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Vol. 3

Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage

Prepared by
J. Lambright, S. Ross, J. Lynch, J. Yakle

Sandia National Laboratories
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Sandia Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

This report presents the details of the analysis of core damage frequency due to fire during shutdown Plant Operational State 5 at the Grand Gulf Nuclear Station. Insights from previous fire analyses (Peach Bottom, Surry, LaSalle) were used to the greatest extent possible in this analysis. The fire analysis was fully integrated utilizing the same event trees and fault trees that were used in the internal events analysis.

In assessing shutdown risk due to fire at Grand Gulf, a detailed screening was performed which included the following elements:

a) Computer-aided vital area analysis

b) Plant inspections

c) Credit for automatic fire protection systems

d) Recovery of random failures

e) Detailed fire propagation modeling

This screening process revealed that all plant areas had a negligible (< 1.0E-8 per year) contribution to fire-induced core damage frequency.

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Foreword

(NUREG/CR-6143 and 6144)
Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analyses that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Volume 1: Summary of Results
Volume 2: Analysis of Core Damage Frequency from Internal Events for Plant Operational State 5 During a Refueling Outage
- Part 1: Main Report
  - Part 1A: Sections 1 - 9
  - Part 1B: Section 10
  - Part 1C: Sections 11 - 14
- Part 2: Internal Events Appendices A to H
- Part 3: Internal Events Appendices I and J
- Part 4: Internal Events Appendices K to M

Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage
Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
- Part 1: Main Report
- Part 2: Supporting MELCOR Calculations

Foreword (Continued)

For Surry:

NUREG/CR-6144 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1

Volume 1: Summary of Results
Volume 2: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations
- Part 1: Main Report
  - Part 1A: Chapters 1 - 6
  - Part 1B: Chapters 7 - 12
- Part 2: Internal Events Appendices A to D
- Part 3: Internal Events Appendix E
  - Part 3A: Sections E.1 - E.8
  - Part 3B: Sections E.9 - E.16
- Part 4: Internal Events Appendices F to H
- Part 5: Internal Events Appendix I

Volume 3: Analysis of Core Damage Frequency from Internal Fires During Mid-loop Operations
- Part 1: Main Report
- Part 2: Appendices

Volume 4: Analysis of Core Damage Frequency from Internal Floods During Mid-loop Operations
Volume 5: Analysis of Core Damage Frequency from Seismic Events During Mid-loop Operations
Volume 6: Evaluation of Severe Accident Risks During Mid-loop Operations
- Part 1: Main Report
- Part 2: Appendices

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Executive Summary

A detailed fire risk assessment at shutdown has been performed for Grand Gulf Nuclear Station. Although the plant may be in various Plant Operational States (POS) at shutdown it was decided to examine one POS in detail. POS 5 was chosen for this analysis. POS 5 can be entered with decay heat as high as 0.9% of full power (34 MW); however, after refueling decay heat will not be higher than 0.16% of full power (6MW) and this is the situation during hydro testing. For non-hydro conditions the plant state can be maintained with Residual Heat Removal (RHR), or with Alternate Decay Heat Removal System (ADHRS) 24 hours after shutdown; during hydro testing, shutdown cooling is with Reactor Water Cleanup (RWCU) letdown and Control Rod Drive (CRD) makeup, but during hydro testing RHR and ADHRS are also available. Recirculation can be either forced or natural. Initial level can be normal when on forced recirculation, raised when on natural circulation, or the vessel can be full as is the case during hydro testing. Containment can be open or closed. Suppression Pool water level can be 18 feet 4 inches, 12 feet 8 inches, or empty if 170,000 gal of Condensate Storage Tank (CST) water is available to HPCS. Numerous components can be out of service for maintenance in POS 5; our model assumes all of train A is unavailable due to maintenance in POS 5. The techniques used in this assessment have made full use of insights gained during the past 15 years in fire risk assessment. This methodology utilized previously completed internal event fault/event tree models. Thus, the level of detail of the fire analysis is consistent with the level of detail of the internal events analysis.

A detailed screening analysis was performed which showed most plant areas had a negligible contribution to fire induced core damage frequency. For four fire zones, a detailed fire propagation analysis was performed. There were no plant areas which were found to have a contribution to core damage frequency of greater than 1.0E-8/ry.

The fire-induced core damage frequency is lower than other fire risk assessments at power due to a number of factors. First, this plant operational state represents only three percent of the time at shutdown and shutdown fire frequencies are similar to those at power. This provides an immediate reduction in core damage frequency. Second, even if active electro-mechanical safety-related equipment is damaged by fire, an initiating event may not necessarily occur. For instance, for the loss of TBCW (Turbine Building Cooling Water) initiator to result from fire- related damage, multiple operational pumps must fail. These pumps and their associated cabling have sufficient separation such that it is highly unlikely that a single fire could lead to failure of all pumps. Many initiating events at shutdown were screened because of physical separation criteria. Even for the unscreened initiating events, very few fire zones were found to be applicable because of physical separation criteria. Also, relative to other plants, Grand Gulf utilizes more automatic fire protection systems in critical safety-related areas which in turn reduces the probability of damage due to a fire. Therefore, taking credit for physical separation of safety-related functions, automatic fire protection systems, lower fire initiating event frequencies, and manual fire suppression most initiating events at shutdown and many fire zones were screened from further analysis.

For those remaining initators and respective fire zones, a detailed fire propagation analysis was performed. It was found that only in very limited areas could fire damage result in both the initiating event and other fire-related failures which were required to lead to core damage. Even in these situations other random failures (non fire-related) were also required to lead to core damage. Therefore, when taking into account reduction in fire frequency due to the limited area of influence and other random failures which were required to lead to core damage, all remaining fire scenarios were found to be less than 1E-08 per reactor year.

In all areas additional random failures of equipment (damage not related to the fire itself) were required to occur in order to obtain core damage. Adequate separation of equipment (and/or) cabling between redundant functions and the presence of automatic fire suppression systems had the effect of reducing core damage frequency for those areas.

1. Introduction

1.1 The Grand Gulf Shutdown Fire Analysis

This report describes the shutdown fire risk analysis for the Grand Gulf nuclear plant. Although this risk assessment is intended to be complete, budget and schedule considerations have dictated the use of past probabilistic risk assessment (PRA) experience (Refs. 1.1,1.2), generic data bases, and other defensible simplifications to the maximum extent possible.

Besides simplification in terms of cost reduction and minimization of execution time, the simplified fire risk assessment described here will also meet the following additional objectives:

a. To be consistent with internal event analyses. The same event trees/fault trees and random, common mode failure, and test/maintenance data are used.

b. To be transparent. A standard report format should enable the reader to reproduce any of the results.

c. To be realistic. Best estimate data and models are used as much as possible. Important plant specific failure modes have been analyzed.

d. To be comparable. The fire analysis is directly comparable with internal event analyses due to common generic data, common methodology, common level of detail, and presentation of results.

1.2 Steps in the Analyses

1.2.1 Plant Walkdown and Data Gathering

T'.e Grand Gulf shutdown fire analysis began with a plant visit in June 1991. The initial visit served as the basis for the initial plant information request submittal. Prior to the first plant visit, the fire analysis team was briefed by the internal events systems analyst as to the general character of safety systems, support systems, system success criteria and critical interdependencies identified to date.

The team consisted of the following personnel:

PRA Project Manager - J. Lambright (Sandia National Laboratories)
Team Leader - J. Lambright
Fire Propagation Analyst - S. Ross (SEA, Inc.)
Fire Brigade Analyst - M. DiMascio (Solutions Engineering, Inc.)

During the initial walkdown, team members visited all areas containing safety or support equipment. The two full days spent on site were considered adequate for this initial visit. At the completion of this initial visit, the following information had been obtained:

a. For each room or compartment containing essential safety equipment, identification of fire sources (power cables, pump motors, solvents and other transient combustibles, etc.) locations of fire barriers, fire/smoke detectors, separation of cable trains, etc., and a list of equipment in the room.

b. Plant layout drawings which formed the basis for division of the plant into fire areas.

c. A list of key plant personnel to be contacted later if specific questions arose.

Following the initial plant visit, a list of drawings and documentation needed for further study was prepared and sent to the designated plant contacts.

Subsequent visits to the plant were made by the fire analysis personnel to conduct cable path tracing and verification and to set up physical models for the fire propagation analysis.

1.2.2 Fire Risk Assessment Methodology

Based on an update to the nuclear power plant operating experience in Reference 1.3, it has been observed that typical nuclear power plants will have three to four significant fires, on average, over their operating lifetime. Previous probabilistic risk assessments (PRAs) at power have shown that fires are often a significant contributor to the overall core damage frequency (Ref. 1.4), contributing anywhere from 7 percent to 50 percent of the total core damage frequency (considering contributions from internal, seismic, flood, fire, and other events). Because of this potentially significant core damage contribution, it is important that fires be examined in detail. An overview of the simplified fire PRA methodology, which is outlined in Reference 1.5, is as follows:

A. Initial Plant Visits

Based on the internal event and seismic analyses, the general location of safety-related components of the systems of interest is known. The plant visit provides the analyst with a means of verifying the physical arrangements in each of these areas. The analyst completes a fire zone checklist which aids in the screening analysis and in the quantification of risk.

The second purpose of the initial plant visit is to confirm with plant personnel that the documentation being used is in fact the best available information and to get clarification about any questions that might have arisen in a review of the documentation.

Also, a thorough review of fire fighting procedures is conducted. This review is performed to determine the probability of manual suppression in any given length of time for all critical plant areas. The results of this analysis are given in Appendix A.

B. Screening

It is necessary to select those fire locations within the power plant having the greatest potential for producing risk-dominant accident sequences. The objectives of location selection are somewhat competing and should be balanced in a meaningful risk assessment study. The first objective is to maximize the possibility that all important locations are analyzed, and this leads to the consideration of a potentially large number of candidate locations. The second objective is to minimize the effort spent in the quantification of event trees and fault trees for fire locations that turn out to be unimportant. A proper balance of these objectives is one that results in an ideal analysis allocation of resources and efficiency of assessment.

The screening analysis is comprised of:

1. Identification of potentially important fire areas. Fire areas which have either safety-related equipment or power and control cables for that equipment were identified as requiring further analysis. This group of fire areas is briefly described in Section 3.2. All critical safety components within these fire areas are given in Appendix B.

2. Screen fire areas for probable fire-induced initiating events. Determination of the fire frequency for all plant locations and determination of the resulting fire-induced initiating events and "off-normal" plant states are delineated in Sections 3.4 and 3.5, respectively.

3. Screen fire areas both on order and frequency of cut sets. A cut set is defined as a minimal combination of fire-related and random failures which lead to core damage.
4. Each fire area remaining is numerically evaluated and culled on frequency. The screening methodology (Section 3.5) describes how reduction of the initial group of locations from Section 3.2 to the four remaining which underwent detailed fire propagation modeling was accomplished.

C. Final Quantification

After the screening analysis has eliminated all but the probabilistically significant fire areas, quantification of dominant cut sets is completed as follows:

1. The temperature response in each fire area for each postulated fire is determined.

2. Fire fragilities are computed. The latest version of the fire growth code COMPBRN (Ref. 1.6) was used to calculate fire propagation and equipment damage.

3. The probabilities of barrier failure for all remaining combinations of adjacent fire areas are assessed. A barrier failure analysis was conducted for those combinations of two adjacent fire areas which, with or without additional random failures, remained after the screening analysis. The methodology to assign barrier failure probability to the fire area combinations is described in Section 3.7.

4. A recovery analysis is performed. In a similar fashion as in the internal events analysis, recovery of non-fire related random failures was addressed. Appropriate modifications to recovery probabilities were made as described in Section 3.8.

5. An uncertainty analysis is performed to estimate error bounds on the computed fire-induced core damage frequencies. The IRRAS code (Ref. 1.7) is utilized in the uncertainty analysis as described in Section 3.9. Since all remaining fire areas were found to have a core damage frequency contribution of <1.0E8/y, no uncertainty analysis was performed for this study.

In Section 3.10, a detailed description of all fire scenarios which survived the initial screening process and their associated fire areas is given. Descriptions of all factors used in the final quantification of these fire areas are delineated.

1.3 References

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1.6 V. Ho, et al., COMPBRN IIIe: An Interactive Computer Vode for Fire Risk Analysis, EPRI NP-7282, May 1991.

1.7 K. D. Russell et al., Integrated Reliability and Risk Analysis System (IRRAS) Version 4.0, NUREG/CR-5813, EGG-2664, EG&G Idaho, January 1992.

2. Plant Description

2.1 Introduction

The Grand Gulf Nuclear Station has one Boiling Water Reactor (BWR) of 1250 megawatts (electrical) capacity and is housed in a Mark III containment. The plant is operated by Entergy Incorporated, is located in Port Gibson, Mississippi, and began commercial operation in July 1985.

2.1.1 Selection and Characterization of Plant Operational State (POS) 5

This section discusses the POSs at Grand Gulf Nuclear Station and describes the characteristics of POS 5 which was selected for this analysis. A POS is a plant condition for which the status of plant systems (operating, standby, unavailable) can be specified with sufficient accuracy to model subsequent accident events. A POS is not identical to a mode or operating condition, but is defined based on operating conditions. The operating conditions at Grand Gulf are defined as follows:

(1) OC 1, Power Operation: Mode Switch in Run, Any Temperature

(2) OC 2, Startup: Mode Switch in Startup/Hot Standby, Any Temperature

(3) OC 3, Hot Shutdown: Mode Switch in Shutdown, Temperature Greater than 200°F

(4) OC 4, Cold Shutdown: Mode Switch in Shutdown, Temperature 200°F or Lower

(5) OC 5, Refueling: Fuel in Vessel with Head Detensioned or Removed, Mode Switch in Shutdown or Refuel, Temperature 140°F or Lower.

The POSs are then defined based upon the operating conditions stated above as follows:

(1) POS 1 consisting of: OC 1 and OC 2 with pressure at rated conditions (about 1000 psig) and thermal power no greater than 15%

(2) POS 2 consisting of: OC 3 from rated pressure to 500 psig

(3) POS 3 consisting of: OC 3 from 500 psig to where RHR/SDC is initiated (about 100 psig)

(4) POS 4 consisting of: OC 3 with the unit on RHR/SDC

(5) POS 5 consisting of: OC 4 ($T \leq 200$ degrees F) and OC 5 until the vessel head is off

(6) POS 6 consisting of: OC 5 with the head off and level raised to the steam lines

(7) POS 7 consisting of: OC 5 with the head off, the upper pool filled, and the refueling transfer tube open.

Grand Gulf can be in POS 5 with a variety of conditions that affect the availability of mitigating equipment should an accident occur. POS 5 can be entered with decay heat as high as 0.9% of full power (34 MW); however, after refueling decay heat will not be higher than 0.16% of full power (6MW). This is the situation during hydro testing. For the non-hydro situation the plant state can be with RHR, or with ADHR 24 hours after shutdown; during hydro testing, shutdown cooling is provided by the combination of RWCU letdown and Control Rod Drive (CRD) makeup, but during hydro testing RHR and ADHR are also available. Recirculation can be either forced or natural. Initial level can be normal when on forced recirculation, raised when on natural circulation, or the vessel can be full as is the case during hydro testing. Containment can be open or closed. Suppression Pool water level can be 18 feet 4 inches, 12 feet 8 inches, or empty if 170,000 gal of CST water is available to HPCS. Numerous components can be out of service for maintenance in POS 5; our model assumes all of train A is unavailable due to maintenance during operation in POS 5.

2.2 Description of Plant Systems

The following sections provide the system descriptions and system models of the major front line and support systems identified as important to mitigating the effects of fire-induced plant transients during POS 5 (Ref. 2.1). Event trees and component fault trees were developed by the internal events analysts. This study utilizes the same event trees, fault trees, and accident sequences developed during the internal events analysis to ensure consistency (Ref. 2.2). The discussion of the systems that follow include:

a. A brief functional description of each system with reference to the one-line diagrams that were developed to indicate which components were included in the model.

b. Interfaces and dependencies between the frontline systems and the support systems.

2.2.1 High Pressure Core Spray System (HPCS)

In POS 5 the function of the HPCS system is to provide coolant makeup to the reactor vessel in order to maintain proper water level and/or flood the reactor. The HPCS system consists of a single train with motor-operated valves and a motor driven pump. Suction is taken from either the Condensate Storage Tank (CST) or the suppression pool. Injection to the reactor vessel is via a spray ring mounted inside the core shroud. The pump is capable of delivering 550 gpm against a reactor pressure of 1177 psig and a full flow of 7115 gpm against a reactor pressure of 200 psig. The total maximum pump run out flow is 9100 gpm. A simplified schematic of the HPCS is provided by Figure 2.1. Major system components are represented with valves shown in their normal standby position.

The HPCS system is automatically initiated and controlled. However, operator intervention is required to throttle flow to prevent the HPCS injection valve from opening and closing in response to the reactor vessel level. The operator may also be required to manually start the system, if an automatic start failure occurs.

The success criterion for the HPCS system is injection of flow to the reactor vessel.

The HPCS system major dependencies are DC control power for initiating the actuation relay logic and HPCS pump breaker, AC power for operating the HPCS pump and valves, and HPCS pump room cooling.

2.2.2 Control Rod Drive (CRD) System

The CRD hydraulic system was modeled as a backup source of high pressure injection.

The CRD pumps take suction from the condenser hotwell makeup/reject line. Makeup to the condenser hotwell is provided by the CST. Excess condensate from the condenser is rejected to the CST by the condensate system. From the condenser hotwell makeup/reject lines, water flows to the CRD pumps through a pump backwash suction filter and one of two pump suction filters. A simplified schematic of the CRD system is provided by Figure 2.2.

The CRD pumps, together, can achieve a flow rate of approximately 238 gpm with the reactor at 1103 psia. Each pump is provided with a minimum flow line which recirculates 20 gpm back to the CST. This minimum flow line prevents the pump from operating at shutoff head where the pump would overheat and possibly be damaged.

Two discharge paths are provided for the CRD pumps. The first path is through the Hydraulic Control Units' (HCUs) cooling headers. Flow is controlled by one of two air operated control valves. When instrument air is lost, the control valves fail "as is". The second path is through the HCU charging headers. This path is upstream of the control valves and fails open on loss of air. However, with both CRD pumps running and the reactor at nominal pressure, the second discharge path restricts flow, by means of an orifice, to approximately 165 gpm. This flow rate is assumed insufficient for core cooling and thus no credit is taken for this discharge path.

Normally one CRD pump is running with the suction and discharge valves to the standby pump being open. Should the operator be required to realign the CRD system as a source of early high pressure injection, the standby CRD pump must be placed into operation and one air operated control valve must be fully opened to achieve sufficient flow to the reactor vessel.

The CRD success criteria require that both pumps be running and the HCU cooling header discharge path be available when CRD is required at the start of the accident as the only coolant makeup source. However, when coolant makeup has been provided for a period of time and then lost, only one CRD pump is required.

CRD Pump A is powered from Division 1 4160 V AC Bus 15 AA with control and actuation power supplied by Division 1 125 V DC Bus 11 DA. CRD Pump B is powered from Division 2 4160 V AC Bus 16AB with control and actuation power supplied by Division 2 125 V DC Bus 11 DB.

2.2.3 Suppression Pool Makeup (SPMU) System

The SPMU system provides water from the upper containment pool to the suppression pool following a LOCA. Water which gravity flows from the upper containment pool to the suppression pool is of sufficient quantity to keep the uppermost drywell vents covered for most conceivable accidents.

The SPMU system consists of two lines which penetrate the side walls in the separator storage area of the upper containment pool. These lines are routed down to the suppression pool on either side of the steam tunnel. A simplified schematic of the SPMU system is provided by Figure 2.3.

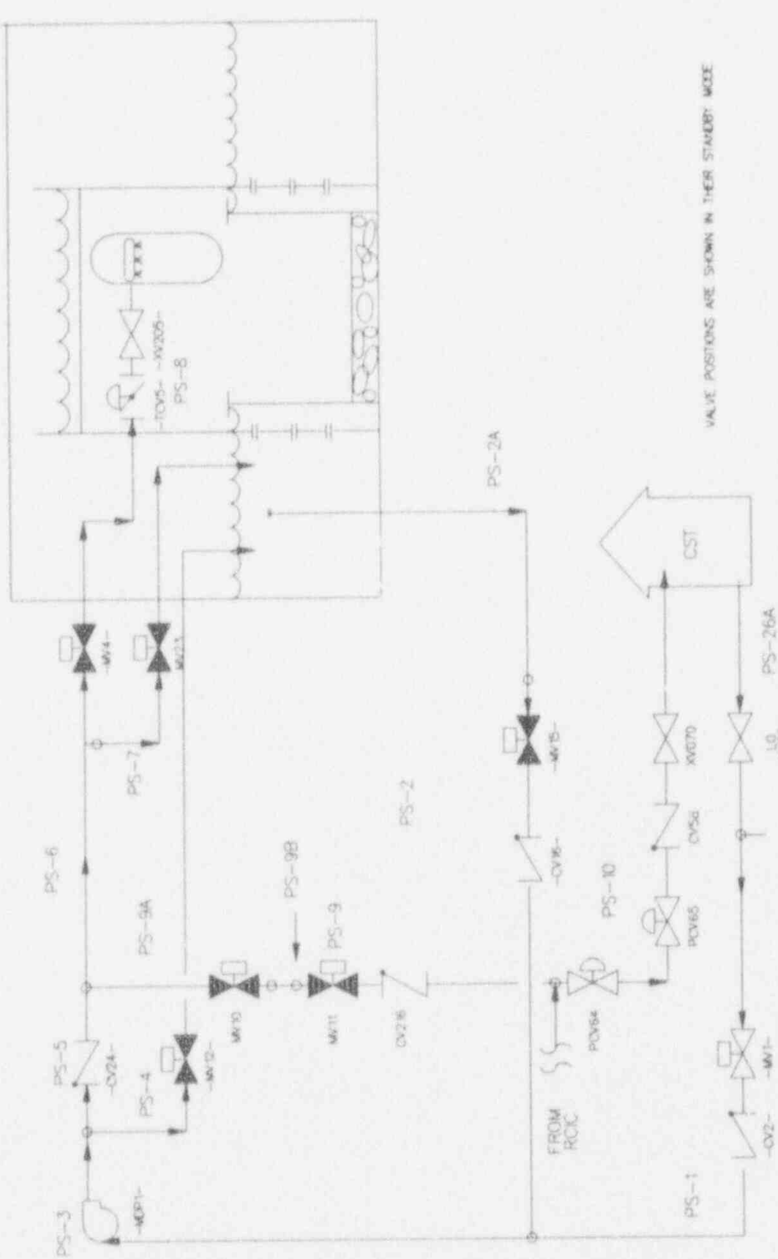

Figure 2.1 HPCS System Schematic

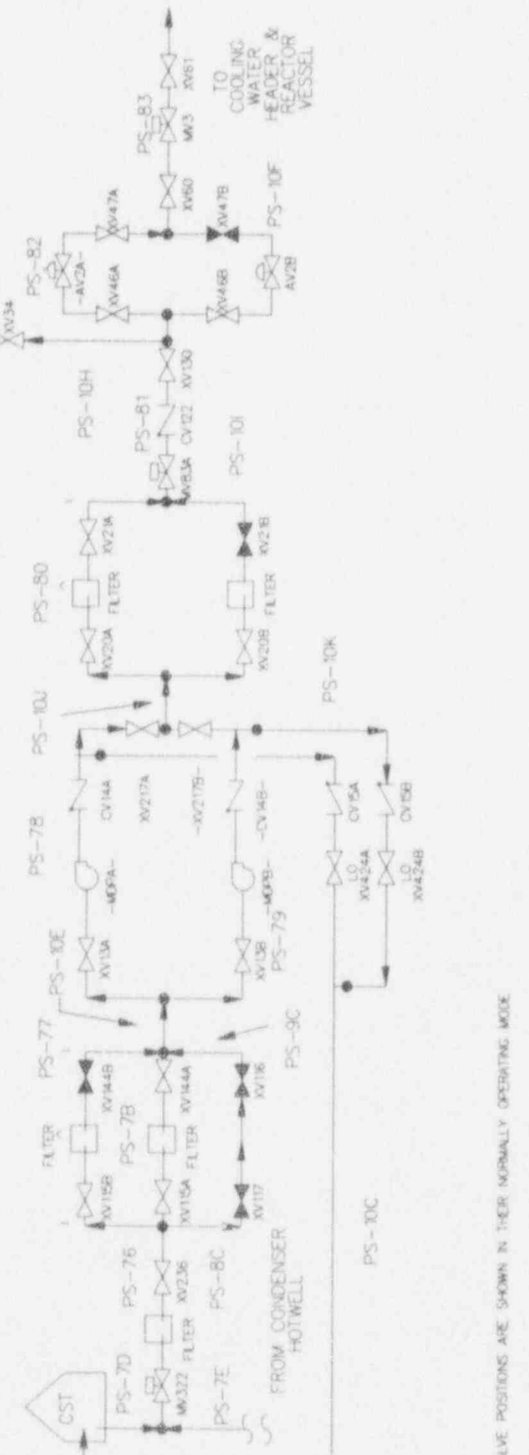

Figure 2.2 CRD System Schematic

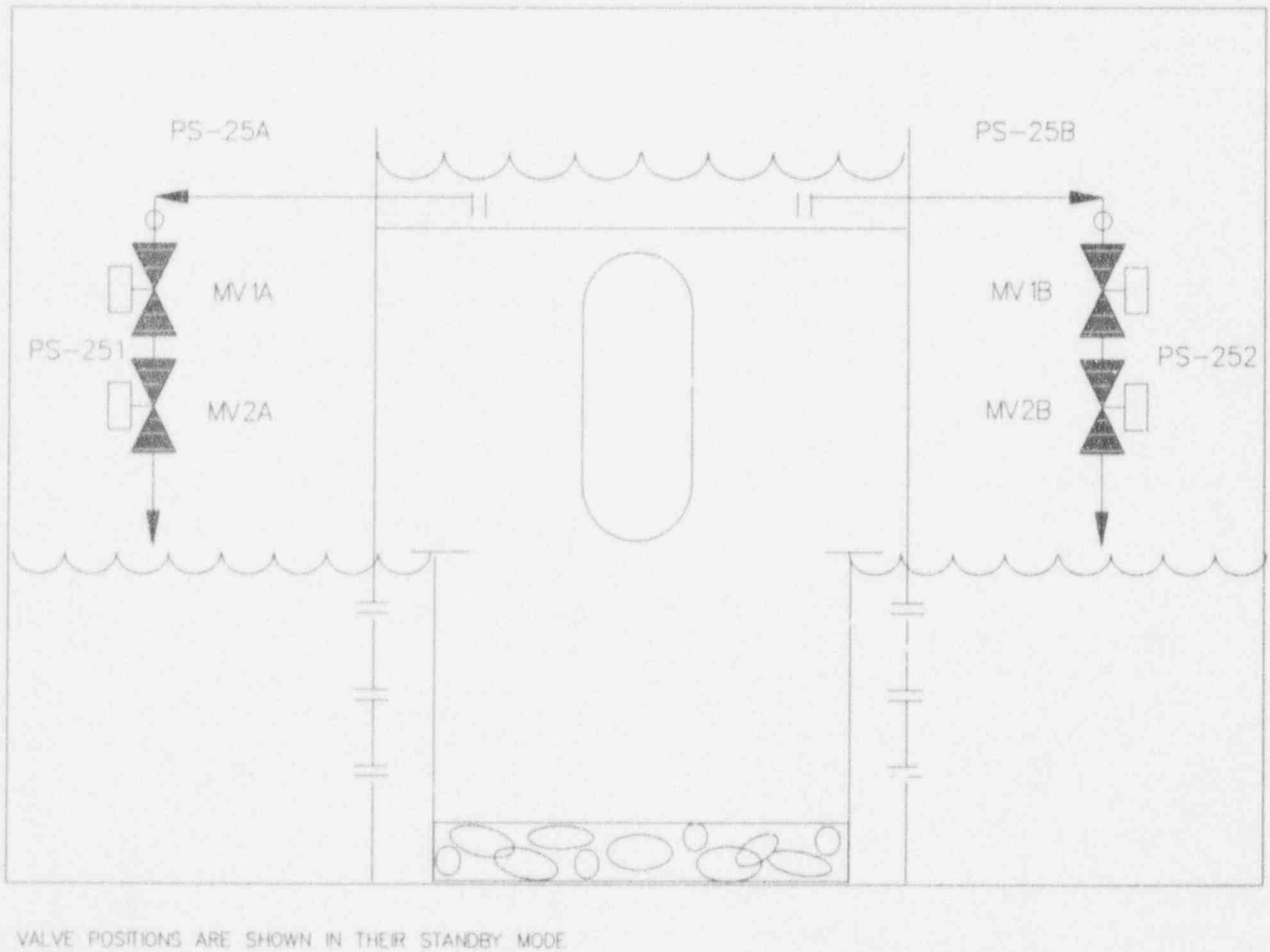

Figure 2.3 SPMU System Schematic

The pool makeup lines each have two normally closed, motor-operated butterfly valves in series. All motor-operated valves are powered from onsite emergency power sources maintaining divisional separation and redundancy.

The upper pool is dumped by gravity flow when the valves receive a divisionally separate but simultaneous signal to open. The open signal for each valve division is generated in any of three ways:

a. By low-low suppression pool level, providing a LOCA signal has been generated or the Emergency Core Cooling System (ECCS) has been manually initiated. This ensures adequate water volume in the suppression pool to keep the suppression pool vents covered for all break sizes.

b. By a timer, thirty minutes after a LOCA signal has been generated. This ensures an adequate long term heat sink is available regardless of break size.

c. By manual initiation, provided a LOCA signal is present or the ECCS has been manually initiated.

In addition, in order to actuate each SPMU valve by any of the three methods listed above, the mode selector handswitch for each division must be in AUTO position and the reactor mode switch must not be in REFUEL position.

The SPMU system requires electrical power for operation. The two redundant SPMU lines are each powered from separate emergency electrical buses. The Train A valves are powered by emergency AC Division 1 MCC 15B21 while the Train B valves are powered by AC Division 2 MCC 16B41. The initiation logic for Train A and B is powered by 125 V DC Divisions 1 and 2, respectively.

2.2.4 Condensate (CDS) System

Credit for the condensate system as a low pressure injection system is taken in this study. The condensate system has three main condenser units, three condensate pumps, three condensate booster pumps, three strings of four low pressure heaters, a condensate drain tank and associated valves, piping, instrumentation, and controls. The condensate system supplies water to the reactor vessel through the feedwater startup valve AV513. A simplified schematic of the condensate system for use as a low pressure injection system is shown in Figure 2.4. The success criteria for the condensate system in POS 5 is one of three condensate pumps operating with a flow path to the reactor through the feedwater start-up flow control valve.

The system dependencies for the condensate system as a low pressure injection system are less demanding than during normal plant operation. The systems required to maintain condenser vacuum are not required since the condensate system is unaffected by loss of condenser vacuum. In addition, parts of the condensate system (e.g., the low pressure heaters) are not required to function. The condensate pumps are powered by non-safety 4.16 kV buses. Power to condensate system motor-operated valves is also provided by non-safety related buses (480 V). The instrument air system is required to supply air to the condenser makeup valve and also to open the feedwater startup valve. Makeup to the condenser is provided by the condensate and refueling water storage and transfer system.

2.2.5 Low Pressure Core Spray (LPCS) System

The function of the LPCS System is to provide coolant to the reactor vessel during accidents in which vessel pressure is low. The LPCS system is a single train system consisting of motor-operated and manual valves and a motor-driven pump. The LPCS pump is rated at 7115 gpm with a discharge head of 319 psig. The LPCS pump takes water from the suppression pool through strainers located 10 feet above the suppression pool floor. A simplified schematic of the LPCS is provided by Figure 2.5.

The LPCS system is automatically initiated and controlled. The operator may be required to manually start the system if an automatic actuation failure occurs.

The success criterion for the LPCS system is injection at rated flow to the reactor vessel.

The LPCS system major dependencies are DC control power for initiating the actuation relay logic and LPCS pump breaker, AC power for operating the LPCS pump and valves, and LPCS pump room cooling.

The DC power is provided by Division 1 125 V DC Panel 1E12-JB1. Power for the LPCS pump is provided by Division 1 4160 V AC Bus 15AA, and power for the valves is provided by Division 1 480 V AC MCC 15B11.

2.2.6 Low Pressure Coolant Injection (LPCI) System

The function of the LPCI system is to provide coolant to the reactor vessel during accidents in which system pressure is low. The LPCI system is one mode of the RHR system and, shares components with other modes. The LPCI system is a three train system consisting of motor-operated valves and motor-driven pumps. The three pumps are each rated at 7450 gpm. Trains A and B each have two heat exchangers in series downstream of the pump. Train C is injection dedicated and has no heat exchangers. Cooling

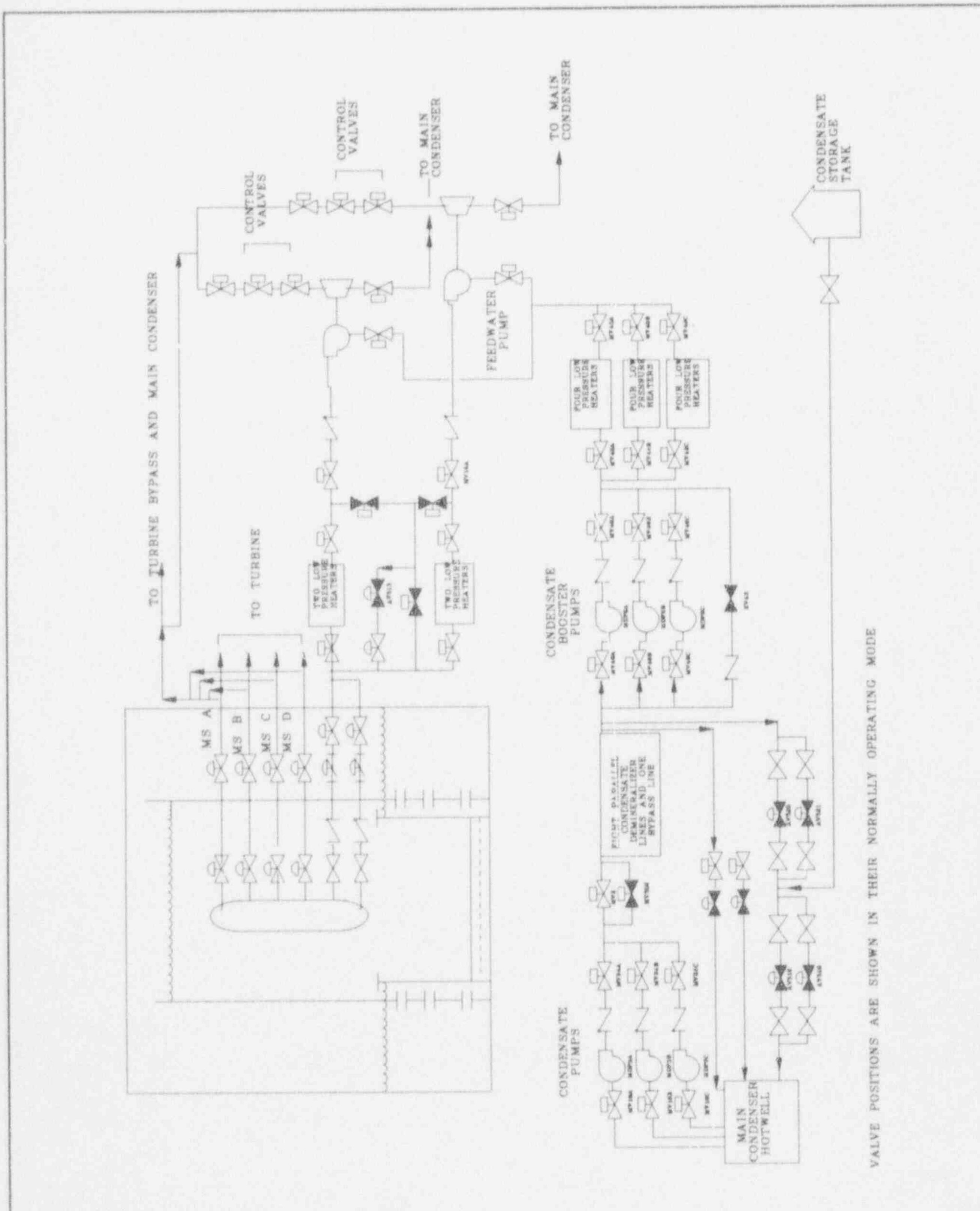

Figure 2.4 Condensate System Schematic

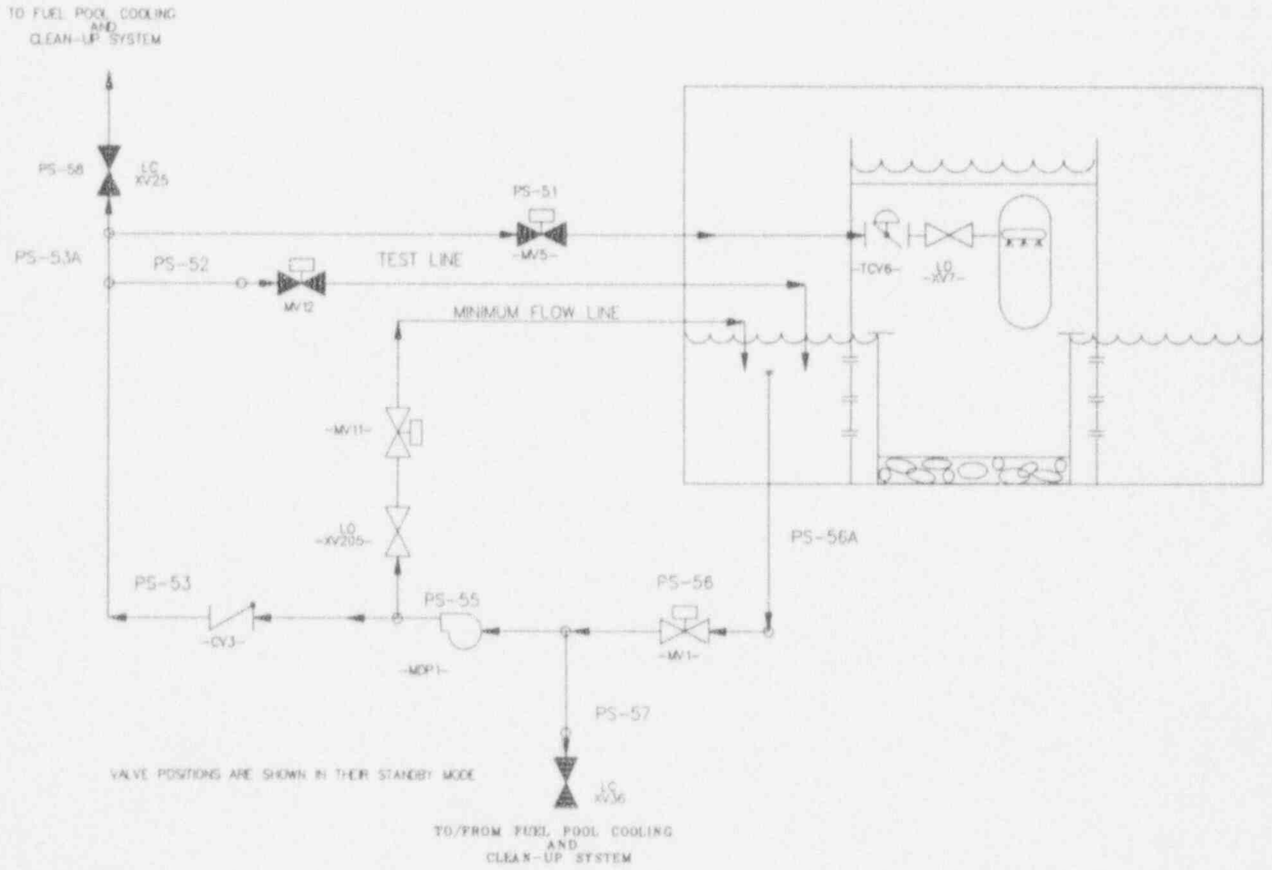

Figure 2.5 LPCS System Schematic

water flow to the heat exchangers is not required for the LPCI mode. A simplified schematic of the LPCI system is provided in Figure 2.6.

The LPCI system is automatically initiated and controlled. However, operator intervention may be required to manually realign and start the system in POS 5 since the individual RHR Trains A and B could be aligned for either shutdown cooling or ADHRS, and Train C could be aligned for ADHRS. Given any of these configurations, the associated train would not automatically initiate for LPCI operation.

The success criterion for the LPCI system is injection of flow from any one pump to the reactor vessel.

The LPCI system major dependencies are DC control power for initiating the actuation relay logic and RHR pump breakers, AC power for operating the RHR pumps and valves, RHR pump cooling, and RHR pump room cooling.

The DC power to Train A is provided by Division 1 125 V DC; for Trains B and C it is provided by Division 2 125 V DC. Power for RHR Pump A is provided by Division 1 4160 V AC Bus 15AA. Power for RHR Pump B and LPCI Pump C is provided by Division 2 4160 V AC Bus 15AB. All pumps require pump cooling.

2.2.7 Standby Service Water Cross-Tie (SSWXT) System

The SSW cross-tie system is used to provide a coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed. The SSW cross-tie system is comprised of Train B of the SSW system and Train B of the LPCI system.

The SSW cross-tie system uses SSW Pump B to inject water into the reactor via the LPCI system Train B injection lines. A simplified schematic of the SSW cross-tie system is provided in Figure 2.7. Major system components are shown in their normal standby position with valves shown. The SSW cross-tie system has no automatic actuation. The system must be manually aligned and manually actuated.

The dependencies for this system are the same as those for SSW Train B (Section 2.2.14) and LPCI Train B (Section 2.2.6).

2.2.8 Firewater (FW) System

The firewater system was modeled as a backup source of low pressure injection as well as a source for automatic and manual fire suppression. The firewater system is a three train system consisting of one motor driven pump and two diesel-driven pumps. The pumps can each provide 1500 gpm at 125 psig. The pumps feed into a common header that supplies water to the fire hoses and sprinkler systems. The pumps take suction from two 300,000 gallon water storage tanks. Any pump can take water from either tank. The fire hoses are connected, via an adapter to various test connections in the auxiliary building. These connections feed into various injection systems and water can be injected through the systems' injection valve. The pumps are located in the firewater pump house. A simplified schematic of the firewater system is provided in Figure 2.8.

The firewater system, when used for injection, must be manually initiated and controlled. The operator is required to align the system and to start the pumps.

The success criteria for the firewater system is injection of flow from any one pump.

The two diesel-driven firewater pumps have no outside interfaces or dependencies, each pump has self-contained batteries that provide it with starting power. The electric motor-driven pump requires balance of plant AC power.

2.2.9 Residual Heat Removal: Suppression Pool Cooling (SPC) System

The function of the SPC system is to remove decay heat from the suppression pool during an accident. The SPC system is but one mode of the RHR system and, as such, shares components with other modes.

The SPC system is a two train system consisting of motor-operated valves and motor driven pumps. Both trains have two heat exchangers in series downstream of the pump. Each pump is rated at 7450 gpm. Cooling water flow to the heat exchanger is required for the SPC mode. The SPC suction source is the suppression pool. A simplified schematic of the SPC (RHR) system is provided by Figure 2.9. Major system components are shown with valves shown in their normal standby operating position.

The SPC system is manually initiated and controlled. The operator is required to align the system and to start the pumps. In POS 5 the RHR system configuration is accident sequence dependent, i.e., RHR Train B can be aligned in (1) Standby LPCI mode, (2) SDC mode, or (3) ADHRS. The operator action to align for SPC is dependent on the prior RHR configuration.

The success criterion for the SPC system is injection of flow from any one pump/heat exchanger train to the suppression pool.

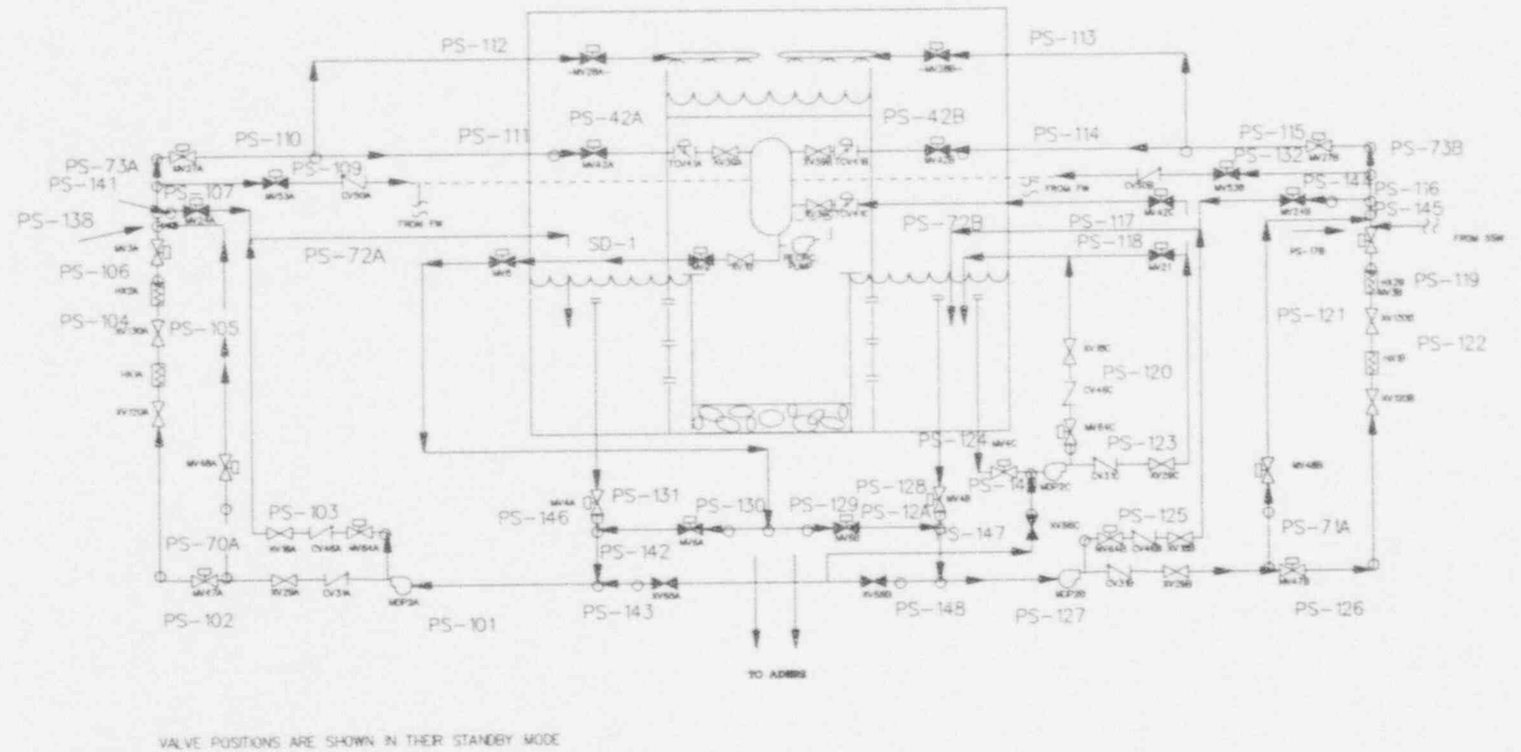

Figure 2.6 LPCI System Schematic

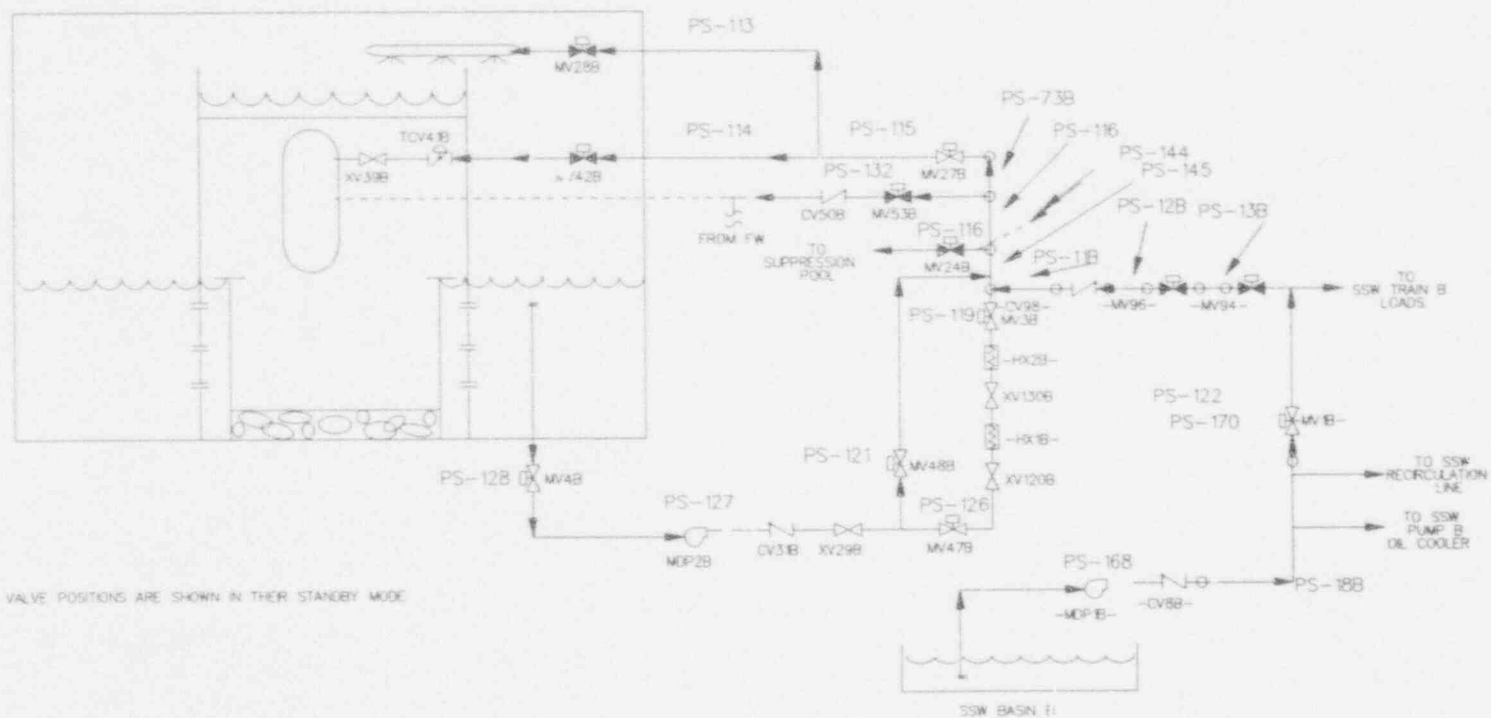

Figure 2.7 SSW Crosstie System Schematic

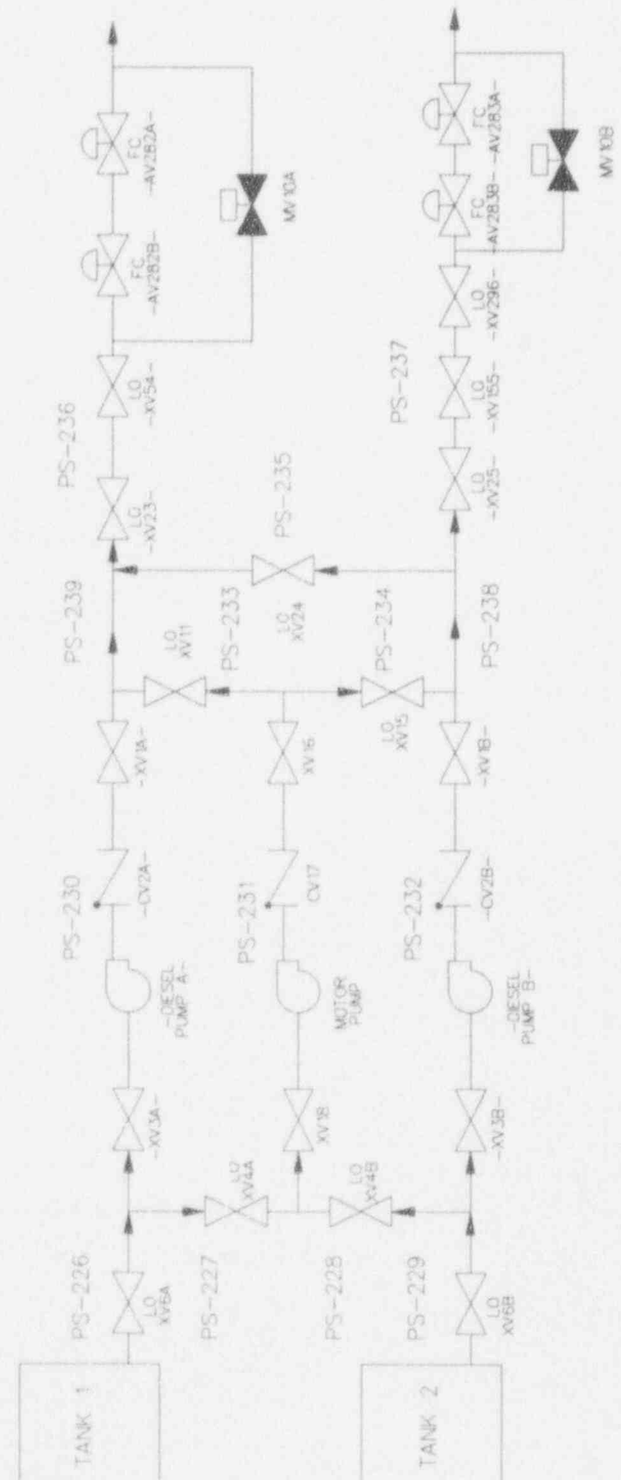

Figure 2.8 Firewater System Schematic

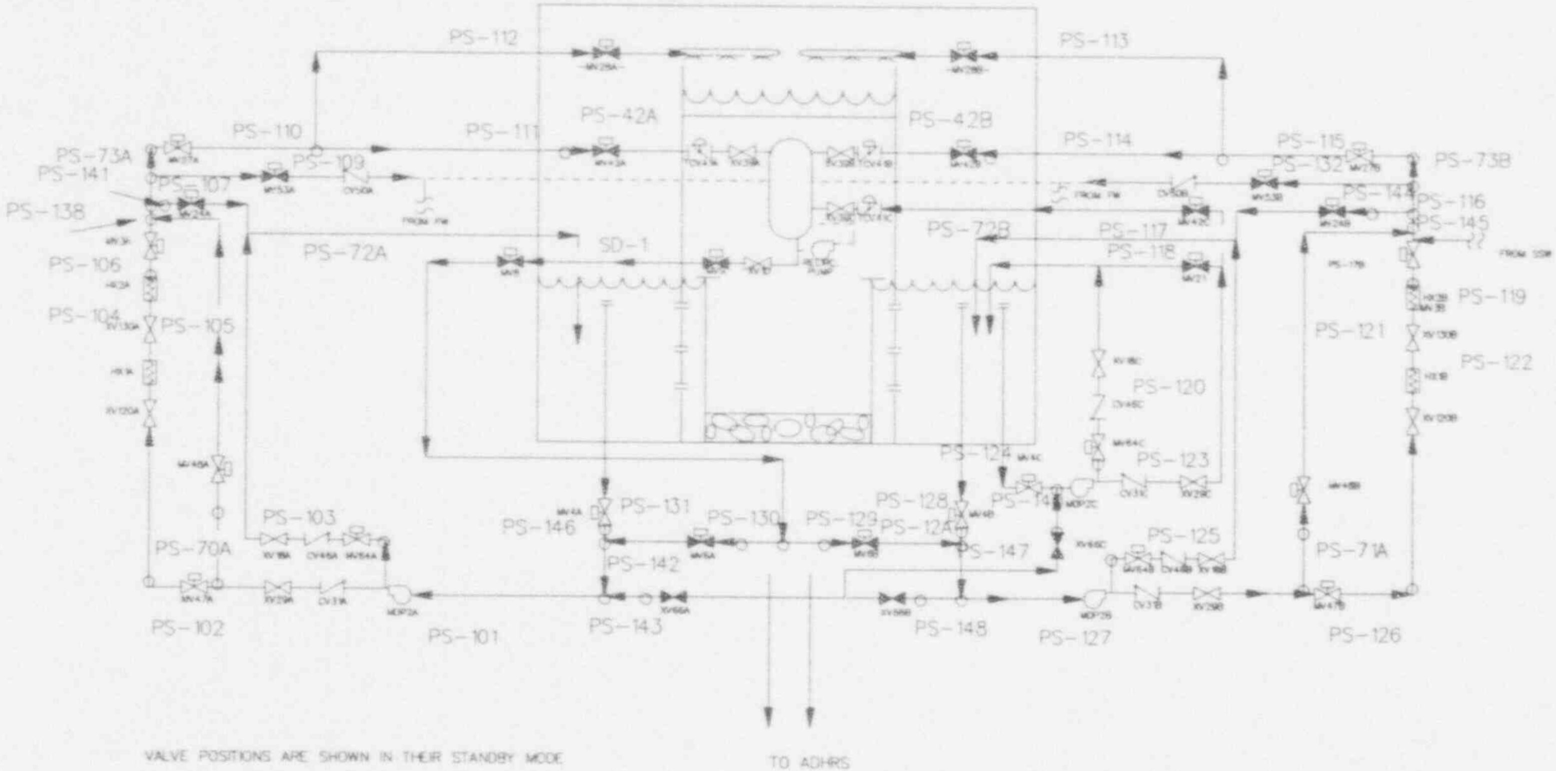

Figure 2.9 SPC System Schematic

The SPC system major dependencies are DC control power for actuation, AC power for operating the RHR pumps and valves, RHR pump cooling, and RHR pump room cooling.

The DC power to Train A is provided by Division 1 125V DC; for Train B it is provided by Division 2 125 V DC. Power for RHR Pump A is provided by Division 1 4160 V AC Bus 15AA. Power to RHR Pump B is provided by Division 2 4160 V AC Bus 16AB. Both pumps require pump cooling.

2.2.10 Residual Heat Removal: Shutdown Cooling (SDC) System

The function of the SDC system in POS 5 is to remove decay heat during shutdown and during accidents in which reactor vessel integrity is maintained. The SDC system is one of the modes of the RHR system and shares components with other modes.

The SDC system is a two train system consisting of motor-operated valves and motor driven pumps. Both trains have two heat exchangers in series downstream of the pump. Each pump is rated at 7450 gpm. Cooling water flow to the heat exchanger is required for the SDC mode. The SDC system suction source is one recirculation pump's suction line. A simplified schematic of the SDC (RHR) system is provided by Figure 2.10. Major system components are shown with valves in their normal standby operating position.

In POS 5 either SDC or ADHRS is in operation removing decay heat. If ADHRS is in operation one Train of SDC is in standby. From standby, the SDC is manually aligned and started when placed in operation. Note that for the POS 5 analysis Train A of RHR has been assumed to be unavailable due to maintenance.

The success criterion for the SDC system is injection of flow from any one pump/heat exchanger train to the reactor vessel.

The SDC system major dependencies are DC control power for actuation, AC power for operating the RHR pumps and valves, RHR pump cooling, and RHR pump room cooling.

The DC power to Train A is provided by Division 1 125 V DC, and for Train B it is provided by Division 2 125 V DC. Power for RHR Pump A is provided by Division 1 4160 V AC Bus 15AA. Power to RHR Pump B is provided by Division 2 4160 V AC Bus 16AB. All pumps require pump cooling.

2.2.11 Residual Heat Removal: Containment Spray (CS) System

The function of the CS system is two-fold: (1) to suppress pressure in the containment during an accident and (2) to remove fission products from the containment atmosphere following core damage. The CS system is one of the modes of the RHR system and shares components with other modes.

The CS system is a two loop system consisting of motor-operated valves and motor-driven pumps. There are two heat exchangers in series per loop. Each pump is rated at 7450 gpm. Cooling water flow to the heat exchanger is required for CS when used to suppress pressure in the containment. The CS suction source is the suppression pool. A simplified schematic of the CS system is provided by Figure 2.11. Major system components with valves are shown in their normal standby operation position.

The CS system is automatically initiated and may be controlled. However, operator intervention is required to manually realign and start the system. In POS 5 RHR Train B can be aligned for either shutdown cooling or ADHRS. Given these configurations, the associated train would not automatically initiate and spray.

The success criterion for the CS system is injection of flow from any one pump/heat exchanger train to the spray ring.

The CS system major dependencies are DC control power for actuation, AC power for operating the RHR pumps and valves, RHR pump cooling, and RHR pump room cooling.

The DC power to Train A is provided by Division 1 125 V DC and for Train B it is provided by Division 2 125 V DC. Power for RHR Pump A is provided by Division 1 4160 V AC Bus 15AA. Power to RHR Pump B is provided by Division 2 4160 V AC Bus 16AB. Both pumps require pump cooling.

2.2.12 Containment Venting System (CVS)

When suppression pool cooling and containment sprays have failed to reduce primary containment pressure, the CVS is used to prevent a primary containment pressure limit from being exceeded.

The vent path used is a 20-inch diameter purge exhaust line which is part of the containment ventilation and filtration system. This line includes four air-operated dampers which are normally closed. All four fail closed on loss of air.

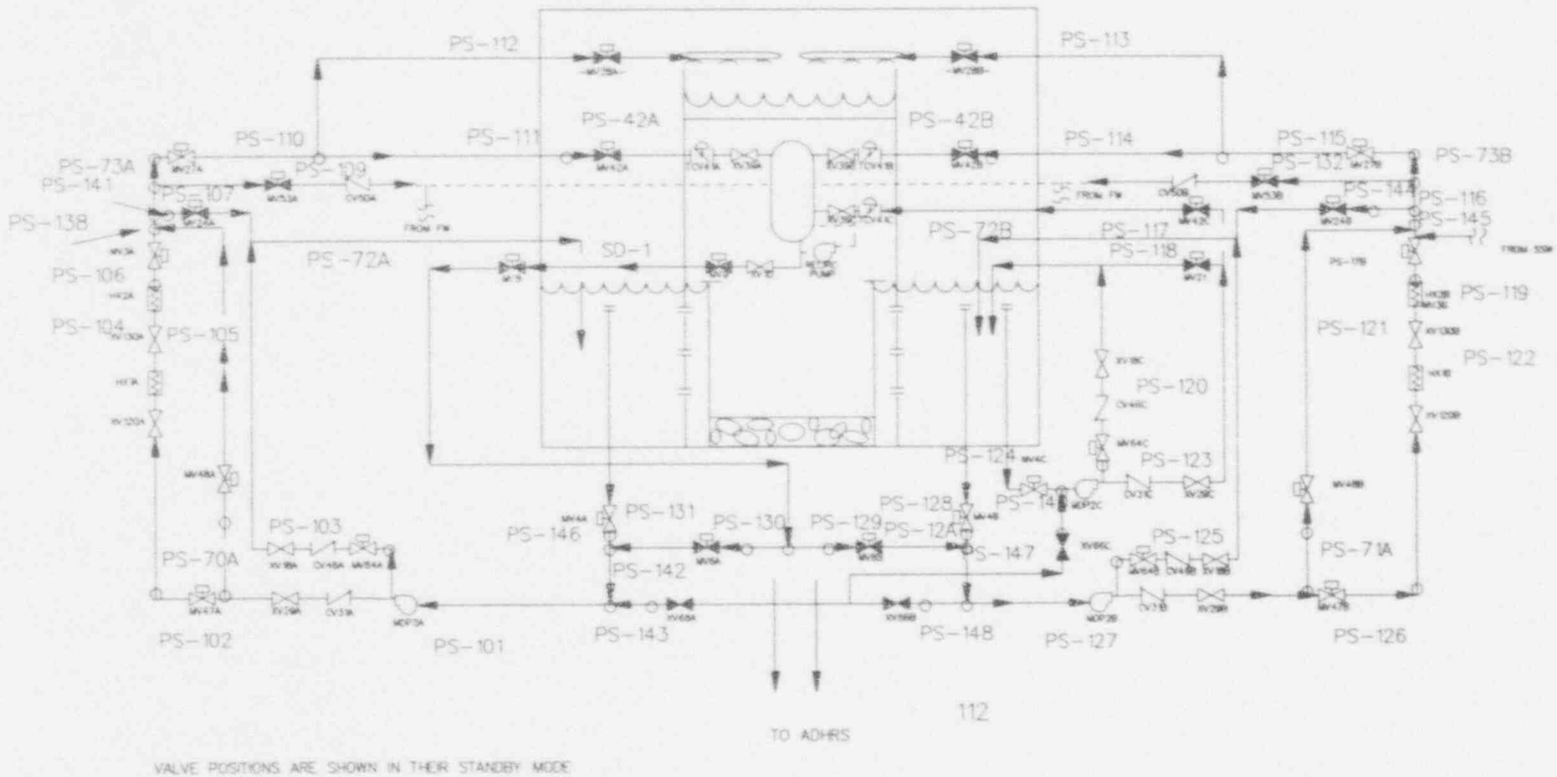

Figure 2.10 SDC System Schematic

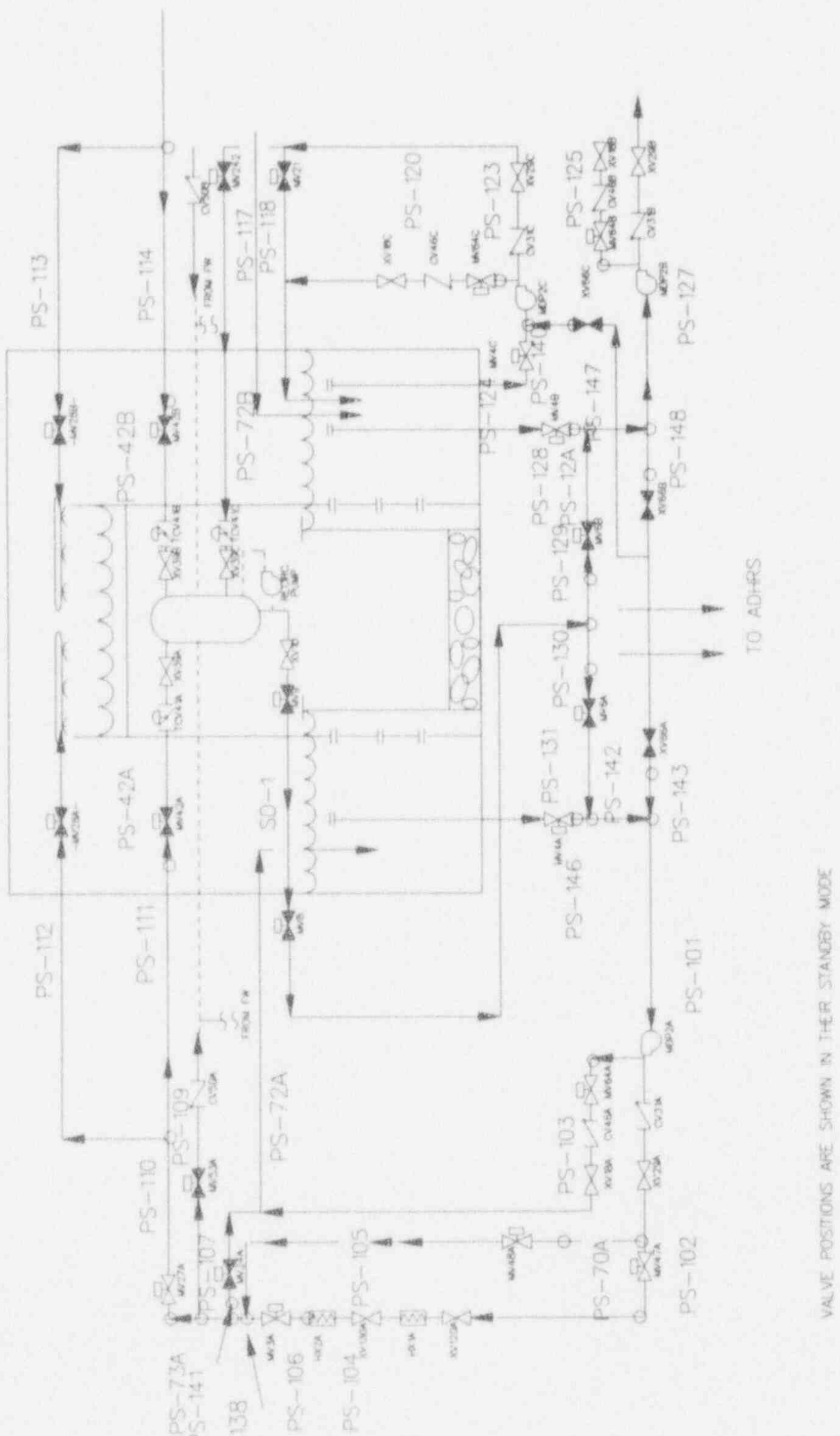

Figure 2.11 CS System Schematic

Two of the dampers are closed by a containment isolation signal. The other two are closed by the standby gas treatment system initiation. The CVS discharges to the roof of the auxiliary building. A schematic of the CVS is shown in Figure 2.12.

The venting procedure requires containment venting when the pressure exceeds 17.25 psig. Venting requires that the operator jump the isolation relays for each damper and then open them (they are located on back panels in the control room). The actual venting procedure can only be initiated by order of the emergency director.

Containment venting requires instrument air for opening the air-operated dampers. The dampers also require power from emergency AC Divisions 1 and 2 for operation of the solenoids.

Instrument air to the auxiliary building is isolated by a LOCA signal (high drywell pressure); and instrument air to the containment is isolated by a containment isolation signal. Since both signals are expected to be present during a venting situation, the operator must also restore instrument air to successfully vent.

2.2.13 Emergency Power System (EPS)

The EPS consists of the AC and DC power divisions required by all systems (except firewater) needed to mitigate postulated accidents. This includes Balance of Plant (BOP) and Engineered Safety Feature (ESF) buses. Both ESF AC and DC power are divided into three separate divisions. Two of the divisions (1 and 2) are for the majority of the ESF and the third (3) is dedicated to the HPCS system and its required support systems. The EPS is shown in Figure 2.13.

The ESF AC divisions normally receive power from one of three offsite sources through ESF (34.5 kV/4.16 kV). In addition to the normal supply from the ESF transformers, each ESF 4.16 kV bus has a standby diesel generator which is available to supply bus loads upon a loss of normal AC power. These diesels may be started manually or automatically. The diesels supplying Divisions 1 and 2 buses are rated at 7000 kW and start on a loss of normal AC power to the associated bus, low reactor level of -150 inches, or high drywell pressure of +2 psig. The diesel supplying Division 3 buses (rated at 3300 kW) is exclusively for the HPCS and starts on a loss of normal AC power, low reactor water level (-42 inches), and high drywell pressure signal of +2 psig. For Divisions 1 and 2, the transfer of power from normal to backup or emergency power supplies is controlled by the load shedding and sequencing system.

For Divisions 1 and 2, when a loss of normal power signal occurs, the diesel generators automatically start and connect to the associated ESF bus if no other source of power is available. To prevent overloading the diesel generator when no alternate source is available, unnecessary loads are shed from the associated bus and those loads required for plant safety are sequenced onto the bus. For Division 3, when a loss of normal power occurs, the diesel generator will start and automatically close on the bus when at speed and voltage.

If Divisions 1 and 2 diesel generators fail to power their buses, power can be supplied to certain Division 1 or 2 loads from the HPCS diesel generator. This is accomplished by isolating the normal Division 3 loads from the diesel and connecting either the Division 1 or 2 loads to the HPCS diesel generator. The electrical equipment that is involved in accomplishing this is shown in Figure 2.14.

The ESF 125 V DC system includes three divisions, each consisting of two battery chargers which normally supply the load and a bank of batteries which functions as a backup. Divisions 1 and 2 of the ESF DC (Buses 11DA and 11DB, respectively) system supply the majority of the ESF loads. Both are rated at 1600 amperes. Division 3 (Bus 11DC) is dedicated to the HPCS system and is rated at 100 ampere hours.

The battery chargers normally supplying power to each ESF bus are silicon controlled, rectifier type chargers rated at 400 amperes, 125 V DC. The ESF battery chargers maintain the terminal voltage of the associated batteries above a minimum of 1.75 volts per cell. Either charger can restore the batteries from this minimum voltage to their fully charged state within eight hours under normal plant operating conditions.

Each ESF DC battery bank consists of sixty lead-calcium type cells connected in series to produce the rated output of 125 V DC. Each ESF battery bank can supply the required DC loads for eleven hours after a loss of AC power if unnecessary loads are shed.

Each diesel generator has six subsystems required for its operation: (1) fuel oil subsystem, (2) air starting subsystem, (3) lube oil subsystem, (4) jacket water cooling subsystem, (5) combustion air intake, exhaust and crankcase ventilation, and (6) standby generator excitation subsystem. All of these subsystems are normally treated as part of the diesel generator. However, some of these other subsystems are dependent on operation of other systems.

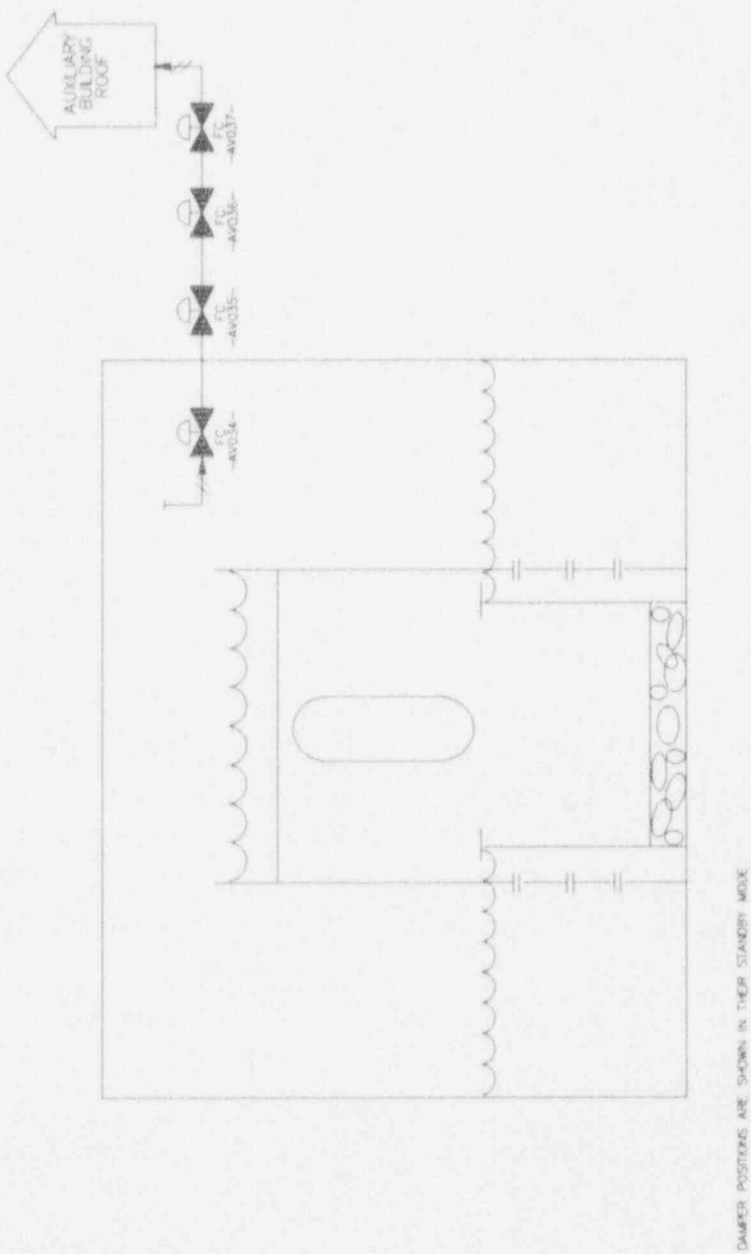

Figure 2.12 CVS System Schematic

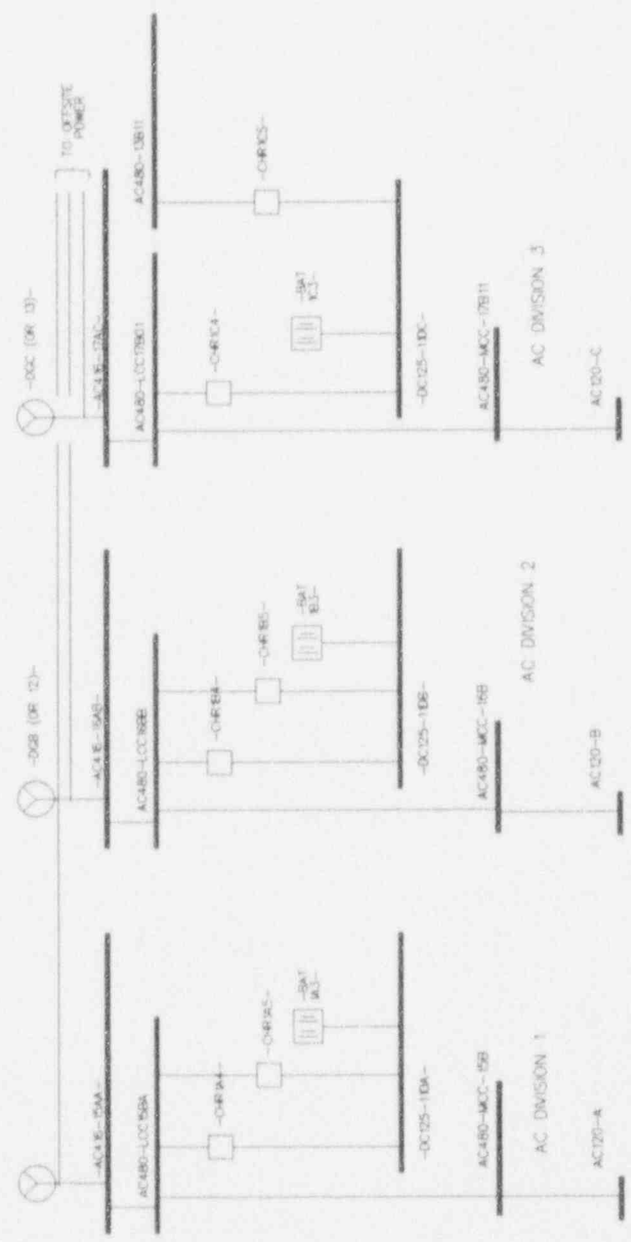

Figure 2.13 EPS System Schematic

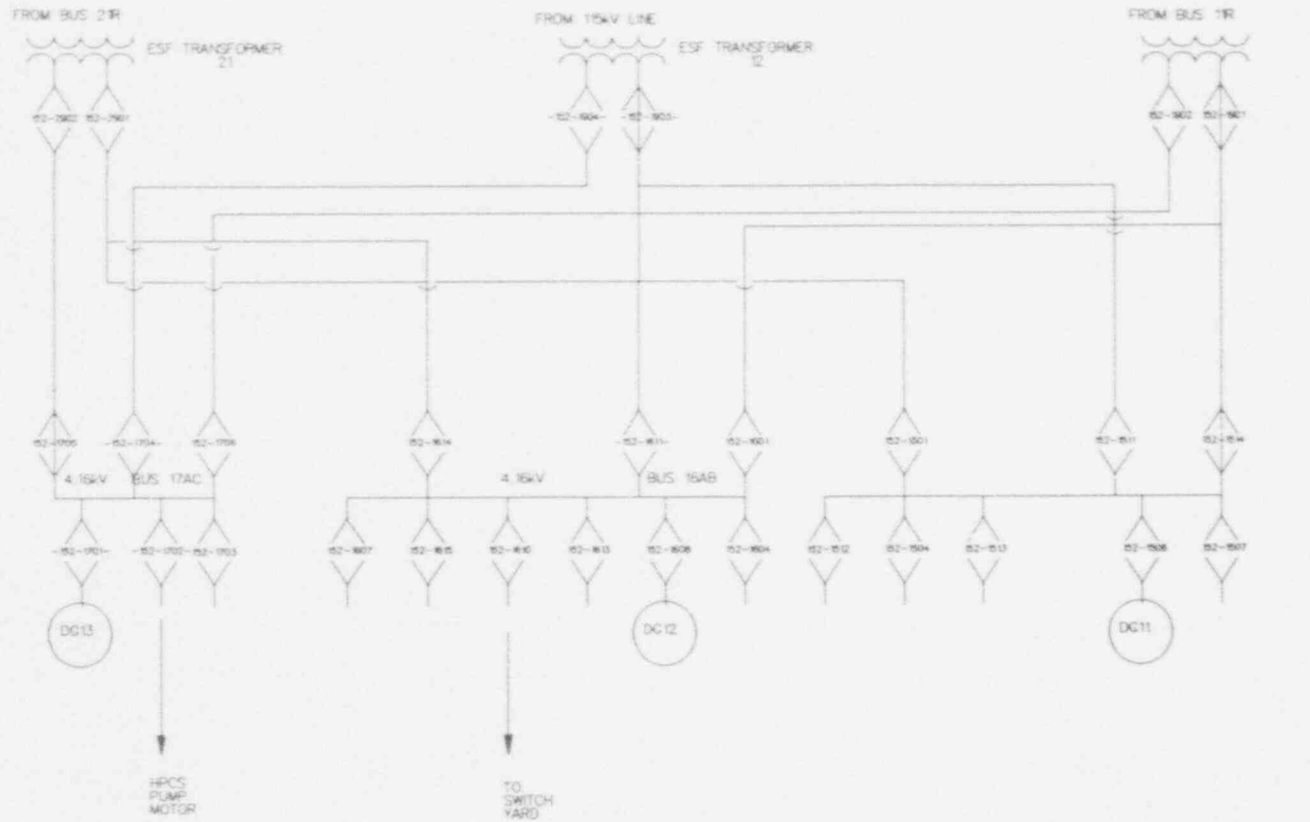

Figure 2.14 Diesel Generator Cross-Tie Schematic

2.2.14 Standby Service Water (SSW) System

The function of the SSW system is to provide heat removal from plant auxiliaries that require cooling water during an emergency shutdown of the plant.

The SSW system is made up of three independent trains. Each train consists of a motor-driven pump, motor-operated valves, and heat exchangers. Train C is dedicated to the HPCS system.

SSW Pumps A (unavailable for POS 5) and B are vertical, centrifugal pumps, each with a 12,000 gpm capacity. SSW Pump C is also a vertical, centrifugal pump, but with only a 1300 gpm capacity. Each pump takes water from the cooling tower basins, circulates water through the heat exchangers for each load, and returns the water to the tower through a motor-operated discharge valve. Each train has its own discharge valve. A simplified schematic of the SSW system is provided in Figure 2.15.

The SSW system is automatically initiated and controlled. However, operator intervention is required to manually start the system given an auto-start failure.

The SSW system major dependencies are DC control power for initiating the actuation relay logic, and AC power for operating the SSW pumps and valves. The pumps are self-cooled.

The DC power to Trains A, B and C is provided by the Division 1 125 V DC, Division 2 125 V DC, and Division 3 125 V DC buses, respectively. Power for SSW Pump A is provided by Division 1 4160 V AC Bus 15AA. Power for SSW Pump B is provided by Division 2 4160 V AC Bus 16AB. Power to SSW Pump C is provided by Division 3 480 V AC Bus 17B01.

2.2.15 Emergency Ventilating System (EVS)

The objective of the EVS is to maintain suitable temperatures in safety related equipment rooms to preclude component failures.

The EVS cools the following: (1) standby diesel generator rooms, (2) service water pump rooms, (3) pump rooms for the RHR, RCIC, HPCS, and LPCS pumps, and four rooms containing electrical switchgear. The service water pump room system is assumed not to be required. Three independent subsystems, one per diesel generator room, each having 100% capacity, are provided for the emergency diesel generator rooms to maintain a design temperature of 120°F. Each diesel unit is provided with a fan damper system connected to the respective diesel engineered safety features bus. The fan is controlled to start on diesel generator startup and stop on diesel generator shutdown. The damper opens on the same signals. A heating coil has been provided to maintain the minimum required air temperature during cold weather. The ventilation system for the diesel generator rooms is shown in Figure 2.16.

The cooling and emergency ventilation systems for the safety-related pump rooms are shown schematically in Figure 2.17. Each safety-related pump room is provided with one full-capacity fan-coil unit to prevent the room temperature from exceeding 150°F during pump operation.

The SSW system provides cooling water for the fan-coil units. The units start automatically when the associated ECCS pump starts.

During normal plant operation, the safety-related pump rooms and penetration rooms are maintained at a slight negative pressure with respect to the corridors by the fuel handling area ventilation system. Supply air is provided from the auxiliary building ventilation system. Air is drawn from the safety-related pump rooms and discharged by the fuel handling area exhaust fans to the fuel handling area vent. The success criterion for the diesel generator rooms requires operation of the fan with the damper opened.

Success of the ECCS pump room systems involves operation of the fan-coil units with associated cooling by the Standby Service Water System.

It is assumed that failure of the EVS would fail operating diesel generators in fifteen minutes. The low pressure ECCS pumps are assumed to fail within four hours after loss of the associated room cooling. HPCS is assumed to fail within twelve hours after loss of room cooling.

The ECCS pump room coolers are all cooled by the Standby Service Water system. Power for the fans for the RCIC, LPCS, and RHR-A pump room is provided by AC Division 1. The fans for RHR-B and RHR-C pump rooms are powered by Division 2. The HPCS pump room fan is powered by HPCS dedicated AC Division 3.

2.2.16 Instrument Air System (IAS)

The IAS provides a pneumatic supply to support operation of safety related equipment.

The Instrument Air System for each unit consists of one full-capacity, multistage, packaged centrifugal compressor, complete with inlet filter, inlet air controller, and aftercooler. The compressor has a receiver and a regenerative desiccant air dryer. The Unit 2 IAS compressor, receiver, and dryer are present and operational.

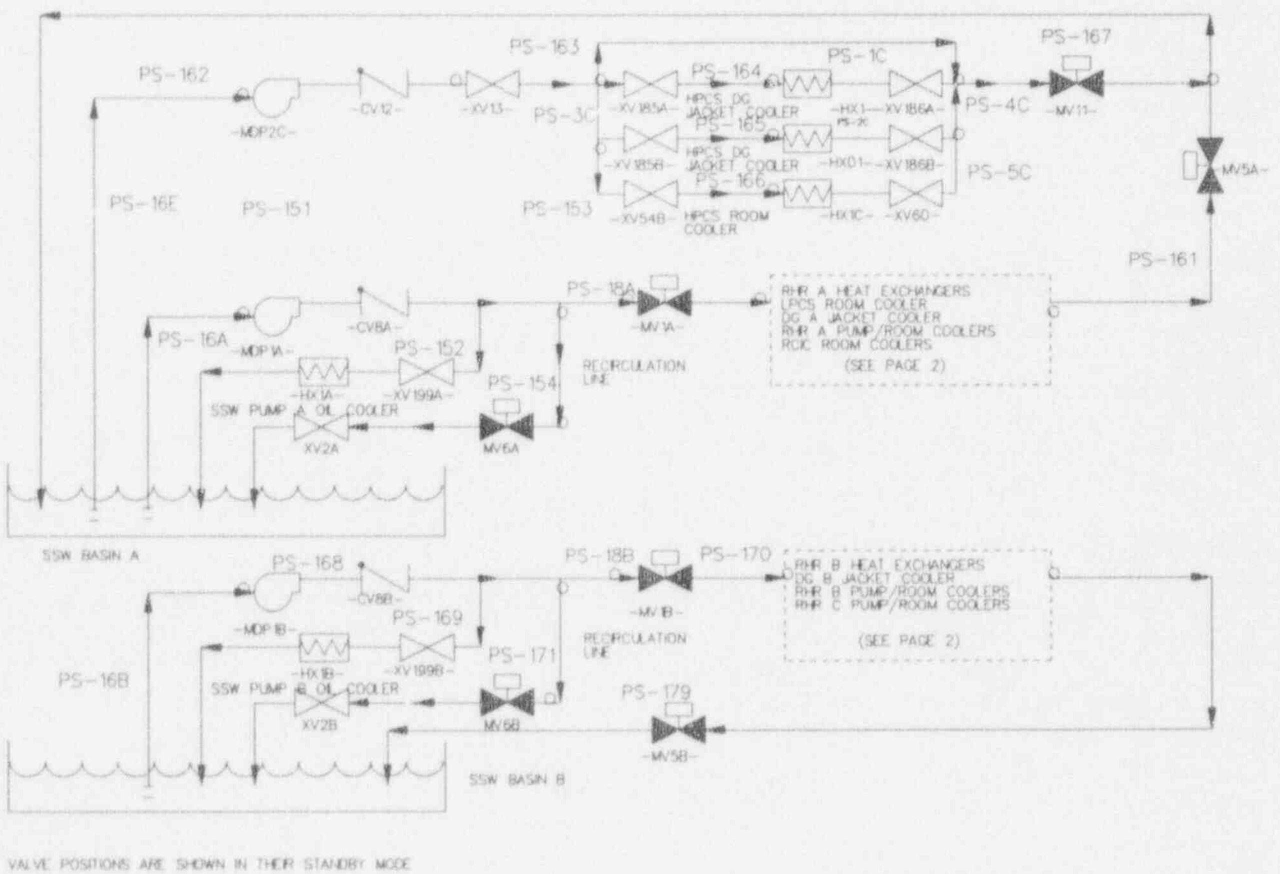

Figure 2.15 SSW System Schematic (Page 1 of 2)

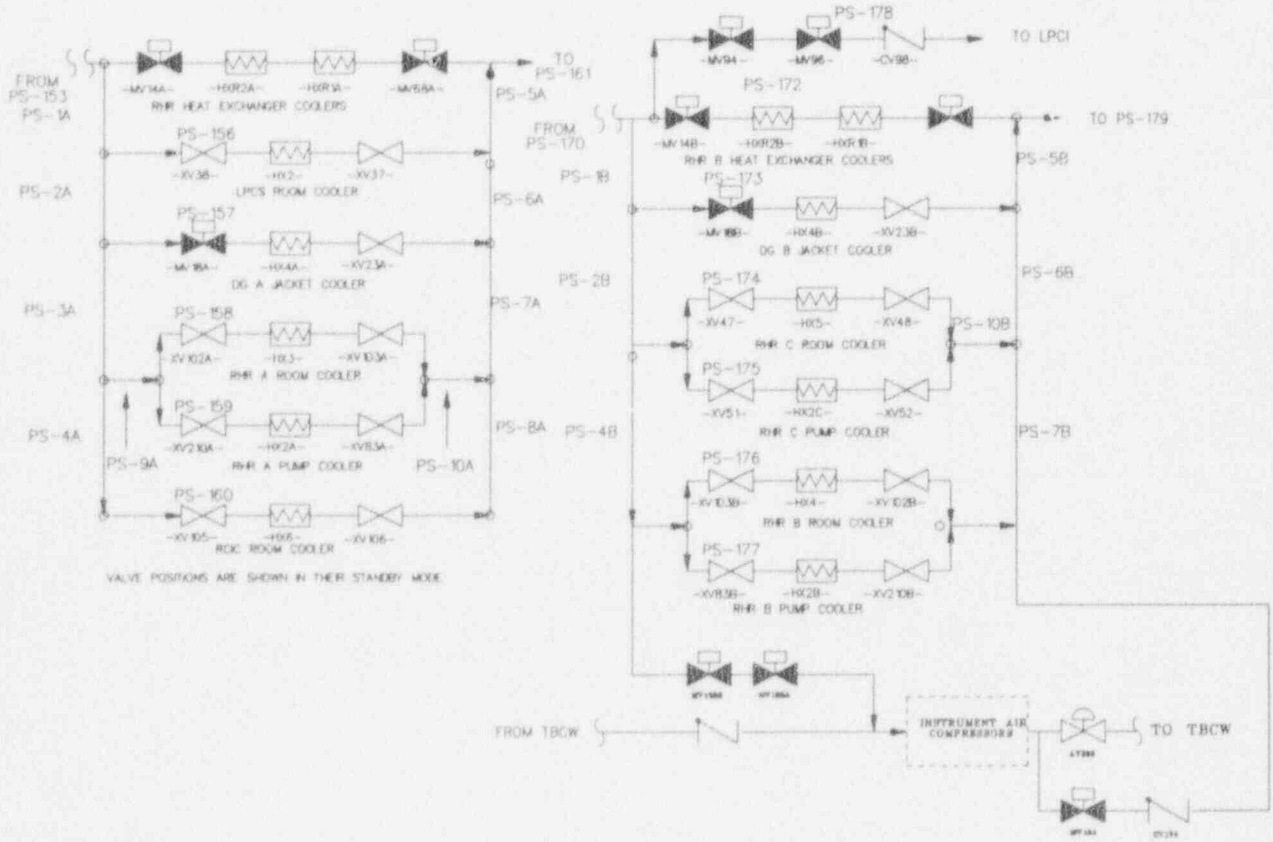

Figure 2.15 SSW System Schematic (Page 2 of 2)

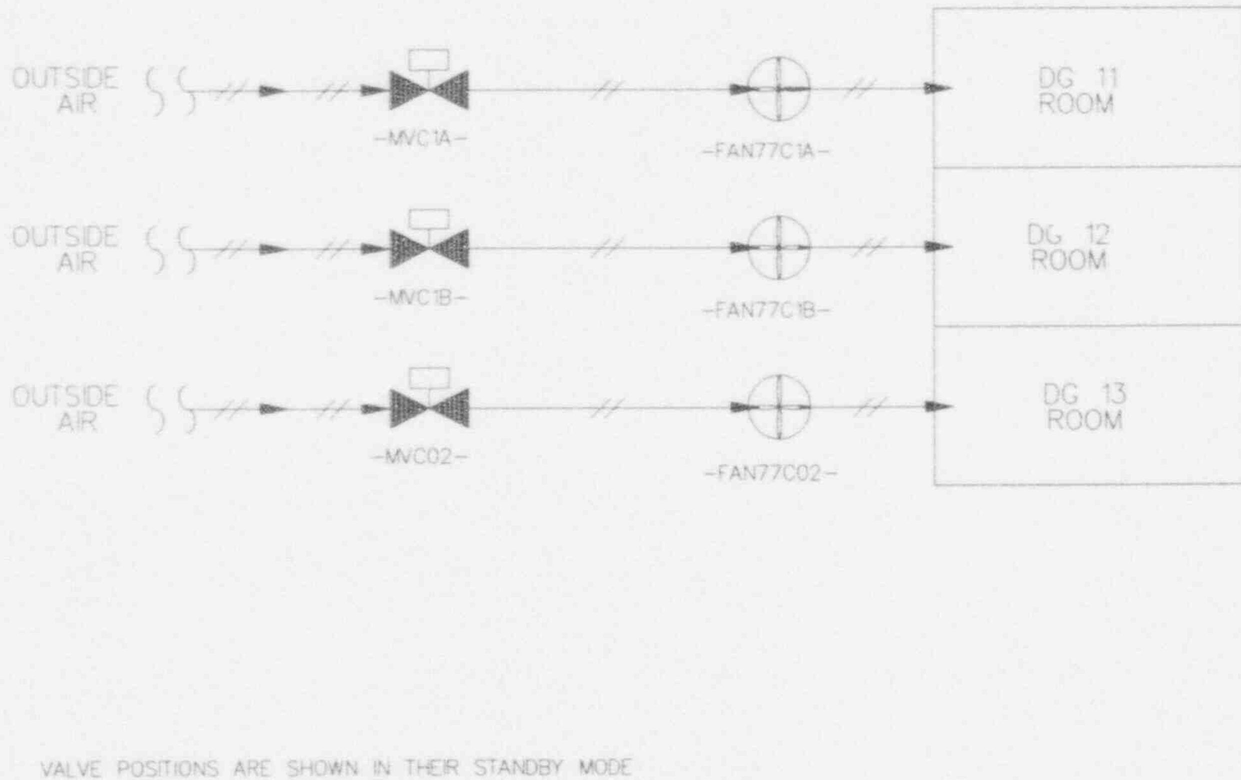

Figure 2.16 EVS System Schematic for Diesel Generator Rooms

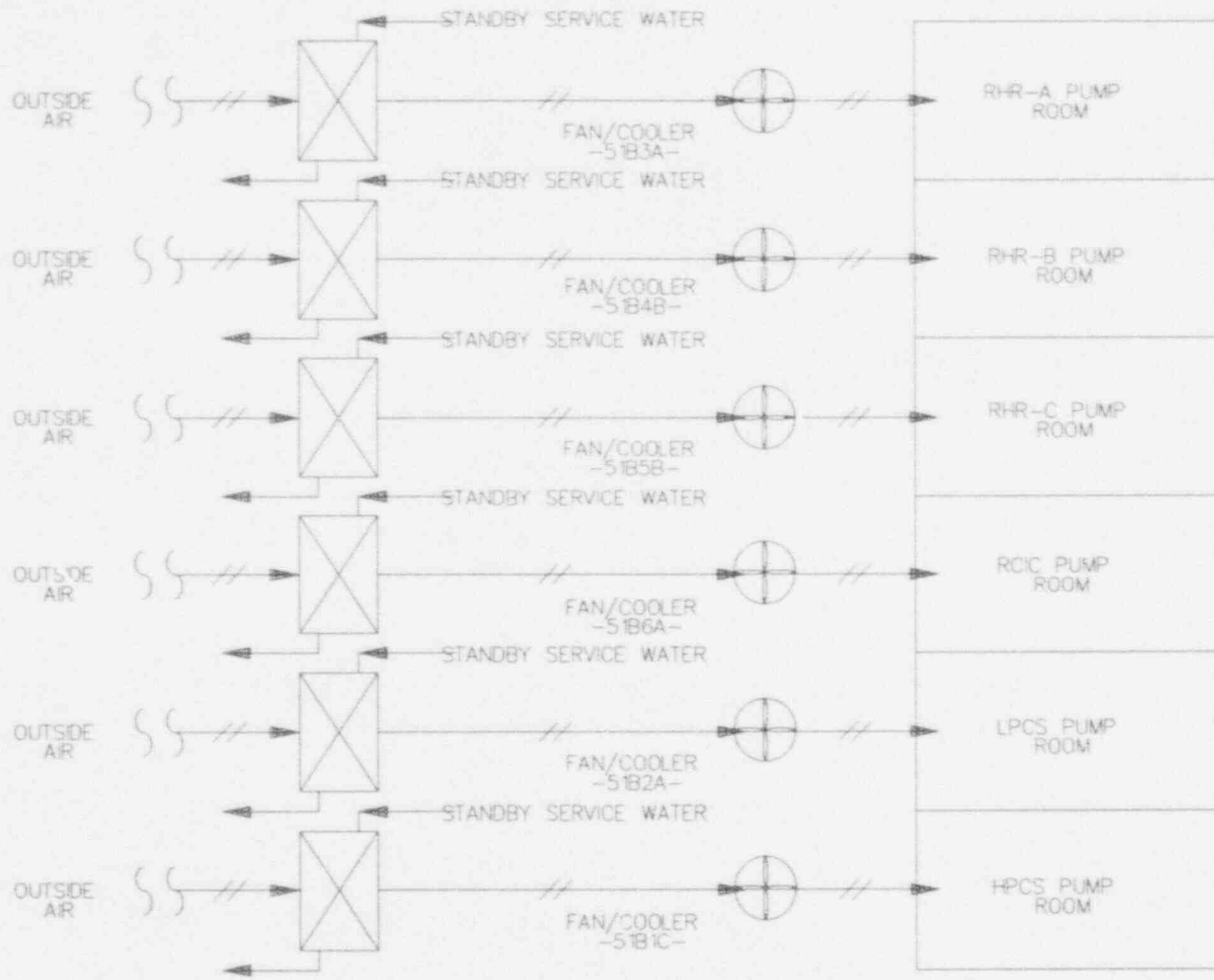

Figure 2.17 EVS Systems Schematic for Safety-Related Pump Rooms

Instrument air is distributed to the plant after drying to a dew point of -40°F and filtration of particles 0.9 microns and larger. In addition, each instrument or group of instruments has a pressure regulating valve with an integral filter located in its instrument air supply. The instrument air systems of Units 1 and 2 can be cross-connected by opening two air-operated valves from the control room. One instrument air compressor can supply all instrument air demands with the other compressor as a backup. The air-operated cross-connect valves fail open upon loss of air to their operators, thereby assuring instrument air supply to Unit 1 from either instrument air system.

The Service Air System (SAS) is also arranged as an automatic backup supply to the instrument air system through a control valve that opens upon reduced line pressure in the instrument air system. The backup connection is upstream of the instrument air dryers. Credit for the SAS is taken in this study and included under the IAS discussion.

The SAS consists of two full-capacity, multistage, packaged centrifugal compressors, each complete with inlet filter, inlet air flow controller, aftercooler, and receiver. A simplified schematic of the IAS and SAS is provided in Figure 2.18.

The IAS supplies clean, dry, oil-free air to EVS air valves, the CRD control system, condensate system valves, containment venting air valves, main steam isolation valves, RWCU AOVs, FW AOVs, PSW AOVs, and the SRV valves (a nitrogen system backs up the IAS supply to these valves).

The success criterion for the IAS is that either of the IAS compressors or one of the SAS compressors must supply air to system pneumatic loads.

Failure of the IAS does not directly fail any related safety systems.

Cooling requirements of the IAS and SAS air compressors and aftercoolers are normally supplied by the Turbine Building Cooling Water (TBCW) system. In the event of offsite power failure, the SSW system cools the air compressors and aftercoolers.

The Unit 1 IAS air compressor is powered from emergency AC Division 2. The Unit 2 IAS and the SAS air compressors are powered from non-safety buses. Following a loss of offsite power, standby onsite power is provided to the Unit 1 IAS air compressors to replenish compressed air storage as required.

2.2.17 Alternate Decay Heat Removal (ADHR) System

The function of the ADHR system is to provide an alternate method of decay heat removal during cold shutdown and refueling when maintenance is being performed on the RHR shutdown cooling loops or associated support systems.

The ADHR system consists of components common to the RHR system including the RHR common suction line, fuel pool cooling and cleanup piping, and the RHR Train C LPCI injection header. Components exclusive to ADHR include two ADHR pumps, two heat exchangers, associated piping, valves, instrumentation and controls.

The ADHR can operate in four main modes of which one is modeled for POS 5. In POS 5, ADHR is used in the reactor vessel cooling mode via RHR B. During the reactor vessel cooling mode, ADHR draws water from the existing RHR common suction line. The reactor coolant is then pumped from the reactor recirculation loop through valves F066A and F006A or valves F066 and F006B to the ADHR pumps, then to the heat exchangers and back to the reactor vessel via RHR C LPCI injection line. A schematic of the ADHR system is shown in Figure 2.19.

Control for the ADHR system is remote manual from the control room. Flow and temperature indications are provided in the control room for ADHR heat exchangers while individual manual control of pump operation with pump running status lights is provided.

The success criterion for the ADHR system is to provide cooling to the reactor vessel at rated flow.

The ADHR major dependencies are AC Division 1 and 2 power for motor operated valves F066A and F066B respectively, and BOP AC bus 14HE for the ADHR pumps and motor operated valve F424. Plant Service Water (PSW) cools the heat exchangers.

2.2.18 Reactor Water Cleanup (RWCU) System

The function of the RWCU system is to provide continuous purification of reactor water to reduce the fouling of heat transfer surfaces, minimize
secondary sources of radiation, and maintain water clarity during refueling. The system also acts as a decay heat removal system and reduces stratification in the reactor vessel by providing a discharge path for excess water to either the condenser hotwell or to the radwaste system.

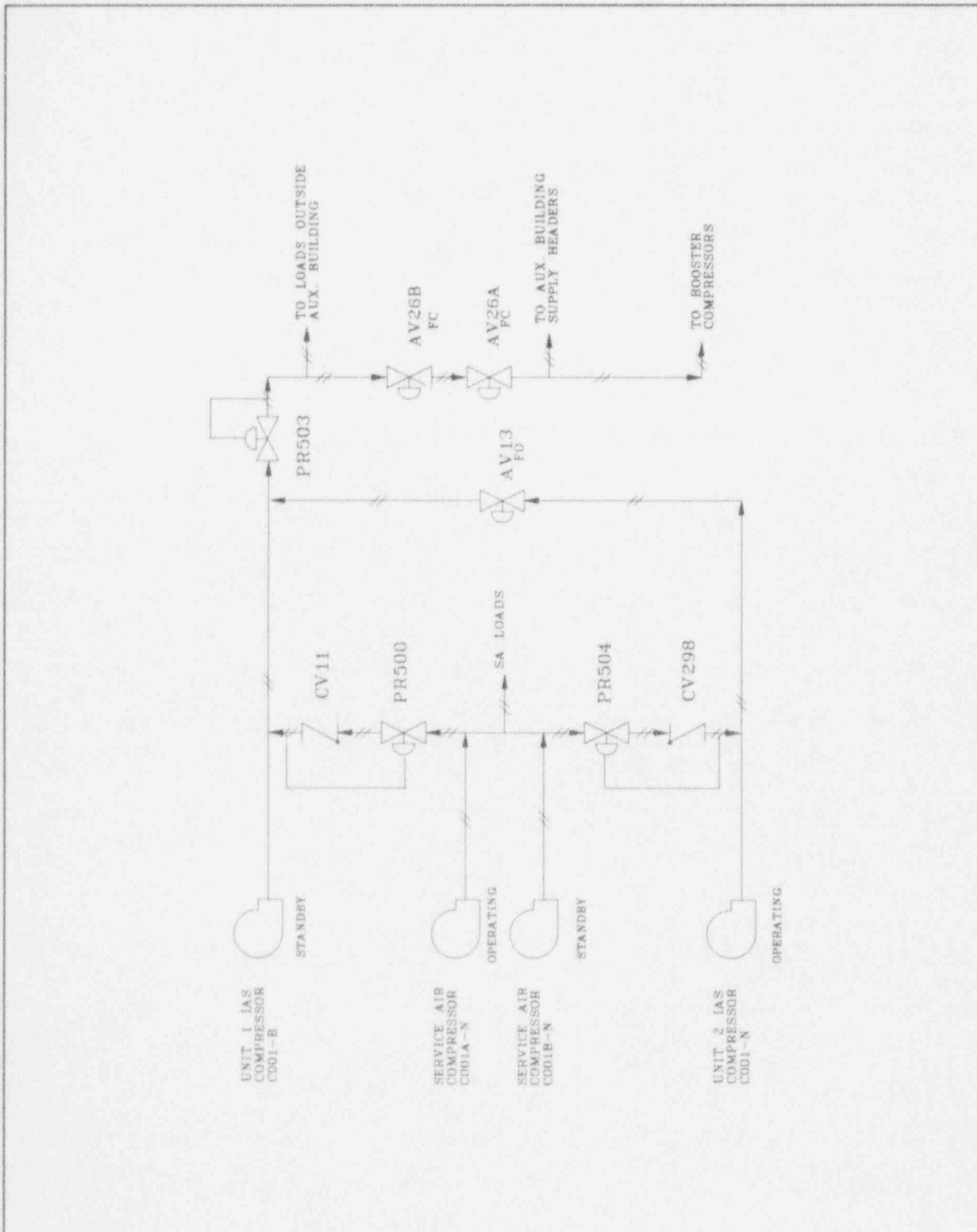

Figure 2.18 IAS System Schematic

Figure 2.19 ADHR System Schematic

Water is drawn into the RWCU system from both recirculation lines and the vessel drain. The flow is then pumped to a series of three regenerative heat exchangers. From the regenerative heat exchangers, the water is routed to the non-regenerative heat exchangers and the filter-demineralizers and finally to the reject lines of either the radwaste or the main condenser. See Figure 2.20 for a flow diagram. Major system components are shown in their standby configurations.

Operation of the RWCU system is controlled from the main control room. The outboard isolation valve will close automatically to prevent damage to the filter demineralizers resins when the outlet temperature of the non-regenerative heat exchangers is too high.

The success criterion for the RWCU in POS 5 is (1) the discharge of reactor water to either the main condenser or radwaste system in order to maintain proper level in the reactor vessel, and (2) to remove decay heat from the vessel during hydro testing.

The RWCU system's major dependencies are AC Division 1 and 2 power for the RWCU pump suction and containment isolation MOVs. BOP AC power is supplied to the RWCU pumps, heat exchanger and filter demineralizes bypass valves, instrument air for the system's blowdown valves to the main condenser and radwaste, and CCW for cooling the non-regenerative heat exchangers and pumps.

2.2.19 Reactor Recirculation System (RRS)

The function of the Reactor Recirculation System (RRS) is to provide mixing of water in the downcomer region and forced circulation through the reactor core. The RRS prevents stagnation and stratification of the core region causing possible transition boiling or film boiling and high temperatures for the cladding of the fuel rods. The RRS insures nucleate boiling for optimum heat transfer, improved efficiency, and lowering of fuel rod cladding temperatures. In the event of a RRS failure, the reactor water level can be raised to initiate natural circulation through the core region.

The RRS consists of two loops external to the reactor vessel (RRS Pump A is unavailable in POS 5). Each loop contains an electric motor-driven centrifugal pump, two electric motor-driven gate valves (for isolating the pump), and a hydraulic flow control valve. A simplified schematic of the RRS system is shown in Figure 2.21.

The Flow Control Valves (FCV), one in each loop, are hydraulically operated and are located immediately after the discharge of the pump and before the discharge gate valve. The flow control valve has a range of operation from 25% to full flow. These valves are mechanically restrained to prevent operation beyond these limits.

The RRS major dependencies are the Component Cooling Water (CCW) system for pump cooling and AC power for operating the recirculation pumps and valves.

The recirculation pumps are driven by squirrel cage induction motors. At full speed the pumps are powered from 4160V AC BOP buses 11HD for loop A and 12 HE for loop B. At 25% rated speed the pumps are powered from separate 15Hz motor-generator sets. The motor-generator sets are in turn powered from 4160V AC BOP buses 13AD for loop A and 14AE for loop B.

The motor-operated valves in the RRS are powered from 480V AC BOP MCC 11B51. The flow control valves are hydraulic-operated valves and are mechanically limited from going more than 25% closed. The hydraulic units for these valves are powered from 480V AC BOP MCC 11B51. The flow control valves fail as is on loss of power to the hydraulic power units. Therefore, loss of power at BOP MCC 11B51 would not necessarily fail the RRS if the system is already operating and properly aligned.

2.2.20 Component Cooling Water (CCW) System

The function of the CCW system is to provide a closed cooling loop between certain plant auxiliary components and the PSW system or the SSW system. The CCW system consists of: (1) three pumps each with a 50% capacity; (2) three heat exchangers each with a 50% capacity; (3) a 550 gal. capacity surge tank; and (4) a 50 gal. capacity chemical addition tank. Each pump is of a single-stage, horizontal, centrifugal, double-suction design powered by a 480V AC, 100 hp electric motor. Each pump has a 1987 gpm flow rate. Each heat exchanger is cooled by the plant service water system and is of a straight tube, single pass, counterflow design with a 1987 gpm capacity. Figure 2.22 shows the CCW system diagram.

During normal operation, two of the three pumps are operating with the third in standby, with both isolation valves open (the check valve prevents backflow). The standby pump auto starts on a low pressure signal of 100 psig. Two of the three heat exchangers are operating with the third isolated by two manual valves from the rest of the system. The surge tank is connected to the system, but the chemical addition tank is generally not connected until needed for addition of chemicals.

The Component Cooling Water (CCW) system supplies cooling to the following loads:

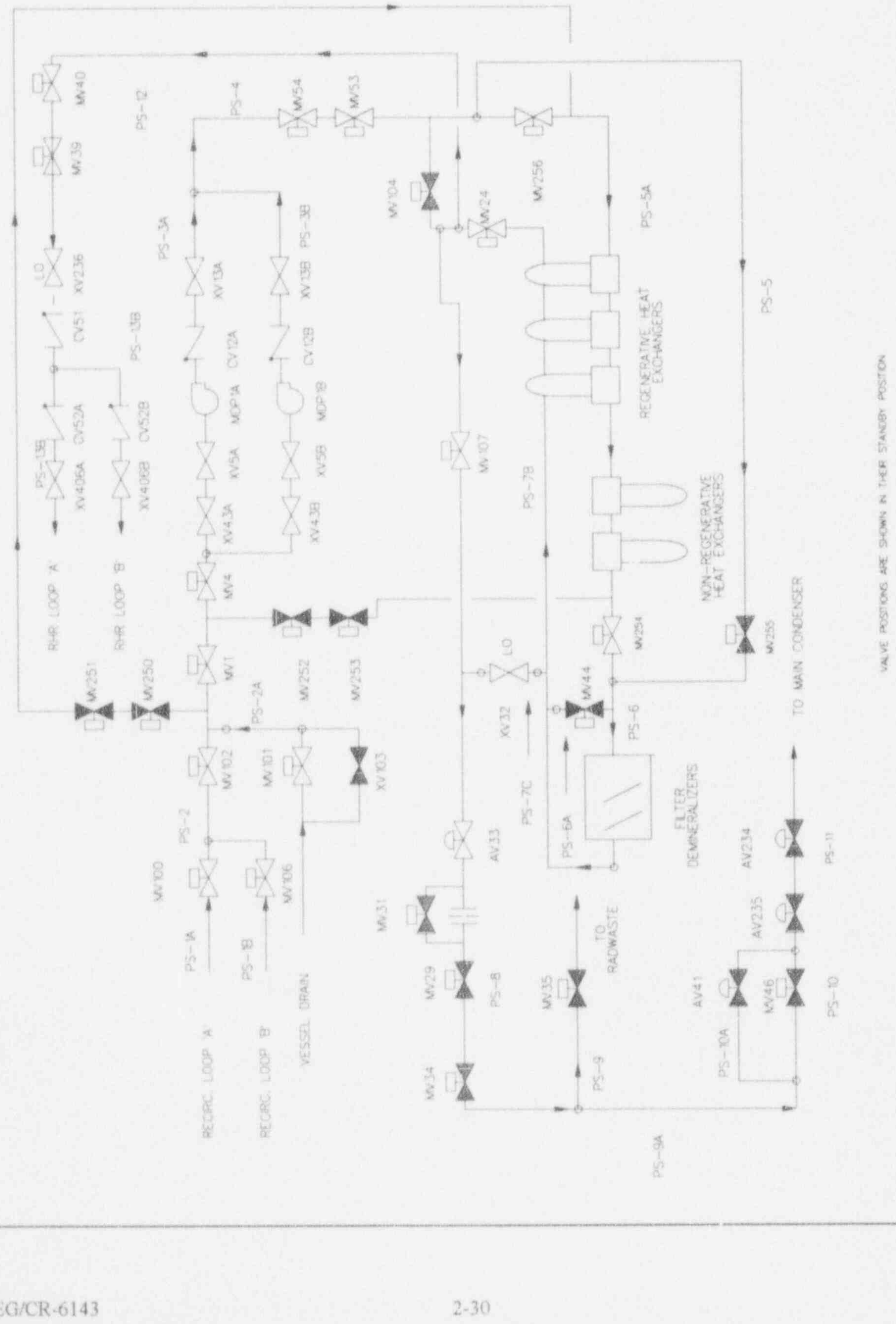

Figure 2.20 RWCU System Schematic

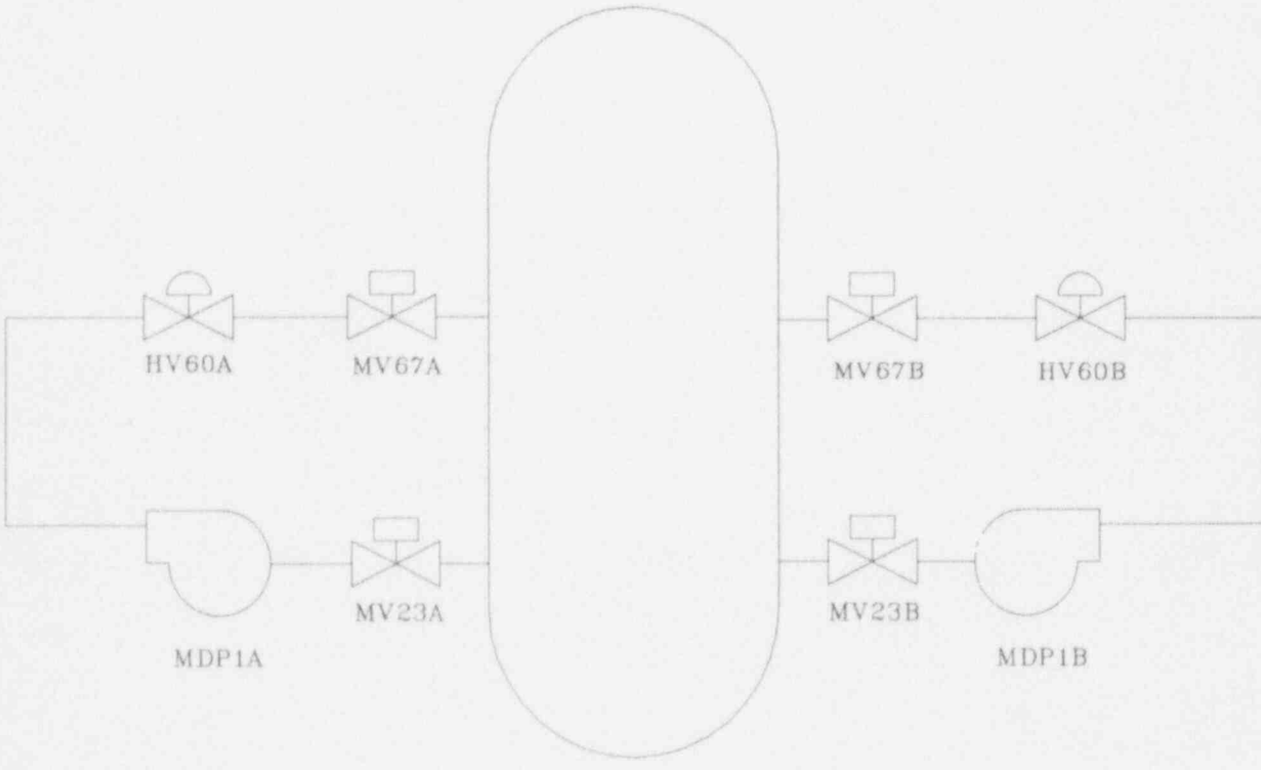

Figure 2.21 RRS System Schematic

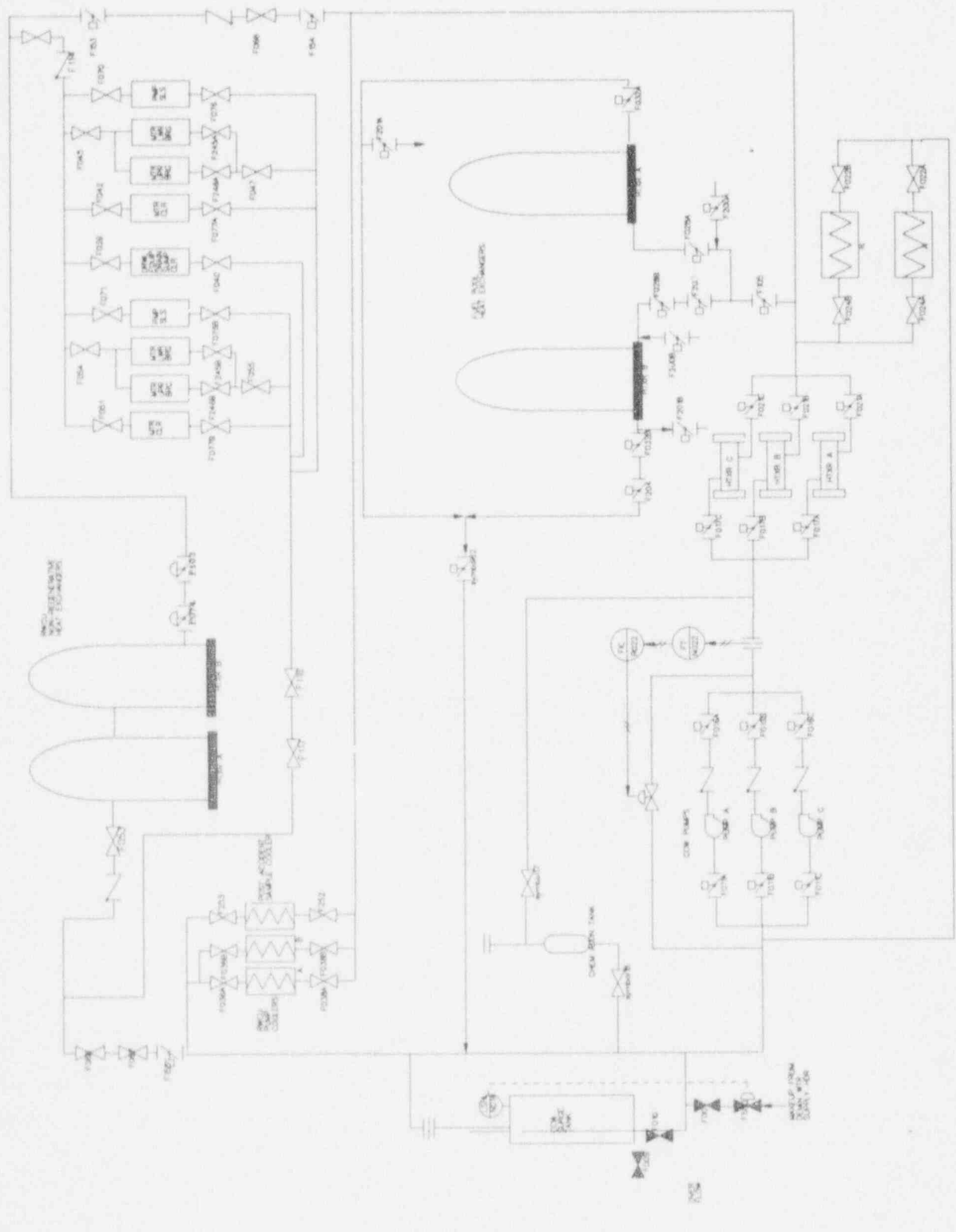

Figure 2.22 CCW System Schematic

- Recirculation pump seals and motor bearings,
- RWCU pumps and non-regenerative heat exchangers,
- CRD pump coolers,
- Fuel pool heat exchangers,
- Post accident sample cooler, and
- Drywell equipment drain sump cooler.

The CCW success criteria requires that two of the three CCW pumps and two of the three CCW heat exchangers be in operation. During LOSP, one pump operation can successfully cool essential loads if the non-essential loads (fuel pool heat exchangers and Reactor Water Clean Up non-regenerative heat exchangers) properly isolate.

The CCW system's major dependencies are: (1) the PSW system; (2) the SSW system; (3) 480V AC power; (4) 120V AC power; (5) 125V DC power; and (6) Instrument Air. The three heat exchangers all use PSW system water for cooling, with the SSW system as a backup source of cooling water in the event of LOSP or loss of PSW. The SSW system is also a backup cooling water source for the Fuel Pool heat exchangers during a LOSP when the Fuel Pool heat exchangers are isolated from the CCW system. Cooling from both the PSW and SSW systems is inhibited if a Loss Of Coolant Accident (LOCA) signal is present. A LOCA signal can consist of either a high drywell pressure or a low reactor vessel level signal. CCW pumps A, B, and C are powered from buses 11BD5, 16BB3, and 12BE2, respectively. The 120V AC power is used to control the surge tank level, RWCU isolation valve control, all alarms, test circuits, and motor-operated valve heaters. The 125V DC power is used as control power for the CCW pumps, the power to auxiliary relays for alarms, and the fuel pool heat exchanger A valves. The instrument air operates the air operated valves in the system.

2.2.21 Plant Service Water (PSW) System

The function of the PSW System is to provide cooling to various plant heat exchangers including the CCW heat exchangers.

The PSW system is an open-loop system which uses the Mississippi river bank radial wells as its source and the discharge basin and circulating water pit as its two sinks. See Figure 2.23 (1 of 2 and 2 of 2) for the PSW diagram. The PSW system draws water from four radial wells on the bank of the Mississippi river. Each well provides water to the suction of two pumps in parallel. The pumps are 4 stage, 5000 gpm centrifugally driven by a 500 hp motor. These eight pumps all feed the common PSW header to supply the loads.

The PSW system uses this untreated water to cool various heat loads and provide makeup to various treated water systems. The PSW supplies cooling water to the CCW heat exchangers, the Steam Jet Air Ejection (SJAE) intercondensers, the ADHR heat exchangers and A/C, and the Turbine Building Cooling Water (TBCW) heat exchangers. The PSW provides makeup for the following treated water systems: (1) the Circulating Water system; (2) the SSW system; and (3) the Firewater System and the Makeup Water system.

The PSW system's major dependencies are 4160 V, 480 V, and 120 V AC power and the Instrument Air System (IAS).

2.3 References

2.1 System Energy Resources, Inc., Grand Gulf Updated Final Safety Analysis Report, 1992.

2.2 D. Whitehead, et. al., Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1, Analysis of Core Damage Frequency from Internal Events for Plant Operation State 5 During a Refueling Outage, Main Report, NUREG/CR-6143, Vol. 2, Part 1, SAND93-2440, Vol. 2, Part 1, Sandia National Laboratories, Albuquerque, NM, 1994.

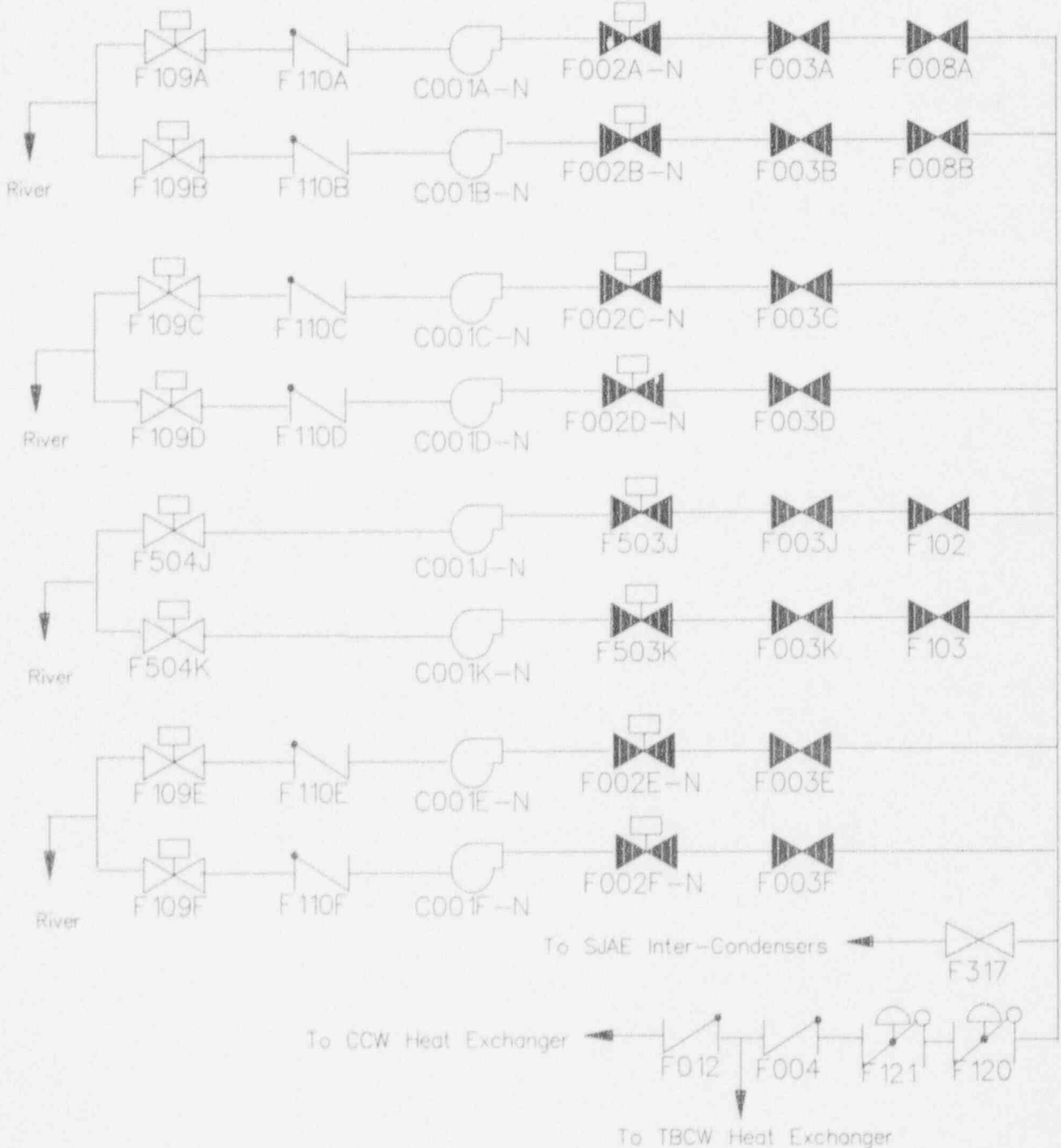

Figure 2.23 PSW System Schematic (page 1 of 2)

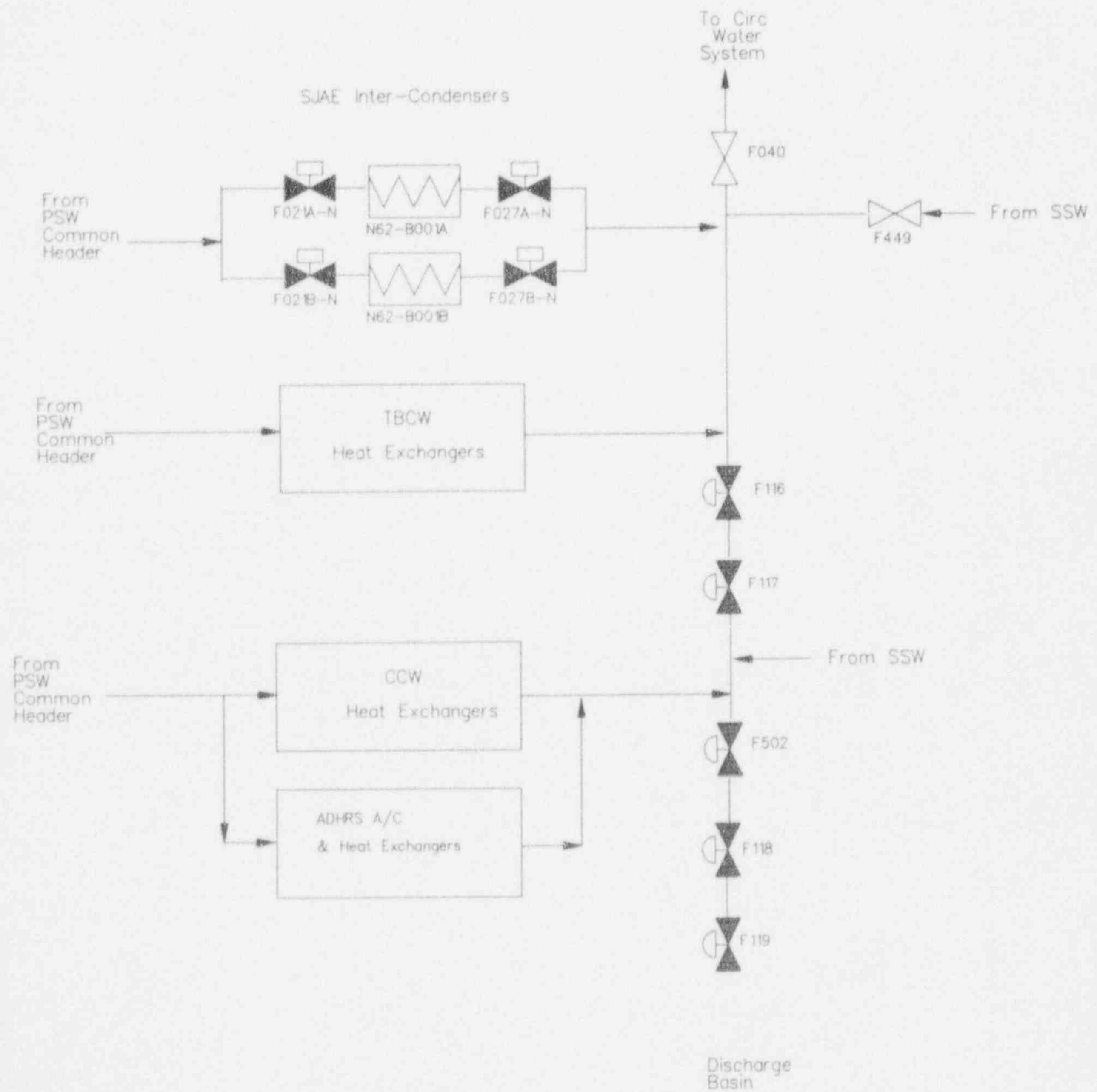

Figure 2.23 PSW System Schematic (page 2 of 2)

3. Grand Gulf Fire Analysis

3.1 Introduction

The objective of this analysis is to estimate the contribution of fire-induced events to core damage frequency during plant shutdown. This is a report of the analysis of such risk performed for the Grand Gulf plant in Plant Operational State (POS 5). During this state of plant operation, the plant is being maintained within the required parameters by operation of the RHR system in the SDC mode of operation. For a complete description of how the POSs were defined, see Section 3.2 the Grand Gulf Internal Events Analysis (NUREG/CR-6143) (Ref. 3.1).

3.2 Fire Locations Analyzed

In this section, the plant fire zones that were analyzed are listed and discussed. Table 3.1 provides a comprehensive listing of the fire zones and a brief physical description of each. All other fire zones not contained in Table 3.1 were eliminated from the analysis because they did not contain either vital equipment or cabling for safety- related equipment. Figure 3.1 presents a general plant layout drawing which illustrates main buildings where the fire zones in Table 3.1 are located.

Identification of the fire zones of interest, and characterization of the boundaries or barriers between zones was accomplished by a comprehensive analysis of plant layout drawings and the Grand Gulf Fire Hazards Analysis (Ref. 3.2) supplemented by plant walkdowns. The walkdowns were conducted to verify the selections made and to obtain clarifications where adequate information was not available from documentation. Typically, fire zones are bounded by 3-hr rated fire barriers; fire doors, fire-resistant walls, and fire dampers. The subdivision of the plant into fire zones is also accomplished by the division of the building into different levels (elevations). In the cases where open gratings separated levels, the rooms at the different elevations were considered part of a single fire zone.

3.3 Initiating Event Frequencies

3.3.1 Fire Data

Data on fires in commercial Light Water Reactors have been analyzed in several studies (Refs. 3.3, 3.4, 3.5). Although these studies have been done independently, they have some common characteristics. For example, almost all studies have used Licensee Event Report (LER) data from the Nuclear Regulatory Commission (NRC). All have reported the overall frequency of fires of approximately 0.16 incidents per reactor year on a plant wide basis. To determine fire initiating event frequencies, two kinds of information are needed: (1) the number of fire incidents that have occurred in specific compartments at shutdown during commercial operation, and (2) the number of compartment years that the nuclear industry has accumulated. Most of the data for the first part comes from reports of insurance inspectors to American Nuclear Insurers (ANI), although other sources are also used, e.g., the U.S. Nuclear Regulatory Commission.

While the NRC requires the reporting of fires that, in some way, affect the safety of the plant, the ANI has more stringent requirements in the sense that all fire events resulting in a property loss claim must be reported. Compartment years at shutdown are computed by adding the age of all compartments (within a certain category of compartments) of units that were in commercial operation by the end of December 1989. The age is defined as the time at shutdown between first commercial operation and the end of December 1989 (or date of decommissioning).

Events were only included if they occurred during shutdown. Eight general areas are typically found in nuclear power plants. These are (1) the control room, (2) cable spreading room, (3) diesel generator room, (4) reactor building, (5) turbine building, (6) auxiliary building, (7) electrical switchgear room, and (8) battery room. In most plants, the first three areas, and the electrical switchgear room and battery room, are single compartments while the other three are typically large buildings.

The fire events and operating years for the eight plant areas were obtained by using an update to the fire data base developed by Wheelis (Ref. 3.6).

To determine operating years for electrical switchgear rooms and battery rooms, auxiliary building operating years were doubled. A survey of all U.S. light water reactors indicated that there is an average of 2.25 trains of emergency switchgear and their associated batteries per plant. However, it is known that some plants such as Surry locate both trains of their emergency switchgear in one fire area. So it was assumed that an average number would be two per plant for both types of rooms.

To obtain fire area-specific initiating frequencies, a partitioning method is required. Partitioning is a process in which the analyst subdivides the frequency of fire occurrence from a large building (e.g., auxiliary building) among specific rooms or fire zones within that building. Also, further partitioning can occur within a specific room or area. One method of partitioning is accomplished by ratioing the areas of fire areas within a building. The assumption here is that the frequency of fire occurrence is dependent only upon the amount of area a fire area contains. Another method of partitioning examines

Table 3.1 Grand Gulf Fire Zone Descriptions

Fire Zone	Building	Elevation	Physical Description
1A101	Auxiliary	93' & 103'	Passage
1A102	Auxiliary	93'	RHR A Heat Exchanger Room
1A103	Auxiliary	93'	RHR A Pump Room
1A105	Auxiliary	93'	RHR B Pump Room
1A106	Auxiliary	93'	RHR B Heat Exchanger Room
1A109	Auxiliary	93'	HPCS Pump Room
1A110C1	Containment	135'4"	Electrical Containment Penetration Area
1A110C3	Containment	135'4"	Electrical Containment Penetration Area
1A110D3	Containment	161'10"	Containment Cooler
1A112	Containment	100' 9'	Drywell Area
1A114	Auxiliary	93' & 103'	Fan Coil Area
1A117	Auxiliary	93' & 103'	Miscellaneous Equipment Area
1A120	Auxiliary	93'	CCW Pump and Heat Exchanger Area
1A128	Auxiliary	93'	RHR A Heat Exchanger Room
1A129	Auxiliary	106'	RHR B Heat Exchanger Room
1A201	Auxiliary	119'	Passage
1A202	Auxiliary	119'	RHR A Heat Exchanger Room
1A203	Auxiliary	119'	RHR A Piping Penetration
1A204	Auxiliary	119'	Access to RCIC Room
1A205	Auxiliary	119'	RHR B Piping Penetration
1A207	Auxiliary	119'	Electrical Switchgear Room Div II
1A208	Auxiliary	119'	Electrical Switchgear Room Div I
1A210	Auxiliary	115'	RWCU Recirculation Pump B Room
1A211	Auxiliary	119'	Misc. Equipment Area

Table 3.1 Grand Gulf Fire Zone Descriptions (Continued)

Fire Zone	Building	Elevation	Physical Description
1A215	Auxiliary	119'	Fan Coil Area
1A222	Auxiliary	119'	Motor Control Center Area
1A301	Auxiliary	139'	Corridor
1A302	Auxiliary	139'	Corridor
1A305	Auxiliary	140'	Main Steam Tunnel
1A307	Auxiliary	139'	RHR B Heat Exchanger Room
1A308	Auxiliary	139'	Electrical Penetration Room Div. II
1A309	Auxiliary	139'	Electrical Penetration Room Div. I
1A313	Containment	135' 4"	CRD Hydraulic Control Area
1A314	Auxiliary	139'	Passage
1A316	Auxiliary	139'	North Passage Area
1A318	Auxiliary	139'	Electrical Penetration Room
1A322	Auxiliary	139'	Centrifugal Chiller Area
1A401	Auxiliary	166'	Passage
1A403	Auxiliary	166'	Passage
1A407	Auxiliary	166'	Motor Control Center
1A410	Auxiliary	166'	Motor Control Center
1A414	Containment	170'	RWCU Heat Exchanger Room
1A417	Auxiliary	166'	Misc. Equipment Area
1A420	Auxiliary	166'	Misc. Equipment Area
1A428	Auxiliary	166'	Passage
1A539	Auxiliary	185'	Cable Space Div I & II
1M110	SSW Pump House	133'	SSW Pump House Div I
2M110	SW Pump House	133'	SSW Pump House Div II
1D310	Diesel Generator	133'	Div I Diesel Generator

Table 3.1 Grand Gulf Fire Zone Descriptions (Concluded)

Fire Zone	Building	Elevation	Physical Description
OC116	Control	93'	Hot Machine Shop
OC202	Control	111'	Div I Switchgear Area
OC207	Control	111'	Div I Battery Room
OC208	Control	111'	Emergency Hot Shutdown Room Div II
OC208A	Control	111'	Emergency Hot Shutdown Room Div I
OC209	Control	111'	Div III Battery Room
OC210	Control	111'	Div III Switchgear Room
OC211	Control	111'	Div II Battery Room
OC215	Control	148'	Lower Cable Spreading Room
OC402	Control	148'	Corridor
OC403	Control	148'	Computer & Control Panel
OC407	Control	148'	Instrument Motor Generator
OC409	Control	148'	Electrical Space
OC503	Control	166'	Control Room Area
OC504	Control	174' 6"	Suspended Ceiling above Instr Rack
OC601	Control	177'	Viewing Gallery
OC702	Control	189'	Upper Cable Spreading Room
OC703	Control	190'	Control Cabinet Area Div I
OM101	Fire Water Pumphouse	133'	Diesel Driven Fire Pump Room A
OM103	Fire Water Pumphouse	133'	Diesel Driven Fire Pump Room B
IT118	Turbine Building	93'	TBCW Pump Area
IT132	Turbine Building	93'	East Corridor
IT214	Turbine Building	113'	Motor Control Center Area
IT325	Turbine Building	133'	Filters Instr. Rack Area
IT327	Turbine Building	133'	Steam Instr. Rack Area
IT404	Turbine Building	166'	Hatches Area

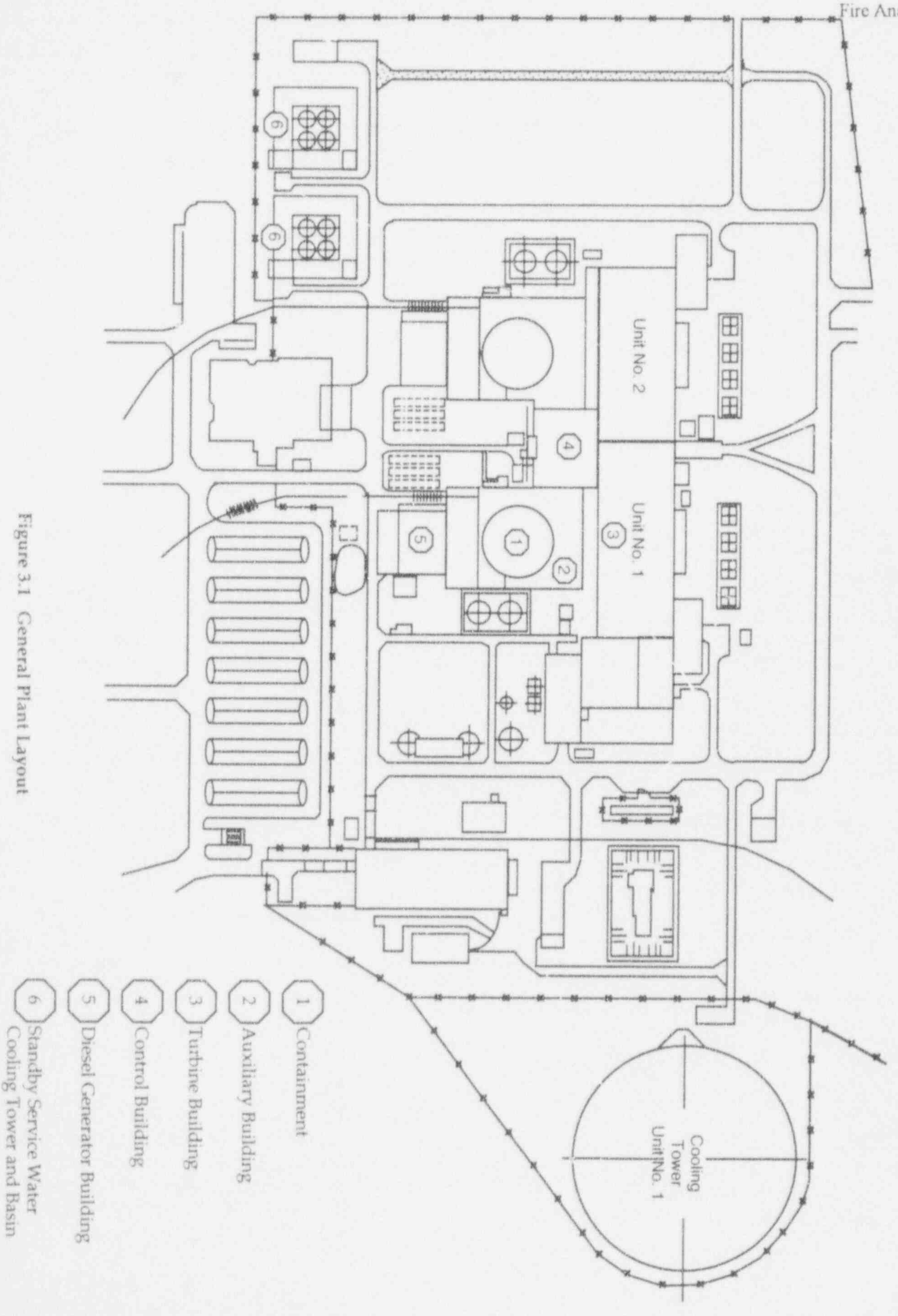

Figure 3.1 General Plant Layout

area-specific factors important to the probability of fire initiation. These factors are the amount of electrical components and cabling, the fire loading, whether the fire area is controlled, and how often the fire area is occupied.

To assist in the process of partitioning in a large building or within a specific fire zone in that building, a checklist was used on the initial plant visit to determine the most probable fire initiating sources. Also, data on past fire occurrences was thoroughly reviewed. This review revealed that there were three fire occurrences at Grand Gulf and all events were pre-operational involving fires in the diesel generator building, and thus were not counted in the computation of Grand Gulf fire frequency.

3.3.2 Bayesian Updating of Fire Frequencies

The generic fire occurrence data were updated using a method developed by Iman (Ref. 3.7) to determine plant-specific LOSP frequencies. By changing the data from LOSP events to fire events and determining explicitly the number of rooms, this method can be used to calculate plant-specific fire-initiating event frequencies at shutdown. This Bayesian approach models the incidence rate of fires for each plant relative to the incidence rates of all other plants, and the posterior distribution is found for the incidence rate of each plant. For this analysis, the gamma distribution is used as a model as described in Reference 3.7, although many other distributions could be used. The probability density function for the two-parameter gamma distribution is:

$$h(\lambda) = f_{\gamma}(\lambda|\alpha, \beta) = \beta^{\alpha}[\Gamma(\alpha)]^{-1} \lambda^{\alpha-1} e^{-\beta\lambda} \; \lambda \geq 0, \alpha, \beta > 0$$

The parameters α and β are unknown, and the non-informative prior is:

$$P(\alpha,\beta) \;\; \alpha \; 1/(\alpha\beta) \;\; \alpha,\beta > 0$$

The likelihood function of the datum (s_i,t_i) is Poisson

$$L(s_i, t_i | \lambda_i) = (\lambda_i t_i)^{s_i} \;\; e^{-\lambda_i t_i / s_i !}$$

The posterior density can, therefore, be expressed as:

$$p^{*}(\alpha,\beta,\lambda_0,...\lambda_n) =$$

$$\frac{p(\alpha,\beta) \prod_{i=0}^{n} [h(\lambda_i) \, L(s_i,t_i|\lambda_i)]}{\int_0^{\infty} \int_0^{\infty} \int_0^{\infty} \cdots \int_0^{\infty} p(\alpha,\beta) \prod_{i=0}^{n} [h(\lambda_i) \, L(s_i,t_i|\lambda_i)] \, d\alpha \, d\beta \, d\lambda_0 \ldots d\lambda_n}$$

Using the above relationships, plant-specific fire-initiating event frequencies and distributions were developed.

3.4 Determination of Fire- Induced "Off-Normal" Plant States

One of the most critical steps in a fire analysis is to determine, on a plant-specific basis, which of a wide range of possible initiating events have the potential to be induced due to a fire occurrence.

As in the Grand Gulf internal events analysis, a comprehensive list of initiators was identified for further study. It is known from a review of previous fire PRAs that only a limited set of initiating events have the potential to be significant contributors to fire-induced core damage frequency. Typically, initiating events such as large or medium LOCAs caused directly by the fire have not been analyzed because the vulnerabilities of a piping system or tanks to fire events are considered insignificant. Table 3.2 lists the initiating events that were determined to have potential to be induced due to a fire occurrence.

The same fault trees and event trees that were used in the internal events analysis were utilized in the fire analysis. Thus, the level of analytical detail was consistent with the level in the internal event analysis.

3.4.1 Initiating Events

T_{5A}: Loss of all Standby Service Water (SSW)

For POS 5 it is assumed that train A of the SSW system is unavailable. Therefore this event is the loss of SSW pump B.

T_{5B}: Loss of all Turbine Building Cooling Water

For POS 5 this event involves the loss of 3 of 3 TBCW pumps.

T_{5C}: Loss of all Plant Service Water (includes Radial Well)

The success criteria for PSW is the continued operation of 1 of 8 PSW pumps. The loss of PSW would prevent the use of Component Cooling Water (CCW) system, the Circulating Water system, the Turbine Building Cooling Water (TBCW) system and the Alternate Decay Heat Removal System (ADHRS).

Table 3.2 Grand Gulf Fire-Induced Initiating Events Analyzed

Initiating Events	Designator	Screening Criteria
Loss of SDC common suction line	E2B4H	1,2,3
Loss of SDC loop B only	E2T4H	1,2,3
Loss of all SSW	T5A4H	1,4
Loss of all TBCW	T5B4H	1,4
Loss of all PSW (includes Radial Well)	T5C4H	1,4
Loss of all CCW	T5D4H	1,4
Loss of 1E 4160 V AC Bus B	TAB4H	1,2,3
Loss of 1E125 V DC Bus B	TDB4H	1,2,3
Loss of instrument air	TIA4H	1,2,3
Inadvertent overpressurization (makeup greater than letdown)	TIOP4	1,3
Inadvertent overpressurization via spurious HPCS actuation	TIHP4	1,3
Inadvertent overfill via LPCS or LPCI	TIOF4	1,3
Loss of makeup (CRD)	TLM5H	1, 2, 3, 4, 5
Loss of Recirculation Pump	TRPT5	1, 2, 3, 4, 5

Screening Criteria

1. Computer Aided
2. Credit for Automatic FPS Coverage
3. Recovery of Random Failures
4. Plant Inspection determined initiator could not be induced by fire
5. Detailed fire propagation modeling

T_{SD}: Loss of all Component Cooling Water

The loss of the CCW pumps would prevent the use of either recirculation pump, CRD pumps, FPCCU and RWCU. The success criteria for the CCW pumps is that 2 of 3 pumps and 2 of 3 heat exchangers be in operation.

T_{AB}: Loss of 1E 4160 V AC Bus B

This event involves the loss of 1E 4160 V AC Bus B and its associated loads. For this event, we screened out loss of non-1E buses as initiating events.

T_{DB}: Loss of 1E 125 V DC Bus B

The loss of a DC bus generally falls into one of two categories. They are: 1) failure to provide DC power on demand as characterized by the loss of charger output, and 2) operational, test, or maintenance errors resulting in the loss of DC power during normal plant operation. The first category can be fire-induced while the second category is random.

The principal cause of failure for the first category involves operation of the DC power system with one or more batteries unable to provide sufficient power to the bus if battery charger output is lost.

The loss of non-1E buses was screened out as initiating events.

T_{IA}: Loss of Instrument Air

This event involves loss of the service and instrument air compressors or their associated power supplies and cabling.

T_{IOP}: Inadvertent Overpressurization (Makeup Greater Than Letdown)

For POS 5 the T_{IOP} initiator is essentially loss of Reactor Water Cleanup system (RWCU) (letdown) and failure of the operator to stop CRD or control pressure.

T_{IHP}: Inadvertent Pressurization via Spurious HPCS Actuation

This event involves the spurious actuation of the High Pressure Core Spray system and subsequent inadvertent pressurization.

T_{IOF}: Inadvertent Overfill via LPCS or LPCI

This event involves the inadvertent overfill of the vessel via the Low Pressure Core Spray or the Low Pressure Core Injection system.

T_{LM}: Loss of Makeup (CRD)

This event involves the loss of CRD which is a two pump system. One CRD pump is sufficient for level control, but two CRD pumps are required to provide adequate makeup for steaming the core.

T_{RPT}: Loss of Recirculation Pump

This event involves the loss of Recirculation pump B.

E_{21}: Loss of Common RHR Suction Line

This event involves the loss of either the inboard and/or outboard MOV(s) in the common RHR suction line.

E_{2B}: Loss of Operating RHR-Shutdown Cooling System

This event involves major interruptions in the operating SDC system. These events essentially cause a long term loss of SDC from the previously operating train(s).

3.5 Detailed Description of the Screening Analysis

A comprehensive screening analysis is required to reduce the number of potential fire-induced scenarios to only those which have the potential to be probabilistically significant contributors to core damage frequency.

The screening analysis was composed of the following four steps:

Step 1. Identification of Relevant Fire Areas

Fire areas containing equipment or cables associated with safety-related systems which mitigate the effects of the unscreened fire-induced "off-normal" plant states were identified. All other fire areas were eliminated from further analysis. This resulted in the identification of relevant fire areas which are listed in Table 3.1.

Step 2. Further Screen Fire Areas Based on Vital Area Analysis

The remaining fire areas were subjected to a vital area analysis (location mapping) of components and cables (both control and power cables) that were located within these areas. This information was integrated into the PRA with the IRRAS computer code (Ref. 3.8), which was then used to solve all front line systems and solve all of the identified sequences.

Fire occurrence frequency for each area was set to 1.0 and, given a fire, all components within that area were assumed to fail. The output of this analysis is a series of accident sequences expressed in terms of cut sets involving single or multiple fire areas as well as random failures (i.e., not fire related).

Cut sets which required three or more fire zones were eliminated. This was deemed appropriate since these cut sets imply the failure of two or more fire barriers or large fires of sufficient duration that there is a high likelihood of suppression. Cut sets which contained two fire zones were screened on the following two criteria: (1) no adjacency between areas and (2) no penetrations in the adjacency between areas. An additional screening step to be performed on the cut sets which contained two fire zones is numerical culling with barrier penetration failure set to a screening value of 0.1. It is known from the analysis of many fire barriers that typical random failure rates are on the order of 10^{-2} to 10^{-3}. Therefore, this screening value has been set high enough to ensure that potentially important fire area combinations are not lost.

Truncation of cut sets at a random failure probability of 10^{-4} was performed. When all fire-related factors are taken into account, the frequency of any scenario is at least three orders of magnitude less than this random failure truncation probability.

Important information gained from the analysis of these cut sets was identification of the remaining plant locations where area-to-area barriers needed to be analyzed. Dominant cut sets which contain adjacent fire areas were analyzed for barrier failure in the quantification process.

Step 3. Cull Fire Areas on Frequency and Non-recovery

Cut sets not eliminated in the first two screening steps are requantified with fire area specific initiating event frequencies that were calculated utilizing the methodology described in Section 3.3.

Additionally, operator recovery of non-fire-related random failures was analyzed. For screening purposes only, all short term (less than 24 hr) recovery actions (of non-fire failures) were increased from their respective internal events probabilities by a factor of five to allow for the additional confusion of the fire situation occurring in conjunction with other random failures. Where recovery actions were long term (greater than 24 hr), no modifications to internal event probabilities were deemed necessary. It was assumed that after 24 hours, the fire would be extinguished and any spurious signals will have terminated in open circuits.

Step 4. Confirmatory Plant Visit

For the fire areas surviving the screening process to this point, all scenarios are then associated with equipment fire-related failures that were identified during the plant walkdown. A scenario can be thought of as a combination of one or more fire-related equipment failures within a fire area with or without additional non-fire-related (random) failures outside of the fire area. These failure combinations must minimally lead to core damage. Each fire area can have one or more scenarios depending on the equipment combinations which might fail due to the fire in that particular area. Additional plant visits were conducted to determine which of the postulated scenarios were valid based on cable or equipment locations within a particular fire area. Past experience with fire code calculations, discussed in the following section, and fire testing results provide the basis for assessing the validity of scenarios.

For example, if a scenario required the fire-related failure of cabling for components A and B and it could be shown that these cables were always separated by greater than 40 ft in an area of sufficient size to preclude buildup of a hot gas layer, or at least one of the component's cabling was in a 3-hr rated fire wrap, then that scenario was eliminated from further analysis.

3.6 Fire Propagation Modeling using COMPBRN IIIe

3.6.1 Introduction

The COMPBRN fire growth code (Ref. 3.9) was used to calculate fire propagation and equipment damage. COMPBRN was developed specifically for use in nuclear power plant fire PRAs. The code calculates the time required to damage critical equipment given that a fire has started. This failure time is then used in conjunction with plant specific information on fire suppression to obtain the probability that a given fire will cause equipment failure which leads to core damage before the fire can be suppressed.

COMPBRN employs a quasistatic approach to simulate the process of fire during the preflashover period in an enclosed space. COMPBRN uses an area model which breaks the fire environment into three sub-areas: flame/plume, hot gas layer, and ambient (see Figure 3.2). Simple fire and heat transfer models and correlations are employed to predict the thermal environment as a function of time. The thermal response of various targets in the fire scenario is modeled to predict the amount of time required for a fire to damage or ignite critical equipment. The critical equipment is generally taken to be a cable tray carrying cables necessary for safe shutdown of the plant,

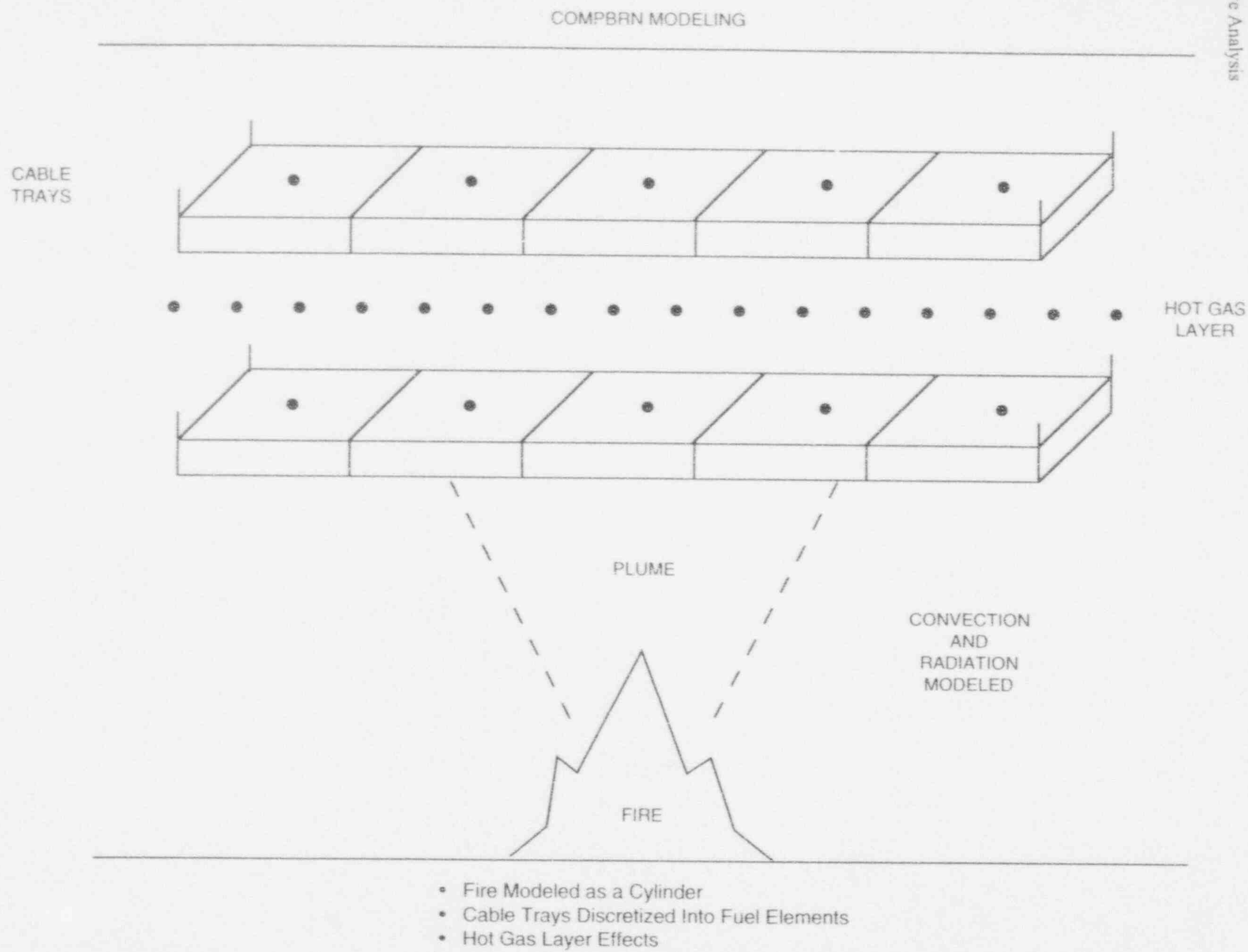

Figure 3.2 COMPBRN Zone Model

although other critical components, such as pumps, may be modeled.

3.6.2 Grand Gulf Fire Propagation Modeling

Walkdowns of the Grand Gulf nuclear plant were conducted to obtain the code input information required to perform the COMPBRN IIIe fire damage assessment calculations. This information includes the location of critical equipment and cable trays, separation between redundant trains, types of cable present, and any shielding or fire barrier material that may be present.

Both small and large fires were postulated in the calculations for Grand Gulf. A small fire was assumed to be 2 ft. (0.61m) in diameter and consist of 1 gal (3.8 l) of oil. A large fire was assumed to be 3 ft. (0.91 m) in diameter and consist of 10 gal (38 l) of oil. Analysis of a data base of transient combustible fuel sources found at nuclear power plants (Ref. 3.9) indicates that oil sources less than or equal to 1 gal (3.8 l) were found approximately 70 percent of the time. Oil sources larger than this were found approximately 30 percent of the time. A similar partitioning between small and large quantities in terms of heat content (BTU or KJ) can be made for other credible transient combustible sources such as solvents or trash paper. Again, analysis indicates that a 70/30 partitioning between small and large fuel sources is appropriate (within $\pm$10 percent). It can also be shown that 10 gal (38 l) of oil bounds any large solvent or trash paper combustible source in terms of heat content and is, therefore, an appropriate upper bound on transient combustible fuel source size.

Cable insulation ignition and damage thresholds are currently not well known. For this study, a cable insulation ignition temperature of 773 K (932°F) was assumed along with a damage temperature of 623 K (662°F). For the large fire simulations, these thresholds are not as critical to the fire damage time calculations because of the intensity of the flames. The uncertainty in the flammability parameters along with the uncertainty in other parameters was considered by the fire experts in interpreting the COMPBRN results and in constructing the probability distributions on time to damage.

A list of input parameters for the COMPBRN calculations is shown in Table 3.3. These parameters were selected based on past fire analyses at commercial nuclear facilities to represent typical qualified cable insulation. It was assumed, based on cable tray inspection in some critical fire areas that the cabling in all the areas of interest at Grand Gulf included typical brands of nuclear qualified cable. Because of the good flame resistance properties of these cables, no self-ignited (electrically initiated) cable tray fires were postulated.

The COMPBRN results are shown in Table 3.4 for the critical areas. A number of scenarios were considered for many of these areas. In most cases, a "zone of influence" was determined for the equipment and fire sizes modeled. In other words, the fire location was varied in the COMPBRN models to determine the maximum distance the fire could be away from the critical equipment and still cause damage. The times to damage increase exponentially as the fire distance increases. The numbers given in Table 3.4 represent the combination of greatest distance and longest times to damage. Using these results, the floor area in which a fire would have to occur to damage critical cables can be estimated. An area ratio can then be calculated by dividing this area by the total floor area of the room, fire area, or building (as appropriate). This reduction factor can then be multiplied by the initiating frequency to estimate the frequency of fires which occur in a critical portion of a given room.

It should be noted that a small fire, except for zone of influence cases, does not yield damage in many of the areas. Prior experience with COMPBRN shows that a small fire must be very close to its target to yield damage. Large fires, however, can and do yield damage in most of the areas. In large open rooms such as the auxiliary building corridor, the larger fire must still be within about 3 ft. horizontally of the target cable trays (assuming typical tray heights). The major exception is in small closed rooms in which a hot gas layer rapidly develops. In such cases, the hot gas layer effects become quite significant.

It has been found in past experience with COMPBRN and in some of the simulations for Grand Gulf that the COMPBRN results can be quite sensitive to fires located adjacent to walls which are in close proximity to the target cable trays. Using the typical model of the wall as one section results in unrealistic radiative heat fluxes from the wall to the cable trays of interest. For these cases, the wall was divided into several sections to more realistically calculate the wall thermal response.

The following subsections discuss the approach and results for the scenarios analyzed in each of the critical fire areas. Scenarios can be thought of as unique equipment combinations which must sustain fire damage. All entries in Table 3.4 which are blank indicate no predicted fire damage.

3.6.3 Fire Zone 1A117, North Hallway 93'/103' Elevation of the Auxiliary Building

3.6.3.1 Discussion

The scenario in this area calls for damage to two cable trays and involves the loss of CRD pump A and a RHR valve.

Table 3.3 COMPBRN IIIe Input Parameters

Cable Insulation Parameters	
Density	1715 kg/m^3
Specific Heat	1045 J/kg-K
Thermal Conductivity	0.092 W/m-K
Heat of Combustion	1.85-2.31E-7 J/kg
Combustion Efficiency	0.6-0.8
Critical Temperature	
Piloted Ignition	773K
Spontaneous Ignition	773K
Damage	623K
Surface Controlled Burning Rate	0.0001-0.0075 kg/m^2-S
Burning Rate Radiation Augmentation	1.86E-7 kg/J-m^2
Radiative Fraction	0.3-0.5
Smoke Attenuation Factor	1.4
Reflectivity	0.1-0.3
Oil Parameters	
Density	900 kg/m^3
Specific Heat	2100 J/kg-K
Heat of Combustion	4.67E7 J/kg
Combustion Efficiency	0.9
Surface Controlled Burning Rate	0.06
Radiative Fraction	0.3-0.5
Mass of Oil	3.4-34.0 kg

Table 3.4 Summary of COMPBRN Results

		Large Fires		Small Fires	
Fire Zone	**Location**	**Time**	**Area Ratio**	**Time**	**Area Ratio**
1A117	93'/103' Elevation	15 min.	4.0E-2	----	----
1A201	119' Elevation	5 min.	1.7E-2	-----	----
1A211 (Scen 1)	119' Elevation	2 min.	3.0E-2	3 min.	3.0E-3
1A211 (Scen 2)	119' Elevation	2 min.	3.0E-2	3 min.	3.0E-3
1A316 (Scen 1)	139' Elevation	2 min.	4.0E-2	3 min.	7.0E-3
1A316 (Scen 2)	139' Elevation	2 min.	4.0E-2	3 min.	7.0E-3

The closest point of approach of both cable trays to each other is 3.6 m or 11.8 ft. This hallway area is part of a larger hallway area that surrounds containment on this elevation (see Figure 3.3). Since this area is so large, no hot gas layer was modeled.

3.6.3.2 Results

No small fire was found to damage to both cable trays simultaneously. However, a large fire caused damage to both trays in 15 minutes.

3.6.4 Fire Zone 1A201, East Hallway, 119' Elevation of the Auxiliary Building

3.6.4.1 Discussion

The fire scenario in this area calls for the damage to two cable trays containing cabling for a RHR valve and recirculation pump B. The cable tray containing the RHR valve is approximately 9.4 ft off the floor with another cable tray situated 3 ft below it. The cable tray with recirc pump B cabling runs perpendicular to and above the cable tray with the RHR valve. The closest point of approach for these cable trays is approximately 4 ft. This area is part of a larger hallway area which surrounds the containment on this elevation (see Figure 3.4). Therefore, a hot gas layer was not modeled.

3.6.4.2 Results

For this scenario a small fire was found not to damage to both cable trays. A large fire caused damage to both trays in 5 minutes.

3.6.5 Fire Zone 1A211, North Hallway 119' Elevation, Auxiliary Building

3.6.5.1 Discussion

There are two scenarios which could lead to core damage. Both scenarios involve cable trays in the same general area.

Scenario 1 - Cable trays along the north wall

The target cable trays for this scenario run from the floor to the ceiling along the north wall in this hallway area, where partial sprinkler coverage protection is provided. This first scenario for this area involves two cable trays separated by 2.5 ft. Both of these trays are 1 ft wide and are situated approximately 2 ft from the wall. Since these trays are located in a large open hallway (see Figure 3.5), no hot gas layer was modeled.

Scenario 2 - Cable trays along the north wall

The target cable trays for this scenario include one from the previous scenario and an additional cable tray which is 3 ft wide and also is situated approximately 2 ft from the wall. These cable trays are also separated by approximately 2.5 ft. Since these cable trays are located in a large open hallway, no hot gas layer was modeled.

3.6.5.2 Results

For both scenarios, a small fire was found to cause damage to both cable trays in approximately 3 minutes. A large fire caused damage to both trays in approximately 2 minutes.

3.6.6 Fire Zone 1A316, North Hallway Area, 139' Elevation, Auxiliary Building

3.6.6.1 Discussion

There are two scenarios which could lead to core damage. Both scenarios involve cable trays in the same general area.

Scenario 1 - Cable trays along the north wall

The target cable trays for this scenario run from the floor to the ceiling along the north wall in this hallway area, where partial sprinkler coverage protection is provided. This first scenario for this area involves two cable trays separated by 2.5ft. Both of these trays are 1 ft wide and are situated approximately 2 ft from the wall. Since these trays are located in a large open hallway (see Figure 3.6), no hot gas layer was modeled.

Scenario 2 - Cable trays along the north wall

The target cable trays for this scenario include one from the previous scenario and an additional cable tray which is 3 ft wide and also is situated approximately 2 ft from the wall. These cable trays are also separated by approximately 2.5 ft. Since these cable trays are located in a large open hallway, no hot gas layer was modeled.

3.6.6.2 Results

For both scenarios, a small fire was found to cause damage to both cable trays in approximately 3 minutes. A large fire caused damage to both trays in approximately 2 minutes.

3.7 Barrier Failure Analysis

In the unscreened cut sets where a potential for barrier failure was identified, barrier failure probability was estimated using barrier failure rates developed as part of the

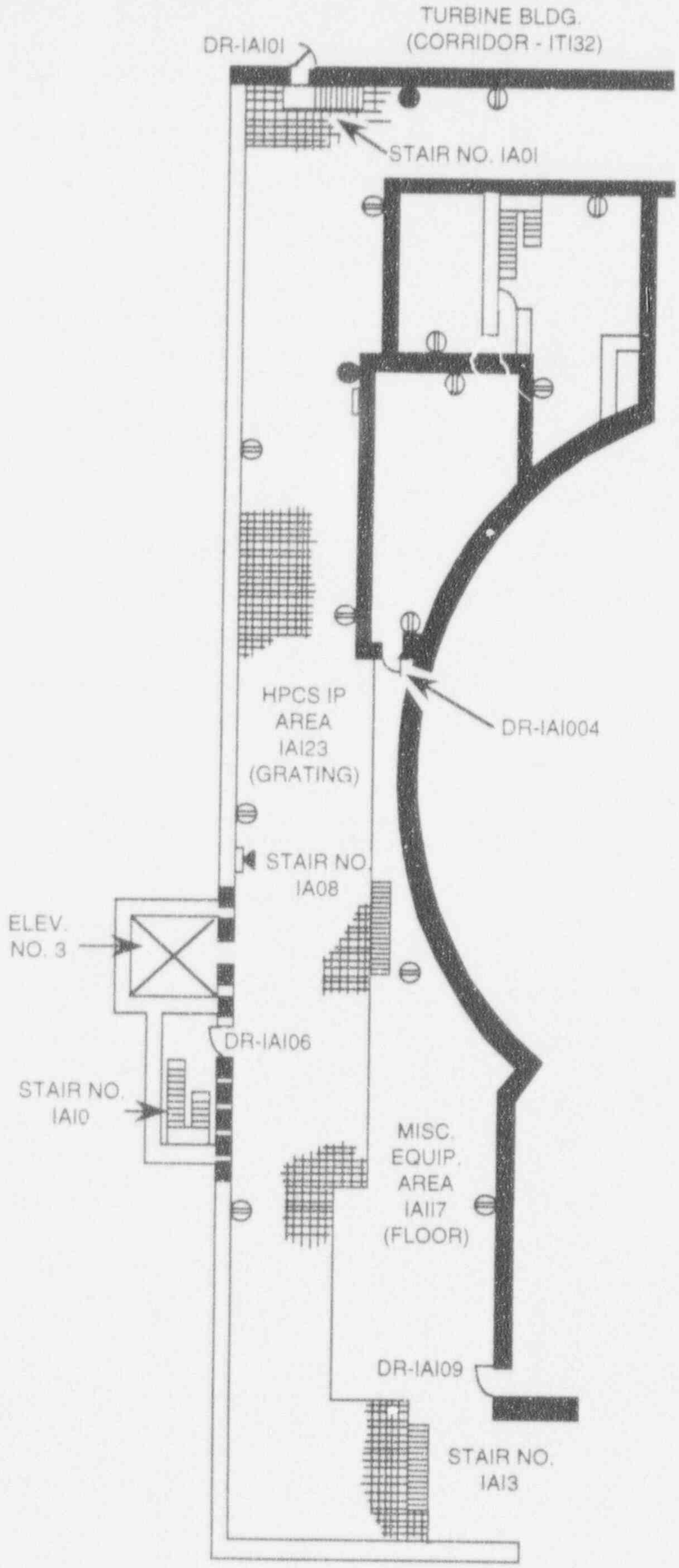

Figure 3.3 Fire Zone 1A117 Layout

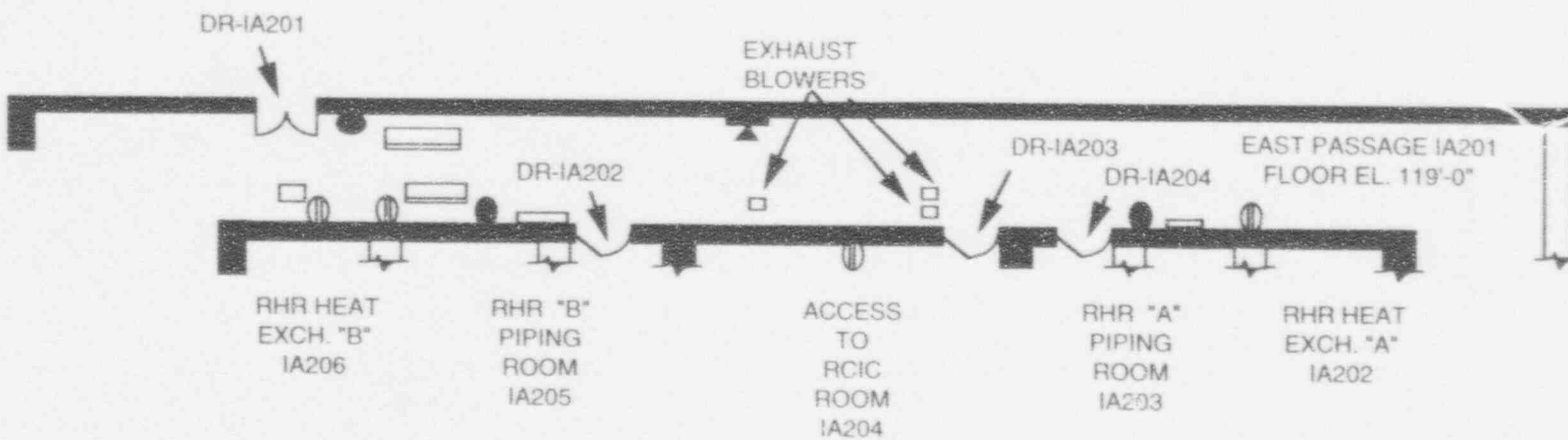

Figure 3.4 Fire Zone 1A201 Layout

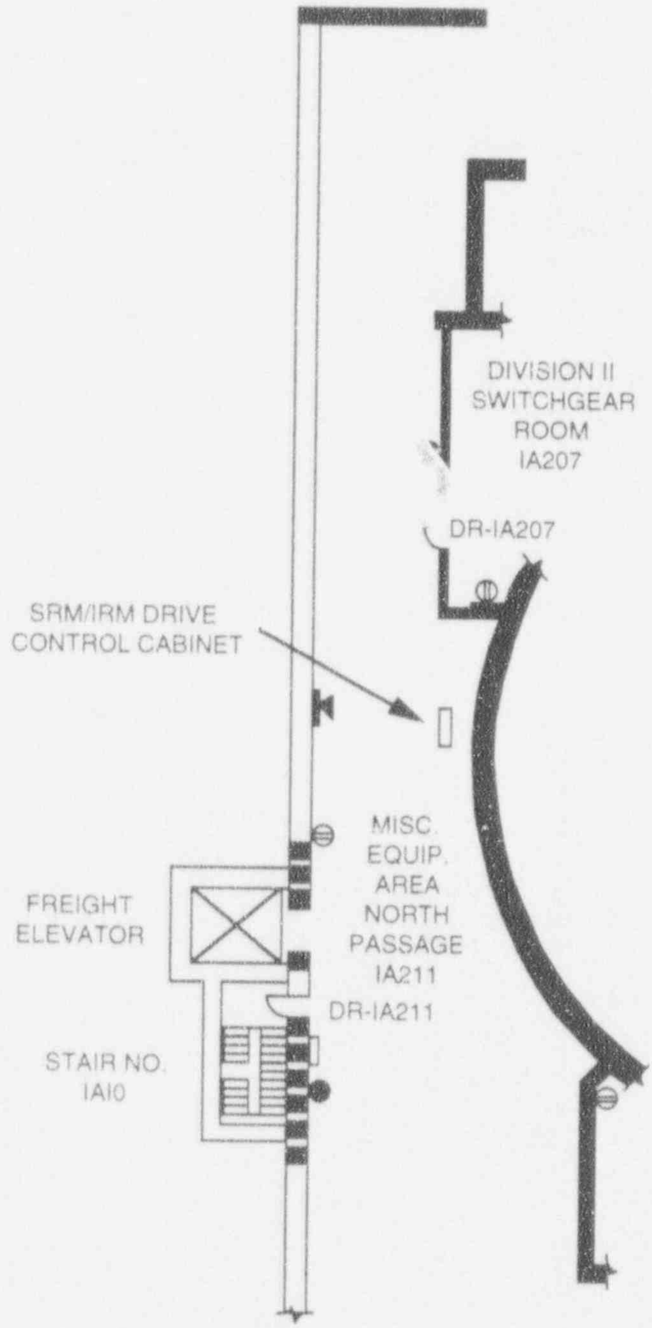

Figure 3.5 Fire Zone 1A211 Layout

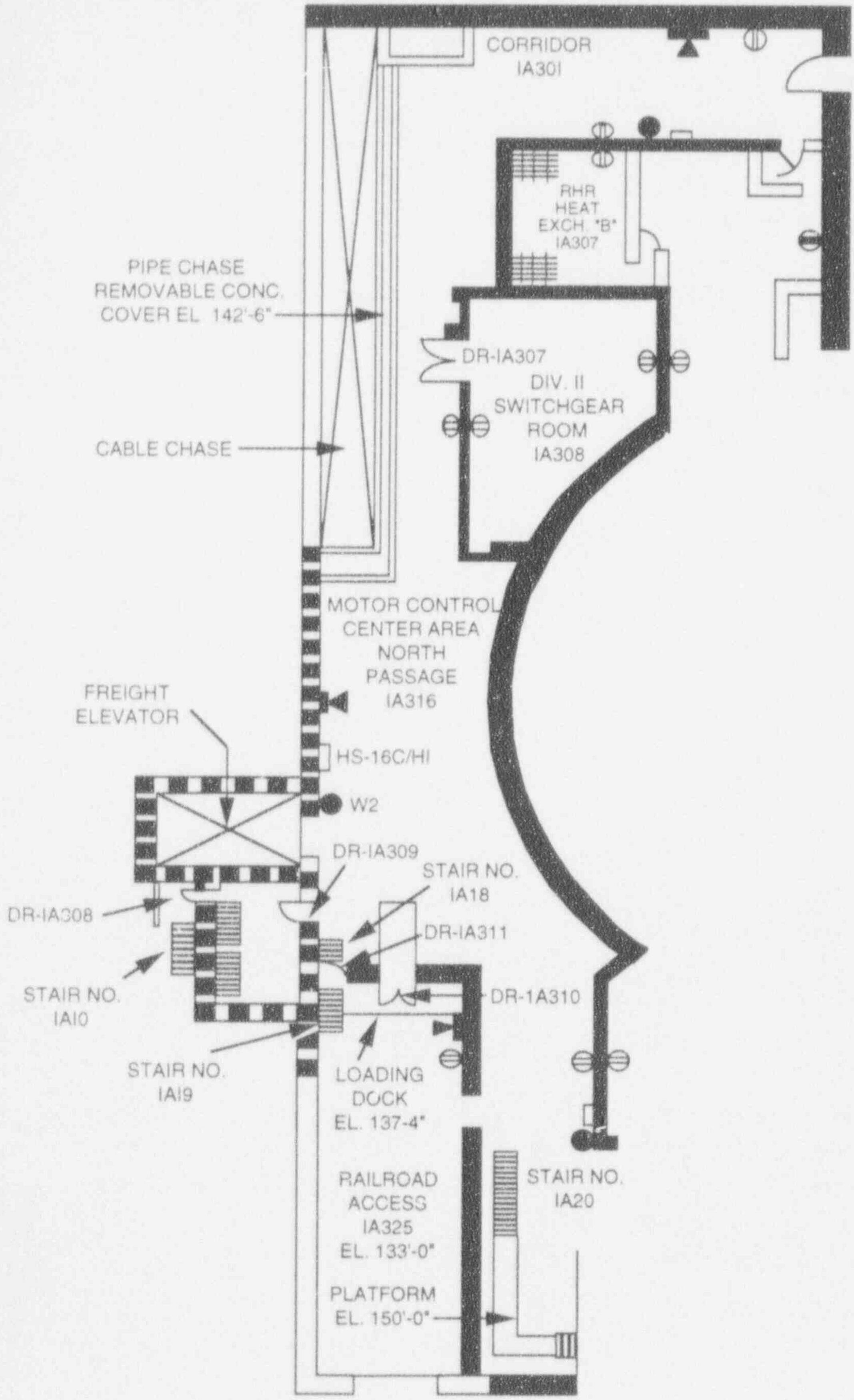

Figure 3.6 Fire Zone 1A316 Layout

Risk Methods Integration and Evaluation Program (RMIEP) analysis (Ref. 3.10). Barriers were grouped into three types: (1) fire doors, security doors, watertight doors, and fire curtains, (2) fire dampers and ventilation dampers, and (3) penetration seals and fire walls. The data base contains 628 records from start of construction on any give plant to the end of June 1985. The number of barriers of each type at a plant is required to estimate the rate at which a specific component fails. The number is not known precisely for each plant, but a nominal figure that has been estimated for each barrier type is given in Table 3.5.

The statistical uncertainty of each estimate, reflecting sampling variation and plant-to-plant variation, is represented by 90 percent confidence bounds. These estimates and confidence bounds are given in Table 3.6 where units of both estimates and bounds are failures/year.

After allowing for credit for recovery of random failures during the screening process, all fire scenarios which required fire-related failures in adjacent fire zones were eliminated from further consideration.

3.8 Recovery Analysis

For those remaining cut sets which survived the screening process and where the COMPBRN code predicted fire damage would occur, recovery of random failures and credit for extinguishment of the fire before the COMPBRN predicted time to fire damage was applied.

Recovery of random failures (non-fire related) has been treated as in the internal events analysis. All operator recovery actions that were used in the internal events analysis were assessed for use where appropriate on the remaining cut sets.

3.9 Uncertainty Analysis

An uncertainty analysis was not performed since none of the scenarios analyzed resulted in core damage frequencies greater than 1.0E-8 per year.

3.10 Quantification of Unscreened Fire Induced Core Damage Scenarios and Their Associated Fire Zones

3.10.1 Introduction

This section describes the fire scenarios and their associated Fire Zones which survived the computer-aided screening process. All other fire zones were assessed to be below 10^{-8} per reactor year after recovery of non-fire related failures, and barrier failure probabilities were applied. Core damage frequencies for the following scenarios can be found in Table 3.7.

3.10.2 Fire Zone 1A117

Fire Zone 1A117 is a hallway area located on the 93'/103' elevation of the auxiliary building. Fire-related failures in this zone were found to lead to initiating event TLM5H (Loss of Makeup-CRD). Fire-related damage failed all or parts of the following systems:

a. CRD
b. RHR

Additional random failures were required to lead to core damage and are represented by the term Σ_{R1A117}.

In the case of Fire Zone 1A117 only a large fire was found to be capable of damaging the critical equipment. Therefore, there are associated terms for area and severity ratios for only a large fire.

The core damage equation for Fire Zone 1A117 is as follows:

$$\Phi_{cm} = \lambda_{aux}\ f_{A1A117}\ Q_1(\tau_G)\ \Sigma_{R1A117}\ f_{A1}\ f_{S1}$$

Where:

Φ_{cm} = fire-induced core damage frequency for Fire Zone 1A117,

λ_{aux} = auxiliary building fire frequency,

f_{A1A117} = area ratio of Fire Zone 1A117 to that of the auxiliary building,

$Q_1(\tau_G)$ = probability that the fire will not be manually suppressed before the critical components are damaged,

f_{A1} = area ratio within Fire Zone 1A117 where a large fire can damage the critical components, and

f_{S1} = severity ratio for a large fire.

Table 3.8 gives the values of all terms in the core damage equation (except for random failures) for Fire Zone 1A117.

3.10.3 Fire Zone 1A211

Fire Zone 1A211 is a hallway area located on the 119' elevation of the auxiliary building. Fire-related failures in

Table 3.5 Approximate Number of Barriers at a Plant

Type	Nominal
1	150
2	200
3	3000

Table 3.6 Estimates of Single Barrier Failure Rate

Barrier Type	Barrier/ Unit	Estimate	5 Percent Confidence Bound	95 Percent Confidence Bound
1	150	7.4E-3	0.0	2.4E-1
2	200	2.7E-3	0.0	2.2E-1
3	3000	1.2E-3	0.0	3.7E-2

Table 3.7 Accident Sequence Core Damage Frequency Contributors

Sequence	Fire Zone	Point Estimate Core Damage Frequency (/ry.)
TLM5H*	1A117 Auxiliary Building Hallway Area, 109' Elevation	$<1x10^{-8}$
TLM5H	1A211Auxiliary Building Hallway Area, 119' Elevation	$<1x10^{-8}$
TLM5H	1A316 Auxiliary Building Hallway Area, 139' Elevation	$<1x10^{-8}$
TRPT5**	1A201 Auxiliary Building Hallway Area, 119' Elevation	$<1x10^{-8}$
TRPT5	1A211 Auxiliary Building Hallway Area, 119' Elevation	$<1x10^{-8}$
TRPT5	1A316 Auxiliary Building Hallway Area, 139' Elevation	$<1x10^{-8}$

* Loss of Makeup (CRD)
** Loss of Recirculation Pump

Table 3.8

Fire Zone 1A117 Core Damage Equation Terms (Point Estimate Values)

λ_{aux}	1.8E-2
f_{A1A117}	4.0E-2
$Q_1(\tau_G)$	0.9
f_{A1}	3.2E-3
f_{S1}	0.3

this zone were found to lead to either initiating event TRPT5 (Loss of Recirculating Pump) or initiating event TLM5H (Loss of Makeup-CRD). Fire-related damage failed all or parts of the following systems:

a. CRD
b. RHR
c. RRS

Additional random failures were required to lead to core damage and are represented by the terms $\Sigma_{R1A211A}$ and $\Sigma_{R1A211B}$.

In the case of Fire Zone 1A211 both a small and a large fire were found to be capable of damaging the critical equipment. Therefore, there are associated terms for area and severity ratios for both a small and a large fire.

The core damage equation for Fire Zone 1A211 is as follows:

$$\Phi_{cm} = \lambda_{aux}\, f_{A1A211}\, Q_2(\tau_G)[\, f_{A2}\, f_{S1} + f_{B1}\, f_{S2}\,]\, [\Sigma_{P1A211A} + \Sigma_{R1A211B}]$$

Where:

Φ_{cm} = fire-induced core damage frequency for Fire Zone 1A211,

λ_{aux} = auxiliary building fire frequency,

f_{A1A211} = area ratio of Fire Zone 1A211 to that of the auxiliary building,

$Q_2(\tau_G)$ = probability that the fire will not be manually suppressed before the critical components are damaged,

f_{A2} = area ratio within Fire Zone 1A211 where a large fire can damage the critical components,

f_{S1} = severity ratio for a large fire

f_{B1} = area ratio within Fire Zone 1A211 where a small fire can damage the critical components, and

f_{S2} = severity ratio for a small fire.

Table 3.9 gives the values of all terms in the core damage equation (except for random failures) for Fire Zone 1A211.

3.10.4 Fire Zone 1A201

Fire Zone 1A201 is a hallway area located on the 119' elevation of the auxiliary building. Fire-related damage failed all or parts of the following systems:

a. RRS
b. RHR

Additional random failures were required to lead to core damage and are represented by the term Σ_{R1A201}.

In the case of Fire Zone 1A201, only a large fire was found to be capable of damaging the critical equipment. Therefore, there are associated terms for area and severity ratios for a large fire only.

The core damage equation for Fire Zone 1A201 is as follows:

$$\Phi_{cm} = \lambda_{aux}\, f_{A1A201}\, Q_3(\tau_G)\, \Sigma_{R1A201}\, f_{A3}\, f_{S1}$$

Where:

Φ_{cm} = fire-induced core damage frequency for Fire Zone 1A201,

λ_{aux} = auxiliary building fire frequency,

f_{A1A201} = area ratio of Fire Zone 1A201 to that of the auxiliary building,

$Q_3(\tau_G)$ = probability that the fire will not be manually suppressed before the critical components are damaged,

f_{A3} = area ratio within Fire Zone 1A201 where a large fire can damage the critical components, and

f_{S1} = severity ratio for a large fire.

Table 3.10 gives the values of all terms in the core damage equation (except for random failures) for Fire Zone 1A201.

3.10.5 Fire Zone 1A316

Fire Zone 1A316 is a hallway area located on the 139' elevation of the auxiliary building. Fire-related failures in this zone were found to lead to either initiating event

Table 3.9 Fire Zone 1A211 Core Damage Equation Terms (Point Estimate Values)

λ_{aux}	1.8E-2
f_{A1A211}	3.1E-2
$Q_2(\tau_G)$	0.95
f_{A2}	4.8E-3
f_{S1}	0.3

Table 3.10 Fire Zone 1A201 Core Damage Equation Terms (Point Estimate Values)

λ_{aux}	1.8E-2
f_{A1A201}	1.7E-2
$Q_3(\tau_G)$	0.9
f_{A3}	3.0E-2
f_{S1}	0.3

TRPT5 (Loss of Recirculation Pump) or initiating event TLM5H (Loss of Makeup-CRD). Fire-related damage failed all or parts of the following systems:

a. CRD
b. RHR
c. RRS

Additional random failures were required to lead to core damage and are represented by the term $\Sigma_{R1A316A}$ and $\Sigma_{R1A316B}$.

In the case of Fire Zone 1A316 only a large fire was capable of damaging the critical equipment. Therefore, there are associated terms for area and severity ratios for a large fire only.

The core damage equation for Fire Zone 1A316 is as follows:

$$\Phi_{cm} = \lambda_{aux} f_{A1A316} Q_4(\tau_G))[f_{A4} f_{S1} + f_{B2} f_{S2}] \left[\Sigma_{R1A316A} + \Sigma_{R1A316B}\right]$$

Where:

Φ_{cm} = fire-induced core damage frequency for Fire Zone 1A316

λ_{aux} = auxiliary building fire frequency,

f_{A1A316} = area ratio of Fire Zone 1A316 to that of the auxiliary building

$Q_4(\tau_G)$ = probability that the fire will not be manually suppressed before the critical components are damaged,

f_{A4} = area ratio within Fire Zone 1A316 where a large fire can damage the critical components,

f_{S1} = severity ratio for a large fire,

f_{B2} = area ratio within Fire Zone 1A211 where a small fire can damage the critical components, and

f_{S2} = severity ratio for a small fire.

Table 3.11 gives the values of all terms in the core damage equation (except for random failures) for Fire Zone 1A316.

3.11 Conclusions

The overall fire-induced core damage frequency during POS 5 for the Grand Gulf Nuclear Station was found to be less than 1.0E-8/ry. The fire-induced core damage frequency at Grand Gulf Nuclear Station was found to be lower than other at power fire risk assessments due to the following reasons:

a. The plant operational state analyzed represents only three percent of the time at shutdown and fire frequencies at shutdown are similar to those at power. This provides an immediate reduction in core damage frequency.

b. Even if active electro-mechanical safety-related equipment is damaged by fire, an initiating event may not occur. For instance, for the loss of Turbine Building Cooling Water (TBCW) initiator to result from fire related damage, multiple operational pumps must fail. These pumps and their associated cables have sufficient separation such that it is highly unlikely that a single fire could lead to failure of all pumps. As a result, many at shutdown initiating events were screened due to physical separation criteria. Even for unscreened initiators, very few fire zones were found to be applicable due to physical separation.

c. Relative to other plants, Grand Gulf utilizes more automatic fire protection systems in critical safety-related areas which in turn reduces the probability of core damage due to a fire.

Table 3.11 Fire Zone 1A316 Core Damage Equation Terms (Point Estimate Values)

λ_{aux}	1.8E-2
f_{1A1316}	4.0E-2
$Q_4(\tau_G)$	0.9
f_{A4}	1.0E-2
f_{S1}	0.3
f_{B2}	7.0E-3
f_{S2}	0.7

3.12 References

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APPENDIX A

MANUAL FIRE SUPPRESSION ANALYSIS

A.1 Backgound and Analysis

A.1.1 Fire Brigade Training, Organization and Standard Operating Procedures

The Fire Brigade at Grand Gulf is a well trained and organized brigade by all standards. It appears that the brigade has the complete and active backing of management at the plant. The management support is shown by the participation of plant and shift superintendent in the brigade as part of the command structure and as actual members of the brigade.

The training program for all members is well organized with regular meetings and drills. Any member not meeting the standards set by the plant for participation in the training is not included in the minimum manning levels set for the plant. This suspension is continued until the requisite training is complete. Record keeping for all aspects of the brigade is excellent. The records include fire drills, quarterly, annual and biannual training, daily manning levels, meeting attendance, classes, live fire experience and NRC required information.

The shift superintendent is the chief of the brigade on each shift. The plant supervisor is the brigade leader on each shift. The operators are given a list of all brigade members on each shift to ensure manning levels are adequate prior to previous shift members leaving. The normal manning level for the brigade is between 8 and 10 members. The minimum manning level is 5 qualified members and the maximum is about 15. During outages these levels may double. All brigade members are from the operations side of the plant. Total brigade membership is in the range of 100 people. Approximately ten percent of the members are also volunteer or call firefighters.

In a fire incident, the brigade members are officially notified of an alarm by the public address system. All members also carry radios which is the preferred method of communication for the members. Upon receiving the alarm all members report to one of the equipment cages strategically located throughout the plant to suit up. The members then respond to the incident area. The brigade leader follows this same procedure instead of responding directly to the incident site for initial size up and attack planning. This is viewed as a weakness in the standard operating procedure. Time used by other members to suit up could be used by the leader to locate the fire and plan the initial attack. Although the time savings in the incidents analyzed would be minimal saving in other locations may be substantial.

The times used for fire brigade response in the analysis are based on records of actual fire drills conducted at the plant. The drills were all unannounced, the only person given advanced warning of the drill being the shift superintendent. The times were modified slightly based on the location of the analyzed spaces within the plant and the location of the equipment cages relative to the spaces.

A.1.2 Fuel and Fire Development

The fuel used for the analysis is lubricating oils. Two scenarios were analyzed, they are a spill of approximately fifty kilograms over an area of 0.67 square meters and five kilograms over an area of 0.29 square meters. The fires develop from ignition to full involvement in sixty seconds and continue at that level throughout the scenarios unless other combustibles in the room become involved. In those instances the heat release rates are adjusted upward to reflect the other material contribution. In all scenarios, information on the fire development through nine hundred seconds is used in modeling. All scenarios are predicted to be terminated prior to this time.

A.1.3 Smoke Detector Response (DETACT)

An estimate of the time for the response of the smoke detectors in the space was determined using a model developed at the National Institute of Standards and Technology called DETACT (<u>DET</u>ector <u>ACT</u>ivation). It is a routine included in a collection of models/routines called FPETOOL.

The input for the model is the time heat release rate growth parameters, room height, detector spacing and properties and ambient temperature. The output is the heat release rate necessary for detector activation and the time of activation.

The model is designed for unconfined ceilings as found in the spaces analyzed. Prior to activation of the detector there should be no significant buildup of hot gas below the ceiling. Due to the rapid fire growth used in the analysis, such a heat buildup is expected in the early stages soon after detector operation. In a completely open space, the devices may have responded faster than indicated by this routine, however, the majority of the detectors were blocked by cable trays located below them. The response time index is normally a measurement of the delay of the device in responding to a flow of hot gas due to its mass, material, and configuration. To account for the time for the smoke to penetrate the cable trays, a higher response time index than normally used for smoke detectors was utilized.

In modeling the smoke detector response, a first order approximation of the response of the smoke detector was made by assuming that a temperature rise of about 15C for

a response device with a response time index of 10 approximates a condition at which the smoke detector would respond. The program does not account for the travel time from the generation of the smoke at the point of combustion to the detector. Since devices are relatively close to the fire source this a negligible factor.

A.1.4 Time From Detection to Alarm

The time to alarm is based on the requirements for alarm system in the National Fire Protection Association standards for alarm transmission. The maximum allowable time for a device activation to result in an alarm transmission and receipt at the control/annunciator location is ninety seconds. This time was used as the maximum estimated time in the analysis. The minimum time used was thirty seconds.

A.1.5 Fire and Time Related Space Conditions (ASETBX Room Fire Model)

Locating the fire, agent application and extinguishment are all affected by the conditions within the spaces during the period of time they are being accomplished. To estimate the conditions within the spaces, the fires used to do the COMPBRN analyses were effectively placed in the spaces using a computer model. The fires used were severe by most standards due to the rapid growth in the heat release rate (from 0 KW to 735 KW and 1650 KW in 60 seconds). The use of these fires account for the rapid response of the detection system and the rapid deterioration of conditions in the spaces in a relatively short time.

An estimate of the time related conditions within the space was determined using a model developed at the National Institute of Standards and Technology called ASET (Available Safe Egress Time). It is a routine included in a collection of models/routines called FPETOOL.

ASET is a simple mathematical model for estimating the rise in temperature and the descent of the fire produced hot gas layer in the room of fire origin. It considers conditions up to the point that the room approaches flashover but does not predict the conditions following the occurrence of flashover. ASET uses a simple point source entrainment and filling calculation that requires the user estimate the average portion of the energy produced by the fire that is lost from the flame or smoke layer. ASET has a fixed assumption that 35% of the energy in the flame is lost by radiation. The user is required to enter a value representing the total heat loss fraction. The choice made has a major impact on the calculated temperature of the hot gas layer and a lesser impact on the calculated position of the bottom of the layer.

The input for the model is the time and heat release rate growth parameters, room geometry, fire location, ambient temperature and the heat loss fraction (the amount of heat losses to the space enclosure as a function of its geometry). The output is the depth of the smoke layer and the temperature of the smoke layer, both as a function of time.

ASET is a zone model which means it divides the room or space being analyzed into two zones. The upper zone is the smoke layer and the lower zone is the ambient air layer. Being a zone model the output presents the layers as being two distinct homogeneous layers. In reality the temperature is more a function of the distance from the ceiling of the space with the highest temperatures closest to the ceiling. A similar condition applies to the smoke with it being thickest near the ceiling and the density of the smoke decreasing with distance from the ceiling.

A.1.6 Locating the Fire

The estimate of time to locate the fire is a function of the size of the space, the configuration, time of fire brigade arrival and the predicted conditions within the space at that time. The spaces analyzed are large, open, uncluttered areas with good visibility throughout. Any difficulty in locating the fire will be a function of the smoke level in the space. For most scenarios the smoke level was high enough that it had little impact on locating the fire. Therefore, the time associated with this phase is minimal.

A.1.7 Agent Application

The estimate of the time to agent application is based on the location of manual suppression equipment relative to the incident area. As the standpipe system outlets are located at regular intervals throughout the plant and fire extinguishers are readily available in all areas, the time to agent application is again minimal. The same time was used for all scenarios within each space as it is assumed a portion of the brigade will be laying lines or retrieving fire extinguishers while other members are locating the fire. Therefore, additional time for this phase is built into the previous phase.

A.1.8 Extinguishment

The fires modeled are relatively large with regard to heat release rate, but are relatively small in area. Based on this condition and the fact a small pool fire can be easily extinguished by well trained personnel, times used for this phase are again minimal. Although the time to extinguishment is reported, once agent application begins the effect of the fire is going to be greatly diminished as the cooling/smothering affect of the agent will quickly reduce the heat release rate associated with the fire.

A.2 Results

The following tables detail the results of the analysis for the fire brigade manual firefighting response to a fire incident in the Grand Gulf Nuclear Power Plant. The tables present three levels of response. The variability between the levels is diminished from similar analyses done for other plants due to the following three items; (1) the fire brigade is located at the plant instead of several miles away, (2) standpipe outlets for suppression activities are located conveniently throughout the plant and (3) all areas analyzed are covered by an automatic fire alarm system annunciated at a constantly attended location with area smoke detection provided at the ceiling.

A.3 Summary

The analysis of the manual suppression of fire incidents in several areas at the Grand Gulf Nuclear Power Plant was based on several factors that impact the different phases of any fire incident. These factors include the fuel involved, ignition, fire growth, detection, alarm, manual suppression response, manual suppression equipment location, and the ability of the fire brigade. Information on all these factors is provided below but a review of several factors that most impacted the analysis is provided herewith.

The Grand Gulf Nuclear Power Plant is a well protected plant when compared to the industry as a whole. All main plant areas surveyed were provided with smoke detection, standpipe outlets within the area or nearby and fire extinguishers and many areas had either full or partial automatic sprinkler protection. The plant has a well staffed and very well trained fire brigade. Training for the brigade is conducted on site in classes or in the training building capable of live fire and smoke situations. This ensures higher than average participation by all members in actual fire fighting activities.

Based on the high level of protection and the quality of the fire brigade, the times to manual suppression are much less than that which would be expected in an average plant. None of the maximum scenario times exceed fifteen minutes.

Table A-1 Grand Gulf Nuclear Power Plant Zone 1A117 - Small Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	66	120	93
2, Alarm	96	210	153
3. Fire Brigade response	246	570	408
4. Locate fire	256	590	423
5. Agent application	316	650	483
6. Extinguishment	346	710	528

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-2 Grand Gulf Nuclear Power Plant Zone 1A117 - Large Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	35	70	53
2. Alarm	65	160	113
3. Fire Brigade response	215	520	368
4. Locate fire	235	580	408
5. Agent application	295	640	468
6. Extinguishment	325	700	513

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-3 Grand Gulf Nuclear Power Plant Zone 1A201 - Small Pool Fire Passage Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	45	90	68
2, Alarm	75	180	128
3. Fire Brigade response	225	540	383
4. Locate fire	235	555	395
5. Agent application	295	615	455
6. Extinguishment	325	675	500

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-4 Grand Gulf Nuclear Power Plant Zone 1A201 - Large Pool Fire Passage Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	27	60	44
2, Alarm	57	150	104
3. Fire Brigade response	207	510	359
4. Locate fire	227	600	414
5. Agent application	287	690	489
6. Extinguishment	317	780	549

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-5 Grand Gulf Nuclear Power Plant Zone 1A211 - Small Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	45	90	68
2, Alarm	75	180	128
3. Fire Brigade response	225	540	383
4. Locate fire	235	555	395
5. Agent application	295	615	455
6. Extinguishment	325	675	500

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-6 Grand Gulf Nuclear Power Plant Zone 1A211 - Large Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	27	60	44
2, Alarm	57	150	104
3. Fire Brigade response	207	510	359
4. Locate fire	217	570	394
5. Agent application	277	630	454
6. Extinguishment	307	690	499

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-7 Grand Gulf Nuclear Power Plant Zone 1A316 - Small Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	47	96	72
2. Alarm	77	186	132
3. Fire Brigade response	257	606	432
4. Locate fire	267	666	467
5. Agent application	327	726	527
6. Extinguishment	367	786	572

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

Table A-8 Grand Gulf Nuclear Power Plant Zone 1A316 - Large Pool Fire Miscellaneous Equipment Area

AUTOMATIC DETECTION:	Smoke Detection
AUTOMATIC SUPPRESSION:	Partial Automatic Sprinkler Protection
STANDPIPE OUTLET LOCATION:	Within Space
EXTINGUISHER LOCATION:	Within Space

Event/Phase Description	Cumulative Time (Seconds)		
	Minimum	Maximum	Average
1. Detection	30	60	45
2, Alarm	60	150	105
3. Fire Brigade response	240	570	405
4. Locate fire	250	630	440
5. Agent application	310	690	500
6. Extinguishment	370	750	560

NOTE: The cumulative time starts at the time of established burning, any time associated with a smoldering ignition has been ignored.

APPENDIX B

GRAND GULF CRITICAL COMPONENTS BY FIRE ZONE

Appendix B

This appendix lists the major mechanical equipment or cables associated with that equipment for each of the dominant fire zones.

In most of the rooms, the major effect of the fire is to damage or destroy the vital cables passing through the room.

Fire Zone	Critical Components or Cables for Critical Components
1A101	LPCS MOV 5A, RHR MOV 96, RHR MOV 94
1A102	RHR A Heat Exchanger, LPCI MOV 48A
1A103	RHR A Pump, RHR MOV 24A, LPCI MOV 48A
1A105	RHR B Pump, RHR Pump 2B, LPCI MOV 48B
1A106	RHR B Heat Exchanger, LPCI MOV 48B
1A109	HPCS Pump, HPCS Breaker 2
1A110C1	RHR MOV 42B, HPCS Breaker 2
1A110C3	Recirc MOV 67B, SPMU MOV 1B & 2B, Recirc MOV 23B
1A110D3	SPMU MOVs 1B & 2B, CS MOV 28B
1A112	RHR MOV 9B
1A114	LPCS MOV 5A
1A117	ADHRS Pump A, ADHRS Pump B, SSW Pump C, CCW Pump 1B, RHR MOV 96, RHR MOV 94, LPCI MOV 242, CRD Pump 1A, RHR Pump 2C, CRD Pump 1B, RHR Pump 2C
1A119	LPCS MOV 5A
1A120	CCW Pump 1B, CCW Pump 1A, CCW Pump 1C
1A128	LPCI MOV 48A
1A129	LPCI MOV 48B
1A201	Recirc Pump B, Recirc MOV 67A, Recirc MOV 23A, FWS MDP, CCW Pump 1A, LPCI MOV 48A RWCU Pump 1B, RWCU Pump 1A, IA Comp (1B), CRD Pump 1A, RHR MOV 8A, RHR MOV 2A, RHR MOV 42A, SDC MOV 53A, CS MOV 28A, SDC MOV 6A, RCIC Logic A&B, LPCS Pump 1A, RHR MOV 24A, RHR MOV 8A, RHR Pump A, CS MOV 28A, Unit 1 IA Compressor 1B
1A202	RHR MOV 24A
1A204	RHR MOV 8A, RWCU Pump 1A
1A205	CS MOV 28B, LPCI MOV 48B, HPCS MOV 4, RHR MOV 24B, SDC MOV 6B, SDC MOV 53B, RHR MOV 24B, SDC MOV 6B, SDC MOV 53B, LPCI MOV 48B, HPCS MOV 4
1A207	Electrical Switchgear Room Division II, RHR MOV 24B, RHR MOV 42B, SDC MOV 6B, RHR Pump 2C, SDC MOV 53B, RHR MOV 9B, RHR MOV 96, RHR MOV 94, CS MOV 28B, LPCI MOV 48B
1A208	Electrical Switchgear Room Division I, RHR MOV 8A, HPCS Breaker 2, RHR MOV 24A, LPCI MOV 48A
1A210	RWCU Recirculation Pump Room, RHR MOV 8A

Fire Zone	Critical Components or Cables for Critical Components
1A211	Recirc Pump B, CCW Pump 1B, ADHRS Pump A, ADHRS B, SSW Pump C , RHR MOV 96, RHR MOV 94, CS MOV 28B, LPCI MOV 242, LPCI MOV 48B, LPCI MOV 48A, CCW Pump 1C,HPCS MOV 4, RWCU Pump 1B, IA Comp (1B), RHR MOV 8A, RHR MOV 2A, RHR MOV 24B, RHR MOV 42B, RHR MOV 42A, SDC MOV 53A, SDC MOV 6B, RHR Pump 2C, RHR Pump 2B, SDC MOV 53B, HPCS Breaker 2, SDC MOV 6A, RCIC Logic A&B, RHR MOV 9B, LPCS MOV 5A, RHR MOV 24A, CRD Pump 1B, SSW Pump 1A, RHR MOV 8A, RHR Pump A, RHR MOV 24B, RHR MOV 42B, SDC MOV 6B, RHR Pump 2B, SDC MOV 53B, RHR MOV 9B, LPCS MOV 5A, RHR MOV 24A, CRD Pump 1B, Unit 1 IA Compressor 1B
1A215	CS MOV 28A, LPCS Pump 1A, FWS MDP, CCW Pump 1A, LPCI 48A, RWCU Pump 1A, Unit 1 IA Compressor 1B
1A219	LPCS MOV 5A
1A220	LPCS Pump 1A, LPCS MOV 5A, LPCI MOV 242
1A221	RHR Pump 2C, CCW Pump 1B, LPCI MOV 242
1A222	Recirc MOV 67A, Recirc MOV 23A, ADHRS Pump A, ADHRS Pump B, CCW Pump 1B, CCW Pump 1A, LPCI MOV 242, CCW Pump 1C, RWCU Pump 1B, RWCU Pump 1A, RHR Pump 2C, LPCS MOV 5A, RHR MOV 8A, LPCS Pump 1A, LPCS MOV 5A
1A301	FWS MDP, CCW Pump 1A, RWCU Pump 1A
1A308	RHR MOV 42B, RHR MOV 9B, Recirc MOV 23B, CS MOV 28B
1A309	CS MOV 28A
1A311	Recirculation MOV 23B, CS MOV 28B
1A316	Recirc Pump A, Recirc MOV 67B, ADHRS Pump A, ADHRS Pump B, SPMU MOV 1B, SPMU MOV 2B, Recirc MOV 23B, SSW Pump C, CCW Pump 1B, CS MOV 28B, LPCI MOV 242, LPCI MOV 48A, CCW Pump 1C, RWCU Pump 1B, RHR MOV 8A, RHR MOV 42A, SDC MOV 53A, RHR Pump 2C, SDC MOV 6A, RCIC Logic A&B, LPCS Pump 1A, LPCS MOV 5A, RHR MOV 24A, SSW Pump 1A, LPCS MOV 5A
1A401	FWS MDP, CCW Pump 1A, RWCU Pump 1A
1A417	RHR MOV 8A, RCIC Logic A&B, LPCS Pump 1A, LPCS MOV 5A, SSW Pump 1A, LPCI 48A
1A539	Recirc Pump A, Recirc MOV 67A, Recirc MOV 23A, SPMU MOV 1A, SPMU MOV 2A, CCW Pump 1A, LPCI MOV 48A, SSW Pump A, CRD Pump 1A, RHR MOV 8A, RHR MOV 2A, RHR MOV 42A, SDC MOV 53A, CS MOV 28A, SDC MOV 6A, RCIC Logic A&B, LPCS Pump 1A, LPCS MOV 5A, RHR MOV 24A, SSW Pump 1A, RHR MOV 8A, RHR Pump A, CS MOV 28A, LPCS MOV 5A, RHR MOV 24A
1M110	SSW Pump C, SSW Pump A
2M110	SSW Pump B, SSW MOV 5B, SSW MOV 1B
2M112	SSW MOV 5B
OC116	HPCS Breaker 2
OC202	SPMU MOV 1B, SPMU MOV 2B, LPCI MOV 48A, SSW Pump A, CRD Pump 1A, RHR MOV 8A, RHR MOV 2A, RHR MOV 42A, SDC MOV 53A, SDC MOV 6A, LPCS Pump 1A, RHR MOV 24A, SSW Pump 1A, RHR MOV 8A, RHR Pump 2A, RHR MOV 42A, SDC MOV 53A, SDC MOV 6A, LPCS Pump 1A, SSW Pump A

Fire Zone	Critical Components or Cables for Critical Components
OC207	SDC MOV 53A
OC208A	CRD Pump 1A, RHR MOV 8A, SDC MOV 53A, SDC MOV 6A, SSW Pump 1A, LPCI MOV 48A, SSW Pump A
OC208	RHR Pump 2A, RHR Pump 2B, SSW MOV 5B
OC209	CRD Pump 1B, SSW Pump 1B
OC210	Division III Switchgear, CRD Pump 1A, RHR MOV 24B, RHR MOV 42B, SDC MOV 6B, RHR Pump 2B, SDC MOV 53B, HPCS Breaker 2, RHR MOV 9B, CRD Pump 1B, SSW Pump 1B, SSW MOV 5B, SSW Pump C (HPCS), RHR MOV 96, RHR MOV 94, HPCS MOV 4
OC211	Div. II Battery Room
OC215	SSW MOV 5B, SSW MOV 1B, CCW Pump 1B, RHR MOV 96, RHR MOV 94, LPCI MOV 48B, 1A COMP (1B), RHR MOV 24B, RHR MOV 42B, RHR Pump 2C, RHR Pump 2B, SDC MOV 53B, RHR MOV 9B, CRD Pump 1B, SSW Pump 1B, RHR MOV 24B, RHR MOV 42B, SDC MOV 53B, RHR MOV 9B, CRD Pump 1B, Unit 1 IA Compressor 1B
OC402	RHR Pump 2C, RHR Pump 2B, SDC MOV 53B, CRD Pump 1B, SSW MOV 5B, Recirc MOV 23B, TBCW Pump 1B, CCW Pump 1A, TBCW 1C Pump, CS MOV 28B, LPCI MOV 48B, COND Pump 3B, COND Pump 3C, CCW Pump 1C, SAS 1A Unit 1 Service Air Compressor
OC407	CRD Pump 1B
OC409	ADHRS Pump A, CRD Pump 1B
OC504	Recirc Pump A, Recirc MOV 67A, Recirc MOV 23A, FWS MDP, CCW Pump 1A, COND Pump 3A, FW MOV 1, RWCU Pump 1A, SAS Comp 1A, RWCU Pump 1B
OC702	CRD Pump 1A, RHR MOV 8A, RHR Pump 2A, RHR MOV 42A, SDC MOV 53A, CS MOV 28A, SDC MOV 6A, RCIC Logic A&B, LPCS Pump 1A, LPCS MOV 5A, TBCW Pump 1A, CCW Pump 1A, LPCI MOV 48A, COND Pump 3A, FW MOV 1, SSW Pump A, SAS 1A Unit 1 Service Air Compressor
OC703	CRD Pump 1A, RHR Pump 2A, RHR MOV 42A, SDC MOV 53A, CS MOV 28A, SDC MOV 6A, RCIC Logic A&B, LPCS Pump 1A, LPCI 48A
OM101	Diesel Driven Fire Pump A
OM102	Diesel Driven Fire Pump B, FWS MDP
OM119	SAS 1A Unit 1 Service Air Compressor, Unit 2 IA Compressor P52C1, Unit 1 IA Compressor 1B
IT118	TBCW Pump 1A, TBCW Pump 1B
IT132	Recirc Pump B, Recirc Pump A, MOV AV504, COND Pump 3A, COND Pump 3B, COND Pump 3C
IT214	TBCW Pump 1B, TBCW Pump 1C, SAS COMP 1A
IT218	Air Comp 3BN, TBCW Pump 1B, TBCW Pump 1C, COND Pump 3A, COND Pump 3B, COND Pump 3C, 1A Comp Unit 2, SAS 1A Unit 1 Service Air Compressor, Unit 2 IA Compressor P52C1
IT219	TBCW Pump 1B, TBCW IC Pump, COND Pump 3B, COND Pump 3C
IT220	COND Pump 3A, Unit 2 IA Compressor P52C1
IT224	COND Pump 3B, COND Pump 3C

Fire Zone	Critical Components or Cables for Critical Components
IT226	MOV AV504, TBCW Pump 1A, TBCW Pump 1B, TBCW Pump 1C, COND Pump 3A, COND Pump 3B, COND Pump 3C
IT306	MOV AV504
IT308	FW MOV 1
IT315	FW MOV 1
IT317	FW MOV 1
IT322	Recirc Pump B, Recirc Pump A, MOV AV504, TBCW Pump 1A, TBCW Pump 1B, TBCW Pump 1C, Cond Pump 3A, SAS COMP 1A, SAS 1A Unit 1 Service Air Compressor
IT323	TBCW Pump 1A, COND Pump 3A
IT324	TBCW Pump 1A, TBCW Pump 1B, TBCW Pump 1C, COND Pump 3A, SAS COMP 1A
IT325	Recirc Pump B, Recirc Pump A, MOV AV504, TBCW Pump 1A, COND Pump 3A, SAS COMP 1A
IT327	Recirc Pump B, Recirc Pump A, MOV AV504, TBCW Pump 1A, TBCW 1B, TBCW Pump 1C, COND Pump 3A, FW MOV 1, COND Pump 3B, COND Pump 3C, SAS COMP 1A
IT403	TBCW Pump 1B, TBCW Pump 1C, SAS 1A Unit 1 Service Air Compressor
IT404	TBCW Pump 1A, COND Pump 3A, FW MOV 1, SAS COMP 1A
IT502	Recirc Pump A, FW MOV 1

Appendix C

AUXILIARY BUILDING SHUTDOWN FIRE EVENT DATA

Presented below is a listing and short description of fires which have occurred at shutdown in nuclear power plants auxiliary buildings thru December 1989.

This information is used in Section 3.3 of the main report to calculate the fire initiating event frequency for the auxiliary building.

Fire Event Data-Auxiliary Building Fires				
Plant Name	**Date of Occurrence**	**Plant Status**	**Fire Type**	**Remarks**
Trojan	3/4/76	Hot Shutdown	Insulation	A short caused by breakers not properly engaging caused the ignition of insulation.
Millstone 2	3/24/76	Hot Shutdown	Motor Control Center	Fire resulted from arcing of a supply lead. Extinguished by de-energizing.
Millstone 2	11/15/76	Hot Shutdown	Relay--MCC	Relay fire in motor control center.
Three Mile Island 2	11/22/76	Cold Shutdown	MCC	A misaligned stab connected to a MCC branch breaker associated with the nuclear service pump starter.
Peach Bottom	7/29/77	Cold Shutdown	Relay	Improper installation of relay contact arm retainers.
Arkansas Nuclear One 1	8/16/78	Hot Shutdown	Pump Motor	LPSI pump motor on fire (being used for shutdown cooling) due to incorrect installation of motor bearings resulting in shorting of rotor with the stator.
Hatch [1]	11/23/81	Cold Shutdown	Relay	Insulation breakdown caused fire in a reactor low-low RPS relay.

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11. ABSTRACT (200 words or less)

This report presents the details of the analysis of core damage frequency due to fire during shutdown Plant Operational State 5 at the Grand Gulf Nuclear Station. Insights from previous fire analyses (Peach Bottom, Surry, LaSalle) were used to the greatest extent possible in this analysis. The fire analysis was fully integrated utilizing the same event trees and fault trees that were used in the internal events analysis.

In assessing shutdown risk due to fire at Grand Gulf, a detailed screening was performed which included the following elements:

a) Computer-aided vital area analysis
b) Plant inspections
c) Credit for automatic fire protection systems
d) Recovery of random failures
e) Detailed fire propagation modeling

This screening process revealed that all plant areas had a negligible (<1.0E-8 per year) contribution to fire-induced core damage frequency.

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