HOPE CREEK GENERATING STATION

Individual Plant Examination for External Events

Public Service Electric and Gas Company

July 1997

FOREWORD

The objectives of the Hope Creek IPEEE Project are:

- To develop an appreciation of severe accident behavior at the Hope Creek Generating Station.
- To develop an understanding of the most likely severe accident sequences that could occur at the Hope Creek Generating Station under full power operating conditions.
- To develop a qualitative understanding of the overall likelihood of core damage and fission product releases at the Hope Creek Generating Station under a variety of external events.
- If necessary, and where appropriate, to suggest evaluation of hardware or procedures that would help to prevent or mitigate severe accidents at the Hope Creek Generating Station and thereby reduce the overall likelihood of core damage and radioactive material releases.

The IPEEE is not a design basis document
and may not be used as such.

ACKNOWLEDGMENT

Although it was the policy of Public Service Electric and Gas Company to use its own employees throughout the Hope Creek IPEEE, technical support was supplied by various recognized experts and firms. The following are recognized for their efforts in developing the Hope Creek Generating Station IPEEE.

<u>Seismic Analysis</u>

The seismic plant review was performed by a number of contractors managed by PSE&G staff responsible for the technical adequacy of the analysis. The PSE&G project manager worked closely to coordinate site walkdowns, implement required analyses, and manage internal Project Review efforts. PSE&G performed the seismic system analysis. PSE&G's consultants for engineering activities in the seismic area were:

- EQE International, Inc. (EQE) - seismic walkdowns, fragility analysis, detritus risk evaluation, and probabilistic seismic response analysis.

- Halliburton NUS (Now SCIENTECH) - seismic walkdown and detritus risk evaluation.

- Woodward-Clyde Consultants - Soil system interactions including soil liquefaction and slope stability issues.

- SFA Independent Review of Seismic Analysis.

<u>Internal Fires</u>

The internal fires analysis was performed by PSE&G with significant support from Safety Factor Associates (SFA). The assessment of internal fire impacts on core damage frequency, for the various seismic damage states, was performed by PSE&G and SFA with project review by PSE&G and SCIENTECH.

<u>High Winds, Floods and other External Environments</u>

The assessment of "Other" external environments, i.e., severe winds, tornadoes, transportation and nearby facility accidents, etc. was performed by PSE&G with review by Scientech and SFA.

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LIST OF ACRONYMS

AB	Auxiliary Building
AC	Alternating Current
ACP	Alternating Current Power
ADS	Automatic Depressurization System
AFST	Auxiliary Feedwater Storage Tank
AFW	Auxiliary Feedwater
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOV	Air-Operated Valve
ASCE	American Society of Civil Engineers
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Scram
AUX	Auxiliary Building
BHEP	Basic Human Error Probability
BKR	Circuit Breaker
BOP	Balance of Plant
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CA	Chatter Acceptable
CACS	Containment Atmosphere Control System
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CET	Containment Event Tree
CFCU	Containment Fan Cooler Unit
CIS	Containment Isolation System
CO_2	Carbon Dioxide
COMPBRN	Computer Code for Fire Analysis
CR	Control Room
CRD	Control Rod Drive
CREF	Control Room Emergency Filtration
CRH	Control Room Habitability

LIST OF ACRONYMS (Continued)

CRIDS	Control Room Integrated Display System
CS	Containment Spray, Core Spray
CSS	Containment spray System, Core Spray System
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DC	Direct Current
DCP	Direct Current Power
DGS	Diesel Generator System
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
FCIA	Fire Compartment Interaction Analysis
FDS	Fire Damage State
FHA	Fire Hazards Analysis
FIVE	Fire Induced Vulnerability Evaluation
FRH	Fire Risk Hazard
FRSS	Fire Risk Scoping Study
FRVS	Filtration, Recirculation, and Ventilation System
FSAR	Final Safety Analysis Report
GI	Generic Issue
GIP	Generic Implementation Procedures
GL	Generic Letter
GPM	Gallons per Minute
GSI	Generic Safety Issue
HCGS	Hope Creek Generating Station
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HMR	Hydro-Metrological Reports
HPCI	High Pressure Core Injection
HRA	Human Reliability Analysis
HRR	Heat Release Rate
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
I&C	Instrumentation and Controls
IAS	Instrument Air System

LIST OF ACRONYMS (Continued)

IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated Leak Rate Testing
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IORV	Inadvertent Opening of a Relief Valve
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISDS	Ignition Source Date Sheets
ISLOCA	Interfacing System Loss of Coolant Accident
JCO	Justification for Continuation of Operation
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LRR	Low Ruggedness Relays
MCC	Motor Control Center
MFW	Main Feedwater
MMIS	Managed Maintenance Information System
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MW	Megawatts
MWe	Megawatts Electric
NCO	Nuclear Control Operator
NEDO	Nuclear Energy Division (O - indicates non proprietary) General Electric Company
NEI	Nuclear Energy Institute
NEO	Nuclear Equipment Operator
NEP	Non - Exceedance Probability
NFPA	National Fire Protection Association
NISS	Not Important to Seismic Safety
NJAC	New Jersey Administrative Code
NOAA	National Oceanic and Atmospheric Aaministration
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center

LIST OF ACRONYMS (Continued)

NSSS	Nuclear Steam Supply System
NTS	National Technical Systems - Nuclear Test Site
NUREG	Nuclear Regulation
NWS	National Weather Service
OBE	Operating Basis Earthquake
OFRNARA	Office of the Federal Register National Archives and Records Administration
PBAPS	Peach Bottom Atomic Power Station
PCIG	Primary Containment Instrument Gas
PCIS	Primary Containment Isolation System
PCIV	Primary Containment Isolation Valve
PDS	Plant Damage State
PFS	Potential Fire Spread
PGA	Peak Ground Acceleration
PMH	Probable Maximum Hurricane
PMP	Probable Maximum Precipitation
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSE&G	Public Service Electric and Gas
PSID	Pounds per Square Inch Differential
PSIG	Pounds per Square Inch Gauge
PSV	Primary Safety Valve
PTI	Plant Trip Initiator
PWR	Pressurized Water Reactor
RACS	Reactor Auxiliaries Cooling System
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFPT	Reactor Feed Pump Turbine
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRCS	Redundant Reactivity Control System
RSP	Remote Shutdown Panel
RW	Radiation Waste
RWCU	Reactor Water Cleanup

LIST OF ACRONYMS (Continued)

SACS	Safety Auxiliaries Cooling System
SBO	Station Blackout
SCBAs	Self-Contained Breathing Apparatus
SDC	Shutdown Cooling
SDS	Seismic Damage State
SEWS	Screening and Evaluation Worksheet
SFPE	Society of Fire Protection Engineers
SGS	Salem Generating Station
SGS 1	Salem Generating Station Unit 1
SGS 2	Salem Generating Station Unit 2
SI	Safety Injection
SLC	Standby Liquid Control
SNL	Sandia National Laboratory
SORV	Stuck Open Relief Valve
SPC	Suppression Pool Cooling
SPT	Standard Penetration Test
SQUG	Seismic Quantification Utility Group
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSD	Safe Shutdown
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSI	Soil Structure Interaction, Spatial Systems Interaction
SSW	Station Service Water
TACS	Turbine Auxiliaries Cooling System
SWIS	Service Water Intake Structure
SWS	Service Water System
TB	Turbine Building
TRS	Test Response Spectra
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
UHS	Uniform Hazard Spectra
USI	Unresolved Safety Issue
VCC	Vital Control Center
ZPA	Zero Period Acceleration

SECTION 1
EXECUTIVE SUMMARY

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SECTION 1

EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This report was developed in response to the Nuclear Regulatory Commission (NRC) request that each licensee perform an Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities for each of its nuclear plants, as detailed in Generic Letter 88-20, Supplement 4, issued in June 1991 (NRC, 1991a). With the performance of the work described in this report, Public Service Electric and Gas (PSE&G) Company has fulfilled all the objectives of the generic letter for its Hope Creek Generating Station (HCGS). The principal objectives of the IPEEE as outlined in the generic letter are, for the case of external initiating events:

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

In NUREG-1407 (NRC, 1991b), the NRC specifically identified the following external events as those to be included in the scope of the IPEEE:

- Seismic events
- Internal fires
- High winds and tornadoes
- External floods
- Transportation and nearby facility accidents

In addition, it was requested that a search for unique and plant specific events be made.

This document summarizes the results of the IPEEE for the HCGS in a manner consistent with the submittal guidance provided in the generic letter and in NUREG-1407 [NRC, 1991b].

In response to the original Generic Letter 88-20, published in November 1988 (NRC, 1988e), PSE&G submitted its Individual Plant Examination (IPE) for the HCGS (PSE&G, 1994a), which addressed the risk from internal initiating events. This IPEEE evaluation builds upon the plant models created for that study, and subsequent updates to those models.

The information provided in this submittal is supported by extensive documentation in the form of analysis reports, notebooks, and files. The organization of the documentation is designed to support a detailed review of the analysis.

This executive summary provides a brief description of the study and its results. Section 2 of this report is a description of the overall scope of the IPEEE and a summary of the methods used for the analyses. Section 3 provides a summary of the analysis of seismic events. Section 4 describes the analysis of the risks from fires, and Section 5 describes the analysis of the risks from all other external events. Section 6 describes PSE&G Independent Review activities. Section 7 discusses the plant improvements that have been identified as a result of this investigation, and Section 8 provides the summary and overall conclusions. References are provided separately at the end of each report section.

1.2 PLANT FAMILIARIZATION

The HCGS is owned* and operated by PSE&G. The station is located on the northern part of the PSE&G reactor site located on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. The HCGS is located on a 700 acre site owned by PSE&G. The Salem Generating Station, a dual unit plant, is also located on this same 700 acre site. Each SGS unit employs a Westinghouse 4-loop Pressurized Water Reactor (PWR) rated at a thermal power of 3411 MW. The nuclear steam supply system (NSSS) for each SGS unit is enclosed by a large, dry, reinforced concrete, steel-lined containment. SGS Unit 1 began commercial operation in June 1977, and SGS Unit 2 in October 1981. The HCGS employs a General Electric boiling water reactor (BWR/4) and is operated at a core thermal power of 3293 MW (100% steam flow) with a gross electrical output of

* Atlantic Electric owns 5%

approximately 1118 MWe and net electrical output of approximately 1067 MWe. The HCGS began commercial operation in December 1986. The HCGS dual barrier containment system consists of a pressure suppression primary containment system (type 4g-Mark I) and a secondary containment system consisting of a dome-shaped reactor building. The reactor building (or secondary containment) is a concrete-reinforced structure which houses the primary containment system, and the fuel storage area. It is capable of containing any radioactive materials released into it subsequent to a design basis loss of coolant accident (LOCA) so that offsite doses remain below 10CFR100 requirements. Figure 1-1 shows the geographical location of PSE&G's nuclear site. Figure 1-2 shows an overall view of the HCGS. Figure 1-3 shows the HCGS primary and secondary containment. The site is considered a soil site for purposes of seismic evaluation and is located within tornado intensity Region I (NRC, 1974b - Figure 1) for the purposes of wind classification.

The HCGS reactor building is equipped with blowout panels to limit internal pressures during specific accidents. The primary containment system consists of a drywell housing, which is connected to a suppression pool. There are vacuum breakers between the suppression pool and drywell (eight - one in each vent pipe), and between the reactor building and suppression pool to ensure integrity of the primary containment.

The Emergency Core Cooling Systems (ECCSs) at the HCGS are similar to the ones used in the NUREG-1150 reference plant, Peach Bottom Atomic Power Station (PBAPS) (NRC, 1987a). A brief explanation of some important safety systems and certain plant-specific designs, features, and procedures follows:

1. There are Residual Heat Removal (RHR) subsystems with one pump in each loop and one heat exchanger in two of the four loops. Loop "B" of RHR can be operated in the shutdown cooling (SDC) mode via the remote shutdown panel. The four primary modes of the RHR system are: 1) to provide Low Pressure Coolant Injection (LPCI), 2) to provide Suppression Pool Cooling (SPC), 3) to provide Shutdown Cooling (SDC), and 4) to provide Containment Spray (CS). Procedures for using the Reactor Water Cleanup (RWCU) system for decay heat removal are in place, although no credit has been taken for them in the IPE or IPEEE.

2. The HCGS is equipped with two high pressure steam driven pumps: The safety related High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) pumps. The turbine of each of these pumps exhausts into the torus. The HPCI and RCIC pumps trip when the torus pressure (turbine exhaust back pressure) exceeds 140.0 PSIG and 25 PSIG, respectively. The (HPCI) system injects 3000 gpm through the Feedwater system and 2600 gpm through the Core Spray (CS) system spargers. RCIC injects 600 gpm through the Feedwater System.

3. The Core Spray System (CSS) has two loops with two pumps in each loop. The CSS pumps take their suctions from the torus; however, they can be manually aligned to take suction from the Condensate Storage Tank (CST). The CSS Loop A spray spargers are shared with HPCI and Standby Liquid Control (SLC) systems.

4. The Automatic Depressurization System (ADS) is an ECCS utilizing five of 14 Target Rock Safety Relief Valves (SRVs).

5. The Standby Liquid Control (SLC) system injects to the vessel automatically when initiated by the Redundant Reactivity Control System (RRCS) in response to Anticipated Transient Without Scram (ATWS) scenarios. It can also be initiated manually.

6. There are four Emergency Diesel Generators (EDGs) at the HCGS.

7. There is a connection point at RHR Loop B for Station Service Water (SSW) System injection to the reactor vessel or for containment flooding. In addition, both the diesel-driven and motor-driven fire pumps can be connected to the RHR system to provide additional alternate methods of injection to the RPV. Loop B of the RHR system can divert flow to the reactor vessel head spray line.

8. The HCGS is equipped with a twelve-inch hard pipe vent which originates from the top of the torus. This vent can be opened remotely from the control room when ac or dc power is available. The vent can also be operated locally, in the absence of any electric power. The HCGS is also equipped with a six-inch hard pipe vent, which is used for Integrated Leak Rate Testing (ILRT). Some credit is given to this pipe for containment venting through both the drywell and the torus. The HCGS is also equipped with

ducts, which can be used for venting; however, no credit is taken for them in the IPE or IPEEE.

9. There are three trains of feedwater/condensate, each containing one feedwater pump, one secondary pump and one primary condensate pump in series. Primary and secondary condensate pumps can inject to the vessel at pressures of up to 550 PSIG and 202 PSIG, respectively.

10. The Control Rod Drive (CRD) pumps are powered by the Class 1E electrical buses, through two in-series breakers (one Class 1E and one non-Class 1E). Upon receipt of a LOCA signal, the Class 1E breaker trips and the non-Class 1E breaker opens on undervoltage.

11. The blowout panels in various locations of the reactor building protect the primary containment against high external pressure.

12. The HCGS primary containment has an internal design pressure of 56.0 PSIG, a maximum calculated internal design pressure of 58.0 PSIG with an allowable maximum internal design pressure of 62.0 PSIG (110 percent of design pressure based on the ASME code). The primary containment maximum external design pressure is 3 PSID, and its design temperature is 340°F.

13. The Emergency Operating Procedures (EOPs) are based on Revision 4 of the Boiling Water Reactor Owner's Group (BWROG) Emergency Procedure Guidelines (EPGs) (NEDO, 1987a).

The HCGS was designed in the early 1970's. The seismic design basis safe shutdown earthquake (SSE) of the unit is 0.2g peak horizontal ground acceleration for all Seismic Category 1 structures, systems and components. The maximum vertical ground acceleration was specified as two-thirds of the horizontal acceleration. The free-field ground response spectra from USNRC Regulatory Guide 1.60 ground response spectra anchored to 0.2g SSE level was used in the design basis. A positive feature of the plant is that CO_2 fire suppression components have been seismically qualified. Such systems protect the diesel generator rooms and the control room equipment room mezzanine area.

Although the unit is not an Appendix R plant, detailed fire hazard analysis including a comparison with Appendix R regulations is part of the design basis. An important design feature of the HCGS with regard to fires is its compartmentalization into hundreds of rooms, even in the reactor building.

The performance of the IPEEE required additional knowledge of the plant over and above that which was required for the performance of the IPE. In particular, the physical characteristics of the plant, including detailed knowledge of the location of equipment and details of its anchorage was required. This plant familiarization was brought into the project by involving engineers with detailed specialized knowledge. (e.g., fire protection engineers, structural engineers), and also by performing walkdowns. Several walkdowns with different objectives were performed, some to address seismic issues, others to address fire related issues, as-well-as walkdowns to confirm the general characteristics of the plant and the site and its response to other potential external events.

Plant walkdowns constituted an important part of the plant familiarization effort. Plant walkdowns were performed by a PSA analyst accompanied by environmental experts, (e.g., seismic and fire) as applicable. When information from various data sources, such as drawings or specifications, could not be independently confirmed, it was supplemented by HCGS Senior Reactor Operators (SROs), system managers, or other cognizant engineering personnel. Plant system managers and other engineering personnel, as well as the fire crew at the HCGS, are intimately familiar with the plant configuration and continually perform "walkdowns" as part of their daily responsibilities.

1.3 OVERALL METHODOLOGY

The IPEEE for the HCGS was performed using methods identified in NUREG-1407 (NRC, 1991b). The methodology used to perform the analysis is described in detail in Section 2; however, a brief summary of the IPEEE analysis method is presented below.

1.3.1 SEISMIC ANALYSIS

A Seismic Probabilistic Safety Assessment (PSA) analysis approach was taken to identify any potential seismic vulnerabilities at the HCGS. The Seismic PSA method is an acceptable seismic evaluation methodology identified in NUREG-1407 (NRC, 1991b). This PSA technique includes consideration of the following elements:

- Seismic Hazard Analysis
- Seismic Fragility Assignment
- Seismic Systems Analysis
- Quantification of the Seismically Induced Core Damage Frequency

Additional seismic analysis at the HCGS was focused to evaluate other seismic vulnerabilities through the evaluation of:

- Human interactions and recovery actions under seismic conditions
- Relay chatter during a seismic event
- Soil seismic liquefaction and slope stability effects
- Containment seismic performance

Seismic hazard analysis was conducted to identify the sources, frequency of occurrence, and intensity of earthquakes that may impact the Hope Creek site. This information was then evaluated to estimate the frequency of exceedance for selected levels of ground motion for the HCGS. For the Hope Creek site, there are two published site-specific hazard studies (EPRI, 1989a; NRC, 1994b), and the results from these studies were utilized in this IPEEE. The potential for soil liquefaction and settlement under earthquake conditions at the HCGS site was evaluated in the IPEEE using a probabilistic approach consistent with EPRI NP-6041 (EPRI, 1991b).

Considerable information exists for the estimation of seismic fragilities for structures and equipment similar to those of the HCGS, as the fragilities of components for over 30 nuclear power plants have been estimated in the last ten years. Potentially risk significant HCGS structures and equipment were identified for detailed seismic fragility evaluations and subjected to seismic walkdowns to search for seismic vulnerabilities, to assist in screening out high capacity components, and to collect additional data on approximately 100 components needing detailed fragility analysis. Screening of relays was conducted to determine if relays susceptible to relay chatter during a seismic event are used in electrical or instrumentation circuits vital to the safe shutdown of the plant. The major containment structures and systems whose performance failures could result in early failure of containment were evaluated through walkdowns and seismic PSA capacity calculations.

Seismic system analysis was conducted to define the potential seismic-induced structure and equipment failure scenarios that could occur after a seismic event, to quantify the frequencies of those scenarios, to quantify the conditional core damage probabilities for these scenarios, and to quantify the overall frequency of seismic-induced core damage. The event and fault tree models developed for the HCGS internal events IPE (PSE&G, 1994a) were used as the starting point for the seismic IPEEE models. Traditional event tree techniques were used to delineate the potential combinations of seismic-induced failures, and resulting seismic scenarios, which were termed "seismic damage states" (SDS). The frequencies of these seismic damage states were quantified by convoluting the earthquake hazard curve with the structure and equipment seismic fragility curves. These calculations incorporate random failures of equipment and operator actions. Special attention was given to human interactions and recovery actions under seismic conditions.

1.3.2 INTERNAL FIRES

The analysis of the impact of internal fires consisted of a screening of fire areas based on EPRI FIVE methodology guidelines, (EPRI, 1993b) and the development of a PSA for the detailed evaluation of unscreened fire areas. The detailed evaluation developed the likelihood and resulting impact of intermediate fire growth stages within each fire area, rather than assuming the contents of the entire area are immediately damaged, as in the screening evaluation. Equipment damage resulting from the thermal effects of fire are considered as well as the random unavailability of components unaffected by fire. Potential vulnerabilities raised in the Sandia Fire Risk Scoping Study (NRC, 1989b) related to seismic/fire interactions, effects of suppressants on safety equipment and control system interactions are addressed through walkdowns, as defined in the FIVE methodology (EPRI, 1993b).

The fire models were developed in a systematic manner and included fire initiation frequency, potential detection and suppression actions, hot short potential, operator recovery actions, and the IPE based conditional core damage probability.

The fire evaluation was performed on the basis of fire areas which are plant locations completely enclosed by at least two hour rated fire barriers. The fire area boundaries which meet the FIVE fire barrier criteria are assumed to be effective in preventing a fire from spreading from the originating area to another area. The fire area boundaries recognized in this study are identical to those identified in the HCGS Updated Final Safety Analysis Report (PSE&G, 1995f). In some cases, these fire areas were further subdivided into compartments in the detailed PSA evaluation where it could be demonstrated that the space was bounded by non-combustible barriers where heat and products of combustion would be substantially confined (EPRI, 1993b).

The analysis was conducted in three main stages as follows:

Stage 1 was a systematic qualitative and quantitative screening analysis of all plant fire areas. The screening analysis was based largely on information already available in the plant's Safe Shutdown Analysis (SSA) and the IPE study. At this stage all equipment and cable in a fire area were assumed to be damaged. The damage was assessed qualitatively to determine if the effects were significant; that is, whether the fire would cause a plant shutdown or trip, or lead to loss of safe shutdown equipment. Fire areas not screened out qualitatively were then subject to a determination of their associated fire frequency (F_1) and conditional core damage probability (P_2), given loss of all functions which may be impacted by the fire. If the resulting fire induced core damage frequency ($F_1 \times P_2$) was less than 1.0E-06 per year, the fire area was screened out.

Stage 2 was a detailed evaluation of the fire areas which did not previously screen out, using fire probabilistic safety assessment (PSA) techniques as well as methods and data provided in the FIVE technical report (EPRI, 1993b). The principal difference in this stage of the analysis is that the resulting impact of intermediate fire growth stages and suppression within each fire area was assessed rather than assuming the entire contents are immediately damaged.

The third stage of the fire evaluation was an evaluation of the Sandia Fire Risk Scoping Study (FRSS) issues (NRC, 1989b) using the tailored walkdown approach provided in the FIVE methodology. Containment performance was also examined at this stage to evaluate the performance of containment systems and equipment when challenged by internal fire initiators.

1.3.3 HIGH WINDS, FLOODS AND OTHER EXTERNAL EVENTS

The method of progressive screening, per NUREG-1407 (NRC, 1991b - Section 5), was used in this assessment. The plant specific hazard data and licensing bases were reviewed. The HCGS is a 1975 Standard Review Plan (SRP) plant (NRC, 1975a), therefore, all aspects of its licensing basis as documented in the HCGS Updated FSAR (e.g., tornado wind loads, nearby facility and transportation characteristics) do conform to 1975 SRP criteria.

1.4 SUMMARY OF MAJOR FINDINGS

This section summarizes the major findings from the external events evaluation for the HCGS. Fire and seismic events were the only important external event contributors to core damage frequency at the HCGS. The IPEEE evaluation predicts a fire related core damage frequency (CDF) of 8.1E-05 per year and a seismic related core damage frequency of 3.6E-06 per year if the conservative Livermore seismic hazard curve is used. If the EPRI hazard curve is employed a seismic core damage frequency of 1.0E-06 per year results. The industry judges that the EPRI hazard curve is more realistic. These CDFs were conservatively assessed.

The evaluation of "other" external events were screened out by compliance with SRP criteria or by demonstration that their predicted CDF fell below the IPEEE screening criteria.

This IPEEE evaluation identified the need for a missile shield installation in front of door 19 in room 5619 to protect against tornado missiles which could otherwise jeopardize operability of the "A" loop of the Control Room Emergency Filtration Units [PSE&G, 1997d]. While not considered a vulnerability, the condition is being corrected to assure that the system complies with its design basis. The missile shield is scheduled to be installed in 1997.

The Sandia Fire Risk Scoping Study Safety issues have been adequately addressed at the HCGS. No vulnerabilities which could cause early failures of containment, or containment bypass were identified.

1.4.1 SEISMIC EVENTS

The total CDF from seismic events at the HCGS was calculated to be 3.6E-06 per year if the Livermore (LLNL) seismic hazard curve is used and 1.0E-06 per year if the EPRI hazard curve is employed. The most important seismic sequences are (LLNL values reported):

- SDS 36 (S-IC1) A seismic induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ481. Core damage is assumed (69.4 percent of the seismic PSA result).

- SDS 37 (S-DC) A seismic induced failure of 1E power to all four 125Vdc distribution panels 1A/B/C/D417. Core damage is assumed (12.2 percent of the seismic PSA result).

- SDS-26 (S-OP-HP) A seismic-induced loss of offsite power and failure of high pressure injection, with simultaneous random failures which result in core damage. The random failures which cause core damage are dominated by reactor depressurization failures which result in inadequate ECCS injection or Emergency Diesel Generator (EDG) failures which result in station blackout (5.3 percent of the seismic PSA result).

- SDS-35 (S-IC2) A seismic induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ482. Credit is taken for manual system control to prevent core damage, but failure of both results in core damage and primary containment isolation failure (4.4 percent of the seismic PSA result).

- SDS-18 (S-OP) A seismic-induced loss of offsite power with subsequent random failures which result in core damage. The random failures are dominated by Emergency Diesel Generator failures which result in station blackout (3.6 percent of the seismic PSA result).

The above five SDSs represent 95% of the total core damage frequency for seismic events, with SDS-36 (S-IC1) being the largest single contributor at 69.4% of the total seismic CDF. Based on these results, none of the seismic sequences investigated represent new or unique significant plant vulnerabilities.

No relay chatter interactions requiring human actions are needed based on the low ruggedness relay evaluation. It is concluded that relay chatter is not significant to safe shutdown after a seismic event at the Hope Creek plant.

Containment performance systems and equipment were explicitly included in the walkdowns and seismic PSA. No vulnerabilities which could cause early failures of containment, or containment bypass were identified.

The principal conclusion is that the seismic evaluations did not identify any unique or new vulnerabilities for the Hope Creek plant. These results and conclusions are developed in detail in the Tier 2 documentation.

1.4.2 INTERNAL FIRES

A total CDF from fire events at the HCGS was calculated to be 8.1E-05 per year.

This CDF should be viewed as an upper bound because of the extremely conservative assumptions in the fire damage modeling. The most important buildings are described in Table 1-1:

Table 1-1 Fire IPEEE CDF by Building

Building	CDF/Year>	Percent of Total
Control/Diesel	6.7E-05	85.9
Reactor	8.0E-06	10.3
Turbine	2.0E-06	2.6
Radwaste	7.3E-07	0.9
Switchyard	3.0E-07	0.4

More than 200 fire compartments were analyzed in the IPEEE Fire Study. Thirty-eight fire compartments did not screen out in the Fire IPEEE study using the FIVE criteria (CDF/Year <1E-06). Table 1-2 shows the top 16. These 16 represent more than 95% of the total Fire IPEEE CDF.

The Fire Risk Scoping Study (NRC, 1989b) Safety issues were addressed during the fire analysis and it was found that each of the issues has been adequately addressed at the HCGS.

1.4.3 HIGH WINDS, FLOODS AND OTHER EXTERNAL EVENTS

Beginning with the list of external events found in NUREG/CR-2300 (NRC, 1983a), the class of external events termed "other external events" have been screened out either by compliance with the 1975 SRP criteria or by bounding probabilistic analyses that demonstrate a core damage frequency of less than the IPEEE screening criterion. The study provides confidence that no plant-unique external event is known that poses a significant threat of severe accidents. The study also provides confidence that the HCGS units are not vulnerable to other external events.

1.4.4 RESOLUTION OF UNRESOLVED SAFETY ISSUES

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

The IPEEE concludes that Unresolved Safety Issues (USIs) with respect to the HCGS are satisfactorily resolved.

- USI A-45 Shutdown Decay Heat Removal - no explicit new vulnerabilities were identified in either the fire, seismic, or other external events IPEEE analysis. This issue is considered closed for the HCGS.

- Charleston Earthquake Issue - issue closed for Hope Creek.

- USI A-17 Systems Interaction - issue satisfied in parallel with IPEEE seismic walkdowns and evaluations.

- GI-57 Seismic induced fire/flood interaction issues, including spurious actuation of the fire protection systems, were evaluated and no unique vulnerabilities were identified.

The Sandia Fire Risk Scoping Study Safety issues were addressed during the fire analysis and it was found that each of the issues has been adequately addressed (PSE&G, 1997c) at the HCGS.

1.5 SUMMARY

The results of the Hope Creek Generating Station study of external events indicate that the core damage frequency due to seismic events is 3.6E-06 per year using the conservative LLNL hazard curve or 1.0E-06 using the more site realistic EPRI hazard curve. The core damage frequency due to fire events is 8.1E-05 per year. All "Other" external events were screened out. No unique core damage vulnerabilities were identified. No unique containment performance vulnerabilities were found. HCGS Unresolved Issues A-17, A-45, GI-57, and the Eastern U.S. Seismicity Issue are satisfactorily resolved.

The examination of external event severe accident vulnerabilities, as requested by the NRC in Generic Letter 88-20, Supplement 4 has been completed for the Hope Creek Generating Station.

Table 1-2 Fire IPEEE CDF by Fire Compartment

Building/ Elevation	Room	Description	Initiating Event	CDF/Year	Percent of Total
Aux - 137'	5510, 5511	Control Room	MSIV Closure LOOP SORV Loss of HVAC Loss of SWS Loss of SACS	2.5E-05	30.86
Aux - 130'	5416, 5417	Class 1E (Ch. A) Switchgear Room	MSIV Closure	1.3E-05	16.05
Aux - 102'	5307	Diesel Generator (Ch. A)	LOOP MSIV Closure	5.3E-06	6.54
RB - 77	4202	CRD Pump Area	MSIV Closure	4.2E-06	5.19
Aux - 102'	5306	Diesel Generator (Ch. B)	LOOP MSIV Closure	4.1E-06	5.06
Aux - 102'	5305	Diesel Generator (Ch. C)	LOOP MSIV Closure	3.7E-06	4.57
Aux - 130'	5412, 5413	Class 1E (Ch. B) Switchgear Room	MSIV Closure	3.0E-06	3.70
Aux - 137'	5501	Electrical Access	MSIV Closure	3.0E-06	3.70
Aux - 102'	5339	Electrical Access	LOOP MSIV Closure	2.7E-06	3.33
Aux - 163.6'	5605, 5631	Upper Control Eqpt. Computer Rooms	MSIV Closure	2.7E-06	3.33
Aux - 102'	5304	Diesel Generator (Ch. D)	LOOP MSIV Closure	2.6E-06	3.21
Aux - 124'	5401, 3425	Electrical Access	MSIV Closure	2.0E-06	2.47
RB - 102'	4301, 4309, 4310, 4311	North Side and Div. 1 SACS Area	MSIV Closure	1.8E-06	2.22
Aux - 102'	5302	Lower Control Electrical Eqpt. Room	LOOP SORV MSIV Closure	1.7E-06	2.10
TB - 102'	1315, 1316, 1317, 1320, 1321, 1322	Access and Unloading Area	LOOP	1.2E-06	1.48
RB - 102'	4303	MCC Area	MSIV Closure	1.2E-06	1.48
Total of Top Sixteen Compartments				7.72E-05	95.29

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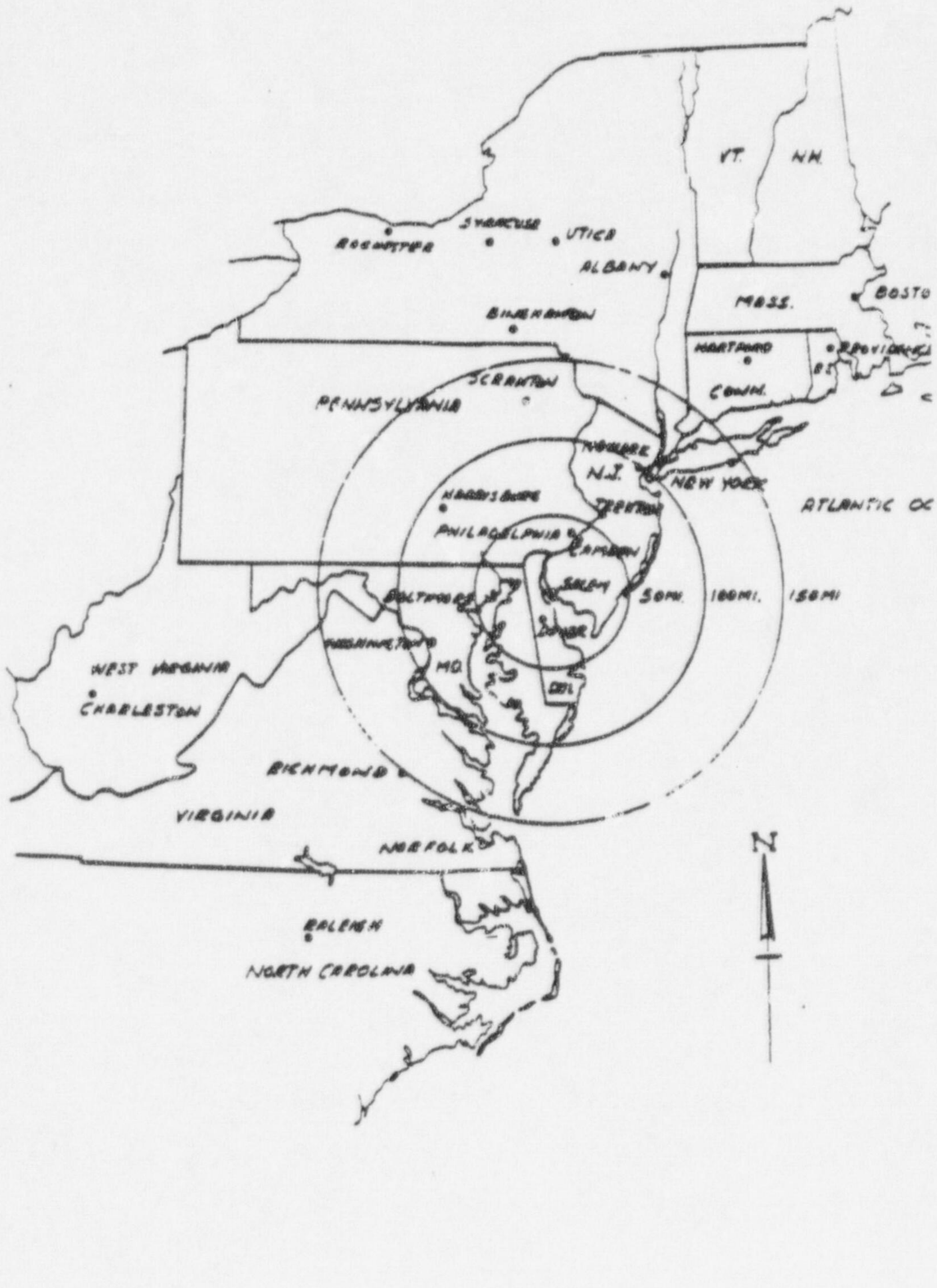

Figure 1-1
HCGS Site Location

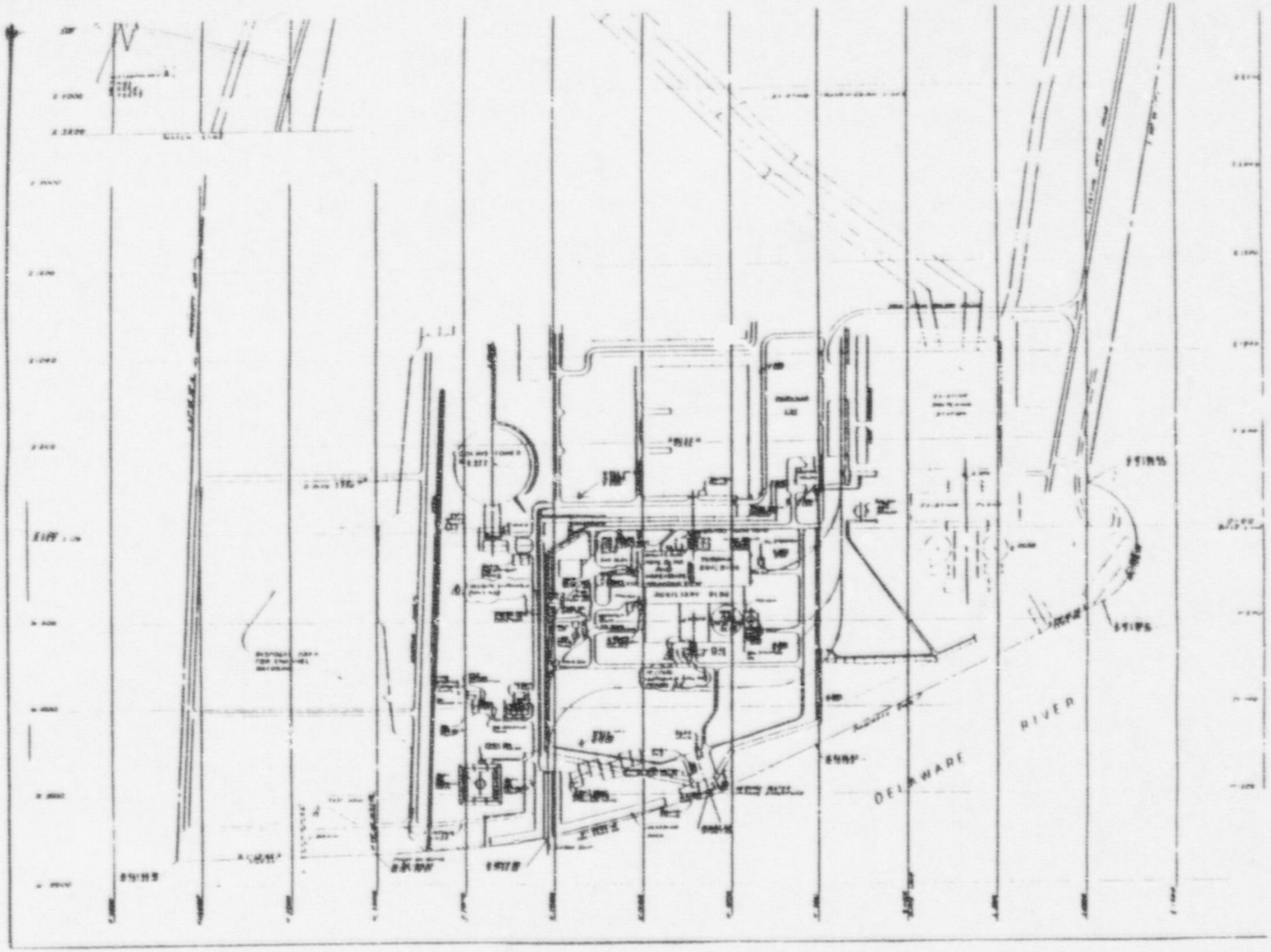

Figure 1-2
HCGS Site Building Layout

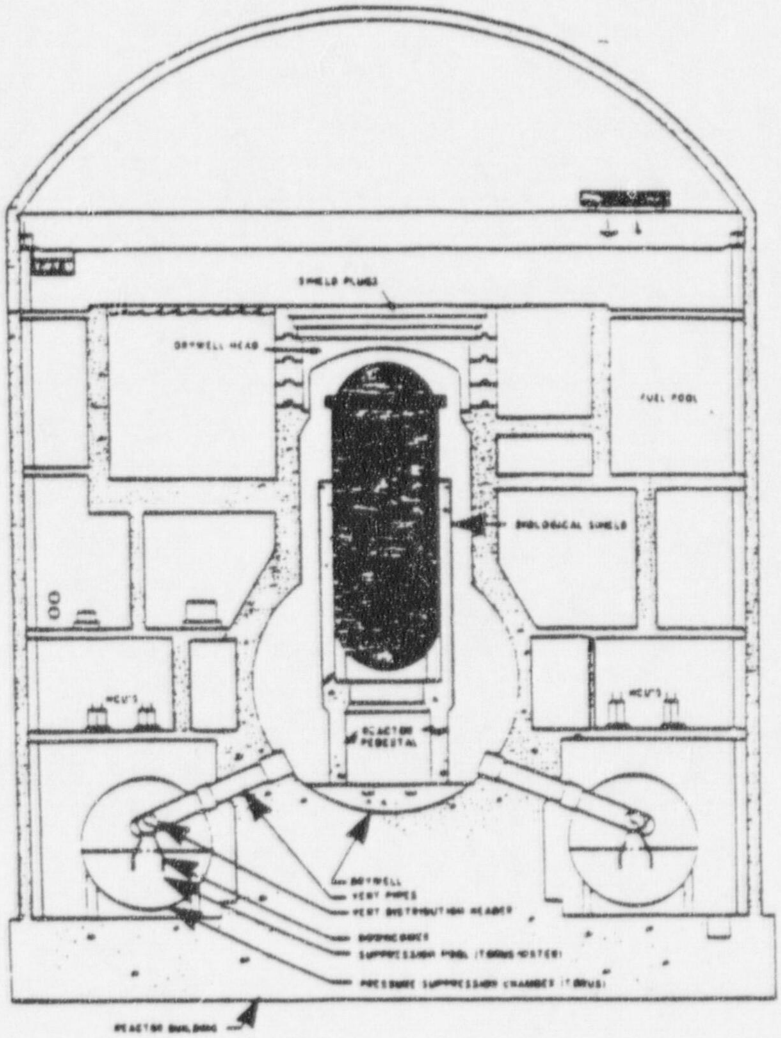

Figure 1-3
Primary and Secondary Containment Layout

SECTION 2
EXAMINATION DESCRIPTION

SECTION 2

EXAMINATION DESCRIPTION

2.1 INTRODUCTION

As part of the implementation of the Severe Accident Policy, the Nuclear Regulatory Commission (NRC) issued Generic Letter 88-20 (NRC, 1988e) on November 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding. To comply with the generic letter, Public Service Electric and Gas Company (PSE&G) submitted the IPE report for its Hope Creek Generating Station (HCGS) in April 1994 (PSE&G, 1994a). In supplement 4 to the generic letter (NRC, 1991a), the NRC requested that the licensee extend its examination to include what have become known as external initiating events.

This report presents the Individual Plant Examination of External Events (IPEEE) for the HCGS in response to that request. The general objectives of the IPEEE are similar to that of the IPE, namely: (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at the plant under full power conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) if necessary, to reduce the overall likelihood of core damage and fission product release by modifying, where appropriate, hardware and procedures to help prevent or mitigate severe accidents. With the performance of the work described in this report, PSE&G has fulfilled all the objectives of the generic letter for its HCGS.

For the purposes of this report, a vulnerability is a scenario which contributes inordinately to the HCGS core damage frequency (CDF), as compared to other plants of similar type and vintage (as available from published risk assessment results), representing a substantial design weakness of the plant.

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

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

PSE&G has performed an IPEEE pursuant to 10 CFR 50.54, as invoked by Generic Letter 88-20, Supplement 4 (NRC, 1991a).

The IPEEE generic letter and report guidance document, NUREG-1407 (NRC, 1991b), requests that licensees consider five specific external events in performing their IPEEEs: seismic events, internal fires, high winds, floods (external), and transportation and nearby facility accidents. Licensees are also asked to confirm that no other plant unique external events, with potential for severe accidents, are being excluded. The IPEEE subsumes the external events aspects of several ongoing NRC programs, such as Unresolved Safety Issue (USI) A-45 (decay heat removal); their resolutions are also required to be explicitly addressed in the IPEEE response.

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

2.2.1 IDENTIFICATION OF EXTERNAL HAZARDS

The specific external hazards that should be addressed by the study are identified in the IPEEE generic letter supplement (NRC, 1991a) and NUREG-1407 (NRC, 1991b) as:

- Seismic events
- Internal fires
- High winds and tornadoes
- External floods
- Transportation and nearby facility accidents

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

2.2.2 METHODS OF EXAMINATION

The response to the IPEEE for fires has been met by performing a Fire Probabilistic Safety Assessment (PSA). Portions of the EPRI Fire Induced Vulnerability Evaluation

(FIVE) (EPRI, 1993b) methodology have been adopted, particularly in the areas of location screening and fire frequency evaluation for the fire PSA. A seismic PSA analysis approach was taken to identify any potential seismic vulnerabilities at the HCGS.

All other external events including external flooding, high winds and tornadoes, and transportation and nearby facility accidents have been analyzed using the approach discussed in NUREG-1407 (NRC, 1991b). The Hope Creek Generating Station complies with the 1975 Standard Review Plan Criteria. Engineering assessments and bounding calculations were performed to verify current plant status.

2.2.3 COORDINATION WITH OTHER EXTERNAL EVENT PROGRAMS

A number of programs are subsumed in the IPEEE: (1) the external event portion of USI A-45, (2) GI-57, (3) USI A-17 and (4) the Eastern U.S. Seismicity Issue (Charleston Earthquake Issue). These issues are considered closed as a result of completion of this IPEEE.

The Sandia Fire Risk Scoping Study (FRSS) issues, NUREG/CR-5088 (NRC, 1989b), were examined using the guidance and detailed outline provided in the FIVE document (EPRI, 1993b). The walkdown was specifically tailored to aid in the assessment of these issues.

The FRSS issues are discussed in Paragraph 4.8. The issue of seismic-fire interactions is addressed in Paragraph 3.1.7.

2.2.4 CONTAINMENT PERFORMANCE

In accordance with Generic Letter 88-20, Supplement 4, for the IPEEE, it is necessary to investigate mechanisms that could lead to containment failure, particularly early containment failure or bypass which are different from those identified in the IPE. Four areas were investigated with respect to the fire and seismic analyses: containment bypass, containment isolation, performance of containment heat removal systems, and structural integrity.

For fires, a detailed systematic evaluation was performed of the potential for bypass, isolation failure, direct containment integrity failure, and containment system degradation or failure for each fire compartment whose screening core damage frequency was greater than 1E-06/yr. The potential for interfacing system LOCAs was also analyzed. This containment evaluation made use of the list of containment systems and components developed for the IPE in addition to the list of equipment, cabinets, and cables compiled for the fire PSA from several different sources. The effects of core damage sequences and direct containment failure were considered, both included the potential for hot shorts.

NUREG-1407 (NRC, 1991b - Paragraph 3.2.6), provides additional guidance on the content of the seismic containment performance analysis. For the IPEEE, containment performance systems and equipment were explicitly included in the seismic walkdowns. The seismic fragility evaluation for the containment structures and unscreened components was performed, and the seismic core damage sequence analysis was developed. Specific aspects of containment performance included in the analysis were: structures and major components, containment isolation (mechanical components and electrical actuation), containment bypass, containment hatches, and containment pressure suppression and heat removal. Containment performance is discussed in more detail in paragraphs 3.1.6 and 4.7 of this report. Containment systems examined and the method employed for their seismic evaluation include:

- Structures and Major Components

  The major structures and systems whose failure could result in early failure of containment were evaluated through walkdowns and seismic capacity calculations. These included the Reactor Building, the Auxiliary Building, the Station Service Water Intake Structure, interior structures such as the Torus and the Drywell, reactor coolant support and piping, main steam lines, and nearby structures. No issues or potentials for failure of these items were noted during the walkdowns. Particular attention was given to the adequacy of seismic gaps between major structures. The fragility calculations (EQE, 1996b) demonstrated that all of these structures had high seismic capacity (with median PGA capacities greater than 1.5g), and could be screened from the analysis.

- Containment Isolation

  Mechanical and electrical penetrations were included in the walkdown to ensure that there would not be failures of the mechanical penetrations or piping, electrical penetration assemblies, isolation valves and associated cables, piping supports, anchorages, or spatial interactions or differential motion which could cause failure of containment isolation or integrity. The Hope Creek Generating Station does not have any primary containment penetrations which require cooling, and no isolation valves require air to close. Therefore, on the basis of the walkdowns, capacity judgments, and the design of the Hope Creek containment isolation and penetrations, there are no vulnerabilities in the mechanical or electrical penetration systems, or in the containment isolation valves and piping.

- Containment Bypass

The potential for seismic-induced Interfacing Systems Loss of Coolant Accidents (ISLOCA) involves the failure of the Reactor Coolant System (RCS) pressure boundary leading to a LOCA outside the containment boundary. The internal events IPE (PSE&G, 1994a) has identified all potential ISLOCA paths, and was used as the initial basis for this seismic containment bypass analysis. Valves in each of the ISLOCA paths were reviewed for inclusion on the seismic equipment list, and then included in the seismic capacity walkdowns. Paths with check valves and normally closed manual valves for isolation have high capacity, therefore these paths were not evaluated further. For the remaining paths, the MOVs were included in the seismic equipment list and walkdowns. These valves were also determined to have high seismic capacities, so they were screened from further analysis. The relays associated with these valves , including isolation actuation systems, were included in the relay chatter evaluation (PSE&G, 1996b), Based on the ISLOCA evaluation, there are no seismic vulnerabilities associated with these paths, or with the valves and associated relays. No additional containment performance modeling is necessary.

- Containment Hatches

The Hope Creek Generating Station does not have inflatable seals on its hatches, so there is no concern about the loss of air to the hatches. This, along with the review of the hatches during the walkdowns, lead to the conclusion that there are no vulnerabilities associated with the containment hatches.

- Containment Isolation Actuation

The sensors, transmitters, power supplies, logic and relay cabinets for the Primary Containment Isolation System (PCIS) were included in the seismic walkdowns. All components have high capacities and were screened from further evaluation, except for the logic cabinet 120 Vac 1E power supplies from the 1A/B/C/DJ482 distribution panels. These panels distribute power to the logic cabinets 1A/B/C/DC652, respectively. The 1A/B/C/DC652 logic cabinets provide automatic LOCA and high radiation isolation signals to non-NSSS Primary Containment Isolation Valves (PCIVs). Manual actuation of the PCIS is still possible from the Control Room, even if the automatic signals fail. In the seismic event tree, event IC2 represents failure of the 1A/B/C/DJ482 distribution panel and failure to perform the necessary actions manually. The Seismic Damage State (SDS 35 or SDS S-IC2) results directly in core damage. Because the event IC2 includes the failure to perform the manual actions necessary to avoid core damage, it is assumed that this would also include a failure to manually close the PCIVs.

Therefore, SDS 35 results directly in core damage and in early containment failure, with a frequency of 1.6E-7 per year when using the LLNL hazard curve (NRC, 1994b), and 4.6E-8 per year when the EPRI curve (EPRI, 1989a) is used. This early release frequency is relatively small when compared to the total seismic core damage frequency (Four percent of the LLNL CDF).

The early release frequency is also small when compared to the total internal events early release frequency. The HCGS IPE (PSE&G, 1994a - Table 4.7-21) indicates that the frequency of a high early release is 9.4E-6 per year, and the total frequency of all early releases is 2.8E-5 per year. Therefore, the early release frequency of SDS 35 is only two percent of the large early release frequency in the HCGS IPE (PSE&G, 1994a), and it is only 0.6% of the total IPE early release frequency. (Note: It is necessary to mention that the HCGs release frequencies are based on highly conservative assumptions).

- Containment Pressure Suppression and Heat Removal

The seismic PSA included containment pressure suppression and heat removal functions in the RHR System. All of the RHR components modeled in the PSA were determined to have high seismic capacity, and were screened from further analysis.

2.2.5 ACCIDENT MANAGEMENT - VULNERABILITY SCREENING

The evaluation of severe accident vulnerabilities was accomplished by reference to the Severe Accident Issue Closure Guidelines NEI 91-04 (NEI, 1994a). Core damage sequences were grouped by categories (e.g., fire induced loss of core cooling) and the group frequency compared to the closure guidelines which are provided in NEI, 1994a - Tables 1 and 2.

2.2.6 CERTIFICATION OF TECHNICAL ADEQUACY

Extensive sensitivity analyses and discussion of uncertainties are included in the seismic and fire analyses and documentation. In addition, special emphasis was placed on the identification of significant assumptions. The significance of the results, the role of assumptions, and the key uncertainties are understood. The significant core damage sequences, their likelihood and the overall likelihood of fire and seismic core damage sequences are understood. The major uncertainties and sensitivities are also understood. Therefore, this submittal is a technically adequate response to GL 88-20, Supplement 4.

2.2.7 DOCUMENTATION OF EXAMINATION RESULTS

The documentation of the IPEEE study has three components. The first is this report which summarizes the results and findings of the IPEEE analyses given the current plant status, and constitutes the Tier 1 documentation. The second is a series of reports, referenced herein, which document the detailed analyses that support this Tier 1 report. These are considered Tier 2 level reports. The third is the set of supporting documentation about the HCGS design as well as prior analyses, which are referenced in the tier 2 reports. The HCGS PSA model is also a part of this supporting information. The HCGS PSA model served as the basis for the evaluation of the conditional core damage probabilities for the fire and seismic sequences.

This report follows the format specified in NUREG-1407, (NRC, 1991b - Appendix C). Information retained for audit corresponds to that specified in NUREG-1407, (NRC, 1991b - Paragraph 8.2).

2.3 GENERAL METHODOLOGY

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

2.3.1 SEISMIC ANALYSIS

A Seismic Probabilistic Safety Assessment (PSA) analysis approach was taken to identify any potential seismic vulnerabilities at the Hope Creek Generating Station. The Seismic PSA method is an acceptable methodology identified in NUREG-1407 (NRC, 1991b). This PSA technique includes consideration of the following elements:

- Seismic hazard analysis
- Seismic fragility assessment
- Seismic systems analysis
- Quantification of the seismically induced core damage frequency

The seismic analysis of the HCGS also included the following elements:

- Human interactions and recovery actions under seismic conditions
- Relay chatter during a seismic event
- Soil seismic liquefaction, settlement, and slope stability effects
- Containment performance during a seismic event

Seismic hazard analysis was performed to estimate the annual frequency of exceeding different levels of seismic ground motion at the plant site. The seismic hazard analysis focus is on the identification of the sources of earthquakes that may impact the Hope Creek site, evaluation and assessment of the frequencies of occurrence of earthquakes of different magnitudes, estimation of the intensity of earthquake-induced ground motion [e.g., peak ground acceleration] (PGA) at the site, and finally, the integration of this information to estimate the frequency of exceedance for selected levels of ground motion. For the Hope Creek site, there are two published site-specific hazard studies (EPRI, 1989a and NRC, 1994b). The results of these studies were used in this IPEEE.

The seismic walkdown followed the procedures given in the EPRI seismic margin assessment methodology report (EPRI, 1991b). It was conducted to assist in screening out high capacity components, clearly define failure modes, identify spatial systems interactions, evaluate fire protection systems for inadvertent actuation and seismic induced fire interactions, and to collect additional data on components needing detailed fragility analysis. A seismic evaluation walkdown sheet (SEWS) was prepared for each component on the equipment list. Approximately 100 components were selected for detailed fragility analysis.

Screening of components was performed 1) by satisfaction of the caveats in EPRI, 1991b for the range of 0.8g to 1.2g spectral accelerations, 2) by demonstration that the median PGA capacity of the component is greater than 1.5g by comparing the median floor response spectra with the design floor response spectra, or 3) demonstration that the HCLPF capacity is greater than 0.5g by comparing the 84 percentile floor response spectra with the design floor response spectra. Another requirement for screening a component is that the anchorage of the screened equipment is adequate to provide a HCLPF at 0.5g pga.

Response of structures was performed using the 10,000 year median Uniform Hazard Spectra per the request of NUREG-1407 (NRC, 1991b). Soil Structure Interaction analysis was performed with the substructure approach using foundation impedance and wave scattering functions assuming a fixed base. The variability in soil and structure properties were accounted for in the probabilistic response model.

The seismic fragility of a structure or equipment is measured by the conditional probability of its failure for a given level of seismic input parameter. Seismic fragilities are needed to estimate the frequency of occurrence of initiating events and to quantify the fault trees used to obtain the seismically induced accident sequence frequencies. Considerable information exists for the estimation of seismic fragilities for structures and equipment similar to those of the HCGS, as the fragilities of components for over 30 nuclear power plants have been estimated in the last ten years. The seismic fragility evaluation process begins with the identification of potentially risk significant structures and equipment for a detailed fragility evaluation, collecting available design and seismic qualification information for these items, and identifying their critical failure modes.

The purpose of seismic system analysis is to define the potential seismic-induced structure and equipment failure scenarios that could occur after a seismic event, to quantify the frequencies of those scenarios, to quantify the conditional core damage probabilities for these scenarios, and to quantify the overall frequency of seismic-induced core damage.

The event and fault tree models developed for the Hope Creek Generating Station internal events IPE (PSE&G, 1994a) were used as the starting point for the seismic IPEEE models.

Seismic system analysis was conducted to define the potential seismic-induced structure and equipment failure scenarios that could occur after a seismic event, to quantify the frequencies of those scenarios, to quantify the conditional core damage probabilities for these scenarios, and to quantify the overall frequency of seismic-induced core damage. A seismic event tree (SET) was used to delineate the potential successes and failures that could occur due to a seismic event, based on the structures and components and their fragilities. Boolean equations were developed for each of the SET top events, based on the logic and seismic fragility information (PSE&G, 1996a). Each seismic sequence equation represents the Boolean logic associated with its corresponding seismic damage state (SDS). The frequencies of these seismic damage states were quantified by convoluting the earthquake hazard curve with the appropriate structure and equipment seismic fragility curves. In particular, the seismic hazard information, structural/component fragilities, and SDS equations were then input to the NUS SEISMIC code, (NUS, 1993a) to quantify the frequency of the SDSs.

For those scenarios that required additional non-seismic failures to occur to result in core damage, the IPE internal events model (event trees and fault trees), with appropriate changes for the seismic damage state, was used to develop conditional core damage probabilities. These calculations incorporated random failures of equipment and operator actions. The NUS PRA Workstation code (NUS, 1992a) was used to calculate the conditional core damage probabilities. To obtain the overall results (i.e., CDFs), the frequency of each seismic damage state (SDS) was multiplied by the conditional core damage probability (CCDP) for that SDS. Human interactions and recovery actions, specific to seismic sequences, were included in this analysis.

A Hope Creek IPEEE seismic relay evaluation, (PSE&G, 1996b), was conducted to determine if any relays which may be susceptible to relay chatter during a seismic event are used in electrical or instrumentation circuits that are vital to the safe shutdown of the plant. The method used to identify potential low ruggedness relays (LRR), and to evaluate the impact of LRR chatter during a seismic event followed the guidance of (EPRI, 1990a). A HCGS relay walkdown was performed in conjunction with the equipment capacity walkdowns to verify that the relays were mounted in a sound manner. Several component and relay data bases were used during the plant-wide search for low ruggedness relays. When LRRs were found, additional evaluations were performed with respect to the specific failure mode, the effect on plant response, and the ability to recover equipment functionality.

Soil failure analysis was conducted to evaluate slope stability, lateral spreading and liquefaction (Woodward - Clyde, 1995 a and b). The Seismic Category I structures at the HCGS are founded on the Vincentown formation. Seismically induced settlement of the Vincentown formation and its effect on the HCGS was assessed using a probabilistic approach based on procedures described by Seed (Seed, 1990a). The liquefaction potential is a function of two probabilities: the probability that the soil will exhibit a given cyclic strength, and the probability of liquefaction given both the cyclic strength and the cyclic stress ratio associated with a given earthquake magnitude.

The soil liquefaction potential for the HCGS site was assessed using a probabilistic approach consistent with EPRI NP-6041 (EPRI, 1991b). Seismically induced settlement of the Vincentown sands and its effect on the HCGS was also assessed using a probabilistic approach based on procedures described by Tokimatsu and Seed (Tokimatsu and Seed, 1987a). Soil liquefaction and settlement evaluations were performed using a Monte Carlo simulation. The probabilistic soil liquefaction and settlement consequence impacts on the HCGS were evaluated in the IPEEE (Woodward-Clyde, 1995b).

The containment performance systems and equipment were explicitly included in the walkdowns and seismic PSA. The major containment structures and performance systems whose failure could result in early failure of containment were evaluated through walkdowns and seismic PSA capacity evaluations. This included the containment building, internal structures, the primary piping, main steam lines, and nearby structures. Particular attention was paid to the adequacy of seismic gaps between major structures.

2.3.2 INTERNAL FIRES

The technical basis of the HCGS fire IPEEE was a new fire probabilistic safety assessment (PSA) performed in a manner consistent with the guidance in

NUREG/CR-2300 [NRC, 1983a] and NUREG/CR-4840 [NRC, 1990a]. The approach taken for the fire PSA was to perform a scenario-by-scenario analysis of unscreened compartments accounting for the relative location of ignition sources and targets. Fire damage calculations were performed to determine the extent of potential damage from each postulated fire source. Openings in walls as well as open active fire dampers were included in the assessment of the extent of fire damage.

In addition to items requested in NUREG-1407 (NRC, 1991b), a special feature of this submittal is an analysis of high hazard (which are not necessarily high risk) rooms at the HCGS. These are rooms which contain a somewhat larger amount of combustible materials (other than normal cables).

The PSA was preceded by 1) a fire compartment interaction analysis (FCIA) per FIVE guidance [EPRI, 1993b] and 2) a quantitative screening analysis also performed in a manner consistent with FIVE guidance. A qualitative screening analysis was not performed for the HCGS IPEEE. That is, no compartments were eliminated from quantitative consideration owing to qualitative factors alone.

The HCGS is composed of hundreds of identifiable rooms. Each room has an associated number. Many of the areas identified as rooms do not qualify as fire compartments using the EPRI, 1993b definitions. The FCIA was performed to establish the combinations of rooms that have boundaries which meet the FIVE criteria. Therefore, many fire compartments analyzed in this study consist of multiple rooms and are so identified in this document. The result of the FCIA was a total of 209 fire compartments which met the FIVE criteria. These compartments included the turbine building, reactor building, control/diesel building, radwaste building, service water intake structure, and transformer yard.

It is computationally unreasonable to perform a detailed PSA on all of the 209 compartments identified from the Fire Compartment Interaction Analysis and the transformer array in the yard. The objective of the screening assessment was to reduce the number of compartments on which detailed fire risk assessments must be performed. A conservative, screening assessment of core damage frequency (SCDF) is used to achieve this objective. The screening assessment was composed of three major parts as follows:

A fire ignition frequency, using the five method of EPRI, 1993b - Attachment 10.3, was developed for each of the 209 fire compartments. This method was implemented using a Fire Compartment Ignition Source Data Sheet (ISDS) for each compartment.

Then, with the assumption that all equipment and cables in a compartment are failed by any fire in that compartment, a conservative reactor trip transient (initiating event) was identified from among those used in the HCGS PSA [PSE&G, 1994b]. The assignment of an initiating event to a compartment determines the event tree, derived from the HCGS PSA, to be used for the screening conditional core damage probabilities (SCCDP). Of the initiating events considered in the HCGS PSA, the following were found to conservatively represent the range of plant responses owing to fires: MSIV closure, inadvertent opening of safety relief valves, loss of offsite power (LOOP), loss of HVAC, and loss of station service water (SSW) or safety auxiliary cooling system (SACS).

Then, with the assumption that all equipment and cables in a compartment are failed by any fire in that compartment, a conservative SCCDP was calculated using the HCGS PSA model. These calculations were conservatively carried out by assuming that instrument air, feedwater, control rod drive pumps, and all human recovery actions had failed.

This study recognized the importance of a fire walkdown in order to 1) assure that documentation, particularly for cable routing and fixed combustible, represents the as-built plant, 2) uncover potential intercompartment interactions associated with openings in walls or inadequate fire barriers, 3) aid in addressing the Sandia Fire Risk Scoping Study issues, 4) assess the likelihood of critical transient combustible loading, 5) review fire protection features of the plant, 6) develop fire scenarios of unscreened compartments, and 7) verify the assumptions used in fire damage propagation analyses. Walkdown checklists were developed and completed for each compartment of the plant.

Fire damage calculations were performed using a modified version of the formulation found in the Fire Screening Methodology User Guide [EPRI, 1993b - Attachment 10.4]. Four types of fire damage mechanisms were modeled: plume effects, ceiling jet effects, hot gas layer effects, and thermal radiation effects. Modifications were made to the FIVE formulation in order to introduce more realism, such as conservation of energy. Both steady state and pseudo-transient calculations were performed. The former estimated the potential for damage assuming an infinite quantity of fuel and the latter considered the potential for extinguishment before damage taking into account fuel consumption.

Each of the unscreened compartments was subjected to a detailed scenario-by-scenario probabilistic analysis. A fire scenario is defined as a unique source, fire intensity, target, and initiating event combination. The basic formulation of the fire PSA is given in Equation 1 of Section 4. The total core damage frequency of each compartment was evaluated using a quadruple summation over fire sources, targets, intensities, and initiating events of the following factors:

$f(S_{j,k})$ = fire ignition frequency of the j^{th} ignition source having intensity (or fire size) k

$P(T_l / S_{j,k})$ = probability of damaging target l (T_l) with source j of intensity k without consideration of suppression

$P(E_{j,k} < T_l)$ = probability of not extinguishing fire from source $S_{j,k}$ before damage to T_l

$P(I_m / T_l)$ = probability of occurrence of initiating event, I_m, given damage to T_l

$P(CD / I_m, T_l)$ = probability of core damage given initiating event, I_m, and damage to T_l.

The HCGS PSA model was used for the conditional core damage probabilities. The PSA also included treatment of hot shorts, considered the potential for openings and failure of active fire barriers to create a path for propagation of damage, and included the potential for inadvertent safety relief valve opening and interfacing system LOCAs. Only two recovery actions, use of the remote shutdown panel and recovery of HVAC, were used in this study.

The Fire Risk Scoping Study issues [NRC, 1989b] were thoroughly treated by document reviews, seismic and fire walkdowns, system analyses of the potential for damage owing to inadvertent suppression system actuation, and the fire PSA. The analysis and reporting followed the checklist and guidance found in FIVE [EPRI, 1993b - Section 7 and Attachment 10.5]. The reporting includes the basis, assumptions, findings, and conclusions with respect to these issues.

2.3.3 HIGH WINDS, FLOODS AND OTHER EXTERNAL EVENTS

The analysis of other external events was performed using the progressive screening approach described in NUREG-1407 (NRC, 1991b - Section 5). Based on the work in NUREG/CR-5042 (NRC, 1989c) and other subsidiary studies, NUREG-1407 suggests specific external events for close examination in the IPEEE. These are internal fires, earthquakes, external floods, high winds and tornadoes, and transportation and nearby facility accidents. It also asks for "a certification that no other plant-unique external event is known that poses any significant threat of severe accident". Using the approach and results of NUREG/CR-2300 (NRC, 1983a) and NUREG/CR-5042 (NRC, 1989c), a screening assessment of potential external events was performed for the Hope Creek site. The list provided in NUREG/CR-2300 (NRC, 1983a - Table 10-1), was used as a starting point for a screening assessment that reduced this list to the following which received more detailed plant specific assessment:

- High Winds and Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Release of On-Site Chemicals
- Detritus

The screening assessment took advantage of the fact that the Hope Creek Generating Station (HCGS) meets the 1975 SRP criteria (NRC, 1975a).

A thorough review of documentation was performed to determine significant changes (if any) with respect to military and industrial facilities within five miles of the site, on-site storage of hazardous materials, transportation, and other recent developments. The documentation review was verified by plant walkdowns when applicable.

Probabilistic hazard screening analyses were performed to screen out river explosions and ship impact on the Service Water Intake Structure. The Service Water Intake Structure is designed for the design basis tornado.

Detritus, which has been postulated to have the potential of affecting service water intake, was also evaluated by a screening analysis. It was found that a large perturbation in the river, such as an earthquake, could initiate a detritus event that might affect all service water intakes. The frequency of an earthquake induced detritus event was found to be below the IPEEE screening criterion.

2.4 INFORMATION ASSEMBLY

The first step in the performance of the IPEEE tasks was the assembly and review of plant specific and generic information which would form the basis for the study. The references provide a list of the primary documents relied upon. Some of these are the UFSAR, seismic design documentation and previous HCGS seismic risk assessments, the Safe Shutdown Analysis (SSA), the Fire Hazards Analysis (FHA), area layout drawings, abnormal and emergency operating procedures, relay usage/installation listings, plant equipment listings, the HCGS IPE study and the HCGS PSA models, and the EPRI FIVE methodology and supporting documents. The analysis of "other" external events, relied on the HCGS UFSAR (PSE&G, 1995f) for much of its needed information. These information sources were supplemented by specific data collection and confirmatory walkdowns when considered necessary. A precise description of how the information in each of these documents was used is provided in Sections 3, 4, and 5.

All study information has been documented and is maintained in the Tier 2 documentation. Tier 2 documentation includes the following PSBP reports:

REPORT	PSBP #		REPORT	PSBP #
EQE, 1995a	320273		PSE&G, 1997a	322798
EQE, 1996a	321025		PSE&G, 1997b	322799
EQE, 1996b	321024		PSE&G, 1997c	322802
NUS, 1995b	320536		PSE&G, 1997f	322800
PSE&G, 1995n	320538		PSE&G, 1997g	322801
PSE&G, 1996a	322117		PSE&G, 1997h	323021
PSE&G, 1996b	320808		Woodward-Clyde, 1995a	320163
PSE&G, 1996j	322216		Woodward-Clyde, 1995b	320164
PSE&G, 1996p	322427			

2.5 IPEEE EXAMINATION CONDUCT AND QUALITY REVIEW

2.5.1 IPEEE STUDY CONDUCT

PSE&G technical staff have been involved from the beginning of the IPEEE study in all aspects of the analysis, project review, and quality affirmation. This involvement has ensured that PSE&G personnel are fully conversant with the IPEEE methods used for the analysis and are in a position to fully integrate the knowledge gained from performing the work into operating procedures, training programs and appropriate hardware changes.

The conduct of the study followed a defined Project Plan (PSE&G, 1992a) that integrated the activities of PSE&G, its prime contractors, and a number of subcontractors in specialized technical areas. The seismic PSA study was performed by a team comprised of engineers and technical specialists from PSE&G, NUS (now SCIENTECH), EQE International, and Woodward-Clyde Consultants. The seismic analysis was independently reviewed by the PSE&G Civil Engineering Group and Safety Factor Associates. The fire analysis was performed by Safety Factor Associates and PSE&G and independently reviewed by PSE&G (Hope Creek System Engineering - Supervisor - BOP). The evaluation of "other" external events was conducted by PSE&G engineering staff with independent review by Safety Factor Associates and SCIENTECH.

2.5.2 IPEEE PROJECT QUALITY REVIEW ACTIVITIES

All technical analysis and supporting Tier 1 and 2 documentation were subjected to multiple in-depth reviews performed by selected PSE&G engineers and consultants forming the "Project Review Team" (PRT). Additionally, an Independent Review was conducted on each of the separate analyses that make up the study (Section 6 discusses the independent review activities).

2.5.2.1 Project Review Team (PRT)

All Tier 1 documents and supporting Tier 2 analysis files, reports, and documents were subjected to review and approval by the PRT. The composition of the review team ensured that all technical supporting products (Tier 2 analysis files) and summary information (Tier 1 information) received competent discipline review.

Care was taken to ensure that PRT review was performed by PSE&G staff and consultants who were independent of the conduct of the technical task being reviewed. PRT review occurred at the preliminary draft and final draft stages for all documents. Documents satisfying the PRT review were placed into the PSE&G Public Service Blue Print (PSBP) document control system and assigned a document control and tracking PSBP number. Upon acceptance into the PSBP system, the documents were then provided to the Independent Reviewers for the next level of quality review.

The composition of the Project Review Team:

V. Amaraksha	PSE&G, I&C and Relay
A. Caplinger	PSE&G, Nuclear Safety and Fire Protection
F. Dombek	SCIENTECH
M. Frank	Safety factor Associates
J. Gebely	PSE&G, Nuclear Safety and Fire Protection
A. Johnson	PSE&G, Civil Engineering
J. Leary	PSE&G, PSA Staff
J. Materazo	PSE&G, I&C AND Relay
I. Nag	PSE&G, Fire Protection/Safe Shutdown Engineering
M. Phillips	PSE&G, PSA Supervisor
C. Pupek	PSE&G, PSA Staff
K. Sarkar	PSE&G, Electrical Engineering
G. Schroeder	PSE&G, Fire Protection Engineering
S. Seyehosseini	PSE&G, PSA Staff
Y. Shyu	PSE&G, Seismic & Soils Analysis
J. Thompson	PSE&G, Hope Creek Technical Staff
T. Weir	PSE&G, PSA Staff

2.5.2.2 Independent Review

Section 6 of this report provides a detailed discussion of the Independent Review quality affirmation activities.

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3.0 METHODOLOGY SELECTION

3.0.1 OUTLINE OF METHODOLOGY

The following material describes, in overview fashion, the features of the seismic PSA analysis approach taken to evaluate the Hope Creek seismic risk. This discussion includes discussion of the methodology chosen for:

- Seismic Hazard Analysis
- Seismic Fragility Assessment
- Seismic Systems Analysis
- Quantification of the Seismically Induced Core Damage Frequency

3.0.1.1 Seismic Hazard Analysis

Seismic hazard analysis is performed to estimate the annual frequency of exceeding different levels of seismic ground motion at the plant site. The steps of this analysis are as follows:

1. Identification of the sources of earthquakes, such as faults and seismotectonic provinces.

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities.

3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site.

4. Integration of the above information to estimate the frequency of exceedance for selected levels of ground motion.

The hazard estimate depends on uncertain estimates of attenuation, upper bound magnitudes, and the geometry of the postulated sources. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these parameters. A probability distribution for the frequency of occurrence is thereby developed. The annual frequencies for exceeding specified values of the ground motion parameter are displayed as a family of curves with different probabilities; they are presented in terms of median, mean, 15 percentile and 85 percentile curves.

For the Hope Creek site, there are two published site-specific hazard studies (EPRI, 1989a and NRC, 1994b). The results from these studies were used in this IPEEE.

3.0.1.2 Seismic Fragilities

Seismic fragility of a structure or equipment is the conditional probability of its failure for a given level of seismic input parameter e.g., 0.4g spectral acceleration or 0.3g peak ground acceleration. Seismic fragility was first introduced in nuclear plant seismic risk assessment studies. Seismic fragilities have been estimated for structures and equipment in over 30 nuclear power plants in the last 15 years. The methodology is described in a number of papers and reports (Kennedy, 1980a), (Ravindra, 1983a), (Kennedy, 1984a), (NRC, 1983a), (NRC, 1985a), (Casciati and Faravelli, 1991a), (Reed and Kennedy, 1993a) and the results of applications to nuclear power plant seismic PSAs and margin studies are discussed in (Kennedy, 1988a and b), (Kipp, 1988a), (Ravindra, 1987a), and (Ravindra, 1988a).

Seismic fragilities are needed to estimate the frequency of occurrence of initiating events and to quantify the fault trees for obtaining the seismically induced accident sequence frequencies. Seismic fragility is described by means of a family of fragility curves reflecting the uncertainty in the parameter values and in the models. A subjective probability is assigned to each curve representing the degree of belief in the set of parameter values and the model that yielded that curve. It is customary to show the median fragility curve and the 95% confidence fragility curve and the 5% confidence curve (Figure 3-1). Using the double lognormal model, the fragility family is concisely described by means of three parameters: A_m, the median capacity of the component, the logarithmic standard deviation β_R reflecting the randomness in the capacity, and the logarithmic standard deviation β_U reflecting the uncertainty in the median capacity. These parameters are evaluated for each component for all critical failure modes using the design information, earthquake experience database, and qualification and fragility test data. In many applications, it is sufficient to use the mean fragility curve whose parameters are A_m and β_c where β_c is the composite variability given by $(\beta_R^2 + \beta_U^2)^{1/2}$. Figure 3-1 also shows this mean curve.

In this study, seismic fragility is developed in terms of the peak ground acceleration at the site. Therefore, the median capacity A_m is the median ground acceleration capacity. From Figure 3-1, it can be seen that at each peak ground acceleration value, the probability of failure of the component is a variable reflecting the uncertainty in the probability estimate; this is very similar to the failure rate distribution given a demand (the demand in this case is the earthquake of a specified PGA). Based on the double lognormal model for fragility and the parameters A_m, β_R, and β_U, the probability (uncertainty) distribution of the failure probability (also called "failure fraction") could be developed for each peak ground acceleration value.

Another important parameter in the seismic fragility and margin assessment is the High Confidence of Low Probability of Failure (HCLPF) capacity of the component. In this study, this parameter is defined as the acceleration level at which the component has a probability of failure of less than five percent corresponding to a confidence level of 95 percent, i.e.,

$$HCLPF = A_m \exp[-1.65 (\beta_R + \beta_U)]$$

The process of seismic fragility evaluation can be described by the following steps:

1. Based on the preliminary systems analysis and on previous seismic PSAs, a set of structures and equipment (about 100 items) is selected for fragility evaluation.

2. Plant design and seismic qualification information is collected.

3. Probabilistic floor and structural response are developed by analysis or by appropriate extrapolation of the design information.

Perform plant walkdowns to search for seismic vulnerabilities, to assist in screening out high capacity components and to collect additional data on components needing detailed fragility analysis. Procedures for seismic walkdowns are given in the EPRI seismic margin assessment methodology report (EPRI 1991b). Typically, about 100 components are selected for detailed fragility analysis. For each component, the critical failure modes are identified. Past seismic PSAs can be used as a guide in this identification. It is important to relate the failure mode of the component to the consequence on the component's function in the system. The median capacity of the component in each failure mode is estimated using the appropriate data sources (i.e., seismic analysis, qualification test data, fragility test data and earthquake experience data). The randomness and uncertainty variability are also estimated using the same data sources.

3.0.1.3 Systems Analysis

The purposes of seismic system analysis are to define the potential seismic-induced structure and equipment failure scenarios that could occur after a seismic event, to quantify the frequencies of those scenarios, to quantify the conditional core damage probabilities for these scenarios, and to quantify the overall frequency of seismic-induced core damage.

3.0.1.3.1 System Analysis Methodology

The event and fault tree models developed for the Hope Creek Generating Station (HCGS) internal events Probabilistic Safety Assessment (PSA) have been used as the starting point for the seismic IPEEE models (PSE&G, 1994b). Traditional event tree techniques were used to delineate the potential combinations of seismic-induced failures, and resulting seismic scenarios, which were termed "seismic damage states." The frequencies of these seismic damage states were quantified by convoluting the earthquake hazard curve with the structure and equipment seismic fragility curves.

Sensitivity studies which examine the importance of different input information and assumptions for the HCGS site are described in Paragraph 3.1.5.6.

3.0.1.3.2 Non-Seismic Failures and Human Reliability Analysis

For those scenarios that required additional non-seismic failures to occur to result in core damage, the PSA internal events model (event trees and fault trees) was used to develop conditional core damage probabilities, with appropriate changes given the seismic damage state. These calculations incorporate random failures of equipment and operator actions.

Special attention was given to human interactions and recovery actions under seismic conditions. Offsite power recovery within the first 24 hours was not credited. No relay chatter interactions requiring human actions were needed based on the low ruggedness relay evaluation (PSE&G, 1996b). Two operator actions were explicitly included in the seismic event tree analysis: (1) Establishing alternate ventilation to the Class 1E Panel Room after a loss of panel room HVAC, and (2) Safe shutdown from outside the control room (remote shutdown). The alternate ventilation action was based on the similar recovery identified in the internal events IPE/PSA, but the Human Error Probability (HEP) was made more conservative by one order of magnitude to reflect additional stresses placed on personnel by a seismic event. The remote shutdown HEP was specifically derived for the seismic event for the IPEEE.

Several human actions were also credited in the conditional core damage probability (non-seismic failures) calculations. For these human actions, similar to establishing alternate ventilation as described above, the HEP calculated for the IPE/PSA was used, but was conservatively increased by one order of magnitude to reflect additional stresses placed on personnel by a seismic event.

The details of the Human Reliability Analysis are provided in Paragraph 3.1.5.3.2. Sensitivity studies examining the importance of human actions were conducted and are described in Paragraph 3.1.5.6.2.

3.0.1.3.3 Relay Chatter Evaluation Methodology

For IPEEE purposes, Hope Creek is classified as a focused scope plant in NUREG-1407 (NRC-1991b). Hope Creek is a Standard Review Plant. The methodology used for the relay chatter evaluation is specified in the Hope Creek Unit 1 IPEEE Relay Chatter Evaluation Report - PSBP 320808 (PSE&G, 1996b). This methodology is based upon EPRI NP-7148-SL (EPRI, 1990a), Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality, with supporting seismic ruggedness information from EPRI NP-7147-SL (EPRI, 1991c), Seismic Ruggedness of Relays. Relay capacity evaluations for low ruggedness relays (LRR) were performed to the IPEEE review level earthquake, which is 0.3g for Hope Creek. The overall approach is as follows:

1. Develop lists of potential low ruggedness relays based upon the EPRI LRR List.

2. Develop an additional list of relays with unknown relay type and/or manufacturer.

3. Screen out relays based upon no impact to seismic safe shutdown equipment.

4. Evaluate the remaining relays to determine if contact chatter is acceptable.

5. Evaluate those relays for which contact chatter is not acceptable for:

   - Operator recovery action in the Control Room
   - Operator recovery action outside the Control Room.
   - Adequate seismic capacity for the plant-specific location.

6. Perform a seismic capacity walkdown for relays.

This process was developed to address the content of Hope Creek-specific databases and other plant information and to be efficient in the relay evaluation process. The results of this process are discussed in Paragraph 3.1.5.4.

3.0.1.3.4 Quantification of Core Damage Frequency

The seismic event tree (SET) is used to delineate the potential successes and failures that could occur due to a seismic event, based on the structures and components and their fragilities. Boolean equations were developed for each of the SET top events, based on the logic and seismic fragility information. Each seismic sequence equation represents the Boolean logic associated with its corresponding seismic damage state (SDS).

The seismic hazard information, structural/component fragilities, and SDS equations were then input to the NUS SEISMIC code, version 1.1, (NUS, 1993a) to quantify the frequency of the SDSs. The quantification included dependent and correlated failures, and appropriate success states.
The NUS Workstation code (NUS, 1994a) was used to calculate the conditional core damage probabilities (Table 3-8). To obtain the overall results, the frequency of each seismic damage state (SDS) is multiplied by the conditional core damage probability (CCDP) for that SDS. Since each SDS is independent of the others, the total core damage frequency due to seismic events is simply the summation of the individual SDS sequence frequencies.

3.1 SEISMIC PSA

3.1.1 SEISMIC HAZARD ANALYSIS

The seismic hazard defines the probability that specified levels of ground motion will be exceeded at the plant site in a given period of time, usually one year. The results of two comprehensive seismic hazard studies are available for use in seismic risk quantification: Electric Power Research Institute study (EPRI, 1989a) and Lawrence Livermore National Laboratory study (NRC, 1994b). These studies have provided seismic hazard curves in terms of peak ground acceleration at the site and ground response spectra with a mean return period of 10,000 years. The ground response spectra shape is used in the evaluation of the probabilistic seismic response of structures and equipment as described in Paragraph 3.1.3.

Figure 3-1 shows the seismic hazard curves developed by LLNL for the Hope Creek site; shown are the mean, median, 15th percentile and 85th percentile curves. NUREG-1407 (NRC, 1991b) requires that the mean (arithmetic) hazard curve be used to obtain point (mean) estimate of core damage frequency. The LLNL mean hazard curve is used to calculate the core damage frequency as the base case. In order to assess the sensitivity of the insights and conclusions, the EPRI hazard curve (Figure 3-2) has also been used for core damage frequency calculations. Paragraph 3.1.5.6 describes the methods used and the results of this sensitivity study.

For the purposes of uncertainty evaluation, we have used the full uncertainty distribution on the seismic hazard as shown by the 15th percentile and the 85th percentile hazard curves. The results of the uncertainty evaluation are included in Paragraph 3.1.5.6.

Although both EPRI and LLNL seismic hazard curves are provided only up to about 1.0g, they have been extrapolated to 1.5g [Table 3-14 and Figure 3-13] in a sensitivity study and uncertainty discussion (Paragraph 3.1.5.6) in calculating the seismic induced accident sequence frequencies.

3.1.1.1 Soil Liquefaction

3.1.1.1.1 Site Subsurface Conditions

Hydraulic fill, placed as a result of dredging operations in the Delaware River, is typically encountered to depths of 30 feet to 35 feet below the ground surface at the site. The fill consists primarily of silty clay to silty sand but is also composed of irregular and sometimes discontinuous layers and pockets of fine to medium sand and organic material. This soil is generally soft to loose, with blow counts generally between two blows per foot (bpf) and ten bpf.

The hydraulic fill has been replaced by engineered backfill in areas surrounding buildings and safety-related underground piping systems. The engineered backfill extends to the Kirkwood formation discussed below and was specified to be compacted to a minimum of 95% to 98% compaction.

Underlying the hydraulic fill is a two foot to eight foot thick layer of river bottom sand consisting of fine to coarse grained sand with blow counts ranging from about 20 to 85 bpf. This material formerly comprised the bed of the Delaware River.

The clays of the Kirkwood formation are found below the river sand. These clays are typically medium stiff to stiff and vary in thickness from about 20 feet to 30 feet. Underlying the clay are the basal sands of the Kirkwood Formation, which consist of micaceous fine to medium grained sands varying in thickness from about two feet to six feet and having blow counts ranging from approximately 20 bpf to 70 bpf.

The sands of the Vincentown formation, which is a Tertiary formation, are below the basal sands and consist of silty sands in the upper zones to poorly graded sands in the middle zones to silty sands in the lower zones. The Vincentown varies in thickness from about 50 feet to 70 feet. The blow counts in these sands vary over a wide range, from about ten bpf to sampler refusal, but are generally about 30 bpf. The Vincentown sands also have varying degrees of cementation, which may partially account for the significant variation in blow counts throughout the layer. The occasional presence of low blow counts in this formation may be due to localized soildisturbance during the drilling operation. The other soils encountered below the Vincentown formation generally consisted of very dense silty sands.

3.1.1.1.2 Soil Liquefaction Potential

Liquefaction potential was assessed using a probabilistic approach (Woodward-Clyde, 1986a) based on the deterministic semi-empirical procedures developed by Seed and his co-workers as described more recently in Seed and Harder (1990a). The probabilistic liquefaction potential is a function of the following two probabilities:

1. The probability that the soil will exhibit a certain cyclic strength as represented by corrected Standard Penetration Test (SPT) blow counts, $(N_1)_{60}$.

2. The probability of liquefaction given a value of $(N_1)_{60}$ and a cyclic stress ratio associated with a given earthquake shaking level from a given magnitude event.

By combining the above two probability functions, the probabilistic liquefaction potential at the site can be computed. The evaluation of liquefaction potential was limited to the Vincentown formation because the Seismic Category I civil structures at the site are founded on this formation. The compacted fill areas containing piping systems are addressed in Paragraph 3.1.4.4.

For a given statistical distribution of $(N_1)_{60}$, the evaluation of the conditional probability of liquefaction given a shaking level a, $P(L \mid Z=a)$, is performed through a Monte Carlo simulation (e.g., Ang and Tang, 1984a). As illustrated in Figure 3-3, the Monte Carlo simulation involves the random selection of a value of $(N_1)_{60j}$ from a known distribution of this parameter. For each value of $(N_1)_{60j}$ selected randomly, the conditional probability of liquefaction given a shaking level a and $(N_1)_{60j}$, $P(L \mid Z=a, (N_1)_{60j})$, and the probability of $(N_1)_{60j}$ occurring, $P[(N_1)_{60j}]$, are computed. By repeating this process over the entire distribution of $(N_1)_{60}$, the conditional probability, $P(L \mid Z=a)$ of liquefaction can be computed.

The results of the conditional probability of liquefaction for each structure and each distribution of $(N_1)_{60}$ are tabulated in Table 3-2 for the SSE and Table 3-3 for 3 * SSE levels. The input earthquake ground motion was defined by the 10,000-year EPRI median UHS anchored to the SSE and 3 * SSE levels, respectively. These results indicate that the highest conditional probability of liquefaction occurs beneath the Reactor Building (Woodward-Clyde, 1995b - Table 5-7a).

In this report, the terms DBE (Design Basis Earthquake) and SSE (Safe Shutdown Earthquake) are used interchangeably.

3.1.1.1.3 Seismically Induced Settlement

Seismically induced settlement of the Vincentown sands was also assessed using a probabilistic approach based on procedures described by (Tokimatsu and Seed, 1987a). Settlement evaluation were also performed using a Monte Carlo simulation.

The total seismically induced settlement at each location of the structure (corner and center) were computed and are tabulated in Table 3-2 and Table 3-3 for the SSE and 3 * SSE levels. The results indicate that the total seismically induced settlements, at the 84th percentile, are less than 1/4-inch for the SSE level. For the 3 * SSE level, the total settlements could be as high as 1-1/4-inch for the 84th percentile. For the 84th percentile, differential settlements are less than 1/4 inch for the SSE. The maximum differential settlements, up to the 84th percentile, at the 3 * SSE level is about 3/4-inch. As for the conditional probability of liquefaction, highest settlements occur at the Reactor Building. The computed settlements are not a concern to either the structural integrity of the building or the critical piping systems connected to the buildings.

The lateral spreading potential at the site is addressed in Paragraph 3.1.4.4.2.

3.1.2 REVIEW PLANT INFORMATION AND WALKDOWN

3.1.2.1 Plant Information

3.1.2.1.1 General Description of Plant

The Hope Creek Nuclear Generating Station has a boiling water reactor of General Electric BWR4 design, with a Mark I containment, with nominal capacity of 1100 MWe. The station was designed in the early 1970's. Figure 3-4 is a layout of the station showing the major structures which include the reactor building, auxiliary building, turbine building, and service water intake structure. The reactor building consists of a reinforced concrete containment, drywell, concrete internal structure, and suppression chamber (i.e., torus). The auxiliary building consists of a control area, diesel generator area and radwaste area. The turbine building, which is not a Category I structure, was included in the IPEEE study since it houses the instrumentation air system (IAS). Seismic Category I structures are founded at the top of Vincentown Formation. Other structures considered in the IPEEE study are the ground-mounted storage tanks, i.e., condensate storage tank and firewater tanks.

3.1.2.1.2 Seismic Design Basis

The Hope Creek Nuclear Generating Station was designed in the 1970's in accordance with criteria and codes in effect at that time. The design criteria included the effects of simultaneous earthquake and loss-of-coolant-accident (LOCA) conditions. The plant was designed to withstand both a Safe Shutdown Earthquake (SSE) and an Operating Basis Earthquake (OBE). The chosen levels of SSE and OBE were 0.20 g and 0.10 g peak horizontal ground accelerations for all Seismic Category I structures, systems and components. The maximum vertical ground acceleration was specified as two-thirds of the horizontal acceleration. The free-field ground response spectra shown in Figures 3-5 and 3-6 were used in the design of Hope Creek Generating Station. These are the USNRC Regulatory Guide 1.60 [NRC, 1973b] ground response spectra anchored to 0.2g SSE level.

3.1.2.1.3 Review Of Documents

The following plant seismic design documents were reviewed to gain an understanding of the plant layout and seismic qualification of critical equipment:

- Updated Final Safety Analysis Report (UFSAR)
- As-built structural drawings
- Seismic qualification reports for different equipment
- Representative equipment anchorage calculations
- Selected equipment specifications

3.1.2.2 Plant Walkdowns

3.1.2.2.1 Walkdown Procedures and Areas

Plant walkdowns were performed following the procedures given in the EPRI Seismic Margins Methodology report (EPRI, 1991b). The first walkdown was conducted in October 1992 during a refueling outage. Subsequent walkdowns were performed to review components outside the RCA that were not reviewed during the initial walkdown and the components in the containment.

The first activity in the walkdown preparatory work was the development of a components list which included:

- Critical components as identified in the internal events PSA model.
- Components in the containment systems needed for the containment performance evaluation.
- Components in the systems needed to be addressed for other issues (subsumed programs, seismic induced fires and floods).
- Certain passive components that may not be explicitly included in the internal events system failure models because of low failure rates, but that could have significant conditional probabilities of failure from seismic events.
- Components that could inadvertently change position during a seismic event and cause a flow diversion.
- Instrumentation, racks, cabinets, transformers, switchgear, motor control centers, and panels that provide essential signals, power, or control room indication.
- All structures that house the components identified above.
- The Walkdown considered relay mounting but it was not practical to evaluate chatter potential.

For each component in the list, a specific walkdown data sheet was prepared to allow a methodical examination and documentation. These seismic evaluation walkdown sheets (SEWS) were filled out in the office as much as possible such that the time for filling out the form in the plant was minimized.

The purpose of the walkdowns were to:

- Pre-screen all equipment items that have sufficiently high seismic capacities.
- Clearly define the failure modes of elements which are not prescreened. Review and gather detailed information and measurements on equipment and structures for performing seismic fragilities.
- Identify spatial system interaction (SI) concerns that are judged to be potentially serious problems (such as heavy, questionably secured space heaters or ceiling fixtures over critical batteries, etc.).
- Evaluate the fire protection systems in the plant for seismic induced fire and inadvertent actuation of fire protection system issues.

The following criteria were used to screen out certain components from the IPEEE list of components based only on the walkdown:

- The majority of the valves (MOVs and AOVs) were screened out in the walkdown so long as no adverse system-interactions were observed and the valve operator heights, yokes and operator mounting met the GIP caveats (SQUG, 1992a). Although EPRI NP-6041 (EPRI, 1991b) recommends that only representative piping runs and associated valves be included in the walkdown, a detailed walkdown was conducted for all accessible valves identified by the systems analysts. The focus was to ascertain that no potential systems interactions (e.g., impact of valve operators on adjacent structures) exist. The large sample of accessible valves examined in this walkdown has provided sufficient basis for screening out valves except as noted on the walkdown data sheets.
- Horizontal pumps and compressors could generally be screened out irrespective of their locations in the buildings.

- Wall or ceiling mounted small instruments (e.g., pressure transmitters and temperature sensors) were screened out based on a sampling review of similar instruments. Some instruments mounted in floor mounted enclosures were not screened out because of concerns of anchorage of the enclosures.

- For components on redundant safety trains, the walkdown focused on one train; the other trains were "walked-by" to verify similarity and that no adverse systems interaction issues exist. The screening of components or development of fragilities as equal for all redundant components is conservative.

- Distributed subsystems such as piping, cable trays and HVAC ducts were screened out based on the walkdown observations of representative runs.

3.1.2.2.2 Walkdown Team

The Hope Creek seismic walkdown team consisted of the following members:

- PSE&G Personnel: The PSE&G personnel participating in the seismic walkdowns provided plant knowledge and PSA expertise. The walkdowns were also used for training the PSE&G staff for future seismic evaluations. The utility team consisted of:

  A. J. Sanders
  W. T. Weir

- Halliburton NUS (now Scientech): The primary responsibility of the systems engineer is to identify all components and structures for which fragility estimation would be required. The systems engineers on the walkdown team were knowledgeable about the power plant equipment, normal and emergency operating procedures, and operator response to abnormal situation. The team consisted of:

  D. L. Moore
  V. A. Mohrenweiser
  L. Sarmanian

- *Seismic Capability Engineers* (EQE): The main activity for the seismic capability engineers was to review all components on the IPEEE list identified by the systems engineers and to establish whether the components could be screened out during the walkdown or if additional field data must be obtained in order to perform fragility analyses of the components. The team consisted of:

M. K. Ravindra
W. H. Tong
R. W. Cushing

These engineers have performed several seismic walkdowns and have completed the SQUG Walkdown Screening and Seismic Evaluation Training Course.

3.1.2.2.3 Walkdown Findings

During the conduct of the walkdown, the walkdown findings were recorded on SEWS forms. A summary of walkdown screening information for the IPEEE equipment is provided in Table 3-4 including the system and equipment identifications, equipment locations, and the equipment screening results.

The screening of components was verified during the walkdown as to whether they meet the caveats given in EPRI NP-6041 (EPRI, 1991b). Major attention was paid to the anchorage details. For a selected number of equipment items, anchorage calculations were made to verify that their HCLPF capacities exceed 0.5g PGA. This selection followed the guidance given in Appendix D of EPRI NP-6041 (EPRI, 1991b). For the components that could not be screened out, fragility calculations were made which also focused on the anchorage of the equipment [EQE, 1996b].

Photographs were also used to record details of the walkdown. Photographs provide a permanent record of what was reviewed and support any notes or details taken during the walkdown. System interaction concerns were typically documented with photographs. The completed SEWS forms and photographs are included in a five volume Hope Creek Walkdown Report (EQE, 1996a).

3.1.2.2.4 Walkdown Documentation

Walkdown documentation for equipment and structures consisted of recording the findings using SEWS forms and photographs. The SEWS forms were developed for each particular class of equipment indicating specific information required to confirm the high seismic capacity of the component in place as well as to record details sufficient to perform a seismic fragility evaluation if necessary.

Photographs were also used to record details of the walkdown. Photographs provide a permanent record of what was reviewed and support any notes or details taken during the walkdown. System interaction concerns were typically documented with photographs. The completed SEWS forms and photographs are included in a five volume Hope Creek Walkdown Report (EQE, 1996a).

Walkdowns were also conducted in support of the Fire Risk Scoping Study Issues Assessment (PSE&G, 1997c). Three concerns addressed were the potential for seismically-induced fires, the potential for seismically-induced actuation of fire suppression systems, and the potential for seismically-induced degradation of fire suppression systems. It was concluded that the first two areas present no significant new challenges and that the potential unavailability of fire water after an earthquake is the principal mode of seismically-induced fire suppression degradation. The effects of fire protection system actuation on safety-related equipment (GL-57) is also discussed in Paragraph 4-10.

3.1.2.3 Screening of Components

In a seismic PSA, the plant walkdown is used to collect as built information of the equipment and to observe any conditions (e.g., spatial systems interaction or deficiencies such as missing bolts, supports or excessive concrete cracking or corrosion) that may render the components seismically vulnerable. Based on the walkdown and review of design/qualification information, seismically high capacity components are screened out in order to reduce the seismic PSA systems analysis and quantification effort (Table 3-4).

The screening of Hope Creek components was done in the following two stages by the seismic capability engineers:

1. At the first level of screening, only the component seismic capacity was used as an indicator of the seismic risk contribution. This was because the realistic Hope Creek building response (EQE, 1995a) was not generated at the time of the walkdowns. The walkdowns identified those components

that are seismically rugged regardless of where they are located in the buildings. Valves, horizontal pumps and distribution systems generally come under this group.

2. A second level of screening was performed after the Hope Creek probabilistic floor response spectra (EQE, 1995a) were developed. Certain components were reviewed and were judged to have high seismic capacities with respect to the realistic seismic demands, and were screened out as not requiring further fragility evaluations. This screening employed a median peak ground acceleration of 1.5g which corresponds to a 0.5g peak ground acceleration HCLPF capacity as the screening level.

For the remaining components, either specific or generic fragilities were developed with the Hope Creek median floor response spectra and incorporated into the quantification process.

Parametric studies conducted using the seismic hazard curves (EPRI and LLNL developed) and generic component fragility curves indicated that components with median seismic capacities larger than 1.5g PGA (equivalently, the HCLPF capacities larger than 0.5g PGA) do not significantly contribute to the seismic core damage frequency (their contribution to mean core damage sequence frequencies are less than 1×10^{-6} per year). The walkdown focused on these components to ensure that they indeed have such high seismic capacities, i.e., there are no potential spatial systems interactions that would jeopardize the functionality of the component and that the existing condition of the component in terms of anchorage and lateral seismic support does not lower the seismic capacity below the screening levels.

The functionality of the screened-out components was verified in the following way. The screening tables in EPRI NP-6041 (EPRI, 1991b) give the caveats to be satisfied in order to claim that the components have HCLPF capacities up to 0.5g PGA (or equivalently, 1.2g spectral acceleration for 5% damping). The screening performed in the Hope Creek IPEEE study ensured that the caveats given in EPRI NP-6041 (corresponding to the range between 0.8g and 1.2g spectral accelerations) were satisfied for each component. These screening tables have been developed on the observations and judgment that those components meeting the caveats shown in these tables generally have the corresponding High Confidence of Low Probability of Failure (HCLPF) seismic capacities. The 50 percentile EPRI uniform hazard spectrum (UHS) has an amplification of peak ground acceleration of about 1.74 (peak spectral acceleration at 5% damping/peak ground acceleration), so meeting the EPRI NP-6041 screening guidelines for the 0.8g to 1.2g spectral acceleration bin results in a HCLPF relative to the 50 percentile EPRI UHS of 1.2/(exp(0.2))/1.74 = 0.56g PGA. The exponent of 0.2 is used to convert the HCLPF spectral acceleration from the 84 percentile space to the 50th percentile space. The corresponding median PGA capacity is about 1.5g. Thus, those components which were screened out can be considered to have a HCLPF of greater than 0.5g and a median capacity of greater than 1.5g.

The walkdown focused on these components to ensure that they indeed have such high seismic capacities, i.e., there are no potential spatial systems interactions that would impair the functionality of the component and that the existing condition of the component in terms of anchorage and lateral seismic support does not lower the seismic capacity below the screening levels.

In the second stage of screening, the median and the 84 percentile floor response spectra were used to further screen the components by:

- Demonstrating that the median PGA capacity of the component is greater than 1.5g by comparing the median floor response spectra with the design floor response spectra, or

- Demonstrating that the HCLPF capacity is greater than 0.5g by comparing the 84 percentile floor response spectra with the design floor response spectra.

Another requirement of the screening is that the anchorage of the screened equipment is adequate to provide a HCLPF capacity of 0.5g PGA. Calculations performed using the 84% floor response spectra from (EQE 1995a) for flexible equipment showed that there is a (deterministic) margin of at least 2.5 over the SSE of 0.2g PGA. The major contributor to this margin is the reduced floor spectra arising from realistic analysis and site-specific ground motion input.

3.1.3 PLANT STRUCTURE RESPONSE ANALYSIS

3.1.3.1 Structure Response

3.1.3.1.1 Plant Structural Models

Probabilistic seismic response analysis (EQE, 1995a) was performed for the following buildings using the 10,000 year median Uniform Hazard Spectra developed by (EPRI 1989a) and (NRC, 1994b):

- Reactor Building
- Auxiliary Building
- Turbine Building
- Service Water Intake Structure

The end products of the probabilistic response analysis are probability distributions of in-structure response, i.e., loads in structural elements and floor response spectra which define the seismic demand on equipment housed in buildings. Ordinarily, probabilistic seismic response analyses are performed for a number of different free-field peak ground acceleration levels, usually multiples of the SSE level, and the acceleration levels at which failure of structures and equipment are calculated to occur are obtained by interpolation, or more commonly by extrapolation of the analytical data. It has been shown in past studies that reasonable estimates can be obtained by performing analyses for a single acceleration level that is near the failure levels of critical equipment and assuming that the probabilistic response is linearly proportional to the acceleration level. Therefore, probabilistic response analysis was performed for free-field input motions selected to match the 10,000 year UHS shape anchored to 3 x SSE (i.e., 0.60g peak ground acceleration). An ensemble of time histories was generated such that their median response spectra match the median 10,000 year EPRI UHS. Variability in the time histories corresponds to the peak to valley variability in real earthquake ground motion spectra. Thirty earthquake motions, three components each, were generated such that their median 5% damped spectra matched the EPRI UHS with a coefficient of variation of 0.20.

The Soil Structure Interaction analysis (EQE, 1995a) utilized the substructure approach; structural models for this approach are fixed-base and SSI effects are incorporated using foundation impedance and wave scattering functions. Structural models were recorded from the dynamic building modes used for the design analyses (Impell, 1983a). These models are more elaborate than typical stick models consisting of several vertical sticks coupled with horizontal members representing floor slabs. The modal damping ratios used for the building models were the upper bound damping values from NUREG/CR-0098 (Newmark and Hall, 1978a) corresponding to the at-yield values. The variability in soil and structure properties was incorporated in the probabilistic response analysis by performing a Latin Hypercube Simulation from lognormal probability distributions with the following coefficients of variations:

Soil shear modulus:	0.35
Soil material damping:	0.50
Structural frequencies:	0.25
Structural modal damping:	0.35

The median and 84% non-exceedance probability (NEP) responses were calculated for each selected in-structure response. These included peak accelerations, maximum member forces and floor response spectra at chosen elevations as needed for equipment fragility estimation.

3.1.3.1.2 Ground Response Spectra

The median ground response spectrum which provides the basis for the probabilistic response analysis (EQE, 1995a) was selected as the 50th percentile, 10,000 year return period spectrum from EPRI (1989a). This spectrum is smoothed to eliminate the peaks and valleys inherent in the response spectra resulting from natural earthquakes. The time histories selected from the probabilistic response analysis correspond approximately to the median response spectrum. The variability in the response spectra considered the peak-to-valley variability (randomness β of 0.2). It was assumed that the uncertainty in the ground motion input is accounted completely through the uncertainty in the seismic hazard curves (i.e., PGA hazard curves). This uncertainty is the result of the uncertainty in the hazard modeling expressed as opinions by the hazard experts. Some differences in the shapes of 85 percentile and median uniform hazard spectra at 10,000 year return period were observed leading one to suspect that there is uncertainty in the spectral shape. However, the effect of this uncertainty was considered to be small for the following reasons: 1) the difference in the spectral shape seen for the 5% damped spectra may not be relevant to Hope Creek because of the high composite soil-structure damping (over 10%) present for Hope Creek, 2) the uncertainty arises because of the uncertainty in the spectral attenuation relationships which is already considered in the PGA attenuation relationships and 3) the effect of adding the spectral shape uncertainty is not significant considering the overall uncertainties in the fragilities and hazard curves.

The probabilistic response analyses of the plant structures were performed for the 3*SSE level with the median horizontal PGA of 0.6g and vertical PGA equal to 2/3 the horizontal. The selection of the control point used for the specification at the input motion in the SSI analysis was based on analysis of the free-filed response. This analysis showed that the top soil layer (hydraulic fill) is very soft and incapable of amplifying higher frequency input. Therefore, the control point was taken on an outcrop at the top of the River Bottom Sand soil layer (Elevation about 70 ft) immediately below the hydraulic fill. This is compatible with the EPRI seismic hazard analysis report (EPRI, 1989a), the designation of the Hope Creek site as a deep soil site and methods of developing site amplification factors.

3.1.3.1.3 Floor Response Spectra

Figures 3-7 through 3-9 show floor response spectra at 5% damping at selected elevations in the reactor building , the auxiliary building and the service water intake structure. A comparison of these spectra with those of the free-field motions shows the building responses to be significantly reduced. The median PGA's of the free-field motions are about 0.6g while the median ZPA's in the buildings is about 0.2g to 0.3g. This significant reduction of the response of power block structures (reactor buildings and auxiliary buildings) is, in part, attributable to the low frequencies of the SSI system and the spectral shape of the EPRI uniform hazard spectrum. Another contributing factor to the reduced response is the large size of the foundations and their radiation damping. A third factor is wave scattering due to embedment effects.

The reduction in the building response of the intake structure is due primarily to embedment effects. Its SSI system frequency is about five Hertz. In this frequency range, the reduction in the foundation input motion due to wave scattering is about 50% to 60%. For foundations with the same embedment ratio as the intake structure (E/R of about 2/3) this reduction is not uncommon.

3.1.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

Seismic fragilities of structures and equipment were estimated using the procedures described in, (Kennedy, 1984a) and (Reed and Kennedy, 1993a). As stated in Paragraph 3.0.1.2, seismic fragilities in this study have been developed in terms of the peak ground acceleration capacity of structures and equipment. As such, the three fragility parameters A_m, β_R and β_U have been calculated for each screened-in component in its significant failure modes. A brief description of the methods used to calculate the fragility parameters and the results are given in the report by EQE, (EQE, 1996b).

3.1.4.1 Structure Fragilities

Seismic fragilities of important structures, tanks, and block walls were estimated for significant failure modes using a combination of new probabilistic response analysis (EQE, 1995a) and existing analysis together with a knowledge of the SSE design criteria utilized. Structures were deemed to fail when their inelastic deformation exceeds the level which interferes with the operability of the equipment housed inside or mounted on the structure. In some instances, structures were considered to fail when the sliding displacements exceeded the deformation capability of attached piping. Tanks were considered to fail when they loose their contents. Block walls were deemed to fail when they either collapse on adjacent components or suffer large deformations that may interfere with the functionality of attached equipment.

For each structure, the ground acceleration capacity in the critical failure mode was estimated as a safety factor times the SSE peak ground acceleration. The factor of safety was estimated using the conservatism and non-conservatism inherent in the design, qualification and construction of the structure. It was separated into factors on capacity and response.

The factor of safety for the structure seismic capacity consisted of:

1. The strength factor, F_S, based on the ratio of actual member strength to the design forces.

2. The inelastic energy absorption factor, F_u, related to the ductility of the structure and to the earthquake magnitude range that is believed to contribute most to the seismic risk.

The factor of safety, F_R, related to building response was determined from a number of variables which include:

1. The response spectra used for design compared to the median centered spectra judged to be appropriate for the Hope Creek site.

2. Damping used in the analysis compared with damping expected at failure.

3. Modal combination methods.

4. Combination of earthquake components.

5. Modeling accuracy.

6. Soil-structure interaction effects.

Median factor of safety, F_m, and variability, β_R and β_U, estimates were made for each of the parameters affecting capacity and response. These median and variability estimates were then combined to obtain A_m, β_R and β_U in order to develop the fragility curves for the structure under consideration.

The median capacities of all the structures, including the reinforced masonry walls in the turbine building, were estimated to exceed 1.5g PGA. The controlling failure mode for the condensate storage tank was found to be the buckling of the tank shell near the base. The median acceleration capacity for this failure mode was determined to be 0.95g with the associated randomness and uncertainty variability of 0.27 and 0.36, respectively.

3.1.4.2 Equipment Fragilities

The procedure used in deriving the equipment fragility was similar to that used for structures in that median factors of safety and their variability were first developed for equipment capacity and equipment response. These two factors, along with the median factor of safety on structural response, were then multiplied together to obtain an overall median factor of safety for the equipment item:

$F = (F_C)(F_{RE})(F_{RS})$

F_C is the capacity factor of safety for the equipment relative to the floor acceleration used for design, F_{RE} is the factor of safety inherent in the computation of the equipment response, and F_{RS} is the factor of safety in the structural response analysis that resulted in the floor spectra for equipment design. The overall factor of safety F is multiplied by the SSE PGA to obtain the ground acceleration capacity of the equipment.

The following categories of failure modes were considered in the equipment fragility evaluation:

1. Elastic functional failures
2. Brittle failures
3. Ductile failures

Elastic functional failures involve the loss of intended function while the component is stressed below its yield point. Examples of this type of failure include: elastic buckling in tank walls or component supports; chatter and trip in electrical components; excessive blade deflection in fans; and shaft seizure in pumps.

Brittle failures are defined in this study as those failure modes which have little or no system inelastic energy absorption capability. Examples include: anchor bolt failures; component support weld failures; and shear pin failures.

Ductile failure modes are those in which the structural system can absorb a significant amount of energy through inelastic deformation. Examples include: pressure boundary failure of piping, structural failure of cable trays, and structural failure of ducting.

Table 3-5 summarizes the seismic fragilities developed for the equipment included in the systems models. The table identifies the equipment and lists the values of A_m, β_R and β_U. Comments indicate the basis for the fragility derivations, i.e., analysis, earthquake experience database, PSA database, etc. When the capacity is listed as greater than 1.5g, the component was screened out either by demonstrating that the HCLPF capacity exceeded 0.5g PGA or that the median ground acceleration capacity exceeded 1.5g. β_R and β_U values are not specified for these components.

3.1.4.3 Relay Chatter Fragilities

PSE&G searched for low ruggedness relays present in plant systems-equipment modeled in the IPEEE. If such relays were discovered, analyses were performed to assess whether the electrical circuitry was sensitive to the chatter of these relays and if they could be recovered from changes of state and associated false alarms or other problems. For those essential relays (i.e., relay contact chatter affects the system in an unrecoverable way), EQE was to examine their seismic adequacy and develop the necessary seismic fragilities. The only LRRs of concern are the SSC-T relays associated with the diesel generator backup lockout relaying for over-current which could potentially prevent a running diesel generator from tripping on over-current as designed.

The Hope Creek IPEEE Seismic Walkdown Report (PSE&G, 1996s - Paragraph 3.3.2) discusses the relay review performed during the walkdown. It concludes that the overall review of relays and mountings provides high assurance that essential relay are properly mounted.

3.1.4.4 Soil Failures

3.1.4.4.1 Slope Stability

The site is generally level with no significant natural or constructed slopes beyond the shoreline. The shoreline consists of riprap slopes at the southern portion of the site, vegetated slopes between the Salem and Hope Creek Service Water Intake Structures (SWISs), and a bulkhead north of the Hope Creek SWIS. These site conditions indicate that flow failures, typically associated with steep slopes are not a concern.

3.1.4.4.2 Lateral Spreading

Lateral spreading is characterized primarily by horizontal displacement of surficial soil layers as a consequence of liquefaction. The potential for lateral spreading of the hydraulic fill layer at the site was assessed using the empirical equations proposed by Youd (Youd, 1994a). A reduction in the lateral spreading displacements is anticipated in the vicinity of the engineered backfill areas, such as the service water trenches, thereby reducing the displacements experienced by the critical piping systems. These site conditions were taken into account by modeling the engineered fill soil as a block on top of the Kirkwood clay, assuming that the hydraulic fill around the compacted soil liquefies, and evaluating the potential earthquake-induced displacements of the block along the compacted soil-Kirkwood clay interface. The results of this analysis, along with the results of the lateral spreading displacement analysis and engineering judgment, were used to assess the ground surface displacements in the vicinity of the critical piping systems for several earthquake-induced peak ground accelerations.

The estimated lateral spreading displacements of the hydraulic fill layer as a function of peak ground acceleration are shown in Figure 3-10. The estimated earthquake-induced displacements of the compacted fill are shown in Figure 3-11. These relationships are employed to evaluate the lateral displacement potential for soil and seismic failure potential for buried piping and structures important to safety (Woodward-Clyde, 1995a and b).

3.1.4.4.3 Soil Liquefaction Evaluation

Woodward-Clyde Consultants performed a probabilistic evaluation for liquefaction potential, seismically induced settlement and lateral spreading at the Hope Creek and Salem sites (Woodward-Clyde, 1995 a and b). The computed probabilities of liquefaction and seismically induced foundation settlements are very small even at a peak ground acceleration as high as 0.6g. This is due to the fact that the Vincentown formation is very old (Tertiary) and has very high (about 2000 fps) shear wave velocity.

The liquefaction evaluation was performed for the Hope Creek buildings considered in the IPEEE study. At the peak ground acceleration of 0.6g, the conditional probabilities of liquefaction with a 84th percentile confidence level varied from 1.68E-02 at the service water intake structure to 4.95E-02 at the reactor building. The corresponding maximum differential settlements were estimated to be 0.32" and 0.83", respectively. The HCLPF capacities (in terms of PGA) for liquefaction were estimated to be 0.5g based on the liquefaction evaluation results (Woodward-Clyde, 1995b). The HCLPF value for an extensive liquefaction at the Vincentown formation such that the formation may lose its load bearing strength was estimated to be in excess of 0.6g PGA.

It is expected that the hydraulic fill near the plant grade level may liquefy at an acceleration level much lower than 0.6g. This would potentially result in an increase of lateral pressure exerting on the subgrade walls of the building structures. However, the seismic fragility evaluation of the structures has included the effects of hydrostatic and hydrodynamic pressure and the static and dynamic lateral earthquake pressure acting on the below grade exterior walls. Thus, liquefaction of the hydraulic fill was judged not to have a significant impact on the seismic fragilities of the Hope Creek structures.

The buried portion of the service water piping in the yard is contained in the service water trench extending from the service water intake structure to the control/diesel building. The trench was excavated to Elevation 60 feet and backfilled with compact soil to 98% compaction. The pipe trench is running primarily perpendicular to the bank of Delaware River. The service water piping is 36 inches in diameter and was constructed of pre-stressed concrete with an embedded steel cylinder complying to American Water Works Association C-301. The HCLPF capacity of the piping was evaluated by the following steps:

- Estimated the lateral ground deformation capacity of the piping using the minimum angular rotation capacity of the pipe joints and an assumed deformation pattern for the pipe segment located within the deformation zone. The minimum deformation capacity perpendicular to the pipe was determined to be 0.62 feet.

- This deformation was related to the peak ground acceleration using the soil liquefaction and slope stability evaluation results (Woodward-Clyde, 1995b). The peak ground acceleration to produce the 0.62 feet ground deformation was estimated to be 1.48g.

The high HCLPF capacity for Hope Creek service water piping is a direct result of the ground deformation for Hope Creek being half of that for Salem.

3.1.4.5 HCLPF Calculations

For the screened-in components, the HCLPF capacity was calculated and reported in Table 3.5. Note that this HCLPF capacity is what is conventionally known as the $HCLPF_{50}$. In the seismic margin approach, a ground motion spectral shape that represents an 84% non-exceedance level is used whereas a median or 50% non-exceedance level is used in the seismic fragility analysis. The seismic fragilities were calculated using the median site-specific 10,000 year uniform hazard spectrum shape. This is a smoothed spectrum. The 84% non-exceedance spectrum could be constructed by taking into consideration the variability due to peaks and valleys in the spectrum derived from real earthquakes. The logarithmic standard deviation of this random variable was estimated to be 0.20. Therefore, the HCLPF corresponding to the 84% non-exceedance spectrum is obtained as exp (-0.20) [$HCLPF_{50}$] = 0.82 [$HCLPF_{50}$].

The HCLPF capacities of accident sequences and plant systems are described in Paragraph 3.1.5.

3.1.5 ANALYSIS OF PLANT SYSTEMS AND SEQUENCES

3.1.5.1 Seismic Screening of Plant Structures and Equipment

A systematic analysis of potential seismic failures and impacts was conducted prior to the construction of seismic event trees and fault trees. This was necessary to prevent unnecessary calculations on insignificant contributors to seismic core damage frequency (CDF) and risk.

Most structures and components that were included on the walkdown list were screened out based on their high seismic capacities. The screening criteria were: 1) median acceleration capacity greater than 1.5g, and 2) HCLPF greater than 0.5g.

These criteria were chosen because parametric studies conducted using the EPRI and LLNL seismic hazard curves and generic component fragility curves indicated that components with median seismic capacities larger than 1.5g PGA or HCLPF capacities larger than 0.5g PGA do not significantly contribute to the seismic core damage frequency (EQE, 1996b).

The seismic-induced failures that were not screened based on high capacity are summarized in Table 3-6, Hope Creek Seismic Fragilities, and documented in the HCGS Seismic Fragilities Report (EQE, 1996b). This report includes the plant systems providing "level 1" safety functions, as well as the structures, equipment, and actuation components necessary for the "PSA Level 2" functions of containment integrity, containment pressure suppression, containment heat removal, containment radioactivity removal, and containment isolation.

Of the screened-in components presented in Table 3-6, the following were not evaluated in the Systems Analysis/Quantification (PSE&G, 1996a):

- Firewater Tank and Firewater Pumps: In the IPE/PSA fault tree models, firewater was given minimal credit in preventing core damage. To simplify the HCGS IPEEE quantification, firewater is given no credit in preventing core damage.

- Air Operated Valve 1EGHV-2325H: This valve supplies/isolates SACS flow to the Core Spray room cooler 1H-VH211. In the HCGS IPE, a room heat-up calculation for the Core Spray room showed that ventilation was not required for pump operation within the IPE/IPEEE mission time of 24 hours (PSE&G, 1994a).

- Battery Room Exhaust Fans 1A/B-V-416: Room heat-up calculations performed for the HCGS IPE demonstrated that these fans are not required for equipment operation within the 24 hour mission time (PSE&G, 1994a).

- Diesel Generator Area Battery Room Exhaust Fans 1A/B/C/D-V-406: Room heat-up calculations performed for the HCGS IPE demonstrated that these fans are not required for equipment operation within the 24 hour mission time (PSE&G, 1994a).

3.1.5.2 Seismic Event Trees and Seismic Damage States

3.1.5.2.1 Development of the Seismic Event Tree

The seismic event tree (SET), depicted in Figure 3-12, is used to delineate the potential successes and failures that could occur due to a seismic event, based on the structures and components listed in Table 3-6. The SET only treats seismic-induced failures. Success of equipment in the SET does not imply success from non-seismic causes. Non-seismic failures, such as random failure of a pump or an operator error, are included in the overall quantification (Paragraph 3.1.5.3), but not explicitly in the SET evaluation.

Potential impacts of seismic-induced relay chatter are not included in the SET model or the quantification, based on the relay chatter evaluation (Paragraph 3.1.5.4). Additionally, the Reactor Protection System (RPS) and reactor vessel internals were determined to be screened from further analysis based on their high seismic capacity. Therefore, it was not necessary to include seismic-induced ATWS events in the quantitative analysis.

The definitions of the top events in the SET are shown in Figure 3-12.

3.1.5.2.2 Seismic Event Tree Sequence Quantification

Boolean equations were developed for each of the SET top events, based on the logic and seismic fragility information discussed previously. Table 3-6 provides a cross reference between the abbreviations used in the equations, the structure/component description, and the fragility information. The failure equations for each top event are:

S = (no equation needed since this is the seismic event)
HV = PNLHVC * HVREC
DC = 125Vdc
IC1 = PNL481
IC2 = PNL482 * RSDOWN
OP = SWYRD
CR = CREFA * CRS * RSDOWN
HP = 250MCC + 250BUS
CT = CSTNK
CV = CNTVNT
S2 = SLOCA

These equations, which represent the seismic failure of structures and components, are then combined into the seismic sequence equations as delineated by the SET. Both failures and successes are included in these seismic sequence equations. Each seismic sequence equation represents the Boolean logic associated with its corresponding seismic damage state (SDS). The complete detail on sequence equations may be found in the PSE&G Seismic Quantification Report (PSE&G, 1996a).

The seismic hazard information, structural/component fragilities, and SDS equations were then input to the SEISMIC code (NUS, 1993a) to quantify the frequency of the SDSs. In essence, the SEISMIC code uses a Monte Carlo sampling process at each seismic magnitude interval to combine the seismic hazard frequency information with the seismic fragility information for each structure/component in the SDS equation. Successes, failures, and Boolean intersects are treated in this calculation. The code repeats this process for each seismic magnitude, and then sums the results to obtain the SDS frequency. This process is then repeated for each SDS equation until all equations are quantified. The concepts and algorithms used in the SEISMIC code are documented in the SEISMIC User's Manual (NUS, 1993a).

The results of the SDS quantification are presented in Table 3-7, Hope Creek Seismic Damage State Frequencies. While the values presented in Table 3-7 are given to two significant figures, it should be noted that values less than 1E-7 should be regarded as order of magnitude estimates based on the sample size of 1,000 used in the Monte Carlo sampling process. Since these sequences are insignificant to plant risk, and therefore do not pose any plant vulnerabilities, more precise calculations through increased sample size were not required.

The quantification of non-seismic failures use these SDS frequencies as initiating event frequencies, including the seismic failures as house events (guaranteed failures).

3.1.5.3 Non-Seismic Failures and Human Reliability Analysis

3.1.5.3.1 <u>Non-Seismic Failures</u>

For those Seismic Damage States (SDSs) with a frequency greater than 1E-7/year, the impact on the plant and plant systems was evaluated, using the internal events PSA model and dependency matrices as the primary basis. Only 18 SDSs met this criterion, as shown in Table 3-8, Hope Creek Seismic Core Damage Frequencies. Of these 18 SDSs, four (SDSs 35, 36, 37 and 38) result directly in core damage and loss of containment heat removal systems. Therefore, no Conditional Core Damage Probability (CCDP) calculation of non-seismic failures is needed, since the plant and containment damage states are delineated. As presented in Table 3-8, the CCDP, given the seismic failures, is 1.0, or a guaranteed failure.

The internal events PSA models were used to determine CCDPs for the remaining SDSs, as discussed below. The complete detail of these calculations can be found in the Hope Creek seismic quantification report (PSE&G, 1996a).

SDS 2 (S-S2) is a seismic-induced small LOCA, with no other equipment damaged by the seismic event. Therefore, the CCDP is taken from the HCGS Probabilistic Risk Assessment (PSE&G, 1994b). The total CDF for the Small LOCA initiating event when all Human Error Probabilities (HEPs) are as presented in Paragraph 3.1.5.3.2 is 5.2E-7/year. The S2 initiating event frequency is 8E-3/year, so the CCDP given a small LOCA is 5.2E-7/year / 8E-3/year = 6.5E-5.

SDS 3 (S-CV) is a seismic-induced failure of containment venting. The CCDP is 5.8E-5, with the dominant failures being random failures of high pressure injection and reactor de-pressurization, followed by random failures of decay heat removal which result in core damage due to the unavailability of venting.

SDS 5 (S-CT) is a seismic-induced failure of the Condensate Storage Tank. The CCDP is 4.2E-5, with the dominant failures being random failures of high pressure injection and reactor de-pressurization.

SDS 9 (S-HP) is a seismic-induced failure of 1E 250Vdc (high pressure injection). The CCDP is 4.8E-2, with the dominant failures being random failures to de-pressurize the reactor.

SDS 18 (S-OP) is a seismic-induced Loss of Offsite Power (LOOP), with no other seismic failures. The CCDP is 2.1E-3, with the dominant failures being random failures of the Emergency Diesel Generators (EDGs) and their support systems, which result in a Station Blackout (SBO).

SDS 19 (S-OP-S2) is seismic-induced LOOP and small LOCA. The CCDP is 2.1E-3, with the dominant failures being random failures of the EDGs and their support systems, which result in a SBO.

SDS 20 (S-OP-CV) is a seismic-induced LOOP and failure of containment venting. The CCDP is 2.1E-3, with the dominant failures being random failures of the EDGs and their support systems, which results in a SBO.

SDS 22 (S-OP-CT) is a seismic-induced LOOP and failure of the Condensate Storage Tank. The CCDP is 2.1E-3, with the dominant failures being random failures of the EDGs and their support systems, which results in a SBO.

SDS 24 (S-OP-CT-CV) is a seismic-induced LOOP with failure of the Condensate Storage Tank and Containment Venting. The CCDP is 2.1E-3, with the dominant failures being random failures of the EDGs and their support systems, which results in a SBO.

SDS 26 (S-OP-HP) is a seismic-induced LOOP with failure of 1E 250Vdc (high pressure injection). The CCDP is 5.1E-2, with the dominant failures being random failures to de-pressurize the reactor, followed by random failures of the EDGs and their support systems causing a SBO.

SDS 27 (S-OP-HP-S2) is a seismic-induced LOOP and small LOCA with failure of 1E 250Vdc (high pressure injection). The CCDP is 7.8E-2, with the dominant failures being random failures to de-pressurize the reactor, followed by random failures of the EDGs and their support systems causing a SBO.

SDS 28 (S-OP-HP-CV) is a seismic-induced LOOP with failure of 1E 250Vdc (high pressure injection) and containment venting. The CCDP is 5.1E-2, with the dominant failures being random failures to de-pressurize the reactor, followed by random failures of the EDGs and their support systems causing a SBO.

SDS 30 (S-OP-HP-CT) is a seismic-induced LOOP with failure of 1E 250Vdc (high pressure injection) and the Condensate Storage Tank. The CCDP is 5.0E-2, with the dominant failures being random failures to de-pressurize the reactor, followed by random failures of the EDGs and their support systems causing a SBO.

SDS 32 (S-OP-HP-CT-CV) is a seismic-induced LOOP with failure of 1E 250Vdc (high pressure injection), the Condensate Storage Tank, and Containment Venting. The CCDP is 5.1E-2, with the dominant failures being random failures to de-pressurize the reactor, followed by random failures of the EDGs and their support systems causing a SBO.

3.1.5.3.2 Human Reliability Analysis

As noted in previous Paragraphs, special attention was given to human interactions and recovery actions. Because of the unusual nature of seismic events, some factors that were considered in the HCGS internal events analysis Human Reliability Analysis (HRA) change in this IPEEE analysis. For example, recovery actions that occur shortly after the seismic event involve additional stress factors above those present when the action is performed without a seismic event. Other actions might not be possible at all, depending on the damage caused by the seismic event. On the other hand, for recovery actions that occur a substantial time after the initiating event (and are possible after a seismic event), it would generally be appropriate to assume the IPEEE Human Error Probability (HEP) is similar to the internal events HEP.

It is assumed that a seismicly induced loss of offsite power would involve substantial switchyard damage and possibly other damages offsite. Therefore, recovery of offsite power within the first 24 hours is not credited.

No relay chatter interaction requiring human actions was needed (Paragraph 3.1.5.4.3) based on the relay chatter evaluation report (PSE&G, 1996b).

The HCGS IPEEE credited some of the internal events recovery actions. These are listed in Table 3-9, with the HEP calculated in the internal events analysis, and with the HEP used in the IPEEE. As seen in Table 3-9, for those recovery actions credited, the HEP calculated for the internal events was conservatively increased by a factor of 10.

Only two operator actions were explicitly included in the seismic event tree analysis: (1) Establishing alternate ventilation to the Class 1E Panel Room after a loss of panel room HVAC, and (2) Safe shutdown from outside the control room (remote shutdown). Similar to the HEPs used in the non-seismic failure calculations above, the HEP for establishing alternate ventilation was taken from the HCGS IPE (PSE&G, 1994a) and increased by one order of magnitude. The remote shutdown recovery HEP was not developed in the HCGS internal events IPE/PRA, so the HEP was developed for the IPEEE.

The development of the HEP for remote shutdown is presented in Table 3-10. Note that equipment failures were not considered for remote shutdown because the total failure probability of RSDOWN is assumed to be dominated by the HEP of 6.3E-2. Other seismic failures are addressed explicitly in the SET, and non-seismic failures are expected to be two or more orders of magnitude less likely that 6.3E-2.

3.1.5.4 Relay Chatter Evaluation

3.1.5.4.1 <u>Relay Identification and Screening</u>

The Hope Creek IPEEE seismic relay evaluation (PSE&G, 1996b) was conducted to determine if any relays, which may be susceptible to relay contact chatter during a seismic event, are used in electrical or instrumentation circuits that are vital to the safe shutdown of the plant. The methods used to identify potential low ruggedness relays (LRR), and to evaluate the impact of LRR chatter during a seismic event are described in Paragraph 3.0.1.3.3 and reference PSE&G, 1996b.

Potential LRRs were identified using searches through the computerized plant data bases, searches of Hope Creek responses to NRC information notices on relays, discussions with Hope Creek staff, and review of electrical schematics. Of the thousands of relays considered, less than 100 were determined to be LRRs or relays of unknown manufacturer type. A detailed evaluation of these relays was performed by a review of schematics and drawings showing the relays and their contacts. Each portion of the circuits was evaluated to determine the impact of relay contact chatter of the potential low ruggedness relay. To determine the potential impact of chatter, it was assumed that the contacts essentially changed state, energizing or de-energizing other relays as applicable.

During the detailed review, most of the relays were determined not to be low ruggedness relays and they were not further evaluated. For many relays, the contacts only provide continued indication for the circuit or an indication that DC power is available. These cases are not considered important to seismic safety (NISS) and Chatter Acceptable (CA) since indication will be restored following the seismic event. This is in accordance with the SQUG guidance on relays used for indications and alarms only, as discussed in EPRI NP-7148-SL (EPRI, 1990a - Paragraph 3.5.3 and Appendix H, p.3). Malfunction of these relays would not prevent the system from accomplishing its safe shutdown function.

3.1.5.4.2 Relay Seismic Capacity Evaluation

Four of the GE PVD21 low ruggedness relays associated with the 4kV vital busses were important to seismic safety for the Hope Creek Generating Station. The remaining GE PVD21 relays are associated with group busses and the switchyard. The HCLPF for these relays was calculated to be greater than 0.5g (PSE&G, 1996b - Reference 2). Since Hope Creek has a review level earthquake of 0.3g from NUREG-1407 (NRC, 1991b), the LRR has a seismic capacity greater than the review level earthquake (HCLPF > 0.3g).

The 12 Westinghouse SSC-T (three in each panel) instantaneous over-current relays associated with the diesel generators could potentially prevent a running diesel generator from tripping on an over-current as designed. This would require that the diesel was running and that the over-current existed. Even if the diesel is running and an over-current occurs simultaneously with contact chatter of the 2-3 contacts of a (1)50 instantaneous over-current relay, then this could prevent the (1)51 over-current relay from tripping and providing its input to the two of three phase relay logic to trip the diesel. This failure effectively makes the over-current logic two of two for the remaining phases.

3.1.5.4.3 Results of Relay Chatter Evaluation

In summary, a through review of documentation and data bases was performed to identify and evaluate any potential impacts from relay contact chatter of LRRs (PSE&G, 1996b).

Although there are several types of LRRs at Hope Creek, the contact chatter and seismic capacity evaluations demonstrated that none of these relays would impact the safe shutdown of the plant or containment performance after an earthquake based upon the following criteria:

- The LRR is not associated with seismic safe shutdown or containment performance equipment.
- Chatter of the LRR contacts is acceptable (does not impact safe shutdown of the plant or containment performance)
- The LRR has seismic capacity greater than the review level earthquake (HCLPF > 0.3g)

In conjunction with the equipment seismic capacity walkdowns, the relays and mountings were examined and verified to be well anchored.

It is therefore concluded that relay chatter is not significant to safe shutdown or containment performance after a seismic event at the Hope Creek Generating Station.

3.1.5.5 Seismic Core Damage Frequency and Seismic Sequences

3.1.5.5.1 Hope Creek Core Damage Frequencies

To obtain the overall results, the frequency of each seismic damage state (SDS) is multiplied by the CCDP for that SDS. Since each SDS is independent of the others, the total core damage frequency due to seismic events is simply the summation of the individual SDS sequence frequencies.

Table 3-8 presents the HCGS seismic core damage frequencies for the dominant seismic sequences of the study. For the baseline analysis, which employs the conservative LLNL seismic hazard curve, only one sequence has a core damage frequency (CDF) greater than 1E-6/year, and only four others have a CDF greater than 1E-7/year. These are:

Seismic Damage State	CDF (per year)	Description
SDS 36 (S-IC1)	2.5E-6	A seismic-induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ481. Core damage is assumed.
SDS 37 (S-DC)	4.4E-7	A seismic-induced failure of 1E power to all four 125Vdc distribution panels (1A/B/C/D-D-417). Similar to SDS 35, core damage is assumed.
SDS 26 (S-OP-HP)	1.9E-7	A seismic-induced loss of offsite power and failure of high pressure injection, with subsequent random failures which result in core damage. The random failures causing core damage are dominated by reactor de-pressurization failures which result in inadequate ECCS injection, or Emergency Diesel

		Generator (EDG) failures which result in a Station Blackout.
SDS 35 (S-IC2)	1.6E-7	A seismic-induced failure of all four divisions of 1E 120Vac instrumentation distribution panels. 1A/B/C/DJ482 panels. Credit is taken for manual system control to prevent core damage, but failure of both results in core damage and primary containment isolation failure.
SDS 18 (S-OP)	1.3E-7	A seismic-induced loss of offsite power, with subsequent random failures which result in core damage. The random failures are dominated by Emergency Diesel Generator failures, resulting in a Station Blackout.

The total seismic CDF for all HCGS Seismic Damage States (SDSs) is 3.6E-6/year, using the LLNL Hazard Curve. This is a factor of 3.5 less than the internal events CDF of 1.3E-5/year calculated in the HCGS PSA (PSE&G, 1994b), and a factor of 13 less than the CDF of 4.58E-5/year calculated for the HCGS IPE (PSE&G, 1994a). The above five SDSs represent 95% of the total CDF for seismic events, with SDS 36 being the largest single contributor at 69% of the total seismic CDF. While this one sequence has a large CDF relative to the other HCGS SDSs, its calculation was conservative in that no credit was taken for shutdown without 1E instrumentation. Additionally, since the magnitude of this SDS is small relative to the internal events CDF, none of the seismic sequences investigated would represent new or unique significant plant vulnerabilities.

3.1.5.5.2 Containment Performance Assessment for Seismic Sequences

Each of the dominant sequences was evaluated to determine the containment performance, particularly with respect to early containment failure and early or large releases. The equipment list contained containment systems such as the isolation system and containment heat removal systems. No low capacity components were identified in the isolation system, so early isolation failures are not anticipated.

The dominant seismic core damage sequences are assumed to end in a plant damage state similar to a Station Blackout (SBO). In the HCGS IPE (PSE&G, 1994a), the containment assessment found that approximately 40% of SBO core damage sequences result in large early or medium early radionuclide releases. Since the core damage frequency due to seismic events is several times lower than the core damage frequency due to internal events, as calculated in the HCGS IPE/PSA assessments, and since the containment response is identical, the magnitude of the large and medium early seismic containment failure frequencies would be small relative to the internal events frequencies.

In summary, there are no containment performance vulnerabilities that are unique or of a comparable magnitude to the internal events containment performance vulnerabilities.

3.1.5.6 Sensitivity Studies and Uncertainty Discussion

Several sensitivity studies were performed to examine different input information and assumptions. These studies are described subsequently.

3.1.5.6.1 Seismic Hazard Curve

The importance of the selection of the seismic hazard curve on the assessment of the HCGS seismic core damage sequence frequencies was examined through a separate sensitivity analysis. The LLNL seismic hazard curve (NRC, 1994b) was employed for the base case analysis of the HCGS IPEEE, and was selected to be a conservative representation of the seismic risk. To examine the sensitivity of the results to the seismic hazard curve used, the SEISMIC code was employed to evaluate the use of both the LLNL (base case - conservative) and the EPRI (site representative) seismic hazard curves.

The EPRI seismic hazard curve (EPRI, 1989a) is believed to more accurately represent actual geotechnical conditions at the HCGS plant. Table 3-8 presents the calculated HCGS seismic core damage frequency (CDF) for the important seismic damage states of this study. From the information of Table 3-8, it can be seen that, uniformly for all of the sequences evaluated, the LLNL hazard curve produces higher seismic CDF than those of the EPRI hazard curve. With both curves, the largest SDS contributor to CDF is SDS 36, a seismic-induced failure of the 1E instrumentation distribution panels. For the LLNL curve, SDS 36 contributes 2.5E-6/year to the CDF, and for the EPRI curve it contributes 6.7E-7/year. Overall, the CDF using the LLNL curve is 3.6E-6/year; with the EPRI curve, the CDF is 1.0E-6/year.

Therefore, this sensitivity analysis demonstrates that the choice of the LLNL hazard curve for the baseline assessment is conservative in relation to the EPRI hazard curve. The overall difference in CDF of approximately a factor of four constrains the level of uncertainty with respect to input hazard curves.

Input for the SEISMIC code (NUS, 1993a) requires midpoint values for the acceleration intervals, and slopes (termed seismogenic frequencies) for the frequency of the interval. The Hope Creek quantification/system analysis report (PSE&G, 1996a) contains the details of the calculations necessary to determine these SEISMIC code input parameters for the LLNL hazard curve. Comparing the Livermore and EPRI approaches indicates that, at lower accelerations, below about 0.15g, the LLNL and EPRI curves are similar in their prediction of seismic hazard. However, at higher accelerations on the order of 0.4g and above, the LLNL curve predicts considerably higher cumulative hazard values, as much as factors of three to seven times higher. While the industry judges that the EPRI hazard curve is more realistic, the baseline analysis for NRC submission has employed the more conservative Livermore curves in the quantification process.

3.1.5.6.2 Seismic Related Human Interactions

The HCGS IPEEE credited some of the internal events recovery actions. These are listed in Table 3-9, with the HEP calculated in the internal events analysis, and with the HEP used in the IPEEE. As seen in Table 3-9, for those recovery actions credited, the HEP calculated for the internal events was conservatively increased by a factor of ten.

Only two operator actions were included in the seismic event tree analysis: (1) Establishing alternate ventilation to the Class 1E Panel Room after a loss of panel room HVAC, and (2) Safe shutdown from outside the control room (remote shutdown). The remote shutdown recovery HEP was not developed in the HCGS internal events IPE/PRA. Its derivation is presented in Table 3-10.

As a sensitivity analysis, the IPEEE recovery actions shown in Table 3-9 were assigned the same HEP as was credited in the internal events analysis

(PSE&G, 1994a). For example, RHR initiation in the baseline IPEEE had an HEP of 5.0E-4. For this sensitivity analysis, the HEP was changed to 5.0E-5, the HEP credited in the internal events models. The results are presented in Table 3-9.

Table 3-11 shows that the seismic CDF is relatively insensitive to the additional credit for recovery actions. The CDF (using the LLNL hazard curve) only drops from 3.6E-6/year to 3.4E-6/year, a 5.6% decrease, when the internal events HEPs are used. This small change is due to the fact that the seismic CDF is dominated by SDSs 35, 36 and 37, the loss of 1E instrumentation and the loss of 125Vdc power. Since these three SDSs do not credit any of the internal events operator recoveries, taking full credit for the internal events HEPs has little effect on the overall seismic CDF.

3.1.5.6.3 Safe Shutdown Without 1E Instrumentation

As presented in Figure 3-12, core damage was assumed if 120Vac 1E instrumentation distribution panels 1A(B,C,D)J481 were all failed in a seismic event. This assumption was conservative because, while it is not proceduralized, manual control of equipment could be accomplished even without instrumentation.

This sensitivity analysis examines the effect of crediting the ability to safely shut down the HCGS without 1E instrumentation. Two cases were considered. In the first, shutdown without 1E instrumentation is given a 50% probability of success, and in the second it is given a 90% probability of success.

The only Seismic Damage State affected by this sensitivity is S36 (S-IC1). The results are presented in Table 3-12.

Giving a 50% success probability to shutdown without 1E instrumentation would reduce the baseline seismic CDF by 33%. Giving a 90% success probability would reduce the baseline seismic CDF by 61%. Further credit would have little effect on the seismic CDF.

Therefore, this sensitivity analysis has demonstrated that even marginal credit for shutdown capability without 1E instrumentation would have a relatively large effect on the overall seismic CDF. However, the magnitude of the baseline seismic CDF is relatively small when compared to the CDF from all internal events (4.6E-5/year in the HCGS IPE [PSE&G, 1994a], and subsequently calculated to be 1.3E-5/year in the HCGS PSA [PSE&G, 1994b]).

so the overall importance of the conservative 0% success assumption is not significant.

3.1.5.6.4 Extrapolation of the Seismic Hazard Curve

The published LLNL seismic hazard curve is limited to an acceleration of 1.0g, and the EPRI seismic hazard curve is limited to 0.8g. This truncation in the reported seismic hazard curves models the body of expert opinion which tends to limit the upper bound accelerations resulting from the largest site earthquakes. Since the EPRI and LLNL seismic hazard curves are provided only up to 0.8g and 1.0g respectively, the current base line analysis (employing the LLNL seismic hazard curve) was convoluted in the SEISMIC code to 1.0g. To examine the importance of higher extrapolations of the LLNL seismic hazard curve, a sensitivity analysis was performed employing the LLNL curve extrapolated to an acceleration of 1.5g.

Figure 3-13 presents a manual extrapolation of the LLNL seismic hazard curve to 1.5g. Note that the extrapolation represents an assumed continuation lognormal curve, and has no geotechnical basis. Table 3-14 compares the seismic core damage frequency when using the baseline LLNL curve with the extrapolated LLNL curve.

As seen in Table 3-14, the seismic CDF increases from 3.6E-5/year to 5.8E-6/year (approximately 61%) when extrapolated LLNL curve is used. The majority of this increase is from sequences S-12, S-IC1 and S-DC, which collectively account for 96% of the increase.

Overall, sequences S-IC2, S-IC1 and S-DC account for 90% of the seismic CDF with the extrapolated LLNL curve, compared to 83% with the baseline LLNL curve. No unique or new plant vulnerabilities were identified as a result of this sensitivity analysis. It is therefore concluded that the baseline analysis with the reported LLNL seismic hazard curve is adequate for the identification of seismic risk and of any vulnerabilities for Hope Creek.

3.1.5.6.5 Uncertainty Evaluation

Statistical and/or modeling uncertainty in the seismic CDF results can come from the hazard curve uncertainty, the fragilities uncertainties, and non-seismic uncertainties in the CCDP calculations. The sensitivity studies above examined some of the modeling uncertainties with respect to the LLNL or EPRI hazard curves, and modeling uncertainties with respect to human responses to an improbable seismic event. The hazard curve uncertainty is not examined in this report, except in so much as the differences between the LLNL and EPRI mean hazard curves were considered in Paragraph 3.1. A complete hazard uncertainty analysis is not required in NUREG-1407 (NRC, 1991b).

The SEISMIC code quantification included the fragilities uncertainties, expressed by the random and modeling uncertainty parameters given in Table 3-6. Statistical uncertainties in the CCDP calculations were not modeled for this analysis, but based on the internal events IPE, would be about a factor of three to five for the 95 percent confidence level.

Based on the above discussion, it is judged that a more detailed quantitative uncertainty analysis would not change or alter the results, identification of dominant sequences, contributors, or vulnerabilities.

3.1.6 ANALYSIS OF CONTAINMENT PERFORMANCE

NUREG-1407 (NRC, 1991b - Paragraph 3.2.6), provides guidance on the content of the seismic containment performance analysis. The purpose is to identify vulnerabilities that involve early failure of containment functions, including containment integrity, containment isolation, prevention of bypass functions, and some specific systems depending on containment design. The HCGS IPE (PSE&G, 1994a) was used to determine the scope of systems for the examination.

Table 3-13 lists the components examined in the IPEEE that pertain to the containment and the containment systems.

3.1.6.1 Structures and Major Components

The major structures and systems whose failure could result in early failure of containment were evaluated through walkdowns and seismic capacity calculations. These included the Reactor Building, the Auxiliary Building, the Station Service Water System Intake Structure, interior structures such as the torus and the drywell, reactor coolant system support and piping, main steam lines, and nearby structures. No issues or potential for failure of these items was noted in the walkdowns. Particular attention was paid to the adequacy of seismic gaps between major structures. The fragility calculations demonstrated that all of these structures and items had high seismic capacity (with median PGA capacities greater than 1.5g), and could be screened from the analysis.

3.1.6.2 Containment Isolation

Mechanical and electrical penetrations were included in the walkdown to ensure that there would not be failures of the mechanical penetrations or piping, electrical penetration assemblies, isolation valves and associated cables, piping supports, anchorages, or spatial interactions or differential motion which could cause failure of containment isolation or integrity. Hope Creek does not have any primary containment penetrations which require cooling, and no isolation valves require air to close. Therefore, on the basis of the walkdowns, capacity judgments, and on the design of the Hope Creek containment isolation and penetrations, there are no vulnerabilities in the mechanical and electrical penetration systems, or the containment isolation valves and piping.

3.1.6.3 Containment Bypass

The potential for seismic-induced Interfacing Systems Loss of Coolant Accidents (ISLOCAs) involves the failure of the Reactor Coolant System pressure boundary leading to a LOCA outside the containment boundary. The internal events IPE identified all of the potential ISLOCA paths, and was used as the initial basis for this seismic analysis. Valves in each of the ISLOCA paths were reviewed for inclusion on the seismic equipment list, and then included in the seismic capacity walkdown. Paths with check valves and normally closed manual valves for isolation have high capacity, and these paths were not evaluated further. For the remaining paths, the MOVs were included in the seismic equipment list and walkdown. These valves were also determined to have high seismic capacities, so they were screened from further analysis. The relays associated with these valves, including isolation actuation systems, were included in the relay chatter evaluation. Based on the ISLOCA evaluation, there are no seismic vulnerabilities associated with these paths, or with the valves and associated relays. No additional containment performance modeling is necessary.

3.1.6.4 Containment Hatches

Hope Creek does not have inflatable seals on the hatches, so there is no concern of the loss of air to the hatches. This, along with the review of the hatches during the walkdown, lead to the conclusion that there are no vulnerabilities associated with the containment hatches.

3.1.6.5 Containment Isolation Actuation

The sensors, transmitters, logic and relay cabinets, and power supplies for the Primary Containment Isolation System (PCIS) were included in the walkdown. All components had high capacities and were screened from further evaluation, except for the logic cabinet 120Vac 1E power supplies from the 1A/B/C/DJ482 distribution panels. These panels distribute power to the logic cabinets 1A/B/C/DC652, respectively. The 1A/B/C/DC652 logic cabinets provide automatic LOCA and high radiation isolation signals to non-NSSS Primary Containment Isolation Valves (PCIVs). Manual actuation of the PCIS is still possible from the control room, even if the automatic signals fail. In the seismic event tree, event IC2 represents failure of the 1A/B/C/DJ482 distribution panel, and failure to perform the necessary actions manually. This Seismic Damage State (SDS 35, or S-IC2) results directly in core damage. Because the event IC2 includes the failure to perform the manual actions necessary to avoid core damage, it is assumed that this would also include the failure to manually close the PCIVs.

Therefore, SDS 35 results directly in core damage and early containment failure, with a frequency of 1.6E-7/year when using the LLNL hazard curve, and 4.6E-8/year with the EPRI curve. This early release frequency is relatively small when compared to the total seismic core damage frequency (4% of the LLNL CDF).

The early release frequency is also small when compared to the total internal events early release frequency. From the HCGS IPE (PSE&G, 1994a - Table 4.7-21), the frequency of the high, early release is 9.4E-6/year, and the total frequency of all early releases is 2.8E-5/year. Therefore, the early release frequency of SDS 35 is only 2% of the large, early release frequency in the HCGS IPE, and it is only 0.6% of the total IPE early release frequency.

3.1.6.6 Containment Pressure Suppression and Heat Removal

The seismic PSA included the containment pressure suppression and heat removal functions in the RHR system. All of the RHR components modeled in the PSA were determined to have high seismic capacity, and were screened from further analysis.

3.1.6.7 Containment Failure Modes

The dominant seismic core damage sequences involve loss of all 1E instrumentation or the loss of all 1E 125Vdc power. If either of these systems were to fail, it is likely that operating equipment would continue to operate (e.g., a pump running off 4160Vac power would continue to run), and that stand-by equipment could be manually started if needed. However, in the conservative baseline seismic IPEEE assessment, it is assumed that the loss of all instrumentation or control power results in the loss of the equipment fed by that power. Therefore, the dominant seismic core damage sequences would each be considered station blackout (SBO) sequences.

The containment response, given core damage from a SBO, was analyzed in the HCGS IPE (PSE&G, 1994a), and the containment response given a SBO after a seismic event would be identical (approximately 40% of the IPE SBO core damage sequences resulted in a high early or medium early containment failure). Since the seismic CDF is small relative to the internal events CDF, and since the containment response is identical, the magnitude of the large and medium early seismic containment failure frequencies would be small relative to the internal events frequencies.

3.1.6.8 Containment Performance Results

In summary, containment performance systems and equipment were explicitly included in the walkdowns and seismic PSA. No significant vulnerabilities which could cause early failures of containment, or containment bypass, were identified.

3.1.7 SEISMIC INDUCED FIRE/FLOOD INTERACTIONS

The potential for seismic induced fire interactions was evaluated during the walkdowns (EQE, 1996a) and is also discussed in Section 4 of this report.

3.2 USI A-45, GI-131 AND OTHER SEISMIC SAFETY ISSUES

The following paragraphs describe related seismic safety issues and how these were considered in the seismic IPEEE.

3.2.1 SHUTDOWN DECAY HEAT REMOVAL (USI A-45)

The primary decay heat removal system at the HCGS is the RHR system. It is supported by the HVAC system for room cooling and the SACS and service water systems for heat exchanger cooling. The secondary decay heat removal system is containment venting featuring a hard torus pipe. This is a manual system supported by either 120Vac or dc power for operation of pneumatic valves. Nitrogen bottles provide motive force to backup the instrument air system. The vent path can be opened either from the control room or from the manual station on 102' elevation.

The seismic walkdown and screening assessment considered all components of the RHR and hard pipe vent systems. Regarding the RHR system: pumps, pipes, cable trays, and heat exchangers needed to provide decay heat removal were screened out. This includes RHR, SACS and SSW systems. All remaining components and structures (e.g., valves, electrical panels, control panels, relay boards, and expansion tanks) were shown to have median capacities in excess of 1.5g PGA.

Regarding the hard pipe vent system: the nitrogen accumulators, valves, pipes, and cable trays were screened out. All remaining components and structures associated with this system, except 1E 120Vac panels, were found to have median capacities greater than 1.5g PGA. The 1E 120Vac panels were calculated to have a median capacity slightly larger than 1.0g PGA. However, this system can operate from the dc power panels as well. These panels have median capacities in excess of 1.5g PGA.

Because of the high capacity of components and structures, as well as the ability to remove decay heat by either the RHR or hard pipe vent systems, the decay heat removal function at the HCGS is seismically robust.

3.2.2 CHARLESTON EARTHQUAKE ISSUE

The NRC states in response to industry question 7.13 on page D-13 of NUREG-1407 (NRC, 1991b) as " The issue of the 1886 Charleston Earthquake has been resolved. The issue of eight outlier plants identified through the Eastern U.S. Seismicity Program has been subsumed in the IPEEE and no specific reporting is required to close this issue." Note that Hope Creek was not one of these outlier plants. Therefore, this issue is considered closed.

3.2.3 USI A-17 SYSTEMS INTERACTION

Although USI A-17 is not a Hope Creek issue, the seismic walkdowns explicitly considered the USI A-17 concerns. The seismic spatial interactions observed during the walkdowns are incorporated in the screening and fragility evaluation of components or in added sequences such as the failure of a blockwall causing the failure of nearby components. The seismic, fire, and flooding examinations for the IPEEE incorporate the walkdown findings and specific insights pertaining to those hazards, for the USI A-17 related concerns. The issue has been addressed satisfactorily.

3.2.4 USI A-40 SEISMIC DESIGN CRITERIA

This issue does not apply since the Hope Creek Generating Station is a Standard Review Plan (SRP) plant.

3.2.5 USI A-46 VERIFICATION OF SEISMIC ADEQUACY OF ELECTRICAL AND MECHANICAL EQUIPMENT (GL 87-02)

This issue does not apply to the Hope Creek Generating Station.

3.2.6 GI-57 EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY RELATED EQUIPMENT

This topic is covered in Section 4 of this report.

3.2.7 GI-131 POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE IN-CORE FLUX MAPPING SYSTEM USED IN WESTINGHOUSE PLANTS

This issue does not apply to the Hope Creek Generating Station.

3.3 CONCLUSIONS

A seismic PSA approach was taken to evaluate the HCGS seismic risk. The predicted HCGS seismic core damage frequency is 3.6E-6/year using the conservative LLNL seismic hazard curve, and 1.0E-6/year using the EPRI seismic hazard curve. This frequency is an order of magnitude less than the internal events IPE and updated PSA frequencies.

Five SDSs represent 95% of the total seismic core damage frequency, with SDS 36 being the largest single contributor at 69% of the total seismic CDF. This SDS represents a seismic-induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ481. Core damage is assumed in this case, but this is a conservative assumption since a safe shutdown could still be possible without 1E instrumentation.

Containment performance systems and equipment were explicitly included in the walkdowns and seismic PSA. No vulnerabilities which could cause early failures of containment, or containment bypass were identified.

Several sensitivity studies were performed to examine different input information and assumptions including: the importance of the selection of the seismic hazard curve, the sensitivity of the seismic results to the assumptions regarding the Human Reliability Analysis, and the importance of the assumption that the loss of 1E instrumentation goes directly to core damage. No unique or new plant vulnerabilities were identified as a result of these sensitivity analyses.

No relay chatter interactions requiring human actions were needed based on the low ruggedness relay evaluation. It is concluded that relay chatter is not significant to safe shutdown after a seismic event at the HCGS.

The potential for seismic-induced fire interactions was evaluated during the walkdowns. No potential seismic-fire interactions were identified in or outside the containment. No significant potential seismic-induced flooding or spray interactions were identified.

The IPEEE concludes that USI issues with respect to the Hope Creek Generating Station are either satisfactorily resolved or are actively being investigated.

- USI A-45 Shutdown Decay Heat Removal - no new vulnerabilities identified in the IPEEE seismic survey.

- Charleston Earthquake Issue - issue closed for the Hope Creek Generating Station.

- USI A-17 Systems Interaction - issue satisfied in parallel with IPEEE seismic walkdowns and evaluations.

- USI A-40 Seismic Design Criteria - issue does not apply to the Hope Creek Generating Station.

- USI A-46 Seismic Adequacy of Electrical/Mechanical Equipment - issue does not apply to the Hope Creek Generating Station.

- GI-57 Seismic induced fire/flood interaction issues, including spurious actuation of the fire protection systems, were evaluated and no unique vulnerabilities were identified (Paragraph 4.10).

- GI-131 Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System - This issue does not apply to the Hope Creek Generating Station.

The principal conclusion is that the seismic evaluations did not identify any unique or new vulnerabilities for the HCGS.

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Table 3-1
Annual Probability of Exceedance of Peak Ground Acceleration at The Hope Creek Site from LLNL Study (NRC, 1994b)

Acceleration (cm/sec/sec)	Mean	Percentiles 15th	50th	85th
50.	.9721E-03	.9990E-04	.3990E-03	.1680E-02
75.	.5512E-03	.4420E-04	.2060E-03	.9370E-03
150.	.1836E-03	.8760E-05	.5530E-04	.3030E-03
250.	.7227E-04	.1900E-05	.1780E-04	.1110E-03
300.	.5028E-04	.9970E-06	.1090E-04	.7550E-04
400.	.2735E-04	.3100E-06	.4790E-05	.3810E-04
500.	.1651E-04	.1070E-06	.2350E-05	.2190E-04
650.	.8770E-05	.2890E-07	.9040E-06	.1050E-04
800.	.5156E-05	.8920E-08	.3870E-06	.5570E-05
1000.	.2826E-05	.2240E-08	.1450E-06	.2750E-05

Table 3-2
Summary of Liquefaction Potential and Seismically Induced Settlement Evaluation at the Hope Creek Generating Station SSE Level

Building	Case	Settlement inches			Conditional Probability of Liquefaction
		Center	Corner	Differ.	
Turbine	16th %	0.00	0.00	0.00	4.38E-07
	50th %	0.01	0.00	0.01	1.29E-04
	84th %	0.08	0.03	0.05	1.70E-03
Service Water Intake	16th %	0.00	0.00	0.00	7.67E-07
	50th %	0.01	0.00	0.01	1.70E-04
	84th %	0.09	0.04	0.05	1.92E-03
Auxiliary	16th %	0.00	0.00	0.00	2.56E-06
	50th %	0.02	0.00	0.02	4.61E-04
	84th %	0.16	0.05	0.11	3.73E-03
Reactor	16th %	0.00	0.00	0.00	5.15E-06
	50th %	0.02	0.00	0.02	5.97E-04
	84th %	0.19	0.06	0.13	4.49E-03

Table 3-3
Summary of Liquefaction Potential and Seismically Induced Settlement Evaluation at the Hope Creek Generating Station 3 * SSE Level

Building	Case	Settlement inches			Conditional Probability of Liquefaction
		Center	Corner	Differ.	
Turbine	16th %	0.00	0.00	0.00	1.42E-04
	50th %	0.14	0.02	0.13	4.76E-03
	84th %	0.60	0.15	0.45	1.93E-02
Service Water Intake	16th %	0.00	0.00	0.00	1.65E-04
	50th %	0.13	0.03	0.10	4.38E-03
	84th %	0.51	0.19	0.32	1.68E-02
Auxiliary	16th %	0.02	0.00	0.02	9.60E-04
	50th %	0.34	0.05	0.29	1.43E-02
	84th %	1.06	0.31	0.76	4.20E-02
Reactor	16th %	0.03	0.00	0.03	1.32E-03
	50th %	0.41	0.07	0.34	1.78E-02
	84th %	1.21	0.38	0.83	4.95E-02

Table 3-4
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
ACP	10Y406	120 Vac CONTROL POWER BUS 10Y406	Control Diesel	5101	54'	No	5	C-293
ACP	1AJ481	CLASS 1E 120V PANEL 1AJ481/1CJ481	Control Diesel	5501	137'	No	5	
ACP	1BJ481	CLASS 1E 120V PANEL 1BJ481/1DJ481	Control Diesel	5448	124'	No	3, 5	
ACP	1BJ484	120 Vac CONTROL ROOM PWR-UPS- BUS 1BJ484	Control Diesel	5624	163'	Yes		C-251,252
ACP	1AJ482	CLASS 1E 120V PANEL 1AJ482/1CJ482	Control Diesel	5616 / 5613	163'	No	5	
ACP	1CJ483	FUSE PANEL YF407-120 Vac BUS 1CJ483	Control Diesel	5102	54'	Yes		C-258
ACP	1BJ482	CLASS 1E 120V PANEL 1BJ482/1DJ482	Control Diesel	5607	163'	No	41	C-253,254,255
ACP	10A401	CLASS 1E 4.16kV BUS 10A401- DIV A/10A402-DIV B/10A403-DIV C/10A404 DIV D	Control Diesel	5411	130'	No	52	C-148,149
ACP	10B411	CLASS 1E 480Vac MCC 10B411 DIV A/10B421 DIV B/ 10B431 DIV C/10B441 DIV D	Control Diesel	5411	130'	No	5	C-145
ACP	10B410	CLASS 1E 480Vac UNIT SUBST 10B410-A/10B420-B/10B430-C/10B440-D	Control Diesel	5411	130'	No	4	
ACP	10B451	CLASS 1E 480Vac MCC 10B451 DIV A/10B461 DIV B/10B471 DIV C/10B481 DIV D	Control Diesel	5411	130'	No	5	
ACP	10B450	CLASS 1E 480Vac UNIT SUBST 10B450A/10B460-B/10B470-C/10B480-D	Control Diesel	5411	130'	No	4, 5	
ACP	10B553	480ac BUS 10B553/563/573/583	SWIS	SWIS	102'	No	42, 5	C-152
ACP	1AD481	1AD481 & 1AD482 AUCTIONEERNG CIRC INCL. INVERTER,STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	Control Diesel	5501	137'	No	1, 2	C-234, 235

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
ACP	1BD481	1BD481 & 1BD482 AUCTNEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	Control Diesel	5448	124'	No	2	
ACP	1CD481	1CD481 & 1CD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	Control Diesel	5501	137'	No	2	C-236, 237
ACP	1DD481	1DD481 & 1DD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	Control Diesel	5448	124'	No	2	
ACP	10B212	1E 480Vac MCC 10B212 DIV A/ 10B222 DIV B/ 10B232 DIV C/ 10B242 DIV D	Reactor	4201 / 4303	77' / 102'	No	6, 5	C-27, 73, 74
ACP	10B252	NON 1E 480Vac MCC 10B252A/10B262B/10B272/10B282 272 DIV C/ 10B282 DIV D	Reactor	4215	77'	No	7, 5	C-75
ACP	10B313	NON 1E 480Vac MCC 10B313 DIV A/ 10B323 DIV B	Reactor	3602	153'	No	42, 5	C-262,263,264,265,26
ACP	10B474	NON 1E DIV C 480VAC MCC 1OB474	Control Diesel	5619	160'	No	42, 5	C-260,261
ACP	1AC421	LOCL GEN CNTRL PNL 1AC421/ 1BC421/ 1CC421/ 1DC421	Control Diesel	4304 / 4307	102'	No	7, 5	C-93,94,95,96
ACP	1AC422	REM GEN CNTRL PNL 1AC422/ 1BC422/ 1CC422/ 1DC422	Control Diesel	5410	130'	No	5	C-139,140,197,198
ACP	1A-C423	DIESEL PANELS - 1A/B/C/D - C423	Control Diesel	5412	130'	No	8	C-148, 149
ACP	1A-C428	D/G LOAD SEQUENCE PANEL 1A/B/C/D - C428	Control Diesel	5410	130'	No	43, 5	C137,138,141,196,20

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
ACP	1AC652	PNL 1AC652/ 1BC652/ 1CC652/1DC652	Control Diesel	5302	102'	No	5	C-292
ACP	10B263	NON-1E 480V MCC 10B263	Reactor	4307	102'	No	2	C-5
ACP	1AX-400	TRANSFORMER 1AX/BX/CX/DX-400	Control Diesel	5411	130'	Yes		
ACP	1AX-401	TRANSFORMER 1AX/BX/CX/DX-401	Control Diesel	5411	130'	Yes		
ADS	AT210	ADS SRV F013A/B/C/D/E AIR ACCUMULATOR - A/B/C/D/ET210	Reactor	4202	77'	Yes		
ADS	3652A	ADS SRV F013A SOV 'A' - 3652A/ SOV'B' -3652B	Reactor	4202	77'	Yes		
ADS	3653A	ADS SRV F013B SOV 'A' - 3653A/ SOV'B' -3653B	Reactor	4202	77'	Yes		
ADS	3654A	ADS SRV F013C SOV 'A' -3654A /SOV'B' -3654B	Reactor	4202	77'	Yes		
ADS	3655A	ADS SRV F013D SOV 'A' - 3655A / SOV'B' -3655B	Reactor	4202	77'	Yes		
ADS	3665A	ADS SRV F013E SOV 'A' - 3665A/ SOV'B' -3665B	Reactor	4202	77'	Yes		
ADS	F013A	ADS SRV F013A /B/C/D/E	Reactor	4202	77'	Yes		
CAC	HV-4956	AOV-HV-4956/4958/4964/4979	Reactor	4321/ 4102	102'	No	44	
CAC	SV-4956	SOLENOID VALVE 4956/ 4958/ 4964/ 4978/ 4979	Reactor	4321/ 4102	102'	Yes		
CAC	4964	ACCUMULATOR 4964/49XX	Reactor	4220	102'	Yes	51	
CAC	HV-4950	HV-4950, 4951, 4952, 4962, 4963, 4980 CONTAINMENT ISOLATION VALVES	Reactor	4410/, 4411	132'	Yes		C-121, 122, 275

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
CAC	1GSHV-4978	N2 SUPPLY ISOLATION VALVE	Reactor	4317	102'	Yes		C-291
CAC	1GSHV-11541	TORUS VENT ISO MANUAL OPERATION FOR AOV-11541	Reactor	4102	77'	No	46	C-275, 289
CAC	1GSHV-4962/4964	OUTBOARD CONTAINMENT ISOLATION MANUAL OVERRIDE	Reactor	4317	102'	No	46	C-275
CAC	HV-4956	AOV-HV-4956/4958/4964/4979	Reactor	4321/ 4102	102'	No	44	
CAC	SV-4956	SOLENOID VALVE 4956/ 4958/ 4964/ 4978/ 4979	Reactor	4321/ 4102	102'	Yes		
CAC	4964	ACCUMULATOR 4964/49XX	Reactor	4220	102'	Yes	51	
CAC	HV-4950	HV-4950, 4951, 4952, 4962, 4963, 4980 CONTAINMENT ISOLATION VALVES	Reactor	4410/, 4411	132'	Yes		C-121, 122, 275
CAC	1GSHV-4978	N2 SUPPLY ISOLATION VALVE	Reactor	4317	102'	Yes		C-291
CAC	1GSHV-11541	TORUS VENT ISO MANUAL OPERATION FOR AOV-11541	Reactor	4102	77'	No	46	C-275, 289
CAC	1GSHV-4962/4964	OUTBOARD CONTAINMENT ISOLATION MANUAL OVERRIDE	Reactor	4317	102'	No	46	C-275
CAC	11558A/B/C	TORUS VENT ISO N2 ACCUM 1KBPCV11558A/B/C	Reactor	4317	102'	Yes		C-290
CHC	TV9634A	AOV TV9634A/B	Control Diesel	5605	171'	Yes		C-181, 182
CHC	TV9637A	AOV TV9637A /B	Control Diesel	5602/ 5630	155'	Yes		
CHC	AK-400	CNTRL AREA CHILLER AK-400/ BK-400	Control Diesel	5602	155'	No	9	C-210,224
CHC	AP-400	RECIRC PUMP AP-400 / BP400	Control Diesel	5630	155'	No	10	
CHS	TV9667A	AOV TV9667A /B	Control Diesel	5620	162'	Yes		C-208

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
CHS	AK-403	CNTRL AREA CHILLER AK-403/BK-403	Control Diesel	5704	178'	Yes		C-183
CHS	AP-414	RECIRC PUMP AP-414 / BP-414	Control Diesel	5704	178'	Yes		C-183,184
CHS	VH316A/B	REMOTE SHUTDOWN PANEL ROOM UNIT VH316A/VH316B	Control Diesel	5501	137'	No	5	C-294
CHS	VH314A/B	TECH SUPPORT CENTER A/C UNITS VH314A/314B	Aux/Rad	3613	152'	No	5	C-295
CHS	AT413	CHS EXPANSION TANKS A/BT413	Control Diesel	5704	173'	Yes		C-287
CNS	HV F011A	MOV HV-F011A /B	Reactor	4220	120'	Yes		C-195
CRH	F002B	CRH FLOW CONTROL VALVE AOV-F002B / F002A	Reactor	4317	102'	Yes		C-36
CRH	AF201	CRH PUMP SUCTION FILTER AF201/ BF201	Reactor	4202	77'	Yes		C-71
CRH	AF204	DRIVE WATER FILTER AF204 / BF204	Reactor	4319	102'	Yes		C-37
CRH	AP207	DRIVE WATER PUMP AP207 / BP207	Reactor	4202	77'	Yes		C-72
CRH	HV-4005	ISOLATION VLV MOV-HV-4005	Reactor	4104	54'	Yes		C-125
CRH	HV-F003	PCV CONTROL VLV MOV-HV-F003	Reactor	4317	102'	Yes		
CRH	CRDACCUM	SAMPLING OF CRD ACCUMULATORS	Reactor	4319	102'	Yes		C-33,34
CRH	XV-126	SAMPLING OF CRD SCRAM INLET AND OUTLET VALVES	Reactor	4317	102'	No	47	C-35,267

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
CSS	AP206	CSS PUMP A/B/C/D P206	Reactor	4116	54'	No	12	C-106,107,108, 110,11
CSS	F031A	MOV F031A /B	Reactor	4102	54'	Yes		C-45
CSS	F005A	MOV F005A /B	Reactor	4329/4321	102'	Yes		C-180, C-300
CSS	F004A	MOV F004A/B	Reactor	4329/4321	102'	Yes		C-299
CSS	F001A	MOV F001A/B/C/D	Reactor	4102	054'	Yes		
DCP	10D410	CLASS 1E 125Vdc BUS 10D410 DIV A/10D420 DIV B/ 10D430 DIV C/ 10D440 DIV D	Control Diesel	5411	130'	No	13	C-146,147
DCP	10D436	CLASS 1E 125Vdc BUS 10D436 DIV C/10D446 DIV D	Control Diesel	5607	160'	No	14	
DCP	1AD318	NON 1E 125Vdc DISTR PNL 1AD318/1BD318/1CD318/1DD318318	Aux/ Rad	3449	124'	No	34	C-307
DCP	1AD417	125Vdc 1E PWR TO LOADS IN PNL 1AD417/ 1BD417/ 1CD417/ 1DD417	Control Diesel	5411	130'	No	15, 5	C-143,144
DCP	10D251	250 Vdc MCC 10D251/ 10D261	Reactor	4112/4108	54'	No	16	C-57,67,68
DCP	10D450	CLASS 1E 250VDC BUS 10D450 DIV A/ 10D460 DIV B	Control Diesel	5128(near)	54'	No	7, 5	
DCP	10D421	250 VDC BAT 10D421/ 10D431-DIV A	Control Diesel	5104	54'	Yes		C-240,241
DCP	1AD411	125 VDC BAT 1AD411-DIV A/ 1BD411-DIV B/ 1CD411-DIV C/ 1DD411-DIV D	Control Diesel	5539	146'	No	18	C-129,131,132
DCP	1CD447	125 VDC BAT 1CD447-DIV C/ 1DD447-DIV D	Control Diesel	5614	163'	Yes		C-239
DCP	10D423	250 Vdc BAT CHGR 10D423 /10D433	Control Diesel	5128(near)	54'	No	20	C-242

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
DCP	1CD444	125 Vdc BAT CHGR 1CD444/1DD444	Control Diesel	5613/5607	163'	Yes	5	C-238, C-296
DCP	1AD413/ 1AD414	125 Vdc BAT CHGR 1A/B/C/DD413 / 1A/B/C/DD414	Control Diesel	5538	146'	No	19	C-126,127
DCP	10D422	FUSE SWTCH BOX 10D422-250Vdc A	Control Diesel	5128(near)	54'	Yes		
DCP	10D432	FUSE SWTCH BOX 10D432-250Vdc B	Control Diesel	CORDR 5128	54'	Yes		
DCP	1AD412	FUSE SWTCH BOX 1AD412-125VDC A/ 1BD412-B/ 1CD412-C/ 1DD412-D	Control Diesel	multiple	146'	Yes		C-128
DCP	1CD448	FUSE SWTCH BOX 1CD448-125VDC C/ 1DD448-D	Control Diesel	5613	160'	Yes		
DGS	1AG400	DIVISION A DIESEL 1AG400/ 1BG400/ 1CG400/ 1DG400	Control Diesel	5303	102'	Yes		C-89,90
DGS	1A-C420	EXCITER PANELS, 1A/B/C/D C420	Control Diesel	5303/5305	102'	No	7	C-97
DGS	AT403	FUEL STORAGE TANKS, A-HT403	Control Diesel	5110	54'	Yes		C-100
DGS	HL-7530A	SDG A/B/C/D DAYTNK LVL SWTCH - LSHL7530A/B/C/D	Control Diesel	5303	102'	Yes		C-81,82
DGS	1AP401	SDG A/B/C/D FOTP 1AP401/ 1BP401/ 1CP401/ 1DP401/1EP401/ 1FP401/ 1GP401	Control Diesel	5107	54'	Yes		C-101
DGS	1AP402	SDG A/B/C/D ELEC FUEL PMP 1AP402/ 1BP402/ 1CP402/ 1DP402	Control Diesel	5303	102'	Yes		C-91

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
DGS	PS7508A	FUEL OIL PRESS SWTCH PSL7508A /B/C/D	DG skid	5303	102'	Yes		
DGS	7534A	EDLOS SOV7534A /B/C/D	Control Diesel	5303	102'	Yes		C-88
DGS	7535A	SDG A SOV 7535A/B/C/D	Control Diesel	5304/5307	102'	Yes		
DGS	7536A	SDG A SOV 7536A /B/C/D	Control Diesel	5304/5307	102'	Yes		
DGS	TV6606A	THERMSTAT CNTRL VLV TV 6606A/B/C/D	Control Diesel	5303/5305	102'	No	45, 46	
DGS	TV6618A	THERMSTAT CNTRL VLV TV 6618A/B/C/D	Control Diesel	5303/5305	102'	No	45, 46	C-276
DGS	TV7722A	THERMSTAT CNTRL VLV TV 7722AB/C/D	Control Diesel	5303/5305	102'	No	45, 46	C-92
DGS	AT404	SDG A/B/C/D DAYTANK - AT404/ BT404/ CT404/ DT404	Control Diesel	5305	102'	Yes		C-79,80,83
DGS	AT406	EDLOS MAKEUP TANK -AT406/ BT406/ CT406/ DT406	Control Diesel	5305	102'	Yes		
DGS	AT407	SDG A/B/C/D JWCL EXPANSION TNK -AT407/ BT407/ CT407/ DT407	Control Diesel	5303	102'	Yes		C-86,87
DGS	AT408	SDG A/B/C/D START AIR RECEIVER - AT/BT/CT/DT/ET/FT/GT/HT 408	Control Diesel	5303	102'	No	21	C-84,85
DGS	1DG403	NEUTRAL GROUNDING AND TFM - 1A/B/C/DG403	Control Diesel	5304	102'	Yes		C-98,99

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
ESF	LI402A	LEVEL XMTER SA-LI-402AB/E/F	Reactor	4215	77'	Yes		C-269
ESF	LIN091A	LEVEL XMTER BB-LI-N091A/B/C/D/E/F/G/H LEVEL L2	Reactor	4215	77'	Yes		C-269
ESF	LIN095B	LEVEL XMTER BB-LI-N095B/D LEVEL L2	Reactor	4215	77'	Yes		C-269
ESF	PSN058A	PRESS SENSOR/XMTER E11-N058A/B/C/D	Reactor	4215	77'	Yes		
ESF	PSN090A	PRESS SENSOR/XMTER B21-N090A /B/E/F/J/K/N/P	Reactor	4215	77'	Yes		
ESF	PT403A	PRESS XMTER SA-PT-403A/B/E/F	Reactor	4215	77'	Yes		C-269
ESF	PTN094A	PRESS XMTER BB-PT-N094A/B/C/D/E/F/G/H DW PRESS.	Reactor	4605	162'	Yes		C-38,41
HPI	8278	MOV HV-8278	Reactor	4316	102'	Yes		
HPI	F001	MOV HV-F001	Reactor	4111	54'	Yes		C-62
HPI	F006	MOV HV-F006	Reactor	4329	102'	Yes		C-174
HPI	F042	MOV HV-F042	Reactor	4102	54'	Yes		
HPI	4880	MOV FV-4880	Reactor	4111	54'	Yes		
HPI	4879	MOV FV-4879	Reactor	4111	54'	Yes		
HPI	F002	N.O. MOV HV-F002	Reactor	4220	102'	Yes		
HPI	F003	N.O. MOV HV-F003	Reactor	4327	102'	Yes		C-298
HPI	F004	MOV HV-F004	Reactor	4111	54'	Yes		
HPI	F059	MOV HV-F059	Reactor	4111	54'	Yes		C-60
HPI	F007	N.O. MOV HV-F007	Reactor	4111	54'	Yes		C-176
HPI	F071	N.O. MOV HV-F071	Reactor	4102	77'	Yes		
HPI	OP204	HPCI TDP OP204 (incl. pmp OP217 & turbine OS211)	Reactor	4111	54'	Yes		C-297
HPI	LT4805	BJLT-4805-1/2, SUPPRESSION POOL LEVEL TRANSMITTER	Reactor	4115	54'	Yes		C-51,52
IAS	PCV7668	SELF-CONTAINED VALVE PCV7668	Reactor	4317	102'	Yes		C-268
IAS	10-T106	EMERGENCY IAS AIR RECEIVER TANK	Turbine	1401	120'	No	34	
IAS	10-F104	IAS AIR DRYER	Turbine	1401	120'	No	34	
IAS	1A-T132	IAS AIR RECEIVER TANK	Turbine	1401	120'	No	34	

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
IGS	5125	AIR OPERATED VALVE 5125	Reactor	4413	132'	Yes		C-118
IGS	AK202	COMPRESSOR AK202 / BK202	Reactor	4413	132'	Yes		C-117
IGS	5126A	MOV 5126 A/B	Reactor	4329/4321	102'	Yes		
IGS	5152A	MOV 5152A/B	Reactor	4220	110'	Yes		C-301
IGS	5160A	MOV 5160A/B	Reactor	4412	132'	Yes		C-119
IGS	BT-201	IGS RECEIVER BT-201	Reactor	4412	132'	Yes		C-42
IGS	AT-201	IGS RECEIVER AT-201	Reactor	4413	132'	Yes		C-42
IGS	AE-214	AFTERCOOLER MOISTURE SEPARATOR AE-214/BE-214	Reactor	4413	132'	Yes		
IGS	AS-208	IGS AIR DRYERS AS-208/BS-208	Reactor	4413	132'	Yes		
IGS	AF-216	IGS OUTLET FILTERS AF-216/BF-216	Reactor	4412/4413	132'	Yes		
IGS	AF-215	IGS INLET FILTERS AF-215/BF-215	Reactor	4412/4413	132'	Yes		
IGS	5124A	MOV 5124 A/B	Reactor	4220	120'	Yes		C-306
IGS	1A-C213	1A/B-C213, A AND B IGS CONTROL PANELS	Reactor	4413	132'	No	7	C-120
PCS	HV-F028A	HVF028A/B/C/D - MAIN STEAM LINE ISOLATION VALVE	Reactor	4316	102'	Yes		
PCS	HV-F022A	HVF022A/B/C/D - MAIN STEAM LINE ISOLATION VALVE	Reactor	4220	102'	Yes		
RAC	2601	AOV-2601	Reactor	4209-4210	77'	Yes		C-2
RAC	2617	AOV TV-2617	Reactor	4209	77'	Yes		C-78
RAC	AE-217	RACS HX AE-217/ BE-217	Reactor	4211	77'	No	30	C-77,270,271
RAC	AP-209	RACS PUMP AP-209 / BP-209	Reactor	4209	77'	No	24	C-1

TABLE 3-4 (Continued)
SCREENING OF HOPE CREEK SEISMIC IPEEE COMPONENTS

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
RCI	F010	MOV F010	Reactor	4110	54'	Yes		C-17
RCI	F013	MOV F013	Reactor	4316	102'	Yes		
RCI	F031	MOV F031	Reactor	4102	54'	Yes		C-45
RCI	F045	MOV F045	Reactor	4110	54'	Yes		C-19
RCI	4282	N.O. MOV HV-4282	Reactor	4110	54'	Yes		C-20
RCI	4283	MOV HV-4283	Reactor	4110	54'	Yes		C-21
RCI	F008	N.O. MOV HV-F008	Reactor	4319	102'	Yes		
RCI	F012	N.O. MOV HV-F012	Reactor	4110	54'	Yes		C-18
RCI	F059	N.O. MOV HV-F059	Reactor	4102	77'	Yes		
RCI	V012	TEST ISOLATION MOV 1BDV-012 (F002)	Reactor	4203	77'	Yes		C-180
RCI	4405	SOV 4405	Reactor	4102	54'	Yes		
RCI	F019	SOV F019	Reactor	4102	54'	Yes		
RCI	OF209	SUCTION STRAINER OF209	Reactor	Torus	54'	Yes		
RCI	OP203	TURBINE-DRIVEN PUMP - OP203 (includes Turbine OS212)	Reactor	4110	54'	Yes		C-16
RHS	F075	MOV F075	Reactor	4208	93'	Yes		
RHS	AE 205	RHS HEAT EXCHANGER A/B E205	RHR Heat Exchanger Room	4113/4109	54'	Yes		C-178,179
RHS	1A-P202	RHS PUMP A/B/C/D P202	Reactor	4115	54'	No	25	C-48,55
RHS	F015A	RHS MOV F015A /B	Reactor	4321/4329	102'	Yes		C-174
RHS	F016A	RHS-MOV-F016A/B	Reactor	4329	102'	Yes		C-175
RHS	F017A	RHS MOV F017A /B/C/D	Reactor	4321/4329	102'	Yes		
RHS	F021A	RHS-MOV-F021A/B	Reactor	4329	102'	Yes		

TABLE 3-4 (Continued)
SCREENING OF HOPE CREEK SEISMIC IPEEE COMPONENTS

SYS	EQUIP ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
RHS	F024A	RHS MOV F024A /B	Reactor	4102	54'	Yes		
RHS	F027A	RHS MOV F027A /B	Reactor	4102	77'	Yes		
RHS	F047A	RHS MOV F047A /B	Reactor	4214/4208	77'	Yes		C-177
RHS	F006A	RHS MOV F006A/B	Reactor	4109	54'	Yes		C-176
RHS	F007A	RHS MOV F007A /B/C/D	Reactor	4102	54	Yes		
RHS	F008	RHS MOV F008	Reactor	4329	102'	Yes		C-175
RHS	F009	RHS MOV F009	Reactor	4220	100'	Yes		
RHS	F048A	RHS MOV F048A/B	Reactor	4208	77'	Yes		
RHS	F004A	RHS MOV F004A /B /C /D	Reactor	4102	54'	Yes		
RHS	A-F211	RHS SUCTION STRAINER A/B/C/D-F211	Reactor	Torus	54'	Yes		
RHS	F010	RHS RETURN TEST VALVES. HV-F010A/B	Reactor	4114/4107	54'	Yes		C-49
RWC	F001	RWCU ISOLATION VALVE F001	Drywell	4220	145'	Yes		
SAC	2290A	AOV HV2290A/B/C/D/E/F/G/H (RHR)	Reactor	4113/4214/	54'	No	26	C-47
SAC	2292A	AOV HV2292A /B (HPCI)	Reactor	4111	54'	Yes		C-61
SAC	2293A	AOV HV2293A/B (RCIC)	Reactor	4110	54'	No	27	C-63
SAC	2325A	AOV HV2325A/B/C/D/E/F/G/H (CSS)	Reactor	4116/4118	54	No	28	C-46,44,109,113,116
SAC	2395A	AOV HV2395A/B/C/D (SAC)	Control Diesel	5210/5211	77'	Yes		C-243
SAC	2398A	AOV HV2398A /B/C/D/E/F/G/H	Control Diesel	5211	77'	Yes		C-103, 279, 280
SAC	2520A	AOV 2520A/B/C/D	Reactor	4113/4109	54'	Yes		C-50,54
SAC	A1E-201	HX A1/B1E-201	Reactor	4307	102'	Yes		C-3,4,29
SAC	A2E-201	HX A2/B2E-201	Reactor	4307	102'	Yes		C-3,4,29
SAC	AP-210	SACS PUMP AP-210/ BP-210/ CP-210/ DP-210	Reactor	4309	102'	Yes		C-6
SAC	2512A	MOV 2512A/B	Reactor	4208	77'	Yes		

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP. ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
SAC	2520A	SOV 2520A/B/C/D	Reactor	4107/ 4114	54'	Yes		
SAC	AT205	SAC EXPANSION TANKS A/BT205	Reactor	4710	201'	No	30	C-302
SAC	2496A-D	TACS RETURN ISOLATION VALVES HV-2496A-D	Reactor	4307	102'	Yes		C-9
SAC	2522A-D	TACS SUPPLY ISOLATION VALVES HV-2522A-D	Reactor	4309	102'	Yes		C-10
SAC	A-BE-202	FUEL POOL HEAT EXCHANGERS	Reactor	4627	162'	Yes		
SAC	1AC201	SAC CONTROL PANELS, 1A/B/C/D-C201	Reactor	4309	102'	No	34	C-7, 8, 272, 273, 274.
SWS	AP-502	SWS PUMP AP-502/ BP-502/ CP-502/ DP-502	SWIS	208	93'	Yes		C-155
SWS	AP-507	SWS-TWS PUMP AP-507 / BP-507/ CP-507/ DP-507	SWIS	SWIS	77'	Yes		C-156
SWS	2197A	STRNR MOV HV-2197A /B/C/D	SWIS	0204	102'	Yes		
SWS	2198A	SWS MOV HV-2198A /B/C/D	SWIS	0204	93', 102'	Yes		
SWS	2225A	SWS MOV HV-2225A/B/C/D	SWIS	0204	93', 102'	Yes		
SWS	2355-A	MOV 2355-A /B	Reactor	4307/4309	102'	Yes		C-30
SWS	2371-A	MOV 2371-A/B	Reactor	4307/4309	102'	Yes		C-31
SWS	HV-2207	MOV HV-2207 (RACS HX)	Reactor	4309	102'	Yes		
SWS	HV-2346	MOV HV-2346 (RACS HX)	Reactor	4211	77'	Yes		
SWS	HV-2203	SWS MOV HV-2203	Reactor	4309	102'	Yes		C-32
SWS	HV-2204	SWS MOV HV-2204	Reactor	4307	102'	Yes		
SWS	SV-2247A	SWS SV-2247A/B/C/D	SWIS	SWIS	102'	Yes		C-157
SWS	AF-509	FILTER AF/BF/CF/DF-509	SWIS	99BYD	93'	Yes		C-154
SWS	OT-543	TANK OT-543/545	SWIS	SWIS	122'	Yes		
SWS	EP-AS501	SWS-TWS EP- AS501/BS501/CS501/DS501	SWIS	SWIS	122'	Yes		

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	Building	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
SWS	F073	SW-RHR VALVES HV-F073/HV-F075	Reactor	4102/4209	77'	Yes		C-124
SWS	PMPCNTRL	SW PUMP CONTROL PANELS	SWIS	SWIS	110'	No	5	C-165
SWS	AC/EC 581	SWIS HVAC CONTROL PANEL AC/EC 581	SWIS	SWIS	93'	No	8	
SWS	1AC515/ CC515	SWIS CONTROL PANEL 1AC515/CC515	SWIS	0292	107'	No	53	
SWS	1DC514	SW PUMP LUBE WATER CONTROL PANEL 1DC514	SWIS	SWIS	102'	Yes		
VAS	AVH208 AHU	HT EXCH A/BVH208 RCIC UNIT COOLERS	Reactor	4110	54'	Yes		C-64,65
VAS	AVH209 AHU	HT EXCH A/BVH209 HPCI UNIT COOLERS	Reactor	4111	54'	Yes		
VAS	AVH210 AHU	HT EXCH A/B/C/D/E/F/G/HVH210 RHR UNIT COOLERS	Reactor	4107	54'	Yes		
VAS	AVH211 AHU	HT EXCH A/B/C/D/E/F/GVH211 CSS UNIT COOLERS	Reactor	4116/4118	54'	Yes		C-111,112
VAS	AVH214 AHU	HT EXCH A/B/C/DVH214 SACS UNIT COOLERS	Reactor	4307/4309	102'	Yes		
VAS	AVH208 FAN	FAN A/BVH208 RCIC UNIT COOLER FANS	Reactor	4110	54'	Yes		
VAS	AVH209 FAN	FAN A/BVH209 HPCI UNIT COOLER FANS	Reactor	4111	54'	Yes		
VAS	AVH210 FAN	FAN A/B/C/D/E/F/G/HVH210 RHR UNIT COOLER FANS	Reactor	4107/4109	54'	Yes		C-53
VAS	AVH211 FAN	FAN A/B/C/D/E/F/GVH211 CSS UNIT COOLER FANS	Reactor	4116/4118	54'	Yes		C-111,112
VAS	AVH214 FAN	FANS A/B/C/DVH214 SACS UNIT COOLER FANS	Reactor	4309	102'	Yes		
VAS	TEMPSENS	TEMP. SENSORS FOR ROOM COOLER	Reactor	4107/4110	54'	Yes		C-114

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
VCA	AVH400AHU	CONTROL ROOM EMERGENCY FILTER SYSTEMS AHU AVH400/BVH400	Control Diesel	5602/ 5630	155'	Yes		C-211,212, 213
VCA	AVH407AHU	CONTROL EQUIP. ROOM SUPPLY AHU AVH407/BVH407	Control Diesel	5703	178'	Yes		C-281,282 ,287
VCA	AVH403AHU	CONTROL ROOM SUPPLY SYSTEM AHU AVH403/BVH403	Control Diesel	5602/ 5630	155'	Yes		
VCA	AVH403FAN	CONTROL ROOM SUPPLY SYSTEM FANS AVH403/BVH403	Control Diesel	5602/ 5630	155'	Yes		
VCA	AV415FAN	CONTROL ROOM SUPPLY SYSTEM FANS AV415/BV415	Control Diesel	5630	155'	Yes		C-215
VCA	AV400FAN	CONTROL ROOM EMERGENCY FILTER SYSTEM FANS AV400/BV400	Control Diesel	5602	155'	Yes		C-211,213
VCA	AVH403F	FILTER AVH403/BVH403	Control Diesel	5602/ 5630	155'	Yes		
VCA	AVH400DF	DOWNSTREAM FILTER AVH400/ BVH400	Control Diesel	5602/ 5630	155'	Yes		
VCA	AVH400UF	UPSTREAM FILTER AVH400/BVH400	Control Diesel	5602/ 5630	155'	Yes		
VCA	AVH407F	FILTER A/BVH407	Control Diesel	5703	178'	Yes		C-283
VCA	AVH400CF	CHARCOAL FILTER AVH400	Control Diesel	5602	155'	Yes		
VCA	BVH400CF	CHARCOAL FILTER BVH400	Control Diesel	5630	155'	Yes		
VCA	HD9593A	DAMPER HD9593A/B	Control Diesel	5630	155'	No	32	C-227
VCA	HD9594A	DAMPER HD9594A/B	Control Diesel	5630	155'	No	32	C-218,219, 226
VCA	HD9588AA	DAMPER HD9588AA/AB/BA/BB	Control Diesel	5630	155'	No	32	C-220
VCA	HD9589A1	DAMPER HD9589A1/ A2/ B1/ B2	Control Diesel	5630	155'	No	33	C-187,188, 189,190
VCA	HD9603A1	DAMPER HD9603A3 CNTRL EQUIP AREA	Control Diesel	5703	178'	No	48	C-192,193, 284,285

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
VCA	HD9603B1	DAMPER HD9603B1/ B2/ B3 CNTRL EQUIP AREA	Control Diesel	5703	178'	No	48	C-194
VCA	FD9589A	DAMPER FD9589A/B	Control Diesel	5602/5630	155'	Yes		
VCA	AVH407	FAN UNIT AVH407/ BVH407	Control Diesel	5703	178'	Yes		C-281,282
VCA	FD9595A	DAMPER FD9595A/B	Control Diesel	5630	155'	Yes		C-214,225
VCA	HD9595A	DAMPER HD9595A /B	Control Diesel	5630	155'	Yes		
VCA	PDD9587A	DAMPER PDD9587A & B	Control Diesel	5602/ 5630	155'	Yes		C-186,191
VDG	AE412	HEAT EXCHANGER AE412/BE412/CE412/DE412/ EE412/FE412/GE412/HE412	Control Diesel	5211	77'	No	49	C-104
VDG	D472A	DAMPER D472A/D472B/D472C/D472D/ D472E/D472F/D472G/D472H	Control Diesel	Unknown	77'	Yes		C-278
VDG	AV412	FAN AV412/BV412/CV412/ DV412/EV412/FV412/ GV412/HV412	Control Diesel	5208	77'	Yes		C-102
VIS	D503A	DAMPER D503A /D503B/D503C/D503D	SWIS	SWIS	122'	Yes		
VIS	D504A	DAMPER D504A/D504B/D504C/D504D	SWIS	SWIS	122'	Yes		
VIS	AV503	FAN AV503/BV503/CV503/DV503	SWIS	SWIS	122'	No	30	

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
VIS	AV504	FAN AV504/BV504/CV504/DV504	SWIS	SWIS	122'	No	30	
VIS	9773A1	DAMPER TD9773A1/B1/C1/D1	SWIS	SWIS	122'	Yes		
VIS	9773A3	DAMPER TD9773A3/B3/C3/D3	SWIS	SWIS	122'	Yes		
VPR	AVH408 AHU	COOLING COIL AVH408/BVH408	Control Diesel	5620	163'	Yes		C-256
VPR	GM497A	DAMPER GM497A/B	Control Diesel	5620	163'	Yes		
VPR	GM498A	DAMPER GM498A /B	Control Diesel	5620	163'	Yes		C-257
VPR	GM499A	DAMPER GM499A/B	Control Diesel	5704	178'	Yes		
VPR	AVH408 FAN	FAN AVH408 /BVH408	Control Diesel	5620	163'	No	35	
VPR	AV416	FAN AV416/BV416	Control Diesel	5704	178'	No	17	C-166
VPR	AVH408F	FILTER AVH408/BVH408	Control Diesel	5620	163'	Yes		C-256
VPR	9558A1	DAMPER HD9558A1/B1	Diesel Generator Room	5620	160'	Yes		C-207
VPR	9558A	DAMPER FD9558A/B	Diesel Generator Room	5620	160'	Yes		
VSW	AVH401AAHU	HT EXCH AVH401A/BVH401A/ CVH401A/DVH401A	Control Diesel	5606	163'	Yes		
VSW	AVH401B AHU	HT EXCH AVH401B/BVH401B/CVH401B/ DVH401B (COIL B)	Control Diesel	5606	163'	Yes		C-203A,259
VSW	AVH401 FAN	FAN AVH401/BVH401/CVH401/DVH401	Control Diesel	5606	160'	Yes		C-158,204

Table 3-4 (Continued)
Screening of Hope Creek Seismic IPEEE Components

SYS	EQUIP ID. NO.	NAME	BUILDING	ROOM NO.	FL_EL	SCREEN	NOTE	PHOTO
VSW	AV406FAN	FAN AV406/BV406/CV406/DV406 BATTERY ROOM	Control Diesel	5606	160'	No	11	C-206
VSW	AVH401F	FILTER AVH401/BVH401/CVH401/DVH40 1	Control Diesel	5606	163'	Yes		C-259
VSW	9547A	DAMPER HD9547A /B/C/D BATTERY ROOM	Control Diesel	5606	160'	Yes		C-205
VSW	9549A	DAMPER HD9549A/B/C/D	Control Diesel	5606	160'	Yes		C-203B
VTS	D505A	DAMPER D505A	SWIS	TS Area	122'	Yes		
VTS	AV558FAN	FAN AV558/BV558	SWIS	TS Area	122'	Yes		
VTS	9774A1	DAMPER TD9774A1/B1/A2/B2	SWIS	TS Area	122'	Yes		
AP	10T522	CONDENSATE STORAGE TANK	YARD	N/A	GRADE	No	36	C-151
	1YF401	120V AC FUSE PANELS, 1YF401-4	Control Diesel	5302	102'	No	30	C-249,250
	10C620	HPCI RELAY VERTICAL BOARD (H11-P620)	Control Diesel	5302	102'	No		C-247,248
	10C617	RHR&CS RELAY BOARDS (10-C617-41)	Control Diesel	5302	102'	No	5	C-245,246
	10C621	RCIC RELAY VERTICAL BOARD	Control Diesel	5302	102'	No		
	10C628	ADS RELAY VERTICAL BOARDS	Control Diesel	5302	102'	No	5	
	10C399	AUXILIARY SHUTDOWN PANEL	Aux/Rad	3579	137'	No	39	C-172,173
MISC	10651A-E	OPERATORS CONSOLE	Control Diesel	5510	137'	No	5	
MISC	CEILING	SUSPENDED CEILING	Control Diesel	5510	137'	Yes		C-169,170
MISC	INSTRUM RACKS	INSTRUMENT RACKS	Various	N/A	VARIOUS	Yes		C-69, 70, 269
	STRUCTURES	SEISMIC CATEGORY I CIVIL STRUCTURES AND TURBINE BUILDING	--	--	GRADE	No	--	--

Table 3-5
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
ACP	120Vac CONTROL POWER BUS 10Y406	10Y406	Control/Diesel	54	Functionality & Anchorage	>1.50				Replace missing nut for one of the bolts.
ACP	CLASS 1E 120V PANEL 1AJ481/1CJ481	1AJ481	Control/Diesel	137	Anchor Tabs	1.08	0.33	0.36	0.35	
ACP	CLASS 1E 120V PANEL 1BJ481/1DJ481	1BJ481	Control/Diesel	124	Anchor Tabs	1.08	0.33	0.36	0.35	
ACP	CLASS 1E 120V PANEL 1AJ482/1CJ482	1AJ482	Control/Diesel	163	Anchor Tabs	1.03	0.33	0.36	0.33	
ACP	CLASS 1E 120V PANEL 1BJ482/1DJ482	1BJ482	Control/Diesel	163	Anchor Tabs	1.03	0.33	0.36	0.33	
ACP	CLASS 1E 4.16kV BUS 10A401- DIV A/10A402-DIV B/10A403-DIV C/10A404 DIV D	10A401	Control/Diesel	130	Functionality & Anchorage	>1.50				Functionality fragility is based on Low Ruggedness Relay GE PVD GERS capacity.
ACP	CLASS 1E 480Vac MCC 10B411-A/10B421- B/10B431-C/10B441-D	10B411	Control/Diesel	130	Functionality & Anchorage	>1.50				

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
ACP	CLASS 1E 480Vac UNIT SUBST 10B410-A/10B420- B/10B430-C/10B440-D	10B410	Control/Diesel	130	Functionality & Anchorage	>1.50				Since relays identified in the walkdown are not on the Low Ruggedness Relay list, and that functional Am>1.5g based on the fragility evaluation, the interaction concern of pounding adjacent cabinets is screened out.
ACP	CLASS 1E 480Vac MCC 10B451A/10B461-B/10B471-C/10B481-D	10B451	Control/Diesel	130	Functionality & Anchorage	>1.50				
ACP	CLASS 1E 480Vac UNIT SUBST 10B450A/10B460-B/10B470-C/10B480-D	10B450	Control/Diesel	130	Functionality & Anchorage	>1.50				Since relays identified in the walkdown are not on the Low Ruggedness Relay list, and that functional Am>1.5g based on the fragility evaluation, the interaction concern of pounding adjacent cabinets is screened out.
ACP	480Vac BUS 10B553/573/583	10B553	SWIS	93	Functionality & Anchorage	>1.50				
ACP	480Vac BUS 10B563	10B563	SWIS	93	Functionality & Anchorage	>1.50				No Low Ruggedness Relay was identified by the system analyst. Therefore, the interaction concern was dismissed.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
ACP	1AD481 & 1AD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	1AD481	Control/Diesel	137	Functionality & Anchorage	>1.5				Light fixture is judged not heavy enough to be of interaction concern.
ACP	1BD481 & 1BD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	1BD481	Control/Diesel	124	Functionality & Anchorage	>1.5				
ACP	1CD481 & 1CD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	1CD481	Control/Diesel	137	Functionality & Anchorage	>1.5				
ACP	1DD481 & 1DD482 AUCTIONEERNG CIRC INCL. INVERTER, STATIC SWITCH LOGIC CIRCUIT, 480-130 AC-DC VOLT RECT.	1DD481	Control/Diesel	124	Functionality & Anchorage	>1.5				
ACP	1E 480Vac MCC 10B212 DIV A/ 10B222 DIV B/ 10B232 DIV C/ 10B242 DIV D	10B212	Reactor	77	Functionality & Anchorage	>1.5				
ACP	NON 1E 480Vac MCC 10B252 DIV A/ 10B262 DIV B/ 10B272 DIV C/ 10B282 DIV D	10B252	Reactor	77	Functionality & Anchorage	>1.5				

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
ACP	NON 1E 480Vac MCC 10B313 DIV A/ 10B323 DIV B	10B313	Reactor	153	Functionality & Anchorage	>1.5				Seismic Interaction from cable tray is judged not to be a concern based on the low seismic response. Also relays are not on Low Ruggedness Relay list so pounding interaction is not a concern.
ACP	NON 1E DIV C 480Vac MCC 1OB474	10B474	Control/ Diesel	160	Functionality & Anchorage	>1.5				Seismic Interaction from cable tray is judged not to be a concern based on the low seismic response from the response analysis (EQE, 1995a).
ACP	LOCL GEN CNTRL PNL 1AC421/ 1BC421/ 1CC421/ 1DC421	1AC421	Control/ Diesel	102	Functionality & Anchorage	>1.5				
ACP	REM GEN CNTRL PNL 1AC422/ 1BC422/ 1CC422/ 1DC422	1AC422	Control/ Diesel	130	Functionality & Anchorage	>1.5				
ACP	DIESEL PANELS - 1A/B/C/D - C423	1A-C423	Control/ Diesel	130	Functionality & Anchorage	>1.5				

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
ACP	D/G LOAD SEQUENCE PANEL 1A/B/C/D - C428	1A-C428	Control/Diesel	130	Functionality & Anchorage	>1.50				Fragility estimate is based on the assumption that anchorage installation of the cabinet is complete.
ACP	PNL 1AC652/ 1BC652/ 1CC652/1DC652	1AC652	Control/Diesel	102	Functionality & Anchorage	>1.50				
ACP	NON-1E 480V MCC 10B263	10B263	Reactor	102	Functionality & Anchorage	>1.50				
ACP	OFFSITE POWER	--	Yard	Grade	Ceramic Insulators	0.31	0.25	.43	0.10	Generic value determined based on a walkdown review of switchyard component.
CAC	AOV-HV-4956/4958/4964/4979	HV-4956	Reactor	102	Anchorage				>0.5	The support of valve HV-4956 was judged to have HCLPF > 0.5g based on the low response of the building.
CAC	TORUS VENT ISO MANUAL OPERATION FOR AOV-11541	1GSHV-11541	Reactor	77	Anchorage	>1.50				
CAC	OUTBOARD CONTAINMENT ISOLATION MANUAL OVERRIDE	1GSHV-4962/4964	Reactor	102	Anchorage	>1.50				

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
CHC	CNTRL AREA CHILLER AK-400/ BK-400	AK-400	Control/Diesel	155	Anchorage				>0.5	Remove wood plank noted in the walkdown.
CHC	RECIRC PUMP AP-400 / BP400	AP-400	Control/Diesel	155	Anchorage				>0.5	Screened based on HCLPF > 0.5g.
CHS	REMOTE SHUTDOWN PANEL ROOM UNIT VH316A/VH316B	VH316A/ B	Control/Diesel	137	Functionality & Anchorage	>1.50				Equipment is mounted on vibratory equipment, therefore relay chatter is not judged to be a concern.
CHS	TECH SUPPORT CENTER A/C UNITS VH314A/314B	VH314A/ B	Control/Diesel	152	Functionality & Anchorage	>1.50				Equipment is mounted on vibratory equipment, therefore relay chatter is not judged to be a concern.
CRH	SAMPLING OF CRD SCRAM INLET AND OUTLET VALVES	XV-126	Reactor	102	Pipe Stresses	>1.50				1/2" pipe is judged to be adequately supported around the valve.
CSS	CSS PUMP A/B/C/D P206	AP206	Reactor	54	Anchorage				>0.5	Screened based on HCLPF > 0.5g.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
DCP	CLASS 1E 125Vdc BUS 10D410 DIV A/10D420 DIV B/ 10D430 DIV C/ 10D440 DIV D	10D410	Control/ Diesel	130	Functionality & Anchorage	>1.50				
DCP	CLASS 1E 125Vdc BUS 10D436 DIV C/10D446 DIV D	10D436	Control/ Diesel	160	Functionality & Anchorage	>1.50				
DCP	125Vdc 1E PWR TO LOADS IN PNL 1AD417/ 1BD417/ 1CD417/ 1DD417	1AD417	Control/ Diesel	130	Tab Plate Anchorage	1.47	0.17	0.40	0.57	Housekeeping items noted in the walkdown need to be resolved.
DCP	250Vdc MCC 10D251/ 10D261	10D251	Reactor	54	Functionality	0.73	0.25	0.30	0.29	The TRS used for qualifying these MCCs were significantly lower than the TRS used for other MCCs.
DCP	CLASS 1E 250Vdc BUS 10D450 DIV A/ 10D460 DIV B	10D450	Control/ Diesel	54	Functionality	1.36	0.20	0.34	0.56	
DCP	125Vdc BAT 1AD411-DIV A/ 1BD411-DIV B/ 1CD411-DIV C/ 1DD411-DIV D	1AD411	Control/ Diesel	146	Functionality & Anchorage	>1.50				Overhead drain pan was judged not posing an interaction concern due to its light weight.
DCP	250Vdc BAT CHGR 10D423 /10D433	10D423	Control/ Diesel	54	Functionality & Anchorage	>1.50				Mobil crane noted in the walkdown should be restrained or moved away from safety-related equipment.
DCP	NON 1E 125Vdc DISTR PNL 1AD318/ 1BD318/ 1CD318/ 1DD318	1AD318	Aux/ Rad	124	Functionality & Anchorage	>1.50				Remove the interaction source for AD, CD, and DD panels, noted in the walkdown.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
DCP	125Vdc BAT CHGR 1CD444/1DD444	1CD444	Control/ Diesel	163	Functionality & Anchorage	>1.5				
DCP	125Vdc BAT CHGR 1A/B/C/DD413 / 1A/B/C/DD414	1AD413/1 AD414	Control/ Diesel	146	Functionality & Anchorage	>1.5				
DGS	EXCITER PANELS, 1A/B/C/D C420	1A-C420	Control/ Diesel	102	Functionality & Anchorage	>1.5				
DGS	THERMSTAT CNTRL VLV TV 6606A/B/C/D	TV6606A	Control/ Diesel	102	Functionality	>1.5				Engineering judgment. Since there is no heavy external operator mounted on the valve body, cast iron valve body is not a concern.
DGS	THERMSTAT CNTRL VLV TV 6618A/B/C/D	TV6618A	Control/ Diesel	102	Functionality	>1.5				Engineering judgment. Since there is no heavy external operator mounted on the valve body, cast iron valve body is not a concern.
DGS	THERMSTAT CNTRL VLV TV 7722AB/C/D	TV7722A	Control/ Diesel	102	Functionality	>1.5				Engineering judgment. Since there is no heavy external operator mounted on the valve body, cast iron valve body is not a concern.
DGS	SDG A/B/C/D START AIR RECEIVER - AT/BT/CT/DT/ET/ FT/GT/HT 408	AT408	Control/ Diesel	102	Anchorage	>1.5				Engineering judgment. The air receiver tank is screened out based on low building response and as-built anchorage. It is bolted to the skid and to the wall.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
IAS	EMERGENCY IAS AIR RECEIVER TANK	10-T106	Turbine	120	Functionality & Anchorage	>1.5				Adjacent masonry block wall has a median capacity greater than 1.5g. Thus, the wall poses no interaction concern.
IAS	IAS AIR DRYER	10-F104	Turbine	120	Functionality & Anchorage	>1.5				Adjacent masonry block wall has a median capacity greater than 1.5g. Thus, the wall poses no interaction concern.
IAS	IAS AIR RECEIVER TANK	1A-T132	Turbine	120	Functionality & Anchorage	>1.5				Adjacent masonry block wall has a median capacity greater than 1.5g. Thus, the wall poses no interaction concern.
IGS	1A/B-C213, A AND B IGS CONTROL PANELS	1A-C213	Reactor	132	Functionality & Anchorage	>1.5				
RAC	RACS HX AE-217/ BE-217	AE-217	Reactor	77	Anchorage	>1.5				The HX is screened out based on drawing review and engineering judgment.
RAC	RACS PUMP AP-209 / BP-209	AP-209	Reactor	77	Anchorage				>0.5	Screened based on HCLPF > 0.5g.
RHS	RHS PUMP A/B/C/D P202	1A-P202	Reactor	54	Anchorage				>0.5	Screened based on HCLPF > 0.5g.
SAC	AOV HV2290A B/C/D/E/F/G/H (RHR)	2290A	Reactor	54	Operator yoke stresses	>1.5				
SAC	AOV HV2293A/B (RCIC)	2293A	Reactor	54	Functionality	>1.5				Based on the low response and pipe flexibility, interaction was judged not a concern.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	b_r	b_u	HCLPF (g)	NOTES
SAC	AOV HV2325A/B/C/D/E/F/G (CSS)	2325A	Reactor	54	Functionality	>1.50				
SAC	AOV HV2325H (CSS)	2325H	Reactor	54	Functionality	0.89	0.25	0.29	0.37	Close proximity of the valve operator to concrete wall. The reported generic fragility is based on a HCLPF peak spectral acceleration of 0.8g and a beta-C of 0.4.
SAC	SAC CONTROL PANELS, 1A/B/C-C201	1AC201	Reactor	102	Functionality & Anchorage	>1.50				
SAC	SAC EXPANSION TANKS A/BT205	AT205	Reactor	201	Functionality & Anchorage	>1.50				
SAC	SAC CONTROL PANELS, 1D-C201	1DC201	Reactor	102	Functionality & Anchorage	>1.50				The overhead fan noted in the walkdown was judged not to be a concern.
SWS	SW PUMP CONTROL PANELS	PMPCN TRL	SWIS	110	Functionality & Anchorage	>1.50				
SWS	SWIS HVAC CONTROL PANEL AC/EC 581	AC/EC 581	SWIS	93	Functionality & Anchorage	>1.50				
SWS	SWIS CONTROL PANEL 1AC515/CC515	1AC515 /CC515	SWIS	107	Functionality	>1.50				Since no relay on the Low Ruggedness Relay list was identified in the walkdown, and that functional A_m>1.5g, the interaction concern of pounding adjacent cabinets is screened out.
VCA	DAMPER HD9593A/B	HD9593 A	Control/ Diesel	155	Equipment	>1.50				The damper and its operator (MOV) are well supported on a floor mounted steel frame.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
VCA	DAMPER HD9594A	HD9594A	Control/ Diesel	155	Equipment	0.89	0.25	0.29	0.37	The seismic fragility was estimated using the HCLPF peak spectral acceleration of 0.8g and a beta-C of 0.4.
VCA	DAMPER HD9594B	HD9594B	Control/ Diesel	155	Equipment	>1.50				The damper and its operator are well supported.
VCA	DAMPER HD9588AA/AB/BA/BB	HD9588A	Control/ Diesel	155	Anchorage	0.89	0.25	0.29	0.37	The seismic fragility was estimated using the HCLPF peak spectral acceleration of 0.8g and a beta-C of 0.4.
VCA	DAMPER HD9589A1/ A2/ B1/ B2	HD9589A1	Control/ Diesel	155	Anchorage	>1.50				Further review of the walkdown data dismissed the interaction concern.
VCA	DAMPER HD9603A3 CNTRL EQUIP AREA	HD9603A1	Control/ Diesel	178	Anchorage	>1.50				Binding of the linkage due to differential displacement was judged not to be a concern based on low response & stiff supports.
VCA	DAMPER HD9603B1/ B2/ B3 CNTRL EQUIP AREA	HD9603B1	Control/ Diesel	178	Anchorage	>1.50				Binding of the linkage due to differential displacement was judged not to be a concern based on low response & stiff supports.
VDG	HEAT EXCHANGER AE412/BE412/CE412/DE412/EE412/FE412/GE412/HE412	AE412	Control/ Diesel	77	Anchorage	>1.50				The cooling coil support frame is attached to wall and slabs along three sides. The equipment is screened out based on the low building response and the as-built conditions.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
VIS	FAN AV503 /BV503/CV503/DV503	AV503	SWIS	122	Anchor Bolts	>1.50				
VIS	FAN AV504/BV504/CV504/ DV504	AV504	SWIS	122	Anchor Bolts	>1.5				
VPR	FAN AVH408 /BVH408	AVH408 FAN	Control /Diesel	163	Anchorage	0.50	0.25	0.25	0.22	No access to review the fans. Fans are assumed on vibration isolators for reporting seismic fragility.
VPR	FAN AV416/BV416	AV416	Control /Diesel	178	Anchorage	0.50	0.25	0.25	0.22	Fans are mounted on vibration isolators.
VSW	FAN AV406/ BV406/CV406/ DV406 BATTERY ROOM	AV406 FAN	Control /Diesel	160	Anchorage	0.50	0.25	0.25	0.22	Fans are mounted on vibration isolators.
AP	CONDENSATE STORAGE TANK	10T522	Yard	Grade	Tank Shell	0.95	0.27	0.36	0.34	
	120V AC FUSE PANELS, 1YF401-4	1YF401	Control /Diesel	102	Anchor Tabs	1.10	0.39	0.41	0.29	
	HPCI RELAY VERTICAL BOARD (H11-P620)	10C620	Control /Diesel	102	Functionality & Anchorage	>1.5				Median capacity was estimated using qualification data and median floor response spectra.
	RHR&CS RELAY BOARDS (10-C617-41)	10C617	Control /Diesel	102	Functionality & Anchorage	>1.5				Median capacity was estimated using qualification data and median floor response spectra.

Table 3-5 (Continued)
Hope Creek IPEEE Screened-In Components Seismic Fragilities

SYS	Equipment Name	Equip ID	BLDG.	El (ft)	FAILURE MODE	A_m (g)	β_r	β_u	HCLPF (g)	NOTES
	RCIC RELAY VERTICAL BOARD	10C621	Control /Diesel	102	Functionality & Anchorage	>1.50				Median capacity was estimated using qualification data and median floor response spectra.
	ADS RELAY VERTICAL BOARDS	10C628	Control /Diesel	102	Functionality & Anchorage	>1.50				Median capacity was estimated using qualification data and median floor response spectra.
	REMOTE SHUTDOWN PANEL	10C399	Aux/ Rad	137	Functionality & Anchorage	>1.50				
MISC	OPERATORS CONSOLE	10651A-E	Control /Diesel	137	Functionality & Anchorage				>0.5	The HCLPF capacity is estimated based on design FRS and the 84th floor response spectra.
	REACTOR BUILDING. AUX BUILDING, STATION SERVICE WATER INTAKE STRUCTURE, TURBINE BUILDING	--	--	--	--	>1.50	--	--	--	Structure response factors included in the equipment fragility evaluation were considered as discussed in Paragraphs 4.1.3 and 5.1.4.

Table 3-6
Hope Creek Seismic Fragilities

Equipment Name	Abbrev	Median Accel (g)	b_r	b_u	HCLPF (g)
Offsite Power (Station power transformers)	SWYRD	0.31	0.25	0.43	0.10
1E 120V Instrumentation Distribution Panels 1A(B,C,D)J481	PNL481	1.08	0.33	0.36	0.35
1E 120V Instrumentation Distribution Panels 1A(B,C,D)J482	PNL482	1.03	0.33	0.36	0.33
125V DC 1E power to panels 1A(B,C,D)417	125Vdc	1.47	0.17	0.40	0.57
250V DC MCC 10D251/10D261	250MCC	0.73	0.25	0.30	0.29
1E 250Vdc buses 10D450 and 10D460	250V BUS	1.36	0.20	0.34	0.56
Firewater tanks 0A-T-508 and 0B-T-508	Not used	0.73	0.27	0.36	0.26
Firewater Pumps (fragility governed by tanks)	Not used	0.73	0.27	0.36	0.26
SACS AOV 1EGHV-2325H	Not used	0.89	0.25	0.29	0.37
Damper 1GKHD-9594A	CREFA	0.89	0.25	0.29	0.37
Dampers 1GKHD-9588AA/AB/BA/BB	CRS	0.89	0.25	0.29	0.37
Fans 1A/B-VH408	PNLHVC	0.50	0.25	0.25	0.22
Fans 1A/B-V-416	Not used	0.50	0.25	0.25	0.22
Fans 1A/B/C/D-V-406	Not used	0.50	0.25	0.25	0.22
Condensate Storage Tank	CSTNK	0.95	0.27	0.36	0.34
120V AC fuse panels 1Y-F-401/402/403/404	CNTVNT	1.10	0.39	0.41	0.29
Small LOCA due to seismic event	SLOCA	1.50	0.30	0.50	0.40
RANDOM FAILURES		Mean	EF		
Recovery of Panel Room HVAC	HVREC	3.0E-3	5.0		
Remote Shutdown	RSDOWN	6.3E-2	5.0		

Table 3-7
Hope Creek Seismic Damage State Frequencies

Sequence		Seismic Damage State Frequency (per year) LLNL Hazard	EPRI Hazard	Core Damage (CD) or Developed Further (DF)
2	S-S2	1.8E-7	7.9E-8	DF
3	S-CV	6.0E-7	4.4E-7	DF
4	S-CV-S2	6.9E-9	2.7E-9	DF
5	S-CT	4.0E-7	2.6E-7	DF
6	S-CT-S2	6.1E-9	3.6E-9	DF
7	S-CT-CV	1.9E-8	7.4E-9	DF
8	S-CT-CV-S2	1.0E-9	<1E-9	DF
9	S-HP	8.1E-7	4.4E-7	DF
10	S-HP-S2	2.0E-8	5.0E-9	DF
11	S-HP-CV	5.8E-8	1.9E-8	DF
12	S-HP-CV-S2	3.2E-9	<1E-9	DF
13	S-HP-CT	6.1E-8	2.7E-8	DF
14	S-HP-CT-S2	4.3E-9	<1E-9	DF
15	S-HP-CT-CV	1.1E-8	1.8E-9	DF
16	S-HP-CT-CV-S2	<1E-9	<1E-9	DF
17	S-CR	2.2E-9	<1E-9	DF
18	S-OP	6.3E-5	5.9E-5	DF
19	S-OP-S2	5.4E-7	1.6E-7	DF
20	S-OP-CV	1.6E-6	6.4E-7	DF
21	S-OP-CV-S2	6.9E-8	9.9E-9	DF
22	S-OP-CT	1.4E-6	4.4E-7	DF
23	S-OP-CT-S2	8.2E-8	1.2E-8	DF
24	S-OP-CT-CV	2.3E-7	3.7E-8	DF
25	S-OP-CT-CV-S2	2.4E-8	1.6E-9	DF
26	S-OP-HP	3.8E-6	1.1E-6	DF
27	S-OP-HP-S2	2.4E-7	3.4E-8	DF
28	S-OP-HP-CV	6.7E-7	1.0E-7	DF
29	S-OP-HP-CV-S2	6.8E-8	4.7E-9	DF
30	S-OP-HP-CT	8.1E-7	1.0E-7	DF
31	S-OP-HP-CT-S2	9.8E-7	6.4E-9	DF
32	S-OP-HP-CT-CV	2.5E-7	1.7E-8	DF
33	S-OP-HP-CT-CV-S2	3.9E-8	1.1E-9	DF
34	S-OP-CR	4.6E-8	3.7E-9	DF
35	S-IC2	1.6E-7	4.6E-8	CD
36	S-IC1	2.5E-6	6.7E-7	CD
37	S-DC	4.4E-7	5.5E-8	CD
38	S-HV	5.4E-8	2.1E-8	CD

Table 3-8
Hope Creek Seismic Core Damage Frequencies

Sequence		Seismic Damage State Frequency		Conditional	CDF (per year)	
		EPRI Hazard	LLNL Hazard	CDP	EPRI	LLNL
S 2	S-S2	7.9E-08	1.8E-07	6.5E-05	5.1E-12	1.2E-11
S 3	S-CV	4.4E-07	6.1E-07	5.8E-05	2.6E-11	3.5E-11
S 5	S-CT	2.6E-07	4.0E-07	4.2E-05	1.1E-11	1.7E-11
S 9	S-HP	4.4E-07	8.2E-07	4.8E-02	2.1E-08	3.9E-08
S 18	S-OP	5.9E-05	6.3E-05	2.1E-03	1.2E-07	1.3E-07
S 19	S-OP-S2	1.6E-07	5.4E-07	2.1E-03	3.4E-10	1.1E-09
S 20	S-OP-CV	6.4E-07	1.6E-06	2.1E-03	1.3E-09	3.4E-09
S 22	S-OP-CT	4.4E-07	1.4E-06	2.1E-03	9.2E-10	2.9E-09
S 24	S-OP-CT-CV	3.7E-08	2.3E-07	2.1E-03	7.8E-11	4.8E-10
S 26	S-OP-HP	1.1E-06	3.8E-06	5.1E-02	5.6E-08	1.9E-07
S 27	S-OP-HP-S2	3.4E-08	2.4E-07	7.8E-02	2.7E-09	1.9E-08
S 28	S-OP-HP-CV	1.0E-07	6.7E-07	5.1E-02	5.1E-09	3.4E-08
S 30	S-OP-HP-CT	1.0E-07	8.1E-07	5.0E-02	5.0E-09	4.1E-08
S 32	S-OP-HP-CT-CV	1.7E-08	2.5E-07	5.1E-02	8.7E-10	1.3E-08
S 35	S-IC2	4.6E-08	1.6E-07	1.0E+00	4.6E-08	1.6E-07
S 36	S-IC1	6.7E-07	2.5E-06	1.0E+00	6.7E-07	2.5E-06
S 37	S-DC	6.8E-08	4.4E-07	1.0E+00	6.8E-08	4.4E-07
S 38	S-HV	2.1E-08	5.4E-08	1.0E+00	2.1E-08	5.4E-08
				Total CDF	1.0E-06	3.6E-06

Table 3-9
HCGS IPEEE Recovery Actions

Recovery Action	Description	IPE/PRA HEP	Seismic IPEEE HEP
1. NR-HVC-PNRM-12	Fail to provide alternate ventilation within 12 hours after loss of Class 1E Panel Room HVAC	3.0E-4	3.0E-3
2. NR-HVC SWGR-24	Fail to provide alternate ventilation within 24 hours after loss of Switchgear Room ventilation	1.6E-4	1.6E-3
3. NR-RHR-INIT	Fail to initiate RHR for decay heat removal	5.0E-5	5.0E-4
4. NR-SACS-SHED	Fail to align the SACS for long-term operation with one operating SACS pump in each loop	1.0E-2	1.0E-1
5. NR-U1X-DEP-40M	Fail to manually depressurize the RPV within 40 minutes	5.2E-3	5.2E-2
6. NR-U1X-DEP-60M	Fail to manually depressurize the RPV within 60 minutes	4.6E-3	4.6E-2
7. NR-UV-ECCS-1	Fail to manually initiate ECCS within 1 hour	3.9E-2	3.9E-1
8. NR-UV-WTLVL-20M	Fail to control RPV water level with high pressure injection systems	4.3E-2	4.3E-1
9. NR-VENT-5	Fail to initiate containment venting	2.0E-3	3.0E-2
10. NR-WW1-SWP-*	Fail to manually start SACS or SSWS pumps in time periods ranging from 40 minutes to 20 hours	range 1.6E-2 to 7.4E-5	1.6E-1

Table 3-10
HEP Calculation for Remote Shutdown after a Seismic Event

Description: Upon Loss of instrumentation or habitability the control room is evacuated, the operator activates the scram switches in the control room or scrams the reactor by opening breakers on the RPS Power Distribution panels in the RPS MG set room. After the reactor is scrammed the operator proceeds to the RSP to manually operate all transfer switches on panel 10C399 (RSP). Thereafter systems controlled from the RSP are completely isolated from the control room.

HRA Methodology: The HRA methodology originally used in the HCGS IPE used a hybrid of the THERP and EPRI 6560L Time-Reliability Correlation methodologies. This approach is especially useful when analyzing individual actions within the context of a larger procedure. Because the Remote Shutdown Procedure is largely a collection of high level tasks it is not easily amenable to this methodology. Therefore, the simpler but more conservative ASEP methodology was chosen for this analysis.

Procedures: HC.OP-AB.ZZ-0130(Q), CONTROL ROOM EVACUATION, September 23,1994;(PSE&G, 1994b) HC.OP-IO.ZZ-0008(Q), SHUTDOWN FROM OUTSIDE CONTROL ROOM, Revision 9, December 18, 1994(PSE&G, 1994m).

Detailed Description/Timeline: Once the need for control room evacuation is ascertained, it is expected that the reactor will be scrammed and the turbine tripped immediately. Subsequently the operators will close the MSIVs, secure condensate pumps and break the main condenser vacuum. Prior to breaking the vacuum, approximately 15 minutes are required for the turbine speed to decrease to the recommended speed of less than 1200 rpm. Depending on the severity of conditions affecting the control room, some members of the crew may remain in the control room while others begin to establish control at the Remote Shutdown Panel. Under the most extreme conditions, requiring complete abandonment of the control room, necessary prerequisite announcements and gathering of keys and radios is estimated to take 5 minutes.

The Remote Shutdown Panel is located approximately 50-100 feet down a corridor from the Control Room and is easily accessible. Card reader identification is necessary before entry. Therefore, this activity is expected to require no more than five minutes.

Table 3-10 (Continued)
HEP CALCULATION FOR REMOTE SHUTDOWN AFTER A SEISMIC EVENT

Prior to initiating shutdown activities at the panel, communications between requisite local panels and breakers are established. If it could not be completed prior to the evacuation of the control room, the reactor is scrammed by locally opening circuit breakers. Upon the initial reactor scram, it is likely that either HPCI or RCIC was initiated. If HPCI were injecting, it would be tripped and RCIC would be placed in service. Otherwise, RCIC is left or is placed in service. These actions are expected to take up to 20 minutes, including transfer of controls to the Remote Shutdown Panel and verification of automatic actions. Based on thermal hydraulic calculations performed in support of the HCGS IPE, it is assumed core uncovery will occur within 60 minutes of the reactor scram.

Once RCIC is in service, operators establish reactor level between 12.5" and 54.0". Based on previous actions, such as opening of SRVs, this activity is estimated to take 20 to 30 minutes.

Having established injection and reactor level control, decay heat removal must be established. This is accomplished by placing the RHR system in the suppression pool cooling mode. Prerequisite to this, SACS and SSWS Loop B are placed in service, if they are not already in service, for control from the remote shutdown panel. Thermal hydraulic calculations completed to support the HCGS IPE indicate that 24 hours can pass from the initial reactor scram without suppression pool cooling before core damage occurs. However, during that period operators are required to insure that the Control Area Chilled Water and 1E Panel Room Chilled Water Systems are in service. (There is also a note in the procedure, that if time and personnel are available, to place the A RHR loop in the Suppression Pool Cooling Mode using local operations as described in an attachment to the procedure. This activity is not credited because of the subjectivity of the entry conditions and the increased burden of local equipment operation.)

Placing the SACS and SSWS B Loops in service is accomplished with controls at the remote panel. Based on simulator exercises it is estimated that the crews will take 20 to 30 minutes to place both of these in service. If necessary, the operators may be required to place the Control Area and Class 1e Panel Room Chilled Water Systems in

Table 3-10 (Continued)
HEP Calculation for Remote Shutdown after a Seismic Event

service. Manipulation times for both remote and local operations are expected to take 15 minutes. Room heat-up calculations performed to support the HCGS IPE indicate that 5.4 hours is the most limiting time required to place these in service to insure adequate room cooling. Finally, Loop B of the RHR system is placed in the Suppression Pool Cooling Mode. Actual control manipulations and system delays are expected to take less than 1 hour.

It should be noted that redundant Loop A equipment is identified in the procedure, but it is not credited in this analysis since equipment failures are not assumed and local operations are more complex.

HEP Calculation: A Human Error Probability was calculated for Remote Shutdown as shown on the following page. The calculated value is 0.063.

Table 3-11
HCGS Seismic Core Damage Frequencies with IPEEE Human Error Probabilities Equal to IPE/PRA HEPs (LLNL Curve)

Sequence		SDS Frequency		Conditional CDP		CDF (per year)	
		Baseline HEPs	Internal Events HEPs	Baseline HEPs	Internal Events HEPs	Baseline HEPs	Internal Events HEPs
S 2	S-S2	1.8E-07	1.8E-07	6.5E-05	2.0E-05	1.2E-11	3.6E-12
S 3	S-CV	6.1E-07	6.1E-07	5.8E-05	1.7E-05	3.5E-11	1.0E-11
S 5	S-CT	4.0E-07	4.0E-07	4.2E-05	7.7E-06	1.7E-11	3.1E-12
S 9	S-HP	8.2E-07	8.2E-07	4.8E-02	5.5E-03	3.9E-08	4.5E-09
S 18	S-OP	6.3E-05	6.3E-05	2.1E-03	1.5E-03	1.3E-07	9.5E-08
S 19	S-OP-S2	5.4E-07	5.4E-07	2.1E-03	1.5E-03	1.1E-09	8.1E-10
S 20	S-OP-CV	1.6E-06	1.6E-06	2.1E-03	1.5E-03	3.4E-09	2.4E-09
S 22	S-OP-CT	1.4E-06	1.4E-06	2.1E-03	1.5E-03	2.9E-09	2.1E-09
S 24	S-OP-CT-CV	2.3E-07	2.3E-07	2.1E-03	1.5E-03	4.8E-10	3.5E-10
S 26	S-OP-HP	3.8E-06	3.8E-06	5.1E-02	7.1E-03	1.9E-07	2.7E-08
S 27	S-OP-HP-S2	2.4E-07	2.4E-07	7.8E-02	9.9E-03	1.9E-08	2.4E-09
S 28	S-OP-HP-CV	6.7E-07	6.7E-07	5.1E-02	7.3E-03	3.4E-08	4.9E-09
S 30	S-OP-HP-CT	8.1E-07	8.1E-07	5.0E-02	7.1E-03	4.1E-08	5.8E-09
S 32	S-OP-HP-CT-CV	2.5E-07	2.5E-07	5.1E-02	7.3E-03	1.3E-08	1.8E-09
S 35	S-IC2	1.6E-07	1.6E-07	1.0E+00	1.0E+00	1.6E-07	1.6E-07
S 35	S-IC1	2.5E-06	2.6E-06	1.0E+00	1.0E+00	2.5E-06	2.6E-06
S 36	S-DC	4.4E-07	5.0E-07	1.0E+00	1.0E+00	4.4E-07	5.0E-07
S 37	S-HV	5.4E-08	5.8E-09	1.0E+00	1.0E+00	5.4E-08	5.8E-09
					Total CDF	3.6E-06	3.4E-06

Table 3-12
Core Damage Frequency With Varying Credit For Shutdown Without 1E Instrumentation (LLNL Curve)

	0% Success (baseline)	50% Success	90% Success
SDS 36 (S-IC1)	2.5E-6	1.3E-6	2.5E-7
Total Seismic CDF	3.6E-6	2.4E-6	1.4E-6

Table 3-13
Components of Containment System Examined in the IPEEE

- Containment vent valves and nitrogen accumulators
- Containment spray pumps and valves
- Main Steam Isolation Valves (MSIVs)
- Activation sensors and system for containment isolation
- Containment hatches and seals

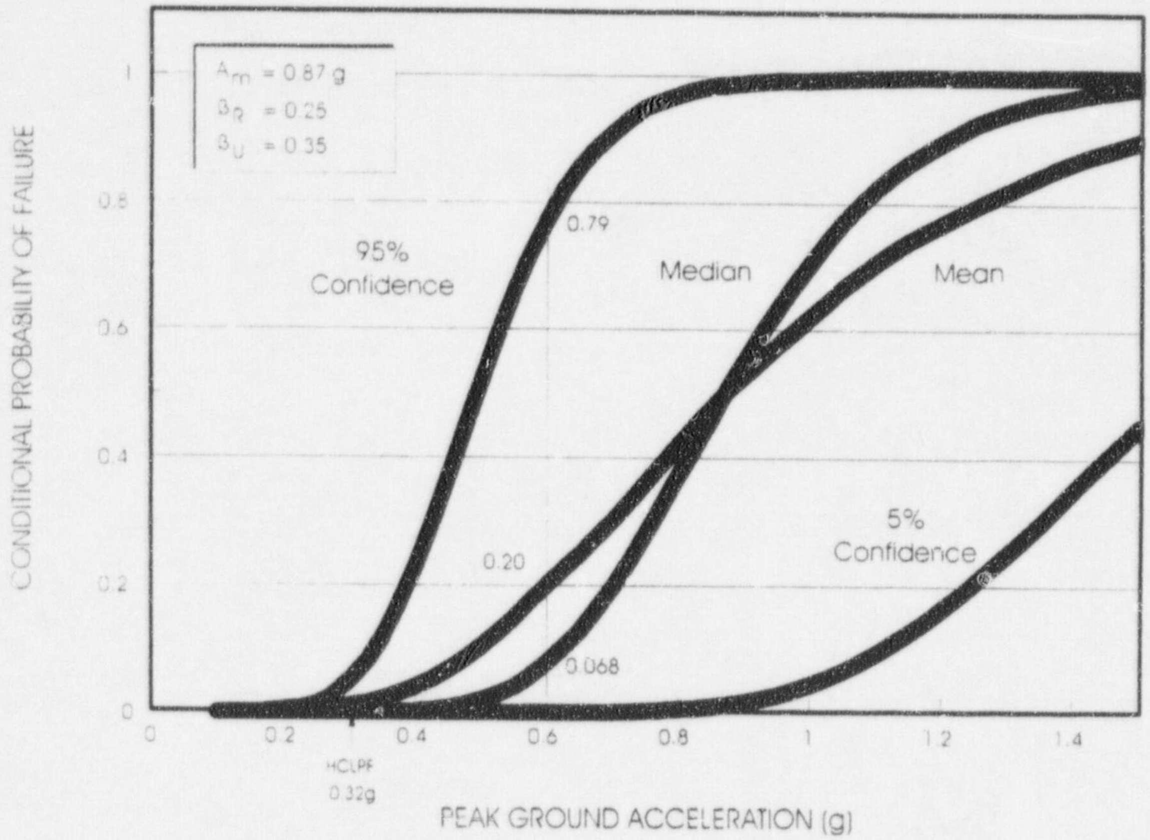

Figure 3-1
Component Fragility Curves

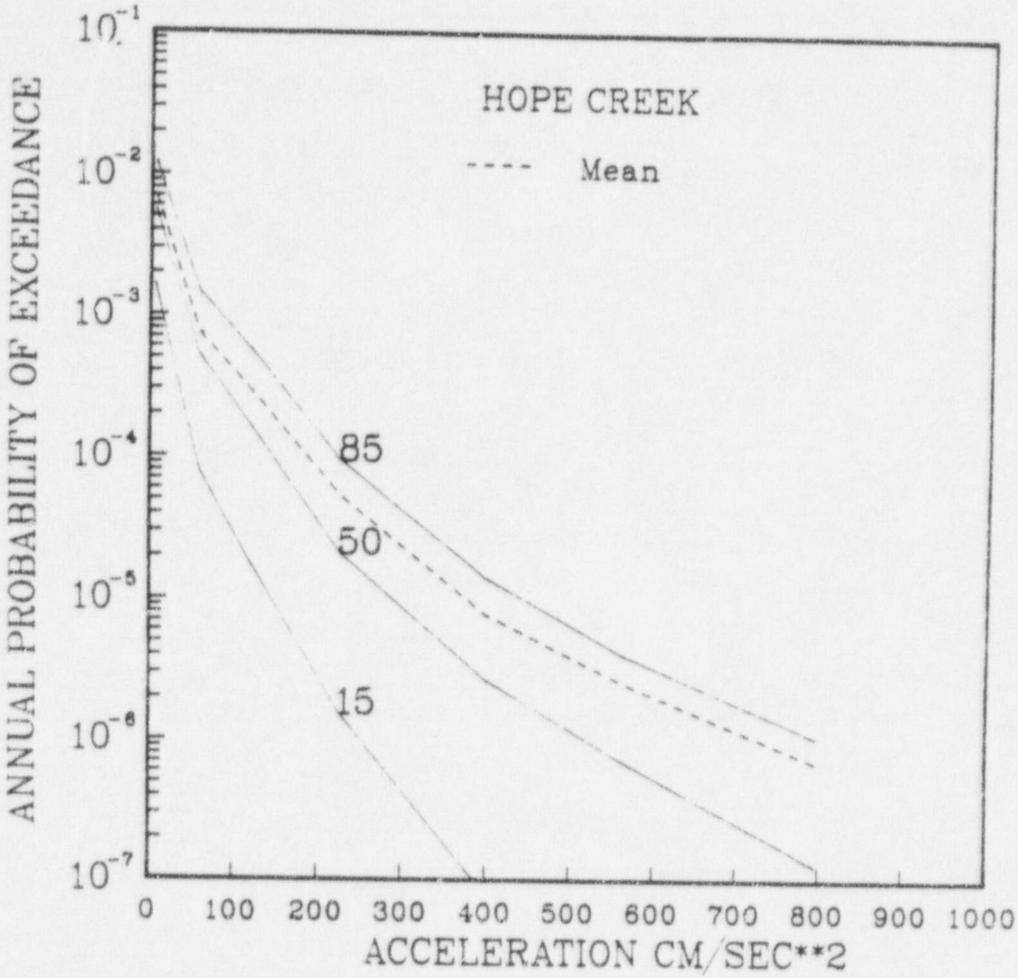

Figure 5-1. 15th, 50th, 85th fractiles and mean annual probability of exceedance of peak ground acceleration.

Figure 3-2
Annual Probability of Exceedance of Peak Ground Acceleration at the Hope Creek Site from EPRI Study (EPRI, 1989a)

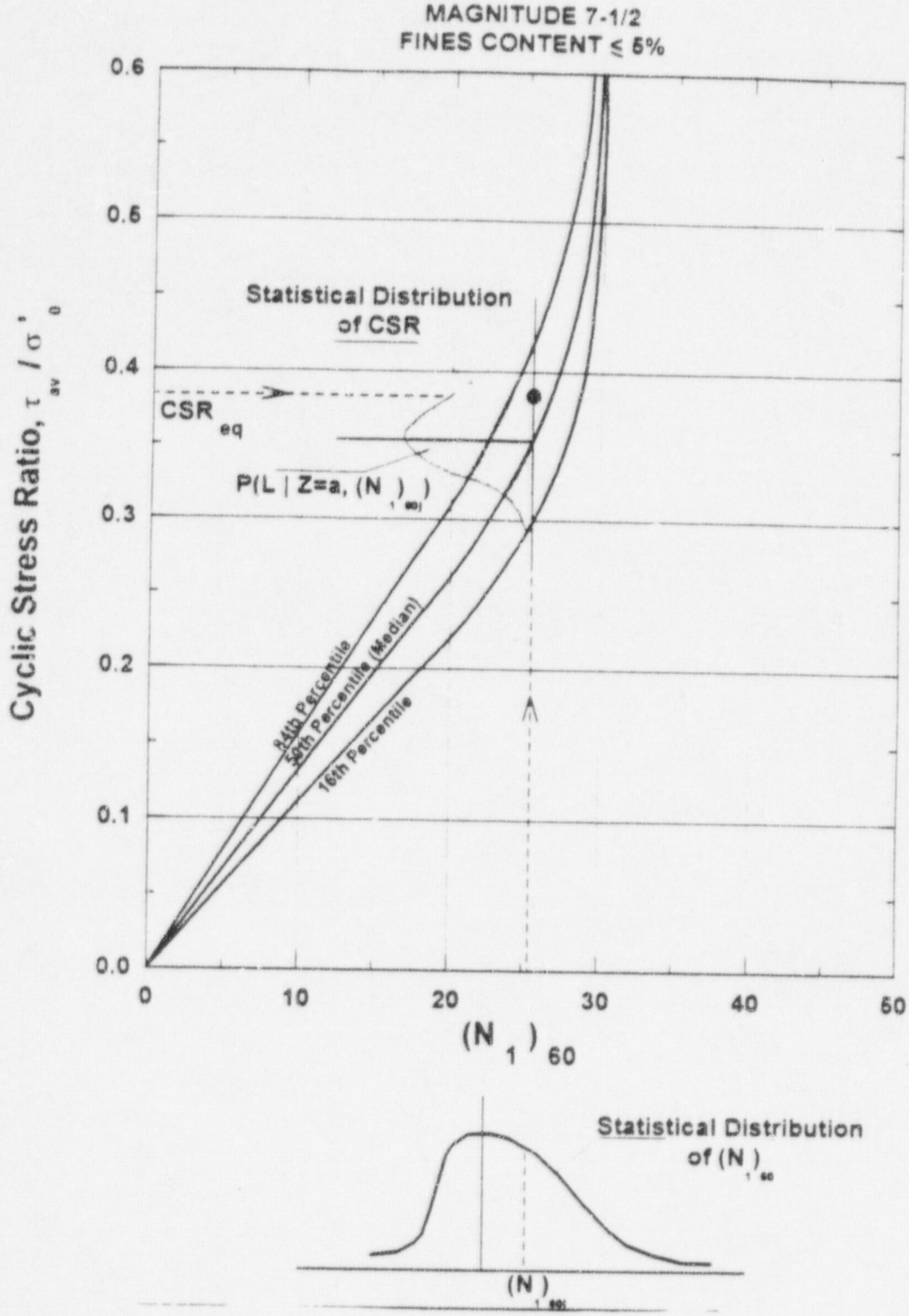

Figure 3-3

Procedure for Computing the Conditional Probability of Liquefaction

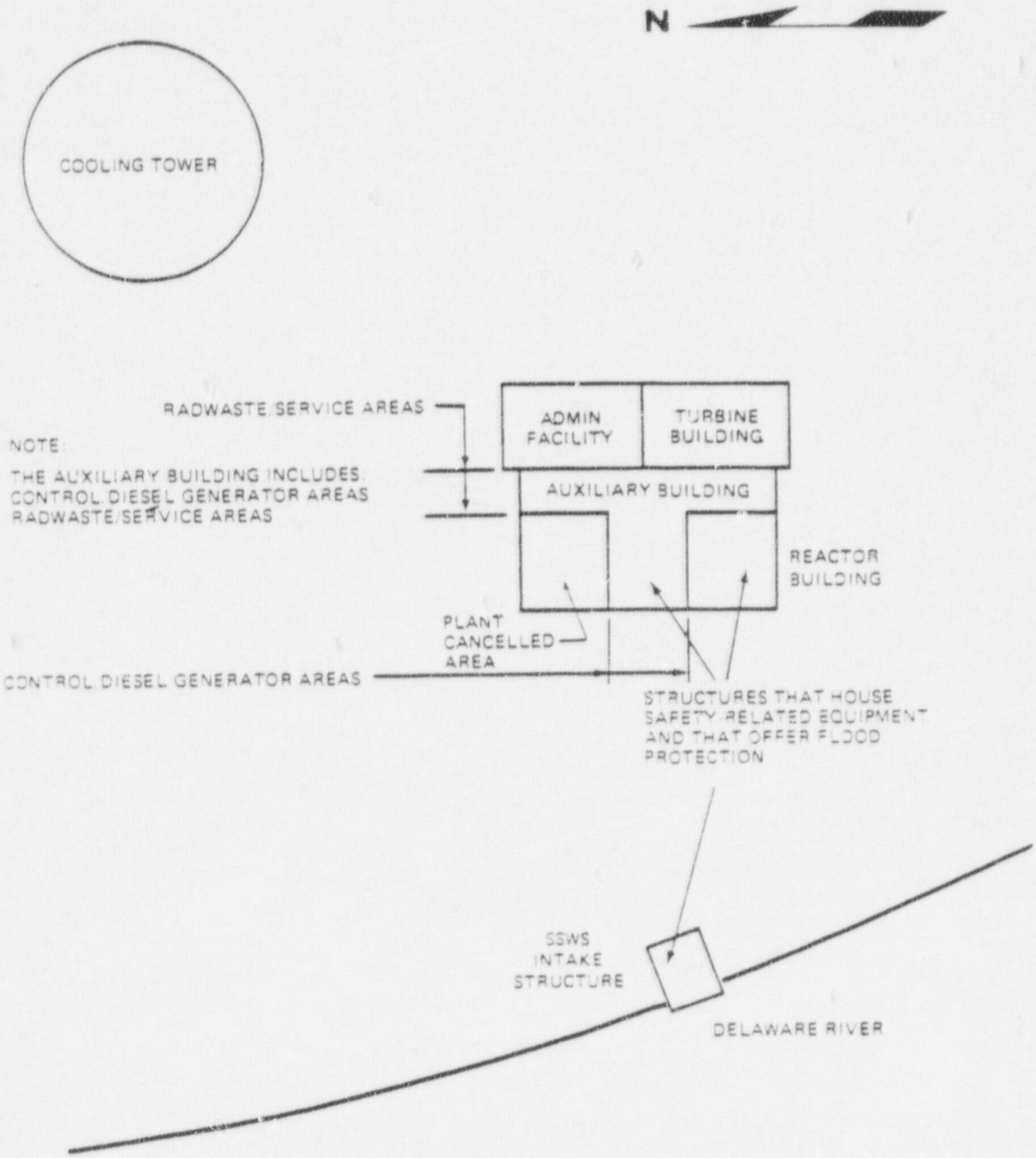

Figure 3-4
General Plot Plan - Hope Creek Nuclear Generating Station

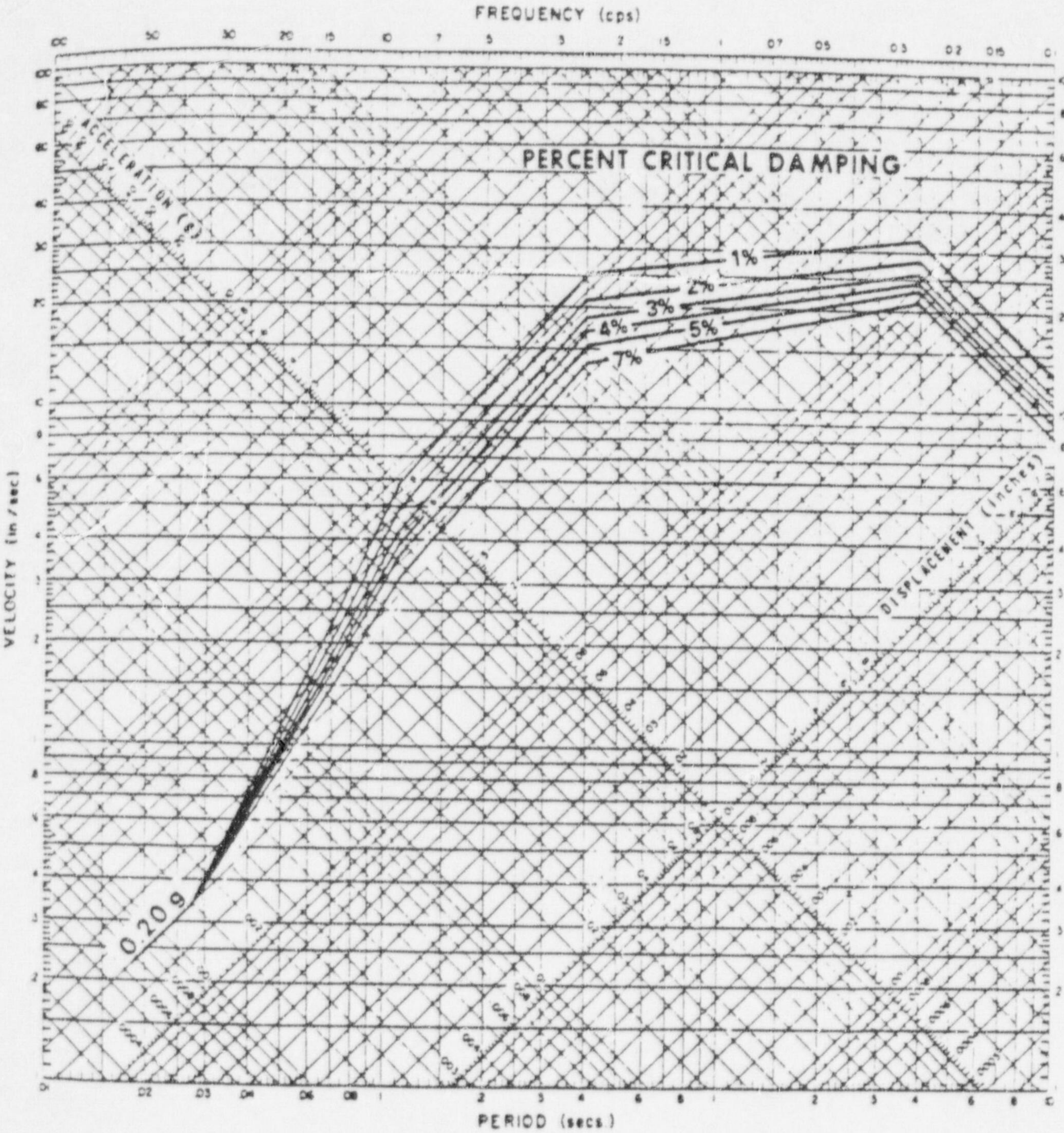

Figure 3-5
SSE Horizontal Ground Response Spectra (0.2g)

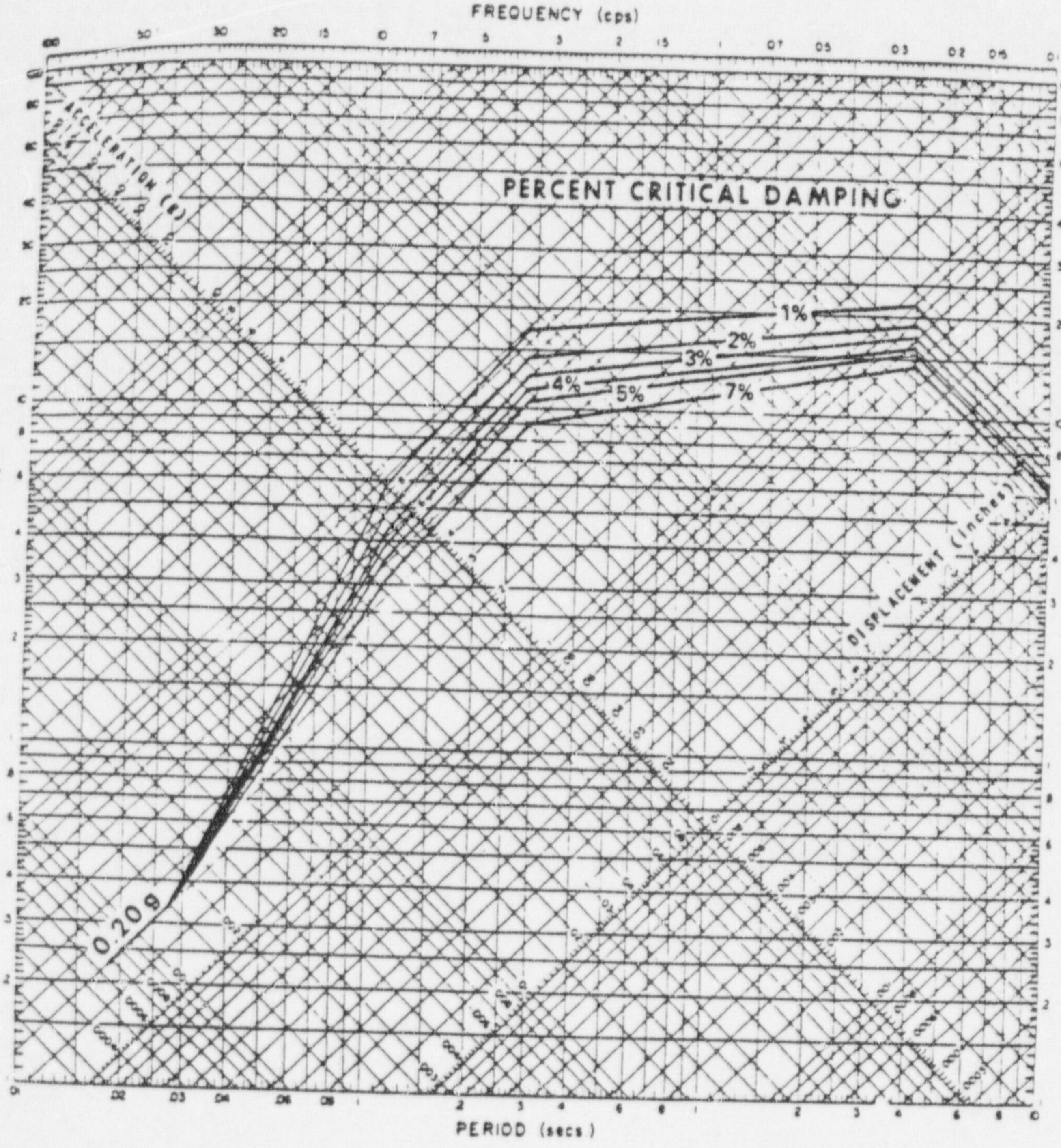

Figure 3-6
SSE Vertical Ground Response Spectra (0.2g)

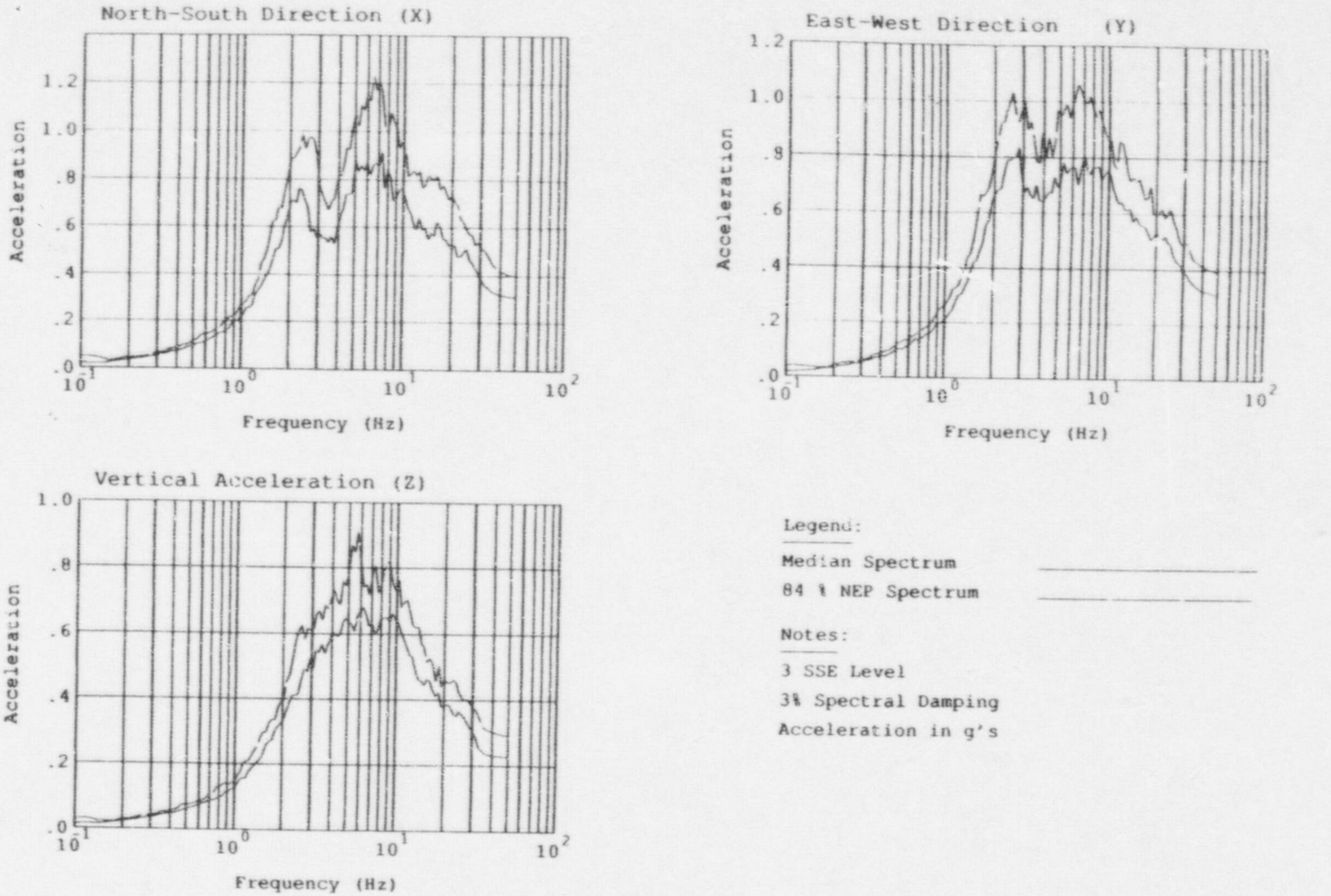

Figure 3-7
Median and 84% NEP Response Spectra for 3* SSE
Reactor Building Operating Floor, Elevation 102 ft.

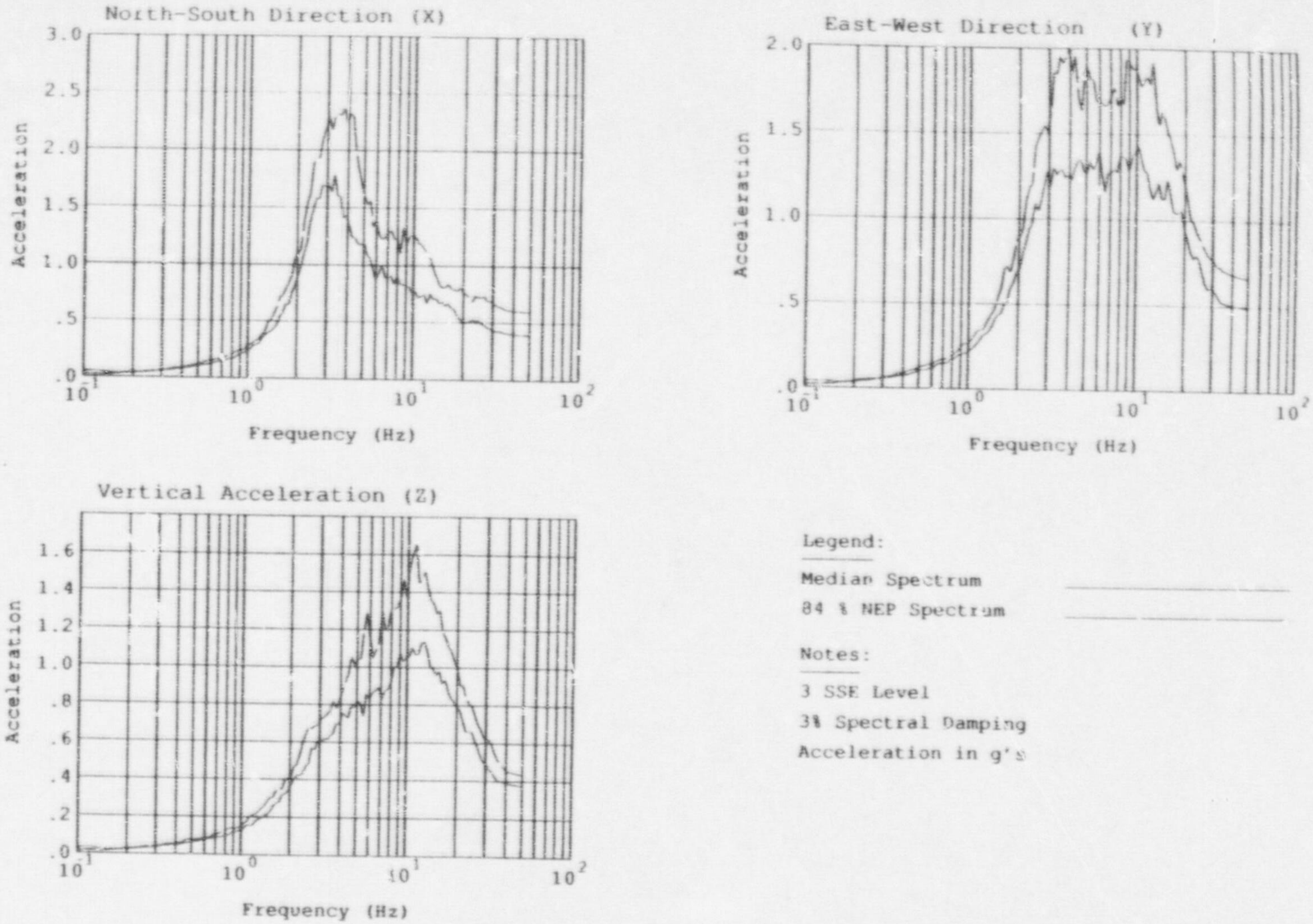

Figure 3-8
Median and 84% NEP Response Spectra for 3* SSE
Auxiliary Building, Floor Elevation 130 ft.

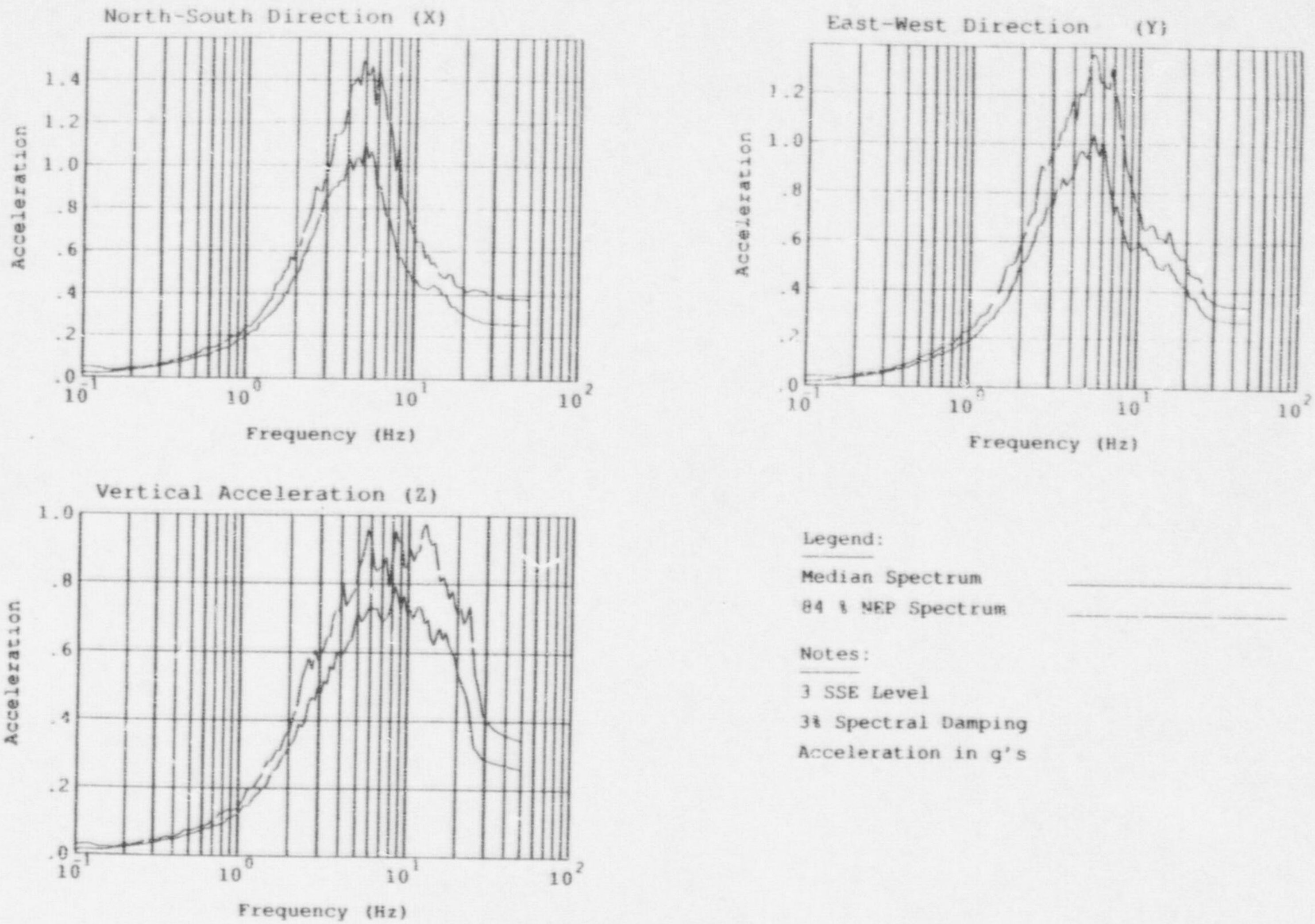

Figure 3-9
Median and 84% NEP Response Spectra for 3* SSE
Service Water Intake Structure, Elevation 122 ft.

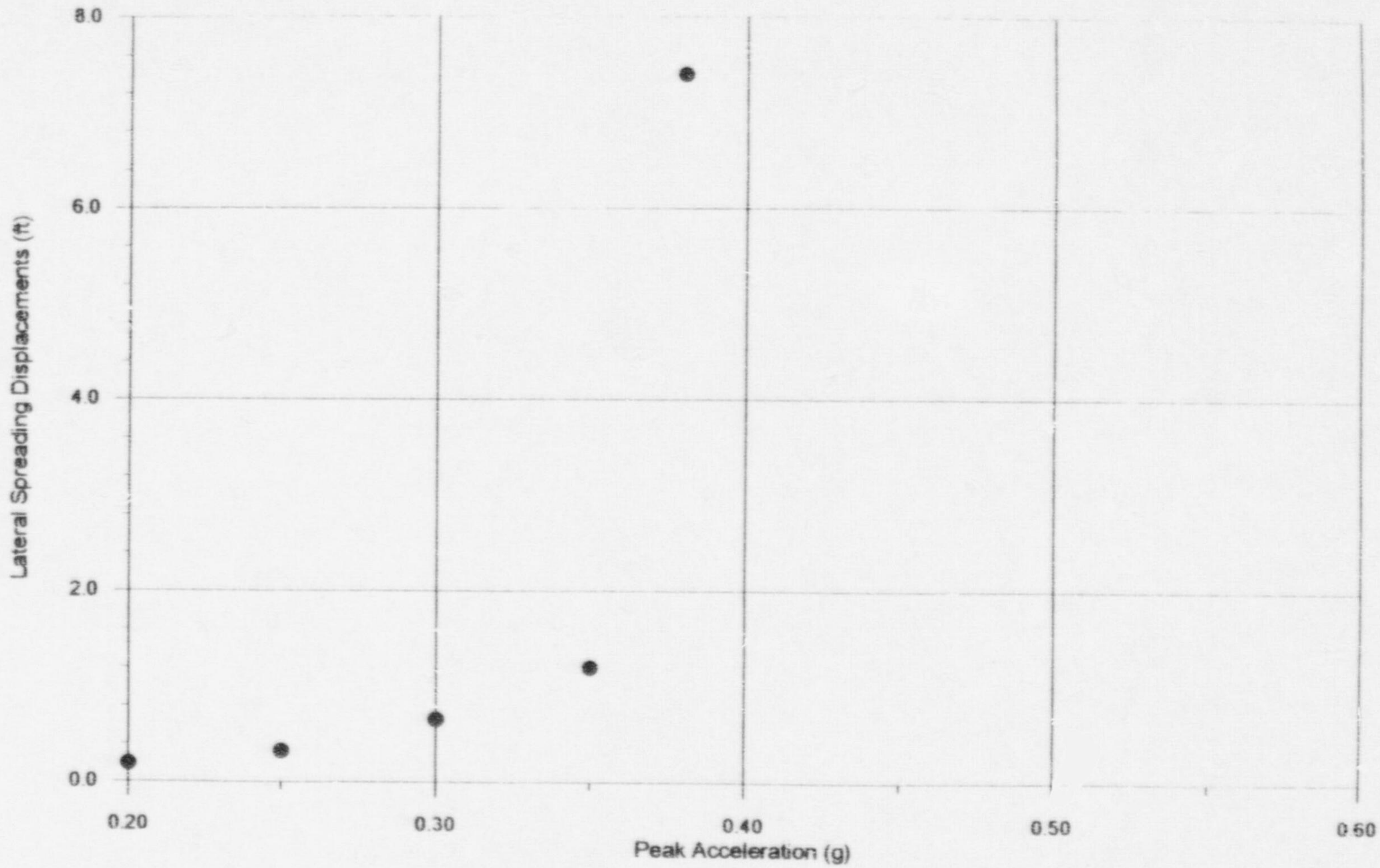

Figure 3-10
Lateral Spreading Displacements as a Function of Peak Accelerations
Salem and Hope Creek Stations

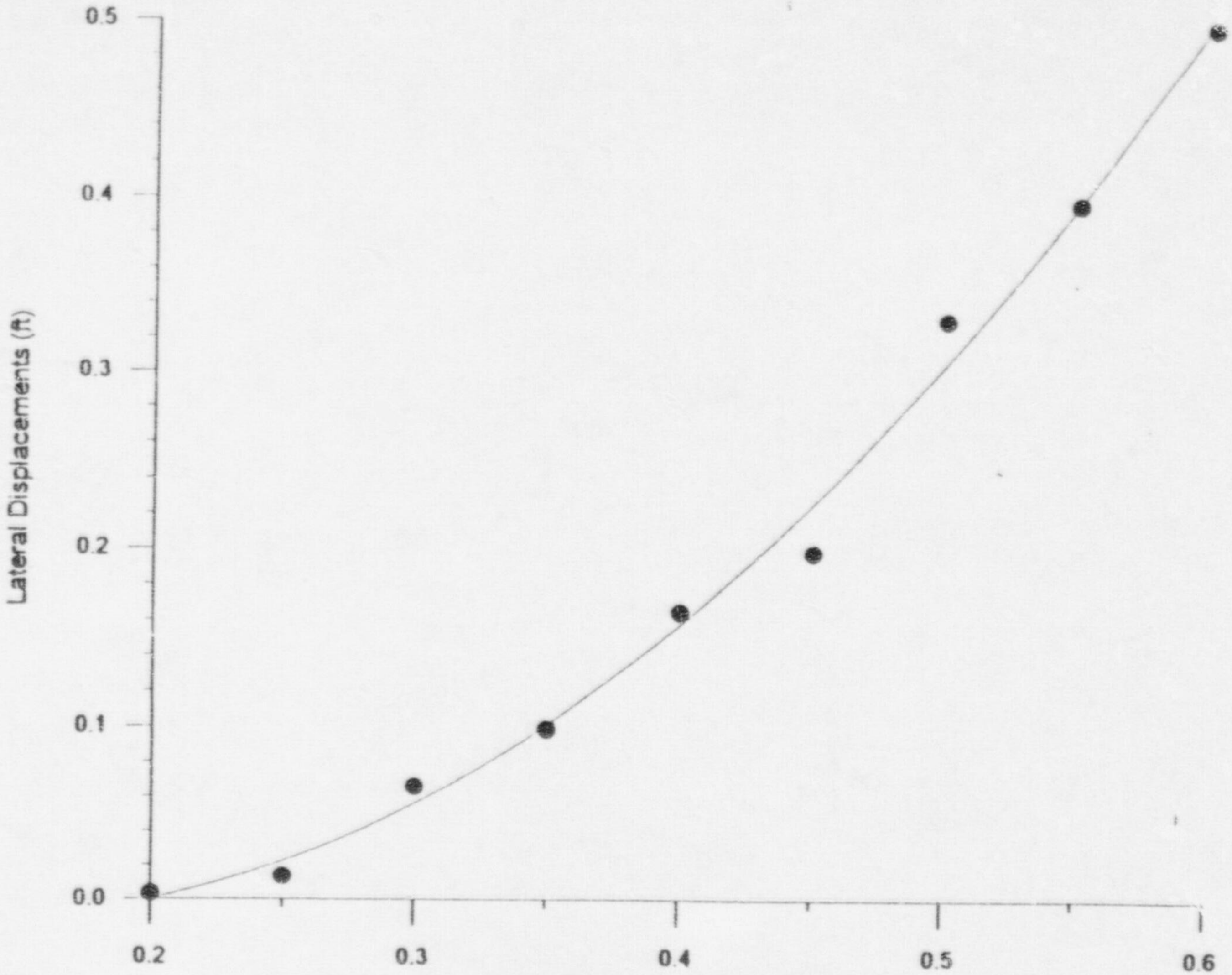

Figure 3-11
Earthquake-Induced Lateral Displacements of Compacted Fill

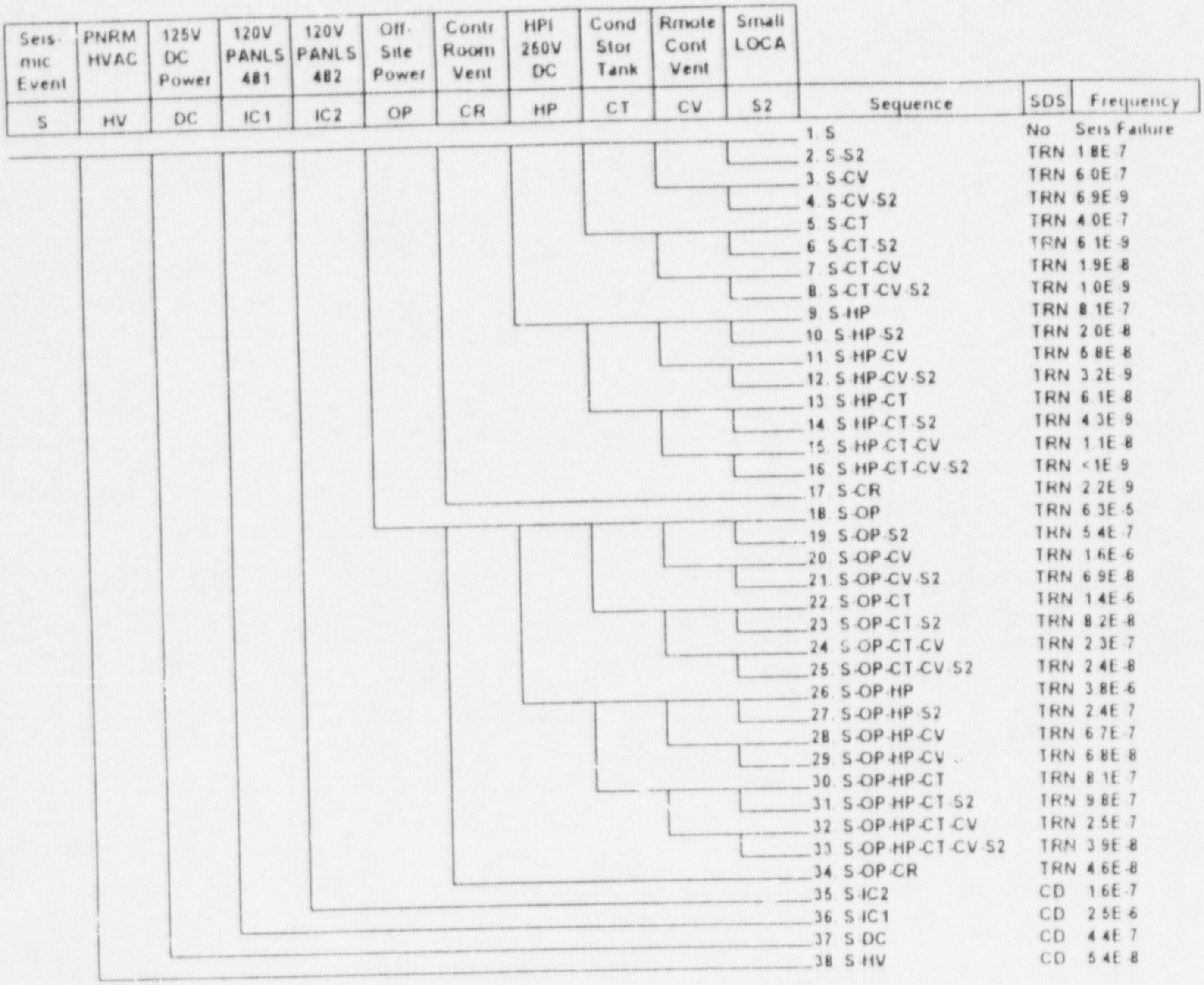

Figure 3-12
Hope Creek Seismic Event Tree

Top Event Success/Failure Description

S Success implies that there is no seismic event greater than 0.05g. Failure indicates a seismic event greater than 0.05g. Since the HCLPF of offsite power is 0.1g, and the design basis of the HCGS is 0.2g, any seismic event of lower magnitude than 0.05g would very likely have offsite power and all safety systems available.

HV Success implies that 1E Panel Room ventilation is available. A seismic-induced failure of the panel room (diesel area) fans 1A-VH408 and 1B-VH408 would result in a failure of ventilation to the Class 1E Panel Room. Consistent with the HCGS IPE, a procedure for loss of ventilation in this room is credited to provide alternate forced ventilation. Event HV represents a failure of the fans AND a failure to provide alternate ventilation. A total loss of ventilation to the Class 1E Panel Room is conservatively assumed to result in core damage.

DC Success implies that 1E 125V DC power is available. A failure of power to DC Panels 1A/B/C/D-D-417 would mean a loss of DC control power to the safety-related systems. While manual control would be possible, it would be difficult to credit operation without 125V DC power, and core damage is conservatively assumed if top event DC fails.

IC1 Success implies that 120V AC instrumentation power is available from 1E panels 1A/B/C/DJ481. If all four of these panels fail, core damage is conservatively assumed. The 1A/B/C/DJ481 panels distribute instrumentation power to diesel generator control panels; various SACS, RHR, Core Spray, HPCI and RCIC valves and/or control panels; class 1E 4160V AC switchgear; class 1E 125V DC and 1E 250V DC battery chargers and switchgear; Remote Shutdown Panel instrumentation; and various other 1E loads. In reality, if no other equipment was affected, a loss of the 1A/B/C/DJ481 panels would not necessarily result in core damage. Innovative operator actions allow manual control of the plant, but these actions are not credited in this baseline assessment. In Paragraph 3.3, a sensitivity analysis shows the effects of crediting operator action after a loss of the 1A/B/C/DJ481 panels.

Figure 3-12
Hope Creek Seismic Event Tree (Continued)

Top Event Success/Failure Description

IC2 Success implies that 120V AC instrumentation power is available from 1E panels 1A/B/C/DJ482. If all four of these panels fail, operator action can still prevent core damage. The 1A/B/C/DJ482 panels distribute 120V 1E AC power to various 1E logic cabinets. The failure of these logic cabinets causes a substantial loss of automatic actuation of 1E equipment, including diesel generator load sequencing and automatic Primary Containment Isolation System signals. However, manual operation of this equipment and manual diesel generator loading is still possible (e.g., at the Remote Shutdown Panel), and procedural guidance is available. The remote shutdown operator action described in Paragraph 3.2 is conservatively used to represent this recovery action. This is conservative since manual actions can be performed directly from the control room.

OP Success implies that offsite power remains available. The frequency of a seismic failure of offsite power is dominated by the failure of the ceramic insulator columns in either the switchyard or the incoming transformers.

CR Success implies that control room ventilation is available, or that remote shutdown is successful. Event CR represents the failure of the 1GKHD-9588AA/AB/BA/BB dampers AND the 1GKHD-9594A damper AND the failure to utilize the remote shutdown panel after a control room evacuation. Control Room ventilation is normally supplied by the 100% redundant Control Room Supply (CRS) system, which circulates 3,000 cfm of outside air and 15,500 cfm of recirculated air. A failure of the outside air intake dampers (1GKHD-9588AA/AB/BA/BB) would isolate the 3,000 cfm of outside air. The remaining 15,500 cfm of recirculated air would probably be sufficient to maintain control room habitability for a long period, but for this analysis, the failure of the outside air supply is conservatively assumed to fail all normal CRS. Given a CRS failure, the Control Room Emergency Filtration (CREF) system can be used in conjunction with the CRS to maintain control room habitability.

Figure 3-12
Hope Creek Seismic Event Tree (Continued)

Top Event Success/Failure Description

The CREF has two redundant trains, of which one has a damper that appears in Table 5. Damper 1GKHD-9594A is the inlet recirculation air damper for CREF fan 1A-VH400. If a failure of this damper were to occur in conjunction with a failure of the 9588 dampers, core damage could still be averted if the "B" train of CREF successfully operates, or if shutdown outside the control room is successful.

HP Success implies that 250V DC power to HPCI/RCIC is available. The failure of 250V DC MCCs 10-D-251 and 10-D-261, or the failure of the 250V DC 1E buses 10-D-450 and 10-D-460 would cause the loss of the HPCI and RCIC systems. Depressurization and low pressure injection would then be required to prevent core damage.

CT Success implies that the Condensate Storage Tank (CST) is available. Failure implies that the CST fails. The CST is not required for ECCS injection to the reactor. This is consistent with the IPE assumption that HPCI and RCIC, whose suction are normally aligned to the CST, would need to have their suction transferred to the torus during the course of an accident (when CST volume is exhausted). Therefore, failure of the CST only affects the ability to use CST water for long term makeup (top event Uv in the IPE event trees).

CV Success implies that remote operation of containment venting is possible. Failure of the 120V AC fuse panels 1Y-F-401/402/403/404 results in a failure of remote containment venting. Local operation of containment venting was credited in the IPE, but due to the uncertain nature of the state of the plant after a large seismic event, local operation of containment venting is not credited here. While this assumption is conservative, it has a negligible impact on the IPEEE results presented in Section 5.0.

S2 Success implies that no seismic-induced LOCA occurs. Failure implies that a small LOCA (pipe breaks with an area of less than 0.005 ft^2 or steam line breaks with an area of less than 0.1 ft^2) has occurred due to the seismic event. This could be caused, for example, by multiple failures in the small instrument lines connected to the reactor coolant system.

Figure 3-12
Hope Creek Seismic Event Tree (Continued)

Top Event Success/Failure Description

The column labeled "Sequence" in Figure 1 provides the usual sequence notation for the failures in the accident sequence, which is used as the seismic damage state number. The column labeled "SDS" tells whether the sequence leads directly to core damage (CD) or transfers (TRN) to a PSA event tree. The column "Frequency" presents the frequency of the seismic damage state. For sequences that are "CD", the frequency represents the core damage frequency. For sequences that are "TRN", the frequency represents the frequency that is transferred to an PRA event tree for further evaluation.

Figure 3-12
Hope Creek Seismic Event Tree (Continued)

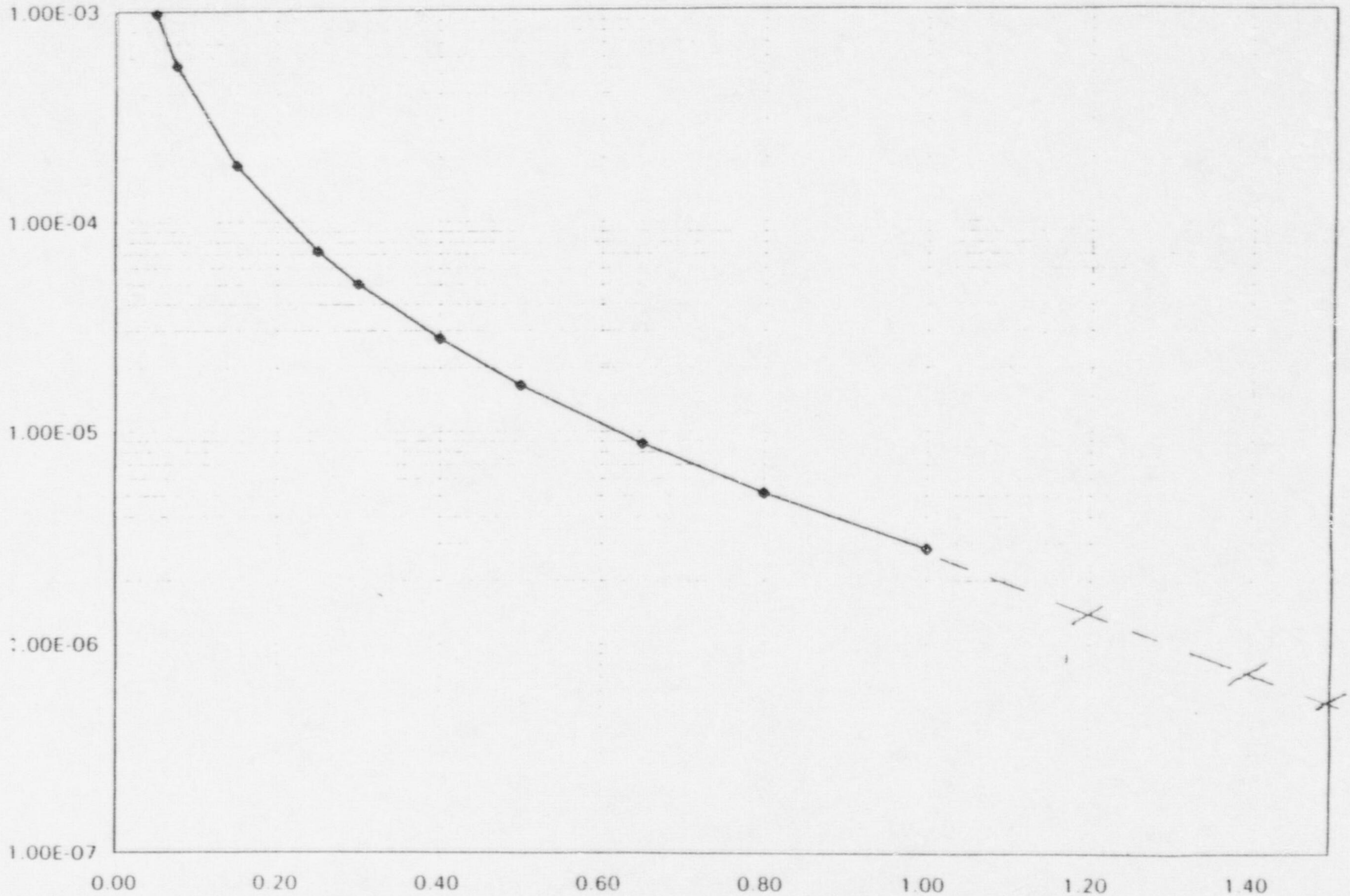

Figure 3-13
Extrapolation Of The LLNL Seismic Hazard Curve To 1.5g

SECTION 4

INTERNAL FIRES ANALYSIS

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

SECTION 4

INTERNAL FIRE ANALYSIS

4.0 METHODOLOGY SELECTION

4.0.1 OBJECTIVES

The objectives of the HCGS fire IPEEE may be stated as follows:

- develop and understand the most likely severe accident sequences that could occur from fires
- develop an appreciation of and insights into potential core damage sequences initiated by fires
- gain an understanding of the overall likelihood of core damage owing to fires at the HCGS, and
- suggest areas, if any, in which it would be reasonable to improve the plant's resilience to fire induced core damage sequences.

4.0.2 OVERVIEW OF TECHNICAL APPROACH

The technical basis of the HCGS fire IPEEE was a new fire probabilistic safety assessment (PSA) performed in a manner consistent with the guidance in NUREG/CR-2300 [NRC, 1983a] and NUREG/CR-4840 [NRC, 1989g]. The approach taken for the PSA was to perform a scenario-by-scenario analysis of unscreened compartments accounting for the relative location of ignition sources and targets. Fire damage calculations were performed to determine the extent of potential damage from each postulated fire source. Openings in walls as well as open active fire dampers were included in the assessment of the extent of fire damage. The PSA methodology is described in more detail in Paragraph 4.0.3 and subsequent paragraphs.

The PSA was preceded by 1) a fire compartment interaction analysis (FCIA) per FIVE guidance [EPRI, 1993b] and 2) a quantitative screening analysis also performed in a manner consistent with FIVE guidance.

The HCGS is composed of hundreds of identifiable rooms. Each room has an associated number. Many of the areas identified as rooms do not qualify as fire compartments using the definitions in Paragraph 5.3.6 of EPRI, 1993b. The FCIA was performed to establish the combinations of rooms that have boundaries which meet the FIVE criteria. Therefore, many fire compartments analyzed in this study consist of multiple rooms and are so identified in this document. The result of the FCIA was a total of 209 fire compartments which met the FIVE criteria. These compartments covered the turbine building, reactor building, control/diesel building, radwaste building, service water intake structure, and yard. The FCIA methodology is described in more detail in Paragraph 4.1.1.

A quantitative screening analysis was performed on each of the 209 compartments as well as the transformer array (station service and main) in the yard. A qualitative screening analysis was not performed for the HCGS IPEEE. That is, no compartments were eliminated from quantitative consideration owing to qualitative factors. The objective of the quantitative screening analysis was to determine which fire compartments might have the potential of being significant to the risk associated with fire induced core damage sequences. The quantitative screening methodology is described in more detail in Paragraphs 4.1.2 and 4.1.3.

This document is fully responsive to the standard outline and information requested in NUREG-1407 [NRC, 1991b - Section C]. Table 4.1 maps the information request of Paragraph C.3 of NUREG-1407 to Paragraphs of this report.

In addition to the requested items, a special feature of this submittal is an analysis of high hazard (which are not necessarily high risk) rooms at the HCGS. These are rooms which contain a somewhat larger amount of combustible materials (other than cables). The analysis is provided in Paragraph 4.1.4.

4.0.3 KEY ASSUMPTIONS

Detailed assumptions are found with the associated detailed discussion of methods. This section provides an overview of the key general assumptions that form the basis of this analysis:

1. Room inventory is as deduced by a review of the UFSAR, MMIS lists, pre-fire plans and as witnessed during the walkdown.
2. Fire barriers for compartments defined for this analysis were defined in accordance with the EPRI, 1993b, Paragraph 5.3.6.
3. The occurrence of damage of cables owing to fires is as determined by the fire damage calculations described in Paragraphs 4.3 and 4.4. These calculations use a modification of the FIVE Fire Screening Methodology [EPRI, 1993b- Attachment 10.47]. Similarly, the occurrence of suppression

before damage to a target is determined by the fire damage calculations described in Paragraph 4.5 which include detection and extinguishment delay times.

4. If two pumps or compressors (e.g., chillers) are located within close proximity, a fire in one is assumed to disable both.
5. Fire damage calculations were used to assess the spread of damage, owing to a hot gas layer, through openings in walls. In these calculations all walls in the source room, below the level of the opening, were assumed to vanish.
6. The fire source (i.e., pump, cabinet, transformer, compressor, etc.) is completely disabled by the fire.
7. All fire damage calculations assume cables are unprotected even if they are in conduit, protected by a cable tray bottom, or protected by an enclosed cable tray. Furthermore, if any cable in a stack of trays was calculated to be damaged, all of the cables in the stack were assumed to be damaged. Neither shielding nor fire growth from tray to tray were considered in the fire damage calculations.
8. Lack of knowledge about the termination points (i.e., functions) of specific cables in a compartment was treated as causing failure of the entire channel in which the cable belongs, if one cable was calculated as damaged.
9. Conditional core damage probabilities (CCDPs) are calculated using the HCGS PSA model [PSE&G, 1994b and NUS, 1995a] as modified by forcing failure of equipment affected by damage caused by each fire scenario. In these calculations feedwater, condensate, and control rod drive systems are assumed to be unavailable.
10. CCDPs are calculated using the pre-initiator operator actions modeled in the PSA model with human error probabilities (HEPs) unmodified from the internal event values. Only two recovery actions are used in the Fire PSA as will be described in Paragraph 4.6.5. The HEPs of all other recovery actions, used in the IPE, were set to unity for this fire assessment. Normal, post initiator operator actions, such as inhibit of ADS, also used HEP values unmodified from the internal event values unless control room abandonment resulted from the scenario. For scenarios involving control room abandonment, the only human action considered was failure to continue operating the plant using the alternate shutdown procedure with the remote shutdown panel and local manual controls.

11. CCDPs are calculated assuming that the least severe initiating event is a transient with MSIV closure. More severe initiating events which were found to be potentially caused by fire induced scenarios are LOOP, loss of SWS/SACS, loss of HVAC, and ISORV.
12. Operators will initiate a plant trip regardless of the level of damage for all fires in the control room. Such a plant trip is assumed to be accompanied by MSIV closure or more severe initiating event.
13. Check valves, manual valves and valve bodies are unaffected by fires.
14. Fires at the valve operator fail the valve in its fail safe position. For example, MOVs fail as is.

4.0.4 FIRE PSA AND QUANTITATIVE SCREENING METHODOLOGY OVERVIEW

The quantitative screening analysis resulted in screening out all but 38 of the 209 compartments defined by the FCIA. Each of the 38 unscreened compartments was subjected to a detailed scenario-by-scenario probabilistic analysis. A fire scenario is defined as a unique source, fire intensity, target, and initiating event combination. Fire damage calculations and fire damage time versus suppression time calculations supported the probabilistic analysis. These are described in more detail in Paragraphs 4.3, 4.4 and 4.5. Equations 1 and 2, below summarize the concept of the fire PSA employed in this analysis.

$$F_n = \sum_{j}^{sources} \sum_{k}^{intensities} \sum_{l}^{targets} f(S_{j,k}) P(T_l / S_{j,k}) P(E_{j,k} < T_l) \sum_{m}^{initiating\,event} P(I_m / T_l) P(CD / I_m, T_l)$$

(Equation 1)

and

$$F_{total} = \sum_{n}^{38\,compartments} F_n$$

(Equation 2)

where,

F_{total} = the total core damage frequency associated with analyzed scenarios in the 38 unscreened compartments.

F_n = the core damage frequency associated with scenarios of each of the 38 unscreened compartments.

$f(S_{j,k})$ = fire ignition frequency of the jth ignition source having intensity (or fire size) k

$P(T_l / S_{j,k})$ = probability of damaging target l (T_l) with source j of intensity k without consideration of suppression. This quantity is often 0 or 1. That is, either deterministic fire damage calculations show that it is possible to damage the target or they show that the target can not be damaged. However, when applicable, this quantity also represents the fraction of compartment area (i.e., a geometric factor) that can be affected by a postulated transient combustible fire.

$P(E_{j,k} < T_l)$ = probability of not extinguishing a fire from source $S_{j,k}$ before damage to T_l. Determination of this quantity involves a calculation to obtain the relative time of occurrence of damage and extinguishment.

$P(I_m / T_l)$ = probability of occurrence of initiating event, I_m, given damage to T_l.

$P(CD / I_m, T_l)$ = probability of core damage given initiating event, I_m, and damage to T_l.

Details of the implementation of these equations are provided in Paragraphs 4.3, 4.4, 4.5, and 4.6.

The quantitative screening analysis followed the philosophy found in FIVE, (EPRI, 1993b), but did not include suppression. It can be described as a simplification of the above equations. Each of the 209 compartment were subject to Equation 1 with the following simplifications:

- Each source has a fire intensity that will completely engulf the compartment failing all components and cables. That is k = 1 and l = 1. Each source has the same target, which is the entire room.
- Therefore, $f(S_{j,k}) = f(S_j)$ and $P(T_l / S_{j,k}) = 1$ for all l, j, and k.
- Fire suppression is not considered. Therefore, $P(E_{j,k} < T_l) = 1$ for all l, j, and k.
- The entire compartment is represented by the worst initiating event. Therefore, there is only a single I_m analyzed in the screening analysis, and
- $P(I_m / T_l) = 1$ and $P(CD / I_m, T_l) = P(CD)$

The result of these simplifications to Equation 1 yields Equation 3 which conceptually represents the screening analysis for each compartment.

$$F_{n,\,screening} = \sum_{j}^{sources} f(S_j) P_n(CD) \quad \text{for compartment n.} \qquad \text{\{Equation 3\}}$$

Equation 2 is not used in the quantitative screening analysis as the screening core damage frequencies are not summed. Both the screening analysis and the PSA used fire sources, found in the as-built plant, with fire initiation frequencies calculated by the FIVE method (EPRI, 1993b - Attachment 10.3). More detail in the implementation of the quantitative screening assessment is provided in Paragraph 4.1.2.

4.0.5 DISCUSSION OF STATUS OF APPENDIX R

The HCGS is not an Appendix R plant. However, a complete fire hazard analysis was performed and documented in the HCGS UFSAR [PSE&G, 1995f - Table 9A-6 to Table 9A-102]. Furthermore, a comparison of the Hope Creek design and fire protection features was made with the Appendix R requirements and documented in the HCGS UFSAR [PSE&G, 1995f - Appendix 9A]. Finally, the HCGS complies with the requirements of Branch Technical Position CMEB 9.5-1 [PSE&G, 1995f - Section 9].

4.1 FIRE HAZARD ANALYSIS

4.1.1 FIRE COMPARTMENT INTERACTION ANALYSIS

The HCGS is a highly compartmentalized unit, when compared to other BWRs; equipment is located in hundreds of separate rooms. The difference is particularly evident in the reactor building, where all elevations in many other BWRs are open to the refueling deck elevation. In contrast, the four lowest elevations (54' through 132') of the HCGS reactor building are not open to the refueling deck.

Rooms in the HCGS have designation numbers which follow the conventions shown in Table 4.2. For example, room 3501 designates room 01 on floor 5 of the radwaste building (3).

The objective of the FCIA was to define compartments suitable for an efficient and accurate fire PSA of the HCGS, accounting for the potential of the spread of damage owing to fire among rooms. This analysis used the FIVE methodology for defining compartment boundaries. The FCIA was performed to establish the combinations of rooms that have boundaries which meet the FIVE criteria [EPRI, 1993b - Paragraph 5.3.6]. This reference lists six criteria suitable for defining a compartment boundary. In this study, if all boundaries of a location, often comprised of more than one room, met at least one of the six criteria, then the location was considered a fire compartment for purposes of the HCGS fire IPEEE. Fire compartment boundaries were developed for the buildings shown in Table 4.2. No rooms were eliminated or screened out by the fire compartment interaction analysis. A conservative assumption in this analysis and in the entire PSA is that each compartment was assumed to cause a plant trip.

A Fire Compartment Interaction Analysis Data Sheet template was created, along the lines of the FIVE methodology [EPRI, 1993b - Table 2 - Attachment 10.1]. An exemplar FCIA data sheet is provided as Table 4.3. One such spreadsheet was completed for each compartment and required the following information:

- a brief description of the exposing room and its location; In the example sheet, the identified compartment (5307) happens to correspond to a single room.
- a summary of the contents of the compartment along with the sources of this information .
- whether the equipment in the compartment could cause a plant trip.
- whether it contains Appendix R safe shutdown equipment. The references to DWG refer to PSE&G drawing numbers. The references to Tables 9A-x refers to the specific fire hazard analysis summary sheet found in [PSE&G, 1995f].
- A list of surrounding rooms under the title Exposed Compartments. Each row designates whether fire can spread to the room (PFS), whether the exposed compartment can be a plant trip initiator (PTI), and whether the exposed compartment contains Appendix R safe shutdown equipment (SSE).

The information basis is noted under Comments/References as well as elsewhere in the spreadsheet. The following information served as the basis for defining compartments:

- "Hope Creek Generating Station Updated Final Safety Analysis Report", [PSE&G, 1995f - Chapter 9.5 and Appendix 9A].
- "Hope Creek Generating Station Probabilistic Risk Assessment", [PSE&G, 1994b].
- "Hope Creek Generating Station Individual Plant Examination", [PSE&G, 1994a]
- PSE&G Drawings M-5001 to M-5013.
- PSE&G Drawings M-5101 to M-5124.
- PSE&G Drawings E-0000-0 to E-0006-1.
- PSE&G Drawings A-0531-0 to A-0546-0.
- Hope Creek Pre-Fire Plan Documents FRH-II-314 through FRH-III-719.
- Fire Walkdown [PSE&G, 1997b].

4.1.2 FIRE IGNITION FREQUENCY OF EACH COMPARTMENT

4.1.2.1 Method and Assumptions for Quantitative Screening Analysis

This paragraph describes the development of the fire ignition frequency, $f(S_i)$, of Equation 3. A fire ignition frequency, using the method of FIVE [EPRI, 1993b -

Attachment 10.3], was developed for each of the 209 fire compartments. This method was implemented using a Fire Compartment Ignition Source Data Sheet (ISDS) for each compartment. An exemplar sheet is shown in Table 4.4. These data sheets were completed using the fire data base [EPRI, 1993b - Attachment 10.3 - Tables 1.1 and 1.2]. A review of the HCGS plant specific fire data [PSE&G, 1997c - Appendix] demonstrated that, on a compartment-by-compartment basis, the plant specific fire data is too sparse to be of practical use in modifying the generic database of FIVE. However, the database did reflect a 4kV station service transformer fire in the yard. The analysis of the transformer array in the yard is provided in Table 4.11 herein.

Because the FIVE fire data base is divided into relatively large categories (e.g., Reactor Building, Diesel Generator Room, Switchgear Room, Cable Spreading Room, Intake Structure), the fire IPEEE analyst has the responsibility of assigning a category to each compartment. Table 4.5 provides the assignments used in this analysis.

Use of the Ignition Source Data Sheets requires a weighting factor (WF_L) which adjusts the generic location frequencies [EPRI, 1993b - Attachment 10.3, - Table 1.2] for specific plant numbers of rooms of each category. The method used to obtain these weighting factors is detailed [EPRI, 1993b - Attachment 10.3 - Table 1.1]. Table 4.6 provides the worksheet that reflects the development of the HCGS specific weighting factors.

The ISDS method also requires an estimate of the transient fire frequency in each compartment. This was done in a conservative implementation of Note D, Reference Table 1.2 of FIVE. The note assigns multiplicative factors to various kinds of transient combustibles. The maximum multiplicative factor if all transient combustibles are present is thirteen. The HCGS fire protection program [PSE&G, 1996f] prohibits cigarette smoking and requires all hot work to be accompanied by a fire watch. By eliminating hot pipe and cigarette transient ignition sources, this analysis used a factor of 10 (of 13) for each compartment. Transient combustibles were not a significant contributor to the fire ignition frequency of a compartment if fixed ignition sources were present in that compartment.

The completion of these data sheets requires a detailed knowledge of the inventory of equipment in each compartment. PSE&G has developed and maintains a Managed Maintenance Information System (MMIS) for the HCGS. This system was used to obtain equipment inventory lists of each compartment [PSE&G, 1996p - Attachment 1]. In compiling the information used in the ISDS analysis, components were not counted if 1) they were listed as removed, spared, or retagged, or 2) they were not located in one of the areas of Reference Table 1.2, Attachment 10.3, EPRI, 1993b. This information was supplemented by the walkdowns [e.g., PSE&G, 1996s and 1997b], Hope Creek Pre-Fire Plan Documents FRH-II-314 through FRH-III-719, and the "Hope Creek Generating Station Updated Final Safety Analysis Report" [PSE&G, 1995f - Chapter 9.5 and Appendix 9A].

Column 5 of Table 4.8 provides the fire ignition frequency calculated for each compartment. These fire ignition frequencies were used for the quantitative screening analysis described in Paragraph 4.1.3 and are called screening fire ignition frequencies. Table 4.7 is a summation of the compartment screening fire ignition frequency results for each building.

4.1.2.2 Method for Development of Fire Ignition Frequencies Used in the PSA of Unscreened Fire Compartments

This paragraph describes the development of the fire ignition frequency of Equation 1, $f(S_{j,k})$ = fire ignition frequency of the jth ignition source having intensity (or fire size) k", used for the scenarios developed for compartments which were not screened out by the quantitative screening analysis represented by Equation 3.

While the screening fire frequency was developed by summing the frequencies of all fire ignition sources in a compartment, the fire PSA was performed on a source-by-source basis. That is, the fire ignition frequency of each scenario was developed separately for each identified source in a compartment. This was easily derived from the ISDS analysis of each compartment, described above, by simply using the frequency of the individual sources. Using Table 4.4 for an RHR pump room as an example, each pump (the jockey pump and RHR pump) are individual fire ignition sources and a fire scenario was developed for each. The fire ignition frequency of each scenario is 2.85×10^{-4} /yr., which is one half of the value for two pumps shown in column F_{if} of Table 4.4.

The ISDS analysis included items called plant wide ignition sources. This was an attempt to include ignition sources that might be found in many locations of the plant, without having an accurate room by room inventory. The walkdown, coupled with the reviewed documentation of the HCGS, provided an accurate accounting of the 'as-built' ignition sources in each room. If an ignition source was not present in a room, it was not included in the PSA. Typically, for the unscreened compartments, the sum of the scenario fire ignition frequencies did not (and should not) equal the ISDS derived frequency of a compartment. For example, because there is no hydrogen in the RHR pump room, a hydrogen fire scenario was not developed.

The PSA included transient combustible sources as individual scenarios. The method to obtain the frequency of transient combustible fire ignition frequencies was derived from FIVE [EPRI, 1993b -Paragraph 6.3.7.2]. The derivation of the transient fire ignition frequencies is described below.

$$F_{tc} = F_{comp} * P_{tc}$$

$$P_{tc} = P_{fst} * u * p * w$$

where,

F_{tc} = transient combustible fire frequency (/yr.)

F_{comp} = total compartment fire frequency (/yr.) from ISDS analysis of a compartment. This is the same as $f(S_j)$ of Equation 3.

P_{tc} = conditional probability that fire is caused by transient combustibles.

P_{fst} = probability that suppression fails to prevent damage to target. Based on the fire growth and damage calculations, no credit was taken for suppression of transient combustible fires. Therefore, P_{fst} = 1.

u = probability that transient ignition source is located within an area that will cause damage to the target. This is a room specific quantity which was numerically equal to the fraction of floor area over which fires have been analyzed to affect cables. It was calculated from the following ratio:

$$\frac{\text{Area of compartment formed by the locus of points at which source } S_{j,k} \text{ can damage target } T_i}{\text{Total Area of the Compartment}}$$

Relative to Equation 1, this ratio is also $P(T_i / S_{j,k})$ for transient combustibles. The discussion of Paragraph 4.3 explains the method used to develop the numerator of this ratio.

p = probability that the combustible is exposed (e.g., container is open). This factor was not used in the HCGS analysis because this factor is already accounted for, at least partially, in F_{total} and w.

w = the probability that inspection has not found and removed unauthorized transient combustibles before ignition

$$w = (x/2)^* \ln(1/x) \text{ where } x = F_{ccl}/F_w$$

F_{ccl} = incidents (per year) of finding an unauthorized amount of critical combustibles. This analysis used 1 as a conservative estimate because the walkdown discovered no such incidents.

F_w = the frequency (number of times per year) of inspections for transient combustibles. During normal operation, the HCGS has good control procedures and practices with respect to transient combustibles. Daily inspections of fire doors and for transient combustible are performed at the HCGS [PSEG, 1996q]. In addition operators walk down the plant each shift and report anything unusual. Furthermore, security guards note anything that is not normal. This analysis used a daily inspection interval. Therefore, F_w = 365, in this study.

With the above inputs,

$$x = 0.003$$

$$w = 0.008$$

Therefore,

$$P_{tc} = 1 * u * 1 * w = u * 0.008$$

$$F_{tc} = F_{total} * P_{tc}$$

$$F_{tc} = F_{total} * w * u = F_{total} * 0.008 * u$$

F_{total} was taken directly from the ISDS analysis and u was developed from the fire damage studies described in Paragraph 4.3.

Table 4.33 provides the scenario by scenario fire ignition frequencies of the unscreened compartments.

4.1.3 QUANTITATIVE SCREENING ANALYSIS TO IDENTIFY POTENTIALLY RISK SIGNIFICANT FIRE COMPARTMENTS

4.1.3.1 Objective

It is computationally unreasonable to perform a detailed PSA on all of the 209 compartments identified from the Fire Compartment Interaction Analysis. The objective of this screening assessment was to reduce the number of compartments in which detailed fire risk assessments must be performed. A conservative, screening assessment of core damage frequency (SCDF) is used to achieve this objective. As indicated by Equation 3 and the accompanying description, the assessment is composed of three major parts as follows:

- Development of the screening fire ignition frequency for each compartment in units of fires per year per compartment (described in Paragraph 4.1.2).
- With the assumption that all equipment and cables in a compartment are failed by any fire in that compartment, identify a conservative reactor trip

transient (initiating event) from among those used in the HCGS PSA [PSE&G, 1994b].

- With the assumption that all equipment and cables in a compartment are failed by any fire in that compartment, calculate a conservative, screening conditional core damage probability (SCCDP) using the HCGS PSA model.

Development of the initiating events and SCCDPs result in the factor $P_n(CD)$ in Equation 3. The SCDF of each compartment is calculated as the product of the screening fire ignition frequency and the SCCDP. This assumes that 1) every fire in the fire database used in this study will propagate to damage all equipment and cables in the compartment and 2) every fire causes the conservative initiating event identified. A compartment was considered removed from further consideration (i.e., screened out) if the SCDF for that compartment, calculated as described above, was found to be less than 1E-06/yr. Potentially risk significant compartments were identified for more detailed probabilistic safety assessment in this manner.

4.1.3.2 Initiating Event Selection for Quantitative Screening

The assignment of an initiating event to a compartment determines the event tree, derived from the HCGS PSA, to be used for the SCCDP. This was done by the collective judgment of the analysts, and reviewed by plant operations personnel, after performing the FCIA and fire ignition frequency analyses. The constraints, listed below, were followed in order to assure a conservative assignment of initiating events. In this context, conservative means leading to a higher than expected SCCDP assuming all equipment and cables in a compartment were disabled. The constraints were:

- All compartments lead to a reactor trip or more severe transient.
- Reactor trip or manual shutdown were modeled using the MSIV closure event tree.
- Compartments in which SRV related cables or equipment are located were evaluated using the most conservative of either the Inadvertent Open Relief Valve event tree or the appropriate non-LOCA transient initiating event (e.g., MSIV closure) in which an SRV was forced to fail open. A sensitivity study found that for such compartments, use of the MSIV closure model yielded a higher SCCDP than use of the inadvertent open SRV event tree model [PSE&G, 1996p]. Therefore, the former was used for the screening analysis of these compartments.

The assumption of a transient at least as severe as an MSIV closure for a compartment was made because of the unfeasibility of complete and precise knowledge of the cable inventory in each compartment. This lack of knowledge uncertainty, therefore, led to a highly conservative assumption. In general, cable inventories were known at the level of an electrical channel rather than component termination points.

The initiating events considered for each compartment [PSE&G, 1994b] are listed below:

- Turbine Trip
- Loss of Condenser Vacuum
- Loss of Feedwater
- MSIV Closure
- Loss of Instrument Air/Service Air
- Loss of Service Water/SACS
- Loss of RACS
- Loss of HVAC
- Loss of Off-site Power (LOOP)
- Inadvertent Open SRV
- Small LOCA
- Intermediate LOCA
- Large LOCA
- ISLOCA

Of the first five above listed initiating events, MSIV closure yields the highest SCCDP. Therefore, it was used as the model for any reactor trip or turbine trip transient (in the absence of a more severe initiating event). A complete Loss of Service Water/SACS or RACS can not occur from fire induced loss of any single compartment in the absence of hot shorts. Therefore, these initiating events were not used for the screening analysis. (Hot shorts were treated in the fire PSA of the unscreened compartments).

Loss of HVAC is narrowly defined in the HCGS PSA. This initiating event is appropriate only for the situations in which either all Class 1E Panel Room HVAC is lost or all Switchgear Room Cooling (all four channels) is [illegible] Only three compartments (5604, 5620 and 5703/5704) are capable of causing this initiating event.

Loss of off-site power is possible for several compartments in the Turbine Building, Radwaste Area, and Control/Diesel Building and the yard because of the presence of 4kV 1E offsite power bus bars or transformer protection relays.

LOCAs, other than inadvertent SRV opening, and ISLOCA can not be caused by fire in any single compartment. As discussed further in Paragraph 4.6.6, SRV opening may be caused by hot shorts. All compartments in which an inadvertent SRV opening could occur are also susceptible to MSIV closure.

Column 3 of Table 4.8 presents the initiating event assigned to each compartment for the screening analysis. Column 4 of that table is a condensed explanation of the assessment which determined the initiating event of each compartment. The explanations fell into eight categories as follows:

1. Reactor trip (or operator shutdown) is unlikely to be initiated by a fire in this location. The SCCDP was conservatively estimated by assuming a MSIV closure initiating event.
2. Reactor trip (or operator shutdown) is likely to be initiated by a large fire in this location. The SCCDP was conservatively estimated by assuming a MSIV closure initiating event.
3. Off-site power or control cables are located within this compartment. A LOOP initiating event was used to estimate the SCCDP for this location.
4. A fire in this location, coupled with a hot short, might cause spurious actuation of one or more SRVs. With no credit taken for the conditional probability of a hot short (given) a fire, the highest SCCDP among the alternative initiating events was used.
5. Loss of a Class 1E AC or DC 1E channel does not cause a reactor trip. The SCCDP was conservatively estimated by assuming a MSIV closure initiating event.
6. Contains cable, switchgear or equipment whose loss would cause loss of 1 Division of 1E Panel Room HVAC or Control Area HVAC.
7. Fire that fails all equipment in this location would cause a loss of HVAC initiating event as defined in the HCGS PSA.
8. Divisions I and II of Service Water are in separate rooms with a room in between. A complete loss of service water initiating event can not occur owing to a fire in an individual compartment. The SCCDP was

conservatively estimated by assuming a MSIV closure initiating event with the additional failure of a division of service water.

4.1.3.3 Calculation of Screening Conditional Core Damage Probabilities

The initiating event, in effect, designates the event tree to be used for the calculation of screening conditional core damage probability. The presumption that a fire is the cause of the transient carries with it the assumption that the fire may have caused failure of equipment or cables. In this assessment, all equipment and cables known to be in a compartment were presumed to be failed.

The initiating event selection process resulted in the need for two event trees from the HCGS PSA: the MSIV Closure event tree and the Loss of Offsite Power event tree. Figure 4.1 and 4.2, respectively, show these event trees. The MSIV Closure event tree was modified from the original internal event model by 1) assigning failure of HVAC directly to core damage, and 2) assigning all recovery branches a failure probability of unity. The Loss of Offsite Power event tree was modified from the original internal event model by 1) assigning failure of HVAC to core damage, 2) assigning inadvertent opening of an SRV to core damage, and 3) assigning all recovery branches a failure probability of unity.

In addition to the conservative selection of initiating events and the conservative modification of the event trees, the following assumptions applied for each run:

- Control rod drive pumps and feedwater pumps were assumed to be unavailable throughout the analysis. This assumption is due to insufficient information about the locations of cabling for control rod drive pumps, and RACS related equipment.
- Instrument air was assumed to be unavailable unless associated with the use of the condensate pumps.
- Human recovery actions (in Event Uv in the event trees) were presumed failed.
- Condensate pumps were assumed to be unavailable for fire initiated in the Turbine building. Note, however, that condensate was assumed to be unavailable for all unscreened compartments, treated in the PSA, which went into the calculation of the core damage frequency.

In addition, neither the screening fire frequency calculations nor the screening conditional core damage probability calculations took credit for fire suppression.

In the absence of recovery actions, the conditional core damage probability of a complete loss of HVAC, as defined in the HCGS PSA, is unity. Therefore, an event tree was not required for this initiating event.

The method used for the screening analysis, does not apply to certain compartments in the plant. These are the cable spreading room (5202), upper control equipment room (5605), lower control equipment room (5302), control room equipment room mezzanine (5403), and control room (5510). The screening conditional core damage probability of these rooms was assumed to be unity. None of them were screened out.

Table 4-9 lists the unscreened compartments that resulted from the quantitative screening analysis. These compartments were analyzed using a scenario-by-scenario fire probabilistic safety assessment as described in Paragraphs 4.3, 4.4, 4.5, and 4.6.

4.1.3.4 Results of the Quantitative Fire Screening Analysis

Table 4.8 provides the results of the quantitative screening analysis, represented by Equation 3. Column 6 provides the calculated screening conditional core damage probabilities. Column 7 is the screening core damage frequency of the compartment which is the product of screening fire ignition frequency (column 5) and the screening conditional core damage (column 6). Columns 8 and 9 indicate whether or not a compartment met the 10^{-6}/yr. screening criterion. Table 4-9 lists the unscreened compartments that resulted from the quantitative screening analysis. These compartments were analyzed using a scenario-by-scenario fire probabilistic safety assessment as described in Paragraphs 4.3, 4.4, 4.5, and 4.6.

4.1.4 SPECIAL TREATMENT OF HIGH HAZARD AREAS

The quantitative screening analysis and fire PSA treat the frequency of fires and the conditional probability that a fire will lead to core damage. The ground rules for a PSA or a FIVE analysis include treating fire barriers as an effective fire containing structure. The PSA, however, did not adequately consider the possibility of, and potential consequences of an extremely large fire, which by its intensity and duration might breach a fire barrier. It simply used the FIVE criterion to define fire compartment barriers. This Paragraph summarizes the method and results of the investigation into high hazard areas of the HCGS. High hazard areas are defined as those which contain a sufficient quantity of flammable or combustible materials that, if fully involved in a fire, might have the ability to breach barriers and damage structures. No distinction was made between flammable and combustible materials in this analysis and the word 'combustible' is used for both.

4.1.4.1 High Hazard Area Analysis Method

The quantity of combustible (solid or liquid) materials in each room in the plant was reviewed. Those areas that normally contain, during operation, at least 50 gallons of liquid combustibles, an excessive amount of solid combustibles (excluding cables), or explosive material (e.g., hydrogen) were further analyzed. These areas are noted in Table 4.10.

A large fire scenario is postulated in each of these rooms (Table 4.10). The fire is postulated to involve all of the fuel and be uncontrolled by fire suppression or natural fire limiting phenomena (e.g., oxygen availability). The potential consequences of this worst case fire are discussed in terms of fire barrier failure, structural failure, and potential to affect safety related equipment. Finally, the frequency of such a large, uncontrolled fire is estimated. This is done by using the FIVE data base to estimate the fire frequency and by using conservative estimates of the complete failure of automatic fire suppression systems, manual fixed fire suppression systems, and fire brigade fire suppression efforts. The fundamental assumption in this analysis is that a large, uncontrolled fire, which can damage fire barriers and structures, can occur only if all fire suppression efforts fail.

4.1.4.2 High Hazard Area Analysis Results

Table 4.11 is a summary of a more detailed study of the worst case consequences and the estimated frequency of the worst case fire scenario of high hazard areas at the HCGS [PSE&G, 1997f]. Note that this Estimated Fire Frequency is just that. It is the frequency of the worst case fire scenario, accounting for suppression efforts. It is not a core damage frequency. The worst case consequences were assigned by analysts' judgment. All of the following rooms were screened out during the quantitative screening analysis, previously discussed, except the diesel generator rooms (5304, 5305, 5306, 5307) and the turbine building equipment unloading area (1316). None of the identified high hazard compartments, other than the diesel generator rooms and turbine building equipment unloading area, are risk significant.

4.1.4.3 Conclusions

The high hazard area analysis demonstrates that such areas, other than those already identified as potentially risk significant and treated in the fire PSA, do not significantly contribute to the overall risk associated with fires at the HCGS.

4.2 REVIEW OF PLANT INFORMATION AND WALKDOWN

4.2.1 DOCUMENT REVIEW

Much of the documentation used for this study has been referenced as part of the discussion in other paragraphs. This paragraph reviews the documentation used for this study. This is not a complete list. An additional list is found in the references at the end of this IPEEE report.

The following references were used for understanding of plant design, operation, and configuration related to this fire analysis:

- HCGS Electrical drawings E-0001 through E-0006; control room console and board drawings J-0399-0, J-0600-0, J-0602-0, J-0648, J-0649, J-0650-1; lower control equipment room panel drawings 5-0601 and 5-0605; upper control equipment room panel drawings 5-0610 and 5-0611; architectural drawings A-5651 through A-5659, A-4641 through A-4648, A-3640, A-3630 through A-3638, and A-1611 through A-1617.
- HCGS Pre-Fire Plans
- HCGS IPE and PSA [PSE&G, 1994a and b]
- PSE&G MMIS

In addition specific cable traces were obtained via the HCGS CARTS computerized cabling database.

The following references were used for understanding fire protection features and fire barriers:

- HCGS UFSAR Paragraph 9.5 and Appendix 9A [PSE&G, 1995f]
- PSE&G drawings: Fire Protection and Detection drawing M-5001 through M-5013; BTP CMEB 9.5-1 Fire Barriers drawings M-5101 through M-5111; Fire Area Floor Plan drawings M-5112 through M-5124
- PSE&G Operational Fire Protection Program [PSE&G, 1996f]

The following references were used for fire event data:

- FIVE [EPRI, 1993b]
- Sandia fire database with 1994 updates [NRC, 1986a] and [Sandia 1994a]
- Major Common Cause Initiating Event Study, Shoreham Nuclear Power Station [NUS, 1984a]

The following references were used for fire growth and damage related data:

- FIVE [EPRI, 1993b]
- "A Summary of Nuclear Power Plant Fire Safety Research at Sandia National Laboratories, 1975 - 1987" [NRC, 1989f]
- "An Investigation of the Effects of Thermal Aging on the Fire Vulnerability of Electrical Cables" [NRC, 1991c]
- SFPE Handbook of Fire Protection Engineering [NFPA, 1995a]
- Marks Standard Handbook for Mechanical Engineers [Marks, 1986a]

Generally, documentation about plant design, configuration and operation were adequate for this study. The documentation carried cable inventory information for each room at the level of electrical divisions. The walkdown was needed to provide information at the level of channels. It was this level of cable information that was generally used in the study. In a few cases (e.g., for the RPS MG set room 5105) individual cable identifiers were traced using the CARTS system and cable drawings to determine the precise end points.

4.2.2 WALKDOWN OBJECTIVES

NUREG-1407 and FIVE recognize the importance of a thorough fire walkdown in order to 1) assure that documentation, particularly for cable routing and fixed combustible, represents the as-built plant, 2) uncover potential intercompartment interactions associated with openings, walls or inadequate fire barriers, 3) aid in addressing the Sandia Fire Risk Scoping Study issues, 4) assess the likelihood of critical transient combustible loading, 5) review fire protection features of the plant, and 6) verify the assumptions used in fire damage propagation analyses. This walkdown, additionally, developed fire scenarios which were later quantified as part of the fire PSA.

4.2.3 WALKDOWN TEAM

The fire IPEEE walkdown team consisted of a fire protection engineer from PSE&G, a PSA analyst from PSE&G, a fire PSA and fire modeling specialist from Safety Factor Associates, Inc., and a fire PSA specialist from Kazarians & Associates. The latter two have participated in several fire PSAs and walkdowns. The principle walkdown was supplemented by several as-needed walkdowns as the work progressed. The total walkdown effort encompassed several hundred person hours and was achieved during October through December of 1996.

4.2.4 WALKDOWN METHOD AND PROCEDURE

Three sets of walkdown checklists were developed and entitled as follows:

- Fire Hazard
- Fire Protection
- Fire Barriers

Each checklist was contained in a notebook and a separate checklist sheet was completed for each room. Thus, each notebook had over 200 checklist sheets with supplemental pages as described below.

The Fire Hazard Checklist required the walkdown to provide the information related to:

- cable tray location and protection features,
- equipment located in each room,
- the susceptibility of this equipment to suppression system actuation,
- paths for hot gas layers,
- potential fire scenarios including ignition sources, target combustible, and potentially affected equipment and cables
- existence of transient combustibles.

Table 4.12 shows an exemplar fire hazard walkdown checklist. In addition to this sheet, the notebook contained for each compartment: 1) a diagram showing room layout and adjacent rooms from the pre-fire plans, and 2) the fire hazard summary sheet from the UFSAR , [PSE&G, 1995f - Appendix 9A, Fire Hazard Analysis Tabulation].

The Fire Protection Checklist required the walkdown to provide information related to:

- amount of foot traffic (only obviously high areas of foot traffic were noted)
- ignition sources (in-situ and transient)
- combustible loading
- fire detection features
- fire suppression systems
- drainage

Table 4.13 contains an exemplar fire protection walkdown checklist. In addition to this sheet, the notebook contained for each compartment a diagram showing room layout and adjacent rooms from the pre-fire plans.

The Fire Barrier Checklist required the walkdown to provide information related to the following:

- existence of doors, their type and whether they are normally open
- ventilation openings
- dampers and closure mechanism
- openings in walls and ceilings
- FCIA criteria verification
- ventilation duct paths

Table 4.14 contains an exemplar fire barrier walkdown checklist. In addition to this sheet, the notebook contained for each compartment: 1) a diagram showing room layout and adjacent rooms from the pre-fire plans and 2) the FCIA data sheet.

Guidance on the principles and use of the checklists was developed and is shown as Table 4.15. The walkdown checklists and guidance were developed to encompass the collective guidance in FIVE [EPRI, 1993b] and NUREG/CR-4840, [NRC, 1989g], and to provide all the information needed to meet the requirements of GL 88-20, Supplement 4 as interpreted in NUREG-1407 (including the Fire Risk Scoping Study Issues). In addition, the walkdown was used to develop fire scenarios (fire ignition sources, targets, distances from source to target, etc.) to be used for the fire PSA described in Paragraphs 4.3, 4.4, 4.5, and 4.6.

The walkdown encompassed all rooms in the Reactor Building, Radwaste Building, Turbine Building, and Service Water Intake Structure, Control/Diesel Building, and Yard. For high radiation and inaccessible areas, a "virtual" walkdown was performed by reviewing the digitized photographs of these rooms. These photos are computer displayed such that the view changes as the observer "walks" through the plant using a joystick. Use of the digitized photographs was consistent with the ALARA policy of PSE&G. The walkdown notes indicates each room in which a "virtual" walkdown was performed.

Before the walkdown began, the participants studied how to perform their tasks by reviewing the walkdown guidance. Particular emphasis was placed on noting the location of cables and on cross zone fire spread owing to openings. The first assured that the fire IPEEE used as-built information only. The location of cable runs in the unscreened rooms was diagrammed in the walkdown notes. The second assured that potential for damage owing to hot gas layers was comprehensively considered and treated.

4.2.5 WALKDOWN OBSERVATIONS, FINDINGS, AND CONCLUSIONS

4.2.5.1 General Observations about the HCGS Electrical Design

Cables and conduits were well labeled with color coding indicating their electrical channel. HCGS has four Class 1E electrical channels and two mechanical equipment divisions coded as follows.

Color Coding	Channel Designation	Division
Green	A (alpha)	I
Purple	B (bravo)	II
Blue	C (charley)	I
Orange	D (delta)	II

Non-1E cables and conduits are coded as white.

The walkdown notes (via the checklists) recorded the location and color of cables in each compartment. Therefore, use of as-built conditions for this analysis has been assured.

The following are generally observed trends in the electrical design of the plant which were evident during the walkdown:

- When more than one channel of cable is present in a single room, they are not of redundant equipment.
- When more than one channel of cable is present in a single room, cable is either within conduits or within fully enclosed cable trays.
- Clusters of 2, 3 or 4 single cable conduits running in parallel were found in many rooms. These were generally associated with the reactor protection system, recirculation pump control and instrumentation, reactor vessel pressure and level instrumentation, and ADS logic.
- Cable in battery rooms were always in conduits.

4.2.5.2 General Findings

The following are general findings that arose from the walkdown and a review of the completed walkdown checklists.

1. Fire barriers, dampers (with fusible link closure mechanisms), doors, and penetrations appear to be well maintained. The walkdown notes [PSE&G, 1997b] recorded openings between rooms and assessed their significance with respect to spread of hot gases. These openings were generally unused penetrations, unsealed penetrations and cable tray runs. Rooms were combined for the probabilistic analysis when such openings were judged to be significant. The FCIA sheets were updated to reflect the walkdown findings.

2. Several rooms were combined as a result of the walkdown finding openings through which a fire might propagate. These rooms were also combined for the purposes of quantitative screening and fire PSA analyses. Conversely, several rooms which were previously analyzed as open to each other were found to be separate with respect to propagation of hot gases.

3. The walkdown found that all screened out compartments were properly screened out.

4. The walkdown found no fire suppression susceptibility. This means that either hoses are positioned such that doors need not be kept open to suppress a fire, or, if a door is to be kept open to suppress a fire, there is no susceptible equipment in the adjacent room. Paragraph 4.8 provides a discussion of damage owing to inadvertent fixed suppression system actuation.

5. Although a complete tracing of HVAC ducts was not performed, a general observation was that smoke and hot gas from fires will not propagate through return ducts to cause failure of equipment in rooms not involved with fire. Also, hot gas from a fire will not enter operating discharge flow ducts because pressure in discharge ducts is higher than in the room.

6. All stairwells were found to be surrounded by two hour fire barriers and open to all elevations. Hot gas and smoke would gather at the ceiling at the highest elevation. No cables were found in stairwells although some were used to store non-combustible materials.

7. No incidents of a transient critical combustible loading (either authorized or unauthorized) were found during the walkdown.

8. There are no normally open fire doors in the plant. No fire doors were found open without a fire watch posted. There are no doors with fusible link (or

other automatic) closure mechanism.

4.2.5.3 Selection of Specific Observations

The complete set of specific observations encompasses hundreds of pages [PSE&G, 1997b]. The following is a small selection of such observations to provide the reviewer with examples. All of the observations were used when performing the quantitative screening analysis and the fire PSA.

Selected Observations in the Reactor Building

- Torus Water Cleanup Room 4101 has two six ft^2 openings in the ceiling to MCC Room 4201. Room 4201 has Channel D cables and Channel A, B and C conduits. These conduits are associated with the main steam system and reactor vessel pressure instrumentation.
- RHR pump room 4107 contains 1E Channel C and D cables and conduits and non-1E conduits.
- RHR pump room 4114 has Channel C cable only and no openings to other rooms.
- Channel B cable trays (associated with RCIC among other systems) is on the south wall of Corridor 4203. These trays traverse into CRD Pump room 4202. Channel A conduit is related to the HPCI test return line valves which are MOVs. Thus a fire that fails the cables within this conduit does not fail HPCI.
- Rooms 4205 and 4207 are connected with an eight ft^2 opening and were treated as a single compartment.
- Rooms 4301, 4309, 4310, and 4311 were treated as a single compartment, because either there is no wall separation or large openings for cable trays pass through the rooms. Room 4311 has no safety related equipment or cables. Room 4313 is an equipment airlock with no openings and no safety related equipment.
- The upper parts of the reactor building have large rooms that are open to each other and open to an open crane hoistway. The hoistway is open to the ceiling of the reactor building above the refueling deck. The following rooms were treated as connected to each other:

  4407, 4408, 4408A, 4410, 4411, 4412, 4413, 4501, 4504, 4508, 4509, 4601, 4602, 4603, 4604, 4605, 4607, 4608, 4614, 4615, 4616, 4617, 4618, 4626, 4627, and 4628

Fires in this area will do local damage only, failing the equipment and possibly cables in the immediate vicinity of the fire. Failure of cables not in the immediate vicinity, owing to a hot gas layer, is unlikely because of the interconnection with the top of the reactor building.

Selected Observations in the Control/Diesel Building

- The HPCI Battery Room 5104 contains Channel A cables only, which are all located in conduits.

- The RPS MG Set Room 5105 contains the RPS MG sets with all but two Channel A conduits terminating in the room. The terminating conduits are related to the RPS function. The non-terminating conduits are related to HP battery room heater function.

- All cables in the Diesel Fuel Storage Tank Rooms (5107 to 5710) are in conduits and connected to the diesel fuel oil transfer pumps.

- Room 5202 is protected by an automatic preaction sprinkler system. The system is designed to control and minimize fire damage within the room. However, the system is not designed to protect one cable tray from a fire in another cable tray directly underneath it. The only ignition sources in this room would come from transient combustibles, because cable initiated fires are not considered credible. As indicated in the UFSAR, all cables at the HCGS are IEEE qualified. A single trash can was found in this room. The east half of this room contains non-1E cables and the west half contains 1E cables, all in uncovered cable trays. The east half of the room serves as a pathway for personnel access from 5207 to 5234.

  All vertical cable chases, listed below, contain only one 1E channel. They are surrounded by at least two hour fire barriers and have automatic sprinkler systems with heat detectors. The sprinkler heads are located above cables at every elevation.

- Electrical Access Room 5339 contains Channel C cable trays, Channel A conduits, and offsite power bus bars in an enclosure. There are no stationary ignition sources in the room. This room is adjacent to the diesel generator rooms and the intervening door is not curbed.

- Diesel Generator Rooms 5304, 5305, 5306 and 5307 each contain a diesel generator, a diesel fuel oil day tank, auxiliary equipment, and an automatic CO_2 fire suppression system. Offsite power bus bars (from both redundant sources) in their enclosures run along the ceiling of these rooms.

Room Number	Compartment Description
5203/5323/5405/5531	Vertical Cable Chase (D)
5204/5324/5406/5532	Vertical Cable Chase (B)
5205/5325/5407/5533	Vertical Cable Chase (C)
5206/5326/5408/5534	Vertical Cable Chase (A)
5331/5419/5531	Vertical Cable Chase (D)
5332/5420/5532	Vertical Cable Chase (B)
5333/5421/5533	Vertical Cable Chase (C)
5334/5422/5534	Vertical Cable Chase (A)

- Battery Charger Room 5540 has 125 Vdc Channel B cabinets and cable only.
- The HVAC Equipment Room 5620 contains the air handling units for both trains of 1E Panel Room HVAC and associated panels and cables. It also contains cables in conduits and enclosed trays for power to the BOP and NSSS computers. There is a small space heater in the room. Cables are either in conduits or enclosed trays, are about ten feet from the floor and five feet from the ceiling. The air handling units are essentially non-combustible with no intervening combustibles. A flammable liquid storage cabinet was found in the room at least 25 feet from the nearest electrical panel or air handling units and about ten feet horizontal displacement from the nearest overhead cable runs.
- DG Area HVAC Equipment Area 5703/5704 contains, among other items, panels, both compressors and both pumps for chilled water for 1E Panel Room HVAC. This area along with Room 5620 and Room 5604 are the only areas in which a large fire might potentially cause a complete loss of 1E Panel Room HVAC. The large compressors are located within a few feet of each other.

4.2.5.4 Walkdown Conclusions

The walkdowns were instrumental in the successful and accurate completion of this study. They provided an accurate picture of fire sources, targets, and relative distances for the construction of fire scenarios. The walkdown notes also provided locations of sources and targets with respect to walls, floor, ceiling, and catwalks. They were also able to supplement documented information about compartment inventories, openings between rooms, room drainage, fire doors and dampers, and effectiveness of the locations of fire detection and suppression equipment. Scenarios were able to include the potential of spread to other compartments because of the information about openings in walls and ceilings provided by the walkdowns.

4.3 FIRE GROWTH AND PROPAGATION

4.3.1 INTRODUCTION AND PERSPECTIVE

Table 4.9 provides a list of the unscreened compartments. Each compartment was analyzed by a detailed fire PSA. An overview of the fire PSA methodology was provided in Paragraph 4.0.3 and centered around Equations 1 and 2. In particular, the fire growth and propagation analysis was used to develop the information for the following factors in Equation 1:

$P(T_l / S_{j,k})$ = probability of damaging target l (T_l) with source j of intensity k without consideration of suppression. This quantity is often 0 or 1. That is, either a deterministic fire damage calculation shows that it is possible to damage the target or it shows that the target can not be damaged.

$P(E_{j,k} < T_l)$ = probability of not extinguishing fire from source $S_{j,k}$ before damage to T_l. Determination of this quantity involves a calculation to obtain the relative time of occurrence of damage and extinguishment.

Numerous fire growth and propagation studies, including sensitivity studies, were performed for the unscreened compartments. However, preceding the discussion of these studies, the procedure used in the detailed fire PSA is provided in order to clearly explain the context of these studies.

4.3.2 IMPLEMENTATION PROCEDURE OF THE FIRE PSA OF UNSCREENED FIRE COMPARTMENTS

For all compartments but the control room, the practical application of the formulation of Equation 1 was implemented using the step-wise procedure given as follows:

1. The compartment inventory, as derived from all sources previously described in Paragraph 4.2, the reference list, and the walkdown, was listed. This served as the basis for defining stationary and transient combustible ignition sources.

2. The automatic and manual detection and suppression equipment in the compartment was identified.
3. The walkdown notes and insights that influence the development of fire scenarios (e.g., sources and distances to targets) were documented.
4. Large openings between compartments were noted.
5. Significant fire protection and drainage features were listed.
6. The quantity of lubrication or fuel oil found in the compartment during the walkdown was indicated.
7. Scenarios initiated by stationary fire sources in the compartment were described in detail. This included source, fire size, target, and initiating event. Also included were scenarios that have a potential for fire spread through openings to other compartments.
8. Scenarios initiated by postulated transient combustible fuel sources in the compartment were described in detail. This included source, fire size, target, and initiating event. Also included were scenarios that have a potential for fire spread through openings to other compartments.
9. Using the fire growth and propagation methodology (FireTran.xlt), derived from the FIVE methodology, determine the potential for target damage for each fire scenario. If target damage can occur and a fire suppression system is present in the compartment, also determine the relative timing of damage and suppression.
10. Establish a conclusion as to whether fire suppression can be credited.
11. The initiating event was identified for each scenario.
12. The possibility of hot shorts to compromise or actuate equipment owing to fires in this compartment was evaluated and noted.
13. The possibility of transient induced LOCAs owing to fires in this compartment was evaluated and noted.
14. The core damage frequency of each scenario was quantified using the fire frequency, probability of non-extinguishment, likelihood of target damage, and conditional core damage probability (CCDP). The fire frequency of stationary sources was derived from the ISDS analysis for each ignition source. The transient combustible fire frequency is calculated using the method in Paragraph 4.1.2.
15. The scenario frequencies were summed, per Equation 2, to obtain a total compartment core damage frequency.

16. The total HCGS fire CDF is taken to be the total of all scenario frequencies of the unscreened compartments per Equation 3.

This procedure was documented, for each unscreened compartment, using a template as shown in Table 4.24. An example analysis is provided in Table 4.25.

4.3.3 FIRE GROWTH AND SUPPRESSION METHODOLOGY

The objectives of the fire growth and suppression calculations were to:

- conservatively determine if a source has the potential to damage targets in the vicinity
- if damage is possible, determine if extinguishment can occur before damage

For transient exposure fires, an additional objective was to determine the distance from the source over which fire damage can occur. For transient combustible fires, the quantity $P(T_i / S_{j,k})$ of Equation 1 is estimated as the following ratio:

$$\frac{\text{Area of compartment formed by the locus of points at which source } S_{j,k} \text{ can damage target } T_i}{\text{Total Area of the Compartment}}$$

Four types of fire damage mechanisms were modeled: plume effects, ceiling jet effects, hot gas layer effects, and thermal radiation effects. Fire damage calculations were performed using a modified version of the formulation found in the Fire Screening Methodology User Guide [EPRI, 1993b - Attachment 10.4]. This methodology was implemented in a series of EXCEL templates, called FireTran.xlt. Three subsidiary templates were developed in order to specialize the inputs to specific fire sources as follows:

- liqpoolstd.xlt for liquid fuel pool fires
- cabinet.xlt for cabinet initiated fires
- trash.xlt for trash can initiated fires

With the exception of the diesel-generator rooms, liquid pool fires were modeled as unconfined because this maximizes the total heat release (BTU/sec) for a given heat generation flux (BTU/sec per ft^2 of pool surface). Spacing and drainage in the diesel generator room limits a liquid pool to about 100 square feet.

Quantities of five gallons through 20 gallons were investigated in this study depending on the compartment. The PSE&G operational fire protection program [PSE&G, 1996f - Paragraph 5.3] states that no more combustibles than needed are allowed in a room and use and storage of flammable liquids shall be limited to five gallons. The walkdown did not find stored transient combustibles in excess of three gallons per container. Because the diesel-generators are sources for larger leaks, a variety of diesel fuel oil liquid pools up to 20 gallons was investigated for the diesel-generator rooms.

Cabinet fires were simulated by a fire located at the top of the cabinet with a heat release rate of 1233 BTU/sec which corresponds to a peak heat release rate for control cabinet fires [NRC, 1989f]. The analyzed trash can fire was 32 gallons using the maximum heat release rate found in FIVE [EPRI, 1993b - Figure 4 - Attachment 10.4]. Pump fires were treated as unconfined liquid pool fires located at the elevation of the pump using a quantity of fuel that conservatively bounds the amount of lube oil in the pump (usually one or two gallons).

An example of the calculation templates for liquid pool fires is shown in Tables 4.16 through 4.19. These tables do not show the embedded equations, comments and notes but they do show the inputs, outputs, and general approach taken. Two types of computations are modeled: steady state and pseudo-time dependent. The former, exhibited in Tables 4.16 through 4.18, assume that the fire burns at the peak heat release rate of a material for infinite time (i.e., with no fuel consumption). The latter (Table 4.19) is an attempt, using a steady state formulation, to estimate the time to damage target materials versus the time that fire detection and suppression would occur. Fuel consumption is also included in that the fire is assumed to burn at the peak heat release rate until the fuel is completely consumed.

Steady state plume effects (Table 4.16) are modeled using the equations and formulation of Worksheet 1, Attachment 10.4 of FIVE. Steady state ceiling jet effects (Table 4.17) are modeled using Worksheet 2, Attachment 10.4 of FIVE. Both of these conservatively add the heat available in the hot gas layer to either the plume or ceiling jet in order to determine if the target damage criterion has been exceeded. Steady state thermal radiation effects (Table 4.18) are modeled using Worksheet 3, Attachment 10.4 of FIVE. The pseudo-time dependent computations (Table 4.19), which calculated the competition between fire damage time and fire suppression time were modeled after Worksheet A-1, Attachment 10.4 of FIVE. However, because of the limitations of the FIVE formulation, the following modifications were made to the FIVE Fire Screening Methodology in an attempt to insert somewhat more realism into the model.

- The FIVE model resulted in unrealistically high plume and ceiling jet temperatures, far higher than the flame temperature of the source. Therefore, temperature of the ceiling jet and plume regions was limited to the typical hydrocarbon adiabatic limiting flame temperature which is approximately 2600°F [NFPA, 1995a - Page 1-86].

- When calculating the time to damage a target, the FIVE model does not include an estimate of the time that the fuel would be consumed by its own combustion. In this report, this time is estimated from the total quantity of energy in the fuel (BTU) divided by the heat release rate of the fuel (BTU/sec).

- In calculating the time to detection, the FIVE model assumes heat is available essentially instantly at the detector which is a substantial underestimate of the detector actuation time, which in turn, might over predict the effectiveness of detection and suppression. This model includes a delay associated with hot gas rise velocity as well as the delay associated with detector placement.

- The FIVE model does not estimate the time to fire extinguishment. In this report, applicable delay times such as suppression actuation delay and cable tray soak times were added to the detection and extinguishment times to estimate actual suppression time. The cable tray soak time was taken as five minutes from NUREG/CR-5384, [NRC, 1989f - Table 4-1]. Suppression actuation delay times arise, particularly for automatic carbon dioxide systems, to allow time for personnel to evacuate the compartment. The actual system specifications for each compartment were used.

- The FIVE model does not conserve energy. In this work, total energy contained in the fuel, Q_{total} (in BTU), was calculated from the specific heat release quantity of the fuel (BTU/lbm) and the postulated amount of fuel (e.g., five gallons). Physical constants of the fuel were either taken from the FIVE document, Marks Standard Handbook for Mechanical Engineers [Marks, 1986a], or the Handbook of Chemistry and Physics, 77th edition [Lide, 1996A]. Q_{total} was conserved by adjusting the energy of the hot gas layer to equal Q_{total} minus the energy in the plume minus the energy in the ceiling jet minus the energy in thermal radiation.

- FIVE uses the horizontal distance to the target to estimate thermal radiation effects. The actual line of sight distance was used in this model.

The modified formulation also led to a difference between how FIVE and FireTran.xlt use the input quantities. The FireTran.xlt use of the quantities is described below:

- The fuel heat release rate, in BTU/sec per ft^2 of area , was used only in the calculation for total heat release rate (BTU/sec) by multiplying it by the area of the fire. The results are somewhat sensitive to the assumed BTU/sec per ft^2. An extensive set of sensitivity studies helped establish a reasonable number for this quantity, which is 120 BTU/sec per ft^2.
- A fuel density was used to convert the volume of fuel (given in gallons) to the mass of fuel. The mass of fuel was used only to obtain Q_{total} as described above.
- The specific area (ft^2/gal) of unconfined liquid pools was used to define the total heat release rate (BTU/sec) for unconfined fires. The specific area and the input volume of fuel defines the spill radius which, in turn, defines the extent of effect of the plume. The radius of influence of the plume was assumed to be proportional to the radius of the liquid pool. The specific area also influences the time for fuel consumption (called the burn out time) via the calculation of total heat release rate. An extensive set of sensitivity studies helped establish a reasonable number for this quantity, which is 120 ft^2/gal.

4.3.4 TYPICAL INPUT DATA AND SENSITIVITY STUDIES

A sensitivity study investigated the effect of the heat release rate in BTU/sec per ft^2 of pool surface area for liquid pool fires. A constant value of 120 BTU/sec per ft^2 was used for all liquid pool fires. The sensitivity study demonstrated an insensitivity, with respect to cable damage, for a typical HCGS room configuration over the range of 50 to 200 BTU/sec per ft^2. That is cables exposed to ceiling jet or plume effects were calculated to be damaged over the range of 50 to 200 BTU/sec per ft^2 . Cables exposed only to hot gas layer effects would not be damaged in typical HCGS room configurations over the range of 50 to 200 BTU/sec per ft^2.

A sensitivity study investigated the effect of spill specific area (ft^2/gal) for liquid pool fires. A constant value of 120 ft^2/gal [EPRI, 1993b - Table 3 - Attachment 10-4] was used for all liquid pool fires. For unconfined liquid pool fires the spill specific area and the heat release rate are used only as a product. Therefore, the insensitivity of cable damage to changes in this quantity is similar to that of the heat release rate explained above.

Other typical input data used in the fire damage calculations are shown in Table 4.20.

4.3.5 ASSUMPTIONS AND APPROXIMATIONS FOR FIRE GROWTH AND SUPPRESSION

The FIVE fire damage formulation makes the following conservative assumptions and approximations:

- Exposure fires instantly attain their peak intensities and remain there for the duration of the fire.
- Unit heat release rates (BTU/sec-ft^2) are associated with fully involved conditions.
- Targets respond with no delay to temperature changes in the surrounding environment.
- Heat loss to boundaries is 70% of the heat of the fire.
- Heat loss by convection in ventilated room fires is neglected.
- Plume and hot gas layer temperature effects are superimposed to determine if targets have been damaged.

The following assumptions and approximations were used in carrying out the calculations of fire damage in the FireTran.xlt model:

- The qualification of HCGS cables exclude self-ignited cable fires.
- The peak heat release rate was used for the entire fire [e.g., as shown in FIVE (EPRI, 1993b - Figure 4 - Attachment 10.4) for trash can fires] rather than the time dependent or average heat release rate over the fire duration.
- The ceiling jet layer is the top 15% of the height of the room. The hot gas layer is effective in the plume and ceiling jet layer. That is, the heat content of the hot gas layer was added to that of the plume or ceiling jet layer to estimate damage.
- Temperature damage criteria were used.
- Plume effects are limited to a cylinder above the source of the radius of the liquid pool.
- No credit was taken for cabling protected by conduit or enclosed cable trays.
- Calculations assume that there are no intervening combustibles or barriers separating the source from the target.
- All smoke and heat detectors at the HCGS were found, via the walkdown, to be located at the ceilings. Therefore, all detectors were assumed to be in the ceiling jet.

- Pump fires are modeled as liquid pool fires of quantity one or two gallons, whichever conservatively bounds the amount of lube oil in the pump.

- When calculating the spread of fire between rooms, it was assumed that the wall below the elevation of the openings found in the wall was not effective in preventing the spread of fire.

- When determining damage to target cables from a specific source (in the absence of suppression), if any elevation of cable was calculated to be damaged, all of the cables were assumed to be damaged.

- When determining whether target cables were damaged from a specific source before extinguishment, if any elevation of cable was calculated to be damaged before extinguishment, all of the cables were assumed to be damaged.

4.3.6 TREATMENT OF ACTIVE FIRE BARRIERS AND OPENINGS BETWEEN ROOMS

The walkdown revealed that there are no active fire doors at the HCGS. Also documented in the walkdown notes is the observation that all doors are normally closed unless a fire watch is posted. Some doors are also designated as "Security" doors. When these are opened additional security personnel are also present.

Openings between rooms were treated during the FCIA, the walkdown, in the quantitative screening analysis, and during calculations of fire growth during the PSA. During the FCIA and walkdown, large openings that could not contain a hot gas layer within a single room were treated by ignoring the existence of the entire wall. That is, the rooms were combined within a single compartment. This was reflected in the identification of compartments and the quantitative screening analysis (Table 4.8). Openings, owing to cable tray runs and unsealed penetrations, exist between rooms of the same compartment. Fire growth calculations were performed for unscreened compartments to investigate whether fires in one room could affect cables in the adjacent room. This was done by assuming 1) a fire is located at the wall of the source room, and b) a hot gas layer is at the elevation of the opening between rooms. A determination was made, by the use of FireTran.xlt, as to whether this layer was capable of damaging cables in the target room.

The walkdown also observed numerous fusible link active fire dampers, the locations of which are documented in the walkdown notes [PSE&G, 1997b]. For unscreened compartments, fire growth and damage calculations were performed for these dampers as if they were simple openings as described above. However, the overall treatment of openings and fire dampers within this IPEEE depended on their locations with respect to screened and unscreened compartments. Table 4.21 shows how they were treated.

4.3.7 SENSITIVITY STUDIES AND RESULTS OF FIRE DAMAGE AND SUPPRESSION CALCULATIONS

Each Fire Scenario Analysis worksheet (see, for example, Table 4.26) documents the fire scenarios applicable to the analyzed compartment and summarizes the results of the fire damage and suppression calculations performed for the associated compartment. The summaries in the Fire Scenario Analysis worksheets are actually the conclusions drawn from performing sensitivity calculations. Typically, the height and horizontal displacement of the target is varied in order to develop an overall picture of how the source fire affects the targets. In addition, the size of the source fire (in terms of gallons of liquid fuel) is sometimes varied in order to gain an understanding of the threshold sizes for damage. Therefore, numerous sensitivity studies were performed. Table 4.22 summarizes the results of these studies, compartment by compartment.

Note the following about the entries in this table:

- Fire damage is said to occur if any elevation of target cables was calculated as damaged.
- Therefore, an unqualified 'no' as an entry means that all elevations of target cables were calculated to be undamaged.
- A superscript "3" after a 'no' indicates that the steady state formulation calculated damage but the pseudo-time dependent formulation, which included fuel consumption, calculated fuel depletion before damage.
- An entry such as 'yes, <13' above cabinet' means that damage was calculated to occur for cables of elevation less than 13 feet from the top of the cabinet, and damage was not calculated to occur for cable elevations above this.
- An entry of n/a means that the target was not subject to the damage mechanism. For example, the target was not within the plume.
- Entries for source and target, such as 'liquid pool in 4202' and 'cables in 4203', are calculations to investigate the spread of damage through openings or dampers in the walls.

4.3.8 GENERAL INSIGHTS FROM FIRE DAMAGE AND SUPPRESSION CALCULATIONS

The following general insights have been developed about the characteristics of fire damage at the HCGS, as calculated using the method, assumptions, and data described above:

- Cabinet fires generate sufficient heat to damage only those cables directly above them and only from plume effects. Cabinet fires do not generate sufficient heat to damage cables in the ceiling jet region at the HCGS. A contributing factor is the high ceilings and large rooms at the HCGS. Similarly, the hot gas layer below the ceiling jet does not contain sufficient heat to damage cables.
- The typical radius of damage owing to radiation effects of a five gallon liquid pool fire, assuming that the fire has infinite fuel and stays at its peak heat release rate, is about 14 feet.
- Cables in plumes of five gallon liquid pool fires were calculated as being damaged using the steady state formulation. However, using the pseudo-time dependent formulation, the fire burned out before cables, located more than approximately 18' above the fire, were damaged.
- Cables affected only by the hot gas layer of a five gallon liquid pool fire were not calculated as damaged in the steady state formulation.
- Targets in the ceiling jet layer above a five gallon liquid pool fire were calculated as being damaged using the steady state formulation. However, the pseudo-time dependent formulation revealed that the fuel would be consumed before damage could occur.
- Damage was usually, but not always, calculated to occur before extinguishment was calculated to occur.

The unscreened compartments often included multiple rooms for situations in which the FCIA and walkdown resulted in the judgment that barriers either did not exist or were ineffective. In all cases in which large openings exist between the unscreened rooms or compartments, the openings were far enough from the ceiling so that ceiling jet effects were not applicable and the hot gas layer had insufficient energy to damage cables in the adjacent rooms or compartments. Therefore, inter-room or intercompartment fire damage does not occur owing to hot gas layer effects using the multiple room definition of compartments that emerged during the FCIA and screening studies.

4.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

Paragraph 4.3 provided a detailed description of fire growth, damage, and suppression methodology and results. This brief paragraph provides the damage criteria used in these calculations. Such damage criteria in a fire PSA may be thought of as analogous to component fragilities and failure modes per NUREG/CR-2300 [NRC, 1983a].

4.4.1 CABLES

All safety related cables at HCGS are IEEE 383 qualified and composed of EPR with Hypalon jacketing. Using the results of Sandia electrical cable fire induced damage tests [e.g., see NRC, 1991c] the damage temperature criterion of EPR/Hypalon was taken to be 698°F (370°C). Because the cables are qualified, self ignited cable fires have been assumed to be insignificant to risk.

Cable damage was calculated assuming all cables are unprotected even if surrounded by conduit or cable tray enclosures. Cable damage was assumed for all cables if any elevation was damaged.

4.4.2 4kV BUS BARS

There are three 4 inch by 4 inch Aluminum bus bars, one for each phase of offsite 4kV 1E power, in a single bus bar enclosure. The enclosure itself is Aluminum. Two bus bar enclosures run from the yard station service transformers, through parts of the Turbine Building, and Radwaste Building, into the Control/Diesel Building (Rooms 5301, 5339, diesel generator rooms) and terminate in the four 1E switchgear rooms by penetrating the ceiling of each diesel generator room. The failure mode was assumed to be a short circuit caused by molten aluminum from the enclosure either falling on adjacent bus bars or causing a circuit from the bus bar through the duct. This is assumed to cause a complete and unrecoverable loss of offsite 1E 4kV power. The damage criterion of the 4kV bus bars, therefore, is taken to be the melting temperature of the surrounding Aluminum duct (1220°F). Because of the close proximity of the 4kV bus bars, if either one is calculated to be damaged, both are assumed to be damaged.

One of the bus bars in the A and B diesel generator rooms is surrounded by a stainless steel mesh blanket. The effect of this blanket on retarding damage to the enclosure was estimated by solving the appropriate set of steady state heat transfer differential equations. A variety of heat-in/heat-out boundary conditions, which encompassed constant temperature, constant heat flux, and constant volumetric heat production were used during the investigation. The bus bars are located on the ceiling of the diesel generator rooms. Therefore, the heat transfer calculations assumed that the bus bars would be surrounded by the ceiling jet layer. The bus bars are no more than 15 feet from the top of the diesel generators. The diesel generator exhaust manifold actually has a closest approach which is much less than 15 feet. The only available heat removal mechanism is conduction along the axis of the bus bars to the walls and adjacent rooms. These calculations showed that 1) this heat removal mechanism is not sufficient to prevent calculated damage before calculated extinguishment, and 2) inclusion of the blanket does not change this result.

4.4.3 PUMPS AND COMPRESSORS

1E Panel Room HVAC chillers in 5703/5704, each have a sufficient quantity of oil and are located sufficiently close to each other so that both were assumed to fail for a fire in either one. Each SACS pump room (4307 and 4309) has two SACS pumps located in close proximity. Both pumps in each room were assumed to fail for a fire in either pump.

All CRD pumps, RACS pumps, condensate pumps and feedwater pumps have been assumed to fail for all core damage frequency calculation of unscreened compartments analyzed in the PSA.

4.4.4 OTHERS

All other components were assumed to fail for a fire ignited within it (i.e., it was a fire source). No components other than those discussed above were used as targets. That is, all other components (e.g., cabinets, HVAC equipment, pipes, heat exchangers, valve bodies, etc.) were assumed to be susceptible to the effects of fire only for fires ignited within them.

4.5 FIRE DETECTION AND SUPPRESSION

Paragraphs 4.3 and 4.4 provided a detailed description of fire growth, damage, and suppression methodology and results. The calculated fire suppression results, presented in the previous paragraph, can not be used to justify suppression before damage in any analyzed fire compartment. This paragraph presents the fire suppression formulation in more detail and reviews information about the input quantities associated with fire detection and suppression used in this analysis.

4.5.1 FIRE DETECTION AND SUPPRESSION INPUTS TO FIRE DAMAGE CALCULATIONS

The following formulations were used for the time of fire extinguishment:

$$t_E = t_D + t_S \qquad \text{\{Equation 4\}}$$

$$t_D = -\ln\{1 - \frac{T_R - T_A}{T_G - T_A}\} * \tau + \frac{H_D}{V_G} \qquad \text{\{Equation 5\}}$$

$$t_S = t_A + t_I \qquad \text{\{Equation 6\}}$$

where,

In Equation 4, the time of extinguishment (t_E) equals the detection time (t_D) plus the suppression time (t_S).

In Equation 5, the first term on the right is directly from Equation 21, Attachment 10.4 of FIVE [EPRI, 1993b]and the second term is a delay time for hot gas to flow past the detector. This term was added to the FIVE formulation because it made the non-conservative assumption that the hot gas is instantly at the detector.

T_R is the rated detector actuation temperature, T_A is the room ambient temperature, T_G is the hot gas temperature at the detector, τ is the time constant of the detector, H_D is the distance from the source to the detector, and V_G is the velocity of hot gas flow to and past the detector.

In Equation 6, the suppression time equals the delay time for actuation of the suppression system given an actuation signal (t_A) plus the soak (or immersion) time to accomplish extinguishment given the system actuates (t_I). The travel time for the fire suppression agent to make its way from source to nozzle is assumed to be negligible. Table 4.23, provides typical values used for the above quantities and the references from which they were drawn.

4.5.2 GENERAL PERSPECTIVE ON SUPPRESSION SYSTEMS AT THE HCGS AND THE IPEEE FIRE SUPPRESSION ANALYSIS

The fire walkdown team found the suppression system well designed and the spray heads located, in general, such that they would be effective in suppressing a fire. This means that the suppression systems are well designed to contain and extinguish a fire before it becomes large enough to threaten equipment and personnel away from the immediate vicinity of the ignition point. It is valid, therefore, to use the existence of suppression systems for defining the boundaries of compartments per the FIVE FCIA compartment criteria. No fire suppression system in a nuclear unit, however, is designed to avoid failure of equipment that is the source of the fire. This is particularly true in the case of in-cabinet initiated fires with detectors located on the ceiling of the room.

Fire damage models (e.g., EPRI, 1991a and EPRI, 1993a and b), which are currently available for fire risk analyses, contain unrealistic assumptions and provide results that exaggerate the amount of damage and minimize the time to cause this damage. These models were developed to predict the time to fail overhead cables or horizontally displaced cables in enclosed or ventilated rooms from a single fire source. As indicated earlier, the fire is assumed to reach the peak intensity of a fully developed fire immediately, the heat release rate is assumed by the model to stay at this peak intensity for the duration of the fire, and the heat is deposited at the target cables immediately. In this study, therefore, damage to overhead or nearby cables from a fire source was calculated to occur in a matter of tens of seconds to a couple of minutes. While this time may not be short in comparison with the time for automatic detection of a fire, Sandia fire suppression data [NRC, 1989f] shows that this time is short in comparison with the time required to suppress a fire. Therefore, use of these models in this study could not justify taking credit for automatic fire suppression.

The site fire protection group is a well trained fire fighting force as indicated by fire drill test data. In all drill tests, the first officer arrived on the scene in a matter of a few to several minutes and extinguishment of the fire occurred sometime later. Although first arrival followed by brigade arrival, containment, and extinguishment were all accomplished expeditiously, the fire damage models predicted fire damage to cables in the immediate vicinity of the fire before extinguishment by a fire brigade could possibly occur. Therefore, use of the fire damage and suppression models could not justify taking credit for manual suppression.

4.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSE

Equations 1 and 2 provided the conceptualization of the method used to calculate the scenario by scenario fire core damage frequencies for the HCGS fire PSA. Paragraph 4.1.2.2 described the development of $f(S_{j,k})$, the fire ignition frequency of the jth ignition source having intensity (or fire size) k.

Paragraphs 4.3, 4.4, and 4.5 described development of

- $P(T_l / S_{j,k})$, the probability of damaging target l (T_l) with source j of intensity k without consideration of suppression.
- $P(E_{j,k} < T_l)$, the probability of not extinguishing fire from source $S_{j,k}$ before damage to T_l.

This paragraph reviews the PSA method used and gives an example of the PSA performed for a compartment. It then describes the methods used to designate the initiating event of each scenario and the development of

- $P(I_m / T_l)$, the probability of occurrence of initiating event, I_m, given damage to T_l
- and the development of the conditional core damage probability,
- $P(CD / I_m, T_l)$ = probability of core damage given initiating event, I_m, and damage to T_l, of Equation 1.

The control room analysis also followed Equation 1 but the specific implementation is somewhat different from that used in the other 37 unscreened compartments. The method used to analyze the control room is described in this paragraph. This paragraph also reviews the human error probabilities used in the conditional core damage probability assessment, and discusses the treatment of hot shorts, LOCAs, and interfacing system LOCAs. Finally, a detailed presentation of results of the PSA is provided along with conclusions, insights and sources of uncertainty.

4.6.1 REVIEW AND EXAMPLE OF PSA METHOD

Paragraphs 4.0.1 and 4.3.1 provided an overview of the PSA approach and a step by step procedure for its implementation. This procedure was documented for each unscreened compartment by use of a template, which is presented as Table 4.24. For example in completed template for a diesel generator room is provided as an example in Table 4.25 to demonstrate the depth of analysis performed for each compartment.

4.6.2 DEVELOPMENT OF INITIATING EVENTS FOR SCENARIOS IN UNSCREENED COMPARTMENTS

This paragraph describes the method used to identify initiating events used for the computation of conditional core damage probabilities during the PSA of unscreened compartments and calculation of the quantity,

$P(I_m / T_i)$ = probability of occurrence of initiating event, I_m, given damage to T_i.

of Equation 1. The general approach was similar to that described in Paragraph 4.1.3.2 for the quantitative screening analysis.

As described in that paragraph, the following initiating events were identified as appropriate for fire compartments at the HCGS: MSIV closure, loss of 1E offsite station power, loss of service water/SACS, loss of HVAC, and LOCA owing to inadvertent/stuck open relief valve. The following ground rules were used to designate a scenario's initiating event:

- Each scenario was presumed to cause a reactor trip which was modeled as an MSIV closure, unless a more severe event could be identified. A more severe event means one which would lead to a higher conditional core damage probability given the complement of equipment damaged by the fire scenario.
- If a fire source/target scenario was calculated to damage the 1E 4kV bus bars, a loss of 4kV 1E power (LOOP) was designated.
- If a fire source/target scenario in a cabinet could cause a hot short that might cause offsite power circuit protection relays to trip, then a fraction of the fire frequency was designated as loss of offsite power and the remaining fraction was designated as an MSIV closure.
- If a fire source/target scenario in a cabinet could cause a hot short that might cause loss of service water or SACS, then a fraction of the fire frequency was designated as loss of service water and the remaining fraction was designated as MSIV closure. Note that no single fire associated with the pumps or power cables can cause a loss of service water or SACS.
- If a fire source/target scenario can cause loss of 1E Panel Room or Switchgear Room HVAC, then loss of HVAC was designated.
- If a fire source/target scenario in a cabinet could cause a hot short that might lead to an ADS or SRV actuation, then a fraction of the fire frequency was designated as MSIV closure with Inadvertent/Stuck Open Relief Valve and the remaining fraction was designated as MSIV closure.

The designation of initiating events was reviewed by HCGS operations personnel. Paragraph 4.6.3 describes the development of conditional core damage probabilities for scenarios of each of these initiating events. Table 4.33 identifies the initiating events used for each fire scenario.

The quantity, $P(I_m / T_i)$, of Equation 1, was given a value of unity unless a hot short could produce one of the above initiating events. In that case, it was assigned the probability of the occurrence of the hot short (e.g., 0.3) for the initiating event caused by the hot short, and it was assigned a value of the complement of the hot short probability (e.g., 0.7) for MSIV closure. Thus, the total scenario fire frequency was preserved. The treatment of hot shorts is discussed in Paragraph 4.6.6.

4.6.3 CALCULATION OF CONDITIONAL CORE DAMAGE PROBABILITY FOR PSA

This paragraph discusses the approach used to calculate the conditional core damage probability,

$P(CD / I_m, T_i)$ = probability of core damage given initiating event, I_m, and damage to T_i, of Equation 1.

The approach is similar to that described for the quantitative screening analysis (Paragraph 4.1.3.) and depends on the designated initiating event.

4.6.3.1 MSIV Closure

The event tree of Figure 4.1 with the associated fault trees from the HCGS PSA [PSE&G, 1994b] was used. After determining the extent of equipment and cable damage from the fire growth and propagation studies (Paragraph 4.3, 4.4, and 4.5), input files were constructed that related the damaged equipment to basic events in the PSA model. These basic events were set to a unity probability in the model. The MSIV closure was assumed to be unrecoverable.

4.6.3.2 LOOP

The event tree of Figure 4.2 with the associated fault trees from the HCGS PSA [PSE&G, 1994b] was used. After determining the extent of equipment and cable damage from the fire growth and propagation studies (Paragraph 4.3, 4.4, and 4.5), input files were constructed that related the damaged equipment to basic events in the PSA model. These basic events were set to a unity probability in the model. The loss of offsite power was assumed to cause a turbine trip and be unrecoverable.

4.6.3.3 Loss of Service Water or SACS

This occurred only for cabinet fires in the control room and then only if a hot short occurred. The conditional core damage probability was simply set to unity.

4.6.3.4 Loss of HVAC

Without recovery the conditional core damage probability for this event is unity. However, the internal events PSA [PSE&G, 1994b] determined that unrecoverable damage owing to heat up of 1E panel rooms or switchgear rooms would occur only after 12 to 24 hours. An alternate room cooling procedure [PSE&G, 1994b - Appendix A] was developed for this occurrence. The probability of failure to provide alternate cooling to the 1E panel and switchgear rooms was developed for the internal events study. The conditional core damage probability for loss of HVAC scenarios in the fire PSA was set to the calculated non-recovery probability (HEP = 3E-04) of the internal events study. See Paragraph 4.6.5 for more discussion about recovery and human error probabilities.

4.6.3.5 LOCA Owing to Stuck Open Pressure Relief Valves

This occurred for some of the scenarios in the control room and the lower control equipment room. A sensitivity study determined that a reasonable and conservative method for developing the conditional core damage probability was by using the MSIV closure event tree (Figure 4.1) and setting the failure probability of the SORV event (P2) equal to unity. After determining the extent of additional equipment and cable damage from the fire growth and propagation studies, input files were constructed that related the damaged equipment to basic events in the PSA model. These basic events were set to a unity probability in the model.

4.6.3.6 Control from Remote Shutdown Panel

Some of the scenarios in the control room and control room equipment room mezzanine involved either loss of control in the control room or abandonment of the control room owing to an adverse environment. For these scenarios, the operators would attempt to regain control using the remote shutdown panel [PSE&G, 1995i, PSE&G, 1996g, and PSE&G, 1996r]. For these scenarios, the conditional core damage probability was assigned the probability of unsuccessfully regaining control using the remote shutdown panel. See Paragraph 4.6.5 for more discussion about this.

4.6.3.7 Additional Assumptions

In addition to the conservative selection of initiating events and the conservative modification of the event trees, the following assumptions applied for each run:

- Control rod drive pumps, condensate pumps, and feedwater pumps were assumed to be unavailable throughout the analysis.
- Human recovery actions (in Event Uv in the event trees) were presumed failed, except as indicated above.

4.6.4 ANALYSIS OF THE CONTROL ROOM

The analysis of the control room used a different format than, but the same general approach as, the other unscreened compartments. Fire scenarios were postulated; a set of failed equipment was obtained; an initiating event was designated for each scenario; a fire ignition frequency was calculated; and a conditional core damage probability was determined. The specific procedure, analysis considerations, and assumptions are described below.

The PSE&G policy is that there will always be a minimum of three operators in the Control Room, at least one of which is certified as a Senior Reactor Operator. The shift complement is three SROs, two NCOs, four NEOs and two Radwaste Operators. This is a sufficient number to provide liaison with the HCGS dedicated fire brigade both in and outside the Control Room. The Hope Creek control room features are similar to that of a typical BWR control room for a nuclear power plant in the United States. Fire loading of the control room and the control cabinets are minimal.

The control room contains an inner horseshoe of cabinets with a control console in the middle. In back of the inner horseshoe is an outer horseshoe and in back of this are rows of vertical cabinets. The inner horseshoe has a large connected open area of circuits beneath floor level. All other cabinets are the typical stand alone variety. The insides of the control panels have been visually inspected by the fire IPEEE analysts. In all inspected cases, it was verified that the cables or wires traverse only a short horizontal distance. Typically, they either rise vertically or traverse a few horizontal feet to their termination point. The fire database indicates that in-cabinet fires in the control room tend to be small and damage a limited number of circuits. Therefore, this analysis was performed using two types of fires: small and large.

- In a small fire, ignition occurs inside a cabinet and the damage is limited to the component at which ignition has occurred and other components adjacent to the point of origin. For these fires, the remote shutdown panel is used, if needed, to regain control of failed components.

- A large fire is defined as one in which the operators are forced to leave the control room because of adverse environmental effects caused by the fire. An ignition occurs either outside the cabinets or inside the cabinets and operators fail to extinguish the fire before abandonment becomes necessary. The remote shutdown panel is relied upon for all postulated large fires.

Each scenario is a postulated fire in a single cabinet in the control room. Transient combustible fire scenarios were screened out as being less than 1% of the CDF owing to cabinet fire scenarios.

An initiating event (e.g., MSIV closure, LOOP) is assigned to the system response caused by failure of equipment in the cabinet. Hot shorts were considered in assessing the applicable initiating event. The initiating events are specific to individual cabinets or individual locations within a cabinet.

If the result of postulated failure of the equipment in a cabinet was no more than a reactor trip and the CDF was estimated, by judgment, to be less than 1E-06/yr., the scenario was screened out. This occurred if the worst result of a fire would be a reactor trip without the need to use the remote shutdown panel. Only unscreened scenarios were carried through the quantification. In all, a total of 68 scenarios were identified. For purposes of quantification and documentation, these scenarios were grouped by initiating event.

Frequency of a small fire within a cabinet is computed from the following equation:

$$(1-0.028)*\lambda_{CR}*L_{cabinet\ x} / L_{total}$$

where $L_{cabinet\ x}$ is the length of cabinet x, L_{total} is the total length of the electrical cabinets, and λ_{CR} is the total control room fire frequency of 9.6E-03/yr. One scenario is postulated per cabinet. The frequency of large fire within a cabinet is the complement of the above formula. That is, the numerical factor is 0.028 instead of (1-0.028). How was this factor of 0.028 derived?

Sandia cabinet fire tests [NRC, 1989f] indicate that a room becomes filled with smoke to the point of total loss of visibility of the panels between six and 15 minutes after fire ignition. If a fire is not suppressed and smoke removed within this time interval, then abandonment might become necessary. NSAC-181[EPRI, 1993a] noted that 0 fires out of 12 recorded control room fires caused abandonment. Application of Bayes Theorem using a uniform prior and the evidence of 0 in 12 provided a probability of large fires of 0.028. This factor, therefore, represents the fraction of all fires in a control room, which has the potential to cause abandonment and which was not suppressed by the operators or fire brigade before they abandoned.

The conditional core damage probability was calculated as discussed in the previous paragraph.

The core damage frequency of a fire scenario is as expressed in Equation 1. However, no fire growth and suppression studies were performed for the control room. Thus, the effective equation to determine the core damage frequency is:

$$F_{CR} = \sum_{j}^{sources} \sum_{k}^{intensities} \sum_{l}^{targets} f(S_{j,k}) \sum_{m}^{initiating\,event} P(I_m / T_l) P(CD / I_m, T_l) \qquad \text{\{Equation 7\}}$$

where there are only two intensities, small and large, and the targets are the cabinets themselves. The quantity P(Im/Ti) quantified as the probability of hot shorts or its complement, as was described in Paragraph 4.6.2.

Other considerations relevant to the selected method of control room analysis are as follows:

- There are no overhead cables in the control room.
- There is no kitchen within the control room.
- There are no power circuits (i.e., above 125 volts nominal or high amperage circuits) in the control room for plant and safety equipment. All circuits inside the control room are at or below 120 Volts nominal ac or 125 volts nominal dc except neutron monitoring which has high voltage (600v) in control room back panels, power supply to the detectors.
- Smoking is not allowed inside the control room.
- There are three entry/exit points for the control room. The two that are located on the east side are under normal usage.
- The remote shutdown panel room is about 50 feet from a normal exit door down a wide well lit corridor. Only those personnel with the proper security clearance can enter the remote shutdown panel room.
- There is a dedicated HVAC system for this room. The HVAC system can be reconfigured for a one pass suction from outside and venting to the outside. This mode can be used to vent off smoke in the area.
- Smoke detectors are located inside the control cabinets and under the suspended ceiling.

- Fire protection for this room is provided by hand held halon extinguishers, and a manually activated fixed halon system for inside the control boards. Operators need to connect a hose (curled and hung from a receptacle inside the control room) to the injection ports of the control cabinet and then activate the halon system.

Assumptions include the following:

- All cabinets are similar with respect to the features that affect the possibility of fire ignition, propagation and severity of the fire itself (i.e., flame temperature, amount of smoke and height of flames), and detection and suppression of fire. This assumption is based on the fact that all cabinets contain switches, readouts, controls, chart recorders and other similar devices. The variations among the cabinets in terms of number and types of devices is deemed to be minimal with regard to fire severity issues.
- All cabinets, regardless of their height, have the same likelihood of fire ignition per linear foot (lengthwise).
- Given a fire in the control room, it is conservatively assumed that the operators will initiate a plant trip regardless of the level of damage. It is assumed that MSIV closure cannot be prevented by a fire in the control room after a reactor trip.
- The computer room and therefore the computer are not considered as part of the control room.
- The approximate position of various circuits can be known in this room by simply inspecting the control board.

4.6.5 HUMAN ERROR PROBABILITY REVIEW

The CCDP calculations of the Fire PSA took advantage of only two recovery actions: 1) recovery of alternate ventilation following a loss of 1E Panel Room HVAC, and 2) control of the plant from the remote shutdown panel following a fire that compromises the ability of operators to completely control the plant from the control room. Room heat-up calculations, performed for the internal events PSA, indicated that 12 hours are available to perform recovery of alternate ventilation. Fires that could potentially cause loss of 1E Panel Room HVAC could occur in rooms 5604, 5620 and 5703 on the 163' and 178' elevations of the Control/Diesel Building. These fires would affect 1E control equipment in rooms, other than these rooms, on elevations 163' and lower. The actions involved in alternate ventilation are the opening of doors and installation of fans creating ventilation paths to/from rooms that are unaffected by the loss of ventilation. The long time scale available for the required action, coupled with the non-coincident location of fire and actions, leads to the conclusion that the recovery action is valid for fire scenarios at the same probability as in the internal events case.

The recovery of plant control using the remote shutdown panel was analyzed for the HCGS seismic IPEEE. The Accident Sequence Evaluation Program method [NRC, 1987b] was used to estimate the HEP associated with failure to act given successful diagnosis. The derived HEP, which was developed independent of seismic initiating event, was 0.06. This assumed, however, that the HEP for diagnosis was zero. For fire scenarios, the need to use the RSP derives from 1) fires that cause a loss of control from the control room owing to failure of cables and control/switching equipment, and 2) fires in the control room that cause abandonment owing to a severe environment (e.g., low visibility or inability to breath). In either case, the ability to correctly diagnose the need to use the RSP has a high probability.

Control room fire scenarios may cause reactor trip transients or more severe initiating events such as loss of service water, inadvertent/stuck open SRV, and loss of offsite power. The actions to cope with scenarios, other than those initiated by reactor trip, are somewhat more complex than those analyzed for the seismic sequences in that operators would have more local manual actions than for a reactor trip transient. These more complex scenarios represent about 10% of the total frequency of control room abandonment. Therefore, for 90% of control room abandonment scenarios, the HEP developed for seismic events is reasonable. It was judged that a reasonable way to assess a conditional probability of failure to control the plant following control room abandonment for scenarios other than reactor trip would be to increase the HEP to 0.1. Therefore, an HEP of 0.06 was used for failure to control the plant for scenarios in which the remote shutdown panel was assumed following an assumed MSIV closure reactor trip, and an HEP of 0.10 was used for failure to control the plant for scenarios in which the remote shutdown panel and associated local manual operations was assumed following all other initiating events (e.g., loss of SWS, LOOP, loss of HVAC, SORV).

4.6.6 TREATMENT OF HOT SHORTS, LOCAs AND INTERFACING SYSTEM LOCAs

Typically, fire induced short circuits are short lived, becoming open circuits. Most shorts are to ground. Those that persist long enough usually cause circuit protection features (e.g., fuses, and circuit breakers,) to actuate. Such is the case for all power cables. Another kind of short is a "hot" short in which control wiring or contacts which should be insulated from one another come in contact in a way that allows power to the controlled component. For example, this may occur if two wires, from the opposite poles of the switch, contact each other either directly or indirectly. Such shorts sometimes have the capability of creating an inadvertent signal in equipment which would either initiate an unwanted change of state (e.g., starting or stopping a pump, or opening a closed valve) or in the case of certain closed motor operated valves, unwanted motor operation. Control cables or wires that run through the plant and terminate at the equipment to be controlled do not cause such "hot" shorts. Only cables or wires within the logic control box itself are capable of causing such hot shorts. These hot shorts cause damage only if circuit protection devices do not actuate in a reasonable time.

This assessment considered the possibility of hot shorts for each scenario and commented on the possibility under the heading Initiating Event(s) within the Fire Scenario Analysis worksheets. Only the control room, lower control equipment room, and switchyard blockhouse were found susceptible to hot short actuation of equipment. The occurrence of hot shorts might cause an I/SORV (LOCA), LOOP, or Loss of SWS/SACS. These effects were considered during the calculation of core damage frequency

The experience with such hot shorts is limited. However, abnormal conditions have been observed in electrical fires. For example, in the Browns Ferry fire, operators observed brightening of the indication lights on the control board. Inadvertent operation of equipment has also been observed but has not been reported rigorously. Therefore, the fraction of fire events in control cables that may lead to a hot short cannot be determined from statistical evidence. The current state of fire PSA treats this quantity simply by judgment. Typically a number on the order of one-tenth is considered reasonably conservative. This assessment used a value of 30%. That is given a fire scenario in which a hot short might cause unwanted effects, the likelihood of those effects is 30% of the likelihood of the fire scenario. The remaining 70% of the fire scenario is treated as if hot short did not occur.

Using this highly conservative value of the conditional probability of hot shorts, the total core damage frequency associated with hot shorts was found to be approximately 7E-06/yr.

This assessment considered the possibility of fire induced LOCAs for each scenario and commented on the possibility under the heading Initiating Event(s) within the Fire Scenario Analysis worksheets (e.g., Tables 4.24 and 4.25). Fire induced LOCAs were found to occur only because of hot shorts, as described above, in cabinets that contain control wiring for SRVs or ADS. This can occur only in the control room and lower control equipment room. Using the highly conservative value of the conditional probability of hot shorts, the total core damage frequency associated with fire induced LOCAs was found to be approximately 4E-07/yr.

The possibility of transient induced LOCAs for scenarios in which fire induced LOCAs did not occur (e.g., MSIV closure followed by a random SORV) was also included in the PSA calculation. Occurrence of transient induced LOCAs are included in the assessment of the CCDPs in every fire scenario by virtue of the use of the entire IPE event tree and fault tree model for these calculations.

An analysis of the interfacing high to low pressure systems was performed for the HCGS PSA (PSE&G, 1994b). This analysis was reviewed for applicability to fire scenarios. As shown by the results of this review, presented in Table 4.26, no high to low pressure interface is susceptible to fire scenarios. With one exception, this is because all boundaries are protected by at lease two diverse, closed isolation valves, one of which is a check valve or stop check valve. Even if a sustained hot short opened an MOV, the check valves are not susceptible to opening by fire scenarios. The one exception to this is the RHR shutdown cooling suction lines which are isolated by two closed MOVs. The shutdown cooling suction valve (BCHVF008) is disabled at the circuit breaker by a key switch to prevent inadvertant opening during fires.

4.6.7 RESULTS, INSIGHTS, AND SOURCES OF UNCERTAINTY

4.6.7.1 Core Damage Frequency and Overall Conclusions

As this paragraph demonstrates, this study has met the IPEEE objectives of GL 88-20, Supplement 4, as stated in Paragraph 4.0.1 of this report. A risk assessment was performed with a conservative set of methods and assumptions which were applied uniformly across the unscreened rooms. Uniform application of assumption and methods was done for the purpose of providing an even-handed picture, among compartments, for 1) an appreciation of and insights into potential fire induced core damage sequences; 2) an understanding of the most likely core damage sequences; and 3) an understanding of the overall likelihood of core damage. The conservative set of assumptions and methods have combined to provide an overestimate of the core damage frequency associated with fire induced sequences at the HCGS.

As this paragraph demonstrates, there are no areas of the plant for which corrective actions should be taken with respect to reduction in the likelihood or severity of fire induced core damage scenarios. This conclusion has been arrived at on the basis of a detailed, conservative probabilistic fire risk analysis.

Insights are gained by the ability to analyze and view the results at various levels of detail. The level of least detail is the total fire induced core damage frequency of the 38 unscreened compartments as shown in Table 4.27.

4.6.7.2 Distribution of Core Damage Frequency and Insights

The next level shows the distribution of this CDF over the buildings included in the scope of this analysis. Table 4.28 shows this distribution.

The Control/Diesel Building, which houses the control area and the diesel generators, is the most significant building contributing 86% of the fire induced CDF. This was expected because of the good separation of equipment in the Reactor Building and the lack of safety related equipment in the other buildings. Typically, the fire risk is dominated by rooms or areas in which there is a confluence of equipment and/or cables from different electrical divisions. This occurs in the Control/Diesel Building at the HCGS, particularly in the cable spreading room, lower control equipment room, control equipment room mezzanine, upper control equipment room, diesel generator rooms, electrical access rooms, and control room.

The next level of detail is the breakdown of the CDF into the analyzed compartments. This provides the locations in the plant that have the highest risk contribution with respect to fire induced core damage scenarios. Shown in Table 4.29 are the 16 compartments with a calculated CDF greater than or equal to 10^{-6} per year along with the initiating events that were assumed to be able to occur from fires in these compartments. These 16 compartments account for 95% of the total CDF.

The Control room, Diesel-Generator rooms (taken together), Electrical Access Area 5339, and the 1E Switchgear Rooms account for 70% of the fire induced CDF of the unscreened rooms. Excluding these rooms leaves a CDF of 2.5E-05/yr. for the remaining unscreened compartments.

The study took no credit for conduit and enclosed cable trays in the fire damage calculations. This allowed the inherent divisional separation of the plant to be displayed. A key observation during the walkdown was that equipment of different channels were often divided into different rooms (e.g., inverter rooms, battery rooms, RHR pump room, CS pump rooms, HPCI room, RCIC room, switchgear rooms, diesel-generator rooms). The compartments that emerged as the most important are those in which 1) multiple channels are found in close proximity so they can be affected by the same fire (e.g., control room, electrical access rooms, control equipment rooms, MCC areas, SACS equipment room), or 2) compartments in which a LOOP can occur (e.g., diesel generator rooms, electrical access area 5339, access and unload area in the turbine building).

At the next level of detail of the results is the distribution of core damage frequency with respect to:

- fire ignition source
- initiating event caused by the fire
- affected electrical channel or division

Tables 4.30, 4.31 and 4.32 provide these results.

Cabinets and diesel generators are the most important ignition sources at the HCGS. The contribution of heaters is due solely to the close proximity of heaters to division II cables in electrical access area 5401.

Even though transient combustibles, of sufficient quantity to damage cables, were not found in any compartment at the HCGS, a thorough transient combustible analysis was performed. Each compartment included consideration of transient combustibles. This analysis assumed that transient combustibles could be located anywhere in the plant. The fire damage models (described in Paragraph 4.3), which provided a very conservative method for damage assessment, were used to calculate the maximum area over which transient combustible could damage targets. This model was used with a conservative estimate of the amount of transient combustible liquids which could be exposed and ignited. The analysis used a five gallon unconfined liquid pool fire, to maximize the heat release rate (BTU/sec), even though transient quantities this large were not found during the walkdown. Thus, the transient combustible CDF shown in Table 4.30 is an overestimate because the fire damage models exaggerated the area over which such fires could damage cables and because the input to the model exaggerated the severity of damage.

MSIV closure and loss of 4kV offsite station power are the two most risk significant initiating events. The significance of MSIV closure stems from the assumption that each room would be analyzed as if MSIV closure were the minimum severity initiating event. Its apparent significance, therefore, is an artifact of an assumption and should be viewed with skepticism. The significance of loss of 4kV offsite power is largely associated with the diesel generator rooms and reflects an unusual aspect of the HCGS in which both sets of 4kV offsite power bus bars run through all four diesel generator rooms.

None of the electrical channels or divisions of the HCGS appear particularly risk significant or vulnerable to fires. The CDF associated with channel A (or B) is primarily due to the channel A (or B) switchgear room and arises from the importance of these channels in providing power to safety related equipment. It is not a result of a vulnerability of channel A (or B) cables or equipment to fires.

NUREG - 1407 [NRC, 1991b] requests results on a scenario by scenario basis. This is provided in Table 4.33.

4.6.7.3 Discussion of Risk Significant Fire Compartments

Control Room

Cabinet initiated fires in the control room were divided into large and small. Small fires were those exemplified in the fire database, which either self-extinguished or were quickly extinguished by operators before damage spread to circuits beyond those in the point of origin. Plant control from the control room was not lost in any of the fires found in the database. Large fires were those postulated to be unsuppressed before they caused abandonment of the control room owing to adverse environmental conditions. The data, showing zero in twelve such control room cabinet fires (EPRI, 1973a), was analyzed using Bayes Theorem with a uniform prior and yielded a probability of 0.028 given a control room fire. This probability implicitly includes the contribution of operator suppression, as found in the database, because only fires that are not expeditiously suppressed can cause abandonment. The value of 0.028 falls within the spread of values (0.10 to 0.01) commonly assumed for operator failure to suppress a fire. The value used in this study, however, was not an assumption. It was derived from the database.

Even though the frequency of large fires is small (on the order of 10^{-4}/yr.), these fires were calculated to dominate the fire risk of the control room. This is because the conditional core damage probability includes the human error probability for failing to successfully gain control of the plant from the remote shutdown panel. This human error probability causes the large fire CCDPs used in this analysis to be, generally, about one to two orders of magnitude larger than the CCDPs used for small fires.

Control room fire scenarios in all plants are similar and are dominated by scenarios that postulate abandonment of the control room and subsequently regaining of control from the remote shutdown panel. Control rooms are typically one of the top five risk significant rooms in a unit. The HCGS calculated value of 2.5E-05/yr. is typical of values found for other units.

Diesel-Generator Rooms

Table 4.25 provides the detailed assessment of a diesel generator room. The assessment of the other three diesel generator rooms is similar. The analysis was divided into small (type I) fires and large (type 2) fires. The break point between large and small fires was defined as the ability to affect the 4kV bus bars on the ceiling. The fire growth calculations indicate that a liquid pool fire of about 11 gallons or less would not damage the 4kV bus bars.

The diesel generator rooms are equipped with CO_2 total flooding systems. They emerge as important fire risk locations at the HCGS because of an unusual configuration in which both sets of Class 1E 4kV bus bars run along the ceiling of these rooms. A loss of offsite 4kV power was assumed for fires large enough to be calculated as causing a short circuit of the bus bars. The bus bars, which are Aluminum are surrounded by an Aluminum duct/enclosure. The assumed damage mechanism was a fire that melted the Aluminum enclosure such that molten or softened Aluminum of the enclosure contacted the bus bars causing a short circuit. Because both sets of bus bars run in relatively close proximity to each other, at the diesel exhaust manifold end of the room , the loss of both bus bars was assumed to occur simultaneously.

The 1994 Sandia fire database (Sandia, 1994a) shows 27 diesel-generator fires. In one of these the automatic suppression system failed to operate and the fire burned for 25 minutes before being extinguished (Grand Gulf, 1982). This is the only fire that might potentially have been severe enough to cause damage to the bus bars, if it had occurred in a HCGS diesel-generator room. All other fires were small and self-extinguished, were extinguished by automatic suppression, or where manually extinguished (usually with portable extinguishers). Because the database does not contain enough information to make the determination of whether it would have caused loss of the 4kV bus bars, a Bayesian analysis of the data under two hypotheses was performed: 1) the fire would cause damage to the bus bars and 2) the fire would not cause damage to the bus bars. Each hypothesis was given a 50% probability of being the correct one. The analysis yields a probability of a fire sufficiently large to damage the bus bars of 0.025. This value implicitly includes failure of suppression, as recorded in the database, because only a fire that is not expeditiously suppressed can grow large enough to damage the bus bars. For comparison, the probability of failure of CO_2 suppression systems is typically about 0.04 [EPRI, 1993b - Table 2]. Even though the frequency of a large fire is low (approximately 2E-04/yr.), the loss of 4kV power scenario represents more than 95% of the calculated CDF of the diesel-generator rooms. This is because such a fire was also assumed to disable the diesel-generator which initiated the fire. The CCDP is dominated by common cause failure of the remaining diesel-generators.

<u>Electrical Access Room 5339</u>

The risk significant scenario of Room 5339 is one in which burning liquid fuel from a diesel-generator room fire leaks under the door separating the diesel-generator room from Room 5339. This area contains Division I cables, both sets of 1E 4kV bus bars, and the control power supply cables for diesel generators A and C. The dominant scenario is a large fire, from one of the four diesel-generator rooms, which spreads into this room causing loss of 4kV station power, loss of cables of Division I, and loss of the ability to start diesel-generators A and C. Calculation of the fire frequency for this scenario was taken to be 50% of the frequency of large fires calculated for all of the diesel generator rooms. That is, leakage under the door was assumed to occur for half of the scenarios in which a large pool might collect in any of the diesel generator rooms.

Switchgear Rooms

The Channel A and B switchgear rooms (5416 and 5412, respectively) emerge as important because these are the most important channels with respect to providing electrical power to safety related equipment. This analysis assumed, as is usually done, that any cabinet fire in these rooms can cause loss of a channel. Relaxation of this assumption would require detailed knowledge of cable end points in these rooms. These rooms do not have automatic suppression systems.

CRD Pump Room

Room 4202 is the CRD pump area and it contains Division II cables passing over cabinets. The fire damage calculations indicate that cables passing directly over cabinets may be damaged by fully developed cabinet fires using peak heat release rates from Sandia cabinet fire tests [NRC, 1989f]. Therefore, Division II cables were calculated as failing with the frequency of cabinet fires in this compartment. This room does not contain automatic suppression. A complete failure of Division II was assumed. Relaxation of this assumption would require detailed knowledge of the cable end points passing within and through this room.

4.6.7.4 Sources of Uncertainty

Per NUREG-1407, this paragraph discusses uncertainties associated with fire ignition data and the estimation of core damage frequency.

Sources of Uncertainty Regarding Fire Ignition Data

The FIVE database of fire ignition frequencies is an interpretation of the EPRI fire database. Use of the FIVE database and method for a specific plant introduces the following sources of uncertainty and variability:

- The information presented in the FIVE database represents generic "average" values from the EPRI database. When applying such a database to a specific unit, plant-to-plant variability associated with fire protection features, plant configuration, plant walkdowns/inspections, operational practices, and equipment/cable qualifications are not included.

- Construction of a fire database 1) is, of necessity, subject to interpretation of the raw fire incident data, 2) is constrained by the raw data sources available, and 3) is constructed over a specific time interval. For example, the EPRI fire database which served as the basis for the FIVE method has over 800 events over a period from 1965 to 1988. The Sandia database [NRC, 1989f], as updated in 1994 [Sandia, 1994a], has a different number of fire incidents over a different time period and uses different data sources.
- The configuration of the HCGS is different from the configuration envisioned by the creators of the FIVE database. For example, the equipment usually associated with a reactor building in a BWR is distributed among the reactor and control/diesel building at the HCGS. Therefore, the association of FIVE database categories with the HCGS compartments is a matter of judgment.

An uncertainty is also introduced because it is not feasible to assure that the compartment inventory of equipment and cables was completely accurate.

<u>Sources of Uncertainty Regarding Estimation of Core Damage Frequency</u>

The largest sources of uncertainty arise from analytical assumptions and the fire damage model and calculations. All such assumptions tended to overestimate the calculated fire CDF. Conservative aspects of the fire damage model have been thoroughly discussed in previous paragraphs and will not be repeated here. Other than these, it is judged that the most significant of the analytical assumptions are as follows:

- Fires in all compartments were assumed to induce a reactor trip. This reactor trip was modeled as an MSIV closure, unless a more severe initiating event was identified.
- If a cable within an electrical channel in a compartment was found to exceed the cable damage criterion, the entire channel was assumed to be disabled. No credit was taken for protection owing to conduits or enclosed cable trays.
- Thirty percent of fires in cabinets in the control room, lower control equipment room, and switchyard blockhouse were assumed to cause hot shorts.
- All large fires in the diesel generator room were assumed to cause a loss of all 4kV 1E power.

<u>Suppression System Effectiveness Sensitivity Study</u>

Of the top 16 rooms, the diesel generator rooms, electrical access areas (rooms 4301, 5339 and 5401), and turbine building unloading area (room 1315) have suppression systems. As discussed previously, the fire damage modeling did not predict that suppression would occur in time to prevent damaging cables above (and in the near vicinity) of fire ignition sources. In recognition of the conservatism of these calculations, a sensitivity study was performed which assumed that suppression could occur in time to prevent damage. This assumption has the effect of reducing the core damage frequency of the following compartments: 4301, 5401, and 1315. The CDF of these rooms is reduced by a factor of 20 because the CDF is multiplied by the unavailability of preaction sprinkler systems, which is 0.05 (EPRI, 1993b - Table 10.2). This would reduce the total fire induced CDF to 7.6E-05/yr. from 8.1E-05/yr. or about 6%. In the other compartments which contain suppression systems (diesel generator rooms and 5339), the Bayesian data analysis approach discussed previously implicitly includes the non-suppression probability. Therefore, these rooms were not included in the sensitivity study.

4.7 CONTAINMENT PERFORMANCE REVIEW

4.7.1 BACKGROUND

Supplement 4 of Generic Letter 88-20 [NRC, 1991a - Appendix 2] states that the evaluation of the containment performance of external events should be directed toward a systematic examination 1) to determine the existence of containment failure modes owing to fire induced sequences that are distinctly different from sequences found in the IPE internal events evaluation and 2) to determine if fires can contribute significantly to direct functional failure of the containment which is not a result of a core damage sequence. The generic letter further suggests that the information developed for the IPEEE should be used to:

1. Identify mechanisms that could lead to containment bypass.
2. Identify mechanisms that could cause containment isolation failure.
3. Assess the effect of fires on containment heat removal and pressure control systems to ascertain if the effects of fire induced sequences are significantly different from those evaluated for internal event sequences.

Should such sequences be discovered, then a Level II analysis, similar to that performed in the IPE, should be performed and the results reported to the NRC [NRC, 1991b - Section 4 and Appendix C].

4.7.2 METHOD AND RESULTS

The occurrence of interfacing system LOCAs which is the predominant form of bypass events for the HCGS was discussed in Paragraph 4.6.6. Bypass events owing to fires were not found to be susceptible to fire events. Table 4.34 summarizes a detailed systematic evaluation of the potential for bypass, isolation failure, direct containment integrity failure, and containment system degradation or failure for each of the 38 unscreened compartments for which a detailed fire PSA was performed [PSE&G, 1997a]. The detailed evaluation made reference to the list of equipment, cabinets, and cables compiled for the fire PSA as well as the effects of failure of those components, including the effects of hot shorts.

4.7.3 CONCLUSION

The conclusion of this evaluation is that there are no fire induced containment failure modes that are significantly different from those treated in the HCGS IPE [PSE&G, 1994a]. Therefore, no further containment performance analysis is needed.

4.8 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

Per Generic Letter 88-20 Supplement 4 (NRC, 1991a) and NUREG-1407 (NRC, 1991b), this paragraph discusses the Fire Risk Scoping Study Issues [NRC, 1989b]. These issues have been categorized as follows:

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

This paragraph describes the basis, assumptions, findings, and conclusions with respect to these issues. The outline of this paragraph follows the checklist found in FIVE [EPRI, 1993b - Attachment 10.5]. The guidance in Section 7 of that document was also used. This paragraph is a summary of a more detailed evaluation [PSE&G, 1997c].

4.8.1 SEISMIC /FIRE INTERACTIONS

The seismic/fire interaction involves three concerns:

- The potential for seismically, induced fires
- The potential for seismically-induced actuation of fire suppression systems.
- The potential for seismically-induced degradation of fire suppression systems.

4.8.1.1 Seismically Induced Fires

This issue concerns 1) the potential leakage or rupture of flammable/combustible liquid or gas lines, tanks, or containers during a seismic event and 2) unanchored non-1E cabinets in close proximity to safety related equipment or cabinets. The equipment that was addressed in this investigation include:

- Hydrogen piping and storage tanks
- Emergency diesel generator fuel oil piping, day tanks, and storage tanks
- Turbine lubricating oil storage tanks and associated piping
- Turbine generator (Hydrogen envelope)
- Hydrogen seal oil unit and associated piping and tanks
- Other sources of flammable or combustible liquids and gases with associated lines, tanks and containers (e.g., reactor recirculation pumps, waste oil drain tanks, and pump lubricating oil sight glasses)
- Unanchored non-1E cabinets in close proximity to 1E cabinets or safety related equipment

The specific location of flammable/combustible liquid and gas containing equipment is delineated in PSE&G, 1995f - Paragraph 9.5 - Table 9.5-3. During the seismic walkdown, the team focused on equipment whose failure could be a fire source that would damage equipment important to seismic safety. No credible failures were found. The emergency diesel generator fuel oil day tanks and storage tanks were found to be seismically rugged. All piping associated with the above equipment was found to be sufficiently seismically rugged not to pose a significant fire risk.

Both 1E and non-1E cabinet anchorages were included in the seismic walkdown and assessment. All non-1E cabinet anchorages were either screened out or found to have median capacities in excess of 1.5g. Therefore, seismic interactions of non-1E cabinets and 1E equipment is not a significant fire risk.

4.8.1.2 Seismic Actuation of Fire Suppression Systems

The HCGS uses preaction water sprinklers, $C0_2$ deluge systems and backup manually actuated water deluge systems for safety related areas in which automatic fire suppression is provided. Preaction sprinklers require two independent failures to allow water discharge.

The seismic walkdown examined the potential for pipe or sprinkler head failure. Fire water piping was adequately supported such that there are no potential seismically induced systems interactions resulting in release of fire water or CO_2.

The HCGS-UFSAR [PSE&G, 1995f - Paragraph 9.5.1.1.4] notes the following:

- All fire protection system components are designed so that a failure or inadvertent operation does not result in loss of function of plant systems important to safety.
- Piping located over safety-related equipment is supported such that it will not fail during a safe shutdown earthquake.
- Water system pressure is low enough so that pipe whip protection is not required.
- The pressurized ponion of the piping of CO_2 systems is located outdoors for the tank serving safety-related areas.
- Automatic CO_2 fire suppression systems serving safety-related equipment have seismically qualified components to avoid inadvertent discharge during a seismic event.

A low ruggedness relay evaluation was performed for the HCGS [PSE&G, 1996b]. A total of 12 panels were identified that contained low ruggedness relays. In addition another 38 miscellaneous low ruggedness relays were identified. None of these were in the fire protection or detection systems.

It is concluded that seismic actuation of fire suppression systems does not pose a significant risk of flood or a significant likelihood of disabling safety related equipment.

4.8.1.3 Seismic Degradation of Fire Suppression Systems

As noted in the previous paragraph, the piping, sprinkler heads, relays and other components of the preaction water suppression and CO_2 suppression equipment are seismically robust.

However, the fire water pumps are located in a Fire Water Pump House which is a block wall structure that is not seismically qualified. The fire water tanks are located outside of this structure and are not seismically qualified. The limiting seismic failure of the fire water system is failure of the tanks. The seismic core damage frequency assessments did not take credit for the fire water system because of its perceived lack of robustness against earthquakes. The fire core damage frequency assessment also did not take credit for fire water suppression systems.

It is concluded that the unavailability of fire water after an earthquake is the principle mode of seismically induced fire suppression system degradation.

4.8.2 FIRE BARRIER QUALIFICATIONS

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

4.8.2.1 Fire Barriers

Fire barriers and penetration seals are subject to periodic surveillance in accordance with the fire surveillance and periodic test program, Procedure ND.FP-AP.ZZ-0005(Q) [PSE&G, 1995j]. Periodic surveillance of fire-rated assemblies is conducted on 18-month intervals or less. For fire barrier penetration seals, the surveillance is conducted on a ten percent sampling basis. The implementation of the required surveillance is specified in detail by procedures.

The HCGS Fire Protection Program provides adequate fire barrier and penetration seal control measures.

4.8.2.2 Fire Doors

Fire doors are included in the ND.FP-AP.ZZ-0005(Q) fire protection surveillance requirements [PSE&G, 1995j], and are subject to inspection on a six-month interval. The implementation of the required surveillance is provided by inspection procedures HC.FP-SV.ZZ-0027(F) [PSE&G, 1996h] and HC.FP-SV.ZZ-0058(F) [PSE&G, 1996q].

The HCGS Fire Protection Program provides adequate door control measures.

4.8.2.3 Penetration Seal Assemblies

The specific surveillance criteria and methodology for penetration seal assemblies are presented in procedure HC.FP-SV.ZZ-0026(F) [PSE&G, 1996i]. The programmatic standard for penetration seal assemblies is HC.DE-PS.ZZ-0021(F) [PSE&G, 1994k] which establishes the acceptance criteria.

The FIVE Methodology identifies three NRC I&E Notices which have specific applicability to fire barrier penetration seals:

- 88-56: Potential Problems with Silicone Foam Fire Barrier Penetration Seals (NRC, 1988c).
- 88-04 Supplement 1: Inadequate Qualification and Documentation of Fire Barrier Penetration Seals (NRC, 1988b).
- 88-04: Inadequate Qualification and Documentation of Fire Barrier Penetration Seals (NRC, 1988a).

The concerns raised by these Information Notices have been addressed in internal memos. These memos indicate that the concerns and issues identified in the Information Notices are either adequately controlled or do not apply to the Penetration Seals used at the Hope Creek Generating Station.

4.8.2.4 Fire Dampers

The fire damper inspection and maintenance program is considered in conjunction with overall fire barrier and penetration seal surveillance. The specific surveillance criteria and methodology are presented in the following procedures: HC.FP-SV.ZZ-0028(F) [PSE&G, 1995l] and HC.FP-ST.ZZ-0031(F) [PSE&G, 1995m].

The HCGS Fire Protection Program provides adequate HVAC fire damper control measures.

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

- 89-52: Potential Fire Damper Operational Problems (NRC, 1989e).
- 83-69: Improperly Installed Fire Dampers at Nuclear Power Plants (NRC, 1983b).

IN 89-52 identified potential closing problems with curtain type fire dampers under system ventilation air flow conditions. IN 83-69 identified three specific fire damper installation deficiencies to be addressed including missing fire dampers in HVAC ducts where the ducts penetrate fire barriers, dampers with an improper rating relative to the barrier rating in which they were installed, and dampers installed outside the fire barrier.

The concerns identified in these two INs were addressed in a major fire damper improvement project at the Hope Creek Generating Station in 1985, prior to startup. Although IN 89-52 did not exist in 1985, it was a direct result of a 10CFR Part 21 notification to the NRC in 1984 by Ruskin (manufacturer of the fire dampers) which also prompted the fire damper project at the HCGS in 1985.

Therefore, the issues and concerns identified in these two INs have been adequately addressed. A detailed discussion of the fire damper issues and how they were resolved may be found in the HCGS UFSAR [PSE&G, 1995f - Paragraph 9.5.1.1.15].

4.8.3 MANUAL FIRE FIGHTING EFFECTIVENESS

This issue is focused on the adequacy of training and preparedness of the Hope Creek Generating Station Fire Brigade, and on the general orientation of appropriate plant personnel to fire response requirements.

4.8.3.1 Reporting Fires

As described by PSE&G, 1993a, a support personnel training program is in place, to indoctrinate selected personnel, as appropriate, in topics associated with the Hope Creek Fire Protection Program. This training includes instruction in the proper selection and use of portable fire extinguishers.

The monthly surveillance of portable fire extinguishers distributed throughout the HCGS is implemented by PSE&G, 1996d. The specific type and location of each fire extinguisher is controlled as defined in PSE&G 1996f.

The hot work permit process requires a portable fire extinguisher and a trained fire watch at the work site.

The procedure for reporting of fires and initial response is addressed in the Nuclear Department Operational Fire Protection Program, PSE&G, 1996f. The responsibilities of fire response are applicable to all plant personnel. Notification requirements and processes are covered in site General Employee Training.

PSE&G, 1996f outlines the use of the plant telephone system or the plant page system for reporting of fires to the Control Room. Fire notification may also be accomplished through the use of local manual pull stations.

It is concluded that the established HCGS programmatic measures for training, equipment, and communication are adequate.

4.8.3.2 Fire Brigade

As stipulated in PSE&G, 1995h, a fire brigade of at least six members is maintained on site at all times. PSE&G, 1995h requires at least two members of the fire brigade on duty each shift, and that the fire brigade leader be trained in the plant's safety systems. In addition, PSE&G, 1995h requires that the fire brigade leader be competent to assess the potential safety consequences of a fire and advise control room personnel concerning these consequences.

PSE&G, 1995h requires that brigade members satisfactorily complete an annual physical examination.

PSE&G, 1993b defines the equipment maintained within the plant for dedicated fire brigade use. This procedure also provides a surveillance mechanism, to ensure the availability of the minimum required equipment at all times. The equipment includes:

- Turnout gear, including coats, helmets, gloves, and boots.
- Self-contained breathing apparatus (air packs), a supply of spare bottles, and a recharging station.
- Portable lanterns/flashlights.
- Smoke ejectors with flexible ducts.
- Portable fire extinguishers - throughout the station.
- Portable radios - stored in the Control Room and Fire House.

This equipment is augmented by vehicle-based equipment associated with the fire fighting and rescue vehicles operated by the station fire department. This supplementary equipment is identified and tracked through PSE&G, 1996e.

It is concluded that the established HCGS programmatic measures for fire brigade staffing and equipment are adequate.

4.8.3.3 Fire Brigade Training

The fire brigade classroom training program, as described in PSE&G, 1993a, provides the following elements:

- Indoctrination in the plant fire fighting plan and identification of individual responsibilities of fire brigade members.
- Identification of the fire hazards and associated types of fires that may occur in the plant.

- Identification of the location of fire fighting equipment for each fire area, and familiarization with the layout of the plant, including access and egress routes.
- The proper use of available fire fighting equipment, and the correct method of fighting each type of fire. The types of fires covered include electrical fires, fires in cable trays, hydrogen fires, flammable liquid fires, waste/debris fires, and record file fires.
- The proper use of communication, lighting, ventilation, and emergency breathing equipment.
- The proper method for fighting fires inside buildings and tunnels.
- Review of the latest plant modifications and changes in pre-fire plans.

The fire brigade hands-on training program [PSE&G, 1993a] provides the following elements:

- The proper method for fighting various types of fires of "similar magnitude, complexity, and difficulty" as those which could occur in a nuclear power plant.
- Experience in actual fire extinguishment and in the use of emergency breathing apparatus under strenuous conditions.
- Practice sessions held at recognized training facilities, at regular intervals, not to exceed one year, for each fire brigade member.

The fire brigade drill program described in references PSE&G, 1993a and PSE&G , 1996c, provides the following elements:

- Drills are performed in the plant so that the fire brigade personnel can practice as a team at the actual site. PSE&G, 1996c - Paragraph 6.1 indicates that each shift practices as a unit.
- Fire drills are held at regular intervals, not to exceed three months, for each fire brigade shift.
- Each fire brigade shift participates in at least one unannounced fire drill each year.
- At least one fire drill per year is performed on a backshift for each shift fire brigade.
- Fire drills are pre-planned to establish training objectives. Each fire drill is critiqued to determine how well the training objectives were met. On a

triennial basis, fire drills are critiqued by qualified individuals independent of PSE&G's staff; in accordance with PSE&G, 1996c.

- Pre-Fire Plans have been developed for all plant areas. These Pre-Fire Plans are used as an integral part of fire brigade training. Lesson Plan M10-TNB-200 provides the implementation of this training.
- Fire Brigade Equipment is subject to routine surveillance and maintenance.

In accordance with PSE&G, 1993a, records of training of each fire brigade member are maintained to assure that each fire brigade member receives adequate training, including refresher training.

It is concluded that the established HCGS programmatic criteria for fire brigade classroom and hands-on training, practice/drill, record keeping, and equipment complement are adequate.

4.8.4 TOTAL ENVIRONMENT EQUIPMENT SURVIVAL

4.8.4.1 Potential Adverse Effects on Plant Equipment and Personnel by Combustion Products

The potential effects of smoke on equipment were qualitatively assessed as follows:

- The effect of smoke within a cabinet, in which a fire initiates, can be significant. However, in this study, cabinet initiated fires were assumed to fail the source cabinet. Many cabinets at HCGS are completely enclosed without vents, others are vented without internal circulation, and of the vented cabinets with circulation, only a few do not have filters. Therefore, damage owing to the spread of smoke from one cabinet to another would not be a significant effect.
- Smoke from fires is not expected to have an appreciable affect on cables or mechanical equipment.
- Large safety related pumps (e.g., RHR, CS, SACS) have a partial dependence on room ventilation which might be compromised by a large, smoky fire. However, each of these pumps have redundant pumps in separate, unconnected locations. Furthermore, a pump fire is, itself, the highest frequency fire source in such rooms and the source pump is assumed to fail.
- For locations that have water suppression, the adverse effect of activation of the suppression system in response to a fire, particularly with respect to

water entry into cabinets, would be more of a concern than smoke production.

Therefore, for this study the effect of smoke on equipment is assessed as being insignificant with respect to calculated fire risk. EPRI, 1993a, Page 7-6 concurs with this conclusion.

The effect of smoke on personnel performance was recognized in this study and treated in a conservative manner. Sandia cabinet fire test data (NRC, 1989f) indicates that smoke can obscure visibility in the control room in a relatively short time. Therefore, a fraction of fires within the control room, consistent with the control room fire experience, was assumed to lead to a requirement to control the plant using the remote shutdown panel. This was discussed in Paragraph 4.6. Furthermore, limited credit was taken for recovery actions as was described in Paragraph 4.6.5. No recovery actions, used in this study, involved plant personnel visiting or passing through compartments in which a fire had occurred.

A separate smoke removal system is provided for the control area in the Auxiliary Building to remove combustion products from HVAC room (5602), diesel area HVAC room (5603), control equipment room mezzanine (5403), inverter rooms (5447 & 5448), electrical access area (5501), cable spreading room (5202), control equipment room (5302), and electrical equipment rooms (5102 & 5103). Smoke is removed by manually opening a normally closed shutoff damper and fire damper and then manually starting the control area smoke exhaust fan, which is dedicated to smoke removal [PSE&G, 1995f - Page 9.5-16].

The control area is equipped with redundant exhaust fans located in the HVAC equipment room. The normally used exhaust fans can be used to remove smoke produced by a fire in the main control room by manually bypassing the emergency filter unit [PSE&G, 1995f - Page 9.5-16].

4.8.4.2 Spurious or Inadvertent Fire Suppression Activity

The effect of disabling redundant safe shutdown equipment and components by fire suppression agents due to spurious or inadvertent actuation of fire suppression systems was evaluated. The evaluation considered all areas of the plant that had fixed fire suppression systems and proceeded to screen out these areas or rooms based on the following criteria:

1. Room had a fire suppression system(s) but no safe shutdown equipment or components in the room or area.

2. Room had a gaseous fire suppression system (CO_2 or Halon), which if discharged into the room or area would not affect equipment performance.

3. The room or area is protected by a preaction sprinkler system in which inadvertent actuation is not a credible event. Inadvertent discharge of a preaction sprinkler system requires multiple failures to occur simultaneously.

4. Safe shutdown components or systems are physically separated from their redundant counterparts by a wall or floor. Discharge of fire suppression system on one side of the wall will not affect equipment on the other side.

5. The fire suppression system for a safety related component or system is internal to that specific piece of equipment. Any discharge of the system will be contained within the component and will not affect that component's redundant counterpart (e.g., deluge systems for charcoal filters).

After applying the above screening criteria, only one room was identified as having a potential impact from an inadvertent fire suppression system discharge. Room 5403, the control equipment room mezzanine, has control circuitry for both electrical divisions I and II within the room and has both an automatic CO_2 fire suppression system and a manually activated deluge system. Inadvertent manual activation of the manual deluge system could potentially disable both Division I and II cables in this room by electrical shorting of the cables. However, loss of both shutdown divisions in this room has been previously analyzed in the HCGS UFSAR [PSE&G, 1995f - Table 9A-50] and is based on a fire condition disabling the cables in this room. Table 9A-50 states that safe shutdown can be achieved from the Remote Shutdown Panel in the event both divisions of shutdown cable in this room are lost.

4.8.4.3 Operator Action Effectiveness

The HCGS Safe-Shutdown Analysis, (PSE&G, 1995f - Volume 15, Paragraph 9A), establishes that safe-shutdown can be achieved in the event of a fire in any given fire area. Reference, PSE&G, 1995i and PSE&G, 1996g provide operating instructions for a fire that renders the Control Room inaccessible, or renders normal controls and indications in the Control Room unreliable.

Fire related alternate equipment operating instructions (PSE&G, 1996r) has been developed to provide for alternate/local operational capabilities, in the event of fire-induced damage to normal operating circuits/equipment.

It is concluded that adequate procedures and training are in place for safe shutdown in the control room and outside of the control room. This is discussed in more detail in Paragraph 4.8.5.

FIVE [EPRI, 1993b - Page 7 -6] suggests that adequate operator aids should be provided to allow operators to perform manual actions in plant areas where fire or smoke may be present. It goes on to suggest that operator aids may include 1) color-coded equipment, 2) portable lights, and 3) SCBA and other protective equipment. With regard to item 1, all cable trays, cabinets, and electrical equipment are color coded.

With regard to item 2, the walkdown paid special attention to the adequacy of emergency lighting in each area. The lighting was found to be generally adequate. In addition, portable flashlights are available in the control area in the Auxiliary Building and in the tool sheds in the Turbine Building.

With regard to item 3, self contained breathing apparatus (SCBA) were found during the walkdown in the main control area, with the fire brigade and in all areas covered with a CO_2 suppression systems.

4.8.5 CONTROL SYSTEMS INTERACTIONS

The objective of this element of the study is to verify that alternate safe-shutdown circuits and components are either physically independent of, or can be isolated from, the Main Control Room or both, in the event of a Control Room fire. Conversely, it is necessary to demonstrate that the normal Control Room controls and instrumentation are either electrically independent of, or can be isolated from any local/alternate shutdown control and indication stations. The objective is also to establish assurance that a fire which disables either the alternate/local control station or the normal Control Room controls and instrumentation cannot simultaneously disable the other.

The Hope Creek Generating Station is provided with a Remote Shutdown System (RSS). The RSS is designed to ensure an alternative safe shutdown capability, in the event of a fire in the control complex, or any other event that may require Main Control Room evacuation.

The RSS is comprised of a Remote Shutdown Panel (RSP) and redundant shutdown instrumentation and controls. The RSP serves as the primary control and instrumentation station for the RSS. Once control has been transferred to the RSP from the Main Control Room, the RSP is independent of the Main Control Room and fully capable of performing a safe reactor shutdown to a hot - and ultimately to a cold condition [PSE&G, 1995f - Paragraph 7.4]. Conversely, if a fire destroys the RSP Room 3576, there are sufficient controls and instrumentation available, divided between the RSS and the Main Control Room, to bring the reactor to a safe and orderly shutdown from the Main Control Room [PSE&G, 1995f - Paragraph 7.4.2.4.4 and Appendix 9A].

A large fire, postulated in the following rooms, might result in damage to normal safe shutdown circuits forcing a decision to continue control of the plant using the Remote Shutdown System (RSS): Cable Spreading Room, Lower Control Equipment Room, Control Equipment Room Mezzanine, Main Control Room, 1E Panel Room HVAC Corridor, Upper Control Equipment Room, Diesel Area HVAC Room.

The RSP is designed as the remote panel from which the operators can bring the reactor to a cold shutdown condition in the event of a fire in any of the above listed rooms [PSE&G, 1995f - Appendix 9A]. The likelihood of having to abandon the control room was included in the fire PSA.

The RSP Room is provided with an HVAC System that provides an environment similar to that of the Main Control Room. No common cause failure modes exist, including smoke and toxic fumes, which could cause both the Main Control Room and the RSP Room to be uninhabitable at the same time. As discussed in the HCGS UFSAR [PSE&G, 1995f - Appendix 9A -Paragraph III.L.3] the cable routing from the RSP to the safe shutdown equipment and to the process instrumentation has been verified to be independent of the specific fire areas for which the RSP should be used. The RSP control and instrumentation is available whether offsite power is available or not.

The evacuation of the Main Control Room and local operation of the RSP are performed in accordance with plant operating procedures [PSE&G, 1992d, PSE&G, 1995i, PSE&G, 1996g, and PSE&G, 1996r].

For a postulated fire that may necessitate operation from the RSP, the operator activates the scram switches in the control room or scrams the reactor by opening breakers on the RPS power distribution panels in the RPS MG set room. After the reactor is scrammed, the operator proceeds to the RSP to manually operate all transfer switches on Panel 10C399 (RSP). Thereafter systems controlled from the RSP are completely isolated from the control room.

If offsite power is lost, the diesel generators will start automatically. The diesel generator circuit breaker "closed" indicating lights in the RSP will give the operator a positive indication that all Class 1E buses are energized. If automatic starting of the diesels does not occur, they can be manually started by operator action at the diesel generator control panels located at the 130 foot elevation next to each 1E switchgear room. Sufficient instrumentation and controls are provided on the Remote Shutdown Panel to allow prompt Hot Shutdown of the reactor and to subsequently bring the reactor to a Cold Shutdown Condition. To achieve Cold Shutdown, a few pumps and valves must be operated locally (e.g., starting a pump or opening a valve).

The above discussion demonstrates that safe shutdown circuits in the control room and remote shutdown panel are mutually independent of each other.

4.8.6 IMPROVED ANALYTICAL CODES

The sixth issue, concerning analytical codes, does not require a plant-specific evaluation or response, as the use of the FIVE methodology for fire damage assessment is an approved IPEEE technique.

4.9 USI A-45

The IPE resolved the issue of adequacy of decay heat removal systems for internal events. This Paragraph resolves this issue for fire initiated scenarios.

4.9.1 RELEVANCE OF INITIATING EVENTS TO THE ASSESSMENT

The following discussion refers to the initiating events of Table 4.31. The primary decay heat removal system at the HCGS is the RHR system. It is supported by the HVAC system for room cooling and the SACS and service water systems for heat exchanger cooling. The alternate decay heat removal system (with RHR failure) is containment venting featuring a hard torus pipe. This is a manual system supported by either 120Vac or 120Vdc power for operation of pneumatic valves. Nitrogen bottles provide motive force to backup the instrument air system. The vent path can be opened either from the control room or from the manual station on 102' elevation.

Loss of Offsite 4kV power is not relevant to the assessment of adequacy of the RHR system because these fire induced scenarios are associated with failure of either the 4kV bus bars or actuation of transformer protection relays coupled with common cause failure of the diesel generators.

Use of the RSP is not relevant to the assessment of adequacy of the RHR system because these fire induced scenarios are associated with abandonment of the control room coupled with operator inability to recover control of the plant using the RSP.

Loss of HVAC specifically refers to a complete loss of either 1E panel room or switchgear room HVAC. Loss of switchgear room HVAC, without recovery, could cause failure of MCCs in the switchgear room that might prevent operation of the

RHR system. Loss of 1E panel room HVAC, without recovery, might conceivably have the effect of causing a loss of ability to control from the control room. This effect is not relevant to the assessment of adequacy of the RHR system as it is similar to use of the RSP, as discussed above.

MSIV closure and inadvertent opening of an SRV are both relevant to the assessment of adequacy of the RHR system because LPCI can provide system make-up (after successful depressurization) and because long term decay heat removal requires either the RHR system (event Uv) or the hard pipe vent system (event W1) hard torus vent valves can be operated manually via hydraulic pump. This does not require electric power or air. Note that the total fire induced CDF owing to MSIV closure and IORV is approximately 5.3E-05/yr.

Loss of SWS/SACS is relevant to the assessment of adequacy of decay heat removal systems because it is required for heat removal from the RHR heat exchangers.

4.9.2 FIRE SEQUENCES DIRECTLY AFFECTING DECAY HEAT REMOVAL

Referring to Table 4.33, the following fire induced scenarios directly fail decay heat removal by 1) failing electrical channels A & B, 2) failing SWS or SACS, or 3) failing 1E Panel Room HVAC. These scenarios are summarized in Table 4.35.

4.9.3 OTHER FIRE SEQUENCES WHICH MIGHT DISABLE LONG TERM DECAY HEAT REMOVAL

Other core damage sequences, initiated by MSIV closure or IORV, which might disable long term decay heat removal are divided into two categories. The total core damage frequency of these scenarios is approximately 5.3E-05/yr. as shown in Table 4.31. The first category is comprised of sequences in which there are no previous RHR channel A or B related failures. The second category is comprised of sequences in which one pump or one channel of long term decay heat removal is disabled by the fire and additional hardware failures are needed to cause core damage.

<u>Analysis of the First Category: No Previous Decay Heat Removal Failures</u>

The total CDF of transients or IORV fire initiated scenarios for which there is no a-prior RHR channel A or B failure is 2.3E-05/yr. To develop this number, the frequency of scenarios, which could disable channel A or channel B or both as well as scenarios that could disable either RHR loop A or B, were subtracted from the total MSIV closure and IORV fire CDF of 5.3E-05/yr. This number includes all failure sequences not just loss of long term decay heat removal failure sequences.

The contribution to core damage frequency of the first category, owing to loss of long term decay heat removal only, can be obtained by noting that the fire PSA used the internal events model [PSE&G, 1994b] to calculate the conditional core damage probability. The results of that model are that transients and IORV initiating events which end in core damage owing to a loss of decay heat removal (either RHR or hard pipe vent or both) comprise 2.1% of the total core damage frequency. Dominant contributors to this are TsaW1Uv, TmUvN(R), TtQUvN(R), and TtPP2WUv.

Therefore, it is estimated that the CDF associated with decay heat removal failures with fire initiated MSIV closure or IORV sequences is 0.021 * 2.3E-05/yr. = 4.8E-07/yr. Adding this to the potential of loss of decay heat removal from a single division of SACS pumps yields 5.3E-07/year.

<u>Analysis of Second Category: Partial A-Priori Loss of Decay Heat Removal</u>

Subtracting 2.3E-05/yr. from 5.3E-05/yr. results in 3E-05/yr. which is the approximate fire induced CDF associated with the second category of sequences. Note again that this includes all sequences in the PSA model not just loss of decay heat removal sequences.

A review of Table 4.33 reveals that about half (1.6E-05/yr.) of the 3E-05/yr. is associated with switchgear rooms A & B. In the analysis of these rooms, it was conservatively assumed that any fire in these rooms would disable the entire electrical channel even though only a fraction of such fires would do so. The remaining fire induced core damage frequency of this category (1.4E-05/yr.), is distributed among many rooms in which either cables or cabinets were assumed to fail an entire electrical channel (or division) because of the lack of specific knowledge about cable termination points. Therefore, it was not possible to estimate the contribution of long term decay heat removal failure of these fire induced scenarios without more detailed information about the cable termination points. If, for example, 10% of all fires actually did fail all of decay heat removal, then the additional contribution to the CDF would be a small 3E-06/yr.

4.9.4 CONCLUSION

Sequences which either directly lead to failure of decay heat removal or which indirectly lead to its failure owing to category 1 sequences described above represent an approximate CDF of 6.4E-06/yr. or about 8% of the fire induced core damage frequency. This is an upper bound estimate of the importance of decay heat removal systems because of the assumptions that 1) all scenarios were assumed to cause at least an MSIV closure when, in fact, the plant would operate through many such fires, and 2) most rooms were evaluated with the assumption that an entire channel failed if any cable of that channel was calculated as failing from the effects of fire. Including the potential contribution of category 2 sequences was estimated to have a small additional contribution to the CDF related to decay heat removal.

This analysis indicates that fire induced loss of decay heat removal scenarios are a small fraction (on the order of 10%) of the total fire induced CDF.

4.10 GI-57

GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," addresses the issue of fire protection system actuation either as an inadvertent action or as a response to a fire. The general concern is the potential for fire suppressant agent damage to co-located safety-related equipment.

This IPEEE specifically included, via walkdown and analysis, consideration of the key issues with respect to GI-57. This paragraph summarizes previous discussions in Paragraphs 4.8.1.2, 4.8.4.2, and 3.1.5.4 in order to explicitly address the GI-57 issues.

The HCGS uses preaction water sprinklers, $C0_2$ deluge systems and backup manually actuated water deluge systems for safety related areas in which automatic fire suppression is provided. Preaction sprinklers are not susceptible to seismic actuation because they require two independent, dissimilar failure modes to allow water discharge. Automatic CO_2 fire suppression systems serving safety-related equipment have seismically qualified components to avoid inadvertent discharge during a seismic event. Furthermore, the pressurized portion of the piping of CO_2 systems is located outdoors for the tank serving safety-related areas. Fire water piping is adequately supported such that there are no potential seismically induced systems interactions resulting in release of fire water or CO_2. Therefore, it is unlikely that a seismic event can cause either an inadvertent actuation or diversion of fire suppressant material.

None of the relays identified as low ruggedness or unknown manufacturer are associated with the fire protection circuitry. Therefore, failure of fire protection systems or inadvertent actuation which may cause an interlock-associated trip of a safety system is not an issue at the HCGS.

The automatic fire suppression systems at the HCGS in safety related areas are heat activated, not smoke activated. Therefore, inadvertent actuation from smoke or dust is not an issue in these areas.

The diesel generators at the HCGS are each in a separate fire compartment. The CO_2 suppression systems are independently controlled in each compartment. Furthermore, the diesel generator combustion air intake vents are located in compartments which are completely separate from and an elevation above the diesel generators. The compartments containing diesel generator combustion air intakes do not have automatic fire suppression.

Inadvertent actuation of fire suppression systems has been shown to be a factor in only one room, Room 5403. (See Paragraph 4.8.4.2 for details). This is the control equipment room mezzanine which has control circuitry for both electrical Divisions I and II within the room and has both an automatic CO_2 fire suppression system and a manually activated deluge system. Inadvertent manual activation of the manual deluge system could potentially disable both Division I and II cables in this room by electrical shorting of the cables. Previous analysis [PSE&G, 1995f - Table 9A-50] has indicated that safe shutdown can be achieved from the Remote Shutdown Panel in the event both divisions of shutdown cable in this room are lost.

The conclusion of the above walkdown and analysis derived information regarding GI-57 issues is that they are not a safety concern at the HCGS.

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Table 4.1
Map of NUREG-1407 Documentation Issues to This Report

Item From Paragraph C.3 of NUREG-1407 (NRC, 1991b)	Responsive Paragraphs in this Report
1. Methodology and Key Assumptions	4.1.1, 4.1.2, 4.1.3, 4.2.2, 4.3.1, 4.3.2, 4.3.3, 4.3.4, 4.3.5, 4.4, 4.5, 4.6.1, 4.6.3, 4.6.4, 4.6.7
Status of Appendix R Modification	4.0.4
2. Summary of walkdown findings, walkdown team, and procedures. Include assurance that as-built cable routings are used and dependence between remote shutdown and control room circuitry is addressed.	4.2, 4.8.5
3. Criteria for identification of critical fire areas and a list of critical areas for single and multiple rooms	4.1.1, 4.1.3, 4.1.4, 4.3.6
4. Criteria for fire size and duration. Treatment of cross-zone fire spread.	4.1.1, 4.3
5. Fire initiation database, major assumptions, data handling method, sources of uncertainty	4.1.2, 4.6.7.4
6. Treatment of fire growth and spread including hot gases	4.1.1, 4.3
Treatment of Smoke	4.8.4
7. Fire damage modeling including fire-induced failures of barriers	4.1.1, 4.1.4, 4.3.6, 4.6
and control systems and cabinets	4.4
Discussion of human intervention and combination of fire-induced and other failures	4.6.1, 4.6.3, 4.6.5
8. Treatment of detection and suppression,	4.3, 4.5
treatment of fire fighting procedures, fire brigade training, adequacy of fire brigade equipment, and	4.8
treatment of access routes versus barriers	4.2.4

Table 4.1
Map of NUREG-1407 Documentation Issues to This Report
(Continued)

9.	Functional/systemic event trees associated with fire induced sequences	4.1.3, 4.6.2, 4.6.3
10.	Dominant sequences, frequencies, and percentage contribution to core damage frequency	4.6.7
11.	Estimated core damage frequency, timing, assumptions, and sources of uncertainty	4.6.7
12.	Fire induced containment failures (if any) that are significantly different from those identified in the IPE	4.7
13.	Decay heat removal function and Fire Risk Scoping Study issues	4.8, 4.9
14.	For existing PSA, sensitivity studies and other supplemental studies	not applicable to the HCGS IPEEE

Table 4.2
Room Naming Convention Used in the Fire IPEEE

Building Name	Building Number.	Floor Number	Typical Elevation
Turbine Unit 1	1	1	54'
Radwaste	3	2	77'
Reactor	4	3	102' (grade)
Control/Diesel (Aux)	5	4	120' - 132'
Service Water Intake Structure (SWIS)	-	5	137' - 145'
		6	155' - 163'
		7	171' - 201'

Table 4.3
Exemplar Fire Compartment Interaction Analysis Data Sheet

Exposing Compartment Diesel Generator Rm. 5307 Location (Bldg., Fire Area, etc.): . Aux., 102, 5307

Compartment Contains:
Channel A DG PTI: Y X N
4kV 1E offsite power cables for channel A SSE: Y X N
(Ref. PSE&G 1995f - Tables 9A-1, 9A-46)

Exposed Compartments								
Compartment Identification	PFS		PTI		SSE		Criteria for Screening Boundary	Comments/References
	Y	N	Y	N	Y	N		
Electrical Access - 5339		X	X		X		2,6	DWG. M-5003, M-5103, M-5114
Diesel Generator Room 5306		X	X		X		2,6	DWG. M-5003, M-5103, M-5114; Table 9A-45
Corridor 5315		X	X			X	2,6	DWG. M-5003, M-5103, M-5114; Table 9A-25
Cable Shaft 5331		X	X		X		2,6	DWG. M-5003, M-5103, M-5114; Table 9A-32
Cable Shaft 5332		X	X		X		2,6	DWG. M-5003, M-5103, M-5114; Table 9A-33
Cable Shaft 5333		X	X		X		2,6	DWG. M-5003, M-5103, M-5114; Table 9A-34
Cable Shaft 5334		X	X		X		2,6	DWG. M-5003, M-5103, M-5114; Table 9A-35
Stair No. 53-01		X		X		X	2,6	DWG. M-5003, M-5103, M-5114

PFS: Potential Fire Spread from exposing compartment.
PTI: Plant Trip Indicator
SSE: Safe Shutdown Equipment in compartment.

Table 4.4
Exemplar Ignition Source Data Sheet

Compartment Description					
Fire Compartment: (1)	4114				
Compartment Location: (2)	RB, 54'				
Compartment Description:	RHR Pump (CP202) Room				
Compartment Description Compartment Fire Ignition Frequency					
Selected Plant Location (3)	Reactor Building				
Location Weighting Factor (WFL) (4)	1				
Ignition Source Frequency (Fif) (5)					
Compartment Specific Ignition Sources	(A)	(B)	W_{fis} = A/B	F_f	F_{if}
electrical cabinets	0	368	0.0E+00	5.0E-02	0.0E+00
pumps	2 0	87	2.3E-02	2.5E-02	5.7E-04
Plant Wide Ignition Sources	(A)	(C)	W_{fis}= A/C	F_f	F_{if}
fire protection panels	0	139	0.0E+00	2.4E-03	0.0E+00
recirc. pump/RPS MG sets	0	3	0.0E+00	5.5E-03	0.0E+00
junction box (9)	4292	1288613	3.3E-03	1.6E-03	5.3E-06
transformers	0	151	0.0E+00	7.9E-03	0.0E+00
battery chargers	0	27	0.0E+00	4.0E-03	0.0E+00
Plant Wide Ignition Sources	(A)	(C)	W_{fis}= A/C	F_f	F_{if}
off-gas/H2 recombiner (Room 4602, only)	0	3	0.0E+00	8.6E-04	0.0E+00
hydrogen tanks	0	6	0.0E+00	3.2E-03	0.0E+00
miscellaneous hydrogen fires	1	209	4.8E-03	3.2E-03	1.5E-05
air compressors	0	8	0.0E+00	4.7E-03	0.0E+00
ventilation subsystems (10)	2	199	1.0E-02	9.5E-03	9.5E-05
elevator motors	0	7	0.0E+00	6.3E-03	0.0E+00
dryers	0	10	0.0E+00	8.7E-03	0.0E+00
transient combustibles	10	209	4.8E-02	1.3E-03	6.2E-05
cable fires caused by welding	1	209	4.8E-03	5.1E-03	2.4E-05
transient fires caused by welding and cutting	1	209	4.8E-03	3.1E-02	1.5E-04
Compartment Fire Frequency (F1) (11)					9.2E-04

(1) List the room number(s) for which a fire frequency will be calculated.
(2) List the building and elevation of the compartment
(3) Provide the generic FIVE plant location (see WFL.XLS) assigned to this compartment
(4) Provide the location weighting factor calculated in WFL.XLS

Table 4.4
Exemplar Ignition Source Data Sheet(Continued)

(5) Ignition Source Frequency: Fif = WFl * WFls * Ff
(6) A = Number of ignition sources in compartment
(7) B = Number of ignition sources in selected plant location. Derived from MMIS schedule or UFSAR dwgs unless otherwise noted.
(8) C = Number of ignition sources in the plant. Derived from MMIS schedule or UFSAR dwgs unless otherwise noted.
(9) All SSE and 1E cable is qualified. Approximate method on page 10.3-6 FIVE is used.
(10) Sum of AHUs and fans from MMIS AHU list.
(11) Compartment fire frequency: F1 = SUM(Fif)

TABLE 4.5
COMPARTMENT ASSIGNMENTS FOR ISDS ANALYSIS

Category Hope Creek Compartment	Category [EPRI, 1993b - Attachment 10.3 - Table 1.2]
Control/Diesel Building Compartments (except those locations specifically listed below)	Reactor Building (BWR)
Reactor Building	Reactor Building (BWR)
Service Water Intake Structure	Intake Structure
Turbine Building	Turbine Building
Radwaste Area (Except Rooms that are part of Control Diesel Building compartments)	Radwaste Area
Transformer Yard	Transformer Yard
Remote Shutdown Panel	Reactor Building
Control Room	Control Room
Diesel Generator Rooms (4 compartments)	Diesel Generator Room
Class 1E Switchgear Rooms (4 compartments)	Switchgear Room
Battery Rooms (12 compartments)	Battery Room
Lower Electric Equipment Room (5302)	Cable Spreading Room

Table 4.6
Weighting Factors (WF_L) for the HCGS

Generic Location FIVE Reference Table 1.1	Similar Hope Creek Location	Units (or diesels)	Number of Like Buildings or Rooms Per Site	Location Weighting Factor (WFl)
Reactor Building	Reactor Building	1	1	1.00
Diesel Generator Room	Diesel Generator Rooms	4	4	1.00
Switchgear Room	Switchgear Rooms	1	4	0.25
Battery Room	Battery Rooms	1	12	0.08
Control Room	Main Control Room	1	1	1.00
Cable Spreading Room	Lower equipment room	1	1	1.00
Intake Structure	Service Water Intake Structure	1	1	1.00
Turbine Building	Turbine Building	1	1	1.00
Radwaste Area	Radwaste Building Areas	1	12	0.08
Transformer Yard	Switchyard	1	1	1.00

Table 4.7
Screening Fire Ignition Frequency per Building

Building	Screening Fire Ignition Frequency (per year)
Reactor Building	7.2E-02
Control/Diesel Building	1.3E-01
Turbine Building	7.1E-02
Radwaste Building	1.8E-02
Service Water Intake Structure	8.0E-03
Yard (Switchyard Blockhouse)	1.5E-03
Total	0.3

Table 4.8
Results of the Quantitative Screening Analysis

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
	Reactor Building							
RB, 54', 4101/4201	Torus Water Cleanup Room/MCC Room	MSIV Closure	2	2.1E-03	2.87E-02	6.03E-05		Y
RB, 54', 4103/4104	Vestibule/Core Spray 'BP206' Room	MSIV Closure	2	8.7E-04	7.12E-04	6.19E-07	Y	
RB, 54', 4105	Core Spray DP206 Room	MSIV Closure	2	6.2E-04	2.87E-02	1.78E-05		Y
RB, 54', 4106	CRD-DPW Pump & Sump Room	MSIV Closure	1	1.4E-03	1.26E-05	1.76E-08	Y	
RB, 54', 4107	RHR Pump DP202 Room	MSIV Closure	2,4	1.0E-03	2.82E-02	2.82E-05		Y
RB, 54', 4108	Elec. Eqp't. Room	MSIV Closure	2	9.2E-04	7.71E-04	7.09E-07	Y	
RB, 54'/77' 4109 /4208/4206	RHR HX Room (BP202 & HX BE205)	MSIV Closure	2	6.3E-04	2.83E-02	1.78E-05		Y
RB, 54', 4110	RCIC Pump & Turbine Room	MSIV Closure	1	1.5E-03	1.69E-04	2.54E-07	Y	
RB,54' 4111	HPCI Pump & Turbine Room	MSIV Closure	1	2.6E-03	1.95E-04	5.07E-07	Y	
RB, 54', 4112	Elec. Eqp't. Room	MSIV Closure	1	8.1E-04	1.95E-04	1.58E-07	Y	
RB, 54'/77', 4113./4214/ 4212	Pump AP202 & HX AE205 Room (and Vestibule)	MSIV Closure	2	9.0E-04	3.11E-03	2.80E-06		Y
RB, 54', 4114	RHR Pump (CP202) Room	MSIV Closure	2	9.1E-04	3.41E-05	3.10E-08	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 54', 4115	CRW-DRW Pumps & Sumps Room	MSIV Closure	1	1.4E-03	1.26E-05	1.76E-08	Y	
RB, 54', 4116	Core Spray Pump 1CP206 Room	MSIV Closure	1	6.3E-04	1.94E-05	1.22E-08	Y	
RB, 54', 4117	Vestibule	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 54', 4118	Core Spray Pump AP206 Room	MSIV Closure	1	6.2E-04	1.94E-05	1.20E-08	Y	
RB, 54' to 101', 4102/4409	Torus Area/Steam Vent	MSIV Closure	2	1.3E-03	1.26E-05	1.69E-08	Y	
RB, 77', 4202	CRD Pump Area	MSIV Closure	2	1.2E-03	2.87E-02	3.44E-05		Y
RB, 77', 4203	Corridor	MSIV Closure	2	5.1E-04	7.12E-04	3.63E-07	Y	
RB, 77', 4205/4207	MCC Area/Passageway	MSIV Closure	2	1.2E-03	7.12E-04	8.69E-07	Y	
RB, 77', 4209/ 4211/4213	RACS Pump & HX Area	MSIV Closure	2	1.7E-03	1.53E-02	2.60E-05		Y
RB, 77', 4210	Safeguard Inst. Room	MSIV Closure	1	3.7E-04	1.69E-04	6.25E-08	Y	
RB, 77', 4215	Electrical Equipment Area	MSIV Closure	2,4	8.5E-04	4.25E-02	3.61E-05		Y
RB, 77', 4216	Corridor	MSIV Closure	2	2.4E-04	6.54E-04	1.57E-07	Y	
RB, 77', 4218	MCC Area	MSIV Closure	2	1.4E-03	2.18E-04	3.05E-07	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 77', 4219	Instrument Room	MSIV Closure	1	5.1E-04	1.95E-04	9.95E-08	Y	
RB, 102', 4301 /4309/4310/ 4311	R Bldg. 102' Elevation-North Side & Div I SACS area	MSIV Closure	2	3.4E-03	1.16E-02	3.94E-05		Y
RB, 102', 4303	MCC Area	MSIV Closure	2,4	1.9E-03	7.02E-04	1.33E-06		Y
RB, 102', 4304	Airlock	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 102', 4305	Airlock	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 102', 4307	Div II SACS Area	MSIV Closure	1	1.7E-03	4.28E-05	7.28E-08	Y	
RB, 102'. 4313	Airlock	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB,102',4315/4317/ 4320/4322	El. 102' Inside Cylinder - South Side (Div. II)	MSIV Closure	2	4.0E-03	7.13E-04	2.84E-06		Y
RB, 102', 4316	Steam Tunnel	MSIV Closure	2	2.4E-04	1.76E-04	4.22E-08	Y	
RB, 102', 4318	Neutron Monitoring	MSIV Closure	1,4	2.4E-04	6.95E-04	1.67E-07	Y	
RB, 102', 4319/4321	Pipe Chases	MSIV Closure	1	2.4E-04	1.74E-04	4.18E-08	Y	
RB, 102', 4323/4324	Equipment Access	MSIV Closure	1	3.7E-04	1.26E-05	4.66E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 102', 4326/4333	CRD Removal & Repair	MSIV Closure	1	1.5E-03	6.07E-04	8.98E-07	Y	
RB, 102', 4327/4329	Pipe Chases	MSIV Closure	1	2.4E-04	3.52E-03	8.45E-07	Y	
RB, 102', 4330	Drywell Access Area	MSIV Closure	1	2.4E-04	3.41E-05	8.18E-09	Y	
RB, 102' 4334	Elevator Machine Room	MSIV Closure	1	1.1E-03	1.26E-05	1.39E-08	Y	
RB, 102', 4331 /4328/4332	El. 102' Inside Cylinder - North Side (Div. I)	MSIV Closure	2	2.4E-03	4.55E-04	1.09E-06		Y
RB, 132', 4401/4404	Electrical Equipment Room/Corridor	MSIV Closure	1	2.0E-03	1.26E-05	2.52E-08	Y	
RB, 132', 4402	Pipe Chase	MSIV Closure	1	2.4E-04	1.94E-05	4.66E-09	Y	
RB, 132', 4403	RWCU Pump Room	MSIV Closure	1	5.2E-04	1.26E-05	6.55E-09	Y	
RB, 132', 4405	RWCU Recirc Pump Room	MSIV Closure	1	5.2E-04	1.94E-05	1.01E-08	Y	
RB, 132', 4406	Clean-up Backwash Tank & Pumps	MSIV Closure	1	8.1E-04	1.82E-05	1.47E-08	Y	
RB, 132', 4410A	CRD Control Room Area	MSIV Closure	2	1.4E-03	1.94E-05	2.72E-08	Y	
RB, 132', 4415	TSC Related Room	MSIV Closure	1	6.4E-04	1.26E-05	8.06E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 132', 4418/4419/4420 /4416/ 4417	Technical Support Center (TSC), TSC Ventilation	MSIV Closure	1	7.8E-04	1.26E-05	9.83E-09	Y	
RB, 145', 4502	RWCU Filter Demin Holding Pump Room	MSIV Closure	1	5.2E-04	1.26E-05	6.55E-09	Y	
RB, 145', 4503	RWCU Demin Holding Pump Room	MSIV Closure	1	5.2E-04	1.26E-05	6.55E-09	Y	
RB, 145', 4505	Pipe Chase	MSIV Closure	1	2.4E-04	1.94E-05	4.66E-09	Y	
RB, 145', 4506	RWCU HX Room	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 145', 4511	FRVS Vent Room	MSIV Closure	1	6.0E-04	1.26E-05	7.56E-09	Y	
RB, 145', 4512	FRVS Vent Room	MSIV Closure	1	4.7E-04	1.26E-05	5.92E-09	Y	
RB, 145', 4513	Sample Station Room	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB,145',4514/ 4515/4516/ 4517	Technical Support Center Related Rooms	MSIV Closure	1	3.7E-04	1.26E-05	4.66E-09	Y	
RB, 145', 4518	Steam Tunnel HVAC Equipment Room	MSIV Closure	1	4.0E-04	1.26E-05	5.04E-09	Y	
RB, 162', 4606	Standby Liquid Control Area	MSIV Closure	1	9.5E-04	1.26E-05	1.20E-08	Y	
RB, 162', 4609	Gamma Scan Electronics	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 162', 4610	Vault	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 156', 4611	Cask Loading Pit	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 160', 4613	Gamma Scan Detector Area	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 160', 4619	Electrical Access Area in TSC	MSIV Closure	1	2.4E-04	1.2[illegible]	3.02E-09	Y	
RB, 162', 4620/4621	RWCU Filter Demineralizer Area	MSIV Closure	1	2.4E-04	1[illegible]	3.02E-09	Y	
RB, 162', 4623	Dryer Separator Storage Pool	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RB, 162',4625	Fuel Pool Cooling Pump Room	MSIV Closure	1	5.3E-04	1.26E-05	6.68E-09	Y	
RB, 4407/4408/4408A/ 4410/4411/4412/4413/ 4501/4504/4508/4509/ 4601/4602/4603/4604/ 4605/4607/4608/4614/ 4615/4616/4617/4618/ 4626/4627/4628	Combined Rooms on Elevations 132', 145', 156', 162'	MSIV Closure	1	8.7E-03	2.1E-05	1.8E-07	Y	
RB, 201', 4701	Elevator Machine Room	MSIV Closure	1	1.1E-03	1.26E-05	1.39E-08	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RB, 201', 4703/4705/4706/ 4707/4708/ 4709/ 4710	Refueling Floor	MSIV Closure	1	3.8E-03	1.26E-05	4.79E 08	Y	
	Control/Diesel Building							
AUX. 54', 5101	Vestibule	MSIV Closure	1	3.4E-04	1.26E-05	4.28E-09	Y	
AUX. 54', 5102	Electrical Equipment Room	MSIV Closure	2	2.2E-03	1.26E-05	2.77E-08	Y	
AUX, 54" 5103	125Vdc Equipment Room	MSIV Closure	5	9.8E-04	1.75E-04	1.72E-07	Y	
AUX, 54', 5104	HPCI Battery Room	MSIV Closure	1	5.4E-04	1.97E-04	1.06E-07	Y	
AUX, 54', 5105	RPS MG Set Area	MSIV Closure	2	3.8E-03	1.99E-04	7.56E-07	Y	
AUX, 54', 5106/3110	Controlled Storage Area	MSIV Closure	2	9.8E-04	1.26E-05	1.23E-08	Y	
AUX, 54', 5107	Diesel Fuel Storage Tank Room (Ch. D)	MSIV Closure	2	5.6E-04	1.26E-05	7.06E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX. 54', 5108	Diesel Fuel Storage Tank Room (Ch. B)	MSIV Closure	1	5.4E-04	1.26E-05	6.80E-09	Y	
AUX. 54', 5109	Diesel Fuel Storage Tank Room (Ch. C)	MSIV Closure	1	8.3E-04	1.26E-05	1.05E-08	Y	
AUX. 54', 5110	Diesel Fuel Storage Tank Room (Ch. A)	MSIV Closure	1	1.1E-03	1.26E-05	1.39E-08	Y	
AUX. 54', 5111/5112	Controlled Storage Area/Corridor	MSIV Closure	2	1.1E-03	1.26E-05	1.39E-08	Y	
AUX. 54', 5121	Vestibule	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX. 54', 5126	125 Vdc Battery Room	MSIV Closure	5	8.5E-04	1.26E-05	1.07E-08	Y	
AUX. 54', 5128	RCIC Battery Room	MSIV Closure	1	4.1E-04	7.12E-04	2.92E-07	Y	
AUX. 54', 5129	HPCI Electrical Equipment Room	MSIV Closure	1	5.7E-04	1.99E-04	1.13E-07	Y	
AUX. 54', 5130	RCIC Electrical Equipment Room	MSIV Closure	1	5.7E-04	7.12E-04	4.06E-07	Y	
AUX. 77', 5201	Vestibule	MSIV Closure	1	2.4E-04	2.79E-05	6.70E-09	Y	
AUX. 77', 5202	Cable Spreading Room	N/A	--	4.6E-04	1.00E+00	4.60E-04		Y
AUX. 77'-150',5203/5323/5405/5531	Vertical Cable Chase (Channel D)	MSIV Closure	5,4	2.7E-04	2.79E-05	7.53E-09	Y	
AUX. 77'-150',5204/5324/5406/5532	Vertical Cable Chase (Channel B)	MSIV Closure	5,4	2.7E-04	7.12E-04	1.92E-07	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX, 77'-150', 5205/5325/ 5407/ 5533	Vertical Cable Chase (Channel C)	MSIV Closure	5	2.7E-04	3.41E-05	9.21E-09	Y	
AUX, 77'-150', 5206/5326/ 5408/ 5534	Vertical Cable Chase (Channel A)	MSIV Closure	5	2.7E-04	3.11E-03	8.40E-07	Y	
AUX/RW, 77', 5207/3204	Electrical Access Area/Corridor	MSIV Closure	1	3.4E-04	2.87E-02	9.76E-06		Y
AUX, 77', 5208	H&V Equipment Room (Ch. D)	MSIV Closure	5	3.3E-04	2.00E-05	6.60E-09	Y	
AUX, 77', 5209	H&V Equipment Room (Ch. B)	MSIV Closure	5	3.3E-04	2.25E-05	7.43E-09	Y	
AUX, 77', 5210	H&V Equipment Room (Ch. C)	MSIV Closure	5	3.3E-04	2.11E-05	6.96E-09	Y	
AUX, 77', 5211	H&V Equipment Room (Ch. A)	MSIV Closure	5	3.3E-04	2.56E-05	8.45E-09	Y	
AUX, 77', 5216	Electrical Raceway (Ch. D)	MSIV Closure	5	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 77', 5215/5217	Corridor/Access Area (Div. II SWS/Div. I SWS in conduit)	MSIV Closure	2	2.5E-04	6.77E-04	1.69E-07	Y	
AUX, 77', 5233	Vestibule	MSIV Closure	2	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 77', 5237	Electrical Access Area (Div. I)	MSIV Closure	1	2.9E-04	3.53E-03	1.02E-06		Y

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX, 102', 5301/ 3314	Aux Elect. access Area & Common area in RW Bldg.	LOOP	3	6.5E-04	9.72E-02	6.32E-05		Y
AUX, 102', 5302	Lower Control Equipment Room	N/A	--	3.5E-03	1.00E+00	3.50E-03		Y
AUX, 102', 5303/5316	Corridor & Vestibule	MSIV Closure	1	9.7E-04	1.26E-05	1.22E-08	Y	
AUX, 102', 5304	Diesel Generator Room (Ch. D)	LOOP	3	8.2E-03	1.25E-02	1.03E-04		Y
AUX, 102', 5305	Diesel Generator Room (Ch. B)	LOOP	3	8.2E-03	1.79E-02	1.47E-04		Y
AUX, 102', 5306	Diesel Generator Room (Ch. C)	LOOP	3	8.2E-03	1.99E-02	1.63E-04		Y
AUX, 102', 5307	Diesel Generator Room (Ch. A)	LOOP	3	8.2E-03	2.57E-02	2.11E-04		Y
AUX, 102', 5308/5315	Corridor	MSIV Closure	1	4.6E-04	1.26E-05	5.80E-09	Y	
AUX, 102'-137' 5331/5419/ 5531	Vertical Cable Chase (Ch. D)	MSIV Closure	5	2.7E-04	2.79E-05	7.53E-09	Y	
AUX, 102'-137', 5332/5420/ 5532	Vertical Cable Chase (Ch. B)	MSIV Closure	5	2.7E-04	7.12E-04	1.92E-07	Y	
AUX, 102'-137' 5333/5421/ 5533	Vertical Cable Chase (Ch. C)	MSIV Closure	5	2.7E-04	3.41E-05	9.21E-09	Y	
AUX 102'-137' 5334/5422/ 5534	Vertical Cable Chase (Ch. A)	MSIV Closure	5	2.7E-04	3.11E-03	8.40E-07	Y	
AUX, 102', 5336	Cable Raceway	LOOP	3	2.4E-04	3.29E-03	7.90E-07	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX. 102', 5339	Electrical Access Room	LOOP	3	8.5E-04	9.72E-02	8.26E-05		Y
AUX. 124', 5401/3425	Electrical Access Area	MSIV Closure	2	2.8E-04	1.00E+00	2.80E-04		Y
AUX. 124', 5402	Vestibule	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX. 117.5', 5403/5449	Control Room Equipment Room Mezzanine	N/A	--	3.6E-04	1.00E+00	3.60E-04		Y
AUX. 120', 5404	Corridor	MSIV Closure	1	3.1E-04	1.26E-05	3.91E-09	Y	
AUX. 130', 5409	Corridor	MSIV Closure	1	3.7E-04	1.26E-05	4.66E-09	Y	
AUX. 130', 5410/5411	Class 1E Switchgear Room (Ch. D)	MSIV Closure	5	4.3E-03	2.79E-05	1.20E-07	Y	
AUX. 130', 5412/5413	Class 1E Switchgear Room (Ch. B)	MSIV Closure	5	4.2E-03	7.12E-04	2.99E-06		Y
AUX. 130', 5414/5415	Class 1E Switchgear Room (Ch.C)	MSIV Closure	5	4.2E-03	3.41E-05	1.43E-07	Y	
AUX. 30', 5416/5417	Class 1E Switchgear Room (Ch. A)	MSIV Closure	5	4.2E-03	3.11E-03	1.31E-05		Y
AUX. 130', 5418	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX,130',5423	DG Combustible Air Intake Room	MSIV Closure	2	2.7E-04	2.87E-02	7.75E-06		Y
AUX, 130', 5447	FRVS Panel Room	MSIV Closure	5	6.4E-04	1.26E-05	8.06E-09	Y	
AUX, 130', 5448	Class 1E Inverter Room	MSIV Closure	5	1.6E-03	2.87E-02	4.59E-05		Y
AUX, 130', 5450	DG Combustible Air Intake Room	MSIV Closure	2	3.8E-04	3.53E-03	1.34E-06		Y
AUX, 137', 5501	Electrical Access Area	MSIV Closure	1	7.9E-04	3.53E-03	2.79E-06		Y
AUX,137',5502/ 5503/5504/ 5505/5507/ 5508	Instructional Viewing	MSIV Closure	1	3.9E-04	1.94E-05	7.57E-09	Y	
AUX, 137', 5509	Shift Supervisor Room	MSIV Closure	1	5.1E-04	1.26E-05	6.43E-09	Y	
AUX, 137', 5510/5511	Control Room	N/A	--	9.6E-03	1.00E+00	9.60E-03		Y
AUX, 137', 5512	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 137', 5515	Computer Room	MSIV Closure	1	9.2E-04	1.94E-05	1.78E-08	Y	
AUX, 137', 5522/5523	Corridor/Work Control Center	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 137', 5525	Corridor	MSIV Closure	1	2.4E-04	3.41E-05	8.18E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX,137',5526/5527 /5517	File Room	MSIV Closure	1	5.1E-04	1.26E-05	6.43E-09	Y	
AUX, 150', 5535	H&V Chase	MSIV Closure	1	2.5E-04	1.26E-05	3.15E-09	Y	
AUX, 150', 5536/5537	Corridor	MSIV Closure	1	4.3E-04	1.26E-05	5.42E-09	Y	
AUX, 146', 5538	Battery Charger Room	MSIV Closure	5	7.8E-04	2.79E-05	2.18E-08	Y	
AUX, 146', 5539	Battery Room	MSIV Closure	5	4.1E-04	2.79E-05	1.14E-08	Y	
AUX, 146', 5540	Battery Charger Room	MSIV Closure	5	6.7E-04	7.12E-04	4.77E-07	Y	
AUX, 146', 5541	Battery Room	MSIV Closure	5	4.1E-04	7.12E-04	2.92E-07	Y	
AUX, 146', 5542	Battery Charger Room	MSIV Closure	5	6.7E-04	3.41E-05	2.28E-08	Y	
AUX, 146', 5543	Battery Room	MSIV Closure	5	4.1E-04	1.94E-05	7.95E-09	Y	
AUX, 146', 5544	Battery Charger Room	MSIV Closure	5	7.8E-04	2.66E-05	2.07E-08	Y	
AUX, 146', 5545	Battery Room	MSIV Closure	5	4.1E-04	2.66E-05	1.09E-08	Y	
AUX, 150', 5546	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX,137',5585/5586 /5587/5588/5589/ 5590	Operations Department	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX, 155.3', 5602	Control Area HVAC Equipment Room	MSIV Closure	1	3.7E-03	3.41E-05	1.26E-07	Y	
AUX,163.6',5604/ 5611/5618	Corridors	Loss of HVAC	7	2.6E-04	3.00E-04	7.80E-08	Y	
AUX, 163.6', 5605/5631	Upper Control equipment Room/Computer Rm.	N/A	--	5.5E-03	1.00E+00	5.50E-03		Y
AUX, 163.6', 5606	Switchgear Area HVAC (Div. II)	MSIV Closure	1	4.3E-04	8.61E-06	3.70E-09	Y	
AUX, 163.6', 5607	Inverter Room	MSIV Closure	5	5.2E-04	2.75E-05	1.43E-08	Y	
AUX, 163.6', 5608	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 163.6', 5609	Battery Room	MSIV Closure	5	5.4E-04	1.94E-05	1.05E-08	Y	
AUX, 163.6', 5610/ 5612	Corridors	MSIV Closure	1	3.7E-04	3.41E-05	1.26E-08	Y	
AUX, 163.6', 5613	Inverter Room	MSIV Closure	5	5.2E-04	2.59E-05	1.35E-08	Y	
AUX, 163.6', 5614	Battery Room	MSIV Closure	5	4.1E-04	1.94E-05	7.95E-09	Y	
AUX, 163.6', 5615	Inverter Room	MSIV Closure	5	3.7E-04	2.20E-05	8.14E-09	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX, 163.6', 5616	Inverter Room	MSIV Closure	5	5.1E-04	1.94E-05	9.89E-09	Y	
AUX, 163.6', 5617	Electrical Access Area	MSIV Closure	1	2.4E-04	3.00E-04	7.20E-08	Y	
AUX, 153.3', 5619	TSC Electrical Room	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 163.6', 5620	HVAC Equipment Room	Loss of HVAC	7	1.0E-03	1.00E+00	1.00E-03		Y
AUX, 163.6', 5621	Inverter Room	MSIV Closure	5	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 153.3', 5622	Inverter Room	MSIV Closure	5	3.7E-04	1.26E-05	4.66E-09	Y	
AUX, 163.6', 5623	Inverter Room	MSIV Closure	5	9.2E-04	1.26E-05	1.16E-08	Y	
AUX, 163.6', 5624	Inverter Room	MSIV Closure	5	8.1E-04	1.26E-05	1.02E-08	Y	
AUX, 163.6', 5625	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 163.6', 5626	Battery Room	MSIV Closure	5	7.4E-04	1.26E-05	9.32E-09	Y	
AUX, 163.6', 5627	Battery Room	MSIV Closure	5	4.1E-04	1.26E-05	5.17E-09	Y	
AUX, 163.6', 5628	Inverter Room	MSIV Closure	5	8.1E-04	1.26E-05	1.02E-08	Y	
AUX, 163.6', 5629	Switchgear Area HVAC (Div.I)	MSIV Closure	1	4.3E-04	1.05E-06	4.52E-10	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
AUX, 163.6', 5630	Control Area HVAC Equipment Room	MSIV Closure	1	2.0E-03	2.79E-05	5.58E-08	Y	
AUX, 178', 5702/5706	Corridor	MSIV Closure	1	3.8E-04	1.26E-05	4.79E-09	Y	
AUX, 178', 5703/5704	DG Area HVAC Equipment Room	Loss of HVAC	7	4.8E-03	1.00E+00	4.80E-03		Y
AUX, 178', 5705	Corridor	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
AUX, 51-01	Elevator	MSIV Closure	1	1.1E-03	1.26E-05	1.39E-08	Y	
AUX, 51-02	Elevator	MSIV Closure	1	1.1E-03	1.26E-05	1.39E-08	Y	
	Turbine Building							
TB, 54', 1119	Oil Interceptor Room	MSIV Closure	2	2.9E-04	2.10E-05	6.09E-09	Y	
TB, 54'-132'	Condenser Area & Other areas	MSIV Closure	2	4.8E-02	1.94E-05	9.31E-07	Y	
TB, 77', 1221	Lube Oil Tank Room	MSIV Closure	1	7.7E-04	2.10E-05	1.62E-08	Y	
TB, 102', 1314	Lube Oil Reservoir	MSIV Closure	1	7.9E-04	2.10E-05	1.66E-08	Y	
TB, 102', 1315/1316/1317/ 1320/1321/1322	Access and Unloading Area	LOOP	3	1.5E-03	3.52E-03	5.28E-06		Y

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
TB, 120', 1406/1409	Electrical Equipment Mezzanine	LOOP	3	3.4E-03	3.52E-03	1.20E-05		Y
TB, 137'-155'	TB, Elevations 137' to 163', except rooms 1516 & 1517	MSIV Closure	1	8.7E-03	2.10E-05	1.83E-07	Y	
TB, 137'-155',1516	TB, Recirc Pump MG set	MSIV Closure	2	2.6E-03	2.10E-05	5.46E-08	Y	
TB, 137'-155',1517	TB, Recirc Pump MG Set	MSIV Closure	2	2.6E-03	2.10E-05	5.46E-08	Y	
TB, 155'-171'	TB, El. 155' - 171 , except 1516 & 1517	MSIV Closure	1	2.3E-03	2.10E-05	4.83E-08	Y	
	Service Water Intake Structure							
SWIS, 79'-93', 203/204/112/107	Pump, Booster Pump, MCC Rooms		8	3.6E-03	1.78E-05	6.41E-08	Y	
SWIS, 79'-93', 207/208/114/110	PUMP, Booster Pump, MCC Rooms		8	3.3E-03	1.91E-05	6.30E-08	Y	
SWIS, 122',304/305/306	Intake Structure Fan Rooms		8	5.2E-04	1.78E-05	9.26E-09	Y	
SWIS, 122',310/311/312	Intake Structure Fan Rooms	MSIV Closure		5.7E-04	1.91E-05	1.09E-08	Y	
	Radwaste Building							
RW, 54', 3101 through 3199	Elevation 54' of RW bldg., including rooms 3110 & 5106	MSIV Closure		2.4E-03	1.26E-05	3.02E-08	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RW, 77'-87, 3201 through 3222	2nd floor of RW bldg., except rooms 3204 & 5207	MSIV Closure	1	2.3E-03	1.76E-05	4.05E-08	Y	
RW, 102',3311/3312/3316/3317/3318/3319/ 3324/3328/ 3329	Middle section of the third floor of the RW Bldg.	LOOP	3	1.4E-03	3.29E-03	4.61E-06		Y
RW, 102', 3rd floor north	All rooms north of 3324, 3328 & 3329	MSIV Closure	1	2.3E-03	1.76E-05	4.05E-08	Y	
RW, 102', 3rd floor south	RAW rooms 3303,3305,3307,3308,3309&3310	MSIV Closure	1	2.0E-03	1.45E-05	2.90E-08	Y	
RW, 102', 3301/3301A/3302/ 3342/3313	RW bldg. entrance, vestibule, corridor, hot water heater & lobby	MSIV Closure	1	2.5E-04	1.94E-05	4.85E-09	Y	
RW, 102', 3303	RW Bldg. Men's Toilet	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RW, 102', 3304	RW Bldg. Janitor Closet	MSIV Closure	1	2.4E-04	1.26E-05	3.02E-09	Y	
RW, 4th floor, 120'/132'	Fourth floor of the RW Bldg., Rooms 3401 through 3451	MSIV Closure	1	1.6E-03	1.45E-05	2.32E-08	Y	
RW, 5th floor, 137'/142'	All rooms on 5th floor, except 3576	MSIV Closure	1	1.6E-03	1.26E-05	2.02E-08	Y	

Table 4.8
Results of the Quantitative Screening Analysis (Continued)

LOCATION	DESCRIPTION	INITIATING EVENT USED	CONDENSED INITIATING EVENT EXPLANATION	SCREENING FIRE IGNITION FREQUENCY (per r/yr.)	SCREENING CONDITIONAL CORE DAMAGE PROBABILITY (SCCDP)	SCREENING CORE DAMAGE FREQUENCY (per r/yr.)	SCREENED OUT	FURTHER STUDY NEEDED
RW, 137', 3576	Remote Shutdown Panel	MSIV Closure	1	3.8E-04	1.26E-05	4.79E-09	Y	
RW, 153' to 163'	Sixth floor of the RW Bldg.	MSIV Closure	1	3.1E-03	1.26E-05	3.91E-08	Y	
	Yard							
Switchyard	Switchyard Blockhouse	LOOP	3.2	1.5E-03	3.52E-03	5.28E-06		Y
No. of Locations	209							

Table 4-9
Unscreened Compartments Requiring Detailed Fire PSA

Reactor Building	Description
RB, 54', 4101/4201	Torus Water Cleanup Room/MCC Room
RB, 54', 4105	Core Spray DP206 Room
RB, 54', 4107	RHR Pump DP202 Room
RB, 54/77', 4109/ 4208/4206	RHR HX Room (BP202 & HX BE205)
RB, 54/77',4113/ 4214/4212	RHR Pump Ap202 & HX AE205 Room (and Vestibule)
RB, 77', 4202	CRD Pump Area
RB, 77', 4209/4211/4213	RACS Pump & HX Area
RB, 77', 4215	Electrical Equipment Area
RB, 102', 4301/4309/4310/4311	R Bldg. 102' Elevation-North Side & lv I SACS area
RB, 102', 4303	MCC Area
RB, 102', 4315/4317/4320/4322	El. 102' Inside Cylinder - South Side (Div. II)
RB, 102', 4331/4328/4332	El. 102' Inside Cylinder - North Side (Div. I)

Turbine Building	Description
TB, 102', 1315/1316/1317/ 1320/1321/1322	Access and Unloading Area
TB, 120', 1406/1409	Electrical Equipment Mezzanine

Control/Diesel Building	Description
AUX, 77', 5202	Cable Spreading Room
AUX/RW, 77',5207/3204	Electrical Access Area/Corridor
AUX, 77', 5237	Electrical Access Area (Div. I)
AUX, 102', 5301/ 3314	Aux Elect. access Area & Common area in RW Bldg.
AUX, 102', 5302	Lower Control Equipment Room

Table 4-9
Unscreened Compartments Requiring Detailed Fire PSA (Continued)

AUX, 102', 5304	Diesel Generator Room (Ch. D)
AUX, 102', 5305	Diesel Generator Room (Ch. B)
AUX, 102', 5306	Diesel Generator Room (Ch. C)
AUX, 102', 5307	Diesel Generator Room (Ch. A)
AUX, 102', 5339	Electrical Access Room
AUX, 124', 5401/3425	Electrical Access Area
AUX, 117.5', 5403/ 5449	Control Room Equipment Room Mezzanine
AUX, 130', 5412/5413	Class 1E Switchgear Room (Ch. B)
AUX, 130', 5416/5417	Class 1E Switchgear Room (Ch. A)
AUX, 130', 5423	DG Combustible Air Intake Room
AUX, 130', 5448	Class 1E Inverter Room
AUX, 130', 5450	DG Combustible Air Intake Room
AUX, 137', 5501	Electrical Access Area
AUX, 137', 5510/5511	Control Room
AUX, 163.6', 5605/5631	Upper Control equipment Room/Computer Rm.
AUX, 163.6', 5620	HVAC Equipment Room
AUX, 178', 5703/5704	DG Area HVAC Equipment Room

Radwaste Area	Description
RW, 102', 3311/3312/3316/3317/3318/ 3319/3324/3328/ 3329	Middle section of the third floor of the RW Bldg.

Yard	Description
Switchyard	Switchyard Blockhouse

Table 4.10
Identification of High Hazard Areas

Rooms	Description	Basis for High Hazard
Reactor Building		
4111	HPCI Pump and Turbine Room	175 gallons of lube oil dispersed among several pumps and air cooling units
Auxiliary/Control Building		
5107, 5108, 5109, 5110	Channel D, B, C, A (respectively) Diesel Fuel Storage Tank Rooms	Two 26,500 gallon tanks of No.2 fuel in each room
5208, 5209, 5210, 5211	Channel D, B, C, A (respectively) HVAC Equipment Rooms	Fuel oil transfer pipes traverse vertically through these rooms from the diesel fuel storage tank rooms.
5304, 5305, 5306, 5307	Channel D, B, C, A (respectively) Diesel Generator Rooms	Diesel engine, day tank (550 gallons of No. 2 fuel), lube oil accumulator
Turbine Building		
1119	Oil Interceptor room	150 gallons of oil
1221	Lube Oil Receiving and Storage Room	44,000 gallons of lube oil
1314	Main Turbine Oil Reservoir and Centrifuge Room	12,295 gallons of lube oil
1316	Equipment Unloading Area	2,700 gallons of lube oil distributed among various locations: operations lube oil storage area, tanks, lockers and safety cans.
1302	Electrical Equipment Area	530 gallons of H_2 seal oil
1402, 1403, 1404	RFPT Lube Oil Reservoir Rooms	1,275 gallons of lube oil in each room
1513	Turbine Generator Area	Large quantities of oil and Hydrogen
1516, 1517	Recirculation Pump MG Set Rooms	Lube oil in motor generator sets
1509, 1510, 1511	Reactor Feed Pump Rooms	Lube oil in pump turbines
Service Water Intake Structure		
204	Division I Service Water Pumps	56 gallons of lube oil
208	Division II Service Water Pumps	56 gallons of lube oil
Yard		
	Station Service Transformers and Main Power Transformers	Transformer Oil (A Main Power Transformer fire occurred in the 1980s)

Table 4.11
Summary High Hazard Area Analysis and Results

Description	Worst Case Consequences	Order of Magnitude Estimate of Frequency of Uncontrolled Fire
Reactor Building		
HPCI Pump and Turbine Room	No fire barrier or structural damage would occur owing to insufficient combustibles. Damage limited to loss of functionality of HPCI pump.	10^{-4}/yr.
Auxiliary/Control Building		
Channel D, B, C, A (respectively) Diesel Fuel Storage Tank Rooms	Potential structural damage to walls and ceiling. Fire effects might potentially propagate through the three hour fire barrier to Rooms 5105, 5106, 5111, 5208. Potentially affected safety related equipment includes HPCI battery room heater, battery heater for 250 V_{dc} Channel A batteries, RPS MG sets, diesel generator HVAC equipment.	10^{-9}/yr./room There are no credible ignition sources in these rooms. The estimated frequency assumes ignition.
Channel D, B, C, A (respectively) HVAC Equipment Rooms	Potential breach of the surrounding three hour fire barrier would allow damage to an adjacent HVAC equipment room, a vertical cable chase, corridor 5237 and/or corridor 5217. Each of these rooms are surrounded by a three hour fire barrier as well. The cable chases (each containing cable of a single electrical channel), and corridor 5237 are protected by automatic preaction sprinklers. Corridor 5217 contains no safety related equipment.	10^{-7}/yr./room. A severe fire can occur only if a rupture occurs in the diesel generator fuel oil transfer lines during operation of the fuel oil transfer pump and the pump does not trip owing to high current.
Channel D, B, C, A (respectively) Diesel Generator Rooms	These rooms were not screened out. The possibility of a large fire was included in the fire PSA. The possibility of a fire spreading to the adjacent electrical access area 5339 was included in the PSA. Adjacent corridor 5315 contains no safety related equipment. A fire severe enough to breach the three hour fire barrier into 5339 and into the Switchgear room above could result in loss of off-site power, loss of the 1E electrical channel corresponding to the diesel generator room in which the fire occurred, loss of Division I and loss of diesel generators A & C.	10^{-7}/yr./room.

Table 4.11
Summary High Hazard Area Analysis and Results (Continued)

Description	Worst Case Consequences	Order of Magnitude Estimate of Frequency of Uncontrolled Fire
Turbine Building		
Oil Interceptor Room	This room is surrounded by a three hour fire barrier and contains an automatic wet pipe sprinkler system. Because of insufficient combustible material loading, no fire barrier or structural damage would occur. There is no safety related equipment in this room or adjacent room.	10^{-7}/yr. There are no in-situ ignition sources. The estimated frequency assumes ignition.
Lube Oil Receiving and Storage Room	This room is surrounded by a three hour fire barrier and contains an automatic wet pipe sprinkler system. A worst case fire can damage fire barriers and turbine building structure. Fire propagation through the ceiling might cause damage to 4kV 1E offsite power bus bars. Fire propagation through the walls might cause damage to the condensate system. No safety related equipment would be damaged.	10^{-6}/yr. There are no in-situ ignition sources. The estimated frequency assumes ignition.
Main Turbine Oil Reservoir and Centrifuge Room	This room is surrounded by a three hour fire barrier and contains an automatic preaction sprinkler system. A worst case fire can damage fire barriers and turbine building structure. Fire propagation to adjacent rooms might cause damage to 4kV 1E offsite power bus bars and condensate system. No safety related equipment would be damaged.	10^{-7}/yr. There are no credible ignition sources in this room. Frequency estimate assumes ignition.
Equipment Unloading Area	This room is surrounded by a three hour fire barrier and contains an automatic preaction sprinkler system. The combustibles stored in this area are distributed throughout the area. One or both of the 4kV 1E offsite power bus bars might be damaged. No other safety related equipment would be damaged.	10^{-5}/yr.

Table 4.11
Summary High Hazard Area Analysis and Results (Continued)

Description	Worst Case Consequences	Order of Magnitude Estimate of Frequency of Uncontrolled Fire
Electrical Equipment Area	This area contains an automatic wet sprinkler system. Combustibie materials are distributed throughout the area. Besides potentially affecting the feedwater system, condensate system and generator, no safety related equipment is in this or adjacent areas.	10^{-5}/yr.
RFPT Lube Oil Reservoir Rooms	Each room is enclosed with a three hour fire barrier and contains a automatic wet pipe sprinkler. It contains no safety related equipment and there is no safety related equipment in adjacent rooms.	10^{-6}/yr. These rooms do not contain in-situ ignition sources. The fire frequency assumes ignition.
Turbine Generator Area	The turbine includes an automatic deluge system and associated exposed piping is protected by an automatic preaction sprinkler. The turbine generator area is separated from safety related equipment by the intervening Radwaste Building. The nearest approach of the 4kV 1E offsite power bus bars is on the floor below on the opposite side of the turbine building. Because of the quantity of oil and hydrogen in this area a large fire can occur (and has occurred in the nuclear industry).	10^{-5}/yr.
Recirculation Pump MG Set Rooms	Each room is surrounded by a three hour fire barrier and contains an automatic deluge system. These rooms are on the turbine deck so that potential damage of a fire that breaches the barrier might include the turbine. The Radwaste Building separates these rooms from safety related equipment.	10^{-5}/yr.
Reactor Feed Pump Rooms	These rooms contain an automatic wet pipe sprinkler. These rooms are on the turbine deck so that potential damage of a fire that breaches the barrier might include the turbine. The Radwaste Building separates these rooms from safety related equipment.	10^{-6}/yr.

Table 4.11
Summary High Hazard Area Analysis and Results (Continued)

Description	Worst Case Consequences	Order of Magnitude Estimate of Frequency of Uncontrolled Fire
Service Water Intake Structure		
Division I or II Service Water Pumps	These rooms are separated from each other by two identical walls, one of which has been designated a three hour fire barrier, and an intervening room. They each contain an automatic preaction sprinkler system. A large fire might damage both service water pumps in one room.	10^{-5}/yr.
Yard		
Station Service Transformers, and Main Power Transformers	Transformers are separated from each other by a one hour concrete fire wall and include an automatic deluge system. Furthermore, the transformers are located within 150 feet of the Station Fire Department and are easily visible from the front door of the fire station. The station service transformers are about 20 feet from the turbine building which has a two hour fire wall. The ground around the transformers is a gravel bed to limit the spread of oil owing to spills. The transformer fire that occurred at HCGS did not propagate into the turbine building but smoke residue coated the outside of the building near the transformer. Propagation of a fire into the turbine building could damage the condensate system and the 4kV 1E offsite power bus bars. These areas are protected by an automatic preaction sprinkler system (in the Unloading Area) and a automatic wet pipe sprinkler system in the condensate area. There are two redundant 4kV bus bars and two redundant transformers per bus bar. Therefore, no single transformer failure can fail off-site power to any bus.	10^{-5}/yr.

Table 4.11
Summary High Hazard Area Analysis and Results (Continued)

Description	Worst Case Consequences	Order of Magnitude Estimate of Frequency of Uncontrolled Fire
	To obtain a loss of offsite power, therefore, either all four 1E station service transformers must fail at the same time or a single fire must spread to fail all the transformers. The likelihood of a core damage scenario initiated by a severe fire would involve failure of both the automatic deluge system and the fire brigade to contain the fire. Such an event would cause a loss of offsite power which has been analyzed in the quantitative screening analysis to have a conditional core damage probability of 3.5E-03. The fire ignition frequency of a transformer as found in FIVE is 1.6E-03. Similarly, failure of the deluge system is given in FIVE as 0.05. The failure of the fire brigade to contain the fire is judgmentally assessed as 0.1 given the fact that a transformer fire occurred at the HCGS and was contained by the deluge system and fire brigade. The screening core damage frequency for a transformer fire that can cause loss of offsite power is therefore: SCDF (Station Service Transformers) = 4 * 1.6E-03/yr. * 0.05 * 0.1 * 3.5E-03 = 1E-07/yr. The transformers, therefore, are not a risk significant item.	

Table 4.12
Fire Hazard Walkdown Checklist

HOPE CREEK

IPEEE FIRE ANALYSIS WALKDOWN CHECKLIST

Fire Hazard

Fire Area
Description:
Room Numbers:
Building: Elevation:

Cable Tray Distribution

Susceptibility to Suppression System Actuation

Hot Gas Layer
Can hot gas layer form in the room?

Potential Fire Scenarios

Ignition source	Target combustibles	Target cables and equipment

Other Comments

Table 4.13
Fire Protection Walkdown Checklist

HOPE CREEK IPEEE FIRE ANALYSIS WALKDOWN CHECKLIST Fire Protection
Fire Area Description: Room Numbers: Building: Elevation:
Foot Traffic
Ignition Sources In situ - <u>Item</u> <u>Quantity</u> <u>Panel Voltage</u> Transient
Combustible Loading Are there any flammable liquids or gases? Is there any hydrogen piping?
Fire Detection Are there any detectors that could potentially be masked from a fire?
Fire Suppression Systems Are the automatic suppression systems activated by smoke or by heat? Is there any part of the suppression system(s) that could potentially be ineffective in a fire?
Fire Brigade Access Is there a possibility of entering the room from an adjacent compartment that contains the equipment from an opposite safety train? Are there any areas within the compartment where emergency lighting may not be effective?
Drainage If the area is protected by a water sprinkler system, are the drainage paths in an acceptable condition?
Other Comments

Table 4.14
Fire Barrier Walkdown Checklist

HOPE CREEK

IPEEE FIRE ANALYSIS WALKDOWN CHECKLIST

Fire Barriers

Fire Area
Description:
Room Numbers:
Building: Elevation:

Doors
Are there any normally open doors or other passageways?

Ventilation Openings
Are there any louvers on the doors?
Are there any ventilation openings on the walls?

Dampers

Are the dampers open or closed? Open Closed

Openings in Walls, Ceiling and Floor
Are there any unsealed pipe penetrations?
Are there any open holes in the ceiling, floor or walls?
Are there any unsealed cable or other penetrations.

FCIA Screening Criteria Verification
see attached form

Other Comments

Table 4.15
Walkdown Guidance

HOPE CREEK

IPEEE FIRE ANALYSIS WALKDOWN CHECKLIST GUIDE

Introduction

The main objective of the walkdown is to identify fire vulnerabilities that may not be discovered from the application of analytical fire PSA or FIVE methodologies. The walkdown team should strive to identify situations that may render part of the fire protection capabilities ineffective. The walkdown should be used to verify the adequacy of the FCIA conclusions. Also, the spatial arrangement of ignition sources and targets (items critical to plant safety) should be identified for possible fire propagation modeling. Finally, the walkdown was used to resolve and verify the issues raised in the Sandia Scoping Study.

It should be noted that the purpose of these walkdowns is not to audit the plant for compliance with any regulation or to verify the fire protection plan.

The walkdown forms are intended to be completed in the course of a plant walkdown. The guidance provided in this document should be viewed as the minimum level of detail.

Fire Protection

Foot Traffic
Enter the level of foot traffic through the area in terms of percent of time that there may be a person present in the area under normal plant operating conditions. Note any unusual personnel related issues related to this area.

Ignition Sources
In situ
Make a list of equipment items that may lead to fire ignition. Distinguish among different electrical voltage levels, different sizes of motors, and size of equipment. (This was done prior to the walkdown using information in the UFSAR, Fire Hazard Analysis and the Fire Pre-Plan documentation.) The walkdown is to verify the list of equipment.

Transient
Discuss the possibility and type of transient ignition sources in the area.

Combustible Loading
Identify all flammable liquid or gas storage vessels or piping (e.g., hydrogen).

Table 4.15
Walkdown Guidance (Continued)

Fire Detection

Review the location of the detection devices and express an opinion regarding the spacing, accessibility (e.g., for smoke) to the detector and the expected effectiveness of the detectors. Postulate possible fire events in the area for this purpose.

Fire Suppression Systems

Review the location of the nozzles or sprinkle heads with respect to the items that they are intended to protect. Postulate possible fire events in the area and express an opinion on whether the suppression system will function properly.

Fire Brigade Access

Note the fire brigade access and travel path to arrive at this location. Note the possibility of entering the room from an adjacent compartment that contains the equipment from an opposite safety train.

Drainage

If the area has water suppression systems (e.g., sprinkler or hose station), note the condition of drainage opening.

Fire Barriers

Doors

Note any doors or large passages to adjacent compartments. Note whether the doors are normally closed or open. Review the self closing mechanisms and express and opinion regarding their adequacy.

Ventilation Openings

Note any ventilation openings and their connections to adjacent compartments. Make a special note if there are ventilation opening on the doors.

Dampers

Note whether the dampers are normally closed or open. Review the self closing mechanisms and express and opinion regarding their adequacy.

Openings in Walls, Ceiling and Floor

Note any openings other than those mentioned above in the walls, ceiling and floor. Specify location, size, shape and adjacent compartment.

FICA Screening Criteria Verification

Verify the FICA criteria selected for compartment.

Table 4.15
Walkdown Guidance (Continued)

Fire Hazard
Cable Tray Distribution Note the safety related cables, the train and their relative locations. Sketch their pathways, if necessary.
Susceptibility to Suppression System Actuation Note the equipment that may be adversely affected by inadvertent activation of the suppression systems in the compartment. Note whether suppression activities would have a detrimental effect on fire or hot gas layer spread.
Hot Gas Layer Note whether hot gas layer can form in the room and where it could spread.
Potential Fire Scenarios <u>Ignition source</u>   <u>Target combustibles</u>   <u>Target cables and equipment</u> Enter postulated ignition scenario, the combustibles that may assist in the spread of the fire and list target cables or equipment that may get damaged from the fire. Make a sketch of the fire propagation path if necessary.

Table 4.16
Example Plume Effect Template for Five Gallon Liquid Pool Fire

FIVE WORKSHEET 1: TARGET IN PLUME SCENARIOS			
Box	Description	Value	Units
1	Target Damage Threshold	698	F
2	Height of Target Above Source	15	ft.
3	Height From Source to Ceiling	30	ft.
4	Peak Fire Intensity	72000	Btu/s
5	Fire Location Factor	1	
6	Effective Heat Release Rate	72000	Btu/s
7	Plume Temperature Rise	2520	F
8	Critical Temperature Rise	618	F
9	Plume Temperature Rise minus Critical Temperature Rise	target damage	F

The following adds the effect of hot gas layer heat input to see if damage occurs

Box	Description	Value	Units
10	Net heat addition per volume to achieve temperature rise of box 9	N/A	Btu/ft3
11	Volume of space between source and ceiling	36000	ft3
12	Critical heat addition	N/A	Btu
13	Heat loss factor	0.7	
14	Critical heat addition accounting for heat loss	N/A	Btu
15	Q_{tot} left after subtracting off Q of plume	347568	Btu
		target damage	

Table 4.17
Example Ceiling Jet Effect Template for Five Gallon Liquid Pool Fire

FIVE WORKSHEET 2: TARGET OUTSIDE OF PLUME					
Box	Description	Value	Units		
1	Target damage threshold	698	F		
2	Height of target above source	15	ft.		
3	Height from source to ceiling	30	ft.		
4	Ratio of target height to ceiling height	0.50			
	Boxes 5 to 11 completed for target in ceiling jet				
5	Horizontal distance of target from center of fire source	2	ft.		
6	Horizontal distance to source to ceiling height ratio	0.067			
7	Enclosure width perpendicular to source to target axis	30.0			
8	Source to ceiling height to enclosure width ratio	1.0			
9	Total heat release rate of fire	72000	Btu/s		
10	Fire location factor	1.0			
11	Effective fire heat release rate	72000	Btu/s		
12	Plume temperature rise at ceiling	2032	F		
13	Ceiling jet temperature rise factor at target	0.0	F	1.8	Ceiling jet temperature rise factor at ceiling
14	Ceiling jet temperature rise at target	0	F	3707	Ceiling jet temperature rise at ceiling
15	Critical temperature rise at target	618	F		
16	Critical - ceiling jet temperature rise at target	618	F		
The following adds the effect of hot gas layer heat input to see if damage occurs					
17	Net heat addition per volume to achieve temperature rise of box 16	7	Btu/ft3		
18	Volume of space between source and ceiling	36000	ft3		
19	Critical heat addition	262004	Btu		
20	Heat loss factor	0.7			
21	Critical heat addition accounting for heat loss	873346	Btu		
22	Qtot left after subtracting off Qtot of plume and ceiling jet.	265933	Btu		
		no damage			

Table 4.18
Example Thermal Radiation Effect Template for Five Gallon Liquid Pool Fire

FIVE WORKSHEET 3: RADIANT EXPOSURE SCENARIOS			
Box	Description	Value	Units
1	Critical radial flux to target	1	Btu/s/ft2
2	Heat release rate	72000	Btu/s
3	Fraction of total heat release rate owing to radiation	0.4	-
4	Total radiant heat release rate	28800	Btu/s
5	Critical radiant heat flux distance	47.87	ft
If target distance > critical distance, then no damage			
possible damage			

Table 4.19 Example FireTran.xlt: Template for Fire Damage Time vs. Extinguishment Time (Includes all inputs for other templates as well as this one.)

FIVE WORKSHEET A-1: TRANSIENT ANALYSIS THERMALLY THICK TARGETS

Box	Description	Value	Units
1	Radiative heat flux at target	4.14	Btu/s/ft2
2	Convective heat flux at target	24.50	Btu/s/ft2
3	Total heat flux at target	28.64	Btu/s/ft2
4	Target Thermal Response Parameter	50	
5	Estimated Time to Target Damage	8	sec.
6	Detection device rated temperature rise	80	F
7	Gas temperature rise at detector	2415	F
8	Detector temp. rise/Gas temp. rise	0.03	
9	Dimensionless detector actuation time	0.03	
10	Time constant of detection device	70	sec.
11	Estimated time to Detector Actuation	9	sec.
	Estimated time to fire burn-out	9	sec.
	Estimated time to Suppression	309	

BOX 2: CONVECTIVE HEAT FLUX ALTERNATIVES

Horizontal distance of target from center of fire source (ft) =		5	
Effective radius of fire source (ft) =		14	Target in Plume
Vertical distance of target from source (ft) =		23	
Vertical distance of ceiling from source (ft) =		25	
Ratio of vertical distances =	0.92		

In Plume			Ceiling Jet Sublayer	
24.50	BTU/sec-ft2		5.30	BTU/sec-ft2
Where is fire?=== ===>	n			
Enter c for corner, w for wall				
and n for neither				
Fire Location Factor (k)=	1		Ceiling Jet Adjustment Factor =	0.22
Plume Rise Velocity	4	ft/sec		

Hot Gas Layer	
0.18	BTU/sec-ft2
Qtotal	66840
Spill Area or solid area	600
Fuel Mass	33.4
Heat available for hot gas layer	235079
Volume of hot gas layer	100900
Heat addition of hot gas layer per volume	2.33
Temperature rise above ambient (F) =	

Target damage

Detection calculations assume that detectors are in ceiling jet region

All detectors at HCGS are on the ceiling

BOX 1. Radiant Heat Flux Parameters	
Q = total heat release rate (BTU/s) of fire	72000
R= line-of-sight distance from fire to target (ft.)	23.54
Fr = fraction of total heat release rate owing to radiation	0.4
Qr = total radiant heat release rate (BTU/s) of fire	28800

Room and Target Parameters	
Room Ambient Temperature (F)	80
Target Damage Temperature (F)	698
Is target vertical cable run?	n
Detection Device Temperature Setpoint (F)	160
Detection Device Time Constant	70
Floor Area (ft2)	4036
Floor Width (ft)	30
Suppression Time Given Actuation	300
Source: NUREG/CR-5384 Table 4.1for suppression time given actuation	

Fuel Parameters			Assumptions	
Fuel	Lube oil			
Heat of Combustion	20000	Btu/lbm	conservative value to cover all liquid pool fires	source: Marks' 9th ed.
Density	50	lbm/ft3	approx. value for above substance	source: Marks' 9th ed.; Hdbk of Chem & Physics 77th ed.
Amount of Fuel	5.0	gal	the walkdown found no container larger than 3 gallons	
Flame Temperature	2600	F		
Unconfined Liquid fuel?	y			
Mass loss rate or unit heat release rate	120.0	BTU/s-ft2	approx. value for above substance	source: Marks' 9th ed.
If unconfined liquid fuel: fuel specific area	120	ft2/gal	high value for lube oil	source: FIVE Mobil DTE 797
If solid fuel or confined liquid, estimated area of source	n/a	ft2		
Estimated time to fire burn-out from data (solid or confined liquid)	0	sec.	From Fig. 4 in FIVE	

Table 4.20
Typical Input Data Used in Fire Damage Calculations

Quantity	Liquid Pool Fire	Trash Can Fire	Cabinet Fire	Comment and Reference
Peak Heat Release Rate	120 ft^2 BTU/sec-	350 BTU/sec	1233 BTU/sec (Fire located on top of cabinet.)	NUREG/CR-5384 for cabinet fire; FIVE for others. [NRC, 1989f]
Heat of Combustion	20,000 BTU/lbm	not applicable	not applicable	Conservative value to cover fuel oils from Marks Mechanical Engr. Handbook, 9th edition. [Marks, 1986a]
Burn Time	Calculated from quantity, heat of combustion, and heat release rate	30 minutes	50 minutes	FIVE [EPRI, 1993b]for trash can fire; NUREG/CR-5384 [NRC, 1989f] for cabinet fire.
Fuel Density	50 lbm/ft^3	not applicable	not applicable	Approximate value for fuel oils from Handbook of Chemistry and Physics, 77th edition and Marks, 9th edition.
Detection Device Time Constant	70	70	70	FIVE [EPRI, 1993b - Table A-6E] for HCGS specified spacing of 25 feet and detector actuation temperature of 160F. This is a conservative formulation which does not account for the fast acting, rate sensitive Fenwall detectors installed at the HCGS.

Table 4.21
Treatment of Fire Dampers and Openings

Location of Fire Damper or Opening	Treatment within the IPEEE
Fire dampers or opening between rooms within a single screened out compartment	Such fire dampers or openings are not significant to fire scenarios at the HCGS and were screened out.
Fire dampers or openings between rooms of adjacent screened out compartments	Such fire dampers or openings are not significant to fire scenarios at the HCGS and were screened out.
Fire dampers or openings between rooms within a single unscreened compartment	Such fire dampers were assumed to fail open. That is, they were treated as an opening between rooms at the elevation of the damper vent. The fire PSA included fire scenarios to investigate the possibility of hot gas layer damage to cables in neighboring rooms. The fire damage model calculations demonstrated that the hot gas layer can not damage cables in neighboring rooms. This is because the locations of the damper or openings (more than 6 feet below the ceiling), the height of the cables above the fire source, and the size of the room reduces the volumetric heat content (BTU/ft^3) of the hot gas layer in the neighboring rooms to below that required to cause cable damage.
Fire dampers or openings between rooms of adjacent unscreened compartments.	Same treatment as fire dampers or openings between rooms within a single unscreened compartment.
Fire dampers or openings between rooms in unscreened and screened compartments.	The results of the fire damage studies of unscreened compartments apply here, as well. A specific example would be the dampers between the diesel-generator rooms (unscreened) and corridor 5315 (screened out). Fires in the corridor can not harm equipment, cables or the bus bars in the diesel generator rooms.

Table 4.22
Summary of Fire Damage and Suppression Results and Sensitivity Studies

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before Damage
			Plume	Ceiling Jet	Hot Gas Layer	Radiation	
4101/4201	cabinet	cables	yes, <13' above cabinet	no	no	no	possible for cables between 11' and 12' above source
	liquid pool in 4101 or 4201	cables in 4101 or 4201	yes	no	no	no	
	liquid pool in 4201	cables in 4202	n/a[2]	n/a	no	no	
4105	pump	cables	n/a	no	no	no	no auto-suppression system
	liquid pool	cables	yes	no[3]	no [3]	yes, <4' from source	
	trash can	cables	no	no	no	yes, <2' from source	
4107	jockey pump	cables	n/a	no	no	no	no auto-suppression system
	RHR pump	cables	n/a	no	no	no	

[2] 2 an n/a indicates that target is not within the region affected by this damage mechansim

[3] Fuel is consumed before damage is calculated to occur

Table 4.22
Summary of Fire Damage and Suppression Results and Sensitivity Studies (Continued)

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before
			Plume	Ceiling Jet	Hot Gas Layer	Radiation	
4107	cabinet	cables	yes	n/a	n/a	n/a	
	liquid pool	cables	yes	n/a	no	yes, <13' from source	
	trash can	cables	no	no	no	yes, <3' from source	
	liquid pool in 4107	cables in 4108	n/a	yes, <10' from source	no	no	
4109/4106/4208	RHR pump	cables	no	no	no	no	no auto-suppression system
	liquid pool	cables	no [3]	no [3]	no [3]	no [3]	
4113/4212/4214	RHR pump	cables	no	no	no	no	no auto-suppression system
	liquid pool	cables	no [3]	no [3]	no [3]	no [3]	
4202	CRD pump	cables	n/a	no	no	no	no auto-suppression system
	cabinet 10C026	cables	yes	no	no	no	
	cabinet 10C281	cables	n/a	no	no	no	

3 fuel is consumed before damage is calculated to occur

Table 4.22
Summary of Fire Damage and Suppression Results and Sensitivity Studies (Continued)

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before Damage
			Plume	Ceiling Jet	Hot Gas Layer	Radiation	
4202	liquid pool	cables	yes	no [3]	no	no	
	trash can	cables	no	no	no	no	
	liquid pool in 4202	cables in 4203	n/a	n/a	no	no	
4209/4211/4213	RACS A pump	cables	n/a	n/a	no	no	no auto-suppression system
	RACS B pump	cables	n/a	n/a	no	no	
	cabinet 1AE217	cables	n/a	n/a	no	no	
	cabinet 10C202	cables	yes	n/a	no	no	
	small transformer[4]	cables	yes	n/a	[illegible]	no	
	liquid pool	cables	yes	n/a	no	yes, < 14' from source	

3 fuel is consumed before damage is calculated to occur

4 Small transformer was simulated as a cabinet fire

Table 4.22
Summary of Fire Damage and Suppression Results and Sensitivity Studies (Continued)

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before Damage
			Plume	Ceiling Jet	Hot Gas Layer	Radiation	
4209/4211/4213	trash can	cables	no	no	no	yes, <3' from source	
	liquid pool in 4213	cables in 4215	n/a	n/a	no	no	
4301/4309/4310/4311	cabinet	cables	yes	no	no	no	yes, for cables > 10' above source
	liquid pool	cables	yes	yes	no	no	no
4303	cabinet	cables	yes	no	no	no	no auto-suppression system
	liquid pool	cables	yes	yes	no	no	
4315/4317/4320/4322	liquid pool in 4315	cables in 4315, 4317, 4320, 4322	yes, in 4315 only	n/a	no	yes, in 4315 only	no auto-suppression system
	liquid pool in 4317	cables in 4317, 4320, 4322	yes, in 4317 only	no	no	no[3]	no auto-suppression system
4328/4331/4332	liquid pool	cables	no [3]	n/a	no	no [3]	no auto suppression system
5202	liquid pool under cable tray	cables	yes	no	no	no [3]	no

[3] fuel is consumed before damage is calculated to occur

Table 4.22

Summary of Fire Damage and Suppression Results and Sensitivity Studies (Continued)

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before Damage
			Plume	Ceiling Jet	Hot Gas Layer	Radiation	
5202	liquid pool on floor between cable trays	cables	yes, cables in adjacent trays damaged	no	no	no [3]	no
5207/3204	liquid pool	cables	yes, < 18' above floor	no	no	yes, <14' from source	yes, for cables > 23' above floor
5237	liquid pool	cables	yes, < 18' above floor	no	no	yes, <14' from source	yes, for cables > 23' above floor
5301/3314	liquid pool	4kV bus bars	no	no	no	no	no auto-suppression system
	liquid pool in 5301	cables in 5339	n/a	no	no	no	
5302	cabinets	overhead cables	yes	no	no	no	no auto-suppression system
	cabinets	cables over adjacent row of cabinets	no	no	no	no	
	trash can	cables	no	no	no	no	
5304, 5305, 5306, or 5307	20 gallon liquid pool	4kV bus bars	yes	yes	n/a	n/a	no
	11 gallon liquid pool	4kV bus bars	no[3]	no[3]	n/a	n/a	

[3] fuel is consumed before damage is calculated to occur.

Table 4.22
Summary of Fire Damage and Suppression Results and Sensitivity Studies (Continued)

Compartment	Source	Target	Damage Mechanism				Auto-Suppression Before Damage
			Plume	Ceiling Jet	Hot Gas	Radiation	
5304/5305/ 5306 or 5307	7 gallon liquid pool	4kV bus bars	no	no	n/a	n/a	
5339	liquid pool	cables	yes	no	no	no	yes, > 10' from floor
	liquid pool	4kV bus bars	no, for < 18 gallons	no, for < 18 gallons	no	no	
5401/3425	liquid pool	cables	yes	no	no	yes, < 14' from source	yes, > 10' from floor
5423[5]	liquid pool	cables	yes	no	no	no	no auto-suppression system
5450[5]	liquid pool	cables	yes	no	no	no	no auto-suppression system
5501	liquid pool	cables	yes	yes	no	no	no auto-suppression system
5620	liquid pool	cables	yes, < 10' from floor	n/a	no	no	no auto-suppression system

[5] 5423 and 5450 are diesel engine combustion air intake rooms and are open to the outside. Neither jet nor hot gas layer can form in these rooms.

Table 4.23
Typical Input Values for Detection and Suppression Analysis

Quantity	Reference and Typical Values
T_A	The room ambient temperature was taken to be 80F except for the diesel generator rooms which are typically about 96F when ventilation is running.
T_R	The rated detector actuation temperature is specific to each detection. Typical values range from 140F to 160F.
T_G	The hot gas temperature at the detector was calculated using the formulation shown in Table 4.18, as derived from FIVE. [EPRI, 1993b]
H_D	The distance from the detector to the source varies by scenario. All detectors in unscreened rooms were located on the ceiling. Therefore, all were assumed subject to the ceiling jet.
V_G	Velocity of gas flowing past the detector is a function of the fire heat release rate, ceiling height, and distance from the fire to the detector. It was calculated using Equation 4 in Section 2/Chapter 4 of [NFPA, 1995a].
t_A	The actuation delay time is used at the HCGS for carbon dioxide deluge systems. A 60 second delay time is built into the system in the diesel generator room.
t_I	The soak time for water sprinkler systems was taken from [NRC, 1989f] as five minutes. The soak time for the carbon dioxide system in the diesel generator room was taken from PSE&G flow tests as 65 seconds for a liquid pool fire on the floor.
τ	The detection time constant was derived from Table A-6E of Attachment 10.4 of FIVE. PSE&G specifications typically call for about 20 feet of separation. A typical value used in this analysis is 70. Note that the HCGS uses Fenwall detectors which respond to both the temperature and the rate of temperature rise using an algorithm that provides fast response times. Therefore, use of Table A-6E values would tend to give longer detector response times than would be expected from the Fenwall detectors.

Table 4.24 Fire Scenario Analysis Template Used for Each Unscreened Compartment

Fire Scenario Analysis for Compartment:	Compartment Name:

Room Inventory From Fire Hazard Analysis (PSE&G, 1995f - Table 9A)) and MMIS List Input the compartment inventory.	

Detection and Suppression Features	
Automatic Detection?	List the automatic and manual detection and suppression equipment in the compartment
Automatic Suppression?	
Manual Suppression?	

Walkdown Insights	
Hazards:	Summarize the walkdown notes and insights that inclucence the development of fire scenarios (e.g sources and distances to targets). Also
Barriers:	note large openings between compartments. Note significant fire protection and drainage features. Include quantity of lubrication or fuel oil found in room.
Fire Protection:	

Table 4.24
Fire Scenario Analysis Template Used for Each Unscreened Compartment (Continued)

Stationary Source Scenario Descriptions Describe, in detail, scenarios initiated by stationary fire sources in the compartment. Includes cource, fire size, target, and initiating events. Include scenarios that have a potential for fire spread through openings noted above to other compartments.

Transient Source Scenario Descriptions Describe, in detail, scenarios initiated by postulated transient combustible fuel sources in the compartment Include source, fire size, target, and initiating event. Include scenarios that have a potential for fire spread through openings to other compartments.

Fire Damage and Suppression Studies by Scenario Using the FireTran.xlt calculation sheets, derived from the methodology of FIVE, to determine the potential for target damage for each fire scenario. If target damage can occur and a fire suppression system is present in the compartment, determine the relative timing of damage and suppression.

Table 4.24
Fire Scenario Analysis Template Used for Each Unscreened Compartment (Continued)

Quantification of CDF	
Credit for suppression?	Note the conclusion reached from the above calculations as to whether fire suppression can be credited.
Initiating Event (s):	List the initiating events to be used in the scenario quantification Comment on the possibility of hot shorts to compromise or actuate equipment owing to fires in this compartment Comment on the possibility of transient induced LOCAs owing to fires in this compartment.
Quantify the core damage frequency of each scenario using the fire frequency, probability of non-extinguishment, likelihood of target damage, and conditional core damage frequency.	

Summary CDF	
The total CDF = sum of the scenarios =	
	Sum the above scenario frequencies to obtain a total compartment core damage frequency.
TOTAL CDF ( ) = /yr	

Table 4.25 Exemplar Fire Scenario Analysis Worksheet

Fire Scenario Analysis

Compartment: 5304 Compartment Name: Diesel Generator Room D

Room Inventory From Fire Hazard Analysis (PSE&G, 1995f Table 9A) and MMIS List

Diesel Generator 1DG400
Diesel Generator 1DG400 auxiliary equipment
Channel D cables
Control Panels

Detection and Suppression Features

Automatic Detection?	infrared photoelectric
Automatic Suppression?	heat actuated CO2 total flooding
Manual Suppression?	none

Walkdown Insights

Hazards: All cable is in conduit. Room contains diesel generator and 550 gallon fuel oil day tank. The ceiling is about 28' above floor. Top of diesel generator and top of day tank are about at 15 - 17' above floor.
Both 1E 4kV offsite power bus bar enclosures run through the room over the diesel generator.
The diesel generator is the most significant fire ignition source and it is a source only when operating.
The room is about 30' wide and has an area of 1455 ft^2.

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

	The room is frequently occupied.
Barriers:	sixteen ft^2 vent with fire damper opens to Corridor 5315 The corridor has no additional safe shutdown equipment. Spilled fuel is able to flow under door into corridor.
Fire Protection:	Transient combustibles found include oil cloths Fuel includes #2 fuel oil and lube oil

Stationary Source Scenario Descriptions

A fire in a diesel generator room is not generally a dominant fire risk contributor. At HCGS, however, the 1E 4kV offsite power bus bars run through the room. Loss of the bus bars causes automatic start of the diesel generators and subsequent load shedding leading to reactor trip and a demand to start the diesels. Typically, a postulated fire is assumed to fail the diesel. At HCGS, a fire protection system actuation will cause ventilation isolation of the room and the diesels will trip because of room overtemperature. Core damage could occur if additional random failures would prevent core cooling. The key question, therefore, is whether a fire damages the bus bars before it can be suppressed. An important factor in this determination is the room ventilation while the diesel generators are running. Ventilation tends to disrupt the ceiling jet and hot gas layers delaying the heating of cables or bus bars on the ceiling. The diesel area exhaust system exhausts air to the outside from the diesel generator rooms, the diesel fuel storage tank rooms, diesel generator room recirculation fan rooms, and corridors 5537, 5604 and 5702. This system will remove smoke and hot gas. The total air exchange rate in the room is approximately 18000 cfm. Damage to cables or bus bars is caused by a sustained contact with hot gas or sustained

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

exposure to thermal radiation from a severe fire. Because the bus bars are on the ceiling only the former (hot gas) is a realistic hazard. Because of the high exchange rate, the potential for preventing bus bar failure should be investigated.
A complication is the fact that actuation of the suppression system (set at 160F) for the ceiling located detectors would cause the ventilation system to isolate. Once this has occurred, the benefit of ventilation would be lost but the benefit of suppression before damage would become relevant.

The bus bar duct is aluminum and per discussion of room 5301, the Al melting temperature is considered a good damage criterion (~1220F).

Historically, there have been different types of diesel generator fires. The most common type is a spray of oil or fuel onto a hot surface (the exhaust manifold is the most common ignition surface)

Chemetron Fire Systems conducted tests and calculations for deep seated fires. Assuming a 50% mix of CO_2 is needed in the room to suppress such fires, suppression would occur 117 seconds after discharge begins.

Other types of DG fires are 1) a leak that sprays onto the floor, collects in a pool, and is ignited and 2) an internal explosion of in the diesel generator which releases a large quantity of burning fuel.

Chemetron tests and calculations assume that a 34% mix of CO_2 is needed to suppress such pool fires. Suppression would occur 65seconds after discharge begins.

The system in the diesel generator rooms have a 60 second delay time after a fire is detected by the Fenwall heat detectors which are set at 160F.

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

Based on the fire damage discussion below, three scenarios can be defined for this room

5304_1 Stationary Ignition Source: Type 1 DG fire

Fire resulting from spray onto diesel generator hot surfaces, or fire internal to a DG such as in exhaust system or turbocharger or a pool fire of 11 gallons or less results in loss of diesel generator but not a loss of station power. An MSIV Closure reactor trip is assumed for purposes of calculating conditional core damage frequency.

5304_2 Stationary Ignition Source: Type 2 DG fire

Large pool fire of more than 11 gallons resulting in LOOP and loss of a DG. Sensitivity study below indicates that an 11 gallon pool fire would burn out before bus bar damage.

Transient Source Scenario Descriptions

Transient sources would have less severe effects than the stationary sources and are of much lower frequency. A quantitative probabilistic analysis is unnecessary.

Fire Damage and Suppression Studies by Scenario

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

5304_1pool

The calculation sheet pool4kV5304.xls helps examine the potential failure of the bus bar. The effects of ventilation relied upon Table 9E of the FIVE document.
As an example, using pool4kV5304.xls, a pool with a 1500
BTU/sec
heat release rate is postulated.
The ambient room temperature while the DG is running is about 90F with the ventilation system operating. The calculation sheet indicates that
a ceiling jet temperature rise owing to this fire, in the absence of ventilation, would be about 145F. With ventilation, Table 9E provides, an average room temperature rise of 103F (x=155, y=17500). Either of these would be sufficient to trip the suppression system and isolate
ventilation.
The heat release rate of the fire, however, was held artificially low for this example.

A more realistic heat release rate is derived from Five Table 2E which gives a unit heat release rate of about 120 BTU/s-ft^2 with a postulated 20 gal. pool fire.

A reasonable but conservative fuel confinement area for this room would be 100 ft^2 because of limitations of available floor space. This gives about 12000BTU/sec total heat release rate.
A 20 gallon fire would burn about four minutes with this constant heat release rate.

Again the ceiling jet temperature rise is well above the detector set-point which would isolate ventilation. The model (pool4kV5304.xls) also indicates that the Aluminum duct would reach its damage criterion using the steady state model.
The estimated time to damage is calculated at about two minutes.

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

The estimated time to detector actuation is calculated to be about 20 second.
The suppression system has a built in 60 sec time delay.
Adding a 65 second suppression time after the delay, yields more than two minutes for suppression for a Type 2 DG fire.
Adding a 117 second suppression time after the delay, yields more than three minutes for suppression for a Type 2 DG fire.

Another note on this method is that a standard heat actuated detector delay time of 70 sec. was used here (from FIVE/NFPA data). Credit for possible faster reaction times owing to the Fenwal detectors used at Hope Creek, however, would not change the above conclusion. Even if detector time were 0, it would be difficult to justify suppression before damage.
See pool4kV5304b.xls for calculation showing damage before extinguishment of 20 gallon pool fire using a 65 sec CO_2 exinguishment time.

By way of sensitivities, 1)calculation indicates that an 11 gallon fire would burn out before damage.
2) calculation indicates that 7 gallon fire would not have sufficient energy to cause the bus bar duct damage criterion to be exceeded. (used pool4kV5304a.xls)

Quantification of CDF

Credit for suppression?	no, see above discussion
Initiating Event (s):	per scenario description above

Top Down Overview

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

CCDP (LOOP) = 1.25E-02 (ref: Scrnanlysis.xls - F5304)

Upper Bound CDF = 1.03E-04 /yr

A review of the 1994 version of Sandia Fire Database reveals 27 incidents of DG caused fires which occurred during power operation. All incidents occurred during routine testing of the DGs.

1 of the 27 (Grand Gulf, 9/4/82) might potentially be a Type 2 fire.

In this fire automatic suppression failed to actuate. The fire brigade put the fire out in 25 minutes. The data does not contain the amount of fuel or lube oil involved in the fire. A Type 2 fire, as described above, is at least 11 gallons of liquid fuel in a fully involved fire. The data does not provide the amount of fuel involved or the fire characteristics. Rather than simply assuming that this fire was a Type 2, we can use the data base information in a Bayesian analysis to derive a value that reflects the uncertainty. Let us give a 50% weight to the hypothesis that it was a Type 2 fire. Therefore, the weight to the hypothesis that it was not a Type 2 fire is also 50%.

The probability of a Type 2 DG fire given the hypothesis Grand Gulf fire was Type 2 is 0.037 (=1/27).

The probability of a Type 2 DG fire given the hypothesis that Grand Gulf fire was NOT a Type 2 is 0.012.

Weighing each hypothesis by 0.5 provides the estimated probability of a Type 2 fire, given the evidence,

P(Type 2 Fire/DG fire/evidence) = 0.0245

P(Type 1 Fire/DG fire/evidence) = 0.9755

5304_1 Stationary Ignition Source: Type 1 DG fire

Reactor trip MSIV closure and failure of Channel D DG

Fire Frequency (/yr)= 8.00E-03 (ref: C_dgisds.xls multiplied by P(Type 1 fire//)

CCDP for MSIV closure and DG Failure = 1.26E-05 (ref: Scrnanlysis.xls - F5101)

CDF(5304_1) = 1.01E-07 /yr

5304_2 Stationary Ignition Source: Type 2 DG fire

LOSP and Failure of Channel D DG

Table 4.25 Exemplar Fire Scenario Analysis Worksheet (Continued)

Fire Scenario Analysis

Fire Frequency (/yr)= 2.01E-04 (ref: C_dgisds.xls multiplied by P(Type 2 fire//)
CCDP for LOSP and DG Failure = 1.25E-02 (ref: Scrnanlysis.xls - F5304)
CDF(5304_2) = 2.51E-06 /yr

Summary CDF
The total CDF = sum of the scenarios =

5304_1MDN	1.01E-07	Stationary Ignition Source:	Type 1 DG fire
5304_2LDN	2.51E-06	Stationary Ignition Source:	Type 2 DG fire

TOTAL CDF (5304) = 2.61E-06 /yr

Table 4.26
Review to Interfacing LOCA Pathways for Fire Scenario Susceptibility

Pathway	Isolation	Susceptible to Fire Scenario?
RHR Pump Discharge Lines (LPCI)	Stop check inside containment; Closed MOV outside of containment	No
RHR Shutdown Cooling Suction Line	Closed MOV inside containment; Closed MOV outside of containment	No
RHR Shutdown Cooling Discharge (return to reactor) Lines	Stop check inside containment; Closed MOV outside of containment	No
RHR Vessel Head Spray Line	Check valve and closed MOV inside containment; closed MOV outside of containment	No
Core Spray Discharge Lines	Stop check inside containment; Closed MOV outside of containment	No

Table 4.27 HCGS Fire PSA CDF

The fire induced CDF of the unscreened compartments at the HCGS =	8.1E - 05/yr.

Table 4.28
Fire IPEEE CDF Results by Building

Building	CDF (/yr.)
Reactor Building	8.0E-06
Control/Diesel Building	7.0E-05
Turbine Building	2.0E-06
Radwaste Building	7.3E-07
Switchyard	3.0E-07

Table 4.29
Fire IPEEE CDF Results by Compartment

Location	Description	Initiating Events	CDF (/yr.)
AUX, 137', 5510/5511	Control Room	MSIV Closure, LOOP, SORV, Loss of HVAC, Loss of SWS/SACS	2.5E-05
AUX, 130', 5416/5417	Class 1E Switchgear Room (Ch. A)	MSIV Closure	1.3E-05
AUX, 102', 5307	Diesel Generator Room (Ch. A)	LOOP, MSIV Closure	5.3E-06
RB, 77', 4202	CRD Pump Area	MSIV Closure	4.2E-06
AUX, 102', 5306	Diesel Generator Room (Ch. B)	LOOP, MSIV Closure	4.1E-06
AUX, 102', 5305	Diesel Generator Room (Ch. C)	LOOP, MSIV Closure	3.7E-06
AUX, 130', 5412/5413	Class 1E Switchgear Room (Ch. B)	MSIV Closure	3.0E-06
AUX,137',5501	Electrical Access Room	MSIV Closure	3.0E-06
AUX, 102', 5339	Electrical Access Area	LOOP, MSIV Closure	2.7E-06
AUX, 163.6', 5605/5631	Upper Control Equipment Room/Computer Rm.	MSIV Closure	2.7E-06
AUX, 102', 5304	Diesel Generator Room (Ch. D)	LOOP, MSIV Closure	2.6E-06
AUX, 124', 5401/3425	Electrical Access Area	MSIV Closure	2.0E-06
RB, 102', 4301/4309/4310/4311	Rx Bldg. 102' Elevation-North Side & Div I SACS area	MSIV Closure	1.8E-06
AUX, 102', 5302	Lower (Control) Electric Equipment Rm.	MSIV Closure, LOOP, SORV	1.7E-06
TB, 102', 1315/1316/1317/1320/1321/1322	Access and Unloading Area	LOOP	1.2E-06
RB, 102', 4303	MCC Area	MSIV Closure	1.2E-06
		TOTAL of Top 16 Compartments =	7.7E-05

Table 4.30
Fire IPEEE Results by Ignition Source

Ignition Source	CDF (/yr.)
Cabinet	5.7E-05
Pump	9.9E-08
Transformer	1.6E-07
Diesel-Generator	1.6E-05
Heaters	2.0E-06
Compressor	5.3E-07
Shop	4.9E-07
Transient	5.8E-06

Table 4.31
Fire IPEEE Results by Initiating Event

Initiating Event	CDF(/yr.)
MSIV Closure	5.3E-05
Loss of Offsite 4kV Power	2.2E-05
Loss of HVAC	7.2E-07
LOCA (SRV or ADS Actuation)	4.9E-07
Loss of SWS/SACS	5.1E-06
Use of RSP (for fires outside of control room)	2.9E-07

Table 4.32
Fire IPEEE Results by Electrical Channel or Division

Failed Channel or Division	CDF(/yr.)	Largest Contributor
None	2.0E-05	DG Rooms with LOOP
A only	1.8E-05	Channel A Switchgear Room 5416
B only	5.1E-06	Channel B Switchgear Room 5412
C only	7.9E-08	
D only	4.0E-07	
Division I (A & C)	5.8E-06	Room 5339 with LOOP
Division II (B & D)	6.2E-06	CRD Pump Room 4202
Control from RSP	2.3E-05	Control Room
Other Combinations	3.5E-06	Control Room

Table 4-33 Fire IPEEE Results by Scenario

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)**
RB. 54' 4101/4201	Torus Water Cleanup Room/ MCC Room	Stationary and Transient Fires Initiated in 4101.	MSIV Closure		0.0E-00
		Stationary Ignition Source: Cabinet Fire Fails Channel D.	MSIV Closure	1.1E-03	3.0E-08
		Transient Combustibles Fires Initiated in 4201	MSIV Closure	3.5E-06	9.7E-11
		Transient Combustible Fires at Interface with 4202	MSIV Closure		0.0E-00
RB. 54' 4105	Core Spray DP206 Room	Stationary Pump Fire	MSIV Closure	2.9E-04	5.6E-09
		Liquid Pool Fires	MSIV Closure	1.1E-07	3.1E-09
		Trash Can Fires	MSIV Closure	1.2E-07	3.4E-09
RB. 54' 4107	RHR Pump DP202 Room	Stationary Ignition Source: Jockey Pump	MSIV Closure	2.9E-04	6.0E-09
		Stationary Ignition Source: RHR Pump	MSIV Closure	2.9E-04	6.0E-09
		Stationary Ignition Source: RHR Cabinet	MSIV Closure	1.4E-04	3.9E-09
		Transient Ignition Source: Liquid Pool Fire	MSIV Closure	5.0E-06	1.4E-07
		Transient Ignition Source: Trash Can Fire	MSIV Closure	8.7E-08	2.5E-09
		Transient Pool Fire near Room 4108	MSIV Closure		0.0E-00
RB. 54/77' 4109/4206 4208	RHR HX Room (BP202 and HX BE205)	Stationary Ignition Source: RHR Pump	MSIV Closure	2.9E-04	7.6E-09
		Transient Ignition Source: Liquid Pool Fire	MSIV Closure	5.0E-07	7.6E-09

** A 0.0E-00 result indicates that the fire damage calculations determined that this fire would cause no damage to the target.

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)**
RB. 54/77' 4113/4212 4214	RHR Pump AP202 and HX AE205 Room (and Vestibule)	Stationary Ignition Source: RHR Pump	MSIV Closure	2.9E-04	7.6E-09
		Transient Ignition Source: Liquid Pool Fire	MSIV Closure	1.0E-06	3.1E-09
RB. 77' 4202	CRD Pump Area	Stationary Ignition Source: CRD Pumps	MSIV Closure	5.7E-04	7.2E-09
		Stationary Ignition Source: Cabinet 10C026 - RHR Control Panel	MSIV Closure	1.4E-04	4.0E-06
		Stationary Ignition Source: Cabinet 10C266 - Division II Control Panel	MSIV Closure	1.4E-04	1.8E-09
		Stationary Ignition Source: Cabinet 10C281 - Division II Control Panel	MSIV Closure	1.4E-04	1.8E-09
		Transient Pool Fire Sources	MSIV Closure	5.0E-06	1.4E-07
		Transient Trash Can Fire Source	MSIV Closure		0.0E+00
		Transient Pool Fire Source Near Room 4203	MSIV Closure		0.0E+00
RB. 77' 4209/4211 4213	RACS Pump and HX Area	Stationary Ignition Source: RACS Pumps	MSIV Closure	8.6E-04	1.1E-08
		Stationary Ignition Source: Cabinets 1AE217 and 1BE217	MSIV Closure	2.7E-04	3.4E-09
		Stationary Ignition Source: Cabinet 10C202	MSIV Closure	1.4E-04	9.6E-08
		Stationary Ignition Source: Transformer 1AC267	MSIV Closure	5.2E-05	1.6E-07
		Stationary Ignition Source: Cabinets 10C009 and 10C010	MSIV Closure	2.7E-04	3.4E-09
		Transient Liquid Pool Fire Near Cables and Conduit	MSIV Closure	9.0E-06	3.6E-07
		Transient Pool Fire Source Near Room 4215	MSIV Closure		0.0E+00

** A 0.0E-00 result indicates that the fire damage calculations determined that this fire would cause no damage to the target.

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year) **
RB, 77' 4215	Electrical Equipment Room	Cabinet Fires	MSIV Closure	5.4E-04	1.8E-08
		Transient Liquid Pool Fire: Channel C	MSIV Closure	1.4E-06	4.7E-11
		Transient Liquid Pool Fire: Channel D	MSIV Closure	6.8E-07	1.9E-11
		Pool Fire at the Interface with Room 4216	MSIV Closure		0.0E+00
RB, 102', 4301/43094 310/4311	North Side and Division I SACS Area	Stationary Ignition Source: Breaker Cabinet 1BN205	MSIV Closure	1.4E-04	3.8E-09
		Stationary Ignition Source: Cabinets 1CC281 and 10B232	MSIV Closure	2.7E-04	9.2E-09
		Stationary Ignition Source: SACS Pumps A&C	MSIV Closure	5.7E-04	4.8E-08
		Stationary Ignition Source: Cabinets 10B212, 1AC201, 1CC201, and 1AC281	MSIV Closure	5.4E-04	1.7E-06
		Liquid Pool Fire in North 4301	MSIV Closure	4.9E-06	1.4E-10
		Liquid Pool Fire in East 4310 along North Wall	MSIV Closure	2.3E-06	7.8E-11
		Liquid Pool Fire in 4310 along Curved (South) Wall and Room 4309	MSIV Closure	4.3E-06	1.3E-08
RB, 102' 4303	MCC Area	Stationary Ignition Source: MCC 10B222 and three other cabinets	MSIV Closure	1.6E-03	1.1E-06
		Liquid Pool Fire Under Channel B Cables	MSIV Closure	1.1E-05	8.1E-09
		Liquid Pool Fire Under Channel D Conduit	MSIV Closure	1.8E-06	5.0E-11
		Liquid Pool Fire In West Part of Room 4303	MSIV Closure	7.6E-07	7.5E-09

** A 0.0E-00 result indicates that the fire damage calculations determined that this fire would cause no damage to the target.

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year) **
RB. 102' 4315/4317 4320/4322	Inside Cylinder - South Side (Division II)	Stationary Source: Cabinet Fire in 4317	MSIV Closure	1.5E-03	1.9E-08
		Transient Liquid Pool Fire in 4315 and 4317	MSIV Closure	9.6E-06	2.7E-10
		Stationary Source: Cabinet Fire in 4320	MSIV Closure	1.5E-03	1.9E-08
		Transient Liquid Pool Fire in 4320 and 4322	MSIV Closure	3.2E-06	2.3E-09
RB. 102' 4328/4331 4332	Inside Cylinder - North Side (Div.I)	Transient Liquid Pool Fire Affects Channel A and C Trays in 4328 and 4331	MSIV Closure	1.9E-06	6.6E-09
		Transient Liquid Pool Fire Affects Channel A Conduit in Room 4332	MSIV Clousre	4.7E-07	1.5E-09
Aux. 77' 5202	Cable Spreading Room	Transient Liquid Pool Fire Affects Channels B and D	MSIV Closure	2.9E-08	8.2E-10
		Transient Liquid Pool Fire Affects Channels B,C, &D	MSIV Closure	3.8E-08	1.6E-09
		Transient Liquide Pool Fire Affects Channels A,B, &C	MSIV Closure	3.8E-08	3.8E-08
		Transient Liquid Pool Fire Affects Channels A and C	MSIV Closure	3.8E-08	1.3E-10
		Transient Liquid Pool Fire Affects Channel D Cable	MSIV Closure		0.0E+00
		Cable Fires Caused by Welding - Affect Individual Channels Only	MSIV Closure	2.3E-05	1.8E-08
AUX/RW. 77' 3204/5207	Electrical Access Area/Corridor	Transient Liquid Pool Fire Damages Channel B	MSIV Closure	4.5E-07	3.2E-10
		Transient Liquid Pool Fire Damages Channel D	MSIV Closure	1.8E-06	5.1E-11
		Transient Liquid Pool Fire Damages Channels B and D	MSIV Closure	3.1E-07	8.8E-09

** A 0.0E-00 result indicates that the fire damage calculations determined that this fire would cause no damage to the target.

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX, 77' 5237	Electrical Access Area (Division I)	Transient Liquid Pool Fire Affects Channel A Cable	MSIV Closure	2.3E-06	7.2E-09
		Transient Liequid Pool Fire Affects Channel B Cable	MSIV Closure	2.3E-07	1.7E-10
		Transient Liquid Pool Fire Affects Channel A and B Cable	MSIV Closure	2.3E-07	8.2E-08
AUX, 102' 3314/5301	Aux. Elect. Access Area and Common Area in RW Bldg.	Stationary Ignition Source: Cabinet Fire Affects Channel D	MSIV Closure	1.4E-04	3.9E-09
		Transient Liquid Pool Fire Affects Channel B Cables	MSIV Closure	1.6E-06	1.1E-09
AUX, 102' 5302	Lower (Control) Electric Equipment Room	Stationary Ignition Source: Channel A Cabinet Fire	MSIV Closure	2.9E-04	9.1E-07
		Stationary Ignition Source: Channel C Cabinet Fire	MSIV Closure	2.9E-04	9.9E-09
		Stationary Ignition Source: Channel B Cabinet Fire, ADS Actuation	MSIV Closure with Stuck Open Relief Valve	8.8E-05	1.6E-07
		Stationary Ignition Source: Channel B Cabinet Fire	MSIV Closure	2.0E-04	1.5E-07
		Stationary Ignition Source: Channel D Cabinet Fire	MSIV Closure with Stuck Open Relief Valve	2.9E-04	1.6E-07
		Stationary Ignition Source: BOP and NSSS Computer Cabinets	MSIV Closure	2.9E-04	3.7E-09
		Stationary Ignition Source: Annunciator Logic Cabinets	MSIV Closure	2.9E-04	3.7E-09
		Stationary Ignition Source: NSSS Instrumentation Cabinets	MSIV Closure	2.9E-04	3.7E-09
		Stationary Ignition Source: Offsite Power Protection Relay Panels	LOOP	2.9E-04	3.1E-07
		Stationary Ignition Source: Non-1E Logic Panels	MSIV Closure	1.2E-03	1.5E-08

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX. 102' 5304	Diesel Generator Room (Ch. D)	Stationary Ignition Source: Type 1 DG Fire	MSIV Closure	8.0E-03	1.0E-07
		Stationary Ignition Source: Type 2 DG Fire	LOOP	2.0E-04	2.5E-06
AUX. 102' 5305	Diesel Generator Room (Ch. B)	Stationary Ignition Source: Type 1 DG Fire	MSIV Closure	8.0E-03	1.0E-07
		Stationary Ignition Source: Type 2 DG Fire	LOOP	2.0E-04	3.6E-06
AUX. 102' 5306	Diesel Generator Room (Ch. C)	Stationary Ignition Source: Type 1 DG Fire	MSIV Closure	8.0E-03	1.0E-07
		Stationary Ignition Source: Type 2 DG Fire	LOOP	2.0E-04	4.0E-06
AUX. 102' 5307	Diesel Generator Room (Ch. A)	Stationary Ignition Source: Type 1 DG Fire	MSIV Closure	8.0E-03	1.0E-07
		Stationary Ignition Source: Type 2 DG Fire	LOOP	2.0E-04	5.2E-06
AUX. 102' 5339	Electrical Access Room	Liquid Pool Fire Causes Loss of Divisions I (no LOOP)	MSIV Closure	7.5E-05	2.7E-07
		Liquid Pool Fire Causes Loss of Division I and DGs A&C with LOOP	LOOP	2.5E-05	2.4E-06
AUX. 124' 3425/5401	Electrical Access Area	Stationary Ignition Source: Electrical Heaters Damage Division II Cable	MSIV Closure	6.8E-05	2.0E-06
		Transient Liquid Pool Fire Damages Division II Cable in Rooms 3425 and 5401	MSIV Closure	2.0E-06	5.8E-08
		Transient Liquid Pool Fire Damages Channel A and D Cable in West End of 5401	MSIV Closure	2.2E-07	1.1E-08
AUX. 117.5' 5403/5449	Control Room Equipment Room Mezzanine	Transient Liequid Pool Fire Damages Cables	Transient with Control from RSP	2.9E-06	2.9E-07
AUX. 130' 5412/5413	Class 1E Switchgear Room (Ch. B)	Stationary Ignition Source: Cabinet or Substation Ignited Fire Fails Channel B	MSIV Closure	4.2E-03	3.0E-06
AUX.130' 5416/5417	Class 1E Switchgear Room (Ch. A)	Stationary Ignition Source: Cabinet or Substation Ignited Fire Fails Channel A	MSIV Closure	4.2E-03	1.3E-05

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX. 130' 5423	DG Combustible Air Intake Room	Liquid Pool Fire Damages Channel A Cable	MSIV Closure	1.3E-07	4.0E-10
		Liquid Pool Fire Damages Channels A and B	MSIV Closure	1.6E-07	5.6E-08
AUX. 130' 5448	Class 1E Inverter Room	Stationary Ignition Source: Cabinet 1BD481 Damages Channel B	MSIV Closure	7.1E-04	1.9E-07
		Stationary Ignition Source: Cabinet 1DD481 Damages Channel D	MSIV Closure	2.7E-04	7.5E-09
		Liquid Pool Fire between Cabinets Damages Both Cabinets (Channels B and D)	MSIV Closure	6.2E-07	1.8E-08
AUX. 130' 5450	DG Combustible Air Intake Room	Liquid Pool Fire Damages Channel A Cable	MSIV Closure	4.4E-07	1.4E-09
		Liquid Pool Fire Damages Channel C Cable	MSIV Closure	2.3E-07	7.8E-12
		Liquid Pool Fire Damages Channels A and C	MSIV Closure	4.6E-08	1.6E-10
AUX. 137' 5501	Electrial Access Area	Stationary Ignition Source: Cabinet Fires Affect Channels A and C	MSIV Closure	8.4E-04	3.0E-06
		Stationary Ignition Source: Cabinet Fires Affect Channel C	MSIV Closure	6.8E-04	2.3E-08
		Liquid Pool Fire Affects Cables, Affects Division I	MSIV Closure	1.4E-05	5.1E-08

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Descrip.ion	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX. 137' 5510/5511	Control Room	Small In-Cabinet Fires	MSIV Closure	1.3E-03	3.8E-06
			LOOP	2.0E-04	7.1E-07
			Loss of HVAC	1.5E-04	4.5E-08
			Loss of SWS/ SACS	1.7E-04	4.9E-06
			MSIV Closure with Stuck Open Relief Valve	8.0E-05	1.0E-07
		Large In-Cabinet Fires	MSIV Closure	2.5E-04	1.5E-05
			LOOP	5.8E-06	1.8E-07
			Loss of HVAC	4.4E-06	1.3E-07
			Loss of SWS/ SACS	4.2E-06	1.4E-07
			MSIV Closure with Stuck Open Relief Valve	2.3E-06	6.9E-08
AUX. 163.6' 5605/5631	Upper Control Equiopment Room / Computer Room	Stationary Ignition Source: Cabinets along the South Wall	MSIV Closure	2.4E-03	3.1E-08
		Stationary Ignition Source: Channel A Cabinets	MSIV Closure	6.8E-04	2.1E-06
		Stationary Ignition Source: Channel B Cabintes	MSIV Closure	6.8E-04	4.8E-07
		Stationary Ignition Source: Channel C Cabinets	MSIV Closure	6.8E-04	2.3E-08
		Stationary Ignition Source: Channel D Cabinets	MSIV Clsoure	6.8E-04	1.9E-08
		Stationary Ignition Source: West Side Cabinets	MSIV Closure	1.3E-05	3.4E-09
		Stationary Ignition Source: All Sources in Room 5631	MSIV Closure	2.2E-03	2.8E-08

Table 4-33 Fire IPEEE Results by Scenario (Continued)

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX. 163.6' 5620	HVAC Equipment Room	Liquid Pool Fire Affects Control or Power Cables of Both AHUs	Loss of HVAC	3.0E-05	9.0E-09
AUX. 178' 5703/5704	DG Area HVAC Equipment Room	Fire in Either Compressor Causes Loss of 1E Panel Room HVAC	Loss of HVAC	1.8E-03	5.3E-07
TB. 102' 1315/13161 317/132013 21/1322	Access and Unloading Area	Stationary Ignition Sources: Cabinets under Offsite Power Ducts	LOOP	1.4E-04	4.9E-07
		Liquid Pool Fire Causes Loss of Offsite Power	LOOP	2.1E-04	7.4E-07
TB. 120' 1406/1409	Electrical Equipment Mezzanine	Sationary Ignition Sources: Cabinets Cause Reactor Trip	MSIV Closure	3.0E-03	3.9E-08
		Liquid Pool Fire Causes Loss of Offsite Power	LOOP	2.1E-04	7.4E-07
RW. 102' 3311/33123 316/331733 18/3319332 4/33283329	Middle Section of the Third Floor of the Radwaste Building	Shop Causes Loss of Offsite Power	LOOP	1.4E-04	4.9E-07
		Liquid Pool Fire Causes Loss of Offsite Power	LOOP	6.6E-05	2.3E-07
Switchyard	Switchyard Blockhouse	Stationary Ignition Sources: Panels Cause LOOP	LOOP	8.4E-05	3.0E-07

Table 4.34 Containment Failure Mode Review

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
Reactor Building			
RB.54',4101/ 4201	Torus Water Cleanup Room/MCC Room	Motor Control Center 10B242 (Channel D) Channel B cable for SSW, SACS, RCIC, RHR and CS Channel D power cable for MCC, SSW, SACS, Reactor Water Cleanup, CS, RHR, Control Area Chilled Water	no
RB, 54', 4105	Core Spray DP206 Room	Core spray pump DP206 Unit Coolers DVH211 and HVH211 Division II cable tray and conduit for SSW, CS, SACS, RCIC and RHR	no
RB, 54', 4107	RHR Pump DP202 Room	RHR Pump DP202 Jockey Pump DP228 Division II cable tray and conduit for SACS, RCIC, RHR, ADS Unit Coolers DVH210 and HVH210 RHR Rack 10C069	no
RB,54'/77',4109/ 4208/4206	RHR HX Room (BP202 & HX BE205)	RHR Pump BP202 RHR HX BE205 Division II cable tray and conduit for SSW, SACS, RCIC, RHR, CS, ADS, an RPV instr. Logic for HPCI auto transfer from CST (Div. I) RHR Unit Coolers BVH210 and FVH210	no
RB,54'/77',4113/ 4214/4212	RHR Pump Ap202 & HX AE205 Room (and Vestibule)	RHR Pump AP202 RHR HX AE205 Channel A cable trays and conduit RHR Unit Coolers AVH210 and EH210	no
RB, 77', 4202	CRD Pump Area	Division II cable tray and conduit for SSW, SACS, RCIC, reactor water cleanup, CS, RHR, Main Steam, ECCS Room Unit Coolers, RCIC testable logic Division II control panels, 10C2166 for RHR and 10C026 Reactor level and pressure instrumentation panel Two CRD pumps: 1AP207 and 1BP207	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of inventory	Containment Failure Modes Significantly Different Than IPE?
RB,77',4209/ 4211/4213	RACS Pump & HX Area	Channel A cable for SSW control and instrumentation, SACS, MCC control power, CS, HPCI, RHR, SSW valve HV2357A, RHR HX flow, RHR room cooling Channel C cable for SSW valve HV2357B, RHR Hx flow, RHR room cooling, RCIC trip, SWS bypass, RACS, SACS/TACS isolation valves Channel D cable for RCIC PT-N055D and H (RCIC trip)	no
RB, 77', 4215	Electrical Equipment Room	Division I RPV level and pressure cabinet 10C004 Channel A conduit for Post Accident Monitoring, CS testable logic, HPCI steam flow and pressure instr. RHR flow, RHR press and flow transmitter Channel C cable for SSW Channel B cables for RHR PT-N057, FRVS torus room Channel D cables: ADS logic input Cabinets 10B252 and 10C232	no

Table 4.34 Containment Failure Mode Review(Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
RB, 102', 4301/4309/4310 /4311	R Bldg. 102' Elevation- North Side & Div I SACS area	Channel C cable for SACS C pump, HX valves, RHR C Pump Motor Cooling, RHR Unit Cooler C and G valves, CS Unit Cooler C and G valves and Channel C distribution panel, SSW valves for 1A2E201 SACS HX, HPCI, PCIG isolation valves, suppression pool level instr., RHR pump C, control area chilled water valves to AVH214 and BVH214 SACS unit coolers Channel C MCC 10B232 Channel A cable for Recirc Pump trip Channel D cable for SACS valves to containment instrument gas compressor, RCIC, RHR valves for LPCI injection, shutdown cooling return and head spray, main steam relief valves (ADS) PSV F013 A, B, C, D, E Channel B cable for Recirc. pump trip. SACS Hx 1A1E201 and 1A2E201 SACS pump 1AP210 and 1CP210 SACS control panels: 1AC201, 1CC201; power supply MCC10B212 Unit Coolers 1AVH214 and 1CVH214; Valves HV2491A, HV2522A, HV2496B and HV2496D Division I Power and control cables for SSW, SACS, HPCI-MCC, recirc. pump trip, RHR, CS, HPCI and unit coolers 1AVH214 and 1BVH214 Division II Power and control cables for FT-2544B (Channel B) and FT2544D (Channel D), and unit cooler CVH214 control and instr. cable	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
RB,102', 4303	MCC Area	Channel B MCC 10B222 Channel B conduit and cable trays for MCC power, RCIC-MCC power, recirc pump trip, RHR, reactor vessel instrumentation, CS, relief valves PSV-F013 (14 of them) and primary containment instrument gas.	no
RB,102', 4315/4317/4320 /4322	El. 102' Inside Cylinder - South Side (Div. II)	Channel D SACS control cable for PCIG compressor cooler valves Channel B control cable for recirc. pump trip control power Cable for RPS Channel Y Rod Scram Circuits Channel D cable for SACS valves to PCIG compressor cooler valves (from 4315), RHR valves for LPCI injection, shutdown cooling return and head spray; ADS valves PSV-F013 A, B, C, D and E; RWCU isolation vlvs Channel B cables for SACS, RCIC, CS, RHR, SRVs (all 14), and PCIG Channel D conduit for CACS, RCIC leak detection, and RHR valves Channel A conduit for RHR flow transmitter for head spray flow Channel B cables for SACS, RCIC, CS, RHR, SRVs (all 14), and PCIG Channel D for CACS Channel A conduit for RHR flow transmitter for head spray flow 2 CRD flush tank recirculation pumps	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
RB, 102', 4331/4328/4332	El. 102' Inside Cylinder - North Side (Div. I)	Channel A cable trays for SACS, CS, HPCI, RHR and PCIGS Channel C cable trays for SACS, CS, RHR and PCIGS Channel C cable trays and conduit for SACS, CS, HPCI, RHR and PCIGS Channel B recirculation pump trip conduit Cable for HPCI valves	no
Auxiliary Building			
AUX, 77', 5202	Cable Spreading Room	Power, instrumentation and control cables for all Channels	no
AUX/RW, 77',5207/3204	Electrical Access Area/Corridor	Division II class 1E cable trays for ADS; remote shutdown panel control for RCIC, SSW, SACS, switchgear room cooling, RHS shutdown cooling valves BC-HV-F008	no
AUX, 77', 5237	Electrical Access Area (Div. I)	Division I, Class 1E cable trays containing power, control and instrumentation for PCIG valves, inboard isolation valves, diesel generator breaker indications at RSP, switchgear room cooling indication, control area HVAC, suppression pool temperature monitoring, HPCI, RHR, SSE, SACS, diesel generators	no
AUX,102', 5301/ 3314	Aux Elect. access Area & Common area in RW Bldg.	Channel B cable trays for RCIC, RSP control and instrumentation, recirc pump trip Division II cable for RHR shutdown cooling valves BC-HV-F008 and BC-HV-F015A	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
AUX, 102', 5302	Lower (Control) Electric Equipment Room	All divisions of cable and control panels	no
AUX,102', 5304	Diesel Generator Rooms (Ch. D)	Diesel Generator 1DG400 Diesel Generator 1DG400 auxiliary equipment Channel D cables Control Panels 4kV 1E offsite power bus bar	no
AUX,102', 5305	Diesel Generator Room (Ch. B)	Diesel Generator 1BG400 Diesel Generator 1BG400 auxiliary equipment Channel B cables Control Panels 4kV 1E offsite power bus bar	no
AUX,102', 5306	Diesel Generator Room (Ch. C)	Diesel Generator 1CG400 Diesel Generator 1CG400 auxiliary equipment Channel C cables Control Panels 4kV 1E offsite power bus bar	no
AUX,102', 5307	Diesel Generator Room (Ch. A)	Diesel Generator 1AG400 Diesel Generator 1AG400 auxiliary equipment Channel A cables Control Panels 4kV 1E offsite power bus bar	no
AUX,102', 5339	Electrical Access Room	Diesel Generator Intake silencers Channel A&C cable which includes the diesel A&C control power supply cable Division I cable for HPCI, RHR, SACS, DG A&C, SWS and CS 4kV 1E offsite power bus bar	no
AUX,124', 5401/3425	Electrical Access Area	Division II Cable Circuits for RHR shutdown cooling valves BC-HV-F008 and BC-HV-F015A	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
AUX,117.5',5403 / 5449	Control Room Equipment Room Mezzanine	Division I & II cable trays and conduit leading to the control room. 5449 contains only non-1E cable	no
AUX,130', 5412/5413	Class 1E Switchgear Room (Ch. B)	Channel B 4.16 kV swgr 10A402 Channel B 480V unit substations 10B420, 10B460 Channel B MCC 10B421, 10B461 Channel B 125 V dc load center 10D420 Channel B 125 V dc distribution panel 1BD417 Channel B diesel control panel 1BC422 Channel B generator control panel 1BC422 Channel B cable Channel B diesel generator sequencer	no
AUX,130', 5416/5417	Class 1E Switchgear Room (Ch. A)	Channel A 4.16 kV swgr 10A401 Channel A 480V unit substations 10B410, 10B450 Channel A MCC 10B411, 10B451 Channel A 125 V dc load center 10D410 Channel A 125 V dc distribution panel 1AD417 Channel A diesel control panel 1AD423 Channel A generator control panel 1AD423 Channel A cable Channel A diesel generator sequencer	no
AUX,130', 5423	DG Combustible Air Intake Room	Division II cable trays for RHR, CS, SACS, chilled water, RCIC, 1E inverters, DG power, DG sequencer, RSP indications, control room HVAC, RHR shutdown cooling valve BC-HV-F015A	no
AUX,130', 5448	Class 1E Inverter Room	Division II 1E inverters 1BD481 and 1DD481	no
AUX,130', 5450	DG Combustible Air Intake Room	Division I cable trays for RHR, CS, SACS, chilled water, HPCI, 1E inverters, DG power, DG sequencer, RSP indications, control room HVAC	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
AUX,137', 5501	Electrical Access Area	Division I tray and conduit Power distribution panels 10D946, 1AD481, 0AD495, and 1CD481 RSP room air handling unit 1VH316 (Channel A) Valve GJ-TV-9768A RSP cooling coil Valve GJ-TV-9768B RSP cooling coil	no
AUX,137', 5510/5511	Control Room	Operators monitor console (panel # 1OC649) Unit operators console (inner horse-shoe, panel # 1OC651) Main vertical boards (outer horse-shoe, panel # 1OC650) Control rod test and RPS division 2/4 (panel #s 1OC610 and 611) Post LOCA H2 recomb. control cabinet (panel #s 1AC633 and 1BC633) and rad. monitoring C/D (panel # 1OC636) inst. Cabinet (panel # 1OC636) TIP control (panel # 1OC607), and rad. monitoring A/B (panel # 1OC635) Class 1E rad. monit. cabinet (panel # 1OC604) and safety relief valve monitoring (panel # 1OC605) Power range neutron monitoring (panel # 1OC608) Communications eqt. monit. (panel # 1OC685) and RPS div. 1/3 (panel # 1OC609) Fire detection status cabinet (panel # 1OC671)	no

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
AUX,163.6',5605 /5631	Control Equipment Room/ Computer Room.	Division I: AC680 A, B, C and D; CC680 A, B, C, and D which are 1E electrical auxiliary cabinets for channels A and C respectively. These contain offsite power relays. AC655, CC655 which are 1E analog logic cabinets for channels A & C control room instrumentation. 10C601 is a redundant reactivity control system channel A&C Division II: BC680 A, B, C and D; DC680 A, B, C, and D which are 1E electrical auxiliary cabinets for channels B and D respectively. These contain offsite power relays. BC655, DC655 which are 1E analog logic cabinets for channels B & D control room instrumentation. 10C602 is a redundant reactivity control system channel B&D Non-1E cabinets and RMS computer HVAC unit 10C471	no
AUX,163.6',5620	HVAC Equipment Room	Diesel Area AHU 1AV408 (Channel A) and panel, Panel for Battery Room exhaust fan 1AC486 Cable for FSL-9562A1 & A2 battery room fan auto start and alarm Diesel Area AHU 1BV408 (Channel B) and panel, Panel for Battery Room exhaust fan 1BC486 Cable for FSL-9562B1 & B2 battery room fan auto start and alarm Two cable trays (Channel B and D) containing 1E power to non-1E BOP and NSSS computers	no
AUX,178', 5703/5704	DG Area HVAC Equipment Room	Both control area water chillers (1AK403 and 1BK403) with associated chilled water pumps (1AP414 and 1BP414) and HVAC panels (1A & CC483 and 1B & DC 483) Both control equipment room air handling units (1AVH407 and 1BVH407 Both control area battery unit exhaust fans (1AV410 & 1BV410)	no
Turbine Building			

Table 4.34 Containment Failure Mode Review (Continued)

Location	Description	Synopsis of Inventory	Containment Failure Modes Significantly Different Than IPE?
TB,102', 1315/1316/1317 / 1320/1321/1322	Access and Unloading Area	4kV 1E offsite power bus bars	no
TB,120', 1406/1409	Electrical Equipment Mezzanine	Non-1E Switchgear	no
Radwaste Building			
RW,102', 3311/3312/3316 /3317/3318/ 3319/3324/3328 / 3329	Middle section of the third floor of the RW Bldg.	4kV 1E offsite power bus bars	no
Yard			
Switchyard	Switchyard Blockhouse	Non-1E relays and switchgear	no

Table 4-35 Fire Induced Scenarios which Fail Decay Heat Removal

Location	Location Description	Scenario Description	Initiating Event	Fire Ignition Frequency (/year)	CDF (/year)
AUX. 77' 5237	Electrical Access Area (Division 1)	Transients Liquid Pool Fire affects Channels A and B Cable	MSIV Closure	2.3E-07	8.2E-08
AUX. 77' 5202	Cable Spreading Room	Transients Liquid Pool Fire affects Channels A, B, and C.	MSIV Closure	3.8E-08	3.8E-08
AUX. 130' 5423	DG Combustible Air Intake Room	Liquid Pool Fire Damages Channels A and B	MSIV Closure	1.6E-07	5.6E-08
AUX. 137' 5510/5511	Control Room	Small In-Cabinet Fires	Loss of HVAC	1.5E-04	4.5E-08
			Loss of SWS/SACs		
		Large In-Cabinet Fires	Loss of HVAC	4.4E-06	1.3E-07
			Loss of SWS/SACS	4.2E-06	1.4E-07
AUX. 178' 5703/5704	DG Area HVAC Equipment Room	Fire in either Compressor Causes Loss of 1E Panel Room HVAC	Loss of HVAC	1.8E-03	5.3E-07

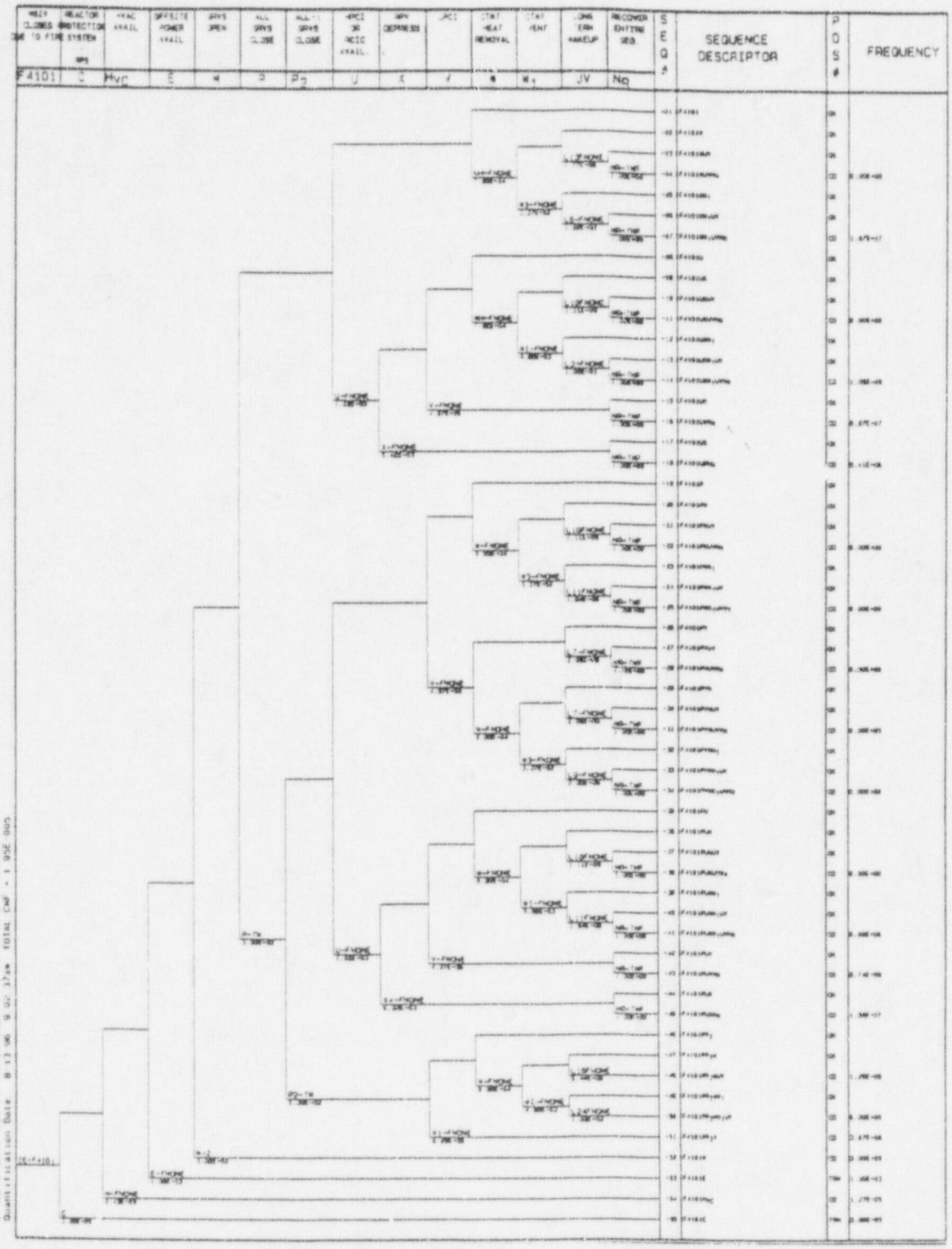

Figure 4.1
MSIV Closure Event Tree Used for the HCGS Fire IPEEE

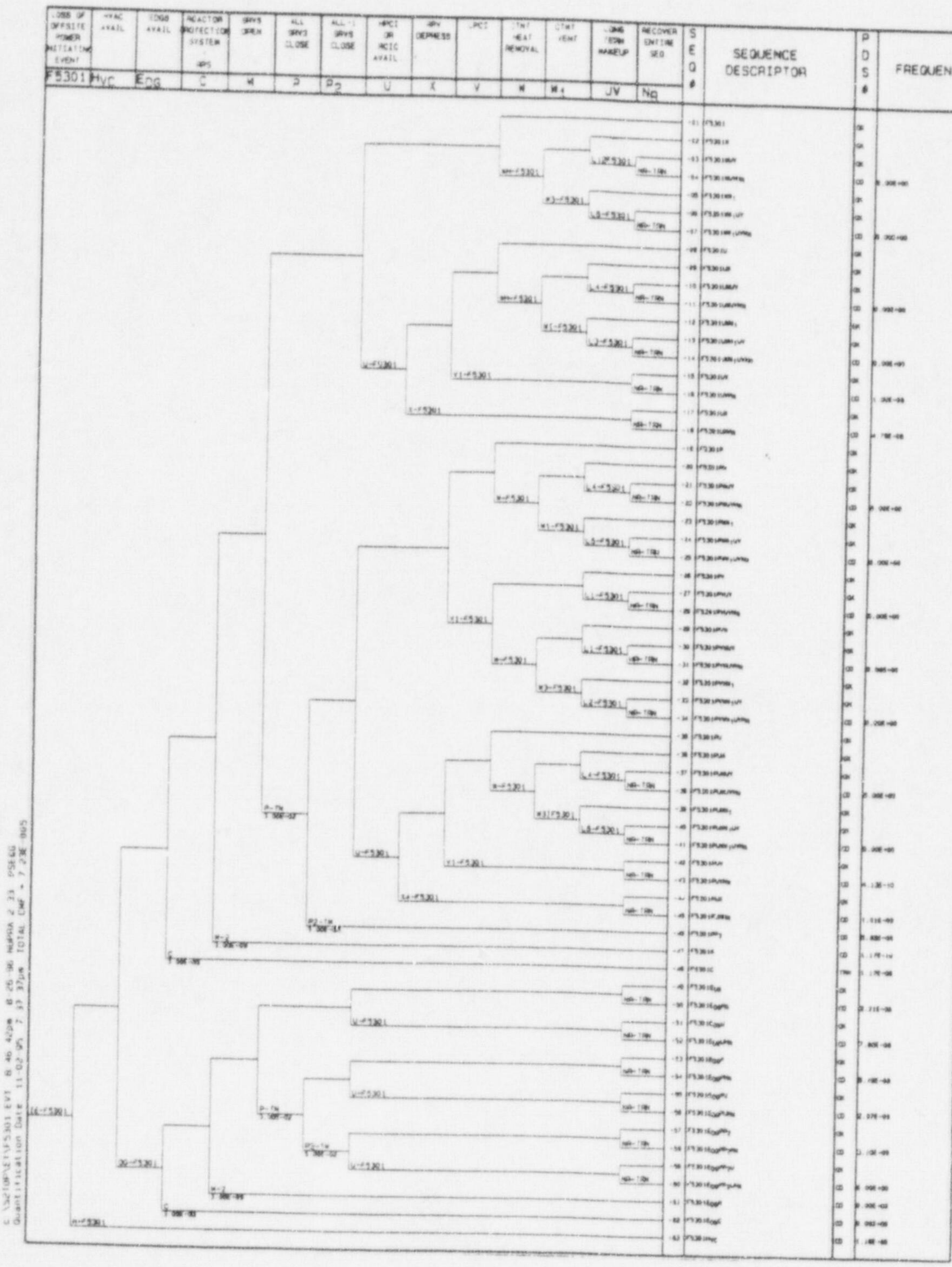

Figure 4.2
Loss of Offsite Power (LOOP) Event Tree for the HCGS Fire IPEEE

SECTION 5

HIGH WINDS, FLOODS AND OTHERS

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SECTION 5

HIGH WINDS, FLOODS, AND OTHERS

5.1 INTRODUCTION

Based on the work in NUREG/CR-5042 (NRC, 1989d) and other subsidiary studies, NUREG-1407 (NRC, 1991b) suggests specific external events for close examination in the IPEEE. These are internal fires, earthquakes, external floods, high winds and tornadoes, and transportation and nearby facility accidents. It also asks for "a certification that no other plant-unique external event is known that poses any significant threat of severe accident." Using the approach and results of NUREG/CR-2300 (NRC, 1983a) and NUREG/CR-5042 (NRC, 1989d), a screening assessment of potential external events was performed for the Hope Creek Generating Station (HCGS) site. The list provided in Table 10-1 of NUREG/CR-2300 (NRC, 1983a) was used as a starting point. The events provided in that list were categorized as follows:

- Transportation and Nearby Facility Accidents
- External Floods (e.g., wind, precipitation, tide, and wave effects)
- Reduction of Secondary Heat Sink (e.g., low river level, ice blockage, detritus)
- High Winds and Tornadoes (e.g., wind and missile effects)
- Internal Fires
- Severe Weather Storms
- Severe Temperature Transients
- Internal Flooding
- Avalanche, Landslide, and Volcanoes
- Lightning
- External Fires

- Release of On-site Chemicals
- Seismic Events
- Soil Failure
- Turbine Missiles
- Extraterrestrial Activity

Table 5.1 shows the results of the screening process that reduced this list to the following, which received more detailed plant specific assessment, documented in this section:

- High Winds and Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Release of On-site Chemicals
- Detritus

The screening assessment took advantage of the fact that the HCGS, is a plant that meets the 1975 Standard Review Plan (SRP) criteria (NRC, 1975a). HCGS is located next to the two unit Salem Generating Station (SGS) plant (Figure 5.1). The plant grade is at the 101.5 foot elevation and the mean sea level (MSL) is at 89.0 feet elevation (PSE&G, 1995f - Paragraph 2.4.2.2).

The assessments performed to develop Table 5.1 demonstrated that no other plant-unique external event is known that poses any significant threat of severe accident to the HCGS.

5.2 METHODOLOGY

The method of progressive screening, per NUREG-1407 (NRC, 1991b - Section 5), was used in this assessment. The plant specific hazard data and licensing bases were reviewed for all items in Table 5.1. This led to screening out most of the items. A thorough review of documentation (see reference list at the end of this section) was performed to determine significant changes (if any) with respect to military and industrial facilities within five miles of the site, on-site storage of hazardous materials, and transportation related aspects. Where possible, the documentation review was verified by either the plant walkdowns or tours of the nearby facilities. As described in Paragraph 5.3, the plant walkdown covered high winds as well as the applicable events from the above list.

A hazard by hazard determination was then made as to conformance with the 1975 SRP criteria. An affirmative determination for a hazard ended work for that hazard on the accepted basis that conformance to the SRP meets the IPEEE screening criterion. Except for explosions from river traffic and ship impact on the Service Water (SW) Intake Structure, transportation and nearby industrial and military facility accidents were screened out on this basis. Screening analyses to demonstrate a hazard frequency that is below the IPEEE screening criteria were performed to screen out river explosions and ship impact on the Service Water Intake Structure.

Tornado missile impact on safety structures and equipment, due to high winds, were also screened out on the basis of meeting the 1975 SRP criteria. However, walkdown in support of IPEEE revealed one exception, with respect to Room 5619, which is discussed further in Paragraph 5.3.3.

Detritus, which was postulated to have the potential of affecting all service water intakes, was evaluated by a screening analysis and review of past incidents at the HCGS. It was found that a large perturbation in the river, such as an earthquake, could initiate a detritus event which would disable all service water intakes. The frequency of an earthquake induced detritus event was found to be below the IPEEE screening criterion (PSE&G, 1995g).

Flood induced leakage into safety structures was screened out on the basis that the plant meets the 1975 SRP criteria.

5.3 WALKDOWNS

Several walkdowns were performed to confirm that no significant changes to the plant and the area near the plant have occurred since the issuance of the HCGS Updated Final Safety Analysis (UFSAR) Report (PSE&G, 1995f) with respect to high winds, floods, transportation and nearby facilities and on-site storage of hazardous materials. A summary of observations of these walkdowns are provided below.

5.3.1 TRANSPORTATION AND NEARBY INDUSTRIAL AND MILITARY FACILITIES

There are no railroads, pipelines, highways, industrial facilities and military facilities within five miles of the site. (This was also verified through phone conversations with the Department of Planning of the nearby counties.) This has not changed since the Updated FSAR. River traffic and air traffic are essentially unchanged since the Updated FSAR. However, commercial explosive shipments on the river momentarily increased significantly in 1996, but was stopped by the US Coast Guard. This is discussed further in Paragraph 5.6.2.

5.3.2 ON-SITE STORAGE OF HAZARDOUS MATERIALS

The walkdown concentrated on location (with respect to the control room) and quantity of hazardous materials. The walkdown participants observed that the plant has not experienced changes in onset storage quantities and locations that would invalidate the Regulatory Guide 1.78 (NRC, 1974a) analysis (PSE&G, 1992b). A new hazardous material storage facility was built in 1994 (PSE&G, 1994h). The hazardous chemicals stored at this facility were reviewed with respect to their impact on control room habitability during a postulated release (PSE&G, 1994h). The hazardous chemicals are stored in 55 gallon drums. Larger, single containers of some of these chemicals, that are listed in Table C-1 of Regulatory Guide 1.78 (NRC, 1974a), were previously evaluated for control room habitability and therefore, separate quantitative evaluations were not warranted. Other hazardous chemicals not listed in NRC, 1974a - Table C-1 were evaluated qualitatively in PSE&G, 1992b, and it was determined that a postulated release would not impact the control room habitability.

5.3.3 HIGH WINDS

The walkdown concentrated on building entrances; building openings such as air intakes, exhaust stacks, and louvers; objects that could become tornado missiles; outdoor tanks; and relative location of non-safety and safety structures or equipment. Table 5.2 is the checklist followed to inspect high wind and tornado related items. The walkdown was made with six types of missiles in mind, as is discussed in Paragraph 5.4.2. The six types of missiles span the range of missile size and velocities to be found at HCGS, except for large transportable cranes that may be in use during outages. Other key observations are:

- The door to the Technical Support Center (TSC) HVAC room (Room # 5619, Door No. 19), at 153 foot elevation (PSE&G, 1995f - Figure 1.2-7) is not strong enough to withstand the impact of a tornado. This door does not have a proper locking mechanism on it. The HVAC to the TSC is not essential for safe plant shutdown; however, this room contains seven safety "Channel C" cables related to the "A" control room air supply, which are contained within conduits that are tightly secured in place. The system design basis requires cables to be protected from missiles. Therefore work has been initiated to install a missile shield in front of door 19, at the entrance to Room No. 5619 (PSE&G, 1997d), with a scheduled project completion date of 1997.

- The HCGS secondary containment is equipped with two sets of exterior blowout panels that open toward the outside environment once 1.5 psid is sensed across them. The first set is located in the Steam Vent at about 145 foot elevation and open towards the West. The other set consists of many blowout panels that are staggered in the steam tunnel, and open towards the south and north at about 155 foot elevation (PSE&G, 1995f - Figure 1.2-10.) Opening of the Steam Vent's blowout panels will not affect the safety equipment within the building, since the pressure tight door at entrance to the Steam Vent, at the 132 foot elevation (PSE&G, 1995f - Figure 1.2-10), will be kept closed in advance of high wind conditions in accordance with procedure HC.OP-AB.ZZ-0139(Q), "Acts of Nature" (PSE&G, 1994g). The other set of external blowout panels are staggered in the Emergency Vent Stack and are designed to protect the steam tunnel against steam line rupture. Should these blowout panels open due to negative pressure caused by a tornado, the steam supply to the main turbine could become vulnerable. However, this will not affect the steam supply to the steam driven Reactor Isolation Cooling (RCIC) and High Pressure Cooling Injection (HPCI) systems.

- In general the non-safety structures do not pose any safety impact on the safety structures; PSE&G, 1995f has adequately discussed this issue. It was observed that in the event of a tornado the lightning mast on top of the Reactor Building could fall on safety related structures, such as the Control/Auxiliary Building containing the Emergency Diesel Generators (EDGs) or the rectangular portions of the Reactor Building. This mast weights 850 lb. and is 65' long (PSE&G, 1992c). This missile is bounded by the impact of the second type of missile attack, as discussed in Paragraph 5.4.2.

- The Condensate Storage Tank (CST) is not a safety related tank and is not designed for tornado or tornado generated missile loads (PSE&G, 1995f -

Table 3.2-1). A Seismic Category I concrete dike surrounds the CST to hold its content and to protect it against tornado winds and tornado generated missiles (PSE&G, 1995f - Paragraph 9.2.6.3).

- Transportable hydrogen tubes are placed far away from the safety structures, in accordance with (EPRI, 1987), so they will not affect the safety structures during tornado (PSE&G, 1994i). Similarly the Liquid Oxygen Tank (LOX) is located sufficiently far away from the safety structures so that their safety related functions will not be compromised by high wind induced motion of the tank (PSE&G, 1994l).

- All exterior ductwork is rated for at least -3.75 to +3.75 inch Water Gauge (W. G.) pressure, and some are rated higher. The tornado dampers are rated for 3 psid, but have not been tested to failure, so they may be able to withstand a higher differential pressure. Structures equipped with tornado dampers include Reactor Building [Filtration, Recirculation and Ventilation System (FRVS) exhaust], the EDG Intake and exhaust, and the Service Water Intake Structure (SWIS) Traveling Screen Room.

In conclusion, high winds do not have any significant impact on the safety structures or equipment at the HCGS.

5.3.4 FLOODS

Flood related observations are:

- The transportable hydrogen tubes and the LOX system could target the safety structures and have a negative impact on the plant. However, procedure HC.OP-AB.ZZ-0139(Q), "Acts of Nature" (PSE&G, 1994g) instructs operators to vent the LOX system and to move the transportable hydrogen tanks away from the site, with advance warning of a potential high water level.

- Exterior entrance doors to the safety structures, are all water tight and procedure HC.OP-AB.ZZ-0139(Q), "Acts of Nature," (PSE&G, 1994g) adequately guide operators in maintaining them closed in advance of flood conditions.

5.4 HIGH WINDS AND TORNADOES

5.4.1 WIND LOADING

Per NUREG 1407 (NRC, 1991b), the design criteria for wind are dominated by tornadoes having an annual frequency of exceedance of about 10^{-7}/yr. The design basis of the HCGS with respect to tornado loading is described in PSE&G, 1995f - Paragraph 3.3.2.1. The maximum wind speed is 360 mph with a radius of 150 feet. This is composed of a maximum rotational speed of 300 mph and a maximum translational speed of 60 mph.

The negative differential pressure design basis is three psid. Regulatory Guide 1.76 (NRC, 1974b) defines the design basis tornado for Region I (which includes the HCGS site) as follows:

- Maximum wind speed of 360 mph comprised of a maximum rotational component of 300 mph and a maximum translational component of 60 mph.

- Negative differential pressure of three psid.

Seismic safety structures exposed to the design basis tornado wind, or missiles associated with this wind, are designed so that they are not affected by these conditions. Furthermore, structures not designed for tornado loads are checked to ensure that during a tornado they do not generate missiles that have more severe effects than the tornado missiles considered in the HCGS UFSAR, (PSE&G, 1995f).

The HCGS plant meets the SRP criteria for tornado loading on its safety structures and no further work is warranted.

5.4.2 TORNADO MISSILES

The minimum thickness of the HCGS safety concrete structure barriers used to resist tornado missiles is 18 inches (PSE&G, 1995f - Paragraph 3.5.1.1). The barriers are evaluated against six types of missiles, with Region I velocity limits in mind (PSE&G, 1995f - Table 3.5-12):

1. Wood plank, 4 in. X 12 in. X 12 ft, weight 115 lb.
2. Steel rod, 1 in. diameter and 3 ft. long, weight 8.8 lb.
3. Steel pipe, 6 in. diameter, schedule 40, 15 ft long, weight 285 lb.
4. Steel pipe, 12 in. diameter, schedule 40, 15 ft long, weight 743 lb.
5. Utility pole, 13-1/2 in. diameter, 35 ft long, weight 1125 lb.
6. Automobile, frontal area 25 ft^2, weight 3990 lb.

This missile spectrum conforms to missile spectrum II of the SRP (NRC,1981a - Paragraph 3.5.1.4).

The rainhoods on the EDG exhaust pipes, located on the EDG roof at 198 foot elevation (PSE&G, 1995f - Figure 1.2-8), are not provided with tornado missile barriers (PSE&G, 1995f - 5.3, Table 1.11-1.) Based on PSE&G, 1995f - Paragraph 9.5.8.6, it is extremely unlikely that a missile will adversely affect the proper functioning of the EDG's. Also, the rainhoods on the vents for the EDG fuel oil storage tanks located at 155 foot elevation (PSE&G, 1995f - Figure 1.2-8) are not missile proofed. However, if a vent line should become damaged by a tornado missile, an alternate vent path will be established to allow venting of the diesel fuel oil storage tanks and the diesel fuel oil day tanks by opening a spare 4-inch flanged connection or the 30-inch manhole located on the diesel fuel oil storage tanks. This vent path will be maintained until the normal vent for the diesel fuel oil storage tanks and the diesel fuel oil day tank can be reestablished (PSE&G, 1995f - Section 3.5.1.1). The HCGS UFSAR requirement is to have a seven day supply of fuel oil and a 14 day supply of lube oil. Through procedure HC.OP-AB.ZZ-0139(Q), "Acts of Nature" (PSE&G, 1994g), this requirement will be met during severe weather conditions.

The HCGS unit contains no vulnerabilities with respect to tornado missiles.

5.5 EXTERNAL FLOODS

The most critical combination of flood producing phenomena results from the postulated occurrence of the probable maximum hurricane (PMH) surge with wave run-up coincident with the ten percent exceedance high tide. The maximum wave run-up to 35.4 feet Mean Sea Level (MSL) occurs along the southeast face of the Reactor Building and a small corner face of the Auxiliary Building. However, the Service Water Intake Structure (SWIS) may be subjected to waves which could overtop the roof of the western portion at Elevation 39 feet MSL. The MSL is at 89 feet elevation and the plant grade is at the 101.5 foot elevation (PSE&G, 1995f - Paragraph 2.4.2.2).

Safety-related systems and components are not affected by a flood when they are located above the postulated maximum flood level. When located below flood level, these systems and components are enclosed in reinforced concrete safety structures that have:

- Exterior walls thickness below flood level of not less than two feet.
- Waterstops provided in exterior wall construction joints and seismic separation joints below flood level.
- A minimum number of openings in exterior walls and slabs below flood level (these openings are designed to prevent intrusion of flood water.)
- Water tight pressure doors installed in exterior walls below flood level.
- Exposed equipment hatches installed above flood level; those below flood level installed behind exterior walls designed to prevent intrusion of water. The exception to this condition is the exterior hatch located at grade level in the north Radwaste Building. This hatch is designed to be water pressure tight.
- Continuous waterproofing systems are applied to the underside of base slabs and on exterior walls to grade.

Doors and hatches in exterior walls below flood elevation (including wave effects), are either provided with a sensor to alarm in the main control room or will be administratively controlled (PSE&G, 1995f - Paragraph 3.4.1). The river level is shown in the control room through a recorder chart; however, there is no annunciation or alarm associated with it. Operators can also readily determine and verify the river level through the computers, such as the CRIDS system, in the control room.

The external flood re-assessment, consisted of a walkdown for the purpose of discovering paths of significant water ingress into safety related structures owing to severe storm induced floods. No vulnerability was discovered.

Since HCGS is an SRP plant it meets the requirements of Regulatory Guide 1.59, (NRC, 1973).

5.5.1 NEW PMP CRITERIA

In 1989 the USNRC issued Generic Letter 89-22 (NRC, 1989a), concerning the potential for the increased roof loads and plant area flood runoff depth due to a change in National Oceanic and Atmospheric Administration (NOAA)/National Weather Service (NWS) hydrometerological reports for maximum precipitation (PMP) developed by the NWS for Hope Creek (PSE&G, 1991). The previous reports generally only forwarded PMP estimates for areas 10 square miles or greater and duration of 6 hours or more. New Hydro-Metrological Reports (HMRs) of NWS provided PMP estimates for drainage areas as small as one square mile and for duration as small as five minutes. PSE&G reviewed the impact of new roof load with respect to existing roof design live loading, in safety-related buildings and buildings that are important to plant processes (e.g., Turbine Building), in HCGS (PSE&G, 1991). The conclusion of this engineering evaluation was that: HCGS has already been designed utilizing the PMP criteria concepts contained within HMRs 51 and 52. Consequently, the requirements delineated in NRC Generic Letter 89-22 (NRC, 1989a) are met, and there were no new increased roof loads and plant area flood runoff depths to evaluate.

The HCGS Reactor Building is composed of a rectangular and a dome shaped structure. Accumulation of neither rain nor snow should be of any concern on the dome shaped section of this building. The roof of the rectangular section is located at 132 Foot Elevation and is surrounded by a berm that is about 5 feet tall (PSE&G, 1995f, - Figure 1.2-10.) This roof has a 15 foot wide opening on its west side and about 15 openings (each about eight inches in diameter) throughout the roof. These openings are located about six inches above the roof and allow for any accumulated water to easily run off to ground level at the 102 Foot Elevation and hence prevent any significant accumulation of water on the roof. The roof of the EDG building also has large runoffs at one end, therefore, accumulation of rain on its roof is not a concern. Similarly, rain accumulation on the SW structure is not concern due to runoff on the roof. On the other hand, snow could accumulate on the roof of the rectangular structures. Paragraph 4.8 of procedure HC.OP-AB.ZZ-0139(Q), "Acts of Nature", (PSE&G, 1994g), requires continuous monitoring of the snow accumulation on the EDG roof once more than 12 inches of snow has accumulated, and clearing the area surrounding the EDG exhaust pipe, once 36" of snow has accumulated.

According to PSE&G, 1995f - Paragraph 2.3.2.1.4, the maximum snowfall ever recorded in this region is 22.0 inches. However, the blizzard of 1996, which resulted in accumulation of about 30 inches of snow at the HCGS site set a new record. Based on PSE&G, 1995f - Paragraph 2.3.2.1.4, we can calculate that the safety structures can withstand about 289 inches of snow before their design limits are exceeded.

In summary, snow and water precipitation on the safety structures, is not of any concern.

5.6 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

All activities and facilities within five miles of the HCGS site are considered in the HCGS UFSAR. No significant manufacturing and chemical plants, oil refineries, storage facilities, military facilities, transportation routes other than the Delaware River, or gas and oil pipelines are located within five miles of the HCGS site, as stated in PSE&G, 1995f - Paragraph 2.2. The accuracy of this data was confirmed on September 29, 1994 through phone conversation with both the New Castle County and the Salem County Department of planning. The only major transportation route within five miles of the site is the Inter-coastal Waterway, 1.5 miles west of the island. NRC, 1978a provides a safe distance, from the impact of explosives point of view, of about 1700 feet for highways and 2100 feet for railroads. Since no major highway or railroad is located within a five mile radius of the plant the impact of any transportation type of accidents on HCGS would be negligible (PSE&G, 1995f - Paragraph 2.2).

Therefore, transportation and nearby industrial and military facility events have been screened out by concurrence with the SRP criteria.

5.6.1 AIRCRAFT

Aircraft hazards are documented in PSE&G, 1995f - Tables 2.2-1 and 2.2-2. PSE&G, 1995f - Paragraph 3.5.1.6.7, based on data prior to 1982, concludes that the probability of an aircraft (any type) strike with a potential for causing radiological consequences in excess of the exposure guidelines of 10CFR100 (OFRNARA, 1992) is 6.7E-8 per year; hence, contribution of air craft crash to the core damage and release frequency is negligible.

Data used in this analysis, was updated to investigate their impact on plant safety. Data in PSE&G, 1995f - Table 2.2-1 indicates that the total number of operations (landings or takeoffs of any type of plane or helicopter) at Greater Wilmington Airport between years 1976 and 1979 averaged about 180,000 per year. On the basis of the actual numbers in this time period (1976 to 1979) the total number of operations for the 1990's was previously estimated to be about 250,000; however, based on response to PSE&G, 1994c, the total number of operations has actually reduced to about 170,000 per year and no increase is expected. This is due to limited working hours of the Wilmington Tower (9:00 A.M. till 11:00 P.M.), which is expected to remain as is, due to economically driven reduction in air traffic activities at the Wilmington Airport. Of the 170,000 operations at the Wilmington Airport, it was estimated that 70% is due to itinerant (student and corporations) activities, 15% due to military activities and remainder is for miscellaneous activities. Recent flight activities at Wilmington Airport are summarized in Table 5.3, based on the response to PSE&G, 1994c.

In Table 5.3, the total number of operations is for both takeoffs and landings. The nearest location of Wilmington Airport's air traffic pattern to the site is within the five mile radius of the plant. Since the total number of operations has not increased, the hazards associated with the flights at Wilmington Airport do not significantly affect the safety of the HCGS site.

A more detailed air traffic pattern within the five mile radius of the plant, for 12 months ending in November of 1994, was provided by "Kalelkar, 1993" for all elevations, as is documented in Table 5.4. Numbers of operations listed in Table 5.4 are within the limits considered in PSE&G, 1995f - Table 3.5-8. Furthermore, no air accidents have occurred within the five mile radius of the plant, since the HCGS was constructed, according to either Walker, 1994 or the sources that responded to PSE&G, 1994d. However, recently a fatal crash occurred at the Wilmington airport (Table 5.3); outside the HCGS five mile radius.

Three smaller airports are within a ten mile radius of the plant. Of these, Summit airport, which is located ten miles to the west-northwest of the HCGS, has the most activity. However, the charter flight at this airport is no longer active and there is no recording of such operations. Through a phone conversation in 1994, the total number of operations at Summit airport was estimated, by an active traffic controller at Summit airport, to be about 70 per day, and this is consistent with previous figures shown in PSE&G, 1995f. There has been no significant accident at Summit airport in the past. The other two airports, Evergreen and Salem, have grass runways and are used mainly for agricultural spraying or recreation. These airports may also be used for emergency landings, although there is no record of such activity in the past.

Furthermore, through both PSE&G, 1994c and PSE&G, 1994d it was confirmed that the number of airways within the five mile radius of the plant has not changed at all, since 1982, as was assumed in PSE&G, 1995f.

On-site Helicopter flight activity has increased since November 1994; on average there are three flights per week to the site. This number is within the limits considered in PSE&G, 1995f. According to PSE&G, 1995f - Paragraph 2.2.2.5.6, the total number of takeoffs and landings allowed at the plant site helipad is 700 per year.

In summary, there is no significant change in flight activities near the HCGS and the plant vulnerability due to flight activities is within the acceptable plant design limits.

5.6.2 RIVER TRAFFIC HAZARDS

Since the HCGS plant is an SRP plant, the hazards associated with the river traffic is within the acceptable limits. However, the HCGS data base was updated to reflect recent activities on the river. Based on verbal response to PSE&G, 1994e, the following is a breakdown of the average yearly river traffic past the HCGS site since 1985.

- Eighteen to a maximum of 44 Liquid Petroleum Gas (LPG) carrying vessels; average of 34 per year.

- Six to a maximum of 18 ore carrying vessels; average of 14 per year.

- Thirty-two to a maximum of 80 bulk oil carrying vessels; average of 52 per year.

Based on the same reference an average of 107 chemical tank river shipments have passed the HCGS site each year. Furthermore, based on limited data from September 1990 through September 1994, the same reference indicates an average of two solid explosive shipments pass the plant each year. However, for a few months in 1996 this frequency increased to one shipment of 750 Ton solid explosives per month. These new explosive loads were being unloaded at Port of Salem. However, since Port of Salem was not authorized for explosive handling, the US Coast Guard has permanently prohibited this facility from storage and handling of explosives. According to the US Coast Guard, currently no facility along the Delaware River has license for storage and handling of explosives (PSE&G, 1997e). When explosive carrying ships travel on the Delaware River, the US Coast Guard maintains a safety distance of 1000 yards, back and forth, and about 500 yards laterally (PSE&G, 1997e).

Data released by the responder of PSE&G, 1994f indicated that the average number of vessels traveling through the Delaware River, past the HCGS between 1969 and 1981, was 4638 per year. However, this number reduced to 2850 per year for years between 1982 and 1993. Responder of PSE&G, 1994e indicated that the distance from the southern part of the HCGS nuclear site to the shipping channel buoy is about 6000 feet.

PSE&G, 1995f, based on 1982 data, indicates that there is an anchorage zone, northwest of the site, designed to be used only for vessels carrying explosives and that the US coast guard has petitioned to relocate it to approximately 8 miles south of the HCGS site. Responder of PSE&G, 1994e confirmed that the anchorage zone still remains at the same location; well within the five mile radius of the HCGS site, and that explosives can be unloaded at any anchorage area along the river.

If we were to assume that the maximum explosive carrying vessel load is 800 tons (as is indicated in Kalelkar, 1993), using NRC, 1978a we have:

$R = K \times W^{1/3}$, Where,

R = Maximum distance affected by explosion of TNT equivalent,

K = A constant value of 18, and
W = Mass of explosives in KG.

$R = 18 \times (800{,}000\ KG)^{1/3} = 18 \times 92.83 = 1671 M = 5482$ Ft.

This means that the closest safety structure at HCGS, namely the Service Water structure, is outside the damage zone of 100% TNT equivalent. However, if explosives are not solid, (i.e., Liquid petroleum gas), a reasonable upper bound to the blast energy would be 240% TNT equivalent mass. In that case R can be calculated as follows:

$R= 18 \times (800{,}000 \times 2.4)^{1/3} = 18 \times 124.3 = 2237.2\ M = 7340$ Ft, which means that the SW intake structure and possibly the Containment Building could be affected. However, the damaging capability of an explosion reduces as the distance from the explosion source increases. The safety structures at HCGS are seismic Category I design with an external pressure rating of 3.0 psig, and can withstand the impact of an earthquake with a magnitude of 0.2g. In other words, the impact of an explosion in the shipping channel on safety structures, which are located about 5700 feet away, would be negligible.

The probability of a ship impacting the SW intake structure is 10^{-7} per year (PSE&G, 1995f - Paragraph 2.2.3.1.5), and hence, there is no concern about explosive carrying vessels colliding with the SW intake. Additionally, with the safety zone maintained around an explosive carrying vessels (1000 yards, back and forth, and about 500 yards, laterally) chance of any other vessel impacting the HCGS due to impact of an explosion on an explosive carrying vessel would be negligible.

In summary, river traffic has reduced substantially, since the plant was built. There is an indication that an average of 50 explosive carrying vessels (34 LPG and two solid explosive - with momentarily increase to 12) have passed the HCGS site each year. However, given the fact that the shipping buoy is located over a mile away from the site and the explosion probability is low, the impact of explosive carrying vessels on the plant safety structures is not significant. Additionally, given that currently no facility along the Delaware River is licensed to carry explosives, supports our conclusion that river traffic hazards do not reveal a vulnerability at the HCGS site.

5.7 ON-SITE CHEMICAL STORAGE

Numerous hazardous chemicals are stored at, delivered to, and used at the HCGS site. PSE&G maintains an inventory of these chemicals, as part of the New Jersey Community Right-To Know Act (NJAC, 1984); however, none of the chemicals stored on-site are regulated by the New Jersey Toxic Prevention Act or 29CFR Part 1910.119. PSE&G has identified hazardous chemicals that may affect Control Room Habitability (CRH), based on Regulatory Guide 1.78 (NRC, 1974a), and has assessed their impacts. The results of PSE&G's assessment, documented in PSE&G 1992b, indicates the following:

The only water treatment chemicals that required quantitative analyses were sulfuric acid, ammonium hydroxide and hydrazine. The HCGS control room meets Regulatory Guide 1.78 (NRC, 1974a) criteria during a postulated release of sulfuric acid, hydrazine and ammonium hydroxide. Sulfuric acid is stored at the HCGS side of the site, while both ammonium hydroxide and hydrazine are stored at the SGS part of the site. No other water treatment chemicals possess the toxicity or physical properties that would enable postulated releases to impact CRH.

Evaluations of the bulk gases stored on-site indicated that CRH would not be impacted during postulated releases. This is due to the relatively small storage containers, locations, high threshold values, and their ability to disperse rapidly in air.

Evaluations of the remaining hazardous chemicals stored on-site conclude that no other chemicals would impact CRH upon a postulated release. This is due to the high threshold values and the relatively small storage containers.

In 1994 a new building for hazardous materials was built (PSE&G, 1994h). This new building will not affect safety of the plant, since it is not the largest on-site source of different chemicals.

In summary, accidents involving release of on-site chemical storage do not pose a vulnerability at HCGS owing to conformance to Regulatory Guide 1.78 (NRC, 1974a).

5.8 DETRITUS

The HCGS SW pumps have experienced problems due to mud and grass buildup in their traveling screens. The extent of these problems has not been significant in the past and has not affected the plant safety significantly. However, recently detritus has significantly affected the SW system availability, such that some improvements to this system became essential.

1) Design was improved to ensure that the SW booster pumps were indeed running prior to the traveling screens operation in low speed (PSE&G, 1996K). Also, the strainer backwash valves are now open all the time, rather than cycling on high strainer differential pressure. This constant backwash reduces buildup of grass and silt in the strainers. Additionally, adjustment was made to the spray pattern on the traveling screens to provide better coverage, especially at the ends, where buildup was the most significant. These modifications substantially reduced mud and grass buildup in the SW traveling screens.

2) The flow switches EPFS-2225A, B, C and D were removed, and replaced with blind flanges, to eliminate the problems with the corrosion of the switches. These flow switches were no longer needed, since their function was replaced with a booster pump run contact to start the traveling screens (PSE&G, 1996l).

3) The SW strainer drain lines were changed from cement lined carbon steel to 6% Moly austenitic stainless steel, to provide a more corrosion resistant design (PSE&G, 1996m).

4) The 28" Aluminum-Bronze body discharge isolation valve for each SW pump was replaced with 6% Moly Stainless body material, to provide a more corrosion resistant design (PSE&G, 1996n).

5) Four new one-inch lines and valves are installed to serve as the backwash line high point vents for maintenance of the strainers and associated components. This new vent line eliminates concern about air ingestion to an operating SW strainer, specially during the Loss of Offsite Power (LOOP) scenario, which could have disabled the SW heat removal capability (PSE&G, 1996o).

6) Procedures were improved to ensure the actuator for the non-safety related SW D-icing valves will not be covered in the water, and that the De-icing valves are maintained operable (PSE&G, 1995o).

Of the four non-safety related SW De-icing valves [DA-HV-2097, is located in one valve pit, and DA-HV-2096, DA-HV-2098 and DA-HV-2099 , are all located in another valve pit. The station procedure, "Preparation for Winter Operations - (PSE&G, 1995o), shows that the preferred SWS de-icing path is circulating water blow down, rather than the SW de-icing path. Failure of the valves in the yard pit could cause a potential common mode failure of all SW pumps due to loss of SW pump suction from service water icing. However, the preparations for winter conditions, PSE&G, 1995o, eliminates this concern.

7) Improvements to the SW System are ongoing. In 1996, "D" SW pump was totally replaced, and early in 1997 the "B" SW pump was replaced. Also, work on strainers, which are the weakest point in the SW system, is ongoing. For example, consideration is being given to changing the strainer mesh size from 1/8 inch to 1/4 inch for better system reliability.

The impact of seismically induced detritus on plant safety is evaluated in PSE&G, 1995n. A large scale river bottom perturbation is required to dislodge sufficient detritus to impact all SW intakes. This reference calculates the frequency of core damage due to a seismic induced detritus event to be in the range of 5.2E-7 to 9.2E-7 per year. This frequency is lower than seismic induced loss of service water which is 1.33E-6 per year.

In summary, detritus induced loss of all service water pumps has been shown to have a frequency less than the IPEEE screening criterion. This, as well as recent improvements to the SW system and establishment of a team to closely monitor the performance of the SW system at HCGS, makes us conclude that contribution of detritus to loss of SW system is negligible.

5.9 SUMMARY

Beginning with the list of external events found in NUREG/CR-2300 (NRC, 1983a), the class of external events termed "other external events" have been screened out either by compliance with the 1975 SRP criteria or by engineering analysis and evaluations. The study provides certification that no plant-unique external event is known that poses a significant threat of severe accidents. The study also provides confidence that the HCGS unit is not vulnerable to other external events.

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Table 5.1 Screening of External Events for HCGS

Event	Generic Basis	Applicability to HCGS
Transportation and Nearby Facility Accidents (i.e., barge, aircraft, railroad, gas pipelines, highways).	GL-88-20 (NRC, 1991a), Supplement 4 requests a plant specific examination for older plants.	The HCGS UFSAR (PSE&G, 1995f) reviewed the location and impact of these events on the site. This section includes a review of changes in areas near the plant to confirm that the UFSAR conclusions are still valid.
External floods (e.g., wind, precipitation, tide, and wave effects).	GL-88-20 (NRC, 1991a), Supplement 4 requests a review of river flooding for older plants. It also requests a review with respect to the latest Probable Maximum Precipitation (PMP) criteria with respect to roof ponding loads.	The plant design meets the 1975 SRP criteria for external floods. This section reviews response of the plant to external floods.
Reduction of secondary heat sink (e.g., low river level, ice blockage, detritus).	Low river level, ice blockage and drought are either part of the plant design basis or technical specification basis for power operation.	Drought and ice blockage induced reduction in heat sink is not applicable to the HCGS site. The HCGS UFSAR (PSE&G, 1995f) analysis shows that low river level has been adequately considered in the design basis of the Service Water System. It is governed by low tide coupled with the worst low water level location during a hurricane. Detritus is discussed in Paragraph 5.8.

Table 5.1 Screening of External Events for HCGS (Continued)

Event	Generic Basis	Applicability to HCGS
High Winds and Tornadoes (e.g., wind, pressure differential and missile effects).	GL-88-20 (NRC, 1991a), Supplement 4 requests a plant specific examination for older plants.	The HCGS UFSAR discussed the design basis. Tornadoes were found to govern with respect to wind speed, pressure differential, and missile generation. The HCGS tornado structural criteria is equivalent to the 1975 SRP criteria for Region I. This section includes a review of HCGS's current status with respect to tornado effects and confirms that the UFSAR conclusions are still valid.
Internal fire.	GL-88-20 (NRC, 1991a), Supplement 4 requests a plant specific examination.	Internal fires are discussed in Section 4 of this submittal.
Severe Weather Storms (e.g., ice and hail storms).	Severe weather storms (other than extreme wind and flood conditions) do not threaten the plant design basis. The potential effects are encompassed by loss of offsite power and station blackout.	Loss of offsite power and station blackout were considered in the HCGS IPE. HCGS does not experience weather storms more severe than considered in the generic basis. Detritus is discussed in Paragraph 5.8.
Severe Temperature Transients.	The potential effects of severe temperature transients may be reduction in ultimate heat sink, loss of offsite power and station blackout.	Loss of offsite power and station blackout were considered in the HCGS IPE. HCGS does not experience temperature transients more severe than considered in the generic basis. Low river level events, unrelated to temperature transients, govern reduction in ultimate heat sink.
Internal Flooding.	Requested in GL 88-20 (NRC, 1991a).	The HCGS IPE included internal flooding.
Avalanche, Landslide, and Volcanoes.	Considered only for sites close to these potential hazards.	The HCGS site is on an artificial island in the Delaware River far from these hazards.

Table 5.1 Screening of External Events for HCGS (Continued)

Event	Generic Basis	Applicability to HCGS
Lightning.	GL-88-20 (NRC, 1991a), Supplement 4 states that the predominate effect of lightning is loss of offsite power which is covered in the IPE studies.	This assessment is applicable to the HCGS.
External Fire.	Potential effects could be loss of offsite power, isolation of plant ventilation and possible control room evacuation. Typically, offsite fires have little or no effect on site because of site clearing during construction. Control room evacuation and plant ventilation are part of the design basis. The IPEs consider loss of offsite power.	Potential sources of external fires at HCGS include marsh grass, ship fires on the Delaware river, and the fuel oil storage tanks. The generic basis for screening out their effects applies to HCGS. (Hydrogen storage tanks and other flammable materials inside the buildings were considered during the internal fire examination.)
Release of On-site Chemicals.	Potential effects are loss of control room habitability.	Discussed in Paragraph 5.7.
Seismic Events.	NUREG-1407 (NRC, 1991b) requests a plant specific examination.	Seismic events are discussed in Section 3 of this submittal.
Soil Failure.	Soil subsidence is part of the plant design basis.	The HCGS UFSAR confirms that subsidence is not an issue. The seismic examination (HCGS IPEEE, Section 3) treats potential earthquake induced soil failures and liquefaction.

Table 5.1 Screening of External Events for HCGS (Continued)

Event	Generic Basis	Applicability to HCGS
Turbine Missiles.	Based on regular inspection of low pressure turbine discs and over-speed protection system, the probability of turbine failure leading to missiles is considered acceptably small so that they need not be considered in the IPEEE.	HCGS is adequately protected from missiles that may be generated by the SGS turbines (PSE&G, 1995f). The HCGS UFSAR (PSE&G, 1995f - Paragraph 3.5.4.2), describes the assessment of the probability of turbine generated missile damage, and finds no significant vulnerability.
Extra-terrestrial Activity.	Meteorite strikes, satellite strikes, and solar disturbances are not plant specific issues and the frequency of occurrence at a site is considered acceptably small.	This is applicable to HCGS.

Table 5.2 Wind Walkdown Checklist

Item	Generic Comment	HCGS Specific Disposition
Locate all safety related equipment and structures.	Observations should focus on protection from wind.	Drawings were used to identify equipment and structures.
Verify thickness of concrete walls protecting equipment.	Inadequate thickness (e.g., less than 12") should be noted.	HCGS category I concrete structure barriers used to resist tornado missiles are at least 18 inches thick which precludes penetration and spalling (PSE&G, 1995f).
Check for metal sided structures, and evaluate the potential of structures that could impact buildings that contain safety related equipment.	The proximity to safety related structures should be noted. Can non-safety buildings fall on safety buildings in a way that damages equipment.	HCGS is an SRP plant and walkdown did not find any vulnerability.

Table 5.2 Wind Walkdown Checklist(Continued)

Item	Generic Comment	HCGS Specific Disposition
Inspect entrances to concrete buildings.	Missile doors or concrete barriers should be noted. Potential paths for tornado missiles entering openings and impacting equipment should be investigated.	Study identified that the safety conduits for the "A" Control Room Emergency Filtration (CREF), contained in Room 5619, were not adequately protected against tornado missiles. As result, a missile shield is being installed in front of door 19, at entrance to Room 5619 (PSE&G, 1997d). All other safety structures at HCGS are well protected against tornado missiles; HCGS is an SRP plant.
Inspect other types of openings to buildings such as air intakes and exhaust, and louvers.	Potential paths for tornado missiles entering openings and impacting equipment should be investigated.	Diesel Generator exhaust rainhoods are not tornado missile proofed. However, study indicates that there is no need for such protection (PSE&G, 1995f).
Note block walls in structures with openings that could fall on safety-related equipment.	The concern is pressure differential across internal block walls.	This does not apply to HCGS.
Investigate the potential for missiles to impact safety related equipment in non-safety structures.	For example, can vital equipment be damaged by missiles that penetrate the Turbine Building.	This does not apply to HCGS.

Table 5.2 Wind Walkdown Checklist, (Continued)

Item	Generic Comment	HCGS Specific Disposition
Inspect outdoor safety related water storage tanks.	Note thickness of concrete barrier, characteristics (e.g., size, number, dimensions, anchorage) of anchor bolts.	No safety related tank is located outdoors. The CST, a non-safety related tank, is contained within a concrete diked area to protect it from possible missile attacks. No other safety related tanks are outside.
Inventory potential tornado missiles.	Number of missiles may be recorded in accordance with flight characteristics.	A detailed survey of objects was judged to be unnecessary because the HCGS UFSAR (PSE&G, 1995f - Section 3.5) demonstrated acceptable plant robustness against tornado missiles, and because HCGS is an SRP plant.

Table 5.3: Recent Flight Activities At Wilmington Airport (Outside the Five Mile Radius of the HCGS.)

Year	Total Operations	Accidents/Incidents
1991	17225	None
1992	17200	One accident on June 17, resulted in four fatalities.
1993	17797	Two incidents.
1994 (As of Sept. 20)	15824	One incident and one accident, which did not result in any fatality.

Table 5.4: Detailed One Year (1994) Flight Activity Within 5 Mile Radius of the HCGS.

SFW-GA (Single Engine)	1200
SFW-GA (Multi Engine)	3850
Air Taxi	6225
Air Carriers	89,117
Military	2492
Helicopters	Exact Numbers Not Available (However, helicopter activities at the site have increased since 1994.)
Accidents	0

FIGURE 5.1
SGS/HCGS SITE BUILDING LAYOUT

SECTION 6
LICENSEE PARTICIPATION AND INDEPENDENT REVIEW TEAM

SECTION 6

LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

This section describes the Public Service Electric and Gas Company and vendor involvement in the performance and peer review of the HCGS IPEEE.

6.1 IPEEE PROJECT ORGANIZATION

Management of the overall HCGS IPEEE Project was provided by Mr. M. A. Phillips, Supervisor - Probabilistic Safety Assessment. He is responsible for all PSA related work performed for the Salem and Hope Creek Generating Stations. Dr. W. T. Weir, Principal Staff Engineer, was responsible for the technical management aspects of both the Salem and Hope Creek IPEEE Projects. PSE&G technical staff contributed to all aspects of performance and peer review of the HCGS IPEEE. A project review team, consisting of in-house operations, seismic and fire protection engineers helped the PSA group validate the technical accuracy of the work generated by PSE&G personnel and vendors.

EQE International provided walkdown and seismic analysis consulting services managed by Dr. M. K. Ravindra. Woodward-Clyde Consultants provided geotechnical consulting services, primarily in the areas of soil liquefaction potential, slope stability, and seismically induced settlement. These services were directed by Dr. Y. Moriwaki and performed primarily by Dr. T. G Thomann both of Woodward-Clyde Consultants. Seismic systems analyses, scenario development and overall integration was performed by Mr. J. D. Leary of PSE&G's PSA staff. The relay chatter assessment was performed by Mr. J. J. Materazo of PSE&G's Instrumentation and Control Group.

Consulting services related to the Fire IPEEE were provided by Safety Factor Associates under the direction of Dr. M. V. Frank. NUS Corporation, now a division of SCIENTECH, provided support in calculating conditional core damage probabilities for the screening analyses. Much of the fire compartment interaction analysis and the system analysis during the PSA of unscreened compartments were performed by Mr. S. Seyedhosseini of PSE&G's PSA staff. The fire walkdown and Control Room analysis benefited from the assistance of Dr. M. Kazarians.

Mr. S. Seyedhosseini was also responsible for the analysis of High Winds, Floods and Other environmental effects on the HCGS (i. e. , Section 5).

Two tiers of review were performed. First by an IPEEE Project review team and then by an Independent Peer Review team. The project review team was composed of the following PSE&G and contractor personnel:

V. Amaraksha	PSE&G, I&C and Relay
C. Atkinson	PSE&G, I&C and Relay
A. Caplinger	PSE&G, Nuclear Safety and Fire Protection
F. Dombek	SCIENTECH
J. Gebely	PSE&G, Nuclear Safety and Fire Protection
A. Johnson	PSE&G, Civil Engineering
J. Leary	PSE&G, PSA Staff
J. Materazo	PSE&G, I&C AND Relay
I. Nag	PSE&G, Fire Protection/Safe Shutdown Engineering
M. Phillips	PSE&G, PSA Supervisor
C. Pupek	PSE&G, PSA Staff
K. Sarka	PSE&G, Electrical Engineering
G. Schroeder	PSE&G, Fire Protection Engineering
S. Seyedhosseini	PSE&G, PSA Staff
Y. Shyu	PSE&G, Seismic & Soils Analysis
J. Thompson	PSE&G, Hope Creek Technical Staff
T. Weir	PSE&G, IPEEE Project Manager

6.2 COMPOSITION OF THE INDEPENDENT REVIEW TEAM

The Independent Peer Review (IRT) team consisted of:

- Dr. Michael V. Frank (SFA), who reviewed the seismic system analysis the overall integration of the seismic study and the High Winds, Floods and Others study. Dr. Frank also prepared Section 6, with input from the other IRT team members.

- Mr. John M. Hilditch (PSE&G), who reviewed the fire study.

- Dr. Gary G. Luh (PSE&G), who reviewed the seismic capability and geotechnical aspects of the seismic study.

A synopsis of their background and qualifications for this assignment is provided:

Dr. Michael V. Frank

Dr. Frank is founder and President of Safety Factor Associates, Inc. Over the last 27 years he has become nationally recognized for his application of safety, reliability, and risk assessment technologies to real world systems. His particular expertise is the assessment and management of all risks associated with engineered systems and the decision-making that accompanies risk management. He has authored over 60 publications and has made numerous presentations to national and international forums. Dr. Frank is a Professional Engineer with a strong background in Mechanical Engineering (heat transfer, fluid flow, and thermal-hydraulics) as well as reliability and risk analysis. Of particular relevance to his role with the HCGS IPEEE Independent Review Team is his review of approximately 20 other PSAs (both internal and external events), his technical contributions to approximately 15 PSAs (both internal and external events), and his publication of VULCAN, a fully probabilistic fire propagation analysis code.

Mr. John M. Hilditch

Mr. Hilditch is a registered professional electrical engineer and has over 14 years of experience in the operation, maintenance, design and evaluation of both commercial and military nuclear power plant process systems. He presently serves as the Hope Creek System Engineering BOP Section Supervisor. He has a unique perspective of the integrated operation of the Hope Creek Generating Station. He is also a member of the Hope Creek Maintenance Rule Expert Panel. Mr. Hilditch was previously employed as an Engineering Consultant with Ogden Environmental and Energy Services Company. He has worked in many design related projects involving the application of Appendix R rules to electrical and mechanical systems in nuclear power plants, including evaluating systems for "hot shorts", high impedance faults, safe shutdown hazards analysis, and fire protection schemes. Mr. Hilditch has also led many multi-discipline safety system functional inspection teams, including an assessment of the Fire Detection and Suppression Systems at the Fermi 2 plant.

Dr. Gary G. Luh

Dr. Luh is Principal Staff Engineer in the Nuclear Business Unit of PSE&G. He held various positions in electric utility companies and engineering/construction firms during his 25 years of experience. He has a total of 17 years with the nuclear power industry. Dr. Luh's areas of specialization are seismic qualification, structural dynamics, geotechnical engineering, and soil-structure interactions. He is the PSE&G project engineer for Salem's USI A-46 program. He is the program sponsor for the equipment seismic qualification and Seismic Category II/I programs at the Salem and Hope Creek Generating Stations. Dr. Luh is a certified Seismic Capability Engineer trained in the EPRI/SQUG methodology.

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

The review examined the methodology, assumptions, relevant data, and the results of the three studies: seismic, fire, and High Winds, Floods, and Others. This paragraph summarizes the major observations and comments of the Independent Review Team.

6.3.1 SEISMIC AND SOIL

A comprehensive review of the seismic, soil, and soil/structure interactions studies was conducted to ascertain that the methods used are adequate and that the results generated are reasonable. Screening Evaluation Work Sheets (SEWS) of selected components were reviewed, and plant walkdowns were performed to verify the information recorded on the SEWS. The IPEEE seismic sections of the tier 2 reports were thoroughly reviewed. Comments were generated and resolved, and changes have been incorporated into the affected calculations and documents. Use of the Livermore seismic hazard information (NRC, 1994b) and walkdown results provided acceptable results. Major review efforts were expended in the following areas:

6.3.1.1 Dynamic Soil Properties

Values for dynamic soil properties are primarily derived from existing soil data extracted from the Salem UFSAR and various Dames & Moore Reports. Due to the close proximity of the Salem and Hope Creek sites, some soil data prepared for the Hope Creek site is also used. Dynamic soil properties used in the report are reasonable and representative of the site soil condition.

6.3.1.2 Soil Liquefaction and Slope Stability

The Hope Creek power block foundation is resting on the Vincentown formation. This is a very old formation and has very high shear wave velocity. Therefore, the computed probabilities of soil liquefaction and seismically induced settlements and differential settlements are very small, as anticipated.

The site is generally level with no significant natural or constructed slopes beyond the shoreline. Site conditions indicate that flow failures, typically associated with steep slopes, do not appear to be a concern.

6.3.1.3 Relay Chatter Evaluation

The relay chatter evaluation involves 1) development of a list for bad actors or Low Ruggedness Relays (LRR) as identified by the NRC and SQUG/EPRI, 2) evaluation of the impacts of potential chatter of the LRRs. The LRR list was developed using five search methods. Sixteen panels which contain LRRs and 38 LRRs were identified. Analysis of these panels and LRRs led to the conclusion that either median capacities are greater than 1.5g for the applicable failure modes or that there is no impact on safe shutdown.

6.3.1.4 Seismic Walkdown

Requirements, results and documentation for seismic walkdowns were reviewed in accordance with EPRI NP-6041 (EPRI, 1991b). An independent sample seismic walkdown was performed on selected equipment to verify the documented results. The sample walkdown results agree with the report.

6.3.1.5 Probabilistic Seismic Response Analyses

Methodology and procedure used in calculating the probabilistic seismic response were reviewed. The methods are reasonable and the results appear to be reasonable and consistent with expectations.

6.3.1.6 Seismic Fragility of Structures and Equipment

A list of structures and equipment is developed for seismic fragility evaluation. Methods for screening and walkdown were reviewed following the criteria established in EPRI NP-6041-M. In the fragility report, major structures (Reactor Building., Auxiliary Building. Diesel/Control Building, Service Water Intake Structure and Turbine Building) are evaluated and screened out if their median PGA capacities are greater than 1.5g. The HCLPF for liquefaction at the Vincentown formation was estimated to be greater than 0.6g PGA. Equipment which had a seismic capacity larger than 1.5g, or HCLPF larger than 0.5g, were screened out. The screened-in components are included in the system analysis.

6.3.1.7 System Analysis and Core Damage Frequency

The seismic system analysis was reviewed for technical accuracy and consistency. Input quantities were checked whenever possible. The overall procedure includes use of both the LLNL and EPRI hazard curves in conjunction with a seismic event tree. The Seismic Code combines the frequency, at each acceleration level, with the mean fragility curves at that level for the components listed in the seismic event tree. Seismic damage states are equated with seismic event tree sequences. Those seismic damage states, which do not end in core damage by themselves, are evaluated further by determining the likelihood that additional internal event failures would lead to core damage. The HCGS PSA model, implemented on the NUS Workstation, is used for the internal event conditional core damage frequency evaluation. This is a reasonable and typical method.

The most significant comments involved treatment of the recovery action RSDOWN, which represents the ability to control the plant from the remote shutdown panel. The calculated HEP assumed that all equipment would be available and that there would be no delay time associated with diagnosing the need for abandonment. Furthermore, this same event appeared in the modeling of two event tree top events without the logical dependencies correctly considered. In other words, RSDOWN could fail and succeed in the same sequence. These concerns were satisfactorily resolved by demonstrating that a) equipment failures are, in fact, a minor contributor to the estimated HEP of 0.06, b) control room abandonment would only be needed as a result of loss of control room ventilation which is easily diagnosed, and c) the effect of neglecting the logical dependency is conservative.

6.3.2 FIRE

The methodology was reviewed with respect to consistency with currently accepted industry practices, especially NUREG-1407 (NRC, 1991b) and GL 88-20 Supplement 4 (NRC, 1991a). The screening criteria appears to be reasonable. The assumptions used appear to be sound and reasonable. The results were spot checked by walking down a strategic sampling of fire compartments. Based on a qualitative knowledge of the plant, it appears that the 38 rooms identified by the screening process pose a higher risk of fire damage than screened compartments.

The High Hazard Area Analysis results appear to be thorough and complete. The conclusion agrees with an independent qualitative assessment of the areas.

The independent review questioned three specific areas:

- The study assumed that valve bodies are not affected by fire. The assumption was satisfactorily justified on the basis that valves were considered disabled if the power or control circuitry at the valve, in cables or in the motor control centers were damaged. In addition, there is a low likelihood of a fire being located at or under a valve such that the plume effects could damage the valve body. Finally, valve bodies have a considerable heat sink because of water within them and connected piping.

- The determination that there is minimum of four operators in the control room at all times was questioned. It has been satisfactorily clarified that the number of operators on duty who are able to respond to a fire, along with the station fire department, is four.

- Finally, a statement that one division of SACS provides cooling to "both RHR heat exchangers A&B using a cross tie" was found inaccurate. The analysis and report was corrected to reflect the actual configuration of the SACS and RHR systems.

6.3.3 HIGH WINDS, FLOODS, AND OTHERS

This work was reviewed for conformance to NUREG-1407, accuracy of technical presentation, and inconsistencies. The screening methodology suggested in NUREG-1407 is correctly applied for the HCGS, which conforms to the 1975 SRP criteria. An overall screening of hazards was performed to certify that all potentially significant hazards are considered. This analysis uncovered detritus buildup as a potential contributor. This is satisfactorily discussed in the submittal. Generally, the technical arguments are consistent, accurate, and well documented.

6.4 RESOLUTION OF COMMENTS

6.4.1 SEISMIC

The seismic studies were performed in accordance with appropriate processes and methodologies. Results appear to be reasonable. The relevant sections of the Tier 1 and Tier 2 reports were thoroughly reviewed. Comments were generated and resolved, and changes have been incorporated into the affected calculations and documents. Use of the Livermore Seismic hazard information and walkdown results provided acceptable results.

With respect to seismic systems analysis, all comments were adequately addressed. The modeling approximations used in the HCGS seismic system analysis produce results that are conservative relative to the more precise technique that were suggested by the review comments.

The integration of the seismic assessment was reviewed for conformance to NUREG-1407, technical inconsistencies and clarity of presentation. A reasonably typical seismic PSA was performed including hazard frequencies, fragilities, and systems analysis of the critical components. The seismic submittal addresses all aspects suggested in NUREG-1407 and GL 88-20, Supplement 4.

6.4.2 FIRE

Fire Analysis methods, assumptions, and results are satisfactory. The methodology in identifying fire compartments, and the subsequent screening process are reasonable. Spot checks of the fire walkdown and analysis reports yielded no deficiencies. Specific comments have been satisfactorily resolved. The fire submittal addresses all aspects suggested in NUREG-1407 and GL 88-20, Supplement 4.

6.4.3 HIGH WINDS, FLOODS, AND OTHERS

High winds, floods, and other environments were evaluated by walkdown and analyses in coincidence with the procedure appropriate for SRP plants. The High Wind, Floods and Other External Environments submittal addresses all aspects suggested in NUREG-1407 and GL 88-20, Supplement 4.

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SECTION 7

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

SECTION 7

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 PLANT IMPROVEMENTS AND SAFETY FEATURES

Defining a plant vulnerability as "a scenario which contributes inordinately to the HCGS core damage frequency", this study found no vulnerabilities owing to external events at the HCGS.

However, this IPEEE identified the need for a missile shield in front of the door to the Technical Support Center (TSC) HVAC room (Room 5619, Door 19), at the 153 foot elevation (PSE&G, 1995f - Figure 1.2-7). In addition to the non-1E cabinets, Room 5619 contains Channel C conduit associated with Loop A of the Control Room Emergency Filtration Units. This missile shield installation is scheduled to be completed in 1997 [PSE&G, 1997d]. The system design basis requies cables to be protected from missiles. The purpose of this shield is to protect against tornado missiles which could damage safety related cables in this room.

7.2 SEISMIC FEATURES

The seismic PSA found that the HCGS is robust against seismic events and the frequency of seismic core damage scenarios is an order of magnitude lower than the internal events core damage frequency. The reason for the low core damage frequency is a combination of a low hazard (mean frequency of the SSE is less than 5E-05/yr.) and generally high capacity of the equipment relative to the hazard. Neither relay chatter nor soil failures are risk significant. The seismic-fire interaction walkdown found that the fire water tanks located outside the fire water blockhouse are not seismically robust. Accordingly, fire water systems were not credited in the seismic PSA as a backup water source.

7.3 FIRE FEATURES

The fire PSA found that the HCGS is generally well designed against fires because of adequate separation of electrical divisions. The rooms which are most important to the calculated core damage frequency are 1) those in which separation can not be achieved or 2) those in which loss of offsite 4kV 1E power might accompany loss of one or more on-site 1E channels owing to a single fire. Therefore, the control room and the four diesel generator rooms, taken together, emerge as the most important contributors to fire induced core damage scenarios. Both channels of offsite 4kV 1E electrical power are contained within Aluminum ducts on the ceiling of the diesel generator rooms. The A and B switchgear rooms also emerge as important contributors owing to the conservative assumption that any fire in these rooms disables the entire corresponding 1E channel.

A major focus of this study was the identification of inter-compartment fire propagation paths via openings in walls, open fire dampers, and normally open fire doors. It was found that the plant does not maintain normally open fire doors unless a full time watch is posted. Using the compartments defined by the FIVE criteria [EPRI, 1993b], it was found by analysis that failure of active fire dampers to close, as well as openings in otherwise solid fire barriers, do not pose a risk of the spread of damage between compartments.

Another focus of the study was the potential of fires in so-called high hazard rooms to breach a solid fire barrier causing either severe structural damage to the plant or significant additions to core damage frequency not usually found by a FIVE or PSA approach. This study found that the location of high hazard rooms within the HCGS is such that none of them pose a significant additional risk even if the fire breaches fire barriers.

7.4 HIGH WINDS, FLOODS, AND OTHERS FEATURES

The HCGS meets the 1975 SRP [NRC, 1975a] requirements for high winds and floods. Therefore, it is considered robust with respect to these external hazards. In addition, analysis has found that this conclusion holds with respect to the new PMP criteria. An update of the potential hazards associated with transportation and nearby facility accidents was conducted for this IPEEE. It confirmed that the acceptable status of these, with respect to the 1975 SRP, has not changed since the operating license was issued. However, the IPEEE was helpful in identifying significant, but temporary, during the summer of 1996, increases in explosive shipments along the Delaware River, and explosive storage near the Hope Creek and Salem Nuclear Generating Stations. PSE&G's investigation revealed that the shipments took place without proper notification to the U. S. Coast Guard, and that the storage facility was not licensed to store explosives. Since its findings about this issue, the U. S. Coast Guard has stopped explosive shipment and storage near the HCGS site and does not see any need for explosive shipment or storage along the Delaware River in the foreseeable future [PSE&G, 1997e].

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SECTION 8
SUMMARY AND CONCLUSIONS
(Including Proposed Resolution of USIs and GIs)

SECTION 8

SUMMARY AND CONCLUSIONS
(Including Proposed Resolutions of USIs and GIs)

8.1 OVERVIEW OF THE IPEEE

The examination of external event severe accident vulnerabilities, as requested by the NRC in Generic Letter 88-20, Supplement 4 [NRC, 1991a] has been completed for the Hope Creek Generating Station. The IPEEE for the HCGS was performed using methods identified in NUREG-1407 (NRC, 1991b). With vulnerabilities defined as "a scenario which contributes inordinately to the HCGS core damage frequency," the principal conclusion is that the IPEEE did not identify any vulnerabilities for the Hope Creek plant.

The IPEEE did identify the need for a missile shield in front of the door to the Technical Support Center (TSC) HVAC room (Room 5619, Door 19). This missile shield installation is scheduled to be completed in 1997 [PSE&G, 1997d]. The IPEEE also identified a temporary increase in explosive shipment and storage along the Delaware River (Section 5).

The methodology used to perform the analysis is described in detail in Sections 2, 3, 4 and 5. As discussed below, the principal objectives of the IPEEE have been met as a result of the work performed for this submittal.

- PSE&G has developed an appreciation of severe accident behavior with regard to earthquakes, fires, and other external events as a result of investigating the nature of these hazards and potential core damage scenarios that might occur as a consequence of these hazards.

- PSE&G understands the most likely earthquake and fire induced severe accident sequences that could occur at the plant under full power conditions because of the PSAs performed for these hazards.

- The evaluation of the frequency of fire and earthquake induced scenarios coupled with the overall containment performance evaluations has allowed PSE&G to gain a qualitative understanding of the overall likelihood of core damage owing to external events.

- Because of the substantial conservatism built into this IPEEE analysis, PSE&G has determined that it is not necessary to reduce the overall likelihood of core damage and fission product release by modifying hardware or procedures.

In addition, this study identified key assumptions and major sources of uncertainties. Sensitivity studies were performed to develop a further understanding of the influence of assumptions and uncertainties on the quantitative core damage frequency results.

8.2 SUMMARY OF MAJOR FINDINGS

This paragraph summarizes the major findings from the external events evaluation of the HCGS. Fire and seismic events were the only important external event contributors to core damage frequency at the HCGS.

8.2.1 SEISMIC EVENTS

The IPEEE evaluation estimates seismic related core damage frequency of 3.6E-06 per year if the conservative Livermore seismic hazard curve [NRC, 1994b] is used. If the EPRI hazard curve [EPRI, 1989a] is employed a seismic core damage frequency of 1.0E-06 per year results. The industry judges that the EPRI hazard curve is more realistic. The most important seismic sequences are (LLNL values reported):

- SDS 36 (S-IC1): A seismic induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ481. Core damage is assumed although recovery actions are possible (69.4 percent of the seismic CDF).

- SDS 37 (S-DC): A seismic induced failure of 1E power to all four 125Vdc distribution panels 1A/B/C/D417. Core damage is assumed although recovery actions are possible (12.2 percent of the seismic PSA result).

- SDS-26 (S-OP-HP): A seismic-induced loss of offsite power and failure of high pressure injection, with simultaneous random failures which result in core damage. The random failures which cause core damage are dominated by reactor depressurization failures which result in inadequate ECCS injection or Emergency Diesel Generator (EDG) failures which result in station blackout (5.3 percent of the seismic PSA result).

- SDS-35 (S-IC2): A seismic induced failure of all four divisions of 1E 120Vac instrumentation distribution panels 1A/B/C/DJ482. Credit is taken for manual system control to prevent core damage, but failure of both results in core damage and primary containment isolation failure (4.4 percent of the seismic PSA result).

- SDS-18 (S-OP) A seismic-induced loss of offsite power with subsequent random failures which result in core damage. The random failures are dominated by Emergency Diesel Generator failures which result in station blackout (3.6 percent of the seismic PSA result).

As indicated by the relatively few components which were shown to have HCLPFs less than 0.5g PGA or median capacities less than 1.5g PGA, the HCGS is a seismically robust unit.

Using the EPRI hazard curve, no seismic core damage sequence exceeds the 1E-06/yr. reporting criterion. Using the LLNL hazard curve, only the first core damage sequence shown above exceeds this criterion.

No relay chatter interactions requiring human actions are needed, based on the low ruggedness relay evaluation. It is concluded that relay chatter is not significant to safe shutdown after a seismic event at the Hope Creek plant.

Containment performance systems and equipment were explicitly included in the walkdowns and seismic PSA. No vulnerabilities, which could cause early failures of containment, or containment bypass, were identified.

Absolute and differential displacements associated with seismically induced soil settlement and lateral spreading were shown to be insignificant.

The key assumptions, which lead to a generally conservative analysis, are as follows:

- Use of LLNL hazard curves.

- Human error probabilities for recovery actions were increased by an order of magnitude over internal event values.

- Failure of both divisions of 120Vac 1E instrumentation distribution panels caused core damage without the possibility of recovery.

Sensitivity studies were performed to investigate the effect of these assumptions on calculated seismic core damage frequency. It is recognized that uncertainties are associated with modeling and data. The hazard and human error probability assessments pose modeling uncertainties. Uncertainties in fragilities are associated with random variability as well as modeling uncertainties. Uncertainties in the conditional core damage probability calculation are due to failure probability parameter uncertainties and modeling uncertainties (or assumptions) associated with the success criteria.

8.2.2 INTERNAL FIRES

A total CDF from fire events at the HCGS was calculated to be 8.1E-05 per year. This frequency is distributed among the HCGS buildings as shown in Table 4.28.

The Control/Diesel Building, which houses the control area and the diesel generators, is the most significant building contributing 86% of the fire induced CDF. This was expected because of the good separation of equipment in the Reactor Building and the lack of safety related equipment in the other buildings. Typically, rooms or areas in which there is a confluence of equipment and/or cables from different electrical divisions dominate the fire risk. This occurs in the Control/Diesel Building at the HCGS, particularly in the cable spreading room, lower control equipment room, control room equipment room mezzanine, upper control equipment room, diesel generator rooms, electrical access rooms, and control room. One of the primary values of a fire PSA is the identification of the most important locations in the plant with respect to the fire risk. Table 4.29 summarizes the contribution of the top 16 compartments . These 16 compartments make up comprise 95% of the total CDF.

A detailed containment performance review was performed with the conclusion that no containment failure modes are unique to fire induced sequences. All containment failure modes, found for fire induced sequences, were treated in the HCGS IPE (PSE&G, 1994a).

The walkdown [PSE&G, 1997b] was valuable in assuring that the study was based on the as-built plant configuration, particularly with respect to cable runs and the Fire Risk Scoping Study issues. It found the plant to be well designed from the perspective of fire damage to redundant equipment. High fire hazard areas, which were screened out of the analysis using the FIVE [EPRI, 1993b] criteria, do not pose a significant risk.

The calculation of core damage frequency is considered very conservative primarily because of the following methods, approximations and assumptions:

- Fires in all compartments were assumed to induce a reactor trip. This reactor trip was modeled as an MSIV closure, unless a more severe initiating event was identified.

- The core damage frequency calculations took credit for only two recovery actions:

    1. Alternate HVAC
    2. Unit control from remote shutdown panel

- If a cable within an electrical channel in a compartment was found to exceed the cable damage criterion, the entire channel was assumed to be disabled.

- No credit was taken for protection owing to conduits or enclosed cable trays.

- Thirty percent of fires in cabinets in the control room, lower control equipment room, and switchyard blockhouse were conservatively assumed to cause hot shorts.

- The fire propagation and damage methodology employed in this study was based on the FIVE Fire Screening Methodology. Conservative application of this methodology included use of peak heat release rates corresponding to fully developed fires over the entire fire duration. Damage to cables was unrealistically calculated to occur in tens of seconds. Fire extinguishment before fire damage could not be demonstrated by the use of this method.

- All large fires in a diesel generator room cause a loss of all 4kV 1E power.

- Any cabinet fire in the 1E switchgear rooms was assumed to cause the loss of the entire corresponding electrical channel.

Sensitivity studies were performed with respect to 1) the height and horizontal displacement of the fire target with respect to the fire source in each unscreened compartment, 2) the size of the source fire in many of the unscreened compartments, 3) the potential to damage the 4kV bus bars in the diesel generator rooms, and 4) the contribution of automatic fire suppression on the calculated core damage frequency.

It is recognized that database uncertainties arise from 1) application of a generic database to a specific plant, 2) selection of a database and the underlying screening of data to create the database. Modeling uncertainties arise from use of the FIVE derived fire damage models and assumptions about success criteria. Parameter uncertainties are inherent in damage thresholds (or fragilities), and non-fire induced component failure probabilities.

8.2.3 HIGH WINDS, FLOODS AND OTHER EXTERNAL EVENTS

The method of progressive screening, per Section 5 of NUREG-1407 (NRC, 1991b), was used in this assessment. Beginning with the list of external events found in NUREG/CR-2300 (NRC, 1983a), the class of external events termed "other external events" have been screened out by 1) compliance with the 1975 SRP criteria [NRC, 1975a], 2) bounding engineering or 3) bounding probabilistic screening analyses. The study provides certification that no plant-unique external event is known that poses a significant threat of severe accidents. The study also provides confidence that the HCGS units are not vulnerable to other external events.

During the course of IPEEE two potential problems were identified and resolved as explained:

a) The first issue was a temporary increase in river explosive shipment and storage near the plant, but outside the five mile zone. This explosive shipment and storage along the Delaware River was halted permanently after the U.S. Coast Guard became aware of the shipment and the fact that the storage facility was not licensed to store explosives. Although the Hope Creek Plant is designed to withstand the impact of such explosives on the channel way of the Delaware River, detonation of these explosives at the storage facility might have impacted a few evacuation routes.

b) The second issue was discovered during the plant walkdown with respect to tornadoes. It revealed that Room No. 5619, which contains the HVACs for the Technical Support Center, contains Safety Conduits related to the "A" Control Room Air Supply. This room has an ordinary door (Door No. 19, at the 153 foot elevation) at its entrance. This door could be damaged due to tornado forces.

Therefore, it was judged that from the plant design point of view, it would be essential to install a missile shield barrier in front of Door 19 to protect conduits in Room 5619 against tornado induced missiles. Installation of this missile shield is to be completed in 1997 (PSE&G, 1997d).

8.3 RESOLUTION OF UNRESOLVED SAFETY ISSUES

The following observations and conclusions regarding unresolved and generic safety issues emerged from this study:

- The USI A-17 issue regarding system interactions is resolved in this study by incorporation of seismic interactions directly into the fragilities and system models.

- The Charleston (Eastern Seismicity) Earthquake issue is resolved because the HCGS is not one of the eight outlier plants identified in NRC, 1991b.

- The USI A-45 issue regarding shutdown decay heat removal was resolved in this study. The seismic walkdown and fragility analyses found that the shutdown decay heat removal systems (RHR and hard pipe containment vent) are seismically robust. The fire PSA found that fire induced loss of decay heat removal scenarios are a small fraction (on the order of 10%) of the total fire induced CDF.

- The GI-57 issue, regarding effects of fire protection system actuation on safety related equipment, was resolved in this study. The walkdown found adequate drainage and an evaluation of spurious actuation of the fire protection system found no vulnerabilities.

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