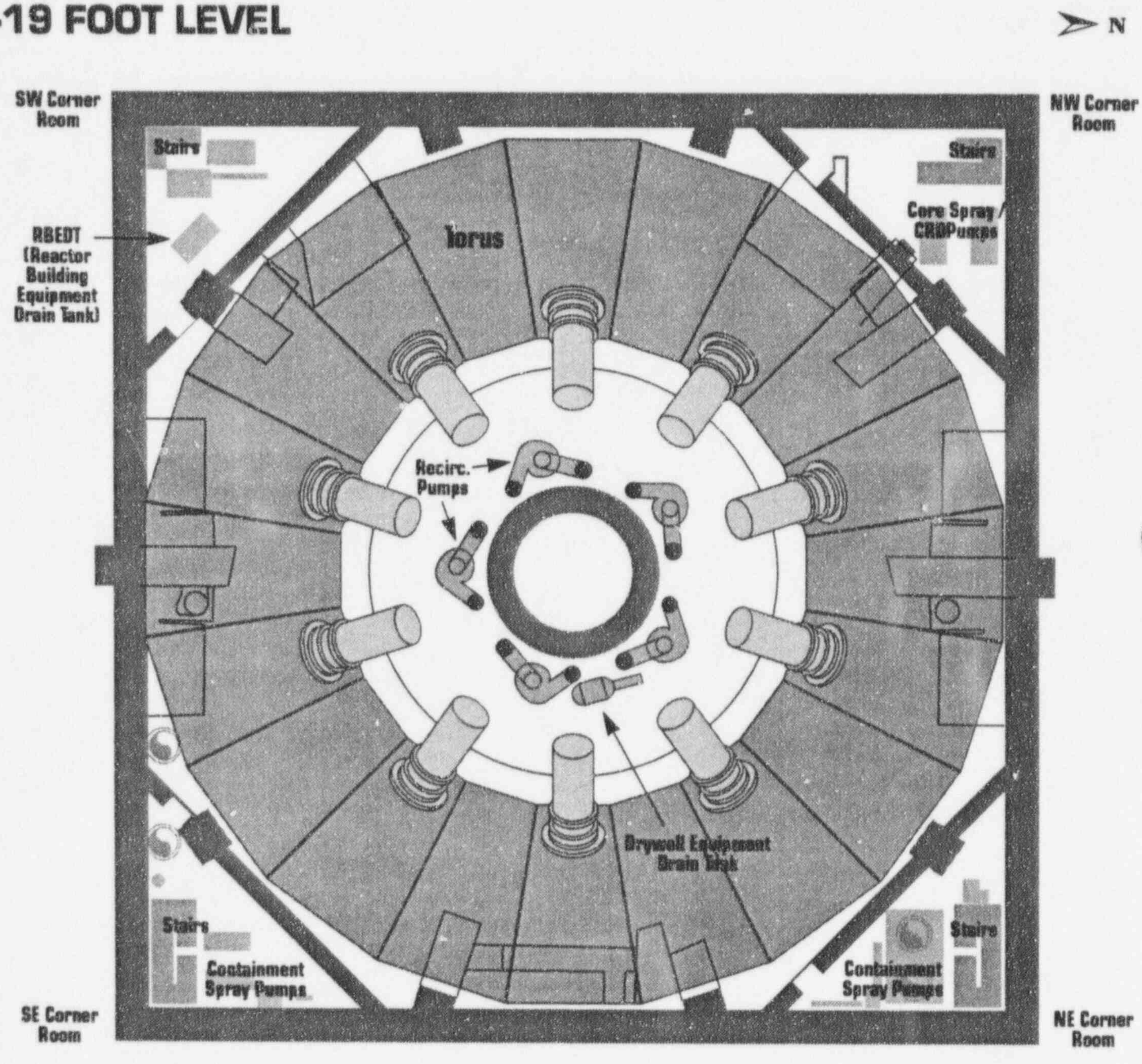
51 FOOT LEVEL



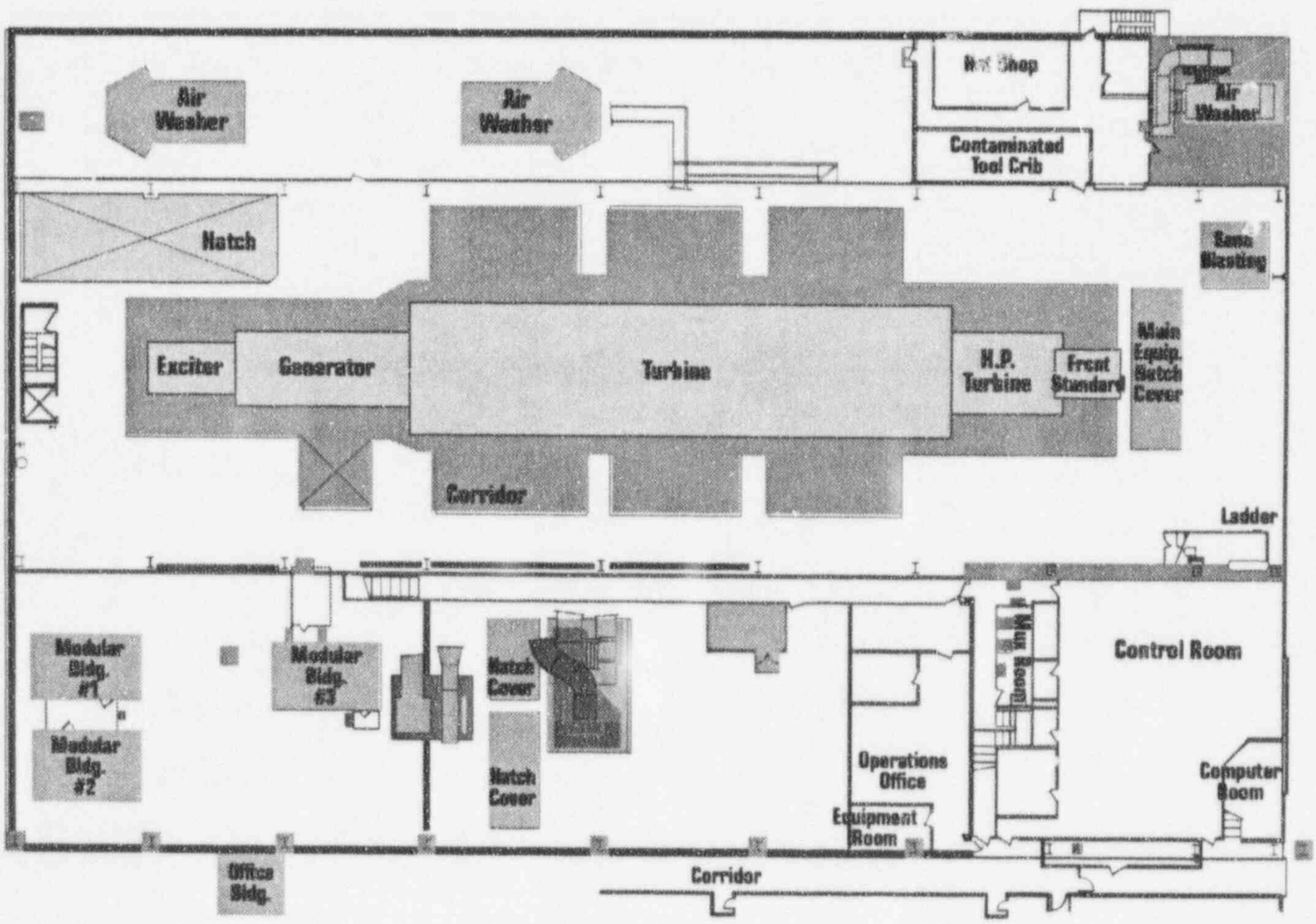
23 FOOT LEVEL



-19 FOOT LEVEL

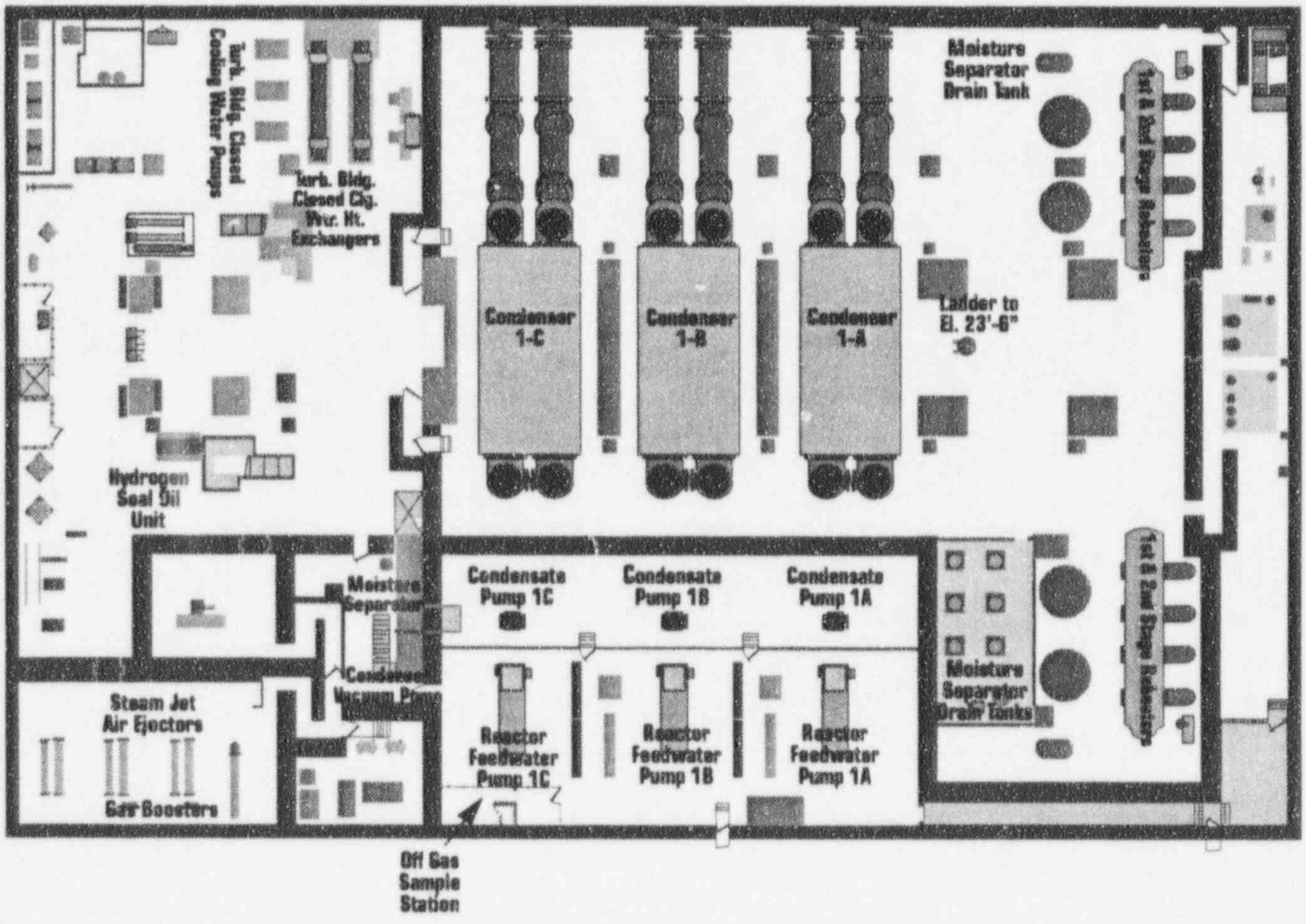


TURBINE BUILDING OPERATING FLOOR



TURBINE BUILDING
23-FOOT ELEVATION

N



4.3 Fire Growth and Propagation

This report section presents the fire barrier effectiveness review task (Task 8) of the Oyster Creek Fire Individual Plant Examination. In this task, fire areas and zones are evaluated for the potential for fire barrier breach, which could impact additional fire areas containing components that are important to risk. This evaluation employs the EPRI Fire induced Vulnerability Evaluation (FIVE) guidance on the qualitative screening of fire barriers

The Review of Fire Barrier Effectiveness, Task 8, comprises a separate path along the general flow of the fire analysis process. In this task, combustible loading, fire barriers and suppression systems are reviewed to determine the potential for fires that could involve more than one fire zone or fire area. These potential "multiple area fires" could then have impacts on plant equipment that are much more severe than fires involving only single plant areas.

- Input to Task 8 is taken from the Fire Hazards Analysis Report and guidance provided by the EPRI FIVE documentation. Combustible loadings for plant areas were obtained from the FHAR documentation.
- The output from Task 8 consists of the list of fire areas and fire zones with boundaries that must be evaluated for potential multiple area fires. In other words, this task provides an independent verification of fire barrier integrity, on a plant-wide basis.

Task 8 consists of the process used to evaluate fire barrier effectiveness at the Oyster Creek plant. This task is performed using a 3 step process:

Step 1. **Review the guidance provided in the EPRI FIVE documentation for implementation**. This guidance is based on 6 criteria, which are listed below. The application of these criteria requires the analyst to obtain the combustible loading for each plant area. These values were taken from the FHAR and are shown in Table 4.3-1.

Step 2. **Review plant layout to determine adjacent fire zones, within the various fire areas.** These fire areas and their associated fire zones are listed in Table 4.6-2.

Step 3. **Perform a qualitative review of fire barrier effectiveness for all plant fire zones**. This review is based on the documentation provided in the FHAR and consists of detailed application of the EPRI FIVE criteria.

This section describes the process used to verify fire barrier effectiveness, on a qualitative basis (Task 8. The methodology used to qualitatively screen the effectiveness of fire barriers between fire areas is based on the guidance provided in the EPRI FIVE report (Section 5.3.6). This review is performed to ensure that fires involving more than one plant area, or "multiple area fires," are not a concern. For reference, the guidance for this review is quoted directly (see page 5-8 of the EPRI FIVE report):

"A common boundary can be evaluated and screened from further consideration based on the following criteria:

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 not adverse effect 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 (See Note 1).*

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 (See Note 1).*

5. *Boundaries where both the exposing and exposed compartment have a very low combustible loading <20,000 Btu per sq. ft. on the basis that a significant fire cannot develop in the area (See Note 1).*

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 (See Note 1).*

NOTE 1:

The utilization of 3, 4, 5 or 6 as the boundary screening basis also require walkdown data indicating that there is not a concentration of combustibles near the boundary or combustible pathway between compartments."

The walkdown process is described in Section 4.2.

Based on the fire boundary evaluation criteria used in the Fire Hazards Analysis Report and the guidance provided in the ERPI FIVE documentation (quoted above), existing evaluations of fire boundary effectiveness between fire areas with rated fire barriers can be judged as adequate for the screening of fire propagation between fire areas at the Oyster Creek plant. This implicitly assumes 100% effectiveness for rated fire barriers. This assumption is judged to be reasonable, given the passive nature of these boundaries and the administrative controls that apply to their maintenance and surveillance. Therefore, all Oyster Creek *Fire Areas* are qualitatively screened using the EPRI FIVE criteria. However, since the barriers between *Fire Zones* may not consist of rated barriers, a qualitative review of these fire zones, using the EPRI FIVE criteria is performed and documented below.

For this evaluation, all fire areas and zones were reviewed, including those that were initially screened from consideration for single area fires in Sections 4.1 and 4.4. Of the areas, only the new warehouse and the torus/drywell are adjacent to other plant areas. The new warehouse boundary with the Turbine Building is located down a hallway approximately 50 feet long. This boundary is fire rated at 3 hours. As described in Section 4.1, the drywell and torus are inerted with Nitrogen during power operation. Therefore, all the fire areas removed from further consideration in Section 4.1 can be qualitatively screened from consideration as potential multiple area fires. These areas are:

RB-FA-2	Drywell and Torus
FW-FA-18	Diesel Driven (Pond) Fire Pump Area
PH-FA-25	Redundant Fire Pump Area
NW-FA-23	New Warehouse
MB-FA-24	Maintenance Building
EB-FA-28	Site Emergency Building

The EPRI FIVE guidance for fire propagation between fire areas is also applied to evaluate fire propagation between fire zones. Since this guidance (Criteria 3, 4, 5 and 6) is based on combustible loading, this information is presented in Table 4.3-1 for the fire areas and zones under consideration (the remaining fire areas and zones). Of the areas listed in Table 4.3-1, the following (indicated on Table 4.3-1 by shading) are either not adjacent to any other fire areas or zones or have fire rated boundaries between areas (Criteria 1 or Criteria 2):

MT-FA-12	Main Transformer and Yard
AB-FA-13	Auxiliary Boiler Building
CW-FA-14	Circulating Water/Intake Area
DG-FA-15	Diesel Generator 1
FS-FA-16	Diesel Generator Fuel Storage Tank (Underground)
DG-FA-17	Diesel Generator 2
OR-FA-19	Old Radwaste Building
NR-FA-20	New Radwaste Building
OG-FA-21	Offgas Building
LL-FA-29	Low Level Radwaste Storage Facility
OB-FA-9	Office Building
TB-FA-3A	4160 VAC Switchgear 1C Vault (TB Mezz 23' El.)
TB-FA-3B	4160 VAC Switchgear 1D Vault (TB Mezz 23' El.)
TB-FA-26	Battery Room South of 4160 VAC Switchgear

Table 4.3-1 Fire Areas Analyzed by Fire Zone in the FHAR						
Fire Area	Fire Area Description	Fire Zones	Fire Zone Description	Total Combustible Loading[1] (BTU/ft^2)		
				Over 80,000	Over 20,000	Less than 20,000
MT-FA-12	Main Transformer and Yard			116,445,185		
AB-FA-13	Auxiliary Boiler Building				39,409	
CW-FA-14	Circ. Water/Intake Area			131,513,409		
DG-FA-15	Diesel Generator 1				51,669	
FS-FA-16	Diesel Generator Fuel Oil Storage Tank (underground)			8,195,644		
DG-FA-17	Diesel Generator 2				49,122	
OR-FA-19	Old Radwaste Building					10,619
NR-FA-20	New Radwaste Building				33,873	
OG-FA-21	Offgas Building				159,732	
LL-FA-29	Low Level Radwaste Storage Facility			No combustible loading produced due to separation from other buildings and lack of safety significance.		
OB-FA-5	Control Room Complex	OB-FZ-4 OB-FZ-5 OB-FZ-22A OB-FZ-22B OB-FZ-22C OB-FZ-22D	Cable Spreading Room (36' El) Control Room (46'-6" El) Upper Cable Spreading Room North Cable Bridge Tunnel (74' El) South Cable Bridge Tunnel (74'El) Mechanical Equipment Room		50,089 42,000	 7,517 3,127 3,014 0
OB-FA-6	480V Switchgear Room	OB-FZ-6A OB-FZ-6B	"A" 480V Switchgear Room "B" 480V Switchgear Room	176,190 91,719		
OB-FA-8	Battery and Equipment Areas	OB-FZ-8A OB-FZ-8B OB-FZ-8C	MG Set Room (23' El) Mechanical Equipment Room A and B Battery Room (35' El)		47,593[3] 28,924	
OB-FA-9	Office Building					2,369

Table 4.3-1 Fire Areas Analyzed by Fire Zone in the FHAR						
Fire Area	Fire Area Description	Fire Zones	Fire Zone Description	Total Combustible Loading[1] (BTU/ft[2])		
				Over 80,000	Over 20,000	Less than 20,000
OB-FA-10	Office Building (Service Area)	OB-FZ-10A	Monitor and Change Areas (46' El)			14,359
		OB-FZ-10B	Chem Lab, PASS Room, Instr. Shop		28,764	
RB-FA-1	Reactor Building	RB-FZ-1A	119 Foot Elevation (Refuel Floor)			1,284
		RB-FZ-1B	95 Foot Elevation			12,654
		RB-FZ-1C	75 Foot Elevation			17,750
		RB-FZ-1D	51 Foot Elevation			19,398
		RB-FZ-1E	23 Foot Elevation		21,898	
		RB-FZ-1F	- 19 Foot Elevation			4,217
		RB-FZ-1G	Shutdown Cooling Area			5,129
		RB-FZ-1H	Trunnion Room			2,078
TB-FA-3A	4160VAC Switchgear C			106,477		
TB-FA-3B	4160VAC Switchgear D				34,598	
TB-FA-11	Turbine Building	TB-FZ-11A	Turbine Operating Floor	110,280		
		TB-FZ-11B	Turbine Lube Oil Area	768,115		
		TB-FZ-11C	Switchgear Room (West Mezzanine)			3,328
		TB-FZ-11D	Basement Area (South End)		39,789	
		TB-FZ-11E	Condenser Bay			6,536
		TB-FZ-11F	Feedwater Pump Area			747
		TB-FZ-11G	Southwest Mezzanine			518
		TB-FZ-11H	Condensate Demineralizer Area			107
TB-FZ-26	Battery Room "C"					5,000

Note 1: The combu tible loadings are taken from Section 8.0 of the Fire Hazards Analysis Report.
Note 2: Fire areas not divided into fire zones have rated fire barriers are screened (qualitatively) from further consideration and appear shaded on this table.
Note 3: Fire zones OB-FZ-8A and OB-FZ-8B are analyzed together. Total combustible loading is given.

Within the FHAR, the following Office Building, Reactor Building and Turbine Building fire areas are subdivided into smaller fire zones (as illustrated in Table 4.3-1 under the "Fire Zone" and "Fire Zone Description" columns). In this analysis, each of these plant areas is then treated as an independent fire area, as though rated fire barriers existed at all boundaries. For fire zones, mitigating or protective features are used in lieu of rated fire barriers to prevent the spread of fires to other adjacent areas.

It should be noted that Office Building fire area **OB-FA-9** is not separated into fire zones for analysis in the FHAR. Since all barriers to adjacent areas are fire rated at least 1 hour and due to the low (2,369 BTU/ft^2) combustible loading for this area, fire growth beyond this area can be screened from further consideration, based on EPRI FIVE criterion 3.

OB-FA-5 (Control Room Complex) is separated from adjacent fire areas (reactor building, turbine building and other office building fire areas) by 3 hour fire rated barriers. Therefore, fire growth beyond this area can be screened from further consideration, based on EPRI FIVE criteria 2 and 3. Within OB-FA-5, the FHAR has analyzed this fire area as 6 distinct fire zones, with fire boundaries and protective features to prevent the spread of fires from one zone to another. Each of these zones is described individually below.

OB-FZ-4 (Cable Spreading Room - 36 Foot Elevation). The floor and south and west walls are fire rated at 3 hours, with the north and east walls fire rated at 2 hours. Combustibles in this zone are protected by an open head water spray system, which is actuated by cross zoned ionization detectors (Criterion 6). Also, there is a hollow block ventilation shaft that communicates with OB-FZ-22A, which is isolated by fire dampers from the control room. Finally, the ceiling (control room floor) is non-fire rated. Due to the presence of automatic detection and suppression, fire growth beyond this area is screened, based on Criterion 6.

OB-FZ-5 (Control Room). The north, east and west walls, as well as that part of the floor adjacent to the Turbine Building, are fire rated at 3 hours. The south wall is fire rated at 2 hours Combustibles in this zone are protected by automatic Halon 1301 suppression for panels 1F-4F (front), 1R-6R (rear), 5F-9F, 7R-11R and 10F and 11F, which are actuated by ionization smoke detectors (Criterion 6). Also, this area is continuously occupied by plant operating staff. The portion of the floor adjacent to OB-FZ-4 is not fire rated. Also, the ceiling, which is adjacent to fire zone OB-FZ-22A, is not fire rated. Due to the presence of automatic detection and suppression and the continuous presence of watchstanding personnel in this area, fire growth beyond this area is screened from further consideration.

OB-FZ-22A (Upper Cable Spreading Room/Mechanical Equipment Room - 63' 9" Elevation). The west wall, adjacent to the Turbine Building, is fire rated at 3 hours, as is the floor adjacent to OB-FZ-10A. Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by an automatic area wide closed head wet pipe sprinkler system. Due to the low (7,517 BTU/ft^2) combustible loading and the presence of automatic detection (Criterion 4) and automatic suppression, fire growth beyond this zone is screened from further consideration.

OB-FZ-22B (North Cable Bridge Tunnel - 75' 1" Elevation). The east wall, adjacent to the Reactor Building, is fire rated at 3 hours. Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by a manually actuated area wide closed head preaction sprinkler system. Due to the low (3,127 BTU/ft^2) combustible loading and the presence of automatic detection (Criterion 4), fire growth beyond this zone is screened from further consideration.

OB-FZ-22C (South Cable Bridge Tunnel - 75' 1" Elevation). The east wall, adjacent to the Reactor Building, is fire rated at 3 hours. Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by a manually actuated area wide closed head preaction sprinkler system. Due to the low (3,014 BTU/ft^2) combustible loading and the presence of automatic detection (Criterion 4), fire growth beyond this zone is screened from further consideration.

OB-FZ-22D (Mechanical Equipment Room - 74' Elevation). Fire zone OB-FZ-22A is the only adjacent area (located below). Due to the extremely low combustible loading in each of these areas (7,517 BTU/ft^2 for fire zone OB-FZ-22A and negligible for OB-FZ-22D), fire growth beyond this zone is screened from further consideration (Criterion 4).

OB-FA-6 (480V Switchgear Rooms) have 3 hour fire rated barriers on all boundaries, except for the north wall of OB-FZ-6B, which is fire rated at 1.5 hours. Fire growth beyond this area is screened from further consideration, based on EPRI FIVE criteria 2 and 3. While the combustible loading for the area facing the north wall is over the 80,000 BTU/ft^2 specified in criterion 3, the fire barrier rating is 50% greater than the required 1 hour rating.

OB-FZ-6A (A 480V Switchgear Room - 23 Foot Elevation). All boundaries for this zone are fire rated at 3 hours, with the exception of the north wall, which is rated at 1 hour. Combustibles in this zone are protected by automatic Halon suppression, which is actuated by a cross zoned area wide fire detection system. While this area has a high combustible loading, primarily due to cable insulation (176,190 BTU/ft^2), fire growth is screened from further consideration, based on Criteria 2 and 6.

OB-FZ-6B (B 480V Switchgear Room - 23 Foot Elevation). All boundaries for this zone are fire rated at 3 hours, with the exception of the north wall, which is rated at 1.5 hours and the south wall, which is rated at 1 hour. Combustibles in this zone are protected by automatic Halon suppression, which is actuated by a cross zoned area wide fire detection system. While this area has a high (91,719 BTU/ft^2) combustible loading, primarily due to cable insulation, fire propagation beyond this area is screened from further consideration, based on Criteria 2 and 6.

OB-FA-8 (A and B Battery Room and MG Set Room) has all boundaries fire rated at 3 hours, except the exterior south wall of OB-FZ-8C. Therefore, fire growth beyond this area is screened, based on Criteria 1, 2 and 3.

OB-FZ-8A (MG Set Room - 23 Foot Elevation) and OB-FZ-8B. All boundaries for OB-FZ-8A are fire rated at 2 or 3 hours, with the exception of the ceiling, adjacent to fire zones OB-FZ-8B and OB-FZ-8C and the exterior portion of the south wall, which is not adjacent to any other plant areas. Combustibles in this zone are protected by an automatic closed head sprinkler system, which provides coverage for all areas of the zone, except for the south end, near the motor control centers. An alarm actuates in the Control Room on flow from the sprinkler system. For purposes of analysis, OB-FZ-8A and OB-FZ-8B are treated as one area. Fire growth beyond these fire zones is screened from further consideration, based on Criterion 6.

OB-FZ-8C (A and B Battery Room, Tunnel and Electrical Tray Room - 35 Foot Elevation). All boundaries for this zone are fire rated at 1 (OB-FA-9) or 3 hours, with the exception of the boundaries to OB-FZ-8A and OB-FZ-8B. Due to the low level of combustible loading for this zone (28,924 BTU/ft^2), a 1 hour barrier to fire area OB-FA-9 is adequate (Criterion 3) to prevent a fire from spreading to this area. Combustibles in this zone are protected by a total flooding Halon 1301 extinguishing system, which provides coverage for all areas of the zone, actuating automatically on actuation of two or more ionization detectors in the area. These detectors are also annunciated in the Control Room. This zone is screened from further consideration, based on Criteria 2, 3 and 6.

OB-FA-10 (Monitor and Change Areas) has boundaries that are fire rated at 3 hours, except adjacent to OB-FA-9, where the barrier is fire rated at 1 hour. Due to the low fire loading of both areas, fire growth beyond OB-FA-9 is screened, based on Criteria 2 and 3.

OB-FZ-10A (Monitor and Change Areas - 46 Foot Elevation). The east and west walls are fire rated at 3 hours, as are the ceiling areas adjacent to OB-FZ-22A and OB-FZ-6B. The portion of the west wall adjacent to the turbine building is fire rated at 2 hours. The south wall adjacent to OB-FA-9 is fire rated at 1 hour. Finally, the floor adjoins fire zone OB-FZ-10B and is not fire rated. Due to the low level of combustible loading for this zone (14,359 BTU/ft^2), a 1 hour barrier to fire area OB-FA-9 is adequate (Criterion 3) to prevent a fire from spreading to this area. Combustibles in this zone are protected by closed head sprinkler systems installed above and below ceiling level. These systems are automatically actuated by particles of combustion smoke detectors. These detectors are also annunciated in the Control Room. This system is judged adequate to prevent the spread of a fire out of this area (Criteria 4, 5 and 6) across this non-fire rated boundary.

OB-FZ-10B (Chemical Lab, PASS Room, Instrument Shop - 35 Foot Elevation). All boundaries for this zone are fire rated at 1 or more hours, with the exception of the boundary to OB-FZ-10A. Due to the low level of combustible loading for this zone (28,764 BTU/ft^2), a 1 hour barrier is adequate (Criterion 3) to prevent a fire from spreading to this area. Area wide detection is provided in this zone, using ionization detectors. These detectors are also annunciated in the Control Room. This system is judged adequate to prevent the spread of a fire out of this area (Criterion 6) across the non-fire rated boundaries.

RB-FA-1 (Reactor Building) is separated from adjacent fire areas by reinforced concrete walls that are rated to provide over 3 hour fire resistance rating. Fire growth beyond this area is screened from further consideration, based on Criteria 2 and 3.

RB-FZ-1A (119 Foot Elevation - Refuel Floor) has a low (1,284 BTU/ft^2) combustible loading. Automatic wet pipe sprinkler protection is provided at ceiling level to provide fire suppression within this area. Due to the presence of automatic suppression and the low combustible loading of adjacent fire zone RB-FZ-1B, fire growth beyond this area is screened from further consideration, based on Criteria 5 and 6.

RB-FZ-1B (95 Foot Elevation) has a low (12,654 BTU/ft^2) combustible loading. Automatic fire detection is provided by ionization detectors. Due to the presence of automatic detection and the low combustible loading of adjacent fire zones RB-FZ-1A and RB-FZ-1C, fire growth beyond this area is screened, based on Criteria 4 and 5.

RB-FZ-1C (75 Foot Elevation) has a low (17,750 BTU/ft^2) combustible loading. Automatic fire detection is provided by ionization detectors. Also, a water curtain is installed to prevent fire growth to lower Reactor Building elevations. Due to the presence of automatic detection and the low combustible loading of adjacent fire zones RB-FZ-1B and RB-FZ-1D, fire growth beyond this area is screened, based on Criteria 4 and 5.

RB-FZ-1D (51 Foot Elevation) has a low (19,398 BTU/ft^2) combustible loading. Automatic fire detection is provided by ionization detectors. Also, a water curtain is installed at the ceiling to prevent fire growth to higher Reactor Building elevations. Due to the presence of automatic detection and the low combustible loading of adjacent fire zones RB-FZ-1C, RB-FZ-1E (see below) and RB-FZ-1G, fire growth beyond this area is screened from further consideration, based on Criteria 4 and 5.

RB-FZ-1E (23 Foot Elevation) has a low (21,898 BTU/ft^2) combustible loading. While this is slightly higher than the 20,000 BTU/ft^2 specified in Criteria 4 and 5, it is felt that this low level of combustible loading meets the intent of these criteria. Automatic fire detection for this area is provided by ionization detectors. Also, a water curtain is installed at the ceiling to prevent fire growth to higher Reactor Building elevations. Therefore, due to the presence of automatic detection and the low combustible loading of adjacent fire zones RB-FZ-1D and RB-FZ-1F, fire growth beyond this area is screened from further consideration, based on Criteria 4 and 5.

RB-FZ-1F (-19 Foot Elevation) has a low (4,217 BTU/ft^2) combustible loading. Automatic fire detection for this area is provided by ionization detectors. Due to the presence of automatic detection and the low combustible loading of adjacent fire zones RB-FZ-1E and RB-FZ-1H, fire growth beyond this area is screened, based on Criteria 4 and 5.

RB-FZ-1G (Shutdown Cooling Area) has a low (5,192 BTU/ft^2) combustible loading. Automatic fire detection for this area is provided by particles of combustion detectors. Due to the presence of automatic detection and the low combustible loading of adjacent fire zone RB-FZ-1D, fire growth beyond this area is screened, based on Criteria 4 and 5.

RB-FZ-1H (Trunnion Room) has 3 hour rated fire barriers on all walls, the ceiling and the floor, except for small openings to the torus room (RB-FZ-1F5) to provide drainage for any leakage. This fire zone has a low (2,078 BTU/ft^2) combustible loading. Due to the low level of combustible loading in this fire zone and the torus room, fire growth beyond this area is screened from further consideration, based on Criterion 5.

TB-FA-11 (Turbine Building) has 3 hour rated fire barriers to the office and reactor building and TB-FA-3A and TB-FA-3B. Fire growth beyond this fire area is screened from further consideration, based on Criterion 2.

TB-FZ-11A (Turbine Operating Floor - 46' Elevation). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11B, C, E and G. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zones TB-FZ-11B, E and G have automatic sprinkler systems installed to protect the boundary area. Also, automatic suppression systems are provided for the lube oil and hydrogen combustibles located in this area. Specifically, an automatic closed head sprinkler system is installed over the turbine bearing lift pumps and turbine bearings. A fixed CO_2 system protects the generator bearings and exciter. Since more than 99% (109,684 BTU/ft^2, of 110,280 BTU/ft^2 total) of combustible loading for this fire zone is the result of "oil and lubricating materials," which are protected by automatic suppression systems, and the fire loading for this area is otherwise negligible, fire growth from this area is screened from further consideration, based on Criteria 4 and 6.

TB-FZ-11B (Turbine Lube Oil Storage, Pumping and Purification Area). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11A, E and F. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zone TB-FZ-11E have an automatic sprinkler system installed to protect the boundary area. Also, automatic suppression systems are provided for the lube oil handling equipment and storage tanks located in this area. Specifically, an automatic closed head sprinkler system is installed over the cable trays, with a water spray deluge system for the oil handling equipment and storage tanks themselves. Also, a closed head sprinkler system, which is activated by thermal detectors, is provided for the bearing lift pump. As described for TB-FZ-11A, above, over 99% (765,984 BTU/ft^2, of 768,115 BTU/ft^2 total) of combustible loading for this fire zone is the result of "oil and lubricating materials," which are protected by automatic suppression systems, and the fire loading for this area is otherwise minimal, fire growth from this area is screened, based on Criteria 4 and 6.

TB-FZ-11C (Switchgear Room, West End of Turbine Building on Mezzanine Level). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11A, C, D, E and G. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zones TB-FZ-11E and G have automatic sprinkler systems installed. Fire detection is from an area wide particles of combustion detection system that alarms in the Control Room. Due to the low level of combustible loading for this area (3,328 BTU/ft^2), fire growth beyond this area is screened from consideration, based on Criterion 4.

TB-FZ-11D (Basement Floor South End). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11C, E, G and H. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zones TB-FZ-11E and G have automatic sprinkler systems installed. This area is protected by an automatic closed head area wide sprinkler system, in addition to an automatic sprinkler system to protect the Hydrogen seal oil unit. Due to the presence of an area wide sprinkler system, in addition to one for the seal oil unit, fire propagation from this area is screened from further consideration, based on Criterion 6. The combustible loading for this fire area 39,789 BTU/ft^2.

TB-FZ-11E (Condenser Bay). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11A, B, C, D, F, G and H. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zones TB-FZ-11A, B, D and G have automatic sprinkler systems installed. This area is protected by an automatic closed head area wide sprinkler system, except for the heater bay area. Since the combustible loading for this fire area is only 6,536 BTU/ft^2, though, fire propagation to TB-FZ-11A, B, D and G is screened from consideration, based on Criterion 4.

Also, Criterion 6 allows the analyst to screen fire growth from consideration where automatic suppression is provided above fire sources in the area. Finally, Criterion 5 allows the analyst to screen from consideration propagation to TB-FZ-11F, G and H, due to the low level of combustible loading in these areas. Therefore, fire propagation from TB-FZ-11E is not a concern.

TB-FZ-11F (Feedwater Pump Area - 3' 6" Elevation). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11B, E and H. All other boundaries are fire rated at 2 or 3 hours. Of the non-rated boundaries, the adjacent portions of fire zone TB-FZ-11E and oil hazards in fire zone TB-FZ-11B have automatic sprinkler systems installed. Since the combustible loading for this fire area is only 747 BTU/ft^2, though, fire propagation to TB-FZ-11B, E and H is screened from consideration, based on Criterion 4. Also, Criterion 5 allows the analyst to screen from consideration propagation to TB-FZ-11E and H, due to the low level of combustible loading in these areas. Therefore, fire propagation from TB-FZ-11F is not a concern.

TB-FZ-11G (Southwest Mezzanine Area and Machine Shop). Adjacent fire areas with non-fire rated boundaries are TB-FZ-11A, D and E. All other boundaries are fire rated at 2 or 3 hours. A closed head sprinkler system protects the machine shop, office area and the area under the turbine. Since the combustible loading for this fire area is only 518 BTU/ft^2, though, fire growth beyond this area is screened from further consideration, based on Criteria 4 and 6.

TB-FZ-11H (Condensate Demineralizer Area). This fire zone has an opening to adjacent fire zone TB-FZ-1E, which is protected with sprinklers. Since the combustible loading for this fire area is only 107 BTU/ft^2, though, fire growth beyond this area is screened from further consideration, based on Criterion 4.

In conclusion, all Oyster Creek fire areas and zones screen (qualitatively) from further consideration based on the EPRI FIVE analysis guidelines. The potential for fire propagation from an individual fire area or zone to additional fire areas or zones (i.e., "multiple area fires") are qualitatively screened from further consideration.

4.4 Evaluation of Component Fragilities and Failure Modes

This report section presents and documents the identification of the risk significant components (Task 2), the identification of the risk significant component and supporting cable locations (Task 3) and the process used to determine the impacts of fires in various plant fire areas (Task 4). The role of Tasks 2, 3 and 4 in the Fire Analysis Development Process is illustrated in Figure 4-1.

- Input to this task is from the Level 1 OCPRA and the Fire Hazard Analysis report.
- The output of these tasks consists of the Fire Initiating Event Impact Table (Task 4) which is utilized as input to the development and quantification of the plant model (Task 5) which is presented in report Section 4.6.

The process of identifying those components that could be impacted by fires in the various locations at the Oyster Creek plant (Task 2) was performed in two steps.

Step 1. Identify Risk Significant Components.

Step 2. Identify Components that are not Susceptible to Fire Damage.

The process of identifying component and supporting cable locations (Task 3) is also a two step process. Taken together these four steps provide the input necessary to develop the Fire Initiating Event Impact Table.

Step 3. Physically Locate Remaining Components.

Step 4. Identify Support Cable Locations.

Utilizing the risk significant component list developed in Task 2 and their supporting cable locations identified in Task 3, a Fire Initiating Event Impact Table (Task 4) is developed. The Fire Initiating Event Impact Table provides the impact of fires, in terms of degraded or failed systems, on the Oyster Creek Risk Model as developed in the Level 1 Oyster Creek PRA.

Section 4.4.1, Identification of the Risk Significant Components (Task 2), presents the methods used to generate the risk significant component list.

Section 4.4.2, Identification of Risk Significant Component and Cable Locations (Task 3), presents the physical plant locations of the risk significant components (Table 4.4-3) and their supporting cables and is comprised of steps 3 and 4, as described above. Section 4.4.2 also presents additional fire areas which are screened from further consideration based on the absence of risk significant components. These screened fire areas are presented in Table 4.4-4.

Section 4.4.3, Development of Fire Initiating Event Impact Table, presents the process used to develop the Fire Initiating Event Impact Table (Table 4.4-5). This table provides the link between risk significant component and cable locations and the system functions (or top events) modeled in the Level 1 OCPRA.

Due to the length of Tables 4.4-1 and 4.4-3, they are presented separately at the end of this report section.

4.4.1 Identification of Risk Significant Components (Task 2)

As described above, the identification of risk significant components is a two step process. First, a list of potentially risk significant components is developed using the system analyses for the Oyster Creek PRA (Appendix F). Components which may not have been modeled in the level 1 PRA due to low probability of failure, subsumed into an initiating event or human action or utilized a special analysis (Appendix B) are added to the list through a review of the OCPRA and the Oyster Creek Fire Hazard Analysis Report.

Second, the candidates for the risk significant list are screened for their susceptibility to fires. The screening criteria are presented in the following paragraphs and are designed to eliminate the components which are not susceptible to fire events. The result of the component screening is Table 4.4-1 which presents the screened and risk significant components. The components which are screened are indicated with a number (1 through 7) in the "Fire Note" column corresponding to the screening criteria presented below. Those components with a letter (A through E) in the "Fire Note" column indicate the component is part of the final risk significant components list and corresponds to the classification criteria provided below. Due to the length of Table 4.4-1, it is presented separately at the end of this report section.

The first column of Table 4.4-1, Component Screening List, provides the component number. The second column of Table 4.4-1 illustrates the OCPRA top event in which the component is modeled. The "Top Event" refers to the OCPRA level 1 system function in which the component is modeled. A list of Oyster Creek PRA Top Event Definitions is provided in Table 4.4-2, below.

The third column presents the component identifier or the "basic event designator". The basic event designator refers to the OCPRA fault tree identifier and provides a reference to the system analyses in Appendix F of the OCPRA.

The fourth column provides the component description. The fifth column provides a categorization of the component. This table entry assists in the determination of the assignment of the fire note. A key to the component category is provided at the end of the table.

The sixth and final column of Table 4.4-1 provides the fire screening note. This note details the basis on which a component was screened from further consideration (numbers 1 through 7) or considered risk significant and classified (letters A through E). During the walkdowns, components which were originally screened from consideration may have been added to the risk significant list based on area specific considerations. The following paragraphs provide a key to the fire screening notes.

Note 1 - Failure of passive components resulting in loss of system integrity. This fire note is applied to component failures which represent a loss of system integrity such as pipe rupture and heat exchanger rupture. This component failure mode is not induced by fire events and is therefore screened.

Note 2 - Check valve failures. Although fire induced failure is possible due to extreme heat (valve internals expansion or leakage due gasket failure) it is unlikely since check valves are typically located within the flowstream which provides cooling. In addition, the heat necessary to cause such damage is extreme. As such, check valves are considered to be non-susceptible to damage due to fires.

Note 3 - Mechanical relief valve failure. Although extreme heat produced by some fires may cause premature opening of spring operated relief valves (i.e., potential setpoint drift) spurious opening of these valves at system design pressure is not expected due to fire event. Therefore, these components are non-susceptible to damage due to fires.

Note 4 - Non-critical reactor protection system components. This note is applied to those components in the reactor protection system that are redundant to other components located outside the area and to components that will not prevent the reactor trip function. This noted includes control rod drive failure (rod failure to insert).

Note 5 - Inadvertent manual valve transfer. The transfer failure of a manual valve (either open or closed) is a time related standby failure as such, manual valve transfer failures are not impacted by fire events.

Note 6 - Non-general transient components. This note applies to the OCPRA Level 1 components that are modeled for system functions (i.e., top events) that are not used in the general transient logic structure. For example, isolation condenser isolation (top event MI) is modeled for the sole purpose of mitigating isolation condenser line break LOCA. Since a fire event cannot cause the event for which these components are modeled they are eliminated (screened) from the risk significant component list.

Note 7 - Components duplicated elsewhere in the component list. In the OCPRA some components are modeled with multiple failure modes. Therefore, some components appear more than once on the original component list. Items indicated with this note already appear elsewhere in the list and are screened.

The components which are listed with a letter (A through E) in the "Fire Note" column on Table 4.4-1 indicate those components which are considered risk significant. The letters correspond to the classification of the component as indicated below:

Note A - Components Modeled in Operator Actions. These components are considered risk significant regardless of the component type since manual manipulation of the component by the operator will require the operator to enter the fire area in which the component is located.

Note B - Electrical and Actuation Components. These components are buses, relays, fuses, circuit breakers and pressure switches which are considered risk significant.

Note C - Motor, Air and Solenoid Operated Valves. This fire note is applied to all motor, air and solenoid valves which are considered risk significant. Included in this note are the primary containment isolation valves.

Note D - Major Rotating Equipment (Pumps, Fans and Compressors). This note is applied to the major rotating equipment such as pumps, compressors and fans which are considered risk significant.

Note E - Individual Fuses, Circuit Breakers and Relays. This note is applied to fuses, circuit breakers and relays which are located within panels, cabinets or switchgear which are considered risk significant. These items are not treated separately since the panel, cabinet or switchgear is already addressed by Fire Note B above. The "Fire Category" column indicates the panel, cabinet or switchgear in which the component is located which then appears as an entry under the "Fire Note" column (Note B).

In total, 1264 components were considered. Of these, 712 line items were screened from consideration and 552 components appear on the risk significant component list. The risk significant components and supporting cables are located within the plant in Section 4.4.2.

Table 4.4-2 Oyster Creek PRA Top Event Definitions

System Analysis Title	OCPRA Section	Top Event	Top Event Description
AC Electric Power	F.3	EA EB EC ED OP	4160V Bus 1A 4160V Bus 1B 4160V Bus 1C 4160V Bus 1D Independent failure of 34.5 kV offsite power
ADS	F.15	AD	Automatic Depressurization System
Circulating Water and Main Condenser	F.14	CW CN	Circulating Water System Condenser Vacuum Available
Condensate and Feedwater	F.13	CP FW OF RF	Condensate System Feedwater System Operator Recovers High Level Transient Operator Controls RPV Level (level setdown)
Condensate Transfer	F.20	CT MU ST	Condensate Transfer System Operator Opens IC Makeup Valve CST Available for Long Term Makeup
Containment Spray and ESW	F.11	CC TC	Containment Spray and Emergency Service Water Operator Aligns Containment Spray to Drywell
Core Spray System	F.10	CS OS	Core Spray System Operator Aligns Core Spray to CST Suction
CRD Hydraulic	F.22	CD	CRD Hydraulic for RPV Inventory
DC Electric Power	F.4	DB DC XB XC	Short Term (3 hr) division II (bus B) DC power Short Term (3 hr) division I (bus C) DC power Long Term (>3 hrs) division II (bus B) DC power Long Term (>3 hrs) division I (bus C) DC power
Engineered Safety Features Actuation System (ESFAS)	F.5	DP RL PR	Logic Actuation on High Drywell Pressure Logic Actuation on Lo-Lo Reactor Vessel Level Logic Actuation on High Reactor Vessel Pressure
Fire Protection System	F.19	FP FS	Fire Protection System Operator Aligns Fire Protection to Core Spray Injection (NPSH)
Instrument Air	F.21	IA	Instrument Air System

System Analysis Title	OCPRA Section	Top Event	Top Event Description
Isolation Condenser	F.1	IC	Isolation Condenser System
Liquid Poison (SLC)	F.16	BI	Standby Liquid Control System (Boron Injection)
Main Steam Isolation	F.9	ME MI MS	MSIV Closure on Low Steamline Pressure IC Isolates on High Steam Flow MSIV Closure on Low-Low RPV Water Level
Main Steam Relief	F.24	SO SR VO VR	Safety Valves Open to Relieve Pressure Open Safety Valves Reseat EMRVs Open to Relieve Pressure Open EMRVs Close
Offsite Power Recovery	B.1	LP LQ	Recovery of One Essential Bus Within 30 Minutes Recovery of Second Essential Bus
Containment Isolation	F.17	PI	Primary Containment Isolation
Recovery of Containment Cooling	B.4	RA RC RD SD RV	Recovery of Instrument Air Within 24 Hours Containment Spray Recovery (When CC1 Failed) Recovery of 125 VDC Bus C Shutdown Cooling System (Recovery Only) Containment Vent Recovery (Loss of Support)
Reactor Building Isolation	F.23	RI	Reactor Building Isolation
Recirc Pump Trip	F.12	RP	Recirculation Pump Trip
Reactor Protection	F.6	RS	Reactor Protection System
Standby Gas Treatment	F.18	SG	Standby Gas Treatment System
SDV Isolation	B.3	VS	Scram Discharge Volume Isolation
Service Water	F.7	SW	Service Water System
Containment Vent	F.25	OV	Containment Venting System
TBCCW	F.8	TB	Turbine Building Closed Cooling Water System
Turbine Trip and Bypass	F.2	TT BT BV	Turbine Trip, Stop/Control Valve Closure Turbine Bypass Valves Close on Low Vacuum Turbine Bypass Valves Open and Throttle

4.4.2 Identification of the Risk Significant Component and Cable Locations (Task 3)

As described in the opening of this report section, the process of identifying the risk significant component and supporting cable locations is a two step process. First, the components which were not screened from consideration in Section 4.4.1, above, must be individually located. Risk significant components are located using the Oyster Creek Fire Mitigation Procedure, Fire Hazard Analysis Report, site layout drawings and various other sources of information as well as plant walkdowns (see Section 4.2).

Once the individual components are located, supporting cables of the risk significant components are located. In the case of supporting cables, the location data is determined using the cable routing data available in the Fire Hazard Analysis Report as well as plant walkdowns and plant layout drawings. Not all the risk significant components identified in Task 2 were analyzed in the Fire Hazard Analysis Report and significant effort was expended to locate the cable routing of these components (See Reference 4.10-32).

The supporting cables include any required electrical cables or other functional system support cables. This requirement includes the possibility of component failure due to "hot shorts" that could cause the component to go to an active failure position.

The result of this task is Table 4.4-3, Location of Risk Significant Components and Associated Cables, which serves as input to Task 4. Due to the length of Table 4.4-3, it is presented separately at the end of this section.

The first column of Table 4.4-3 provides the OCPRA Level 1 top event identifier. The second column provides the component identifier. The third column provides the fire area in which the component or supporting cables are located. The fourth column provides the component description.

Columns five and six indicate whether the actual component is physically located within the fire area or if the line entry represents supporting cables. Cable routing typically transits several fire areas within the plant; therefore, the line items which indicate a component may appear several times in the table within different fire areas. The distinction between actual physical component location and supporting cables is utilized further in the development of the fire initiating event impact table and the detailed fire area analysis task.

In addition, several fire areas do not contain any risk significant components or supporting cables. On this basis, these areas are screened from further consideration (see Table 4.4-4, Oyster Creek Fire Areas Screened from Further Consideration). In all, 7 additional fire zones and areas were removed from further consideration at this point. The 6 fire areas that were removed from further consideration in Section 4.1 are also listed in Table 4.4-4 for reference.

Screened areas are treated in Section 4.3, Fire Growth and Propagation to determine the potential for a fire to begin in these areas and spread to other plant areas which do contain risk significant components or supporting cables.

Table 4.4-4
Oyster Creek Fire Areas Screened from Further Consideration

Area	Description	Assigned Frequency
Areas Screened in Section 4.1 (No Plant Trip due to Fire Event)		
RB-FA-2	Primary Containment and Torus	N/A
FW-FA-18	Diesel Driven (Pond) Fire Pump Area	N/A
PH-FA-25	Redundant Fire Pump Area	N/A
NW-FA-23	New Warehouse	N/A
MB-FA-24	Maintenance Building	N/A
EB-FA-28	Site Emergency Building	N/A
Areas Screened in Section 4.4 (No Risk Significant Components or Cables)		
AB-FA-13	Auxilary Boiler Building	2.49E-3
RB-FZ-1A	Refuel Floor (119 Foot Elevation)	1.08E-2
OR-FA-19	Old Radwaste Building	4.11E-3
NR-FA-20	New Radwaste Building	7.36E-3
OG-FA-21	Augmented Offgas Building	9.08E-2
OB-FZ-22D	Mechanical Equipment Room (74' El.)	4.31E-3
LL-FA-29	Low Level Radwaste Storage Facility	3.64E-3

4.4.3 Fire Initiating Event Impact Table Development

Following the identification of risk significant components (Section 4.4.1) and identification of component and cable locations (Section 4.4.2), the Development of the Fire Initiating Event Impact Table links the risk significant component locations and the cable routing information to determine the impact of fire events for each fire area on the system functions (or top events) modeled in the Level 1 OCPRA.

Table 4.4-2, Oyster Creek Level 1 PRA Top Events, provides the system functions or top events modeled in the OCPRA. With this information, as well as the location of risk signification components and support cables, a table can be produced which provides the impact, in terms of degraded or failed systems, as a function of a fire event in a given fire area. The first column of Table 4.4-2 provides the level 1 OCPRA system analysis title. The second column provides the Level 1 OCPRA Appendix in which the system analysis can be found. The third and fourth columns provide the top event designators and top event descriptions, respectively.

Table 4.4-5, Fire Initiating Event Impact Table, provides a listing of the system functions (top events) impacted by fire events within a fire area. The first column of the matrix provides the general plant area either Reactor Building, Turbine Building Office Building or Miscellaneous Areas. The second column of the matrix provides the fire area or zone designator. Subsequent columns (columns 3 through 41) provide the top events designators.

To determine the impacts for a fire event in a given fire area the matrix is read from left to right. An "X", "c" or number under a top events indicates that this event is impacted by the fire event. An "X" in a row indicates that the fire area contains components which when assumed to fail due to the fire event result in failure of the system function in that column. A "c" in a row indicates that the fire area contains support cables of that system function. A key to the numbered notes used on the matrix is provided in Table 4.4-6.

It should be noted that the fire initiating event impact table assumes that all susceptible components and cables within a fire area are affected. Thus, all fire events are conservatively assumed to be all engulfing fires within the fire area and no fire suppression or detection is modeled. Detailed analysis performed in Section 4.3 and Section 4.6 evaluate the potential for fire detection, suppression and/or other mitigative features. In addition, cable failures are modeled as failing component function either due to cable failure (open circuit) or "hot short" (closed circuit and energized) conditions.

In addition to the above listed walkdowns, walkdowns of the as installed Thermo-Lag barriers were performed. These barriers were credited in 3 plant areas (see notes 54, 55 and 56 in Tables 4.4-5 and 4.4-6).

TABLE 4.4-5 FIRE INITIATING EVENT IMPACT TABLE

Fire Area Designator	OYSTER CREEK LEVEL 1 PRA IMPACTED TOP EVENT																																						
	AD	BI	BT	BV	CC	CD	CP	CS	CT	CW	DB	DC	DP	EA	EB	EC	ED	FP	FW	IA	IC	ME	MS	MU	OV	PI	PR	RF	RI	RL	RP	SD	SW	TB	TT	VO	VR	XB	XC
MISCELLANOUS PLANT FIRE AREAS																																							
CW-FA-14					X					X								1															X						
DG-FA-15																2																							
DG-FA-17																	3																						
FS-FA-16																2	3																						
MT-FA-12									X	c				X	X																								
OFFICE BUILDING FIRE AREAS AND ZONES																																							
OB-FZ-04,5	9				c	c	c	c	c	c	X		c		X	c	c	1	c		c	X	X	c	c	c	c	X		c	c	c	c			c	c	c	
OB-FZ-06A	10				X	c		X		c		42	X			X	5				8				c	c				X		c				10	c	18	
OB-FZ-06B					20	21		22								56	X				16			35								c						c	56
OB-FZ-08A	11				20			23			40						X				8	25	25			c					19	c				c	c		
OB-FZ-08B																																						c	
OB-FZ-08C	11	c			c	c		47		c	X		c		c	4	c				8	25	25		c	c						c				15	14	X	
OB-FA-09	11																										c									14	14	c	
OB-FZ-10A	12				c			7					c				X				X			35		c	c									c	c	X	
OB-FZ-10B																	c																						
OB-FZ-22A																	6				16			c		c	c			c								c	
OB-FZ-22B																					16			34		c				c									
OB-FZ-22C																										c				c									
REACTOR BUILDING FIRE AREAS AND ZONES																																							
RB-FZ-01B		X																						24		c			X										
RB-FZ-01C								27				38									X					X			X										
RB-FZ-01D	10				33			28				39	X								X				X	c	X		X	X		X				X	X		
RB-FZ-01E	13				X	X		29			40		c								c	25	25		c	X						c				c	c		
RB-FZ-01F1					36																																		
RB-FZ-01F2								31									c																						
RB-FZ-01F3						X		30																	c	X													
RB-FZ-01F4					37																																		
RB-FZ-01F5								32																															
RB-FZ-01G																										X						X							
RB-FZ-01H																						26	26																

TABLE 4.4-5 FIRE INITIATING EVENT IMPACT TABLE

Fire Area Designator	OYSTER CREEK LEVEL 1 PRA IMPACTED TOP EVENT																																						
	AD	BI	BT	BV	CC	CD	CP	CS	CT	CW	DB	DC	DP	EA	EB	EC	ED	FP	FW	IA	IC	ME	MS	MU	OV	PI	PR	RF	RI	RL	RP	SD	SW	TB	TT	VO	VR	XB	XC
TURBINE BUILDING FIRE AREAS AND ZONES																																							
TB-FA-03A					c			49								X																							
TB-FA-03B					c			50									X																						
TB-FZ-11A			X	X																															46				
TB-FZ-11B				X				47		c		42			c	41	48	1	43		16										19		c		46	42			X
TB-FZ-11C								47		c		X		X	X	X	54	1	c		16							45			45		c			17			
TB-FZ-11D					c			47		c						44	55	1	c	X													c	X					
TB-FZ-11E			X		c		X	47	c	c						c	48	1	X		16										19		c		46				
TB-FZ-11F							51	47									48		X			X																	
TB-FZ-11G					36		52	32																															
TB-FZ-11H							53	49								c	c			X																			
TB-FA-26												X																											

Table 4.4-6 Impact Matrix Notes	
Note	Impact Matrix Note Description
c	Area contains electrical cables which fail system (no components)
X	Area contains system components which fail system function
1.	Cables for diesel fire pump 1-1 (not separately modeled due to system redundancy)
2.	Fails emergency diesel generator 1. Top event EC fails when EA fails.
3.	Fails emergency diesel generator 2. Top event ED fails when EB fails.
4.	Control cable for supply breaker 1A2M (backed up by breaker 1A2P, not separately modeled)
5.	DG 2 start and idling circuit cables. Only required following EA failure.
6.	Vent fan power cables for FN-56-004, -007 and -008
7.	Main and booster pump A power cables
8.	IC A isolation valve control cables only
9.	Actuation switches and control cables
10.	ADS/EMRV actuation relays/control circuits
11.	ADS control cables
12.	ADS actuation panel ER-642-078 and control cables
13.	System II pressure switches, control cables
14.	EMRV pressure circuit control cables
15.	EMRV actuation panels ER-18A and ER-18B
16.	IC B isolation valve control cables only
17.	Fails only 1 division of DC power to EMRVs
18.	Fails supply fan only (ventilation adequate with exhaust only)
19.	Fails automatic RPT actuation only.
20.	Cables for pumps C and D only
21.	Cables for pump B only
22.	Cables for pumps B and C only
23.	Cables for pumps B and D and system 2 isolation valves
24.	Fails makeup to IC "A" only
25.	Contains cables for inboard MSIVs only (will not prevent outboard MSIV closure)
26.	Contains outboard MSIVs only (will not prevent inboard MSIV closure)
27.	Cables for System II isolation valves only (parallel isolation valves)
28.	A, C booster pumps and system I isolation valves
29.	B, D booster pumps and system II isolation valves
30.	A, C main pumps and system I isolation valves

Table 4.4-6 Impact Matrix Notes	
Note	Impact Matrix Note Description
31.	B, D main pumps and system II isolation valves
32.	Cables for B and D main pumps only
33.	Drywell spray isolation valve V-21-11 (System I)
34.	Control cables for A IC makeup only
35.	Control cables for B IC makeup only
36.	System 2 Containment spray pumps
37.	System 1 Containment spray pumps
38.	MCC-DC2 - supplies IC "B" isolation valves only (impact subsumed in IC)
39.	Cable for MCC-DC2 - supplies IC "B" isolation valves only (impact subsumed in IC)
40.	MCC-DC1 - supplies IC "A" and SDC isolation valves only (impact subsumed in IC and SD)
41.	MCC 1A2 circuit breakers can be tripped locally, if necessary
42.	Cable to DC-F only (IC B, train 1 of ADS/CS/containment spray actuation logic)
43.	Supply breakers can be tripped locally/regulating valves control RPV water level
44.	MCC 1A1/1B1 (normally trips on LOSP), does not fail EC/ED
45.	Supply breaker (trip) impact is overridden by assumed loss of 1A/1B (not modeled)
46.	Due to system redundancy and timing of fire damage, failure of turbine trip due to fire is not judged to be credible.
47.	May disable pump trip from the control room (local breaker trip still available)
48.	Switchgear control from local shutdown panels still available
49.	Pump B and C breakers may have to be tripped from bus 1D
50.	Cables for pumps B and C only (failed by loss of bus 1D)
51.	Feedwater regulating valves A, B and C and startup valves
52.	Cables for feedwater train isolation valves (3)
53.	SJAE isolation valves (6)
54.	1D supply cable from DG 2 protected by 3 hour barrier
55.	1D support cables protected by 1 hour barrier
56.	Vent fan, 125VDC power to 1A2 and battery C charger supply cables protected by 1 hour barriers

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1	CP	FLDMAP	Condensate demineralizer A	DEMIN	1
2	CP	FLDMBP	Condensate demineralizer B	DEMIN	1
3	CP	FLDMCP	Condensate demineralizer C	DEMIN	1
4	CP	FLDMDP	Condensate demineralizer D	DEMIN	1
5	CP	FLDMEP	Condensate demineralizer E	DEMIN	1
6	CP	FLDMFP	Condensate demineralizer F	DEMIN	1
7	IA	FL01CP	Air Intake Filter	FILTER	1
8	IA	FL01DP	Air Intake Filter	FILTER	1
9	EC	FLF1AP	Fuel Oil Filter 1A	FILTER	1
10	EC	FLF1BP	Fuel Oil Filter 1B	FILTER	1
11	ED	FLF2AP	Fuel Oil Filter 2A	FILTER	1
12	ED	FLF2BP	Fuel Oil Filter 2B	FILTER	1
13	IA	FL01BP	Postfilter 1-1	FILTER	1
14	IA	FL01AP	Prefilter 1-1	FILTER	1
15	SG	FL011P	Train Charcoal Filter F-1-11	FILTER	1
16	SG	FL008P	Train Charcoal Filter F-1-8	FILTER	1
17	SG	FL010P	Train HEPA Filter F-1-10	FILTER	1
18	SG	FL007P	Train HEPA Filter F-1-7	FILTER	1
19	SG	FL012P	Train Postfilter F-1-12	FILTER	1
20	SG	FL009P	Train Postfilter F-1-9	FILTER	1
21	SG	FL029P	Train Prefilter F-1-29	FILTER	1
22	SG	FL030P	Train Prefilter F-1-30	FILTER	1
23	TB	HX004P	Heat Exchanger C-5-004	HEAT EX	1
24	CC	HXCCAB	Heat Exchanger H-21-001A	HEAT EX	1
25	CC	HX1AB	Heat Exchanger H-21-001B	HEAT EX	1
26	CC	HXCCCR	Heat Exchanger H-21-001C	HEAT EX	1
27	CC	HXCCDR	Heat Exchanger H-21-001D	HEAT EX	1
28	TB	HX003B	Heat Exchanger(C-5-003)	HEAT EX	1
29	CP	HXACAP	SJAE after condenser A	HEAT EX	1
30	CP	HXACBP	SJAE after condenser B	HEAT EX	1
31	CP	HXACCP	SJAE after condenser C	HEAT EX	1
32	FW	HXDCAP	Drain cooler A	HX PLUG	1
33	CP	HXDCAP	Drain cooler A	HX PLUG	1
34	CP	HXDCBP	Drain cooler B	HX PLUG	1
35	FW	HXDCBP	Drain cooler B	HX PLUG	1
36	CP	HXDCCP	Drain cooler C	HX PLUG	1
37	FW	HXDCCP	Drain cooler C	HX PLUG	1
38	SD	HX11AP	HEAT EXCHANGER 1-1 BLOCKAGE	HX PLUG	1
39	SD	HX12BP	HEAT EXCHANGER 1-2 BLOCKAGE	HX PLUG	1
40	SD	HXSDAP	HEAT EXCHANGER A BLOCKAGE	HX PLUG	1
41	SD	HXSDBP	HEAT EXCHANGER B BLOCKAGE	HX PLUG	1
42	SD	HXSDCP	HEAT EXCHANGER C BLOCKAGE	HX PLUG	1
43	CP	HXHPAP	HP heater A	HX PLUG	1
44	FW	HXHPAP	HP heater A	HX PLUG	1
45	CP	HXHPBP	HP heater B	HX PLUG	1
46	FW	HXHPBP	HP heater B	HX PLUG	1

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
47	CP	HXHPCP	HP heater C	HX PLUG	1
48	FW	HXHPCP	HP heater C	HX PLUG	1
49	CP	HX0A1P	LP heater A-1	HX PLUG	1
50	FW	HX0A1P	LP heater A-1	HX PLUG	1
51	CP	HX0A2P	LP heater A-2	HX PLUG	1
52	FW	HX0A2P	LP heater A-2	HX PLUG	1
53	CP	HX0B1P	LP heater B-1	HX PLUG	1
54	FW	HX0B1P	LP heater B-1	HX PLUG	1
55	CP	HX0B2P	LP heater B-2	HX PLUG	1
56	FW	HX0B2P	LP heater B-2	HX PLUG	1
57	CP	HX0C1P	LP heater C-1	HX PLUG	1
58	FW	HX0C1P	LP heater C-1	HX PLUG	1
59	CP	HX0C2P	LP heater C-2	HX PLUG	1
60	FW	HX0C2P	LP heater C-2	HX PLUG	1
61	CP	HXICAP	SJAE Inter condenser A	HX PLUG	1
62	CP	HXICBP	SJAE Inter condenser B	HX PLUG	1
63	CP	HXICCP	SJAE Inter condenser C	HX PLUG	1
64	CP	HXSSEP	Steam seal exhauster	HX PLUG	1
65	FP	PP001B	Piping Integrity	PIPE	1
66	FP	XXLOWS	Diesel Pump Water Supply	POND	1
67	CD	ST51AT	NC51A Y-Strainer	STRAINER	1
68	CD	ST51BT	NC51B Y-Strainer	STRAINER	1
69	IA	TK001B	Air Receiver 1-1	TANK	1
70	IA	TK002B	Air Receiver 1-2	TANK	1
71	IA	TK003B	Air Receiver 1-3	TANK	1
72	ST	TK001B	CST T-11-001	TANK	1
73	EC	TK001A	Fuel Oil Storage Tank	TANK	1
74	ED	TK001B	Fuel Oil Storage Tank	TANK	1
75	IC	IC00AP	Isolation Condenser "A"	TANK	1
76	IC	IC00BP	Isolation Condenser "B"	TANK	1
77	FP	TK001B	Redundant Fire Water Pump Supply	TANK	1
78	BI	TKSLCL	T-19-001 Liquid Poison Tank	TANK	1
79	CS	CV053D	Booster Discharge CV V-20-53	CHECK VLV	2
80	CS	CV053P	Booster Discharge CV V-20-53	CHECK VLV	2
81	CS	CV054P	Booster Discharge CV V-20-54	CHECK VLV	2
82	CS	CV055D	Booster Discharge CV V-20-55	CHECK VLV	2
83	EC	CV01AD	Check Valve 1A	CHECK VLV	2
84	EC	CV01BD	Check Valve 1B	CHECK VLV	2
85	ED	CV02AD	Check Valve 2A	CHECK VLV	2
86	ED	CV02BD	Check Valve 2B	CHECK VLV	2
87	CT	CV012L	Check Valve V-11-12	CHECK VLV	2
88	CT	CV013D	Check Valve V-11-13	CHECK VLV	2
89	MU	CV033D	Check Valve V-11-33	CHECK VLV	2
90	MU	CV035D	Check Valve V-11-35	CHECK VLV	2
91	MU	CV042L	Check Valve V-11-42	CHECK VLV	2
92	FW	CV280R	Check valve V-2-80	CHECK VLV	2

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
93	CS	CV030D	Check Valve V-20-30	CHECK VLV	2
94	CS	CV031D	Check Valve V-20-31	CHECK VLV	2
95	CS	CV051D	Check Valve V-20-51	CHECK VLV	2
96	CS	CV060D	Check Valve V-20-60	CHECK VLV	2
97	CS	CV061D	Check Valve V-20-61	CHECK VLV	2
98	CS	CV088D	Check Valve V-20-88	CHECK VLV	2
99	CS	CV089D	Check Valve V-20-89	CHECK VLV	2
100	PI	CV515L	Check Valve V-26-15	CHECK VLV	2
101	PI	CV617L	Check Valve V-26-17	CHECK VLV	2
102	EC	CV391D	Check Valve V-39-1	CHECK VLV	2
103	ED	CV391D	Check Valve V-39-1	CHECK VLV	2
104	IA	CV078D	Check Valve V-6-3078	CHECK VLV	2
105	IA	CV079D	Check Valve V-6-3079	CHECK VLV	2
106	FP	CV901D	Check Valve V-9-1	CHECK VLV	2
107	FP	CV902C	Check Valve V-9-2	CHECK VLV	2
108	FP	CV336D	Check Valve V-9-336	CHECK VLV	2
109	EC	CVXXXD	Check Valve XXX	CHECK VLV	2
110	ED	CVYYYD	Check Valve YYY	CHECK VLV	2
111	SD	CV11RP	Discharge Check Valve Transfers Closed	CHECK VLV	2
112	SD	CV12RP	Discharge Check Valve Transfers Closed	CHECK VLV	2
113	CP	CV275R	Discharge check valve V-2-75	CHECK VLV	2
114	CP	CV276R	Discharge check valve V-2-76	CHECK VLV	2
115	CP	CV277R	Discharge check valve V-2-77	CHECK VLV	2
116	CP	CV273T	Discharge check valve V-2-78	CHECK VLV	2
117	FW	CV278T	Discharge check valve V-2-78	CHECK VLV	2
118	CP	CV279T	Discharge check valve V-2-79	CHECK VLV	2
119	FW	CV279T	Discharge check valve V-2-79	CHECK VLV	2
120	CP	CV280T	Discharge check valve V-2-80	CHECK VLV	2
121	FW	CV280T	Discharge check valve V-2-80	CHECK VLV	2
122	CS	CV052D	Discharge Check Valve V-20-52	CHECK VLV	2
123	CC	CV066D	Discharge Check Valve V-3-66	CHECK VLV	2
124	CC	CV002D	Discharge Valve V-21-2	CHECK VLV	2
125	CC	CV068D	ESW Pump 52A Disch. CV V-3-68	CHECK VLV	2
126	CC	CV067D	ESW Pump 52B Disch. CV V-3-67	CHECK VLV	2
127	CC	CV065D	ESW Pump D Discharge CV V-3-65	CHECK VLV	2
128	CC	CVO68D	ESW Pump Discharge CV	CHECK VLV	2
129	CC	PM52B	ESW Pump Discharge CV	CHECK VLV	2
130	CC	CVO66D	ESW Pump Discharge CV V-3-66	CHECK VLV	2
131	CP	CVV271T	Injection line check valve V-2-71	CHECK VLV	2
132	CP	CV272T	Injection line check valve V-2-72	CHECK VLV	2
133	CP	CV273T	Injection line check valve V-2-73	CHECK VLV	2
134	CP	CV274T	Injection line check valve V-2-74	CHECK VLV	2
135	SW	CV062C	Pump Discharge Check Value V-3-62	CHECK VLV	2
136	SW	CV063P	Pump Discharge Check Valve V-3-63	CHECK VLV	2
137	TB	CV016D	Pump Discharge Check Valve V-5-16	CHECK VLV	2
138	TB	CV017T	Pump Discharge Check Valve V-5-17	CHECK VLV	2

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
139	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	2
140	CS	CV016D	Pump Discharge CV V-20-16	CHECK VLV	2
141	CS	CV022D	Pump Discharge CV V-20-22	CHECK VLV	2
142	CS	CV008D	Pump Discharge CV V-20-8	CHECK VLV	2
143	CS	CV009D	Pump Discharge CV V-20-9	CHECK VLV	2
144	CC	CV010D	Pump Discharge CV V-21-10	CHECK VLV	2
145	CC	CV004D	Pump Discharge CV V-21-4	CHECK VLV	2
146	CC	CV008D	Pump Discharge Valve V-21-10	CHECK VLV	2
147	CC	CV008D	Pump Discharge Valve V-21-8	CHECK VLV	2
148	SD	CVSDAD	SDC Pump A Discharge Check Valve Fails to Open	CHECK VLV	2
149	SD	CVSDBD	SDC Pump B Discharge Check Valve Fails to Open	CHECK VLV	2
150	SD	CVSDCD	SDC Pump C Discharge Check Valve Fails to Open	CHECK VLV	2
151	CS	CV150D	Testable CV NZ02A (V-20-150)	CHECK VLV	2
152	CS	CV151D	Testable CV NZ02B (V-20-151)	CHECK VLV	2
153	CS	CV152D	Testable CV NZ02C (V-20-152)	CHECK VLV	2
154	CS	CV153D	Testable CV NZ02D (V-20-153)	CHECK VLV	2
155	CD	CV010L	V-15-10 Stop Check Valve	CHECK VLV	2
156	CD	CV027T	V-15-27 Check Valve	CHECK VLV	2
157	CD	HV028T	V-15-28 Check Valve	CHECK VLV	2
158	CD	CV007C	V-15-7 Stop Check Valve	CHECK VLV	2
159	BI	CV916D	V-19-16 Common Line Check Valve	CHECK VLV	2
160	BI	CV920D	V-19-20 Common Line Check Valve	CHECK VLV	2
161	BI	CV937D	V-19-37 Pump A Discharge Check Valve	CHECK VLV	2
162	BI	CV938D	V-19-38 Pump B Discharge Check Valve	CHECK VLV	2
163	CS	CV050L	Vacuum Break (Sys I) CV V-20-50	CHECK VLV	2
164	FP	RV957T	Relief Valve V-9-57	FP-1	3
165	FP	RV967T	Relief Valve V-9-67	FP-2	3
166	BI	RV942T	V-19-42 Pump A Discharge Relief Valve	RELIEF	3
167	BI	RV943T	V-19-43 Pump B Discharge Relief Valve	RELIEF	3
168	IA	RV301T	Air Compressor 1-1 Relief Valve V-6S-301	RELIEF VLV	3
169	IA	RV081T	Air Compressor 1-2 Relief Valve V-6S-81	RELIEF VLV	3
170	IA	RV298T	Air Compressor 1-3 Relief Valve V-6S-298	RELIEF VLV	3
171	IA	RV317T	Air Receiver 1-1 Relief Valve V-6S-317	RELIEF VLV	3
172	IA	RV008T	Air Receiver 1-1 Relief Valve V-6S-8	RELIEF VLV	3
173	IA	RV318T	Air Receiver 1-2 Relief Valve V-6S-318	RELIEF VLV	3
174	IA	RV009T	Air Receiver 1-2 Relief Valve V-6S-9	RELIEF VLV	3
175	IA	RV319T	Air Receiver 1-3 Relief Valve V-6S-319	RELIEF VLV	3
176	CS	RV024P	Relief Valve V-20-24	RELIEF VLV	3
177	CS	RV025P	Relief Valve V-20-25	RELIEF VLV	3
178	CD	RV045T	V-15-45 Relief Valve	RELIEF VLV	3
179	CD	RV046T	V-15-46 Relief Valve	RELIEF VLV	3
180	SR	SV28DR	Safety Valve (NR-28D) V-1-160	SAFETY	3
181	SO	SV28DD	Safety Valve (NR-28D) V-1-160	SAFETY	3
182	SO	SV28ED	Safety Valve (NR-28E) V-1-161	SAFETY	3
183	SR	SV28ER	Safety Valve (NR-28E) V-1-161	SAFETY	3
184	SO	SV28FD	Safety Valve (NR-28F) V-1-162	SAFETY	3

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
185	SR	SV28FR	Safety Valve (NR-28F) V-1-162	SAFETY	3
186	SO	SV28GD	Safety Valve (NR-28G) V-1-163	SAFETY	3
187	SR	SV28GR	Safety Valve (NR-28G) V-1-163	SAFETY	3
188	SO	SV28HD	Safety Valve (NR-28H) V-1-164	SAFETY	3
189	SR	SV28HR	Safety Valve (NR-28H) V-1-164	SAFETY	3
190	SO	SV28JD	Safety Valve (NR-28J) V-1-165	SAFETY	3
191	SR	SV28JR	Safety Valve (NR-28J) V-1-165	SAFETY	3
192	SO	SV28KD	Safety Valve (NR-28K) V-1-166	SAFETY	3
193	SR	SV28KR	Safety Valve (NR-28K) V-1-166	SAFETY	3
194	SO	SV28LD	Safety Valve (NR-28L) V-1-167	SAFETY	3
195	SR	SV28LR	Safety Valve (NR-28L) V-1-167	SAFETY	3
196	SO	SV28MD	Safety Valve (NR-28M) V-1-168	SAFETY	3
197	SR	SV28MR	Safety Valve (NR-28M) V-1-168	SAFETY	3
198	RS	CRDR12	Control Rod Drives (137)	ONTROL RO	4
199	RS	AVSC12	Scram Valves(137 Outlet Valves CV127s)	FAIL SAFE	4
200	BI	HTSLCR	Tank Discharge Piping Heat Tracing	HT TRACE	4
201	RS	PX018R	Reactor Pressure Transmitter PT-622-1018	PRESS TRANS	4
202	RS	PX019R	Reactor Pressure Transmitter PT-622-1019	PRESS TRANS	4
203	RS	LG018D	Trip Unit PT-622-1018	PRESS TRANS	4
204	RS	LG019D	Trip Unit PT-622-1019	PRESS TRANS	4
205	RS	RL11AD	Reactor Trip Relay 1K21A	RELAY	4
206	RS	RL151D	Reactor Trip Relay 1K51	RELAY	4
207	RS	RL152D	Reactor Trip Relay 1K52	RELAY	4
208	RS	RL1ZAD	Reactor Trip Relay 2K21A	RELAY	4
209	RS	RL251D	Reactor Trip Relay 2K51	RELAY	4
210	RS	RL252D	Reactor Trip Relay 2K52	RELAY	4
211	RS	VSAB1D	ARI Block Valve #1	SOV	4
212	RS	VSAB2D	ARI Block Valve #2	SOV	4
213	RS	VSAR1D	ARI Main Header Vent Valve #1	SOV	4
214	RS	VS450D	Backup Scram Valve I (V-6-450)	SOV	4
215	RS	VS451D	Backup Scram Valve II (V-6-451)	SOV	4
216	RS	VS117X	Scram Pilot Solenoid Valves I (137 SO117s)	SOV	4
217	RS	VS118X	Scram Pilot Solenoid Valves II (137 SO118s)	SOV	4
218	IA	HV207T	Air Dryer A-B Inlet Isolation Valve V-6-207	HOV XFER	5
219	IA	HV222T	Air Dryer A-B Outlet Isolation Valve V-6-22	HOV XFER	5
220	IA	HV205T	Air Dryer Inlet Manual Valve V-6-205	HOV XFER	5
221	IA	HV240T	Air Dryer Outlet Manual Valve V-6-240	HOV XFER	5
222	IA	HV005T	Air Receiver Inlet Manual Valve V-6S-5	HOV XFER	5
223	IA	HV006T	Air Receiver Inlet Manual Valve V-6S-6	HOV XFER	5
224	IA	HV314T	Air Receiver Outlet Manual Valve V-6S-314	HOV XFER	5
225	IA	HV315T	Air Receiver Outlet Manual Valve V-6S-315	HOV XFER	5
226	ST	HV001T	Butterfly valve V-11-1	HOV XFER	5
227	CT	HV015T	Butterfly Valve V-11-15	HOV XFER	5
228	CT	HV008T	Butterfly Valve V-11-8	HOV XFER	5
229	CT	HV009T	Butterfly Valve V-11-9	HOV XFER	5
230	TB	HV058T	Circ. Water to Heat Exchanger Iso. V-3-58	HOV XFER	5

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
231	SW	HV058T	Circ. Water-TBCCW Heat Exchgr Isol.V-3-58	HOV XFER	5
232	SW	HV059T	Common Hdr.-TBCCW Heat Exchgr V-3-59	HOV XFER	5
233	CP	HV484T	Condensate demin. manual isol V-2-484	HOV XFER	5
234	CP	HV488	Condensate demin. manual isol V-2-488	HOV XFER	5
235	CP	HV489T	Condensate demin. manual isol V-2-489	HOV XFER	5
236	CP	HV490T	Condensate demin. manual isol V-2-490	HOV XFER	5
237	CP	HV921T	Condensate demin. manual isol V-2-491	HOV XFER	5
238	CP	HV492T	Condensate demin. manual isol V-2-492	HOV XFER	5
239	CP	HV493T	Condensate demin. manual isol V-2-493	HOV XFER	5
240	CP	HV702T	Condensate demin. manual isol V-2-702	HOV XFER	5
241	CP	HV703T	Condensate demin. manual isol V-2-703	HOV XFER	5
242	CP	HV704T	Condensate demin. manual isol V-2-704	HOV XFER	5
243	CP	HV705T	Condensate demin. manual isol V-2-705	HOV XFER	5
244	CP	HV706T	Condensate demin. manual isol V-2-706	HOV XFER	5
245	CP	HV229T	Condenser Discharge Valve F-2-29	HOV XFER	5
246	CP	HV230T	Condenser Discharge Valve F-2-30	HOV XFER	5
247	CP	HV231T	Condenser Discharge Valve F-2-31	HOV XFER	5
248	CP	HV232T	Condenser Discharge Valve F-2-32	HOV XFER	5
249	CP	HV233T	Condenser Discharge Valve F-2-33	HOV XFER	5
250	CP	HV234T	Condenser Discharge Valve F-2-34	HOV XFER	5
251	SD	HV151T	Discharge Valve V-5-151 Transfers Closed	HOV XFER	5
252	SD	HV152T	Discharge Valve V-5-152 Transfers Closed	HOV XFER	5
253	TB	HVHXBT	Heat Ex (C-5-003) Inlet Isolation V-5-6	HOV XFER	5
254	TB	HVHXBT	Heat Ex (C-5-003) Outlet Isolation Valve V-5-5	HOV XFER	5
255	TB	HVHXAT	Heat Ex (C-5-004) Inlet Isolation V-5-8	HOV XFER	5
256	TB	HVHXAT	Heat Ex (C-5-004) Outlet Isolation V-5-7	HOV XFER	5
257	SD	HV685T	Heat Ex 1-1 Bypass V-5-685 Transfers Open	HOV XFER	5
258	SD	HV130T	Heat Ex 1-1 Suction V-5-130 Transfers Closed	HOV XFER	5
259	SD	HV131T	Heat Ex 1-1 Suction V-5-131 Transfers Closed	HOV XFER	5
260	SD	HV684T	Heat Ex 1-2 Bypass V-5-684 Transfers Open	HOV XFER	5
261	SD	HV133T	Heat Ex 1-2 Discharge V-5-133 Transfers Closed	HOV XFER	5
262	SD	HV132T	Heat Ex 1-2 Suction V-5-132 Transfers Closed	HOV XFER	5
263	SD	HV136T	Heat Ex Bypass Valve V-5-136 Fails Open	HOV XFER	5
264	TB	HV009T	Heat Exchanger Bypass Valve V-5-9	HOV XFER	5
265	TB	HVHXBT	HX (C-5-003) Cool Water Outlet Iso. V-3-76	HOV XFER	5
266	TB	HVHXBT	HX (C-5-003) Cool.Water Inlet Iso.V-3-74	HOV XFER	5
267	TB	HVHXAT	HX (C-5-004) Cool.Water Inlet Iso.V-3-75	HOV XFER	5
268	TB	HVHXAT	HX (C-5-004) Cool.Water Outlet V-3-77	HOV XFER	5
269	CP	HV235T	Injection line manual isol V-2-35	HOV XFER	5
270	CP	HV236T	Injection line manual isol V-2-36	HOV XFER	5
271	CT	HV010T	Isolation Valve V-11-10	HOV XFER	5
272	CT	HV011T	Isolation Valve V-11-11	HOV XFER	5
273	FP	HV904T	Manual Discharge Valve V-9-4	HOV XFER	5
274	FP	HV905T	Manual Discharge Valve V-9-5	HOV XFER	5
275	CP	HV161T	Manual isolation valve V-2-161	HOV XFER	5
276	CP	HV290T	Manual isolation valve V-2-90	HOV XFER	5

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
277	FP	HV913T	Manual Isolation Valve V-9-13	HOV XFER	5
278	FP	HV918T	Manual Isolation Valve V-9-18	HOV XFER	5
279	FP	HV933T	Manual Isolation Valve V-9-33	HOV XFER	5
280	FP	HV440T	Manual Isolation Valve V-9-440	HOV XFER	5
281	FP	HV907T	Manual Isolation Valve V-9-7	HOV XFER	5
282	FP	HV908T	Manual Isolation Valve V-9-8	HOV XFER	5
283	FP	HV334T	Manual Suction Valve V-9-334	HOV XFER	5
284	CT	HV160T	Manual Valve V-11-160	HOV XFER	5
285	ST	HV161T	Manual Valve V-11-161	HOV XFER	5
286	ST	HV002T	Manual Valve V-11-2	HOV XFER	5
287	CT	HV004T	Manual Valve V-11-4	HOV XFER	5
288	CT	HV041T	Manual Valve V-11-41	HOV XFER	5
289	MU	HV044T	Manual Valve V-11-44	HOV XFER	5
290	MU	HV063T	Manual Valve V-11-463	HOV XFER	5
291	MU	HV049T	Manual Valve V-11-49	HOV XFER	5
292	CP	HV236T	Manual valve V-2-236	HOV XFER	5
293	CP	HV237T	Manual valve V-2-237	HOV XFER	5
294	CP	HV226T	Manual valve V-2-26	HOV XFER	5
295	CP	HV240T	Manual valve V-2-40	HOV XFER	5
296	CP	HV241T	Manual valve V-2-41	HOV XFER	5
297	CP	HV254T	Manual valve V-2-54	HOV XFER	5
298	CP	HV255T	Manual valve V-2-55	HOV XFER	5
299	CP	HV256T	Manual valve V-2-56	HOV XFER	5
300	CS	HV017T	Manual Valve V-20-17	HOV XFER	5
301	CS	HV023T	Manual Valve V-20-23	HOV XFER	5
302	CS	HV082T	Manual Valve V-20-82	HOV XFER	5
303	CS	HV083T	Manual Valve V-20-83	HOV XFER	5
304	CS	HV090T	Manual Valve V-20-90	HOV XFER	5
305	CS	HV091T	Manual Valve V-20-91	HOV XFER	5
306	EC	HV049T	Manual Valve V-36-49	HOV XFER	5
307	ED	HVZZZT	Manual Valve V-36-ZZ	HOV XFER	5
308	EC	HV017T	Manual Valve V-39-17	HOV XFER	5
309	EC	HV392T	Manual Valve V-39-2	HOV XFER	5
310	ED	HV392T	Manual Valve V-39-2	HOV XFER	5
311	EC	HV003T	Manual Valve V-39-3	HOV XFER	5
312	ED	HVVVVT	Manual Valve V-39-VVV	HOV XFER	5
313	ED	HVWWWT	Manual Valve V-39-WWW	HOV XFER	5
314	IA	HV177T	Manual Valve V-6-3177	HOV XFER	5
315	IA	HV179T	Manual Valve V-6-3179	HOV XFER	5
316	IA	HV180T	Manual Valve V-6-3180	HOV XFER	5
317	IA	HV028T	Manual Valve V-9-28	HOV XFER	5
318	IA	HV028T	Manual Valve V-9-28	HOV XFER	5
319	IA	HV048T	Manual Valve V-9-48	HOV XFER	5
320	IA	HV074T	Manual Valve V-9-74	HOV XFER	5
321	IA	HV075T	Manual Valve V-9-75	HOV XFER	5
322	IA	HV076T	Manual Valve V-9-76	HOV XFER	5

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
323	IA	HV243T	Postfilter 1-1 Isolation Valve V-6-243	HOV XFER	5
324	IA	HV244T	Postfilter 1-1 Isolation Valve V-6-244	HOV XFER	5
325	IA	HV193T	Prefilter 1-1 Isolation Manual Valve V-6-193	HOV XFER	5
326	IA	HV194T	Prefilter 1-1 Isolation Manual Valve V-6-194	HOV XFER	5
327	TB	CV010T	Pump Discharge Iso. Valve V-5-10	HOV XFER	5
328	TB	CV011T	Pump Discharge Iso. Valve V-5-11	HOV XFER	5
329	TB	HV012T	Pump Discharge Iso. Valve V-5-12	HOV XFER	5
330	SW	HV033T	Pump Discharge Isolation Valve V-3-3	HOV XFER	5
331	SW	HV034T	Pump Discharge Isolation Valve V-3-34	HOV XFER	5
332	CP	HV227T	Pump manual isolation valve V-2-27	HOV XFER	5
333	CP	HV228T	Pump manual isolation valve V-2-28	HOV XFER	5
334	CP	HV037T	Pump manual isolation valve V-2-37	HOV XFER	5
335	CP	HV238T	Pump manual isolation valve V-2-38	HOV XFER	5
336	CP	HV239T	Pump manual isolation valve V-2-39	HOV XFER	5
337	TB	HV013T	Pump Suction Iso. Valve V-5-13	HOV XFER	5
338	TB	HV014T	Pump Suction Iso. Valve V-5-14	HOV XFER	5
339	TB	HV015T	Pump Suction Iso. Valve V-5-15	HOV XFER	5
340	CS	HV027T	Recirc. to Torus Valve V-20-27	HOV XFER	5
341	CS	HV026T	Recirc. Torus Valve V-20-26	HOV XFER	5
342	SD	HV149T	Suction Valve V-5-149 Transfers Closed	HOV XFER	5
343	SD	HV150T	Suction Valve V-5-150 Transfers Closed	HOV XFER	5
344	CP	HV566T	TBCCW isolation valve V-5-66	HOV XFER	5
345	CP	HV567T	TBCCW isolation valve V-5-67	HOV XFER	5
346	CP	HV571T	TBCCW isolation valve V-5-71	HOV XFER	5
347	CP	HV572T	TBCCW isolation valve V-5-72	HOV XFER	5
348	CP	HV576T	TBCCW isolation valve V-5-76	HOV XFER	5
349	CP	HV577T	TBCCW isolation valve V-5-77	HOV XFER	5
350	IA	HV053T	TBCCW Inlet Manual Valve V-5-53	HOV XFER	5
351	FW	HV563T	TBCCW manual isol V-5-63	HOV XFER	5
352	FW	HV565T	TBCCW manual isol V-5-65	HOV XFER	5
353	FW	HV568T	TBCCW manual isol V-5-68	HOV XFER	5
354	FW	HV570T	TBCCW manual isol V-5-70	HOV XFER	5
355	FW	HV573T	TBCCW manual isol V-5-73	HOV XFER	5
356	FW	HV575T	TBCCW manual isol V-5-75	HOV XFER	5
357	IA	HV055T	TBCCW Outlet Manual Valve V-5-55	HOV XFER	5
358	IA	HV060T	TBCCW Outlet Manual Valve V-5-60	HOV XFER	5
359	CD	HV026T	V-15-26 Manual Valve	HOV XFER	5
360	CD	CV029T	V-15-29 Manual Valve	HOV XFER	5
361	CD	HV030T	V-15-30 Manual Valve	HOV XFER	5
362	CD	HV005T	V-15-5 Manual Valve	HOV XFER	5
363	CD	HV006T	V-15-6 Manual Valve	HOV XFER	5
364	BI	HV919T	V-19-19 Injection Path Inboard Isolation	HOV XFER	5
365	BI	HV925T	V-19-25 Injection Path Outboard Isolation	HOV XFER	5
366	BI	HV194T	V-19-4 Tank Isolation Valve	HOV XFER	5
367	BI	HV195T	V-19-5 Pump B Suction Valve	HOV XFER	5
368	BI	HV196T	V-19-6 Pump A Suction Valve	HOV XFER	5

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
369	BI	HV197T	V-19-7 Pump A Discharge Valve	HOV XFER	5
370	BI	HV198T	V-19-8 Pump B Discharge Valve	HOV XFER	5
371	RI	AV011D	CRD Rebuild Area&Gen Area Sup V-28-11	AOV	6
372	RI	AV012D	CRD Rebuild Area&Gen Area Sup V-28-12	AOV	6
373	RI	AV021D	RB General Area Exhaust Valve V-28-21	AOV	6
374	RI	AV022D	RB General Area Exhaust Valve V-28-22	AOV	6
375	RI	AV010D	RB General Area Supply Valve V-28-10	AOV	6
376	RI	AV013D	RB General Area Supply Valve V-28-13	AOV	6
377	RI	AV014D	RB General Area Supply Valve V-28-14	AOV	6
378	RI	AV003D	RB General Area Supply Valve V-28-3	AOV	6
379	RI	AV036D	RB General Area Supply Valve V-28-36	AOV	6
380	RI	AV037D	RB General Area Supply Valve V-28-37	AOV	6
381	RI	AV004D	RB General Area Supply Valve V-28-4	AOV	6
382	RI	AV005D	RB General Area Supply Valve V-28-5	AOV	6
383	RI	AV006D	RB General Area Supply Valve V-28-6	AOV	6
384	RI	AV007D	RB General Area Supply Valve V-28-7	AOV	6
385	RI	AV008D	RB General Area Supply Valve V-28-8	AOV	6
386	RI	AV009D	RB General Area Supply Valve V-28-9	AOV	6
387	SG	AV823D	Train Inlet Valve V-28-23	AOV	6
388	SG	AV827D	Train Inlet Valve V-28-27	AOV	6
389	SG	AV826D	Train Outlet Valve V-28-26	AOV	6
390	SG	AV830D	Train Outlet Valve V-28-30	AOV	6
391	SG	AV819T	Common Inlet Valve V-28-19	AOV XFER	6
392	EE	XXEAFF	4160V AC Bus 1A Failed	BUS	6
393	EE	XXEBFF	4160V AC Bus 1B Failed	BUS	6
394	EE	CB01CO	Breaker 1C	CKT BKR	6
395	EE	CB01DO	Breaker 1D from 4,160VAC Bus 1B	CKT BKR	6
396	EE	CBDG1C	Breaker EDG1	CKT BKR	6
397	EE	CBDG2C	EDG Output Breaker EDG2	CKT BKR	6
398	CF	CBCSAO	Pump 51A (1-1) Supply Breaker	CKT BKR	6
399	CF	CBCSCO	Pump 51C (1-3) Supply Breaker	CKT BKR	6
400	SG	DA049T	Train Inlet Damper DM-28-0049	DAMPER	6
401	SG	DA050T	Train Inlet Damper DM-28-0050	DAMPER	6
402	SG	FNTNAR	Train A Fan	FAN	6
403	SG	FNTNBR	Train B Fan	FAN	6
404	MI	MV035D	Condensate Isolation Valve MOV-14-35	MOV	6
405	MI	MV037D	Condensate Isolation Valve MOV-14-37	MOV	6
406	MI	MV034D	Condensate Return Iso. Valve MOV-14-34	MOV	6
407	MI	MV036D	Condensate Return Iso. Valve MOV-14-36	MOV	6
408	CF	MV211D	Drywell Header Inlet V-21-11	MOV	6
409	MI	MV032D	Steam Isolation Valve MOV-14-32	MOV	6
410	MI	MV033D	Steam Isolation Valve MOV-14-33	MOV	6
411	MI	MV030D	Steam Supply Isolation Valve MOV-14-30	MOV	6
412	MI	MV031D	Steam Supply Isolation Valve MOV-14-31	MOV	6
413	CF	ZHECF1	Operator closes drywell supply	OP ACT	6
414	CF	ZHECF1	Operator closes drywell supply	OP ACT	6

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
415	MI	ZHEMI2	Operator Prevents IC Actuation	OP ACT	6
416	CF	SP15AD	Drywell Pressure Switch IP-15A	PRESS SW	6
417	CF	SP15BD	Drywell Pressure Switch IP-15B	PRESS SW	6
418	CF	SP15CD	Drywell Pressure Switch IP-15C	PRESS SW	6
419	CF	SP15DD	Drywell Pressure Switch IP-15D	PRESS SW	6
420	MI	SP1A1D	IC Condensate Line DP Switch 11A1	PRESS SW	6
421	MI	SP1A2D	IC Condensate Line DP Switch 11A2	PRESS SW	6
422	MI	SP1B1D	IC Condensate Line DP Switch 11B1	PRESS SW	6
423	MI	SP1B2D	IC Condensate Line DP Switch 11B2	PRESS SW	6
424	MI	SP5A1D	IC Steam Line DP Switch 5A1	PRESS SW	6
425	MI	SP5A2D	IC Steam Line DP Switch 5A2	PRESS SW	6
426	MI	SP5B1D	IC Steam Line DP Switch 5B1	PRESS SW	6
427	MI	SP5B2D	IC Steam Line DP Switch 5B2	PRESS SW	6
428	SG	SP255D	Train Flow Switch FIS-28-255	PRESS SW	6
429	RI	RLK37D	Actuation Relay 6K37	RELAY	6
430	CF	RLK2AD	Pump Start Relay 16K2A	RELAY	6
431	CF	RLK2BD	Pump Start Relay 16K2B	RELAY	6
432	CF	RL25AD	Relay 16K25A	RELAY	6
433	CF	RL25BD	Relay 16K25B	RELAY	6
434	CF	RL26AD	Relay 16K26A	RELAY	6
435	CF	RL26BD	Relay 16K26B	RELAY	6
436	MI	RL03AD	Relay 6K3A	RELAY	6
437	MI	RL03BD	Relay 6K3B	RELAY	6
438	MI	RL04AD	Relay 6K4A	RELAY	6
439	MI	RL04BD	Relay 6K4B	RELAY	6
440	MI	RL05AD	Relay 6K5A	RELAY	6
441	MI	RL05BD	Relay 6K5B	RELAY	6
442	MI	RL06AD	Relay 6K6A	RELAY	6
443	MI	RL06BD	Relay 6K6B	RELAY	6
444	MI	RL07AD	Relay 6K7A	RELAY	6
445	MI	RL07BD	Relay 6K7B	RELAY	6
446	MI	RL08AD	Relay 6K8B	RELAY	6
447	MI	RL08BD	Relay 6K8B	RELAY	6
448	ST	AV235T	Air operated valve V-2-235	AOV	7
449	MU	AV034T	Air-Operated Valve V-11-34	AOV	7
450	MU	AV036T	Air-Operated Valve V-11-36	AOV	7
451	PI	AV202T	Air-Operated Valve V-22-2	AOV	7
452	PI	AV778T	Air-Operated Valve V-22-28	AOV	7
453	PI	AV779T	Air-Operated Valve V-22-29	AOV	7
454	PI	AV313T	Air-Operated Valve V-23-13	AOV	7
455	PI	AV314T	Air-Operated Valve V-23-14	AOV	7
456	PI	AV315T	Air-Operated Valve V-23-15	AOV	7
457	PI	AV316T	Air-Operated Valve V-23-16	AOV	7
458	PI	AV317T	Air-Operated Valve V-23-17	AOV	7
459	PI	AV318T	Air-Operated Valve V-23-18	AOV	7
460	PI	AV319T	Air-Operated Valve V-23-19	AOV	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
461	PI	AV320T	Air-Operated Valve V-23-20	AOV	7
462	PI	AV321T	Air-Operated Valve V-23-21	AOV	7
463	PI	AV322T	Air-Operated Valve V-23-22	AOV	7
464	PI	AV817T	Air-Operated Valve V-28-17	AOV	7
465	RI	AV011T	CRD Rebuild Area&Gen Area Sup V-28-11	AOV	7
466	RI	AV012T	CRD Rebuild Area&Gen Area Sup V-28-12	AOV	7
467	FW	AV732T	Feed regulating valve AOV-2-732	AOV	7
468	RF	AV732C	Feed regulating valve AOV-2-732 (A)	AOV	7
469	FW	AV733T	Feed regulating valve AOV-2-733	AOV	7
470	RF	AV733C	Feed regulating valve AOV-2-733 (B)	AOV	7
471	FW	AV734T	Feed regulating valve AOV-2-734	AOV	7
472	RF	AV734C	Feed regulating valve AOV-2-734 (C)	AOV	7
473	CS	AV093T	Min. Flow Valve V-20-93	AOV	7
474	CS	AV094T	Min. Flow Valve V-20-94	AOV	7
475	CS	AV092T	Min.Flow Valve V-20-92	AOV	7
476	RI	AV021T	RB General Area Exhaust Valve V-28-21	AOV	7
477	RI	AV022T	RB General Area Exhaust Valve V-28-22	AOV	7
478	RI	AV010T	RB General Area Supply Valve V-28-10	AOV	7
479	RI	AV013T	RB General Area Supply Valve V-28-13	AOV	7
480	RI	AV014T	RB General Area Supply Valve V-28-14	AOV	7
481	RI	AV015T	RB General Area Supply Valve V-28-15	AOV	7
482	RI	AV016T	RB General Area Supply Valve V-28-16	AOV	7
483	RI	AV003T	RB General Area Supply Valve V-28-3	AOV	7
484	RI	AV036T	RB General Area Supply Valve V-28-36	AOV	7
485	RI	AV037T	RB General Area Supply Valve V-28-37	AOV	7
486	RI	AV004T	RB General Area Supply Valve V-28-4	AOV	7
487	RI	AV005T	RB General Area Supply Valve V-28-5	AOV	7
488	RI	AV006T	RB General Area Supply Valve V-28-6	AOV	7
489	RI	AV007T	RB General Area Supply Valve V-28-7	AOV	7
490	RI	AV008T	RB General Area Supply Valve V-28-8	AOV	7
491	RI	AV009T	RB General Area Supply Valve V-28-9	AOV	7
492	SG	AV823T	Train Inlet Valve V-28-23	AOV	7
493	SG	AV827T	Train Inlet Valve V-28-27	AOV	7
494	SG	AV826T	Train Outlet Valve V-28-26	AOV	7
495	SG	AV830T	Train Outlet Valve V-28-30	AOV	7
496	DB	BTDCBR	125V DC Battery B	BATTERY	7
497	DC	BTDCCR	125V DC Battery C	BATTERY	7
498	EE	BTDB1D	Diesel Generator Battery #1	BATTERY	7
499	EE	BTDG2D	Diesel Generator Battery #2	BATTERY	7
500	CS	CV053L	Booster Discharge CV V-20-53	CHECK VLV	7
501	CS	CV054D	Booster Discharge CV V-20-54	CHECK VLV	7
502	CS	CV055L	Booster Discharge CV V-20-55	CHECK VLV	7
503	CS	CV055P	Booster Discharge CV V-20-55	CHECK VLV	7
504	CT	CV012P	Check Valve V-11-12	CHECK VLV	7
505	CT	CV013P	Check Valve V-11-13	CHECK VLV	7
506	MU	CV033P	Check Valve V-11-33	CHECK VLV	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
507	MU	CV035P	Check Valve V-11-35	CHECK VLV	7
508	CT	CV042P	Check Valve V-11-42	CHECK VLV	7
509	FW	CV278R	Check valve V-2-78	CHECK VLV	7
510	FW	CV279R	Check valve V-2-79	CHECK VLV	7
511	CS	CV060T	Check Valve V-20-60	CHECK VLV	7
512	CS	CV061T	Check Valve V-20-61	CHECK VLV	7
513	CS	CV088T	Check Valve V-20-88	CHECK VLV	7
514	CS	CV089T	Check Valve V-20-89	CHECK VLV	7
515	FP	CV901D	Check Valve V-9-1	CHECK VLV	7
516	FP	CV901P	Check Valve V-9-1	CHECK VLV	7
517	FP	CV902D	Check Valve V-9-2	CHECK VLV	7
518	FP	CV902P	Check Valve V-9-2	CHECK VLV	7
519	FP	CV336P	Check Valve V-9-336	CHECK VLV	7
520	CP	CV275T	Discharge check valve V-2-75	CHECK VLV	7
521	CP	CV276T	Discharge check valve V-2-76	CHECK VLV	7
522	CP	CV277T	Discharge check valve V-2-77	CHECK VLV	7
523	CS	CV052P	Discharge Check Valve V-20-52	CHECK VLV	7
524	CC	CV065D	Discharge Check Valve V-3-65	CHECK VLV	7
525	CC	CV065L	Discharge Check Valve V-3-65	CHECK VLV	7
526	CC	CV066D	Discharge Check Valve V-3-66	CHECK VLV	7
527	CC	CV067L	ESW Pump Discharge CV V-3-67	CHECK VLV	7
528	CC	CV068L	ESW Pump Discharge CV V-3-68	CHECK VLV	7
529	SW	CV062L	Pump Discharge Check Value V-3-62	CHECK VLV	7
530	SW	CV062D	Pump Discharge Check Valve V-3-62	CHECK VLV	7
531	SW	CV062P	Pump Discharge Check Valve V-3-62	CHECK VLV	7
532	SW	CV063D	Pump Discharge Check Valve V-3-63	CHECK VLV	7
533	SW	CV063L	Pump Discharge Check Valve V-3-63	CHECK VLV	7
534	TB	CV016L	Pump Discharge Check Valve V-5-16	CHECK VLV	7
535	TB	CV016T	Pump Discharge Check Valve V-5-16	CHECK VLV	7
536	TB	CV016T	Pump Discharge Check Valve V-5-16	CHECK VLV	7
537	TB	CV017L	Pump Discharge Check Valve V-5-17	CHECK VLV	7
538	TB	CV017D	Pump Discharge Check Valve V-5-17	CHECK VLV	7
539	TB	CV017T	Pump Discharge Check Valve V-5-17	CHECK VLV	7
540	TB	CV018D	Pump Discharge Check Valve V-5-18	CHECK VLV	7
541	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	7
542	TB	CV018T	Pump Discharge Check Valve V-5-18	CHECK VLV	7
543	CS	CV016P	Pump Discharge CV V-20-16	CHECK VLV	7
544	CS	CV022P	Pump Discharge CV V-20-22	CHECK VLV	7
545	CS	CV022R	Pump Discharge CV V-20-22	CHECK VLV	7
546	CS	CV008P	Pump Discharge CV V-20-8	CHECK VLV	7
547	CS	CV009P	Pump Discharge CV V-20-9	CHECK VLV	7
548	CC	CV010L	Pump Discharge CV V-21-10	CHECK VLV	7
549	CC	CV002L	Pump Discharge CV V-21-2	CHECK VLV	7
550	CC	CV004D	Pump Discharge CV V-21-4	CHECK VLV	7
551	CC	CV008L	Pump Discharge CV V-21-8	CHECK VLV	7
552	CC	CV002D	Pump Discharge Valve V-21-2	CHECK VLV	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
553	CC	CV004L	Pump Discharge Valve V-21-4	CHECK VLV	7
554	CC	CV008D	Pump Discharge Valve V-21-8	CHECK VLV	7
555	CS	CV150P	Testable CV NZ02A (V-20-150)	CHECK VLV	7
556	CS	CV151P	Testable CV NZ02B (V-20-151)	CHECK VLV	7
557	CS	CV152P	Testable CV NZ02C (V-20-152)	CHECK VLV	7
558	CS	CV153P	Testable CV NZ02D (V-20-153)	CHECK VLV	7
559	CD	CV010D	V-15-10 Stop Check Valve	CHECK VLV	7
560	CD	CV010T	V-15-10 Stop Check Valve	CHECK VLV	7
561	CD	CV007L	V-15-7 Stop Check Valve	CHECK VLV	7
562	CD	CV007D	V-15-7 Stop Check Valve	CHECK VLV	7
563	CD	CV007T	V-15-7 Stop Check Valve	CHECK VLV	7
564	BI	CV916T	V-19-16 Common Line Check Valve	CHECK VLV	7
565	BI	CV920T	V-19-20 Common Line Check Valve	CHECK VLV	7
566	BI	CV937T	V-19-37 Pump A Discharge Check Valve	CHECK VLV	7
567	BI	CV938T	V-19-38 Pump B Discharge Check Valve	CHECK VLV	7
568	EC	CB1A1T	Breaker 1A1M	CKT BKR	7
569	ED	CBB1MT	Breaker 1B1M from Transformer to Bus 1B1	CKT BKR	7
570	EA,EB,E	CBS1AT	Breaker S1A	CKT BKR	7
571	EA,EB,E	CBS1BT	Breaker S1B	CKT BKR	7
572	IA	CP011S	Air Compressor 1-1	COMPRESS	7
573	IA	CP012S	Air Compressor 1-2	COMPRESS	7
574	BV	EPR01D	Electrical Pressure Regulator	CONTROL	7
575	TT	EPR01T	Electrical Pressure Regulator	CONTROL	7
576	EC	DG001S	Emergency Diesel Generator #1	DIESEL	7
577	EE	DG001R	Emergency Diesel Generator #1	DIESEL	7
578	EE	DG001S	Emergency Diesel Generator #1	DIESEL	7
579	ED	DG002S	Emergency Diesel Generator #2	DIESEL	7
580	EE	DG002R	Emergency Diesel Generator #2	DIESEL	7
581	EE	DG002S	Emergency Diesel Generator #2	DIESEL	7
582	PI	AV701T	Air-Operated Valve V-22-1	DUPL	7
583	PI	AV818T	Air-Operated Valve V-28-18	DUPL	7
584	CS	MV015T	Parallel Isol. MOV V-20-15	DUPL	7
585	EC	FN008S	Alternate Exhaust Fan FN-56-008	FAN	7
586	SG	FNTNAS	Train A Fan	FAN	7
587	SG	FNTNBS	Train B Fan	FAN	7
588	TB	HX004P	Heat Exchanger C-5-004	HEAT EX	7
589	CC	HXCCAP	Heat Exchanger H-21-001A	HEAT EX	7
590	CC	HXCCAS	Heat Exchanger H-21-001A	HEAT EX	7
591	CC	HX1AB	Heat Exchanger H-21-001B	HEAT EX	7
592	CC	HX1AB	Heat Exchanger H-21-001B	HEAT EX	7
593	CC	HXCCCP	Heat Exchanger H-21-001C	HEAT EX	7
594	CC	HXCCCS	Heat Exchanger H-21-001C	HEAT EX	7
595	CC	HXCCDP	Heat Exchanger H-21-001D	HEAT EX	7
596	CC	HXCCDS	Heat Exchanger H-21-001D	HEAT EX	7
597	TB	HX003P	Heat Exchanger(C-5-003)	HEAT EX	7
598	TB	HV058T	Circ. Water to Heat Exchanger Iso.V-3-58	HOV XFER	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
599	IA	HV048T	Manual Valve V-9-48	HOV XFER	7
600	IA	HV074T	Manual Valve V-9-74	HOV XFER	7
601	TB	HV059T	Serv. Water to Heat Exchanger Iso.V-3-59	HOV XFER	7
602	IA	HV058T	TBCCW Inlet Manual Valve V-5-58	HOV XFER	7
603	MI	MV035T	Condensate Isolation Valve MOV-14-35	MOV	7
604	MI	MV037T	Condensate Isolation Valve MOV-14-37	MOV	7
605	MI	MV034T	Condensate Return Iso. Valve MOV-14-34	MOV	7
606	MI	MV036T	Condensate Return Iso. Valve MOV-14-36	MOV	7
607	CC	MV011D	Drywell Header Inlet V-21-11	MOV	7
608	CF	MV211D	Drywell Header Inlet V-21-11	MOV	7
609	CC	MV005T	Drywell Header Inlet V-21-5	MOV	7
610	CC	MV005T	Drywell Header Inlet V-21-5	MOV	7
611	CC	MV013T	Dynamic Test (Torus) Return	MOV	7
612	CS	MV021T	Parallel Isol. MOV V-20-21	MOV	7
613	CS	MV040T	Parallel Isol. MOV V-20-40	MOV	7
614	CS	MV041T	Parallel Isol. MOV V-20-41	MOV	7
615	MI	MV032T	Steam Isolation Valve MOV-14-32	MOV	7
616	MI	MV033T	Steam Isolation Valve MOV-14-33	MOV	7
617	MI	MV030T	Steam Supply Isolation Valve MOV-14-30	MOV	7
618	MI	MV031T	Steam Supply Isolation Valve MOV-14-31	MOV	7
619	BI	MV161T	V-16-1 RWCU Inboard Isolation	MOV	7
620	IC	MV034T	Condensate Isolation valve MOV-14-34	MOV XFER	7
621	IC	MV035T	Condensate Isolation valve MOV-14-35	MOV XFER	7
622	BI	MV164T	Condensate Isolation valve MOV-14-35	MOV XFER	7
623	CC	MV011T	Drywell Header Inlet V-21-11	MOV XFER	7
624	CC	MV013T	Dynamic Test (Torus) Return	MOV XFER	7
625	FW	MV210T	FW train A isol MOV-2-10	MOV XFER	7
626	FW	MV211T	FW train B isol MOV-2-11	MOV XFER	7
627	FW	MV212T	FW train C isol MOV-2-12	MOV XFER	7
628	ME	NV03AT	Inboard MSIV NS3A	MSIV	7
629	MS	NV03AD	Inboard MSIV NS3A	MSIV	7
630	MS	NV03AT	Inboard MSIV NS3A	MSIV	7
631	ME	NV03BT	Inboard MSIV NS3B	MSIV	7
632	MS	NV03BD	Inboard MSIV NS3B	MSIV	7
633	MS	NV03BT	Inboard MSIV NS3B	MSIV	7
634	ME	NV04AT	Outboard MSIV NS4A	MSIV	7
635	MS	NV04AD	Outboard MSIV NS4A	MSIV	7
636	MS	NV04AT	Outboard MSIV NS4A	MSIV	7
637	ME	NV04BT	Outboard MSIV NS4B	MSIV	7
638	MS	NV04BD	Outboard MSIV NS4B	MSIV	7
639	MS	NV04BT	Outboard MSIV NS4B	MSIV	7
640	IA	OPIA4F	Operator Action	OP ACT	7
641	AD	ZHEAD3	Operator Actuates ADS	OP ACT	7
642	AD	SL18CD	Mech Switch RE18C (Triple low)	PRESS SW	7
643	AD	SL18DD	Mech Switch RE18C (Triple low)	PRESS SW	7
644	CS	PM03AS	Booster Pump NZ03A	PUMP	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
645	CS	PM03BS	Booster Pump NZ03B	PUMP	7
646	CS	PM03CS	Booster Pump NZ03C	PUMP	7
647	CS	PM03DS	Booster Pump NZ03D	PUMP	7
648	CT	PM002S	Condensate Transfer Pump 1-2	PUMP	7
649	CS	PM01AS	Core Spray Pump NZ01A	PUMP	7
650	CS	PM01BS	Core Spray Pump NZ01B	PUMP	7
651	CC	PM52A	ESW Pump 52A (P-3-003A)	PUMP	7
652	CC	PM52B	ESW Pump 52B (P-3-003B)	PUMP	7
653	CC	PM52C	ESW Pump 52C (P-3-003C)	PUMP	7
654	CC	PM52D	ESW Pump 52D (P-3-003D)	PUMP	7
655	FP	PD001S	Fire Pump 1	PUMP	7
656	FP	PD002S	Fire Pump 2	PUMP	7
657	EC	PM01AS	Fuel Oil Transfer Pump 1A	PUMP	7
658	EC	PM01BS	Fuel Oil Transfer Pump 1B	PUMP	7
659	ED	PM02AS	Fuel Oil Transfer Pump 2A	PUMP	7
660	ED	PM02BS	Fuel Oil Transfer Pump 2B	PUMP	7
661	CD	PM08AS	NC08A CRD Pump	PUMP	7
662	CD	PM08BR	NC08B CRD Pump	PUMP	7
663	BI	PM91AS	P-19-001A SLC Pump A	PUMP	7
664	BI	PM91BS	P-19-001B SLC Pump B	PUMP	7
665	CC	PM51A	Pump 51A (1-1)	PUMP	7
666	CC	PM51B	Pump 51B (1-2)	PUMP	7
667	CC	PM51C	Pump 51C (1-3)	PUMP	7
668	CC	PM51D	Pump 51D (1-4)	PUMP	7
669	CS	PM01CS	Pump NZ01C	PUMP	7
670	CS	PM01DS	Pump NZ01D	PUMP	7
671	SW	PM01AS	Pump P-3-001A	PUMP	7
672	SW	PM01BS	Pump P-3-001B	PUMP	7
673	TB	PM003S	Pump P-5-003	PUMP	7
674	TB	PM003S	Pump P-5-004	PUMP	7
675	TB	PM003S	Pump P-5-005	PUMP	7
676	SD	PMR12S	RBCCW PUMP 1-2 FAILS TO START	PUMP	7
677	FP	PM003S	Redundant Fire Pump	PUMP	7
678	SD	PMSD1S	SDC PUMP A FAILS TO START	PUMP	7
679	SD	PMSDBS	SDC PUMP B FAILS TO START	PUMP	7
680	SD	PMSDCS	SDC PUMP C FAILS TO START	PUMP	7
681	PI	SV810T	Solenoid Valve V-38-10	SOV	7
682	PI	SV816T	Solenoid Valve V-38-16	SOV	7
683	PI	SV817T	Solenoid Valve V-38-17	SOV	7
684	PI	SV837T	Solenoid Valve V-38-37	SOV	7
685	PI	SV838T	Solenoid Valve V-38-38	SOV	7
686	PI	SV839T	Solenoid Valve V-38-39	SOV	7
687	PI	SV840T	Solenoid Valve V-38-40	SOV	7
688	PI	SV841T	Solenoid Valve V-38-41	SOV	7
689	PI	SV843T	Solenoid Valve V-38-43	SOV	7
690	PI	SV8445	Solenoid Valve V-38-44	SOV	7

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
691	PI	SV846T	Solenoid Valve V-38-46	SOV	7
692	PI	SV809T	Solenoid Valve V-38-9	SOV	7
693	IA	SV054T	TBCCW Inlet Solenoid Valve V-5-54	SOV	7
694	IA	SV059T	TBCCW Inlet Solenoid Valve V-5-59	SOV	7
695	BV	BV30AD	Turbine Bypass Valve V-1-130A	TBV	7
696	BV	BV30BD	Turbine Bypass Valve V-1-130B	TBV	7
697	BV	BV30CD	Turbine Bypass Valve V-1-130C	TBV	7
698	BV	BV30DD	Turbine Bypass Valve V-1-130D	TBV	7
699	BV	BV30ED	Turbine Bypass Valve V-1-130E	TBV	7
700	BV	BV30FD	Turbine Bypass Valve V-1-130F	TBV	7
701	BV	BV30GD	Turbine Bypass Valve V-1-130G	TBV	7
702	BV	BV30HD	Turbine Bypass Valve V-1-130H	TBV	7
703	BV	BV30ID	Turbine Bypass Valve V-1-130I	TBV	7
704	BV	BV30AD	Turbine Bypass Valves V-1-130	TBV	7
705	TT	CV001T	Turbine Control Valve CV-1	TCV	7
706	TT	CV002T	Turbine Control Valve CV-2	TCV	7
707	TT	CV003T	Turbine Control Valve CV-3	TCV	7
708	TT	CV004T	Turbine Control Valve CV-4	TCV	7
709	TT	SV001T	Turbine Stop Valve SV-1	TSV	7
710	TT	SV002T	Turbine Stop Valve SV-2	TSV	7
711	TT	SV003T	Turbine Stop Valve SV-3	TSV	7
712	TT	SV004T	Turbine Stop Valve SV-4	TSV	7
713	CC	ZHECC6	Actuate Spray during ATWS	OP ACT	A
714	CC	ZHECC5	Alignment to spray containment	OP ACT	A
715	SW	ZHESW1	Manual pump start by operator	OP ACT	A
716	RS	ZHERS4	Manual Scram inititiated after logic failure	OP ACT	A
717	RS	ZHERS3	Manual Scram inititiated after TT failure	OP ACT	A
718	IC	ZHEIC4	Op. actuates IC after logic failure	OP ACT	A
719	RP	ZHERP2	Oper trips recirc pumps & actuates BI	OP ACT	A
720	FP	OP001F	Operator Action	OP ACT	A
721	AD	ZHEAD3	Operator Actuates ADS	OP ACT	A
722	AD	ZHEAD4	Operator Actuates ADS	OP ACT	A
723	BI	ZHEBI1	Operator actuates liquid poison	OP ACT	A
724	BI	ZHEBI2	Operator actuates liquid poison	OP ACT	A
725	BI	ZHEBI3	Operator actuates liquid poison	OP ACT	A
726	BI	ZHEBI4	Operator actuates liquid poison	OP ACT	A
727	RI	ZHERI3	Operator Actuates RB Iso. on Relay Failure	OP ACT	A
728	CS	ZHECS5	Operator aligns fire protection to CS	OP ACT	A
729	XB	OP001F	Operator aligns standby charger	OP ACT	A
730	XC	OP001F	Operator aligns standby charger	OP ACT	A
731	MU	ZHEMU1	Operator aligns system	OP ACT	A
732	MU	ZHEMU2	Operator aligns system	OP ACT	A
733	ME	ZHEME2	Operator Closes MSIVs After RL Failure	OP ACT	A
734	MS	ZHEME2	Operator Closes MSIVs After RL Failure	OP ACT	A
735	RF	ZHERF1	Operator controls level	OP ACT	A
736	SD	ZHESD1	OPERATOR FAILS TO ALIGN SYSTEM	OP ACT	A

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
737	TT	ZHETT3	Operator Fails To Trip Turbine	OP ACT	A
738	PI	ZHEP12	Operator isolates containment after logic failure	OP ACT	A
739	CS	ZHECS4	Operator manually starts core spray	OP ACT	A
740	IA	OPIA3F	Operator Recovers from LOSP	OP ACT	A
741	IA	OPIA4F	Operator Recovers from Loss of TBCCW	OP ACT	A
742	RF	ZHE0F1	Operator recovers level control	OP ACT	A
743	CC	ZHECC3	Operator Starts Cool Torus Cooling	OP ACT	A
744	CD	ZHECD1	Operator Starts Injection	OP ACT	A
745	CD	ZHECD4	Operator Starts Injection	OP ACT	A
746	CC	ZHECC4	Operator Starts Pump 51B(1-2)	OP ACT	A
747	CC	ZHECC4	Operator Starts Pump 51D	OP ACT	A
748	CC	ZHECC4	Operator Starts Pump 52B	OP ACT	A
749	CC	ZHECC4	Operator Starts Pump 52D	OP ACT	A
750	OV	ZHEOV1	Operator vents - no core damage	OP ACT	A
751	OV	ZHEOV2	Operator vents with core damage	OP ACT	A
752	TB	ZHETB5	Pumps Restart. If Loss of Offsite Power	OP ACT	A
753	TB	ZHETB4	SW Aligned to Hx on loss of Circ. Water	OP ACT	A
754	DB	BTDCBD	125V DC Battery B	BATTERY	B
755	DC	BTDCCD	125V DC Battery C	BATTERY	B
756	DB	BSDCBR	125V DC Bus B	BUS	B
757	DC	BSDCCR	125V DC Bus C	BUS	B
758	EA	BS01AR	4,160V AC Bus 1A	BUS	B
759	EB	BS01BR	4,160V AC Bus 1B	BUS	B
760	EC	BS01CR	4,160V AC Bus 1C	BUS	B
[illegible]	ED	BS01DR	4,160V DC Bus 1D	BUS	B
[illegible]	EC	XXEAFF	4160V AC Bus 1A Failed	BUS	B
763	ED	XXEBFF	4160V AC Bus 1B Failed	BUS	B
764	EC	BS1A1R	480V AC Bus 1A1	BUS	B
765	EC	BS1A2R	480V AC Bus 1A2	BUS	B
766	ED	BS1B1R	480V AC Bus 1B1	BUS	B
767	ED	BS1B2R	480V AC Bus 1B2	BUS	B
768	FP	BS1E1R	480V AC Bus 1E1	BUS	B
769	XB	MGDCBR	Battery Charger B Motor Generator Set	CHARGER	B
770	XC	BCDC1R	Battery Charger C1	CHARGER	B
771	XB	BCDABR	Standby Static Charger A/B	CHARGER	B
772	XC	BCDC2R	Standby Static Charger C2	CHARGER	B
773	FP	CBE17T	480V AC Feeder Breaker to 1E17	CKT BKR	B
774	FP	CB1E1T	480V AC Feeder Breaker to Bus 1E1	CKT BKR	B
775	XC,DC	CB018T	AC Circuit Breaker from 1A2	CKT BKR	B
776	XB,DB	CB009T	AC Circuit Breaker from 1B2	CKT BKR	B
777	XC	CB015T	AC Circuit Breaker from VMCC 1A2	CKT BKR	B
778	XB	CB006T	AC Circuit Breaker from VMCC 1B2	CKT BKR	B
779	EA	CB01AO	Circuit Breaker 1A	CKT BKR	B
780	EC	CB1A1D	Circuit Breaker 1A1M	CKT BKR	B
781	EC	CB1A2T	Circuit Breaker 1A2M	CKT BKR	B
782	EC	CBA2PT	Circuit Breaker 1A2P to Bus 1A2	CKT BKR	B

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
783	EB	CB01BO	Circuit Breaker 1B	CKT BKR	B
784	ED	CBB1MT	Circuit Breaker 1B2M from Transformer to Bus 1B1	CKT BKR	B
785	ED	CBB2MT	Circuit Breaker 1B2M from Transformer to Bus 1B2	CKT BKR	B
786	ED	CBB2PT	Circuit Breaker 1B2P to Bus 1B2	CKT BKR	B
787	EC	CB01CO	Circuit Breaker 1C	CKT BKR	B
788	ED	CB01DO	Circuit Breaker 1D from 4,160VAC Bus 1B	CKT BKR	B
789	EC	CBDG1C	Circuit Breaker EDG1	CKT BKR	B
790	EA	CBS1AC	Circuit Breaker S1A	CKT BKR	B
791	EB	CBS1BC	Circuit Breaker S1B	CKT BKR	B
792	VO,DB	CB162T	DC Circuit Breaker CB62-162 (Panel F)	CKT BKR	B
793	VO,DB	CB163T	DC Circuit Breaker CB62-163 (Panel F)	CKT BKR	B
794	VO,DB	CB069T	DC Circuit Breaker CB62-69 (Panel D)	CKT BKR	B
795	VO,DB	CB079T	DC Circuit Breaker CB62-79 (Panel D)	CKT BKR	B
796	DB	CB003T	DC Circuit Breaker to AC Switchgear	CKT BKR	B
797	DC	CB012T	DC Circuit Breaker to AC Switchgear	CKT BKR	B
798	DC	CB014T	DC Circuit Breaker to Computer Supply	CKT BKR	B
799	XB,DB	CB007T	DC Circuit Breaker to DC Bus B	CKT BKR	B
800	XB,DB	CB008T	DC Circuit Breaker to DC Bus B	CKT BKR	B
801	DB	CB001T	DC Circuit Breaker to DC Bus B	CKT BKR	B
802	XC,DC	CB016T	DC Circuit Breaker to DC Bus C	CKT BKR	B
803	XC,DC	CB017T	DC Circuit Breaker to DC Bus C	CKT BKR	B
804	DC	CB010T	DC Circuit Breaker to DC Bus C	CKT BKR	B
805	DB	CB002T	DC Circuit Breaker to MCC DC-1	CKT BKR	B
806	DC	CB011T	DC Circuit Breaker to MCC DC-2	CKT BKR	B
807	DB	CB004T	DC Circuit Breaker to Panel DC-D	CKT BKR	B
808	DC	CB013T	DC Circuit Breaker to Panel DC-F	CKT BKR	B
809	DB	CB005T	DC Ckt Bkr to Inst. Bus 3 Rotary Inverter	CKT BKR	B
810	ED	CBDG2C	EDG Output Breaker EDG2	CKT BKR	B
811	RF	CBFWAO	FWP A supply breaker	CKT BKR	B
812	RF	CBFWBO	FWP B supply breaker	CKT BKR	B
813	RF	CBFWCO	FWP C supply breaker	CKT BKR	B
814	RP	BKMGAO	M/G Set Supply Breaker A	CKT BKR	B
815	RP	BKMGBO	M/G Set Supply Breaker B	CKT BKR	B
816	RP	BKMGCO	M/G Set Supply Breaker C	CKT BKR	B
817	RP	BKMGDO	M/G Set Supply Breaker D	CKT BKR	B
818	RP	BKMGEO	M/G Set Supply Breaker E	CKT BKR	B
819	TT	EPR01D	Electrical Pressure Regulator	CONTROL	B
820	RF	SMFWAR	FRV A controller	CONTROL	B
821	RF	SMFWBR	FRV B controller	CONTROL	B
822	RF	SMFWCR	FRV C controller	CONTROL	B
823	RF	LGLLSD	Low level setdown	CONTROL	B
824	RF	SMMASR	Master RPV level controller	CONTROL	B
825	EC	BTDB1D	Diesel Generator Battery #1	DG 1	B
826	EC	DG001R	Emergency Diesel Generator #1	DG 1	B
827	EC	PM01AR	Fuel Oil Transfer Pump 1A	DG 1	B
828	EC	PM01BR	Fuel Oil Transfer Pump 1B	DG 1	B

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
829	ED	BTDG2D	Diesel Generator Battery #2	DG 2	B
830	ED	DG002R	Emergency Diesel Generator #2	DG 2	B
831	ED	PM02AR	Fuel Oil Transfer Pump 2A	DG 2	B
832	ED	PM02BR	Fuel Oil Transfer Pump 2B	DG 2	B
833	AD		Relay Panel ER-18A (Individual Relays - Note E)	ER-18A	B
834	AD		Relay Panel ER-18B (Individual Relays - Note E)	ER-18B	B
835	AD		Relay Panel ER-642-078 (Individual Relays - Note E)	ER-642-078	B
836	AD		Relay Panel ER-642-112 (Individual Relays - Note E)	ER-642-112	B
837	AD		Relay Panel ER-642-113 (Individual Relays - Note E)	ER-642-113	B
838	AD		Relay Panel ER-642-114 (Individual Relays - Note E)	ER-642-114	B
839	AD		Relay Panel ER-642-115 (Individual Relays - Note E)	ER-642-115	B
840	AD		Relay Panel ER-8A (Individual Relays - Note E)	ER-8A	B
841	AD		Relay Panel ER-8B (Individual Relays - Note E)	ER-8B	B
842	OV		Panel 11F (For Individual Components - See Note E)	Panel 11F	B
843	AD	SP40AD	Differential Pressure Switch RV40A	PRESS SW	B
844	AD	SP40BD	Differential Pressure Switch RV40B	PRESS SW	B
845	AD	SP40CD	Differential Pressure Switch RV40C	PRESS SW	B
846	AD	SP40DD	Differential Pressure Switch RV40D	PRESS SW	B
847	CC	SP15AD	Drywell Pressure Switch IP-15A	PRESS SW	B
848	CC	SP15BD	Drywell Pressure Switch IP-15B	PRESS SW	B
849	CC	SP15CD	Drywell Pressure Switch IP-15C	PRESS SW	B
850	CC	SP15DD	Drywell Pressure Switch IP-15D	PRESS SW	B
851	DP	SP46AD	Drywell Pressure Switch RV46A	PRESS SW	B
852	DP	SP46BD	Drywell Pressure Switch RV46B	PRESS SW	B
853	DP	SP46CD	Drywell Pressure Switch RV46C	PRESS SW	B
854	DP	SP46DD	Drywell Pressure Switch RV46D	PRESS SW	B
855	BI	FIL06D	FS 1L06 Flow Sensor To Isolate RWCU	PRESS SW	B
856	IC	PSCA1I	IC A isolation condensate line PS 1	PRESS SW	B
857	IC	PSCA2I	IC A isolation condensate line PS 1	PRESS SW	B
858	IC	PSSA1I	IC A isolation steamline pressure switch 1	PRESS SW	B
859	IC	PSSA2I	IC A isolation steamline pressure switch 2	PRESS SW	B
860	IC	PSCB1I	IC B isolation condensate line PS 1	PRESS SW	B
861	IC	PSCB2I	IC B isolation condensate line PS 1	PRESS SW	B
862	IC	PSSB1I	IC B isolation steamline pressure switch 1	PRESS SW	B
863	IC	PSSB2I	IC B isolation steamline pressure switch 2	PRESS SW	B
864	CC	SP17AD	Low Flow Pressure Switch PS17A	PRESS SW	B
865	CC	SP17BD	Low Flow Pressure Switch PS17B	PRESS SW	B
866	ME	SP23AD	Main Steam Low Pressure Switch RE-23A	PRESS SW	B
867	ME	SP23BD	Main Steam Low Pressure Switch RE-23B	PRESS SW	B
868	ME	SP23CD	Main Steam Low Pressure Switch RE-23C	PRESS SW	B
869	ME	SP23DD	Main Steam Low Pressure Switch RE-23D	PRESS SW	B
870	AD	SL18AD	Mech Switch RE18A (Triple low)	PRESS SW	B
871	AD	SL18BD	Mech Switch RE18C (Triple low)	PRESS SW	B
872	CS	SP40AD	Pressure Switch RV40A	PRESS SW	B
873	CS	SP40BD	Pressure Switch RV40B	PRESS SW	B
874	CS	SP40CD	Pressure Switch RV40C	PRESS SW	B

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
875	CS	SP40DD	Pressure Switch RV40D	PRESS SW	B
876	CS	SP29AD	PS RV29A - System I Booster Pumps	PRESS SW	B
877	CS	SP29BD	PS RV29B - System II Booster Pumps	PRESS SW	B
878	CS	SP29CD	PS RV29C - SystemI Booster Pumps	PRESS SW	B
879	CS	SP29DD	PS RV29D - System II Booster Pumps	PRESS SW	B
880	VO,VR	SP23AD	Reactor Pressure Switch PS-IDA83A	PRESS SW	B
881	VR	SP83AD	Reactor Pressure Switch PS-IDA83A	PRESS SW	B
882	VO,VR	SP23BD	Reactor Pressure Switch PS-IDA83B	PRESS SW	B
883	VR	SP83BD	Reactor Pressure Switch PS-IDA83B	PRESS SW	B
884	VO,VR	SP23CD	Reactor Pressure Switch PS-IDA83C	PRESS SW	B
885	VR	SP83CD	Reactor Pressure Switch PS-IDA83C	PRESS SW	B
886	VO,VR	SP23DD	Reactor Pressure Switch PS-IDA83D	PRESS SW	B
887	VR	SP83DD	Reactor Pressure Switch PS-IDA83D	PRESS SW	B
888	VO,VR	SP23ED	Reactor Pressure Switch PS-IDA83E	PRESS SW	B
889	VR	SP83ED	Reactor Pressure Switch PS-IDA83E	PRESS SW	B
890	PR	SP15AD	Reactor Pressure Switch RE15A	PRESS SW	B
891	PR	SP15BD	Reactor Pressure Switch RE15B	PRESS SW	B
892	PR	SP15CD	Reactor Pressure Switch RE15C	PRESS SW	B
893	PR	SP15DD	Reactor Pressure Switch RE15D	PRESS SW	B
894	RL	LS02AD	Reactor Water Level Switch RE02A	PRESS SW	B
895	RL	LS02BD	Reactor Water Level Switch RE02B	PRESS SW	B
896	RL	LS02CD	Reactor Water Level Switch RE02C	PRESS SW	B
897	RL	LS02DD	Reactor Water Level Switch RE02D	PRESS SW	B
898	CS	RL12AD	Booster Time Delay Relay 112A	RELAY	B
899	CS	RL12BD	Booster Time Delay Relay 112B	RELAY	B
900	CS	RL12CD	Booster Time Delay Relay 112C	RELAY	B
901	CS	RL12DD	Booster Time Delay Relay 112D	RELAY	B
902	BT	RLMT2D	Condenser Vacuum Trip No. 2	RELAY	B
903	CC	RL27AD	Low Flow Relay 16K27A	RELAY	B
904	CC	RL27BD	Low Flow Relay 16K27B	RELAY	B
905	ME	RL117D	Main Steam Low Pressure Relay 1K117	RELAY	B
906	ME	RL118D	Main Steam Low Pressure Relay 1K118	RELAY	B
907	ME	RL217D	Main Steam Low Pressure Relay 2K117	RELAY	B
908	ME	RL218D	Main Steam Low Pressure Relay 2K118	RELAY	B
909	TT	RLMT3D	Main Trip Solenoid #3	RELAY	B
910	TT	RLMT1D	Main Trip Solenoid No. 1	RELAY	B
911	MS,ME	RL173D	MSIV Actuation Relay 1K73	RELAY	B
912	ME	RL173D	MSIV Actuation Relay 1K73	RELAY	B
913	MS,ME	RL174D	MSIV Actuation Relay 1K74	RELAY	B
914	ME	RL174D	MSIV Actuation Relay 1K74	RELAY	B
915	MS,ME	RL273D	MSIV Actuation Relay 2K73	RELAY	B
916	ME	RL273D	MSIV Actuation Relay 2K73	RELAY	B
917	MS,ME	RL274D	MSIV Actuation Relay 2K74	RELAY	B
918	ME	RL274D	MSIV Actuation Relay 2K74	RELAY	B
919	CS	RL04AD	Pump Time Delay Relay 104A	RELAY	B
920	CS	RL04BD	Pump Time Delay Relay 104B	RELAY	B

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
921	CS	RL04CD	Pump Time Delay Relay 104C	RELAY	B
922	CS	RL04DD	Pump Time Delay Relay 104D	RELAY	B
923	CS	RL05AD	Relay 105A	RELAY	B
924	CS	RL05BD	Relay 105B	RELAY	B
925	CS	RL05CD	Relay 105C	RELAY	B
926	CS	RL05DD	Relay 105D	RELAY	B
927	CS	RL11AD	Relay 111A	RELAY	B
928	CS	RL11BD	Relay 111B	RELAY	B
929	CS	RL11CD	Relay 111C	RELAY	B
930	CS	RL11DD	Relay 111D	RELAY	B
931	CS	RL14AD	Relay 114A	RELAY	B
932	CS	RL14BD	Relay 114B	RELAY	B
933	CS	RL14CD	Relay 114C	RELAY	B
934	CS	RL14DD	Relay 114D	RELAY	B
935	RL	RL10AD	Relay 16K110A	RELAY	B
936	RL	RL10BD	Relay 16K110B	RELAY	B
937	RL	RL10CD	Relay 16K110C	RELAY	B
938	RL	RL10DD	Relay 16K110D	RELAY	B
939	CC	RLK2AD	Relay 16K2A	RELAY	B
940	CC	RLK6AD	Relay 16K6A	RELAY	B
941	CC	RLK6BD	Relay 16K6B	RELAY	B
942	CC	RLK8AD	Relay 16K8A	RELAY	B
943	CC	RLK8BD	Relay 16K8B	RELAY	B
944	BI	RL3K7D	Relay 3K7 in RWCU Isolation Logic	RELAY	B
945	RP	RL1OAD	Relay 6K10A	RELAY	B
946	RP	RL0AAD	Relay 6K10AA	RELAY	B
947	RP	RL1OAD	Relay 6K11A	RELAY	B
948	RP	RL1AAD	Relay 6K11AA	RELAY	B
949	RP	RL1ZAD	Relay 6K12A	RELAY	B
950	RP	RL2AAD	Relay 6K12AA	RELAY	B
951	RP	RLK9AD	Relay 6K9A	RELAY	B
952	RP	RL9AAD	Relay 6K9AA	RELAY	B
953	CC	RLTK1D	Relay TK1	RELAY	B
954	CC	RLTK3D	Relay TK3	RELAY	B
955	CC	RLK4AD	Time Delay Relay 16K4A	RELAY	B
956	CC	RLK4BD	Time Delay Relay 16K4B	RELAY	B
957	RP	TC0AID	Trip Coil A-1	RELAY	B
958	RP	TCAIID	Trip Coil A-II	RELAY	B
959	RP	TC0BID	Trip Coil B-I	RELAY	B
960	RP	TCBIID	Trip Coil B-II	RELAY	B
961	RP	TC0CID	Trip Coil C-I	RELAY	B
962	RP	TCCFFD	Trip Coil C-II	RELAY	B
963	RP	TC0DFD	Trip Coil D-I	RELAY	B
964	RP	TCDIID	Trip Coil D-II	RELAY	B
965	RP	TC0EID	Trip Coil E-I	RELAY	B
966	RP	TCEIID	Trip Coil E-II	RELAY	B

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
967	TT	SWNLSC	No Load Switch	SWITCH	B
968	FP	XR480R	34.5 kV 480V AC Transformer	XFORMR	B
969	EC	XR1A1R	4,160V/480V Transformer to Bus 1A1	XFORMR	B
970	EC	XR1A2R	4,160V/480V Transformer to Bus 1A2	XFORMR	B
971	ED	XR1B1R	4160V 480V Transformer to Bus 1B1	XFORMR	B
972	ED	XR1B2R	4160V 480V Transformer to Bus 1B2	XFORMR	B
973	EA	XR0SAR	Startup Transformer SA	XFORMR	B
974	EB	XR0SBR	Startup Transformer SB	XFORMR	B
975	ST	AV016T	Air operated valve V-2-16	AOV	C
976	OV	AV271D	Air operated valve V-27-1	AOV	C
977	OV	AV272D	Air operated valve V-27-2	AOV	C
978	OV	AV818D	Air operated valve V-28-15	AOV	C
979	OV	AV817D	Air operated valve V-28-16	AOV	C
980	MU	AV034D	Air-Operated Valve V-11-34	AOV	C
981	MU	AV036D	Air-Operated Valve V-11-36	AOV	C
982	PI	AV701L	Air-Operated Valve V-22-1	AOV	C
983	PI	AV202D	Air-Operated Valve V-22-2	AOV	C
984	PI	AV778D	Air-Operated Valve V-22-28	AOV	C
985	PI	AV779D	Air-Operated Valve V-22-29	AOV	C
986	PI	AV313D	Air-Operated Valve V-23-13	AOV	C
987	PI	AV314D	Air-Operated Valve V-23-14	AOV	C
988	PI	AV315D	Air-Operated Valve V-23-15	AOV	C
989	PI	AV316D	Air-Operated Valve V-23-16	AOV	C
990	PI	AV317D	Air-Operated Valve V-23-17	AOV	C
991	PI	AV318D	Air-Operated Valve V-23-18	AOV	C
992	PI	AV319D	Air-Operated Valve V-23-19	AOV	C
993	PI	AV320D	Air-Operated Valve V-23-20	AOV	C
994	PI	AV321D	Air-Operated Valve V-23-21	AOV	C
995	PI	AV322D	Air-Operated Valve V-23-22	AOV	C
996	PI,OV	AV701T	Air-Operated Valve V-27-1	AOV	C
997	PI,OV	AV702T	Air-Operated Valve V-27-2	AOV	C
998	PI	AV817D	Air-Operated Valve V-28-17	AOV	C
999	PI	AV818D	Air-Operated Valve V-28-18	AOV	C
1000	CP,ST	AV216T	Control valve V-2-16	AOV	C
1001	CP	AV235T	Control valve V-2-235	AOV	C
1002	CP	AV732T	Feed regulating valve AOV-2-732	AOV	C
1003	CP	AV733T	Feed regulating valve AOV-2-733	AOV	C
1004	CP	AV734T	Feed regulating valve AOV-2-734	AOV	C
1005	CS	AV093D	Min. Flow Valve V-20-93	AOV	C
1006	CS	AV094D	Min. Flow Valve V-20-94	AOV	C
1007	CS	AV092D	Min.Flow Valve V-20-92	AOV	C
1008	RI,OV	AV015D	RB General Area Supply Valve V-28-15	AOV	C
1009	RI,OV	AV016D	RB General Area Supply Valve V-28-16	AOV	C
1010	FW	AV564T	Temperature control valve AOV-5-64	AOV	C
1011	FW	AV569T	Temperature control valve AOV-5-69	AOV	C
1012	FW	AV574T	Temperature control valve AOV-5-74	AOV	C

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1013	PI	AV616T	Air-Operated Valve V-26-16	AOV XFER	C
1014	PI	AV618T	Air-Operated Valve V-26-18	AOV XFER	C
1015	PI	AV703T	Air-Operated Valve V-27-3	AOV XFER	C
1016	PI	AV704T	Air-Operated Valve V-27-4	AOV XFER	C
1017	PI	AV847T	Air-Operated Valve V-28-47	AOV XFER	C
1018	O,VR,A	EMRVAD	EMRV (NR108A) V-1-173	EMRV	C
1019	O,VR,A	EMRVBD	EMRV (NR108B) V-1-174	EMRV	C
1020	O,VR,A	EMRVCD	EMRV (NR108C) V-1-175	EMRV	C
1021	O,VR,A	EMRVDD	EMRV (NR108D) V-1-176	EMRV	C
1022	O,VR,A	EMRVED	EMRV (NR108E) V-1-177	EMRV	C
1023	IC	MV034D	Condensate Isolation valve MOV-14-34 (IC A)	MOV	C
1024	IC	MV035D	Condensate Isolation valve MOV-14-35	MOV	C
1025	SD	MV205D	Discharge Valve V-17-205 Fails to Open	MOV	C
1026	SD	MV206D	Dishcarge Valve V-17-206 Fails to Open	MOV	C
1027	SD	MV207D	DISCHARGE VALVE V-17-207 Fails to Open	MOV	C
1028	SD	MV755D	DISCHARGE VALVE V-17-55 Fails to Open	MOV	C
1029	SD	MV756D	DISCHARGE VALVE V-17-56 Fails to Open	MOV	C
1030	SD	MV757D	DISCHARGE VALVE V-17-57 Fails to Open	MOV	C
1031	CC	MV011D	Drywell Header Inlet V-21-11	MOV	C
1032	CC	MV005D	Drywell Header Inlet V-21-5	MOV	C
1033	CC	MV013D	Dynamic Test (Torus) Return V-21-13	MOV	C
1034	CC	MV017D	Dynamic Test (Torus) Return V-21-17	MOV	C
1035	TT	VTEDYD	Emergency Trip Oil Dump Valve	MOV	C
1036	FW,CP	MV027T	FW train A isol MOV-2-7	MOV	C
1037	FW,CP	MV028T	FW train B isol MOV-2-8	MOV	C
1038	FW,CP	MV029T	FW train C isol MOV-2-9	MOV	C
1039	CS	MV015D	Parallel Isol. MOV V-20-15	MOV	C
1040	CS	MV021D	Parallel Isol. MOV V-20-21	MOV	C
1041	CS	MV040D	Parallel Isol. MOV V-20-40	MOV	C
1042	CS	MV041D	Parallel Isol.MOV V-20-41	MOV	C
1043	SD	MV754D	SDC Return Isolation Valve V-17-54 Fails to Open	MOV	C
1044	SD	MV719D	SDC Supply Isolation V-17-19 Fails to Open	MOV	C
1045	SD	MV171D	Train A Suction Valve V-17-1 Fails to Open	MOV	C
1046	SD	MV172D	Train B Suction Valve V-17-2 Fails to Open	MOV	C
1047	SD	MV173D	Train C Suction Valve V-17-3 Fails to Open	MOV	C
1048	BI	MV161D	V-16-1 RWCU Inboard Isolation	MOV	C
1049	BI	MV164D	V-16-14 RWCU Outboard Isolation	MOV	C
1050	IC	MV036T	Condensate Isolation valve MOV-14-36 (IC A)	MOV XFER	C
1051	IC	MV037T	Condensate Isolation valve MOV-14-37	MOV XFER	C
1052	CS	MV012T	Discharge MOV V-20-12	MOV XFER	C
1053	CS	MV018T	Discharge MOV V-20-18	MOV XFER	C
1054	CW	MVO10T	Discharge Valve MOV V-3-10	MOV XFER	C
1055	CW	MV311T	Discharge Valve MOV V-3-11	MOV XFER	C
1056	CW	MVO38T	Discharge Valve MOV V-3-8	MOV XFER	C
1057	CW	MVO39T	Discharge Valve MOV V-3-9	MOV XFER	C
1058	CP	MV210T	FW train A isol MOV-2-10	MOV XFER	C

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1059	CP	MV027T	FW train A isol MOV-2-7	MOV XFER	C
1060	CP	MV211T	FW train B isol MOV-2-11	MOV XFER	C
1061	CP	MV028T	FW train B isol MOV-2-8	MOV XFER	C
1062	CP	MV212T	FW train C isol MOV-2-12	MOV XFER	C
1063	CP	MV029T	FW train C isol MOV-2-9	MOV XFER	C
1064	CC	MV087T	HX ESW Outlet Isol V-21-87	MOV XFER	C
1065	CC	MVO88T	HX ESW Outlet Isol V-21-88	MOV XFER	C
1066	CC	MV009T	Pump 51A(1-1) Suction Valve V-21-9	MOV XFER	C
1067	CC	MV007T	Pump 51B(1-2) Suction Valve V-21-7	MOV XFER	C
1068	CC	MVO01T	Pump 51C Suction Valve V-21-1	MOV XFER	C
1069	CC	MV003T	Pump 51D Suction Valve V-21-3	MOV XFER	C
1070	CS	MV003T	Pump Suction MOV V-20-3	MOV XFER	C
1071	CS	MV032T	Pump Suction MOV V-20-32	MOV XFER	C
1072	CS	MV033T	Core Spray Pump Suction MOV V-20-33	MOV XFER	C
1073	CS	MV004T	Core Spray Pump Suction MOV V-20-4	MOV XFER	C
1074	CP	MV021T	SJAE Isolation MOV-2-1	MOV XFER	C
1075	CP	MV022T	SJAE isolation valve MOV-2-2	MOV XFER	C
1076	CP	MV023T	SJAE isolation valve MOV-2-3	MOV XFER	C
1077	CP	MV024T	SJAE isolation valve MOV-2-4	MOV XFER	C
1078	CP	MV025T	SJAE isolation valve MOV-2-5	MOV XFER	C
1079	CP	MV026T	SJAE isolation valve MOV-2-6	MOV XFER	C
1080	IC	MV030T	Steam Isolation valve MOV-14-30 (IC A)	MOV XFER	C
1081	IC	MV031T	Steam Isolation valve MOV-14-31 (IC A)	MOV XFER	C
1082	IC	MV032T	Steam Isolation valve MOV-14-32	MOV XFER	C
1083	IC	MV033T	Steam Isolation valve MOV-14-33	MOV XFER	C
1084	BI	MV162T	V-16-2 RWCU Aux Cleanup Pump Isolation	MOV XFER	C
1085	MS,ME	NV03AD	Inboard MSIV NS3A	MSIV	C
1086	MS,ME	NV03BD	Inboard MSIV NS3B	MSIV	C
1087	MS,ME	NV04AD	Outboard MSIV NS4A	MSIV	C
1088	MS,ME	NV04BD	Outboard MSIV NS4B	MSIV	C
1089	PI	SV810D	Solenoid Valve V-38-10	SOV	C
1090	PI	SV816D	Solenoid Valve V-38-16	SOV	C
1091	PI	SV817D	Solenoid Valve V-38-17	SOV	C
1092	PI	SV837D	Solenoid Valve V-38-37	SOV	C
1093	PI	SV838D	Solenoid Valve V-38-38	SOV	C
1094	PI	SV839D	Solenoid Valve V-38-39	SOV	C
1095	PI	SV840D	Solenoid Valve V-38-40	SOV	C
1096	PI	SV841D	Solenoid Valve V-38-41	SOV	C
1097	PI	SV843D	Solenoid Valve V-38-43	SOV	C
1098	PI	SV844D	Solenoid Valve V-38-44	SOV	C
1099	PI	SV846D	Solenoid Valve V-38-46	SOV	C
1100	PI	SV809D	Solenoid Valve V-38-9	SOV	C
1101	OV	SV591D	Solenoid valve V-6-591	SOV	C
1102	OV	SV592D	Solenoid valve V-6-592	SOV	C
1103	OV	SV595D	Solenoid valve V-6-595	SOV	C
1104	OV	SV596D	Solenoid valve V-6-596	SOV	C

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1105	IA	SV054D	TBCCW Inlet Solenoid Valve V-5-54	SOV	C
1106	IA	SV059D	TBCCW Inlet Solenoid Valve V-5-59	SOV	C
1107	IA	SV211T	Air Dryer Four-Way Valve V-6-211	SOV XFER	C
1108	BI	EV944D	V-19-44 Squib Valve B	SQUIB	C
1109	BI	EV945D	V-19-45 Squib Valve A	SQUIB	C
1110	BT	BV30AD	Turbine Bypass Valve V-1-130A	TBV	C
1111	BT	BV30BD	Turbine Bypass Valve V-1-130B	TBV	C
1112	BT	BV30CD	Turbine Bypass Valve V-1-130C	TBV	C
1113	BT	BV30DD	Turbine Bypass Valve V-1-130D	TBV	C
1114	BT	BV30ED	Turbine Bypass Valve V-1-130E	TBV	C
1115	BT	BV30FD	Turbine Bypass Valve V-1-130F	TBV	C
1116	BT	BV30GD	Turbine Bypass Valve V-1-130G	TBV	C
1117	BT	BV30HD	Turbine Bypass Valve V-1-130H	TBV	C
1118	BT	BV30ID	Turbine Bypass Valve V-1-130I	TBV	C
1119	TT	CV001D	Turbine Control Valve CV-1	TCV	C
1120	TT	CV002D	Turbine Control Valve CV-2	TCV	C
1121	TT	CV003D	Turbine Control Valve CV-3	TCV	C
1122	TT	CV004D	Turbine Control Valve CV-4	TCV	C
1123	TT	SV001D	Turbine Stop Valve SV-1	TSV	C
1124	TT	SV002D	Turbine Stop Valve SV-2	TSV	C
1125	TT	SV003D	Turbine Stop Valve SV-3	TSV	C
1126	TT	SV004D	Turbine Stop Valve SV-4	TSV	C
1127	BV,TT	EPR01T	Electrical Pressure Regulator	TT	C
1128	BT	TVBBVD	Trip Oil Dump Valve	VALVE	C
1129	IA	ADOABP	Air Dryer A-B	AIR DRIER	D
1130	IA	CP011R	Air Compressor 1-1	COMPRESS	D
1131	IA	CP012R	Air Compressor 1-2	COMPRESS	D
1132	XB	FNE20R	A/B Battery Room Exhaust Fan EF-1-20	FAN	D
1133	XB	FNS20R	A/B Battery Room Supply Fan SF-1-20	FAN	D
1134	EC	FN008R	Alternate Exhaust Fan FN-56-008	FAN	D
1135	ED	FNE21R	Exhaust Fan EF 1-21	FAN	D
1136	EC	FN007R	Exhaust Fan FN-56-007	FAN	D
1137	EC	FN004R	Supply Fan FN-56-004	FAN	D
1138	ED	FNS21R	Supply Fan SF 1-21	FAN	D
1139	CS	PM03AR	Core Spray Booster Pump NZ03A	PUMP	D
1140	CS	PM03BR	Core Spray Booster Pump NZ03B	PUMP	D
1141	CS	PM03CR	Core Spray Booster Pump NZ03C	PUMP	D
1142	CS	PM03DR	Core Spray Booster Pump NZ03D	PUMP	D
1143	CW	PMO2AR	Circulating Water Pump P-3-002A	PUMP	D
1144	CW	PMO2BR	Circulating Water Pump P-3-002B	PUMP	D
1145	CW	PMO2CR	Circulating Water Pump P-3-002C	PUMP	D
1146	CW	PMO2CR	Circulating Water Pump P-3-002D	PUMP	D
1147	CP	PMCPAR	Condensate pump 1-A	PUMP	D
1148	CP	PMCPBR	Condensate pump 1-B	PUMP	D
1149	CP	PMCPCR	Condensate pump 1-C	PUMP	D
1150	CT	PM001R	Condensate Transfer Pump 1-1	PUMP	D

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1151	CT	PM002R	Condensate Transfer Pump 1-2	PUMP	D
1152	CS	PM01AR	Core Spray Pump NZ01A	PUMP	D
1153	CS	PM01BR	Core Spray Pump NZ01B	PUMP	D
1154	CC	PM52A	ESW Pump 52A (P-3-003A)	PUMP	D
1155	CC	PM52B	ESW Pump 52B (P-3-003B)	PUMP	D
1156	CC	PM52C	ESW Pump 52C (P-3-003C)	PUMP	D
1157	CC	PM52D	ESW Pump 52D (P-3-003D)	PUMP	D
1158	FW	PMFWAR	Feedwater pump A	PUMP	D
1159	FW	PMFWBR	Feedwater pump B	PUMP	D
1160	FW	PMFWCR	Feedwater pump C	PUMP	D
1161	FP	PD001R	Fire Pump 1	PUMP	D
1162	FP	PD002R	Fire Pump 2	PUMP	D
1163	CD	PM08AR	NC08A CRD Pump	PUMP	D
1164	CD	PM08BS	NC08B CRD Pump	PUMP	D
1165	BI	PM91AR	P-19-001A SLC Pump A	PUMP	D
1166	BI	PM91BR	P-19-001B SLC Pump B	PUMP	D
1167	CC	PM51A	Containment Spray Pump 51A(1-1)	PUMP	D
1168	CC	PM51B	Containment Spray Pump 51B(1-2)	PUMP	D
1169	CC	PM51C	Containment Spray Pump 51C(1-3)	PUMP	D
1170	CC	PM51D	Containment Spray Pump 51D(1-4)	PUMP	D
1171	CS	PM01CR	Core Spray Pump NZ01C	PUMP	D
1172	CS	PM01DR	Core Spray Pump NZ01D	PUMP	D
1173	SW	PM01AR	Service Water Pump P-3-001A	PUMP	D
1174	SW	PM01BR	Service Water Pump P-3-001B	PUMP	D
1175	TB	PM003R	TBCCW Pump P-5-003	PUMP	D
1176	TB	PM003R	TBCCW Pump P-5-004	PUMP	D
1177	TB	PM003R	TBCCW Pump P-5-005	PUMP	D
1178	SD	PMR11R	RBCCW PUMP 1-1 FAILS TO RUN	PUMP	D
1179	SD	PMR12R	RBCCW PUMP 1-2 FAILS TO RUN	PUMP	D
1180	FP	PM003R	Redundant Fire Pump	PUMP	D
1181	SD	PMSD1R	SDC PUMP A FAILS TO RUN	PUMP	D
1182	SD	PMSDBR	SDC PUMP B FAILS TO RUN	PUMP	D
1183	SD	PMSDCR	SDC PUMP C FAILS TO RUN	PUMP	D
1184	CS	RL02AD	Booster Pump Actuation Relay 102A	ER-18A	E
1185	CS	RL02CD	Booster Pump Actuation Relay 102C	ER-18A	E
1186	CS	RL09AD	Booster Pump Auxiliary Relay 109A	ER-18A	E
1187	CS	RL09CD	Booster Pump Auxiliary Relay 109C	ER-18A	E
1188	VO	RL16AD	DC Transfer Relay 16K216A	ER-18A	E
1189	VO	RL16CD	DC Transfer Relay 16K216C	ER-18A	E
1190	VO	RL16ED	DC Transfer Relay 16K216E	ER-18A	E
1191	CS	RL01AD	Pump Actuation Relay 101A	ER-18A	E
1192	CS	RL01CD	Pump Actuation Relay 101C	ER-18A	E
1193	CS	RL03AD	Pump Auxiliary Relay 103A	ER-18A	E
1194	CS	RL03CD	Pump Auxiliary Relay 103C	ER-18A	E
1195	DP	RL15AD	Relay 115A	ER-18A	E
1196	DP	RL5AXD	Relay 115AX	ER-18A	E

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1197	DP	RL15CD	Relay 115C	ER-18A	E
1198	DP	RL5CXD	Relay 115CX	ER-18A	E
1199	AD	RL14AD	Relay 16K114A	ER-18A	E
1200	AD	RL14CD	Relay 16K114C	ER-18A	E
1201	AD	RL01AD	Relay 16K201A	ER-18A	E
1202	AD	RL24AD	Relay 16K214A	ER-18A	E
1203	AD	RL25AD	Relay 16K215A	ER-18A	E
1204	AD	RL27AD	Relay 16K217A	ER-18A	E
1205	AD	RL27CD	Relay 16K217C	ER-18A	E
1206	AD	RL26AD	Switching Relay 16K216A	ER-18A	E
1207	AD	RL26CD	Switching Relay 16K216C	ER-18A	E
1208	AD	RL26ED	Switching Relay 16K216E	ER-18A	E
1209	AD	RL02AD	Timer Relay 16K202A	ER-18A	E
1210	CS	RL02BD	Booster Pump Actuation Relay 102B	ER-18B	E
1211	CS	RL02DD	Booster Pump Actuation Relay 102D	ER-18B	E
1212	CS	RL09BD	Booster Pump Auxiliary Relay 109B	ER-18B	E
1213	CS	RL09DD	Booster Pump Auxiliary Relay 109D	ER-18B	E
1214	VO	RL16BD	DC Transfer Relay 16K216B	ER-18B	E
1215	VO	RL16DD	DC Transfer Relay 16K216D	ER-18B	E
1216	CS	RL01BD	Pump Actuation Relay 101B	ER-18B	E
1217	CS	RL01DD	Pump Actuation Relay 101D	ER-18B	E
1218	CS	RL03BD	Pump Auxiliary Relay 103B	ER-18B	E
1219	CS	RL03DD	Pump Auxiliary Relay 103D	ER-18B	E
1220	DP	RL15BD	Relay 115B	ER-18B	E
1221	DP	RL5BXD	Relay 115BX	ER-18B	E
1222	DP	RL15DD	Relay 115D	ER-18B	E
1223	DP	RL5DXD	Relay 115DX	ER-18B	E
1224	AD	RL14BD	Relay 16K114B	ER-18B	E
1225	AD	RL14DD	Relay 16K114D	ER-18B	E
1226	AD	RL01BD	Relay 16K201B	ER-18B	E
1227	AD	RL24BD	Relay 16K214B	ER-18B	E
1228	AD	RL25BD	Relay 16K215B	ER-18B	E
1229	AD	RL27BD	Relay 16K217B	ER-18B	E
1230	AD	RL27DD	Relay 16K217D	ER-18B	E
1231	AD	RL02BD	Timer Relay 16K202B	ER-18B	E
1232	AD	RLAX1D	Relay 16K115AX1	ER-642-078	E
1233	AD	RLAX2D	Relay 16K115AX2	ER-642-078	E
1234	AD	RLBX1D	Relay 16K115BX1	ER-642-078	E
1235	AD	RLBX2D	Relay 16K115BX2	ER-642-078	E
1236	AD	RLCX1D	Relay 16K115CX1	ER-642-078	E
1237	AD	RLCX2D	Relay 16K115CX2	ER-642-078	E
1238	AD	RLDX1D	Relay 16K115DX1	ER-642-078	E
1239	AD	R-642-07	Relay 16K115DX2	ER-642-078	E
1240	AD	RLDX2D	Relay 16K115DX2	ER-642-078	E
1241	AD	R-642-11	Initiation Relay 16K115AX3	ER-642-112	E
1242	AD	RLAX3D	Initiation Relay 16K115AX3	ER-642-112	E

TABLE 4.4-1 OYSTER CREEK FIRE INDIVIDUAL PLANT EXAMINATION COMPONENT LIST					
Item No.	Top Event	OCPRA I.D.	Component Description	Component Category	Fire Note
1243	AD	R-642-11	Initiation Relay 16K115CX3	ER-642-113	E
1244	AD	RLCX3D	Initiation Relay 16K115CX3	ER-642-113	E
1245	AD	R-642-11	Initiation Relay 16K115BX3	ER-642-114	E
1246	AD	RLBX3D	Initiation Relay 16K115BX3	ER-642-114	E
1247	AD	R-642-11	Initiation Relay 16K115DX3	ER-642-115	E
1248	AD	RLDX3D	Initiation Relay 16K115DX3	ER-642-115	E
1249	AD	RL20AD	Timer Relay 16K220A	ER-8A	E
1250	AD	RLT4CD	Timer Relay 16K224C	ER-8A	E
1251	AD	RLT4ED	Timer Relay 16K224E	ER-8A	E
1252	AD	RLT5CD	Timer Relay 16K225C	ER-8A	E
1253	AD	RLT5ED	Timer Relay 16K225F	ER-8A	E
1254	CC	RLK2BD	Relay 16K2B	ER-8B	E
1255	AD	RL20BD	Timer Relay 16K220B	ER-8B	E
1256	AD	RLT4BD	Timer Relay 16K224B	ER-8B	E
1257	AD	RLT5BD	Timer Relay 16K225B	ER-8B	E
1258	OV	RL102D	Control relay 6K102	Panel 11F	E
1259	OV	RL103D	Control relay 6K103	Panel 11F	E
1260	OV	RL108D	Control relay 6K108	Panel 11F	E
1261	OV	RL109D	Control relay 6K109	Panel 11F	E
1262	OV	RL661D	Drywell vent permissive relay 6K61	Panel 11F	E
1263	OV	RL659D	Mode switch relay 6K59	Panel 11F	E
1264	OV	RL660D	Torus vent permissive relay 6K60	Panel 11F	E

Component Categories:

- AOV - Air Operated Valve
- AOV XFER - Air Operated Valve with transfers closed failure mode
- BATTERY - DC Battery (125 VDC System)
- HOV XFER - Manual Valve with transfers closed failure mode
- MAN VALVE - Manual Valve modeled in human action
- MOV XFER - Motor Operated Valve with transfers close failure mode
- MOV - Motor Operated Valve
- SOV - Solenoid operated valve
- CHECK VLV - Check Valve
- HEAT EX - Heat Exchanger
- PIPE - Pipe integrity
- DEMIN - Demineralizers
- FILTER - Filters
- STRAINER - Strainers
- SWITCH - Hand Switches
- FLOW TRANS - Flow Transmitter
- LEV SW - Level Switch
- LEV TRANS - Level Transmitter
- PRESS SW - Pressure Switch
- PRESS TRANS - Pressure Transmitter
- FAIL TRIPPED - Component fails in the tripped or actuated position
- SUCCESSFUL - Successful turbine trip assumed as a result of seismic event.
- CONTROL - Solid state control circuitry
- INSTRUMENT - Instrumentation
- CKT BKR - Circuit Breaker or Fuse
- DUPL - Duplicate Component due to dual failure modes modeled in the PRA
- CONTROL ROD - Control rod drive failure

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
Fire Zone OB-FZ-04,05			**Cable Spreading Room / Control Room**		**Location 1 and 2**	
AD	DPS-RV40C	OB-FZ-04,5	Sys. I, Booster pump disch. press. switch		X	C-4.2
AD	ER-18A	OB-FZ-04,5	Relay Panel ER-18A to ADS		X	C-4.6
AD	ER-18B	OB-FZ-04,5	Relay Panel ER-18B to ADS		X	C-4.6
AD	NR108A	OB-FZ-04,5	EMRV (NR108A) V-1-173		X	C-4.3
AD	NR108B	OB-FZ-04,5	EMRV (NR108B) V-1-174		X	C-4.3
AD	NR108C	OB-FZ-04,5	EMRV (NR108C) V-1-175		X	C-4.3
AD	NR108D	OB-FZ-04,5	EMRV (NR108D) V-1-176		X	C-4.3
AD	NR108E	OB-FZ-04,5	EMRV (NR108E) V-1-177		X	C-4.3
CC	PMP 51A	OB-FZ-04,5	Containment Spray Pump 1-1		X	C-4.5
CC	PMP 51B	OB-FZ-04,5	Containment Spray Pump 1-2		X	C-4.5
CC	PMP 51C	OB-FZ-04,5	Containment Spray Pump 1-3		X	C-4.5
CC	PMP 51D	OB-FZ-04,5	Containment Spray Pump 1-4		X	C-4.5
CC	PMP 52A	OB-FZ-04,5	ESW Pump 52A (P-3-003A)		X	C-4.6
CC	PMP 52B	OB-FZ-04,5	ESW Pump 52B (P-3-003B)		X	C-4.6
CC	PMP 52C	OB-FZ-04,5	ESW Pump 52C (P-3-003C)		X	C-4.6
CC	PMP 52D	OB-FZ-04,5	ESW Pump 52D (P-3-003D)		X	C-4.6
CC	PS-15A	OB-FZ-04,5	Low Flow Pressure Switch PS17A		X	JMM-01
CC	PS-15B	OB-FZ-04,5	Low Flow Pressure Switch PS17B		X	JMM-02
CC	PS-IP15A	OB-FZ-04,5	Drywell Pressure Switch IP-15A		X	C-4.9
CC	PS-IP15C	OB-FZ-04,5	Drywell Pressure Switch IP-15C		X	C-4.9
CC	V-21-01	OB-FZ-04,5	Containment Spray Suction Valve		X	C-4.5
CC	V-21-03	OB-FZ-04,5	Containment Spray Suction Valve		X	C-4.5
CC	V-21-05	OB-FZ-04,5	Drywell Spray Valve - System II		X	C-4.5
CC	V-21-07	OB-FZ-04,5	Pump 51B(1-2) Suction Valve V-21-7		X	C-4.5
CC	V-21-09	OB-FZ-04,5	Pump 51A(1-1) Suction Valve V-21-9		X	C-4.5
CC	V-21-13	OB-FZ-04,5	Dynamic Test (Torus) Return V-21-13		X	C-4.5
CC	V-3-87	OB-FZ-04,5	ESW HX Outlet (ESW side) - II		X	C-4.6
CC	V-3-88	OB-FZ-04,5	ESW Sys 1 HX Outlet (ESW side)		X	C-4.6
CD	NC08A	OB-FZ-04,5	CRD Pump A		X	C-4.1
CD	NC08B	OB-FZ-04,5	CRD Pump B		X	C-4.1
CP	ID11A	OB-FZ-04,5	Feed Reg Valve A		X	C-4.11
CP	ID11B	OB-FZ-04,5	Feed Reg Valve B		X	C-4.11
CP	ID11C	OB-FZ-04,5	Feed Reg Valve C		X	C-4.11
CP	ID12A	OB-FZ-04,5	Startup Reg Valve A		X	C-4.11
CP	ID12C	OB-FZ-04,5	Startup Reg Valve C		X	C-4.11
CP	V-2-10	OB-FZ-04,5	FW train A isol MOV-2-10		X	C-4.11
CP	V-2-11	OB-FZ-04,5	FW train B isol MOV-2-11		X	C-4.11
CP	V-2-12	OB-FZ-04,5	FW train C isol MOV-2-12		X	C-4.11
CS	DPS-RV40C	OB-FZ-04,5	Sys. I, Booster pump disch. press. switch		X	C-4.2
CS	ER-18A	OB-FZ-04,5	Relay Panel ER-18A to ADS		X	C-4.6
CS	ER-18B	OB-FZ-04,5	Relay Panel ER-18B to ADS		X	C-4.6
CS	NZ01A	OB-FZ-04,5	Core Spray Pump NZ01A		X	C-4.1
CS	NZ01B	OB-FZ-04,5	Core Spray Pump NZ01B		X	C-4.1
CS	NZ01C	OB-FZ-04,5	Core Spray Pump NZ01C		X	C-4.1
CS	NZ01D	OB-FZ-04,5	Core Spray Pump NZ01D		X	C-4.1
CS	NZ03A	OB-FZ-04,5	Core Spray Booster Pump NZ03A		X	C-4.1
CS	NZ03B	OB-FZ-04,5	Core Spray Booster Pump NZ03B		X	C-4.1
CS	NZ03C	OB-FZ-04,5	Core Spray Booster Pump NZ03C		X	C-4.1
CS	NZ03D	OB-FZ-04,5	Core Spray Booster Pump NZ03D		X	C-4.1
CS	V-20-03	OB-FZ-04,5	CS Pump A Suction Valve		X	C-4.1
CS	V-20-04	OB-FZ-04,5	Core Spray Pump Suction MOV V-20-4		X	C-4.1
CS	V-20-15	OB-FZ-04,5	CS I Containment Isolation		X	C-4.1
CS	V-20-21	OB-FZ-04,5	CS II Containment Isolation		X	C-4.1
CS	V-20-32	OB-FZ-04,5	CS Pump C Suction Valve		X	C-4.1
CS	V-20-33	OB-FZ-04,5	Core Spray Pump Suction MOV V-20-33		X	C-4.1

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CS	V-20-40	OB-FZ-04,5	CS I Containment Isolation		X	C-4.1
CS	V-20-41	OB-FZ-04,5	CS II Containment Isolation		X	C-4.1
CT	PMP 1-2	OB-FZ-04,5	Condensate Transfer Pump 1-2		X	C-4.5
CW	P-03-002A	OB-FZ-04,5	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	OB-FZ-04,5	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	OB-FZ-04,5	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	OB-FZ-04,5	Circulating Water Pump 1-4		X	JMM-66
DB	PNL DC-E	OB-FZ-04,5	125v DC Panel "E"	X	X	C-4.8
DP	ER-18A	OB-FZ-04,5	Relay Panel ER-18A to ADS		X	C-4.6
DP	ER-18B	OB-FZ-04,5	Relay Panel ER-18B to ADS		X	C-4.6
DP	PS-RV/46A	OB-FZ-04,5	Drywell Pressure Switch RV46A		X	C-4.3
DP	PS-RV/46B	OB-FZ-04,5	Drywell Pressure Switch RV46B		X	C-4.3
EB	SWGR 1B	OB-FZ-04,5	125v DC Dist. Ctr. B to SWGR 1B	X		C-4.6
EC	BKR 1A1M	OB-FZ-04,5	Circuit Breaker 1A1M		X	C-4.7
EC	BKR 1A1P	OB-FZ-04,5	Circuit Breaker 1A1P		X	C-4.7
EC	BKR 1A2M	OB-FZ-04,5	Circuit Breaker 1A2M		X	C-4.7
EC	BKR 1A2P	OB-FZ-04,5	Circuit Breaker 1A2P		X	C-4.7
EC	BKR 1C	OB-FZ-04,5	Circuit Breaker 1C		X	C-4.7
EC	EDG-1	OB-FZ-04,5	Emergency Diesel Generator #1		X	C-4.7
EC	FN-56-004	OB-FZ-04,5	Supply Fan		X	C-4.10
EC	FN-56-007	OB-FZ-04,5	Exhaust Fan		X	C-4.10
ED	BKR 1B1M	OB-FZ-04,5	Circuit Breaker 1B1M		X	C-4.7
ED	BKR 1B2M	OB-FZ-04,5	Circuit Breaker 1B2M		X	C-4.7
ED	BKR 1B2P	OB-FZ-04,5	Circuit Breaker 1B2P		X	C-4.7
ED	BKR 1D	OB-FZ-04,5	Circuit Breaker 1D		X	C-4.7
ED	EDG-2	OB-FZ-04,5	125v DC Dist. Ctr. B to EDG2 SWGR		X	C-4.6
ED	EDG-2	OB-FZ-04,5	Emergency Diesel Generator #2		X	C-4.7
ED	SF/EF-1-21	OB-FZ-04,5	Supply Fan		X	C-4.10
FP	DFP 1-1	OB-FZ-04,5	Diesel Driven Fire Pump 1-1		X	C-4.10
FW	PMP 1A	OB-FZ-04,5	Feedwater Pump A		X	C-4.11
FW	PMP 1B	OB-FZ-04,5	Feedwater Pump B		X	C-4.11
FW	PMP 1C	OB-FZ-04,5	Feedwater Pump C		X	C-4.11
IC	V-14-30	OB-FZ-04,5	IC A Steam Line Valve (AC)		X	C-4.2
IC	V-14-31	OB-FZ-04,5	IC A Steam Line Valve (DC)		X	C-4.2
IC	V-14-32	OB-FZ-04,5	IC B Steam Line Valve (AC)		X	C-4.2
IC	V-14-33	OB-FZ-04,5	IC B Steam Line Valve (DC)		X	C-4.2
IC	V-14-34	OB-FZ-04,5	IC A Condensate Valve (DC)		X	C-4.2
IC	V-14-35	OB-FZ-04,5	IC B Condensate Valve (DC)		X	C-4.2
IC	V-14-36	OB-FZ-04,5	A IC condensate return valve (AC)		X	C-4.2
IC	V-14-37	OB-FZ-04,5	B IC condensate return valve (AC)		X	C-4.2
IC	IC A Isol	OB-FZ-04,5	IC A isolation logic cable 63-361		X	B-3.7
IC	IC B Isol	OB-FZ-04,5	IC B isolation logic cable 63-361		X	B-3.7
ME	1K073	OB-FZ-04,5	MSIV Actuation Relay 1K73	X		OCP
ME	1K074	OB-FZ-04,5	MSIV Actuation Relay 1K74	X		OCP
ME	1K117	OB-FZ-04,5	Main Steam Low Pressure Relay 1K117	X		OCP
ME	1K118	OB-FZ-04,5	Main Steam Low Pressure Relay 1K118	X		OCP
ME	2K073	OB-FZ-04,5	MSIV Actuation Relay 2K73	X		OCP
ME	2K074	OB-FZ-04,5	MSIV Actuation Relay 2K74	X		OCP
ME	2K117	OB-FZ-04,5	Main Steam Low Pressure Relay 2K117	X		OCP
ME	2K118	OB-FZ-04,5	Main Steam Low Pressure Relay 2K118	X		OCP
ME	NS03A	OB-FZ-04,5	Inboard MSIVs		X	C-4.9
ME	NS03B	OB-FZ-04,5	Inboard MSIVs		X	C-4.9
ME	NS04A	OB-FZ-04,5	Outboard MSIV (V-1-9)		X	C-4.9
ME	NS04B	OB-FZ-04,5	Outboard MSIV (V-1-10)		X	C-4.9

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
MS	1K073	OB-FZ-04,5	MSIV Actuation Relay 1K73	X		OCP
MS	1K074	OB-FZ-04,5	MSIV Actuation Relay 1K74	X		OCP
MS	2K073	OB-FZ-04,5	MSIV Actuation Relay 2K73	X		OCP
MS	2K074	OB-FZ-04,5	MSIV Actuation Relay 2K74	X		OCP
MS	NS03A	OB-FZ-04,5	Inboard MSIVs		X	C-4.9
MS	NS03B	OB-FZ-04,5	Inboard MSIVs		X	C-4.9
MS	NS04A	OB-FZ-04,5	Outboard MSIV (V-1-9)		X	C-4.9
MS	NS04B	OB-FZ-04,5	Outboard MSIV (V-1-10)		X	C-4.9
MU	V-11-34	OB-FZ-04,5	Shell side makeup to B IC		X	C-4.3
MU	V-11-36	OB-FZ-04,5	Shell side makeup to A IC		X	C-4.2
OV	V-27-01	OB-FZ-04,5	Air-Operated Valve V-27-01		X	JMM-19
OV	V-27-02	OB-FZ-04,5	Air-Operated Valve V-27-02		X	JMM-20
PI	V-22-01	OB-FZ-04,5	Air-Operated Valve V-22-1		X	JMM-03
PI	V-22-02	OB-FZ-04,5	Air-Operated Valve V-22-2		X	JMM-04
PI	V-22-28	OB-FZ-04,5	Air-Operated Valve V-22-28		X	JMM-05
PI	V-22-29	OB-FZ-04,5	Air-Operated Valve V-22-29		X	JMM-06
PI	V-23-13	OB-FZ-04,5	Air-Operated Valve V-23-13		X	JMM-07
PI	V-23-14	OB-FZ-04,5	Air-Operated Valve V-23-14		X	JMM-08
PI	V-23-15	OB-FZ-04,5	Air-Operated Valve V-23-15		X	JMM-09
PI	V-23-16	OB-FZ-04,5	Air-Operated Valve V-23-16		X	JMM-10
PI	V-23-17	OB-FZ-04,5	Air-Operated Valve V-23-17		X	JMM-11
PI	V-23-18	OB-FZ-04,5	Air-Operated Valve V-23-18		X	JMM-12
PI	V-23-19	OB-FZ-04,5	Air-Operated Valve V-23-19		X	JMM-13
PI	V-23-20	OB-FZ-04,5	Air-Operated Valve V-23-20		X	JMM-14
PI	V-23-21	OB-FZ-04,5	Air-Operated Valve V-23-21		X	JMM-15
PI	V-23-22	OB-FZ-04,5	Air-Operated Valve V-23-22		X	JMM-16
PI	V-26-16	OB-FZ-04,5	Air-Operated Valve V-26-16		X	JMM-17
PI	V-26-18	OB-FZ-04,5	Air-Operated Valve V-26-18		X	JMM-18
PI	V-27-03	OB-FZ-04,5	Air-Operated Valve V-27-03		X	JMM-21
PI	V-27-04	OB-FZ-04,5	Air-Operated Valve V-27-04		X	JMM-22
PI	V-28-17	OB-FZ-04,5	Air-Operated Valve V-28-17		X	JMM-23
PI	V-28-18	OB-FZ-04,5	Air-Operated Valve V-28-18		X	JMM-24
PI	V-28-47	OB-FZ-04,5	Air-Operated Valve V-28-47		X	JMM-25
PI	V-38-43	OB-FZ-04,5	Air-Operated Valve V-38-43		X	JMM-43
PI	V-38-46	OB-FZ-04,5	Air-Operated Valve V-38-46		X	JMM-45
PR	PT-RE-15A	OB-FZ-04,5	Reactor Pressure Switch RE15A		X	JMM-46
PR	PT-RE-15B	OB-FZ-04,5	Reactor Pressure Switch RE15B		X	JMM-47
PR	PT-RE-15C	OB-FZ-04,5	Reactor Pressure Switch RE15C		X	JMM-48
PR	PT-RE-15D	OB-FZ-04,5	Reactor Pressure Switch RE15D		X	JMM-49
RF		OB-FZ-04,5	FRV A controller	X		OCP
RF		OB-FZ-04,5	FRV B controller	X		OCP
RF		OB-FZ-04,5	FRV C controller	X		OCP
RL	RE02A	OB-FZ-04,5	Reactor Water Level Switch RE02A		X	C-4.9
RL	RE02B	OB-FZ-04,5	Reactor Water Level Switch RE02B		X	C-4.9
RL	RE02C	OB-FZ-04,5	Reactor Water Level Switch RE02C		X	C-4.9
RL	RE02D	OB-FZ-04,5	Reactor Water Level Switch RE02D		X	C-4.9
RP	PMP TRIP 1	OB-FZ-04,5	Recirculation Pump Trip, Division I		X	C-4.8
RP	PMP TRIP 2	OB-FZ-04,5	Recirculation Pump Trip, Division II		X	C-4.8
SD	NU02A	OB-FZ-04,5	SDC pump A		X	B-3.15
SD	NU02B	OB-FZ-04,5	SDC pump B		X	B-3.15
SD	NU02C	OB-FZ-04,5	SDC pump C		X	ABN
SD	RBCCW Pmp 1	OB-FZ-04,5	RBCCW Pump 1-1		X	B-5.29
SD	RBCCW Pmp 2	OB-FZ-04,5	RBCCW Pump 1-2		X	B-5.29
SD	V-17-001	OB-FZ-04,5	SDC suction valve		X	ABN
SD	V-17-002	OB-FZ-04,5	SDC suction valve		X	B-3.16
SD	V-17-003	OB-FZ-04,5	SDC suction valve		X	ABN

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
SD	V-17-019	OB-FZ-04,5	SDC supply isolation valve		X	B-3.16
SD	V-17-054	OB-FZ-04,5	SDC return isolation valve		X	B-3.16
SD	V-17-056	OB-FZ-04,5	SDC discharge valve		X	B-3.16
SD	V-17-057	OB-FZ-04,5	SDC discharge valve		X	ABN
SW	PMP 1-1	OB-FZ-04,5	Service Water Pump P-3-001A		X	C-4.9
SW	PMP 1-2	OB-FZ-04,5	Service Water Pump P-3-001B		X	C-4.9
VO	ER-18A	OB-FZ-04,5	Relay Panel ER-18A to ADS		X	C-4.6
VO	ER-18B	OB-FZ-04,5	Relay Panel ER-18B to ADS		X	C-4.6
VO	NR108A	OB-FZ-04,5	EMRV (NR108A) V-1-173		X	C-4.3
VO	NR108B	OB-FZ-04,5	EMRV (NR108B) V-1-174		X	C-4.3
VO	NR108C	OB-FZ-04,5	EMRV (NR108C) V-1-175		X	C-4.3
VO	NR108D	OB-FZ-04,5	EMRV (NR108D) V-1-176		X	C-4.3
VO	NR108E	OB-FZ-04,5	EMRV (NR108E) V-1-177		X	C-4.3
VR	NR108A	OB-FZ-04,5	EMRV (NR108A) V-1-173		X	C-4.3
VR	NR108B	OB-FZ-04,5	EMRV (NR108B) V-1-174		X	C-4.3
VR	NR108C	OB-FZ-04,5	EMRV (NR108C) V-1-175		X	C-4.3
VR	NR108D	OB-FZ-04,5	EMRV (NR108D) V-1-176		X	C-4.3
VR	NR108E	OB-FZ-04,5	EMRV (NR108E) V-1-177		X	C-4.3
XB	EF-1-20	OB-FZ-04,5	Exhaust Fan		X	C-4.10
XB	SF-1-20	OB-FZ-04,5	Supply Fan		X	C-4.10
Fire Zone OB-FZ-05			**Control Room (46'6")**			**Location 2**
PI	V-38-09	OB-FZ-05	Air-Operated Valve V-38-09		X	JMM-27
PI	V-38-10	OB-FZ-05	Air-Operated Valve V-38-10		X	JMM-28
PI	V-38-16	OB-FZ-05	Air-Operated Valve V-38-16		X	JMM-32
PI	V-38-17	OB-FZ-05	Air-Operated Valve V-38-17		X	JMM-37
PI	V-38-37	OB-FZ-05	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	OB-FZ-05	Air-Operated Valve V-38-38		X	JMM-39
PI	V-38-39	OB-FZ-05	Air-Operated Valve V-38-39		X	JMM-40
PI	V-38-40	OB-FZ-05	Air-Operated Valve V-38-40		X	JMM-41
PI	V-38-41	OB-FZ-05	Air-Operated Valve V-38-41		X	JMM-42
PI	V-38-44	OB-FZ-05	Air-Operated Valve V-38-44		X	JMM-44
Fire Zone OB-FZ-06A			**"A" 480 VAC Switchgear Room**			**Location 2**
AD	DPS-RV40A	OB-FZ-06A	Sys. I, Booster pump disch. press. switch		X	C-6.2
AD	DPS-RV40B	OB-FZ-06A	Sys. II, Booster pump disch. press. switch		X	C-6.2
AD	DPS-RV40C	OB-FZ-06A	Sys. I, Booster pump disch. press. switch		X	C-6.2
AD	DPS-RV40D	OB-FZ-06A	Sys. II, Booster pump disch. press. switch		X	C-6.2
AD	ER-18A	OB-FZ-06A	Control Panel	X		C-6.2
AD	ER-18B	OB-FZ-06A	Control Panel	X		C-6.2
AD	ER-8A	OB-FZ-06A	Containment Spray Control Panel	X	X	C-6.4
AD	ER-8B	OB-FZ-06A	Containment Spray Control Panel	X	X	C-6.4
AD	NR108A	OB-FZ-06A	EMRV ADS control circuit		X	C-6.2
AD	NR108B	OB-FZ-06A	EMRV ADS control circuit		X	C-6.2
AD	NR108C	OB-FZ-06A	EMRV ADS control circuit		X	C-6.2
AD	NR108D	OB-FZ-06A	EMRV ADS control circuit		X	C-6.2
AD	NR108E	OB-FZ-06A	EMRV ADS control circuit		X	C-6.2
CC	ER-8B	OB-FZ-06A	Containment Spray Control Panel	X	X	C-6.4
CC	PMP 51A	OB-FZ-06A	Containment Spray Pump 1-1 power and control		X	C-6.4
CC	PMP 51B	OB-FZ-06A	Containment Spray Pump 1-2 power and control		X	C-6.4
CC	PMP 51C	OB-FZ-06A	Containment Spray Pump 1-3 power and control		X	C-6.4
CC	PMP 51D	OB-FZ-06A	Containment Spray Pump 1-4 power and control		X	C-6.4
CC	PS-15A	OB-FZ-06A	Low Flow Pressure Switch PS17A		X	JMM-01
CC	PS-IP15A	OB-FZ-06A	Drywell Pressure Switch IP-15A		X	C-6.3
CC	PS-IP15B	OB-FZ-06A	Drywell Pressure Switch IP-15B		X	C-6.3
CC	PS-IP15C	OB-FZ-06A	Drywell Pressure Switch IP-15C		X	C-6.3
CC	PS-IP15D	OB-FZ-06A	Drywell Pressure Switch IP-15D		X	C-6.3
CC	RLK2A	OB-FZ-06A	Relay 16K2A	X		OCP

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CC	RLK6A	OB-FZ-06A	Relay 16K6A	X		OCP
CC	RLK6B	OB-FZ-06A	Relay 16K6B	X		OCP
CC	RLK8A	OB-FZ-06A	Relay 16K8A	X		OCP
CC	RLK8B	OB-FZ-06A	Relay 16K8B	X		OCP
CC	V-21-01	OB-FZ-06A	Containment Spray Suction Valve		X	C-6.4
CC	V-21-03	OB-FZ-06A	Containment Spray Suction Valve		X	C-6.4
CC	V-21-05	OB-FZ-06A	Drywell Spray Valve - System II		X	C-6.4
CC	V-21-07	OB-FZ-06A	Pump 51B(1-2) Suction Valve V-21-7		X	C-6.4
CC	V-21-09	OB-FZ-06A	Pump 51A(1-1) Suction Valve V-21-9		X	C-6.4
CC	V-21-11	OB-FZ-06A	Interlock		X	C-6.4
CC	V-21-13	OB-FZ-06A	Interlock		X	C-6.4
CC	V-21-17	OB-FZ-06A	Dynamic Test (Torus) Return V-21-17		X	C-6.4
CC	V-3-88	OB-FZ-06A	ESW Sys 1 HX Outlet (ESW side)		X	C-6.4
CD	NC08A	OB-FZ-06A	CRD Pump A		X	C-6.1
CD	NC08A	OB-FZ-06A	CRD Pump A		X	C-6.1
CD	NC08B	OB-FZ-06A	CRD Pump B		X	C-6.1
CS	DPS-RV40A	OB-FZ-06A	Sys. I, Booster pump disch. press. switch		X	C-6.2
CS	DPS-RV40B	OB-FZ-06A	Sys. II, Booster pump disch. press. switch		X	C-6.2
CS	DPS-RV40C	OB-FZ-06A	Sys. I, Booster pump disch. press. switch		X	C-6.2
CS	DPS-RV40D	OB-FZ-06A	Sys. II, Booster pump disch. press. switch		X	C-6.2
CS	ER-18A	OB-FZ-06A	Control Panel	X		C-6.2
CS	ER-18B	OB-FZ-06A	Control Panel	X		C-6.2
CS	ER-8B	OB-FZ-06A	Containment Spray Control Panel	X	X	C-6.4
CS	NZ01A	OB-FZ-06A	CS pump control circuit		X	C-6.1
CS	NZ01B	OB-FZ-06A	CS pump control circuit		X	C-6.1
CS	NZ01C	OB-FZ-06A	CS pump control circuit		X	C-6.1
CS	NZ01D	OB-FZ-06A	CS pump control circuit		X	C-6.1
CS	NZ03A	OB-FZ-06A	CS booster pump control circuit		X	C-6.1
CS	NZ03B	OB-FZ-06A	CS booster pump control circuit		X	C-6.1
CS	NZ03C	OB-FZ-06A	CS booster pump control circuit		X	C-6.1
CS	NZ03D	OB-FZ-06A	CS booster pump control circuit		X	C-6.1
CS	PS-RV-29A	OB-FZ-06A	Sys. I, Discharge Pressure Switch		X	C-6.2
CS	PS-RV-29B	OB-FZ-06A	Sys. II, Discharge Pressure Switch		X	C-6.2
CS	PS-RV-29C	OB-FZ-06A	Sys. I, Discharge Pressure Switch		X	C-6.2
CS	PS-RV-29D	OB-FZ-06A	Sys. II, Discharge Pressure Switch		X	C-6.2
CS	RL04A	OB-FZ-06A	Pump Time Delay Relay 104A	X		OCP
CS	RL04B	OB-FZ-06A	Pump Time Delay Relay 104B	X		OCP
CS	RL04C	OB-FZ-06A	Pump Time Delay Relay 104C	X		OCP
CS	RL04D	OB-FZ-06A	Pump Time Delay Relay 104D	X		OCP
CS	RL05A	OB-FZ-06A	Relay 105A	X		OCP
CS	RL05B	OB-FZ-06A	Relay 105B	X		OCP
CS	RL05C	OB-FZ-06A	Relay 105C	X		OCP
CS	RL05D	OB-FZ-06A	Relay 105D	X		OCP
CS	RL11A	OB-FZ-06A	Relay 111A	X		OCP
CS	RL11B	OB-FZ-06A	Relay 111B	X		OCP
CS	RL11C	OB-FZ-06A	Relay 111C	X		OCP
CS	RL11D	OB-FZ-06A	Relay 111D	X		OCP
CS	RL12A	OB-FZ-06A	Booster Time Delay Relay 112A	X		OCP
CS	RL12B	OB-FZ-06A	Booster Time Delay Relay 112B	X		OCP
CS	RL12C	OB-FZ-06A	Booster Time Delay Relay 112C	X		OCP
CS	RL12D	OB-FZ-06A	Booster Time Delay Relay 112D	X		OCP
CS	RL14A	OB-FZ-06A	Relay 114A	X		OCP
CS	RL14B	OB-FZ-06A	Relay 114B	X		OCP
CS	RL14C	OB-FZ-06A	Relay 114C	X		OCP
CS	RL14D	OB-FZ-06A	Relay 114D	X		OCP
CS	V-20-03	OB-FZ-06A	CS Pump A Suction Valve		X	C-6.1

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CS	V-20-04	OB-FZ-06A	Core Spray Pump Suction MOV V-20-4		X	C-6.1
CS	V-20-15	OB-FZ-06A	CS I Containment Isolation		X	C-6.1
CS	V-20-21	OB-FZ-06A	CS II Containment Isolation		X	C-6.2
CS	V-20-32	OB-FZ-06A	CS Pump C Suction Valve		X	C-6.1
CS	V-20-33	OB-FZ-06A	Core Spray Pump Suction MOV V-20-33		X	C-6.1
CS	V-20-40	OB-FZ-06A	CS I Containment Isolation		X	C-6.2
CS	V-20-41	OB-FZ-06A	CS II Containment Isolation		X	C-6.2
CW	P-03-002A	OB-FZ-06A	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	OB-FZ-06A	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	OB-FZ-06A	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	OB-FZ-06A	Circulating Water Pump 1-4		X	JMM-66
DC	PNL DC-F	OB-FZ-06A	Power Panel DC-F	X		C-6.7
DP	ER-18A	OB-FZ-06A	Control Panel	X		C-6.2
DP	ER-18B	OB-FZ-06A	Control Panel	X		C-6.2
DP	PS-RV/46A	OB-FZ-06A	Drywell Pressure Switch RV46A		X	C-6.2
DP	PS-RV/46B	OB-FZ-06A	Drywell Pressure Switch RV46B		X	C-6.3
DP	PS-RV/46C	OB-FZ-06A	Drywell Pressure Switch RV46C		X	C-6.3
DP	PS-RV/46D	OB-FZ-06A	Drywell Pressure Switch RV46D		X	C-6.3
EC	EDG-1	OB-FZ-06A	Emergency Diesel Generator 1		X	C-6.7
EC	FN-56-004	OB-FZ-06A	Supply Fan		X	C-6.8
EC	FN-56-007	OB-FZ-06A	Exhaust Fan		X	C-6.8
EC	USS 1A2	OB-FZ-06A	460v Unit Substation 1A2	X	X	C-6.4
ED	EDG-2	OB-FZ-06A	Emergency Diesel Generator 2 Idling/Start Circuit		X	C-6.7
IC	V-14-30	OB-FZ-06A	IC A Steam Line Valve (AC)		X	C-6.2
IC	V-14-36	OB-FZ-06A	A IC condensate return valve (AC)		X	C-6.2
IC	IC A Isol	OB-FZ-06A	IC A isolation signal (63-359)		X	B-3.7
OV	V-27-01	OB-FZ-06A	Air-Operated Valve V-27-01		X	JMM-19
OV	V-27-02	OB-FZ-06A	Air-Operated Valve V-27-02		X	JMM-20
PI	V-22-01	OB-FZ-06A	Air-Operated Valve V-22-1		X	JMM-03
PI	V-22-02	OB-FZ-06A	Air-Operated Valve V-22-2		X	JMM-04
PI	V-22-28	OB-FZ-06A	Air-Operated Valve V-22-28		X	JMM-05
PI	V-22-29	OB-FZ-06A	Air-Operated Valve V-22-29		X	JMM-06
PI	V-23-21	OB-FZ-06A	Air-Operated Valve V-23-21		X	JMM-15
PI	V-23-22	OB-FZ-06A	Air-Operated Valve V-23-22		X	JMM-16
PI	V-28-17	OB-FZ-06A	Air-Operated Valve V-28-17		X	JMM-23
PI	V-28-18	OB-FZ-06A	Air-Operated Valve V-28-18		X	JMM-24
PI	V-28-47	OB-FZ-06A	Air-Operated Valve V-28-47		X	JMM-25
RL	16K110A	OB-FZ-06A	Relay 16K110A	X		OCP
RL	16K110B	OB-FZ-06A	Relay 16K110B	X		OCP
RL	16K110C	OB-FZ-06A	Relay 16K110C	X		OCP
RL	16K110D	OB-FZ-06A	Relay 16K110D	X		OCP
SD	NU02A	OB-FZ-06A	SDC pump A		X	B-3.15
SD	NU02B	OB-FZ-06A	SDC pump B		X	B-3.15
SD	NU02C	OB-FZ-06A	SDC pump C		X	ABN
SD	RBCCW Pump 1	OB-FZ-06A	RBCCW Pump 1-1		X	B-5.29
SD	RBCCW Pump 2	OB-FZ-06A	RBCCW Pump 1-2		X	B-5.29
SD	V-17-019	OB-FZ-06A	SDC supply isolation valve		X	B-3.16
SD	V-17-054	OB-FZ-06A	SDC return isolation valve		X	B-3.16
VO	ER-18A	OB-FZ-06A	Control Panel	X		C-6.2
VO	ER-18B	OB-FZ-06A	Control Panel	X		C-6.2
VO	NR108A	OB-FZ-06A	EMRV (NR108A) V-1-173		X	C-6.2
VO	NR108B	OB-FZ-06A	EMRV (NR108B) V-1-174		X	C-6.2
VO	NR108C	OB-FZ-06A	EMRV (NR108C) V-1-175		X	C-6.2
VO	NR108D	OB-FZ-06A	EMRV (NR108D) V-1-176		X	C-6.2
VO	NR108E	OB-FZ-06A	EMRV (NR108E) V-1-177		X	C-6.2
VO	PNL DC-F	OB-FZ-06A	Power Panel DC-F	X		C-6.7

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
VR	NR108A	OB-FZ-06A	EMRV (NR108A) V-1-173		X	C-6.2
VR	NR108B	OB-FZ-06A	EMRV (NR108B) V-1-174		X	C-6.2
VR	NR108C	OB-FZ-06A	EMRV (NR108C) V-1-175		X	C-6.2
VR	NR108D	OB-FZ-06A	EMRV (NR108D) V-1-176		X	C-6.2
VR	NR108E	OB-FZ-06A	EMRV (NR108E) V-1-177		X	C-6.2
XB	SF-1-20	OB-FZ-06A	Supply Fan		X	C-6.8
Fire Zone OB-FZ-06B			**"B" 480 VAC Switchgear Room**			**Location 4**
CC	PMP 51C	OB-FZ-06B	Containment Spray Pump 1-3 power and control		X	C-6.11
CC	PMP 51D	OB-FZ-06B	Containment Spray Pump 1-4 power and control		X	C-6.11
CD	NC08B	OB-FZ-06B	CRD Pump B		X	C-6.9
CD	NC08B	OB-FZ-06B	CRD Pump B		X	C-6.9
CS	NZ03B	OB-FZ-06B	CS booster pump power and control		X	C-6.9
CS	NZ03C	OB-FZ-06B	CS booster pump power and control		X	C-6.9
CS	PS-RV-29A	OB-FZ-06B	Sys. I, Discharge Pressure Switch		X	C-6.9
CS	PS-RV-29C	OB-FZ-06B	Sys. I, Discharge Pressure Switch		X	C-6.9
CS	V-20-03	OB-FZ-06B	CS Pump A Suction Valve		X	C-6.9
CS	V-20-32	OB-FZ-06B	CS Pump C Suction Valve		X	C-6.9
EC	FN-56-004	OB-FZ-06B	Supply Fan power circuit		X	C-6.12
EC	FN-56-007	OB-FZ-06B	Exhaust Fan power circuit		X	C-6.12
EC	FN-56-008	OB-FZ-06B	Alternate Exhaust Fan		X	ABN-53
EC	USS 1A2	OB-FZ-06B	125 VDC to USS 1A2		X	C-6.11
ED	SF-1-21	OB-FZ-06B	Supply Fan		X	C-6.12
ED	USS 1B2	OB-FZ-06B	460V Unit Substation Transformer	X	X	C-6.11
IC	V-14-32	OB-FZ-06B	IC B Steam Line Valve (AC)		X	C-6.9
IC	V-14-33	OB-FZ-06B	IC B Steam Line Valve (DC)		X	C-6.9
IC	V-14-35	OB-FZ-06B	IC B Condensate Valve (DC)		X	C-6.9
IC	V-14-37	OB-FZ-06B	B IC condensate return valve (AC)		X	C-6.10
MU	V-11-34	OB-FZ-06B	Shell side makeup to B IC		X	C-6.10
SD	NU02A	OB-FZ-06B	SDC pump B		X	B-3.15
SD	RBCCW Pump 2	OB-FZ-06B	RBCCW Pump 1-2		X	B-5.29
XB	EF-1-20	OB-FZ-06B	Exhaust Fan		X	C-6.12
XC	Batt Chg. C1	OB-FZ-06B	460V power feed to battery charger C1		X	C-6.11
Fire Zone OB-FZ-08A			**MG Set Room**			**Location 5**
AD	DPS-RV40B	OB-FZ-08A	Sys. II, Booster pump disch. press. switch		X	C-8.1
AD	DPS-RV40D	OB-FZ-08A	Sys. II, Booster pump disch. press. switch		X	C-8.1
AD	NR108A	OB-FZ-08A	EMRV (NR108A) V-1-173		X	C-8.1
AD	NR108B	OB-FZ-08A	EMRV (NR108B) V-1-174		X	C-8.2
AD	NR108C	OB-FZ-08A	EMRV (NR108C) V-1-175		X	C-8.2
AD	NR108D	OB-FZ-08A	EMRV (NR108D) V-1-176		X	C-8.2
AD	NR108E	OB-FZ-08A	EMRV (NR108E) V-1-177		X	C-8.2
CC	PMP 51C	OB-FZ-08A	Power cable		X	C-8.3
CC	PMP 51D	OB-FZ-08A	Power cable		X	C-8.3
CC	PS-15B	OB-FZ-08A	Low Flow Pressure Switch PS17B		X	JMM-02
CC	V-21-01	OB-FZ-08A	Containment Spray Suction Valve		X	C-8.3
CC	V-21-03	OB-FZ-08A	Containment Spray Suction Valve		X	C-8.3
CC	V-21-05	OB-FZ-08A	Drywell Spray Valve - System II		X	C-8.3
CC	V-21-13	OB-FZ-08A	Interlock		X	C-8.3
CC	V-3-87	OB-FZ-08A	ESW HX Outlet (ESW side) - II		X	C-8.3
CS	DPS-RV40B	OB-FZ-08A	Sys. II, Booster pump disch. press. switch		X	C-8.1
CS	DPS-RV40D	OB-FZ-08A	Differential Pressure Switch RV40D		X	C-8.1
CS	NZ03B	OB-FZ-08A	Power cable		X	C-8.1
CS	NZ03D	OB-FZ-08A	Power cable		X	C-8.1
CS	PS-RV-29B	OB-FZ-08A	PS RV29B - System II Booster Pumps		X	C-8.1
CS	PS-RV-29D	OB-FZ-08A	PS RV29D - System II Booster Pumps		X	C-8.1
CS	V-20-03	OB-FZ-08A	CS Pump A Suction Valve		X	C-8.1

Table 4.4-3 Location of Risk Significant Components and Associated Cables

Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CS	V-20-04	OB-FZ-08A	Core Spray Pump Suction MOV V-20-4		X	C-8.1
CS	V-20-21	OB-FZ-08A	CS II Containment Isolation		X	C-8.1
CS	V-20-33	OB-FZ-08A	Core Spray Pump Suction MOV V-20-33		X	C-8.1
CS	V-20-41	OB-FZ-08A	CS II Containment Isolation		X	C-8.1
DB	MCC DC-1	OB-FZ-08A	125 VDC MCC		X	C-8.3
ED	USS 1B3	OB-FZ-08A	125 VDC to USS 1B3	X		C-8.3
ED	460V SWGR 1D	OB-FZ-08A	125VDC to 460VAC Switchgear 1D	X		C-8.3
IC	V-14-31	OB-FZ-08A	IC A Steam Line Valve (DC)		X	C-8.1
IC	V-14-34	OB-FZ-08A	IC A Condensate Valve (DC)		X	C-8.1
ME	NS03A	OB-FZ-08A	Inboard MSIVs		X	C-8.3
ME	NS03B	OB-FZ-08A	Inboard MSIVs		X	C-8.3
MS	NS03A	OB-FZ-08A	Inboard MSIVs		X	C-8.3
MS	NS03B	OB-FZ-08A	Inboard MSIVs		X	C-8.3
PI	V-23-15	OB-FZ-08A	Air-Operated Valve V-23-15		X	JMM-09
PI	V-23-16	OB-FZ-08A	Air-Operated Valve V-23-16		X	JMM-10
PI	V-23-19	OB-FZ-08A	Air-Operated Valve V-23-19		X	JMM-13
PI	V-23-20	OB-FZ-08A	Air-Operated Valve V-23-20		X	JMM-14
PI	V-26-16	OB-FZ-08A	Air-Operated Valve V-26-16		X	JMM-17
PI	V-26-18	OB-FZ-08A	Air-Operated Valve V-26-18		X	JMM-18
RP	PNL RY21A	OB-FZ-08A	MG Set Control Panel Recirc. Pp NG01A	X	X	C-8.3
RP	PNL RY21B	OB-FZ-08A	MG Set Control Panel Recirc. Pp NG01B	X	X	C-8.3
RP	PNL RY21C	OB-FZ-08A	MG Set Control Panel Recirc. Pp NG01C	X	X	C-8.3
RP	PNL RY21D	OB-FZ-08A	MG Set Control Panel Recirc. Pp NG01D	X	X	C-8.3
RP	PNL RY21E	OB-FZ-08A	MG Set Control Panel Recirc. Pp NG01E	X	X	C-8.3
SD	V-17-001	OB-FZ-08A	SDC suction valve		X	ABN
SD	V-17-002	OB-FZ-08A	SDC suction valve		X	B-3.16
SD	V-17-003	OB-FZ-08A	SDC suction valve		X	ABN
SD	V-17-056	OB-FZ-08A	SDC discharge valve		X	B-3.16
SD	V-17-057	OB-FZ-08A	SDC discharge valve		X	ABN
VO	IA83A	OB-FZ-08A	Press Control for NR108A		X	C-8.1
VO	IA83B	OB-FZ-08A	Press Control for NR108B		X	C-8.2
VO	IA83C	OB-FZ-08A	Press Control for NR108C		X	C-8.2
VO	IA83D	OB-FZ-08A	Press Control for NR108D		X	C-8.2
VO	IA83E	OB-FZ-08A	Press Control for NR108E		X	C-8.2
VO	NR108A	OB-FZ-08A	EMRV (NR108A) V-1-173		X	C-8.1
VO	NR108B	OB-FZ-08A	EMRV (NR108B) V-1-174		X	C-8.2
VO	NR108C	OB-FZ-08A	EMRV (NR108C) V-1-175		X	C-8.2
VO	NR108D	OB-FZ-08A	EMRV (NR108D) V-1-176		X	C-8.2
VO	NR108E	OB-FZ-08A	EMRV (NR108E) V-1-177		X	C-8.2
VR	IA83A	OB-FZ-08A	Press Control for NR108A		X	C-8.1
VR	IA83B	OB-FZ-08A	Press Control for NR108B		X	C-8.2
VR	IA83C	OB-FZ-08A	Press Control for NR108C		X	C-8.2
VR	IA83D	OB-FZ-08A	Press Control for NR108D		X	C-8.2
VR	IA83E	OB-FZ-08A	Press Control for NR108E		X	C-8.2
VR	NR108A	OB-FZ-08A	EMRV (NR108A) V-1-173		X	C-8.1
VR	NR108B	OB-FZ-08A	EMRV (NR108B) V-1-174		X	C-8.2
VR	NR108C	OB-FZ-08A	EMRV (NR108C) V-1-175		X	C-8.2
VR	NR108D	OB-FZ-08A	EMRV (NR108D) V-1-176		X	C-8.2
VR	NR108E	OB-FZ-08A	EMRV (NR108E) V-1-177		X	C-8.2
Fire Zone OB-FZ-08B			**Mechanical Equipment Room**			**Location 5**
XB	SF-1-20	OB-FZ-08B	Supply Fan		X	C-8.4
Fire Zone OB-FZ-08C			**A & B Battery Room, Tunnel and Electrical Tray Room**			**Location 6**
AD	DPS-RV40B	OB-FZ-08C	Sys. II, Booster pump disch. press. switch		X	C-8.6
AD	DPS-RV40C	OB-FZ-08C	Sys. I, Booster pump disch. press. switch		X	C-8.6
AD	DPS-RV40D	OB-FZ-08C	Sys. II, Booster pump disch. press. switch		X	C-8.6
AD	ER-18A	OB-FZ-08C	Relay Panel ER-18A (See Note E)		X	C-8.8

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
AD	ER-18B	OB-FZ-08C	Relay Panel ER-18B (See Note E)		X	C-8.8
AD	NR108A	OB-FZ-08C	EMRV (NR108A) V-1-173		X	C-8.6
AD	NR108A	OB-FZ-08C	EMRV (NR108A) V-1-173		X	C-8.6
AD	NR108B	OB-FZ-08C	EMRV (NR108B) V-1-174		X	C-8.6
AD	NR108C	OB-FZ-08C	EMRV (NR108C) V-1-175		X	C-8.6
AD	NR108D	OB-FZ-08C	EMRV (NR108D) V-1-176		X	C-8.6
AD	NR108E	OB-FZ-08C	EMRV (NR108E) V-1-177		X	C-8.6
BI	V-16-14	OB-FZ-08C	V-16-14 RWCU Outboard Isolation		X	C-8.9
BI	V-16-2	OB-FZ-08C	V-16-2 RWCU Aux Cleanup Pump Isolation		X	C-8.9
CC	PMP 51A	OB-FZ-08C	Containment Spray Pump 51A(1-1)		X	C-8.7
CC	PMP 51B	OB-FZ-08C	Containment Spray Pump 51B(1-2)		X	C-8.7
CC	PMP 51C	OB-FZ-08C	Containment Spray Pump 51C(1-3)		X	C-8.7
CC	PMP 51D	OB-FZ-08C	Containment Spray Pump 51D(1-4)		X	C-8.7
CC	PS-15A	OB-FZ-08C	Low Flow Pressure Switch PS17A		X	JMM-01
CC	PS-15B	OB-FZ-08C	Low Flow Pressure Switch PS17B		X	JMM-02
CC	PS-IP15A	OB-FZ-08C	Drywell Pressure Switch IP-15A		X	C-8.6
CC	PS-IP15C	OB-FZ-08C	Drywell Pressure Switch IP-15C		X	C-8.6
CC	V-21-01	OB-FZ-08C	Containment Spray Suction Valve		X	C-8.7
CC	V-21-03	OB-FZ-08C	Containment Spray Suction Valve		X	C-8.7
CC	V-21-05	OB-FZ-08C	Drywell Spray Valve - System II		X	C-8.7
CC	V-21-07	OB-FZ-08C	Pump 51B(1-2) Suction Valve V-21-7		X	C-8.7
CC	V-21-11	OB-FZ-08C	Interlock only		X	C-8.7
CC	V-21-13	OB-FZ-08C	Dynamic Test (Torus) Return V-21-13		X	C-8.7
CC	V-21-17	OB-FZ-08C	Dynamic Test (Torus) Return V-21-17		X	C-8.7
CC	V-3-87	OB-FZ-08C	ESW HX Outlet (ESW side) - II		X	C-8.7
CC	V-3-88	OB-FZ-08C	ESW Sys 1 HX Outlet (ESW side)		X	C-8.7
CD	NC08A	OB-FZ-08C	CRD Pump A		X	C-8.5
CD	NC08B	OB-FZ-08C	CRD Pump B		X	C-8.5
CS	DPS-RV40B	OB-FZ-08C	Sys. II, Booster pump disch. press. switch		X	C-8.6
CS	DPS-RV40C	OB-FZ-08C	Sys. I, Booster pump disch. press. switch		X	C-8.6
CS	DPS-RV40D	OB-FZ-08C	Sys. II, Booster pump disch. press. switch		X	C-8.6
CS	ER-18A	OB-FZ-08C	Relay Panel ER-18A (See Note E)		X	C-8.8
CS	ER-18B	OB-FZ-08C	Relay Panel ER-18B (See Note E)		X	C-8.8
CS	NZ01A	OB-FZ-08C	Power and control circuits		X	C-8.5
CS	NZ01B	OB-FZ-08C	Power and control circuits		X	C-8.5
CS	NZ01C	OB-FZ-08C	Power and control circuits		X	C-8.5
CS	NZ01D	OB-FZ-08C	Power and control circuits		X	C-8.5
CS	NZ03A	OB-FZ-08C	Core Spray Booster Pump NZ03A		X	C-8.5
CS	NZ03B	OB-FZ-08C	Core Spray Booster Pump NZ03B		X	C-8.5
CS	NZ03C	OB-FZ-08C	Core Spray Booster Pump NZ03C		X	C-8.5
CS	NZ03D	OB-FZ-08C	Core Spray Booster Pump NZ03D		X	C-8.5
CS	PS-RV-29B	OB-FZ-08C	PS RV29B - System II Booster Pumps		X	C-8.6
CS	PS-RV-29D	OB-FZ-08C	PS RV29D - System II Booster Pumps		X	C-8.6
CS	V-20-03	OB-FZ-08C	CS Pump A Suction Valve		X	C-8.5
CS	V-20-04	OB-FZ-08C	Core Spray Pump Suction MOV V-20-4		X	C-8.5
CS	V-20-15	OB-FZ-08C	CS I Containment Isolation		X	C-8.5
CS	V-20-21	OB-FZ-08C	CS II Containment Isolation		X	C-8.5
CS	V-20-32	OB-FZ-08C	CS Pump C Suction Valve		X	C-8.5
CS	V-20-33	OB-FZ-08C	Core Spray Pump Suction MOV V-20-33		X	C-8.5
CS	V-20-40	OB-FZ-08C	CS I Containment Isolation		X	C-8.5
CS	V-20-41	OB-FZ-08C	CS II Containment Isolation		X	C-8.5
CW	P-03-002A	OB-FZ-08C	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	OB-FZ-08C	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	OB-FZ-08C	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	OB-FZ-08C	Circulating Water Pump 1-4		X	JMM-66

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
DB	BATT. B	OB-FZ-08C	125V DC Battery "B"	X	X	C-8.8
DB	DIST.CTR."B"	OB-FZ-08C	125V DC Distribution Center "B"	X	X	C-8.8
DB	MCC DC-1	OB-FZ-08C	125 VDC MCC		X	C-8.8
DB	PNL D	OB-FZ-08C	125v DC Power Panel D	X	X	C-8.8
DB	ROT.INV	OB-FZ-08C	Continuous Power Unit	X		C-8.8
DB	ROT.INV.PNL	OB-FZ-08C	460V AC		X	C-8.7
DP	ER-18A	OB-FZ-08C	Relay Panel ER-18A (See Note E)		X	C-8.8
DP	ER-18B	OB-FZ-08C	Relay Panel ER-18B (See Note E)		X	C-8.8
DP	PS-RV/46A	OB-FZ-08C	Drywell Pressure Switch RV46A		X	C-8.6
DP	PS-RV/46C	OB-FZ-08C	Drywell Pressure Switch RV46C		X	C-8.6
EB	SWGR 1B	OB-FZ-08C	4,160V AC Bus 1B		X	C-8.8
EC	BKR 1A2M	OB-FZ-08C	Circuit Breaker 1A2M		X	C-8.7
ED	BKR 1B2M	OB-FZ-08C	Circuit Breaker 1B2M		X	C-8.7
ED	EDG-2	OB-FZ-08C	Diesel Generator 2 Switchgear		X	C-8.8
ED	EDG-2	OB-FZ-08C	Emergency Diesel Generator 2		X	C-8.9
ED	SWGR 1D	OB-FZ-08C	4160V AC Switchgear "D"		X	C-8.8
ED	USS 1B2	OB-FZ-08C	480V AC Bus 1B2		X	C-8.8
IC	V-14-30	OB-FZ-08C	IC A Steam Line Valve (AC)		X	C-8.6
IC	V-14-31	OB-FZ-08C	IC A Steam Line Valve (DC)		X	C-8.6
IC	V-14-34	OB-FZ-08C	IC A Condensate Valve (DC)		X	C-8.6
IC	V-14-36	OB-FZ-08C	A IC condensate return valve (AC)		X	C-8.6
IC	IC A Isol	OB-FZ-08C	IC A isolation signal (63-359)		X	B-3.7
ME	NS03A	OB-FZ-08C	Inboard MSIVs		X	C-8.9
ME	NS03B	OB-FZ-08C	Inboard MSIVs		X	C-8.9
MS	NS03A	OB-FZ-08C	Inboard MSIVs		X	C-8.9
MS	NS03B	OB-FZ-08C	Inboard MSIVs		X	C-8.9
OV	V-27-01	OB-FZ-08C	Air-Operated Valve V-27-01		X	JMM-19
OV	V-27-02	OB-FZ-08C	Air-Operated Valve V-27-02		X	JMM-20
PI	V-22-01	OB-FZ-08C	Air-Operated Valve V-22-1		X	JMM-03
PI	V-22-02	OB-FZ-08C	Air-Operated Valve V-22-2		X	JMM-04
PI	V-22-28	OB-FZ-08C	Air-Operated Valve V-22-28		X	JMM-05
PI	V-22-29	OB-FZ-08C	Air-Operated Valve V-22-29		X	JMM-06
PI	V-23-15	OB-FZ-08C	Air-Operated Valve V-23-15		X	JMM-09
PI	V-23-16	OB-FZ-08C	Air-Operated Valve V-23-16		X	JMM-10
PI	V-23-19	OB-FZ-08C	Air-Operated Valve V-23-19		X	JMM-13
PI	V-23-20	OB-FZ-08C	Air-Operated Valve V-23-20		X	JMM-14
PI	V-23-21	OB-FZ-08C	Air-Operated Valve V-23-21		X	JMM-15
PI	V-23-22	OB-FZ-08C	Air-Operated Valve V-23-22		X	JMM-16
PI	V-26-16	OB-FZ-08C	Air-Operated Valve V-26-16		X	JMM-17
PI	V-26-18	OB-FZ-08C	Air-Operated Valve V-26-18		X	JMM-18
PI	V-28-17	OB-FZ-08C	Air-Operated Valve V-28-17		X	JMM-23
PI	V-28-18	OB-FZ-08C	Air-Operated Valve V-28-18		X	JMM-24
PI	V-28-47	OB-FZ-08C	Air-Operated Valve V-28-47		X	JMM-25
SD	NU02A	OB-FZ-08C	SDC pump A		X	B-3.15
SD	NU02B	OB-FZ-08C	SDC pump B		X	B-3.15
SD	NU02C	OB-FZ-08C	SDC pump C		X	ABN
SD	RBCCW Pump 1	OB-FZ-08C	RBCCW Pump 1-1		X	B-5.29
SD	RBCCW Pump 2	OB-FZ-08C	RBCCW Pump 1-2		X	B-5.29
SD	V-17-001	OB-FZ-08C	SDC suction valve		X	ABN
SD	V-17-002	OB-FZ-08C	SDC suction valve		X	B-3.16
SD	V-17-003	OB-FZ-08C	SDC suction valve		X	ABN
SD	V-17-019	OB-FZ-08C	SDC supply isolation valve		X	B-3.16
SD	V-17-054	OB-FZ-08C	SDC return isolation valve		X	B-3.16
SD	V-17-056	OB-FZ-08C	SDC discharge valve		X	B-3.16
SD	V-17-057	OB-FZ-08C	SDC discharge valve		X	ABN

Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
VO	ER-18A	OB-FZ-08C	Relay Panel ER-18A (See Note E)		X	C-8.8
VO	ER-18B	OB-FZ-08C	Relay Panel ER-18B (See Note E)		X	C-8.8
VO	NR108A	OB-FZ-08C	EMRV (NR108A) V-1-173		X	C-8.6
VO	NR108B	OB-FZ-08C	EMRV (NR108B) V-1-174		X	C-8.6
VO	NR108C	OB-FZ-08C	EMRV (NR108C) V-1-175		X	C-8.6
VO	NR108D	OB-FZ-08C	EMRV (NR108D) V-1-176		X	C-8.6
VO	NR108E	OB-FZ-08C	EMRV (NR108E) V-1-177		X	C-8.6
VO	PNL D	OB-FZ-08C	125v DC Power Panel D	X	X	C-8.8
VR	NR108A	OB-FZ-08C	EMRV (NR108A) V-1-173		X	C-8.6
VR	NR108B	OB-FZ-08C	EMRV (NR108B) V-1-174		X	C-8.6
VR	NR108C	OB-FZ-08C	EMRV (NR108C) V-1-175		X	C-8.6
VR	NR108D	OB-FZ-08C	EMRV (NR108D) V-1-176		X	C-8.6
VR	NR108E	OB-FZ-08C	EMRV (NR108E) V-1-177		X	C-8.6
XB	DIST.CTR."B"	OB-FZ-08C	125V DC Distribution Center "B"	X	X	C-8.8
XB	MG SET B	OB-FZ-08C	460V AC		X	C-8.7
XB	MG SET B	OB-FZ-08C	"B" Battery Charger MG Set B.	X	X	C-8.8
XB	SF-1-20	OB-FZ-08C	Supply Fan		X	C-8.10
Fire Area OB-FA-09			**Office Building**			**Location 7**
AD	NR108A	OB-FA-09	EMRV (NR108A) V-1-173		X	C-9
AD	NR108C	OB-FA-09	EMRV (NR108C) V-1-175		X	C-9
AD	NR108D	OB-FA-09	EMRV (NR108D) V-1-176		X	C-9
PR	PT-RE-15C	OB-FA-09	Reactor Pressure Switch RE15C		X	JMM-48
PR	PT-RE-15D	OB-FA-09	Reactor Pressure Switch RE15D		X	JMM-49
VO	NR108A	OB-FA-09	EMRV (NR108A) V-1-173		X	C-9
VO	NR108C	OB-FA-09	EMRV (NR108C) V-1-175		X	C-9
VO	NR108D	OB-FA-09	EMRV (NR108D) V-1-176		X	C-9
VR	NR108A	OB-FA-09	EMRV (NR108A) V-1-173		X	C-9
VR	NR108C	OB-FA-09	EMRV (NR108C) V-1-175		X	C-9
VR	NR108D	OB-FA-09	EMRV (NR108D) V-1-176		X	C-9
XB	SF-1-20	OB-FA-09	Supply Fan		X	C-9
Fire Zone OB-FZ-10A			**Monitoring and Change Room (46')**			**Location 8**
AD	DPS-RV40C	OB-FZ-10A	Sys. I, Booster pump disch. press. switch		X	C-10.1
AD	ER-642-078	OB-FZ-10A	EMRV Aux. Relay Cabinet	X		C-10.1
AD	NR108A	OB-FZ-10A	EMRV (NR108A) V-1-173		X	C-10.1
AD	NR108B	OB-FZ-10A	EMRV (NR108B) V-1-174		X	C-10.1
AD	NR108C	OB-FZ-10A	EMRV (NR108C) V-1-175		X	C-10.1
AD	NR108D	OB-FZ-10A	EMRV (NR108D) V-1-176		X	C-10.1
AD	NR108E	OB-FZ-10A	EMRV (NR108E) V-1-177		X	C-10.1
CC	PS-IP15A	OB-FZ-10A	Drywell Pressure Switch IP-15A		X	C-10.2
CC	PS-IP15C	OB-FZ-10A	Drywell Pressure Switch IP-15C		X	C-10.2
CC	V-21-11	OB-FZ-10A	System 1 drywell isolation		X	C-10.2
CS	DPS-RV40C	OB-FZ-10A	Sys. I, Booster pump disch. press. switch		X	C-10.1
CS	NZ01A	OB-FZ-10A	Power cable		X	ABN-69
CS	NZ03A	OB-FZ-10A	Power cable		X	C-10.1
CS	V-20-15	OB-FZ-10A	CS I Containment Isolation		X	C-10.1
DP	PS-RV/46A	OB-FZ-10A	Drywell Pressure Switch RV46A		X	C-10.2
DP	PS-RV/46B	OB-FZ-10A	Drywell Pressure Switch RV46B		X	C-10.2
DP	PS-RV/46C	OB-FZ-10A	Drywell Pressure Switch RV46C		X	C-10.2
DP	PS-RV/46D	OB-FZ-10A	Drywell Pressure Switch RV46D		X	C-10.2
ED	EF-1-21	OB-FZ-10A	Exhaust Fan	X	X	C-10.3
ED	SF-1-21	OB-FZ-10A	Supply Fan	X	X	C-10.3
ED	USS 1B2	OB-FZ-10A	125VDC Distribution Center to		X	C-10.3
IC	IC A isol	OB-FZ-10A	IC A isolation logic cable 63-361		X	B-3.7
IC	V-14-32	OB-FZ-10A	IC B Steam Line Valve (AC)		X	C-10.1
IC	V-14-33	OB-FZ-10A	IC B Steam Line Valve (DC)		X	ABN-70a

Table 4.4-3 Location of Risk Significant Components and Associated Cables

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
IC	V-14-35	OB-FZ-10A	IC B Condensate Valve (DC)		X	C-10.1
IC	V-14-37	OB-FZ-10A	B IC condensate return valve (AC)		X	C-10.1
IC	IC B isol	OB-FZ-10A	IC B isolation logic cable 63-361		X	B-3.7
MU	V-11-34	OB-FZ-10A	Shell side makeup to B IC		X	C-10.1
PI	V-23-13	OB-FZ-10A	Air-Operated Valve V-23-13		X	JMM-07
PI	V-23-14	OB-FZ-10A	Air-Operated Valve V-23-14		X	JMM-08
PI	V-23-17	OB-FZ-10A	Air-Operated Valve V-23-17		X	JMM-11
PI	V-23-18	OB-FZ-10A	Air-Operated Valve V-23-18		X	JMM-12
PI	V-27-03	OB-FZ-10A	Air-Operated Valve V-27-03		X	JMM-21
PI	V-27-04	OB-FZ-10A	Air-Operated Valve V-27-04		X	JMM-22
PI	V-38-43	OB-FZ-10A	Air-Operated Valve V-38-43		X	JMM-43
PI	V-38-46	OB-FZ-10A	Air-Operated Valve V-38-46		X	JMM-45
PR	PT-RE-15A	OB-FZ-10A	Reactor Pressure Switch RE15A		X	JMM-46
PR	PT-RE-15B	OB-FZ-10A	Reactor Pressure Switch RE15B		X	JMM-47
PR	PT-RE-15C	OB-FZ-10A	Reactor Pressure Switch RE15C		X	JMM-48
PR	PT-RE-15D	OB-FZ-10A	Reactor Pressure Switch RE15D		X	JMM-49
VO	NR108A	OB-FZ-10A	EMRV (NR108A) V-1-173		X	C-10.1
VO	NR108B	OB-FZ-10A	EMRV (NR108B) V-1-174		X	C-10.1
VO	NR108C	OB-FZ-10A	EMRV (NR108C) V-1-175		X	C-10.1
VO	NR108D	OB-FZ-10A	EMRV (NR108D) V-1-176		X	C-10.1
VO	NR108E	OB-FZ-10A	EMRV (NR108E) V-1-177		X	C-10.1
VR	NR108A	OB-FZ-10A	EMRV (NR108A) V-1-173		X	C-10.1
VR	NR108B	OB-FZ-10A	EMRV (NR108B) V-1-174		X	C-10.1
VR	NR108C	OB-FZ-10A	EMRV (NR108C) V-1-175		X	C-10.1
VR	NR108D	OB-FZ-10A	EMRV (NR108D) V-1-176		X	C-10.1
VR	NR108E	OB-FZ-10A	EMRV (NR108E) V-1-177		X	C-10.1
XB	EF-1-20	OB-FZ-10A	Exhaust Fan	X	X	C-10.3
Fire Zone OB-FZ-10B			**Chem Lab, Laundry, Instrument Shop (35')**			**Location 9**
ED	USS 1B2	OB-FZ-10B	125VDC Distribution Center to USS 1B2		X	C-10.4
Fire Zone OB-FZ-22A			**New Cable Spreading Room (Mech Equp Rm)**			**Location 10**
ED	SF/EF-1-21	OB-FZ-22A	Supply Fan		X	C-22.2
IC	V-14-33	OB-FZ-22A	IC B Steam Line Valve (DC)		X	C-22.1
IC	V-14-35	OB-FZ-22A	IC B Condensate Valve (DC)		X	C-22.1
IC	IC B isol	OB-FZ-22A	IC B isolation logic cable 306RC0947		X	B-3.7
MU	V-11-34	OB-FZ-22A	Shell side makeup to B IC		X	C-22.1
MU	V-11-36	OB-FZ-22A	Shell side makeup to A IC		X	C-22.1
PI	V-38-09	OB-FZ-22A	Air-Operated Valve V-38-09		X	JMM-27
PI	V-38-10	OB-FZ-22A	Air-Operated Valve V-38-10		X	JMM-28
PI	V-38-16	OB-FZ-22A	Air-Operated Valve V-38-16		X	JMM-32
PI	V-38-17	OB-FZ-22A	Air-Operated Valve V-38-17		X	JMM-37
PI	V-38-37	OB-FZ-22A	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	OB-FZ-22A	Air-Operated Valve V-38-38		X	JMM-39
PI	V-38-39	OB-FZ-22A	Air-Operated Valve V-38-39		X	JMM-40
PI	V-38-40	OB-FZ-22A	Air-Operated Valve V-38-40		X	JMM-41
PI	V-38-41	OB-FZ-22A	Air-Operated Valve V-38-41		X	JMM-42
PI	V-38-44	OB-FZ-22A	Air-Operated Valve V-38-44		X	JMM-44
PR	PT-RE-15A	OB-FZ-22A	Reactor Pressure Switch RE15A		X	JMM-46
PR	PT-RE-15B	OB-FZ-22A	Reactor Pressure Switch RE15B		X	JMM-47
PR	PT-RE-15C	OB-FZ-22A	Reactor Pressure Switch RE15C		X	JMM-48
PR	PT-RE-15D	OB-FZ-22A	Reactor Pressure Switch RE15D		X	JMM-49
RL	RE02A	OB-FZ-22A	Reactor Water Level Switch RE02A		X	C-22.3
RL	RE02B	OB-FZ-22A	Reactor Water Level Switch RE02B		X	C-22.3
RL	RE02C	OB-FZ-22A	Reactor Water Level Switch RE02C		X	C-22.3
RL	RE02D	OB-FZ-22A	Reactor Water Level Switch RE02D		X	C-22.3
XB	EF-1-20	OB-FZ-22A	Exhaust Fan		X	C-22.2

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
Fire Zone OB-FZ-22B			**North Cable Bridge Tunnel (74' 6")**			**Location 11**
IC	IC B isol	OB-FZ-22B	IC B isolation logic cable 306RC0947		X	B-3.7
MU	V-11-36	OB-FZ-22B	Shell side makeup to A IC		X	C-22.5
PI	V-38-37	OB-FZ-22B	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	OB-FZ-22B	Air-Operated Valve V-38-38		X	JMM-39
PI	V-38-39	OB-FZ-22B	Air-Operated Valve V-38-39		X	JMM-40
PI	V-38-40	OB-FZ-22B	Air-Operated Valve V-38-40		X	JMM-41
RL	RE02A	OB-FZ-22B	Reactor Water Level Switch RE02A		X	C-22.5
RL	RE02C	OB-FZ-22B	Reactor Water Level Switch RE02C		X	C-22.5
Fire Zone OB-FZ-22C			**South Cable Bridge Tunnel**			**Location 12**
PI	V-38-09	OB-FZ-22C	Air-Operated Valve V-38-09		X	JMM-26
PI	V-38-10	OB-FZ-22C	Air-Operated Valve V-38-10		X	JMM-28
PI	V-38-16	OB-FZ-22C	Air-Operated Valve V-38-16		X	JMM-32
PI	V-38-17	OB-FZ-22C	Air-Operated Valve V-38-17		X	JMM-36
PI	V-38-41	OB-FZ-22C	Air-Operated Valve V-38-41		X	JMM-42
PI	V-38-44	OB-FZ-22C	Air-Operated Valve V-38-44		X	JMM-44
RL	RE02B	OB-FZ-22C	Reactor Water Level Switch RE02B		X	C-22.6
RL	RE02D	OB-FZ-22C	Reactor Water Level Switch RE02D		X	C-22.6
Fire Area TB-FA-03A			**4160 VAC Switchgear 1C Vault (TB 23' Elevation)**			**Location 14**
CC	PMP 52A	TB-FA-03A	ESW Pump (trip from Control Room)		X	C-3.1
CC	PMP 52B	TB-FA-03A	ESW Pump (trip from Control Room)		X	C-3.1
CC	PMP 52C	TB-FA-03A	ESW Pump		X	C-3.1
CC	PMP 52D	TB-FA-03A	ESW Pump		X	C-3.1
CS	NZ01A	TB-FA-03A	Pump trip from control room		X	C-3.1
CS	NZ01B	TB-FA-03A	Core Spray Pump NZ01B		X	C-3.1
CS	NZ01C	TB-FA-03A	Core Spray Pump NZ01C		X	C-3.1
CS	NZ01D	TB-FA-03A	Pump trip from control room		X	C-3.1
EC	EDG-1	TB-FA-03A	Emergency Diesel Generator #1		X	C-3.1
EC	SWGR 1C	TB-FA-03A	4,160V AC Bus 1C	X	X	C-3.1
EC	USS 1A1	TB-FA-03A	480V AC Bus 1A1		X	C-3.1
EC	USS 1A2	TB-FA-03A	480V AC Bus 1A2		X	C-3.1
Fire Area TB-FA-03B			**4160 VAC Switchgear 1D Vault (TB 23' Elevation)**			**Location 15**
CC	PMP 52C	TB-FA-03B	ESW Pump		X	C-3.2
CC	PMP 52D	TB-FA-03B	ESW Pump		X	C-3.2
CS	NZ01B	TB-FA-03B	Core Spray Pump NZ01B		X	C-3.2
CS	NZ01C	TB-FA-03B	Core Spray Pump NZ01C		X	C-3.2
ED	EDG-2	TB-FA-03B	Emergency Diesel Generator #2		X	C-3.2
ED	SWGR 1D	TB-FA-03B	4,160V DC Bus 1D	X	X	C-3.2
ED	USS 1B1	TB-FA-03B	480V AC Bus 1B1		X	C-3.2
ED	USS 1B2	TB-FA-03B	480V AC Bus 1B2		X	C-3.2
Fire Zone TB-FA-11A			**Main Turbine Operating Floor**			**Location 16**
BT	CVT 2	TB-FZ-11A	Condenser Vacuum Trip No. 2	X		OCP
BT	TODV	TB-FZ-11A	Trip Oil Dump Valve	X		OCP
BV	EPR	TB-FZ-11A	Electrical Pressure Regulator	X		OCP
TT	EPR	TB-FZ-11A	Electrical Pressure Regulator	X		OCP
TT	ETODV	TB-FZ-11A	Emergency Trip Oil Dump Valve	X		OCP
TT	MTS 1	TB-FZ-11A	Main Trip Solenoid No. 1	X		OCP
TT	MTS 3	TB-FZ-11A	Main Trip Solenoid #3	X		OCP
TT	NO LOAD SW	TB-FZ-11A	No Load Switch	X		OCP
Fire Zone TB-FZ-11B			**Lube Oil Storage, Pumping and Purification**			**Location 17**
BV	EPR	TB-FZ-11B	Pressure transmitter (EPR)	X		WD
CS	NZ01A	TB-FZ-11B	Core Spray Pump NZ01A		X	C-11.1
CS	NZ01C	TB-FZ-11B	Core Spray Pump NZ01C		X	C-11.1
CS	NZ01D	TB-FZ-11B	Core Spray Pump NZ01D		X	C-11.1

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CW	P-03-002A	TB-FZ-11B	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	TB-FZ-11B	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	TB-FZ-11B	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	TB-FZ-11B	Circulating Water Pump 1-4		X	JMM-66
DC	MCC DC-2	TB-FZ-11B	125v DC Dist. to MCC DC-2		X	C-11.2
DC	PNL DC-F	TB-FZ-11B	125v DC Dist. to panel DC-F		X	C-11.2
EB	SWGR 1B	TB-FZ-11B	125v DC Dist. to Swgr 1B		X	C-11.2
EC	BKR 1A1M	TB-FZ-11B	460V AC		X	C-11.1
EC	BKR 1A1P	TB-FZ-11B	4160V AC		X	C-11.1
EC	BKR 1A2P	TB-FZ-11B	4160V AC		X	C-11.1
EC	BKR 1C	TB-FZ-11B	Circuit Breaker 1C		X	C-11.1
EC	EDG-1	TB-FZ-11B	Start Circuit for EDG		X	C-11.2
EC	FN-56-004	TB-FZ-11B	Supply Fan		X	C-11.3
EC	FN-56-007	TB-FZ-11B	Exhaust Fan		X	C-11.3
EC	USS 1A2	TB-FZ-11B	AC Circuit Breaker from 1A2		X	C-11.1
EC	USS 1A2	TB-FZ-11B	125v DC Control Power		X	C-11.2
ED	BKR 1B1M	TB-FZ-11B	460V AC		X	C-11.1
ED	BKR 1B1P	TB-FZ-11B	SWGR 1D		X	C-11.1
ED	BKR 1D	TB-FZ-11B	Circuit Breaker 1D from 4,160VAC Bus 1B		X	C-11.1
ED	USS 1B2	TB-FZ-11B	460V AC		X	C-11.1
FP	DFP 1-1	TB-FZ-11B	Diesel Driven Fire Pump 1-1		X	C-11.3
FW	PMP 1A	TB-FZ-11B	Feedwater Pump A		X	C-11.3
FW	PMP 1B	TB-FZ-11B	Feedwater Pump B		X	C-11.3
FW	PMP 1C	TB-FZ-11B	Feedwater Pump C		X	C-11.3
IC	IC B Isol	TB-FZ-11B	Isolation Condenser B isolation logic cable 67-40		X	B-3.7
RP	PMP TRIP 1	TB-FZ-11B	Recirculation Pump Trip, Division I		X	C-11.2
RP	PMP TRIP 2	TB-FZ-11B	Recirculation Pump Trip, Division II		X	C-11.2
SW	PMP 1-1	TB-FZ-11B	Service Water Pump P-3-001A		X	C-11.2
SW	PMP 1-2	TB-FZ-11B	Service Water Pump P-3-001B		X	C-11.2
TT	EPR	TB-FZ-11B	Pressure transmitter (EPR)	X		WD
VO	PNL DC-F	TB-FZ-11B	125v DC Dist. to panel DC-F		X	C-11.2
XC	BATT CHG C1	TB-FZ-11B	460V Power to Battery Charger C1		X	C-11.2
Fire Zone TB-FZ-11C			**4160 VAC Switchgear 1A and 1B**			**Location 18**
CS	NZ01A	TB-FZ-11C	Pump control circuit		X	C-11.4
CS	NZ01B	TB-FZ-11C	Pump control circuit		X	C-11.4
CS	NZ01C	TB-FZ-11C	Pump control circuit		X	C-11.4
CS	NZ01D	TB-FZ-11C	Pump control circuit		X	C-11.4
CW	P-03-002A	TB-FZ-11C	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	TB-FZ-11C	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	TB-FZ-11C	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	TB-FZ-11C	Circulating Water Pump 1-4		X	JMM-66
DC	MCC DC-2	TB-FZ-11C	125v DC Distribution Center to MCC DC-2		X	C-11.5
DC	PNL DC-C	TB-FZ-11C	125v DC Distribution Panel "C"		X	D02-003
DC	PNL DC-F	TB-FZ-11C	125v DC Distribution Center to panel DC-F		X	C-11.5
EA	SWGR 1A	TB-FZ-11C	4160v Switchgear 1A	X		C-11.4
EA	SWGR 1A	TB-FZ-11C	125v DC Distribution Center to		X	C-11.5
EB	SWGR 1B	TB-FZ-11C	4160v Switchgear 1B	X		C-11.4
EB	SWGR 1B	TB-FZ-11C	125v DC Distribution Center to		X	C-11.5
EC	BKR 1A1P	TB-FZ-11C	Circuit Breaker 1A1P		X	C-11.4
EC	BKR 1A2P	TB-FZ-11C	Circuit Breaker 1A2P to Bus 1A2		X	C-11.4
EC	BKR 1C	TB-FZ-11C	Circuit Breaker 1C		X	C-11.4
EC	EDG-1	TB-FZ-11C	Idling Start		X	C-11.4
EC	EDG-1	TB-FZ-11C	Emergency Diesel Generator 1		X	C-11.5
EC	SWGR 1C	TB-FZ-11C	4,160V AC Bus 1C		X	C-11.4
EC	SWGR 1C	TB-FZ-11C	125v DC Control Power		X	C-11.5
EC	USS 1A2	TB-FZ-11C	125v DC Control Power	X	X	C-11.5

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
ED	BKR 1B2P	TB-FZ-11C	Circuit Breaker 1B2P to Bus 1B2		X	C-11.4
ED	BKR 1D	TB-FZ-11C	Circuit Breaker 1D from 4,160VAC Bus 1B		X	C-11.4
ED	EDG-2	TB-FZ-11C	DG 2 power supply to bus 1D		X	C-11.4
ED	EDG-2	TB-FZ-11C	Idling Start		X	C-11.4
FP	DFP 1-1	TB-FZ-11C	Diesel Driven Fire Pump 1-1		X	C-11.6
FW	PMP 1A	TB-FZ-11C	Feedwater Pump A		X	C-11.6
FW	PMP 1B	TB-FZ-11C	Feedwater Pump B		X	C-11.6
FW	PMP 1C	TB-FZ-11C	Feedwater Pump C		X	C-11.6
IC	IC B Isol	TB-FZ-11C	Isolation Condenser B isolation logic cable 67-40		X	B-3.7
RF		TB-FZ-11C	FWP A supply breaker	X		OCP
RF		TB-FZ-11C	FWP B supply breaker	X		OCP
RF		TB-FZ-11C	FWP C supply breaker	X		OCP
RP	PMP TRIP 1	TB-FZ-11C	Recirculation Pump Trip, Division I		X	C-11.6
RP	PMP TRIP 2	TB-FZ-11C	Recirculation Pump Trip, Division II		X	C-11.6
SW	PMP 1-1	TB-FZ-11C	Service Water Pump P-3-001A		X	C-11.6
SW	PMP 1-2	TB-FZ-11C	Service Water Pump P-3-001B		X	C-11.6
VO	PNL DC-F	TB-FZ-11C	125v DC Distribution Center to	X	X	C-11.5
XC	BATT CHG C1	TB-FZ-11C	"C" Battery Charger C1	X	X	C-11.5
XC	PNL DC-C	TB-FZ-11C	125v DC Distribution Panel "C"	X		D02-003
Fire Area TB-FZ-11D			**Basement Floor - South End**			**Location 19**
CC	PMP 52A	TB-FZ-11D	ESW Pump Control		X	C-11.7
CC	PMP 52B	TB-FZ-11D	ESW Pump Control		X	C-11.7
CC	PMP 52C	TB-FZ-11D	ESW Pump Control		X	C-11.7
CC	PMP 52D	TB-FZ-11D	ESW Pump Control		X	C-11.7
CS	NZ01A	TB-FZ-11D	CS pump power and control cables		X	C-11.7
CS	NZ01B	TB-FZ-11D	CS pump power and control cables		X	C-11.7
CS	NZ01C	TB-FZ-11D	CS pump power and control cables		X	C-11.7
CS	NZ01D	TB-FZ-11D	CS pump power and control cables		X	C-11.7
CW	P-03-002A	TB-FZ-11D	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	TB-FZ-11D	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	TB-FZ-11D	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	TB-FZ-11D	Circulating Water Pump 1-4		X	JMM-66
EC	BKR 1A1M	TB-FZ-11D	Circuit Breaker 1A1M		X	C-11.9
EC	BKR 1A1P	TB-FZ-11D	Circuit Breaker 1A1P to Bus 1A1		X	C-11.8
EC	BKR 1A2P	TB-FZ-11D	Circuit Breaker 1A2P to Bus 1A2		X	C-11.8
EC	BKR 1C	TB-FZ-11D	Circuit Breaker 1C		X	C-11.8
EC	EDG-1	TB-FZ-11D	Emergency Diesel Generator #1		X	C-11.7
EC	EDG-1	TB-FZ-11D	Emergency Diesel Generator #1		X	C-11.9
EC	SWGR 1C	TB-FZ-11D	4,160V AC Bus 1C		X	C-11.8
EC	USS 1A1	TB-FZ-11D	480V AC Bus 1A1		X	C-11.8
EC	USS 1A1	TB-FZ-11D	460v Unit Substation	X		C-11.9
EC	USS 1A1	TB-FZ-11D	480V AC Bus 1A1	X		WD
EC	USS 1A2	TB-FZ-11D	480V AC Bus 1A2		X	C-11.17
EC	USS 1A2	TB-FZ-11D	480V AC Bus 1A2		X	C-11.8
ED	BKR 1B2P	TB-FZ-11D	Circuit Breaker 1B2P to Bus 1B2		X	C-11.9
ED	EDG-2	TB-FZ-11D	Emergency Diesel Generator #2		X	C-11.7
ED	EDG-2	TB-FZ-11D	Emergency Diesel Generator #2		X	C-11.9
ED	SWGR 1D	TB-FZ-11D	4,160V DC Bus 1D		X	C-11.9
ED	USS 1B1	TB-FZ-11D	460v Unit Substation	X	X	C-11.9
ED	USS 1B2	TB-FZ-11D	480V AC Bus 1B2		X	C-11.9
FP	DFP 1-1	TB-FZ-11D	Diesel Driven Fire Pump 1-1		X	C-11.10
FW	PMP 1A	TB-FZ-11D	Feedwater Pump A		X	C-11.10
FW	PMP 1B	TB-FZ-11D	Feedwater Pump B		X	C-11.10
FW	PMP 1C	TB-FZ-11D	Feedwater Pump C		X	C-11.10

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
IA	COMPR. 1-1	TB-FZ-11D	Instrument Air Compressor 1-1	X		D02-002
IA	COMPR.1-2	TB-FZ-11D	Instrument Air Compressor 1-2	X		D02-002
IA	DRYER A-B	TB-FZ-11D	Instrument Air Dryer A-B	X		D02-002
SW	PMP 1-1	TB-FZ-11D	Service Water Pump P-3-001A		X	C-11.10
SW	PMP 1-2	TB-FZ-11D	Service Water Pump P-3-001B		X	C-11.10
TB	PMP 1-1	TB-FZ-11D	TBCCW Pump P-5-003	X		D02-002
TB	PMP 1-2	TB-FZ-11D	TBCCW Pump P-5-004	X		D02-002
TB	PMP 1-3	TB-FZ-11D	TBCCW Pump P-5-005	X		D02-002
Fire Zone TB-FZ-11E			**Condenser Bay - 3' 6" Elevation**			**Location 20**
BT	V-1-130A	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130B	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130C	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130D	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130E	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130F	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130G	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130H	TB-FZ-11E	Turbine Bypass Valve	X		02-007
BT	V-1-130I	TB-FZ-11E	Turbine Bypass Valve	X		02-007
CC	PMP 52A	TB-FZ-11E	ESW Pump Control		X	C-11.11
CC	PMP 52B	TB-FZ-11E	ESW Pump Control		X	C-11.11
CC	PMP 52C	TB-FZ-11E	ESW Pump Control		X	C-11.11
CC	PMP 52D	TB-FZ-11E	ESW Pump Control		X	C-11.11
CP	ID11A	TB-FZ-11E	Feed Reg Valve A		X	C-11.13
CP	ID11B	TB-FZ-11E	Feed Reg Valve B		X	C-11.13
CP	ID11C	TB-FZ-11E	Feed Reg Valve C		X	C-11.13
CP	ID12A	TB-FZ-11E	Startup Reg Valve A		X	C-11.13
CP	ID12C	TB-FZ-11E	Startup Reg Valve C		X	C-11.13
CP	V-2-07	TB-FZ-11E	FW train A isol MOV-2-7	X		MSB
CP	V-2-10	TB-FZ-11E	FW train A isol MOV-2-10	X	X	C-11.13
CP	V-2-10	TB-FZ-11E	FW train A isol MOV-2-10	X		MSB
CP	V-2-11	TB-FZ-11E	FW train B isol MOV-2-11	X	X	C-11.14
CP	V-2-11	TB-FZ-11E	FW train B isol MOV-2-11	X		MSB
CP	V-2-12	TB-FZ-11E	FW train C isol MOV-2-12	X	X	C-11.14
CP	V-2-12	TB-FZ-11E	FW train C isol MOV-2-12	X		MSB
CP	V-2-8	TB-FZ-11E	FW train B isol MOV-2-8	X		MSB
CP	V-2-9	TB-FZ-11E	FW train C isol MOV-2-9	X		MSB
CS	NZ01A	TB-FZ-11E	CS pump control cables		X	C-11.11
CS	NZ01A	TB-FZ-11E	CS pump power cables		X	C-11.11
CS	NZ01B	TB-FZ-11E	CS pump control cables		X	C-11.11
CS	NZ01C	TB-FZ-11E	CS pump control cables		X	C-11.11
CS	NZ01C	TB-FZ-11E	CS pump power cables		X	C-11.11
CS	NZ01D	TB-FZ-11E	CS pump control cables		X	C-11.11
CT	PMP 1-1	TB-FZ-11E	Control circuit		X	C-11.11
CT	PMP 1-2	TB-FZ-11E	Control circuit		X	C-11.11
CW	P-03-002A	TB-FZ-11E	Circulating Water Pump 1-1		X	JMM-50
CW	P-03-002B	TB-FZ-11E	Circulating Water Pump 1-2		X	JMM-56
CW	P-03-002C	TB-FZ-11E	Circulating Water Pump 1-3		X	JMM-61
CW	P-03-002D	TB-FZ-11E	Circulating Water Pump 1-4		X	JMM-66
EC	BKR 1A1M	TB-FZ-11E	Circuit Breaker 1A1M		X	C-11.11
EC	BKR 1A1P	TB-FZ-11E	Circuit Breaker 1A1P to Bus 1A1		X	C-11.11
EC	BKR 1A2P	TB-FZ-11E	Circuit Breaker 1A2P to Bus 1A2		X	C-11.11
EC	EDG-1	TB-FZ-11E	Emergency Diesel Generator #1		X	C-11.11
ED	BKR 1B1M	TB-FZ-11E	Circuit Breaker 1A1M		X	C-11.12
ED	BKR 1B2P	TB-FZ-11E	Circuit Breaker 1B2P to Bus 1B2		X	C-11.12
ED	EDG-2	TB-FZ-11E	Emergency Diesel Generator #2		X	C-11.11
FP	DFP 1-1	TB-FZ-11E	Diesel Driven Fire Pump 1-1		X	C-11.13

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
FW	PMP 1A	TB-FZ-11E	Feedwater Pump A		X	C-11.13
FW	PMP 1B	TB-FZ-11E	Feedwater Pump B		X	C-11.13
FW	PMP 1C	TB-FZ-11E	Feedwater Pump C		X	C-11.13
FW	V-2-07	TB-FZ-11E	FW train A isol MOV-2-7	X		MSB
FW	V-2-8	TB-FZ-11E	FW train B isol MOV-2-8	X		MSB
FW	V-2-9	TB-FZ-11E	FW train C isol MOV-2-9	X		MSB
IC	IC B isol	TB-FZ-11E	Isolation Condenser B isolation logic cable 67-40		X	B-3.7
RP	PMP TRIP 1	TB-FZ-11E	Recirculation Pump Trip, Division I		X	C-11.12
RP	PMP TRIP 2	TB-FZ-11E	Recirculation Pump Trip, Division II		X	C-11.12
SW	PMP 1-1	TB-FZ-11E	Service Water Pump P-3-001A		X	C-11.13
SW	PMP 1-2	TB-FZ-11E	Service Water Pump P-3-001B		X	C-11.13
TT	CV-1	TB-FZ-11E	Turbine Control Valve	X		02-007
TT	CV-2	TB-FZ-11E	Turbine Control Valve	X		02-007
TT	CV-3	TB-FZ-11E	Turbine Control Valve	X		02-007
TT	CV-4	TB-FZ-11E	Turbine Control Valve	X		02-007
TT	SV-1	TB-FZ-11E	Turbine Stop Valve	X		02-007
TT	SV-2	TB-FZ-11E	Turbine Stop Valve	X		02-007
TT	SV-3	TB-FZ-11E	Turbine Stop Valve	X		02-007
TT	SV-4	TB-FZ-11E	Turbine Stop Valve	X		02-007
Fire Zone TB-FZ-11F			**Feedwater Pump Area, 0' 6" and 3' 6" Elevations**			**Location 21**
CP	ID11A	TB-FZ-11F	Feed Reg Valve A	X	X	C-11.15
CP	ID11B	TB-FZ-11F	Feed Reg Valve B	X	X	C-11.15
CP	ID11C	TB-FZ-11F	Feed Reg Valve C	X	X	C-11.15
CP	ID12A	TB-FZ-11F	Startup Reg Valve A	X	X	C-11.15
CP	ID12C	TB-FZ-11F	Startup Reg Valve C	X	X	C-11.15
CP	PMP A	TB-FZ-11F	Condensate Pump 1-A	X		Dwg
CP	PMP B	TB-FZ-11F	Condensate Pump 1-B	X		Dwg
CP	PMP C	TB-FZ-11F	Condensate Pump 1-C	X		Dwg
CP	V-2-732	TB-FZ-11F	Feed regulating valve AOV-2-732	X		MSB
CP	V-2-733	TB-FZ-11F	Feed regulating valve AOV-2-733	X		MSB
CP	V-2-734	TB-FZ-11F	Feed regulating valve AOV-2-734	X		MSB
CS	NZ01A	TB-FZ-11F	CS pump power circuit		X	C-11.15
CS	NZ01C	TB-FZ-11F	CS pump power circuit		X	C-11.15
ED	SWGR 1D	TB-FZ-11F	Control Power to 4160 VAC Bus 1D		X	C-11.15
FW	PMP 1A	TB-FZ-11F	Feedwater Pump A	X		Dwg
FW	PMP 1B	TB-FZ-11F	Feedwater Pump B	X		Dwg
FW	PMP 1C	TB-FZ-11F	Feedwater Pump C	X		Dwg
ME	PS RE-23A	TB-FZ-11F	Main Steam Low Pressure Switch RE-23A	X		OCP
ME	PS RE-23B	TB-FZ-11F	Main Steam Low Pressure Switch RE-23B	X		OCP
ME	PS RE-23C	TB-FZ-11F	Main Steam Low Pressure Switch RE-23C	X		OCP
ME	PS RE-23D	TB-FZ-11F	Main Steam Low Pressure Switch RE-23D	X		OCP
Fire Zone TB-FZ-11G			**Mezzanine Level SE Corner and Machine Shop**			**Location 22**
CC	PMP 52C	TB-FZ-11G	ESW pump control circuit		X	C-11.16
CC	PMP 52D	TB-FZ-11G	ESW pump control circuit		X	C-11.16
CP	V-2-10	TB-FZ-11G	FW train A isol MOV-2-10		X	C-11.16
CP	V-2-11	TB-FZ-11G	FW train B isol MOV-2-11		X	C-11.16
CP	V-2-12	TB-FZ-11G	FW train C isol MOV-2-12		X	C-11.16
CS	NZ01B	TB-FZ-11G	CS pump control circuit		X	C-11.16
CS	NZ01C	TB-FZ-11G	CS pump control circuit		X	C-11.16
Fire Zone TB-FZ-11H			**Demin Tank and SJAE Area - 23' 6" Elevation**			**Location 23**
CP	ID11C	TB-FZ-11H	Feed Reg Valve C		X	C-11.17
CP	V-2-01	TB-FZ-11H	SJAE isolation valve	X		MSR
CP	V-2-02	TB-FZ-11H	SJAE isolation valve	X		MSR
CP	V-2-03	TB-FZ-11H	SJAE isolation valve	X		MSR
CP	V-2-04	TB-FZ-11H	SJAE isolation valve	X		MSR

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
CP	V-2-05	TB-FZ-11H	SJAE Isolation valve	X		MSR
CP	V-2-06	TB-FZ-11H	SJAE Isolation valve	X		MSR
CS	NZ01A	TB-FZ-11H	CS pump power cable		X	C-11.17
CS	NZ01B	TB-FZ-11H	CS pump power cable		X	C-11.17
CS	NZ01C	TB-FZ-11H	CS pump power cable		X	C-11.17
CS	NZ01D	TB-FZ-11H	CS pump power cable		X	C-11.17
EC	USS 1A2	TB-FZ-11H	Power feeder cable to USS 1A2		X	C-11.17
ED	4160V BUS 1D	TB-FZ-11H	125 VDC control power to 4160VAC Bus 1D		X	C-11.17
ED	USS 1B2	TB-FZ-11H	Power feeder cable to USS 1B2		X	C-11.17
IA	V-5-54	TB-FZ-11H	TBCCW Inlet Solenoid	X		MSR
IA	V-5-59	TB-FZ-11H	TBCCW Inlet Solenoid	X		MSR
IA	V-6-211	TB-FZ-11H	Air Dryer 4-way valve	X		MSR
Fire Area TB-FA-26			**Battery Room "C"**			**Location 24**
DC	Battery C	TB-FA-26	125 VDC Battery C	X	X	C-26
Fire Zone RB-FZ-01B			**Reactor Building 95' Elevation**			**Location 26**
BI	FS-1L06	RB-FZ-01B	FS 1L06 Flow Sensor To Isolate RWCU	X		MSB
BI	PMP	RB-FZ-01B	Poison pumps	X		WD
BI	V-19-44	RB-FZ-01B	Squib Valve B	X		MSB
BI	V-19-45	RB-FZ-01B	Squib Valve A	X		MSB
MU	V-11-36	RB-FZ-01B	Shell side makeup to A IC	X	X	C-1.1
PI	V-38-37	RB-FZ-01B	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	RB-FZ-01B	Air-Operated Valve V-38-38		X	JMM-39
PI	V-38-39	RB-FZ-01B	Air-Operated Valve V-38-39		X	JMM-40
PI	V-38-40	RB-FZ-01B	Air-Operated Valve V-38-40		X	JMM-41
RI	V-28-05	RB-FZ-01B	RB General Area Supply	X		MSB
RI	V-28-06	RB-FZ-01B	RB General Area Supply	X		MSB
RI	V-28-07	RB-FZ-01B	RB General Area Supply	X		MSB
RI	V-28-08	RB-FZ-01B	RB General Area Supply	X		MSB
Fire Zone RB-FZ-01C			**Reactor Building 75' Elevation**			**Location 27**
CS	V-20-18	RB-FZ-01C	CS II Test Isolation	X		C-1.2
CS	V-20-21	RB-FZ-01C	CS II Containment Isolation	X	X	C-1.2
CS	V-20-41	RB-FZ-01C	CS II Containment Isolation	X	X	C-1.2
DC	MCC DC-2	RB-FZ-01C	125 VDC MCC DC-2	X	X	C-1.4
IC	V-14-30	RB-FZ-01C	IC A Steam Line Valve (AC)	X	X	C-1.2
IC	V-14-31	RB-FZ-01C	IC A Steam Line Valve (DC)	X	X	C-1.2
IC	V-14-32	RB-FZ-01C	IC B Steam Line Valve (AC)	X	X	C-1.2
IC	V-14-33	RB-FZ-01C	IC B Steam Line Valve (DC)	X	X	C-1.3
IC	V-14-34	RB-FZ-01C	IC A Condensate Valve (DC)	X	X	C-1.3
IC	V-14-35	RB-FZ-01C	IC B Condensate Valve (DC)	X	X	C-1.3
PI	V-23-13	RB-FZ-01C	Air-Operated Valve V-23-13	X		JMM-07
PI	V-23-14	RB-FZ-01C	Air-Operated Valve V-23-14	X		JMM-08
PI	V-23-17	RB-FZ-01C	Air-Operated Valve V-23-17	X		JMM-11
PI	V-23-18	RB-FZ-01C	Air-Operated Valve V-23-18	X		JMM-12
PI	V-27-03	RB-FZ-01C	Air-Operated Valve V-27-03	X		JMM-21
PI	V-27-04	RB-FZ-01C	Air-Operated Valve V-27-04	X		JMM-22
PI	V-38-09	RB-FZ-01C	Air-Operated Valve V-38-09		X	JMM-26
PI	V-38-10	RB-FZ-01C	Air-Operated Valve V-38-10		X	JMM-29
PI	V-38-16	RB-FZ-01C	Air-Operated Valve V-38-16		X	JMM-33
PI	V-38-17	RB-FZ-01C	Air-Operated Valve V-38-17		X	JMM-36
PI	V-38-37	RB-FZ-01C	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	RB-FZ-01C	Air-Operated Valve V-38-38	X		JMM-39
PI	V-38-39	RB-FZ-01C	Air-Operated Valve V-38-39	X		JMM-40
PI	V-38-40	RB-FZ-01C	Air-Operated Valve V-38-40	X		JMM-41
PI	V-38-41	RB-FZ-01C	Air-Operated Valve V-38-41	X		JMM-42
PI	V-38-43	RB-FZ-01C	Air-Operated Valve V-38-43	X		JMM-43

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
PI	V-38-44	RB-FZ-01C	Air-Operated Valve V-38-44	X		JMM-44
PI	V-38-46	RB-FZ-01C	Air-Operated Valve V-38-46	X		JMM-45
RI	V-28-09	RB-FZ-01C	Valve V-28-09 (Not Used - See Note 6)	X		MSB
RI	V-28-10	RB-FZ-01C	Valve V-28-10 (Not Used - See Note 6)	X		MSB
RI	V-28-11	RB-FZ-01C	Valve V-28-11 (Not Used - See Note 6)	X		MSB
RI	V-28-12	RB-FZ-01C	Valve V-28-12 (Not Used - See Note 6)	X		MSB
Fire Zone RB-FZ-01D			**Reactor Building 51' Elevation**			**Location 28**
AD	DPS-RV40A	RB-FZ-01D	CS I Booster Pump Disch PS	X	X	C-1.5
AD	DPS-RV40C	RB-FZ-01D	CS I Booster Pump Disch PS	X	X	C-1.6
AD	ER-642-112	RB-FZ-01D	Relay Cabinet for PS-RV/46A	X		C-1.8
AD	ER-642-113	RB-FZ-01D	Relay Cabinet for PS-RV/46B	X		C-1.9
AD	ER-642-114	RB-FZ-01D	Relay Cabinet for PS-RV/46C	X		C-1.9
AD	ER-642-115	RB-FZ-01D	Relay Cabinet for PS-RV/46D	X		C-1.9
AD	RE18A	RB-FZ-01D	Mech Switch RE18A (Triple low)	X		OCP
AD	RE18C	RB-FZ-01D	Mech Switch RE18C (Triple low)	X		OCP
CC	PS-IP15A	RB-FZ-01D	Drywell Pressure Switch	X	X	C-1.10
CC	PS-IP15B	RB-FZ-01D	Drywell Pressure Switch	X	X	C-1.10
CC	PS-IP15C	RB-FZ-01D	Drywell Pressure Switch	X	X	C-1.10
CC	PS-IP15D	RB-FZ-01D	Drywell Pressure Switch	X	X	C-1.11
CC	V-21-11	RB-FZ-01D	Drywell Spray Valve	X	X	C-1.13
CS	DPS-RV40A	RB-FZ-01D	CS I Booster Pump Disch PS	X	X	C-1.5
CS	DPS-RV40C	RB-FZ-01D	CS I Booster Pump Disch PS	X	X	C-1.6
CS	NZ03A	RB-FZ-01D	CS Booster Pump A	X	X	C-1.5
CS	NZ03C	RB-FZ-01D	CS Booster Pump C	X	X	C-1.5
CS	V-20-12	RB-FZ-01D	CS I Test Isolation	X		C-1.5
CS	V-20-15	RB-FZ-01D	CS I Containment Isolation	X	X	C-1.5
CS	V-20-40	RB-FZ-01D	CS I Containment Isolation	X	X	C-1.5
CS	V-20-92	RB-FZ-01D	Min. Flow Valve	X		MSR
CS	V-20-94	RB-FZ-01D	Min. Flow Valve	X		MSR
DC	MCC DC-2	RB-FZ-01D	125 VDC MCC DC-2		X	C-1.12
DP	ER-642-112	RB-FZ-01D	Relay Cabinet for PS-RV/46A	X		C-1.8
DP	ER-642-113	RB-FZ-01D	Relay Cabinet for PS-RV/46B	X		C-1.9
DP	ER-642-114	RB-FZ-01D	Relay Cabinet for PS-RV/46C	X		C-1.9
DP	ER-642-115	RB-FZ-01D	Relay Cabinet for PS-RV/46D	X		C-1.9
DP	PS-RV/46A	RB-FZ-01D	Drywell Hi Press Control		X	C-1.8
DP	PS-RV/46A	RB-FZ-01D	Drywell Hi Press Control		X	C-1.8
DP	PS-RV/46A	RB-FZ-01D	Drywell Hi Press Switch	X		C-1.8
DP	PS-RV/46B	RB-FZ-01D	Drywell Hi Press Control		X	C-1.8
DP	PS-RV/46B	RB-FZ-01D	Drywell Hi Press Control		X	C-1.8
DP	PS-RV/46B	RB-FZ-01D	Drywell Hi Press Switch	X		C-1.8
DP	PS-RV/46C	RB-FZ-01D	Drywell Hi Press Control		X	C-1.9
DP	PS-RV/46C	RB-FZ-01D	Drywell Hi Press Control		X	C-1.9
DP	PS-RV/46C	RB-FZ-01D	Drywell Hi Press Switch	X		C-1.9
DP	PS-RV/46D	RB-FZ-01D	Drywell Hi Press Control		X	C-1.9
DP	PS-RV/46D	RB-FZ-01D	Drywell Hi Press Control		X	C-1.9
DP	PS-RV/46D	RB-FZ-01D	Drywell Hi Press Switch	X		C-1.9
IC	PSC A1	RB-FZ-01D	IC A isolation condensate line PS 1 (11A1)	X		OCP
IC	PSC A2	RB-FZ-01D	IC A isolation condensate line PS 1 (11A2)	X		OCP
IC	PSC B1	RB-FZ-01D	IC B isolation condensate line PS 1 (11B1)	X		OCP
IC	PSC B2	RB-FZ-01D	IC B isolation condensate line PS 1 (11B2)	X		OCP
IC	PSS A1	RB-FZ-01D	IC A iso. steamline press. switch 1 (5A1)	X		OCP
IC	PSS A2	RB-FZ-01D	IC A iso. steamline press. switch 2 (5A2)	X		OCP
IC	PSS B1	RB-FZ-01D	IC A iso. steamline press. switch 1 (5B1)	X		OCP
IC	PSS B2	RB-FZ-01D	IC A iso. steamline press. switch 1 (5B2)	X		OCP
IC	SP1A1D	RB-FZ-01D	IC Condensate Line DP Switch 11A1	X		OCP
IC	SP1A2D	RB-FZ-01D	IC Condensate Line DP Switch 11A2	X		OCP

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
IC	SP1B1D	RB-FZ-01D	IC Condensate Line DP Switch 11B1	X		OCP
IC	SP1B2D	RB-FZ-01D	IC Condensate Line DP Switch 11B2	X		OCP
IC	SP5A1D	RB-FZ-01D	IC Steam Line DP Switch 5A1	X		OCP
IC	SP5A2D	RB-FZ-01D	IC Steam Line DP Switch 5A2	X		OCP
IC	SP5B1D	RB-FZ-01D	IC Steam Line DP Switch 5B1	X		OCP
IC	SP5B2D	RB-FZ-01D	IC Steam Line DP Switch 5B2	X		OCP
IC	V-14-30	RB-FZ-01D	IC A Steam Line Valve (AC)		X	C-1.6
IC	V-14-31	RB-FZ-01D	IC A Steam Line Valve (DC)		X	C-1.6
IC	V-14-32	RB-FZ-01D	IC B Steam Line Valve (AC)		X	C-1.6
IC	V-14-33	RB-FZ-01D	IC B Steam Line Valve (DC)		X	C-1.6
IC	V-14-34	RB-FZ-01D	IC A Condensate Valve (DC)		X	C-1.6
IC	V-14-35	RB-FZ-01D	IC B Condensate Valve (DC)		X	C-1.6
IC	V-14-37	RB-FZ-01D	B IC condensate return valve (AC)		X	C-1.6
IC	IC A Isol	RB-FZ-01D	IC A isolation logic cable 63-361		X	B-3.7
IC	IC B Isol	RB-FZ-01D	IC B isolation logic cable 63-361		X	B-3.7
OV	V-28-15	RB-FZ-01D	General Area Supply	X		MSR
OV	V-28-16	RB-FZ-01D	General Area Supply	X		MSR
PI	V-23-13	RB-FZ-01D	Air-Operated Valve V-23-13		X	JMM-07
PI	V-23-14	RB-FZ-01D	Air-Operated Valve V-23-14		X	JMM-08
PI	V-23-17	RB-FZ-01D	Air-Operated Valve V-23-17		X	JMM-11
PI	V-23-18	RB-FZ-01D	Air-Operated Valve V-23-18		X	JMM-12
PI	V-27-03	RB-FZ-01D	Air-Operated Valve V-27-03		X	JMM-21
PI	V-27-04	RB-FZ-01D	Air-Operated Valve V-27-04		X	JMM-22
PI	V-38-09	RB-FZ-01D	Air-Operated Valve V-38-09		X	JMM-26
PI	V-38-10	RB-FZ-01D	Air-Operated Valve V-38-10		X	JMM-29
PI	V-38-16	RB-FZ-01D	Air-Operated Valve V-38-16		X	JMM-33
PI	V-38-17	RB-FZ-01D	Air-Operated Valve V-38-17		X	JMM-36
PI	V-38-37	RB-FZ-01D	Air-Operated Valve V-38-37		X	JMM-38
PI	V-38-38	RB-FZ-01D	Air-Operated Valve V-38-38		X	JMM-39
PI	V-38-39	RB-FZ-01D	Air-Operated Valve V-38-39		X	JMM-40
PI	V-38-40	RB-FZ-01D	Air-Operated Valve V-38-40		X	JMM-41
PI	V-38-43	RB-FZ-01D	Air-Operated Valve V-38-43		X	JMM-43
PI	V-38-46	RB-FZ-01D	Air-Operated Valve V-38-46		X	JMM-45
PR	PT-RE-15A	RB-FZ-01D	Reactor Pressure Switch RE15A	X		JMM-46
PR	PT-RE-15B	RB-FZ-01D	Reactor Pressure Switch RE15B	X		JMM-47
PR	PT-RE-15C	RB-FZ-01D	Reactor Pressure Switch RE15C	X		JMM-48
PR	PT-RE-15D	RB-FZ-01D	Reactor Pressure Switch RE15D	X		JMM-49
RI	V-28-13	RB-FZ-01D	General Area Supply	X		MSR
RI	V-28-14	RB-FZ-01D	General Area Supply	X		MSR
RI	V-28-15	RB-FZ-01D	General Area Supply	X		MSR
RI	V-28-16	RB-FZ-01D	General Area Supply	X		MSR
RL	RE02A	RB-FZ-01D	RPV Lo-Lo Level Xmitter	X	X	C-1.11
RL	RE02B	RB-FZ-01D	RPV Lo-Lo Level Xmitter	X	X	C-1.11
RL	RE02C	RB-FZ-01D	RPV Lo-Lo Level Xmitter	X	X	C-1.11
RL	RE02D	RB-FZ-01D	RPV Lo-Lo Level Xmitter	X	X	C-1.11
SD	BCCW Pump 1-	RB-FZ-01D	RBCCW Pump 1-1	X	X	B-5.29
SD	BCCW Pump 1-	RB-FZ-01D	RBCCW Pump 1-2	X	X	B-5.29
VO	IA83A	RB-FZ-01D	Press Control for NR108A	X	X	C-1.7
VO	IA83B	RB-FZ-01D	Press Control for NR108B	X	X	C-1.7
VO	IA83C	RB-FZ-01D	Press Control for NR108C	X	X	C-1.7
VO	IA83D	RB-FZ-01D	Press Control for NR108D	X	X	C-1.8
VO	IA83E	RB-FZ-01D	Press Control for NR108E	X	X	C-1.8
VR	IA83A	RB-FZ-01D	Press Control for NR108A	X	X	C-1.7
VR	IA83B	RB-FZ-01D	Press Control for NR108B	X	X	C-1.7

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
VR	IA83C	RB-FZ-01D	Press Control for NR108C	X	X	C-1.7
VR	IA83D	RB-FZ-01D	Press Control for NR108D	X	X	C-1.8
VR	IA83E	RB-FZ-01D	Press Control for NR108E	X	X	C-1.8
Fire Zone RB-FZ-01E			**Reactor Building Main Floor (23' Elevation)**			**Location 29**
AD	DPS-RV40A	RB-FZ-01E	CS Booster Pump Disch Press		X	C-1.15
AD	DPS-RV40B	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.15
AD	DPS-RV40D	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.15
AD	NR108A	RB-FZ-01E	EMRV (NR108A) V-1-173		X	C-1.16
AD	NR108B	RB-FZ-01E	EMRV (NR108B) V-1-174		X	C-1.16
AD	NR108C	RB-FZ-01E	EMRV (NR108C) V-1-175		X	C-1.16
AD	NR108D	RB-FZ-01E	EMRV (NR108D) V-1-176		X	C-1.16
AD	NR108E	RB-FZ-01E	EMRV (NR108E) V-1-177		X	C-1.16
CC	PMP 51A	RB-FZ-01E	Motor Feeder		X	C-1.20
CC	PMP 51B	RB-FZ-01E	Motor Feeder		X	C-1.20
CC	PMP 51C	RB-FZ-01E	Motor Feeder		X	C-1.20
CC	PMP 51D	RB-FZ-01E	Motor Feeder		X	C-1.20
CC	PS-15A	RB-FZ-01E	Low Flow Pressure Switch PS17A		X	JMM-01
CC	PS-15B	RB-FZ-01E	Low Flow Pressure Switch PS17B		X	JMM-02
CC	PS-IP15B	RB-FZ-01E	Drywell Pressure Switch Control Ckt		X	C-1.20
CC	PS-IP15D	RB-FZ-01E	Drywell Pressure Switch Control Ckt		X	C-1.20
CC	V-21-01	RB-FZ-01E	Containment Spray Suction Valve		X	C-1.20
CC	V-21-03	RB-FZ-01E	Containment Spray Suction Valve		X	C-1.20
CC	V-21-05	RB-FZ-01E	Drywell Spray Valve - System II	X	X	C-1.20
CC	V-21-07	RB-FZ-01E	Pump 51B(1-2) Suction Valve V-21-7		X	C-1.20
CC	V-21-09	RB-FZ-01E	Pump 51A(1-1) Suction Valve V-21-9		X	C-1.20
CC	V-21-11	RB-FZ-01E	System 1 drywell isolation		X	C-1.20
CC	V-21-13	RB-FZ-01E	Dynamic Test Valve II	X	X	C-1.20
CC	V-21-17	RB-FZ-01E	Dynamic Test Valve I	X	X	C-1.21
CC	V-3-87	RB-FZ-01E	ESW HX Outlet (ESW side) - II	X	X	C-1.21
CC	V-3-88	RB-FZ-01E	ESW Sys 1 HX Outlet (ESW side)	X	X	C-1.21
CS	DPS-RV40A	RB-FZ-01E	CS Booster Pump Disch Press		X	C-1.15
CS	DPS-RV40B	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.15
CS	DPS-RV40D	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.15
CS	NZ03B	RB-FZ-01E	CS Booster Pump B - II	X	X	C-1.15
CS	NZ03D	RB-FZ-01E	CS Booster Pump D - II	X	X	C-1.15
CS	PS-RV-29B	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.25
CS	PS-RV-29D	RB-FZ-01E	CS Booster Pump Disch Press	X	X	C-1.25
CS	PT-RV-29B	RB-FZ-01E	CS BP Disch Press Xmitter	X	X	C-1.25
CS	V-20-03	RB-FZ-01E	CS Pump A Suction Valve		X	C-1.15
CS	V-20-04	RB-FZ-01E	Core Spray Pump Suction MOV V-20-4		X	C-1.15
CS	V-20-21	RB-FZ-01E	CS II Containment Isolation		X	C-1.15
CS	V-20-32	RB-FZ-01E	CS Pump C Suction Valve		X	C-1.15
CS	V-20-33	RB-FZ-01E	Core Spray Pump Suction MOV V-20-33		X	C-1.15
CS	V-20-40	RB-FZ-01E	CS I Containment Isolation		X	C-1.15
CS	V-20-41	RB-FZ-01E	CS II Containment Isolation		X	C-1.15
CS	V-20-92	RB-FZ-01E	Min. Flow Valve	X		WD
CS	V-20-93	RB-FZ-01E	Min. Flow Valve	X		MSR
CS	V-20-95	RB-FZ-01E	Min. Flow Valve	X		MSR
DB	MCC DC-1	RB-FZ-01E	125 VDC MCC	X	X	C-1.21
DP	PS-RV/46B	RB-FZ-01E	ADS Control Logic		X	C-1.17
DP	PS-RV/46D	RB-FZ-01E	ADS Control Logic		X	C-1.17
IC	V-14-30	RB-FZ-01E	IC A Steam Line Valve (AC)		X	C-1.16
IC	V-14-31	RB-FZ-01E	IC A Steam Line Valve (DC)		X	C-1.16
IC	V-14-33	RB-FZ-01E	IC B Steam Line Valve (DC)		X	C-1.16
IC	V-14-34	RB-FZ-01E	IC A Condensate Valve (DC)		X	C-1.16
IC	V-14-35	RB-FZ-01E	IC A Condensate Valve (DC)		X	C-1.16

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
IC	V-14-36	RB-FZ-01E	A IC condensate return valve (AC)		X	C-1.16
IC	V-14-37	RB-FZ-01E	B IC condensate return valve (AC)		X	C-1.16
IC	IC A isol	RB-FZ-01E	IC A isolation logic cable 63-359		X	B-3.7
ME	NS03A	RB-FZ-01E	Inboard MSIVs		X	C-1.22
ME	NS03B	RB-FZ-01E	Inboard MSIVs		X	C-1.22
MS	NS03A	RB-FZ-01E	Inboard MSIVs		X	C-1.22
MS	NS03B	RB-FZ-01E	Inboard MSIVs		X	C-1.22
OV	V-27-01	RB-FZ-01E	Air-Operated Valve V-27-01		X	JMM-19
OV	V-27-02	RB-FZ-01E	Air-Operated Valve V-27-02		X	JMM-20
PI	V-22-01	RB-FZ-01E	Air-Operated Valve V-22-1		X	JMM-03
PI	V-22-02	RB-FZ-01E	Air-Operated Valve V-22-2		X	JMM-04
PI	V-22-28	RB-FZ-01E	Air-Operated Valve V-22-28		X	JMM-05
PI	V-22-29	RB-FZ-01E	Air-Operated Valve V-22-29		X	JMM-06
PI	V-23-15	RB-FZ-01E	Air-Operated Valve V-23-15	X		JMM-09
PI	V-23-16	RB-FZ-01E	Air-Operated Valve V-23-16	X		JMM-10
PI	V-23-19	RB-FZ-01E	Air-Operated Valve V-23-19	X		JMM-13
PI	V-23-20	RB-FZ-01E	Air-Operated Valve V-23-20	X	X	JMM-14
PI	V-23-21	RB-FZ-01E	Air-Operated Valve V-23-21		X	JMM-15
PI	V-23-22	RB-FZ-01E	Air-Operated Valve V-23-22		X	JMM-16
PI	V-26-16	RB-FZ-01E	Air-Operated Valve V-26-16	X		JMM-17
PI	V-26-18	RB-FZ-01E	Air-Operated Valve V-26-18	X		JMM-18
PI	V-28-17	RB-FZ-01E	Air-Operated Valve V-28-17		X	JMM-23
PI	V-28-18	RB-FZ-01E	Air-Operated Valve V-28-18		X	JMM-24
PI	V-28-47	RB-FZ-01E	Air-Operated Valve V-28-47		X	JMM-25
PI	V-38-09	RB-FZ-01E	Air-Operated Valve V-38-09	X		JMM-26
PI	V-38-10	RB-FZ-01E	Air-Operated Valve V-38-10	X		JMM-28
PI	V-38-16	RB-FZ-01E	Air-Operated Valve V-38-16	X		JMM-32
PI	V-38-17	RB-FZ-01E	Air-Operated Valve V-38-17	X		JMM-36
PI	V-38-37	RB-FZ-01E	Air-Operated Valve V-38-37	X		JMM-38
SD	NU02A	RB-FZ-01E	SDC pump A		X	B-3.15
SD	NU02B	RB-FZ-01E	SDC pump B		X	B-3.15
SD	NU02C	RB-FZ-01E	SDC pump C		X	ABN
SD	RBCCW Pump 1	RB-FZ-01E	RBCCW Pump 1-1		X	B-5.29
SD	RBCCW Pump 2	RB-FZ-01E	RBCCW Pump 1-2		X	B-5.29
SD	V-17-001	RB-FZ-01E	SDC suction valve		X	ABN
SD	V-17-002	RB-FZ-01E	SDC suction valve		X	B-3.16
SD	V-17-003	RB-FZ-01E	SDC suction valve		X	ABN
SD	V-17-019	RB-FZ-01E	SDC supply isolation valve		X	B-3.16
SD	V-17-054	RB-FZ-01E	SDC return isolation valve		X	B-3.16
SD	V-17-056	RB-FZ-01E	SDC discharge valve		X	B-3.15
SD	V-17-057	RB-FZ-01E	SDC discharge valve		X	ABN
VO	IA83A	RB-FZ-01E	Press Control for NR108A		X	C-1.16
VO	IA83B	RB-FZ-01E	Press Control for NR108B		X	C-1.17
VO	IA83C	RB-FZ-01E	Press Control for NR108C		X	C-1.17
VO	IA83D	RB-FZ-01E	Press Control for NR108D		X	C-1.17
VO	NR108A	RB-FZ-01E	EMRV (NR108A) V-1-173		X	C-1.16
VO	NR108B	RB-FZ-01E	EMRV (NR108B) V-1-174		X	C-1.16
VO	NR108C	RB-FZ-01E	EMRV (NR108C) V-1-175		X	C-1.16
VO	NR108D	RB-FZ-01E	EMRV (NR108D) V-1-176		X	C-1.16
VO	NR108E	RB-FZ-01E	EMRV (NR108E) V-1-177		X	C-1.16
VR	IA83A	RB-FZ-01E	Press Control for NR108A		X	C-1.16
VR	IA83B	RB-FZ-01E	Press Control for NR108B		X	C-1.17
VR	IA83C	RB-FZ-01E	Press Control for NR108C		X	C-1.17
VR	IA83D	RB-FZ-01E	Press Control for NR108D		X	C-1.17
VR	NR108A	RB-FZ-01E	EMRV (NR108A) V-1-173		X	C-1.16
VR	NR108B	RB-FZ-01E	EMRV (NR108B) V-1-174		X	C-1.16

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
VR	NR108C	RB-FZ-01E	EMRV (NR108C) V-1-175		X	C-1.16
VR	NR108D	RB-FZ-01E	EMRV (NR108D) V-1-176		X	C-1.16
VR	NR108E	RB-FZ-01E	EMRV (NR108E) V-1-177		X	C-1.16
Fire Zone RB-FZ-01F			**Torus Area and Corner Rooms (-19' Elevation)**			**Location 30**
OV	V-27-01	RB-FZ-01F	Air-Operated Valve V-27-01	X		JMM-19
OV	V-27-02	RB-FZ-01F	Air-Operated Valve V-27-02	X		JMM-20
PI	V-23-21	RB-FZ-01F	Air-Operated Valve V-23-21	X		JMM-15
PI	V-23-22	RB-FZ-01F	Air-Operated Valve V-23-22	X		JMM-16
PI	V-28-17	RB-FZ-01F	Air-Operated Valve V-28-17	X		JMM-23
PI	V-28-18	RB-FZ-01F	Air-Operated Valve V-28-18	X		JMM-24
PI	V-28-47	RB-FZ-01F	Air-Operated Valve V-28-47	X		JMM-25
CC	PMP 51C	RB-FZ-01F1	Containment Spray Pump 1-3	X	X	C-1.24
CC	PMP 51D	RB-FZ-01F1	Containment Spray Pump 1-4	X	X	C-1.24
CC	V-21-01	RB-FZ-01F1	Containment Spray Suction Valve	X	X	C-1.24
CC	V-21-03	RB-FZ-01F1	Containment Spray Suction Valve	X	X	C-1.24
CS	NZ01B	RB-FZ-01F2	Core Spray Pump B	X	X	C-1.25
CS	NZ01D	RB-FZ-01F2	Core Spray Pump D	X	X	C-1.25
CS	PS-RV-29B	RB-FZ-01F2	CS Booster Pump Disch Press	X	X	C-1.25
CS	PS-RV-29D	RB-FZ-01F2	CS Booster Pump Disch Press	X	X	C-1.25
CS	V-20-04	RB-FZ-01F2	Core Spray Pump B Suction Valve	X	X	C-1.25
CS	V-20-33	RB-FZ-01F2	Core Spray Pump D Suction Valve	X	X	C-1.25
ED	SWGR 1D	RB-FZ-01F2	Control Power to 4160 VAC Bus 1D		X	C-1.25
CD	NC08A	RB-FZ-01F3	CRD Pump A	X	X	C-1.26
CD	NC08B	RB-FZ-01F3	CRD Pump B	X	X	C-1.26
CS	NZ01A	RB-FZ-01F3	CS Pump A	X	X	C-1.26
CS	NZ01C	RB-FZ-01F3	CS Pump C	X	X	C-1.26
CS	PS-RV-29A	RB-FZ-01F3	CS I Disch Pressure Switch	X	X	C-1.26
CS	PS-RV-29C	RB-FZ-01F3	CS I Disch Pressure Switch	X	X	C-1.26
CS	V-20-03	RB-FZ-01F3	CS Pump A Suction Valve	X	X	C-1.26
CS	V-20-32	RB-FZ-01F3	CS Pump C Suction Valve	X	X	C-1.26
CC	PMP 51A	RB-FZ-01F4	Containment Spray Pump 1-1	X	X	C-1.27
CC	PMP 51B	RB-FZ-01F4	Containment Spray Pump 1-2	X	X	C-1.27
CC	V-21-07	RB-FZ-01F4	Containment Spray Suction Valve	X	X	C-1.27
CC	V-21-09	RB-FZ-01F4	Containment Spray Suction Valve	X	X	C-1.27
CS	NZ01B	RB-FZ-01F5	Core Spray Pump B		X	C-1.28
CS	NZ01D	RB-FZ-01F5	Core Spray Pump D		X	C-1.28
Fire Zone RB-FZ-01G			**Shutdown Cooling Area (38' and 51' Elevations)**			**Location 31**
PI	V-22-01	RB-FZ-01G	Air-Operated Valve V-22-1	X		JMM-03
PI	V-22-02	RB-FZ-01G	Air-Operated Valve V-22-2	X		JMM-04
PI	V-22-28	RB-FZ-01G	Air-Operated Valve V-22-28	X		JMM-05
PI	V-22-29	RB-FZ-01G	Air-Operated Valve V-22-29	X		JMM-06
SD	NU02A	RB-FZ-01G	SDC pump A	X	X	B-3.15
SD	NU02B	RB-FZ-01G	SDC pump B	X	X	B-3.15
SD	NU02C	RB-FZ-01G	SDC pump C	X	X	ABN
SD	V-17-001	RB-FZ-01G	SDC suction valve	X	X	B-3.16
SD	V-17-002	RB-FZ-01G	SDC suction valve	X	X	B-3.16
SD	V-17-003	RB-FZ-01G	SDC suction valve	X	X	ABN
SD	V-17-055	RB-FZ-01G	SDC discharge valve	X	X	B-3.16
SD	V-17-056	RB-FZ-01G	SDC discharge valve	X	X	B-3.15
SD	V-17-057	RB-FZ-01G	SDC discharge valve	X	X	ABN
SD	V-17-205	RB-FZ-01G	SDC pump A discharge valve	X	X	B-3.16
SD	V-17-206	RB-FZ-01G	SDC pump B discharge valve	X	X	B-3.16
SD	V-17-207	RB-FZ-01G	SDC pump C discharge valve	X	X	ABN

Table 4.4-3 Location of Risk Significant Components and Associated Cables						
Top Event	Component Identifier	Fire Area or Zone	Component Description	Component	Associated Cables	Reference
Fire Zone RB-FZ-01H			**Trunnion Room (23' 6" Elevation)**			**Location 32**
ME	NS04A	RB-FZ-01H	Outboard MSIV (V-1-9)	X	X	C-1.31
ME	NS04B	RB-FZ-01H	Outboard MSIV (V-1-10)	X	X	C-1.31
MS	NS04A	RB-FZ-01H	Outboard MSIV (V-1-9)	X	X	C-1.31
MS	NS04B	RB-FZ-01H	Outboard MSIV (V-1-10)	X	X	C-1.31
Fire Area MT-FA-12			**Main Transformer and CST Area**			**Location 34**
CT	PMP 1-1	MT-FA-12	CST Pump 1-1	X	X	ABN-74
CT	PMP 1-2	MT-FA-12	CST Pump 1-2	X	X	C-12
CW	P-03-002A	MT-FA-12	Circulating Water Pump 1-1		X	JMM-54
CW	P-03-002B	MT-FA-12	Circulating Water Pump 1-2		X	JMM-58
CW	P-03-002C	MT-FA-12	Circulating Water Pump 1-3		X	JMM-63
CW	P-03-002D	MT-FA-12	Circulating Water Pump 1-4		X	JMM-68
EA	SA	MT-FA-12	Startup Transformer SA	X		D01-001
EB	SB	MT-FA-12	Startup Transformer SB	X		D01-001
Fire Area DG-FA-15			**No. 1 Diesel Generator Room**			**Location 35**
EC	EDG-1	DG-FA-15	Emergency Diesel Generator	X	X	C-15
Fire Area DG-FA-17			**No. 2 Diesel Generator Room**			**Location 36**
ED	EDG-2	DG-FA-17	Emergency Diesel Generator	X	X	C-17
Fire Area FS-FA-16			**Emergency Diesel Generator Fuel Oil Storage Area**			**Location 37**
EC	EDG-1	FS-FA-16	Diesel Fuel Oil Storage	X	X	C-15
ED	EDG-2	FS-FA-16	Diesel Fuel Oil Storage	X	X	C-17
Fire Area CW-FA-14			**Circulating Water Intake Area**			**Location 38**
CC	PMP 52A	CW-FA-14	ESW Pump 1-1	X	X	C-14.2
CC	PMP 52B	CW-FA-14	ESW Pump 1-2	X	X	C-14.2
CC	PMP 52C	CW-FA-14	ESW Pump 1-3	X	X	C-14.2
CC	PMP 52D	CW-FA-14	ESW Pump 1-4	X	X	C-14.2
CW	P-03-002A	CW-FA-14	Circulating Water Pump 1-1	X		JMM-50
CW	P-03-002B	CW-FA-14	Circulating Water Pump 1-2	X		JMM-56
CW	P-03-002C	CW-FA-14	Circulating Water Pump 1-3	X		JMM-61
CW	P-03-002D	CW-FA-14	Circulating Water Pump 1-4	X		JMM-66
CW	V-3-08	CW-FA-14	Discharge Valve MOV V-3-8	X		MSB
CW	V-3-09	CW-FA-14	Discharge Valve MOV V-3-9	X		MSB
CW	V-3-10	CW-FA-14	Discharge Valve MOV V-3-10	X		MSB
CW	V-3-11	CW-FA-14	Discharge Valve MOV V-3-11	X		MSB
FP	DFP 1-1	CW-FA-14	Diesel Fire Pump		X	C-14.2
SW	PMP 1-1	CW-FA-14	SW Pump 1-1	X	X	C-14.1
SW	PMP 1-2	CW-FA-14	SW Pump 1-2	X	X	C-14.1

4.5 Fire Detection and Suppression

The Oyster Creek fire detection and suppression systems are discussed and documented in various sections of this report. This information is collected and presented here to provide a single section where all information on Oyster Creek fire detection and suppression can be reviewed. The documentation of the Oyster Creek fire detection and suppression systems is divided into two sub-sections. The first presents the automatic detection and suppression systems. The second presents the manual fire suppression equipment.

4.5.1 Automatic Detection and Suppression Systems

A diverse variety of automatic detection and suppression systems are employed to fight fires at Oyster Creek. The type of fire fighting system used in a given area depends on the type of equipment in the area (fire ignition sources), the combustible loading and the potential effects of the use of a particular fire fighting system (type of targets). Often, fire fighting systems are installed to prevent the spread of a fire from its origin and therefore are installed over the most significant source (or possibly the most significant target). The paragraphs are arranged below by fire area and zone and present the type of detection and suppression system installed. Only fire areas which were not screened from further consideration in either Section 4.1 (fire events which do not result in plant trip) or Section 4.2 (fire events which do not impact risk significant components or cables) are documented.

OFFICE BUILDING FIRE AREAS AND ZONES

OB-FZ-4 Cable Spreading Room - 36' Elevation

This area is protected by a particles of combustion detection system, with an automatic open head water spray system to suppress a cable fire and limit any water damage to other electrical equipment. Also, Maronite boards are installed to direct water streams to affected fire sources only, preventing any collateral damage due to water sprays.

OB-FZ-5 Control Room - 46' 6" Elevation

In addition to being constantly occupied by watch section personnel, this area is protected by ionization detection systems, with three high-flow Halon systems to protect critical panels. The Halon systems are actuated by cross-zoned particles of combustion detectors located within the panels. The main control panel sections protected by these Halon systems are separated by a fire barrier between panels 6R and 7R and panels 4F and 5F. The general functions performed with the controls on these panels are:

1F through 4F	Isolation condensers, core spray, reactor recirculation	Halon system A
1R through 6R	Control rod position control, ADS	Halon system A
5F through 9F	Main feedwater, turbine controls, instrument air	Halon system B
7R through 11R	Essential AC power, service and circulating water	Halon system B
10F through 11F	Primary containment control	Halon system C

OB-FZ-6A "A" 480 VAC Switchgear Room

This area is protected by ionization and photoelectric detection systems, with a fixed, total flooding Halon 1301 extinguishing system to suppress and to prevent the spread of fires.

OB-FZ-6B "B" 480 VAC Switchgear Room

Combustibles in this zone are protected by automatic Halon suppression, which is actuated by a cross zoned area wide fire detection system.

OB-FZ-8A/B MG Set Room/Mechanical Equipment Room

Combustibles in this zone are protected by an automatic closed head sprinkler system, which provides coverage for all areas of the zone, except for the south end, near the motor control centers. An alarm actuates in the Control Room on flow from the sprinkler system.

OB-FZ-8C A and B Battery Room, Tunnel and Electric Tray Room (35')

This area is protected by a particles of combustion detection system, with a fixed, total flooding Halon 1301 extinguishing system to prevent spread of fires.

OB-FA-9 Office Building

Products of combustion detectors are installed in the hallways of each floor. Detector actuation is annunciated locally and in the control room. Fire extinguishers are provided on each floor and hose stations are provided on the second and third floors.

OB-FZ-10A Monitoring and Change Room - 46'

Combustibles in this zone are protected by closed head sprinkler systems installed above the ceiling (in the computer room hallway to protect cable trays) and below the ceiling (in the monitoring and change room, the cable tray closet). These systems are automatically actuated by particles of combustion smoke detectors. These detectors are also annunciated in the Control Room.

OB-FZ-10B Chem Lab, Laundry, Instrument Shop - 35'

Area wide detection is provided in this zone, using ionization detectors. These detectors are also annunciated in the Control Room.

OB-FA-22A New Cable Spreading Room (Mech Equip Room) 63' 6" El.

Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by an automatic area wide closed head wet pipe sprinkler system.

OB-FA-22B North Cable Bridge Tunnel, 74' 6"

Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by a manually actuated area wide closed head preaction sprinkler system.

OB-FZ-22C South Cable Bridge Tunnel

Area wide fire detection is provided by ionization type POC detectors, which alarm locally and in the Control Room. Suppression is provided by a manually actuated area wide closed head preaction sprinkler system.

TURBINE BUILDING FIRE AREAS AND ZONES

TB-FA-3A 4160 VAC Switchgear 1C Vault (TB Mezz - 23')

This area is protected by a manually actuated, totally flooding CO_2 system, which is supplied by a low pressure CO_2 tank. The system is manually actuated by the operator upon receipt of an area fire alarm. Particles of combustion detectors are installed at the vault ceiling and alarm locally and in the control room.

TB-FA-3B 4160 VAC Switchgear 1D Vault (TB Mezz - 23')

This area is protected by a manually actuated, totally flooding CO_2 system, which is supplied by a low pressure CO_2 tank. The system is manually actuated by the operator upon receipt of an area fire alarm. Particles of combustion detectors are installed at the vault ceiling and alarm locally and in the control room.

TB-FZ-11A Turbine Operating Floor - 46' Elevation

Automatic suppression systems are provided for the lube oil and hydrogen combustibles located in this zone. Specifically, an automatic closed head sprinkler system is installed over the turbine bearing lift pumps and turbine bearings.

TB-FZ-11B Lube Oil Storage, Pumping and Purification Areas, 0' and 27'

This fire zone is protected by thermal detectors, which actuate closed head sprinkler systems over zone cable trays, a water spray system located over oil handling equipment and the oil storage tank and a closed head sprinkler system to protect the bearing lift pump.

TB-FZ-11C Switchgear Room, West End of TB on Mezz Level (Elev. 23' 6")

Fire detection is from an area wide particles of combustion detection system that alarms in the Control Room.

TB-FZ-11D Basement Floor, South End

This area is protected by an automatic closed head area wide sprinkler system, in addition to an automatic sprinkler system to protect the Hydrogen seal oil unit.

TB-FZ-11E Condenser Bay, Elevation 0' 6"

This area is protected by an automatic closed head area wide sprinkler system, except for the heater bay area.

TB-FZ-11F Feedwater Pump Area, 0'6" & 3'6" Levels

No fire detection or automatic suppression is installed in this area. Manual suppression is available from local fire extinguishers and hose stations.

TB-FZ-11G Mezz Level Southeast Corner and Machine Shop, Elev 23' 6"

A closed head sprinkler system protects the machine shop, office area and the area under the turbine.

TB-FZ-11H Demineralizer Tank and Steam Jet Air Ejector Area, Elev 23' 6"

No fire detection or automatic suppression is installed in this area. Manual suppression capability is provided by a hose station and CO_2 fire extinguishers. The opening to fire zone TB-FZ-11E is protected by a sprinkler system.

TB-FA-26 Battery Room South of 4160 VAC Switchgear

This area is protected by a particles of combustion detection system, with carbon dioxide and hose stations outside for manual fire suppression.

REACTOR BUILDING FIRE ZONES

RB-FZ-1A 119 Foot Elevation

Automatic wet pipe sprinkler protection is provided at ceiling level to provide fire suppression within this area.

RB-FZ-1B 95 Foot Elevation

Automatic fire detection is provided by ionization detectors.

RB-FZ-1C 75 Foot Elevation

Automatic fire detection is provided by ionization detectors. Also, a water curtain is installed to prevent fire growth to lower Reactor Building elevations.

RB-FZ-1D 51 Foot Elevation

This fire zone is protected by an ionization detection system, with a water curtain installed at ceiling elevation to prevent fire growth to RB-FZ-1C, above. Also, fixed automatic water spray systems are installed to protect the grouped cables.

RB-FZ-1E Main Floor (23 Foot Elevation)

This area is protected by an ionization detection system, with a water curtain installed at ceiling elevation to prevent fire growth to RB-FZ-1D. Also, fixed automatic water spray systems are installed to protect the grouped cables in this area.

RB-FZ-1F Torus Area & Corner Rooms (-19')

The fire protection for this zone is provided by a particles of combustion detection system.

RB-FZ-1G Shutdown Cooling Area (38' and 51')

Automatic fire detection for this area is provided by particles of combustion detectors.

RB-FZ-1H Trunnion Room (23' 6" Elevation)

No fire detection or automatic suppression is installed in this area. Manual fire suppression is available from local fire extinguishers and hose stations.

OTHER PLANT FIRE AREAS

MT-FA-12 Main Transformer and CST

The transformers are protected by a thermally actuated water spray system. Water spray protection is also provided where the main transformer buswork enters the Turbine Building.

DG-FA-15 No. 1 Diesel Generator Room

Thermally actuated rate of rise heat detectors and ionization type POC detection systems are installed in this area. These systems alarm locally and in the control room. Manual fire protection is provided by a fire hydrant and hose house with AFFF (foam) located approximately 100 feet north of the diesel generator building.

DG-FA-17 No. 2 Diesel Generator Room

Same as fire area DG-FA-15.

FS-FA-16 Emergency Diesel Fuel Storage Area

Same as fire area DG-FA-15.

CW-FA-14 Circulating Water Intake Area

No fire detection or suppression is installed in this area. Manual fire suppression is provided by a hydrant and a hose house located within 100 feet and CO_2 fire extinguishers. Transformers are diked to contain oil spills to the immediate area.

4.5.2 Manual Fire Fighting

Manual fire fighting is performed using portable fire extinguishers, which are distributed throughout the plant in clearly marked areas, manual actuation of installed systems and use of manual hose stations that are also distributed throughout the plant.

Plant personnel are assigned general responsibilities for the response to fire by the procedure "General Response to Fires", procedure number 120.4. This procedure gives guidance for response to fires in non-radiation controlled areas as well as radiation controlled areas. Plant personnel are assigned as follows:

1. The person discovering a fire has the responsibility to report the fire to the control room.

2. The control room operator general responsibilities include sounding the station fire alarm, announcing the fire location over the plant paging system and dispatching the fire brigade.

3. The group shift supervisor determines the applicability of the site emergency plan procedures beginning with EPIP-1, coordinates with the fire brigade leader to determine if the fire will impact safety related equipment and, if necessary, coordinates with the offsite fire department personnel.

4. The fire brigade leader and the fire brigade are responsible for the suppression of the fire including status reports to the control room.

5. The emergency director is responsible for taking actions as directed by the site emergency plan procedure EPIP-1.

In the case of a fire in a radiation controlled area responsibilities are also assigned for radiation control personnel.

Plant operation, including previously identified listings of equipment that could potentially be affected by fires in any given plant fire area or zone are identified in "Response to Fire" (procedure 2000-ABN-3200.29). This procedure directs the control room operator to evaluate plant conditions for entry into the plant Emergency Operating Procedures (EOPs) and, if necessary, to execute the fire response concurrently with plant shutdown through the guidance provided in the EOPs.

The control room operator is also directed by the "Response to Fire" procedure to perform notifications, in accordance with "Fires" procedure 120.4 if time permits. The operator is directed to dispatch a radio equipped operator to the fire scene when fire alarms are received without prior notification from someone in the plant. Upon confirmation of a fire the operator sounds the station fire alarm and announces the location of the fire over the plant paging system and directs the fire brigade to one of four pre-assigned turnout locations.

In addition to the procedures mentioned above several other procedures govern the Oyster Creek response to fires depending on the location and severity of the fire. These including:

2000-ABN-3200.30, Control Room Evacuation,
346, Operation of Local and Remote Shutdown Panels
120.7, Appendix "R" Emergency Preparedness
6430-IMP-1300.01, Classification of Emergency Conditions

Details on the training of the fire brigade and Sandia Fire Risk Scoping Study issues including, "manual fire fighting effectiveness", are contained in Section 4.8.

4.6 Analysis of Plant Systems, Sequences and Plant Response

This report section presents the quantitative evaluation tasks (Task 5, Task 6 and Task 7) as well as the Presentation of Results (Task 9). In Tasks 5, 6, and 7, the plant fire areas and zones which can potentially contribute to core damage due to plant fires during power operation are evaluated. The role of the quantitative evaluation tasks (Task 5, 6, and 7) and the presentation of results (Task 9) in the fire analysis development process are discussed below and illustrated on Figure 4-1 and Figure 4-4.

Develop and Quantify Fire Risk Model (Task 5)

Task 5, develops and quantifies the fire risk model. The fire risk model is developed by incorporating the fire initiating event impacts (Task 4) into the Level 1 OCPRA plant model. This model is then quantified using the fire initiating event frequencies (Task 1). Since fire suppression, mitigation or other mitigative features are not modeled, the result is an "all engulfing fire" in which all components and cables within the fire area or zone are assumed to fail. The core damage frequency estimate produced in this evaluation is termed an "upper bound core damage frequency". If the upper bound core damage frequency is less than $1x10^{-6}$ then the fire area or zone is screened from further consideration. This step in the quantitative evaluation is presented in report Section 4.6.1, Estimation of Upper Bound Core Damage Frequency.

Those fire areas or zones whose upper bound core damage frequency is greater than $1x10^{-6}$ are re-evaluated to determine if assumptions regarding the "all engulfing fire" can be relaxed. These conservatisms include allowing the recovery of 4160 VAC ventilation or the elimination of guaranteed failure events where control or instrument cables are damaged. Those fire areas or zones whose revised upper bound core damage frequency estimates are less than $1x10^{-6}$ are screened from further consideration. This step in the quantitative evaluation is depicted on Figure 4-1 and Figure 4-4 as a dotted line labeled "iterations". This step of the analysis is presented in report Section 4.6.2, Revised Estimation of Upper Bound Core Damage Frequency.

The results of the Development and Quantification of the Fire Risk Model (Task 5) is the Oyster Creek fire areas and zones which are screened from further consideration.

Detailed Fire Propagation Analysis (Task 7) and Fire Mitigation Potential (Task 6)

Those fire areas and zones whose initial or revised upper bound core damage frequency is greater than $1x10^{-6}$ (not screened) serve as input to the Detailed Fire Propagation Analysis. For each of these fire areas, the fire mitigation potential is evaluated (Task 6). This evaluation consists of the determination and application of mitigating factors such as the "fire severity factor" and/or detection and suppression probabilities. These tasks (Tasks 6 and 7) are documented in report Section 4.6.3, Detailed Evaluation of Core Damage Frequency due to Fires. The output of Task 7 is a list of fire areas and zones that can be screened from further consideration.

Presentation of Results (Task 9)

Input to the Presentation of Results (Task 9) is from the quantitative evaluation Tasks (Task 5, 6 and 7). The results of the quantitative tasks are outlined in report Section 4.6.4 and are used to address fire event related issues in report Sections 4.7, 4.8 and 4.9.

4.6.1 Estimation of Upper Bound Core Damage Frequency

The first step in the quantitative evaluation of fire areas and zones consists of an estimation of upper bound core damage frequency (illustrated on Figure 4-1 as Task 5). The core damage frequency produced is termed an "upper bound" since all fire events are assumed to be "all engulfing fires". That is, all fire events fail all equipment and cables in the fire area. Also, the failure of control, power or instrument cables which transit a fire area are assumed to fail due to the fire event and result in the guaranteed failure of the equipment to which they are associated. In addition, all failure modes are addressed including "hot shorts" which are assumed to cause the active failure of equipment.

The estimation of the upper bound core damage frequency is performed for the 35 fire areas that were not screened from further consideration in Section 4.1 (due to no impact on plant operation) or in Section 4.4 (due to the absence of risk significant components or associated cables). (See Table 4.4-3, Oyster Creek Fire Areas Screened from Further Consideration).

The evaluation of upper bound core damage frequency is performed by using the Level 1 plant model from the Oyster Creek PRA, as described in Section 7 of the Level 1 OCPRA report. The Level 1 plant model was then adapted to perform fire evaluations as follows:

- Unless fire-specific impacts indicate a specific form of plant trip, such as turbine trip, the fire risk model was quantified as a "general" transient. In other words, no initiator-specific changes to the model logic were incorporated. This treatment requires 4 of 5 EMRVs to open for all fire initiators. This is conservative for the case where the fire initiating event is best represented by a reactor trip or other initiating event which require less restrictive post trip pressure relief via the EMRVs. For example, a reactor trip initiated by a fire event requires no post trip pressure relief through the EMRVs when condenser vacuum is available (the most limiting case) and only 2 of 5 EMRVs with condenser vacuum unavailable.

- The Level 1 OCPRA plant model is adjusted to incorporate the fire initiating event impacts. The Fire Initiating Event Impact Table (Table 4.4-5) serves as the source of the impacts, in terms of degraded or failed top events. These impacts are incorporated into the Level 1 OCPRA plant model for each of the 35 fire initiating events. The result is a logic structure which models the fire initiating events.

- The fire initiating event frequencies calculated in Section 4.1 (Table 4.1-11) are input to the model. The incorporation of all of the above into the Level 1 OCPRA produces the "Fire Risk Model" which is then used to calculate an estimate of the upper bound core damage frequency for each fire area.

The core damage frequencies produced by this quantification of the fire risk model are extremely conservative due to the assumptions regarding "all engulfing fires", "hot short" failures and "general" transient initiator however this conservative value is adequate for screening purposes. The estimation of the upper bound core damage frequency is completed following the screening from further consideration those areas whose total upper bound core damage frequency less than $1x10^{-6}$. The estimates of upper bound core damage frequency of all 35 fire areas are shown in Table 4.6-1 with areas that screen from further consideration shown shaded.

Table 4.6-1 Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation					
Fire Area	Description	Failed Top Events	Degraded Top Events	Frequency	CDF[2]
OFFICE BUILDING FIRE AREAS AND ZONES					
OB-FZ-4	Cable Spreading Room - 36' Elevation	DB, EC, SW, DP, PR, RL, ME, MS, CP, CW, IC, VR, OL, PI, CD, CC, RC, OV, AD, CS		1.11E-02	1.1E-02
OB-FZ-5	Control Room - 46' 6" Elevation	DB, EC, SW, DP, PR, RL, ME, MS, CP, CW, IC, VR, OL, PI, CD, CC, RC, OV, AD, CS		1.16E-02	1.2E-02
OB-FZ-6A	"A" 480 VAC Switchgear Room	CC, RC, CD, CS, CW, DP, EC, OV, PI, RL, SD, VO	IC "A"	4.18E-03	1.5E-04
OB-FZ-6B	"B" 480 VAC Switchgear Room	ED, SD		4.00E-03	**3.1E-07**
OB-FZ-8A/B	MG Set Room/ Mechanical Equipment Room	PI, SD, VO, ED	1 train of CS, CC, RC, auto RPT (RP)	7.40E-03	**8.7E-07**
OB-FZ-8C	A and B Battery Room, Tunnel and Electric Tray Room (35' Elevation)	BI, CC, RC, CD, CW, DB, DP, EB, ED, PI, VO, OV		3.71E-03	1.3E-04
OB-FA-9	Office Building	PR, VO, XB, IC		3.96E-03	**2.3E-07**
OB-FZ-10A	Monitoring and Change Room - 46'	ED, AD, DP, PR, PI, XB		9.28E-03	2.4E-06
OB-FZ-10B	Chem Lab, Laundry, Instrument Shop - 35'	ED		8.68E-04	**6.8E-08**
OB-FA-22A	New Cable Spreading Room (Mech Equip Room) 63' 6" El.	PI, RL, PR, MU, XB, IC, ED		3.58E-03	3.9E-06
OB-FA-22B	North Cable Bridge Tunnel, 74' 6"	PI, RL, MU		1.69E-03	**3.3E-09**
OB-FZ-22C	South Cable Bridge Tunnel	PI, RL		1.69E-03	**6.9E-10**
TURBINE BUILDING FIRE AREAS AND ZONES					
TB-FA-3A	4160 VAC Switchgear 1C Vault (TB Mezz - 23')	EC, CC, RC		4.79E-03	**4.6E-07**
TB-FA-3B	4160 VAC Switchgear 1D Vault (TB Mezz - 23')	ED, CC, RC		3.64E-03	**4.2E-07**
TB-FZ-11A	Turbine Operating Floor - 46' Elevation	BT, BV		2.08E-02	**1.2E-08**
TB-FZ-11B	Lube Oil Storage, Pumping and Purification Areas, 0' and 27' Levels	EB, EC, SW, BV, XC	IC "B"	1.19E-02	4.3E-05
TB-FZ-11C	Switchgear Room, TB (Mezz Level - 23'6")	EA, EB, EC, SW, DC	IC "B"	3.28E-03	1.6E-05

Table 4.6-1 Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation					
Fire Area	Description	Failed Top Events	Degraded Top Events	Frequency	CDF [2]
TB-FZ-11D	Basement Floor, South End	CW, SW, CC, RC, FW, CS, IA, TB		1.09E-02	6.8E-04
TB-FZ-11E	Condenser Bay, Elevation 3' 6"	BT, CP, FW, CT, CC, RC, EC, SW	IC "B"	5.92E-03	1.7E-06
TB-FZ-11F	Feedwater Pump Area, 0'6" & 3'6" Levels	CP, ME, ED		8.07E-03	4.7E-06
TB-FZ-11G	Mezzanine Level Southeast Corner and Machine Shop, Elevation 23' 6"		1 train FW train 2 of CS, CC, RC	4.73E-03	2.0E-07
TB-FZ-11H	Demineralizer Tank and Steam Jet Air Ejector Area, Elevation 23' 6"	IA, CP, EC, ED	CS pumps B & D	4.73E-03	3.4E-04
TB-FA-26	Battery Room South of 4160 VAC Switchgear	DC		2.47E-03	1.5E-06
REACTOR BUILDING FIRE ZONES					
RB-FZ-1B	95 Foot Elevation	BI, RI, PI, MU		2.61E-03	8.4E-09
RB-FZ-1C	75 Foot Elevation	IC, RI, PI, CS2		1.03E-02	4.2E-08
RB-FZ-1D	51 Foot Elevation	DP, IC, OV, RI, PI, RL, PR, SD, VR, AD	Train 1 - CS, CC, RC	2.53E-02	2.4E-05
RB-FZ-1E	Main Floor (23 Foot Elevation)	CC, RC, DP, IC, SD, AD, CD, OV, PI	Train 1 CS, outboard MSIVs (ME MS)	1.55E-02	5.6E-05
RB-FZ-1F	Torus Area & Corner Rooms (-19')	CC, RC, CS, CD, ED, OV, PI		8.05E-03	2.9E-06
RB-FZ-1G	Shutdown Cooling Area (38' and 51')	SD, PI		3.39E-03	3.4E-09
RB-FZ-1H	Trunnion Room (23' 6" Elevation)	ME, MS		8.70E-04	4.2E-10
OTHER PLANT FIRE AREAS					
MT-FA-12	Main Transformer and CST	EA, EB, CT, CW		2.14E-02	6.1E-06
DG-FA-15	No. 1 Diesel Generator Room		Diesel Gen 1	1.50E-02	2.5E-08
DG-FA-17	No. 2 Diesel Generator Room		Diesel Gen 2	1.50E-02	2.3E-08
FS-FA-16	Emergency Diesel Fuel Storage Area		DG 1 and 2	1.02E-03	1.6E-08
CW-FA-14	Circulating Water Intake Area		CW, SW, CC, RC	1.05E-02	1.4E-06

NOTE 1: Shaded areas indicate that the fire area is screened from further consideration based on the calculated upper bound core damage frequency of less than 1×10^{-6} per year.

NOTE 2: The "CDF" column reports the upper bound core damage frequency. These values are only used for screening purposes and represent the highest possible core damage frequency for the fire initiating event.

4.6.2 Revised Upper Bound Core Damage Frequency

The revision of the upper bound core damage frequency consists of a review and potential relaxation of the conservative assumptions used during the initial evaluation of the upper bound core damage frequency. The revised estimation of upper bound core damage frequency is represented on Figure 4-1 as a diamond shape and the dashed line labeled "iterations". The eighteen (18) fire areas and zones which were not screened in the initial estimation of upper bound core damage frequency are analyzed here. For reference, seventeen (17) fire areas and zones were screened from further consideration in the initial evaluation of upper bound core damage frequency.

During the initial estimation of the upper bound core damage frequency, many assumptions were employed to simplify the analysis. These assumptions are related to the development and growth of the fire (phenomenolgical assumptions) and to the method chosen to quantify the upper bound core damage frequency (fire risk modeling assumptions). These assumptions are:

Phenomological Assumptions

- "All Engulfing Fire". It was assumed that all fires would totally engulf the affected area, damaging all plant equipment and cables in the area.

- It was also assumed that the failure of control, power and instrument cables (due to the "all engulfing fire") would fail the associated component.

- "Hot Short". All failure modes of equipment were addressing including hot shorts which are assumed to cause active failure of equipment.

Fire Risk Model Assumptions

- "General Transient". This assumption used the restrictive EMRV post trip pressure relief requirement (4 of 5 EMRVs open) for all fire initiators to simplify the fire risk model.

- Recovery actions are failed when the equipment of cables associated with the action are in or transit through the fire area.

- Fire detection and suppression is not modeled.

Each of the eighteen (18) fire areas and zones which were not screened in the initial evaluation are reviewed to determine if relaxation of one or more of the above conservative assumptions is appropriate. In addition, other fire risk model assumptions regarding potential recovery actions may also be addressed, such as allowing recovery for switchgear room ventilation failure. In each case, justification is made for the less conservative analysis.

It should be noted that the revised estimation of upper bound core damage frequency does NOT relax assumptions which concern the "all engulfing fire", "hot shorts" and fire detection and suppression. Relaxation of these assumptions is considered in the detailed evaluation of fire areas presented in report Section 4.2.3.

Following a review and relaxation of the appropriate assumptions the Fire Risk Model is re-quantified and those fire initiating events whose total upper bound core damage frequency contributes less than $1x10^{-6}$ are screened from further consideration.

Ten (10) of the eighteen (18) fire areas and zones are evaluated in the revised estimation of upper bound core damage report sections below. The remaining eight (8) fire areas and zones are analyzed in the detailed fire propagation analysis (report Section 4.6.3).

4.6.2.1 Revised Estimation of Upper Bound CDF for the Office Building Fire Areas

A total of six (6) office building fire zones have not been screened from further consideration in Step 1 (report Section 4.6.1). The evaluations for fire zones OB-FZ-10A and OB-FZ-22A are described in detail below the remaining office building fire area and zones are presented in the Detailed Analysis.

The core damage frequencies produced in this report sub-section are the result of the relaxation of one the conservative assumptions made in the initial estimation of upper bound core damage frequency. The revised estimation of upper bound core damage frequency is the result. The revised estimate remains an upper bound since fire detection and suppression is not modeled and assumptions regarding the "all engulfing fire" and "hot shorts" remain.

OB-FZ-10A (Monitoring and Change Room - 46 Foot Elevation) has an effective floor area of 2,019 square feet. The area is protected by particles of combustion detectors and closed head automatic sprinkler systems in the monitoring and change room itself, the cable tray closet and above the ceiling in the computer room hallway to protect cable trays.

This fire zone was modeled in the initial estimate of upper bound core damage frequency by failing 4160 VAC switchgear 1D, ADS actuation, high drywell and RPV actuation logic, primary containment isolation and long term 125 VDC power. (Top Events ED, AD, DP, PR, PI, and XB). These impacts gave an initial core damage frequency of greater than $1x10^{-6}$ per year ($2.4x10^{-6}$).

The assumed failure of essential 4160 VAC switchgear 1D is caused by possible damage to supply cables for switchgear room ventilation supply and exhaust fans. The recovery of room cooling by opening doors, installation of portable vent fans, or recovery of damage to cables, was not initially modeled. This form of recovery is simulated (numerically) within the fire risk model by assuming that the diesel generator is required for success of bus 1D for all scenarios. This results in a failure rate for this essential switchgear of approximately 0.066 (split faction EDD) or an increase of a factor of approximately 100 from the nominal, all support available case (split fraction ED1) and a factor of 20 decrease from the initial, assumed guaranteed failure, evaluation). As before, all other equipment and cables in the area are assumed to fail due to any and all fires in this area. Operator procedural guidance for this recovery action is provided in procedure 2000-ABN-3200.29, Response to Fire, on Attachment 1, page E1-71. This procedure also directs the operator to recovery actions in procedures 331 and 346.

The revised estimate of upper bound core damage frequency for fire zone OB-FZ-10A is less than $1x10^{-6}$ per year ($4.9x10^{-7}$) and the area can be screened from further consideration.

OB-FZ-22A (New Cable Spreading Room/Mechanical Equipment Room - 63' 6") has an effective floor area of 3,435 square feet and is protected by an ionization detection system and a closed head wet pipe automatic sprinkler system.

This fire zone was modeled in the initial estimate of upper bound core damage frequency by failing primary containment isolation, low-low RPV water level and high RPV pressure actuation logic, isolation condenser makeup, long term 125 volt DC power, both isolation condensers and 4160 VAC switchgear 1D. (Top Events PI, RL, PR, MU, XB, IC and ED). These impacts gave an initial upper bound core damage frequency greater than $1x10^{-6}$ per year ($3.9x10^{-6}$).

As in the case with fire zone OB-FZ-10A the failure of essential 4160 VAC switchgear 1D was caused by possible damage to supply cables for switchgear room ventilation supply and exhaust fans. The Response to Fire procedure, 2000-ABN-3200.29, directs the operator to the applicable sections of the Control Room Evacuation procedure (2000-ABN-3200.30) and Operation of Remote Shutdown Panels (Procedure 346) for all fires in the control room complex. Procedure 346 then provides direction on recovery of essential switchgear room ventilation, if necessary.

Since recovery of room cooling by opening doors, portable ventilation, or recovery of damage to cables, was not originally modeled, this form of recovery was simulated (numerically) within the fire risk model by assuming that the diesel generator is required for success of bus 1D for all scenarios. This effectively increases the failure rate for this essential switchgear by a factor of approximately 100 from the nominal, all support available case. As before, all other equipment and cables in the area are assumed to fail due to any and all fires in this area.

The revised estimate of upper bound core damage frequency for fire zone OB-FZ-22A is less than $1x10^{-6}$ per year ($2.6x10^{-7}$). Therefore, this fire zone can be screened from further consideration.

4.6.2.2 Revised Estimation of Upper Bound CDF for Turbine Building Fire Areas and Zones

A total of six (6) turbine building fire areas and zones (fire zones TB-FZ-11B, TB-FZ-11C, TB-FZ-11E, TB-FZ-11F, TB-FZ-11H and fire area TB-FA-26) have not been screened from further consideration in report Section 4.6.1. The evaluations for all six of these fire areas and zones are described in detail below.

TB-FZ-11B (Lube Oil Storage, Pumping and Purification Area) consists of a long open area, running along the north wall of the Turbine Building. This area measures approximately 22 by 150 feet and has an effective floor area of 3,175 square feet. It is protected by thermal detectors, which actuate closed head sprinkler systems over area cable trays, a water spray system located over oil handling equipment and the oil storage tank and a closed head sprinkler system to protect the bearing lift pump.

This fire zones modeled in the initial estimate of upper bound core damage frequency by failing 4160 VAC switchgear 1B and 1C, long term division 1 DC power, isolation condenser "B", service water and main turbine bypass control. (Top Events EB, EC, XC, IC "B", SW and BV). These impacts gave an initial upper bound core damage frequency of $4.3x10^{-5}$.

The assumed failure of essential 4160 VAC switchgear 1C is caused by possible damage to control cables for diesel generator 1 and possible loss of control room operation for the supply breakers for 480 VAC switchgear. It was noted during this review that protective functions would remain available for these circuit breakers and their associated buses, including undervoltage protection for bus 1A1, and local control of these supply breakers would remain available from the switchgear itself (operator guidance for recovery of control for this area is provided by Attachment 1, page E1-25 of "Response to Fire", procedure 2000, ABN-3200.29). Therefore, the guaranteed failure condition, which was assumed was relaxed to an assumed failure of diesel generator 1.

This evaluation remains an upper bound estimate since all fires are still modeled as being severe, and fire suppression and operator tripping the reactor (reducing the number of EMRVs required to actuate following plant trip) are not modeled. That is, all fires are modeled as "general" transients. The revised estimate of upper bound core damage frequency for fire zone TB-FZ-11B is less than $1x10^{-6}$ per year ($3.4x10^{-7}$) and the area can be screened from further consideration.

TB-FZ-11C (Switchgear Room, West End of Turbine Building on Mezzanine Level) is rectangular in shape and measures approximately 33 by 90 feet, with an effective floor area of 2,666 square feet. This area is protected by particles of combustion detectors, which actuate alarms in the Control Room.

This fire zone was modeled in the initial estimate of upper bound core damage frequency by failing 4160 VAC switchgear 1A, 1B and 1C, isolation condenser "B", service water and 125 Division 2 VDC power. (Top Events EA, EB, EC, IC "B", SW and DC). These impacts gave an initial upper bound core damage frequency of $1.56x10^{-5}$.

The assumed failure of both non-essential 4160 VAC buses would result in a loss of offsite power initiating event. The 125 VDC power cables in this area supply power to isolation condenser "B" valves (see FHAR Section 3) which is reflected in the failure of IC B at top event IC. Therefore, the guaranteed failure of short term division 1 DC power (top event DC) was removed from the model. It should be noted that, due to an assumed failure of essential 4160 VAC bus 1C, division 1 long term DC power (top event XC) is assumed to fail after battery discharge due to loss of power to the battery chargers. As before, all other equipment and cables in the area are assumed to fail due to any and all fires in this area.

The revised estimate of upper bound core damage frequency for fire zone TB-FZ-11C is less than $1x10^{-6}$ per year ($4.6x10^{-7}$) and this fire zone can be screened from further consideration.

TB-FZ-11E (Main Condenser Bay Area) is a large rectangular area, measuring approximately 140 by 180 feet, with an effective floor area of 26,427 square feet. This area is protected by closed head sprinklers located over cable trays throughout the condenser bay.

This fire zone was modeled in the initial estimate of upper bound core damage frequency by failing turbine bypass valve trip, main condensate and feedwater, condensate transfer, containment spray, 4160 VAC switchgear 1C, isolation condenser "B" and service water. (Top Events BT, CP, FW, CT, CC, RC, EC, IC "B" and SW). These impacts produced an initial upper bound core damage frequency of $1.8x10^{-6}$.

The assumed failure of essential 4160 VAC switchgear 1C was caused by possible damage to control cables for diesel generator 1 and possible loss of control room operation of supply breakers for 480V switchgear. It was noted during this review that protective functions would remain available for these circuit breakers and their associated buses, including undervoltage protection for bus 1A1, and local control of the supply breakers would remain available from the switchgear itself (operator guidance for recovery of control for this area is provided by Attachment 1, page E1-34 of "Response to Fire", procedure 2000, ABN-3200.29). Also, the circuit breaker control cables are routed through different cable trays (trays 6 and 9) than the diesel generator control circuits (tray 2 - see FHAR Appendix C-11). Therefore, the guaranteed failure condition, which was assumed during the initial evaluation, was relaxed to an assumed failure of diesel generator 1. Also, since main condensate and feedwater are assumed to fail, this was modeled as a loss of feedwater initiating event. As before, all other equipment and cables in the area are assumed to fail due to any and all fires in this area.

The revised estimate of upper bound core damage frequency for fire zone TB-FZ-11E is less than $1x10^{-6}$ per year ($8.9x10^{-7}$) and this fire zone can be screened from further consideration.

TB-FZ-11F (Main Feedwater Pump Area) is rectangular in shape, measuring approximately 100 by 65 feet, with an effective floor area of 5,650 square feet.

This area was initially modeled by failing main condensate, MSIV closure on low main steamline pressure (due to sensor failure) and 4160 VAC switchgear 1D. (Top Events CP, ME and ED). These impacts gave an initial core damage frequency of $4.74x10^{-6}$.

Review of the impacts that could be created by fires in this area showed that the assumed failure of main condensate would result in a loss of main feedwater initiating event. Also, the division 2 power cables located in this area (top event ED) are only required to trip the division 2 core spray pumps from the control room (see FHAR Volume 2, Section 1). Response to Fire (Procedure 2000-ABN-3200.29, page E1-36) directs the operator to open these circuit breakers in accordance with the Core Spray operating procedure (Procedure 308), provided that these pumps are not required to assure adequate core cooling. Therefore, the guaranteed failure of AC power (top event ED) was removed from this evaluation. As before, all other equipment and cables in the area are assumed to fail due to any and all fires in this area.

The revised estimate of upper bound core damage frequency for fire zone TB-FZ-11F is less than $1x10^{-6}$ per year ($1.1x10^{-7}$) and the area can be screened from further consideration.

TB-FZ-11H (Basement Southeast End) is rectangular in shape, with an effective floor area of 3,944 square feet.

This area was modeled in the initial estimation of upper bound core damage frequency by failing instrument air, main condensate and 4160 VAC switchgear 1C and 1D. (Top Events IA, CP, EC and ED). These impacts gave an initial upper bound core damage frequency of $3.36x10^{-4}$.

Review of the impacts that could be created by fires in this area showed that the division 1 power cables (power circuit 14-12) are routed in a conduit in a cable pull pit in this fire zone. The pit is covered with a steel plate and these circuits are separated from the general zone area by a vertical concrete wall and a horizontal 6 inch thick seal of RTV foam below the plate. There are no combustible materials located in the pit (see FHAR Volume 2, Section 3, page 39). Based on this review, the assumed impact on top event EC was removed from the evaluation. As before, all other equipment and cables in the area are assumed to fail due to all fires in this area.

The revised estimate of upper bound core damage frequency for fire zone TB-FZ-11H is less than $1x10^{-6}$ per year ($7.3x10^{-7}$) and the area can be screened from further consideration.

4.6.2.3 Revised Estimation of Upper Bound CDF of Reactor Building Fire Zones

One of the three remaining reactor building fire zones is evaluated (RB-FZ-1F) below. The CDF produced is the result of the relaxation of one of the conservative assumptions made in the initial estimation of upper bound core damage frequency. The revised estimation of upper bound core damage frequency is the result. The revised estimate remains an upper bound since fire detection and suppression is not modeled and assumptions regarding the "all engulfing fire" and "hot shorts" remain.

RB-FZ-1F (-19 Foot Elevation) is a large rectangular area, approximately 137 by 106 feet, with an effective floor area of 12,140 square feet. Review of the layout of RB-FZ-1F shows it to be a large open area, with 5 distinct compartments, consisting of 4 corner rooms, separated by the torus area. The risk-significant equipment for this area is primarily located in the corner rooms, with very little located in the torus area.

The fire protection for this zone is provided by a particles of combustion detection system. Fire suppression in this area is manual.

This fire zone modeled in the initial estimate of upper bound core damage frequency by failing containment spray, core spray, control rod drive, primary containment isolation, containment vent and essential 4160 VAC switchgear 1D. (Top Events CC, RC, CS, CD, PI, OV and ED). These impacts gave an initial upper bound core damage frequency of $2.9x10^{-6}$.

Since a fire spreading from one corner room too another requires crossing the torus area where combustible loading is low, the spread of a fire to all 5 areas was judged to be unlikely. Therefore, for purposes of a revised screening evaluation, all fires for fire zone RB-FZ-1F were modeled as starting in the northwest corner room, which results in potential impacts to the control rod drive and control power to essential 4160 VAC bus 1D, in addition to one train of core spray. It should be noted that this form of modeling assumes that 3 of the 4 core spray pumps are failed due to fire (2 main pumps located in the northwest corner room, plus one failed by loss of support to bus 1D).

The revised estimate of upper bound core damage frequency for this fire zone (RB-FZ-1F) is less than $1x10^{-6}$ per year ($9.0x10^{-7}$) and the area can be screened from further consideration.

4.6.2.4 Revised Estimation of Upper Bound CDF of Other Plant Fire Zones and Areas

Two plant areas designated as "other" plant areas, remain for further consideration (MT-FA-12 and CW-FA-14), following the initial screening evaluation performed in Section 4.6.1. Both of these areas are described in detail below.

MT-FA-12 (Main Transformer and CST Area) is an outdoor area, with the main transformers located approximately 32 feet from the Turbine Building wall. The transformers are protected by a thermally actuated water spray system. Water spray protection is also provided where the main transformer buswork enters the Turbine Building

This fire area was modeled in the initial estimate of upper bound core damage frequency with the "general" transient logic structure. Incorporating the assumed failure of non-essential 4160 VAC switchgear 1A and 1B (top events EA and EB), circulating water (top event CW) and condensate transfer (top event CT) in this model gave an initial damage frequency of 6.0×10^{-6}.

Upon review, this fire initiating event most closely resembles a loss of offsite power initiating event, which results in the removal of several conservatisms from the fire risk model. Most notably, a less stringent requirement for RPV pressure relief following plant trip (top event VO and, subsequently, EMRV closure at top event VR).

By evaluating this fire area (MT-FA-12) as a loss of offsite power, revised estimate of upper bound core damage frequency is less than 1×10^{-6} per year (8.1×10^{-7}) and the area can be screened from further consideration.

CW-FA-14 (Circulating Water/Intake Area) is an outside area, located approximately 100 feet from the nearest structure. Review of the layout of the intake structure area shows it to be open, with large spaces between components.

This fire area was modeled in the initial estimate of upper bound core damage frequency by failing circulating water, service water and containment spray. (Top Events CW, SW, CC and RC). These impacts gave an initial core damage frequency of 1.4×10^{-6}.

Due to the open layout of this area, recovery of at least one of the four installed ESW pumps within a 24 hour period is judged to be reasonable, particularly when all fires are still assumed to damage all other equipment and cables located in this fire area. During this review, it was noted that the cables for ESW pumps 52A and 52B are routed through different cable trays (D9 and D11 and conduits 14-33 and 14-34) from the cables for ESW pumps 52C and 52D (trays D6 and D8 and conduits 14-31 and 14-32 - see FHAR Appendix B-5, page 5).

By allowing long term recovery of at least one ESW pump, the revised estimate of upper bound core damage frequency for this fire area (CW-FA-14) is less than 1×10^{-6} per year (5.6×10^{-7}) and can be screened from further consideration.

As is the case with all the revised estimates of upper bound core damage frequency the analysis remains an upper bound since fire detection and suppression is not modeled and assumptions regarding the "all engulfing fire" and "hot shorts" remain.

Table 4.6-2 Revised Estimation of Upper Bound Core Damage Frequency					
Fire Area	Description	Degraded or Failed Top Events	Revised Impacts [3,4]	Fire Event Frequency	Upper Bound CDF [2]
OFFICE BUILDING FIRE AREAS AND ZONES					
OB-FZ-10A	Monitoring and Change Room - 46'	ED, AD, DP, PR, PI, XB	4160 VAC 1D Ventilation Recovery	9.28E-03	4.9E-07
OB-FA-22A	New Cable Spreading Room (Mech Equip Rm)	PI, RL, PR, MU, XB, IC, ED	4160 VAC 1D Ventilation Recovery	3.58E-03	2.6E-07
TURBINE BUILDING FIRE AREAS AND ZONES					
TB-FZ-11B	Lube Oil Storage, Pumping and Purification Areas	EB, EC, SW, BV, XC, Isolation Cond "B"	Same Except EC Success	1.19E-02	3.4E-07
TB-FZ-11C	Switchgear Room, West End of TB on Mezzanine Level (Elev. 23' 6")	EA, EB, EC, SW, DC Isolation Cond "B"	Same Except LOSP Event and DC Success	3.28E-03	4.6E-07
TB-FZ-11E	Condenser Bay (Elevation 3' 6")	BT, CP, FW, CT, CC, RC, EC, SW	Same Except EC Success	5.92E-03	8.9E-07
TB-FZ-11F	Feedwater Pump Area (0'6" & 3'6" Levels)	CP, ME, ED, Isolation Cond. "B"	Same Except LOFW Event and ED Success	8.07E-03	1.1E-07
TB-FZ-11H	Demin Tank and SJAE Area (23'6")	IA, CP, EC, ED	EC Success	4.73E-03	7.3E-07
REACTOR BUILDING FIRE ZONES					
RB-FZ-1F	Torus Area & Corner Rooms (-19')	CC, RC, CS, CD, ED, OV, PI	CD, ED Train 1 of CS and CC, RC	8.05E-03	9.0E-07
OTHER PLANT FIRE AREAS					
MT-FA-12	Main Transformer and CST	EA, EB, CT, CW	Same Except LOSP Event	2.14E-02	8.1E-07
CW-FA-14	Circulating Water Intake Area	CW, SW, CC, RC	Same Except RC Success	1.05E-02	5.6E-07

Note 2: The "Upper Bound CDF" column reports the upper bound core damage frequency. These values are for screening purposes only.

Note 3: The term "Success" refers to the non-guaranteed failure of the event. The independent failure probabilities are used in these cases.

Note 4: The term "Event" is used to describe the initiating event which is modeled in place of the "general" transient initiating event used in the fire risk model.

4.6.3 Detailed Evaluation of Core Damage Frequency Due to Fire

Eight (8) fire areas and zones are not screened from further consideration in either Section 4.6.1 (Estimate of Upper Bound Core Damage Frequency) or Section 4.6.2 (Revised Estimate of Upper Bound Core Damage Frequency). These seven fire areas are:

RB-FZ-1D	51 Foot Elevation
RB-FZ-1E	Main Floor (23 Foot Elevation)
OB-FZ-4	Cable Spreading Room - 36 Foot Elevation
OB-FZ-5	Control Room - 46' 6" Elevation
OB-FZ-6A	"A" 480 VAC Switchgear Room
OB-FZ-8C	A and B Battery Room, Tunnel and Electric Tray Room - 35 Foot Elevation)
TB-FZ-11D	Turbine Building, Basement Southend
TB-FA-26	Battery Room South of 4160 VAC Switchgear

Each of the above fire areas and zones are reviewed on a detailed basis for specific fire hazard and target locations, providing a more realistic evaluation of existing fire hazards in these areas. This detailed evaluation consists of the evaluation of fire risk model assumptions (impacted top events) as well as those assumptions regarding the growth and propagation of fires within the fire area or zone (fire severity factor, fire detection and suppression and the use of fire modeling computer codes). The evaluation of fire risk model impacts is discussed in previous report sections and adjustments of any impacts from earlier quantitative evaluations are described in the individual fire area paragraphs below. The assumptions regarding fire growth and propagation used in detailed analysis: fire severity factor, fire detection and suppression and fire modeling computer codes are detailed in the following sub-sections.

Fire Severity Factor

In overview, the "fire severity factor" is a numerical factor applied to the fire frequency based on a detailed review of the events which were considered in the database and used to calculate this fire frequency.

The fire severity factor is developed through a review of the industry experience used in the development of the fire initiating event frequencies presented in Section 4.1. The review of industry experience of the documented fire events provides insight into the basis for the frequency of occurrence of fires at nuclear power stations as well as data on the type, causes and severity of the fire events. This information is used to develop a factor which adjusts the fire initiating event frequency to account for the severity of fire events in the database. The source used in the development of the fire frequencies is the EPRI Fire Events Database (NSAC-178L).

A two step review process is used. The first review of the fire database consists of a review of the plant equipment which is damaged as a result of the fire event. The second review of the database evaluates the economic impact of the fire event on the plant. The second review functions as a verification of the first review since, in some cases, data is not available on the equipment damaged. In these cases data is available on the economic loss. Therefore, fire events in which no data is available on equipment damage can be screened from the database on the basis of limited or minimal economic loss. These two reviews, taken together, provide a complete evaluation of the severity of the fire events contained in the database.

A total of 753 fire events are listed in NSAC-178L. Of these 753 events 545 fires have specific entries describing the equipment that was damaged by the fire (datafield "COMBEFFECT"). These 545 fire events are divided into categories as follows:

- Forty-four (44) entries indicate that no plant material was damaged by the fire. These fire entries would appear to be those events which involve smoke or potential fire events but not actually involved fires.

- Four hundred (400) entries indicate that only one item in the plant was damaged. This was typically the fire source itself and frequently included non-plant equipment, such as on site automobiles and cleaning gear.

- Eight-five (85) entries indicate that two items were damaged. Of these,

    - One (1) resulted in a 90 day outage and was estimated to have a direct cost of over $1 million. This fire was attributed to arson.

    - One (1) fire in a PWR reactor building damaged a pump (believed to be a reactor coolant pump) and a cable, resulting in a 36 day outage and was estimated to have a direct cost of $500K - $1000K.

    - No other fires resulted in a plant outage longer than 5 days.

- Twenty-two (22) entries indicate that 3 items were damaged.

- Three (3) entries indicate that 4 items were damaged.

- One (1) entry indicated that 8 items were damaged by the fire. This was due to a diesel generator fire, with most of the damaged components associated with support of diesel generator operation.

It should be noted that items damaged by the fire include extension cords, plant cleaning equipment, personal automobiles and other non-critical components, as well as plant equipment.

For the 208 entries which have no listing which describes the equipment damaged (i.e., no listing under the "COMBEFFECT" datafield), a second datafield which reports direct economic losses (datafield DIRECTLOSS) was reviewed. Of the entries which report direct economic losses, only

5 caused a direct economic loss of more than $5,000 and these five events were listed as causing $5,000 - $50,000 in loss. None of these events are considered severe. On this basis, the 208 database entries without equipment damage listings are no longer considered in the estimation of severity factor. This assumption results in a lower population of events (545 versus 753). Given that more severe fires are most likely the better documented events than less severe fires; this assumption results in the elimination of the least severe fires from the database and is therefore conservative.

The entries listed with damaged equipment ("COMBEFFECT" data field) had the following distribution of direct economic loss (where this information was available in the database):

8	None
169	Less than $5,000
54	$5,000 to $50,000
7	$50,000 to $100,000
8	$100,000 to $500,000
3	$500,000 to $1,000,000. These were noted as: Main Coolant Pump (noted above) Main generator Hydrogen leak Diesel generator fuel line rupture
6	Over $1,000,000 (of these, 3 were yard transformer fires)

From the review of the fire events database, specifically the equipment damaged with additional review of the direct economic loss, two conclusions and observations ca[illegible] made:

- First, the first four categories (none, one item, two items or three items damaged by the fire) can be modeled as corresponding to the "nominal" case used in the general screening analysis. The "nominal" case being defined as one piece of equipment within the fire area or zone is destroyed by the fire. This corresponds to 99.3% of the available data.

- Second, the categories of fires that have 4 and 8 items damaged could be used to conservatively model the screening quantification case with damage to all cables in the area, in addition to all equipment located in the area (an "engulfing fire"). This corresponds to 0.7% of the available data.

The calculation of the severity factor remains conservative for the following for several reasons. First, the value used for screening in the first case is rounded up to 0.01, as opposed to 0.007, as indicated above, for the "engulfing fire" case. Second, the listing of items damaged within the database was extremely conservative in that numerous non-plant equipment items were included in the database and for completeness and to ensure a conservative evaluation, all items were counted during this review. Third, 208 entries in the database are excluded since there is no entry in the damaged equipment field. In these cases direct economic loss was small and indicates that the fire event damaged little if any plant equipment. Inclusion of these entries would result in lower fire severity factors.

In this study fire severity factor is used in the detailed fire evaluations to divide fire events into cases for further study. The division of the fire events into several cases results in the reduction of the "all engulfing fire" frequency (by a factor of 0.01) with less severe fires constituting the remaining frequency (a factor of 0.99). Total fire frequency for a given fire area or zone remains the same. The severity factor corresponds to the frequency of severe fires in the database and although no correlation is implied the fire severity factor is roughly an order of magnitude greater than the probabilities of both manual and automatic suppression.

The limitations of the fire severity factor require it to be used only for screening evaluations. All of the Oyster Creek fire areas and zones evaluated with the fire severity factor contain fire detection and automatic suppression. In addition, fire severity factor of 1% is used only for those areas which have large open spaces with little intervening combustibles. A fire severity factor of 10% is used for fire areas or zones which consist of smaller spaces or where more intervening combustibles exist. Justification is provided for each individual use of fire severity factor in the detailed analysis.

Fire Detection and Suppression

Fire detection and suppression probabilities are addressed in cases where severity factor use may not be appropriate. Fire detection and suppression probabilities are used to divide the fire initiating event frequency into case studies. As with the fire severity factor, the cases analyzed represent the "all engulfing fire" which is applied where suppression has failed and a less severe fire in cases where fire suppression is successful. The total fire initiating event frequency for the fire area or zone remains the same. In cases where suppression is successful a guaranteed failure of the source is assumed. Where multiple sources exist the guaranteed failed source is chosen is based on the most limiting (i.e., highest probabilistic importance) piece(s) of equipment or cable(s) within the fire area or zone.

It is appropriate to model detection and suppression probabilities in small spaces where combustible loading is high since automatic fire suppression initiation is based on either particle combustion detectors or relatively low ambient (when compared to cable damage) temperature heat devices. Also, the placement of either the particles of combustion detectors or the heat initiation devices are such that they are normally in the plume of the source which they protect or they are in the ceiling jet of any possible source in the area. Automatic fire system suppression failure probabilities are taken from the EPRI FIVE documentation.

Fire Modeling Computer Codes

It was not necessary to run fire modeling codes (COMPBRN IIIe) in the detailed fire evaluations since the screening probabilities used for severity factor and other mitigative features reduced the upper bound core damage frequency below the screening levels. Experience with the computer fire modeling codes indicate that their used would further reduce the frequency of core damage since the "all engulfing fire" assumption, which is used prevalently in this study, would be removed in many cases. Experience with the fire modeling codes has shown that component damage can occur due to the fire through any of four processes:

1. Damage due to the target being in the plume of the fire. In general, this will only damage electrical cables and other components that are located directly in the plume (i.e., directly over the fire source), particularly those located within 5 to 10 feet above the top of the fire source. This is due to the relatively high failure temperatures of cables.

2. Damage due to target near the plume at the ceiling, due to ceiling jet effects or development of a hot gas layer. This effect can damage electrical cables and other components located within 15% of the distance to the ceiling of the fire source, particularly if they are located radially within half of the distance fore the fire source to the ceiling. At Oyster Creek many plant fire areas such as the reactor building and the turbine building have large open areas and floors which are connected through equipment hatches which prevent the formation of hot gas layers. In addition, high ceiling are prevalent in these buildings and serve to reduce the impact of ceiling jet impingement.

3. Ambient heat rise in an area. Due to the large volumes involved in many plant areas this effect is minimal, except for severe fires is small spaces in which electronic equipment may be susceptible due to exceeding environmental qualification limits.

4. Radiant heat transfer. In general, this phenomenon will only affect those targets located within a close distance (5 to 10 feet) of the fire source.

In summary, fire modeling code experience shows that a severe or "all engulfing fire" is much less likely than was assumed in this analysis. This is primarily due to the large volumes available to distribute heat away from all but the largest fire sources. In addition, few intervening combustibles exist along which a fire could spread. This is especially true for the Oyster Creek reactor building in which the only items found along which a fire could spread from the north to the south side was a single electrical cable.

Each fire area or zone for detailed evaluation is described in the following paragraphs. Fire areas and zones whose screening core damage frequency is less than $1x10^{-6}$ are screened from further consideration. Any fire area or zone which is not screened in this step of the analysis is discussed in Section 4.6.4, Summary of Quantitative Evaluations.

OB-FZ-4 (Cable Spreading Room - 36 Foot Elevation) is a rectangular area, approximately 50 feet by 60 feet in size (see general arrangement drawing 3E-151-02-004). The FHAR gives the effective surface area of this fire zone as 2,543 square feet. The equipment layout in this area consists of a series of RPS electrical protection assembly (EPA) panels and motor generator sets, arranged along the north wall.

The significant targets for fire damage, including the RPS components noted above, are the individual system cables and 125 VDC panel E, which is located near the central area of OB-FZ-4. Also, it was confirmed during reviews for this analysis that the isolation condenser cables, which were failed for the initial analysis for this area, are protected by conduit, which would act to dissipate heat away from the cables. This also applies to the ADS/EMRV control cables running through this area (see FHAR table C-4). Review of cables that might impact division 1 essential AC switchgear (top event EC) revealed that hot short conditions would be required to fail any of the components (by inserting a breaker trip signal). Due to the unlikely nature of this form of failure and the recovery actions available (including local closure of the impacted breaker), this impact was also removed.

This area is protected by a particles of combustion detection system, with an automatic open head water spray system to suppress a cable fire and limit any water damage to other electrical equipment.

Due to the layout of this area and the small number of components located in OB-FZ-4, the detailed evaluation was performed using the following three cases:

Case 1 A fire starts anywhere in OB-FZ-4. Following suppression system failure, the fire spreads to engulf the area. This fire is assumed to fail 125 VDC power (top event DB), main circulating water (CW), service water (top event SW), high drywell and RPV actuation logic (top events DP and PR), low-low RPV water level (top event RL), MSIV actuation logic (top events ME and MS), condensate (top event CP), operator ATWS response (top event OL), primary containment isolation (top event PI), control rod drive (top event CD), containment spray (top events CC and RC), containment vent (top event OV) and core spray (top event CS). Due to the failure of actuation logic, manual operation of Isolation Condensers is required. Also, since failure of condensate is assumed, this case was quantified using the total loss of feedwater logic.

Although fire severity factor indicates a 1% probability of a fire of this magnitude, a factor of 10% is used. The increase in fire severity factor from 1% to 10% is warranted due to the relatively smaller areas with larger intervening combustible loadings of this fire zone. In addition, a 2% unavailability for wet pipe fire suppression systems (EPRI FIVE) is applied. Manual fire suppression or remote shutdown panel capability is NOT modeled. This case is assigned a frequency of 2.22×10^{-5}.

Case 2 A fire starts in or near panel DC-E. The automatic fire suppression system is successful. Due to the potential to damage the fire source, prior to fire suppression, this fire is assumed to fail 125 VDC power (top event DB) and is assigned a frequency of 8.0×10^{-4}.

Case 3 A fire starts in OB-FZ-4, but is suppressed before spreading. Due to the potential for this fire to cause damage to the fire source, this case is assumed to fail actuation sensor logic (top events PR, RL and DP). This is judged to bound other single cable failures that could occur in this area, prior to automatic fire suppression. This case is assigned the remainder of fire frequency for this area, after accounting for case 1 and 2, or 1.03×10^{-2}.

Evaluation of OB-FZ-4 Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail DB, SW, CW, DP, PR, RL, ME, MS, CP, OL, PI, CD, CC, RC, OV and CS, manual IC actuation, LOFW logic	2.22E-05	1.3E-06
Case 2	Fail DB	8.00E-04	4.8E-07
Case 3	Fail DP, PR and RL	1.03E-02	8.3E-07
Total Core Damage Frequency for Fire Zone OB-FZ-4		1.11E-02	2.6E-06

Since the total core damage frequency for these three cases is more than 1×10^{-6}, this area is NOT be screened.

This evaluation remains conservative since recovery of impacted systems and actuation logic is not considered. Also, only automatic suppression is modeled. Manual suppression of the fire is not modeled. Finally, the installed remote shutdown capability is not modeled. This would recover much of the impacted equipment functions, including division 2 of AC and DC power (see FHAR Appendix D for further discussion of remote shutdown capabilities for Oyster Creek).

OB-FZ-5 (Control Room - 46 Foot Elevation) is rectangular in shape, approximately 50 feet by 60 feet in size (see general arrangement drawing 3E-151-02-006). The FHAR gives the effective surface area of this fire zone as 2,764 square feet. The area consists of a series of panels, arranged into a front and a rear panel area.

The significant targets for fire damage are the individual system controls located on the various panels. These targets are separated by panel sections, with metal barriers between vertical panel sections. Also, during the review of potential impacts due to fires, the impact on division 2 125 VDC (top event DB) was lifted, since panel DC-E is actually located in the cable spreading room.

In addition to being constantly occupied by watch section personnel, this area is protected by ionization detection systems, with three high-flow Halon systems to protect critical panels. The Halon systems are actuated by cross-zoned particles of combustion detectors located within the panels. The main control panel sections protected by these Halon systems are separated by a fire barrier between panels 6R and 7R and panels 4F and 5F. The general functions performed with the controls on these panels are:

1F through 4F	Isolation condensers, core spray, reactor recirculation	Halon system A
1R through 6R	Control rod position control, ADS	Halon system A
5F through 9F	Main feedwater, turbine controls, instrument air	Halon system B
7R through 11R	Essential AC power, service and circulating water	Halon system B
10F through 11F	Primary containment control	Halon system C

In total, there are 39 panels in the control room. Therefore, fire initiating event frequency was segmented as follows:

10 panels	Protected by Halon system A
12 panels	Protected by Halon system B
2 panels	Protected by Halon system C
15 panels	Not protected by Halon

Due to the complex geometry and the location of components and cables within the area, the detailed evaluation of fires in OB-FZ-5 was performed by evaluating the following five (5) cases:

Case 1 A fire starts in one of the control room panels that is protected by Halon system A. Following successful discharge of the Halon system, the fire would be suppressed. For this case, core spray train 1 is assumed to be damaged by the fire, before suppression. This case is judged to bound other cases where only the fire source is damaged by the fire. With 10 panels protected by Halon system A and assuming a 5% unavailability/failure rate for Halon systems (provided in the EPRI FIVE documentation), this case is assigned a frequency of ($1.16 \times 10^{-2} \times 10/39 \times 0.95 =$) 2.83×10^{-3}.

Case 1A A fire starts in one of the control room panels that is protected by Halon system A. Following Halon system failure, a fire could spread to the fire barriers between these panels. Therefore, this case is assumed to fail actuation logic (top events DP, PR and RL), both isolation condensers (top event IC), operator response to ATWS (top event OL), control rod drive (top event CD), containment spray (top events CC and RC), ADS actuation (top event AD) and core spray (top event CS). This case is assigned a frequency of ($1.16 \times 10^{-2} \times 10/39 \times 0.05 =$) 1.49×10^{-4}.

Case 2 A fire starts in one of the control room panels that is protected by Halon system B. This fire is then suppressed by discharge of Halon within the area. This fire is assumed to result in a total loss of main feedwater. This is judged to bound other cases, where only the fire source itself is damaged by the fire. With 12 panels protected by Halon system B and assuming a 5% unavailability/failure rate for Halon systems (provided in the EPRI FIVE documentation), this case is assigned a frequency of ($1.16x10^{-2}$ x 12/39 x 0.95 =) $3.40x10^{-3}$.

Case 2A A fire starts in one of the control room panels that is protected by Halon system B. Following Halon system failure, a fire could spread to the fire barriers between these panels. Therefore, this case is assumed to fail circulating water (top event CW), 4160 VAC switchgear 1C (top event EC), service water (top event SW), instrument air (top event IA) and main condensate (top event CP). Case 2 frequency is ($1.16x10^{-2}$ x 12/39 x 0.05=) $1.78x10^{-4}$.

Case 3 A fire starts anywhere in the control room, besides the panels protected by Halon system A or B. Due to the less critical nature of the controls and instrumentation in these panels, plant trip may or may not occur. For purposes of analysis, though, a plant trip is assumed, along with a loss of offsite power, as might occur for an unsuppressed fire in panel 12F. This case is assigned the remaining fire frequency of (17/39 x $1.16x10^{-2}$ =) or $5.01x10^{-3}$.

Evaluation of OB-FZ-5 Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail one train of CS	2.83E-03	>1.0E-09
Case 1A	Fail AD, CS, DP, PR, RL, IC, OL, CD, CC and RC	1.49E-04	1.1E-07
Case 2	Total Loss of Feedwater	3.40E-03	6.2E-09
Case 2A	Fail EC, SW, CP, IA and CW	1.78E-04	2.6E-08
Case 3	Loss of Offsite Power	5.01E-03	1.9E-07
Total Core Damage Frequency for Fire Zone OB-FZ-5		1.16E-02	3.3E-07

Since the total core damage frequency for these five cases is less than 10^{-6}, this area can be screened from further consideration. The results for case 3 are dominated by scenarios with discharge of heat to the primary containment and independent failure of containment spray.

This evaluation remains conservative for several reasons. First, recovery of impacted control circuits, either through repair or enabling the plant remote shutdown capability is not considered. Second, only automatic Halon suppression is modeled, with no allowance for manual suppression. Third, fire severity factor is not modeled (i.e., cases where the source of the fire burns out without damaging risk-significant plant equipment).

OB-FZ-6A ("A" 480 VAC Switchgear Room - 23 Foot Elevation) is approximately 35 feet by 35 feet in size (see general arrangement drawing 3E-153-02-002). The FHAR gives the effective surface area of this fire zone as 1,157 square feet. The area consists of a main hallway running along the west wall of the reactor building. Off of this main hallway are 3 alleys which contain risk-significant equipment.

The significant targets for fire damage located in the north alley are USS Transformer 1A2 and MCC 1A21. These targets are separated from the second and third hallway by a concrete wall, which stretch from floor to ceiling, but are not complete boundaries from west to east. The significant targets for fire damage located in the second and third hallways are relay panels ER-8A and B and ER-18A and B, each of which supply actuation logic for one division of containment and core spray. MCC 1A23 and MCC 1B23 are located in the southeast corner.

This area is protected by ionization and photoelectric detection systems, with a fixed, total flooding Halon 1301 extinguishing system to suppress and to prevent the spread of fires.

Due to the geometry of the area and the location of plant components and control cables, the detailed evaluation of fires in OB-FZ-6A was performed by evaluating the following three cases:

Case 1 A fire starts anywhere in OB-FZ-6A. Through failure of the Halon system to discharge, this fire is assumed to spread to engulf the area. This fire is assumed to fail containment spray (top events CC and RC), control rod drive (top event CD), core spray (top event CS), circulating water (top event CW), high drywell pressure logic (top event DP), 4160 VAC bus 1C (top event EC), containment vent (top event OV), primary containment isolation (top event PI), low-low RPV water level logic (top event RL), shutdown cooling (top event SD), isolation condenser A (top event IC). Assuming a 5% unavailability for Halon systems (EPRI FIVE documentation), this case is assigned a frequency of 2.09×10^{-4} (5% of 4.18×10^{-3}).

Case 2 A fire starts in USS 1A2. This fire is then suppressed by discharge of Halon within the area. This fire is assumed to fail division 2 AC power (top event EC) and a number of nearby cables. This case is assigned 1/3 of the total frequency for this area, after accounting for case 1, 32% of total or 1.34×10^{-3}.

Case 3 A fire starts in the second or third hallway in OB-FZ-6A, but is suppressed by Halon before damaging any other equipment. This case is assigned 63% of the fire frequency (the remainder of the fire frequency, after accounting for case 1 and 2), or 2.63×10^{-3}.

Fire severity factor is NOT applied in any of cases above. The presence of a large transformer (Unit Sub-Station Transformer 1A2) containing approximately 225 gallons of flammable mineral oil provides the potential for fire growth which can involve the entire small space given halon suppression failure.

Evaluation of OB-FZ-6A Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail CD, CW, DP, OV, PI, RL, SD, EC, CC, RC, CS and one train of IC	2.09E-04	2.7E-06
Case 2	Fail CD, CW, DP, OV, PI, RL, SD and EC	1.34E-03	8.0E-07
Case 3	Fail CD, CW, DP, OV, PI, RL, SD and one train of CC, RC, CS and IC	2.63E-03	1.6E-06
Total CDF for fire zone OB-FZ-6A		4.18E-03	5.1E-06

Since the total core damage frequency for these three cases is more than 1×10^{-6}, this area can not be screened from further consideration.

The results for case 2 are dominated by scenarios with discharge of heat to containment and independent failure of the available train of containment spray. With the guaranteed failure of containment vent and of all division 1 essential AC power, these scenarios lead to core damage.

This evaluation is conservative since recovery of impacted logic for core and containment spray is not considered. Only automatic Halon suppression of fires in this area is modeled, with no allowance for manual suppression or for cases where the source of the fire burns out before damaging risk-significant plant equipment (fire severity factor).

OB-FZ-8C (A and B Battery Room - 35 Foot Elevation) is rectangular in shape, approximately 50 feet by 31 feet (see general arrangement drawing 3E-156-02-002). The FHAR gives the effective surface area of this fire zone as 1,292 square feet, indicating that approximately 16% of the available floor area is occupied by walls and other interference. This acts to give this fire zone a complex shape and from the aspect of fire evaluation, makes it unlikely that a fire will spread from any location in OB-FZ-8C to a significantly larger area.

The significant targets for fire damage are located in the north half of this area, along with the "B" 125 VDC battery, which is separated from the "A" battery by a roll-up door. Also, it was noted during this review that the impacts incorporated during the initial evaluation of OB-FZ-8C on VO and RC do not actually fail these system functions. In the case of EMRV operation (VO), the only system circuitry impacted would be valve position indication, which would not fail EMRV operation itself. Also, a fire has the potential to fail the remote operation of containment spray (top events CC and RC), which would not impact the recovery of containment spray.

This area is protected by a particles of combustion detection system, with a fixed, total flooding Halon 1301 extinguishing system to prevent spread of fires within this area.

Due to the geometry of the area and the location of plant components, the detailed evaluation of fires in OB-FZ-8C was performed by evaluating the following three cases:

Case 1 A fire starts anywhere in OB-FZ-8C. Through failure of the Halon system, the fire is assumed to spread to engulf the area. This fire fails liquid poison injection (top event BI), containment spray (top events CC and RC), control rod drive (top event CD), circulating water (top event CW), 125 VDC power (top event DB), high drywell pressure actuation logic (top event DP), 4160 VAC buses 1B and 1D (top events EB and ED) and top events PI, VO and OV.

Although fire severity factor indicates a 1% probability of a fire of this magnitude a factor of 10% is used. The increase in fire severity factor from 1% to 10% is warranted due to the relatively smaller area with larger intervening combustible loadings of this fire zone. In addition, assuming a 5% unavailability/failure rate for Halon systems (provided in the EPRI FIVE documentation), this case is assigned a frequency of 1.86×10^{-5} ($0.1 \times 0.05 \times 3.71 \times 10^{-3}$).

Case 2 A fire starts in battery B in OB-FZ-8C. This fire is then suppressed by discharge of Halon within the area. This fire is assumed to fail division 2 DC power (top event DB). Through loss of control power, this also fails division 2 of plant AC power (top events EB and ED). This case is assigned half of the total frequency for this area as a battery room ($0.50 \times 1.60 \times 10^{-3} = 8.00 \times 10^{-4}$).

Case 3 A fire starts anywhere in OB-FZ-8C, but is suppressed by Halon discharge before damaging any risk-significant equipment. This case is assigned the remainder of the fire frequency for this area, ($3.71 \times 10^{-3} - 8.00 \times 10^{-4} - 1.86 \times 10^{-4} = 2.72 \times 10^{-3}$).

Evaluation of OB-FZ-8C Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail Top Events BI, CC, CD, CW, DB, DP, EB, ED, PI, OV	1.86E-05	3.3E-08
Case 2	Fail Top Event DB	8.00E-04	4.8E-07
Case 3	No impact, beyond plant trip	2.72E-03	>1.0E-09
Total Core Damage Frequency for fire zone OB-FZ-8C		3.71E-03	5.1E-07

Since the total core damage frequency for these 3 cases is less than 10^{-6}, this area can be screened from further consideration.

The scenarios for case 1 are dominated by scenarios with discharge of heat to the containment and. With the guaranteed failure of isolation condensers, containment spray and containment vent, these scenarios lead directly to core damage.

The scenarios for case 2 are dominated by failure of the second train of electrical power (top events DC, EC, EA), which leads to an assumed unrecoverable loss of all site electrical power.

This evaluation is conservative since recovery of DC or AC power is not considered. Also, it is assumed that more than 25% of the fires lead to an unrecoverable loss of division 2 of 125 VDC.

TB-FA-26 ("C" Battery Room) is rectangular in shape, located in the Turbine Building on the 23 foot elevation, approximately 14 feet by 10 feet (see general arrangement drawing 3E-151-02-003). The FHAR gives the effective surface area of this fire zone as 140 square feet. This area is protected by a particles of combustion detection system, with carbon dioxide and hose stations outside for manual fire suppression.

Case 1 A fire starts anywhere in TB-FA-26 and progresses to the point of failing battery C, which is conservatively assumed to fail Division 1 of the plant 125 VDC power system. This case is modeled as occurring for half of the fires that start in this area.

Case 2 A fire starts in TB-FA-26 and results in a plant trip, followed by long term battery failure at top event XC. This case is assumed to occur for all other fires in the C battery room.

Evaluation of TB-FA-26 Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail DC (before plant trip)	1.23E-03	7.6E-07
Case 2	Division 1 DC power failure after plant trip (top event XC)	1.24E-03	<1.0E-09
Total Core Damage Frequency for fire zone TB-FA-26		2.47E-03	7.6E-07

Since the total core damage frequency for these cases is less than 10^{-6}, this area can be screened from further consideration.

As shown in the results table above, Case 1 dominates the results, primarily due to an assumed battery failure prior to plant trip, resulting in a loss of AC bus control power for Division 1 equipment. No recovery actions are modeled for this failure.

This evaluation is conservative in that all fires in this area result in plant trip with equipment damage to division 1 DC power. Case 1 is modeled with DC bus failure prior to plant trip and does not allow bus recovery with the battery charger only. Case 2 is modeled with DC bus failure following plant trip. Again, bus recovery is not considered.

TB-FZ-11D (Turbine Building, Basement Southend) is rectangular in shape along the south and west walls of the turbine building. The effective floor area is given as 9668 square feet with a low combustible loading (less than 40,000 BTU/ft^2). A closed head automatic sprinkler system provides area wide coverage and a water spray system with directional nozzles protect the hydrogen seal oil unit located in this area.

The initial estimate of upper bound core damage frequency failed the circulating water (top event CW), service water (top event SW), TBCCW (top event TB), feedwater (FW), containment spray/ ESW (top events CC and RC), core spray (CS) and instrument air (IA) systems. Given the fire zone geometry and the layout of equipment in the area three cases present themselves:

Case 1 A fire starts anywhere in fire zone TB-FZ-11D. Through failure of the area wide sprinkler system, the fire spreads to engulf the entire area. This event fails circulating water (top event CW), service water (top event SW), TBCCW (top event TB), instrument air (top event IA), feedwater (top event FW), core spray (top event CS) and containment spray and ESW (top event CC and RC).

A fire severity factor indicates a 1% probability of a fire of this magnitude. In addition, a 2% unavailability for wet pipe fire suppression systems (EPRI FIVE) is applied. Manual fire fighting or remote shutdown panel capability is NOT modeled. This case is assigned a frequency of (0.01 x 0.02 x 1.09×10^{-2}) or 2.18×10^{-6}.

Case 2 A fire starts in the southeast section of fire zone TB-FZ-11D and involves the instrument air system (i.e., air compressors) at top event IA and the hydrogen seal oil unit. Suppression with the area wide sprinkler system is successful. This case is assigned 50% of the remaining fire frequency or 5.45×10^{-3}.

Case 3 A fire starts in the northwest section of fire zone TB-FZ-11D and involves the Turbine Building Closed Cooling (TBCCW) system (top event TB) service and circulating water (top event SW and CW). Fire suppression with the area wide sprinkler system is successful and the fire contained. Case 3 is assigned 50% of the remaining frequency or 5.45×10^{-3}.

Evaluation of TB-FZ-11D Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail top events CW, SW, TB, IA, FW, CS, CC and RC.	2.18E-06	1.3E-07
Case 2	Fail top event IA.	5.45E-03	2.4E-09
Case 3	Fail top event TB, CW and SW.	5.45E-03	7.6E-08
Total Core Damage Frequency for fire zone RB-FZ-1D		1.09E-02	2.1E-07

Since this fire zone contributes less than 1×10^{-6} to total core damage frequency this fire zone can be screened from further consideration.

Case 1 dominates the results primarily due to a loss of RPV injection (feedwater, condensate and core spray systems failed) with a stuck open EMRV at split fraction VR2. It should be noted that feedwater control is failed due to fires in this zone and local control of feedwater remain available. The loss of TBCCW cooling is lost to both the condensate and feedwater pumps and bearings however feedwater would remain available in the short term. The short term availability of feedwater presents additional unmodeled success paths where decay heat is eventually matched by CRD injection or the RPV is depressurized through the stuck open relief and fire water is injected through core spray.

Although the full severity factor (1%) as well as the unavailability of wet pipe sprinklers (2%) are both applied manual suppression and remote shutdown capability are not modeled and as such the evaluation is considered appropriate.

RB-FZ-1D (Reactor Building 51 Foot Elevation) is rectangular in shape along the outside boundary of the reactor building, approximately 137 feet by 106 feet (see general arrangement drawing 3E-151-02-003). The FHAR gives the effective surface area of this fire zone as 9,100 square feet, indicating that approximately 40% of the available floor area is occupied by the primary containment/drywell area and shutdown cooling fire zone RB-FZ-1G. The drywell area and RB-FZ-1G act to give this fire zone a U-shape, reducing it to a hallway near RB-FZ-1G. Also, the reactor water cleanup equipment rooms are located along the south wall (sludge tank, pump and heat exchanger rooms. Each of these rooms, in addition to the CRD rebuild area located along the northeast corner, is of concrete wall construction, with complex paths for entry, primarily to prevent radiation streaming effects from RWCU hot spots. From the aspect of fire growth and propagation, this geometry makes it unlikely that a fire will spread from any location in RB-FZ-1D to any significantly larger area.

The core and containment spray equipment is located in the northwest corner, which is separated from the rest of RB-FZ-1D by the drywell, fire zone RB-FZ-1G and a hallway that runs for approximately 25 feet.

This fire zone is protected by an ionization detection system, with a water curtain installed at ceiling elevation to prevent fire growth to RB-FZ-1C, above. Also, fixed automatic water spray systems are installed to protect the grouped cables.

Due to the zone geometry and the location of components and cables the evaluation is performed using three cases:

Case 1 A fire starts anywhere in RB-FZ-1D and spreads to all areas within the fire zone. This case is identical to the initial evaluation for the fire zone, except only 1% of the fire events are expected to reach a severe level. The use of a fire severity factor of 1% is appropriate due to the geometry and large open spaces with little intervening combustibles.

Case 2 A fire starts in the northwest corner of RB-FZ-1D. The fire is contained or suppressed within this area. Since the RPS instrument rack is in this area, top events DP and RL are assumed to fail. Also, shutdown cooling cables transit the area and top event SD is assumed to fail. Train 1 of core and containment spray (top events CS, CC and RC) fail. This case is assigned 25% of the total fire zone frequency or 6.33×10^{-3}.

Case 3 A fire starts in any of the remaining areas of RB-FZ-1D. This fire is then prevented from spreading to the northwest corner area. This case is assumed to fail isolation condensers (top event IC, due to isolation logic pressure switch failure), containment vent (top event OV) and reactor building isolation (top event RI). This case is assigned the remainder (74%) of the fire frequency or 1.87×10^{-2}).

Evaluation of RB-FZ-1D Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail DP, IC, OV, PI, PR, RI, RL, SD, VR and train 1 of core spray and containment spray (top events CC, RC and CS)	2.53E-04	1.1E-07
Case 2	Fail DP, RL, SD and train 1 of core spray, containment spray (top events CC, RC, CS)	6.33E-03	1.5E-08
Case 3	Fail IC, OV, RI	1.87E-02	1.5E-07
Total Core Damage Frequency for fire zone RB-FZ-1D		2.53E-02	2.7E-07

Since the total core damage frequency for these 3 cases is less than 10^{-6}, this area can be screened from further consideration.

As shown in the results table above, Case 1 and 3 dominate the results. In case 1 this is primarily due to an assumed EMRV pressure switch failure (high), concurrent with failure of one train of containment spray and failure of containment vent. Failure of the remaining train of containment spray, either directly or due to other dependencies, such as division 2 of AC or DC power, is assumed to lead directly to core damage. Case 3, which contributes mainly due to a high initiating event frequency (a factor of 75 higher than Case 1), contains similar results with the exception that the entire containment spray/ESW system is failed independently. No recovery actions are modeled.

RB-FZ-1E (Reactor Building 23 Foot Elevation) is rectangular in shape along the outside boundary of the reactor building, approximately 137 feet by 106 feet (see general arrangement drawing 3E-153-02-002). The effective surface area of this fire zone as 12,140 square feet, indicating that approximately 16% of the available floor area is occupied by the primary containment/drywell area. This acts to give this fire zone a U-shape, with the trunnion room preventing communication from one end to the other. This geometry makes it unlikely that a fire will spread from any location in RB-FZ-1E to any significantly larger area.

The system 2 core spray booster pumps that are located in this area are both in the southwest corner, approximately 15 feet from the west wall of the reactor building. The south banks of CRD HCUs are located nearby. Also, during the review of fire zone geometry and cables that were susceptible to fire damage, it was noted that the EMRV actuation cables in this area are contained in conduit (see FHAR appendix C-1), which would act to dissipate heat until fire suppression. The portion of the EMRV circuits that could potentially hot short is contained inside a penetration box, which is itself protected by electrical fault devices (see FHAR Volume 2, Section 1, Page 18). Therefore, hot short failure of EMRVs was removed from the model.

This area is protected by an ionization detection system, with a water curtain installed at ceiling elevation to prevent fire growth to RB-FZ-1D, above. Also, fixed automatic water spray systems are installed to protect the grouped cables in this area. Due to the geometry of the fire zone and the location of plant components and control cables, the detailed evaluation was performed using three cases:

Case 1 A fire starts anywhere in RB-FZ-1E and spreads to all areas within the fire zone. This case is identical to the initial evaluation except that only 1% of the fires are expected to reach a severe level. The use of a severity factor of 1% is appropriate due the large open spaces with small amounts of intervening combustibles.

Case 2 A fire starts anywhere on the south side of RB-FZ-1E. This fire is then contained or suppressed within this area. This fire is assumed to fail division 2 containment spray cables (top events CC and RC), high drywell pressure actuation logic (top event DP), cables for the south isolation condenser (top event IC), division 2 core spray booster pumps (top event CS) and outboard MSIV isolation (top events ME and MS). This case is assigned 50% of the total fire zone frequency (7.67×10^{-3}).

Case 3 A fire starts anywhere on the north side of RB-FZ-1E. This fire is then contained or suppressed within this area. This fire is assumed to fail division 1 containment spray cables (top events CC and RC), high drywell pressure actuation logic (top event DP), cables for the north isolation condenser (top event IC), shutdown cooling (top event SD) and outboard MSIV isolation (top events ME and MS). This case is assigned 50% of the total fire zone frequency (7.67×10^{-3}).

Evaluation of RB-FZ-1E Cases

Case	Fire Risk Model Impacts	Fire Frequency	Damage Frequency
Case 1	Fail CC, RC, CD, DP, IC, OV, PI, SD and train 2 of core spray and outboard MSIVs	1.55E-04	1.1E-07
Case 2	Fail DP and train 2 of core and containment spray (top events CC, RC and CS), train 2 IC and outboard MSIVs	7.67E-03	2.8E-09
Case 3	Fail DP, SD and train 1 of containment spray (top events CC and RC), train 1 IC and outboard MSIVs	7.67E-03	1.8E-08
Total Core Damage Frequency for Fire Zone RB-FZ-1E		1.55E-02	1.3E-07

Since the total core damage frequency for these three cases is below 1×10^{-6}, this area can be screened from further consideration. Case 1 dominates the results, primarily due to scenarios with discharge of heat (e.g., EMRV failure to close) to the primary containment and an assumed failure of containment vent. With the guaranteed failure of the isolation condensers and containment spray system, these scenarios lead directly to core damage.

4.6.4 Summary of the Quantitative Fire Analysis (Task 9)

Table 4.6-6 summarizes the results of the quantitative fire analysis; Tasks 5, 6, and 7. Fires areas which are screened from further consideration are shaded. In total 35 fire areas and zones are evaluated quantitatively. Of these, 18 fire areas or zones are screened in the initial quantification, 10 fire areas or zones are screened in the revised quantification and 5 fire areas and zones are screened in the detailed evaluation. Two fire areas did not screen from consideration.

OB-FZ-4 Cable Spreading Room - 36' Elevation
OB-FZ-6A "A" 480 VAC Switchgear Room

Fire zone OB-FZ-4, Cable Spreading Room - 36' Elevation

Total Fire Ignition Frequency	=	1.11×10^{-2} per year
Total Upper Bound CDF	=	2.6×10^{-6} per year

The total upper bound CDF estimate is 2.6×10^{-6} per year. This fire zone was modeled in three cases with the "all engulfing fire" contributing the largest fraction (50%) to the results.

A fire severity factor of 10% as well as an automatic suppression failure rate 0.02 were applied. The fire severity factor of 10% is applied versus 1% due to the relatively small area and amount of intervening combustibles in the fire zone. Due to the number of sources and targets in the area fire modeling codes are not used. Manual fire suppression and use of the remote shutdown panel are not modeled and consideration of these factors would reduce the CDF.

Fire scenarios in this fire zone are dominated by the failure of an EMRV to reclose after post-trip pressure relief (split fraction VR1) and a loss of containment heat removal (failure of containment spray/emergency service water). Although the fire zone is not screened, the screening core damage frequency is considered low and no action is recommended.

Fire zone OB-FZ-6A, "A" 480 VAC Switchgear Room

Total Fire Ignition Frequency	=	4.18×10^{-3} per year
Total Upper Bound CDF	=	5.1×10^{-6} per year

This fire zone was modeled in three cases with the "all engulfing fire" dominating the results. No fire severity factor is modeled due to the presence of a large transformer and its cooling oil (225 gallons of mineral oil). An automatic suppression probability of failure of 0.05 was modeled. Fire modeling codes are not used. Manual fire suppression and use of the remote shutdown panel are not modeled and consideration of these factors would reduce the core damage frequency.

Fire scenarios are dominated with the decay heat discharge to the containment (split fraction VR1) and a failure of containment heat removal (both the containment vent and containment spray/emergency service water). Although the fire zone is not screened, the screening core damage frequency is considered low and no action is recommended.

The following six (6) fire areas and zones were screened from further consideration but are summarized in the paragraphs below:

OB-FZ-5	Control Room - 46' 6" Elevation
OB-FZ-8C	A and B Battery Room, Tunnel and Electric Tray Room (35' Elevation)
TB-FA-26	Battery Room South of 4160 VAC Switchgear
TB-FZ-11D	Turbine Building Basement Southend
RB-FZ-1D	51 Foot Elevation
RB-FZ-1E	Main Floor (23 Foot Elevation)

Fire zone OB-FZ-5, Control Room - 46' 6" Elevation

Total Fire Ignition Frequency = $1.16x10^{-2}$ per year
Total Upper Bound CDF = $3.3x10^{-7}$ per year

No fire severity factor is modeled. The three (3) Halon suppression systems are modeled. The evaluation is considered conservative since fire severity factor, manual suppression and enabling yremote shutdown capability are not modeled. The control room screens from further consideration largely due to the probability of automatic suppression with the three independent Halon systems. No action is recommended.

The scenarios produced in Case 3, an assumed loss of offsite power (with a frequency of $5.19x10^{-3}$) dominate the results. With an assumed failure of offsite power these scenarios are similar loss of offsite power sequences produced in the Level 1 OCPRA. Scenario 1 produced in this model is the same as scenario 1 of the Oyster Creek Level 1 PRA.

OB-FZ-8C, A/B Battery Room, Tunnel and Electric Tray Room (35' Elevation)

Total Fire Ignition Frequency = $3.71x10^{-3}$ per year
Total Upper Bound CDF = $5.1x10^{-7}$ per year

This fire zone is modeled with a fire severity factor of 10%. This 10% factor is used (versus 1%) due to the relatively small room area and the larger amount of intervening combustibles. In addition to the severity factor a 5% unavailability of Halon systems is also used in the evaluation. This fire zone is screened from further consideration and no action is recommended. The use of a 1% severity factor would reduce the upper bound core damage frequency by a factor of 10 to approximately $5x10^{-8}$.

With the guaranteed failure of the A/B battery in most fire scenarios (94% by upper bound CDF) the results are dominated by the failure of battery "C" (at split fraction DC1). These scenarios are similar to the OCPRA total loss of DC power scenarios.

Fire zone TB-FA-26, Battery Room "C" (South 4160 VAC Switchgear)

Total Fire Ignition Frequency	= 2.47×10^{-3} per year
Total Upper Bound CDF	= 7.6×10^{-7} per year

Fire zone TB-FA-26, the Battery Room "C", core damage frequency is estimated at 7.6×10^{-7} per year. This fire area was modeled by approximating the timing of battery failure. That is, since the area has fire detection, plant personnel would be immediately aware of a fire in this area. Therefore, fires in this area were divided into two cases, depending on when plant trip was assumed to occur, relative to battery failure. For half the fire ignition frequency for this area, division 1 DC power was assumed to fail prior to plant trip, which is modeled as failing division 1 AC power shift to the startup transformer. The remaining fire frequency was assigned to the case where DC power remains available during plant trip, but long term failure is then assumed to occur at top event XC. Where DC power is available offsite power remain available.

As in the case with the A/B Battery room the scenario results are dominated by the independent loss of the A/B battery at split fraction DB1. These results are similar to the total loss of DC power scenarios in the Level 1 OCPRA. This fire area is screened from further consideration and no action is recommended.

Fire Zone TB-FZ-11D, Turbine Building Basement Southend

Total Fire Ignition Frequency	= 1.09×10^{-2} per year
Total Upper Bound CDF	= 2.1×10^{-7} per year

TB-FZ-11D, Turbine Building Basement Southend, core damge frequency is estimated at 2.1×10^{-7} per year. A closed head automatic sprinkler system provides area wide coverage and a water spray system with directional nozzles protect the hydrogen seal oil unit located in this area.

The fire zone is modeled in three case. A fire severity factor of 1% is applied as well as an unavailability of wet pipe sprinklers of 2%. The results are dominated by Case 1 which represents the "all engulfing fire". This approach is judged to be acceptable since manual fire fighting and remote shutdown capability are not addressed.

Case 1 dominates the results primarily due to a loss of RPV injection (feedwater, condensate and core spray systems failed) with a stuck open EMRV at split fraction VR2. It should be noted that feedwater control is failed due to fires in this zone and local control of feedwater remain available. With the loss of TBCCW cooling is lost to both the condensate and feedwater pumps and bearings however feedwater would remain available in the short term. The short term availability of feedwater presents additional unmodeled success paths where decay heat is eventually matched by CRD injection or the RPV is depressurized through the stuck open relief and fire water is injected through core spray.

In case 2 the only impact is the failure of instrument air at top event IA. This is representative of the loss of instrument air initiating event in the Level 1 OCPRA. Case 3, failure of TBCCW, service and circulating water, which contributes (approximately 33%) to the total core damage frequency for this fire zone is best represented by a loss of TBCCW initiating event and a loss of condenser vacuum or condenser support.

The fire zone screens from further consideration and no other action is recommended.

Fire zone RB-FZ-1D, Reactor Building 51 Foot Elevation

Total Fire Ignition Frequency	= 2.53×10^{-2} per year
Total Upper Bound CDF	= 2.7×10^{-7} per year

Fire zone RB-FZ-1D, Reactor Building 51 Foot Elevation screening core damage frequency is estimated as 2.7×10^{-7}. A fire severity factor of 1% is applied based on the large open area of the fire zone which contains little intervening combustibles. Neither automatic or manual fire suppression is modeled and results in a conservative evaluation.

Case 1 and 3 dominate the results. In case 1, all engulfing fire case, this is primarily due to an assumed EMRV pressure switch failure (high), concurrent with failure of one train of containment spray and failure of containment vent. Failure of the remaining train of containment spray, either directly or due to other dependencies, such as division 2 of AC or DC power, is assumed to lead directly to core damage.

In Case 3, which contributes mainly due to a high initiating event frequency (a factor of 75 higher than Case 1), contains similar results with the exception that the entire containment spray/ESW system if failed independently. No recovery actions are modeled. No action is recommended.

Fire zone RB-FZ-1E, Reactor Building Main Floor (23 Foot Elevation)

Total Fire Ignition Frequency	= 1.55×10^{-2} per year
Total Upper Bound CDF	= 1.3×10^{-7} per year

This evaluation applies a severity factor of 1% based on large open area with little intervening combustibles. Neither automatic or manual suppression is modeled. Scenarios are dominated by loss of containment heat removal scenarios due to a stuck open EMRV and failure of the containment vent as well as containment spray.

This fire zone screens from further consideration and no action is recommended.

Table 4.6-3 Oyster Creek Plant Fire Areas and Zones Quantitative Evaluation Results							
Fire Area	Fire Area Description	Fire Frequency	Initial Quantitative Evaluation [1]	Revised Quantitative Evaluation [2]	Detailed Evaluation [3]		
					Severity Factor	Other Factors [3]	Detailed Evaluation
OFFICE BUILDING FIRE AREAS AND ZONES							
OB-FZ-4	Cable Spreading Room - 36' Elevation	1.11E-02	1.1E-02		0.10	0.02	2.6E-06
OB-FZ-5	Control Room - 46' 6" Elevation	1.16E-02	1.2E-02			0.05	**3.3E-07**
OB-FZ-6A	"A" 480 VAC Switchgear Room	4.18E-03	1.5E-04			0.05	5.1E-06
OB-FZ-6B	"B" 480 VAC Switchgear Room	4.00E-03	**3.1E-07**				
OB-FZ-8A/B	MG Set Room / Mechanical Equipment Room	7.40E-03	**8.7E-08**				
OB-FZ-8C	A and B Battery Room, Tunnel and Electric Tray Room (35' Elevation)	3.71E-03	1.3E-04		0.10	0.05	**5.1E-07**
OB-FA-9	Office Building	3.96E-03	**2.3E-07**				
OB-FZ-10A	Monitoring and Change Room - 46'	9.28E-03	2.4E-06	**4.9E-07**			
OB-FZ-10B	Chem Lab, Laundry, Instrument Shop - 35'	8.68E-04	**6.8E-08**				
OB-FA-22A	New Cable Spreading Room (Mech Equip Room) 63' 6" El.	3.58E-03	3.9E-06	**2.6E-07**			
OB-FA-22B	North Cable Bridge Tunnel, 74' 6"	1.69E-03	**3.3E-09**				
OB-FZ-22C	South Cable Bridge Tunnel	1.69E-03	**6.9E-10**				
TURBINE BUILDING FIRE AREAS AND ZONES							
TB-FA-3A	4160 VAC Switchgear 1C Vault (TB Mezz - 23')	4.79E-03	**4.6E-07**				
TB-FA-3B	4160 VAC Switchgear 1D Vault (TB Mezz - 23')	3.64E-03	**4.2E-07**				
TB-FZ-11A	Turbine Operating Floor - 46' Elevation	2.08E-02	**1.2E-08**				
TB-FZ-11B	Lube Oil Storage, Pumping and Purification Areas, 0' and 27' Levels	1.19E-02	4.3E-05	**3.4E-07**			
TB-FZ-11C	Switchgear Room, West End of TB on Mezzanine Level (Elev. 23' 6")	3.28E-03	1.6E-05	**4.6E-07**			
TB-FZ-11D	Basement Floor, South End	1.09E-02	6.8E-04		0.01	0.02	2.1E-07
TB-FZ-11E	Condenser Bay, Elevation 3' 6"	5.92E-03	1.7E-06	**8.9E-07**			

Table 4.6-3 Oyster Creek Plant Fire Areas and Zones Quantitative Evaluation Results							
Fire Area	Fire Area Description	Fire Frequency	Initial Quantitative Evaluation [1]	Revised Quantitative Evaluation [2]	Detailed Evaluation [3]		
					Severity Factor	Other Factors [3]	Detailed Evaluation
TB-FZ-11F	Feedwater Pump Area, 0'6" & 3'6" Levels	8.07E-03	4.7E-06	**1.1E-07**			
TB-FZ-11G	Mezzanine Level Southeast Corner and Machine Shop, Elevation 23' 6"	4.73E-03	**2.0E-07**				
TB-FZ-11H	Demineralizer Tank and Steam Jet Air Ejector Area, Elevation 23' 6"	4.73E-03	3.4E-04	**7.3E-07**			
TB-FA-26	Battery Room South of 4160 VAC Switchgear	2.47E-03	1.5E-06		0	0	**7.6E-07**
REACTOR BUILDING FIRE ZONES							
RB-FZ-1B	95 Foot Elevation	2.61E-03	**8.4E-09**				
RB-FZ-1C	75 Foot Elevation	1.03E-02	**4.2E-08**				
RB-FZ-1D	51 Foot Elevation	2.53E-02	2.4E-05		0.01		**2.7E-07**
RB-FZ-1E	Main Floor (23 Foot Elevation)	1.55E-02	5.6E-05		0.01		**1.3E-07**
RB-FZ-1F	Torus Area & Corner Rooms (-19')	8.05E-03	2.9E-06	**9.0E-07**			
RB-FZ-1G	Shutdown Cooling Area (38' and 51')	3.39E-03	**3.4E-09**				
RB-FZ-1H	Trunnion Room (23' 6" Elevation)	8.07E-04	**4.2E-10**				
OTHER PLANT FIRE AREAS							
MT-FA-12	Main Transformer and CST	2.14E-02	6.1E-06	**8.1E-07**			
DG-FA-15	No. 1 Diesel Generator Room	1.50E-02	**2.5E-08**				
DG-FA-17	No. 2 Diesel Generator Room	1.50E-02	**2.3E-08**				
FS-FA-16	Emergency Diesel Fuel Storage Area	1.02E-03	**1.6E-08**				
CW-FA-14	Circulating Water Intake Area	1.05E-02	1.4E-06	**5.6E-07**			

NOTE 1. The "Initial Quantitative Evaluation" column reports the upper bound core damage frequency from the first screening evaluation. These values are only used for screening purposes.

NOTE 2. The "Revised Quantitative Evaluation" column reports the upper bound core damage frequency following modeling revisions.

NOTE 3. The "Detailed Evaluation" columns present the factors used in the determination of the core damage frequency of the remaining fire areas. Other factors include manual and automatic suppression and remote shutdown panel use.

4.7 Containment Failure Modes Due To Fire

In NUREG-1407 Section 4.1.5 Perform Containment Analysis, states *"Perform containment analysis if containment failure modes differ significantly from those found in the IPE internal events evaluation. Perform in a fashion similar to an internal-initiator PRA."* Section 4.7.1 of this report presents the methods and approach used in the determination of the containment failure modes and impacts. Section 4.7.2 presents a summary of the impacts of fire events on the failure modes of containment by fire area and zone.

4.7.1 Containment Failure Impacts Method and Assumptions

The approach to determine the impacts of fire events on the failure modes of containment is a two step process. First, create a list all fire areas and zones that had a total core damage sequence frequency greater than 10^{-8} events per year. Second, from this list, the impacts of the fire events on containment isolation were derived (based on the component locations and cable locations). If the physical equipment is located in the fire area or zone, it is assumed failed by the fire and no recovery action is modeled. Recovery actions are modeled where remote manipulation of equipment is required by either the base OCPRA or where remote capability is lost due to cable damage. Manual operator actions are not modeled (guaranteed failed) if there is no path available to the component from the control room.

Information from the Location of Critical Components and Associated Cables matrix (Table 4.4-3) was used to determine the impacts of the "all engulfing fire" in each of the fire areas on the containment isolation. Containment cooling functions were not separately considered for this analysis, since these functions were explicitly assigned during the evaluation of core damage frequency. Each area fire impact was reviewed to determine if the all engulfing fire in the area could impact the primary or secondary containment isolation functions. Two types of impacts were identified:

- A partial failure of the containment isolation or secondary containment isolation. This type of failure is used when the redundant train is available or a recovery action is possible for the operation of valves.
- A total failure of a containment isolation or secondary containment isolation. A total failure of the top event or function was assigned if the fire could damage cables to redundant trains of equipment other than valves, such as actuation logic. It should be noted that this form of failure would require containment vent to be in progress or the simultaneous "hot short" failure of two valves in the same line to fail primary containment isolation (top event PI)

Failure of top event PI (Primary Containment Isolation) can occur if the containment vent valves are open and fail to close or a hot short occurs in control circuits that results in the opening of two containment isolation valves in the same line. The valves in this system are predominantly air operated, with solenoid valves installed in several sample lines and are designed to fail closed on loss of instrument air or power. Therefore, for purposes of quantification, the frequency for "hot shorts" of both the containment isolation valves for a single purge line was approximated by using manual containment isolation as an equivalent failure rate. This is judged to be conservative, since the operator error rate for this action is approximately 1.5×10^{-2}, which approximates a probability of a "hot short" failure rate of 10%, or 1 in 10 failures for each of the two isolation valves cables.

Failure of top event RI (Reactor Building Isolation) can occur if both valves in any of the nine (9) ventilation paths fail to isolate on loss of power or low-low RPV water level or high drywell pressure. This top event does not affect the frequency of core damage, its failure could impact the assignment of plant damage endstates however the Level 2 OCPRA does not model the effects of fission product scrubbing in the secondary containment due the assumed energetic failure of containment. Therefore, the status of this top event does not impact the type or timing of any potential fission product release and is not considered in the containment failure modes due to fire. Nevertheless, fire areas or zones which could impact secondary containment isolation (top event RI) are indicated as such.

Failure of top event SG (Standby Gas Treatment) can occur due to failure of the common supply line or failure of both filter/fan trains. As in the case above, the standby gas treatment system is not modeled in the Level 2 OCPRA due to the assumed energetic failure mode of the containment. Therefore, the status of this top event does not impact the type or timing of the fission product release. As such, the Standby Gas Treatment is not considered further in the evaluation of the effect of fire event on the failure modes of containment.

4.7.2 Containment Failure Impacts by Fire Zones

The results of a review of the containment isolation functions impacted due to fires in the various Oyster Creek fire zones are described below.

4.7.2.1 Reactor Building Fire Zone Impacts

Some fires in the Reactor Building could cause a failure of cabling to one of two valves in a containment penetration or some of the building isolation functions. These fire area impacts are listed as partial failures in the fire zone impact list.

> Fires in **Fire Zone RB-FZ-1A (Refuel Floor, 119 Foot Elevation)** have no impacts on primary containment isolation. There are no identified impacts on reactor building isolation. This area was screened from further consideration in Report Section 4.4.
>
> Fires in **Fire Zone RB-FZ-1B (95 Foot Elevation)** were screened to less than 1×10^{-8} events per year.
>
> Fires in **Fire Zone RB-FZ-1C (75 Foot Elevation)** could impact several primary containment isolation valves, in addition to several reactor building isolation valves. Due to the low upper bound core damage frequency generated for this area (4.2×10^{-8}) during the initial screening quantification, the potential failure of

either of these containment functions, when taken together with the conservative level of evaluation, this area can be removed from consideration.

Fires in **Fire Zone RB-FZ-1D (51 Foot Elevation)** could potentially impact control cables for several primary containment isolation valves. This area also contains several reactor building isolation valves. The upper bound core damage frequency for this fire area is $2.7x10^{-7}$ per year. this plant area was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone RB-FZ-1E (Main Floor, 23 Foot Elevation)** have the potential to impact several containment isolation valves. There were no identified impacts on reactor building isolation functions. The upper bound core damage frequency for this fire area is $1.3x10^{-7}$ per year. This fire area was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

A fire in **Fire Zone RB-FZ-1F (Torus Area and Corner Rooms, -19 Foot Elevation**) could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $9.0x10^{-7}$ per year. This fire area was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone RB-FZ-1G (Shutdown Cooling Area, 38 and 51 Foot Elevations)** were screened to less than $1x10^{-8}$ events per year.

Fires in **Fire Zone RB-FZ-1H (Trunnion Room)** were screened below $1x10^{-8}$ events per year.

4.7.2.2 Office Building Fire Zone Impacts

Some of the potential fires in the Office Building could cause a failure of cabling to one or two valves in a containment penetration. Since these valves are normally closed, both components would have to fail in an active, or "hot short" manner in order to open the valve and maintain it open. There are no identified impacts for reactor building isolation.

Fires in **Fire Zone OB-FZ-4** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $2.6x10^{-6}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore retained for further analysis.

Fires in **Fire Zone OB-FZ-5** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $3.3x10^{-7}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone OB-FZ-6A** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $5.1x10^{-6}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore retained for further analysis.

Fires in **Fire Zone OB-FZ-6B** do not have the potential to impact primary containment isolation.

Fires in **Fire Zones OB-FZ-8A and OB-FZ-8B** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for these fire areas is $8.7x10^{-8}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone OB-FZ-8C** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $5.1x10^{-7}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Area OB-FA-9** do not have the potential to impact primary containment isolation.

Fires in **Fire Zone OB-FZ-10A** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $4.9x10^{-7}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone OB-FZ-10B** do not have the potential to impact primary containment isolation.

Fires in **Fire Zone OB-FZ-22A** could potentially impact several primary containment isolation valves. There are no identified impacts on reactor building isolation. The upper bound core damage frequency for this fire area is $2.6x10^{-7}$ per year and was generated by assuming failure of primary containment isolation. This fire zone was therefore screened from further consideration based on an upper bound core damage frequency of less than $1x10^{-6}$ per year.

Fires in **Fire Zone OB-FZ-22B** were screened to less than $1x10^{-8}$ events per year.

Fires in **Fire Zone OB-FZ-22C** were screened to less than $1x10^{-8}$ events per year.

4.7.3 Review of Fire Impacts on Containment Failure Modes Results

The fire events that were not screened from further consideration in the initial screening process (fires in OB-FZ-6A and OB-FZ-4) were quantified using the Oyster Creek Level 1 IPE plant model. The model was exercised to confirm that these fires do not cause plant damage state results that significantly differ from or are significantly higher than those identified in the results from the Level 1 PRA analysis of internal events. The dominant plant damage states from this quantification are show in Table 4.7-1 below.

Table 4.7-1 Damage State Contributions for Unscreened Fires

Plant Damage State	Percentage of CDF from Unscreened Fires	Containment State
xLxx	60%	Late Containment Failure
xIxx	21%	Containment Intact
xKxx	13%	Early Containment Failure
xJxx	6%	Containment Bypassed
All PDS	100%	All Containment States

The following comments can be made concerning these results:

1. Endstates with containment bypassed (xJxx damage states) total 6%, which compares favorably with the value of 6% for with the Level 1 PRA report.

2. Endstates with early containment failure (xKxx damage states) total 13%, which, while higher, compares favorably with the value of 7% given in the Level 1 PRA report. This value is primarily due to the assumed immediate failure of all actuation logic top events (DP, PR and RL) for cable spreading room fires.

3. The late containment failure contribution is more significant than shown in the Level 1 PRA report, this is primarily due to the assumed unrecoverable failure of the containment spray system for several of these fires.

Therefore, as a result of this review, no containment failure modes have been found due to a fire that differ significantly from those found in the Oyster Creek IPE internal events evaluation.

4.8 Treatment of Sandia Fire Risk Scoping Study Issues

The EPRI FIVE documentation discusses the attributes of an adequate fire protection program. These attributes, which were originally taken from the Fire Risk Scoping Study (NUREG/CR-5088) performed by Sandia Laboratories (the Sandia Fire Risk Scoping Study Issues) are discussed below. The specific responses for each of these concerns for Oyster Creek are listed in italics directly below the description of the Sandia issue in the following report sections.

4.8.1 Seismic/Fire Interactions

The issues associated with seismic/fire interactions center on fire protection and suppression system design features which may prove to be vulnerable to beyond design basis seismic excitations. During a dedicated walkdown, several areas of the plant were reviewed for seismic/fire interaction issues involving both the fire systems and plant design features. Specifically, issues associated with the following were reviewed:

- Section 4.8.1.2 Seismic Induced Fire Initiation
- Section 4.8.1.3 Inadvertent Seismic Induced Actuation of Fire Suppression Systems
- Section 4.8.1.4 Seismic Induced Degradation or Diversion of Fire Suppression Systems

As a result, the potential for seismically induced fires is evaluated based upon a review of combustible materials and ignition sources in various critical plant areas, and information is gathered leading to the resolution of Generic Issue (GI) 57, "Effects of Fire Protection System Actuation on Safety Related Equipment", and GI 106, "Piping and the Use of Highly Combustible Gases in Vital Areas". In addition, those concerns expressed in Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety Related Equipment" have also been addressed.

Individual report sub-sections describe the above mentioned issues in further detail. Report sub-sections 4.8.1.1 describes the dedicated walkdown and report sub-section 4.8.1.5 provides a summary and conclusion of the seismic/fire walkdown analysis.

4.8.1.1 Seismic/Fire Interaction Walkdown

The following paragraphs describe the dedicated seismic/fire interaction walkdown of the Oyster Creek plant site. Included in the following paragraphs is a description of the walkdown team as well as a description of the fire protection systems and features reviewed and a description of the plant areas which were included in the walkdowns with the specific feature which warranted the area inclusion. This dedicated walkdown was performed following detailed walkdowns in support of the seismic fragility development (Seismic IPEEE) in which several walkdowns team members also participated.

Walkdown Team Description

The plant walkthrough was performed on October 17 and 18, 1994. The walkdown team was comprised of the following experienced personnel:

NAME	AFFILIATION	AREA OF EXPERTISE
Thomas Kipp	EQE International	Seismic Capability
Kenneth Canavan	GPU Nuclear	Risk & Systems Analysis
Raymond Daley	GPU Nuclear	Fire Protection
Timothy Trettel	GPU Nuclear	Fire Protection

Mr. Canavan is intimately familiar with the operation of the Oyster Creek safety systems and is responsible for the development of the seismic PRA risk model. Mr. Daley and Mr. Trettel are Fire Protection Systems Engineers at Oyster Creek and are intimately familiar with the details of the control and operation of the fire protection detection and suppression system in each fire zone. Mr. Kipp is a trained seismic capability engineer, has been intimately involved in the development and application of the seismic fragility methodology, and is responsible for development of the Oyster Creek seismic fragility descriptions for SSEL equipment.

Fire Protection Systems

1. Manual water hose station systems: Constitute the backup method of fire suppression in the plant fire areas with installed fixed suppression. In plant areas without fixed suppression, hose stations are the main method of fire suppression. Hose stations service major areas of virtually all important safety related buildings.

2. Wet pipe water sprinkler systems: Service general areas of the Office Building, the Turbine Building including several areas designated as Office Building fire zones, and selected areas of the Reactor Buildings as well as the Maintenance Building and the New Warehouse.

3. Dry, open head, water deluge systems: Manual and automatic systems service specific areas of the Office and Reactor Buildings primarily providing suppression for important cable trays, and specific areas of the Turbine Building, Fire Water Pump House, and the Yard where suppression is provided for the Turbine

Generator lube oil tank, the Fire Pump fuel oil tanks, and for the station transformers, respectively.

4. Dry, closed head, water preaction systems: Manually actuated systems service the North and South Cable Bridge Tunnel. No automatic preaction systems exist.

5. Carbon dioxide systems: Service the 4160 VAC C and D Switchgear Vaults and the Turbine exciter and bearing 10 areas of the Turbine Building.

6. Halon systems: Provide suppression for the Control Room panels in the Turbine Building, the 480VAC Switchgear rooms located in the Reactor Building, and the A and B Battery Rooms located in the Office Building, all of which are designated as Office Building fire zones.

Oyster Creek Plant Areas

The evaluations were conducted by plant walkdowns of specific areas identified by GPUN personnel, review of the various Fire Service Operating Procedures, review of automatic suppression system control panel electrical drawings, and interviews with Oyster Creek staff responsible for fire protection and for IPEEE fire evaluations. A summary of the fire detection and suppression equipment available in each of the fire areas/zones is listed in Table 4.8-1. Areas visually evaluated were selected on the basis of concentration of safety related components whose failure could affect plant performance, concentrations of combustible materials, and location of important fire protection components. Oyster Creek plant areas reviewed included:

AREA	REASON FOR INCLUSION
FW-FA-18	Diesel Driven Fire Pump; Fire Pump Fuel Oil Tanks
OB-FZ-4	Cable Spreading Room; Deluge system; Deluge Control Panel; Deluge Valves
OB-FZ-5A	Open control panels; Halon system
OB-FZ-6A	480V Load Center; Oil-filled transformers; Halon system
OB-FZ-6B	Similar to OB-FZ-6A
OB-FA-22A	Typical wet pipe sprinkler system
RB-FZ-1C	Oxygen Bottle Racks
RB-FZ-1D	Deluge System; Piping with mechanical couplings
TB-FA-3A	4160V Switchgear Room; CO_2 suppression
TB-FZ-11A	Turbine Generator combustion sources; Hydrogen piping
TB-FZ-11B	Turbine lube oil sources; Deluge valve station
TB-FZ-11D	Seal Oil Vacuum Tank; Hydrogen piping
YARD	Transformer deluge system: Low pressure CO_2 system storage tank

4.8.1.1 Seismic Induced Fire Initiation

Visual reviews of potential fire ignition sources and combustible inventories were performed for seismic vulnerability. In general, plant components which are not nuclear safety related but are located in safety related areas appear to exhibit design details able to resist nominal seismic loadings. Anchorage for such equipment appears to exhibit similar features. The anchorage for certain floor mounted safety related equipment is being upgraded as part of the A-46 evaluation of the plant and such upgrades may not apply to non-safety related components. Specific cases were noted in which non-class 1E electrical equipment, and/or single bays within a string of non-class 1E electrical equipment, were unanchored or the anchorage exhibited limited capacity (e.g., in the vicinity of the 480V Switchgear and in the vicinity of the 125VDC Batteries).

Samples of combustible and transient inventories were reviewed during the area walkthroughs. Areas reviewed included the Turbine Generator Hydrogen Seal Oil Unit, Turbine Generator Lube Oil Tank, oil-filled 480V Load Center Transformers, hydrogen piping located at the Turbine deck and at lower elevations in the Turbine Building, oxygen bottle stations in the Reactor Building, and several areas with large cable concentrations.

It appeared, in general, that the combination of ignition sources and combustion sources were well controlled. For example:

1. The 3" turbine generator lube oil lines which pass over transformer bus ducts and across the C and D 4160V Switchgear rooms is of welded pipe construction, and the lube oil manifold includes a guard pipe in this region.

2. The hydrogen lines located in various areas of the Turbine Building are generally well supported. However the hydrogen manifold located at the turbine deck elevation is at one place supported off the elevator shaft block wall which is not seismically designed or evaluated.

3. A small oil filter unit off of the Turbine Generator Hydrogen Seal Oil Unit is supported only by a vertical stanchion, and no lateral support is provided . The small diameter piping to and from the filter forms an approximately 8-foot cantilever which was found to be very flexible.

4. Oxygen bottles located in the Reactor Building are well supported.

5. Air locks, penetration seals, and concrete walls provide a significant barrier to the progress of any fire from the Turbine Building to the Reactor Building.

6. A 3-hour fire rated boundary exists between the general turbine generator areas and the 4160 VAC C and D Switchgear Vaults and C Battery Rooms.

7. No significant debris was observed, other than from materials being used during the outage activities. It appears that good housekeeping practices are being regularly followed.

Although all observed conditions are not considered ideal, they were judged unable to contribute significantly to a seismically induced fire scenario. No other unusual or unique seismic vulnerabilities were observed.

4.8.1.2 Seismic Induced Actuation of Fire Suppression Systems

Fire suppression systems are potentially susceptible to inadvertent actuation due to excitations occurring during a seismic event. Actions which could result in inadvertent actuation and/or discharge of fire suppressant include:

1. Change of state of mercury switches in automatic control systems due to seismic excitation or impact. Mercury switches are very sensitive to low frequency excitation.

2. Chatter of relays in automatic control systems due to excitation or relay cabinet impact.

3. Improper identification of seismic induced dust/debris movement as smoke by ionization detectors.

4. Release of ball in dropweight actuated deluge valves due to seismic excitation.

5. Shearing off of a sprinkler head due to impact with adjacent structural elements resulting from the large displacement response of the flexible fire protection system piping during the seismic event.

As noted above, several different fire suppression systems are used at Oyster Creek. These include manual hose, wet pipe sprinkler, automatic dry open head deluge, and manual dry closed head preaction water discharge systems, a Halon system, and a CO_2 system.

<u>Water Suppression Systems</u>

All water suppression systems located in plant areas containing safety related equipment appeared to be generally well supported for seismic loadings. Piping and hanger details that were reviewed incorporated structural detailing which appeared to be consistent with that utilized in commercial fire protection systems designed to NFPA 13 seismic requirements. For example, the wet pipe sprinkler and deluge systems were generally rod hung with occasional lateral support. Systems protecting cable trays included additional lateral support. It was noted that sprinkler heads generally had adequate clearance to nearby structural elements or other commodities to accommodate relative displacements. The deluge valve stations appear to be supported consistent with commercial practice (see Figures 1 and 2), some with very limited lateral support.

The water hose systems and wet pipe sprinkler systems compose the bulk of the water suppression systems. The water hose systems are all manual and thus are not susceptible to inadvertent actuation. For the wet pipe sprinkler systems, the detection equipment is most often for alarm purposes only. Since the wet pipe systems utilize closed head sprinklers, a fusible link must be melted or broken or a head must be broken off in order to have water discharged from the system. Mercury switches are only present in the alarm circuit for the Condenser Bay sprinkler system.

The automatic water deluge systems contain no mercury switches, and were generally found to require redundant (dual zone) product of combustion sensor/relay closure or thermal sensor/relay closure to result in inadvertent actuation of the deluge valves and resultant water release. Fire zones equipped with only ionization detectors require dual zone detection prior to actuation of the deluge system control valves except at the 75-foot elevation of the Reactor Building. Class 1E electrical panels located in fire zones employing automatic or wet pipe sprinkler suppression systems are generally sealed or equipped with sheet metal spray shielding to inhibit water intrusion. The deluge valves are equipped with a "pin and ball" (i.e., dropweight) actuation system which has been shown to be sensitive to shaking in some cases. This is considered a potential vulnerability for inadvertent actuation considering the limited lateral support provided for the valves and the likely impact with adjacent walls.

The open head deluge systems, except for the Transformer and Turbine Generator Lube Oil Tank systems, are configured with Pyrotronics System 3 fire detection and control equipment. The panels are wall mounted units secured by bolting to Unistrut members. A review of these systems indicated that they are of good industrial quality. Based on interviews with Oyster Creek fire protection personnel, the system controls are backed up with self-contained critical DC power supplies. The deluge system control equipment enclosures, conduit, and junction boxes in the control systems are not sealed against water intrusion. Particularly, the control panels located in the Turbine Building are situated in areas with non-safety water systems. Seismic induced failures of non-safety water systems could provide a mechanism for water intrusion into fire detection and control equipment which could in turn result in system actuation. In addition, a detection signal from a deluge system sensor closes the control relay (Model AE30 or SR-32) which then signals the deluge valve to open. The control/logic system requires either a dual signal from the detection system or the simultaneous chatter of the control relays from two zones to actuate the control valve. Thus, simultaneous chatter of two control relays during a seismic event is required to actuate the open head deluge system resulting in inadvertent water discharge. In one specific case, the control panel for the lower Cable Spreading Room suppression system is mounted to a non-seismic block wall. The increased panel response, should the block wall begin to fail during a seismic event, would increase the potential for simultaneous chatter of the relays within the panel and for inadvertent water discharge.

Similarly, the Transformer and Turbine Generator Lube Oil Tank Systems are activated from a single Hoffman control panel which receives fire indications from Fenwall thermal sensors. A detection signal from a thermal sensor closes the control relay (Line Electric Model MKH2D) which then signals the deluge valve to open). The panel is again a wall mounted unit secured by bolting to Unistrut members and appears to be of good industrial quality.
The fire suppression system for the Station Black Out (SBO) Transformer is a closed head, pressurized dry pipe system with nitrogen purge gas. Loss of pressure in the line due to breaking of one of the discharge heads or rupture of the piping operates the discharge valve. The discharge header is supported off a non-seismic structure before reaching the SBO Transformer as shown in Figure 3. The distribution system piping in the vicinity of the transformer is shown in Figures 4 and 5. It can be seen that an unanchored Arrowhead Industrial Water trailer is situated within 6 inches of the distribution system support columns. Such proximity poses a vulnerability for inadvertent water discharge of the deluge system.

CO_2 Suppression Systems

Two CO_2 suppression systems provide for protection of the Turbine Generator Exciter and Turbine bearing No. 10 and for protection of essential electrical panels. The Exciter and Turbine bearing system is a high pressure system with CO_2 flow being directed from eight (8) bottles located in the Turbine building at elevation 23'-6". These are shown in Figure 6. The approximately 10" diameter by 60" tall bottles are attached together to a support frame, but the frame is secured to the floor by only (2) 1/4" bolts which provide virtually no restraint against overturning The support frame is not secured to the wall. The Fenwall local control panel is located near the high pressure bottles and receives fire indication from Fenwall thermal detectors. A detection signal from one of the thermal sensors closes the control relay (Model SR-32) which in turn signals the discharge valve to open. However, the high pressure CO_2 system is discharged in timed intervals over 30 minutes during Turbine coast down, thus mitigating the effects of loss of suppressant during the seismic event. Control system issues related to seismically induced inadvertent actuation of the high pressure CO_2 system are similar to those for the water deluge systems identified above.

The low pressure CO_2 system, which provides fire suppression capability in the vicinity of the C and D 4160V Switchgear Vaults, is a manually activated system with the detection system for alarm purposes only. The CO_2 is supplied from the CO_2 Storage Tank, fabricated by Chemtron and located outside the Turbine Building (see Figures 7 and 8). The tank is set on two channel runners which are each secured by two (2) visible anchor bolts. Attachment of the tank to the runners could not be determined. It is believed that the tank is unanchored. An approximately 3' length of flexible hosing is provided between the CO_2 tank and the Turbine Building wall penetration to accommodate some lateral movement of the tank. The manually activated CO_2 discharge valve is located near the 4160V Switchgear Room. Inadvertent actuation is not considered an issue for this system..

Halon Suppression Systems

The Halon suppression system is provided for protection of the Control Room Panels, 480V Switchgear Rooms, and the A and B Battery Rooms. The Control Room Panels are open back/open top panels. The Control Room Panel halon system is divided into three (3) sections with two (2) ionization sensors each and constitutes a local application of the area system. The A and B sections are tied together with only one operational at any given time and the C section is independent. The detection and control panels for the Control Room Panel halon system are secured to the end of the various Control Room Panel strings. The halon is ejected from a 600 psi pressurized system and actuation would likely stir up dust and debris. Under such conditions, the remaining active section could also be activated.

The halon system for the A and B Battery Rooms includes dual zone ionization detection. A Firelite control system is employed for these Battery Rooms.

The detection system for the 480V Switchgear Rooms consists of ionization and photoelectric sensors, both of which must provide indication of fire to activate the halon system. The halon control panel for the area is mounted to a seismically evaluated block wall for which Unistrut bracing is provided. Control system issues related to seismically induced inadvertent actuation of the halon system are similar to those for the water deluge systems identified above. The halon bottles for the 480V Switchgear Rooms were visually reviewed and were found to be well supported. Six bottles, which are approximately 16" in diameter by 4' tall, provide suppression for the area and are supported by two structural steel support stands (3 bottles each). Three steel straps are used to secure each bottle to a support stand. The bolts for the straps are threaded into the web of structural channel sections which form part of the support stand. No unusual or unique vulnerabilities were observed.

4.8.1.3 Seismic Induced Degradation or Diversion of Fire Suppression Systems

The Fire Protection systems were reviewed to assess the extent to which such systems would be available to suppress a fire following a significant seismic event.

Water Suppression Systems

The review of the water suppression system included a review of the two diesel powered fire pumps, associated control panels and ancillary equipment, and the distribution system. The diesel powered fire pumps (Figures 9 through 12) are located in the remote Fire Water Pump House which is set in a small fresh water pond created by a man-made dam. The dam, which was damaged under heavy runoff conditions after deterioration of the riprap supporting the wooden weir, has recently been rebuilt and strengthened and appears adequate for potential earthquake levels. The pumps are small vertical column pumps which appear to be well anchored. These are typically robust and perform well during earthquakes. The small diesel engines are also well anchored by six (6) 5/8" to 3/4" embedded bolts (3 each side). The distribution piping is deadweight supported only except that the penetration through the Pump House floor, leading to the underground run, provides some lateral support. The four (4) batteries for each diesel are set unanchored in a "coffin" container with each battery set in its own 1"x 1" angle iron cradle, precluding impact between batteries. The two fuel tanks are located outside the Pump House. The tanks are stitch welded to two saddles (outside only) which are in turn bolted to concrete pedestals. Four (4) 1/2" bolts are provided for each tank. The diesel control panel also appears to be adequately anchored. The FWPH appears to be well designed and is constructed of corrugated steel above the ground level.

The fire protection system in the pumphouse includes a small pipe-mounted deluge valve and a Pyrotronics, System 3 control panel which is not sealed. The small diameter distribution system is rod hung with horizontal vise clamps. Sufficient clearance is provided for the twelve (12) distribution heads to preclude impact.

It was noted that the fire system is designed with common pumps and delivery headers which provide protection for safety related areas inside the main power block and non safety areas outside the power complex. The fire water system services several general purpose buildings and plant facilities. These include structures with limited seismic capacity including office and warehouse facilities. The ability of the fire water system to deliver adequate flow to any post-seismic fire demands inside the plant safety related areas could be compromised due to failures of fire water lines in some of the non-safety structures. Actions by the plant operators or the fire brigade would be required to isolate any damaged portions of the system. The portion of the system servicing the non-safety buildings can be isolated within a reasonable time frame by valves provided at the connection to each non-safety building and/or by valves in the underground manifold. The pumping capacity of fire pumps is 2000 GPM each or in combination, at a pressure of 156 psi. This is likely sufficient to meet potential safety related area demands even considering failure of some non safety systems.

Some of the vulnerabilities of the fire water system noted above in connection with inadvertent actuation also limit the ability of the fire protection system to perform subsequent to a seismic event. These include control panels mounted to block walls which are not seismically designed or evaluated, an unanchored water van located in close proximity to the SBO Transformer fire protection system distribution piping, and limited lateral support for the deluge valve stations. Another identified vulnerability is related to headers with mechanical couplings. Such mechanical couplings are common in commercial applications but are unable to carry significant piping moments which could occur at the joint in response to seismic excitation.

CO_2 Suppression Systems

As noted above, the low pressure CO_2 system storage tank together with the sensing and distribution lines (Figures 7 and 8) is located outside the Turbine Building. The sense and distribution lines run a somewhat circuitous and unprotected path through the Turbine Building to the C and D 4160V Switchgear Vaults. Due to the myriad of non-safety piping systems and equipment found in the Turbine Building along the path, it is possible that failure of some non-safety component during a significant seismic event could sever the lines rendering the CO_2 system inoperable for providing suppressant to the Switchgear vault. In addition significant displacement of the unanchored storage tank or toppling of the unanchored CO_2 cylinder rack located 7' from the storage tank (the rack holds up to sixty (60) cylinders which could potentially become missiles if the valves are broken) could also render the CO_2 system inoperable for providing suppressant to the Switchgear Vault. In such cases, other manual fire fighting methods available in the area would have to be used to combat a seismic induced fire.

The lack of adequate restraint for the high pressure CO_2 bottles could also render the fire suppression system for the Turbine Exciter and Turbine bearing #10 inoperable. Other manual fire fighting methods available in the area would have to be used to combat a seismic induced fire in this case as well.

4.8.1.4 Seismic/Fire Interaction Analysis Conclusions

Based on a review of the Oyster Creek plant site including the fire protection systems, specific plant areas and features the following conclusions can be made:

- Based on the review of the fire protection system documented above, no sources of seismic induced fire initiation at reasonable levels of earthquake beyond the design basis were identified.

- For inadvertent actuation, it was noted that in areas where safety related equipment is located, the fire protection systems are either manual or are automatic deluge systems requiring redundant sensor/relay circuit closure to result in actuation of control valves and local head actuation for water release. Potential inadvertent actuation related to the deluge valve dropweight actuation mechanism, fire control panels mounted on non-seismic block walls, and proximity of an unanchored van to distribution system piping were noted. Very few instances were noted where sprinkler heads were in close proximity to other plant features. Sensitive electrical equipment is sealed and/or equipped with sheet metal or plastic spray shields to protect them from the effects of water intrusion.

- Some potential fire suppression flow diversion paths were noted following a seismic event. Actions by plant operators following the seismic event could isolate flow diversions of the fire suppression system and in the case of 4160 VAC C and D switchgear manual fire fighting methods can be used. Since a seismic induced fire does not appear to be a very probable event and manual action can be used to support fire fighting this issue is not considered a potential vulnerability. As such the issue is considered closed.

The potential for seismically induced fire events is low. Although some potential for seismic actuation of suppression systems and flow diversion exists, these represent only potential plant improvements since the low likelihood of fire initiation coupled with manual actions to restore fire suppression systems, results in low total risk. Seismic/Fire Interaction issues are considered closed.

Table 4.8-1 Oyster Creek Seismic Fire Interaction Walkdowns									
Fire Area/Zone	Fire Area Description	Fire Detection Equipment			Fire Suppression Equipment				Comments
		Product of Combustion	Thermal	Photo-electric	Deluge Systems	Wet Pipe Sprinkler	Halon	CO_2	
OFFICE BUILDING FIRE AREAS AND ZONES									
OB-FZ-4	Cable Spreading Room - El 36'	X[1]			X(2)				Installed to protect cable. Two (2) separate deluge systems.
OB-FZ-5	Control Room - 46' 6" Elevation	X[1]					X(3)		Auto Halon actuated by cross-zoned POC (inside panels (3 separate systems)).
OB-FZ-6A	"A" 480 VAC Switchgear Room	X		X			X		Actuation of both POC and photoelectric detectors required for halon discharge.
OB-FZ-6B	"B" 480 VAC Switchgear Room	X		X			X		
OB-FZ-8A/B	MG Set Room/ Mechanical Equipment Room					X			NOTE 2
OB-FZ-8C	A and B Battery Room, Tunnel and Electric Tray Room (35' Elevation)	X[1]					X		NOTE 2
OB-FA-9	Office Building	X							Manual suppression.
OB-FZ-10A	Monitoring and Change Room - 46'	X				X			Single zone - detection actuates alarm only.
OB-FZ-10B	Chem Lab, Instrument Shop-35'	X							Manual Suppression.
OB-FA-22A	New Cable Spreading Room (Mech Equip Room) 63' 6" El.	X				X			Single zone detection actuates alarm only.
OB-FA-22B	North Cable Bridge Tunnel, 74' 6"	X				X			Manually actuated closed head sprinkler.
OB-FZ-22C	South Cable Bridge Tunnel	X				X			Manually actuated closed head sprinkler.
OB-FZ-22D	Mech Equipment Room (74')								Manual Suppression.
TURBINE BUILDING FIRE AREAS AND ZONES									
TB-FA-3A	4160 VAC Switchgear 1C - Mezz 23'	X						X	Manually actuated low pressure CO_2 system. Detectors are alarm only.
TB-FA-3B	4160 VAC Switchgear 1D - Mezz 23'	X						X	
TB-FZ-11A	Turbine Operating Floor - 46'El.		X			X		X	Wet pipe for bearings only. High pressure CO_2 systems (2) are for exciter and turbine bearing No. 10. Actuated by thermal detection. Manual chemical ext.
TB-FZ-11B	Lube Oil Storage, Pumping and Purification Areas, 0' and 27' Levels		X		X	X			Deluge system over tank. Wet pipe sprinkler is area wide.

Table 4.8-1 Oyster Creek Seismic Fire Interaction Walkdowns									
Fire Area/Zone	Fire Area Description	Fire Detection Equipment			Fire Suppression Equipment				Comments
		Product of Combustion	Thermal	Photo-electric	Deluge Systems	Wet Pipe Sprinkler	Halon	CO_2	
TB-FZ-11C	Switchgear Room, SW End of Mezzanine Level (Elev. 23' 6")	X							Manual Suppression.
TB-FZ-11D	Basement Floor, South End					X			NOTE 2
TB-FZ-11E	Condenser Bay, Elevation 3' 6"					X			NOTE 2
TB-FZ-11F	Feedwater Pump Area, 0'6" & 3'6"								Manual Suppression.
TB-FZ-11G	Mezzanine Lvl SE Corner, 23' 6"					X			NOTE 2
TB-FZ-11H	Demineralizer Tank and Steam Jet Air Ejector Area, Elevation 23' 6"								Manual Suppression.
TB-FA-26	Battery Room "C"	X							Manual Suppression.
REACTOR BUILDING FIRE ZONES									
RB-FZ-1A	119 Foot Elevation					X			NOTE 2
RB-FZ-1B	95 Foot Elevation	X							Manual Suppression.
RB-FZ-1C	75 Foot Elevation	X			X	X			Wet pipe sprinkler over west side (spent fuel pool cooling pumps). Water curtain below slab over equip. hatch. Spray nozzle over NW floor opening. Both are extensions of deluge system in RB-FZ-1D.
RB-FZ-1D	51 Foot Elevation	X[1]			X(2)				Two deluge systems which provide (1) spray to protect grouped cables in trays, (2) water curtain below floor over equip. hatch, and (3) spray nozzle over NW floor opening.
RB-FZ-1E	Main Floor (23 Foot Elevation)	X[1]			X(2)				Duct penetration by equipment hatch protected by water spray at 35' el. Ceiling opening at northwest stairwell is protected by spray nozzles.
RB-FZ-1F	Torus Area & Corner Rooms (-19')	X							Manual Suppression. No detection in torus area.
RB-FZ-1G	Shutdown Cooling Area (38' and 51')	X							Manual Suppression.
RB-FZ-1H	Trunnion Room (23' 6" Elevation)								Manual Suppression.

Table 4.8-1 Oyster Creek Seismic Fire Interaction Walkdowns

Fire Area/Zone	Fire Area Description	Fire Detection Equipment			Fire Suppression Equipment				Comments
		Product of Combustion	Thermal	Photo-electric	Deluge Systems	Wet Pipe Sprinkler	Halon	CO_2	
OTHER PLANT FIRE AREAS									
MT-FA-12	Transformers (Yard)		X		X(3)				Three thermally actuated water spray systems.
AB-FZ-13	Auxiliary Boiler House		X						Manual suppression.
DG-FA-15	No. 1 Diesel Generator Room	X	X						Manual Suppression. POC in switchgear and thermal in building.
DG-FA-17	No. 2 Diesel Generator Room	X	X						
FW-FA-18	Fire Water Pump House		X		X				Deluge system. Sprinklers inside building are closed head. Open head sprinklers over outside fuel tanks.
OG-FA-21	Augmented Off Gas Building	X	X						Manual Suppression.
NW-FA-23	New Warehouse					X			
MB-FA-24	Maintenance Building					X			
PH-FA-25	Redundant Fire Pump House								Manual Suppression.
OR-FA-19	Old Radwaste Building	X							Manual Suppression.
NR-FA-20	New Radwaste	X	X						Manual Suppression.
FS-FA-16	Emergency Diesel Fuel Storage Area		X						Manual suppression.
CW-FA-14	Intake Area								Manual Suppression.

Notes:
1. Cross zone POC detection required for automatic system actuation in these areas.
2. Wet pipe sprinklers in these areas have flow or pressure switch which provide alarms in the control room.
3. Manual Suppression as indicated in the comments column is performed using nearby hose stations or portable extinguishers.
4. The following buildings and structures are not included:
   - Site Emergency Building (EB-FA-28)
   - Low Level Radwaste Storage Facility (LL-FA-29)
   - New Administration Building
   - Rad Con Building No. 3 (wet pipe sprinklers)
   - Maintenance Shop Building No. 4 (wet pipe sprinklers)
   - Heater Bay Roof Buildings (dry pipe sprinklers)
   - Respirator Building (fire detection system)
5. With few exceptions, all fire areas and zones have nearby hose stations.
6. Generally, isolation valves are available in the fire header and/or at fire protection line entry into buildings.

Figure 1:
Typical Support for Deluge Valve Bank

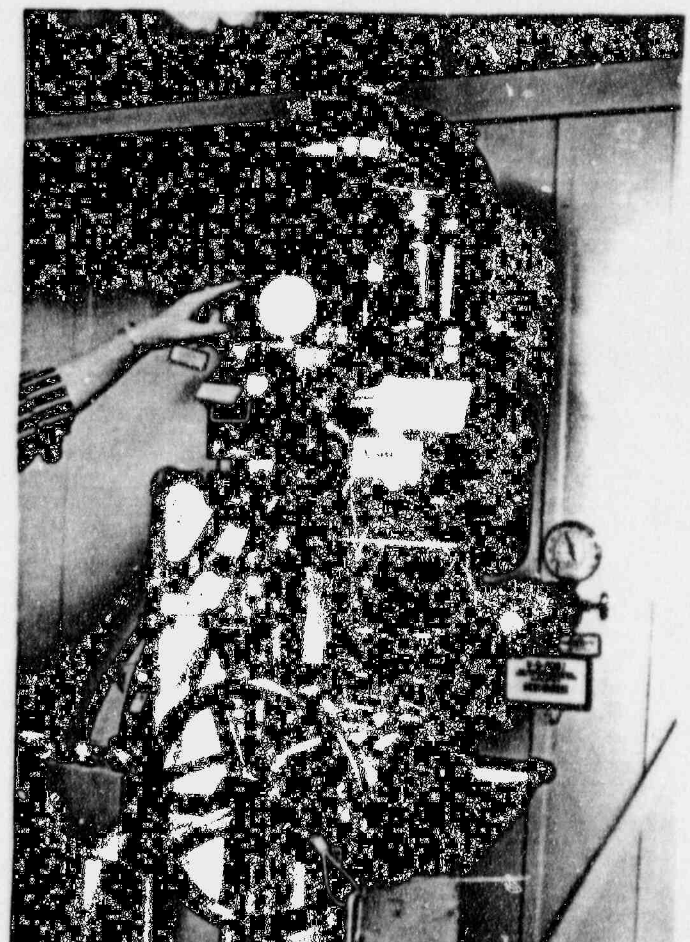


Figure 2:
Recently Upgraded Deluge Valve Station

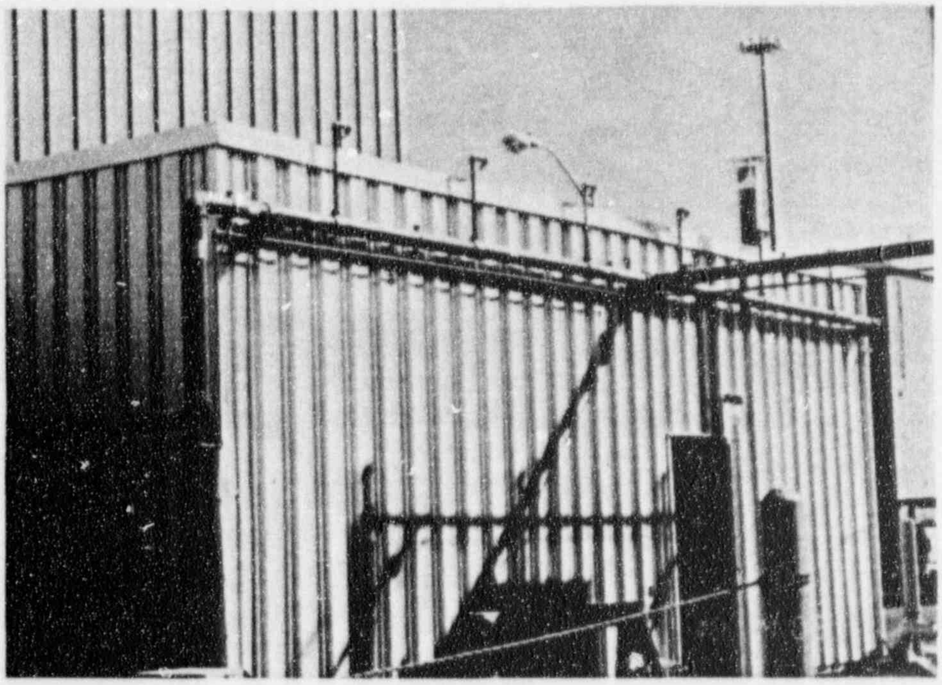


Figure 3: Support for the Fire System Header Leading to the SBO Transformer

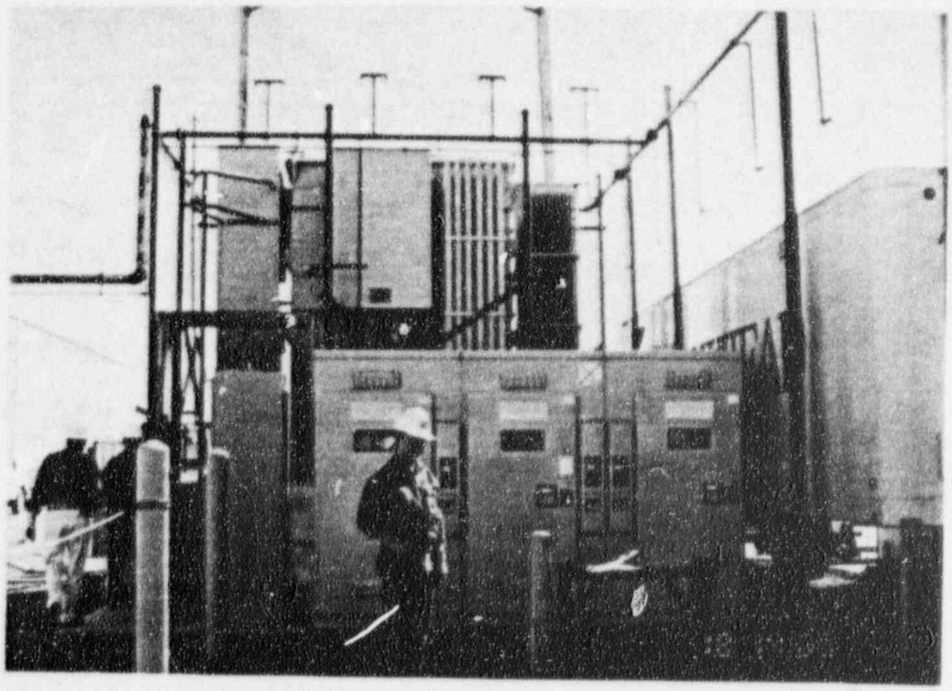


Figure 4: SBO Transformer Fire Suppression Distribution Piping and Nozzles

Figure 5:
Unanchored
Industrial Water Van
in Close Proximity to
the SBO
Transformer Fire
Suppression
Distribution Piping
and System Support

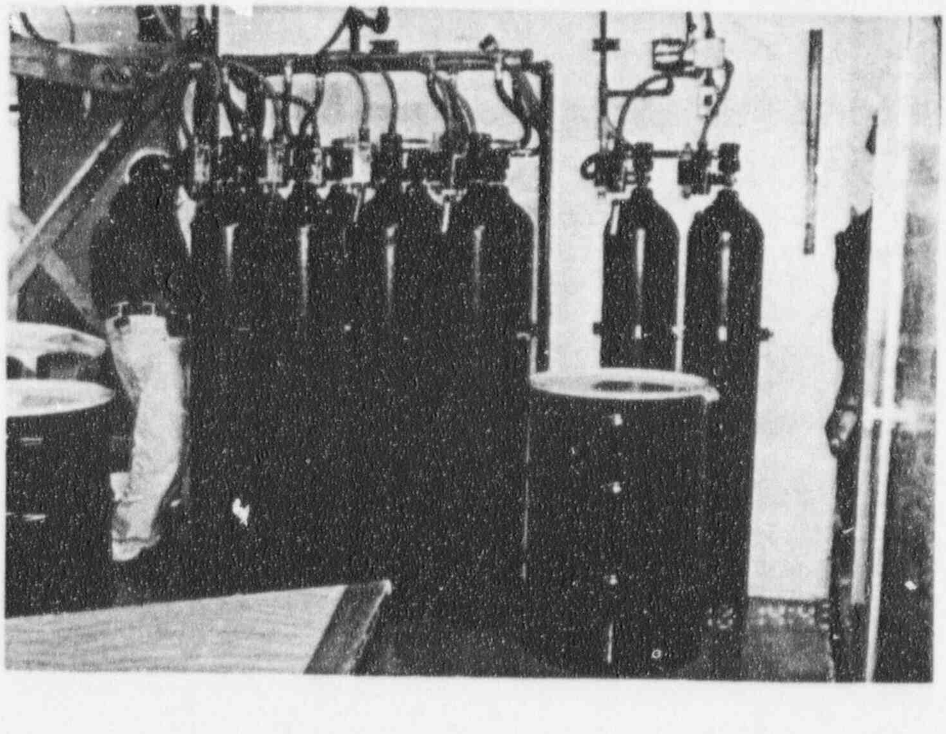
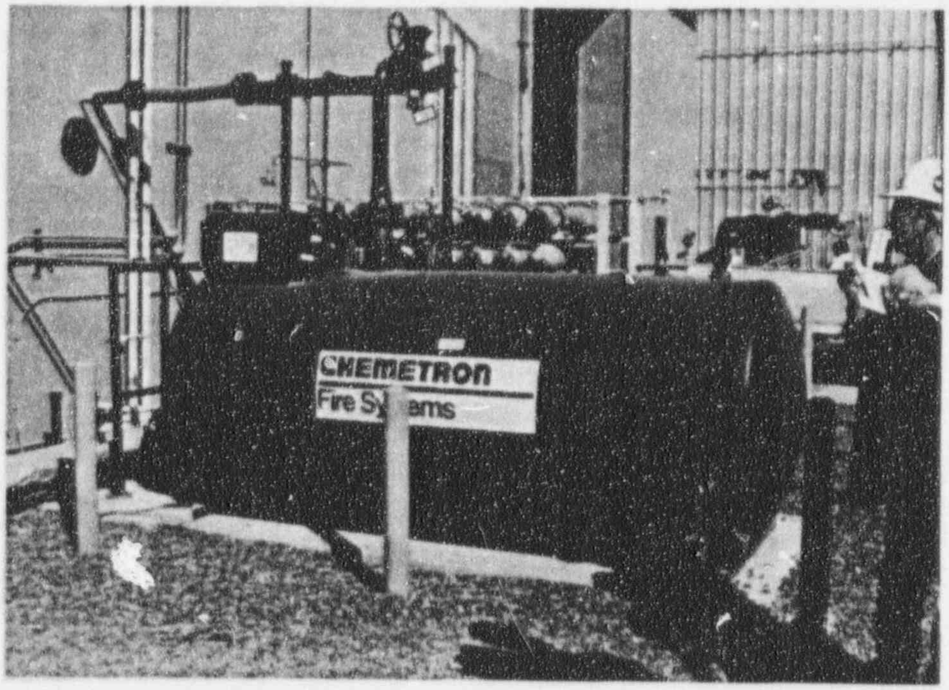


Figure 6:
High Pressure
CO_2 Storage
Bottles and
Support Frame

Figure 7:
Low Pressure CO_2 Storage Tank and Interface Piping



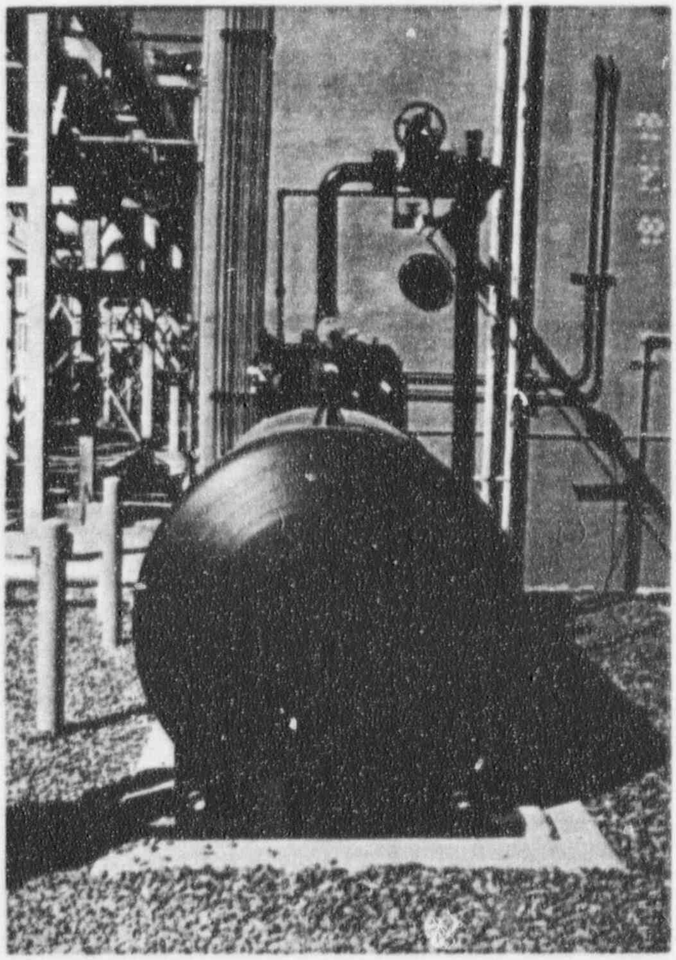


Figure 8:
Low Pressure CO_2 Storage Tank Mounting

EQE

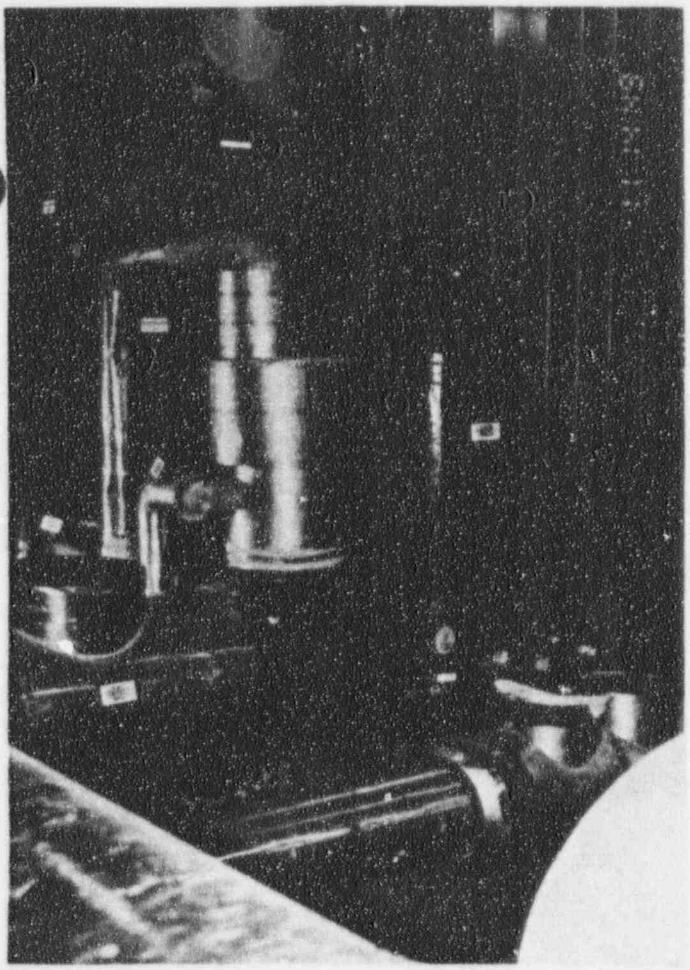


Figure 9: Diesel Driven Fire Water Pump

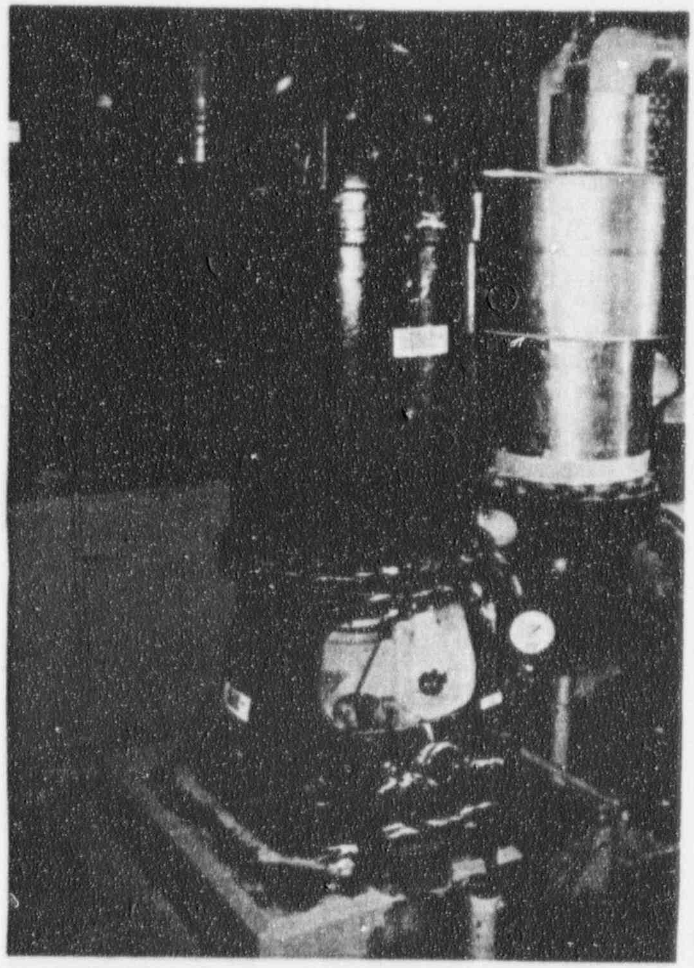
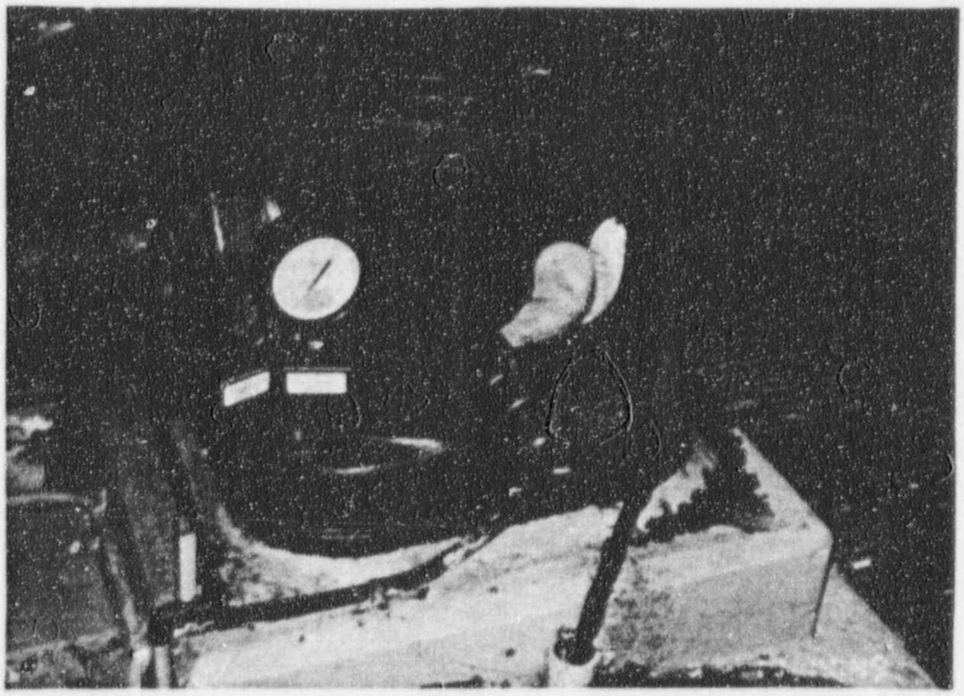


Figure 10:
Diesel Driven Fire Water Pump and Engine

Figure 11:
Diesel Driven
Fire Water
Pump
Anchorage



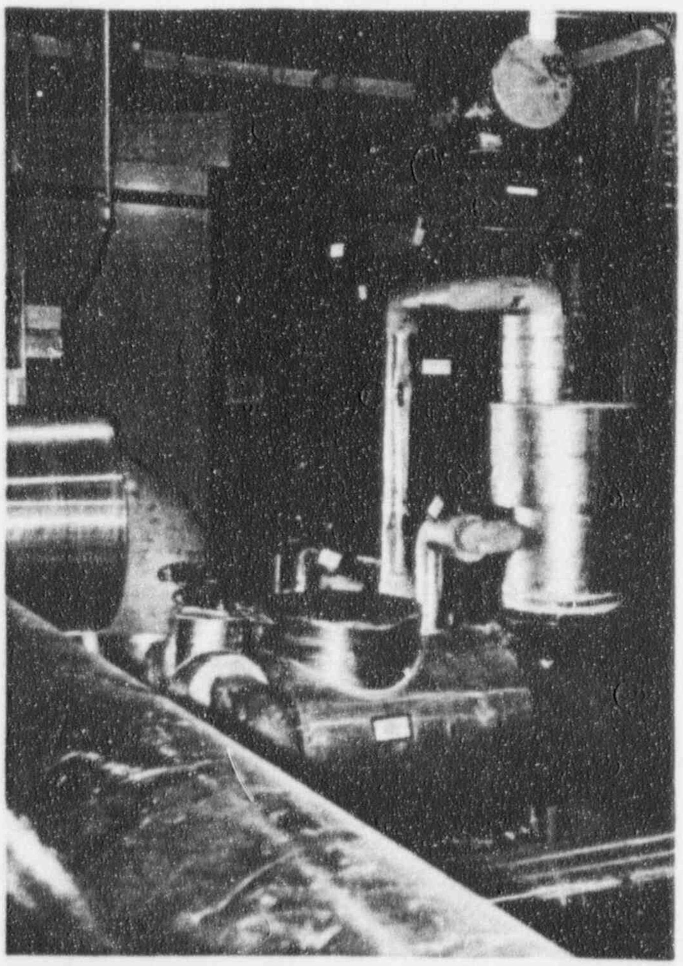


Figure 12:
Diesel Driven Fire Water Pump
Engine and Control Panel

4.8.2 Fire Barrier Qualifications

The concern for fire barrier qualification centers on the following 4 areas of interest:

- Fire barrier surveillance program.
- Inspection and maintenance of fire doors.
- Installation, inspection, surveillance and maintenance of penetration seal assemblies.
- Inspection, testing and maintenance of fire dampers.

Fire Barrier Surveillance

Fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers are included in the plant surveillance program.

Fire barriers are included in the plant surveillance program. Surveillance procedure 645.6.017 is used to visually inspect fire barrier penetration seals each refueling interval and following repairs or maintenance on the affected seal. The seals are inspected to verify that significant degradation has not occurred, the seal has not separated from the sealing surface, or excessive shrinkage has not occurred. The barriers are checked for defects such as rupture or puncture of the barrier surface, gaps, voids, or cracks in the joint area. Fire barrier maintenance and repair is controlled by procedure A100-GMM-3900.55, Fire Barrier Penetration Seal Repair and Installation Requirements.

Fire Doors

A fire door inspection and maintenance program should be implemented at the plant.

The operation of fire doors is specified in the Oyster Creek Fire Hazard procedure, number 120, which lists the plant fire doors in Attachment 120-1. Also, the Fire Protection Inspection procedure, number 119.1, verifies the requirements of procedure 120 on no less than a monthly basis, and weekly during outages.

Penetration Seal Assemblies

a. A penetration seal inspection and surveillance program should be implemented at the plant.

   Fire barriers penetration seals are included in the plant surveillance program. Surveillance procedure 645.6.017 is used to visually inspect fire barrier penetration seals each refueling interval and following repairs or maintenance on the affected seal. The seals are inspected to verify that significant degradation has not occurred, the seal has not separated from the sealing surface, or excessive shrinkage has not occurred. The barriers are checked for defects such as rupture or puncture of the barrier surface, gaps, *voids, or cracks in the joint area. Fire barrier maintenance and repair is controlled by procedure A100-GMM-3900.55, Fire Barrier Penetration Seal Repair and Installation Requirements.*

b. Fire barrier penetration seals have been installed and maintained to address concerns such as those identified in NRC Information Notice 88-04.

NRC Information Notice 88-04 and Supplement 1 to Information Notice 88-04 were reviewed by the onsite Fire Protection Engineer. The concerns identified were evaluated and found to be satisfactorily addressed by the programs in place to install, repair, and test the fire seals. (Ref.: Licensing Action Item No. 88021.02 and GPUN Memorandum 2200-88-560B)

Fire Dampers

a. An inspection and maintenance program for fire dampers should be implemented at the plant.

Fire dampers are inspected as part of the monthly fire protection inspection, as noted in procedures 119.1 and 120.

b. Damper installations address concerns such as those identified in NRC Information Notice 89-52, "Potential Fire Damper Operational Problems," dated June 8, 1989 and NRC Information Notice 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," dated October 21, 1983.

NRC Information Notices 89-52 and 83-69 were reviewed by the onsite Fire Protection Engineer. The identified concerns were evaluated and found to be satisfactorily addressed by the programs in place to test the fire dampers and to ensure damper closure in case of fire. (References 4.10-33 and 4.10-35).

4.8.3 Manual Fire Fighting Effectiveness

The concern for manual fire fighting effectiveness centers on the following 6 areas of interest:

- Fire reporting, including the use and availability of portable fire extinguishers and plant procedures for reporting fires, including plant communication.
- Fire brigade makeup and equipment.
- Fire brigade training in the classroom.
- Fire brigade practice in hands-on structural fire training and in the use of equipment.
- Fire brigade drills.
- Fire brigade training records.

Reporting Fires

a. Appropriate plant personnel are knowledgeable in the use of portable fire extinguishers.

Shift personnel are periodically trained on the usage of portable fire extinguishers.

b. Portable extinguishers are located throughout the plant.

Numerous portable extinguishers are located throughout the plant and their locations are visibly marked.

c. A plant procedure is in use for reporting fires in the plant.

All personnel that are granted unescorted access to the plant receive yearly General Employees Training regarding the actions to be taken when a fire is discovered. This training includes information on fire extinguisher types, selection, and use. Procedure 120.4, General Response to Fire, directs plant personnel to immediately report all fires to the control room, evacuate unnecessary personnel in the area and attempt to fight the fire with portable extinguishers if it is safe to do so.

d. A plant communication system that includes contact to the control room is operable at the plant.

The plant has several communications systems available for calling or paging the control room.

Fire Brigade Makeup and Equipment

1. A fire brigade that is made up of at least 5 trained people on each shift should be maintained at the plant.

   In accordance with procedure 101.2, Fire Protection Program, the Oyster Creek fire brigade, which is made up of at least five members, is maintained on site at all times.

2. The fire brigade leader and at least two other brigade members on each brigade shift should be knowledgeable in plant systems and operations.

   The Fire Brigade Leader is a licensed operator and the fire brigade contains at least four other trained members.

3. Each brigade member should receive an annual review of physical condition to evaluate his ability to perform fire fighting activities.

Procedure 101.2, Fire Protection Program, requires the fire brigade members to complete the fire school training program and satisfactory completion of an annual physical examination.

4. A minimum amount of equipment should be provided for the on site fire brigade:

The fire fighting emergency equipment inventory verification procedure is performed on a weekly basis to determine the availability of equipment for fire brigade use. The fire protection engineer is tasked with replenishing the supply of fire brigade equipment if the inventory falls below the minimum level listed in the procedure. The fire brigade equipment is located in several locations: Equipment Stowage Building (adjacent to building 3), Monitor and Change Room, locked tool box at the redundant fire system pump house, hose house, fire brigade van and the new warehouse fire lockers.

Also, procedure 120.7, Appendix "R" Emergency Preparedness, requires inventory and inspection of all Appendix R equipment annually and after each use.

a. Personal protective equipment should be provided such as SCBA, turnout coats, boots, gloves, and hard hats.

SCBA equipment is located in strategic areas of the plant for emergency use. In addition, the following equipment is located in the Equipment Stowage Building, the Monitoring and Change Room and the New Radwaste Building:

	Equipment Stowage Building	*Monitoring and Change Room*	*New Radwaste Building*
Helmets	*11 each*	*5 each*	*5 each*
Coats	*11 each*	*5 each*	*5 each*
Boots	*12 pairs*	*5 pairs*	*8 pairs*
Gloves	*11 pairs*	*5 pairs*	*5 pairs*
MSA Air Packs	*10 each*	*5 each*	*3 each*
Portable Lights	*11 each*	*5 each*	*5 each*
MSA Bottles	*9 each*		
Locks	*2 each*		
Spanner Wrenches	*5 each*		*5 each*
Nomex Hoods	*11 each*		*5 each*
Axe	*1 each*		
Pry Bar	*1 each*		
ABC Dry Chem Ext.	*2 each*		
Incident Command Vest	*2 each*	*1 each*	

In addition to the above, 17 hose houses located throughout the Oyster Creek site also *contain various fire fighting equipment. These hose house are locked closed. If hose house locks have been tampered with full inspection of the contents is performed. Lastly, the fire brigade van is fully equipped with hoses, wrenches, pike poles, megaphone, spare air bottles and various other fire fighting equipment.*

b. Emergency communications equipment should be provided for fire brigade use.

Numerous fire brigade members carry portable radios during their normal duties in the plant, therefore, no additional radios are required to carry out fire brigade duties. These radios can also be used to communicate directly with the control room. Other means of communication, such as in-plant phones and sound powered phones are also available, if needed.

c. Portable lights should be provided for fire brigade use.

A total of 21 portable lights are available from the Equipment Stowage Building (11), the Monitoring and Change Room (5) and the New Radwaste Building (5).

d. Portable ventilation equipment should be provided for fire brigade use.

Fire Brigade Van	*One (1) 16 inch 4500 cfm electric smoke ejector fan with door bar, reel extension cord, flexible duct collar and elephant trunk (16 inch x 20 feet round duct).*
Monitor and Change Rm	*One (1) 16 inch 4500 cfm electric smoke ejector fan: with door bar, reel extension cord, flexible duct collar and elephant trunk (16 inch x 20 feet round duct).*

Additional smoke ejector fan available in the Equipment Stowage Building. Currently procuring a portable cart with installed electrical breaker and power cord for use with a 220 VAC fan to be staged in Turbine Building mezzanine area.

e. Portable extinguishers should be provided for fire brigade use.

Portable fire extinguishers are strategically located throughout the plant. The fire brigade team is instructed to bring extinguishers from the areas they pass through on their way to the fire scene.

Fire Brigade Training

Brigade members should receive an initial classroom instruction program consisting of the following:

a. A review of the plant fire fighting plan and identification of each individual's responsibilities.

The Fire Protection Program, procedure 101.2, requires all fire brigade members to complete an initial fire school training program, as well as brigade training to maintain qualification status. This training includes indoctrination to the plant fire fighting plan, with specific coverage of each individual's responsibilities.

b. Identification of typical fire hazards and associated types of fires that may occur in the plant.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum: coverage of fire hazards and the types of fires that may occur in the plant; and an identification of the location of such hazards.

c. Identification of the location of fire fighting equipment and familiarization with the layout of the plant, including access and egress routes.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum: identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes.

d. Training on the proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays and fires involving flammable and combustible liquids and gases.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum the proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered include electrical fires, fires in cables and cable trays, hydrogen fires, flammable liquid fires and waste debris fires.

e. Training on the proper use of communication, lighting, ventilation and emergency breathing equipment.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum the proper use of communication, lighting, ventilation and emergency breathing equipment.

f. Training on techniques for fighting fires inside buildings and confined spaces.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum the proper method for fighting fires inside confined spaces.

g. A review of fire fighting strategies and procedures.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to complete an initial training program that includes as a minimum a detailed review of fire fighting procedures and procedure changes and changes in fire fighting plans.

Fire Brigade Practice

Fire brigade members should receive hands-on structural fire fighting training at least once a year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus.

The Oyster Creek Fire Protection Training Program requires the fire brigade members to annually attend practice sessions to provide experience in actual fire extinguishment and in the use of emergency breathing apparatus under strenuous conditions. The fires are of a similar type and magnitude as those which can be expected to occur in the plant. Also, annual physical, respirator, SCBA training and SCBA fit qualifications are required to be maintained annually, as specified in the Fire Protection Program, procedure 101.2.

Fire Brigade Drills

a. Fire brigade drills are performed in the plant so that each fire brigade shift can practice as a team.

The Oyster Creek Fire Protection Program, procedure 101.2, requires the fire brigade members to complete an initial training program that includes as a minimum fire drills are performed at least quarterly. These drills are normally performed inside the plant, at numerous locations, with varying classes of fires assumed.

b. Drills should be performed at regular intervals for each shift fire brigade.

The Oyster Creek Fire Protection Program, procedure 101.2, requires fire drills to be performed at least quarterly for each shift fire brigade (1 drill per shift per quarter). Each shift fire brigade member should participate in each drill. To ensure coverage of all individuals, each fire brigade member must participate in at least one fire drill per year. Those individuals who do not participate in at least one fire drill per year are placed on an "inactive (drill required) list" until the drill requirement is met.

c. At least one unannounced fire drill for each shift fire brigade should be performed per year.

The Oyster Creek Fire Protection Training Program requires an unannounced fire drill to be performed at least once per year for each shift fire brigade.

d. At least one drill per year should be performed on a "backshift" for each shift fire brigade.

The Oyster Creek Fire Protection Training Program requires fire drills to be performed at least once per year for each shift fire brigade on a back shift.

e. Drills should be preplanned to establish training objectives and critiqued to determine how well the training objectives have been met.

The fire brigade drills are preplanned with established training objectives and a critique is held after each drill to determine how well the training objectives have been met.

f. At least triennially, an unannounced drill should be performed for and critiqued by qualified individuals, independent of the licensee's staff.

Oyster Creek procedure 101.2, Section 4.10.5 states that an independent triennial audit and inspection of the Oyster Creek Fire Protection Program shall be conducted in accordance with procedure 5000-ADM-7371.01 (EP-14) and Technical Specification 6.5.3.

g. Pre-fire plans should be developed for safety related areas of the plant (as a minimum).

Abnormal operating procedure 2000-ABN-3200.29, Response to Fire, has been developed for all safety related areas in the plant as well as other fire zones. This procedure provides a list of equipment that could be impacted by the fire and guidance for operator response to mitigate the effects of the fire.

h. The pre-fire plans should be updated and used as part of the brigade training.

The Oyster Creek Response to Fire Procedure, 2000-ABN-3200.29, is reviewed and updated biennially, or when plant modifications warrant a change to the procedure.

i. Fire brigade equipment is maintained in good condition and ready for use by the fire brigade.

The fire brigade equipment is checked and verified operable on a weekly basis using Repetitive Task number 00312F. Portable fire extinguisher inspection and surveillance is performed on a monthly basis (Repetitive Task 00306F) to verify that all extinguishers are *in place and are operable.*

Also, Oyster Creek Procedure 120, Fire Hazards, specifies that fire protection equipment is to be used for "Emergency Only." Finally, procedure 120.7, Appendix "R" Emergency Preparedness, specifies that all Appendix R equipment shall be inspected and inventoried after each use and annually.

Fire Brigade Training Records

Records are provided for each fire brigade member, demonstrating the minimum level of training and refresher training has been provided.

Records for each fire brigade member's initial and refresher training and fire drill participation are maintained by the fire protection instructor, as specified by the Fire Protection Program, procedure 101.2. This includes tracking for each fire brigade member's training and drill deficiency status in a report, which is published by the first operating shift of the month. Additionally, the fire protection instructor maintains and publishes, on a quarterly basis, an attendance roster of training conducted, department and personnel failing to attend training and drills.

4.8.4 Total Environment Equipment Survival

The general issue of total environmental equipment survival centers on the following 3 areas of interest:

- Adverse effects of combustion products on plant equipment.
- Spurious or inadvertent fire suppression system actuation.
- Impact on effectiveness of operator actions.

Potential Adverse Effects on Plant Equipment by Combustion Products

a. The FIVE methodology does not currently provide for an evaluation of non-thermal environmental effects of smoke on equipment. See Section 4.2.2 of EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE).

During the screening evaluation of Oyster Creek fire areas, all equipment that was physically located in the impacted area was, as a minimum, assumed to be impacted by the fire. More specific response was modeled during the detailed analysis, however, this is still judged to conservatively bound the impact from non-thermal environmental effects on plant equipment.

Also, many of the effects of smoke on equipment, such as corrosion or degradation due to soot or other products occur over a much longer period than required to establish cold shutdown conditions. These impacts on plant equipment, such as switchgear, would be addressed during the ensuing outage period, as part of corrective maintenance.

b. Plant staff should be aware of and sensitive to the potential impact of smoke and products of combustion on human performance in safe shutdown operations in application of FIVE.

Oyster Creek operations personnel train on the use of SCBA in confined spaces as part of their fire brigade training. Also, operator actions were considered to fail during these screening analyses for fires in the area.

Spurious or Inadvertent Fire Suppression Activation

Verify that the design of fire suppression systems considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on equipment credited for safe shutdown for concerns such as those discussed in NRC I&E Information Notice 83-41.

Spray shields have been installed to protect against the effects of inadvertent suppression system actuation and discharge on important plant equipment. Information Notice 83-41 was reviewed by the onsite Fire Protection Engineer. The identified concerns were evaluated and found to be satisfactorily addressed by the programs in place to verify that water spray from the fire suppression system would not create damage resulting in the loss of all safe shutdown paths. (Reference 38: "Water Impingement from Fire Suppression Systems at Oyster Creek", TDR-864)

Operator Action Effectiveness

a. There are safe shutdown procedures that identify the steps for planned shutdown when necessary, in the event of a fire.

The Oyster Creek Response to Fire Procedure, 2000-ABN-3200.29, instructs the operators on plant control for fires in all fire zones/areas of the plant. The procedure is divided into sections by fire areas and provides guidance on equipment that may be lost due to the fire and lists operator actions to be taken to mitigate consequences of the fire.

b. Operators should receive training on the safe shutdown procedures.

During fire drills, the fire brigade team leader uses the fire mitigation procedure. A training review of fire fighting strategies from the Fire Mitigation Procedure is performed at least every two years by the Oyster Creek licensed operators and shift technical advisors. The fire mitigation procedure and fire fighting preplans are for use as guidance for directing the fire fighting efforts.

c. If, in performance of these procedures, operators are expected to pass through or perform manual actions in areas that may contain fire or smoke suitable SCBA equipment and other protective equipment are available for operators to perform their function.

SCBA equipment is strategically located in the plant to provide a source of air for the *operators. Other protective equipment is located in the Monitoring and Change Room, New Radwaste Building and the Fire Brigade Van.*

4.8.5 Control Systems Interactions

This issue centers on the concern that safe shutdown circuits are physically independent of, or can be isolated from, the control room for a fire in the control room fire area.

The Oyster Creek Remote Shutdown System provides for plant monitoring and control stations to perform a safe shutdown of the plant from outside the control room in the event of circuit destruction caused by a fire in the control room or in the cable spreading room.

The following functions can be performed by the remote shutdown system:

a) *Communications circuits can be isolated from the control and cable spreading room circuits to permit communications in other areas of the plant;*

b) *Power to the essential plant electrical buses can be established and/or verified;*

c) *Reactivity monitoring is available to ensure a continued shutdown condition;*

d) *Heat removal control is available using isolation condensers or shutdown cooling systems;*

e) *RPV pressure and inventory control is available;*

Emergency lighting is provided and maintained in areas needed for operation of safe shutdown equipment and in access and egress routes to the equipment controls. The capabilities of this system are described in detail in Section 1.4 of Volume 2 of the Fire Hazards Analysis Report (Control Room Evacuation). Operation of this system is as detailed in abnormal operating procedure 2000-ABN-3200.30 (Control Room Evacuation) and operating procedure 346 (Operation of the Remote and Local Shutdown Panels).

4.8.6 Improved Analytical Codes

The issue of analytical codes centers on the fire modelling techniques that have been incorporated into the FIVE methodology. These modeling techniques, which are derived from the basic correlations used in the COMPBRN IIIe fire modelling program, have been reviewed for use in the modeling of fire progression.

The detailed fire area analysis did not extend to the point that the EPRI FIVE methodology worksheets were required. These worksheets have been used to document the analysis of fire effects on nearby equipment for several fire areas, the results of these analyses are documented in Section 4.6.4 "Detailed Fire Area Analysis". Otherwise, the EPRI data and methodology were adapted for use from the guidance provided in the FIVE documentation. This use extended from generation of fire event frequency to use of the EPRI fire events database to determine severity factors. The specific fire modeling techniques used for each area are described in Section 4.6.

4.9 USI A-45 and Requirements of NUREG-1407

The resolution of safety issue A-45, regarding the reliability of decay heat removal, and the individual informational requirements of NUREG-1407 are discussed below.

4.9.1 Resolution of USI A-45

The systems used for decay heat removal at the Oyster Creek plant are described in detail in the Level 1 PRA and the Oyster Creek Individual Plant Examination Submittal Report. Essentially, the decay heat removal function is performed in one of three functional paths.

First, and the most preferred decay heat removal path is using the reactor feedwater to maintain reactor vessel inventory and main condenser and turbine bypass systems to remove decay heat. This path is the normal shutdown path at Oyster Creek.

The second decay heat removal path is utilized when the main condenser or turbine bypass system is unavailable. This decay heat removal path uses the isolation condenser system to reject heat to the atmosphere. The isolation condensers are normally maintained in a standby status and removes decay heat removal through condensation of RPV steam following a loss of loss of heat removal through main condenser or the turbine bypass systems. The system consists of two condensers, each capable of removing 45 minutes of decay heat without makeup. Actuation is through operation of a single valve, which is powered by the associated division of 125 VDC power. Shell side inventory to the isolation condensers is supplied by the condensate transfer system or, the less preferred, fire protection system.

The third and least preferred method of decay heat removal is the rejection of heat to the torus (or drywell) and subsequent removal through containment spray/emergency service water systems or the containment hardened vent. This decay heat removal path is used when the main condenser and isolation condenser heat removal paths are unavailable or a loss of reactor coolant exists (i.e., LOCA, stuck open EMRV or stuck open safety valve). In these cases the reactor vessel inventory is maintained by either feedwater, condensate or the core spray system. Decay heat removal is either through the containment spray/emergency service water systems to the discharge canal or through the containment vent system directly to atmosphere.

The components that support the operation of each of these systems are located in numerous areas of the plant, with the main feedwater and turbine bypass system components located predominantly in the Turbine Building. The isolation condenser system and core and containment spray system components are located predominantly in the Reactor Building, with isolation condenser components located on the 95 and 75 foot elevations and core and containment spray components located predominantly on lower elevations.

Due to this separation fire hazard is not a significant concern. This separation continues on a divisional level to include the various support systems, such as essential AC power, required to maintain the decay heat removal function. It should be noted that this separation is maintained for the cable spreading room and the control room through use of the remote shutdown capability. Due to the level of redundancy between and within the various methods for decay heat removal at the Oyster Creek plant and based on the discussions generated during the review of each of the individual fire areas examined in this report, the decay heat removal function is judged to be reliable with regard to hazards resulting from plant fires.

4.9.2 Requirements of NUREG-1407

The analysis described in this report was performed in order to meet the informational requirements of NUREG-1407. In particular, NUREG-1407 specifies the submittal of documentation for the following areas of interest (Appendix C, Section C.3):

1. A description of the methodology and key assumptions used in performing the fire IPEEE and a discussion of the status of Appendix R modifications.

   The methodology used in the Oyster Creek Fire IPE (OCFIPE) consists of a progressive screening form of analysis, based on the EPRI FIVE methodology. The steps performed during this analysis are described in Section 4.0.

   Also, numerous modifications have been made at the Oyster Creek plant in response to the requirements of 10 CFR 50 Appendix R. The majority of these modifications consisted of cable routing changes or additional protection of safe shutdown equipment cables. These modifications have been completed for the Oyster Creek plant.

2. A summary of walkdown findings and a concise description of the walkdown team and the procedures used. This should include a description of the efforts to ensure that cable routing used in the analysis represents as-built information and the treatment of any existing dependence between remote shutdown and control room circuitry.

   In general, site walkdowns confirmed the information contained in the Appendix R documentation, primarily the Fire Hazards Analysis Report and the walkdown team consisted of a GPUN fire protection engineer, GPUN Risk Analysis Engineer and a contractor (Delta Prime Inc.) Risk and Safety Analysis Engineer. As discussed in Section 4.2, walkdowns were performed at several stages of the fire analysis process, including a separate, detailed review of the areas that were not screened from further consideration before detailed fire area analysis.

   Cable routing information was generated from the Fire Hazards Analysis report and engineering review of plant documentation. This information was then confirmed during plant walkdowns. Also, in addition to physical walkdown of plant areas, the interactive plant laserdisk video system was used during several phases of the analysis to confirm component and fire hazard locations. The plant simulator was used to confirm the layout of Control Room components and control panels.

   The remote shutdown capability was only credited for fires in the Cable Spreading Room. The remote shutdown capability was specifically designed to provide an independent control station, including any required control circuitry.

This capability is described in detail in Section 1.4 of Volume 2 of the Oyster Creek Fire Hazards Analysis Report (Alternate Shutdown Capability). Operation of this system is performed as directed in procedures 2000-ABN-3200.30 (Control Room Evacuation) and 346 (Operation of the Remote and Local Shutdown Panels).

3. A discussion of the criteria used to identify critical fire areas and a list of critical areas, including (a) single areas in which equipment failures represent a serious erosion of safety margin, and (b) same as (a), but for double or multiple areas that share common barriers, penetration seals, HVAC ducting, etc.

   Critical fire areas are those that contain either any component modeled in the Level 1 PRA or any support circuitry for these components. Only 13 areas were removed from consideration, based on this criteria. A complete list of Oyster Creek Fire Areas addressed by this analysis is presented in Table 4.1-2.

   Each of the individual fire areas remaining for quantitative analysis was then evaluated, based on a screening cutoff core damage frequency of $1x10^{-6}$ per year, beyond which further quantitative analysis was judged to be unnecessary. These analyses are discussed in Sections 4.6.1 (initial evaluation) and 4.6.2 (revised evaluation) and are summarized in Sections 4.6.2 and 4.6.3. Seven areas remained for detailed fire hazard analysis following these evaluations.

   Each of the remaining areas was then reviewed for remaining conservative assumptions and re-evaluated. Two areas remained for further analysis following this evaluation. This process is described in Section 4.6.3. Each of the fire areas that remained after this analysis are then discussed in Section 4.6.4. These areas are:

OB-FZ-4	*Cable Spreading Room*
OB-FZ-6A	*480 VAC Switchgear Room A*

   Multiple fire area hazards were screened from further consideration based on the fire barrier screening guidelines given in the EPRI FIVE documentation. This process is discussed in detail in Section 4.3.

4. A discussion of the criteria used to determine fire size and duration and the treatment of cross-zone fire spread and associated major assumptions.

   The discussion of the factors that impact fire severity is provided in Section 4.6.3.

   The treatment of fires that could spread across fire boundaries, was based on plant walkdowns, Appendix R documentation and EPRI FIVE guidelines, as discussed in Section 4.3.

5. A discussion of the fire initiating event database, including the plant specific database used. Provide documentation in each case where the plant specific data is less conservative than the data used in the approved fire vulnerability methodologies. Describe methods for handling data, including major assumptions, the role of expert judgement, and the identification and evaluation of sources of data uncertainty.

   Fire frequencies were based on the generic EPRI Fire Events Database, as described in EPRI report NSAC/178L. This database was used to generate the fire frequency guidelines discussed in Section 4.1. A review of the fires that have historically occurred at the Oyster Creek plant found no reason to believe that plant specific data is less conservative (i.e. higher fire frequency) than that generated by this process.

   Due to the use of a progressive screening analysis, data uncertainty was not explicitly modeled. For each of the fires that remained for more detailed analysis, a qualitative discussion of conservative assumptions and potential recovery actions is given in Section 4.6.4. It should be noted that, with the exception of the recovery of at least one train of ESW for fires in the Intake Structure area (CW-FA-14), recovery of damaged equipment is conservatively not modeled in this analysis.

6. A discussion of the treatment of fire growth and spread, the spread of hot gases and smoke, and the analysis of detection and suppression and their associated assumptions, including the treatment of suppression induced damage to equipment.

   For the initial screening analysis, fire growth and spread were implicit in the analysis with the assumption of the "all engulfing fire" in which all equipment and cables were assumed to fail due to the fire. The failure modes included instrument and control cables failures which are conservatively assumed to fail their associated component as well as "hot shorts" which cause the failure of non-active equipment in the active mode. Eighteen (18) of the 35 five areas and zones which are evaluated quantitatively are screened from further consideration. In these cases all equipment is assumed to be failed due to the fire event and no fire suppression is modeled.

   The revised estimation of core damage frequency presented in Section 4.6.2. In this quantitative task the remaining fire areas and zones are evaluated to determine the potential for relaxation of conservative assumptions. In general, this relaxation of conservative assumptions is limited to the modeling of local control of circuit breakers when remote control cables are damaged, ventilation recovery and adjustment of type of initiating event. Ten (10) of the remaining 17 fire areas are screened from further consideration. The fire events are generally assumed to fail all equipment in the area and no fire suppression is modeled.

   The detailed evaluation of core damage frequency is presented in Section 4.6.3. In this quantitative task the remaining seven (7) fire areas are reviewed to remove additional conservative assumptions. This is accomplished through the use of severity and suppression factors. The generation of these factors is discussed in

Section 4.6.3.

In Section 4.3, Fire growth and spread beyond individual fire areas is evaluated by applying EPRI FIVE criteria. Damage due to engulfing fires was assumed for all components in the area.

It should be noted that fire modeling codes are not used in this analysis. Experience with these codes, specifically COMPBRN IIIe, indicates that little additional fire growth information can be obtained for Oyster Creek fire areas. This is primarily due to the communication which exist in the reactor building between floors (equipment hatch) and the turbine building large open areas and connections which prevent the formation of hot gas layers and allow smoke to vent to the highest elevations. Unless components or cables are very near or in the fire plume damage does not occur. During the detailed evaluation of core damage frequency the cases which are analyzed are walkdown to verify equipment and cables which are in the plumes of potential sources. Additional information is documented in Section 4.6.3.

7. A discussion of fire damage modeling, including the definition of fire-induced failures related to fire barriers and control systems and fire induced damage to cabinets. A discussion of how human intervention is treated and how fire induced and non-fire induced failures are combined. Identify recovery actions and types of fire mitigating actions for which credit is taken in these sequences.

Fire-induced failure of fire barriers was evaluated using the screening criteria given in the EPRI FIVE documentation and plant Appendix R documentation, as discussed in Section 4.3.

Fire induced failures of control systems were assumed to fail the impacted system in such a way as to prevent its success within the plant model. For example, hot short failures were assumed when valve control cables passed through a fire area.

For other control systems, such as RPV water level control, failure in such a way to give a high feedwater flow condition is assumed. It should be noted that this eventually results in a failure of the main feedwater function within the plant model. This form of failure is, therefore, judged to be conservative.

Operator failure is assumed for operator actions that would be performed in the affected area. Fire induced and non-fire induced failures were combined by using the Level 1 plant model.

Recovery of plant systems or components that are damaged by fire was not modeled, with the exception of the recovery of at least 1 train of ESW for fires in the Intake area (CW-FA-14).

8. Discuss the treatment of fire detection and suppression, including fire fighting procedures, fire brigade training and adequacy of existing fire brigade equipment and treatment of access routes versus existing barriers.

Fire suppression was not modeled, except during the detailed area analysis, as described in Section 4.6.3. While manual suppression is conservatively not modeled, it is implicitly assumed that fire brigade response will act to minimize the spread of fires. It should be noted that the use of severity factors generated from industry data implicitly includes fire detection data information, as well as the likelihood of self-extinguishing fires.

9. All functional and systemic event trees associated with fire-initiated sequences.

The event tree equivalent for fire severity is described for each fire area in the initial screening analysis described in Sections 4.6.1 and 4.6.2 and in the detailed screening analysis, as described in Section 4.6.3. The plant response event trees are as described in Section 7 of the Oyster Creek PRA Report, incorporating the impacts for each of the various fire areas and the various cases that have been quantified for each.

10. A description of dominant functional and systemic sequences leading to core damage, along with their frequencies and percentage contribution to overall core damage frequency due to fire. Sequence selection criteria are as provided in Generic Letter 88-20 and NUREG-1335. The description of the sequences should include a discussion of specific assumptions and human recovery actions.

Since the IPEEE analysis is based on a progressive screening methodology, the reviewer should be cautioned against arbitrarily adding the final values for various areas. Due to the differing levels of detail required to screen the various areas from further consideration, there will be significant conservative assumptions implicit in some final values, whereas some of these conservative assumptions may have been relaxed for more detailed analysis. Given this introduction, the final upper bound core damage frequencies for areas that were not screened from further consideration during this analysis are shown below (data taken from Table 4.6-6).

A qualitative discussion of the conservative assumptions that remain in these detailed analyses are provided for the various unscreened areas in Section 4.6.4. Human recovery actions from fire induced damage was not modeled, with the singular exception of the recovery of at least one train of ESW following a fire in the intake area (CW-FA-14).

Table 4.9-1 Detailed Fire Evaluation Summary

Fire Area	*Description*	*Upper Bound Core Damage Frequency*	*Comments*
OB-FZ-4	*Cable Spreading Room*	*1.3E-6*	*Unsuppressed fire, assumed to become engulfing*
		4.8E-7	*Fire in or near panel DC-E with suppression*
		8.3E-7	*Fire in area, other than in or near panel DC-E with suppression*
OB-FZ-6A	*480 VAC Switchgear Room A*	*2.7E-6*	*Unsuppressed fire, assumed to become engulfing*
		8.0E-7	*Fire in USS 1A2, suppressed by Halon*
		1.6E-6	*Fire in or near ER-8A, ER-8B, ER-18A or ER-18B, suppressed by Halon*

Fire scenarios were also reviewed against the containment failure criteria given in NUREG-1335. This review is described in Section 4.7 and resulted in no new insights or significant containment bypass scenarios.

11. The estimated core damage frequency, the timing of the associated core damage, a list of analytical assumptions, including their bases, and the sources of uncertainty.

The estimated core damage frequency due to fires is discussed in response to item 10, above. Based on the review of plant damage states performed during the review of containment response described in Section 4.7, no new or significant timing to core damage is introduced by the analysis of internal fires, when compared to the results submitted with the initial IPE report.

Analytical assumptions used for each of the various fire areas are discussed in the associated text. These assumptions are based on those presented in the EPRI FIVE report and are based on the EPRI fire events database. Due to the use of a progressive screening analysis, sources of data uncertainty are not explicitly modeled in a quantitative form. A qualitative discussion of the conservative assumptions remaining for each of the areas that were not screened from further consideration is provided in Section 4.6.4.

12. Any fire induced containment failures identified as being different from those identified in the internal events analysis.

Containment failure during fires was reviewed in Section 4.7. This review concluded that no new containment failures were introduced by the analysis of internal fires. In fact, the percentage of containment bypass events for the fire analysis is smaller than that in the internal events analysis since fire events are unlikely to cause loss of coolant due to pipe ruptures outside containment.

13. Documentation with regard to the decay heat removal function and Fire Risk Scoping Study issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements should be specifically highlighted. Specifically, NUREG-1407 (Section 4) specifies that the submittal should address the following Fire Risk Scoping Study issues:

- Seismic/fire interactions.
- Effect of fire suppressant systems on safety equipment.
- Control system interactions.

The decay heat removal function was reviewed in Section 4.9.1. This review identified no new significant failure mechanisms for decay heat removal. The fire risk scoping study issues were reviewed in detail in Section 4.8, with no new issues identified as a result of this analysis.

14. When an existing PRA is used to address the fire IPEEE, the licensee should describe sensitivity studies related to the use of the initial hazard, supplemental plant walkdown results and subsequent evaluations. The licensee should examine the above list to fill in those items missed in the existing fire PRA.

The plant model was used from the Level 1 PRA. In particular, this model was used specifically to capture the non-fire induced failures that could occur and to model plant response, following the incorporation of fire-induced failures.

4.10 References

1. Electric Power Research Institute, COMPBRN IIIe: An Interactive Computer Code for Fire Risk Analysis, EPRI NP-7282, Final Report, May 1991.

2. Electric Power Research Institute, Fire Induced Vulnerability Evaluation (FIVE), EPRI TR-100370, Final Report, April 1992.

3. Electric Power Research Institute, Fire Events Database for U.S. Nuclear Power Plants, prepared by Science Applications International Corporation, NSAC/178L, June 1992.

4. EQE International, Seismic Fire Interaction Walkdown Report, Report No. P:\42113-08\-OYCEQFIR.DOC, Final Report, August 8, 1995.

5. GPU Nuclear, Oyster Creek Nuclear Generating Station, Fire Hazard Analysis Report, Revision 6, Document Number 990-1746.

6. GPU Nuclear, "Fire Protection Inspection," Oyster Creek Procedure Number 119.1, Revision 13, May 24, 1993.

7. GPU Nuclear, "Fire Hazards," Oyster Creek Procedure Number 120, Revision 16, June 26, 1993.

8. GPU Nuclear, "Cutting, Welding, Miscellaneous Hot Work Administrative Procedure," Oyster Creek Procedure Number 120.1, Revision 13, September 24, 1992.

9. GPU Nuclear, "Fire System Impairment Reporting and Fire Watch Instructions," Oyster Creek Procedure Number 120.2, Revision 9, September 24, 1992.

10. GPU Nuclear, "Portable Heating Equipment and Electrical Cords Administrative Controls," Oyster Creek Procedure Number 120.3, Revision 3, June 26, 1993.

11. GPU Nuclear, "General Response to Fire," Oyster Creek Procedure Number 120.4, Revision 6, April 19, 1992.

12. GPU Nuclear, "Control of Combustibles," Oyster Creek Procedure Number 120.5, Revision 5, May 20, 1993.

13. GPU Nuclear, "Fire Barrier Penetration Data Base Maintenance," Oyster Creek Procedure Number 120.6, Revision 3, May 15, 1993.

14. GPU Nuclear, "Appendix 'R' Emergency Preparedness," Oyster Creek Procedure Number 120.7, Revision 4, June 23, 1991.

15. GPU Nuclear, "Plant Fire Protection System," Oyster Creek Plant Operating Procedure Number 333, Revision 42, August 27, 1993.

16. GPU Nuclear, "Operation of Remote Shutdown Panels," Oyster Creek Plant Operating Procedure Number 346, Revision 6, March 9, 1993.

17. GPU Nuclear, "Response to Fire," Oyster Creek Procedure Number 2000-ABN-3200.29, Revision 13, June 14, 1993.

18. GPU Nuclear, "Control Room Evacuation," Oyster Creek Procedure Number 2000-ABN-3200.30, Revision 10, March 26, 1992.

19. GPU Nuclear, "Oyster Creek Nuclear Generating Station - Fire Hazards Analysis Report," Document No. 990-1746, Revision 7, July 1, 1991.

20. GPU Nuclear, "Oyster Creek Probabilistic Risk Assessment (Level 1)," November 1991.

21. GPU Nuclear, "Oyster Creek Probabilistic Risk Assessment (Level 2)," June 1992.

22. GPU Nuclear, "Oyster Creek Individual Plant Examination Submittal Report," June 1992.

23. GPU Nuclear, "Fire Protection," Oyster Creek Operations Plant Manual, Module 19, Revision 0, March 1987.

24. GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 0'-0" and 3'-6"," GPU Drawing Number 3E-151-02-001, Revision 3, December 4, 1989.

25. GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 6'-0", 0'-0" and 3'-6"," GPU Drawing Number 3E-151-02-002, Revision 4, July 26, 1991.

26. GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-151-02-003, Revision 4, July 26, 1991.

27. GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-151-02-004, Revision 4, August 1, 1991.

28. GPU Nuclear, "General Arrangement - Turbine Building, Plan Floor Elevation 46'-6"," GPU Drawing Number 3E-151-02-006, Revision 4, August 22, 1991.

29. GPU Nuclear, "General Arrangement Reactor Building, Plan Floor Elevation -19'-6"," GPU Drawing Number 3E-153-02-001, Revision 3, December 4, 1989.

30. GPU Nuclear, "General Arrangement - Reactor Building, Plan Floor Elevation 23'-6"," GPU Drawing Number 3E-153-02-002, Revision 4, July 26, 1991.

31. GPU Nuclear, "General Arrangement - Reactor Building, Plan Floor Elevation 51'-3"," GPU Drawing Number 3E-153-02-003, Revision 4, July 26, 1991.

32. GPU Nuclear, "General Arrangement - Office Building, Plan Floor Elevation 23'-6" and 35'," GPU Drawing Number 3E-156-02-001, Revision 3, December 4, 1989.

33. GPU Nuclear, "General Arrangement - Office Building, Plan Floor Elevation 46'-6" and Roof Plan," GPU Drawing Number 3E-156-02-002, Revision 3, December 4, 1989.

34. GPU Nuclear Memorandum 5160-94-007, from J.M. Manoleas to K.T. Canavan, "Cable Tracing for PRA Components," February 22, 1994.

35. GPU Nuclear Memorandum, Parsippany, FSD-85-296, from G. Capodanno to W. Smith, Oyster Creek Non-Functional Fire Dampers, December 18, 1985.

36. GPU Nuclear Memorandum, Oyster Creek, 2200-88-5603B, from J. Ventosa to G. Bush, Response to LAI 88021.01, IEN 88-04, Inadequate Qualification and Documentation of Fire Barrier Penetration Seals, June 10, 1988.

37. GPU Nuclear Memorandum, MCC, 5360-90-115, from M. Calbureanu to M. Heller, OCNGS LAI 89124.01, IE 89-52, Potential Fire Damper Operational Problems, April 9, 1990.

38. GPU Nuclear, Technical Document Report, "Water Impingement from Fire Suppression Systems at Oyster Creek", TDR No. 864, Revision 0, August 17, 1987.

39. Ho, Vincent Simon, PhD., COMBRN IIIe: An Interactive Computer Code for Fire Risk Analysis, University of California, Los Angeles, 1990.

40. Nuclear Regulatory Commission, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Prepared by Sandia National Labs, NUREG/CR-5088, January, 1989.

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

42. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54 (f)," Generic Letter No. 88-20, November 23, 1988.

43. Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Final Report, August 1989.

44. Nuclear Regulatory Commission, Information Notice No. 83-69, Improperly Installed Fire Dampers at Nuclear Power Plants, October 21, 1983.

45. Nuclear Regulatory Commission, Information Notice No. 83-41, Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment, June 22, 1983.

46. Nuclear Regulatory Commission, Information Notice No. 88-04, Inadequate Qualification and Documentation of Fire Barrier Penetration Seals, February 5, 1988.

47. Nuclear Regulatory Commission, Information Notice No. 88-04, Supplement 1: Inadequate Qualification and Documentation of Fire Barrier Penetration Seals, August 9, 1988.

48. Nuclear Regulatory Commission, Information Notice No. 89-52, Potential Fire Damper Operational Problems, June 8, 1989.

49. Nuclear Regulatory Commission, Information Notice No. 94-12, Insights Gained from Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment, February 9, 1994.

50. PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual I: Data Analysis, Volume 1, 1994.

51. PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual II: Systems Analysis, Volume 2, 1994.

52. PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual III: Event Trees, Volume 3, 1994.

54. PLG, Incorporated, PRA Workstation Software, RISKMAN User Manual IV: Important Sequences, Volume 4, 1994.

55. PLG, Incorporated, "Database for Probabilistic Assessment of Light Water Nuclear Plants: Fire Data", PLG-0500, Volume 8, October 1990.

APPENDIX A

ACRONYMS AND GLOSSARY

All Engulfing Fire	The all engulfing fire or the design basis fire are terms used to describe the conservative assumptions used in the initial and revised quantitative screening analysis. These assumptions are used to bound the worst case fire and models the failure of all components and all supporting cables (power, instrumentation and control) within a fire area or zone. The result of "all engulfing fire" is the most conservative fire model impacts possible.
Area	This term is a generic term used throughout the Oyster Creek Fire Individual Plant Examination to refer to either an Oyster Creek fire zone or fire area as described in the FHAR.
Area Geometry	(1) Actual fire area or zone geometry. (2) One of the fire propagation factors used in the quantitative evaluation of fire barrier effectiveness. This factor considers geometry such as potential for hot gas layer formation and horizontal and vertical separation of targets and sources.
CDF	Core Damage Frequency.
COMPBRN IIIe	An interactive fire modeling computer code used in fire risk analysis (an EPRI product).
Electrical Conduit	A rigid or flexible tubing usually steel or aluminum in which electrical cables are run.
EMRV	Electro-Matic relief valve. These relief valves provide the post-trip pressure relief required following various plant trips. The number of valves required to open (top event VO) and subsequently reclose (top event VR) is defined by the type of plant trip (initiating event definition).
EPRI	Electric Power Research Institute.
FHAR	Fire Hazard Analysis Report.
Fire Area	A plant area bounded by construction which will contain a fire to that area without reliance on automatic or manual fire suppression (Oyster Creek FHAR, page 2-2). That is, an area which is bounded by a rated barrier.

Fire Brigade	The team of plant personnel assigned to fire fighting and who are trained in the fighting of fires by an established training program.
Fire Detector	A device designed to automatically detect the presence of fire and initiate an alarm.
Fire Detection	The discovery of fire and notification of appropriate personnel either automatically (see fire detector) or manually by plant personnel.
FIVE	Five Induced Vulnerability Evaluation. An EPRI product which presents a method of fire vulnerability determination designed to respond to the requirements of the fire individual plant examination portion of Generic Letter 88-20 Supplement 4.
Fire Modeling	The deterministic modeling of the growth and propagation of fires usually using a computer code. This modeling technique provides information on component and room heatup as a result of the fire due to hot gas layer formation, conductive and radiant heat transfer.
Fire Modeling Computer Codes	Fire modeling computer codes are those codes which are used in the deterministic modeling of fire growth and propagation. The computer code referred to in this analysis is an EPRI distributed product: COMPBRN IIIe.
Fire Propagation Factors	Fire propagation factors are multiples of the fire initiating event frequency which are used in the quantitative evaluation of fire propagation beyond individual fire areas or zones. These factors include fire source strength, area geometry, fire detection and suppression. These factors are designed to probabilistically evaluate the potential for multiple area fires.
Fire Risk Model	The OCPRA Level 1 plant model is adjusted to account for the impacts of fire events for each fire area and zone at Oyster Creek. Fire initiating event frequencies are added to the Level 1 model. The result of this process is the fire risk model.
Fire Severity Factor	The fire severity factor is a multiple of the fire initiating event frequency designed to adjust the original fire frequency to address the fact that many of the fires events used in the calculation did not fail actual plant equipment or did not cause plant damage.

Fire Source Strength	The fire source strength is one of the fire propagation factors used in the quantitative evaluation of fire barrier effectiveness (Section 4.3.2). This factor addresses the fact that fire areas or zones with lower combustible loads are less likely to be able to support severe fires which can damage other equipment than areas with high combustible loadings. This factor is used to adjust the initiating event frequency.
Fire Suppression	The capability for control and/or extinguishment of fires (fire fighting). Manual fire suppression activities refer to the use of standpipe and hose, portable extinguishers, or actuation of manual systems. Automatic fire suppression refers to fixed systems such as water sprinklers, halon or carbon dioxide.
Fire Zone	A subdivision of a fire are in which the fire suppression systems are designed to combat a particular type of fire. The concept of fire zone aids in defining to the fire fighter the fire parameters and actions which would be necessary to take. A fire zone does not have rated fire barriers.
"General" Transient Initiator	The general transient initiator is assumed in the initial quantification of the fire risk model. This assumption does not specify the type of plant trip (i.e., initiating event) and therefore requires 4 of 5 EMRVs to reclose. This assumption is removed in the revised and detailed quantification analysis, if required.
Hot Short	Is a cable failure mode due to a fire which results in the active failure of a component. That is, a change of state of a component without an authentic signal. For example, a normally closed motor or air operated valve transferring to the open position as a result of a cables short which provides the necessary signal or power.
Initiating Event	The event which begins the sequence of events and results in a demand for a reactor scram. For this analysis, initiating events are fires which occur in the plant and causes damage to equipment or cables which results in a demand for a reactor scram.
Initiating Event Impact Table	The initiating event impact table (Table 4.2-2) provides the impacts of fire events in each fire zones. This table provides the link between the components and cables damaged in a fire area (given a fire) and the fire risk model.

IPE	Individual Plant Examination. Generic Letter 88-20 requests that all holders of nuclear power plant operating licenses perform an investigation for potential for plant specific vulnerabilities.
IPEEE	Individual Plant Examination for External Event Vulnerabilities. Generic Letter 88-20, supplement 4, requests all holders of nuclear power plant operating licenses to perform an investigation for plant specific vulnerabilities due to external events.
Multiple Area Fire	A fire event which begins in an individual fire area or zone and propagates to additional fire areas or zones through failure of suppression or rated (non-rated) barriers.
NRC	Nuclear Regulatory Commission
OCFIPE	Oyster Creek Fire Individual Plant Examination.
OCNGS	Oyster Creek Nuclear Generating Station.
OCPRA	Oyster Creek Probabilistic Risk Assessment.
Plant Area	This term is used to refer to either a plant fire area or fire zone.
Plant Model	The plant model refers to the Oyster Creek Probabilistic Risk Assessment Level 1 plant model which is documented in Section 7 of that reference.
Screened from Further Consideration	Plant fire areas or zones whose total core damage frequency is less than $1x10^{-6}$ per year are not considered risk significant. This cutoff is based on the cutoff frequency for area consideration presented in the EPRI FIVE methodology. In the iterative approach used in this analysis, area or zones whose CDF falls below the cutoff, in any iteration, are not considered in future iterations and terms "screened from further consideration".
Split Fraction	The value used to represent the independent failure probability for a given system (i.e., top event) under a specific set of conditions (i.e., available support systems).

Surrogate Tour	An interactive laserdisk video tour station. This station was used for pre-walkdowns and walkdowns of high contamination and high radiation areas.
Supporting Cables	Electrical power, instrumentation or control cables which provide support to the components modeled in the fire risk model.
System Analysis	OCPRA Level 1 qualitative and quantitative analysis which provide the documentation of the plant model and fire risk model top events and split fractions.
Top Event	The representation of a given system response within the OCPRA Level 1 plant model or the fire risk model. Top events quantified under various boundary conditions (i.e., supporting system availabilities) within the OCPRA Level 1 system analyses to produce split factions.
Upper Bound Core Damage Frequency	The core damage frequency is the result of the quantification of the fire risk model. When the term "upper bound" appears as a prefix, this indicates that conservative assumptions in the analysis (e.g., "all engulfing fire") result in a calculated core damage frequency which provides the maximum possible value. Additional modeling of detection and suppression or the relaxation of other conservative assumptions will result in lower total core damage frequency values.
UBCDF	Upper Bound Core Damage Frequency.
USI	Unresolved Safety Issue.

APPENDIX B

FIRE RISK MODEL SUCCESS CRITERIA

This appendix presents the Fire Risk Model success criteria, the Fire Hazard Analysis Report (FHAR) safe shutdown paths and comparison of the two. The Fire Risk Model success criteria are the same as those used in the OCPRA Level 1 for general transients (Section 8 of that reference) and are summarized below.

B.1 Fire Risk Model Success Criteria

In overview, the successful mitigation of core damage, is asccomplished by decay heat removal through one of the following paths:

- Decay Heat Removal Through the Main Condenser. This is the normal path for decay heat removal and normal shutdown. Use of this decay heat removal path requires that MSIVs are open and that the main condenser and its support systems are available.

- Decay Heat Removal Through the Isolation Condenser. This decay heat removal path is utilized following reactor isolation transients where either the main condenser is unavailable or the MSIVs are closed. This path requires successful initiation of 1 of 2 isolation condensers and successful long term shell side makeup from either the condensate transfer system or the fire protection water system. In this path, decay heat is discharge to the atmosphere via boil-off of the shell side inventory.

- Decay Heat Removal Through Containment Spray/Emergency Service Water. Should the isolation condensers or their support be unavailable, core decay heat is discharged into the containment through the operation of EMRVs or safety valves. The decay heat is removed by the containment spray/emergency service water system to the intake canal.

- Decay Heat Removal Through the Hardened Vent. This decay heat removal path utilizes the hardened vent system following the failure of the containment spray/emergency service water system when decay heat is being rejected to the containment. Decay heat is discharged to the atmosphere via the hardened vent piping and the plant stack. Again, this condition is applied when RPV makeup is available through high or low pressure sources and decay heat removal is through the hardened wetwell vent.

The above overview of success criteria is expressed in the Fire Risk Model as the following detailed logic rule:

```
SUCCESS    RLC*(SHSD+SIC+SATWS)
           +DWHR*(CC=S+OV=S+RC=S+RV=S+SD=S)
           +SLOFC
           +SRVF
```

The Fire Individual Plant Examination uses the OCPRA Level 1 success criteria for general transient linked model. Additional details on this success criteria as well as the assignment of plant damage states are available in Section 8 of the OCPRA Level 1 report. Stated in narrative form, the SUCCESS rule above represents six primary success paths. These are:

1. **Successful Hot Shutdown**. Successful reactor level control (RLC) and (*) hot shutdown with the main condenser and feedwater injection (SHSD). This path is the normal method of cooldown. Interim variable SHSD requires successful turbine and reactor trip, turbine bypass actuation and control, feedwater control during the trip, mode switch to shutdown and long term feedwater availability.

2. **Successful Isolation Condenser Shutdown**. Successful reactor level control (RLC) and successful shutdown with the isolation condenser (SIC). Interim variable SIC is defined as successful hot shutdown using the isolation condenser for decay heat removal. This variable requires: A) the success of top event RS ensures that this variable is not applied to ATWS conditions. B) The success of top events IC and MU to ensure the initiation and shell side make (long term operability) to the isolation condenser. C) Top events for VO and SO ensure adequate early steam relief capability with top events VS, VO and VR ensuring the scram discharge volume and all relief and safety valves are close providing reactor vessel isolation. MSIV closure or successful turbine bypass preclude continued loss of RPV inventory through the main steam system. No reactor vessel makeup is necessary since the reactor vessel is isolated. Varible RLC ensure that RPV level is controlled and ICs do not need to be isolated on high level.

3. **Successful ATWS Mitigation**. Successful reactor level control (RLC) and (*) successful anticipated transient without scram mitigation (SATWS). This success path is not discussed here due to its complex nature. Details on the successful mitigation of ATWS events is contained in Section 8.0 of the OCPRA.

4. **Successful Containment Decay Heat Removal**. Reactor decay heat discharge to the containment with successful containment heat removal via the containment spray system or the containment venting system and successful RPV inventory control. Decay heat is deposited to the containment via the EMRVs or safety valves. This decay heat is successfully removed from the containment via the containment spray system or the containment hardened vent.

5. **Successful Mitigation of Reactor High Level Excursion**. Successful mitigation of extended high RPV water level excursion (SLOFC). This success path relquires the termination of feedwater injection and core cooling provided with the automatic depressurization system and the core spray system.

6. **Successful Mitigation of Relief Valve Failure.** Reactor decay heat discharge to the containment through induced loss of coolant accident following the failure of all relief valves to lift with successful containment heat removal via the containment spray system or the containment venting system and adequate RPV inventory control (SRVF).

B.2 Fire Hazard Analysis Shutdown Paths

10CFR50 Appendix R requires fire protection features to be "capable of limiting fire damage so that a) one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free from fire damage, and b) systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station (s) can be repaired within 72 hours." The success criteria used in the Fire Hazards Analysis Report are summarized in Appendix A, Section 4.0 of that report. This summary is reproduced in its entirety for reference below.

4.0 *SHUTDOWN PATH SUMMARIES*

Figures A-1 and A-2 summarize the systems and components used to achieve and maintain hot and cold shutdown, respectively. Five paths are included for hot shutdown and three paths are included for cold shutdown based on the following:

Hot Shutdown Path	*Decay Heat Removal Component*	*Electrical Power Train*
#1	*Isolation Condenser "B" (Note 1)*	*"B"*
#2	*Isolation Condenser "A" (Note 1)*	*"A"*
#3	*EMRVs*	*"B"*
#4	*EMRVs*	*"A"*
#5	*Isolation Condenser "B"*	*"A"*

Note 1: *For certain fire zones as described on Figure A-1, Isolation Condenser "A" is used for Hot Shutdown Path #1 and Isolation Condenser "B" is used for Hot Shutdown Path #2.*

Cold Shutdown Path		
#1	*Shutdown Cooling Pump "B"*	*"B"*
#2	*Shutdown Cooling Pump "A"*	*"A"*
#3	*EMRVs*	*"B"*

The systems and components summarized for each hot and cold shutdown path are as described in Sections 2.0 and 3.0, respectively.

The applicable fire areas/zones for each hot and cold shutdown path are shown below the system and component summaries on Figure A-1 and A-2. An "x" next to a Fire Area/Zone on the figures indicates that the hot or cold shutdown path can be used in the event of an Appendix "R" fire in that area/zone. The path used for hot shutdown is not necessarily the same path used to achieve cold shutdown for a fire in that area. However, combinations of the fire hot shutdown paths and three cold shutdown paths provide shutdown capability in accordance with 10CFR50, Appendix R for a fire in any fire area/zone.

Hot and cold shutdown Path #1 can be used to provide alternate shutdown capability in *the event of a fire in the control room, cable spreading rooms, cable bridge tunnels or the mechanical equipment room (fire zones OB-FZ-4 and 5 and OB-FZ-22A, 22B, 22C and 22D), that results in a control room evacuation, or the loss of control and indication functions in the control room. Prior to evacuating the control room the operators will scram the reactor, close the main steam isolation valves, and trip the reactor feedwater and reactor recirculation pumps. Control would then be from the RSP located in the "B" 480V switchgear room, fire zone OB-FZ-6B, from LSPs located outside the control and cable spreading rooms, and at various equipment. Refer to Appendix D of this report for further description of this alternate shutdown capability.*

For plant shutdown in the event of a fire outside the control room, cable spreading rooms, cable bridge tunnels or mechanical equipment room, hot shutdown Path #1, #2, #3, #4 or #5 and cold shutdown Path #1, #2 or #3 can be used depending on the location of the fire as summarized on Figures A-1 and A-2 for hot and cold shutdown, respectively. Backup systems or components are used or repairs or manual operations are required to achieve certain functions for a fire in certain areas as indicated by notes referenced in the fire area/zone vs. shutdown path matrices of Figures A-1 and A-2."

The basic philosophy of the Fire Hazard Analysis Report is therefore, to maintain at least one hot shutdown path and one cold shutdown path free from fire damage. Free from fire damage requires that given a fire area or zone either A) no components or support cables are contained in the area from the credited shutdown paths or B) the cables or components which are within the fire area or zone are protected from fire damage.

B.3 Comparison of Fire Risk Model Success Criteria and FHAR Shutdown Paths

The difference between the success criteria used in the Fire Risk Model and the Fire Hazards Analysis Report (FHAR) is a philosophical in nature. The Fire Risk Model success criteria specifies those components and support cables whose independent failure or failure due to fire events results in the inability of that specific system component to mitigate the transient. The FHAR specifies one of five hot shutdown paths and one of 3 cold shutdown paths which remain free of damage following a fire event in a given area.

The FHAR success criteria are more restrictive than those used in the Level 1 model. This is due to four primary reasons:

1. The FHAR does not take credit for safe shutdown due to any other methods than those proposed in the five hot and three cold shutdown paths. The Fire Risk Model allow additional shutdown paths such as shutdown with decay heat removal through the main condenser or the hardened vent. The main condenser with feedwater and turbine bypass is the normal and preferred cooldown path. In the OCPRA as well as the Fire Risk Model a significant number of scenarios use this success path.

2. The FHAR requires both a hot shutdown and a cold shutdown success path for each area. The OCPRA and the Fire Risk Model does not require a cold shutdown path.

3. The Fire Risk Model success paths are based on component and system success and failures. The FHAR success paths are based on the entire path remaining unaffected by the fire event. For example, assume ventilation for switchgear required for event mitigation is affected by a fire event. However, room heatup calculations indicate that serveral hours are available before the switchgear exceeds design temperatures. Operator action to provide temporary ventilation may be modeled.

4. The OCPRA and the Fire Risk Model (in the revised and detailed analysis) model fire supression both automatic and manual. In addition, factors which are related to the probability of fire growth and propagation are also addressed.

In overview, the Fire Risk Model specifies success criteria at a lower level (component and system) than the FHAR (entire success path). This results in the Fire Risk Model having more potential pathway available for fire event mitigation given a fire in any given fire area. Details on the Fire Risk Model success criteria are available in the Section 8 of the Level 1 OCPRA.

APPENDIX C

INDEPENDENT REVIEW

The purpose of the independent review of the fire portion of the Oyster Creek IPEEE was to ensure that the analysis was reasonably accurate and reflects the design and operation of the Oyster Creek plant. The review was performed in a largely collegial process wherein in-house Risk Analysis personnel made presentations on the content of the study, talking through the various portions of the fire evaluation process. Review group members reviewed these portions before the presentations and offered comments orally in a group setting. Many comments during these meetings elicited further comments from other members of the group, resulting in significant amounts of discussion. The recorded comments from these meetings (see below) represent the consensus of the group on a particular issue. The formal portion of the review process consisted of a series of 3 meetings that were held in January and February 1995. These meetings were held at the Oyster Creek site.

The following personnel were in attendance for each of these meetings:

Fred Barbieri	- Mechanical Components, Engineer	
Ray Daley	- Fire Protection, Engineer	
Tim Trettel	- Fire Protection, Engineer	
Lou Lanese	- Risk Analysis, Manager (Acting)	(January 20 and 27 only)
Ken Canavan	- Risk Analysis, Engineer	

Group comments were developed during the review meetings and represent the consensus of the group. These comments are collected and presented by meeting date below.

D.1 Independent Review Group Meeting of January 20, 1995

1. **Comment** Report organization is confusing and does not seem to follow the analysis flow process.

   Resolution *The report structure and text were chosen to reflect the suggested format of NUREG-1407. The assignment of individual report sections in this suggested format was reviewed against the flow process and modified to provide a more organized flow and report structure.*

2. **Comment** The report does not present any sensitivity studies. Are any planned? This is of particular importance since Halon fire protection systems may be removed or replaced in the future.

   Resolution *Due to the nature of a screening evaluation, individual sensitivity studies are not appropriate since the fire areas and zones are treated various levels of detail. No sensitivity studies are presently planned. In the future, sensitivity evaluations on the effects of Halon removal could be performed.*

3. **Comment** The fire zone which contains the diesel generator fuel oil tank is not presented separately.

Resolution *The diesel generator fuel oil tank is analyzed (FS-FA-16) separately. This fire area screens in the initial upper bound estimate of core damage frequency.*

4. **Comment** The operability of the fire protection system is no longer a Technical Specification requirement. It is now a station technical requirement. Does this affect the screening of the fire pond pump house?

Are there any other considerations, given the fact that station operation may continue for 7 days without operable fire protection?

Resolution *Plant trip would not be expected to occur in response to a fire in the fire pond pump house, regardless of whether system operation is a Technical Specification or station technical requirement. Therefore, this concern does not impact the screening of the fire pond pump house. While station operation may continue without operable fire protection, this form of system maintenance is accounted for in the unavailability of plant sprinklers and water suppression systems.*

5. **Comment** Table 4.1-5, Fire Protection Panels Contribution to Fire Area Frequency, indicates that a fire protection panel is inside No. 2 Diesel Generator Room. This panel is located on the outside wall of the Diesel Generator Building and should not be included n the development of a fire initiating event frequency.

Resolution *The fire ignition frequency for this panel was reassigned to area FS-FA-17.*

6. **Comment** Table 4.1-5, Fire Protection Panels Contribution to Fire Area Frequency, indicates that a fire protection panel is located in fire zone RB-FZ-1E. Verify that this panel is located in RB-FZ-1D and adjust the table.

Resolution *Panel location was confirmed and Table 4.1-5 has been modified accordingly.*

7. **Comment** Table 4.1-6, Transformer Contributions to Fire Area Frequency, seems to include the contributions of dry transformers, as well as oil cooled units. Should this be the case?

Resolution *The EPRI fire events database used to generate the fire ignition frequencies for the FIVE methodology does not distinguish between dry or oil cooled transformers. Therefore, the Oyster Creek fire IPEEE does not distinguish* *between these types of components.*

8. **Comment** Table 4.1-7, Non-Qualified Cable and Non Qualified and Qualified Junction Box Contribution to Fire Area Frequency, differentiates between qualified cable and non-qualified cable at Oyster Creek. Please describe in detail the rationale for differentiating and the method used to determine the qualification of cable at Oyster Creek.

Resolution *Types of cable were initially differentiated based on interpretation of the combustible loading assignments for the various Oyster Creek fire zones. This differentiation has since been removed and Table 4.1-7 has been modified accordingly.*

D.2 Independent Review Group Meeting of January 27, 1995

9. **Comment** Table 4.1-7, Non-Qualified Cable and Non-Qualified and Qualified Junction Box Contribution to Fire Area Frequency, reports combustible loading with too many significant figures, which indicates a precision which does not exist. Consider rounding off.

Resolution *Entries were rounded to the nearest 1,000 BTU.*

10. **Comment** The "Hydrogen Tanks" discussion on page 4.1-12 indicates that the main generator hydrogen tanks are located in the yard area, MT-FA-12. This is not the case. These tanks are located outside the protected area and should be removed from MT-FA-12 fire frequency calculation.

Resolution *The fire ignition frequency contribution due to hydrogen tanks was removed for MT-FA-12.*

11. **Comment** The "Other Hydrogen Fires" discussion on page 4.1-12 indicates the fire frequency is allocated to fire zone OB-FZ-10B. No industrial gases are located in OB-FA-10B and the most probable location of other hydrogen fires is either the main generator area or the feedwater hydrogen injection manifold.

Resolution *The fire ignition frequency due to "Other Hydrogen Fires" is reassigned to feedwater hydrogen injection manifold, as described above.*

12. **Comment** The "Air Compressors" discussion on page 4.1-13 indicates only four air compressors on the Oyster Creek site. There are many compressors of

various sizes located on the site. Verify the number of compressors.

Resolution *The current air compressor fire ignition frequency was assigned to the indicated units based on size and regular operation. The EPRI FIVE Fire Ignition Frequency Methodology does not discriminate between the air compressor size and service. Judgement was used in the assignment of fire ignition frequency to the indicated compressors.*

13. **Comment** Table 4.1-7, Ventilation Subsystem Fire Frequency Contribution, fire frequency allocations are incorrect. There are no normally running ventilation fans located in the Reactor Building or in fire zone TB-FZ-11B. In addition, the fans associated with fire zone OB-FZ-8A/B are located outside. Fans associated with the 4160 VAC Switchgear rooms are not identified.

Resolution *In fact, a case can be made for the removal of all ventilation fans from the fire initiating event frequency calculation. Judgement is used in the assignment of the frequency based a review of the EPRI Five Events Database. The contributions of all fans except the Reactor Building corner room fans remain.*

14. **Comment** The "Laundry Dryers" discussion on page 4.1-13 indicates the presence of laundry dryers in fire zone OB-FZ-10B. These dryers have been moved to the new radwaste building and a shed located outside the new maintenance building.

Resolution *Fire ignition frequency for OB-FZ-10B was adjusted as described. The other areas were removed from consideration based on qualitative factors.*

15. **Comment** Table 4.1-8, Transient Combustible Contribution to Fire Area Frequency, does not agree with the current control of combustibles procedure.

Resolution *Table 4.1-8, Transient Combustible Contribution to Fire Area Frequency, was revised to allow up to a pint of acetone (10,000 BTU) to be in any plant area as a transient combustible.*

16. **Comment** Table 4.2-3, Impact Matrix Notes, note 1, does not indicate that these cables are associated with the diesel fire pump manual start.

Resolution *The impact matrix notes was revised as described.*

D.3 Independent Review Group Meeting of January 31, 1995

17. Comment Table 4.2-2, Fire Initiating Event Impact Table, should Isolation Condensers (top event IC) be impacted in fire zone TB-FZ-11C? In fire zone TB-FZ-11D, should 125 VDC distribution bus B, USS 1B1, USS 1A1, 4160 VAC 1C and 1D be impacted?

Resolution *Table 4.2-2, Fire Initiating Event Impact Table, was revised to incorporate the Isolation Condenser "B" (Top Event IC) isolation valve control cables impact as a result of a fire in fire zone TB-FZ-11C.*

Table 4.2-2, Fire Initiating Event Impact Table, was revised to incorporate the impacts of fire events in the fire zone TB-FZ-11D. In addition, the impact matrix was reviewed for all other plant fire areas and zones to verify the impacts in the fire risk model and adjusted accordingly.

18. Comment Table 4.6-1, Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation, fire zone OB-FZ-6A does not list the ADS impact which appears in Table 4.2-2.

Resolution *The EMRV failure mode modeled in Table 4.6-1, Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation, is failure to open to relieve pressure, which subsumes the failure to open for ADS actuation. That is, the EMRVs are already failed as a result of failure to open to relieve pressure.*

19. Comment It is not clear which fire zones have been screened from further consideration (and why) when reviewing Section 4.6. Consider moving Table 4.4-3 to the introduction of Section 4.6.

Resolution *The text of Section 4.6 has been revised to provide additional clarity on the why and at what point fire areas and zones have been screened from further consideration. Additional text has also been added to Section 4.0 which provides an overview of the Fire Individual Plant Examination methodology.*

20. Comment On page 4.6-6, fire zone RB-FZ-1F does not contain any automatic suppression. Fire suppression in this fire zone is all manual.

Resolution *The text has been revised to reflect manual fire suppression.*

21. Comment Review and revise assumptions in the detailed evaluation of fire zones TB-FZ-11C and TB-FZ-11F associated with the 125 VDC cables located in conduits.

Resolution *The assumptions regarding the protection provided by conduits have been reviewed. The text and the evaluation have been revised to eliminate these assumptions.*

22. **Comment** The assumption that a fire in fire area MT-FZ-12 results in a loss of offsite power may be overly conservative. Consider revising initiator impact to reactor trip event with loss of offsite power following failure of transformers.

Resolution *It is acknowledged that the screening methodology used in FIVE and in this analysis is conservative. This evaluation is based on the premise that, if an area can be screened from consideration under this level of conservative assumptions, no further evaluation is warranted and resources can then be directed to the evaluation of other, unscreened areas. The fire zone MT-FZ-12 screens from further consideration in the revised estimate of upper bound core damage frequency.*

23. **Comment** In Table 4.6-2, Revised Estimation of Upper Bound Core Damage Frequency, the revised CDF for fire area CW-FA-14 should be $5.57x10^{-7}$.

Resolution *Table 4.6-2, Revised Estimation of Upper Bound Core Damage Frequency, was revised for all fire areas due to modification of fire ignition frequencies based on previous comments in this report section.*

24. **Comment** For the detailed evaluation of fire zone OB-FZ-5, Control Room, consider manual suppression of fires instead of automatic actuation of fire suppression systems. This is the more likely scenario since the control room is continuously manned.

Resolution *Again, the screening evaluation used in FIVE is intended to provide a conservative evaluation of fire-related core damage. The intention was to show that even without operator response, in a continuously manned area, this fire zone still screens from further consideration. If manual suppression was modeled it is likely that the same results would be achieved. Due to the uncertain future of Halon as a fire suppressant this evaluation will most likely be performed.*

25. **Comment** On page 4.6-28, "C" Battery Room, reword the description to indicate that a carbon dioxide fire extinguisher is located outside the fire area for manual fire suppression.

Resolution *The report text was revised as noted.*

26. **Comment** Fire area TB-FA-26 ("C" Battery Room), does not screen in the detailed quantitative evaluation. This does not seem consistent with the screening of the A/B battery room as well as the other area which did screen in the detailed evaluation. Review the assumptions with regard to the application of fire severity factor in this case.

Resolution *The evaluation of TB-FA-26 ("C" Battery Room) was revised. The area now screens from further consideration. It should be noted that the initial evaluation gave an "unscreened" value that was very close to the screening value of* $1x10^{-7}$.

27. **Comment** Report Section 4.2.3, Quantitative Evaluation of Fire Barrier Effectiveness, is very subjective and, as such, adds little new information to the analysis. In the absence of less subjective inputs, consider the removal of the entire report section.

Resolution *Section 4.2.3 has been removed from the evaluation.*

SECTION 5.1

HIGH WINDS

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

5.1.1 Introduction and Summary

Winds can affect critical structures at the plant site in at least two ways. If wind forces exceed the load capacity of a building or other external facility, the incident walls or framing might collapse or the structure overturn from the excessive loading. If the wind is strong enough, such as in a tornado, it might be capable of lifting materials and thrusting them as missiles against some of these critical facilities. Critical components or other contents of facilities not designed to resist missile penetration might be damaged and lose their function. This section presents an analysis of the risk to the Oyster Creek Nuclear Generating Station from tornado winds and missiles. Hurricanes are also considered to be bounded by this analysis since wind speeds associated with hurricanes are typically much lower than those associated with tornados and as you approach the upper bound region of tornado wind speeds, probability of hurricane wind speeds exceeding such thresholds becomes extremely low. It is concluded that neither a tornado wind load nor a potential missile generated in a tornado event leads to scenarios that exceed the NUREG-1407 screening criteria to require further analysis.

5.1.2 Tornado Wind Hazard and Frequency

This section is based on the NRC sponsored report entitled "Tornado and Straight Wind Hazard Probability for Oyster Creek Nuclear Power Reactor Site, New Jersey" (Reference 1) as presented in Oyster Creek Updated Final Safety Analysis Report, Appendix 2.3A.

5.1.2.1 Methodology

In basic terms, the probability, P, of a tornado striking a point is given by the relation (Reference 2):

$$P = \begin{array}{l}\text{Number of tornados per year}\\ \text{in a geographical region}\end{array} \times \frac{\text{Tornado path area}}{\text{Total area of geographical region}}$$

To compute the probability of a tornado striking a point, the geographical region must be defined first. For Oyster Creek, obviously, the geographic region should include the Oyster Creek site. The geographic region should also be as large as possible and still give a reasonably homogeneous condition for tornado formation, i.e., the conditions for tornado formation anywhere in the region should remain fairly representative of the Oyster Creek site.

A region that satisfies the above criteria is a rectangular area bounded by latitude 38°N to the south, latitude 42°N to the north, longitude 73°W to the east, and longitude 77°W to the west. The geographical region is shown in Figure 1.

To compute the probability of a tornado striking a point within this region, the number of tornados per year and the tornado path area must be known. These parameters are discussed in the next two sections.

5.1.2.2 Tornado Frequency

Figure 2 illustrates the historical trend of tornado frequency for New Jersey and also for an area within 125 nautical miles of the Oyster Creek Nuclear Plant. This data is presented for illustrative purposes only. Similar data for the rectangular geographic region described above was not readily available. Nevertheless, note that the annual number of tornados for the eight-year period beginning in 1973 is much higher than for the previous 23 years. Apparently, a campaign in the early 1970's to encourage tornado reporting and documentation by the National Weather Service, Red Cross, and civil defense officials accounts for most of the increase (Reference 3). Natural climatic variability is not a factor in the increase.

To compute the probability of a tornado striking a point within the rectangular region, the number of tornados per year must be known. Figure 3 shows the annual average number of tornados that have occurred during the 29-year period 1950-1978. The annual average number of tornados is shown as a function of tornado intensity. (See Table 1 for a description of tornado intensities). As can be seen in the figure, slightly over six tornados annually have occurred in the 29-year period. F1 tornados were most frequent, occurring three to four times per year. All tornados reported had an intensity of F3 or less, i.e., wind speeds of 206 mph or less.

A statistical analysis of the data in Figure 3 gives the number of tornados expected to occur during any given year. This is the number that will be inserted in the above relation (Section 5.1.2.1). The number of tornados expected to occur during any given year are shown in Figure 4A. The upper limit of the number of tornados expected during any year are shown in Figure 4B.

As can be seen from Figure 4A, between six and seven tornados can be expected to occur each year. In any given year, it is unlikely there will be more than eight tornados (Figure 4B). The tornado most likely to occur will have winds between 73 and 112 mph and will occur three to four times per year (Figure 4A). A tornado with winds between 261 and 318 mph is expected to occur once every 100 years (Figure 4A), but not more than every 20 years (Figure 4B).

5.1.2.3 Tornado Path Area

The last parameter needed to compute the probability of a tornado striking a point is the tornado path area. The tornado path area is the damage path area (path length times path width) of the tornado. Figure 5 shows the mean damage path area, as a function of tornado intensity, of all tornados that occurred during the nine-year period 1971-1978. (Accurate damage path area statistics were not generally available prior to 1971).

As Figure 5 illustrates, the mean damage path area of the most intense tornado reported (F3 intensity) is smaller than the mean damage path area of the less intense F2 tornado. This may be due more to differences in sample size than in the physics of tornados. For example, the sample size of F3 tornados is only 1/4 the sample size of the F2 tornados.

The bias of sample size is minimized in Figure 6, which is a statistical analysis of the data in Figure 5. Figure 6 shows the expected damage path area as a function of tornado intensity. Not surprisingly, the expected damage path area increases with tornado intensity.

5.1.2.4 Tornado Hazard Probabilities

The tornado hazard probabilities are computed by substituting both the expected damage path area (as a function of tornado intensity) and the expected number of tornados per year (as a function of intensity) into the relation in Section 5.1.2.1. The tornado hazard probabilities computed in this way are shown in the second column of Table 2. The first column of Table 2 shows the mean recurrence interval, which is the inverse of the tornado hazard probability.

It is obvious from Table 2 that the hazard probability decreases as the tornado intensity increases. Also note that the recurrence interval for the weakest tornado is 10,000 years, which is five times longer than the recurrence interval calculated using the method and data of Reference 6.

It should be emphasized that the probability of observing a specific tornado wind speed in a span equal to its recurrence interval is not one. Rather, the probability is 0.63, and this value is approached asymptotically as the recurrence interval approaches infinity (Reference 7).

The tornado hazard probabilities in Table 2 are graphed in Figure 7. For purposes of comparison, the straight wind hazard probabilities are also shown. Note that for wind speeds less than 110 mph, the hazard probability for straight winds is much greater than the hazard probability for tornado winds. Thus, for wind speeds up to 100 mph, the hazard is not likely to be from a tornado. For wind speeds over 100 mph, however, the hazard is likely to be from a tornado.

A summary of wind speed hazard probabilities for Oyster Creek is presented in Table 3. The table summarizes what has just been mentioned. That is, for wind speeds less than 110 mph, straight winds are the hazard. For wind speeds greater than 110 mph, tornado winds are the hazard.

5.1.3 Tornado Wind Fragility of Structures

10 CFR Part 50 (GDC 2), as implemented by SRP Sections 3.3.1 and 3.3.2 and Regulatory Guides 1.76 and 1.117, requires that new plants be designed to withstand the effects of natural phenomena such as wind and tornados.

The effects of tornados were not considered in the original design of the Oyster Creek structural systems.

In Integrated Plant Safety Assessment Report (IPSAR), Sections 4.3.1 through 4.3.9, the staff identified some structures and components important to safety that did not meet current licensing criteria, which require that they be adequate to resist tornado winds of 250 miles per hour and a differential pressure of 1.5 pounds per square inch (Reference 9). The following were identified in the IPSAR as not meeting the prescribed criteria (Reference 9):

(1) reactor building steel structure above the operating floor
(2) ventilation stack

(3) effects of failure of non-qualified structures
(4) components not enclosed in qualified structures
(5) Exterior masonry walls
(6) roof decks
(7) intake structure, oil tanks, and diesel generator building
(8) load combinations
(9) soil and foundation capacities

However, in IPSAR Sections 4.3.4, 4.3.5, and 4.3.9, the staff concluded that further evaluation of items (4), (5), and (9) was not warranted (Reference 9).

GPUN responded to the remaining issues in submittals dated February 2 and October 25, 1983, and February 2, March 13, and June 4, 1984.

On the basis of a letter from the staff to GPUN dated March 8, 1986 (Reference 8), which provided an evaluation of the responses, in IPSAR Supplement 1, Sections 2.3.2, 2.3.3, and 2.3.4, the staff reported that items (2), (3), and (6) were resolved. Item (7) has been resolved on the following basis: The licensee proposed that safe shutdown could be achieved and maintained in case of loss of intake structure, oil tanks, and diesel generator building with makeup water provided to the isolation condenser from the torus by the main core spray pumps. The staff reviewed this proposal and concluded that although the flow path itself is acceptable, the core spray pumps rely on the emergency diesel generators for motive power.

In its safety evaluation dated February 26, 1990, the staff concluded that the walls of the diesel generator vaults and the oil tank compartment are capable of withstanding the loads generated by a tornado having a wind speed of 168 miles per hour and are acceptable. However, the staff required that GPUN provide adequate protection to the outside fuel supply line against the potential missile strike (Reference 9).

In letters dated April 16 and July 27, 1990, GPUN committed to install a safety-grade check valve and a safety-grade gate valve in the supply line inside the emergency diesel generator fuel tank room. The installation of these valves is intended to prevent the fuel oil supply from backflowing out of the 15,000-gallon diesel generator fuel storage tank (day tank) in the event of a rupture of the fuel supply line outside the fuel storage tank room.

The staff reviewed the licensee's proposed changes to the diesel generator supply line and the proposed modification to protect the day tank fuel supply to the diesel generators and found them acceptable in its safety evaluation dated November 28, 1990 (Reference 9).

In IPSAR Supplement 1, Sections 2.3.7 and 2.3.8, the staff identified the following two additional items related to tornado missile damage on the basis of the letter of March 8, 1986: (10) control room and (11) architectural components, respectively.

GPUN addressed items (1), (8), (10), and (11) in letters dated November 15, 1990 and October 3, 1991. In the letters, GPUN described a planned upgrading of the upper reactor building structure by adding cross braces to the roof framing and provided justifications for other items of concern.

NRC issued letters dated July 29, 1992 and December 7, 1992 (References 10 and 11) in which the staff concluded that the upper steel-framed portion of the Reactor Building will be able to withstand the tornado-wind loading generated by a total wind speed (i.e., rotational plus translational) of up to 306 KM/h (190 mi/h), based on the planned modifications to the steel framing completed during Cycle 14, 1993 - 1994 time frame (Reference 12). NRC also considered all other tornado missile related issues resolved by above letters on the basis of having a system consisting of the isolation condenser (IC), torus, and a core spray pump available for both shutting down the plant and maintaining it in a shutdown mode. More discussion of this shutdown path is available in Section 5.1.6 titled: "Tornado Missile Fragility of Structures."

In summary, all the structures and components important to safety are capable of withstanding at least a tornado of a wind speed of 168 miles per hour after all the planned modifications are completed.

This is the wind speed that the staff has concluded that the most vulnerable structure, the walls of the diesel generator vaults and the oil tank compartment, can withstand and the other structures can take much higher wind speeds.

5.1.4 Tornado Wind-Initiated Scenarios

Even conservatively assuming an expected wind speed of 168 miles per hour being the vulnerability threshold for all the structures important to safety at Oyster Creek, based on Figure 7, the probability of exceeding such wind speeds is about 5×10^{-7}. Of course, not all the structures at the plant are expected to be damaged at such tornado wind speeds and the conditional core damage frequency is obviously less than such occurrence probability of 5×10^{-7}.

5.1.5 Tornado Missile Hazard and Frequency

Tornado missile analysis involves information about the likelihood of a spectrum of available missiles in the plant vicinity, representation of the wind field in the tornado, and aerodynamic behavior relative to "liftoff" and flight of the potential missile. The analysis leads to a spectrum of missiles and missile impact velocities with their respective probabilities. A detailed analysis that integrated all these effects for typical plant layouts has previously been performed (Reference 13). The results of that work are considered to be reasonable gross estimates for the hazard of tornado missiles at Oyster Creek Nuclear Generating Station.

In Reference 13, calculations were made using tornado histories of each tornado region defined by the NRC (Ref. 14). The analysis used a typical two-unit plant layout to establish the target envelope and a 26-missile spectrum, which includes the six missiles defined in the NRC Standard Review Plan, Section 3.5.1.4 (wood plank, steel pipe, steel rod, utility pole, and automobile). In general, the 26-missile spectrum of Reference 13 is more conservative than the Standard Review Plan spectrum with respect to damage potential. Calculations were made for several cases including a two-unit plant. Assuming 1,000 available missiles during the operating phase, the study obtains the following upper and lower bounds (at 95% confidence level) for the annual impact and damage frequency for all structures of one unit of a two-unit plant in NRC Region 1:

Upper Bound = 8.63×10^{-7}

Lower Bound = 6.64×10^{-9}

The thickness of the targets considered in this calculation ranges from 12 to 18 inches for targets such as the diesel generator building and service water intake structure, and from 24 to 36 inches for the containment. Outdoor storage tanks are also considered to be enclosed by 12-inch thick concrete walls.

The above impact/damage frequency bounds were calculated on the basis of a tornado strike frequency of 2.3×10^{-3} per year, per square mile, while the strike frequency at Oyster Creek Nuclear Station is 1.11×10^{-4} (total of all different tornado intensities from Table 2) per year, per square mile. Adjusting for this factor results in the following bounds for a two-unit plant:

Upper Bound = 4.16×10^{-8} per year, per square mile

Lower Bound = 3.20×10^{-10} per year, per square mile

In this analysis, the above values are used as the 5th and 95th percentiles of a lognormal distribution (a somewhat conservative use of the confidence bounds) resulting in a mean value of 1.09×10^{-8}.

This mean annual strike/damage frequency of 1.09×10^{-8} is for all the structures at this plant. One could simplistically define missile strike/damage frequency for any structure by multiplying this total mean value by the ratio of the surface area of the target structure over the surface area of all the structures at the plant. Of course, the total mean annual strike/damage frequency for the whole plant would remain the same but each structure's contribution would be better defined.

One should be cautious about the fact that all outdoor tanks in the example case study (Reference 13) were enclosed, whereas at Oyster Creek some outdoor tanks are not. For this reason, if one only looks up tornado missile hit frequency with no conditional enclosure damage criteria attached, the frequency obtained would be more representative for the outdoor tanks at Oyster Creek (even though conservative) than the method described above for other structures. Reference 13 provides a value of 4.92×10^{-7} for such frequency. Obviously, this value dominates the total mean annual strike/damage frequency of 1.09×10^{-8} for the whole plant.

5.1.6 Tornado Missile Fragility of Structures

10CFR (GDC2), as implemented by Regulatory Guide 1.117, prescribes structures, systems, and components that should be designed to withstand the effects of a tornado, including tornado missiles, without loss of capability to perform their safety functions. Regulatory Guide 1.117 requires that structures, systems, and components that should be protected from the effects of a design-basis tornado are (1) those necessary to ensure the integrity of the reactor coolant pressure boundary, (2) those necessary to ensure the capability to shut down the reactor and maintain it in a safe shutdown conditions (including both hot standby and cold shutdown), and (3) those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25% of the guideline exposures of 10 CFR 100 using appropriately conservative analytical methods and assumptions. The physical separation of redundant or alternate structures or components required for the safe shutdown of the plant is not considered acceptable by itself for providing protection against the effects of tornados, including tornado-generated missiles, because of the large number and random direction of potential missiles that could result from a tornado as well as the need to consider the single-failure criterion (Reference 15).

The following structures and components were found vulnerable to tornado missiles at Oyster Creek according to NRC staff IPSAR (Reference 15):

(1) emergency diesel generators and fuel oil day tank
(2) mechanical equipment access area
(3) control room, reactor building, and turbine building heating, ventilating, and air conditioning systems
(4) condensate storage tank
(5) torus water storage tank
(6) service water and emergency service water pumps

Item (3) was considered resolved by the staff on the following basis (Reference 15):

Since the intake for the control room HVAC system is located in the reactor building wall and is not protected from tornado missiles, GPUN has committed to install a remote shutdown capability as part of his Appendix R modification; therefore, the plant would be able to shut down in the event the control room is lost.

The staff has found this acceptable because GPUN has committed that the location of the remote shutdown capability systems and all supporting systems would be protected from the effects of tornado missiles.

As far as the Reactor Building and Turbine Building HVAC is concerned, GPUN has concluded that the plant can be safely shutdown with a loss of HVAC. The demonstration of that ability has been evaluated in conjunction with SECY 82-207, "The Final Rule for Environmental Qualification of Safety Related Electrical Equipment for Nuclear Power Plants."

Items 2, 4, 5 and 6 were found acceptable by the staff (Reference 9 and 11) based on the following available alternate shutdown path (Reference 16):

"GPUN indicated in its August 14, 1987 submittal that a detailed field walkdown identified an existing system interconnection between core spray and the isolation condenser using the suppression chamber (torus) water as the protected water supply. This water supply and the interconnection are located in the Reactor Building, below 119' -3" elevation, where they are protected from tornado missiles or potential external flooding.

During a tornado event at Oyster Creek, decay heat from the reactor will be removed by the isolation condenser where the boiloff from the shell side will be released to the atmosphere. The licensee proposed to supply the necessary makeup, due to the boiloff, by using the system interconnection and the torus as water supply. Credit for torus water is given during a tornado missile event for emergency fill, because a design basis accident, plant transient, or failure of core spray system isolation valves are not postulated to occur coincidentally with a tornado missile event. The fill flow will be pumped by the Core Spray System I main pump from the torus to the isolation condenser through the core spray system fill line. The licensee calculated that by adding 10,000 ft^3 (differential volume between maximum and minimum torus water level) of torus water to the isolation condenser shell along with the amount of water initially in the isolation condenser shell, the system is capable of removing decay heat for about 13 hours which provides sufficient time to ensure an emergency reactor shutdown. Also, the licensee indicated that the inventory water in the isolation condenser could remove decay heat for the initial 1.5 hours or until the fill path is established.

This emergency fill path involves manual operation of the following valves: valves V-11-110, V-11-111, and V-11-346 change position from close to open; valve V-11-15 changes from open to close. The licensee indicated in their August 12, 1988 submittal that these valves are readily accessible for manual operation during either a tornado event or an external flooding event. The operator would take these actions as soon as the procedure for the event is initiated. Because the inventory water in the isolation condenser could remove decay heat for 1.5 hours, the staff concludes that the operator would have sufficient time to manually operate these valves for the emergency fill path.

The licensee indicated that the Core Spray Pump NPSH is approximately 39' available at the minimum torus water level during the transient and that the required NPSH is 12' at 2600 gpm. Therefore, the required NPSH for the Core Spray Pump is about 1/3 of the NPSH available. This ensures the pump sufficient NPSH if torus water level drops to the end of the downcomer.

The licensee performed an analysis which determined that the system flow rate using the system interconnection will provide sufficient decay heat removal for a safe reactor shutdown. The licensee also performed a radioactivity dose calculation for the small amount of radioactive cesium in the torus water. In a tornado missile event, the radioactivity release would be within permissible levels of radiation to an unrestricted area according to 10 CFR 20.105. The calculated dose is only a fraction of the 10 CFR 20.105 limit.

GPUN indicated that the revision schedule for the Iso-Condenser-Diagnostic and Restoration Actions, procedure 2000-OPS-3024.18 is October 30, 1988. This procedure is intended to provide specific guidance to the operators to identify the needed actions to initiate emergency fill path. The training of the operators to follow this procedure and the implementation of this procedure will be completed within a few days after NRC's approval of this proposal."

Based on above, the staff concluded:

"Based on our review, the staff concludes that the proposed emergency fill path is protected from tornado missiles or external flooding and that the reactor emergency safe shutdown pathway during a tornado missile event is ensured for the following reasons:

a. The core spray pump using the system interconnection will have a system flow rate to provide sufficient heat removal, and the system is capable of removing decay heat for 13 hours.

b. The manual operated valves in the emergency fill flow path are readily accessible during either a tornado missile event or an external flooding event.

c. The core spray pump has sufficient NPSH to operate should the torus water level drop to downcomer submergence.

d. The small amount of radioactive cesium in the torus water released to the environment through the isolation condenser shell side boiloff is only a fraction of the 10 CFR 20.105 limit.

e. The licensee's revision schedule for the Iso-Condenser-Diagnostic and Restoration Action procedure is October 30, 1988.

Therefore, the staff concludes that the proposed emergency fill path is acceptable for a tornado missile event or an external flooding event with the loss of all external water sources. The viability of this water supply is dependent on diesel generators for motive power. Protection for the diesel generators including missile protection is being reviewed under a related issue (see Safety Evaluation dated February 26, 1990)."

As discussed in Section 5.1.3, the staff has accepted the changes proposed by GPUN to protect the outside fuel supply line to the diesel generators against potential missile strike (Reference 9) and the issue with the diesel generators protection is considered resolved.

One issue that surfaced during the review of the proposed alternate shutdown path was protection of the controls for this system in the control room, which are located adjacent to the north wall.

The staff, in evaluating the susceptibility of the control room (CR) to damage by tornados, found that the north wall of the CR would be damaged if struck directly by a utility pole of the type and size described in SRP Section 3.5.1.4, "Missiles Generated by Natural Phenomena," during a tornado strike with a wind velocity of 168 MPH. Such a strike could cause spalling of the inner portions of the wall which could, in turn, damage the controls for the proposed shutdown system; there is no potential for missile penetration (Reference 10). However, the staff concluded that (Reference 10): "the utility pole would have to hit the wall almost perpendicularly and in the specific locale of the controls in order to preclude use of the shutdown system. GPUN has an established procedure for shutting down using normal shutdown methods in the event of a tornado watch or tornado warning, thereby reaching shutdown before a tornado could strike the plant. GPUN also has an established procedure for shutting the plant down, employing the system described above (IC, torus and core spray pump) as needed.

In view of the foregoing, the staff considers that GPUN complies with the intent of the criterion of SRP Section 2.2.3 and also, therefore, with the intent of SRP Section 3.5.1.4, "Missiles

Generated by Natural Phenomena," with regard to safe shutdown in the event of tornado missiles."

5.1.7 Tornado Missile-Initiated Scenarios

The low frequency of a tornado missile hitting various critical structures (see Section 5.1.5), combined with the fact that damage to the Class 1 structures would certainly be localized and not enough to impact several vital components, leads to extremely low frequencies of all tornado missile-initiated accident scenarios that can be hypothesized.

Even if one assumes very conservatively that loss of outdoor tanks (i.e., condensate and torus water storage tanks) due to tornado missiles at Oyster Creek leads to core damage, the frequency of such event, which dominates other scenarios, is less than 4.92×10^{-7} based on the discussions in Section 5.1.5. Of course, the alternate shutdown path discussed in previous section 5.1.6 will not be impacted by such scenario and is being conservatively ignored totally for such postulated scenario.

5.1.8 Conclusions

Even if we assume every tornado missile hit causing inside wall scabbing of the concrete critical structures or damage to the outdoor tanks will lead to a guaranteed core damage scenario, the total core damage frequency will be less than 4.92×10^{-7} as discussed in Sections 5.1.5 and 5.1.7. Combining this with the frequency obtained in Section 5.1.4 for the postulated tornado wind load generated scenario of 5×10^{-7}, the total is below the screening criterion specified in NUREG 1407 (1×10^{-6}) to require any further analysis.

FIGURE 1. THE GEOGRAPHICAL REGION FOR WHICH TORNADO HAZARD PROBABILITIES WERE CALCULATED (Geographical region is within dashed lines)

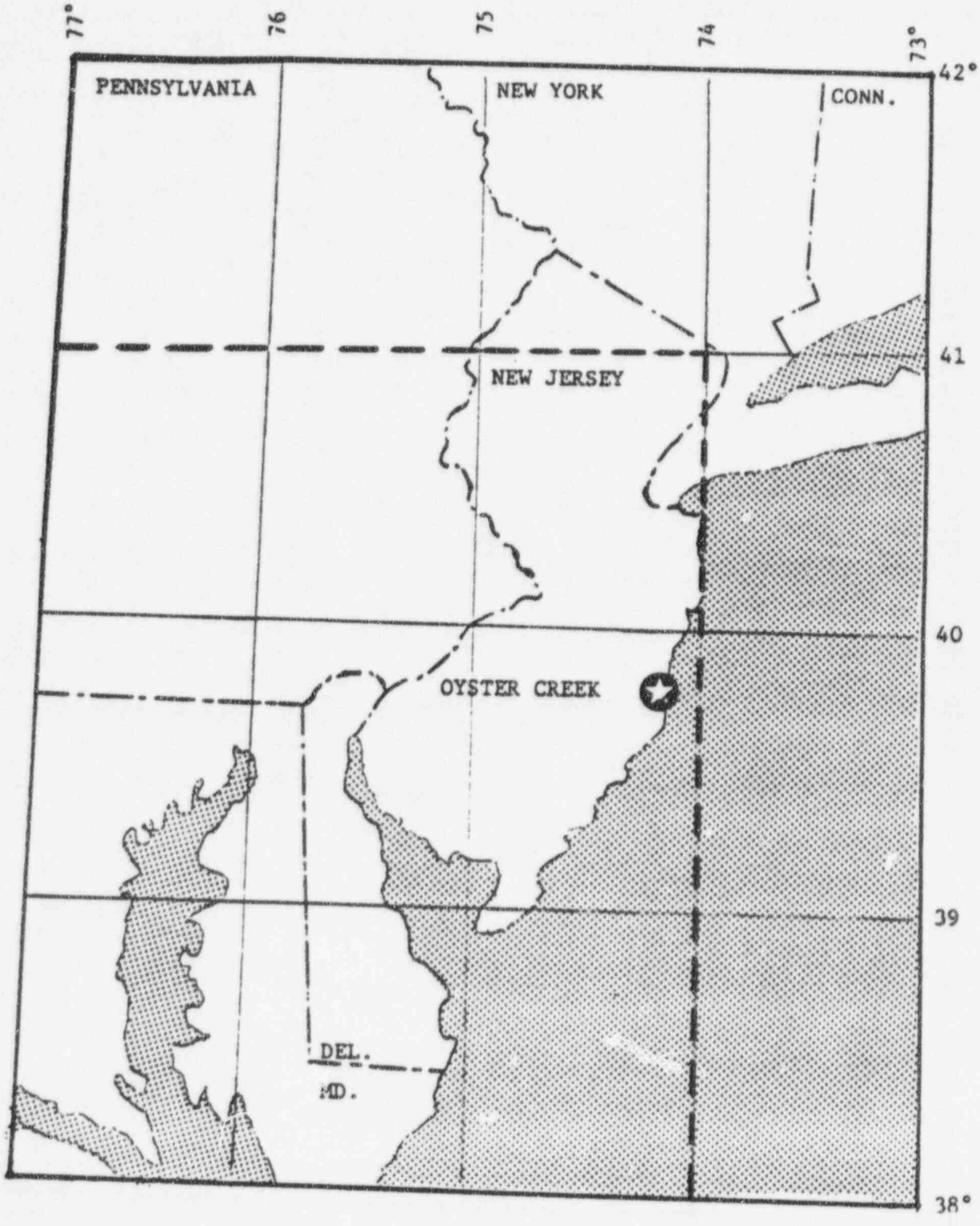


FIGURE 2: ANNUAL COUNT OF TORNADOES
WITHIN 125 NAUTICAL MILES OF THE OYSTER CREEK NUCLEAR PLANT

(TORNADO DATA FROM THE NATIONAL SEVERE STORMS FORECAST CENTER)

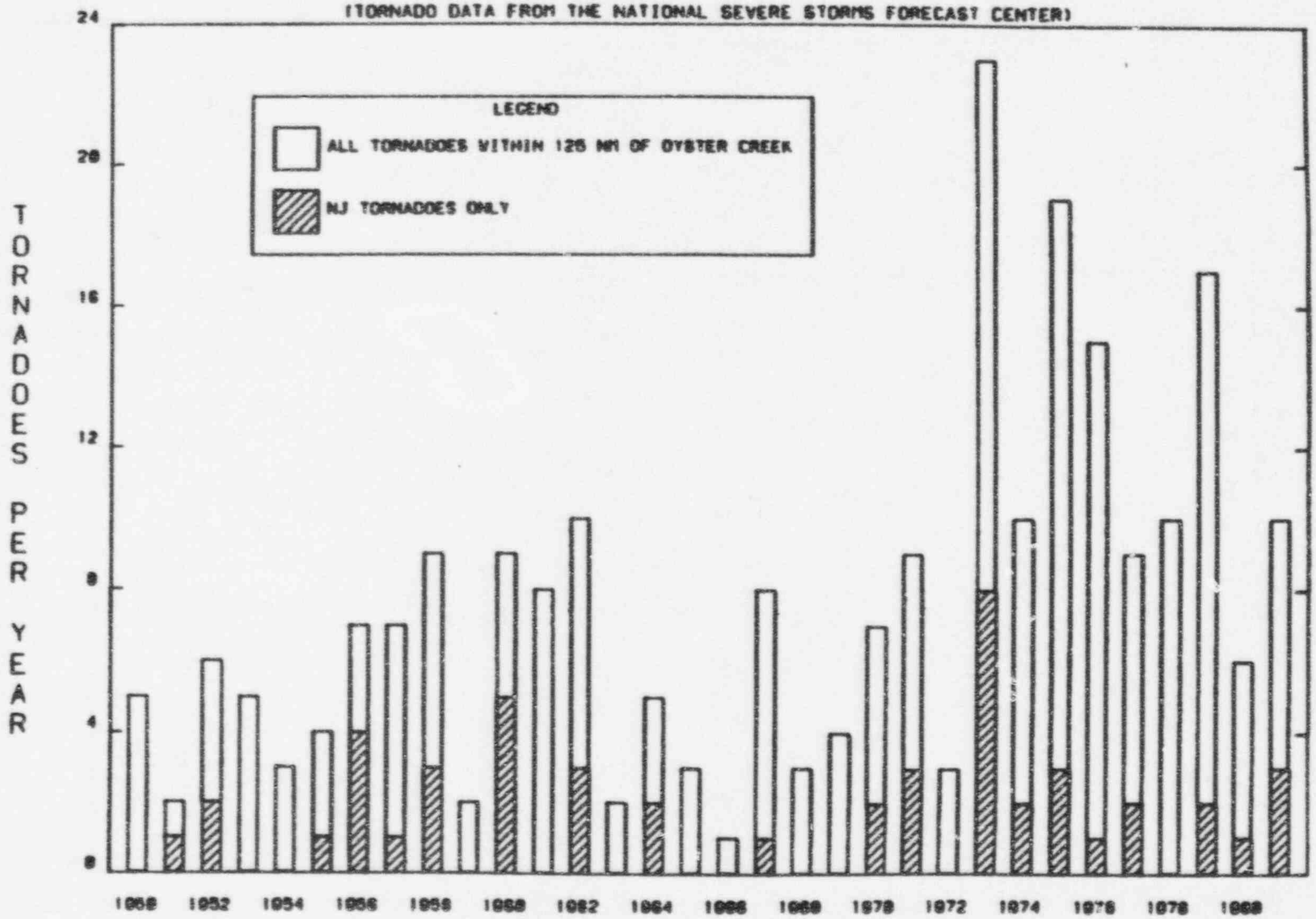


FIGURE 3: AVERAGE NUMBER OF TORNADOES PER YEAR AS A FUNCTION OF TORNADO INTENSITY

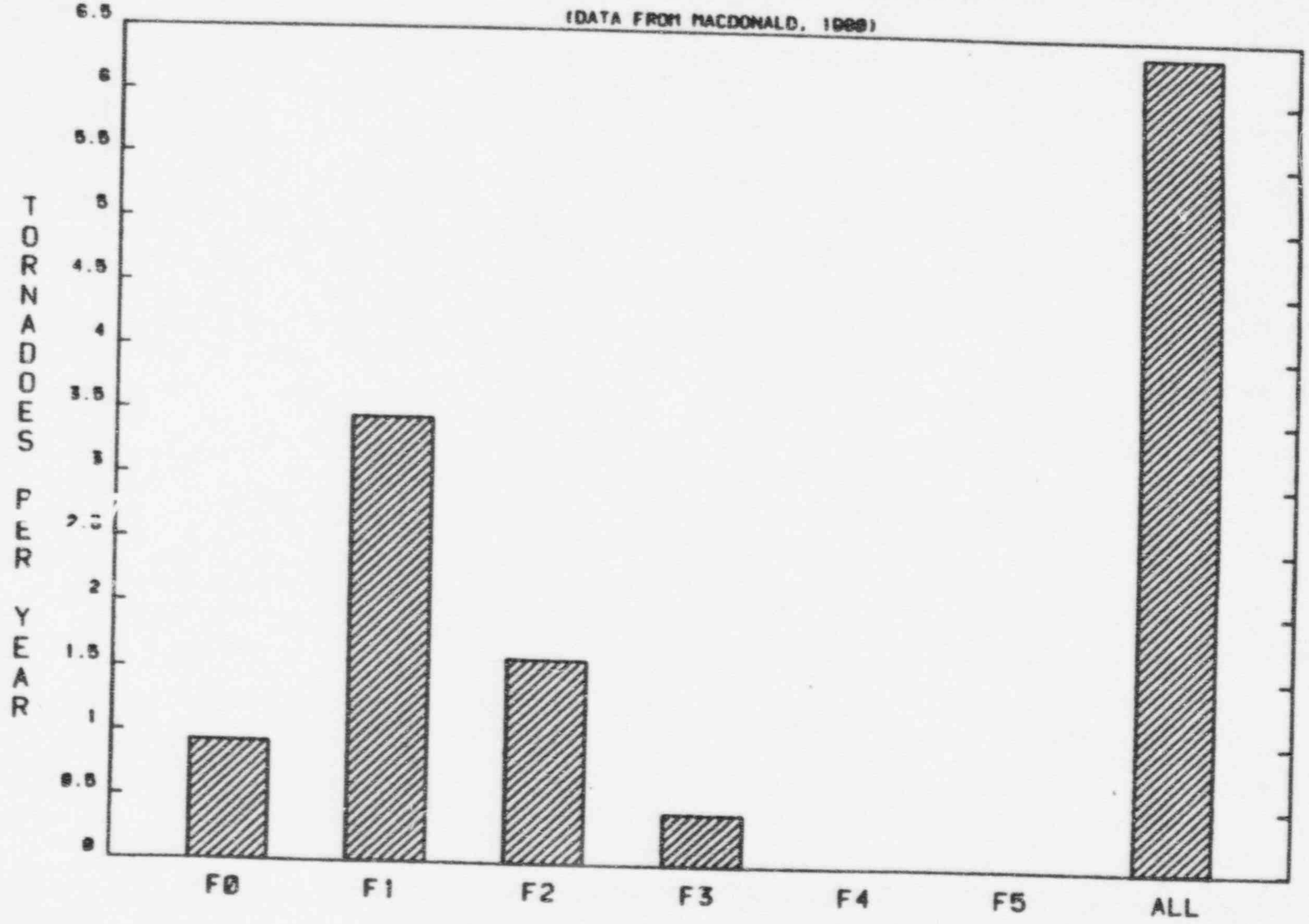


FIGURE 4A: EXPECTED NUMBER OF TORNADOES PER YEAR
AS A FUNCTION OF TORNADO INTENSITY

(DATA FROM MACDONALD, 1988)

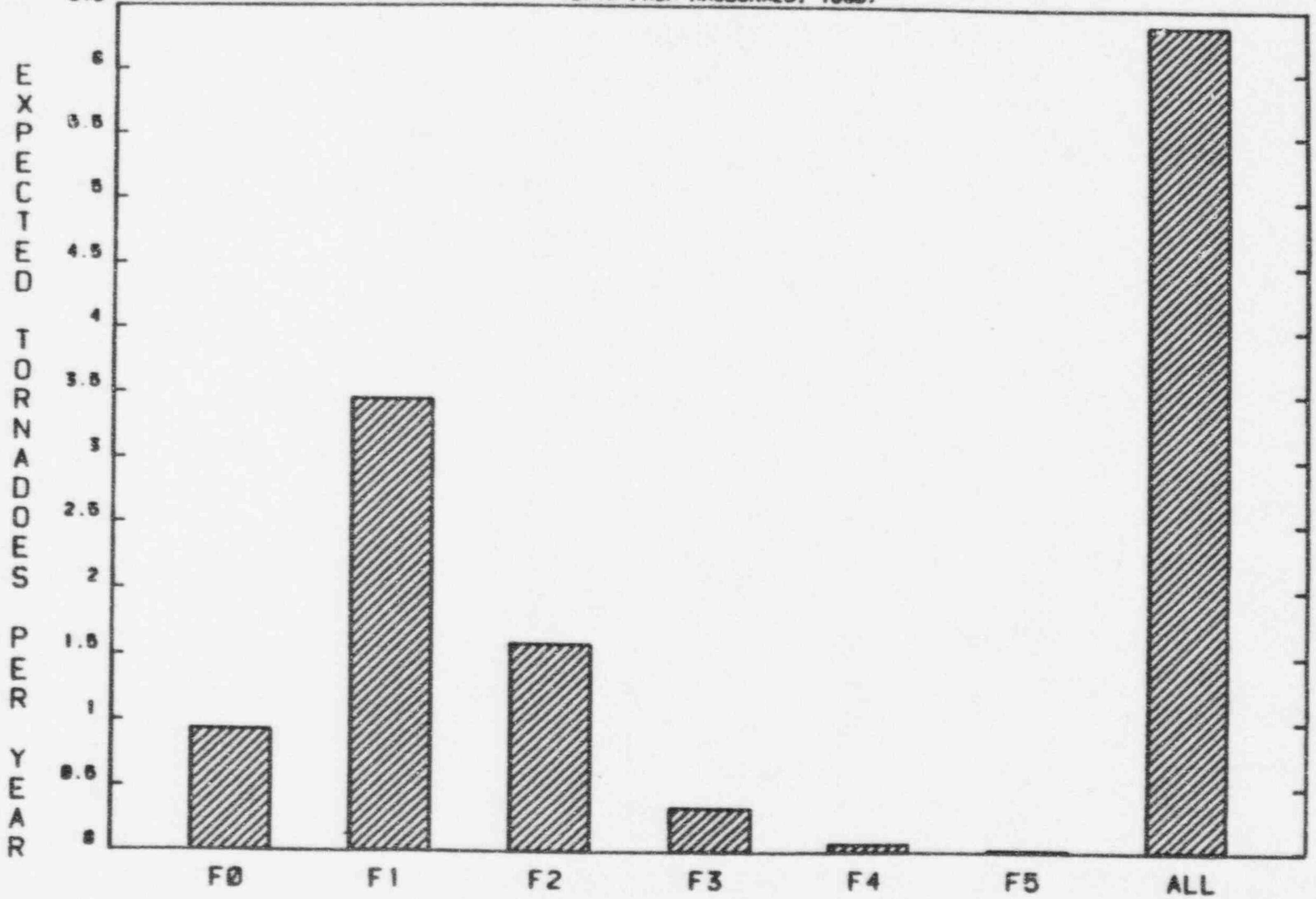


FIGURE 4B: THE UPPER LIMIT OF THE EXPECTED NUMBER OF TORNADOES PER YEAR AS A FUNCTION OF TORNADO INTENSITY

(DATA FROM MACDONALD, 1988)

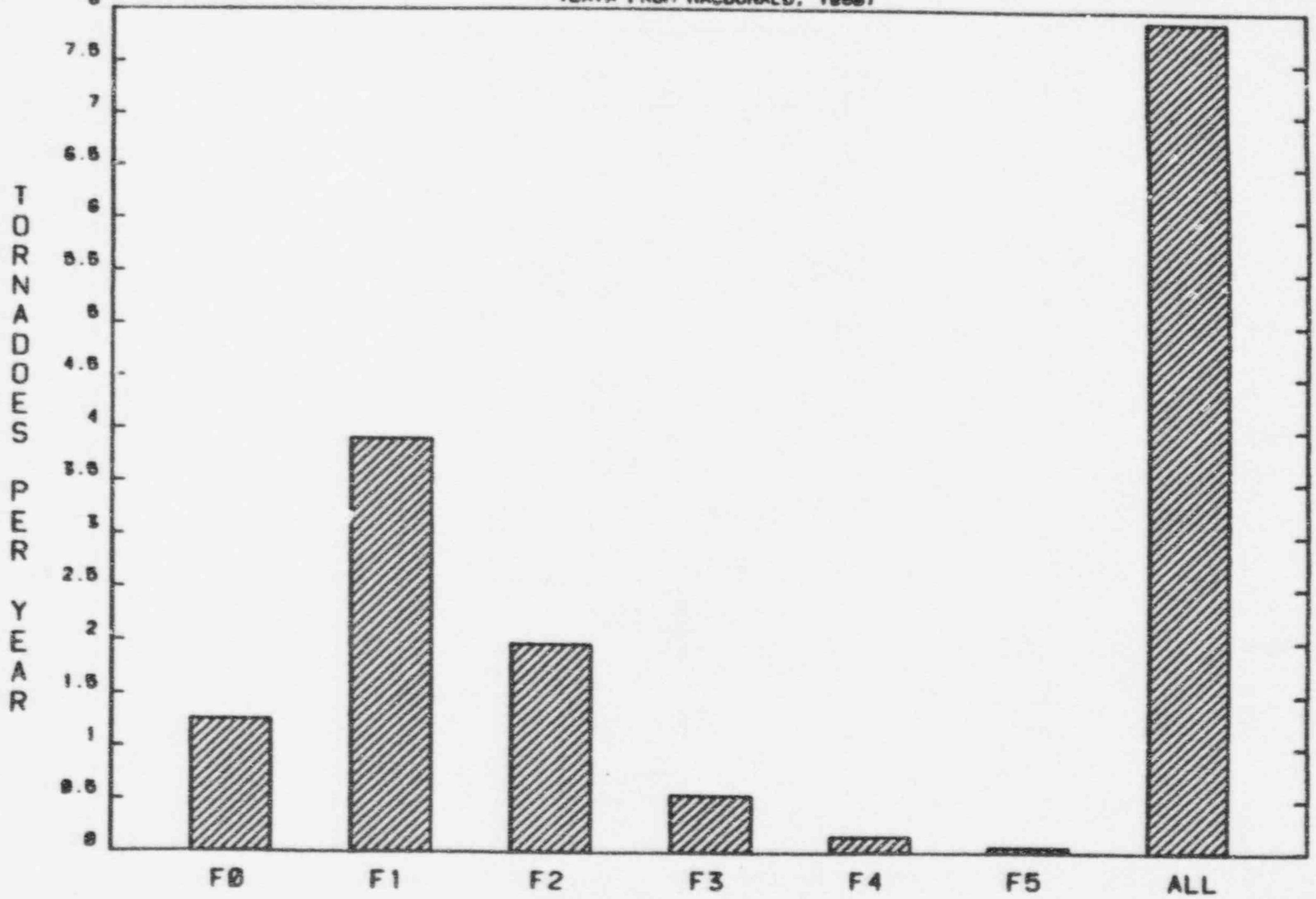


FIGURE 5: MEAN DAMAGE PATH AREA
AS A FUNCTION OF TORNADO INTENSITY

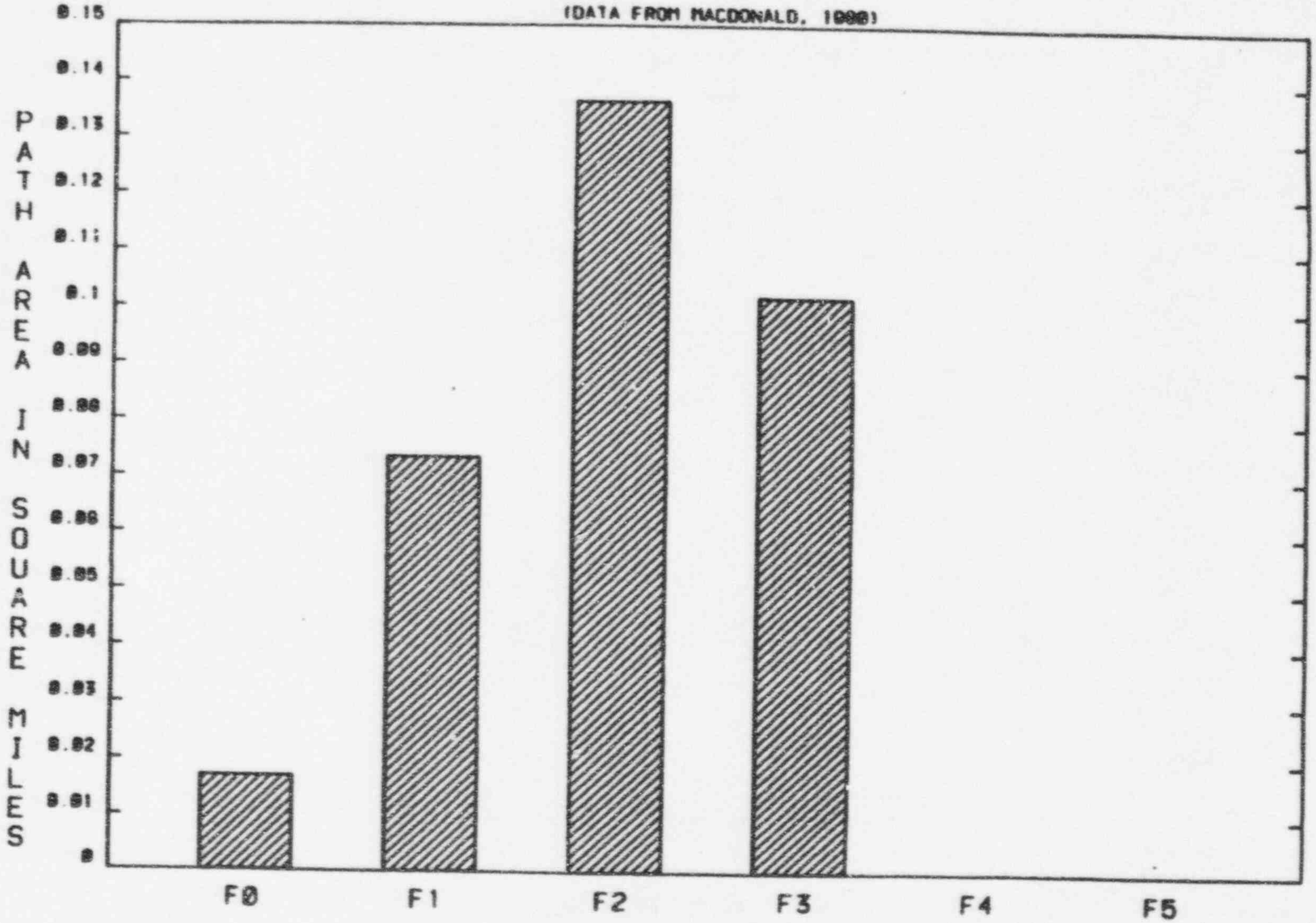


FIGURE 6: EXPECTED DAMAGE PATH AREA
AS A FUNCTION OF TORNADO INTENSITY

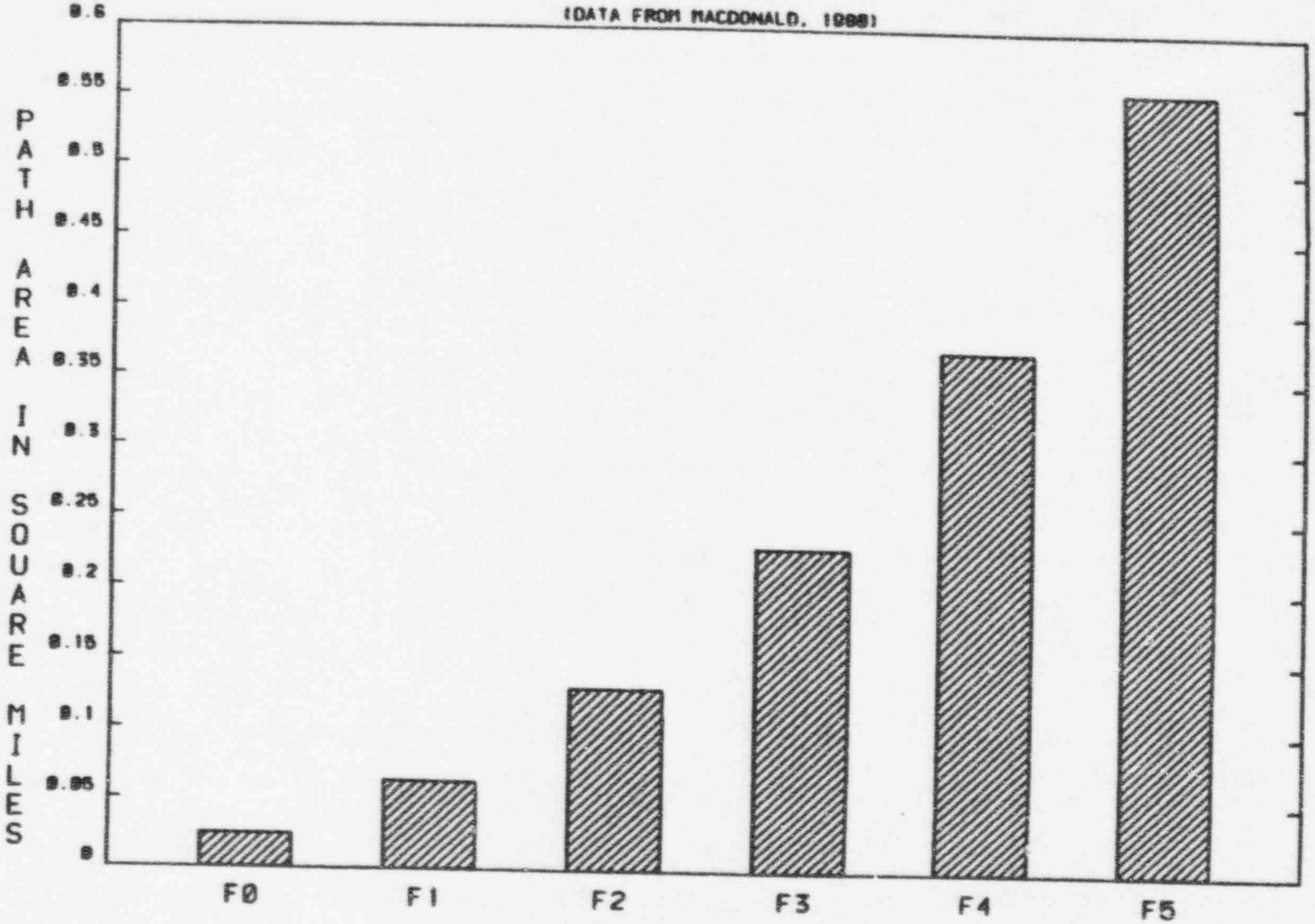


FIGURE 7: TORNADO AND STRAIGHT WIND HAZARD PROBABILITY MODEL FOR OYSTER CREEK POWER REACTOR SITE, NEW JERSEY

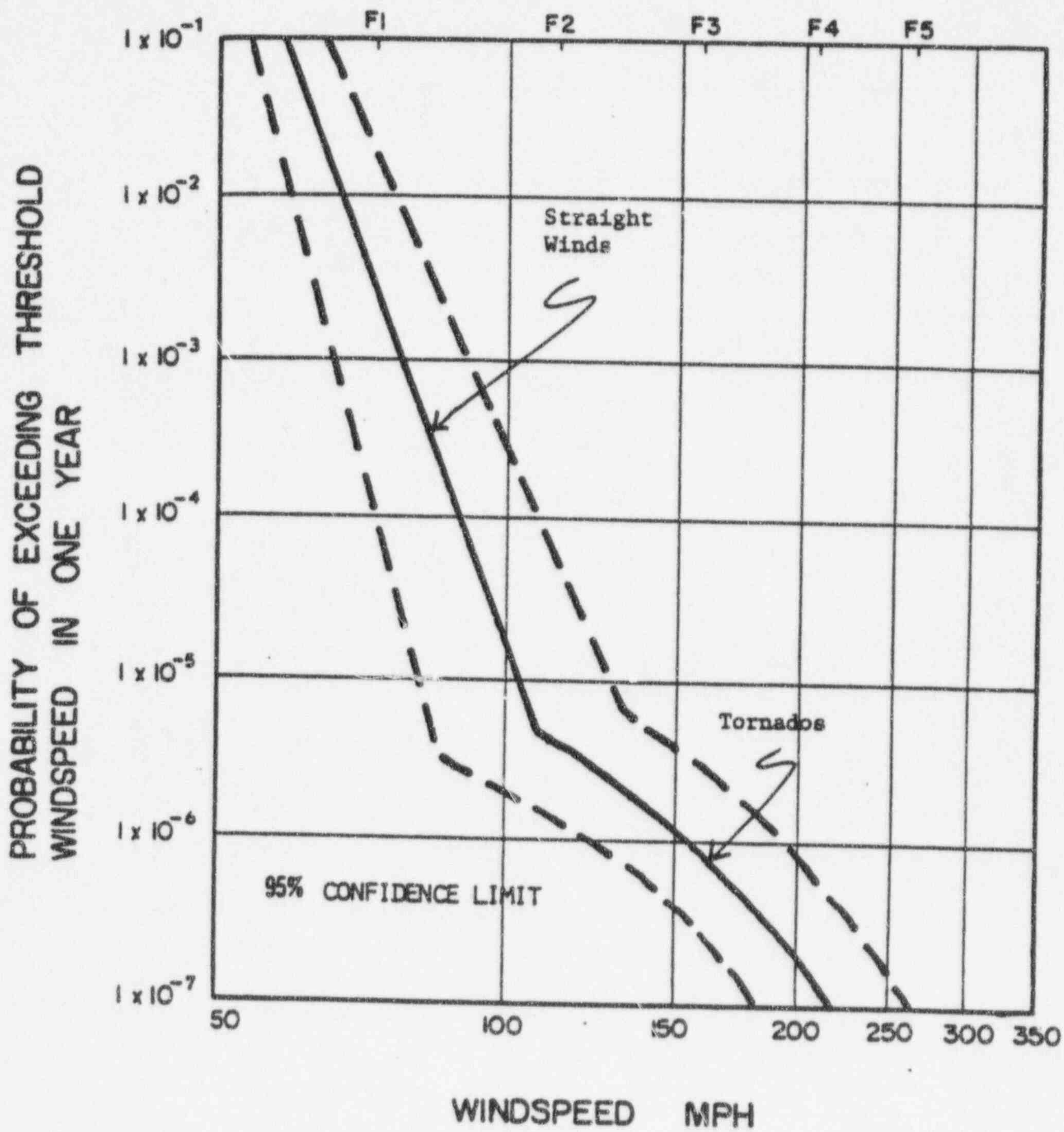


TABLE 1

F-Scale Classification of Tornado Intensity Based on Damage (Reference 5)

Fujita (Reference 4) proposed a rating system, now widely recognized, whereby tornado intensity (i.e., wind speed) is judged on the basis of damage appearance. Six intensity levels of the Fujita scale, or F-scale, were defined. Each intensity classification has an association wind speed range. Table 1 presents a general damage description of the destruction expected within each F-scale classification. The intensity assigned to a tornado is based on the worst damage within the tornado path.

(F0) LIGHT DAMAGE 40-72 mph

This speed range corresponds to Beaufort 9 through 11. Some damage to chimneys or TV antennae; breaks branches off trees; pushes over shallow-rooted trees; old trees with hollow inside break or fall; sign boards damaged.

(F1) MODERATE DAMAGE 73-112 mph

73 mph is the beginning of hurricane wind speed or Beaufort 12. Peels surface off roofs; windows broken; trailer houses pushed or overturned; trees on soft ground uprooted; some trees snapped; moving autos pushed off the road.

(F2) CONSIDERABLE DAMAGE 113-157 mph

Roof torn off frame houses leaving strong upright walls standing; weak structure or outbuildings demolished; trailer houses demolished; railroad boxcars pushed over; large trees snapped or uprooted; light-object missiles generated; cars blown off highway; block structures and walls badly damaged.

(F3) SEVERE DAMAGE 158-206 mph

Roofs and some walls torn off well-constructed frame houses; some rural buildings completely demolished or flattened; trains overturned; steel framed hangar-warehouse type structures torn; cars lifted off the ground and may roll some distance; most trees in a forest uprooted, snapped, or leveled; block structures often leveled.

TABLE 1 (Continued)

F-Scale Classification of Tornado Intensity Based on Damage (Reference 5)

(F4) DEVASTATING DAMAGE 207-260 mph

Well-constructed frame houses leveled, leaving piles of debris; structure with weak foundation lifted, torn, and blown off some distance; trees debarked by small flying debris; sandy soil eroded and gravels fly in high winds; cars thrown some distances or rolled considerable distance finally to disintegrate; large missiles generated.

(F5) INCREDIBLE DAMAGE 261-318 mph

Strong frame houses lifted clear off foundation and carried considerable distance to disintegrate; steel-reinforced concrete structures badly damaged; automobile-sized missiles fly through the distance of 100 yds. or more; trees debarked completely; incredible phenomena can occur.

TABLE 2

Tornado Hazard Probabilities with 95 Percent Confidence Limits

		TORNADO INTENSITY		
Mean recurrence interval	Hazard Probability Per Year	Expected Wind Speed (mph)	Lower Limit of Wind Speed (mph)	Upper Limit of Wind Speed (mph)
10,000	1.0×10^{-4}	<40	<40	<40
100,000	1.0×10^{-5}	84	56	118
1,000,000	1.0×10^{-6}	152	123	188
10,000,000	1.0×10^{-7}	209	174	255

TABLE 3

Summary of Wind Speed Risks with 95 Percent Confidence Limits for Oyster Creek

Recurrence Limit	Probability Per Year	Wind Speeds, mph Expected Value	Lower Limit	Upper Limit	Type of Storm
100	1.0×10^{-2}	69	60	78	Straight Wind
1,000	1.0×10^{-3}	80	66	93	Straight Wind
10,000	1.0×10^{-4}	91	73	109	Straight Wind
100,000	1.0×10^{-5}	102	80	124	Straight Wind
1,000,000	1.0×10^{-6}	152	123	188	Tornado
10,000,000	1.0×10^{-7}	209	174	255	Tornado

5.1.9 References

1. McDonald, J. R., 1982: Tornado and Straight Wind Hazard Probability for Oyster Creek Nuclear Power Reactor Site, New Jersey. Prepared for U. S. Nuclear Regulatory Commission, Site Safety Research Branch, Division of Reactor Safety Research by Institute for Disaster Research, Texas Tech University, Lubbock, Texas.

2. Abbey, Robert F., Jr., 1976: Risk probabilities associated with tornado wind speeds. Proceedings of the Symposium on Tornadoes (Assessment of Knowledge and Implications for Man), June 22-24, 1976, Texas Tech University, Lubbock, Texas.

3. Abbey, R. F., Jr., 1982: Private Communication, Office of Nuclear Regulatory Research.

4. Fujita, T. T., 1971: Proposed characterization of tornadoes and hurricanes by area and intensity. SMRP Res. Paper No. 91, University of Chicago, Chicago, Illinois.

5. Fujita, T. T., 1973: Experimental classification of tornadoes in F P P scale. SMRP Res. Paper No. 98, University of Chicago, Chicago, Illinois.

6. Thom, H. C. S., 1963: Tornado Probabilities. Mon. Wea. Rev., 91, 730-736.

7. Crutcher, H. L., et al, 1975: A preliminary note on return periods and their misuse. Unpublished manuscript available from NOAA National Climate Center, Asheville, North Carolina, 10 pp.

8. Letter, March 8, 1986, from J. A. Zwolinski (NRC) to Fiedler (GPU Nuclear), Subject: Integrated Plant Safety Assessment Section 4.3, Wind and Tornado Loadings - Oyster Creek (TAC 49392).

9. Letter, January 29, 1991, from J. F. Stolz (NRC) to J. J. Barton (GPU Nuclear), Subject: Issuance of Safety Evaluation Report Relating to the Full-Term Operating License for Oyster Creek Nuclear Generating Station.

10. Letter, July 29, 1992, from A. W. Dromerick (NRC) to J. J. Barton (GPU Nuclear), Subject: Adequacy of Shutdown Cooling System in the Event of a Tornado at Oyster Creek Nuclear Generating Station - Wind and Tornado Loads - Item 10, SEP Topic III-2 and Tornado Missiles IPSAR Section 4.6.2 - SEP Topic III - 4.A (TAC No. M79165).

11. Letter, December 7, 1992, from A. W. Dromerick (NRC) to J. J. Barton (GPU Nuclear), Subject: Evaluation of Upper Reactor Building and Non-Safety Architectural Components Subjected to Tornado-Wind Loading - Items 1 and 11 of SEP Topic III-2 (TAC No. M79165).

12. Letter, June 5, 1992, from R. L. Long (GPUN) to NRC, Subject: Project List Semi-Annual Update.

13. Twisdale, L. A., W. L. Dunn, and J. Cho, "Tornado Missile Risk Analysis," Electric Power Research Institute, EPRI NP-768, May 1978.

14. U. S. Atomic Energy Commission Regulatory Guide 1.76: Design Basis Tornado for Nuclear Power Plants, April 1974.

15. NUREG 0822, "Integrated Plant Safety Assessment Systematic Evaluation Program, Oyster Creek Nuclear Generating Station," Final Report, January 1983.

16. Letter, February 28, 1990, from A. W. Dromerick (NRC) to E. E. Fitzpatrick (GPUN), Subject: Oyster Creek Nuclear Generating Station Systematic Evaluation Program (SEP) topics: II-3B, "Flooding Potential and Protection Requirements" and III-4A, "Tornado Missiles" (TAC Nos. 41352 and 49394).

SECTION 5.2

EXTERNAL FLOODS

TABLE OF CONTENTS

5.2.1 Introduction and Summary

Barnegat Bay, on which the Oyster Creek site is located, is a relatively shallow body of water extending in a north-south direction parallel to the New Jersey coastline. It is separated from the Atlantic Ocean by Long Beach Island and Island Beach Peninsula, which are divided from each other by the narrow Barnegat Inlet. The bay itself is approximately twenty miles long and from one to five miles wide and varies in depth between one and ten feet. It is part of the intracoastal waterway and is adjacent to Little Egg Harbor on the south and Silver Bay on the north. On the ocean front at Barnegat Inlet, the mean low water level is 1.5 feet mean sea level (Reference 1).

On the south of the plant site, Oyster Creek flows east to Barnegat Bay. Its drainage basin is 12.4 square miles and consists mostly of pine barrens. It is dammed by a low-head earthen dam known as the Wells Mills Dam, which has a timber spillway and shallow reservoir about four miles upstream from the plant site. Another low-head timber dam on the site forms a pond with a four-acre surface area. It is used to store fire water for use at the plant. Oyster Creek joins the discharge canal approximately 700 feet west of the Route 9 bridge. To the north of the site is South Branch Forked River, which has a watershed area of 2.7 square miles, also flowing west to east in pine barrens land. It is not dammed and empties into the intake canal just upstream of the railroad and the Route 9 bridges crossing the intake canal. The South Branch Forked River discharge flows through two structures before reaching the canal. One is a 12-inch-diameter steel pipe, and the other is a water passageway under the Forked River Nuclear Station site access road (Reference 1).

The plant site covers approximately 800 acres. The plant structures were built on an island created by the intake canal to the north and west, the discharge canal to the south and west, and Barnegat Bay to the east. A dike due east of the reactor and turbine buildings separates the intake and discharge canals and provides ready access to the rest of the site from the island (Reference 1).

Water levels in Barnegat Bay are influenced primarily by winds and tidal actions. Effluents discharged into Barnegat Bay ultimately are mixed with ocean water, with the extent of mixing dependent on the tidal forces, local winds, rainfall runoff, and temperature and salinity gradients. The barrier beach and the shallowness of the bay minimize tidal fluctuations by attenuating the tidal energy (Reference 2).

The potential for flooding due to stream flow was evaluated for the Oyster Creek Nuclear Generating Station (OCNGS) as part of the Systematic Evaluation Program (SEP) and no flooding that would affect safety related structures was postulated for the site (Reference 2).

Two small dams are located on the Oyster Creek. Incremental flood flows were calculated based on their breaching by any unspecified cause as part of the SEP. No flooding again which would affect safety related structures is postulated for the site (Reference 2).

Tsunami events are not typical of the eastern coast of the United States and therefore, have not been addressed for the site (Reference 2).

The site grade elevation is 23 feet mean sea level (MSL). During its review of the hydrology-related topics, the NRC staff identified the following flooding levels as defined by current licensing criteria (Reference 1):

- Probable Maximum Hurricane (PMH) - 22 feet MSL
- Probable Maximum Precipitation (PMP) - 23.5 feet MSL

The probable maximum hurricane storm surge still water level at the site is 22 feet MSL. Less than one foot of wave runup would occur. Wave forces on the intake structure will be minimal because of refraction around the plant island (Reference 1).

The plant island is divided into three drainage basins. The area with the greatest potential for local flooding from probable maximum precipitation is the 5.2-acre area at the north. The storeroom, mobile offices, old and new radwaste buildings, office building, boiler house, and part of the reactor building are located in this area. Existing storm drains functioning at full capacity are assumed to remove six cubic feet per second of runoff, leaving a peak overland flow of 60 cubic feet per second. Ponding (five inches deep) occurs to elevation 23 feet 5 inches mean sea level (MSL) (Reference 1).

5.2.2 Flood History

In March 1962, high winds accompanied by a storm which is considered to have been the most severe ever to strike New Jersey, left water marks which were recorded by the United States Geological Survey immediately north of the Oyster Creek site at Forked River. These water marks showed a high flood elevation of 4.5 feet (Reference 2).

5.2.3 Flood Design Considerations and Protections

The plant site has a general grade elevation of 23 feet MSL. The land slopes down gradually towards the north, south, and east. On the west, the grade meets the top of the intake and discharge canals at an elevation of 23 feet and then drops sharply into the canal bottom elevation of (-)10 ft MSL. The slope at the canal bank is 1:1-1/2 (Reference 2).

As reported in section 5.2.1, the maximum flood level due to PMH will be at elevation 22 ft MSL. The plant grade, elevation 23 ft MSL, is one foot above the PMH flood level. Therefore, the flood will not find its way into the plant buildings, the floor levels of which are generally six inches above grade at elevation 23' -6". The circulating water intake structure with its deck at elevation 6 ft will be under water. This deck supports, apart from the other equipment, the circulating water pumps and the emergency service water pumps. During a PMH flood, the circulating water and service water pumps will become inoperable and thus emergency plant procedures have been instituted which require the plant to be shutdown when flood waters reach a predetermined level as to ensure the capability for safe shutdown under either normal or abnormal conditions (Reference 2).

For flooding of structures caused by local probable maximum precipitation, protection is provided up to a flood level of 23.5 feet MSL. By letter of June 6, 1983, GPU reported to NRC that all sill and entry flood elevations are at or above 23 feet, 6 inches MSL for the reactor building, the turbine building, and the new and old radwaste buildings (Reference 3). However, the two entrances to the emergency diesel generator building were at elevation 23 feet MSL, which was 6 inches below the flooding level which would be caused by local probable maximum precipitation (23.5 ft MSL). A 6-inch high asphalt dike has been provided at these entrances to provide protection against internal flooding of the diesel generator building (Reference 2). Further, GPU confirmed that a review of contour maps of the site had shown no indication of contours that might impound water at the southwest door of the offgas building as NRC had inquired about and, therefore, no modifications were necessary.

As a result of the flooding levels reported for Probable Maximum Hurricane and Probable Maximum Precipitation in Section 5.2.1 here, NRC Staff identified nine issues in the Integrated Plant Safety Assessment Report (IPSAR) pertaining the following: (1) condensate transfer pumps, (2) plant operating limits in the Oyster Creek Technical Specifications (TS) on canal water level, (3) canal water level instrumentation, (4) makeup isolation condenser water sources, (5) plant operating limits in the TS on water level at the service water intake, (6) procedures for a flood, (7) protection during internal flooding, (8) hydrostatic loads on buildings, and (9) reactor and turbine building parapets and scuppers. All these issues have been considered resolved by the NRC Staff and a brief summary of each is presented below (Reference 1):

1) Condensate Water Pumps

> "In IPSAR Section 4.1(1), the staff concluded that two condensate transfer pumps are essential to charge the emergency condenser with cooling water during a hurricane-induced flood. Because the motors of both of these pumps are powered from the same engineered safety features bus, a single failure of the power bus would disable both condensate transfer pumps.
>
> In letters dated August 14, 1987, and August 12, 1988, the licensee stated that through a detailed field walkdown and line-loss analysis of an existing system interconnection between the core spray and condensate and demineralized water transfer systems, it was determined that the existing plant configuration ensures that makeup water can be supplied to the isolation condenser. As discussed in Section 3.5.1.1 of this SER, the staff reviewed the proposal and found the water supply path acceptable. Staff concerns about dependence of this path on diesel generators are addressed for flooding scenarios, as discussed in IPSAR Supplement 1, Section 2.1.3. This item is resolved." (Reference 1).

The IPSAR Supplement 1, Section 2.1.3 referred to above, discusses the issue of the two entrances to the diesel generator building being raised to 23.5 feet MSL elevation (Reference 3). This modification has been completed as discussed before (Reference 2).

2) Flooding Level Procedures

"In its topic evaluation, the staff concluded that the Oyster Creek Technical Specifications should include plant-operating limits when flood water levels at the intake or discharge canals exceed 4.5 feet MSL. This proposed requirement was based on the plant emergency procedure (EP-520), which specified operator actions to be taken when water levels in the intake or discharge canals exceed 4.5 feet MSL.

In IPSAR Section 4.1(2), the staff concluded that procedures are sufficient to specify corrective actions for flooding conditions, and modifications to the plant Technical Specifications were not warranted. Therefore, this item was resolved in the IPSAR." (Reference 1).

See Item 6 "Hurricane Flooding of Pumps" later for updated referenced procedures that formed the basis for NRC's conclusion above.

3) Canal Water Level Instrumentation

"In IPSAR Section 4.1(3), the staff concluded that water level instrumentation in the intake canal was inadequate and there was no water level measurement in the discharge canal. The staff recommended that automatic water level instrumentation be provided so that the operator would be able to implement emergency shutdown procedures when the specified flooding levels occurred. Because these instruments are not intended for post-accident monitoring, they need not necessarily be safety grade. With adequate water level instrumentation in the intake canal, another water level gage in the discharge canal was not necessary because flooding conditions could be identified from the intake canal measurement.

The design-basis hurricane surge for the Oyster Creek plant has a still water elevation of 22.0 feet MSL. Associated wind waves are estimated to be less than one foot. External cooling water for the plant can be supplied from service water, circulating water, and fire water pumps.

The fire pumps are powered by diesel generators, which are located at about elevation 12.0 feet MSL. The service water and circulating water pump motors are located on the deck of the intake structure (6.0 feet MSL) at about elevation 8.0 feet MSL.

Oyster Creek Nuclear Generating Station Procedure Number 2000-ABN-3200.31, "High Winds," requires initiation of plant shutdown if the intake water level exceeds elevation 4.5 feet MSL and reactor scram if the water level exceeds 6.0 feet MSL. There is no technical specification for plant shutdown for either high or low water level.

The staff gage mounted on the wing wall of the intake structure that was used to measure high water level was inadequate because of small-sized markings, missing markings, and wrong elevation datum. The staff requested that the licensee replace the existing gage with a quality automatic water level recording gage compatible with its safety significance. The licensee installed a new staff gage on September 13, 1988. The new gage elevation datum is mean sea level and is the same datum that is used in the associated plant operating procedures. The gage has legible gradations that are easily read from the deck of the intake structure.

Although the staff believes there would be some decrease in the margin of safety using the staff gage rather than a recording gage, the degree of variance is difficult to quantify. The most significant factor is the introduction of the potential for human error. The staff gage must be read visually and still water level interpolated, whereas a recording gage is located in a stilling well that eliminates wave effects. The high water levels most probably will occur during high winds and heavy rain and at night; these factors increase the potential for human error. Conversely, the deck of the intake structure is at elevation 6.0 feet MSL, which is the "reactor scram" control elevation, and it should be fairly easy to determine when water was over the deck; there would still be 2.0 feet of freeboard before the service water pump motors were lost. The operating procedure does not specify the frequency for reading the gage during these adverse conditions. The staff requires that the gage be read at 1/2-hour intervals from the beginning of high winds and until the water level reaches 3.0 feet MSL and then continuously when the level is above 3.0 feet MSL.

The staff finds the new staff gage and revised operating procedures to be an acceptable alternative to the automatic water level recording gage it had requested." (Reference 1).

4) Isolation Condenser Flooding

"In IPSAR Section 4.1(4), the staff stated that the plant did not have a reliable means of maintaining a safe shutdown in light of single-failure and flooding conditions, specifically in regard to the provision of adequate makeup water sources for the isolation condensers.

The staff required the licensee to make procedural revisions to include the fire water storage tank as a redundant source of water supply to the emergency condenser and to include in operating procedures a minimum inventory of water to be maintained in the condensate storage tank.

In IPSAR Supplement 1, Sections 3.1.1 and 4.1.1, the staff identified the licensee procedures that specify actions associated with emergency condenser water supplies. The staff verified these procedures by inspection and found them acceptable. Because full resolution of this issue depends on the resolution of the related issue in IPSAR Section 4.1(1), and the latter issue is reported as resolved in Section 3.4.1.1 of this SER, this issue is also resolved." (Reference 1).

The IPSAR Supplement 1, Section 3.1.1 (Reference 3) referred to above indicates that:

"In its letters dated July 26, 1985, and April 21, 1986, the licensee stated that makeup water to the isolation condensers is provided by the condensate storage tank and the fire water storage tank. These tanks can provide a volume of water of nearly one million gallons, which should be sufficient to maintain the reactor in hot shutdown using the isolation condensers for 10 days. This time is sufficient to take corrective actions to restore submerged components. The tanks and the pumps are above the probable maximum hurricane flooding level of 22 feet MSL.

The licensee explained in the meeting of June 16 and 17, 1986 (meeting summary dated August 1, 1986), that its procedures require a minimum of 20 feet or 250,000 gallons in the condensate storage tank (CST). The intake tour sheet requires a minimum of 350,000 gallons in the fire water storage tank. The high wind conditions for Emergency Procedure 2000-ABN-3200.31 are the following: (1) tornado watch or warning, (2) hurricane watch or warning, (3) tornado funnel cloud in the area, and (4) sustained wind speeds greater than 74 mph. This procedure requires the CST to be filled to 43 feet or 537,500 gallons and the isolation condensers to be filled (50,000 gallons). In its safety evaluation dated November 28, 1986, the staff stated that the licensee had stated that it could, if needed, bring in a fire truck and pump water into the isolation condensers using an alternate connection from the fire water main."

The IPSAR Supplement 1, Section 4.1.1 (Reference 3) referred to above indicates that:

"The following licensee procedures specify actions associated with emergency condenser water supplies:

- Procedure 307, "Isolation Condenser System," states, in relation to filling the isolation condenser, "In emergency situations fire protection shall be used if condensate transfer is not available." The procedure also provides instructions for providing makeup water to the isolation condenser from the fire protection system.

- Procedure 316, "Condensate System," specifies that 20 feet (250,000 gallons) of water should be maintained in the condensate storage system.

- Procedure 333, "Plant Fire Protection System," specifies that 310,000 gallons of water or more should be maintained in the fire water storage tank.

- Procedure 2000-ABN-3200.31, "High Winds," specifies certain actions to be taken at specific sea water levels. Among these actions are filling the isolation condenser to the high level alarm (7.7 feet) and filling the condensate storage tank to the high level alarm (43 feet).'

5) Low Water Level Shutdown

"In addition to the concern related to shutdown under flooding conditions, the staff identified the concern of low water level shutdown in IPSAR Section 4.1(5). The licensee addressed this concern by providing administrative procedures to monitor water level, using the intake canal instrumentation discussed in Section 3.4.1.3 of this SER (IPSAR Section 4.1(3)), and to appropriately respond to low level in the intake canal.

Low water level at the Oyster Creek station may be caused by a hurricane that forces water out of the intake canal, blockage of the canal, or blockage of the intake screens. Two gages (PI-SWS-1 and PI-SWS-2) at the intake structure monitor potential low water level in the intake canal. These gages provide indication of the intake structure's water level that is on the plant side of the traveling screens and therefore includes any reduction that would result from clogging of the screens. These gages are read routinely (i.e., every shift) by a plant operator, and the readings are recorded on the Intake Area Tour Sheet. Operating Procedure 2000-ABN-3200.32, "Response to Loss of Intake," contains operator actions required at various water levels in the intake canal in order to regain level as well as to ensure safe operation of the plant.

The procedure also instructs the operator to monitor service water discharge pressure indication in the control room to avoid possible service water pump cavitation. The service water pumps are expected to reach their minimum required water level at -0.5 foot MSL. Service water may be lost at this level, and the operator is instructed to follow Operating Procedure 2000-ABN-3200.18, "Service Water Failure." The procedure instructs the operator to shut down the plant if the service water system cannot be returned to operation.

The staff concludes that the licensee's procedures and equipment used for monitoring low water level and controlling the plant under low-water-level conditions are acceptable." (Reference 1).

6) Hurricane Flooding of Pumps

"In IPSAR Section 4.1(6), the staff indicated that the licensee had proposed to update emergency procedures, to identify the alternate water sources and flow paths if the intake structure became flooded, and to identify the priority of water sources and flow paths to be used to ensure a safe shutdown.

In IPSAR Supplement 1, Section 4.1.2, the staff reported that the licensee had identified the station procedures which resolve this item." (Reference 1).

The IPSAR Supplement 1, Section 4.1.2 (Reference 3) referred to above indicates that:

"The licensee has provided procedural instructions in Station Procedure 2000-ABN-3200.31, "High Winds," for the actions to be taken in the event of high water level in the intake structure. The instructions include actions to be taken in shutting down the circulating water pumps and the service water pumps. In addition, Station Procedure 307, "Isolation Condenser System," provides instructions for providing makeup water to the isolation condenser using the fire protection system should the preferred condensate transfer system not be available."

7) Protection During Internal Flooding

"In IPSAR Section 4.1(7), the staff stated that protection against internal flooding of structures caused by local probable maximum precipitation should be provided to a flood level of 23.5 feet MSL and that the licensee should verify that all entrance levels were above this level.

In IPSAR Supplement 1, Section 2.1.3, the staff reported that the licensee had verified that all entrances except two entrances to the diesel generator building are not vulnerable to flooding to the 23.5-foot MSL. These two entrances are at elevation 23 feet MSL. The licensee proposed to construct a 6-inch-high asphalt dike at each of the two entrances. The staff found this proposal acceptable for resolving the concern." (Reference 1).

The above modification to the diesel generator building two entrances has been completed as discussed before.

8) Ground Water Elevation and Effects of High Water Level on Structures

"In IPSAR Section 4.4, the staff reported that all issues associated with this topic, except that concerning short-term hydrostatic loads, had been resolved.

In IPSAR Section 4.4(2), the staff concluded that the licensee should demonstrate that safety-related structures would remain functional under a short-term hydrostatic load and could resist flotation for water levels up to 22 feet MSL.

This resolved the remaining item and the issue of groundwater elevation (IPSAR Section 4.1(8))." (Reference 1).

9) Roof Drains

"In IPSAR Section 4.1(9), the staff stated that the licensee had committed to drill holes in the parapets and install scuppers to preclude the potential for buildup of rain water on the roof of either the reactor building or the turbine building.

In IPSAR Supplement 1, Section 4.1.3, the staff reported that the modifications had been completed and verified by inspection. Thus, this issue was resolved." (Reference 1).

Additionally, the NRC staff evaluated the issue of "Inservice Inspection of Water Control Structures" at Oyster Creek and reported the following (Reference 1, Section 3.4.3):

"The licensee identified the following water control structures and components that require surveillance in accordance with 10 CFR Part 50 ((GDC 1) as implemented by Regulatory Guide 1.127: the intake and discharge canals, the intake structure, trash racks, traveling screens, tunnels, pumps, and the fire protection pond.

The licensee has revised the existing inspection program so that it includes the requirement that the program be conducted or overseen by qualified engineering personnel, that a documentation file be established, and that water control structures be inspected following extreme events.

Resolution of the issues associated with this topic is documented in IPSAR Section 4.5 and IPSAR Supplement 1, Section 4.2."

5.2.4 Conclusions

The Systematic Evaluation Program (SEP) was initiated by the U. S. Nuclear Regulatory Commission (NRC) to review the designs of older operating nuclear power plants in order to reconfirm and document their safety. The review provided (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The results of the SEP review of the Oyster Creek plant were published in NUREG-0822, the Final Integrated Plant Safety Assessment Report (IPSAR), dated January 1983, (Reference 4), and its supplement No. 1 dated July 1988 (Reference 3).

The following criteria were used during the review of the flood related topics for the above evaluation:

> "10 CFR Part 50 (GDC 2), as implemented by SRP Sections 2.4.2, 2.4.5, 2.4.10, and 2.4.11 and Regulatory Guides 1.59 and 1.27, requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as flooding. The safety objective of the review under these topics (II-3.B, II-3.B1, and II-3.C) is to verify that operating procedures and/or system design provided to cope with the design-basis flood are adequate." (References 1 and 4)

> "10 CFR Part 50 (GDC 2), as implemented by SRP Section 3.4 and Regulatory Guide 1.59, requires that plant structures be designed to withstand the effects of flooding. The safety objective of the review under this SEP topic is to ensure the function of safety-related structures with hydrostatic loading resulting from design-basis water levels when combined with other non-accident loadings." (Reference 1).

"10 CFR Part 50 (GDC 2, 44, and 45) as implemented by Regulatory Guide 1.127, requires that structures, systems, and components important to safety be designed to withstand natural phenomena such as floods and that a system to transfer heat to an ultimate heat sink be provided. Water control structures used for flood protection and emergency cooling water system are inspected to ensure that water control structures that are part of the ultimate heat sink are available at all times during both normal and accident conditions." (Reference 1).

NRC Generic Letter 88-20, Supplement 4 (Reference 5) and NUREG-1407 (Reference 6) recommend that the licensees compare their plants against the 1975 SRP criteria.

The issues raised as a result of the SEP evaluation in 1983 (Reference 4) were discussed in previous section 5.2.3 here. These issues were results of such comparison against the SRP criteria in effect at the time (1975 SRP) and were enumerated above. All these issues have been considered resolved by NRC as reported in the Oyster Creek Full-Term Operating License Safety Evaluation in 1991 (Reference 1). It is therefore concluded that the intent of the 1975 SRP criteria for the external flooding issue at Oyster Creek is met and no further analysis is required according to NUREG-1407 (Reference 6).

5.2.5 References

1. Letter, January 29, 1991, from J. F. Stolz (NRC) to J. J. Barton (GPU Nuclear), Subject: Issuance of Safety Evaluation Report Relating to the Full-Term Operating License for Oyster Creek Nuclear Generating Station.

2. Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 2.4, Update 7, December 1992.

3. Integrated Plant Safety Assessment Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, NUREG 0822, Supplement No. 1, U. S. Nuclear Regulatory Commission, July 1988.

4. Integrated plant Safety Assessment Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, NUREG 0822, Final Report, U. S. Nuclear Regulatory Commission, January 1983.

5. U. S. NRC Generic Letter No. 88-20, Supplement 4, June 28, 1991, Subject: Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54 (F).

6. U. S. NRC NUREG 1407, Subject: Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, Final Report, June 1991.

SECTION 5.3

TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

TABLE OF CONTENTS

5.3.1 Introduction and Summary

Ocean County's industrial base is small, but diversified. Boat building and the manufacturing of marine equipment were once the dominant industrial activities, but today the industrial activity also includes chemical manufacturing, mining of ilmenite, quarrying of industrial sands, garment manufacturing, food processing, and production of concrete (Reference 1).

There are no large commercial harbors within 10 miles of the site. Public marinas are the chief recreational facilities in the immediate site area. The Intracoastal Waterway is the only inland waterway used for shipping in the area. Major shipping lanes in the Atlantic Ocean are located well off shore (Reference 1).

The nearest railroad corridor is approximately 0.25 miles east of the reactor building. Rail traffic through this corridor has been discontinued, and the railroad tracks have been removed (Reference 1).

The hazards due to natural gas pipelines, airfields and transportation near the site are discussed in the following sections.

5.3.2 Nearby Natural Gas Pipeline Hazard Evaluation

There are four natural gas pipelines along US Route 9 approximately 0.25 miles from the plant. The USNRC Systematic Evaluation Program (SEP) Topic II-1C for Oyster Creek (Reference 2) evaluated the impact of two of these pipelines (6 inch and 8 inch diameters) and concluded that the pipelines do not pose a significant hazard to the plant due to the distance involved. The full-term operating license safety evaluation re-affirmed this conclusion and noted that these pipelines pass through the edge of the exclusion area boundary and therefore do not pose a significant hazard to the plant because of the distance involved (Reference 1).

In 1990, two natural gas pipelines were installed in the vicinity of Oyster Creek Nuclear Generating Station (OCNGS). One pipeline is a 16 inch diameter line which runs parallel to Route 9 on the east side of the plant. This pipeline is outside the plant exclusion zone except where it crosses the intake and discharge canals. The other pipeline is a 16 inch line that runs roughly adjacent to the north side of the intake canal to the combustion turbines owned by JCP&L (Reference 3). Even considering the remote possibility that an explosion of this pipeline along the intake canal would cause blockage to the plant, Oyster Creek has isolation condensers that can be used for cooling to atmosphere with makeup water provided from the condensate transfer system, or if that is not available, from the fire water system with diesel fire pumps which are a long way from the gas pipeline.

A safety evaluation was performed for the installation of these new lines (Reference 4) and it was concluded that while the newly installed gas lines are larger in diameter and were pressurized to higher pressures than those analyzed by NRC, it is reasonable to conclude that the primary factors which influenced NRC's conclusion of very low hazard (i.e., distance from the plant and low probability of failure) would result in a similar conclusion for the new installation.

Also, NUREG 0014 (Reference 5) comprising the USNRC safety assessment for the Construction of TVA Hartsville Nuclear Plants concluded that the existence of a pipeline in the vicinity represents no undue threat to the safe operation of the proposed facility and that accidents occurring to that pipeline need not be considered in the design of the plant.

This conclusion was based on extensive research study performed for TVA (Reference 6).

The differences between the gasline at the TVA plant and the one at Oyster Creek are as follows:

- The distance from the gas line to the plant is approximately one half mile versus approximately one quarter mile at Oyster Creek.
- The diameter of the pipeline is 22 inches versus 16 inches at Oyster Creek.
- The working pressure is 720 psi versus 350 psi (up to 550 psi in the future, in the main line) at Oyster Creek.

As shown in Reference 6, (Risk Assessment Summary Table) the probability of a pipeline accident affecting the TVA facilities is of an order of magnitude of 10^{-7} or less. Taking the above differences into consideration, a qualitative judgment can be made that the USNRC conclusions listed in NUREG-0014 are applicable for the Oyster Creek facility.

In NUREG 0014, the USNRC performed an independent analysis of the consequences of a gas line accident, which included the consideration of a variety of postulated mechanisms of explosion and dispersal of the natural gas, resulting in the following conclusions:

1. There is no identifiable pipeline accident that could credibly lead to flammable gas concentrations at the plant air intakes. Buoyancy causes the flammable region of the cloud to rise above the plant structures before reaching these structures.
2. The state of knowledge concerning the chemical reactions of natural gas mixed with air is sufficiently well established to form a basis for the judgment that the detonation of an unconfined natural gas dispersal in air is not a credible event.
3. Even if a natural gas cloud from a pipeline failure were to detonate, the maximum transient pressures expected at the NSR structures would not produce overpressures in excess of the maximum design pressures for structures.

5.3.3 Hydrogen Water Chemistry System Hydrogen Storage Facility Hazard Evaluation

The Hydrogen Water Chemistry System was installed to mitigate the potential for Intergranular Stress Corrosion Cracking (IGSCC) in the Reactor Coolant piping. This system consists of four subsystems. They are Bulk Gaseous Hydrogen Supply and Storage Facility, Hydrogen Injection Subsystem, Air Injection Subsystem, and a cross-connect piping with a pressure control station to feed H_2 to the Generator Cooling System. The Bulk Hydrogen Supply and Storage Facility consists of two storage modules with storage capacity of up to 14 days (for continuous operation at normal flow and based on a truck delivery of once a week). The hydrogen storage design pressure is at 2450 psig and storage temperature is between -13° to 105°F (Reference 16).

The EPRI guidelines, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision" (Reference 18), require that a non-mechanistic rupture failure of one hydril tube vessel at the hydrogen storage facility be assumed as the criteria for locating the facility to avoid damage to safety related structures and equipment. The conservative assumption that all the tubes at the facility simultaneously rupture and explode has been evaluated with Appendix B to the EPRI guidelines (Reference 18), "Separation Distances Recommended for Hydrogen Storage to Prevent Damage to Nuclear Power Plant Structures from Hydrogen Explosion," June 1986, by R. P. Kennedy. The evaluation shows that the location of the facility, adjacent to the southeast corner of the stone parking lot in the vicinity of Monument-D, is sufficiently separated from the nearest safety related structure (Ventilation Stack, approximately 620 ft. away) to avoid damage (Reference 17).

In addition, the explosion of the total hydrogen supply of one delivery truck at its closest distance to a safety related structure (Ventilation Stack, approximately 390 ft. away) was evaluated and also found acceptable (Reference 17).

5.3.4 Aircraft Hazards

There are no missile sites within a 10-mile radius of the Oyster Creek site. Nine airfields are located within 20 miles of the plant. Two of the airfields are military installations: (1) McGuire Air Force Base, also used by the U.S. Air Force, U.S. Air National Guard, and the Military Air Transport Service, 25 miles to the northwest of the site and (2) Lakehurst Naval Air Station, 20 miles north-northwest of the site. Other airports listed by the Federal Aviation Administration (FAA) are Breton Woods, 17 miles north; Eagle's Nest, 12 miles south-southwest; Coyle Tower, 10 miles west; Ocean County, 9 miles north-northwest; Manahawkin, 9 miles south-southwest; and Beechwood, 8 miles north-northeast. In addition, there is a sod strip 2 miles northeast of Forked River (Reference 1).

Aircraft strike probabilities have been estimated for three size categories including small general aviation, medium sized commercial, and large (heavy) commercial or military aircraft. The nearest airports of significance are at Lakehurst, 16 miles north-northwest and McGuire Air Force Base about 24 miles northwest. At these distances there is no significant hazard due to landing and takeoff activities. Low level military training routes in the area must be kept more than five miles from the plant by agreement between the military and the NRC. There is little traffic along these routes, and at this distance they represent an extremely low hazard to the plant (Reference 7).

Based on evaluation of the available information on air traffic conditions at the site, it was concluded that the only significant hazard is from the traffic along the V312 airway and general aviation in the area. Probabilities for a strike on the plant were developed for three sizes of aircraft based on available traffic information for each size. The largest mean frequency was from general aviation at 4.0×10^{-7} (Reference 7).

The NRC staff has concluded in Reference 1 that the only air corridor in the vicinity of the site is a civilian corridor marked "Victor Air Lane 312," which is aligned east-west and passes over the site. The corridor can be used by all types of aircraft, but the FAA - which controls all civilian aviation - specifies minimum safe altitudes at which planes can be flown in the corridor. The NRC staff considers this issue resolved based on conclusions reached in IPSAR Supplement 1, Section 2.7.1 (Reference 8), which states that because the aircraft strike probabilities are extremely low, aircraft traffic does not pose a significant threat to the Oyster Creek plant.

The FAA lists three restricted areas in the vicinity of the plant. Two of these areas, R5001A and R5001B, are contiguous to Fort Dix, which is 15 miles to the north-northwest of the site. These restricted areas are used mainly as firing ranges for small arms, artillery, and mortars. The third area, R5002, at Warren Grove is a low-level aerial target range used by the U.S. Air National Guard. Its closest boundary to the plant is 7.5 miles. Bombs, rockets, and 20-millimeter guns are used in the target range. The bombs are dummies that give off a flash, but no explosive charge. The rockets do not have explosive charges, only a propellant to deliver the rocket on target, and shells used in the 20-millimeter guns have solid heads without explosives (Reference 1). The likelihood of any of these hazards having any impact on Oyster Creek is judged to be extremely low.

5.3.5 Nearby Transportation Hazard Evaluation

The nearest transportation route to the station is U.S. Route 9, which is located approximately 0.25 miles east of the reactor building. In 1981, Route 9 was not heavily used for shipping in the locality. There were no industries in close proximity to the plant site that were expected to use or store large amounts of explosive or hazardous material. Additionally, Route 9 is a local road with many traffic lights and low speed limits, especially where it passes through towns. Through traffic generally used the Garden State Parkway, a limited access toll road that runs parallel to Route 9. The parkway is about 1.25 miles west of the plant. The separation distance between the highway and the plant exceeds the minimum distance criteria given in Regulatory Guide 1.91 (Reference 9) for truck-size shipments of the explosive materials. Therefore, in a letter dated February 4, 1982, the NRC staff concluded that the transportation of hazardous materials on U.S. Route 9 posed no significant hazard to the plant (Reference 1).

A probabilistic Safety Analysis performed for Oyster Creek in 1979 (Reference 10) estimated the probability of explosions at a mean frequency of 1.5×10^{-8} along Route 9 based on NRC Regulatory Guide 1.91 and information received from New Jersey Department of Highways at that time.

The NRC staff concluded in 1983 in IPSAR (Reference 11) that: "Shipments of explosives are normally made over the Garden State Parkway, which runs parallel to Route 9. In addition, New Jersey law allows only 25,000 lb of TNT to be trucked instead of the 50,000-lb TNT equivalent permitted by Regulatory Guide 1.91. On the basis of the existing State law and the low probability of explosive shipments on Route 9, the staff concludes that the Oyster Creek site meets the intent of Regulatory Guide 1.91 and backfitting is not recommended."

However, in January 1991, the staff issued a Safety Evaluation Report (SER) related to the full-term operating license for Oyster Creek Nuclear Generating Station (Reference 1). One of the issues identified by the staff in Section 2.2 of the SER (NUREG-1382) refers to potential shipments of hazardous materials near the plant. Specifically, the staff had found that current estimates of truck traffic on the nearby U.S. Route 9 were significantly greater than what was estimated previously, at the time of the application for a Provisional Operating License. The staff noted in the SER that the current truck traffic rate on Route 9 had the potential for exceeding the screening criterion given in Regulatory Guide 1.78 (Reference 12). In view of the above, the staff had obtained from GPUN a commitment to assess the transportation on the route and submit the results to the staff (Reference 13).

In May 1992, GPUN met with the staff to describe their assessment of the issue and the means for addressing the potential hazards. As indicated by GPUN, specific shipping data for hazardous materials are not maintained by State or Federal transportation authorities. GPUN corresponded with the New Jersey Department of Transportation, New Jersey Office of Emergency Management, New Jersey Emergency Response Commission, New Jersey State Police Hazardous Material Transportation Unit and New Jersey Department of Environmental Protection but found out that data on frequency of hazardous material shipments on Route 9 in the vicinity of OCNGS is not available (References 14 and 15).

The staff indicated that in the absence of supporting data the potential for frequent shipment of hazardous materials cannot be dismissed. As an alternative, the staff indicated that a reasonable safeguard against traffic accidents involving release of hazardous materials is to provide early warning to the plant. Specifically, early notification to the control room operators would provide sufficient time to implement emergency procedures relating to the potential effects of the hazardous material (e.g., safeguards against airborne toxic or flammable materials).

In response, GPUN requested and obtained from the New Jersey State police a commitment (New Jersey State Police interoffice memo, dated April 13, 1992) regarding hazardous material accident notification. Specifically, current New Jersey State police policy requires their duty officer or hazardous material emergency response personnel to notify promptly the Oyster Creek Group Shift Supervisor or the Control Room in the event of a transportation accident involving the release of airborne hazardous material occurring in Lacey or Ocean Townships.

The staff stated in an SER dated April 6, 1993, (Reference 13) that "In view of the above, and the licensee's Procedure 2000-ABN-3200.33, "Toxic Material/Flammable Gas Release - No Radiation Involved," the staff finds that the risk due to hazardous truck shipments near the OCNGS is acceptably low and the staff considers this issue resolved."

NUREG-0737, "Clarification of TMI Action Plan Requirements," Task Action Plan Item III.D.3.4, "Control Room Habitability," requires that the operators in the control room be adequately protected against the effects of accidental releases of toxic and radioactive gases.

By a confirmatory order dated March 14, 1983, GPUN was required to have NUREG-0737, Item III.D.3.4, fully implemented at the Oyster Creek station before the restart from the Cycle 11 refueling (Cycle 11R) outage. Technical Specifications (TS) related to control room habitability were part of the NUREG-0737 TS requested by the staff in Generic Letter (GL) 83-36, "NUREG-0737 Technical Specifications," dated November 1, 1983. In its letter dated November 22, 1985, the NRC staff evaluated the GPUN response to GL 83-36. By TS Amendment 105 dated July 15, 1986, GPUN was granted a postponement of the full implementation until the Cycle 12 refueling outage, provided interim system upgrades and accident analyses were completed.

Two items - performance of a single-failure analysis of the control room ventilation system and provision of remedial measures, and an assessment of existing diesel generator capability to provide backup power to the control room ventilation system - were postponed. By letter dated April 17, 1989, GPUN indicated that these items had been implemented on March 8, 1989.

Additional TS changes to address the items in GL 83-36 are included in POL Amendment 115, dated March 31, 1987. In the SER accompanying this amendment, the staff identified two GL 83-36 TS items that remain open. These are control room maximum temperature and plant shutdown if the control room heating, ventilation, and air conditioning (HVAC) system (except the dampers) is inoperable in regard to air inflow or control room temperature for more than 7 days.

In a TS change request dated October 18, 1989, as supplemented on February 21, 1990, GPUN addressed these TS open items and other items related to control room habitability. In this submittal GPUN also described modifications that had been made to the Oyster Creek control room HVAC system. With the issuance of POL Amendment 139 dated May 29, 1990, and its accompanying SER, the staff found the licensee's provisions acceptable to resolve this issue (Reference 1)

5.3.6 Conclusions

Based on the information provided in the discussions in previous sections none of the nearby hazards to OCNGS are considered significant from a severe accident likelihood standpoint. The frequency of such hazards is concluded to be acceptably low and all previously defined issues are resolved. On that basis transportation and nearby hazards have been screened from further consideration in the IPEEE.

5.3.7 References

1. Letter, January 29, 1991, from J. F. Stolz (NRC) to J. J. Barton (GPU Nuclear), Subject: Issuance of Safety Evaluation Report Relating to the Full-Term Operating License for Oyster Creek Nuclear Generating Station.

2. Letter, February 4, 1982, from D. M. Crutchfield (NRC) to P. B. Fiedler (GPU Nuclear), Subject: SEP Topics II-1.A, Exclusion Area Authority and Control; II-1B, Population Distribution; II-1C, Potential Hazards due to Nearby Transportation, Institutional, Industrial and Military Facilities - Oyster Creek.

3. GPU Nuclear Internal Memorandum, March 29, 1993, from D. J. Distel (Licensing) to Distribution, Subject: TMI-1 and OCNGS Design Basis for External Hazards, Memo No. C320-93-1060.

4. GPU Nuclear Technical Functions Safety/Environmental Determination and 50.59 Review, June 13, 1990, Safety Evaluation No. 00126-001, Subject: Installation of Natural Gas Pipelines.

5. USNRC NUREG-0014, Safety Evaluation Report, Hartsville Nuclear plants A1, A2, B1, B2.

6. Mechanic's Research Inc., Nuclear Power Plant Risks from a Natural Gas Pipeline, a Research Study Performed for TVA, August 1974.

7. Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 2.2, Update 7, December 1992.

8. Integrated Plant Safety Assessment Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, NUREG 0822, Supplement No. 1, U.S. Nuclear Regulatory Commission, July 1988.

9. USNRC Regulatory Guide 1.91, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants, Revision 1, February 1978.

10. Oyster Creek Probabilistic Safety Analysis, PLG-0100, August 1979.

11. Integrated Plant Safety Assessment Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, NUREG 0822, Final Report, U.S. Nuclear Regulatory Commission, January 1983.

12. USNRC Regulatory Guide 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, June 1974.

13. Letter, April 6, 1993, from A. W. Dromerick (NRC) to J. J. Barton (GPU Nuclear), Subject: Issuance of Amendment (TAC No. M81848).

14. Letter, December 24, 1991, from J. J. Barton (GPU Nuclear) to U. S. Nuclear Regulatory Commission, Subject: Oyster Creek Nuclear Generating Station (OCNGS), Docket No. 50-219, Request to Extend the Duration of the Operating License to Forty Years from the Date of Issuance of the Full Power License.

15. Letter, June 3, 1992, from J. J. Barton (GPU Nuclear) to U. S. Nuclear Regulatory Commission, Subject: Oyster Creek Nuclear Generating Station (OCNGS), Docket No. 50-219, Potential Hazards or Changes in Potential Hazards Due to Transportation.

16. GPU Nuclear System Design Description, Division I Hydrogen Water Chemistry System, Oyster Creek Nuclear Generating Station, SDD-OC-567A, Revision 5, September 1987.

17. GPU Nuclear Technical Functions Safety/Environmental Determination and 50.59 Review, March 2, 1990, Safety Evaluation No. 402840-001, Subject: Hydrogen Water Chemistry System and Generator Cooling System.

18. EPRI Special Report, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision, EPRI NP-5283-SR-A, September 1987.

SECTION 6

OYSTER CREEK IPEEE

PEER REVIEW

6.1 Introduction

Generic Letter 88-20, Supplement 4 "Individual Plant Examination of External Event (IPEEE) for Severe Accident Vulnerabilities" requested each licensee to include a peer review of the IPEEE analysis to ensure the accuracy of the documentation packages and to validate both the IPEEE process and results. Generic Letter 88-20, Supplement 4 requests the IPEEE to be peer reviewed and a description of the peer review process along with the results of the review to be included in the IPEEE submittal report. This section satisfies this request by providing an overview of the peer review process used in the development of the Oyster Creek IPEEE, a list of the participants and describes all significant review comments and their disposition.

6.2 Peer Review Process

A multi-disciplinary and multi-organizational reviews of the drafts of each of the analysis sections (Seismic, Fire and Other External Events) was performed. Each of the analyses were review by GPU Nuclear managers and engineers whose expertise was appropriate for the analyses reviewed. In addition, these multi-disciplinary and multi-organizational review teams consisted of managers and engineers, to the degree possible, who were not involved in the process of developing the analysis thereby ensuring a necessary degree of independence. The individuals were selected on the basis of their knowledge of Oyster Creek plant design, operation and maintenance. Although prior knowledge of PRA techniques was not a prerequisite some members had prior exposure to PRA approaches.

GPU Nuclear Reviewer	Title	IPEEE Section Reviewed
C. D. Adams	Risk Analysis Engineer	Seismic
F. Barbieri	Engineer, Engineering & Design	Fire Analysis
K. Canavan	Risk Analysis Engineer	Other External Events
R. Daley	Fire Protection Engineer	Fire Analysis
D. Distel	Licensing Engineer	Other External Events
L. C. Lanese	Manager Special Projects	Seismic, Fire Analysis and Other External Events
R. Lopez	Contractor, A-46	Seismic (relays only)
R. Panicker	Electrical Power and Instrumentation, Manager	Seismic (relays only)
S. C. Ramdeen	Engineer, Engineer Mechanics	Seismic
T. Trettel	Fire Protection Engineer	Fire Analysis
K. Whitmore	Civil/Structure Manager	Seismic

6.3 Peer Review Comments

The peer review comments on the various sections of the Oyster Creek IPEEE are presented below in individual report sub-sections associated with each of the analyses.

6.3.1 Seismic Analysis Review Comments

The purpose of the independent review of the seismic portion of the Oyster Creek IPEEE was to ensure that the analysis was reasonably accurate and reflects the design and operation of the Oyster Creek plant. The review was performed by individuals from the Risk Analysis Section (seismic risk model), Electrical Power and Instrumentation (Relay Chatter Issues), Seismic and Structural Engineers (structure and component fragilities). Member of the independent review team included:

Charles Adams, Risk Analysis Engineer
Louis Lanese, Manager Risk Analysis
Swadish Ramdeen, Engineering Mechanics
Rene Lopez, Contractor, A-46 Program
Ravi Panicker, Electric Power & Instrumentation, Manager
Kenneth Whitmore, Civil/Structural Manager

The review comments as well as the disposition of each is provided below:

1. **Comment:** A table or discussion of the seismic split fraction values (MFF) and new fragility values used for the LLNL curves would be helpful. This would provide additional insights since the LLNL curves are not the same range as the EPRI curve initiating events.

   Resolution: *Comment accepted. A table of the LLNL seismic split fractions values (MFF) is added to the LLNL evaluation.*

2. **Comment:** Why is there no seismic impact for the 'C' train of batteries?

   Resolution: *There is no seismic impact modeled in the "C" 125 VDC distribution for two reasons. The first and primary reason is that seismic capacity walkdowns indicated that the ventilation system was rugged. The second reason is that the loads in the 125 VDC Battery room "C" consists of the battery bank only. This loads is not consider significant enough to require ventilation for heat removal and the short term hydrogen production is not expected to results in explosive concentrations.*

3. **Comment:** Page 3-92 states that the recovery requirements of the Oyster Creek logic structure are intact from the IPE model. Shouldn't the recovery analysis be modified (error rates increased) based on seismic damage to equipment and the possible interference reducing the accessibility for

operators?

Resolution: *Comment accepted. Additional clarification is provided in the paragraphs which discuss the use of the Oyster Creek PRA logic to indicate that recovery is not modeled in the seismic model with the exception of offsite power. All other recoveries are guaranteed failed since they may not be possible following a seismically initiated event.*

In the case of offsite power recovery, only use of the near site combustion turbines to restore offsite power is modeled. Walkdowns and seismic capacities were evaluated for the combustion turbine, fuel oil tank, and station blackout transformer. In addition, the station blackout controls in the 4160 VAC A/B switchgear rooms are expected to remain accessible either through the turbine building or via an outside route in which access through the turbine building southwest doors is possible.

4. **Comment:** How are the other assumed operator actions (non recovery) modified to account for the seismic event causing accessibility problems?

Resolution: *A sensitivity case is performed in response to the above comment. The non-recovery human actions are reviewed and selected based on the action occurring outside the control room and importance to core damage. These actions are then increased by a factor of 10. The results and additional details of this sensitivity case are presented in report Section 3.1.5.5, Oyster Creek Seismic Sensitivity Studies.*

5. **Comment:** Page 3-139 - Decay heat removal via CS/ESW and through the hardened vent: What is the available makeup source or sources of water to keep the core cool and covered?

Resolution: *In the case of decay heat removal paths of containment spray/emergency service water and hardened vent following a seismic event the method of inventory control is the core spray system. Additional clarification is provided. Comment accepted.*

6. **Comment:** Page 3-123 - More discussion on the significant plant damage states (NIHx, MIAx, and PIEx) would be beneficial to the understanding of the results.

Resolution: *Comment accepted. Addition discussion is added as to the significance of plant damage states listed above.*

7. **Comment:** What is the torus fragility? No discussion was found to determine if it was screened and the reason for screening.

Resolution: *The torus screens based on ruggedness. The torus was included in the seismic capacity walkdowns and was determined to screen. In addition, the torus screened from the A-46 program based on analysis and modifications made in the MARK I Containment System Evaluation Program.*

8. **Comment:** Are the source terms similar or significantly different from the Oyster Creek IPE? (see NUREG 1407 page 6).

Resolution: *The source terms which result from seismically initiated events are lower than those associated with the IPE. This is due to the fact that the seismic core damage frequency is dominated with plant damage states in which containment is intact (>98%) as compared with 83% for the IPE. No seismic failures of containment were identified. This fact in conjunction with no identified seismic LOCAs result in lower contributions of containment bypass events.*

9. **Comment:** Page 3-107 - The rule for EC2 does not agree with the comment for EC2, DY in the comment appears to be wrong.

Resolution: *Comment accepted. The "DY" in the EC2 rule should be "DW".*

10. **Comment:** Table 5-3 should be Table 3-7 starting on page 3-79.

Resolution: *Table is renamed to "Table 3-7" and added to the Table of Contents. Comment accepted.*

11. **Comment:** The diesel driven fire pump has a high capacity at Oyster Creek, yet a similar design at TMI was found to have a low capacity. Please explain the differences between the pump designs.

Resolution: *The fuel oil tanks for the diesel driven fire pumps at TMI were determined not to screen. Although the design of the Oyster Creek Fire Protection system is similar, differences in the size of the tanks (Oyster Creek fire protection diesel fuel oil tanks are smaller), aspect ratio (tanks are lower to the ground) and stitch welding of the tank to its support frame provide a more rugged construction and higher capacity.*

12. **Comment:** Page 3-58 - The second paragraph states that the condensate storage tank is not in the scope of this evaluation. However, a fragility value was calculated for the condensate storage tank.

Resolution: *The statement that the condensate storage tank is not in the scope of this evaluation is given in the "Seismic Fragilities of Civil Structures" report sub-*

section. This is true. The condensate storage tank was not evaluated as a "civil structure". Subsequent evaluations of component fragilities produced a seismic capacity of this tank which is modeled in the seismic logic structure. Comment accepted and additional clarification added.

13. **Comment:** Page 3-95 - Change in the second paragraph "Figure 3-5" to "Figure 3-6".

    Resolution: *Comment accepted.*

14. **Comment:** Page 3-96 - Change in the second paragraph "Table 3.1-8 through 3.1-11" to "Table 3.1-9 through 3.1-12".

    Resolution: *Comment accepted. "Table 3.1-8 through 3.1-11" changed to "Table 3.1-9 through 3.1-12".*

15. **Comment:** Page 3-104 - Last paragraph, last sentence, change "core spray to containment spray", also change the rules for CX2, CX3, CX4 (similar problem).

    Resolution: *Comment accepted. Editorial comments incorporated.*

16. **Comment:** Page 3-105 - In the CZ top event discussion, in the last sentence, change "containment spray" to "core spray".

    Resolution: *Comment accepted and incorporated.*

17. **Comment:** Page 3-116 - The rules for SBGT shows possible successful operation of the system, but on page 3-87 it is stated that the SBGT system is always failed.

    Resolution: *The split fraction assignment rule for standby gas treatment system (SBGTS) guaranteed failure was inadvertently deleted from the table. The logic model does indeed reflect the guaranteed failure of the SBGTS. Comment accepted and the table revised.*

6.3.2 Fire Analysis Review Comments

The purpose of the independent review of the fire portion of the Oyster Creek IPEEE was to ensure that the analysis was reasonably accurate and reflects the design and operation of the Oyster Creek plant. The review was performed in a largely collegial process wherein in-house Risk Analysis personnel made presentations on the content of the study, talking through the various portions of the fire evaluation process. Review group members reviewed these portions before the presentations and offered comments orally in a group setting. Many comments during these meetings elicited further comments from other members of the group, resulting in significant amounts of discussion. The recorded comments from these meetings (see below) represent the consensus of the group on a particular issue. The formal portion of the review process consisted of a series of 3 meetings that were held in January, 1995. These meetings were held at the Oyster Creek site. The following personnel were in attendance for each of these meetings:

Fred Barbieri	- Mechanical Components, Engineer	
Ray Daley	- Fire Protection, Engineer	
Tim Trettel	- Fire Protection, Engineer	
Lou Lanese	- Risk Analysis, Manager (Acting)	(January 20 and 27 only)
Ken Canavan	- Risk Analysis, Engineer	

Group comments were developed during the review meetings and represent the consensus of the group. These comments are collected and presented by meeting date below.

Independent Review Group Meeting of January 20, 1995

1. **Comment:** Report organization is confusing and does not seem to follow the analysis flow process.

   Resolution: *The report structure and text were chosen to reflect the suggested format of NUREG-1407. The assignment of individual report sections in this suggested format was reviewed against the flow process and modified to provide a more organized flow and report structure.*

2. **Comment:** The report does not present any sensitivity studies. Are any planned? This is of particular importance since Halon fire protection systems may be removed or replaced in the future.

   Resolution: *Due to the nature of a screening evaluation, individual sensitivity studies are not appropriate since the fire areas and zones are treated various levels of detail. No sensitivity studies are presently planned. In the future, sensitivity evaluations on the effects of Halon removal could be performed.*

3. **Comment:** The fire zone which contains the diesel generator fuel oil tank is not presented separately.

   Resolution: *The diesel generator fuel oil tank is analyzed (FS-FA-16) separately. This fire area screens in the initial upper bound estimate of core damage frequency.*

4. **Comment:** The operability of the fire protection system is no longer a Technical Specification requirement. It is now a station technical requirement. Does this affect the screening of the fire pond pump house?

Are there any other considerations, given the fact that station operation may continue for 7 days without operable fire protection?

Resolution: *Plant trip would not be expected to occur in response to a fire in the fire pond pump house, regardless of whether system operation is a Technical Specification or station technical requirement. Therefore, this concern does not impact the screening of the fire pond pump house. While station operation may continue without operable fire protection, this form of system maintenance is accounted for in the unavailability of plant sprinklers and water suppression systems.*

5. **Comment:** Table 4.1-5, Fire Protection Panels Contribution to Fire Area Frequency, indicates that a fire protection panel is inside No. 2 Diesel Generator Room. This panel is located on the outside wall of the Diesel Generator Building and should not be included n the development of a fire initiating event frequency.

Resolution: *The fire ignition frequency for this panel was reassigned to area FS-FA-17.*

6. **Comment:** Table 4.1-5, Fire Protection Panels Contribution to Fire Area Frequency, indicates that a fire protection panel is located in fire zone RB-FZ-1E. Verify that this panel is located in RB-FZ-1D and adjust the table.

Resolution: *Panel location was confirmed and Table 4.1-5 has been modified accordingly.*

7. **Comment:** Table 4.1-6, Transformer Contributions to Fire Area Frequency, seems to include the contributions of dry transformers, as well as oil cooled units. Should this be the case?

Resolution: *The EPRI fire events database used to generate the fire ignition frequencies for the FIVE methodology does not distinguish between dry or oil cooled transformers. Therefore, the Oyster Creek fire IPEEE does not distinguish between these types of components.*

8. **Comment:** Table 4.1-7, Non-Qualified Cable and Non Qualified and Qualified Junction Box Contribution to Fire Area Frequency, differentiates between qualified cable and non-qualified cable at Oyster Creek. Please describe in detail the rationale for differentiating and the method used to determine the qualification of cable at Oyster Creek.

Resolution: *Types of cable were initially differentiated based on interpretation of the combustible loading assignments for the various Oyster Creek fire zones. This differentiation has since been removed and Table 4.1-7 has been modified accordingly.*

Independent Review Group Meeting of January 27, 1995

9. **Comment:** Table 4.1-7, Non-Qualified Cable and Non-Qualified and Qualified Junction Box Contribution to Fire Area Frequency, reports combustible loading with too many significant figures, which indicates a precision which does not exist. Consider rounding off.

   Resolution: *Entries were rounded to the nearest 1,000 BTU.*

10. **Comment:** The "Hydrogen Tanks" discussion on page 4.1-12 indicates that the main generator hydrogen tanks are located in the yard area, MT-FA-12. This is not the case. These tanks are located outside the protected area and should be removed from MT-FA-12 fire frequency calculation.

    Resolution: *The fire ignition frequency contribution due to hydrogen tanks was removed for MT-FA-12.*

11. **Comment:** The "Other Hydrogen Fires" discussion on page 4.1-12 indicates the fire frequency is allocated to fire zone OB-FZ-10B. No industrial gases are located in OB-FA-10B and the most probable location of other hydrogen fires is either the main generator area or the feedwater hydrogen injection manifold.

    Resolution: *The fire ignition frequency due to "Other Hydrogen Fires" is reassigned to feedwater hydrogen injection manifold, as described above.*

12. **Comment:** The "Air Compressors" discussion on page 4.1-13 indicates only four air compressors on the Oyster Creek site. There are many compressors of various sizes located on the site. Verify the number of compressors.

    Resolution: *The current air compressor fire ignition frequency was assigned to the indicated units based on size and regular operation. The EPRI FIVE Fire Ignition Frequency Methodology does not discriminate between the air compressor size and service. Judgement was used in the assignment of fire ignition frequency to the indicated compressors.*

13. **Comment:** Table 4.1-7, Ventilation Subsystem Fire Frequency Contribution, fire frequency allocations are incorrect. There are no normally running ventilation fans located in the Reactor Building or in fire zone TB-FZ-11B. In addition, the fans associated with fire zone OB-FZ-8A/B are located outside. Fans associated with the 4160 VAC Switchgear rooms are not identified.

Resolution: *In fact, a case can be made for the removal of all ventilation fans from the fire initiating event frequency calculation. Judgement is used in the assignment of the frequency based a review of the EPRI Five Events Database. The contributions of all fans except the Reactor Building corner room fans remain.*

14. **Comment:** The "Laundry Dryers" discussion on page 4.1-13 indicates the presence of laundry dryers in fire zone OB-FZ-10B. These dryers have been moved to the new radwaste building and a shed located outside the new maintenance building.

Resolution: *Fire ignition frequency for OB-FZ-10B was adjusted as described. The other areas were removed from consideration based on qualitative factors.*

15. **Comment:** Table 4.1-8, Transient Combustible Contribution to Fire Area Frequency, does not agree with the current control of combustibles procedure.

Resolution: *Table 4.1-8, Transient Combustible Contribution to Fire Area Frequency, was revised to allow up to a pint of acetone (10,000 BTU) to be in any plant area as a transient combustible.*

16. **Comment:** Table 4.2-3, Impact Matrix Notes, note 1, does not indicate that these cables are associated with the diesel fire pump manual start.

Resolution: *The impact matrix notes was revised as described.*

<u>Independent Review Group Meeting of January 31, 1995</u>

17. **Comment:** Table 4.2-2, Fire Initiating Event Impact Table, should Isolation Condensers (top event IC) be impacted in fire zone TB-FZ-11C? In fire zone TB-FZ-11D, should 125 VDC distribution bus B, USS 1B1, USS 1A1, 4160 VAC 1C and 1D be impacted?

Resolution: *Table 4.2-2, Fire Initiating Event Impact Table, was revised to incorporate the Isolation Condenser "B" (Top Event IC) isolation valve control cables impact as a result of a fire in fire zone TB-FZ-11C.*

Table 4.2-2, Fire Initiating Event Impact Table, was revised to incorporate the impacts of fire events in the fire zone TB-FZ-11D. In addition, the impact matrix was reviewed for all other plant fire areas and zones to verify the impacts in the fire risk model and adjusted accordingly.

18. **Comment:** Table 4.6-1, Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation, fire zone OB-FZ-6A does not list the ADS impact which appears in Table 4.2-2.

Resolution: *The EMRV failure mode modeled in Table 4.6-1, Oyster Creek Plant Fire Areas and Zones for Quantitative Evaluation, is failure to open to relieve pressure, which subsumes the failure to open for ADS actuation. That is, the EMRVs are already failed as a result of failure to open to relieve pressure.*

19. **Comment:** It is not clear which fire zones have been screened from further consideration (and why) when reviewing Section 4.6. Consider moving Table 4.4-3 to the introduction of Section 4.6.

Resolution: *The text of Section 4.6 has been revised to provide additional clarity on the why and at what point fire areas and zones have been screened from further consideration. Additional text has also been added to Section 4.0 which provides an overview of the Fire Individual Plant Examination methodology.*

20. **Comment:** On page 4.6-6, fire zone RB-FZ-1F does not contain any automatic suppression. Fire suppression in this fire zone is all manual.

Resolution: *The text has been revised to reflect manual fire suppression.*

21. **Comment:** Review and revise assumptions in the detailed evaluation of fire zones TB-FZ-11C and TB-FZ-11F associated with the 125 VDC cables located in conduits.

Resolution: *The assumptions regarding the protection provided by conduits have been reviewed. The text and the evaluation have been revised to eliminate these assumptions.*

22. Comment: The assumption that a fire in fire area MT-FZ-12 results in a loss of offsite power may be overly conservative. Consider revising initiator impact to reactor trip event with loss of offsite power following failure of transformers.

Resolution: *It is acknowledged that the screening methodology used in FIVE and in this analysis is conservative. This evaluation is based on the premise that, if an area can be screened from consideration under this level of conservative assumptions, no further evaluation is warranted and resources can then be directed to the evaluation of other, unscreened areas. The fire zone MT-FZ-12 screens from further consideration in the revised estimate of upper bound core damage frequency.*

23. Comment: In Table 4.6-2, Revised Estimation of Upper Bound Core Damage Frequency, the revised CDF for fire area CW-FA-14 should be $5.57x10^{-7}$.

Resolution: *Table 4.6-2, Revised Estimation of Upper Bound Core Damage Frequency, was revised for all fire areas due to modification of fire ignition frequencies based on previous comments in this report section.*

24. Comment: For the detailed evaluation of fire zone OB-FZ-5, Control Room, consider manual suppression of fires instead of automatic actuation of fire suppression systems. This is the more likely scenario since the control room is continuously manned.

Resolution: *Again, the screening evaluation used in FIVE is intended to provide a conservative evaluation of fire-related core damage. The intention was to show that even without operator response, in a continuously manned area, this fire zone still screens from further consideration. If manual suppression was modeled it is likely that the same results would be achieved. Due to the uncertain future of Halon as a fire suppressant this evaluation will most likely be performed.*

25. Comment: On page 4.6-28, "C" Battery Room, reword the description to indicate that a carbon dioxide fire extinguisher is located outside the fire area for manual fire suppression.

Resolution: *The report text was revised as noted.*

26. Comment: Fire area TB-FA-26 ("C" Battery Room), does not screen in the detailed quantitative evaluation. This does not seem consistent with the screening of the A/B battery room as well as the other area which did screen in the detailed evaluation. Review the assumptions with regard to the application of fire severity factor in this case.

Resolution: *The evaluation of TB-FA-26 ("C" Battery Room) was revised. The area now screens from further consideration. It should be noted that the initial evaluation gave an "unscreened" value that was very close to the screening value of $1x10^{-7}$.*

27. Comment: Report Section 4.2.3, Quantitative Evaluation of Fire Barrier Effectiveness, is very subjective and, as such, adds little new information to the analysis. In the absence of less subjective inputs, consider the removal of the entire report section.

Resolution: *Section 4.2.3 has been removed from the evaluation.*

6.3.3 Other External Events Review Comments

The purpose of the independent review of the Other External Events portion of the Oyster Creek IPEEE was to ensure that the analysis was reasonably accurate and reflects the design and operation of the Oyster Creek plant. The review was performed using in-house Risk Analysis personnel not involved in the development of the analysis (K. Canavan) and a licensing engineer cognizant of the results of the FSAR, IPSAR and other licensing issues related to other potential external hazards. Comments were elicited and collected and are provided below.

1. Comment: Section 5.1.1 states that hurricane winds are bounded by tornado winds because a tornado has a greater likelihood of exceeding the wind speed thresholds. However, there is no discussion of the frequency of occurrence of hurricanes with wind speeds exceeding 168 mph, which could then be compared to the frequency of tornado occurrence.

Resolution: *Figure 7 shows that straight winds (hurricanes) probabilities are much lower than tornadoes in the upper wind speed ranges. It should be noted that due to extreme low probability of hurricane wind speeds exceeding upper 100 MPH, one has to extrapolate such data for comparison with tornadoes at OC from Figure 7.*

2. Comment: In Section 5.1.2.2, Provide the basis for not using data beyond 1978 for tornado frequency and intensity, especially in light of tornadoes identified in the latter part of this time span.

Resolution: *The data used here for tornado frequency is the latest available which was prepared for NRC and spans over a 29-year period. Even though the number of tornadoes is higher in the last 6 years of this time span, this is not due to natural climatic variability but because of encouragement by* *government and various agencies to report and document tornadoes (See Reference 3 for this section). It is suspected that most of these additional reportings are of tornadoes of less severity and the probability of tornado wind speeds exceeding the 168 MPH threshold used here, conservatively estimated at $5x10^{-7}$ from Figure 7, remains unchanged.*

3. **Comment:** Discuss the probabilities associated with the probable maximum flood (PMF) level, and provide basis for not considering the probability of floods beyond the PMF in Section 5.2.1.

   Resolution: *NUREG 1407 recommends the licensees to compare their plants against 1975 SRP criteria. If the comparison indicates that the plant conforms to the 1975 SRP criteria, it is judged that the contribution from that hazard to core damage frequency is less than 10^{-6} per year and the IPEEE screening criterion is met. This is the screening approached used here and probabilities were not discussed.*

4. **Comment:** See suggested word changes on Pages 5.1-4 and 5.1-5 for Section 5.1.3.

   Resolution: *Editorial changes were incorporated.*

5. **Comment:** No Comments on Section 5.3.

   Resolution: *Comment Accepted.*

SECTION 7

PLANT IMPROVEMENTS

AND

UNIQUE SAFETY FEATURES

7.0 Plant Improvements and Unique Safety Features

This report section provides a summary of potential plant improvements as well as a description of the safety features at the Oyster Creek Nuclear Generating Station.

7.1 Plant Improvements

No vulnerabilities were identified in any of the analysis sections of the Oyster Creek IPEEE. However, in some instances potential enhancements could result in increased safety margins. These are listed in report sub-sections below:

Seismic Analysis

The Oyster Creek Seismic IPEEE did not identify any plant vulnerabilities which result from seismically initiated events. However, two potential plant modifications to the combustion turbines that can improve the seismic margin were identified. The effect of these modifications was not included in the Oyster Creek seismic fragility analysis or logic model.

1) Ensure all bolts on the Forked River Combustion Turbine fin-fan coolers are installed and torqued properly. Although adequate capacity for withstanding the range of potential ground motions exist, addition margin could be obtained by ensuring all bolts are installed and torqued properly.

2) Consider the addition of battery spacers in the Combustion Turbine battery compartments. Although the current battery spacing is sufficient to prevent battery failure due to interactions, additional seismic margin could result with the use of battery spacers.

Fire Analysis

The Oyster Creek Fire Analysis did not identify any plant vulnerabilities due to fire initiated events. However, several items were noted in the performance of the fire analysis and seismic/fire interaction walkdowns. These are:

1) Continued transient combustible control and good housekeeping are essential elements of a successful fire protection program. Plant walkdowns done in support of the Fire Analysis confirmed good combustible control and housekeeping. Continue good housekeeping practices and continued attention to the control of transient combustibles.

2) Consider upgrading the anchorage of the high pressure CO_2 system in the Turbine Building. These high pressure cylinders could potentially become missiles following a seismic event. The result is the potential loss of turbine bearing no. 10 and turbine generator exciter fire suppression.

3) Consider additional support of a small oil filter of the Turbine Generator Hydrogen Seal Oil Unit. This unit is supported only by a vertical stanchion, and no lateral support is provided. The small diameter piping to and from the filter forms an approximately 8-foot cantilever which was found to be flexible.

4) Review basis for anchorage of the Arrowhead Demineralizer trailer. During the seismic/fire interaction walkdowns anchorage chain was not attached to embedded eyehook. During a seismic event interaction of trailer with the station blackout transformer fire suppression system could result in inadvertent suppression system actuation.

5) Consider anchoring the high pressure generator purge CO_2 rack outside the turbine building. Following a seismic event these racks could overturn and result in missiles. Low pressure CO_2 tank could be disabled by these missiles due to its close proximity to the high pressure CO_2 storage racks.

6) Consider replacement of the fire protection drop-weight actuated deluge valves. Following seismic event, actuation of these valves could result in diversion of fire suppressant from actual fire events or fire suppressant spray effects on safety related equipment. Although walkdowns verified that electrical panels and safety equipment are generally well sealed or spray protected, new deluge valves with less potential for seismic actuation, would provide additional margin from fire suppressant spray effects and flow diversion.

7) Consider operator and fire brigade training on fire analysis scenarios from the most significant Oyster Creek fire areas (fire areas for which a detailed evaluation is performed). Particular emphasis should be placed on the unscreened fire areas (Cable Spreading Room and "A" 480 VAC Switchgear Room).

Other Events

No vulnerabilities were identified in the examination of other external events. No actions or modifications are warranted or planned.

7.2 Summary Plant Description

The Oyster Creek Nuclear Generating Station (OCNGS) is a single unit General Electric boiling water reactor (BWR-2) of 620 MWe capacity housed in a Mark I containment. The site is located near the Atlantic Ocean about nine (9) miles south of Toms River, New Jersey. Condenser cooling water and cooling water for many plant auxiliaries is drawn from Barnegat Bay through a canal following the south branch of Forked River and discharged through another canal following Oyster Creek back to the Bay. Some important system design features of the Oyster Creek plant are described below:

SUMMARY OF DESIGN FEATURES

COOLANT INJECTION SYSTEMS	High Pressure Systems	Main feedwater provides full capacity makeup with three electric driven pumps supplied from non-IE sources.
		Control Rod Drive provides up to 150 GPM via two (2) electric driven pumps through a manually opened bypass valve.
	Depressurization Systems	Automatic Depressurization System (ADS) depressurizes to pressures at which low pressure systems can inject into the reactor vessel. Consists of five (5) electromatic relief valves (EMRVs) which are DC operated, no air needed.
		Two trains of Isolation Condensers each initiated by opening a single DC powered isolation valve.
	Low Pressure Systems	Core Spray provided by to (2) complete systems each with two (2) redundant booster and two (2) redundant main pumps powered from 1E sources.
		Condensate system provides makeup through the Feedwater System with three (3) electric driven pumps powered from non-1E sources.
		Fire protection water manually aligned through the Core Spray system. The fire header is supplied by three (3) electric driven pumps and two (2) diesel driven pumps.

CORE HEAT REMOVAL SYSTEMS	Main Condenser	Turbine Bypass valves that reject heat to the main condenser: nine (9) valves with a total capacity of 40% reactor power.
	Isolation Condensers	Isolation Condensers that reject heat directly to atmosphere: two (2) trains initiated by the opening of DC powered isolation valves. Makeup to the secondary side of the isolation condensers is provided by the condensate transfer system with two (2) electric pumps or the firewater system supplied by two (2) diesel driven pumps or three (3) electric driven pumps.
	EMRV Cooling	EMRV discharge to the suppression pool with cooling provided by the containment spray/emergency service water system.
	Shutdown Cooling	Shutdown cooling consists of three (3) trains, each with a pump and heat exchanger.
KEY SUPPORT SYSTEMS	DC Power	DC Power from two (2) station batteries with up to a 3 hour capacity. A third battery could be cross-tied to provide DC recovery in some scenarios.
	AC Power	Emergency AC power from two (2) diesel generators.
		An alternate AC source provided by a cross-connection to two combustion turbine generators on the adjacent Forked River site.
REACTIVITY CONTROL SYSTEMS	Normal	137 Hydraulicly positioned control rods.
	Backup	Standby Liquid Control System with two (2) parallel positive displacement pumps to inject enriched boron solution.

CONTAINMENT HEAT REMOVAL SYSTEMS	Containment Spray	Containment spray system with two (2) redundant pumps reject heat to the emergency service water system which has two (2) redundant pumps and two (2) redundant heat exchangers.
	Containment Vent	Hardened containment vent system (installed in 14R) which can be used to protect against overpressure in the event the normal containment cooling function is unavailable.

In summary, Oyster Creek has in addition to the normal heat rejection paths to the main condenser under post trip conditions, the plant is equipped with two redundant isolation condensers (ICs) which initiate independent of AC power in the event of reactor isolation. Multiple makeup sources, condensate transfer and fire protection water (supplied by diesel driven fire protection pumps) make this a very reliable long term means of removing decay heat. If the isolation condenser system is unavailable, DC powered (no air required) Electro-Matic relief valves (EMRVs) can be used to reject heat to the torus for extended periods without cooling. With torus cooling and an RPV injection source this heat rejection path can be maintained indefinitely. Even without cooling a hardened pipe vent can be used to protect containment from overpressure and is sized to remove sufficient decay heat and provided an RPV makeup source is available to prevent core damage. RPV makeup sources include two fully redundant core spray systems, feedwater, condensate and fire protection injection through the core spray system. This versatility provides numerous success paths for cooling the core, all of which have been incorporated into plant procedures.

The studies show that offsite and onsite power are important contributors to, and important mitigators of, core damage frequency. These contributors are ameliorated by a reasonable reliable onsite distribution system and an Alternate AC Source (combustion turbines located on the Forked River Site) as well as the DC powered initiated isolation condenser system.

SECTION 8

SUMMARY AND CONCLUSIONS

8.0 Summary and Conclusions

This section summarizes the systematic, plant-specific examination for external event vulnerabilities to severe accidents at the Oyster Creek Nuclear Generating Station. The results and conclusions presented here are based on the current Oyster Creek Model which has been modified to include the impacts from fire and seismic events.

The results of this study indicate a low risk of core damage from external events. Using the definition of vulnerability from Section 3.2 of the Oyster Creek IPE submittal report, no severe accident vulnerabilities were identified as a result of this analysis. Table 8-1 and Figure 8-1 identify the contributors to the estimated core damage frequency due to external events.

Oyster Creek Fire IPEEE

The Oyster Creek Fire IPEEE is a screening study which reviewed all areas of the plant for vulnerabilities due to fire events. No fire induced vulnerabilities were identified as a result of the analysis, therefore no recommendations for further plant modifications are warranted. The items listed in Section 7 of this report can potentially increase the safety margin with respect to fire initiated events however are classified as preliminary.

The two unscreened plant fire areas result in total core damage frequency of 7.7×10^{-6} per year due to internal fire events. This is the major contributor to external event risk; contributing 62.6%.

Oyster Creek Seismic Analysis

The Oyster Creek Seismic IPEEE uses a modified Oyster Creek IPE RISKMAN model to analyze for seismic event vulnerabilities. No seismic induced vulnerabilities are identified as a result of the analysis, therefore no modifications are warranted. The items listed in Section 7 of this report can potentially increase seismic margins however are classified as preliminary.

Seismic events result in a total calculated core damage frequency of 3.6×10^{-6} per year or 29.3% of the total external event estimated core damage frequency.

Other External Events

None of the other external events analyzed in this report (high winds, floods, transportation and nearby facility accidents) identified plant vulnerabilities and no modifications are identified or warranted. Only high winds are quantified. An upper bound value of 9.9×10^{-7} per year or 8.1% of the total external event estimated core damage frequency was calculated. All other external events are screened.

There are no external event specific contributors that contribute significantly to containment isolation failure or degradation. No modifications are identified or warranted.

Table 8-1
External Event Contributions to Oyster Creek
Calculated Core Damage Frequency

External Event	Core Damage Frequency	Percent of External Event CDF
SEISMIC EVENTS	**3.6E-06**	**29.3%**
FIRE EVENTS (Total) *	**7.7E-06**	**62.6%**
Cable Spreading Room Fire *	2.6E-06	21.1%
"A" 480 VAC Switchgear Room Fire *	5.1E-06	41.5%
OTHER EXTERNAL EVENTS (Total)	**9.9E-07**	**8.1%**
High Winds	9.9E-07	8.1%
External Floods	Screened	N/A
Transportation and Nearby Facility Accidents	Screened	N/A
TOTAL EXTERNAL EVENT CDF	**1.23E-05**	**100%**

* The Oyster Creek Fire IPEEE is a screening analysis with the results listed in the above table for unscreened fire events. The results are conservative.

Figure 8-1
Oyster Creek External Event
Core Damage Frequency Percentage