GRAND GULF NUCLEAR STATION

ENGINEERING REPORT

FOR

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS

SUMMARY REPORT

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1.0 EXECUTIVE SUMMARY

NRC Generic Letter 88-20, Supplement 4, [4] requested that licensees perform an "Individual Plant Examination of External Events (IPEEE)" to assess severe accident vulnerabilities. This report is presented to document the completion of all portions of that evaluation. Internal Flooding was included in the Individual Plant Examination (IPE) for internal events, which was submitted on December 23, 1992, to the NRC via letter no. GNRO-92/00157 [2].

1.1 Background and Objectives

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there was no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. As part of the implementation of the Severe Accident Policy, the Commission issued Generic Letter 88-20 on November 23, 1988, requesting that each licensee conduct an Individual Plant Examination (IPE) for internally initiated events including internal flooding.

Many PRAs indicate that, in some instances, the risk from external events could contribute significantly to core damage. In December 1987, an External Events Steering Group (EESG) was established by the NRC to make recommendations regarding the scope, methods and coordination of the Individual Plant Examination of external events (IPEEE).

In June 1991, the NRC issued Supplement 4 to Generic Letter 88-20 requesting a plant specific analysis of external events. Jointly issued with Supplement 4, NUREG 1407 [5] was issued to give procedural and submittal guidance for the IPEEE.

The objectives of the IPEEE, as outlined in NUREG-1407, are:

1. To develop an appreciation of severe accident behavior.
2. To understand the most likely severe accident sequences that could occur at Grand Gulf Nuclear Station (GGNS) under full power operating conditions.
3. To gain a qualitative understanding of the overall likelihood of core damage and fission product releases.
4. If necessary, to reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

Entergy Operations, Incorporated (EOI) has completed and documented an IPEEE for GGNS. This report contains a summary of the methods, results, and conclusions for the IPEEE. This report complies with the NRC request for information contained in Generic Letter 88-20, Supplement 4 and NUREG 1407.

1.2 Plant Familiarization

Grand Gulf Nuclear Station is located on the east bank of the Mississippi River in Claiborne County, Mississippi about six miles northwest of Port Gibson, 25 miles southwest of Vicksburg and 37 miles northeast of Natchez. GGNS is a General Electric BWR/6 reactor with a Mark III Containment. It has a rated and licensed core thermal output of 3833 MWt that corresponds to a net electrical output of 1254 MWe when the contribution from the reactor coolant pumps is considered. The design power level is 4025 MWt. The Mark III containment incorporates the drywell/pressure suppression concept. This site was originally designed to support two units, but Unit 2 has been canceled. GGNS received its full power operating license on August 31, 1984, and commenced commercial operation on July 1, 1985.

The nuclear boiler system includes a direct cycle, forced circulation boiling water reactor that produces steam for direct use in the steam turbine. The reactor vessel is a 251 inch BWR/6 vessel with 800 fuel assemblies. Overpressure protection is provided by 20 safety/relief valves that discharge into the suppression pool.

The containment is designed to contain the energy released from the design basis loss of coolant accident (LOCA) and provide a leak tight barrier against the uncontrolled release of radioactivity to the environment. Its design incorporates a drywell surrounding the reactor pressure vessel and a large part of the reactor coolant pressure boundary; a suppression pool that serves as a heat sink during normal operational transients and accident conditions; and a steel lined reinforced concrete containment structure.

The plant design incorporates redundant Engineered Safety Features (ESF) which in conjunction with the containment ensure off site radiological consequences do not exceed regulations. Some of the important ESF systems include: Emergency Core Cooling Systems [which include High Pressure Core Spray (HPCS), Automatic Depressurization System (ADS), Low Pressure Core Spray (LPCS) and Low Pressure Core Injection (LPCI)]; Containment Spray (CS) subsystem of Residual Heat Removal (RHR); Standby Gas Treatment System (SGTS); and the Suppression Pool Makeup (SPMU) system. Other systems credited in the IPEEE for mitigating accidents include Reactor Core Isolation Cooling (RCIC), Control Rod Drive (CRD), Power Conversion System (PCS), Standby Service Water (SSW) cross-tie to LPCI, Firewater aligned for vessel injection, the Suppression Pool Cooling (SPC) and Shutdown Cooling (SDC) modes of RHR, Containment Venting, Alternate Rod Insertion (ARI), Recirculation Pump Trip (RPT), and Standby Liquid Control System (SLCS).

GGNS generates electrical power at 22kV that is connected to the 500kV switchyard through a main step up transformer. In the event of loss of normal and preferred auxiliary power sources, the ESF loads and various non-safety loads (under certain conditions) can be supplied from the on-site emergency power sources. The emergency on-site power sources consist of two independent and completely segregated emergency diesel generators, each of which has an adequate capacity to meet the loads required for safe shutdown of the reactor. A third diesel is used as a backup power source to the HPCS. Three divisions of emergency DC power are also available to supply the ESF loads as required.

The SSW system provides required cooling water to ESF equipment served by the system as well as to various non-safety related portions of the plant. That portion of the SSW required for safe shutdown of the plant is designed to meet Seismic Category 1 requirements and the single failure criterion. Non-safety related portions are isolated from the SSW during events resulting in ESF actuation.

Heating Ventilating and Cooling (HVAC) systems are present to cool ESF system equipment such as ECCS pump rooms, diesel generator rooms, SSW pump rooms and electrical switchgear.

1.3 Overall Methodology

The IPEEE consist of three separate analyses:

- Seismic Analysis
- Internal Fire Analysis
- High Winds, Floods and Others Analysis

For Seismic analysis, the EPRI Seismic Margins Methodology (1) was used to perform th analysis. The GGNS Fire PRA was performed using the methods described in EPRI report 3385-01, "Fire Risk Analysis Implementation Guide" (Reference 27). This method calculates a core damage frequency associated with fire events. In doing so it considers both the likelihood of fire in individual locations of the plant and the potential for damage to equipment by the fire as well as the potential for random failures in equipment utilized to mitigate the fire event. For High Winds, Floods and Others analysis, a review was performed to demonstrate conformance to the 1975 SRP. As required in NUREG 1407, evaluations for the effects of localized Probable Maximum Precipitation (PMP) consider the revised estimates published in Hydrometeorological Report (HMR) Nos. 51 and 52.

1.4 Summary of Major Findings

For the Seismic Analysis, Grand Gulf Nuclear Station is identified as a reduced scope plant by NUREG 1407. Therefore, the Safe Shutdown Earthquake (SSE) ground response spectra and corresponding in-structure response spectra were used as the Review Level Earthquake (RLE).

The conclusions of the seismic analysis is that Grand Gulf Nuclear Station is seismically rugged and that all components identified in the Safe Shutdown Path have adequately considered the seismic input. All anchorage to these components was found to be rugged. Only one potential vulnerability to a seismic event was identified, which has been corrected.

The estimated core damage frequency from internal fires at GGNS is 8.76E-06/yr, which is less than the core damage frequency due to internal events of 1.72E-5/yr reported in the GGNS IPE. Fire risk due to the control room contributes 44% of the total internal fire result. Twelve other fire compartments contribute from approximately 11% to 0.1% of the total. The remaining 171 compartments were screened as a part of the analysis and do not contribute to the total. Due to conservatisms in the analysis this is an upper bound estimate of the true core damage frequency. The core damage frequencies estimated for internal fires are not and should not be directly compared to the core damage frequencies calculated in the internal events IPE. The conservatisms and uncertainties associated with a fire PRA are much greater than those associated with an internal events PRA.

For Flooding, Transportation & Nearby Facility Accidents, and Others, GGNS is in compliance with criteria in the '75 Standard Review Plan (SRP). Significant changes to site features were not noted during IPEEE inspections.

With regards to High Winds and Tornadoes, GGNS is in compliance with the '75 SRP with only a few exceptions. A probabilistic evaluation(19) for frequency of occurrence for these exceptions determined a frequency of 0.77 X E-8/yr. This low frequency of occurrence is well below the screening criteria in IPEEE.

As a result of the Flood analysis, enhancements were identified which would help ensure continued compliance with the SRP and mitigate adverse effects of the new PMP criteria. Section 7.3.2 contains details of these enhancements.

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

EOI has completed an IPEEE for Grand Gulf Nuclear Station. This Section provides details on the conformance with the Generic Letter and supporting material, the general methodology and the information assembly.

2.2 Conformance With Generic Letter and Supporting Material

This report conforms with Generic Letter 88-20, Supplement 4 and its supporting material. NUREG-1407 was followed closely in preparing this report. The content and format of this report conforms with the requirements of NUREG-1407.

A major thrust of GL 88-20 is that the Utility should gain the insights into severe accident behavior. EOI has expanded significant resources developing in-house personnel to perform the IPEEE. The majority of contractor work was performed on-site and in-house personnel worked closely with the contractors. Because of this working philosophy, the insights and knowledge gained in performance of the IPEEE have been retained by the utility.

Technical adequacy of the IPEEE is assured by a combination of:

- Use of information from reliable documents.
- Use of knowledgeable individuals
- Evaluation by multiple individuals (where appropriate).
- Performance of a PEER review.

For the Seismic analysis, the Level 1 PRA was used in determining what components are required for safe shutdown. Two "Seismic Capability Engineers" as defined in EPRI NP6041[1] evaluated the component for seismic adequacy. Where questions arose on the seismic walkdowns, additional sources of information, such as existing calculations, were referred to.

There is a high confidence in the technical accuracy of the IPEEE. Knowledgeable individuals performed the analysis using reliable sources of information. Additionally a Peer review was performed which provides additional assurance that the IPEEE is technically accurate.

2.3 General Methodology

The IPEEE consists of three separate analyses:

- Seismic Analysis
- Internal Fire Analysis
- High Winds, Floods and Others Analysis

This Section gives a brief description of the methodology used in performing these analyses.

2.3.1 Seismic Analysis

Grand Gulf Nuclear Station (GGNS) is classified a reduced scope plant as defined NUREG-1407 based on the low seismicity. Therefore, a seismic review of the plant was performed to the plant's original design basis. This was accomplished by performing a Seismic Margins Assessment (SMA) of the Safe Shutdown Equipment List (SSEL) with plant walkdowns in accordance with the guidelines and procedures documented in Electrical Power Research Institute (EPRI) Report NP-6041-SL.[1]

Since Grand Gulf Nuclear Station is a reduced scope plant, the original design basis Safe Shutdown Earthquake (SSE) ground response spectra and corresponding in-structure response spectra were used as the Review Level Earthquake (RLE) input for the walkdown and evaluation, as requested by NUREG-1407. No new in-structure response spectra were developed and those described in the Grand Gulf Nuclear Station Updated Final Safety Analysis Report (UFSAR) (15) were utilized.

Safe shutdown success paths were developed to identify the systems that must function to successfully shutdown and cool the reactor following the occurrence of a RLE. A safe shutdown success path is a string of systems which is used to accomplish all of the required safe shutdown functions. This success path can be depicted in a Shutdown Path Logic Diagram (SPLD). The SPLD identified 44 systems and 567 components which required evaluation for seismic adequacy.

2.3.2 Fire Analysis

The fire PRA methodology used in this study employs NUREG/CR-2300 for the basic framework; however, more recent data sources and technical bases are used. For example, EPRI TR-100370 (FIVE) is one of the more recent fire sources used extensively as a resource. This method draws upon not only FIVE, but data sources in NUREG/CR-4840 and NUREG/CR-2815, and insights from both Sandia's and EPRI's fire research programs. These data and bases allow the removal of many of the conservatisms and non-conservatisms that characterize earlier fire PRAs. The key assumptions and specifics of this approach are described in the Implementation Guide. Additional discussion of the methodology is provided in Section 4.0 of this report.

2.3.3 High Winds, Floods and Others

High Winds, Floods and Others are external events other than seismic, internal fire, or internal flooding events that may be initiators of accident sequences leading to core damage. Such phenomena are potentially important because they affect multiple components. An accident involving a number of different component failures may be nearly incredible in the absence of some external influence, but may be possible or even likely by the occurrence of a tornado, for instance.

As recommended in Generic Letter 88-20, Supplement 4, the methodology employed for analyzing other external events at GGNS was a screening approach. The first step in the screening approach was to determine if the criteria of the 1975 Standard Review Plan (6) are met.

The progressive screening approach as outlined in Generic Letter 88-20, Supplement 4, was used in the evaluation of transportation and nearby facility accidents. Guidance from NUREG-1407 and NUREG/CR-5042 (16) was also used in the evaluation of specific hazards for the determination of potential vulnerabilities and insight for screening criteria.

As required in NUREG 1407, evaluations for the effects of localized Probable Maximum Precipitation (PMP) on site flooding and roof ponding consider the revised PMP estimates published by the National Oceanic and Atmospheric Administration in HMR No. 51. Local intense precipitation on adjacent streams, and the site yard are evaluated applying HMR No. 52, Regulatory Guide 1.59, Regulatory Guide 1.102, and ANS 2.8-1992. A backwater analysis is performed to determine plant water elevations. Food water entering the powerblock is estimated and evaluated to determine if it can jeopardize components required to safely shutdown the plant and/or maintain the plant in a shutdown condition. Evaluations for the effects of local intense precipitation include assessments of the potential hazards resulting from roof ponding.

2.4 Information Assembly

Evaluations were performed using current revisions of As-Built drawings, specifications, standards, plant procedures, and other sources. Design change packages and non-conformance reports were reviewed where applicable. The As- Built conditions of plant features were verified by visual inspection and measurements in the field where required or appropriate. Evaluations for each section of the submittal (i.e., seismic, fire, flooding, high winds, etc.) are documented in detail in separate engineering reports, each prepared in accordance with the GGNS Quality Assurance program. These reports were also reviewed by an independent Peer Reviewer. Similarly, detailed calculations performed in support of these reports were prepared in accordance with the GGNS Quality Assurance program and made available for Peer Review.

Detailed evaluations pertaining to this submittal are contained in the following reports:

- Engineering Report GGNS-93-0001, Rev. 0 "Individual Plant Examination for External Events (External Flooding)".
- Engineering Report GGNS-94-0053, Rev. 0, "IPEEE Reduced Scope Seismic Margins Assessment (SMA)".
- Engineering Report GGNS-93-0048, Rev. 0, "High Wind and Tornado Assessment".
- Engineering Report GGNS-93-0047, Rev. 0, "Transportation & Nearby Facility Accidents".
- Engineering Report GGNS-93-0031, Rev. 0, "An Assessment of the Risk From Lightning Initiators".
- Engineering Report GGNS-95-00041, Rev 0, "Internal Plant Examination of External Events, Fire"

SEISMIC ANALYSIS

3.0 Methodology Selection

Grand Gulf Nuclear Station (GGNS) is classified a reduced scope plant as defined NUREG-1407 based on the low seismicity. Therefore, a seismic review of the plant was performed to the plant's original design basis. This was accomplished by performing a Seismic Margins Assessment (SMA) of the Safe Shutdown Equipment List (SSEL) with plant walkdowns in accordance with the guidelines and procedures documented in Electrical Power Research Institute (EPRI) Report NP-6041-SL [1].

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3.1 EPRI Seismic Margins Method (SMM)

The EPRI Seismic Margins Method (SMM)[1] was used to perform the seismic analysis. This section provides the details about the analysis.

3.1.1 General Plant Information and Seismic Input

This section provides general plant information and the seismic input used in the analysis.

3.1.1.1 General Plant Information

GGNS is located in Claiborne County in southwestern Mississippi. The plant site is on the east side of the Mississippi River about 25 miles south of Vicksburg and 37 miles north-northeast of Natchez. The Grand Gulf Military Park borders a portion of the north side of the plant site property and the community of Grand Gulf is about 1½ miles to the north. The town of Port Gibson is about 6 miles southeast of the plant site.

The site and its environs consist primarily of woodlands and farms. The total area of the plant site is approximately 2100 acres. Within this area are two lakes, Gin Lake and Hamilton Lake. These lakes were once the channel of the Mississippi River and average about 8 to 10 feet above mean sea level (msl).

The western half of the plant site consists of materials deposited by the Mississippi River and extends eastward from the river about 0.8 miles. This area is generally 55 to 75 feet above mean sea level (msl).

The eastern half of the plant site is rough and irregular with steep slopes and deep-cut stream valleys and drainage coursed. Elevations in this portion of the plant site range about 400 feet above mean sea level (msl) occur on the hilltops east and northeast of the site.

Surface material at the site is Pleistocene loess. This material erodes easily forming very steep slopes along stream channels . One such slope, along the Mississippi River flood plain, divides the site so that it lies in two sub provinces of the Central Gulf Coastal Plain Physiographic province. The sub provinces are the Loess or Bluff Hills to the east and the Mississippi alluvial plain to the west.

The site is underlain by approximately 18,000 feet of Cretaceous through Cenozoic sands, gravel's, clays, marls, claystones, sandstones and limestones. These sediments were deposited on middle Jurassic evaporates, the parent material for salt domes found in the area. Regional dip is southward and becomes progressively steeper toward the Gulf Coast. As a result of the steepened dip, most formations tend to be wedge shaped, thickening coastward.

Several domal or structural uplift areas are found within the Gulf Coast Basin. The nearest of these, located about 50 miles east-northeast of the site, is the Jackson Dome. Formation of this structure began in the early Cretaceous period and ended in the middle Tertiary period. A salt dome has been formed as near as 8 miles from the site. The dome was formed from the late Cretaceous period through the Oligocene epoch. No nearer salt domes are known.

Most deep site borings encountered the Miocene age Catahoula formation. The Catahoula consists of a hard-to-very-hard, gray-to-gray-green, silty-to-sandy clay, and clayey silt and sand, with some locally indurated or cemented clay, sand and silt seams. The Catahoula Formation is the bearing stratum for the major plant structures. The maximum estimated thickness of the Catahoula formation at the site is 320 feet.

Uncomformably underlying the Catahoula formation is the Vicksburg Group, a sequence of four formations of Oligocene age. These formations, from youngest to oldest, are the Bucatunna, the Byram, the Glendon, and the Mint Spring. The Bucatunna is a 53 foot thick layer of stiff-to-hard green-black-to-black clay with thin, gray, fine sand seams. The Byram Marl, underlying the Bucatunna, is hard-to-very-hard, green-to-gray, fine sandy, calcareous clay approximately 5 feet thick. The Byram Marl is discontinuous throughout the region. Conformably underlying the Byram Marl is the Glendon Formation. It consists of a series of interbedded, light gray, fossiliferous limestones and hard-to-partly-indurated, grayish-green, fine sandy, calcareous clays.

Total thickness is about 46 feet. Underlying the Glendon is the Mint Spring Marl. Forty feet of the Mint Spring Marl was penetrated at the sit; however, the total thickness of the formation was not determined. The formation consists of hard, grayish green fossiliferous, glauconitic sand and clay.

Aerial photographic interpretation and geologic mapping of outcrops and excavations did not detect the presence of any faults or tectonic structures in the plant vicinity. Inspections of outcrops, exposures in excavation, and subsurface samples have revealed that there are no deformational zones within the Catahoula material, which is the foundation material for the major plant structures. There are no reversals of dip of the Catahoula Formation in the vicinity of the site. Exposures of the Catahoula contain occasional preferred joint orientations of N75°E, N45°W and N45°E. The joints are tight and contain no altered materials. Core samples of Catahoula material at the site do not exhibit shear zones or fractures.

There is no evidence to suggest that surfacial or subsurface materials at the site have been affected by prior earthquake activity. No faults were encountered by the numerous site boring or exposed in any of the excavations.

The Gulf Coast Basin tectonic province, in which the site is located, is characterized by infrequent earthquakes of low epicentral intensities (Modified Mercalli Intensity VI or less), with an attendant low seismic-risk level.

All documented earthquakes of epicentral intensity IV to V or greater which have occurred within 200 miles of the site have been investigated. The historical earthquake which occurred nearest to the site had a maximum intensity of III to IV and occurred on June 28, 1941, at Vicksburg, Mississippi, about 25 Miles north-northeast of the site. It is the only earthquake reported within 100 miles of the site.

An anomalous zone of seismicity, the new Madrid seismic zone, is located within the Mississippi Embayment. The New Madrid seismic zone is characterized by a high level of seismicity and potential intensity (epicentral intensities to XII). The great earthquake series of 1811-1812 occurred in this zone, near New Madrid Missouri, about 325 miles from the site. The nearest approach of this zone to the site is approximately 220 miles. Notwithstanding its relatively great distance from the site, the New Madrid seismic zone is a potential source of ground motion at the Grand Gulf site. Accordingly, all know earthquakes of intensity V or greater which have occurred within the northern Mississippi Embayment were considered in establishing the Seismic Design Basis for the site.

The Uniform Building Code designates the vicinity of the site as Zone 0 on the map intitled. "Map of the United States Showing Zones of Approximate Equal Seismic Probability." The U. S. Coast and Geodetic Survey indicates Zone 0 as an area of no earthquake damage.

The principal buildings and structures include the containment structure, the turbine building, the auxiliary building, the control building, the diesel generator building, the standby service water cooling towers and basins, the enclosure building, the radwaste building, and the natural draft cooling tower.

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

The Containment Structure is a Seismic Category I structure which encloses the reactor coolant system, the drywell, suppression pool, upper pool and some of the engineered safety feature systems and supporting systems. The functional design basis of the containment, including its penetrations and isolation valves, is to contain with adequate design margin the energy released from design basis loss-of-coolant-accident and to provide a leaktight barrier against the uncontrolled release of radioactivity to the environment, even assuming a partial loss of engineered safety features.

The Turbine Building houses all equipment associated with the main turbine generator and other auxiliary equipment. There are safety related instruments in the Turbine Building, but the building will not collapse onto or otherwise adversely affect the systems of which those instruments are part in the event of a postulated accident.

The Auxiliary Building is Seismic Category I structure that contains safety systems, fuel storage and shipping equipment and necessary auxiliary support system. Redundant safety trains in the auxiliary building and all other areas of the plant are separated and protected so that a loss of function of one train will not prevent the other train from performing its safety function.

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

The Diesel Generator Building is a Seismic Category I structure constructed of reinforced concrete. The building contains the three diesel generators, three fuel oil day tanks, six starting air receivers-compressors, air intake vents and filters, mufflers and controls. Each diesel generator and its associated equipment is in an individual room within the Diesel Generator Building.

The Enclosure Building is a limited leakage Seismic Category I structure that encloses the upper portion of the containment above the Auxiliary Building roof level. The enclosure building provides a boundary for the standby gas treatment system, which maintains a negative pressure in the volume between the Containment and Enclosure Building to ensure that leakage of radioactive materials from the containment is filtered prior to releases to the environment in the unlikely event of a loss-of-coolant-accident.

The Radwaste Building contains six major areas: the collection tankage area, a processing area, a pipeway area, a personnel area, a solidification area, and a storage area. The radwaste systems process liquid, solid and gaseous radioactive wastes generated by the plant.

The Natural Draft Cooling Tower is a Concrete, natural draft, hyperbolic structure. The tower is designed to dissipate all excess heat removed from the main condensers and accomplishes this function by the use of a spray network, a film type heat transfer surface, a tower basin, and circulating water pumps, piping and valves.

The Ultimate Heat Sink is a system comprised of two separate, Seismic Category I, mechanical draft cooling tower/pumphouse/basin structures. Each tower consists of four cells; each cell with a separate stack. Only four cells are required to support Unit 1 operation. The towers are constructed of a reinforced concrete frame with air intake louvers in the sides and contains ceramic tile fill blocks within the frame. Each tower is located over a separate concrete cooling water basin. Each pumphouse is located over the southwest corner of the basins, contains vertical pit pumps and is provided with separate tornado missile protection walls on all sides and on the roof.

3.1.1.2 Seismic Input to Structures and Equipment

The Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) design response spectra defined the free field vibratory motions for the Grand Gulf site. The peak horizontal ground acceleration of the OBE has been established to be 0.075 g's and SSE has been established to be 0.15 g's based on the vibratory ground motion studies. The design spectra which were used for the plant seismic design were obtained by modifying Newmark's curves. In developing these spectra, variations in site conditions, foundation properties and effects of focal and epicentral distance from the site were considered.

Seismic Category I structures, systems, and components have also been designed to withstand the effects of vibratory motion of at least the operation basis earthquake in combination with other appropriate loads within allowable stress limits of applicable codes.

For the vertical direction of seismic motion, the corresponding design earthquake was taken to be two-thirds of that established for the horizontal direction at the site.

3.1.2 System Analysis

This section describes the process used in selecting the equipment for which the seismic adequacy was determined.

3.1.2.1 Overall Approach

The project used a Seismic Margins Assessment (SMA) for the means to investigate the seismic external event of the IPEEE. The project identified a preferred and an alternate success path based on operational and systems considerations for GGNS which were originally based on the

PRA model developed for the GGNS IPE[2]. Once the front-line systems were identified, the systems required to support the operation of the front-line systems were identified. The components that must function in order for each system to work were then identified. The Safe Shutdown List (SSEL) was developed which contains the listing of the required components. This list was developed by GGNS engineers and reviewed by both an outside consultant and an independent reviewer. Development of the Shutdown Path Logic Diagrams (SPLD) and the SSEL is documented in Engineering Report GGNS-93-0017, Revision 0 [17].

The SPLD's were developed to identify the systems that must function to successfully shutdown and cool the reactor following the occurrence of a RLE.

3.1.2.2 Assumptions

The following assumptions for the development of the SPLD apply to the Seismic IPEEE are:

1. Offsite power is assumed to be failed due to the SME (Seismic Margin Event) and unrecoverable during the 72 hour time period of interest. Consider possible adverse effects if offsite power is not lost.

2. Path success is defined as the ability to achieve and maintain a stable hot or cold shutdown condition for at least a 72 hour period following the seismic event.

3. Assume a small break loss of coolant accident (LOCA) equivalent to 1" diameter.

4. With regard to success path and equipment selection the following apply:

   - Nonseismic caused unavailability of component or systems is not explicitly addressed but single train systems are to be treated with caution.
   - Identify primary and alternate Safe Shutdown Success paths.
   - Analyze redundant equipment where present in path (NUREG 1407).
   - Highlight non redundant equipment in path (NUREG 1407).
   - Use NUREG/CR-4826 screening approach for single train/multiple train systems (NUREG 1407).
   - To extent possible, select an alternate path involving different systems, piping runs, components, from the preferred success path (NUREG 1407).

5. Verify that postulated operations are viable (NP 6041).

6. Equipment for Seismic Evaluation Includes:

   - Active and passive fluid mechanical components in the safe shutdown success paths
   - Electrical equipment in the safe shutdown success paths

7. Equipment exempt from seismic evaluation includes:

   Check valves
   Valves with external operators that do not change state.

8. No relay evaluation for GGNS

3.1.2.3 Principal Safety Functions

The first step in the development of the SPLDs is to define the safety functions that must be accomplished to achieve and maintain a stable shutdown. The four safety functions are identified in EPRI NP-6041. These functions are reactivity control, reactor coolant system pressure control, reactor coolant inventory control and decay heat removal. The GGNS IPE(2) also identified safety functions that must be accomplished to successfully mitigate the events analyzed in the IPE. The initiating events of interest for this evaluation are Loss of Offsite Power (LOSP) and small LOCA. The safety functions are equivalent to the safety functions from EPRI NP-6041 except for the early containment over pressure protection function for the small LOCA initiating event. This function is accomplished by the successful operation of the vapor suppression system. The vapor suppression system consists of the weir wall inside the drywell, the drywell to suppression pool vents and the suppression pool. These components are all passive in nature and need only maintain their structural integrity in order to accomplish the function. These will be reviewed as part of the containment structural review. Therefore, the four primary safety functions identified above will form the basis for the identification of the front-line systems for the safe shutdown paths.

3.1.2.4 System Success Criteria

The systems that can accomplish each of the primary safety functions were defined. During the performance of the GGNS IPE, combinations of systems that are required to successfully function to successfully accomplish each safety function were identified. The systems that can successfully perform each safety function are summarized below.

Reactivity Control	Reactor Protection System and Control Rod Drive System
	Alternate Rod Insertion (and CRD System) and Reactor Pump Trip
	Manual Rod Insertion (and CRD System) and Reactor Pump Trip
	Standby Liquid Control System and Reactor Pump Trip
Reactor Pressure Control	Steam Line Safety Relief Valves
	Power Conversion System (MSIVs and Condenser)*
Reactor Inventory Control	Feedwater*
	High Pressure Core Spray System#
	Reactor Core Isolation Cooling System#
	Control Rod Drive System (injection mode)

	Low Pressure Core Spray and Depressuization with at least 4 SRVs#
	Low Pressure Core Injection and Depressuization with at least 4 SRVs#
	Condensate and Depressuization with at least 4 SRVs*
	Standby Service Water Crosstie to LPCI and Depressuization with at least 4 SRVs
	Firewater and Depressuization with at least 4 SRVs
	Suppression Pool Makeup
Decay Heat Removal	Power Conversion System*
	Suppression Pool Cooling Mode of RHR
	Containment Spray Cooling Mode of RHR
	Shutdown Cooling Mode of RHR
	Containment Venting

* Not available with Loss of Offsite Power
\# Suppression Pool Makeup required for these systems if LOCA

The above systems also require various support systems for successful operation.

3.1.2.5 Overall Success Path Logic Diagram

With the identification of the primary safety functions and the systems that can accomplish those safety functions an overall success path logic diagram (SPLD) can be developed for GGNS. A SPLD is a graphic representation that shows the combinations of systems whose successful operation will result in long term shutdown following the seismic margin earthquake. It can be envisioned as a simple electrical circuit diagram constructed in a series-parallel fashion. The Seismic margin earthquake is depicted as the node on the left, and the desired long-term safe shutdown condition is depicted as the node on the right. Between these two nodes are a number of system blocks arranged in a series-parallel manner, showing alternate paths of achieving the safe shutdown condition. The selected paths must represent paths that the control room operators will use based upon their training and procedures. Adequate instrumentation must also be available to the operators. An overall SPLD for GGNS is provided in Figure 11.1. This SPLD was constructed using the success criteria for a Loss of Offsite Power and Small LOCA initiators as discussed above. Note that some of the systems that are capable of being utilized are not included in the GGNS overall SPLD since they are not the systems that operators would preferentially use. It should be noted that support systems are also not included in Figure 11.1. Dependencies between front line systems and support systems are identified in Table 10.1.

The first node on the Overall SPLD is the function of Reactivity Control. This node consists of two parallel paths. One path is made up of the control rod drive system which works in conjunction with the reactor protection system. This path represents the insertion of control rods into the core in response to an automatic scram signal generated by RPS to shutdown the reactor. An automatic scram signal can be generated by any one of several results of a seismic event including the loss of offsite power. The second parallel path is made up of the standby liquid control system (SLCS) block. This block represents the injection of sodium pentaborate into the reactor coolant system such that the reactor is shutdown. The SLCS is only actuated manually and must be actuated quickly. EPRI NP-6041 recommends that SLCS not be relied upon because of assumed stress levels on the operators during a seismic event. Even though GGNS does not completely agree with this assumption because of the strength of the emergency operating procedures (EOPs), the high degree of operator training and the culture of adherence to procedure, this assumption will be maintained for this analysis. Therefore, no credit for the SLC System will be taken in the preferred and alternate path selections. The other methods of reactivity control are not included sense they rely on the CRD system also.

The second node in the SPLD represents the function of Reactor Pressure Control. This block represents the opening and closing of the SRVs to control reactor pressure. Because of the assumption of loss of offsite power no credit is taken for the power conversion system as it will lead to the isolation of the main steam isolation valves and prevent use of the condenser to control pressure.

The third node of the SPLD represents the function of Reactor Inventory Control. This node consists of two primary parallel paths. One path represents inventory control with high pressure systems and the other represents inventory control with low pressure systems. The high pressure path consists of two blocks in parallel. The top block represents the injection of water into the core at high pressure by the RCIC system. The lower block represents the injection of water into the core at high pressure by the HPCS system. Other high pressure inventory control methods (feedwater and CRD injection) are not included because of unavailability because of the basic assumptions or low capacity. The second path represents inventory control with low pressure systems and consists of two parts. The first part consists of the Automatic Depressurization System (ADS) block. This block represents depressurization of the reactor using the SRVs. Note that the EOPs direct the operators to inhibit automatic depressurization. Therefore, depressurization is only performed manually since the operators will follow the EOPs. Depressurization can be accomplished through the use of any of the SRVs but only the ADS valves are credited in this analysis. Depressurization is always required for use of the low pressure systems for inventory control. This is true even with the assumption that a small LOCA exists since the assumed break size is too small to depressurize the reactor in sufficient time to allow low pressure systems to inject prior to core damage. The top block represents the low pressure coolant injection (LPCI) mode of RHR. The bottom parallel path represents the low pressure core spray (LPCS) system. Other methods of low pressure inventory control are not included on the SPLD. Condensate would not be available because of the assumption of loss of offsite power. SSW crosstie to LPCI and firewater could be used but SSW crosstie is a lower priority system for injection in the EOPs and firewater would only be effective after inventory had been maintained by some other system for a period of several hours. The final block, which is in series with both parallel paths for reactor inventory control represents the suppression pool makeup (SPMU) system. This system is required only if a LOCA is assumed. It is necessary because of the loss of inventory from the suppression pool to the drywell with a LOCA inside the drywell. Systems such as LPCI and LPCS which take suction on the suppression pool could loose net positive head if this inventory loss if not made up. Both RCIC and HPCS can take suction from the condensate storage tank in addition to the suppression pool. However, the condensate storage tank is not credited for this analysis.

The fourth and final node of the SPLD represents the function of Decay Heat Removal. This node consists of two parallel success paths. The top path represents the removal of decay heat using the suppression pool cooling (SPC) mode of the RHR system. The lower block represents the decay heat removal with the shutdown cooling (SDC) mode of the RHR system. Both of these modes utilize the RHR heat exchangers. Note that for the SPC mode to work, decay heat from the core must be rejected to the suppression pool through the SRVs. This also requires a continued means of making up inventory to the reactor vessel. Other methods of decay heat removal not included on the SPLD are the containment spray mode (CS) of RHR and containment venting. Containment venting is not included since this method requires instrument air which may not be available during a loss of offsite power. CS mode is not included because its use is lower in priority to SPC and SDC in the EOPs.

Primary and Alternate Success Path Selection

The methodology requires the identification of two independent paths, a preferred and alternate path. The alternate path is selected to include equipment or a backup train of equipment so that the plant can be shut down in the event of an active failure or unavailability of a single item of equipment in the preferred path. In selecting the preferred and alternate shutdown paths, emphasis was placed on compatibility with plant procedures, operator training on the use of the systems, system unavailability, and minimizing the number of components required for performance of the plant walkdown and verification. In addition one of the success paths must be capable of mitigating a small break LOCA.

Preferred Path Selection

The preferred path consists of the control rod drive system, SRV division 1, RCIC, and RHR A in suppression pool cooling mode. Figure 11.2 shows the preferred safe shut down path.

In this case if a Seismic Margin Earthquake (SME) occurs coincident with a loss of offsite power, the reactor protection system will provide a SCRAM signal and the CRD system will provide the motive force for driving in rods. (Since the RPS is a de-energize to SCRAM system, any loss of power to the RPS will SCRAM the reactor.) If a loss of offsite power does not occur, a manual SCRAM can be performed. The control room operators verify that the reactor is shut down following the SCRAM signal.

Following a loss of offsite power Safety Relief Valves will be required to open in response to the pressure transient resulting from the closure of the turbine control valve and MSIVs. The MSIVs will close on loss of offsite power due to de-energization of the solenoids. If a loss of offsite power does not occur, the MSIVs can be manually closed or allowed to close on Low Steam Line Pressure. Division 1 (actuation and control logic) of the SRV is the success front line system for the reactor pressure control function in this path.

The RCIC system will automatically start on a low reactor water level signal (Level 2). According to procedure, operators will verify that the RCIC system is operating and providing inventory makeup. RCIC stops injection on high water level (Level 8) to prevent water carry over to the RCIC turbine. No manual actions are required for restarting injection since the system will automatically restart injection on low water level (Level 2). However, the operators will manually control RCIC in accordance with Emergency Procedures to prevent the cycling of RCIC injection. Note that Suppression Pool Makeup (Train A) is required for this path so that the inventory loss from the LOCA to the drywell can be made up. With out this makeup, NPSH for the RCIC and RHR pump could be lost.

The RCIC system, due to poor turbine-driven pump reliability tends to have a relatively high unavailability. EPRI NP-6041-SL recommends that two redundant systems be required for the success path in cases such as these. HPCS is also capable of injecting high pressure water to the reactor vessel although with a much higher flow rate than RCIC. The HPCS system will also automatically start on a low reactor water level signal (Level 2) and isolate at high water level (Level 8). While RCIC is the preferred source because it is more easily controlled, HPCS is an alternate source of high pressure injection to the vessel which is capable providing adequate cooling to the reactor core..

RCIC (or HPCS) will be used to maintain inventory while the SRVs will maintain reactor pressure, either automatically at the set relief pressure or manually in accordance with emergency procedures at a lower pressure range. RHR train A in SPC cooling mode is used to remove the decay heat from the suppression pool and thus protect the containment. This is performed manually in accordance with emergency procedures when the pool temperature increases because of the discharge from the RCIC turbine exhaust and/or the SRVs.

This path is capable of bringing to plant to hot shutdown.

Alternate Path Selection

The alternate path consists of the control rod drive system, ADS Division 2 (actuation and control logic), LPCI C, and RHR B in shutdown cooling mode. Suppression Pool Makeup (Train B) is not required as no LOCA is assumed for this path. Figure 11.3 shows the alternate shutdown path.

A reactor SCRAM will be initiated as described for the preferred path. The EPRI Seismic Margins methodology recommends against using the Standby Liquid Control (SLCS) as a method for reactivity control because of the added stress on the operator. Initial pressure control is also similar to the preferred path except that Division 2 of SRVs is credited for this path. Since no high pressure makeup systems are credited following the scram, the reactor inventory will be reduced as the SRVs are opened. Once reactor level reaches Level 1, ADS will actuate (as long as drywell pressure is high or the ADS bypass timer has timed out) to lower reactor pressure and LPCI C will commence injection when reactor pressure is low enough. The operator will control the depressurization process manually in accordance with emergency procedures..

By the time water level is restored and stabilized, reactor pressure will be below the setpoint for use of shutdown cooling for decay heat removal. Operator action will be required to manually open valve E12F008 as this valve is power from Division 1. This will allow the reactor to be brought to cold shutdown conditions.

Systems Used in Success Paths

Table 10.1 shows a matrix of the front-line systems used in the preferred and alternate success paths. Table 10.2 is a matrix showing front-line to support system dependencies. Table 10.3 is a matrix showing support system to support system dependencies.

Front-line Systems Descriptions

The specific GGNS systems that are available to perform the safety functions identified in Section 3.1.2.4 are shown on the success path logic diagram shown in Figure 11.1. These systems are categorized as front-line systems as they perform a direct safety function. These front line systems are described below.

<u>Control Rod Drive (CRD) System</u> - When a scram is initiated by the reactor protection system (RPS), the CRD system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water from an accumulator for each rod forces each control rod into the core.

<u>Standby Liquid Control (SLC)System</u> - The SLC system is an alternate means of inserting negative reactivity into the reactor. The SLC system serves as an emergency backup system to the control rods and is used if the CRD system fails. The system injects a sufficient quantity of neutron absorber into the reactor to ensure that the reactor is shutdown.

<u>Pressure Control System</u> - A pressure control system consisting of safety relief valves mounted on the main steam lines is provided to prevent excessive pressure inside the nuclear steam system following either normal operations, abnormal operational transients, or accidents. These safety relief valves open automatically at specific pressure settings to ensure that the reactor coolant system is not over pressurized. The valves relieve steam from the reactor vessel into the suppression pool.

<u>Reactor Core Isolation Cooling (RCIC) System</u> - The RCIC system is designed to provide core cooling in the event of a loss of feedwater flow or loss of cooling in the condenser and to cool down and depressurize the plant to the point where the shutdown cooling mode of RHR can be utilized. RCIC uses steam from the reactor to pump water into the reactor vessel to maintain coolant inventory without the use of Emergency Core Cooling Systems. RCIC is capable of providing sufficient coolant for small LOCAs also. The GGNS IPE identified a potential enhancement concerning the RCIC system. It involves a procedural change to allow operators to bypass RCIC isolation on high steam tunnel temperature due to a loss of steam tunnel cooling. As recommended in the GGNS IPE, this enhancement is still being pursued.

High Pressure Core Spray (HPCS) System - The HPCS system provides and maintains an adequate coolant inventory inside the reactor vessel to prevent fuel clad melting as a result of postulated small breaks in the reactor coolant system. The HPCS system will supply makeup water to the reactor if the RCIC system fails to operate or if a LOCA is large enough that RCIC does not have sufficient capacity to maintain inventory.

Automatic Depressurization (ADS) System - The ADS is provided to automatically depressurize the reactor coolant system so that flow from any of the low pressure ECCS can enter the reactor vessel in time to cool the core and limit fuel cladding temperature if the HPCS does not operate. The ADS system consists of eight safety relief valves with redundant actuation capability. The system also includes accumulators and air receivers that maintain ADS capability beyond 72 hours.

Residual Heat Removal (RHR) System - The RHR is a system of pumps, heat exchangers and piping that fulfills the following functions:

- Removal of decay and sensible heat from the reactor coolant system during and after shutdown.

- Removal of stored and decay heat from the reactor coolant system following a design basis loss of coolant accident (LOCA).

- Removal of heat from containment following a LOCA in order to limit the increase in containment pressure. This is accomplished by cooling and recirculating the water in the suppression pool. This function can also be accomplished while spraying the containment.

Redundant trains of RHR are utilized in the preferred and alternate paths. The assumption is that these trains have similar seismic strength . This assumption will be verified during the walkdowns.

Low Pressure Coolant Injection (LPCI) - The LPCI is an operating mode of the RHR system. LPCI uses the pump loops of the RHR system to inject cooling water at low pressure from the suppression pool into the reactor. LPCI is actuated by conditions that detect a break in the nuclear steam system, but water is delivered to the core only after the reactor vessel pressure is reduced. LPCI operation provides the capability of core reflooding following a LOCA in time to prevent fuel clad melting. There are three independent LPCI trains.

Suppression Pool Cooling (SPC) - The suppression pool cooling mode of RHR is placed in operation to limit the temperature of the water in the suppression pool following a design basis LOCA. In this mode, the RHR pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the Standby Service Water (SSW) system. The water is then returned to the suppression pool. There are two independent SPC trains.

Shutdown Cooling Mode (SDC) - The shutdown cooling mode is a function of the RHR system and is placed in operation during a normal shutdown and cooldown. Reactor cooldown is completed by pumping reactor coolant with the RHR pumps from one of the recirculation loops through the residual heat removal heat exchangers where heat is transferred to the SSW. The reactor coolant is then returned to the reactor vessel via feedwater system inlet lines. Except for a common suction line, there are two independent SDC trains.

Suppression Pool Makeup (SPMU) System - The suppression pool makeup system is designed to provide sufficient water from the upper containment pool to the suppression pool to keep the drywell vents covered following a LOCA. This is necessary because of suppression pool inventory loss through the reactor coolant system break to the drywell.

Support Systems Descriptions

In order for the front-line systems to perform their required safety functions, systems categorized as support systems must also remain operable during and after the SME. Support systems provide a support function to front-line systems and/or to other support systems. Plant specific support systems for GGNS are shown in the front-line and support systems dependency matrixes (Tables 10.2 and 10.3). The primary support systems are described below.

Standby AC Power Systems - GGNS is designed to shutdown safely and maintain a safe condition on complete loss of offsite electrical power. Standby AC power is supplied by independent and redundant diesel generators to provide power to systems and components required for safe reactor shutdown. Two identical, independent diesel generators provide power to Divisions 1 and 2 ESF power distribution system. A third independent diesel generator provides power to the Division 3 power distribution system which powers the HPCS pump and associated equipment

DC Power System - In the event that normal plant power sources become unavailable, the DC power system provides power for controls or components required for safe reactor shutdown. The ESF DC power system is divided into three divisions. Each division consists of two battery chargers (Division 3 only has one), a bank of batteries and appropriate distribution panels. In the case of a complete loss of electrical power, the batteries can continue powering the DC system for a period of time.

Standby Service Water (SSW) - The SSW system is designed to provide a reliable source of cooling for plant systems and components that are essential to safe shutdown of the plant. The SSW system supplies cooled water from the SSW cooling tower basin to various components and systems and then returns the water to the SSW cooling tower. The system consists of three different loops, each with a pump and associated valves and piping.

Various HVAC Systems - Equipment area cooling and ventilation systems are provided to maintain the local environment of specific areas at temperatures with design allowable operating ranges for electrical and mechanical components located within different areas. The systems include the Standby Service Water Pump House Ventilation system; the Diesel Generator Room Ventilation system, the ESF Switchgear Room Coolers, the Switchgear and Battery Room HVAC system; the Control Room HVAC system; and, various ECCS Room Coolers. Those systems above that require a cooling water system rely upon Plant Service Water during normal operation but switch to SSW following an accident. Many of these systems were not included in the GGNS IPE. Because of the assumption of the need to maintain the shutdown condition for a 72 hour time period, it was necessary to add these systems.

3.1.2.8 Equipment Identification and Selection

GGNS engineering developed the Safe Shutdown Equipment List (SSEL) for the IPEEE including the requirement of USI A-45 "Shutdown Decay Heat Removal Requirements" and the seismic induced fire and flooding issue for the IPEEE. Redundancy, reliability, path independence, and highly successful operational sequences are attributes of the IPEEE SSEL for the four required safe shutdown function; reactivity control, reactor coolant pressure control, reactor coolant inventory control, and decay heat removal. The additional requirements imposed upon the IPEEE success paths (consideration of SBLOCA, consideration of containment isolation and cooling requirements, and USI A-45) were factored as additional requirements to the SSEL alternatives.

The objective of this task was to determine the optimum safe shutdown alternatives for achieving a safe shutdown condition during a seismic event at the SSE level. Optimum safe shutdown alternatives are defined as safe shutdown paths that not only meet the required safety functions to bring the plant to safe shutdown, but also are: a) reliable; b) seismically rugged at the SSE levels; and , c) involve operational sequences where operator actions are likely to be highly successful.

The equipment identified for seismic evaluation included:

- Active mechanical and electrical equipment that operates or changes state to accomplish a safe shutdown function.
- Active equipment in systems that support the operation of identified safe shutdown equipment (e.g., power supplies, control systems, cooling systems, lubrication systems).
- Instrumentation needed to confirm that the four safe shutdown functions have been achieved and are being maintained.
- Instrumentation needed to operate the safe shutdown equipment.
- Tanks and heat exchangers used by systems on the SSEL.

The following equipment types were not identified for seismic evaluations:

- Equipment that could operate but does not need to operate and that, upon loss of power, will fail in the desired position or state. This type of equipment is defined as passive for the SMA.

- Passive equipment such as piping, filters, and electrical penetration assemblies. Note that although piping was not explicitly included in the SSEL, selected piping was examined during the walkdowns.

- Self-actuated check valves and manual valves.

- Major items of equipment in the nuclear steam supply system, their supports, and components mounted on or within this equipment such as the reactor pressure vessel, reactor fuel assemblies, reactor internals, control rods and their drive mechanisms, reactor coolant pumps, steam generators, pressurizers. and reactor coolant piping.

The performance of the SSEL identified 567 components which were the focus of the seismic walkdowns.

3.1.3 Seismic Walkdown

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

3.1.3.1 Approach

The key element of a reduced-scope evaluation is the plant walkdown. The approach used to perform the systems and element selection walkdown, and the seismic capability walkdown follows the recommendations of EPRI NP-6041. This includes the following parts.

- Selection of the assessment team
- Pre-walkdown preparation
- Systems and element selection for walkdown
- Seismic capacity walkdown

The assessment team, all part of the Seismic Review Team (SRT), was made up of eight members. Six of the members were Entergy Operations Personnel, three of those possessing the qualification requirements of EPRI NP-6041. The remaining two members were outside consultants, also possessing the qualification requirements of EPRI NP-6041. The following is a listing of the SRT members, their affiliation and area of expertise.

Mr. Kyle Grillis	Entergy Operations	Operations
Mr. Mark D. Locke	Entergy Operations	Seismic Capability Engineer

Mr. Joseph D. Malara	Entergy Operations	Seismic Capability Engineer
Mr. James Owens	Entergy Operations	Licensing and Operations
Mr. Amir Shahkarami	Entergy Operations	Seismic Capability Engineer
Dr. John D. Stevenson	Stevenson and Associates	Seismic Capability Engineer
Mr. Mike Sweeney	Entergy Operations	Operations
Mr. George G. Thomas	Stevenson and Associates	Seismic Capability Engineer

Prior to the walkdown, a detailed product plan was prepared, including the detailed technical approach of each task and the interfaces between the team members and GGNS personnel. In preparation for the walkdown, data assembly and evaluations were performed to define a technical baseline for the systems analysis and seismic screening walkdown. Two independent teams were used during the walkdown process.

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

- GGNS Safe Shutdown Paths and Equipment Lists
- plant arrangement drawings
- sections of the GGNS UFSAR relating to the seismic criteria and licensing basis for the plant
- ground response spectra for the SSE
- floor response spectra and how they were generated
- a sample of construction details of the anchorage including drawings and specifications
- a sample of procurement and seismic testing specifications for equipment
- examples of calculations for seismic and anchorage qualifications
- design basis documents for the GGNS structures
- selected evaluations for block walls
- design calculations for a sample of large flat bottom tanks

The structures at GGNS were screened generically. The drawings and analysis models were reviewed for details that might indicate seismic vulnerabilities in accordance with the requirements of a reduced scope SMA. The drawing and structural analysis reviews confirmed that consistent good practice in design detail and analysis was implemented at GGNS. Therefore, it was not necessary to review more than a small sample of the details of connections, reinforcement bar placement, construction joints, etc., to make the judgments on screening.

3.1.3.2 Screening Walkdown

For the SMA at GGNS, rigorous statistically based sampling criteria were neither practical nor desirable. The SMA procedures and guidelines used were heavily reliant on the judgment of the highly experienced engineer and criteria for sampling in this plant likewise are modeled around this judgment.

There are two areas of a reduced scope SMA where sampling was applicable and used at GGNS. They are; 1) screening of structures and components, and 2) walkdowns. Issues that influence the sampling are; redundancy provided by multi-train systems, similarity in design and location of redundant trains, treatment of single failures, access to components during walkdowns, and systems interactions potential including fire and internal flood sources.

The sampling approach described below are appropriate for modern plants of GGNS's vintage. The document review and walkdown verified that uniform practices in accordance with the plant design basis for construction, design and installation were implemented. Therefore, the sampling approach was used throughout the effort.

During the GGNS walkdown, it was confirmed as expected that most items in a given equipment class were either identical or very similar. The plant documentation review and walkdown confirmed that the vast majority of equipment was manufactured, and installed as specified. The screening procedures used at GGNS for generic categories of equipment and structures contained caveats or inclusion rules that were checked during the walkdown. Since the equipment at GGNS was purchased and installed to similar codes and standards the SRT screened generic classes of equipment on the basis of their relative ruggedness. The screening sampling size for identical or very similar equipment in a given class for caveats was one or greater for each walkdown team. The screening size for very similar equipment in a given class with identical or very similar anchorage was two or greater for each walkdown team. The increased sample for anchorage is based on experience at other plants that anchorage installations are not always consistent. This is consistent with the guidance given in Appendix D of EPRI NP 6041-S[illegible]. A 100 percent "walk-by" of all equipment on the SSEL was employed to check for unique equipment details and for seismic interactions.

Distribution systems that were installed in bulk such as piping, cable trays, HVAC ducting, electrical conduit and instrument lines were screened generically after completion of a walkdown with verification that the distribution systems meet the inclusion rules. It was confirmed that the design and installation practice at GGNS are consistent, therefore the screening judgment was based upon a review of the general specifications and drawing for a single run of each generic class of distribution system, i.e., a sample size of one per generic class. As expected the review of the general specifications and drawings did not indicate significant differences in design and installations practice.

3.1.3.3 Walkdowns

Two walkdowns were performed. An off-line outage walkdown was performed during the fall of 1993 (October 12, 1993) that included equipment located in the drywell and on on-line walkdown was performed during the summer of 1993 (August 26-31, 1993). The structures and distribution system review was performed during the course of the on-line walkdown.

A typical day during the walkdown consisted of:

- reviewing issues identified on previous days for determination as to whether the item is screened or an outlier
- planning the day's walkdown effort
- performing the walkdown
- briefing GGNS Design Engineering on the day's progress

3.1.3.4 Seismic Analysis Results

The seismic walkdowns found GGNS is seismically rugged and that there were no outliers affecting plant operability. However, there were several equipment items that could not be initially screened. Some of these items are candidates for voluntary design enhancements by Entergy, and some of these equipment items are potential outliers. Final disposition of these items have been documented and completed.

A complete listing of the items not initially screened during the walkdowns and the resolutions is included in the IPEEE Reduced Scope Seismic Margins Assessment (SMA).[3]

3.2 Conclusions of Seismic Analysis

The conclusions of the seismic analysis is that Grand Gulf Nuclear Station is seismically rugged and that all components identified in the Safe Shutdown Path have adequately considered the seismic input. All anchorage to these components was found to be rugged. Only one potential vulnerability to a seismic event was identified, which has been corrected.

INTERNAL FIRES ANALYSIS

4.0 METHODOLOGY SELECTION

The fire PRA methodology used in this study employs NUREG/CR-2300 for the basic framework; however, more recent data sources and technical bases are used. For example, EPRI TR-100370 (FIVE) is one of the more recent fire sources used extensively as a resource. This method draws upon not only FIVE, but data sources in NUREG/CR-4840 and NUREG/CR-2815, and insights from both Sandia and EPRI fire research programs. These data and bases allow the removal of many of the conservatisms and non-conservatisms that characterize earlier fire PRAs. The key assumptions and specifics of this approach are described in the EPRI Fire Risk Implementation Guide (Reference 27). This document will be referred to as the Implementation Guide throughout this document.

This implementation of NUREG/CR-2300, henceforth knows as the EPRI Fire PRA methodology, evolves in four technical tasks. These tasks follow the progression of the fire accident from fire initiation to core damage and challenge to the containment integrity.

The four tasks of this methodology are:

1. Develop fire-induced sequences;
2. Develop fire scenarios;
3. Evaluate fire damage sequences and their uncertainties; and
4. Document and verify the analysis.

Figure 4-1 shows an overview of the methodology to develop the model and evaluate fire risk. The following describes the methodology and lists key assumptions.

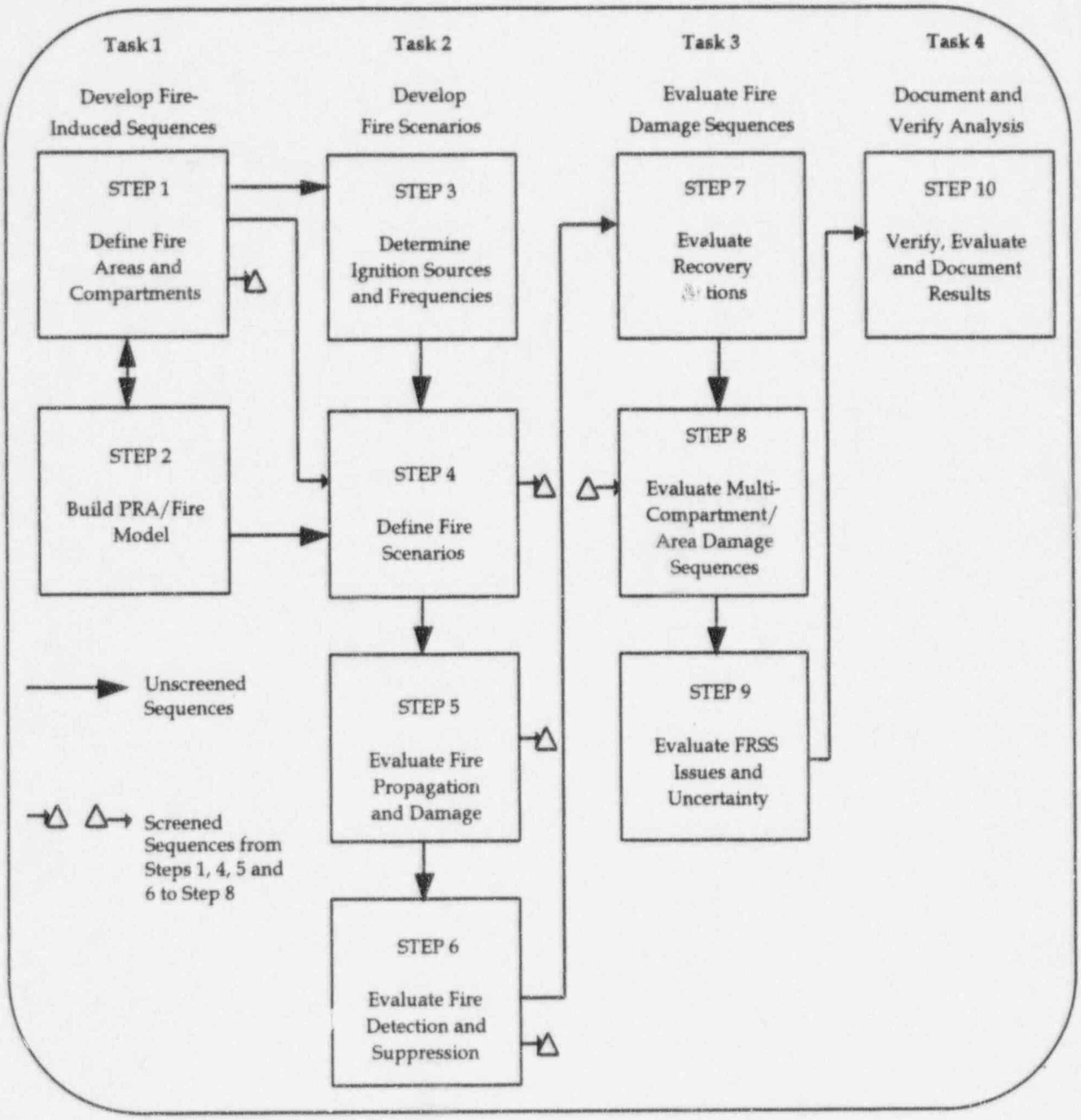


Figure 4-1 Overview of the Fire PRA Approach

<u>Task 1 - Develop Fire-Induced Sequences</u>

The methodology begins by developing a PRA/Fire model containing fire-induced sequences. These fire-induced sequences begin with fire-induced failure (or failures) in an area or compartment followed by a combination of equipment and human failures, both random and fire-induced, which lead to core damage.

This is done in two steps. First, critical areas and compartments are defined consistent with FIVE. As a result, a number of fire zones are combined into single compartments and hot gases are assumed to affect components throughout the compartment unless otherwise evaluated.

Then, the internal events PRA model is modified to incorporate fire-induced initiators and appropriate equipment failure modes. That is, both fire-induced and random failures are combined and treated to the same level of detail as the IPE. (Note however that some components are assumed to be failed by the fire because all their associated cable locations were not readily known and determining these was believed to be of little benefit compared to the level of effort required to retrieve the information). All human actions were evaluated for fire situations, including those actions specific to the compartment. Guidelines for incorporating human actions are described in Appendix B of the Implementation Guide. Also, potential dependencies resulting from the dual role of fire water for suppression and vessel injection were explicitly evaluated.

The resulting PRA/Fire model is linked to the plant equipment location database. An equipment location database is created that:

1. Documents readily available and verifiable location data for equipment modeled in the IPE (and its required cable), and
2. Identifies the type of accident initiators which could result from a fire in a specified plant location.

The key assumptions for this task are:

- Compartment boundaries need not be rated to be barriers to the spread of hot gases.
- Certain specified components whose locations were not determined were assumed failed by any fire.
- Plant trip initiators were assumed to occur in each fire area.

<u>Task 2 - Develop Fire Scenarios</u>

In the second task, the methodology continues by developing fire scenarios for each compartment in a four-step process (Steps 3 through 6 of the methodology). A fire scenario begins with an ignition source and ends with likelihood of equipment damage.

Step 3 of the methodology develops the ignition source model. The likelihood of ignition depends on the type of equipment and activities in the compartment. Fire ignition frequencies are developed based on the EPRI Fire Events Database (NSAC-178L) method also used in FIVE. A review of plant-specific data indicated no trends or anomalies requiring this generic data to be updated.

Step 4 defines the fire scenarios by identifying the combinations of equipment in the compartment that are susceptible to fire damage. Initially, it is assumed that each fire grows sufficiently to fail all equipment in the compartment. The core damage probability calculated based on these assumptions is multiplied by the compartment ignition frequency. If this product is less than 1E-6/yr, the compartment is screened. (The only further analysis performed would be to verify that no fire-induced containment bypass of greater than 1E-7/yr is possible.)

Step 5 evaluates fire propagation and damage. Some fire sources are small enough that they self-extinguish. Larger fire sources may damage equipment in the compartment. If the equipment is combustible or other combustibles are present, the fire may even propagate. Ignition of a fire source and other combustibles heat the compartment (the rate of heating depends on the ignition source and combustibles) and the equipment in it, potentially leading to loss of equipment and functions. The fire model is used to determine the likelihood and time of target or target set(s) failure. This step includes the uncertain phenomena of how equipment is damaged and at what environmental severity level. Unless the fire is detected and suppressed before the damage time, equipment will fail.

Fire modeling is performed primarily using the FIVE hazard worksheets. Key data such as heat release rates and heat loss factors are based on test data. Interpretations of tests are used to model complex situations, such as propagation of fires in cable tray stacks and between cubicles in electrical cabinets.

Fire sizes and durations were selected based primarily on test data from SNL, NIST, etc. The Implementation Guide documents how a HRR (heat release rate) was selected for each ignition source identified using the EPRI method.

Damage criteria for cable was assumed to be 700°F. In general, damage criteria for other components was not significant. (The control room analysis did assume that solid-state circuits in adjacent cabinets could be damaged at 150°F.) There appeared to be little sensitivity to the 700°F criteria because in few, if any, incidences were there any close calls (i.e., if the temperature was near 700°F, e.g., above 600°F, it was often much higher than 700°F, e.g., above 800°F).

Step 6 of the methodology models the response of the fire suppression system to the fire ignition and propagation. This last step in developing fire scenarios considers both automatic or manual means, both of which fire experience indicates are usually quite effective. Automatic systems, fire brigade members and continuous fire watches are considered for their effectiveness of suppression. Automatic suppression probabilities are based on NSAC-179L for automatic system initiation and manual operation. Potential dependencies in automatic systems were evaluated per the guidance contained in Appendix L of the Implementation Guide.
If the fire brigade drill data indicates brigade members would be present when needed, manual suppression is credited. Manual suppression probabilities are based on the EPRI fire events database. The principal data used are the fire events involving cable tray fires, largely because the dominant scenarios involving suppression are those that produce hot gas layers that affect redundant trains. In practice, due to analysis simplifications, automatic and manual suppression by the fire brigade were generally credited only for more slowly-developing scenarios (e.g., hot gas layer development affecting separated equipment).

For welding scenarios, a continuous fire watch was credited with suppression of a fire before a significant heat release rate was developed. The probability of suppression for these fires and cable tray fires was based on data compiled in Appendix Q of the Implementation Guide.

The key assumptions for this task are:

- Unless precluded by procedure, transient ignition source frequencies were assumed the same for each compartment, combustibles and HRRs were selected based on the specific characteristics of the compartment.
- The damaging effects of a fire were assumed to affect all components in a compartment unless detailed fire modeling was done to demonstrate otherwise.
- Heat lost to the walls and ceiling of the compartment for a full compartment scenario (hot gas layer) was assumed to be 94% based on test data.
- One-hour and Nominal one-hour wraps were conservatively assumed to provide 60 and 45 minutes protection, respectively. The former is based on Grand Gulf's current plans to upgrade Thermo-Lag barriers and the latter is based on test data referenced in Grand Gulf's fire protection program.
- The damaging effects of fires were assumed to be limited to hot gases. That is, smoke effects were assumed to be long term in nature.
- Each fire was assumed to reach a fully developed stage and to do so prior to detection. The only exception is the case of transient combustibles ignited by welding in the presence of a continuous fire watch.
- No credit was given to human detection except when a continuous fire watch is required.
- Manual fire suppression was not credited unless at least one fire brigade member could reach the compartment in time.
- Suppression prior to loss of a cabinet's function was not credited. This assumption was particularly important to the control room.
- The loss of a control room cabinet containing divisional equipment was assumed to affect the entire division.
- Evacuation of the control room was assumed to occur at the time smoke visibly obscured the panel.
- The time for evacuation assumed a similar response as indicated by the SNL cabinet fire tests.
- Human detection of control room cabinet fires was not credited except for the manned main control panel and then it was assumed to be only as effective as a smoke detector.

Task 3 - Evaluate Fire Damage Sequences and Their Uncertainties

The third technical task evaluates the fire damage sequence in a three-step process (steps 7 through 9 of the methodology). A fire damage sequence is a fire scenario from Task 2 that leads to one of the fire-induced sequences identified in Task 1. Step 7 of the methodology addresses any potential operator recovery actions available to mitigate damage from the fire. Operators may face a variety of conditions as a result of a fire damage sequence. The plant may trip automatically due to equipment damage; or the operators may trip the plant as a preventive measure. Equipment already damaged by the fire will not respond. Other equipment may fail independent of fire, more equipment may be damaged by the fire, or operators may fail to recover certain plant systems. The fire damage sequence may continue and possibly lead to core damage. This step analyzes the operator's ability to recover other systems even if a safe shutdown system damage occurs. If containment isolation systems are affected, the impact on containment integrity is documented.

Step 8 evaluates multi-compartment scenarios. Multi-compartment scenarios address the effects on plant safety should a fire propagate beyond a single compartment. If the fire is severe enough and if fire protection systems or personnel fail, the fire or its products of combustion may reach another compartment. While this is unlikely, the potentially high consequences of damaging additional shutdown equipment may yield notable risk.

Step 9 considers uncertainties and special issues. Besides the traditional fire analysis, a special set of issues that have been raised by NRC's Research Branch are evaluated. Those issues, called the Fire Risk Scoping Study (FRSS) issues, are analyzed using an approach developed for FIVE.

The key assumptions for this task are:

- Human actions were, if demonstrated to be possible under fire conditions, generally quantified using screening values from the IPE.
- Fire spread to more than one additional compartment was assumed to be probabilistically insignificant.
- Consistent with NUREG/CR-4840, a barrier failure probability of 0.1 was assumed unless an evaluation was performed of the numbers and types of barriers.

Task 4 - Document and Verify the Analysis

The fourth and final technical task (step 10 of the methodology) documents and evaluates the results and verifies key supporting information. This two-step process produces the final pieces of information necessary for developing an IPEEE fire submittal to satisfy Generic Letter 88-20, Supplement 4 and NUREG-1407.

4.1 Fire Hazard Analysis

The GGNS Fire Hazard Analysis (FHA) and supporting documentation forms the core of information utilized for the Fire PRA. Supporting documentation includes location information for cables and components. The FHA assumes a loss of offsite power when ensuring that a safe shutdown path will be available. This assumption means that all of the equipment credited in the

FHA is fed electrically off an emergency (Class 1E) bus that is diesel generator backed. However, not all equipment powered by emergency buses is credited. Therefore, the FHA/supporting documentation information was supplemented with location information for certain systems that could be useful in the mitigation of possible core damage events.

With some minor exceptions the fire zones defined in the FHA were used in the Fire PRA. Some fire zones were combined into single compartments.

4.2 Review of Plant Information and Walkdown

GGNS complies with 10CFR50 Appendix R as described in UFSAR Table 9.5-12 (Fire Protection Program Comparison with Appendix R to 10CFR50). Deviations to these requirements have been evaluated for acceptability and the NRC has documented concurrence in GGNS Safety Evaluation Reports.

4.2.1 Fire Compartment Information

A total of 184 fire compartments were defined based on the guidance provided in the Implementation Guide. Compartments were defined for areas of the plant where damage to equipment located therein could affect the safe shutdown capability of the plant or cause a reactor trip. A compartment is a well defined, enclosed room, not necessarily having fire barriers. However, compartments were generally established along fire barriers. Compartments may include a single Appendix R fire zone or multiple Appendix R fire zones as established in the GGNS Fire Hazards Analysis (Reference 28). Details of the compartment definition are documented in Engineering Report No. GGNS-94-0018 (Reference 29). A listing the compartments is found in Table 4.2-1.

In general, ignition source frequencies and conditional core damage probabilities (CCDPs) were developed for each compartment. Initially a conservative core damage probability was determined based on the assumption that all equipment in the compartment would be failed as a result of a fire. Compartments with core damage probabilities <1.0E-06 were screened from further consideration. The CCDPs for the remaining compartments were further refined by the progressive removal of conservatisms or refinement of the fire scenario. Development of ignition source frequencies is described in Section 4.2.2. Development of CCDPs is described in Section 4.6. The screening process is discussed and results tabulated in Section 4.6.

4.2.2 Ignition Source Frequencies Overview

The general methodology for the preparation of ignition source frequencies is described in the Implementation Guide (Section 4.2 and Appendix D). The method calls for apportioning generic fire frequencies to individual compartments based on the numbers and types of ignition sources in the compartment.

Ignition sources were counted for each compartment based on a report generated from the plant SIMS database. The SIMS report included location information as well as equipment tag numbers and descriptions. The SIMS report was sorted according to equipment categories which were then mapped to the ignition sources identified in the methodology document. An initial count for each compartment developed from the SIMS report, then cross-checked against equipment location drawings. Finally, the ignition source counts were checked for a representative number of compartments by walkdown. Details of the ignition frequency preparation for this study are documented in Engineering Report No. GGNS-94-0018 (Reference 29). Ignition source frequencies used in the screening analysis are reported in Table 4.2-2.

4.2.3 Walkdowns

Walkdowns were performed by GGNS Fire Protection personnel at various stages of the evaluation. The most detailed walkdowns occurred during the data gathering and verification of the detailed fire scenarios. These walkdowns involved field measurements of ignition source dimensions and distance to targets.

The checklist in Attachment E of the Implementation Guide was used in the GGNS walkdowns. The checklist was used primarily only in those compartments that did not screen in the early phases of the analysis.

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
1	CA101	1	1A101, 1A114, 1A117, 1A120	RB
2	CA102	2	1A102, 1A202, 1A303, 1A442	RB
3	CA103	2	1A103, 1A203	RB
4	CA104	2	1A104, 1A204	RB
5	CA105	2	1A105, 1A205	RB
6	CA106	2	1A106, 1A206, 1A307, 1A441	RB
7	CA107	1	1A107	RB
8	CA108	1	1A108	RB
9	CA109	5	1A109	RB
10	CA110	25	1A110A, 1A110B, 1A110C1, 1A110C2, 1A110C3, 1A110D1, 1A110D2, 1A110D3, 1A110E1, 1A110E2, 1A110F1, 1A110F2, 1A110F3, 1A110F4, 1A311, 1A313, 1A411, 1A509, 1A510, 1A513, 1A520, 1A601	CONTAIN (NOTE 2)
11	CA111	1	1A111, 1A127	RB
12	CA112	25	1A112, 1A113	CONTAIN (NOTE 2)
13	CA115	3	1A115, 1A116, 1A118, 1A119, 1A220	RB
14	CA124	2	1A124	RB
15	CA125	2	1A125	RB
16	CA130	1	1A130, 1A131	RB
17	CA132	2	1A132, 1A224, 1A226, 1A305, 1A439, 1A440	RB
18	CA201	6	1A201, 1A211, 1A215, 1A222	RB
19	CA207	7	1A207	SWGR
20	CA208	8	1A208	SWGR
21	CA209	2	1A209	RB
22	CA210	2	1A210	RB
23	CA219	9	1A219	SWGR
24	CA221	10	1A221	SWGR
25	CA225	2	1A225	RB
26	CA301	11	1A301, 1A302, 1A314, 1A316, 1A321, 1A322, 1A324	RB
27	CA304	2	1A304	RB
28	CA306	2	1A306	RB
29	CA308	12	1A308	RB

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
30	CA309	13	1A309	RB
31	CA310	25	1A310	CONTAIN (NOTE 2)
32	CA318	14	1A318	RB
33	CA319	15	1A319	RB
34	CA320	16	1A320	RB
35	CA323	11	1A323	RB
36	CA325	17	1A325	RB
37	CA326	11	1A326	RB
38	CA401	19	1A401, 1A403, 1A417, 1A420, 1A424, 1A427, 1A428,1A434	RB
39	CA402	18	1A402	RB
40	CA404	18	1A404	RB
41	CA405	18	1A405	RB
42	CA406	18	1A406	RB
43	CA407	20	1A407	SWGR
44	CA410	21	1A410	SWGR
45	CA414	25	1A414, 1A507	CONTAIN (NOTE 2)
46	CA419	25	1A419, 1A421	CONTAIN (NOTE 2)
47	CA429	19	1A429	RB
48	CA430	19	1A430	RB
49	CA431	19	1A431, 1A437, 1A438, 1A444, 1A525, 1A528, 1A532, 1A602, 1A603, 1A604, 1A606, 1A607	RB
50	CA432	19	1A432	RB
51	CA433	19	1A433	RB
52	CA436	19	1A436	RB
53	CA443	25	1A443	CONTAIN (NOTE 2)
54	CA506	2	1A506, 1A508, 1A605	RB
55	CA514	25	1A514	CONTAIN (NOTE 2)
56	CA515	25	1A515	CONTAIN (NOTE 2)
57	CA516	25	1A516	CONTAIN (NOTE 2)
58	CA517	25	1A517	CONTAIN (NOTE 2)
59	CA519	19	1A519, 1A523, 1A524, 1A527, 1A531	RB
60	CA529	19	1A529	RB
61	CA530	19	1A530	RB
62	CA533	19	1A533	RB
63	CA534	19	1A534	RB

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
64	CA536	19	1A536	RB
65	CA537	19	1A537	RB
66	CA539	22	1A539	RB
67	CC101	26	0C101, OC103, OC115, OC117, OC217	RB (NOTE 1)
68	CC104	28	0C104, OC109, OC116	RB (NOTE 1)
69	CC125	26	0C125	RB (NOTE 1)
70	CC126	26	0C126	RB (NOTE 1)
71	CC128	27	0C128	RB (NOTE 1)
72	CC202	31	0C202	SWGR
73	CC203	30	0C203	RB (NOTE 3)
74	CC204	30	0C204	RB (NOTE 4)
75	CC205	30	0C205	RB (NOTE 1)
76	CC205A	30	0C205A	RB (NOTE 1)
77	CC206	30	0C206	RB (NOTE 4)
78	CC207	32	0C207	BATTR
79	CC208	33	0C208	RB (NOTE 1)
80	CC208A	34	0C208A	RB (NOTE 1)
81	CC209	35	0C209	BATTR
82	CC210	36	0C210	SWGR
83	CC211	37	0C211	BATTR
84	CC212	30	0C212	RB (NOTE 4)
85	CC213	30	0C213	RB (NOTE 3)
86	CC214	30	0C214	RB (NOTE 3)
87	CC215	38	0C215	SWGR
88	CC216	39	0C216	RB (NOTE 1)
89	CC218	40	0C218	RB (NOTE 1)
90	CC219	30	0C219	RB (NOTE 1)
91	CC301	42	0C301	RB (NOTE 1)
92	CC302	42	0C302	RB (NOTE 1)
93	CC303	42	0C303	RB (NOTE 1)
94	CC304	42	0C304, OC412, OC612	RB (NOTE 1)
95	CC305	42	0C305	RB (NOTE 1)
96	CC306	47	0C306, OC409, OC610, OC709	RB (NOTE 1)
97	CC307	41	0C307	RB (NOTE 1)
98	CC308	42	0C308	RB (NOTE 1)
99	CC309	42	0C309	RB (NOTE 1)
100	CC401	42	0C401	RB (NOTE 1)

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
101	CC402	42	0C402	CSR
102	CC402A	50	0C402A, OC512B	RB (NOTE 1)
103	CC403	45	0C403	RB (NOTE 1)
104	CC404	42	0C404	RB (NOTE 1)
105	CC405	42	0C405	RB (NOTE 1)
106	CC405A	50	0C405A, OC507A	RB (NOTE 1)
107	CC406	46	0C406	RB (NOTE 1)
108	CC406A	42	0C406A, OC518A, OC613A	RB (NOTE 1)
109	CC407	43	0C407	RB (NOTE 1)
110	CC408	42	0C408	RB (NOTE 1)
111	CC409A	47	0C409A, OC512A, OC608B	RB (NOTE 1)
112	CC410	44	0C410	BATTR
113	CC411	42	0C411	RB (NOTE 1)
114	CC412A	42	0C412A, OC507C, OC603B	RB (NOTE 1)
115	CC501	50	0C501	RB (NOTE 1)
116	CC502	50	0C502, OC503, OC504	CR
117	CC507	50	0C507	RB (NOTE 1)
118	CC509	50	0C509, OC511, OC512	RB (NOTE 1)
119	CC510	50	0C510	RB (NOTE 1)
120	CC513	50	0C513	RB (NOTE 1)
121	CC514	50	0C514	RB (NOTE 1)
122	CC515	50	0C515	RB (NOTE 1)
123	CC518	48	0C518, OC611	RB (NOTE 1)
124	CC601	50	0C601, OC602	RB (NOTE 1)
125	CC603	50	0C603	RB (NOTE 1)
126	CC604	51	0C604	RB (NOTE 1)
127	CC606	50	0C606	RB (NOTE 1)
128	CC608	50	0C608	RB (NOTE 1)
129	CC609	52	0C609	RB (NOTE 1)
130	CC613	50	0C613	RB (NOTE 1)
131	CC614	50	0C614	RB (NOTE 1)
132	CC615	50	0C615	RB (NOTE 1)
133	CC616	50	0C616	RB (NOTE 1)
134	CC617	50	0C617	RB (NOTE 1)
135	CC618	50	0C618	RB (NOTE 1)
136	CC619	50	0C619	RB (NOTE 1)
137	CC701	54	0C701	RB (NOTE 1)
138	CC702	47	0C702	CSR

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
139	CC703	53	0C703	RB (NOTE 1)
140	CC704	58	0C704	RB (NOTE 1)
141	CC705	58	0C705	RB (NOTE 1)
142	CC706	58	0C706	RB (NOTE 1)
143	CC707	55	0C707	RB (NOTE 1)
144	CC708	56	0C708, OC710	RB (NOTE 1)
145	CC708A	58	0C708A	RB (NOTE 1)
146	CC711	54	0C711	RB (NOTE 1)
147	CC712	47	0C712	RB (NOTE 1)
148	CC713	58	0C713	RB (NOTE 1)
149	CD301	60	1D301	RB (NOTE 1)
150	CD306	63	1D306	DGR
151	CD308	62	1D308	DGR
152	CD310	61	1D310	DGR
153	CM100	64	BASIN NO. 1	RB (NOTE 1)
154	CM110	64	1M110	RB (NOTE 1)
155	CM112	64	1M112	RB (NOTE 1)
156	CM200	65	BASIN NO. 2	RB (NOTE 1)
157	CM210	65	2M110	RB (NOTE 1)
158	CM212	65	2M112	RB (NOTE 1)
159	CT100		93'-0" ELEVATION , TURBINE BLDG.	TB
160	CT200		113'-0" ELEVATION , TURBINE BLDG.	TB
161	CT212		1T212	BATTR
162	CT219		1T219	SWGR
163	CT300		133'-0" ELEVATION, TURBINE BLDG.	TB
164	CT312		1T312	BATTR
165	CT323		1T323	SWGR
166	CT400		166'-0" ELEVATION , TURBINE BLDG. + 1T502, 1T503	TB
167	CT405		1T405	BATTR
168	CT406		1T406	BATTR
169	CM101		OM101 (CIRC. WATER PUMPHOUSE)	RB (NOTE 1)
170	CM102		OM102 (MTR. DRIVEN FIRE PUMP ROOM)	RB (NOTE 1)
171	CM115		OM115 (ALL WTR. TREATMENT	RB (NOTE 1)

Table 4.2-1 Fire Compartment Definition

No.	Compartment	Fire Area	Fire Zones Within Compartment	Location
			BLDG.)	
172	CRAD		RADWASTE BLDG.	RADWASTE
173	SWHYD		YARD (INCLUDING 11 AND 21 MAIN TRANSFORMERS)	TRANS YARD
174	CTR11		TRANSFORMERS BOP11A, BOP 11B	TRANS YARD
175	CTR12		TRANSFORMERS BOP12A, BOP12B	TRANS YARD
176	CTR14		TRANSFORMERS BOP14, BOP24	TRANS YARD
177	CTRESF11		TRANSFORMER ESF11	TRANS YARD
178	CTRESF12		TRANSFORMER ESF12	TRANS YARD
179	CTRESF21		TRANSFORMER ESF21	TRANS YARD
180	CTRMAIN		TRANSFORMERS MAIN 1A, 1B, 1C, 1D	TRANS YARD
181	CDUCT1		DIVISION 1 DUCT BANK TO SSW COOLING TOWER	RB (NOTE 1)
182	CDUCT2		DIVISION 2 DUCT BANK TO SSW COOLING TOWER	RB (NOTE 1)
183	CDUCT3		DIVISION 3 DUCT BANK TO SSW COOLING TOWER	RB (NOTE 1)
184	CYARD		BALANCE OF YARD AREA	RB (NOTE 1)

Notes for Table 4.2-1

Note 1: Compartments assigned to Reactor Building per Reference 27, Appendix D, Page 4.

Note 2: Containment is not considered in this analysis. (Reference 8.1, Appendix D, Page 4).

Note 3: While identified as switchgear rooms (originally designed for Unit 2), these compartments are assigned to the Reactor Building based on their actual occupancy (see Note 1).

Note 4: While identified as battery rooms (originally designed for Unit 2), these compartments are assigned to the Reactor Building based on their actual occupancy (see Note 1). No batteries are contained in these rooms.

Note 5: Fire zone designations and room numbers are found on drawings A-0630 through 650.

LEGEND:

- BATTR = Battery Room
- CONTAIN = Containment/Drywell
- CR = Control Room
- CSR = Cable Spreading Room
- DGR = Diesel Generator Room
- RADWASTE = Radwaste Building
- RB = Reactor Building
- SWGR = Switchgear Room
- TB = Turbine Building
- TRANS YARD = Transformer Yard

Table 4.2-2 Compartment Ignition Frequencies

	Compartment	Ignition Source Frequency
1	CA101	5.3E-03
2	CA102	2.5E-04
3	CA103	1.2E-03
4	CA104	1.8E-03
5	CA105	1.2E-03
6	CA106	2.5E-04
7	CA107	7.0E-04
8	CA108	7.0E-04
9	CA109	1.2E-03
10	CA110	n/a
11	CA111	3.6E-04
12	CA112	n/a
13	CA115	2.7E-03
14	CA124	2.5E-04
15	CA125	2.5E-04
16	CA130	2.5E-04
17	CA132	3.0E-04
18	CA201	6.9E-03
19	CA207	1.7E-03
20	CA208	1.7E-03
21	CA209	4.8E-04
22	CA210	4.8E-04
23	CA219	1.8E-03
24	CA221	1.8E-03
25	CA225	2.5E-04
26	CA301	1.0E-02
27	CA304	2.5E-04
28	CA306	2.5E-04
29	CA308	7.2E-04
30	CA309	1.7E-03
31	CA310	n/a
32	CA318	2.5E-03
33	CA319	3.0E-04
34	CA320	2.0E-03
35	CA323	3.0E-04
36	CA325	2.5E-04
37	CA326	3.7E-04

Table 4.2-2 Compartment Ignition Frequencies

	Compartment	Ignition Source Frequency
38	CA401	8.2E-03
39	CA402	2.5E-04
40	CA404	2.5E-04
41	CA405	1.0E-03
42	CA406	3.8E-04
43	CA407	1.7E-03
44	CA410	1.7E-03
45	CA414	n/a
46	CA419	n/a
47	CA429	3.6E-04
48	CA430	9.3E-04
49	CA431	4.5E-03
50	CA432	8.1E-04
51	CA433	4.8E-04
52	CA436	2.5E-04
53	CA443	n/a
54	CA506	2.5E-04
55	CA514	n/a
56	CA515	n/a
57	CA516	n/a
58	CA517	n/a
59	CA519	7.7E-04
60	CA529	7.0E-04
61	CA530	7.0E-04
62	CA533	2.5E-04
63	CA534	2.5E-04
64	CA536	2.5E-04
65	CA537	2.5E-04
66	CA539	2.5E-04
67	CC101	5.4E-04
68	CC104	1.1E-03
69	CC125	4.0E-04
70	CC126	7.0E-04
71	CC128	4.7E-04
72	CC202	2.3E-03
73	CC203	3.5E-04
74	CC204	4.6E-04
75	CC205	2.5E-04

Table 4.2-2 Compartment Ignition Frequencies

	Compartment	Ignition Source Frequency
76	CC205A	2.5E-04
77	CC206	2.5E-04
78	CC207	6.5E-04
79	CC208	4.0E-04
80	CC208A	2.5E-04
81	CC209	6.5E-04
82	CC210	2.3E-03
83	CC211	6.5E-04
84	CC212	2.5E-04
85	CC213	5.6E-04
86	CC214	3.5E-04
87	CC215	2.4E-03
88	CC216	2.5E-04
89	CC218	2.5E-04
90	CC219	2.5E-04
91	CC301	2.5E-04
92	CC302	9.3E-04
93	CC303	1.5E-03
94	CC304	2.5E-04
95	CC305	2.5E-04
96	CC306	3.5E-04
97	CC307	2.5E-04
98	CC308	2.5E-04
99	CC309	2.5E-04
100	CC401	2.5E-04
101	CC402	1.8E-03
102	CC402A	2.5E-04
103	CC403	6.0E-03
104	CC404	2.5E-04
105	CC405	2.5E-04
106	CC405A	2.5E-04
107	CC406	2.5E-04
108	CC406A	2.5E-04
109	CC407	4.6E-03
110	CC408	2.5E-04
111	CC409A	2.5E-04
112	CC410	6.5E-04
113	CC411	2.5E-04

Table 4.2-2 Compartment Ignition Frequencies

	Compartment	Ignition Source Frequency
114	CC412A	2.5E-04
115	CC501	2.5E-04
116	CC502	1.0E-02
117	CC507	2.5E-04
118	CC509	2.5E-04
119	CC510	2.5E-04
120	CC513	2.5E-04
121	CC514	2.5E-04
122	CC515	2.5E-04
123	CC518	2.5E-04
124	CC601	2.5E-04
125	CC603	3.8E-04
126	CC604	3.3E-04
127	CC606	2.5E-04
128	CC608	3.5E-04
129	CC609	2.5E-04
130	CC613	2.5E-04
131	CC614	3.1E-04
132	CC615	2.5E-04
133	CC616	2.5E-04
134	CC617	2.5E-04
135	CC618	2.5E-04
136	CC619	2.5E-04
137	CC701	2.5E-04
138	CC702	3.5E-04
139	CC703	3.9E-03
140	CC704	7.2E-04
141	CC705	3.0E-04
142	CC706	2.5E-04
143	CC707	3.7E-03
144	CC708	1.9E-03
145	CC708A	2.5E-04
146	CC711	2.5E-04
147	CC712	2.5E-04
148	CC713	2.5E-04
149	CD301	4.0E-04
150	CD306	2.9E-02
151	CD308	2.9E-02

Table 4.2-2 Compartment Ignition Frequencies

	Compartment	Ignition Source Frequency
152	CD310	2.9E-02
153	CM100	2.5E-04
154	CM110	1.7E-03
155	CM112	3.0E-04
156	CM200	2.5E-04
157	CM210	1.3E-03
158	CM212	3.0E-04
159	CT100	9.5E-02
160	CT200	1.0E-02
161	CT212	6.5E-04
162	CT219	2.0E-03
163	CT300	3.0E-02
164	CT312	6.5E-04
165	CT323	2.1E-03
166	CT400	9.4E-03
167	CT405	6.5E-04
168	CT406	6.5E-04
169	CM101	2.2E-03
170	CM102	3.4E-03
171	CM115	1.4E-02
172	CRAD	1.9E-02
173	SWHYD	7.9E-03
174	CTR11	5.6E-03
175	CTR12	5.6E-03
176	CTR14	1.6E-03
177	CTRESF11	8.7E-04
178	CTRESF12	8.7E-04
179	CTRESF21	8.7E-04
180	CTRMAIN	2.9E-03
181	CDUCT1	2.6E-04
182	CDUCT2	2.6E-04
183	CDUCT3	4.8E-04
184	CYARD	1.9E-04

4.3 Fire Growth and Propagation

Fire growth and propagation were discussed in Section 4.0 as a part of Task 2, step 5.

4.4 Evaluation of Component Fragilities and Failure Modes

Component fragilities and failure modes were discussed in Section 4.0 as a part of Task 2, step 5.

4.5 Fire Detection and Suppression

Fire detection and suppression were discussed in Section 4.0 as a part of Task 2, step 6.

4.6 Analysis of Plant Systems, Sequences, and Plant Response

4.6.1 Plant Model Overview

The Fire PRA methodology utilized by GGNS starts with the development of a PRA/Fire model containing fire-induced sequences. These fire-induced sequences are initiated with a fire-induced failure (or failures) in an area or compartment followed by a combination of equipment and human failures, both random and fire-induced, which lead to core damage. This is done by first screening the fire areas and compartments and then by modifying the internal events PRA model to incorporate fire-induced initiators and appropriate equipment failure modes. The resulting Fire PRA model is linked to a plant equipment location database. The overall model development process is discussed below.

The next step after the definition of fire compartments was the identification of equipment and cables located in the fire compartments. The primary source of this information was the Appendix R Safe Shutdown Analysis. This safe shutdown equipment information was related to the appropriate equipment in the GGNS internal events PRA via two data bases. One data base relates equipment tags to PRA logic model basic events and the other data base relates cable numbers and location information (compartment number) to equipment tags. During initial quantification efforts it was determined that location information for equipment not required by the Safe Shutdown Analysis but included in the PRA model would be necessary to obtain more realistic results in the analysis. Cable routing information was obtained for additional injection trains, offsite power feeds to the three ESF buses and selected instrumentation modeled in the internal events PRA. However, it should be noted that cable routing information for a large number of components modeled in the internal events PRA was not identified. The components without cable routing information were assumed to fail with any fire. This leads to conservatism in the calculational results.

The logic model for the Fire PRA was essentially the same as the logic model for the internal events PRA. However, modifications were required in order to allow for the use of the model with EPRI's Fire Risk Analysis Code (FRANC) software (Reference 31). The primary modification was the merging of sequence top logic into a single top so that an overall conditional core damage probability could be calculated as controlled by FRANC. The actual calculational tools were the same ones used for the internal events PRA (CAFTA, Reference 32). In addition, the probability of most post-initiator human error events was set to a screening value based on the HEP (human error probability) calculated in the internal events PRA. Others were set to failure probabilities of 1.0 or used the original screening human error probability from the internal events IPE. In general, the treatment of human error probabilities was consistent with the guidance provided in Appendix B of the Implementation Guide.

4.6.2 Screening Process and Results

All plant areas were considered in the search for important areas with regard to fire. To identify the important areas, a progressive screening approach was used. The screening approach provided a systematic means of eliminating non-important compartments from further consideration, thereby allowing resources to be focused on the important fire compartments. In general a compartment is screened from further consideration once the product of its compartment ignition frequency and its Conditional Core Damage Probability (CCDP) drops below 1E-6/yr. Discussion of the screening criteria is provided below.

4.6.2.1 Screen 1--Fire Compartments Inside Containment

The containment building is a large cylindrical concrete and steel structure which encloses the drywell and suppression pool. Typically, areas in containment are large, have communication with multiple elevations, and have low combustible loadings. Formation of a hot gas layer due to a fire is unlikely due to the size and configuration of the areas.

The list of Safe Shutdown equipment and cables which are located in containment is limited. Most of the equipment is indication which provides information on vessel and containment variables. Other equipment and cables in containment are for SRVs, and some system isolation valves. The Safe Shutdown Analysis does not credit any fire wrap in containment, indicating that good physical separation between divisions exists. This, coupled with the expectation that a hot gas layer will not form, gives reasonable assurance that redundant trains of critical equipment will not be damaged by a fire.

Based upon this analysis, the compartments in containment were judged to have low risk significance and were screened from further analysis. A list of compartments located inside containment is provided in Table 4.6.2-1

4.6.2.2 Screen 2--Compartments with No Safe Shutdown or PRA Equipment

Not all of the compartments contained safe shutdown equipment or PRA equipment credited in the GGNS Fire PRA. Even though a fire in these compartments could not damage the safe shutdown equipment or other equipment credited in the PRA, a fire could potentially cause a plant trip and thus a need for the plant to mitigate the event. For these compartments, a CCDP was calculated that assumed all equipment without cable routing information in the model failed due to the fire. The resulting CCDP was combined with the fire ignition frequencies for each compartment. Compartments with core damage probabilities less than 1E-06 were screened from further analysis. Table 4.6.2-2 contains the compartments screened in this step.

4.6.2.3 Screen 3--Assuming all Equipment in Compartment Fails

With this step, the assumption is made that all of the equipment associated with the compartment fails due to the fire. Equipment without cable routing information is also assumed failed by the fire. A CCDP was calculated for each of the compartments that were not screened from the previous screening step. If the product of the compartment ignition frequency and its CCDP fell below 1E-06, the compartment was screened from further consideration. A listing of the compartments screened in this step is included in Table 4.6.2-3.

4.6.2.4 Screen 4--Credit Detailed Recovery

As previously discussed, the CCDP's are calculated using screening values for post accident human actions. These screening values were either the same as the initial screening value used in the GGNS Individual Plant Examination (IPE) or were based on the final calculated Human Error Probability (HEP) used in the IPE. Those based on the final IPE HEP values were for actions that take place in the control room which would not be impacted by the fire. Events that take place outside the Control Room used the original screening value or, in some cases, a more conservative value. The individual cut sets for each of the remaining compartments were reviewed and the probabilities of the recovery actions were updated or appropriate recovery events were added to the cut sets. Compartments whose resulting recovered core damage probability fell below 1E-6 were screened from further consideration. Table 4.6.2-4 contains a list of compartments screened with this method.

4.6.2.5 Remaining Compartments

The previous screening steps resulted in fourteen unscreened compartments. Thirteen of these compartments were subjected to a detailed analysis. The detailed analysis methods and results for these remaining compartments are discussed in Section 4.6.3. The last compartment was the switchyard. This compartment was excluded from additional review because it was addressed in the internal events IPE as a part of the loss of offsite power initiator.

Table 4.6.2-1 COMPARTMENTS IN CONTAINMENT
CA110
CA112
CA310
CA414
CA419
CA443
CA514
CA515
CA516
CA517

Table 4.6.2-2 COMPARTMENTS SCREENED BASED ON NO SAFE SHUTDOWN OR PRA EQUIPMENT

COMPARTMENT	IGNITION FREQUENCY	FINAL CDF
CA107	7.0E-04	2.06E-08
CA108	7.0E-04	2.06E-08
CA111	3.60E-04	1.06E-08
CA124	2.50E-04	7.35E-09
CA125	2.50E-04	7.35E-09
CA130	2.50E-04	7.35E-09
CA209	4.8E-04	1.41E-08
CA210	4.8E-04	1.41E-08
CA225	2.50E-04	7.35E-09
CA319	3.0E-04	8.82E-09
CA323	3.0E-04	8.82E-09
CA325	2.50E-04	7.35E-09
CA326	3.7E-04	1.09E-08
CA402	2.50E-04	7.35E-09
CA404	2.50E-04	7.35E-09
CA405	1.00E-03	2.94E-08
CA406	3.80E-04	1.12E-08
CA429	3.60E-04	1.06E-08
CA430	9.30E-04	2.73E-08
CA431	4.50E-03	1.32E-07
CA432	8.10E-04	2.38E-08
CA433	4.80E-04	1.41E-08
CA436	2.50E-04	7.35E-09
CA506	2.50E-04	7.35E-09
CA519	7.70E-04	2.26E-08
CA530	7.00E-04	2.06E-08
CA533	2.50E-04	7.35E-09
CA534	2.50E-04	7.35E-09
CA536	2.50E-04	7.35E-09
CA537	2.50E-04	7.35E-09
CC125	4.00E-04	1.18E-08
CC126	7.00E-04	2.06E-08
CC204	4.60E-04	1.35E-08
CC205	2.50E-04	7.35E-09
CC205A	2.50E-04	7.35E-09
CC206	2.50E-04	7.35E-09

Table 4.6.2-2 COMPARTMENTS SCREENED BASED ON NO SAFE SHUTDOWN OR PRA EQUIPMENT

COMPARTMENT	IGNITION FREQUENCY	FINAL CDF
CC212	2.50E-04	7.35E-09
CC218	2.50E-04	7.35E-09
CC219	2.50E-04	7.35E-09
CC301	2.50E-04	7.35E-09
CC305	2.50E-04	7.35E-09
CC306	3.50E-04	1.03E-08
CC309	2.50E-04	7.35E-09
CC401	2.50E-04	7.35E-09
CC402A	2.50E-04	7.35E-09
CC404	2.50E-04	7.35E-09
CC405	2.50E-04	7.35E-09
CC405A	2.50E-04	7.35E-09
CC406	2.50E-04	7.35E-09
CC406A	2.50E-04	7.35E-09
CC407	4.60E-03	1.35E-07
CC409A	2.50E-04	7.35E-09
CC410	6.50E-04	1.91E-08
CC411	2.50E-04	7.35E-09
CC412A	2.50E-04	7.35E-09
CC501	2.50E-04	7.35E-09
CC507	2.50E-04	7.35E-09
CC509	2.50E-04	7.35E-09
CC510	2.50E-04	7.35E-09
CC513	2.50E-04	7.35E-09
CC514	2.50E-04	7.35E-09
CC515	2.50E-04	7.35E-09
CC518	2.50E-04	7.35E-09
CC603	3.80E-04	1.12E-08
CC604	3.30E-04	9.70E-09
CC606	2.50E-04	7.35E-09
CC608	3.50E-04	1.03E-08
CC609	2.50E-04	7.35E-09
CC613	2.50E-04	7.35E-09
CC614	3.10E-04	9.11E-09
CC615	2.50E-04	7.35E-09
CC616	2.50E-04	7.35E-09
CC617	2.50E-04	7.35E-09

Table 4.6.2-2 COMPARTMENTS SCREENED BASED ON NO SAFE SHUTDOWN OR PRA EQUIPMENT

COMPARTMENT	IGNITION FREQUENCY	FINAL CDF
CC619	2.50E-04	7.35E-09
CC701	2.50E-04	7.35E-09
CC704	7.20E-04	2.12E-08
CC705	3.00E-04	8.82E-09
CC707	3.70E-03	1.09E-07
CC708	1.90E-03	5.59E-08
CC708A	2.50E-04	7.35E-09
CC711	2.50E-04	7.35E-09
CC712	2.50E-04	7.35E-09
CC713	2.50E-04	7.35E-09
CT212	6.50E-04	1.91E-08
CT312	6.50E-04	1.91E-08
CT405	6.50E-04	1.91E-08
CT406	6.50E-04	1.91E-08
CTR14	1.60E-03	4.70E-08
CTRMAIN	2.90E-03	8.53E-08

Table 4.6.2-3 COMPARTMENTS SCREENED ASSUMING ALL EQUIPMENT IN COMPARTMENT IS FAILED

LOCATION	IGNITION FREQUENCY	CONDITIONAL PROBABILITY	FINAL PROBABILITY
CA102	2.50E-04	4.04E-05	1.01E-08
CA103	1.20E-03	4.04E-05	4.85E-08
CA104	1.80E-03	3.75E-05	6.76E-08
CA105	1.20E-03	5.99E-04	7.19E-07
CA106	2.50E-04	3.42E-05	8.56E-09
CA109	1.20E-03	5.26E-04	6.31E-07
CA132	3.00E-04	2.94E-05	8.83E-09
CA304	2.50E-04	2.94E-05	7.36E-09
CA306	2.50E-04	2.94E-05	7.36E-09
CA308	7.20E-04	3.64E-05	2.62E-08
CA309	1.70E-03	2.10E-04	3.57E-07
CA318	2.50E-03	9.63E-05	2.41E-07
CA320	2.00E-03	1.54E-04	3.09E-07
CA407	1.70E-03	2.94E-05	5.00E-08
CA410	1.70E-03	2.94E-05	5.00E-08
CA529	7.50E-04	3.64E-05	2.73E-08
CC128	4.70E-04	2.94E-05	1.38E-08
CC208	4.00E-04	2.19E-04	8.75E-08
CC208A	2.50E-04	1.50E-04	3.75E-08
CC209	6.50E-04	7.12E-04	4.63E-07
CC213	5.60E-04	6.90E-05	3.86E-08
CC216	2.50E-04	2.94E-05	7.36E-09
CC303	1.50E-03	2.94E-05	4.42E-08
CC304	2.50E-04	1.79E-04	4.46E-08
CC308	2.50E-04	2.94E-05	7.36E-09
CC408	2.50E-04	4.21E-04	1.05E-07
CC601	2.50E-04	1.83E-04	4.58E-08
CC618	2.50E-04	4.21E-04	1.05E-07
CC706	2.50E-04	4.21E-04	1.05E-07
CD301	2.60E-04	1.67E-04	4.34E-08
CDUC1	2.60E-04	1.21E-04	3.15E-08
CDUC3	4.80E-04	5.26E-04	2.52E-07
CESF11	8.70E-04	1.83E-04	1.59E-07
CESF12	8.70E-04	2.94E-05	2.56E-08
CESF21	8.70E-04	1.79E-04	1.55E-07
CM100	2.50E-04	1.12E-04	2.80E-08
CM101	2.20E-03	2.94E-05	6.48E-08

Table 4.6.2-3 COMPARTMENTS SCREENED ASSUMING ALL EQUIPMENT IN COMPARTMENT IS FAILED

LOCATION	IGNITION FREQUENCY	CONDITIONAL PROBABILITY	FINAL PROBABILITY
CM102	3.40E-03	3.35E-05	1.14E-07
CM112	3.00E-04	2.63E-03	7.90E-07
CM200	2.50E-04	1.16E-04	2.90E-08
CM210	1.30E-03	5.62E-04	7.31E-07
CM212	3.00E-04	1.34E-04	4.02E-08
CT219	2.00E-03	4.62E-04	9.23E-07
CT323	2.10E-03	4.21E-04	8.84E-07
CTR12	5.60E-03	2.94E-05	1.65E-07

Table 4.6.2-4 COMPARTMENTS SCREENED WITH CREDIT FOR DETAILED RECOVERY

LOCATION	IGF	Recov Cond Prob	Recov Final Prob
CA115	2.70E-03	4.97E-05	1.34E-07
CA207	1.70E-03	2.04E-04	3.47E-07
CA208	1.70E-03	4.79E-04	8.14E-07
CA219	1.80E-03	2.27E-04	4.09E-07
CA221	1.80E-03	2.54E-04	4.57E-07
CA401	8.20E-03	2.37E-05	1.94E-07
CA539	2.50E-04	1.38E-03	3.45E-07
CC101	5.40E-04	1.05E-04	5.67E-08
CC203	3.50E-04	8.41E-6	2.94E-9
CC207	6.50E-04	1.36E-03	8.84E-07
CC211	6.50E-04	4.53E-04	2.94E-07
CC214	3.50E-04	1.21E-03	4.24E-07
CC307	2.50E-04	7.74E-4	1.94E-7
CC403	6.00E-03	1.73E-05	1.04E-07
CC702	3.50E-04	1.48E-03	5.18E-07
CC703	3.90E-03	1.21E-04	4.72E-07
CD308	2.90E-02	1.05E-05	3.05E-07
CD310	2.90E-02	1.20E-05	3.48E-07
CDUC2	2.60E-04	9.51E-04	2.47E-07
CM110	1.90E-03	1.35E-04	2.56E-07
CM115	1.40E-02	1.73E-05	2.42E-07
CRAD	1.90E-02	1.73E-05	3.29E-07
CT300	3.00E-02	1.73E-05	5.19E-07
CT400	9.40E-03	1.73E-05	1.63E-07
CTR11	5.60E-03	3.47E-05	1.94E-07
YARD	1.90E-04	4.06E-03	7.71E-07

4.6.3 Detailed Analysis

4.6.3.1 Overview of the Detailed Analysis

Thirteen unscreened compartments, listed in Table 4.6.3-1, remained at the conclusion of the screening analysis. These thirteen compartments were subjected to detailed analysis consisting of one or more of the following activities:

- Fire modeling,
- Location of critical targets,
- Definition of accident scenarios,
- Evaluation of conditional core damage probabilities for the scenarios,
- Apportioning the compartment fire frequency among the scenarios, and
- Evaluation of the probability of suppression before damage occurs.

These activities are described in the paragraphs below. Assumptions used in the detailed analysis are listed in Table 4.6.3-2. Specific assumptions applicable to individual compartments are provided in the discussion of each compartment in the sections that follow.

Fire Modeling

Fire modeling relied on the techniques described in the Fire Induced Vulnerability Evaluation (FIVE) method (Reference 30) and the Implementation Guide. Objectives of the fire modeling were:

- To determine whether critical temperatures could be reached in the hot gas layer in a compartment. This analysis took into account the probability of igniting intervening combustibles which could contribute heat to the hot gas layer, as well as the exposure fire.
- To evaluate the probability of damage to local targets due to the temperature in the plume and ceiling jet, the temperature in the hot gas layer, and/or radiant heat flux .

The critical parameters used in the fire modeling are listed in Table 4.6.3-3.

Location of critical targets

Cable locations were primarily determined from the Appendix R Safe Shutdown Analysis (Appendix C), Plant Data Management System (PDMS) and cable racc .ay drawings. Detailed routings for a limited set of critical cables, primarily offsite power and HPCS, were confirmed by engineering review of raceway drawings and walkdowns.

Identification of Accident Scenarios

The process of identifying accident scenarios was highly integrated with the other analysis activities. First, the major contributors to a compartment's conditional core damage probability were identified from the Fire PRA model. Then cable routings within the room were determined for the major contributors. Combinations of targets whose failure could result from a single exposure fire were identified based on the location information and the results of the fire modeling. Finally, the impact of both success and failure of suppression on the extent of damage was assessed.

Evaluation of Conditional Core Damage Probabilities

Conditional core damage probabilities were determined for each unique combination of targets arising from the scenarios. The process for evaluating conditional core damage probabilities was described above in Section 4.6.1. Specific assumptions made during the evaluation of individual scenarios are provided in the sections that follow.

Apportioning Compartment Fire Frequencies

Ignition sources were screened from further consideration if fire modeling showed they were not capable of damaging any other equipment or propagating to other combustibles. The remaining ignition sources were apportioned among the scenarios based on extent of damage determined for each ignition source. For example, if critical temperatures in the hot gas layer could not be ruled out, ignition sources were binned according to whether they could or could not contribute to such a hot gas layer. If critical temperatures in the hot gas layer were not feasible, ignition sources were binned according to their proximity to critical targets or sets of targets. Transient fire frequencies were apportioned to scenarios based on area ratios.

Evaluation of the Probability of Suppression before Damage Occurs

Many of the compartments analyzed are provided with automatic suppression systems. Automatic suppression was credited, where appropriate, with preventing damage to protected targets (e.g., targets wrapped with Nominal one-hour wrap), preventing the hot gas layer from reaching critical temperatures, and preventing ignition of overhead cable trays. Automatic suppression was not credited with preventing damage to targets in the plume or ceiling jet or within critical radiant heat distances. The system unavailabilities for automatic suppression systems used in this study are given n the Implementation Guide (Section 4.2) as follows:

System Type	System Unavailability
Wet pipe sprinkler	2.0E-02
Preaction sprinkler	5.0E-02
CO_2	4.0E-02

Credit was given for manual recovery of automatic suppression systems for preventing damage to protected targets (e.g., targets wrapped with Nominal one-hour wrap), preventing the hot gas layer from reaching critical temperatures, and preventing ignition of overhead cable trays. Manual recovery of automatic suppression was not credited with preventing damage to targets in the plume or ceiling jet or within critical radiant heat distances. Roughly two-thirds of automatic suppression system failures are recoverable (Implementation Guide, Section 4.2).

Credit was given for manual suppression activities by the fire watch and by members of the fire brigade. Manual suppression was credited for preventing damage to protected targets, i.e., targets wrapped with Nominal one-hour or One-hour wraps. Nominal one-hour wrap was assumed to provide protection for up to 45 minutes and One-hour wrap was assumed to provide protection for up to 60 minutes.

Based on these times, the following failure probabilities for manual suppression were used in this study (Implementation Guide, Appendix Q):

Protective Wrap	Time Available for Suppression	Manual Suppression Failure Probability
Nominal one-hour	45 minutes	0.15
One-hour	60 minutes	0.10

Manual suppression was also credited for transient fires due to welding and other transient fires. Results of transient fire characterization tests show that transient fires do not reach peak heat release rates for five minutes or more (Reference 35). Therefore, manual suppression by the firewatch may be credited for fires due to welding and cutting operations. For other transient fires, manual suppression by the first arriving member of the fire brigade may be credited. The probability of manual suppression failure by the fire watch is taken to be 0.15 (Implementation Guide, Appendix Q, Figure Q-3). The probability of manual suppression failure by the first arriving member of the fire brigade is taken to be 0.65 (Implementation Guide, Appendix Q, Figure Q-4). A single value for the probability of manual suppression failure of all types of transient fires may be calculated by averaging the probabilities over their frequencies, as follows:

	Suppression Failure Probability A	Frequency B	A x B
Welding Fires	0.15	1.8E-04	2.70E-05
Other Transient Fires	0.65	7.5E-05	4.88E-05
	SUM	2.55E-04	7.58E-05

The probability of failure to manually suppress transient fires is:

$$Pms = 7.58E\text{-}05/2.55E\text{-}04 = 0.3$$

Detailed analysis of the six most important compartments, and the analysis of multiple compartments is discussed in Sections 4.6.3.2 through 4.6.3.7 below.

4.6.3.2 Detailed Analysis of Compartment CA201

4.6.3.2.1 Description

CA201 consists of the corridors on the 119'-0" elevation of the Auxiliary Building. CA201 has a total area of 19,832 ft^2 and a ceiling height of 20 feet. CA201 is provided with automatic detection alarming in the control room, and a partial coverage wet pipe suppression system protecting areas where Division 1 and 2 cables are separated by <20 feet. The targets of interest in this compartment are cables associated with Division 1 and Division 2 safe shutdown equipment and the HPCS system. Division 1 and 2 safe shutdown cables which are separated by <20 feet in this compartment are protected by at least a nominal one-hour fire wrap.

4.6.3.2.2 Assumptions

Assumptions used in analysis of CA201 include the following:

1) Protected Division 1 and 2 cables in CA201 are protected with either nominal one-hour or one-hour wrap. Therefore, Division 1 and 2 cables are assumed to be protected for up to 45 minutes.
2) The CCDP's for the various scenarios were calculated assuming at least one of the following initiators:
    - Loss of Instrument Air
    - Loss of Plant Service Water
    - Loss of Component Cooling Water
    - Loss of Div1 DC Bus
3) Recoveries that were credited include manual actuation of ECCS, realignment of offsite power to ESF bus, and alignment of firewater for injection to the vessel. The recoveries were applied to individual cut sets depending on the events in the cut sets.

4.6.3.2.3 Analysis

Detailed analysis for CA201 consisted of fire modeling, location of critical targets, identification of accident scenarios, evaluation of conditional core damage probabilities for the scenarios, apportioning the compartment fire frequency among the scenarios, and evaluation of the probability of suppression before damage occurs.

Fire modeling for CA201 consisted of:

- Determining the critical combustible load required to achieve critical damage temperatures in the hot gas layer.
- Determining critical damage and ignition heights for ignition sources in the compartment.

- Determining critical radial distances for ignition sources in the compartment.

The fire modeling calculations for CA201 are reported in Reference 33. As a result of the fire modeling, the following conclusions were drawn:

- Ignition sources were screened from further consideration where overhead targets were above critical damage heights.
- Critical temperatures in the hot gas layer are not feasible in this compartment. None of the ignition sources in the room contains enough combustibles to cause critical temperatures in the hot gas layer from the exposure fire alone. Although some ignition sources may ignite overhead cable trays, the additional heat from the cable tray fire would not be sufficient to cause critical temperatures in the hot gas layer.
- Damage from ignition sources in CA201 would be confined to cables and equipment directly impacted by the exposure fire or intervening combustibles ignited by the exposure fire.

Locations of Division 1 and 2 safe shutdown trays and conduits were determined from safe shutdown raceway drawings. Locations of HPCS cables and conduits were obtained from the review of raceway drawings and were confirmed by walkdowns. Locations of other cable and equipment in CA201 were obtained from the Plant Data Management System (PDMS). The PDMS, while not providing precise locations of cable and equipment in the compartment, allows locating equipment within four subzones: 1A201, 1A211, 1A215, 1A222. These subzones are illustrated in Figure 4.6.3-1.

Based on the results of the fire modeling and location information, nine scenarios were defined for CA201. The scenarios are described and conditional core damage probabilities provided in Table 4.6.3-4. Ignition sources not screened during the propagation analysis were apportioned among the eight scenarios. Unscreened ignition sources in this room consist of electrical cabinets, transformers, ventilation subsystems, pumps and transients. Severity factors defined for the Implementation Guide were also applied to the scenario frequencies based on the ignition source component type contribution. No severity factor was applied to the transient fire frequency contribution. The severity factor represents the fraction of fires that is expected to be severe enough to damage other equipment.

Manual suppression was credited for transient fires due to welding and for other transient fires. As discussed earlier, transient fires do not reach peak heat release rates for five minutes or more. Therefore, manual suppression by the fire watch can be credited for fires due to welding and cutting operations. For other transient fires, manual suppression by the first arriving member of the fire brigade can be credited, since CA201 is provided with automatic detection alarming in the control room.

4.6.3.2.4 Results

The core damage equation for CA201 scenarios is:

$$CDF_{CA201-n} = IF_{CA201-n} * SF * CCDP_{CA201-n} * Pms$$

where
$CDF_{CA201-n}$ = core damage frequency for the scenario
$IF_{CA201-n}$ = ignition frequency for the scenario
$CCDP_{CA201-n}$ = conditional core damage probability for the scenario
Pms = probability of failure of manual suppression for the scenario
SF = severity factor for ignition source type

The total core damage frequency for the room is the sum of the core damage frequencies for the scenarios and is equal to 6.19E-7/yr..

4.6.3.3 Detailed Analysis of Compartment CA301

4.6.3.3.1 Description

CA301 consists of the corridors on the 139'-0" elevation of the Auxiliary Building. CA301 has a total area of 16921 ft^2 and a ceiling height of 27 feet. CA301 is provided with automatic detection alarming in the control room, and a partial coverage wet pipe suppression system protecting areas where Division 1 and 2 cables are separated by <20 feet. The targets of interest in this compartment are cables associated with Division 1 and Division 2 safe shutdown equipment and the HPCS system. Division 1 and 2 safe shutdown cables in this compartment are protected with a nominal one-hour wrap where they are separated by <20 feet.

4.6.3.3.2 Assumptions

Major assumptions used in analysis of CA301 include the following:

1) Protected Division 1 and 2 cables in CA301 are predominantly protected with a nominal one-hour wrap. Therefore, Division 1 and 2 cables are assumed to be protected for up to 45 minutes.
2) The CCDP's for the various scenarios were calculated assuming at least one of the following initiators:
   Loss of Instrument Air
   Loss of Plant Service Water
   T2 (Transient with Loss of PCS)
3) Recoveries that were credited include manual actuation of ECCS, realignment of offsite power to ESF bus, failure to control RCIC, and alignment of firewater for injection to the vessel. The recoveries were applied to individual cut sets depending on the events in the cut sets.

4) HPCS failures due to SSW Pumphouse ventilation were not allowed in Scenario 5 since ventilation is not needed for HPCS SSW pump if Division 1 SSW pump is not operating.

4.6.3.3.3 Analysis

Detailed analysis for CA301 consisted of fire modeling, location of critical targets, identification of accident scenarios, evaluation of conditional core damage probabilities for the scenarios, apportioning the compartment fire frequency among the scenarios, and evaluation of the probability of suppression before damage occurs.

Fire modeling for CA301 consisted of:

- Determining the critical combustible load required to achieve critical damage temperatures in the hot gas layer
- Determining critical damage and ignition heights for ignition sources in the compartment.
- Determining critical radial distances for ignition sources in the compartment.

The fire modeling calculations for CA301 are documented in Reference 33. As a result of the fire modeling, the following conclusions were drawn:

- Ignition sources were screened from further consideration if all overhead targets were above critical damage heights.
- Critical temperatures in the hot gas layer are not feasible in this compartment. None of the ignition sources in the room contains enough combustibles to cause critical temperatures in the hot gas layer from the exposure fire alone. Although some ignition sources may ignite overhead cable trays, the additional heat from the cable tray fire would not be sufficient to cause critical temperatures in the hot gas layer.
- D. mage from ignition sources in CA301 would be confined to cables and equipment directly impacted by the exposure fire or intervening combustibles ignited by the exposure fire.

Locations of Division 1 and 2 safe shutdown trays and conduits were determined from safe shutdown raceway drawings. Locations of HPCS cables and conduits were obtained from the review of raceway drawings and were confirmed by walkdowns. Locations of other equipment in CA301 were obtained from the Plant Data Management System (PDMS). The PDMS, while not providing precise locations of equipment in the compartment, allows locating equipment within six subzones: 1A301, 1A302, 1A314, 1A316, 1A321, 1A322. These subzones are illustrated in Figure 4.6.3-2.

Based on the results of the fire modeling and location information, seven scenarios were defined for CA301. The scenarios are described and conditional core damage probabilities provided in Table 4.6.3-5. Ignition sources not screened during the propagation analysis were apportioned among the seven scenarios. Unscreened ignition sources in this room consist of electrical cabinets, ventilation subsystems, pumps and transients. Severity factors defined for the Implementation Guide were also applied to the scenario frequencies based on the ignition source component type contribution. No severity factor was applied to the transient fire frequency contribution. The severity factor represents the fraction of fires that is expected to be severe enough to damage other equipment.

Manual suppression was credited for transient fires due to welding and for other transient fires in the same manner as discussed for CA201.

4.6.3.3.4 Results

The core damage equation for CA301 scenarios is:

$$CDF_{CA301\text{-}n} = IF_{CA301\text{-}n} * SF * CCDP_{CA301\text{-}n} * Pms$$

where	$CDF_{CA301\text{-}n}$	=	core damage frequency for the scenario
	$IF_{CA301\text{-}n}$	=	ignition frequency for the scenario
	$CCDP_{CA301\text{-}n}$	=	conditional core damage probability for the scenario
	Pms	=	probability of failure of manual suppression for the scenario
	SF	=	severity factor for ignition source type

The total core damage frequency for the room is the sum of the core damage frequencies for the scenarios and is equal to 6.7E-07/year.

4.6.3.4 Detailed Analysis of Compartment CC202

4.6.3.4.1 Description

CC202 consists of the Division 1 Switchgear Room on the 111'-0" elevation of the Control Building. CC202 has a total ceiling area of 2510 ft^2, a total floor area of 1723 ft^2 and a ceiling height of 21 feet. CC202 is provided with automatic detection alarming in the control room, and a full coverage CO_2 suppression system. The targets of interest in this compartment are cables associated with offsite power, Division 1 and Division 2 safe shutdown equipment. Division 2 cables in this compartment are wrapped with at least a nominal one-hour wrap.

4.6.3.4.2 Assumptions

Major assumptions used in analysis of CC202 include the following:

1) Division 2 cables in CC202 are protected with nominal one-hour wrap and are assumed to be protected for up to 45 minutes.
2) Solid state equipment in this room consists of the load shedding and sequencing panels, battery chargers and inverters. Since Division 1 equipment in CC202 is assumed to be failed for all sequences, the lower failure temperature criteria for solid state equipment is accounted for in the analysis.
3) Typical transient combustibles in this area are assumed to consist of ordinary maintenance refuse. The heat release rate for typical transient fires involving ordinary maintenance refuse is assumed to be 138 Btu/s.
4) Although possible, the probability of transient ignition sources being located in the mezzanine areas is considered to be very small and is ignored in the analysis.
5) The CCDP's for the various scenarios were calculated assuming at least one of the following initiators:
   Loss of Offsite Power
   Loss of Division 1 AC Bus
   Loss of Division 1 DC Bus
6) Recoveries that were credited include manual actuation of ECCS, realignment of offsite power to ESF bus, failure to start Div 3 diesel, and alignment of firewater for injection to the vessel. The recoveries were applied to individual cut sets depending on the events in the cut sets.

4.6.3.4.3 Analysis

Detailed analysis for CC202 consisted of fire modeling, location of critical targets, identification of accident scenarios, evaluation of conditional core damage probabilities for the scenarios, apportioning the compartment fire frequency among the scenarios, and evaluation of the probability of suppression before damage occurs.

Fire modeling for CC202 consisted of:

- Determining the critical combustible load required to achieve critical damage temperatures in the hot gas layer
- Determining critical damage and ignition heights for ignition sources in the compartment.
- Determining critical radial distances for ignition sources in the compartment.

The fire modeling calculations for CC202 are reported in Reference 33. As a result of the fire modeling, the following conclusions were drawn:

- Ignition sources were screened from further consideration if all overhead targets were above critical damage heights.
- Critical temperatures in the hot gas layer are feasible in this compartment
- None of the ignition sources in the room contain enough combustibles to cause critical temperatures in the hot gas layer from the exposure fire alone. However, some ignition sources may ignite overhead cable trays. The additional heat from the cable tray fire could cause critical temperatures in the hot gas layer.

- If automatic suppression is successful, ignition of overhead cable trays is prevented. The exposure fire is extinguished promptly and the contribution from the hot gas layer to target environment temperature is negligible. However, damage may occur to targets impacted directly by the exposure fire.
- If automatic suppression is not successful, overhead cable trays may be ignited leading to critical temperatures in the hot gas layer. In this case all unprotected equipment in the room may be damaged.

Locations of power and control cables for offsite power were obtained from the review of raceway drawings and were confirmed by walkdowns. A sketch of the room is provided in Figure 4.6.3-3. Based on the results of the fire modeling and location information, six scenarios were defined for CC202. The scenarios are described and conditional core damage probabilities provided in Table 4.6.3-6. Ignition sources not screened during the propagation analysis were apportioned among the six scenarios. Ignition sources in this room consist of electrical cabinets, transformers, battery chargers and transients. Severity factors defined for the Implementation Guide were also applied to the scenario frequencies based on the ignition source component type contribution. No severity factor was applied to the transient fire frequency contribution. The severity factor represents the fraction of fires that is expected to be severe enough to damage other equipment.

The probability of failure of automatic CO_2 suppression systems is given as 0.04 (Implementation Guide, Appendix Q). Manual suppression was credited for preventing failure of Division 2 cables wrapped with nominal one-hour wrap, which is assumed to afford protection for up to 45 minutes. The manual suppression failure probability has two components: 1) failure to manually recover the automatic suppression system, given as 0.33 (Implementation Guide, Section 4.2) and 2) failure of fire fighters to manually suppress the fire, given as 0.15 (Implementation Guide, Appendix Q). Manual suppression was also credited for transient fires due to welding and for other transient fires in the same manner as discussed for CA201.

4.6.3.4.4 Results

The core damage equation for CC202 scenarios is:

$$CDF_{CC202\text{-}n} = IF_{CC202\text{-}n} * SF * CCDP_{CC202\text{-}n} * Pas * Pms$$

where

$CDF_{CC202\text{-}n}$	=	core damage frequency for the scenario
$IF_{CC202\text{-}n}$	=	ignition frequency for the scenario
$CCDP_{CC202\text{-}n}$	=	conditional core damage probability for the scenario
Pas	=	probability of failure of automatic suppression
Pms	=	probability of failure of manual suppression
SF	=	severity factor for ignition source type

The total core damage frequency for the room is the sum of the core damage frequencies for the scenarios and is 9.3E-07/year.

4.6.3.5 Detailed Analysis of Compartment CC210

4.6.3.5.1 Description

CC210 consists of the Division 3 (HPCS) Switchgear Room on the 111'-0" elevation of the Control Building. CC210 has a total area of 800 ft^2 and a ceiling height of 21 feet. CC210 is provided with automatic detection alarming in the control room, and a full coverage CO_2 suppression system. The targets of interest in this compartment are cables associated with Division 2 safe shutdown equipment, some offsite power source equipment and the HPCS system. Division 2 cables are not wrapped in CC210.

4.6.3.5.2 Assumptions

Major assumptions used in analysis of CC210 include the following:

1) Solid state equipment in CC210 consists of inverters and battery chargers. Since Division 3 equipment in CC210 is assumed to be failed for all sequences, the lower failure temperature criteria for solid state equipment is accounted for in the analysis.
2) Typical transient combustibles in this area are assumed to consist of ordinary maintenance refuse. The heat release rate for typical transient fires involving ordinary maintenance refuse is assumed to be 138 Btu/s.
3) The CCDP's for the various scenarios were calculated assuming the following initiator:
   T2 (Transient with Loss of PCS)
4) Recoveries that were credited include manual actuation of ECCS, realignment of offsite power to ESF bus, manual depressurization and alignment of firewater for injection to the vessel. The recoveries were applied to individual cut sets depending on the events in the cut sets.

4.6.3.5.3 Analysis

Detailed analysis for CC210 consisted of fire modeling, location of critical targets, identification of accident scenarios, evaluation of conditional core damage probabilities for the scenarios, and evaluation of the probability of suppression before damage occurs.

Fire modeling for CC210 consisted of:

- Determining the critical combustible load required to achieve critical damage temperatures in the hot gas layer
- Determining critical damage and ignition heights for ignition sources in the compartment.
- Determining critical radial distances for ignition sources in the compartment.

The fire modeling calculations for CC210 are reported in Reference 33. As a result of the fire modeling, the following conclusions were drawn:

- Critical temperatures in the hot gas layer are feasible in this compartment
- Division 2 raceways cannot be directly impacted by any ignition sources in the room. Therefore, their failure can only result from failure of suppression leading to critical temperatures in the hot gas layer.

Locations of Division 2 cables were obtained from review of raceway drawings and were confirmed by walkdowns. Based on the results of the fire modeling and location information, two scenarios were defined for CC210. The scenarios are described and conditional core damage probabilities provided in Table 4.6.3-7. Severity factors were also applied to the scenario frequencies based on the ignition source frequency contribution. Ignition sources in this room consist of electrical cabinets, transformers, battery chargers and transients.

The probability of failure of automatic CO_2 suppression systems is given as 0.04 (Implementation Guide, Section 4.2). Manual suppression was also credited for transient fires as previously discussed.

4.6.3.5.4 Results

The core damage equation for CC210 scenarios is:

$$CDF_{CC210\text{-}n} = IF_{CC210\text{-}n} * SF * CCDP_{CC210\text{-}n} * Pas$$

where
$CDF_{CC210\text{-}n}$ = core damage frequency for the scenario
$IF_{CC210\text{-}n}$ = ignition frequency for the scenario
$CCDP_{CC210\text{-}n}$ = conditional core damage probability for the scenario
Pas = probability of failure of automatic suppression
SF = severity factor for ignition source type

The total core damage frequency for the room is the sum of the core damage frequencies for the scenarios and is 6.08E-07/year.

4.6.3.6 Detailed Analysis of Compartment CC215

4.6.3.6.1 Description

CC215 consists of the Division 2 Switchgear Room on the 111'-0" elevation of the Control Building. CC215 has a total area of 2201 ft^2 and a ceiling height of 21 feet. CC215 is provided with automatic detection alarming in the control room, and a full coverage CO_2 suppression system. The targets of interest in this compartment are Division 1 and Division 2 safe shutdown equipment and the HPCS system. Division 1 cables in this room are wrapped with nominal one-hour wrap.

4.6.3.6.2 Assumptions

Major assumptions used in analysis of CC215 include the following:

1) Division 1 cables in CC215 are protected with nominal one-hour wrap and are assumed to be protected for up to 45 minutes in this room.
2) Solid state equipment in CC215 consists of the load shedding and sequencing panels, battery chargers and inverters. Since all Division 2 equipment in CC215 is assumed to be failed for all sequences, the lower failure temperature criteria for solid state equipment is accounted for in the analysis.
3) Typical transient combustibles in this area are assumed to consist of ordinary maintenance refuse. The heat release rate for typical transient fires involving ordinary maintenance refuse is assumed to be 1[illegible]3 Btu/s.
4) The CCDP's for the various scenarios were calculated assuming at least one of the following initiators:
   - Loss of Division 1 AC Bus
   - Loss of Division 1 DC Bus
   - Loss of Instrument Air
5) Recoveries that were credited include manual actuation of ECCS, realignment of offsite power to ESF bus, failure to start Div 3 diesel, manual depressurization and alignment of firewater for injection to the vessel. The recoveries were applied to individual cut sets depending on the events in the cut sets.

4.6.3.6.3 Analysis

Detailed analysis for CC215 consisted of fire modeling, identification of accident scenarios, location of critical targets, evaluation of conditional core damage probabilities for the scenarios, apportioning the compartment fire frequency among the scenarios, and evaluation of the probability of suppression before damage occurs.

Fire modeling for CC215 consisted of:

- Determining the critical combustible load required to achieve critical damage temperatures in the hot gas layer
- Determining critical damage and ignition heights for ignition sources in the compartment.
- Determining critical radial distances for ignition sources in the compartment

The fire modeling calculations for CC215 are reported in Reference 33. As a result of the fire modeling, the following conclusions were drawn:

- Ignition sources were screened from further consideration if all overhead targets were above critical damage heights.
- Critical temperatures in the hot gas layer are feasible in this compartment
- None of the ignition sources in the room contain enough combustibles to cause critical temperatures in the hot gas layer from the exposure fire alone. However, some ignition sources may ignite overhead cable trays. The additional heat from the cable tray fire could cause critical temperatures in the hot gas layer.
- If automatic suppression is successful, ignition of overhead cable trays is prevented. The exposure fire is extinguished promptly and the contribution from the hot gas layer to target environment temperature is negligible. However, damage may occur to targets impacted directly by the exposure fire.
- If automatic suppression is not successful, overhead cable trays may be ignited leading to critical temperatures in the hot gas layer. In this case all unprotected equipment in the room may be damaged.
- HPCS cables in this room are above the damage height for any ignition sources in the room. However, if an ignition source were to ignite an overhead cable tray, HPCS cables could be damaged by the ensuing cable tray fire.

Based on the results of the fire modeling, four scenarios were defined for CC215. The scenarios are described and conditional core damage probabilities provided in Table 4.6.3-8. Ignition sources in this room consist of electrical cabinets, transformers, battery chargers and transients. Ignition sources not screened during the propagation analysis were apportioned among the four scenarios. Severity factors were also applied to the scenario frequencies based on the ignition source component type.

The probability of failure of automatic CO_2 suppression systems is given as 0.04 (Implementation Guide, Section 4.2). Manual suppression was credited for preventing failure of Division 1 cables wrapped with nominal one-hour wrap, which is assumed to afford protection for up to 45 minutes. The manual suppression failure probability has two components: 1) failure to manually recover the automatic suppression system, given as 0.33 (Implementation Guide, Section 4.2) and 2) failure of fire fighters to manually suppress the fire, given as 0.15 (Implementation Guide, Appendix Q). Manual suppression of transient fires was also credited as discussed earlier.

4.6.3.6.4 Results

The core damage equation for CC215 scenarios is:

$$CDF_{CC215\text{-}n} = IF_{CC215\text{-}n} * SF * CCDP_{CC215\text{-}n} * Pas * Pms$$

where	$CDF_{CC215\text{-}n}$	=	core damage frequency for the scenario
	$IF_{CC215\text{-}n}$	=	ignition frequency for the scenario
	$CCDP_{CC215\text{-}n}$	=	conditional core damage probability for the scenario
	Pas	=	probability of failure of automatic suppression
	Pms	=	probability of failure of manual suppression

SF = severity factor for ignition source type

The total core damage frequency for the room is the sum of the core damage frequencies for the scenarios and is 4.06E-7.

4.6.3.7 Detailed Analysis of the Control Room

4.6.3.7.1 Description

The control room (Compartment CC502) is at elevation 177'-0" of the Control Building. It is held at positive pressure that would prevent smoke intrusion from fires in other compartments. CC502 has a floor area of 8602 ft^2 and a multi-level ceiling. The area above the main control panels (approximately 1275 ft^2) has a ceiling height of 21 ft. and the remaining areas have a suspended ceiling with a height of 8.5 ft. CC502 has three principal subcompartments, an underfloor area containing cable, the normally-occupied area containing the controls for the plant and the area above the suspended ceiling. The underfloor area is protected by an automatic halon system. The remainder of the control room has detectors in every cabinet except P680 which is the console at which an operator normally sits. The targets of interest are controls for offsite power and each of the three divisions. The control room contains 43 panels and 15 termination cabinets.

4.6.3.7.2 Assumptions

The major assumptions used in the Control Room analysis are:

1. The underfloor area poses no fire risk due to its design (i.e., automatic halon protection and cables that cannot burn at atmospheric conditions (ref. NEDO-10466A)).
2. Suppression prior to loss of a cabinet's function was not credited. This assumption was particularly important to the control room.
3. The loss of a cabinet containing divisional equipment was assumed to affect the entire division.
4. Evacuation of the control room was assumed to occur at the time smoke visibly obscured the panel.
5. The time for evacuation assumed a similar response to the SNL tests.
6. Human detection was not credited except for the manned main control panel and then it was assumed to be only as effective as a smoke detector.

4.6.37.3 Analysis

The analysis of the control room followed the guidelines of Appendix J of the Fire Risk Analysis Implementation Guide. The principal steps of the analysis were:

- Determine logical boundaries of the spread of fire initiated in a cabinet.

- Determine the divisions of equipment contained in those cabinets.
- Identify the safe shutdown capability remaining after loss of each cabinet.
- Calculate the corresponding conditional core damage frequency for the cabinet.
- Calculate the fire ignition frequency for the cabinet.
- Determine the probability of cabinet fire non-suppression prior to the need for control room evacuation.
- Consider the effects of fire on adjacent cabinets.
- Calculate core damage frequencies for fires in individual cabinets and sum to obtain the total control room core damage frequency.

The logical boundaries of the spread of fire in a control room cabinet were based on the principal that a double wall with an air gap will contain the spread of a fire. Because each "cabinet" contained solid walls and was separated (generally by an inch) from the adjacent cabinet(s), individual cabinet boundaries as designed on drawing no. E-0777B were used. A walkdown of the control room, including a visual inspection of selected cabinets, confirmed this evaluation.

The walkdown also identified other physical boundaries to the spread of fire contained within the cabinet. For example, cabinets with multiple doors contained single interior walls. Sometimes the walls contained small openings, other times not. When openings were present, generally only one or a few cables passed through the wall. Often cables from the cabinet interior touched the cabinet wall. Because there was not available test data for these configurations and because available test data did indicate some possibility for damage to cables touching an interior wall, these internal boundaries were not credited.

The above finding applied to the 43 control room panels. For the 15 termination cabinets, an additional one or two swing doors were also present inside the cabinet. Test data was available in NEDO-10466A that implied the swing doors would provide some level of protection (e.g., ten minutes) for cables on the side opposite from the fire. However, during the control room walkdown, it was found that these doors often were not in their tested position. Hence, these doors were not credited.

It is important to note that the dominant core damage scenarios in the control room panels and termination cabinets each required fire damage to occur on opposite sides of these internal boundaries. Hence, the assumption about damage to the entire cabinet implies a potentially significant level of conservatism.

The divisions of equipment contained in each cabinet were determined based on drawing E-0777B. It was also determined that the only controls for offsite power are contained in P807 (one termination cabinet also contains offsite power cables) and that all Division 1 equipment is isolated from its remote shutdown controls but that Division 2 equipment is not. Finally, solid-state equipment (which is more susceptible to temperature) was assumed to be contained in all cabinets.

The safe shutdown capability remaining after the loss of each cabinet depends on whether evacuation must occur. Because evacuation could happen if suppression does not occur and because there is a small probability that suppression will not occur, each cabinet has two scenarios. The first scenario involves fire in the cabinet and evacuation required. The second scenario involves a fire in the cabinet and evacuation as an option. In the latter case, fire damage to control functions may necessitate use of a remote shutdown panel even if the control room is usable.

For evacuation scenarios, three outcomes are possible depending on whether a Division 2 cabinet or the offsite power panels are affected. Fourteen panels (four of which are control benchboads) and seven termination cabinets contain Division 2 equipment. Because Division 2 remote shutdown is not capable of being isolated, it is assumed unavailable for these scenarios. Offsite power and Division 1 remote shutdown are available however.

One panel and one termination cabinet contain offsite power controls. For these scenarios, it is assumed that offsite power is lost, but that both Division 1 and 2 remote shutdown is available.

Finally, for 28 panels and 7 termination cabinets, neither Division 2 nor offsite power controls are affected. Consequently, offsite power and both divisions of remote shutdown are available.

For a fire in a control room cabinet without evacuation, nine outcomes are possible, depending on which divisions are available and whether Division 1 and/or 2 are available from the remote shutdown panels. The nine groups are summarized as follows:

- Group 1 includes 25 panels and 4 termination cabinets. These cabinets contain only BOP equipment.
- Group 2 includes 2 panels and 1 termination cabinet. These cabinets include either Division 1 and BOP equipment or only Division 1 equipment.
- Group 3 includes 9 panels and 1 termination cabinet. These cabinets include either Division 2 and BOP equipment or only Division 2 equipment.
- Group 4 includes 1 panel and 1 termination cabinet. These cabinets include either Division 3 and BOP equipment or only Division equipment.
- Group 5 includes 1 termination cabinet. This cabinet contains both Division 1 and Division 3 equipment.
- Group 6 includes 4 panels (3 of which are control benchboards) and 4 termination cabinets. These cabinets contain either Division 1, Division 2 and BOP equipment or only Division 1 and Division 2 equipment
- Group 7 includes 1 termination cabinet. This cabinet contains Division 2, Division 3 and BOP equipment.

- Group 8 includes 1 panel (a control benchboard) and 1 termination cabinet. These cabinets contain Division 1, Division 2, Division 3 and BOP equipment or only Division 1, Division 2 and Division 3 equipment.

- Group 9 includes 1 panel and 1 termination cabinet. These cabinets include offsite power equipment.

The conditional core damage probability (CCDP) for each control room cabinet was calculated using the IPE/Fire model. Various cases were evaluated for loss of individual divisions and combinations of divisions in accordance with the nine outcomes or groups. The following table summarizes the results:

Group	Equipment Affected	In Control Room CCDP	Control Room Evacuation Case CCDP[1]
1	BOP	3.5E-5	1.26E-2[2]
2	Div 1 and BOP	5.21E-5	1.26E-2[2]
3	Div 2 and BOP	6.48E-4	1.39E-1[3]
4	Div 3 and BOP	1.18E-3	1.26E-2[2]
5	Div 1, Div 3 and BOP	2.67E-3	1.26E-2[2]
6	Div 1, Div 2 and BOP	3.00E-3	1.39E-1[3]
7	Div 2, Div 3 and BOP	4.18E-3	1.39E-1[3]
8	Div 1, Div 2, Div 3 and BOP	2.78E-2	1.39E-1[3]
9	Offsite Power	8.24E-4	5.0E-2[2]

Note 1: The calculated CCDPs have been increased by a factor of 5 to account for increased stress of Control Room evacuation.

Note 2: Both Div 1 and Div 2 equipment operable from the remote shutdown panels.

Note 3: Only Div 1 equipment operable from the remote shutdown panel.

The results for equipment actuated from the remote shutdown panel were assumed to be five times the calculated CCDPs in order to correct for human errors which were assumed to occur at a high stress condition.

The fire ignition frequency for the control room is 9.5E-3, which essentially is the frequency of electrical cabinet fires. The eleven electrical cabinet fires in the FEDB can be categorized as caused by the following:

relays - 6 incidences
circuit boards - 3 incidences
other - 2 incidences

Discussions with the electrical design personnel and the control room walkdown indicated that the principal causes of cabinet fires, namely relays and circuit cards, did not apply to termination cabinets. Of the panels, the heavy concentrations of relays and circuit boards tend to be in non-divisional cabinets. A visual inspection of panels containing all three divisions indicated that P601 was lightly loaded with relays and circuit cards and that P680 had no relays but a light to moderate load of circuit cards. Based on this, the control benchboard panels (P601, P864, P870 and P680) were assumed to have a lighter load of circuit cards and relays. The fire ignition frequency for the three types of cabinets were weighted based on the following assumptions and relationships:

- All fire types apply to control panels (y = control panel weighting factor)
- Only the "other" fires apply to termination cabinets (x = 2/11 * y, where x = termination cabinet weighting factor))
- The relay and circuit card loading of control benchboard panels and therefore the fire initiation frequency is assumed to be 1/4 that of control panel cabinets (z = 1/4 * y, where z = control benchboard panel weighting factor)

$$\therefore IF_{Total} = n_T(x)(IF_{aver}) + n_P(y)(IF_{aver}) + n_{Bb}(z)(IF_{aver}) = 58(IF_{aver})$$

where: IF_{Total} = total CR fire frequency
IF_{aver} = average CR cabinet frequency
n_T = number of termination cabinets
n_p = number of control panels
n_{Bb} = number of control benchboards

substituting and solving:

$$15(2/11)y + 39y + 4(1/4)y = 58$$

y = 1.357
x = 0.247
z = 0.339

The probability of cabinet fire non-suppression prior to the need for control room evacuation is 3.4E-3 based on the Implementation Guide Appendix J. This value assumes detection at or prior to about the time that visible smoke appears in the cabinet and an in-cabinet smoke detector actuates. Because of the presence of in-cabinet detectors in all but P680, this value is conservative, but generally insensitive to assumptions about human detection. For P680, the presence of operators at the console justifies early human detection and a similar suppression probability.

The CCDP's in the above table are conservative in that the assumption is made that the entire cabinet fails due to the fire and that any electrical division associated with that cabinet also fails. In order to make the calculation of control room electrical cabinet core damage frequency more realistic, severity factors have been applied for certain cases. For the in-control room case, severity factors have been applied to the 15 cabinets which contain multiple safety related divisions. A severity factor of 0.20 as calculated for the Implementation Guide was utilized. This is the conditional probability that the fire develops sufficiently to damage the entire cabinet and thus multiple divisions. The remaining portion (0.80) of the contribution was also calculated using a CCDP assuming that one less division within the cabinet has failed. The severity factor has also been applied to all cabinets for the control room evacuation case.

The effects on adjacent panels were also considered. Four termination cabinet fires and one panel fire, if unsuppressed, could progress until three divisions failed. However, a significant time is available to suppress the fire before adjacent cabinet temperatures would affect solid-state equipment. Consequently, because the non-suppression probability would be 2E-2 (based on a ten minute time for suppression) or lower, the contribution from these fires is insignificant compared to the direct damage to the cabinet.

4.6.3.7.4 Results

The total core damage frequency for the GGNS control room from internal fire is 3.85E-6 per year. This value includes contribution from fires that not require evacuation (3.53E-6) and those requiring evacuation (3. 23E-7). The low core damage frequency reflects the effectiveness of the GGNS control room design even with the use of conservative assumptions for evaluating control room cabinet fires.

Slightly more than one-third of the risk is in four cabinets, P601, P702, P855 and P625. Two of these cabinets contain all three divisions, one has both Divisions 1 and 2, and the fourth contains only Division 3. The top [illegible] cabinets (out of 58) consist of approximately 75% of the risk. The remaining cabinets after the top four contribute less than five percent each to the total.

Thirty-seven percent of the risk is due to cabinets that will result in the failure of only Division 2 (Group 3 cabinets). This contribution is due to the relative large number of cabinets that contain Division 2 equipment. Twenty-two percent of the risk is due to cabinets with all three divisions (Group 8). While there are only two such cabinets the CCDP for this situation is relatively high. Seventeen percent of the risk is from cabinets that contain both Division 1 and 2 equipment (Group 6). The CCDP for this group is relatively moderate but there are nine Group 6 cabinets. The remaining groups contribute seven percent or less to the total core damage frequency.

Table 4.6.3-1 UNSCREENED COMPARTMENTS SUBJECTED TO DETAILED ANALYSIS

COMPARTMENT	DESCRIPTION
CA101	Auxiliary Building Corridors, 93'-0" Elevation
CA201	Auxiliary Building Corridors, 119'-0" Elevation
CA301	Auxiliary Building Corridors, 139'-0" Elevation
CC104	Hot Machine Shop
CC202	Division 1 Switchgear Room
CC210	Division 3 (HPCS) Switchgear Room
CC215	Division 2 Switchgear Room
CC302	HVAC Equipment Room
CC402	Cable Spreading Room
CC502	Control Room
CD306	Division 3 (HPCS) Diesel Generator Room
CT100	Turbine Building Floor, 93'-0" Elevation
CT200	Turbine Building Floor, 113'-0" Elevation

Table 4.6.3-2 ASSUMPTIONS

1.	All cabinet fires are treated as fully-involved fires.
2.	Equipment not included in the Fire PRA equipment location data bases was assumed failed in every compartment except where locations were ascertained from raceway drawings or by walkdown.
3.	If auto suppression succeeds, an exposure fire does not propagate to overhead cable trays. Therefore, damage is limited to targets in the plume and ceiling jet of the exposure fire with the successful operation of auto suppression systems.
4.	If auto suppression succeeds, temperature rise in the hot gas layer is insignificant. Therefore, only plume or ceiling jet effects are considered when assessing the target environment temperature for scenarios where automatic suppression succeeds.
5.	Exposure fires are assumed to be detected and suppressed within 15 minutes.
6.	Nominal one-hour wrap is assumed to provide protection for up to 45 minutes. One-hour wrap is assumed to provide protection for up to 60 minutes
7.	Failure of control cables for the switchgear controlling alignment of ESF buses to offsite power transformers (ESF11, ESF12 and ESF21) is conservatively assumed to result in failure of the associated switchgear.

Table 4.6.3-3 CRITICAL FIRE MODELING PARAMETERS

Cable Damage Temperature	700 F
Cable Ignition Temperature	932 F
Critical Radiant Flux for Cable	1 Btu/s-ft2
Heat Loss to the Boundaries of the Hot Gas Layer	85%.

Table 4.6.3-4 SCENARIOS DEFINED FOR CA201

SCENARIO	DESCRIPTION	CCDP
CA201-1	All equipment and cables located in room 1A201 (including Division 1 safe shutdown cables)are assumed failed. Division 2 and HPCS are not in this area and are credited.	9.67E-03
CA201-2a	All equipment and cables located in room 1A211, area a (including Division 1 and 2 safe shutdown and HPCS cables) are assumed failed.	6.70E-02
CA201-2b	Local targets directly impacted by lighting transformer in area a fail. Division 1 and HPCS do not fail.	3.33E-04
CA201-3	All equipment and cables located in room 1A211, area b (including Division 2 safe shutdown and HPCS cables) are assumed failed. Division 1 cables are not in this area and are credited.	3.84E-03
CA201-4	All equipment and cables located in room 1A211, area c (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	3.33E-04
CA201-5	All equipment and cables located in room 1A215 (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	5.68E-03
CA201-6	All equipment and cables located in room 1A222, area d (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	3.52E-04
CA201-7	All equipment and cables located in room 1A222, area e (including Division 1 safe shutdown cables) are assumed failed. Division 2 and HPCS cables are not in this area and are credited.	1.23E-04
CA201-8	Equipment and cables located in the northwest corner of CA201 (area f - Drywell Chiller Area) Equipment in portions of both 1A211 and 1A222 near the drywell chillers, as well and Division 2 safe shutdown cables are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	3.77E-04

Table 4.6.3-5 SCENARIOS DEFINED FOR CA301

SCENARIO	DESCRIPTION	CCDP
CA301-1	All equipment and cables located in room 1A301 (including Division 1 safe shutdown cables) are assumed failed. Division 2 and HPCS are not in this area and are credited.	4.27E-04
CA301-2	All equipment and cables located in rooms 1A302 and 1A314 (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	5.31E-04
CA301-3	All equipment and cables located in room 1A321 (including Division 1 safe shutdown cables) are assumed failed. Division 2 and HPCS cables are not in this area and are credited.	8.75E-04
CA301-4	All equipment and cables located in room 1A322 (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	2.56E-05
CA301-5	All equipment and cables located in room 1A316, area a (including Division 1 and 2 safe shutdown cables) are assumed failed. HPCS cables are not in this area and are credited.	5.77E-02
CA301-6	All equipment and cables located in room 1A316, area b (including Division 2 safe shutdown HPCS cables) are assumed failed. Division 1 cables are not in this area and are credited.	3.52E-03
CA301-7	All equipment and cables in room 1A316, area c (including Division 2 safe shutdown cables) are assumed failed. Division 1 and HPCS cables are not in this area and are credited.	4.60E-04

Table 4.6.3-6 SCENARIOS DEFINED FOR CC202

SCENARIO	DESCRIPTION	CCDP
CC202-1a	Automatic suppression fails and is not recovered. The hot gas layer reaches critical damage temperature, resulting in loss of offsite power and failure of unprotected equipment (Division 1). Manual suppression is not successful within 45 minutes, leading to failure of protected equipment (Division 2) as well.	1.63E-01
CC202-1b	Automatic suppression fails, but is either manually recovered, or manual fire fighting activities succeed within 45 minutes. Offsite power and unprotected cables (Division 1) fail, but protected (Division 2) cables do not fail.	1.64E-02
CC202-2a	Automatic suppression succeeds. The hot gas layer does not reach critical temperatures, but the exposure fire directly fails cables for all three offsite power transformers where they are in close proximity (area a). Division 1 (unprotected) cables also fail. Division 2 (protected) cables do not fail.	1.64E-02
CC202-2b	Automatic suppression succeeds. The hot gas layer does not reach critical temperatures, but the exposure fire directly fails cables for ESF12 and 21 (area b). Division 1 (unprotected) cables also fail. ESF11 cables are not in this area and do not fail. Division 2 cables are protected and do not fail.	9.97E-04
CC202-2c	Automatic suppression succeeds. The hot gas layer does not reach critical temperatures, but the exposure fire directly fails cables for ESF12 (area c). Division 1 (unprotected) cables also fail. ESF11 and 12 cables are not in this area and do not fail. Division 2 cables are protected and do not fail.	8.79E-04
CC202-2d	Automatic suppression succeeds. The hot gas layer does not reach critical temperatures. The exposure fire occurs in an area where it cannot directly fail any offsite power cables. Division 1 (unprotected) cables fail, but Division 2 cables (protected) are not failed.	8.79E-04

Table 4.6.3-7 SCENARIOS DEFINED FOR CC210

SCENARIO	DESCRIPTION	CCDP
CC210-1	Automatic suppression fails. The hot gas layer reaches critical damage temperature, resulting in loss of all equipment in the room.	4.91E-03
CC210-2	Automatic suppression succeeds, preventing hot gas layer from reaching critical temperatures. Division 2 cables do not fail since they are above the damage height for any ignition source in the room.	1.96E-03

Table 4.6.3-8 SCENARIOS DEFINED FOR CC215

SCENARIO	DESCRIPTION	CCDP
CC215-1	Fire occurs in an ignition source where propagation to overhead trays is possible. Automatic suppression fails and is not recovered. The hot gas layer reaches critical damage temperature, resulting in failure of unprotected equipment (Division 2 and HPCS). Manual suppression is not successful within 45 minutes, leading to failure of protected equipment (Division 1) as well.	8.25E-01
CC215-2	Fire occurs in an ignition source where propagation to overhead trays is possible. Automatic suppression fails, but is either manually recovered, or manual fire fighting activities succeed within 45 minutes. Unprotected cables (Division 2 and HPCS) fail, but protected (Division 1) cables do not fail.	5.11E-03
CC215-3	Fire occurs in an ignition source where propagation to overhead trays is possible. Automatic suppression succeeds. Protected (Division 1) cables do not fail, and HPCS cables (above the damage height) do not fail.	9.06E-04
CC215-4	Fire occurs in an ignition source where propagation to overhead trays is not possible. The hot gas layer does not reach critical temperatures. Protected (Division 1) cables do not fail, and HPCS cables (above the damage height) do not fail.	9.06E-04

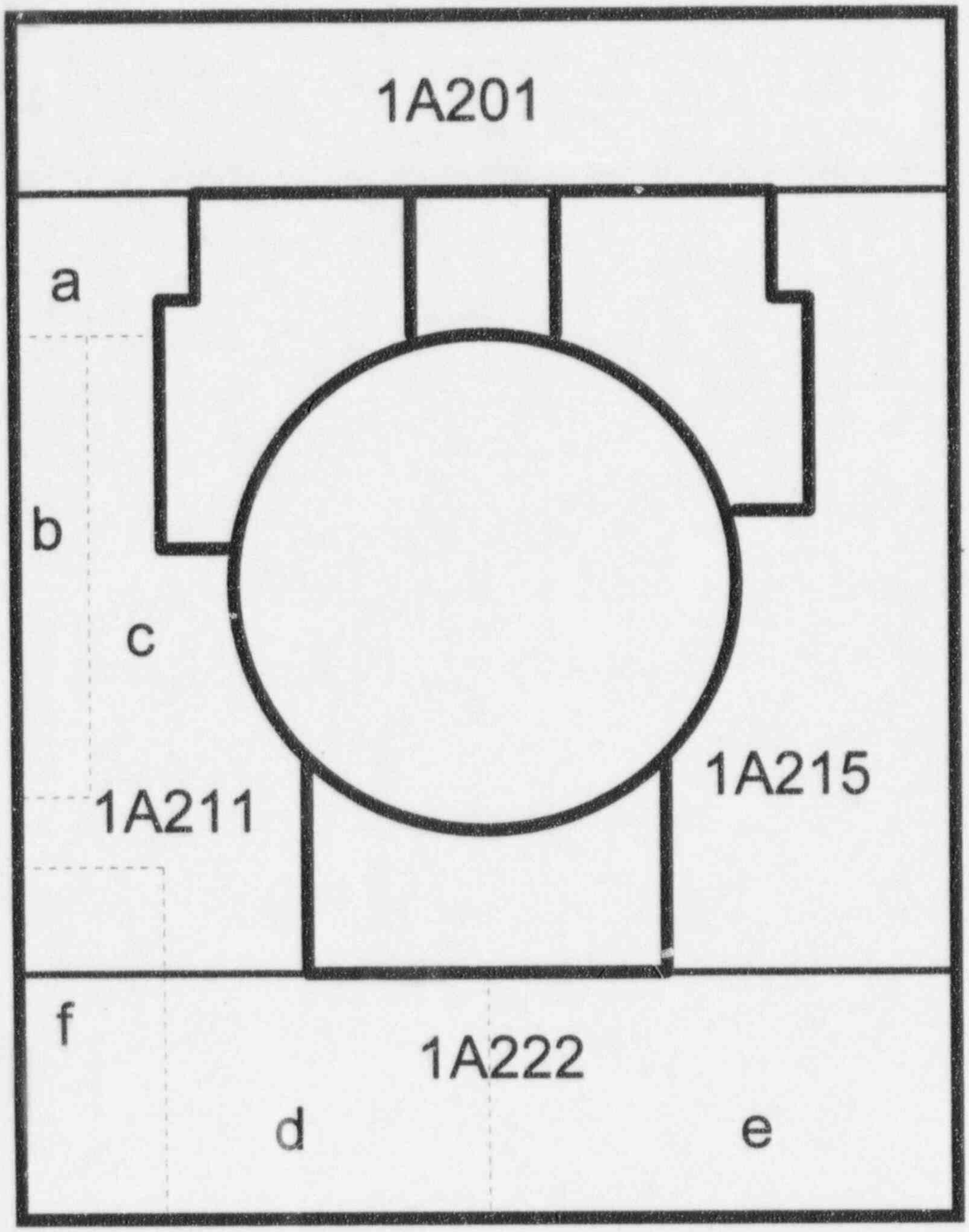


Figure 4.6.3-1 Equipment Location Subzones for CA201

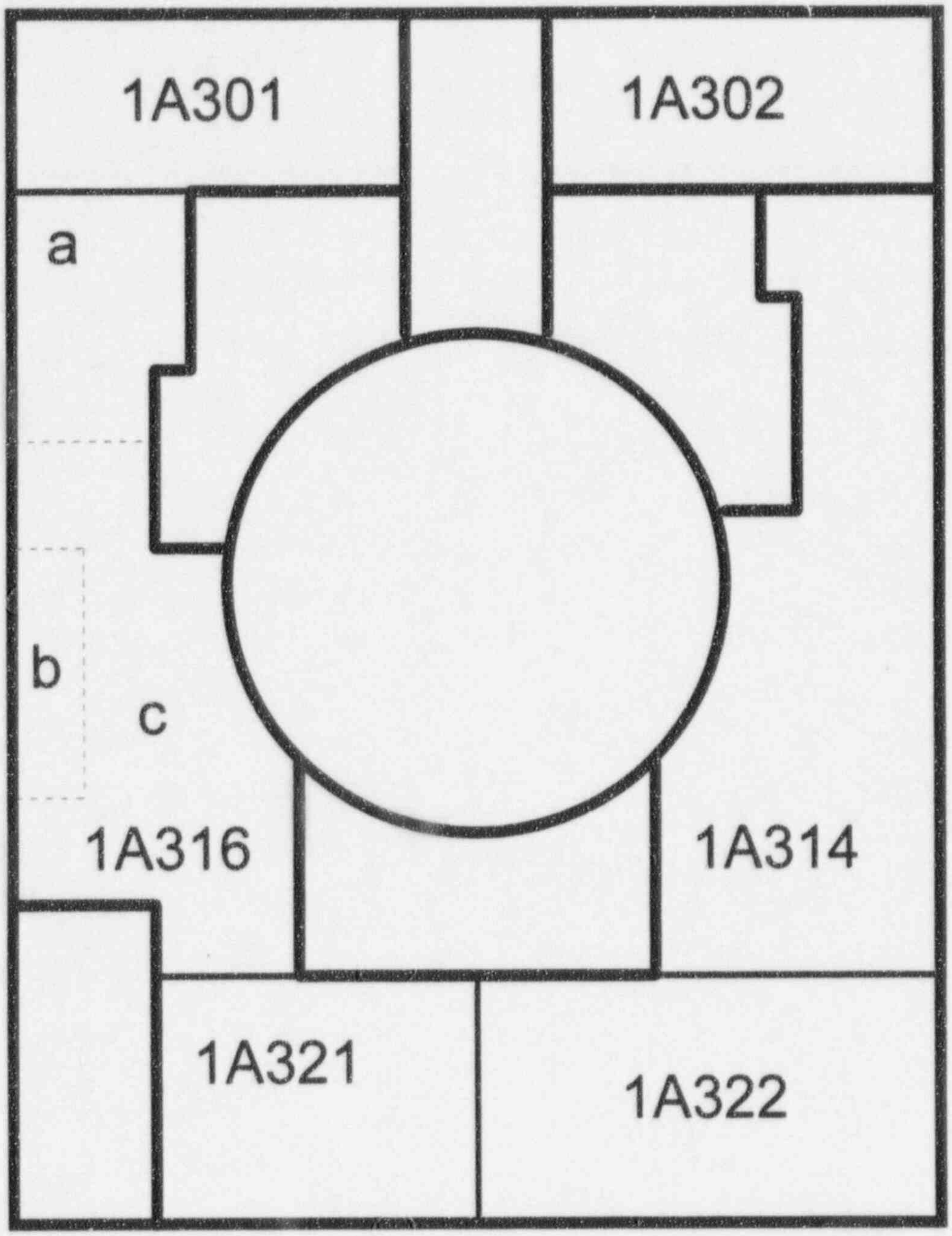
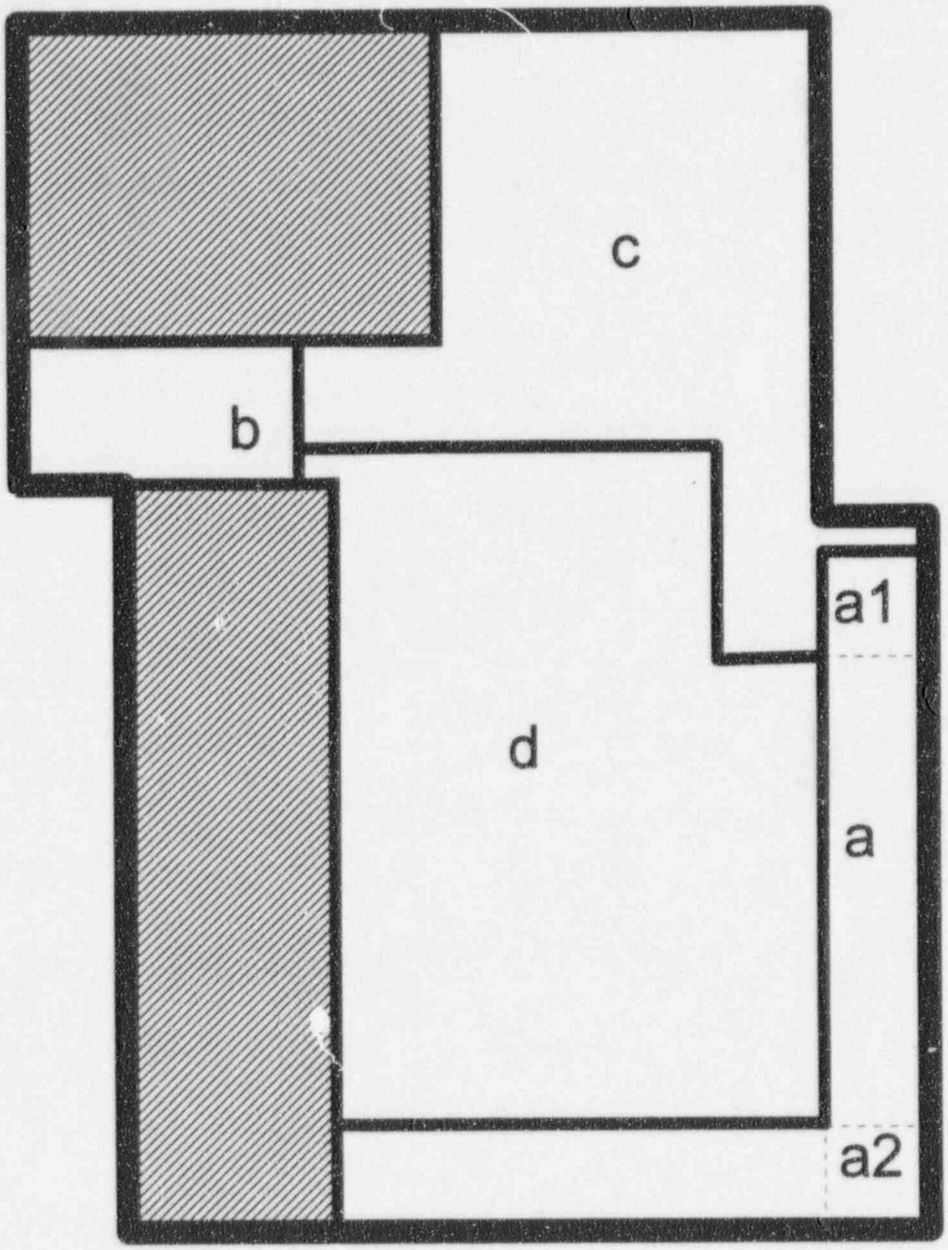


Figure 4.6.3-2 Equipment Location Sub zones for CA301



Cross-hatched area indicates mezzanine areas over the Div 1 and 3 battery rooms and the remote shutdown rooms.

Figure 4.6.3-3 Equipment Location Subzones for CC202

4.6.4 Analysis of Scenarios Involving Multiple Compartments

4.6.4.1 Description

The multi-compartment analysis considers the plant locations identified in Step 1 of the methodology. That is, both screened and unscreened compartments are considered. The analysis considers the effects of a fire in one compartment, called the exposing compartment, generating hot gases that spread to an adjacent compartment. The gases spread to the adjacent compartment due to an active failure of a fire door or damper or a passive failure in the barrier. If the fire in the exposing compartment is not suppressed, sufficient hot gases may accumulate to damage cables and equipment in both the exposing and the adjacent compartment.

4.6.4.2 Assumptions

The major assumptions used in the multi-compartment analysis are:

- Consistent with NUREG/CR-4840, a barrier failure probability of 0.1 was used unless an evaluation was performed of the numbers and types of barriers.
- This barrier failure data conservatively bounds plant-specific data because the source for failures often considers a barrier to be failed when it would in fact prevent the spread of hot gases.
- Fire spread to more than one additional compartment is probabilistically insignificant.
- Hot gases cannot spread to an adjacent compartment on a lower level because it is below the virtual source of the fire.
- Suppression will be reliable enough such that the probability that fire in the exposing compartment will exceed a barrier rating is small compared to the assumed barrier failure probabilities.
- Suppression by a gaseous system in an adjacent compartment was not considered to be effective either in cooling hot gases in the exposing compartment or in extinguishing the fire in the adjacent compartment.
- A hot gas layer of 700°F was assumed to occur, unless: 1) it was shown otherwise in the detailed analysis of individual compartments; or 2) either the exposing or adjacent compartments were judged to be too large to allow it.

4.6.4.3 Analysis

The analysis of multi-compartment fires follows the guidelines of Step 8 of the Implementation Guide. The principal steps of the analysis were:

- Identify logical combinations of exposing and adjacent compartments.
- Determine applicable hot gas layer scenarios.

- Calculate a screening value for the fire propagation frequency and screen compartment combinations.
- Credit automatic suppression for water systems in the adjacent compartment.
- Perform simplified CCDP calculations for unscreened combinations.
- Calculate scenario-specific barrier failure probabilities.
- Perform detailed CCDP calculations for the remaining unscreened combinations.

The logical combinations of exposing and adjacent compartments were determined primarily by inspecting plant drawings. Consistent with the assumptions discussed earlier, combinations of an exposing compartment and single adjacent compartments were identified. It was also designated as to whether the intervening barrier was a ceiling or a wall.

From this list of combinations, deletions were made according to whether a hot gas layer of critical temperature could occur. Certain compartments are simply too large to support hot gas layer development for the applicable sources. A combination was deleted if such compartments were in the exposing or adjacent compartment list. Other compartments might be capable of generating a hot gas layer, but, by a specific evaluation of the ignition source and intervening combustibles in it, they were determined to be incapable of it. This evaluation generally was performed in the detailed analysis of the compartment described earlier in this section. In this case, a combination was deleted if such a compartment was the exposing compartment.

Having deleted the known physically precluded cases, the analysis focused on probabilistic techniques. A fire propagation frequency was calculated that was the product of the following terms:

- The ignition source frequency for ignition sources capable of causing a hot gas layer.
- The failure probability of automatic suppression in the exposing compartment.
- The screening value for barrier failure probability.

The first term value was based on the information available for the compartment. The ignition source frequency from Step 3 (see Section 2.2) was used if the compartment screened in Step 4. If a fire modeling screening walkdown had been performed, the Step 3 frequency was adjusted to remove screened ignition sources because those had been shown incapable of propagating. If detailed modeling of hot gas layer scenarios had been performed, the frequency was further adjusted to remove all ignition sources incapable of causing a hot gas layer.

The second term value the reliability data values for suppression systems from NSAC-179L. However, a specific evaluation was performed to ensure the automatic suppression system would be effective. For water systems, the only systems credited were full compartment systems or systems that protected all sources capable of causing a hot gas layer. For gaseous systems, a more detailed evaluation is required because a barrier failure could allow gas to escape the exposing compartment. If such a dilution was large enough, the gas would be ineffective and the fire in the exposing compartment would not be suppressed.

An evaluation was performed of the active components in the barrier, the gas system design and the total volume of the combined compartments. The following guidelines were used to establish the effectiveness of gaseous systems under those conditions:

- The system was a CO_2 system and the barrier was the ceiling. (CO_2 is heavier than air and will not escape to the adjacent compartment.)
- The total volume of the combined compartments was approximately less than or equal to twice the exposing compartment. (A second shot of gas can be manually initiated and it is likely that the design concentration of the system is twice that needed to suppress the fire.)
- There are no active components in the barrier. (Passive barrier failures rarely if ever involve actual openings for gas to pass through and, if they do, the openings are often small.)

If the situation met one or more of these criteria, the NSAC-179L values were credited.

The last term value used is the screening value for barrier failure probability from NUREG/CR-4840. That value is used in lieu of a specific count of the number and types of barriers. Subsequent evaluations confirmed this value to be conservative, generally by nearly an order of magnitude or more.

The resulting fire propagation frequency was calculated for each combination. If the frequency was below 1E-6/yr, the combination was screened. Unfortunately, many frequencies were just slightly above the screening criteria and few combinations screened in this step.

Because a full compartment water system would cool hot gases in an adjacent compartment, it could prevent a multi-compartment scenario. Hence, for those combinations with a wet pipe or pre-action system in the adjacent compartment, the fire propagation frequency was multiplied by the appropriate NSAC-179L value for the water system's unreliability. However, check was made to determine if the exposing compartment also contained automatic suppression. If so, a dependency analysis consistent with Appendix L of the Implementation Guide was performed to ensure crediting both values was appropriate. A number of compartment combinations screened in this step.

For each of the unscreened combinations, a simple evaluation was performed to obtain a screening CCDP. The evaluation compared the equipment lost in the two compartments. Values for the CCDP were assigned as follows: If the equipment lost in one compartment had the same or greater functional effect than the equipment lost in another, the unrecovered CCDP of that compartment was used. If the equipment lost had a substantially different functional effect, a CCDP of 1.0 was used. If the product of the fire propagation frequency and this CCDP was less than 1E-6/yr, the compartment combination was screened. If not, but if the recovered CCDP would screen the compartments, the recovery analysis was evaluated to ensure it was applicable. If so, recovery was credited and screening performed. This step screened many compartment combinations, including nearly all of the combinations for which the exposing or adjacent compartment contained few if any cables or equipment.

If these simplified CCDP calculations failed to screen the compartment, a barrier evaluation was performed. The evaluation was done consistent with Step 8.4 of the Implementation Guide. For Type 3 barriers, however, the Grand Gulf evaluation of penetration seals for IN 88-04, indicated that penetration seals would be sufficient for the types of scenarios considered. One Type 3 barrier failure was used to represent this experience. Most of the remaining combinations screened at this point.

In a few cases, a detailed evaluation of the CCDP for the combination was required. This analysis was performed using a process similar to that described in Section 3.2. The results of the evaluation indicated that three compartment combinations just barely fail to meet the screening criteria. These were:

Exposing Compartment	Adjacent Compartment	Core Damage Frequency (yr^{-1})
CC202	CC215	1.20E-6
CC207	CC202	1.20E-6
CC210	CC215	1.06E-6

A quantitative evaluation of hot gas layer development, application of fire severity factors and credit for manual suppression would screen these combinations. Hence, these scenarios are considered inappropriate for further consideration and were also screened.

4.6.5 Summary of Results and Conclusions

The overall result of the Fire PRA includes the contributions from all compartments that were not screened in the initial phases of the analysis. The summation of the core damage frequencies from these compartments is the total core damage frequency due to internal fires at GGNS. A listing of these compartments is found in Table 4.6.5-1. The total core damage frequency is 8.76E-06/yr.

The control room is the major contributor to the total core damage frequency with 44% of the total. The high contribution and importance of the control room makes intuitive sense because most of the divisional and important balance of plant equipment is controlled from this location. Each of the remaining individual compartments contribute less than 11% each to the total.

As a group, the control building switchgear rooms are the second major contributor to internal fire risk. These locations collectively account for 22.1% of the internal fire risk. This is expected since power and control feeds for divisional equipment is concentrated in the switchgear rooms. Compartment CC202, the Division 1 Switchgear Room, by itself contributes 10.6% of the total. The room is more important than the other switchgear rooms because of the potential to fail Division 1, some Division 2 equipment and offsite power.

The 93', 119' and 139' elevations of the Auxiliary Building as a group contribute 21% of the internal fire risk. This is primarily due to presence of all three electrical divisions in portions of these compartments.

When considering the types of initiators, the largest contributor is electrical cabinets at 63%. A large portion of this is from the control room. The basic assumption of the cabinet fire causing the failure of the division associated with the cabinet makes this result conservative. The second major contributor by initiator type is transient sources. Transient sources contribute to 24% of the total. This is also somewhat conservative in that it is assumed that the transient source (when suppression fails) will always become a fully developed fire capable of damaging overhead cables or nearby equipment.

Table 4.6.5-1 RESULTS SUMMARY TABLE

COMPARTMENT	DESCRIPTION	FINAL CDF	% OF TOTAL
CC502	Control Room - Control Building, 166'-0"	3.85E-06	44.0%
CC202	Division 1 Switchgear Room - Control Building, 111'-0" Elevation	9.30E-07	10.6%
CA301	Auxiliary Building Corridors, 139'-0" Elevation	6.70E-07	7.7%
CA201	Auxiliary Building Corridors, 119'-0" Elevation	6.19E-07	7.1%
CC210	Division 3 (HPCS) Switchgear Room - Control Building, 111'-0" Elevation	6.08E-07	6.9%
CA101	Auxiliary Building Corridors, 93'-0" Elevation	5.74E-07	6.6%
CC215	Division 2 Switchgear Room - Control Building, 111'-0" Elevation	4.06E-07	4.6%
CT100	Turbine Building, 93'-0" Elevation	3.24E-07	3.7%
CC402	Lower Cable Room - Control Building, 148'-0" Elevation	2.82E-07	3.2%
CC104	Hot Machine Shop - Control Building, 93'-0" Elevation	2.42E-07	2.8%
CD306	Division 3 (HPCS) Diesel Generator Building	1.72E-07	2.0%
CC302	HVAC Equipment Room - Control Building, 133'-0" Elevation	6.99E-08	0.8%
CT200	Turbine Building, 113'-0" Elevation	7.10E-09	0.1%
	TOTAL	8.76E-06	

4.7 Containment Evaluation

The impact of fires within the containment is specifically excluded from analysis as part of the Fire PRA methodology. This is based on the assumptions that a hot gas layer is unlikely to form that could damage cables and the historical frequency and consequences of fires in containments. The accuracy of these assumptions regarding GGNS was reviewed to ensure that there is no significant risk associated with the containment.

The first assumption that hot gas layer formation is unlikely was determined to be applicable to GGNS. The GGNS containment is a General Electric Mark III containment with an internal free volume of 1.67 million cubic feet. A review of potential combustion sources did not identify any source with the potential to form a hot gas layer in this structure.

The second assumption is based on data in the EPRI Fire Events Database which found that the frequency of fires occurring inside the containment was very low. Review of GGNS fire reports did not identify any situation that contradicted the low frequency assumption. Other factors contributing to the assumption of low consequences is that containment fires do not have the potential to damage redundant equipment with the same fire plume. In the GGNS containment, redundant equipment is typically installed with large spatial separation. In most cases, such equipment has either the drywell or the steam tunnel between redundant equipment. Large scale testing (which included combustion of hydrogen in a quarter-scale Mark III containment test facility) and analyses performed by GGNS and the Hydrogen Control Owner's Group support these conclusions.

The impact of a fire on the ability to isolate the containment is another aspect of containment performance. This aspect was assessed by identifying the penetrations that were potential containment bypass candidates from the GGNS IPE. These penetrations were identified in Engineering Report No. GGNS-92-0036. Each of these penetrations was assessed separately. Each was evaluated on the potential for containment isolation valves to fail to close or to inadvertently open. All of the penetrations were screened based on one or more of the following considerations.

- The CDF of the compartment under consideration was less than 1E-7/yr.
- Bypass would require a sustained hot short in a normally shut MOV.
- There was sufficient separation between cabling of two isolation valves that a single fire source could not fail both valves.
- Bypass would require a piping or seal rupture.

4.8 Treatment of Fire Risk Scoping Study Issues

Six issues related to fire risk were identified in the NRC Fire Risk Scoping Study (NUREG CR-5088). The six issues are identified below and as requested, an assessment of these issues as they apply to GGNS has been completed. The assessment for each issue is provided below.

- Seismic/Fire Interactions
- Fire Barrier Qualifications
- Manual Fire Fighting Effectiveness
- Total Environment Equipment Survival
- Control System Interactions
- Adequacy of Analytical Tools

4.8.1 Grand Gulf Nuclear Station Fire Protection Program

The Fire Protection Program at Grand Gulf Nuclear Station is based on the concept of defense-in-depth. This defense-in-depth concept has as its objectives:

- Prevent fires from starting.
- Detect rapidly, control, and extinguish promptly those fires that do occur.
- Provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by fire suppression activities will not compromise the ability to achieve safe shutdown of the plant.

The Fire Protection Program also delineates responsibilities and the methods to be used to accomplish the objectives stated above. The Fire Protection Program interfaces with other GGNS manuals, plans, and procedures to provide an effective and coordinated Fire Protection Program that encompasses all phases of operation, administration, maintenance, and emergency activities. Procedures are in place to ensure implementation of the Fire Protection Program thus reducing the probability of fire at GGNS.

4.8.2 Seismic/Fire Interactions

4.8.2.1 Seismically Induced Fires

As part of the IPEEE Reduced Scope Seismic Margins Assessment (SMA), a Safe Shutdown Equipment List (SSEL) was developed including the requirements of seismic induced fire and seismic induced flooding. The Seismic Review Team (SRT) reviewed the potential for earthquake caused fire hazards and internal flooding in areas containing equipment on the SSEL.

Hydrogen storage tanks are located in the open yard and the hydrogen is piped in the turbine building. Hydrogen lines in the turbine building were not specifically addressed in the SMA, since the only safe shutdown equipment in the turbine building is steam pressure transmitters B21N076A, B, C, & D. The reason for the acceptability of these transmitters in a non-safety related building is that they fail safe and are included on the SSEL only because they are termination points for the Safe Shutdown Paths. In addition, the Fire PRA calculated a total CDP of 3.54E-08 for unscreened zones in the turbine building. Therefore, no significant risk of core damage exists for seismic induced hydrogen fires. Transient sources of flammable gases, such as welding gases and small hydrogen bottles used for calibration, are allowed within the plant; however, they are strictly controlled by plant procedures 10-S-03-3 "Fire Prevention and Control of Ignition Sources" and 10-S-03-4 "Fire Prevention and Control of Combustible Material". As a result, they do not present an unacceptable threat.

4.8.2.2 Seismic Actuation of Fire Suppression System

Actuation of fire suppression systems at GGNS is addressed in Section 4.8.5.2 "Spurious or Inadvertent Fire Suppression Activation".

4.8.2.3 Seismic Degradation of Fire Suppression Systems

Inside hose stations are required to be operable following a seismic event. The only equipment not included in the SSEL for fire protection was inside fire hose stations. Walkdowns performed in the SMA determined that the hose stations were anchored adequately for the seismic loading expected during a safe shutdown earthquake event at GGNS.

Suppression systems located in areas containing seismic class I equipment are analyzed and installed as seismic class II/I. Should a break in a suppression system occur as a result of a seismic event, the issue of internal flooding is bounded by the flood analysis as discussed in the UFSAR Section 3C.3, Moderate Piping Failure, UFSAR Section 3C.4, Compartment Flooding, or Engineering Report No. SERI-89-0007 "Engineering Report for Internal Flooding - Significant Operating Experience Report (SOER) 85-05". Therefore, seismic degradation of fire suppression systems should not degrade seismic class I equipment in the event of a seismic event.

4.8.3 Fire Barrier Qualifications

Fire barriers at GGNS including barrier components (doors, fire dampers, penetrations seal assemblies, etc.) are adequately designed, inspected, tested and maintained. Therefore, a high level of reliability of barrier effectiveness is demonstrated.

4.8.3.1 Fire Barriers

Fire barrier walls, ceilings, and floors are constructed in accordance with the Standard Building Code and designs which have been fire tested and listed by a recognized testing laboratory (UL, FM, etc.). Deviations from tested configurations are evaluated for acceptability by a qualified fire protection engineer. In some locations GGNS utilizes Thermo-Lag 330-1 fire barrier enclosures for separation of redundant safe shutdown raceways within a fire area. These installations have been analyzed as part of the GGNS program to address issues related to the adequacy of the Thermo-Lag fire barrier enclosures. Upgrades of existing Thermo-Lag enclosures to provide a qualified hourly fire endurance rating are scheduled to be completed by the end of 1996. In the mean time hourly fire watches are maintained in the affected areas. Again, deviations from tested configurations are evaluated for acceptability by a qualified fire protection engineer. With the completion of this work, all 10 CFR 50, Appendix R fire barrier enclosures for electrical raceways will be a qualified fire barrier based on fire test and evaluated by a qualified fire protection engineer.

Fire barrier walls, ceilings, & floors and raceway fire barrier enclosures are inspected for operability on an 18 month schedule in accordance with surveillance procedure 06-OP-SP64-R-0047 and 06-OP-SP64-R-0048 respectively. These procedures also address compensatory measures required in the event that inoperable conditions are found. Modifications to these fire barriers are reviewed by Fire Protection personnel in the Nuclear Design Engineering Department. This ensures that modifications made to fire barriers are acceptable with regard to maintaining the fire barrier qualifications.

4.8.3.2 Fire Doors

Most fire doors at GGNS are listed by a recognized testing laboratory (UL, FM, etc.). Those that are not listed for fire service are installed where other regulatory requirements (water tight, containment boundary, etc.) exist and listed doors are not available. In these cases the doors are certified to be built with similar construction as listed fire doors or are evaluated for acceptability based on actual conditions in the area. Fire doors are inspected on a routine basis in accordance with surveillance procedures 06-OP-SP64-D-0043, 0044 and 0045 and the following schedule:

• Locked close fire doors	Once per 7 days.
• Fire doors with automatic hold-open and release mechanisms and unlocked fire doors without electrical supervision (Release mechanism checked on an 18 month schedule).	Once per 24 hours
• Electrically supervised fire doors	Once per 31 days.

These fire door surveillance procedures also address compensatory measures required in the event that inoperable conditions are found.

4.8.3.3 Penetration Seal Assemblies

Penetration seal assemblies at GGNS are designed and installed to maintain a fire resistance rating at least equivalent to that of the fire barrier. A 10% sample of each type penetration seal assembly is inspected in accordance with surveillance procedure 06-OP-SP64-R-0049 on an 18 month frequency. This frequency ensures that each penetration seal is inspected at least once per 15 years. The surveillance procedure also address compensatory measures required in the event that inoperable conditions are found.

NRC Information Notices 88-04, 88-04 Supplement 1, and 88-56 identified potential problems with penetration seal assemblies. Following issuance of these IEN's, GGNS initiated a proactive evaluation of the concerns expressed in these notices to ensure the adequacy of the GGNS Penetration Seal Program. This proactive evaluation is documented in Engineering Report Number 03-0200-1184, dated May 17, 1989; Engineering Report Number GGNS-90-26; and GGNS Information Notice Review dated August 4, 1988.

4.8.3.4 Fire Dampers

Fire dampers at GGNS are designed and installed in all ventilation openings through fire barriers to provide at least an equivalent fire rating as that of the fire barrier in which it is installed. Fire dampers are listed by a recognized testing laboratory (UL, FM, etc.) and installed in accordance with the vendor's installation instructions. The dampers are inspected on an 18 month frequency in accordance with Surveillance Procedure 06-ME-SP64-R-0045. A portion of the fire dampers is drop tested and the remainder are visually inspected in accordance with the above procedure. This fire damper surveillance procedure also addresses compensatory measures required in the event that inoperable conditions are found.

NRC Information Notices 83-69 and 89-52 identified potential problems with fire damper installations. Following issuance of these IEN's, GGNS initiated evaluations of the concerns expressed in these notices to ensure the adequacy of the GGNS fire damper design. These evaluations are documented in internal documents PMI-84/0839, PMI-84/2231, PMI-84/8093, PMI-90/3852, and PMI-90/2546.

4.8.4 Manual Fire Fighting Effectiveness

4.8.4.1 Reporting Fires

All plant personnel with unescorted access to the plant receive instructions on reporting fires in General Employee Training which is provided on an annual frequency. In addition, security personnel receive instructions that address entry procedures for offsite fire departments, crowd control for persons exiting the station, and procedures for reporting fires during their tours of the station. In addition, procedure 10-S-03-2, “Response To Fires” identifies responsibilities with regards to response to fires for individuals, control room operator actions, shift superintendent actions (Shift Fire Chief), plant personnel actions, fire brigade actions, health physicist actions, offsite fire fighting assistance, and actions after the fire.

Fire extinguishers are located throughout the plant in accordance with NFPA 10 requirements. These fire extinguishers are inspected on a routine basis in accordance with procedures 04-S-03-P64-11, 13, and 14.

The plant paging system and the plant telephone system are used for communications with the control room for reporting fire.

4.8.4.2 Fire Brigade

A site fire brigade trained and equipped for fighting fires that may occur at GGNS is established to ensure adequate manual fire fighting capability for all areas of the plant. The fire brigade is composed of at least five members on each shift and these five members are not part of the minimum shift crew necessary for safe shutdown of the unit. The fire brigade leader and at least two fire brigade members have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on the safe shutdown capability.

Fire brigade members receive an annual physical examination to determine acceptability for performing strenuous activity.

A full complement of fire brigade equipment is maintained in each of 5 fire lockers strategically located throughout the plant. Each locker consist of a minimum of turnout coats, boots, gloves, hard hats, self contained breathing apparatus, portable lights, two-way radios, portable ventilation equipment, and portable fire extinguishers. The fire lockers are inspected weekly to ensure proper fire fighting equipment is available.

4.8.4.3 Fire Brigade Training

A fire brigade training program is established and ensures that the capability to fight potential fires at GGNS is established and maintained. The fire training program consists of classroom instruction, actual fire fighting practice, and fire drills. Procedure 10-S-03-7 "Fire Protection Training Program" established requirements for this program. Training and drills meet all the requirements for a trained fire brigade as identified in 10 CFR 50, Appendix R, Section III.I.

Prior to assignment as a fire brigade member, personnel receive instruction in the following topics:

- Identification of fire hazards, their location, and associated types of fire that may occur at GGNS.
- Identification and location of installed and portable fire fighting equipment in the plant.
- Familiarization with plant layout including access and egress routes for each area.
- Proper use of installed and portable fire fighting equipment.
- Correct methods of fighting various types of fires.
- Indoctrination in the Fire Protection Plan. This includes individual and fire brigade responsibilities.
- Proper use of breathing, communications, lighting, and portable ventilation equipment.
- Detailed review of the pre-planned fire fighting strategies.
- Review of modifications, changes, etc. to the physical plant, procedures, fire fighting equipment, or Fire Protection Plan.
- Methods of fighting fires inside buildings and confined spaces.
- The toxic and corrosive characteristics of expected products of combustion.

In addition to the above topics, fire brigade leaders receive training in directing and coordinating fire fighting activities. Refresher training for each brigade member, in the above topics, is conducted annually.

4.8.4.4 Practice

Practice sessions are held for each shift fire brigade on the proper methods of fighting the various types of fires that could occur at GGNS. The practice sessions provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting. Practice sessions for each fire brigade member are conducted a minimum of once per year.

4.8.4.5 Drills

Fire brigade drills are performed in the plant to promote effective teamwork on the fire brigade. Drills generally use actual operation of fire protection equipment or utilize simulated use of equipment for various situations and types of fires. The situation selected for each drill simulates the size and arrangement of a fire that could reasonably occur in the area selected and different areas are selected from previous drills.

Fire brigade drill frequency and guidelines are as follows:

- Drills are performed for each shift at intervals of 92 days.
- Each fire brigade member is required to participate in a minimum of two drills per year.
- One drill per calendar year for each fire brigade shift is unannounced.
- One drill per calendar year for each fire brigade shift is conducted on a backshift.
- Unannounced drills are preplanned and critiqued by a board of responsible management personnel responsible for plant safety and fire protection.
- All drills are preplanned to meet established training objectives and critiqued to determine the effectiveness in these objectives.
- Performance deficiencies of fire brigades or individual members are corrected by providing additional training for noted weakness.
- Unsatisfactory drill performance is corrected by providing additional training for the noted weakness and a repeat drill is held within 30 days.
- Randomly selected unannounced drills are monitored and critiqued by a group of qualified individuals at least once every three years. These qualified individuals are independent of GGNS.

4.8.4.6 Records

Records necessary to establish acceptability of each fire brigade member's qualifications are maintained and are available for review.

4.8.5 Total Environment Equipment Survival

4.8.5.1 Potential Adverse Effects on Plant Equipment by Combustion Products

Products of combustion (smoke) is a recognized hazard to both equipment and personnel. GGNS believes that studies to date are insufficient to adequately quantify the potential problems and solutions associated with non-thermal fire effects on industrial plant equipment. However, the detrimental short term effects of smoke on equipment are not believed to be significant.

Smoke removal is an important aspect of fire fighting and adequate training is accomplished at GGNS on this aspect. Appropriate training is conducted on utilization of installed smoke removal systems as well as portable smoke ejectors, which are provided for fire brigade use.

4.8.5.2 Spurious or Inadvertent Fire Suppression Activation

The impact of spurious or inadvertent fire suppression activation on safe shutdown was analyzed and documented in a "Water Suppression Effects Study" (DCA No. NPE-4-254, Request No. 1 and transmitted by MPT 85/0252). This analysis involved identification of safety related equipment in areas containing water suppression systems and a walkdown to identify equipment which could be adversely affected by system activation. Corrective actions to preclude adverse affects were implemented for equipment identified as potentially affected. Activation of CO_2 or Halon 1301 suppression systems as installed at GGNS have no known adverse affect on operation of safety related equipment.

4.8.5.3 Operator Action Effectiveness

GGNS is designed for alternate shutdown capability for the control room. The plant can be safely shutdown from the Remote Shutdown Panel should a fire cause evacuation of the control room. This remote shutdown capability is electrically independent from cabling, controls, and equipment located in the control room. Training is provided on this remote shutdown capability and the capability was successfully demonstrated to the NRC on February 24-28,1992 and documented in NRC Inspection Report No. 50-416/92-06.

4.8.6 Control System Interactions

GGNS has completed it's Appendix R analysis and modifications. The control room is the only area at GGNS designed for alternate shutdown capability. An NRC inspection team evaluated this alternate shutdown capability in the inspection identified in section 4.8.5.3 above and found it to be acceptable. This remote shutdown capability provides GGNS the ability to safely shutdown the plant with a fire in the main control room.

4.8.7 Adequacy of Analytical Tools

Fire Modeling techniques of the FIVE Methodology were utilized. These fire modeling techniques are derived from the same basic correlations used in COMPBRN IIIe; therefore, no additional evaluation is required for this issue.

5.0 HIGH WINDS, FLOODS, AND OTHERS

5.1 High Winds

The Updated Final Safety Analysis Report (UFSAR), Safety Evaluation Report (SER) and other pertinent design and licensing documents were reviewed in order to find the site specific hazard data and licensing basis for high winds and tornadoes. The findings of this review were compared to the criteria of the 1975 Standard Review Plan (SRP)[15] so that possible differences could be identified. To ensure that any possible significant changes to the reviewed documents were included, recent weather data for the region was compared with data used to formulate the existing high winds hazard basis. Additionally, a review of current site drawings and a walkdown of the area was performed to determine if there are any potential vulnerabilities not included in the original design basis analysis (Reference - Engineering Report No.: GGNS-93-0048, Rev. 0)[18].

At Grand Gulf Nuclear Power Station (GGNS), Seismic Category I structures are designed for the extreme wind and tornado phenomena. The minimum design wind velocity for these structures is 90 mph at 30 feet above ground for a 100 year recurrence interval. This wind speed is the most conservative value based on SRP specified reference codes as well as site and regional meteorology data. A review of new meteorology data for the period after the original evaluation revealed that this existing parameter still represent the most conservative value. The procedures that were used to transform the wind velocity into pressure loadings on the structures and the associated vertical distribution of the wind pressures and gust factors are in accordance with ASCE Paper # 3269[24] and ANSI Standard A58.1[25] and were acceptable as determined in the NRC review per section 3.3.1 of the SER. The tornado loadings are calculated on the basis of a maximum wind velocity of 360 mph which is the vector sum of a maximum peripheral rotational velocity of 290 mph and a translational velocity of 70 mph. The maximum design pressure drop is 3 psi with a maximum rated change of 2 psi/sec. The radius from the center of the tornado at which the maximum wind velocity occurs is 150 ft. These parameters conform to those given in Regulatory Guide 1.76[26]. The methods employed to convert tornado loadings into forces and to distribute them across the structures conform to the requirements in SRP Section 3.3.2.

The Structures, Systems and Components which need protection from externally generated missiles as required by SRP Section 3.5.2 are, in general, protected from the tornado missiles in SRP Section 3.5.1.4, due to their location in or behind missile-proof structures. These structures were designed using methodology that predicts local and overall damage, meeting the intent of SRP Section 3.5.3. However, several isolated components (e.g. Standby Service Water return lines) are not provided this type of protection. A probabilistic evaluation was performed to determine the total annual frequency of occurrence for missiles of the types described in SRP Section 3.5.1.4 in striking these vulnerable areas. This evaluation determined this frequency to be 0.77 X E-8/yr. (Reference - Calc. CC-Q1111-94004, Rev. 0)[19]. This low frequency of occurrence is well below the screening criteria in IPEEE.

In summary this review concludes that GGNS meets the intent/criteria set forth in the 1975 SRP for the High Winds and Tornado Hazard. Some concerns were discovered, but they were evaluated and found to be within the acceptance criteria. No changes to the protection for High Winds and Tornado were identified or required.

5.2 Floods

5.2.1 Hydrologic Conditions and Existing Flood Protection

Grand Gulf Nuclear Station is on the east bank of the Mississippi River near river mile 406, approximately 25 mi. south of Vicksburg, Mississippi, and 6 mi. northwest of Port Gibson, Mississippi. The site is located in the water resources planning area No. 7 of the lower Mississippi River region. It is bounded on the west by the Mississippi River and on the east by loessial bluffs. At the plant site, the river flood plain is about 60 miles wide with elevations ranging from 55 to 75 ft. Mean Sea Level (MSL) Flow is confined to a width of about two to four miles by high bluffs on the east bank and man-made levees on the west bank.

Of immediate relevance to the plant site are two small steep streams. Stream A, north of the site, is perennial and drains Basin A with an area of 2.8 square miles. Stream B, south side of the site, is intermittent and drains Basin B with an area of 0.6 square miles. Both streams drain into Hamilton Lake located in the flood plain of the Mississippi River. Stream A, receives most of its water from the watershed outside the plant area and has a 12-foot culvert under the access road to connect it to the flood plain. Stream B receives most of its water from the site and has a 15-foot culvert under the access road to carry local floods and site drainage.

The plant yard has an average nominal elevation of 132.5 ft. Finished floor elevations for plant at-grade spaces is 133.0 ft. The plant yard is graded to direct runoff away from the buildings, and toward Streams A and B via a combination of drainage swales, ditches, and overland flow.

Flood seals are installed on eleven doors: OC313, and OCT5 in the control building; 1D301, 1D308, 1D309, 1D310, and 1D312 in the diesel generator building; and 1M110, 1M111, 2M110 and 2M111 in the standby service water (SSW) pump houses. Seals, penetration sleeves, toe plates, and curbs are installed in the SSW pump houses to prevent water from reaching safety related equipment.

5.2.2 Original Design Basis Evaluations

The safety-related facilities, systems, and equipment are capable of withstanding the worst flooding caused by a combination of several hypothetical events. These events are: probable maximum flood of the Mississippi River coincident with wind generated waves; seismic failure of upstream dams coincident with the U. S. Army Corps of Engineers design-project flood; ice flooding; probable maximum flood of the two small streams adjacent to the plant; and flooding of the site due to PMP rainfall on the site watershed. A detailed discussion of the external flooding analyses related to the existing GGNS design basis is presented in GGNS UFSAR, Section 2.4 and in Engineering Report GGNS-93-0001[13]. These analyses can be briefly summarized as follows:

(a) Evaluation of the Mississippi River, and adjacent streams A and B indicates that floodwaters associated with their flooding do not result in inundation of the power block structures. Therefore, GGNS may be considered a dry site as defined in Regulatory Guide 1.102, and hydrostatic loading of external walls and structures due to inundation are not applicable. As a result, the only structural concern related to external flooding, is the potential for failure of roofing system due to ponding during heavy precipitation.

Rainfall that falls on the roof of on-site buildings is collected by roof drains and discharged directly into a subsurface drainage system which is designed for the 100-year rainfall. Safety related roofing systems are fitted with overflow scuppers. Water flowing over these scuppers falls to the ground at the side of the building and then flows over the yard surface by natural drainage. Roof structures for the SSW pump rooms, as well as the auxiliary, control, and diesel generator buildings, were evaluated to ensure that during the PMP rainfall the roofing systems will not fail, even if the depth of roof ponding is assumed to exceed parapet height.

(b) Drainage of local intense precipitation is evaluated in accordance with the criteria set forth in Regulatory Guide 1.59. Based upon an average time of concentration in the yard of approximately 30 minutes, the 6 hour PMP estimates from HMR No. 33, and the temporal distribution obtained from EM-1110-2-1411; a rainfall intensity of 16.4 inches per hour is used. The Rational Formula is used to estimate flows for performance of a backwater analysis to determine water surface elevations near plant structures. Maximum calculated water levels near primary power block structures may be taken as 133.20 ft. MSL on the east side of the power block and 133.25 ft. MSL on the west side of the power block; and could exceed elevation 133'-0" for about 7 hours. As a result, flood barriers requiring 6" of freeboard are maintained on all of the at-grade penetration seals in the SSW pump houses, diesel generator building, and control building. Leakage through remaining at-grade power block openings has been estimated and determined to pose no threat to the safe operation of the plant.

5.2.3 Evaluation of Revised Hazards Due to Flooding of the Mississippi River

The Mississippi River PMF, and the U. S. Army Corps of Engineers' Design Project Flood on which it is based, have not increased since the GGNS became operational. The 2 year recurring wind speed at GGNS has also remained the same. Slightly higher levee elevations exist, and future levee elevations of 106 ft. MSL have been approved by the U. S. Army Corps of Engineers. Resulting PMF water levels including wind wave effects would be approximately 116 ft. MSL, which is still well below the elevation of the plant yard.

5.2.4 Review of Flood Hazards Due to Precipitation over the Site Watershed

Applying the revised criteria, the maximum site rainfall intensity for use in the Rational Equation is approximately 28.2 inches of rain per hour. Water levels in the plant yard, are determined using flow models similar to those used in the design basis analysis. In some instances models were altered to consider less conservative flow widths and friction losses, combined weir and culvert flow, and partial by-pass of obstructions. Plant improvements were assumed as indicated in section 7.3.2 For a detailed description of the methods by which site water levels are determined see Engineering Report GGNS-93-0001 and calculations CC-Q1Y13-93001 through CC-Q1Y13-93003. Superimposing the average wave height onto the calculated water surface elevations, and comparing to the As-Built flood level protection the following summary is obtained:

Barrier	Top Elevation(ft. MSL)	WSEL (ft. MSL)	Freeboard
Door OCT5	134.25	134.15	0.10 feet
Door OC313	134.25	134.15	0.10 feet
Door 1D301	134.25	134.21	0.04 feet
Door 1D308	134.25	134.21	0.04 feet
Door 1D309	134.25	134.21	0.04 feet
Door 1D310	134.25	134.21	0.04 feet
Door 1D312	134.25	134.21	0.04 feet
Door 1M110	134.25	134.07	0.18 feet
Door 1M111	134.25	134.07	0.18 feet
Door 2M110	134.25	133.61	0.64 feet
Door 2M111	134.25	133.61	0.64 feet
Other, SSW A	133.62	134.07	- 0.45 feet
Other, SSW B	133.62	133.61	0.01 feet

Therefore, the current level of protection at all structures except the SSW A pump house remain adequate for the new rainfall criteria. Protective barriers around penetrations inside the SSW A structure are sheltered from the wind, and are therefore still adequate using the revised criteria. However, the barrier crossing the exterior equipment hatch would require additional protection to guard against the revised PMP.

Leakage into the power block through unprotected openings was estimated based upon the maximum static water level calculated near the power block and assuming gross failure of non-safety related roof structures. Additionally, the entire six hour PMP can be expected to enter the incomplete decking on the 185 foot level of the former unit 2 auxiliary building. In this event, water entering the power block, would be expected to flood the control, turbine, radwaste, and former unit 2 buildings to an elevation of approximately 99.11 feet MSL All safety related equipment in the control building which is essential in attaining and maintaining a cold safe shutdown is located above elevation 111 ft. and the auxiliary building is watertight up to elevation 114 ft. The design basis circulating water line break is expected to flood these spaces to a height of 108 ft. without functional degradation of any equipment essential to attaining and maintaining a cold safe shutdown. Therefore, flooding of the power block to elevation 99.11 ft. is bounded by the circulation water analysis, and no impact on the ability to attain and maintain a cold safe shutdown will result.

5.2.5 Effects of the New PMP Data on Roof Loading

The control, auxiliary, and diesel generator buildings, as well as the SSW pump houses, have been evaluated for structural adequacy of their roofing systems, and application of either the design basis rainfall data or the new rainfall data can be expected to have no impact on the structural adequacy of these structures. Further, due to the presence of roof drains and overflows, calculations demonstrate that ponded roof depths do not exceed the level at which water could propagate into the buildings through doors, vents, or other openings. Water depths on the auxiliary building roof could exceed the sill elevations on doors 1A502 and 1A504 for a short time. However, these doors are secondary containment (airtight) boundaries fitted with gaskets similar to those making up the PMP door seal assemblies. Therefore, little if any leakage through these doors is expected. The enclosure building roof is not expected to fail during the PMP and ponded depths are not expected to be large enough to result in leakage through roof openings due to the presence of roof drains and overflows. However, the roofing system is not adequate to withstand ponding up to the parapet height. The overflows alone will prevent roof stresses from exceeding material yield stresses. However, the roof drains alone would not be adequate to prevent roof failure. Therefore, the roof drainage system and roof overflows must be relied upon to prevent roof failure.

5.3 Transportation and Nearby Facility Accidents

The analysis for transportation and nearby facility accidents was performed using Standard Review Plan 2.2.1 & 2.2.2 "Identification of Potential Hazards in Site Vicinity" sections from NUREG 75-087[6]. Potential external hazards or hazardous material may be a threat to plant safety if accidents involving nearby industrial, military, and transportation facilities and routes would constitute a design bases event. These facilities and routes include air, ground, and water traffic, pipelines, and fixed manufacturing, processing, and storage facilities were reviewed for location and separation distance. This analysis was divided into the following sections.

- Industrial and military facilities.
- Transportation facilities and routes.
- Mississippi River accidents.
- Significant changes

5.3.1 Industrial and Military Facilities

There is no extensive industrial activity around the Grand Gulf site. There are no military installations, chemical or munition plants, stone quarries, or major gasoline-storage areas located within 5 miles of the plant. The nearest military facility is about 100 miles away. The nearest industrial facilities that have significant quanities of stored chemicals are in Port Gibson about

4.5 miles away and a 4-inch gas line is about 4.5 miles east of the site. Based on the separation distances of the chemical storage facility and gas line, the safety of the plant will not be affected. There are no commercial airports within 10 miles of the site, and no major air routes near the site.

5.3.2 Transportation Facilities and Routes

There are two county roads near the site that carry local traffic. The nearest major highway is U.S. 61 which passes within 4.5 miles east-southeast of the site. The Natchez Trace Parkway is located about 6 miles southeast of the site. The nearest railroad carrying hazardous material is 30 miles south of the site. Based on the separation distances, we conclude that accidents along these routes will not affect the safe operation of the plant. A review of the 10-15-92 edition of the oil and gas map for Mississippi showed that no new pipe lines have been installed within 5 miles of the plant.

5.3.3 Mississippi River Accidents

Mississippi River traffic pose no significant hazard to the plant. The Mississippi River passes 1.34 miles west of the site at its closest point. The plant is protected from the consequences of accidents on the river by distance and a 65 foot earthen east river bank. The effects of toxic clouds resulting from barge accidents has also been analyzed and taken into account. The risk to control room operators is acceptably low due to the low concentration of the chemical upon reaching the control room or the low probability of an accident which falls within the guidelines established in Standard Review Plan section 2.2.3. There is no danger of ships or barges damaging intake structures or of corrosive liquids being drawn into the plant from an accident of the river since there are no intake structures.

5.3.4 Significant Changes

A review was performed to determine if any significant changes have occurred from that reported in the UFSAR. The following is a discussion of some of the findings. Highway 61 is in the process of becoming a 4-lane highway. The additional two lanes are being added on the east side of the existing highway. This activity required the relocation of the existing 4" natural gas transmission line to just west of the highway's right-a-way. Therefore, the closest distance of this highway to the plant site will not change. Additionally, the change from a two lane highway to a four lane highway should help reduce the accident rate on the highway. The Port Facility south of the plant has been built but at the present is only being used by local fishermen and there is no planned additional industrial development for the port at this time. Southern Cotton Oil (formally Port Gibson Oil Works) has replaced the two 12,000 gallon Hexan tanks with one 15,000 gallon Hexan tank which has a leakage monitoring system. The railroad line between Port Gibson and Vicksburg including the spur line to the plant has been removed.

5.3.5 Conclusion

On the basis of the information provided the Grand Gulf facility is protected and can be operated with an acceptable degree of safety considering the activities at nearby transportation, industrial, and military facilities.

5.4 Others

An assessment of the core damage risk from lightning initiators at Grand Gulf Nuclear Station was performed in support of the GGNS IPEEE effort, even though such an assessment was neither a requirement nor lightning an initiating event of special concern.

The Level 1 PSA model was used with updated GGNS-specific initiating event data, including recent lightning-initiated plant scrams to evaluate the core damage risk from lightning initiators (Reference 14). This report is in agreement with the conclusions made by the NRC, that the risk from lightning-initiated severe accidents is insignificant, and there is no need to address lightning as a separate external event initiator in responding to the IPEEE generic letter supplement.

A review for other external events with potential severe accident vulnerability was performed as described in Section 2 of NUREG-1407. Of those items listed there is no prevailing evidence that would indicate that these events should be addressed in the IPEEE process and there are no other known plant-unique external events that should be assessed.

6.0 LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE Program Organization

Entergy Operations, Inc. (EOI) personnel were involved in all aspects of the IPEEE. Contractor expertise was used to supplement in-house capabilities. The vast majority of contractor work was performed on-site with EOI personnel working closely with the contractors. This allowed technology transfer from the contractors and also ensured that the knowledge and insights gained from the analysis remained with the utility.

EOI personnel who performed the Seismic Analysis attended industry training on the EPRI Seismic Margin Methodology. Three individuals attended the EPRI SQUG training course and the EPRI IPEEE Seismic Add-on Course. All three of these individuals meet the qualification requirements of a "Seismic Capability Engineer" as described in EPRI NP-6041.

6.2 Composition of Independent Review Team

Mr. Harry Johnson of Programmatic Solutions and Mr. Robert Budnitz of Future Resources Associates, Inc. performed the peer review for the IPEEE SMA at Grand Gulf Nuclear Station. These contractors have expertise in the Nuclear Systems and Seismic Evaluations and are well recognized in the nuclear industry.

Independent Peer Reviews for the remaining portions of the IPEEE evaluations were performed by EOI design engineering personnel from other nuclear sites as follows:

Portion	Peer Reviewer	Site
External Flooding	Murray Moser	Arkansas Nuclear One (ANO)
Transportation & Nearby Facility Accidents	Roberrt Murillo	Waterford 3 (WF3)
High Winds & Tornado Assessment	Todd Reichardt	Riverbend (RBS)

Fire PRA methods and results were reviewed by a GGNS team with representatives from various groups. They are listed below.

Operations	Walter Cade (SRO)
Fire Protection	Bob Hicks
	Wayne Brown
System Engineering	Scott Kirby
Nuclear Safety and Regulatory Affairs	Wayne Russell (SRO)
Design Engineering (Appendix R/Safe Shutdown Expert)	Thomas Barnett

6.3 Areas of Review and Major Comments

6.3.1 Seismic Review

Mr Johnson's review covered all seismic evaluations portions of the project and included a review of the Project Plan, the draft report, a visit to the site for a sample walkdown and a review of the documentation. Mr. Budnitz's review covered all aspects of the project and included a review of the Project Plan, the SSEL development documentation, the draft report and a review of the documentation.

Mr Johnson and Mr. Budnitz concluded that there were no deficiencies in the IPEEE SMA effort at GGNS in the final report, the SSEL, the walkdown or documentation.

6.3.2 High Winds, Floods, and Others

Peer Review was performed for each major portion of the evaluation and focused on the individual engineering reports summarized in section 2.4. Copies of supporting calculations, UFSAR passages, and other reference material was provided to the reviewer for consideration during review of the report. Review considered the methodology used, the requirements of NUREG 1407, the requirements of the '75 SRP, and the requirements of other applicable regulatory and industry documentation.

The method of documenting Peer Review comments varied with each report. In some instances comments were documented in correspondence between the reviewer and the preparer. In other instances, a summary of Peer Review comments was included as an attachment to the report reviewed. Engineering reports have been signed by the Peer Reviewer's to indicate the satisfactory incorporation of all comments.

6.3.3 Internal Fires Analysis

The team reviewed a summary of the methodology and the results. A meeting was held for additional discussion of the results and resolution of comments. Conclusions and observations from the analysis were also reviewed and agreed upon.

6.4 Resolution of Comments

Where applicable, Peer Review comments have been incorporated into the reports summarized in section 2.4. Similarly, where Peer Review comments affected the draft supporting calculations, the calculations were amended to address the comment. Engineering reports have been signed by the Peer Reviewer's to indicate the satisfactory incorporation of all comments.

7.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

This section defines vulnerability for each external event and then compares Grand Gulf Nuclear Station against the definition to determine if any vulnerabilities exist.

7.1 Seismic Analysis

A vulnerability due to a Seismic event is defined as a component, identified as being needed in the SPLD, which is not capable of surviving the Review Level Earthquake (RLE). The seismic walkdowns found that the Grand Gulf Nuclear Station is seismically rugged and that all components in the SPLD adequately considered the Seismic input. All the SPLD equipment was screened out and the outliers were evaluated. There are no outliers requiring further evaluation.

It is therefore concluded the Grand Gulf Nuclear Station has no vulnerabilities with regard to Seismic events.

7.2 Internal Fire Analysis

7.2.1 Vulnerabilities Due to Internal Fires

One of the main purposes of the IPEEE process is the identification of plant specific vulnerabilities to severe accidents. As stated in the GGNS IPE submittal (Reference 2), GGNS has chosen the general methodology suggested by NEI 91-04 (formerly NUMARC 91-04, Reference 34) for vulnerability screening. This document provides guidance for the closure of severe accident issues in Table 1 of Section 2 of that document. As stated in the GGNS IPE, the definition of a plant specific vulnerability is any condition satisfying the CDFs in the top evaluation category of this table. None of the compartments contributing to fire core damage frequency meet this test; therefore, there is no plant specific vulnerability due to internal fire at GGNS.

The largest contributor is the control room with 44% of the internal fire core damage frequency. While this room does met the lowest category of NEI 91-04, Table 1, no specific action is planned at this time. The suggested "Licensee Response" for this category is "Ensure Severe Accident Management Guidance is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure." No action is necessary because, as stated above, the control room results are conservative and current procedures are considered adequate. GGNS will take the Fire IPE results into consideration during the implementation of the BWR Owners Group Severe Accident Management Guidelines.

Even though there are no plant specific vulnerabilities for GGNS, plant specific observations were made. These are discussed below.

7.2.2 GGNS Internal Fire Observations

The following observations with regard the ability of GGNS to respond to a fire were made during the performance of the GGNS Fire IPEEE. These include the following:

- A large number of fire compartments are unimportant to internal fire risk even without credit for fire detection and/or fire suppression. Essentially all of the compartments that screened (i.e., not subjected to detailed analysis) were screened without taking into account existing fire detection and suppression.
- The control room is the most important room with regard to an internal fire. This observation makes sense as controls for most systems are present in the control room. The most important cabinets in the control room are the P601 and P702 cabinets. Both of these cabinets contain equipment for all three safety related divisions. The control room results are conservative in that a fire in a cabinet is assumed to fail the entire division associated with that cabinet.
- The presence of offsite power cables in a compartment is an important contributor to that compartment's risk. There are several rooms where all sources of off site power are present. However, the areas where all three sources are in close proximity is limited.
- The safety-related switchgear rooms in the control building are also important. The Division 1 switchgear room is most important because of the presence of all offsite power feeds in the compartment.

7.3 High Winds, Floods, and Others

7.3.1 High Winds and Others

A vulnerability to High Winds and Others External Event is defined as a plant non-conformance, with respect to the 1975 SRP, which has a significant contribution to GGNS's CDF. Hazards due to high winds and tornadoes meet the intent/criteria of the 1975 SRP. The threat to plant safety from transportation and nearby facility accidents has not been increased.

7.3.2 Floods

GGNS is in compliance with criteria in the '75 Standard Review Plan. Significant changes to site features were not noted during IPEEE inspections. The design basis evaluations were last revised in 1992; and were performed in accordance with applicable sections of Regulatory Guide 1.59, Rev. 2[20], Regulatory Guide 1.102, Rev. 1[21], and ANS 2.8/N170-1976[22]. HMR 33 and EM-1110-2-1411 are used for determining PMP estimates and temporal distribution. Complete stream blockage is assumed for Stream A. Partial blockage of Stream B is addressed by the restrictions imposed in GGNS Technical Specifications sections 3/4.7.10. With the exception of the enclosure building (designed to loose its roof and siding during design basis tornadoes and wind storms), all safety related structures can withstand roof ponding exceeding the parapet height. Roof drains and overflows are adequate to prevent ponding depths allowing propagating through roof doors, vents, and penetration sleeves. However, the following mechanisms and programmatic changes are being considered to prevent deterioration of site conditions from affecting the analysis.

- Increase maintenance on drainage structures. Maintenance should include cleaning of culverts, concrete repair and removal of vegetation/debris which could obstruct flow. The portion of the south ditch near Culvert No. 1 is currently being well maintained, but additional maintenance on the remaining drainage structures is warranted.

- Plant procedures 05-1-02-VI-1 and 05-1-02-VI-2 currently require plant staff to insure that plant doors are closed during severe weather and in the event of plant flooding (Implicitly including former unit 2 doors). The only former unit 2 doors explicitly required to be closed are the SSW B pump room doors. All other cited doors are Unit 1 doors. Since leakage through an open door could be substantial; revise one or both of these procedures to explicitly include at-grade former unit 2 doors.

- Roof drains and overflows, particularly those on the enclosure building roof, should be periodically inspected to ensure that they are not blocked.

Per NRC request, the new PMP estimates and distribution methods described in HMR's 51 and 52 were evaluated in accordance with applicable sections of Regulatory Guide 1.59, Regulatory Guide 1.102, and ANS 2.8-1992. Appendix B to ANS 2.8-92 discusses the probability of occurrence of the PMP. Based upon portions applicable to the GGNS analysis, the combined probability of inundation of the GGNS site is between 5.0×10^{-6} and 5.0×10^{-8}. Further, the combined probabilities associated with the coincident wind wave activity assumed in the evaluation, would effectively reduce this to between 2.1×10^{-8} and 2.1×10^{-10}.

Applying the new criteria, the bulk of the precipitation would occur over a shorter time frame, and markedly higher rainfall intensities would result. As a result, the GGNS site is not expected to be completely protected against external flooding without making some site modifications. Evaluation reveals that the following site drainage/flood protection improvements would allow for adequate protection of the site against external flooding due to the revised criteria. However, they are not necessarily the only combination of potential changes for consideration. Given the small probability of occurrence for the PMP, as described in the preceding paragraph, the relative cost and benefit for potential improvements will be considered prior to implementation of any physical improvements.

- Remove the wooden foot bridge crossing the northwest ditch near its upstream end.

- Remove the 15" corrugated metal pipe located in the small auxiliary ditch parallel to the northwest ditch (at the same approximate location as the duct bank crossing the northwest ditch). Re-grade the area to provide a gradual transition between the yard upstream, and the auxiliary ditch.

- Re-hang the security fence gates west of the control building to insure that approximately 5" of gap exists between the gate and the road.

- Grade down and remove the access road, the raised berm parallel to the access road, and curbs adjacent to the access road as necessary where they cross Culvert No. 1, such that elevations above the culvert do not exceed 132.7 ft. MSL.

- Replace the C8x11.5 channel forming the flood barrier across the SSW A equipment hatch opening with another member having a minimum depth of approximately 13".

8.0 SUMMARY AND CONCLUSIONS (INCLUDING PROPOSED RESOLUTIONS OF USIs AND GIs)

EOI has performed a complete IPEEE for GGNS. The intent of GL 88-20, Supplement 4 and NUREG 1407 has been met. EOI has expended significant resources developing in-house capabilities in the performance of the IPEEE. The insights and knowledge gained in the process remains with the utility.

8.1 Seismic Analysis

The Seismic Analysis was performed using the EPRI Seismic Margins Methodology for Reduced Scope Plants. The Safe Shutdown Earthquake (SSE) ground response spectra and corresponding in-structure response spectra were used as the Review Level Earthquake (RLW).

The conclusions of the seismic analysis are that Grand Gulf Nuclear Station is seismically rugged and that all components identified in the Safe Shutdown Path have adequately considered the seismic input. All anchorage to these components was found to be rugged.

Only one potential vulnerability to a Seismic event was identified which has been corrected. During the review of pipe supports of Standby Service Water (SSW) piping in the Control Building, it was identified that the grouted condition of the penetration CP9A was not accounted for in the stress analysis of the piping systems. This is an exterior penetration at elevation 105', south wall of the Control Building. Four SSW (System P41) pipes were affected. These pipes traverse from the Auxiliary Building into the Control Building and are the supply and return lines to the Control Room AC units for both A & B loops of the SSW system. The as-found condition had the potential to induce significantly high seismic stresses into the piping between the buildings.

To correct the situation to meet design requirements, the grout was removed and a design change was issued to repair the penetration. The as-found grouted condition was evaluated for operability considerations and was determined not to be an operability concern.

8.2 Fire Analysis:

The overall result of the Fire PRA includes the contributions from all compartments that were not screened in the initial phases of the analysis. The summation of the core damage frequencies from these compartments is the total core damage frequency due to internal fires at GGNS. A listing of these compartments is found in Table 4.6.5-1. The total core damage frequency is 8.76E-06/yr.

The control room is the major contributor to the total core damage frequency with 44% of the total. The high contribution and importance of the control room makes intuitive sense because most of the divisional and important balance of plant equipment is controlled from this location. The remaining individual compartments contribute less than 11% each to the total.

As a group, the control building switchgear rooms are the second major contributor to internal fire risk. These locations collectively account for 22.1% of the internal fire risk. This is expected since power and control feeds for divisional equipment is concentrated in the switchgear rooms. Compartment CC202, the Division 1 Switchgear Room, by itself contributes 10.6% of the total. This room is more important than the other switchgear rooms because of the potential to fail Division 1, some Division 2 equipment and offsite power.

The 93', 119' and 139' elevations of the Auxiliary Building as a group contribute 21% of the internal fire risk. This is primarily due to presence of all three electrical divisions in portions of these compartments.

When considering the types of initiators, the largest contributor is electrical cabinets at 63%. A large portion of this is from the control room. The basic assumption of the cabinet fire causing the failure of the division associated with the cabinet makes this result conservative. The second major contributor by initiator type is transient sources. Transient sources contribute to 24% of the total. This is also somewhat conservative in that it is assumed that the transient source (when suppression fails) will always become a fully developed fire capable of damaging overhead cables or nearby equipment.

One of the main purposes of the IPEEE process is the identification of plant specific vulnerabilities to severe accidents. As stated in the GGNS IPE submittal (Reference 2), GGNS has chosen the general methodology suggested by NEI 91-04 (formerly NUMARC 91-04, Reference 34) for vulnerability screening. This document provides guidance for the closure of severe accident issues in Table 1 of Section 2. As stated in the GGNS IPE, the definition of a plant specific vulnerability is any condition satisfying the CDFs in the top evaluation category of this table. None of the compartments contributing to fire core damage frequency meet this test; therefore, there is no plant specific vulnerability due to internal fire at GGNS.

The largest contributor is the control room with 44% of the internal fire core damage frequency. While this room does met the lowest category of NEI 91-04, Table 1, no specific action is planned at this time. The suggested "Licensee Response" for this category is "Ensure Severe Accident Management Guidance is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure." No action is necessary because, as stated above, the control room results are conservative and current procedures are considered adequate. GGNS will take the Fire IPE results into consideration during the implementation of the BWR Owners Group Severe Accident Management Guidelines.

8.3 High Winds, Floods, and Others Analysis

Potential floodwater elevations at GGNS are controlled by application of the PMP over the site watershed. Three potential vulnerabilities to the design basis evaluation were identified during plant inspection, and corrected by means of the Material Non-Conformance Report process. These challenges consisted of a partially crushed culvert, an abraded PMP door seal gasket, and conditions at a turbine building door which could allow more leakage than previously considered. Several potential programmatic enhancements were noted for consideration as discussed in section 7.3.2. Engineering Report GGNS 91-0055 identifies, in general terms, those areas critical to PMP; and has been revised for the IPEEE to provide additional guidance in the selection of temporary lay-down areas to prevent critical drainage paths from being obstructed for any length of time. Additionally, it was determined that even though the probability of occurrence of the combined PMP event is well below 10^{-6}, some additional vulnerability would result from the application of the revised PMP estimates in HMR 51 and HMR 52. Potential changes for consideration, are as described in section 7.3.2.

The threat to plant safety from transportation and nearby facility accidents has not increased. As reported in section 5.3.4 of those changes that have occured, the threat to the plant has lessened. It is therefore concluded that activities at nearby transportion, industrial, and military facilities will not affect safe plant operation.

Hazards due to high winds and tornadoes meet the intent/criteria of the 1975 SRP. However; during the review it was determined that for some items credit for unit 2 structures as missile barriers, which would not be completed as originally designed, had been taken or the basis for protection was unclear. Therefore, a probabilistic evaluation was performed and determined that the frequency for a tornado generated missile striking these items was 0.77E-8/yr. This low frequency of occurrence is well below the screening criteria in IPEEE.

9.0 REFERENCES

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2. GGNS Individual Plant Examination Summary Report, Entergy Operations, Inc., Submitted via GNRO-92/00157, December, 1991.

3. Grand Gulf Nuclear Station Engineering Report for IPEEE Reduced Scope Seismic Margins Assessment (SMA), GGNS-94-0053, Revision 0.

4. Generic Letter 88-20, Supplement 4, United States Nuclear Regulatory Commission, June 28, 1991.

5. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", United States Nuclear Regulatory Commission, June, 1991.

6. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants", United States Nuclear Regulatory Commission, 1975.

7. NUREG-0831, "Safety Evaluation Report Related to the Operation of Grand Gulf Nuclear Station, Units 1 and 2", United States Nuclear Regulatory Commission, September, 1975.

8. NUREG-0934, "Technical Specifications, Grand Gulf Nuclear Station, Unit No. 1, Docket No. 50-416, Appendix A to License No. NPF-29", United States Nuclear regulatory Commission, October 1984. (As amended February 8, 1994)

9. Generic Letter 89-22, United States Nuclear Regulatory Commission, October 19, 1989.

10. Calculation CC-Q1Y13-93001, Rev. 0, "PMF Hydrographs for Basins A and B (HMR 51 PMP Data)", August 11, 1993.

11. Calculation CC-Q1Y13-93002, Rev. 0, "Backwater Analysis of External Flooding (HMR 51 PMP Data)", October 5, 1994.

12. Calculation CC-Q1Y13-93003, Rev. 0, "In-Leakage Analysis Due to External Flooding (HMR 51 PMP Data)", October 5, 1994.

13. Engineering Report GGNS-93-0001, Rev. 0, "Individual Plant Examination for External Events (External Flooding)", December 5, 1994.

14. Engineering Report GGNS-93-0031, Rev. 0, "An Assessment of the Risk From Lightning Initiators", December 20, 1993

15. GGNS UFSAR

16. NUREG/CR-5042

17. Engineering Report GGNS-93-0017, Revision 0, "Selection of the Safe Shutdown Paths and Equipment For the GGNSD Seismic IPEEE

18 Engineering Report GGNS-93-0048, Revision 0, "High Wind and Tornado Assessment"

19 Calculation CC-Q1111-94004, Revision 0

20 Regulatory Guide 1.59, Rev. 2

21 Regulatory Guide 1.102, Rev. 1

22 ANSI 2.8/N170-1976

23 NUREG /CR-4826

24 ASCE Paper NO. 3269, "Wind Forces on Structures", Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961)

25 ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and other Structures", Committee A58.1, American National Standards Institute (1972)

26 Regulatory Guide 1.76, "Design Bases Tornado for Nuclear Power Plants", April 1974

27 Parkinson, W. J., et. al., "EPRI Fire PRA Implementation Guide", EPRI Report Project 3385-01, Draft, January 1994.

28 Specification M-500.0, Revision 7, GGNS Fire Hazards Analysis.

29 Engineering Report No. GGNS-94-0018, "Grand Gulf Nuclear Station Engineering Report for the Documentaion of Fire Ignition Source Frequencies Used in the Fire Probabilistic Risk Assessment.

30 Fire-induced Vulnerability Evaluation (FIVE), EPRI TR-100370, Final Report, April 1992.

31 Fire Risk Analysis Code, RP3385, Electric Power Research Institute, Palo Alto, CA, January 1994.

32 CAFTA User's Manual," Electric Power Research Institute, Palo Alto, CA, May 29, 1992.

33 Engineering Report No. GGNS-94-0051, "Grand Gulf Nuclear Station Engineering Report: Documentation of Fire Modeling for Fire Probabilistic Risk Assessment.

34 NEI 91-04, Severe Accident Issue Closure Guidelines, Revision 1, Nuclear Energy Institute, Decenber 1994.

35 S.P. Nowlen, Heat and Mass Release for Some Transient Fuel Source Fires, NUREG/CR-4680, October 1986

Table 10.1 Front-line Systems

FUNCTION	PREFERRED PATH	ALTERNATE PATH*
Reactivity Control	CRD	CRD
Pressure Control	SRVs in relief mode for initial transient (Div1)	SRVs in relief mode for initial transient (Div2)
Inventory Control	RCIC HPCS SPMU A	ADS (Div 2) LPCI C
Decay Heat Removal	RHR A in SPC (Hot Shutdown)	RHR B in SDC (Cold Shutdown)

* LOCA is not assumed for the Alternate Path.

Table 10.2 FRONT LINE TO SUPPORT SYSTEM DEPENDENCY MATRIX*

	HPCS	RCIC	CRD	LPCS	LPCI			SSW/ RHR X-TIE	FIRE WATER INJ	ADS	RHR/SDC		RHR/SPC		RHR/CS		PCS	CTMT VENT	SPMU		SLC	
					A	B	C				A	B	A	B	A	B			A	B	A	B
ESF AC DIV I			X	X	X				X		X	X	X		X			X	X		X	
ESF AC DIV II			X			X	X	X	X		X	X		X		X		X		X		X
ESF AC DIV III	X																					
BOP AC			X						X								X					
ESF DC DIV I		X	X	X	X					X	X		X		X			X	X			
ESF DC DIV II		X[9]	X			X	X			X		X		X		X		X		X		
ESF DC DIV III	X																					
BOP DC																	X					
SSW TRAIN A					X						X		X		X							
SSW TRAIN B						X	X	X				X		X		X						
CCW			X																			
TBCW																	X					
PSW																						
CHILLED WTR																						
CIRC WTR																	X					
INST AIR			X[10]						X	X[8]							X	X				
ECCS Rm HVAC	X			X[2]	X [7]	X	X	X			X	X	X	X	X	X						
Steam Tunnel HVAC		X[1]															X					

*Dependencies are indicated for the systems in the column header.

Table 10.3 SUPPORT SYSTEM TO SUPPORT SYSTEM DEPENDENCY MATRIX*

	DG			DGX		SSW			CCW	TBCW	PSW	CHLD WTR	INST AIR	CIRC WTR	DG Rm HVAC	SSW Pump House Vent		ECCS Rm HVAC	STM Tnl HVAC	AC Power			DC Power		
	I	II	III	I	II	A	B	C								A	B			I	II	III	I	II	III
ESF AC DIV I						X					X		X		X	X		X							
ESF AC DIV II							X		X[3]		X		X		X	X	X	X							
ESF AC DIV III				X	X			X							X			X							
BOP AC									X	X	X	X	X	X					X						
ESF DC DIV I	X					X					X		X												
ESF DC DIV II		X					X		X		X		X												
ESF DC DIV III			X	X	X			X																	
BOP DC									X[11]	X[12]	X[11]	X[12]	X[13]	X											
SSW TRAIN A	X																	X							
SSW TRAIN B		X							X[4]				X					X							
SSW TRAIN C			X															X							
TBCW													X												
PSW									X	X		X													
CHILLED WTR																			X						
INST AIR											X	X													
DG Rm HVAC	X	X																							
SSW Pump House Vent A						X[6] X[5]																			
SSW Pump House Vent B							X[5]																		
Switchgerr & Batt Rm Cooling Div I																				X			X		
Switchgerr & Batt Rm Cooling Div 2																					X			X	

*Dependencies are indicated for the systems in the column header.

Notes for Tables 10.2 and 10.3

1. Delayed time dependency. The RCIC pump will operate for 30 minutes after the steam leak detection signal is initiated. No isolation occurs during SBO due to loss of power to the timer.

2. Delayed time dependency. LPCS Pump will operate approximately 10 to 12 hours without room cooling.

3. Train B pump.

4. SSW Train B is alternate source of cooling water under certain conditions.

5. Delayed time dependency. SSW pumps will fail approximately 2.5 hours after loss of HVAC.

6. Delayed time dependency. SSW pumps will fail approximately 2.5 hours after loss of HVAC. No dependency if SSW A pump is not operating.

7. Deleted

8. Backup by accumulators.

9. Required for redundant actuation logic and Level 8 protection instrumentation.

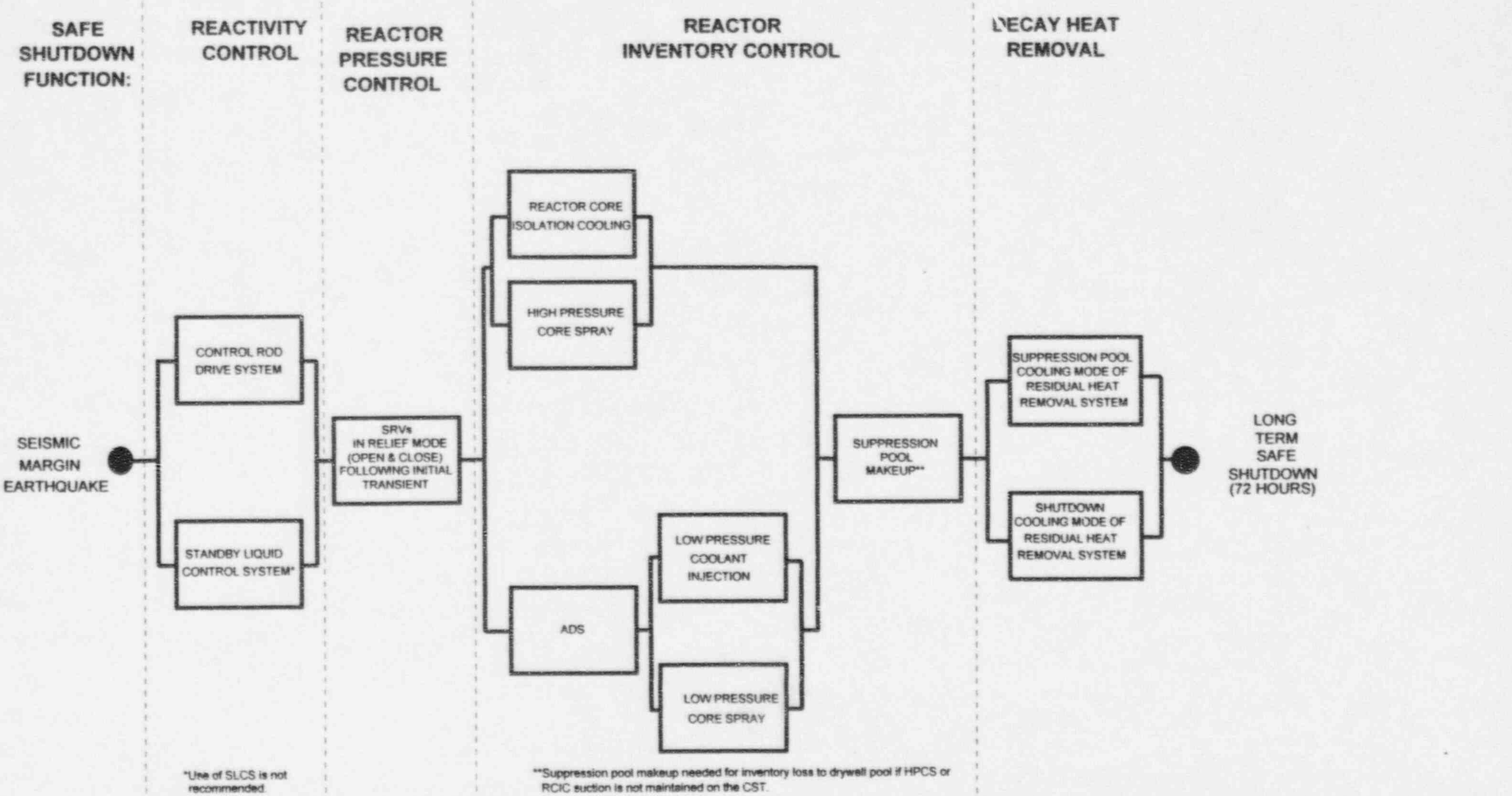
10. Required for enhanced flow mode only.

11. DC power is required to start the pumps. However, the pumps are normally operating and DC power is not modeled.

12. DC power is required to start standby pump.

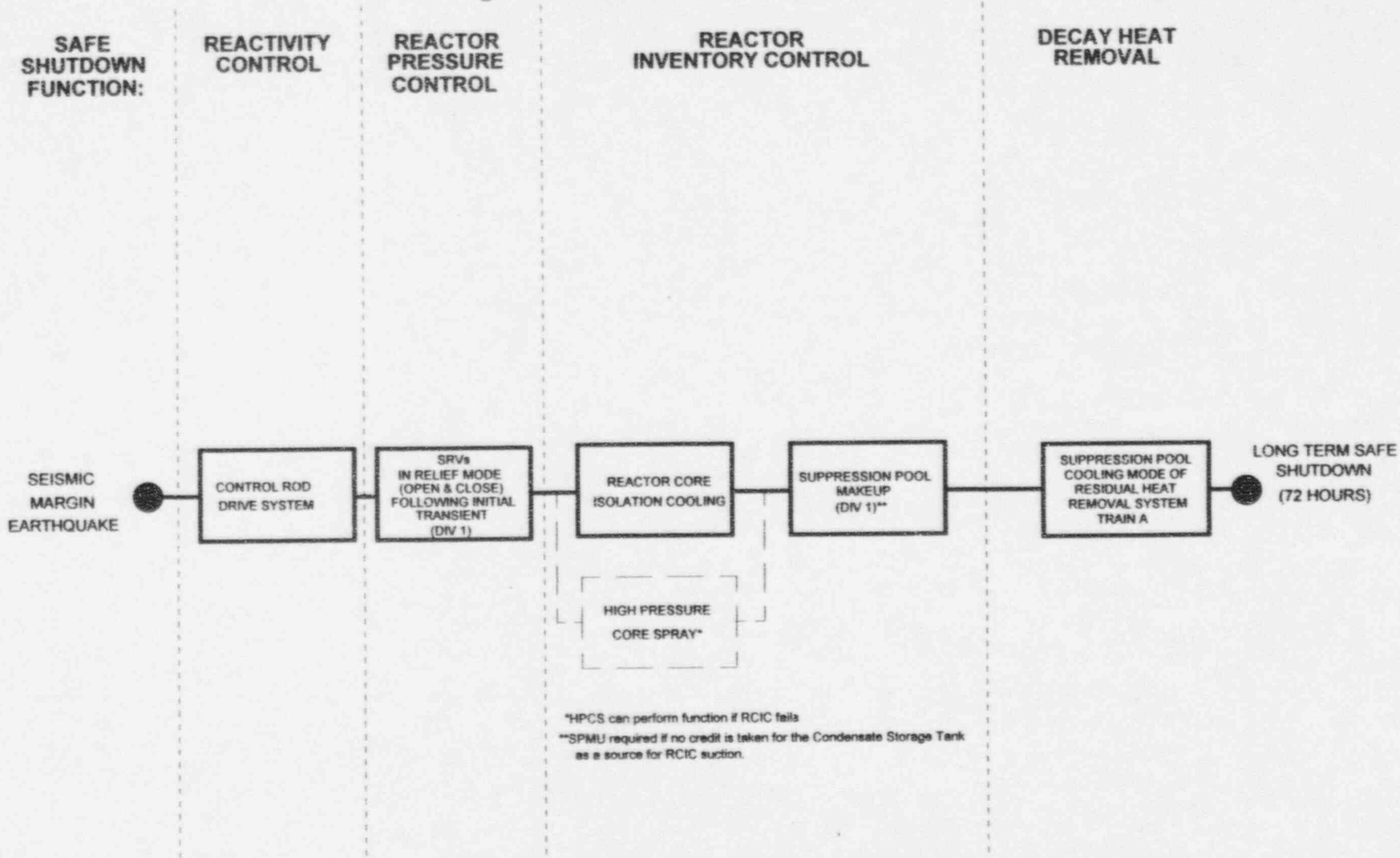
13. DC power is required to start the normally operating compressor. Therefore, DC power is not modeled.

Figure 11.1 GGNS Success Path Logic Diagram



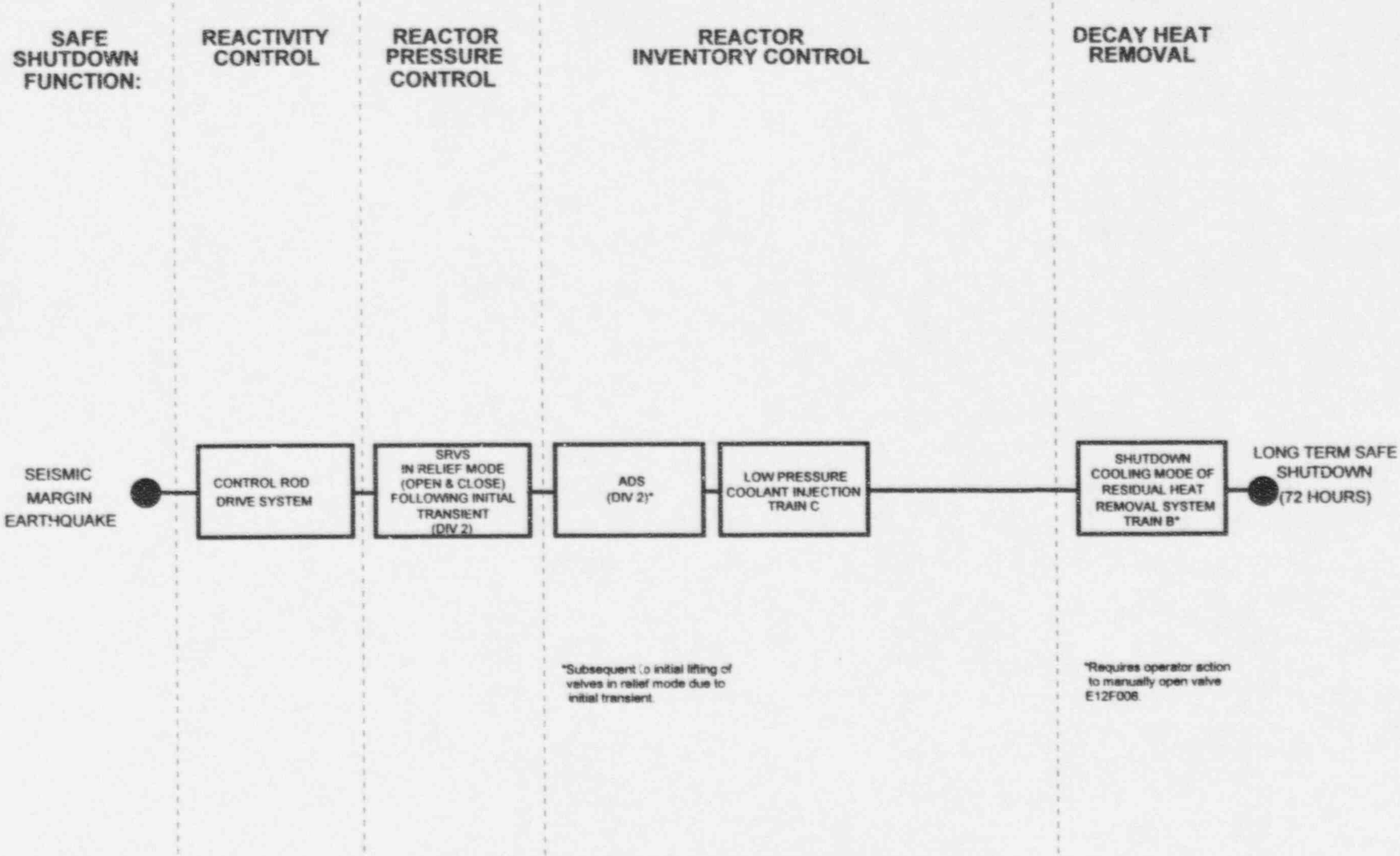
GGNS PROPOSED SUCCESS PATH LOGIC DIAGRAM

Figure 11.2 GGNS Preferred Success Path



GGNS IPEEE SEISMIC MARGIN ASSESSMENT SUCCESS PATH--PREFERRED

Figure 11.3 GGNS Alternate Success Path



GGNS IPEEE SEISMIC MARGIN ASSESSMENT SUCCESS PATH--ALTERNATE (ASSUMES NO LOCA)

SAFETY EVALUATION APPLICABILITY REVIEW FORM

A) Document Evaluated: Engineering Report GGNS-94-0054, Revision 1

B) Description of the Proposed Change: Revision 1 of this report incorporates the Internal Fire Events assessment into a summary report for submittal to the NRC documenting completion of the Individual Plant Examination for External Events (IPEEE). All other events are addressed in Revision 0 of this report. This Applicability Review addresses only those changes associated with the Internal Fire Events assessment since no changes are made to the assessment of other events evaluated in Revision 0. IPEEE assessments were requested from each utility in Generic Letter 88-20, Supplement 4; and are performed in accordance with the guidelines presented in NUREG 1407.

PRE-SCREENING

Check the applicable boxes below. If any of the boxes are checked, neither a safety evaluation applicability review nor a safety evaluation is necessary and steps C, and D may be skipped. The preparer and reviewer must sign at the bottom of the form.

___ The change is editorial only.
___ 10CFR50.54 applies to the change instead of 10CFR50.59.
___ An approved safety evaluation covering all aspects of this subject already exists.
Reference SE# ____________
___ The change, in its entirety, has been approved by the NRC.
Reference ____________
___ The change is an FSAR change that meets the exclusion criteria outlined in Site Directive G4.803.

Safety Evaluation Applicability Review

If any of the following questions are answered "yes", then a full 50.59 Safety Evaluation must be completed.

C) Does the proposed change or activity represent a change to the Technical Specifications?
YES ___ Explain:
NO XX Operating License Amendment No. 82 relocated the Fire Protection Technical Specifications to UFSAR Appendix 16A and Technical Requirements Manual (TRM). Revision 1 of the above report incorporates a summary of risk assessment associated with fire events. The Internal Fire Assessment uses, as a basic framework, NUREG/CR-2300 and applicable requirements set forth in NUREG 1407. Although observations were made as a result of this assessment, no changes were recommended or implemented as a result of this revision. Therefore, there will be no change to TS or the Bases for any Technical Specifications or the TRM.

D) Does the proposed change or activity represent:

(1) A change to the facility which alters, or has the potential to alter, the information, operation, function or ability to perform the function of a system, structure or component described in the SAR?

YES ___ Explain:
NO XX As stated above, no changes resulted from this assessment of fire events; therefore, no change has been made to the facility which alters, or has the potential to alter, the information, function or ability to perform the function of a system, structure or component described in the SAR.

(2) A change to a procedure which alters, or has the potential to alter, a procedure described, outlined or summarized in the SAR?

YES ___ Explain:
NO XX Revision 1 of the above report incorporates a summary of risk assessment associated with fire events. Various GGNS procedures were reviewed and observations were made as a result of this assessment; however, no changes were recommended or implemented as a result of this assessment. Therefore, there is no change to a procedure which alters, or has the potential to alter, a procedure described, outlined or summarized in the SAR.

(3) A test or experiment not described in the SAR or which requires that a system be operated in an abnormal manner that is not described or previously analyzed in the SAR?

YES ___ Explain:
NO XX No test or experiment resulted from this assessment of fire risk.

PREPARER Michael R. Cumbest — Tech Spec IV, Sr Lead — 11/9/95
Michael R. Cumbest — Job Title — Date

REVIEWER Gary W. Smith — Engineer, Sr Staff — 11/9/95
Gary W. Smith — Job Title — Date

If the preparer performed an applicability review, the reviewer should check below to indicate by which means the independent review reached the same conclusions.

X Reviewed the applicability review documentation.

__ Completed an independent applicability review.

__ Performed a verbal review with the preparer.