2

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PRA Applications Program for Inspection at Calvert Cliffs Unit 1 Nuclear Power Plant

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ABSTRACT

The level one probabilistic risk assessment (PRA) for Calvert Cliffs Unit 1 (CC-1) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency, and to identify the primary failure modes of these components. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement Manual. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance, are: Reactor Protection, Auxiliary Feedwater, DC power, AC power, Power Conversion, High Pressure Injection, Room Cooling, Salt Water, Safety Relief Valves, and Chemical Volume and Control. This ranking is based on the Fussel-Vesely measure of risk importance, i.e., the fraction of the total core melt frequency which involves failures of the system of interest.

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SUMMARY

The PRA Applications Program for inspection at Calvert Cliffs Unit 1 (CC-1) was performed for the U.S. Nuclear Regulatory Commission (NRC) at Pacific Northwest Laboratory (PNL). This program applies a previously developed methodology to identify and present information which is useful for the planning and performance of powerplant inspections.

The level one probabilistic risk assessment (PRA) for CC-1 (Payne 1984) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement (IE) Manual (USNRC 1984) which are used by inspectors in the planning and performance of inspections. The body of this report consists of a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the core melt probability resulting from plant operation.

Following a section describing important accident initiators and sequences identified in the PRA, tabulations are presented for ten systems. These system tables are ordered by system risk importance, as measured by the fraction of the total core melt probability associated with failures of each system. Three tables are presented for each system. The first table presents the failure modes identified in the PRA for each important system component. The second table correlates each component with the IE inspection modules most related to ensuring component reliability. The third table provides a modified system check off list identifying the proper line-up of each component during normal operation.

The tabulations were developed by the following analysis procedure. First, the plant systems were ordered according to system risk importance. To accomplish this, the dominant cut sets representing more than 98% of the core melt probability were listed, and the fraction of the total core melt probability which involved failures of components from each system was calculated [this is the Fussel-Vesely Importance measure (Henley 1981)]. Systems were then selected from the ordered list until more than 98% of the

core melt probability was accounted for. Second, for each selected system, the fault tree from the PRA was reanalyzed to rank system components according to their importance to system failure. For each system, components were selected for inclusion in the tabulations until more than 95% of the system failure probability had been addressed.

The tables thus present, in decreasing order of system importance, the failure modes, applicable inspection modules, and a check off list of normal operational state for all components associated with 98% of the core melt probability associated with plant operation. This information allows an inspector to readily identify important systems and components when developing an inspection plan and when walking down systems in the plant.

The information presented in this document allows an inspector to concentrate his efforts on systems important to the prevention of core melt. However, it is essential that inspections not focus exclusively on these systems. Other systems which perform essential safety functions, but are absent from the tables because of high reliability and redundancy, must also be addressed to ensure that their importance is not increased by allowing their reliability to decrease. A balanced inspection program is essential. This information represents but one of the many tools to be used by experienced inspectors.

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CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

This work was performed for the U.S. Nuclear Regulatory Commission (NRC) as part of an extensive program to develop information based on probabilistic risk analyses (PRAs) for use in the planning and performance of nuclear powerplant inspections. Due to the broad scope of this program, project work has been divided among three national laboratories, each of which concentrates upon a particular reactor type. Thus, Brookhaven National Laboratory (BNL) analyzes plants powered by boiling water reactors, and at Idaho National Engineering Laboratory (INEL) analyzes pressurized water reactor plants (PWRs) built by Westinghouse. Pacific Northwest Laboratory (PNL) analyzes PWRs from both Babcock & Wilcox and Combustion Engineering.

In this particular project, information from the Calvert Cliffs Unit 1 (CC-1) PRA (Payne 1984) has been used to identify plant systems and components important to minimizing the probability of core melt, and to identify failure modes for these components. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement (IE) Manual (USNRC 1984) which are used by inspectors in the planning and performance of inspections. The body of this report consists of a series of tables, organized by system and prioritized by system importance, which identify components associated with 98% of the plant core melt probability.

Previous studies in this program (Hinton and Wright 1986, Higgins 1986) have addressed how PRA-based information may be best incorporated into inspection planning, performance and evaluation. The conclusion of this previous work was that the existing IE Manual provides a logical and effective framework for inspection planning. This manual contains an extensive sequence of inspection procedures, or modules, addressing functional areas such as calibration, surveillance, maintenance, engineered safety features (ESF) system walkdown, etc. It also contains a methodology for selecting inspection modules for performance, plus guidance on the frequency at which modules should be performed. It was concluded that this manual should be retained as the general framework for inspection planning. PRA-based information, which is necessarily plant-specific, should be provided for each plant. This information should then be used in the inspection planning process to help focus on areas where public risk is most sensitive to performance degradation.

The NRC program is, therefore, directed towards the preparation of a series of plant-specific appendices to the IE Manual which contain plant-specific information of a common type and safety significance. These appendices are structured according to a common format. Each appendix begins with a description of accident initiators and sequences important at the plant. This is followed by a listing of plant systems associated with 98% of the plant core melt probability, which is ordered according to the importance of each system to plant damage. For each system addressed, the components associated with 95% of the probability of system failure are identified and ranked according to their importance. Three tables are presented for each system. The first identifies the failure modes by which each component contributes to plant damage. The second correlates each component with the IE inspection modules most related to ensuring component reliability. The third provides a modified system check-off list identifying the proper line-up of each component during normal operation. For each system, a diagram is reproduced from the PRA which shows how the system depends on other supporting systems. The body of this report presents the plant-specific appendix developed for the CC-1 nuclear power plant. It follows the format described above.

In addition, a final section has been added which discusses containment protection systems. These systems are not involved in the prevention of core melt, but are of fundamental importance to preventing or minimizing public risk due to the release of radionuclides, if a core melt should occur. This section discusses the Containment Spray System, and the Containment Air Recirculation and Cooling System and identifies failure modes for components in these systems which were found to be important in the analysis of the Level 3 PRA for Oconee Unit 3.

PRAs have been performed for less than one quarter of the nation's nuclear plants. Consequently, a significant aspect of the NRC program addresses the development of generic insights which may be utilized to guide inspection planning for plants without a PRA. As plant-specific appendices are developed, the information is reviewed to identify dominant generic contributors to risk, including: initiating events, accident sequences, important systems and components, component failure modes, significant human errors, and common cause failures.

The compilation of generic insights resulting from the analysis of PRAs indicates systems and components which may have risk importance at other plants. For application to a specific site, plant-specific information must be used to evaluate the relevance and applicability of the generic insights. For instance, important functions may be performed by different systems at different plants, or, systems may be either more vulnerable (single failure dependencies) or less vulnerable (redundancies) at different plants. PNL has performed an analysis of the Rancho Seco plant (no PRA) using the results of PRAs for the Arkansas Nuclear One Unit 1 and Oconee-3 plants, plus a detailed comparison of system designs at the three plants (Gore and Huenefeld 1987). INEL and BNL are performing similar studies using generic insights and plant-specific information to address plants for which PRAs have been performed (Higgins et al. 1987). Future comparison of results from those studies with results obtained from analyzing the plant-specific PRAs will provide an indication of how effective this approach is in identifying important systems and components.

As was noted above, this document reports the results of a detailed analysis of the PRA performed for the CC-1 plant. It was not necessary to utilize generic insights in the performance of this analysis. Rather, the results of this study will contribute to the database of generic information to be utilized in the analyses of plants which lack PRAs. The analysis approach used in this study is discussed in the following Section 2.0. The results of the analysis are presented in Sections 3.0 and 4.0, according to the above-described format for plant-specific appendices to the IE Manual. For completeness, information on containment protection systems is presented in Section 5.0.

2.0 ANALYSIS OF THE CC-1 PRA

The analysis required three major steps to produce the tables presented in Section 4. The first step was the calculation of risk importance for each system from information in the PRA. This was used to select systems to be analyzed for component importances. The second step was the re-analysis of system fault trees from the PRA to identify component importances. The third step was the correlation of components and their dominant failure modes with inspection modules relevant to maintaining component reliability. These steps are discussed below.

2.1 CALCULATION OF SYSTEM IMPORTANCES

The selection of systems for detailed fault tree analysis required that they be ranked according to an appropriate measure of risk. The CC-1 PRA is a level 1 PRA. Core melt probability is addressed in detail, with only a limited analysis of subsequent containment failure mechanisms, and radionuclide releases to the public. Consequently, for this study core melt frequency is used as the risk measure used to rank system importance.

The Fussel-Vesely (F-V) Importance measure (Henley and Kumamoto 1981) applied to core melt frequency was selected as the specific risk measure used to rank systems and components. The F-V Importance is the fraction of the total risk (core melt frequency) which results from failures involving the system or component of interest. Thus, high values of F-V Importance identify systems which are the greatest contributors to risk. In addition, the increase in risk due to a given percentage increase in system failure probability is also highest for systems with highest F-V Importance values. Thus, this measure identifies not only the systems which are the greatest contributors to risk, but also those for which risk is most sensitive to performance degradation. It is therefore the logical measure to use for ranking system importance for inspection attention, to ensure that safety performance is maintained.

Appendix C of the CC-1 PRA presents a detailed listing of initiating events and cut set elements, and associated unavailabilities (both with and without recovery factors). The dominant cut sets presented in the body of the PRA were selected from this list. This listing of more than 400 cut sets was analyzed to determine system importances. Each element of each cut set was analyzed to determine what system was responsible for the root cause failure represented by the cut set element. Core melt frequencies associated with each cut set were input to a spread sheet data file (recovery factors due to operator action from the PRA were included). This file was then manipulated into system-based sub-files, each of which included data only from cut sets involving failures of components in a given system. The F-V Importance of each system was then calculated by summing the sub-files and dividing by the total from all of the cut sets.

Table 2.1 presents the relative system importances. The Reactor Protection System (RPS) has the highest F-V Importance, primarily due to the high probability of a cut set representing common mode failure of both reactor trip breakers to open given a presence of a trip signal, and no credit has been given for recovery actions. Auxiliary Feedwater System (AFW) follows, with each of these systems having an importance of approximately 20%.

TABLE 2.1. Calculated System Importance

System	Relative Importance (%)
RPS	24
AFW	20
Emergency DC Power	10
Emergency AC Power	10
PCS	10
HPSI	6
Room Cooling	6
Salt Water System	5
SRVs	4
Chemical and Volume Control	3
Total Relative Importance	98

The Emergency DC Power, and Emergency AC Power follow, with importance values of about 10%. They are followed by the Power Conversion System (PCS), the High-Pressure Safety Injection System (HPSI), the Room Cooling System, the Salt Water System (SWS), the Safety Relief Valves (SRVs), and the Chemical and Volume Control System (CVCS), all of which have importance values of between 3% and 10%.

It is important to note that the Low-Pressure Safety Injection (LPSI) system does not appear in the list of risk-important systems at Calvert Cliffs (Table 2.1). Specific reasons are outlined below:

1. The principal function of the LPSI system is to perform large-LOCA mitigating activities (i.e., inject borated water into the RCS to cool the reactor core following a large LOCA, decay heat removal from the core for extended periods of time following a LOCA). Of the sixteen dominant accident sequences identified in the PRA, none involve a large-LOCA initiating event. Therefore, since large-LOCA accident sequences do not dominate the accident sequence risk at Calvert Cliffs, then the system that is designed to mitigate these accident sequences should also not be a highly risk-important system.
2. At Calvert Cliffs, the HPSI/R can draw directly from the sump and is the preferred system for recirculation. The LPSI/R system is not normally used in the recirculation mode.

At other plants (e.g., B&W plants), however, the risk of LOCAs (specifically large LOCAs) may be more dominant. Therefore, the LPSI system at other B&W plants may have greater risk importance than that of Calvert Cliff's LPSI system.

2.2 CALCULATION OF COMPONENT IMPORTANCES

Construction of the tables presented in Section 4 of this report required the identification of components associated with at least 95% of the system failure probability for each of the systems selected for anlaysis

(Table 2.1). This required a reanalysis of the fault trees presented in Appendix B of the PRA document to identify the components most important to system failure.

For most systems selected for analysis, the system fault trees published in the PRA were reanalyzed using the Integrated Reliability and Risk Analysis (IRRAS) computer code (Russell et al. 1987) run on an IBM-PC. Other analysis methods were used for three systems: Reactor Protection; Power Conversion, and Safety Relief Valves.

Fault tree gates and component unavailability data were input to the code and processed using the integrated fault tree analysis package. IRRAS identified the dominant minimal cut sets and quantified the fault trees by ordering cut sets by probability. IRRAS also calculated the F-V Importance of both cut sets and of system component failures. The calculated importance of the component failures was then used to select components for inclusion in the tables. For all systems analyzed, components comprising more than 95% of the total component importance were selected for tabulation.

2.3 PREPARATION OF TABLES

For each system, the components selected for inclusion in the tables were grouped according to type for discussion of failure modes [e.g., pump suction and discharge motor-operated valves (MOVs) in parallel trains]. For many components, cut set elements indicated more than one failure mode (e.g., failure to operate, operator failure to initiate, inappropriate change of position). These failure modes were grouped and discussed for each component type in the system failure mode identification tables.

The characteristics of each component were assessed to determine what types of inspection would be most appropriate for ensuring component reliability. This information was then used to prepare a table for each system correlating each of the relevant IE inspection modules with components which should be addressed when the module is applied to the system. This table also contained a cross correlation to the failure modes which would be minimized by the given type of inspection. For instance, pump failure to start and run is addressed in modules for Surveillance, Operational Safety Verification, and ESF System Walkdown. It is also addressed through the Maintenance module, in terms of minimizing unavailability due to maintenance scheduling and work.

For each system, an abbreviated system walkdown table was prepared addressing only the selected system components. This table identifies the normal operating state or position of each component determined to be risk significant from the PRA. It was compiled using information from the PRA, plant system descriptions, operator training information, and plant drawings. In many cases it was possible to correlate and verify this information using system lineup tables from plant operating procedures. In general, these tables are considerably shorter than lineup tables in procedures. They therefore allow an inspector with limited available system walkdown time to concentrate on risk-significant components, without concern that he may be overlooking something important.

2.4 CONCLUSIONS AND RECOMMENDATIONS

In this project, we have identified the systems and components most important to risk during operation of the CC-1 power plant. They are identified in Tables 4.1 through 4.10. Systems are addressed in the order of decreasing importance, as determined by the fraction of the total core melt frequency which involves the failure of each system. This information has been developed from the PRA analysis of the CC-1 plant.

The RPS, AFW, Emergency DC, Emergency AC, and PCS Systems are the most important systems for minimization of core damage. Systems of intermediate importance include the HPI, Room Cooling, and Salt Water Systems. Lower importance systems include the Safety Relief Valves, and CVCS.

The information in these tables allows an inspector to identify quickly the components most important to risk--a combination of failure probability and the consequences of the failure. For this analysis, the sole consequence considered was the occurrence or non-occurrence of a core melt. Thus the risk turns out to be equivalent to the cut set occurrence probabilities. This information allows him to direct his attention to these components preferentially. In particular, by using the system walkdown tables, the inspector can rapidly review the line up of important system components on a routine basis. These tables may also be used when selecting systems for the performance or more detailed inspection activities.

In using these tables, however, it is essential to remember that other systems are also important. If, through inattention, the failure probabilities of other systems were allowed to increase significantly, their risk significance might exceed that of systems in the tables. Consequently, a balanced inspection program is essential to minimizing plant risk. The tables allow an inspector to concentrate on systems of highest risk importance. In so doing, however, he must maintain cognizance of the status of systems performing other essential safety functions, and ensure that their reliability is maintained.

3.0 IMPORTANT ACCIDENT INITIATORS AND SEQUENCES

Two general types of accident initiators are addressed in the PRA: Loss of Coolant Accidents (LOCAs), and transients. However, subsequent event sequences leading to core melt are not distinct, because each of the transient types addressed has the potential for inducing LOCA events. The total core melt frequency for CC-1 was determined to be 1.30E-04/yr and consisted almost entirely of sequences with frequencies greater than 1.00E-06/yr. Table 3.1 identifies the initiating event types and presents the annual core melt frequency estimated in the PRA document for events of each type.

TABLE 3.1. Core Melt Frequencies Associated with Important Initiating Events

Initiating Event	Annual Core-Melt Frequency
Anticipated Transients Without Scram (ATWS)	2.80E-05
Loss-of-Coolant Accidents (LOCAs) Small-Small (1.9" dia)	2.66E-05
Plant Transients	
Loss of 125 VDC Bus 11	2.10E-05
Loss of Offsite Power	1.12E-05
Transients Requiring Primary Relief	7.70E-06
Loss of Secondary Heat Removal	7.10E-06
Station Blackout	4.40E-06
All Other Transients	2.40E-05

The following discussion presents the various types of event sequences identified in the PRA as most likely to lead to core melt following occurrence of these initiating events.

3.1 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

- Sequence 1 - This sequence is an ATWS followed by reduced secondary heat removal capacity (i.e., PCS and/or auxiliary feedwater in a runback mode). The resulting imbalance between the energy production and removal rates leads to the heat-up of the primary system and an increase in system pressure. Primary system pressure boundary failure is expected to occur and result in core melt if the pressure exceeds the Service Level C limit (3200 psia). Such pressures can result in system damage severe enough to make continued reactor core cooling highly questionable. Because of the short time before pressure exceeds Service Level C, no credit has been given for any recovery actions.

3.2 SMALL-SMALL LOCA (1.9" in dia.)

- Sequence 2 - A small-small LOCA occurs followed by successful scram and operation of AFW and HPI, providing both secondary heat removal and primary system makeup. When the Refueling Water Tank (RWT) is depleted and switchover to recirculation occurs (anywhere from 4 to 12 hours into the transient depending on the size of the leak), High Pressure Safety Recirculation (HPSR) fails. Due to the lack of primary makeup, the core then uncovers and core melt ensues. The dominant contributors to this sequence are of two types: 1) failures of HPSR pumps combined with failure of room cooling; or 2) failures of the Component Cooling Water (CCW) system or Salt Water System (SWS) resulting in loss of HPSR pump seal cooling and failure of all HPSR pumps.

- Sequence 3 - A small-small LOCA occurs and is followed by successful scram and secondary heat removal via the AFW system. However, HPI fails and there is no makeup in the injection phase. The initiator can be broken into two parts: 1) reactor coolant pump seal LOCAs; and 2) other small-small LOCAs. The dominant contributors to this sequence are failure of either of the two valves in the common minimum flow recirculation line. These valves are common to all High-pressure Safety Injection (HPSI), Low-pressure Safety Injection (LPSI), and Containment Safety Spray (CSS) pumps. For the small-small LOCA case, if these valves should fail closed, the HPSI pumps are assumed to fail due to pumping against dead head for a significant period of time (i.e., greater than 10 minutes).

3.3 LOSS OF 125 VDC BUS 11

- Sequence 4 - In this sequence, a failure of DC bus 11 results in a trip of Units 1 and 2 followed by failure of the PCS and AFW motor-driven pump 13, with degradation of the safety systems. The plant scrams successfully, but AFW subsequently fails. No credit is given for feed and bleed due to the low head of the HPI pumps. As a result of the lack of secondary heat removal, the core inventory boils off through the intermittent cycling of the power-operated relief valves (PORVs). The dominant contributors to this sequence are single failures in the AFW turbine-driven pump 11 train combined with failure of the operator to start the locked-out turbine-driven AFW pump 12.

3.4 LOSS OF OFFSITE POWER

- Sequence 5 - A loss of offsite power is followed by a transient-induced LOCA. AFW functions but HPSI fails. Due to lack of primary system makeup, the core uncovers in about 1 hour and core melt ensues. The dominant contributors are various failures of AC power train A combined with failures of AC power train B.

- Sequence 6 - A loss of offsite power is followed by failure of AFW. The plant scrams successfully. As a result of the loss of secondary heat removal, the core inventory boils off through the cycling open of the PORVs. No credit is given for feed and bleed due to the low head of the HPI pumps. The dominant contributors to this sequence are: (1) failure of the AFW motor-driven pump due to failure of train A of onsite AC power combined with failure of the AFW turbine-driven pump 11, together with (2) failure of the operator to start the locked-out turbine-driven AFW pump 12, and (3) failure to restore offsite power in order to restart the motor-driven AFW pump.

3.5 TRANSIENTS REQUIRING PRIMARY RELIEF

- Sequence 7 - In this sequence, a transient requiring primary pressure relief occurs followed by a failure to scram (ATWS) and either failure of emergency boration or induced LOCA due to a stuck open relief valve.

The primary system survives the initial pressure transient caused by a run back of the PCS as a result of the initiator. However, due to the high initial rate of coolant loss and low rate of pressure reduction, core uncovery and melt occurs before successful HPSI coolant injection. (This situation is unique to Calvert Cliffs design.) The dominant contributors to this sequence are failure to scram combined with either failure of the operator to initiate emergency boration or failure of a relief valve to reclose.

- Sequence 8 - A transient requires primary pressure relief followed by a loss of PCS and AFW. As a result of the loss of secondary heat removal, the core inventory boils off through the cycling open of the PORVs. No credit is given for feed and bleed in this accident sequence. The dominant contributor to this sequence is failure of 120V AC inverter #11 (which results in failure of the PCS and failure to actuate the motor-driven AFW pump) combined with various single failures of the AFW turbine-driven pump and failure of the operator to manually actuate the motor-driven AFW pump from the control room.

3.6 LOSS OF SECONDARY HEAT REMOVAL

- Sequence 9 - A loss of PCS occurs and is followed by a loss of AFW. As a result of the loss of secondary heat removal, the core inventory boils off through the intermittent cycling of the PORVs. The dominant contributors to this sequence are: 1) the common suction line valve fails closed resulting in failure of all operating AFW pumps combined with failure of the operator to recover by realigning the AFW suction to an alternate supply and starting the locked-out turbine-driven AFW pump; or 2) simultaneous failures of both operating AFW pumps combined with failure of the operator to start the locked-out turbine-driven AFW pump.

3.7 STATION BLACKOUT

- Sequence 10 - In this sequence, a loss of offsite power occurs followed by the loss of all onsite AC power. The AFW system functions until battery depletion occurs approximately four hours into the accident (offsite and onsite AC power not being recovered). Due to a lack of secondary heat removal, the primary system coolant heats up and boils. Within another two hours, core uncovery occurs followed by eventual core melt. All containment Heat Removal systems fail due to the loss of AC power.

3.8 OTHER TRANSIENTS

- Sequence 11 - In these sequences, a transient occurs, followed by failure to scram (ATWS) and an induced LOCA due to a stuck-open relief valve. The PCS is assumed to run back and the operator has successfully initiated emergency boration, and the primary system survives the initial pressure transient. However, due to the high initial rate of coolant loss and the low rate of pressure reduction, core uncovery and melt occurs before pressure drops to the 1275 psi shutoff head of the HPI pumps. The dominant contributor to this sequence is failure to scram combined with failure of a relief valve to reclose.
- Sequence 12 - A transient occurs and is followed by a loss of PCS and AFW. As a result of the loss of secondary heat removal, the reactor core inventory boils off through the cycling open of the PORVs. No credit is given for feed and bleed due to the information presented in the CC-1 PRA. The dominant contributors to this sequence are: 1) operator fails to manually start AFW motor-driven pump, 2) local fault of vital AC inverter #11, and 3) local fault of AFW turbine-driven pump and its associated valves.
- Sequence 13 - A transient is followed by a failure to scram and failure of emergency boration. The reactor has survived due to an assessed PCS runback. Based on information presented in the CC-1 PRA, if the operator fails to start shutting the reactor down within 20-30 minutes, then core melt will result. The dominant contributor to this sequence is failure to scram, combined with failure of the operator to initiate emergency boration within 20-30 minutes.

4.0 SYSTEM INSPECTION TABLES

Tables are presented for each of the systems selected in the analysis which identify important system failure modes, IE modules applicable to the inspection of system components, and the required position of each important component during normal system operation (i.e., system walkdown checklist). The systems are presented in decreasing order of risk importance, and together comprise more than 98% of the risk associated with plant operation. To provide useful information for the inspector, simplified system drawings and/or dependency diagrams are provided at the end of each section, with the exception of the SRV which is independent of all other systems.

4.1 REACTOR PROTECTION SYSTEM

TABLE 4.1A. REACTOR PROTECTION SYSTEM FAILURE MODE IDENTIFICATION

The Reactor Protection System (RPS) continuously monitors selected Nuclear Steam Supply System (NSSS) parameters which are essential to reactor protection. The RPS utilizes trip signals from various protective channels to de-energize and trip the reactor trip breakers. When the reactor trip breakers open, power is removed from the Control Element Drive Mechanism (CEDM) magnetic coils allowing the Control Element Assemblies (CEAs) to fall into the active fuel region of the core, thereby inserting negative reactivity and making the reactor subcritical.

The following failure modes were identified in the PRA document, which did not quantify the probabilities of these failures.

Conditions that Lead to Failure

1. Reactor Trip Breakers Feeding CEDM Groups 1 and 2 Fail to Open

   Simultaneous failure of reactor trip breakers feeding CEDM groups 1 and 2 to open given the presence of a trip signal is the primary contribution to failure of the reactor to trip. The dominant failure cause is hardware failure of the trip breakers. A contributing cause is operator failure to trip these breakers manually. Surveillance of the licensee's periodic testing and preventive maintenance activities and procedures in accordance with the Technical Specifications should reduce the probability of failure. Operator training and awareness of Emergency Operating Procedures will enhance the probability of recovery.

2. Control Rod Element Assemblies Fail to Insert

   Failure of the CEDM hold latches to release, or mechanical disruption of the reactor core could result in a failure to bring the reactor subcritical during a scram condition. Surveillance of the licensee's periodic testing and preventive maintenance activities and procedures in accordance with the Technical Specifications, and relevant NRC bulletins and information notices should reduce the probability of failure.

3. Cable Fault in Groups 1 and 2 CEDM Power Buses

   During normal operation, the CEDMs hold the control rods withdrawn from the core in a static position by means of CEDM hold or gripper, which latches the rods by means of an applied magnetic field. The control rods drop by de-energizing the magnetic coils. Failure of the CEDM power supplies to de-energize the magnetic coils upon interruption of power contributes significantly to the failure to trip event. The cause is hardware-related cable faults or shorts in groups 1 or 2 CEDM power buses. The licensee's periodic testing activities in accordance with the Technical Specifications should detect preexisting failures of this type.

TABLE 4.1B. IE MODULES FOR REACTOR PROTECTION SYSTEM INSPECTION

Module	Title	Components	Failure[a] Mode
41700	Training	Reactor Trip Breakers	1
61701	Surveillance (Complex)	Reactor Trip Breakers CEDM Power Buses Control Rod Element Assemblies	1 2 3
61725	Surveillance Testing and Calibration Program	Reactor Trip Breakers CEDM Power Buses Control Rod Element Assemblies	1 2 3
61726	Monthly Surveillance Observation	Reactor Trip Breakers CEDM Power Buses Control Rod Element Assemblies	1 2 3
62702	Maintenance (Section 02.03, Preventive Maintenance)	Reactor Trip Breakers Control Rod Element Assemblies	1 3
71707	Operational Safety Verification	CEDM Power Buses	2

(a) See Table 4.1A for failure identification.

TABLE 4.1C. MODIFIED REACTOR PROTECTION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
Walkdown is ineffective against risk significant RPS failures.				

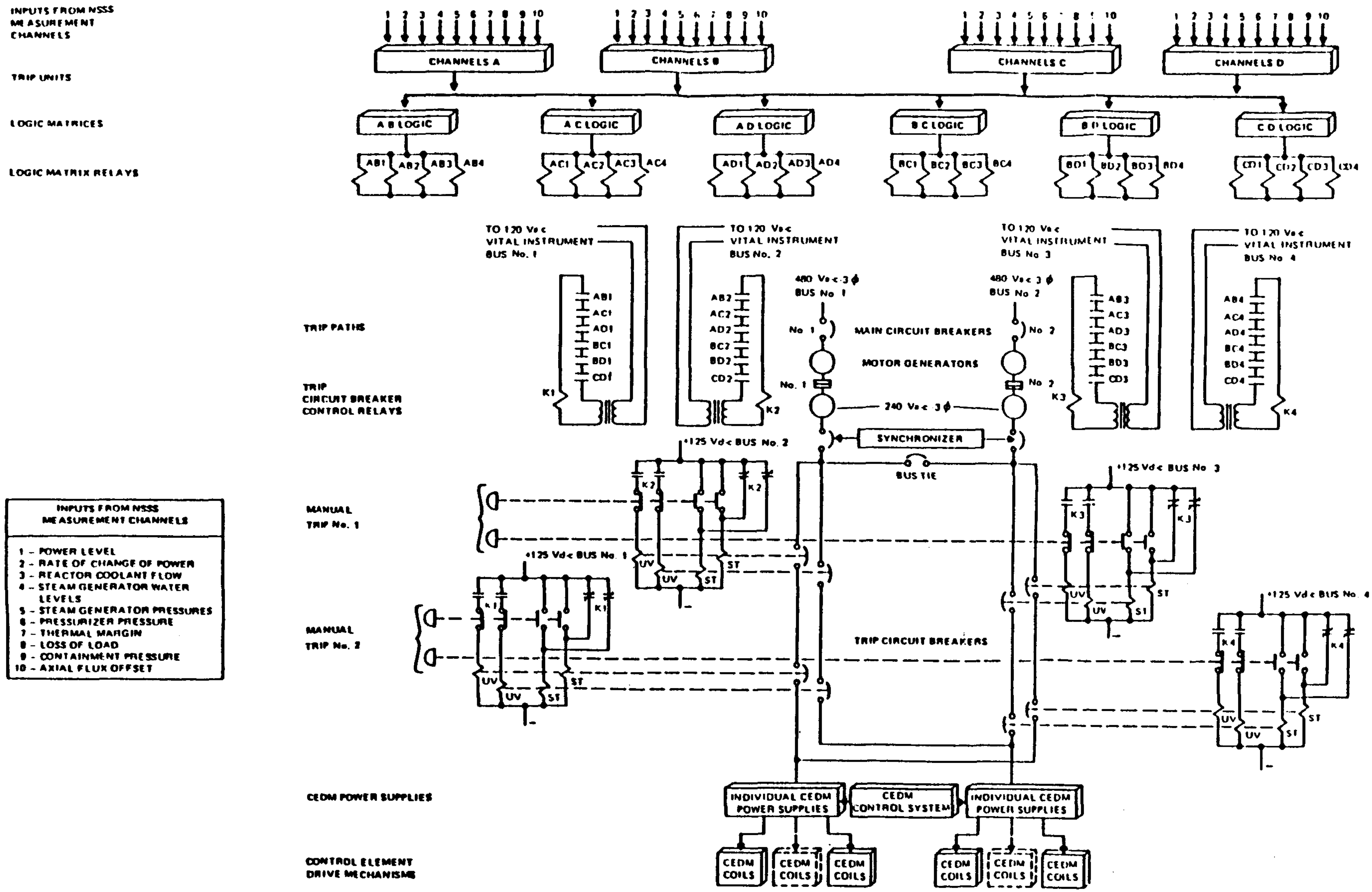


FIGURE 4.1. Simplified Functional Diagram of the Reactor Protection System

4.2 AUXILIARY FEEDWATER SYSTEM

TABLE 4.2A. AUXILIARY FEEDWATER SYSTEM FAILURE MODE IDENTIFICATION

The Auxiliary Feedwater (AFW) System supplies feedwater to the steam generators for evaporation permitting the removal of decay heat. The AFW system is used to cool the primary system to 300°F at which point shutdown cooling can be initiated. The AFW is used whenever the Power Conversion System (PCS) is not available. It consists of a pair of steam turbine-driven feed pumps (#11 and 12) (one of which is locked-out) connected in parallel with a motor-driven feed pump #13. Successful operation of the AFW is defined as the supply of a sufficient flow of feedwater to the steam generators to remove decay heat. This is equivalent to flow from at least one pump through at least two of four headers.

Conditions that Lead to Failure

1. Motor-driven AFW Pump #13 Fails to Start or Run

   Failure of Pump #13 to operate could contribute to failure of flow through the AFW system. This failure, in combination with failure of the turbine-driven pump to provide flow to the steam generators, is the dominant contributor to system failure. The dominant contributor to failure of pump #13 to start or run is a loss of electrical power at the 4KV bus. Other significant contributors to this failure mode are random hardware failures of the pump itself as well as failure of the electric power circuit breaker. Observation and review of surveillance, maintenance, and lineup of this pump will maintain reliability. Training in Emergency Operating Procedures and system malfunction response will enhance recovery when possible.

2. Turbine-driven AFW Pump #11 Fails to Start or Run

   Failure of Pump #11 to start or run could contribute to loss of AFW flow through the AFW system. This failure, combined with a loss of electrical power to the motor-driven pump and failure of the operator to start the locked-out turbine-driven pump #12, can lead to system failure. The dominant contributors to failure of pump #11 are local hardware or electrical faults which cause failure to start. Observation and review of surveillance, maintenance, and lineup of this pump will maintain reliability. Training in Emergency Operating Procedures and system malfunction response will enhance recovery when possible.

3. Manual Valves AFW-0103, -0904 Unavailable Due to Maintenance or Testing

   Manual valve 0103 closed blocks the discharge-flow from turbine-driven pump #11. This, in combination with loss of power to the motor-driven pump #13, can lead to system failure. Similarly, manual valve 0904 closed will prevent discharge flow from motor-driven pump #13. The dominant contributor to this failure mode is valve and pump unavailability due to periodic maintenance or testing. A secondary failure mechanism is failure to restore a valve to the correct position following maintenance or test of an AFW pump. Review of the periodic maintenance and testing procedures, including post-test surveillance, and adherence to the Technical Specifications should maintain valve availability.

4. <u>AFW Pump #11 or Pump #13 Unavailable Due to Maintenance</u>

   Failure of either Pump #11 or Pump #13 to be available when required results in system degradation. Failure of either of these pumps to operate while the other is out of service can lead to system failure. Periodic maintenance, which causes pump unavailability, in conjunction with hardware failure of the remaining pump is the dominant contributor to this failure mode. A review of the scheduled and unscheduled maintenance practices should be performed in order to maintain availability.

5. <u>Manual Valve AFW-0161 Fails to Remain Open</u>

   Failure of this valve to remain open could prevent flow from the condensate storage tank #12 to the Unit 1 AFW pumps, thus blocking flow to the steam generators. The dominant failure mechanism identified in the PRA for this failure mode is hardware failure of the valve. (Valve intervals have been removed since the PRA was performed.)

6. <u>Pneumatic-Hydraulic Valves AFW-3987 or S903 Fail to Remain Open</u>

   Failure of valve 3987 or stop valve S903 to remain open (plug) will prevent steam flow to the turbine-driven AFW pump #11. This failure, in combination with failure of the motor-driven pump, could lead to system failure. The dominant failure mechanisms for these valves are local hardware failures. Observation and review of surveillance and maintenance procedures associated with these valves (including air availability) should ensure reliable operation.

7. <u>Pneumatic-Hydraulic Valves AFW-4522 or 4532 Unavailable due to Maintenance</u>

   Failure of valve 4522 or valve 4532 to remain open prevents flow to steam generators #11 and #12 respectively. The dominant cause of this failure mode is periodic pneumatic valve maintenance. This includes both scheduled and unscheduled maintenance. The performance of maintenance should be reviewed to ensure that efficient scheduling is done, and that repairs are performed correctly, minimizing the unavailability of the valves.

TABLE 4.2B. IE MODULES FOR AUXILIARY FEEDWATER SYSTEM INSPECTION

Module	Title	Components	Failure[(a)] Mode
41700	Training	AFW Pump 13	1
		AFW Pump 11	2
		Valves 0103,0904	3
61725	Surveillance Testing and Calibration Program	AFW Pump 13	1,4
		AFW Pump 11	2,4
		Valves 3987,S903	6
		Valves 4522,4532	7
61726	Monthly Surveillance Observation	AFW Pump 13	1
		AFW Pump 11	2
		Valves 0103,0904	3
		Valve 0161	5
		Valves 3987,S903	6
		Valves 4522,4532	7
62700	Maintenance Program	AFW Pump 13	1,4
		AFW Pump 11	2,4
		Valves 0103,0904	3
		Valve 0161	5
		Valves 3987,S903	6
		Valves 4522,4532	7
62703	Monthly Maintenance Observation	AFW Pump 13	1,4
		AFW Pump 11	2,4
		Valves 0103,0904	3
		Valve 0161	5
		Valves 3987,S903	6
		Valves 4522,4532	7
71707	Operational Safety Verification	AFW Pump 13	1,4
		AFW Pump 11	2,4
		Valves 0103,0904	3
		Valve 0161	5
		Valves 3987,S903	6
		Valves 4522,4532	7
71710	ESF System Walkdown	AFW Pump 13	1,4
		AFW Pump 11	2,4
		Valves 0103,0904	3
		Valve 0161	5
		Valves 3987,S903	6
		Valves 4522,4532	7

(a) See Table 4.2A for failure mode identification.

TABLE 4.2C. MODIFIED AUXILIARY FEEDWATER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
Pump 13	Circuit Breaker	27' Turb Bldg	Racked in/ Closed	______
	Air			
MS-3987	Valve Air Supply	W Wall	On	______
AFW-4522	Valve Air Supply	E Wall SRW Rm	On	______
AFW-4532	Valve Air Supply	E Wall SRW Rm	On	______
	Valves			
AFW-S903	Stop Valve	W Wall	Open	______
MS-3987	Pneumatic-Hydraulic Valve	W Wall	Open	______
AFW-4522	Pneumatic-Hydraulic Valve	E Wall SRW Rm	Open	______
AFW-4532	Pneumatic-Hydraulic Valve	E Wall SRW Rm	Open	______
AFW-0103	Pump 11 Discharge Manual Valve	CST Outlet	Open	______
AFW-0904	Pump 13 Discharge Manual Valve	SRW Rm	Open	______
AFW-0161	CST-12 Outlet Manual Valve[a]	Between CST 11&21 on Stand	Open	______

(a) Valve internals have been removed since PRA, due to the high risk importance of this valve.

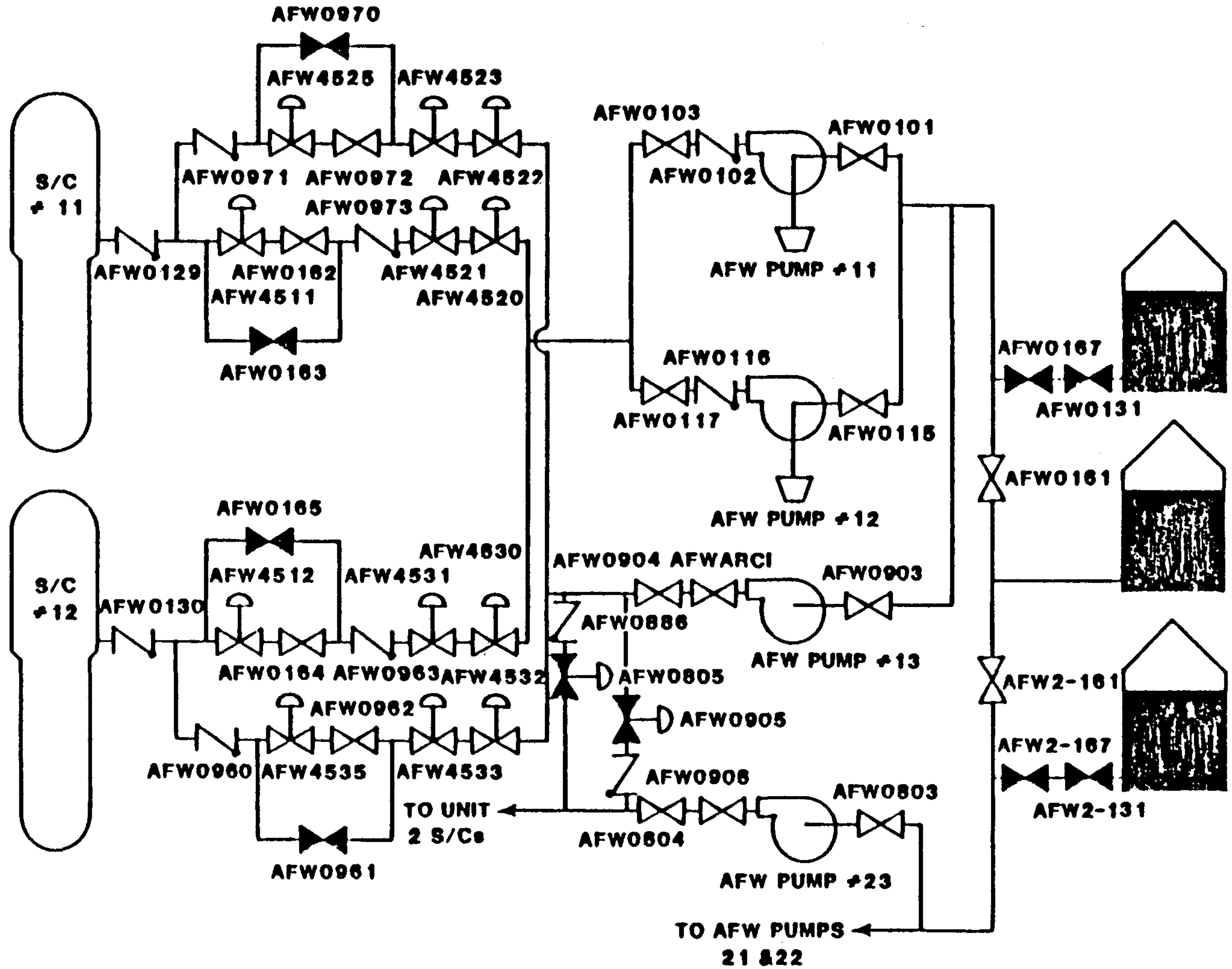


FIGURE 4.2. (Sheet 1) Simplified, System Drawing of AFWS

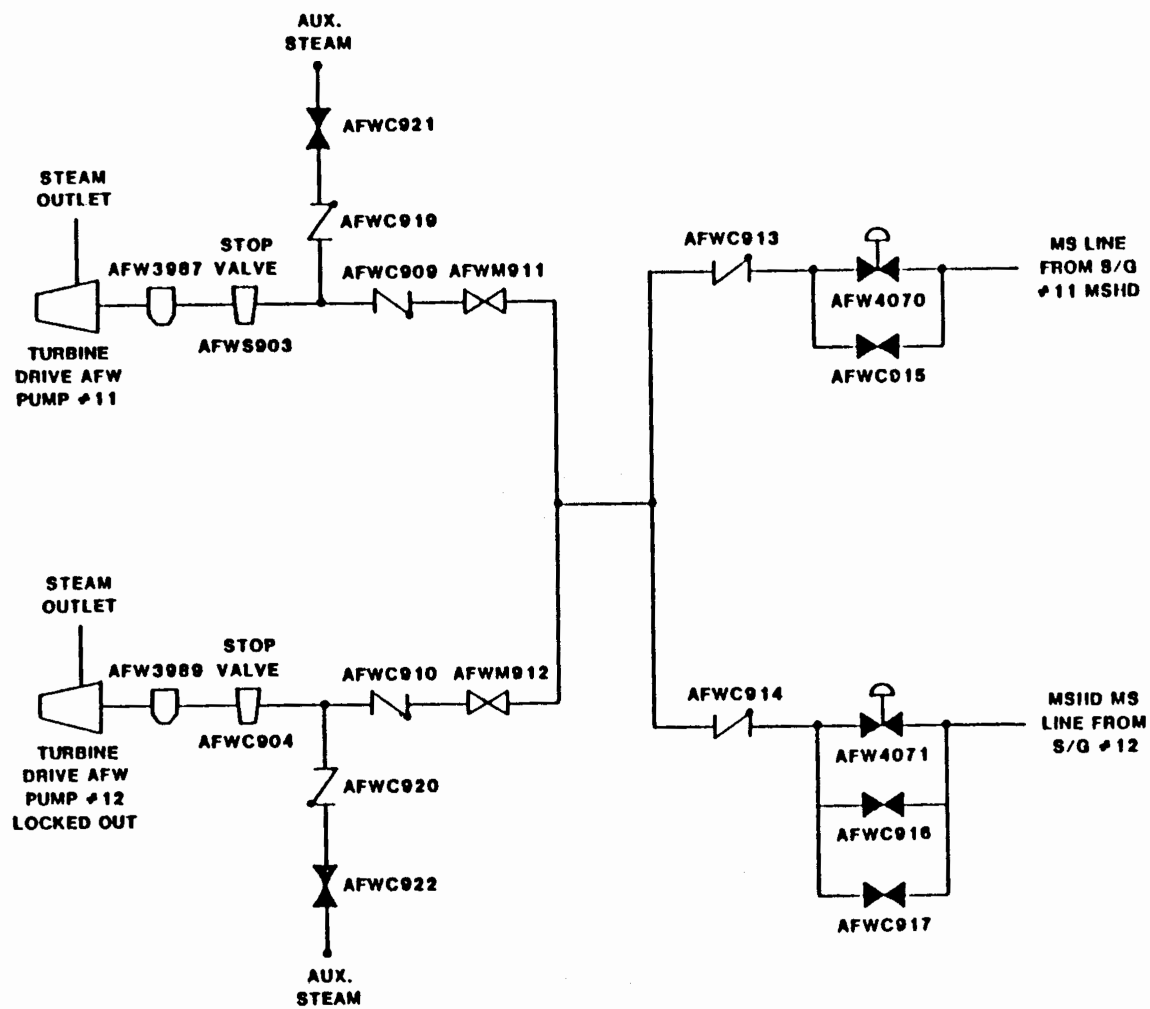


FIGURE 4.2. (Sheet 2) Simplified, System Drawing of AFWS

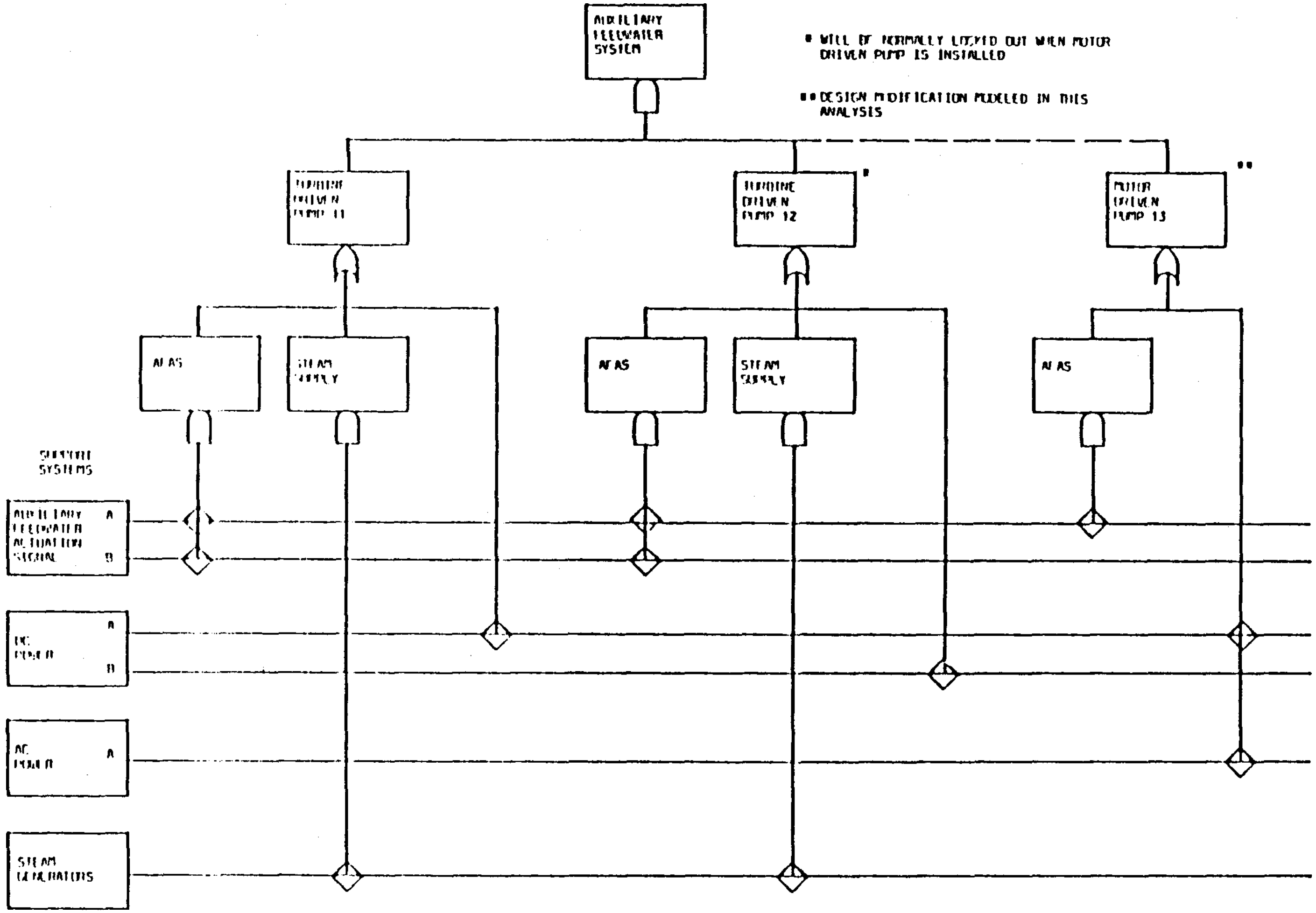


FIGURE 4.3. AFWS Support System Dependency Diagram

4.3 EMERGENCY DC POWER SYSTEM

TABLE 4.3A. EMERGENCY DC POWER SYSTEM FAILURE MODE IDENTIFICATION

The 125V DC system provides continuous power for control, instrumentation, reactor protection, and ESFAS. It powers the vital 120V AC system, which is also included in this analysis. The DC power system creates the field flashing and controls the diesel generator for the emergency AC electrical system. In addition, the DC power system provides control power to the emergency AC circuit breakers for the 4160V switchgear and powers the control valves in the AFWS.

The DC system is composed of four separate trains, each consisting of a 125V DC battery, bus, battery charger, and control panels. Also included in this analysis are those components which supply power from the 125 VDC buses to their respective 120V AC buses. These components include the electric power inverters, circuit breakers, transfer switches, and power cables. Therefore the failure modes identified below reflect the four-way redundancy of the DC power system and indicate conditions that lead to failure of any one train of the DC power system.

Conditions that Lead to Failure

---

1. Failure of Inverters 11A, 12B, 13B, 14A

   This is the dominant contributor to DC subsystem failure. These inverters convert DC power to AC power for the 120V AC distribution panels. Failure of any of these inverters results in loss of power to the affected 120V AC bus. Failure of the inverter may be the result of failure of switches, or hardware or electronic component failures. Observation and review of the periodic maintenance and testing and surveillance procedures should maintain availability.

2. Fault in Power Cables 11A, 12B, 13B, 14A

   Failure of any of these power cables which connect the 125V AC buses and the 120V DC buses will result in loss of power at the corresponding 120V AC bus. Failure of these power cables is typically the result of hardware failures. Observation and review of the periodic maintenance, testing and surveillance procedures should maintain availability.

3. Circuit Breaker 11A, 12B, 13B, 14A Fails Open

   Open circuit failure of any of these normally closed circuit breakers leads to loss of power at the respective 120V AC bus. Hardware or electric component failures are the dominant failure mechanisms. Review of the periodic maintenance and surveillance procedures along with verification of proper breaker position should maintain breaker performance.

TABLE 4.3A (contd)

4. <u>Fault in Power Cable 20A, 26B, 31B, 23A</u>

Failure in any of these power cables between the 125V DC buses and their respective batteries results in a loss of power to the affected 125V DC bus. This failure, coupled with a loss of offsite power, results in loss of power at the respective 120V AC bus. Hardware failures are the dominant failure mechanisms. Review of the periodic maintenance and surveillance procedures, as well as adherence to the Technical Specifications, should maintain power cable performance.

5. <u>Failure of Battery 11A, 12B, 13B, 14A</u>

Failure of any of these batteries results in a loss of power from the battery to its respective 125V DC bus. This failure in combination with other failures can result in a loss of all power at the affected 125V DC bus. Local faults of the battery itself are the dominant failure mechanisms for this failure mode. Periodic testing of battery voltage and specific gravity, in accordance with the Technical Specifications, as well as proper battery maintenance, should be reviewed and monitored.

6. <u>Circuit Breaker 20A, 26B, 31B, 23A Fails Open</u>

Failure of any of these normally closed circuit breakers in the open position leads to loss of power from the associated battery to their respective 125V DC bus. This failure in combination with a loss of offsite power results in a loss of all power at the affected 125V DC bus. Hardware or electric component failures are the dominant failure mechanisms. Periodic maintenance and surveillance of these breakers should be reviewed and proper breaker position should be verified.

TABLE 4.3B. IE MODULES FOR DC POWER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61725	Surveillance Testing and Calibration Program	Batteries 11A,12B,13B,14A	5
61726	Monthly Surveillance Observation	Inverters 11A,12B,13B,14A	1
		Power Cables 11A,12B,13B 14A	2
		Circuit Breakers 11A,12B, 13B,14A	3
		Power Cables 20A,26B,31B, 23A	4
		Batteries 11A,12B,13B,14A	5
		Circuit Breakers 20A,26B 31B,23A	6
62700	Maintenance Program	Inverters 11A,12B,13B,14A	1
		Power Cables 11A,12B,13B 14A	2
		Circuit Breakers 11A,12B, 13B,14A	3
		Power Cables 20A,26B,31B, 23A	4
		Batteries 11A,12B,13B,14A	5
		Circuit Breakers 20A,26B 31B,23A	6
71707	Operational Safety Verification	Inverters 11A,12B,13B,14A	1
		Circuit Breakers 11A,12B, 13B,14A	3
		Power Cables 20A,26B,31B, 23A	4
		Batteries 11A,12B,13B,14A	5
		Circuit Breakers 20A,26B 31B,23A	6
71710	ESF System Walkdown	Inverters 11A,12B,13B,14A	1
		Circuit Breakers 11A,12B, 13B,14A	3
		Batteries 11A,12B,13B,14A	5
		Circuit Breakers 20A,26B 31B,23A	6
		Supply Switches and Transfer Switches	1

(a) See Table 4.3A for failure mode identification.

TABLE 4.3C. MODIFIED EMERGENCY DC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical[a]			
DC Supply Switch	DC Supply Switch to Inverter 11	Cable Spreading Rm	Closed	______
DC Supply Switch	DC Supply Switch to Inverter 12	Cable Spreading Rm	Closed	______
DC Supply Switch	DC Supply Switch to Inverter 13	Cable Spreading Rm	Closed	______
DC Supply Switch	DC Supply Switch to Inverter 14	Cable Spreading Rm	Closed	______
11	Inverter 11 Manual Transfer Switch		Closed	______
12	Inverter 12 Manual Transfer Switch		Closed	______
13	Inverter 13 Manual Transfer Switch		Closed	______
14	Inverter 14 Manual Transfer Switch		Closed	______
3311A	Circuit Breaker 11A		Closed	______
12B	Circuit Breaker 12B		Closed	______
13B	Circuit Breaker 13B		Closed	______
14A	Circuit Breaker 14A		Closed	______
20A	Circuit Breaker 20A		Closed	______
26B	Circuit Breaker 26B		Closed	______
21B	Circuit Breaker 31B		Closed	______
23A	Circuit Breaker 23A		Closed	______

(a) Due to the integrated nature of the emergency DC power system, inspection of the inverters, power cables, and batteries should also be included in system walkdown procedures.

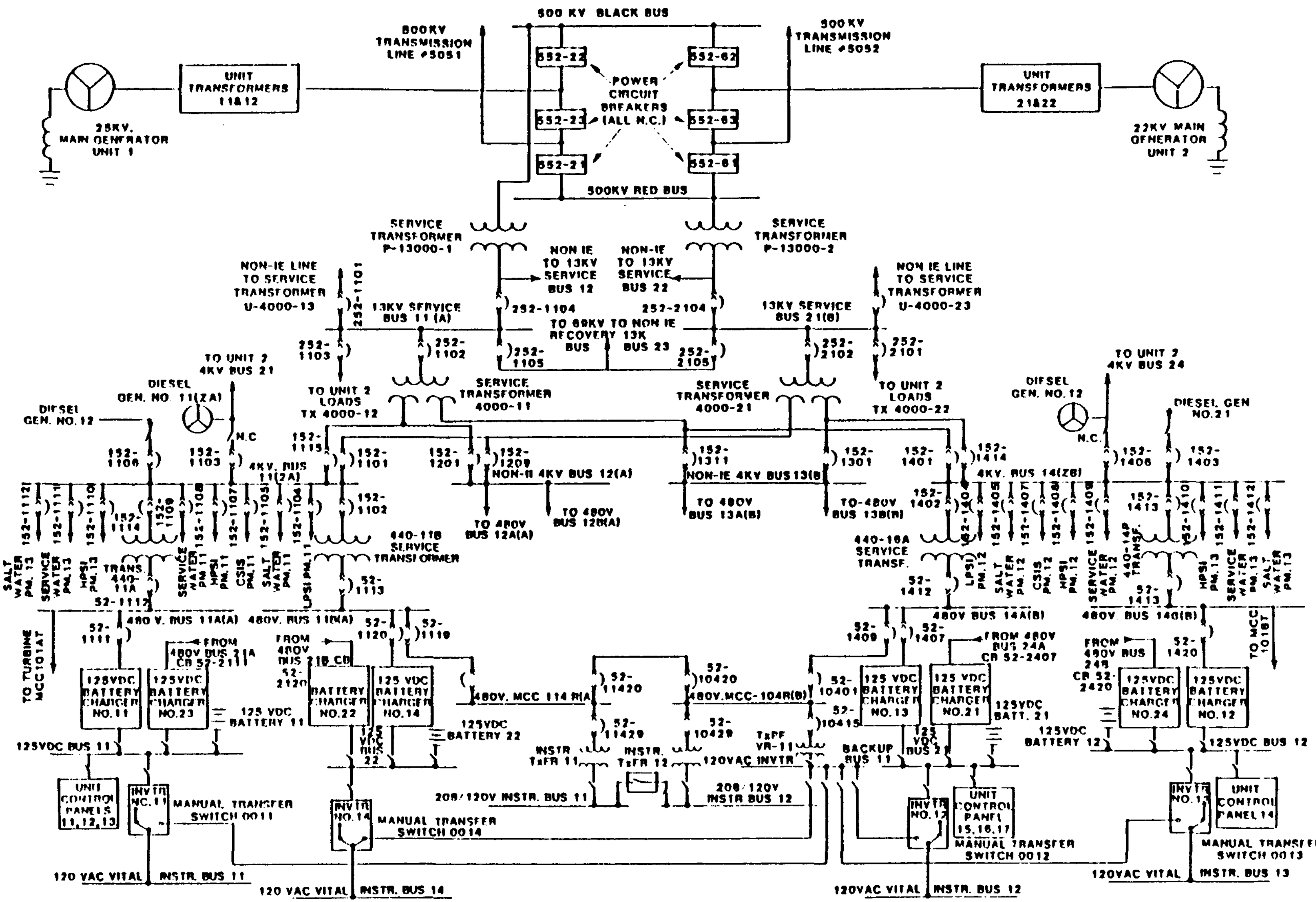
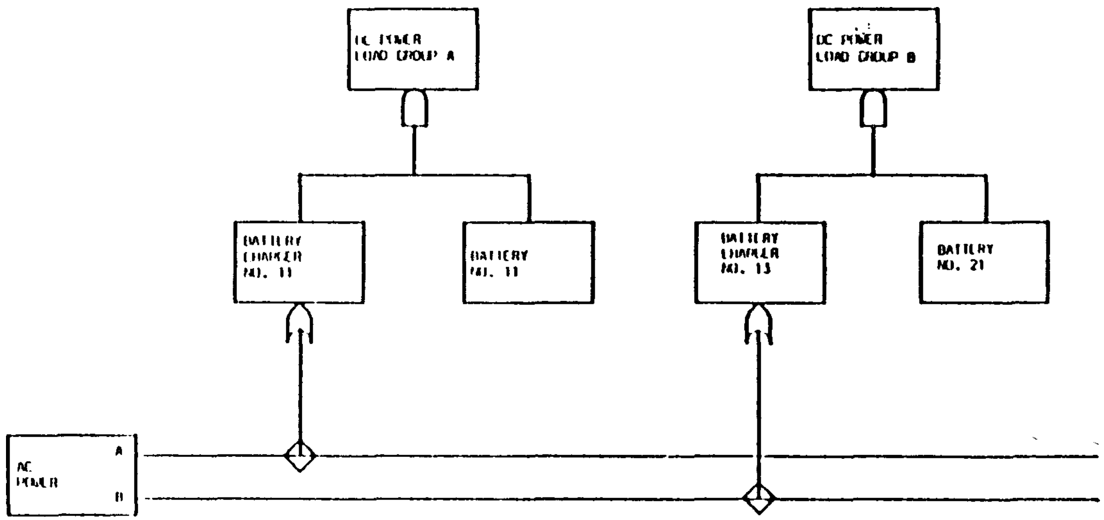


FIGURE 4.4. Simplified System Drawing of Emergency AC and DC System



<u>FIGURE 4.5</u>. Emergency DC Support System Dependency Diagram

4.4 EMERGENCY AC POWER SYSTEM

TABLE 4.4A. EMERGENCY AC POWER SYSTEM FAILURE MODE IDENTIFICATION

The purpose of the emergency AC power system is to provide electrical power to components in vital systems which are needed to mitigate the consequences of LOCAs and transients. Among these vital systems are those which shutdown the reactor, remove decay and sensible heat from the reactor coolant and the containment building, and limit the release of radioactive material from the containment. The AC power is required for the operation of pumps, fans, and MOVs in these systems.

The emergency AC system is composed of two trains, each of which consists of a diesel generator (DG), 4160V switchgear, 480V load centers and motor control centers, 120V instrumentation panels, and the associated transformers and circuit breakers. Therefore, the failure modes identified below reflect the redundancy of the emergency AC power system and indicate conditions that lead to failure of one train of the AC power system.

Conditions that Lead to Failure

---

1. Diesel Generator 12 Unavailable Due to Unit 2 Requirements or Operator Error

   Failure of DG 21 (which normally supports CC-2) requires that DG 12 be aligned to Unit 2. The loss of AC power due to DG 21 failure and subsequent alignment of DG 12 to Unit 2 is the dominant contributor to emergency AC power train failure at Unit 1. This results in emergency AC power system degradation. The dominant contributor to this failure mode is failure of DG 21 to start or run. A secondary contributor to this failure mode is operator failure to properly align DG 12 to Unit 1. Observation and review of the diesel generator maintenance and testing programs will help maintain reliable performance. Operator training regarding emergency operating procedures for proper DG alignment should also be reviewed.

2. Diesel Generator 11, 12 Fails to Start or Run

   Failure of either of the diesel generators to start or run when required will prevent emergency AC power from being supplied to the corresponding safeguards component buses. When combined with a loss of offsite power, a total loss of emergency AC power can result. The dominant contributor to this failure mode is random hardware failures within the diesel generator itself. A secondary contributor is failure of the DG room cooling system resulting in eventual failure of the DG to run, due to overheating. Observation and review of the DG maintenance and testing programs as well as review of the DG room cooling system maintenance program should maintain availability.

TABLE 4.4A (contd)

3. Diesel Generator 11, 12 Unavailable due to Maintenance or Testing

Failure of either of the diesel generators to be available when required will degrade system redundancy and therefore increase the probability of failure of the emergency AC power system. The dominant contributor to this failure mode is the downtime associated with periodic maintenance of the DGs. A secondary contributor to this failure mode is failure to properly restore the DG following a test. In order to assure maximum availability of the diesel generators, the periodic maintenance procedures should be reviewed, including post-test surveillance procedures, to ensure that efficient scheduling of maintenance is done and that repairs are performed quickly and correctly, minimizing DG downtime.

4. Circuit Breaker 1103A, 1406B Fail to Operate

Failure of either of these electric power circuit breakers results in failure of the affected diesel generator's power circuit output. This results in a failure of the DG to provide power to the corresponding safeguards component buses. The dominant contributor to this failure mode is failure of the system to automatically actuate the circuit breakers. A secondary contributor is hardware or electric failures of the circuit breakers themselves. The circuit breaker automatic actuation system should be reviewed to maintain availability. Maintenance and surveillance of these electric power circuit breakers should also be reviewed and observed.

TABLE 4.4B. IE MODULES FOR EMERGENCY AC POWER SYSTEM INSPECTION

Module	Title	Components	Failure[a] Mode
41700	Training	DG-12	1
61701	Surveillance (Complex)	DG-11,12	2
61725	Surveillance Testing and Calibration Program	DG-11,12 Breakers 1103A,1406B	2,3 4
61726	Monthly Surveillance Observation	DG-11,12 Breakers 1103A,1406B	2,3 4
62700	Maintenance Program	DG-11,12 Breakers 1103A,1406B	2,3 4
62707	Operational Safety Verification	DG-11,12 Breakers 1103A,1406B	2,3 4
71710	ESF System Walkdown	DG-11,12 Breakers 1103A,1406B	2,3 4

(a) See Table 4.4A for failure mode identification.

TABLE 4.4C. MODIFIED EMERGENCY AC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
1103A	DG-11 4KV Bus 11 Feed Breaker	27' Switch-gear Rm	Closed	______
1103A	DG-11 Disconnect to Bus 11	27' Switch-gear Rm	Closed	______
1406B	DG-12 4KV Bus 14 Feed Breaker	45' Switch-gear Rm	Closed	______
1406B	DG-12 Disconnect to Bus 14	45' Switch-gear Rm	Closed	______
DG-11	Diesel Generator 11		(a)	______
DG-12	Diesel Generator 12		(a)	______

(a) Due to the integrated nature of the diesel generator failure to start or to run failure modes, the lineup of all automatic diesel generator support functions (service water, fuel oil, starting air, etc.) should be verified.

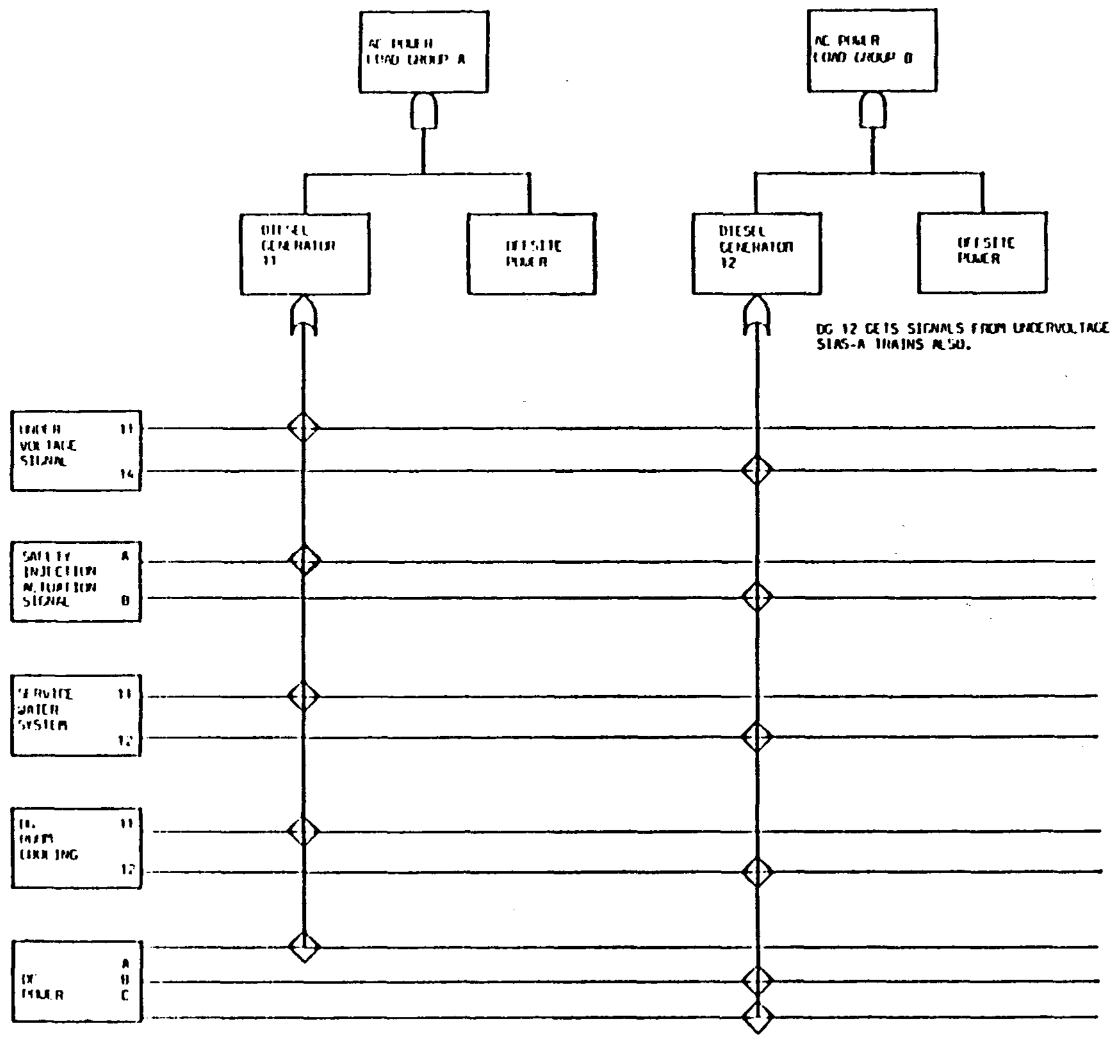


FIGURE 4.6. Emergency AC Support System Dependency Diagram

4.5 POWER CONVERSION SYSTEM

TABLE 4.5A. POWER CONVERSION SYSTEM FAILURE MODE IDENTIFICATION

The Power Conversion System (PCS) at Calvert Cliffs Unit 1 (CC-1) consists of the main feedwater and condensate system (MFWCS), and the steam generators. The PCS is designed to transfer feedwater from the condenser hotwell to the steam generators, while, at the same time, raising the temperature and pressure, and controlling the chemical composition of the feedwater. This system also controls the quantity of feedwater delivered to the steam generators. The PCS operates successfully to provide 5 percent full main feedwater flow to the steam generators if one train of the PCS remains in operation during a transient. One train is defined as one MFW pump, one condensate booster pump, one condensate pump, and the associated valves and piping.

Conditions That Lead to Failure

1. Failure of Turbine-Driven MFW Pumps 11 and 12

   The adequate removal of decay heat after reactor trip requires a supply of at least 750 gpm of feedwater to one of the two steam generators at a pressure of approximately 800 psia. To provide this flow, at least one of the two feedwater pumps must be available. If the transient initiator is the loss of one of the two main-feedwater pumps, failure of the other main-feedwater pump leads to a total loss of main feedwater. Main-feedwater pump failures are attributed to random failures of the pump, operator error in turning the pump off, a pump trip by trip circuits, loss of steam supply to the pump, or loss of cooling-water supplies for the pump. Operator training, awareness of Emergency Operating Procedures, proper maintenance, testing and surveillance should be reviewed and observed to reduce the probability of failure.

2. Failure of Condensate Booster Pumps 11, 12, and 13

   One out of the three condensate booster pumps is required to supply the cooling-water flow to the main feedwater lines during the transient. Simultaneous failure of three pumps will prevent sufficient flow of water to these lines. The failure causes are due to a random failure of the pump, pump(s) in maintenance, trip or start circuit failures, loss of electrical power, or loss of cooling-water supplies for the pumps. Proper maintenance, surveillance, and testing of the pumps which are not in use, according to Surveillance Test Procedures, should reduce the probability of failure.

TABLE 4.5A (contd)

3. Failure of Condensate Motor-Driven Pumps 11, 12, and 13

Three condensate motor-driven pumps are provided for the PCS. At least one out of the three pumps must remain in operation during a transient. Simultaneous failure of three pumps will prevent sufficient flow of water to the feedwater lines. The important failure causes are hardware failure, loss of electrical power, or insufficient inventory of water in the hotwell. Proper maintenance, surveillance, and testing of the pumps which are not in use, according to Surveillance Test Procedure should reduce the probability of failure.

TABLE 4.5B. IE MODULES FOR POWER CONVERSION SYSTEM INSPECTION

Module	Title	Components	Failure[a] Mode
41700	Training	MFW Pumps 11,12	1
61725	Surveillance Testing Calibration Program	MFW Pumps 11,12	1
61726	Monthly Surveillance Observation	MFW Pumps 11,12	1
		Condensate Booster Pumps 11,12,13	2
		Condensate MD Pumps 11,12,13	3
62700	Maintenance	MFW Pumps 11,12	1
		Condensate Booster Pumps 11,12,13	2
		Condensate MD Pumps 11,12,13	3
71707	Operational Safety Verification	MFW Pumps 11,12	1
		Condensate Booster Pumps 11,12,13	2
		Condensate MD Pumps 11,12,13	3

(a) See Table 4.5A for failure identification.

TABLE 4.5C. POWER CONVERSION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
MFW Pump 1	Main Feedwater Pump 11 Trip Circuit Breaker		Closed	______
MFW Pump 2	Main Feedwater Pump 12 Trip Circuit Breaker		Closed	______
Booster Pump 11	Condensate Booster Pump 11 Breaker		Racked in/ Closed	______
Booster Pump 12	Condensate Booster Pump 12 Breaker		Racked in/ Closed	______
Booster Pump 13	Condensate Booster Pump 13 Breaker		Racked in/ Closed	______
MD Pump 11	Condensate Motor-Driven Pump 11 Breaker		Racked in/ Closed	______
MD Pump 12	Condensate Motor-Driven Pump 12 Breaker		Racked in/ Closed	______
MD Pump 13	Condensate Motor-Driven Pump 13 Breaker		Racked in/ Closed	______

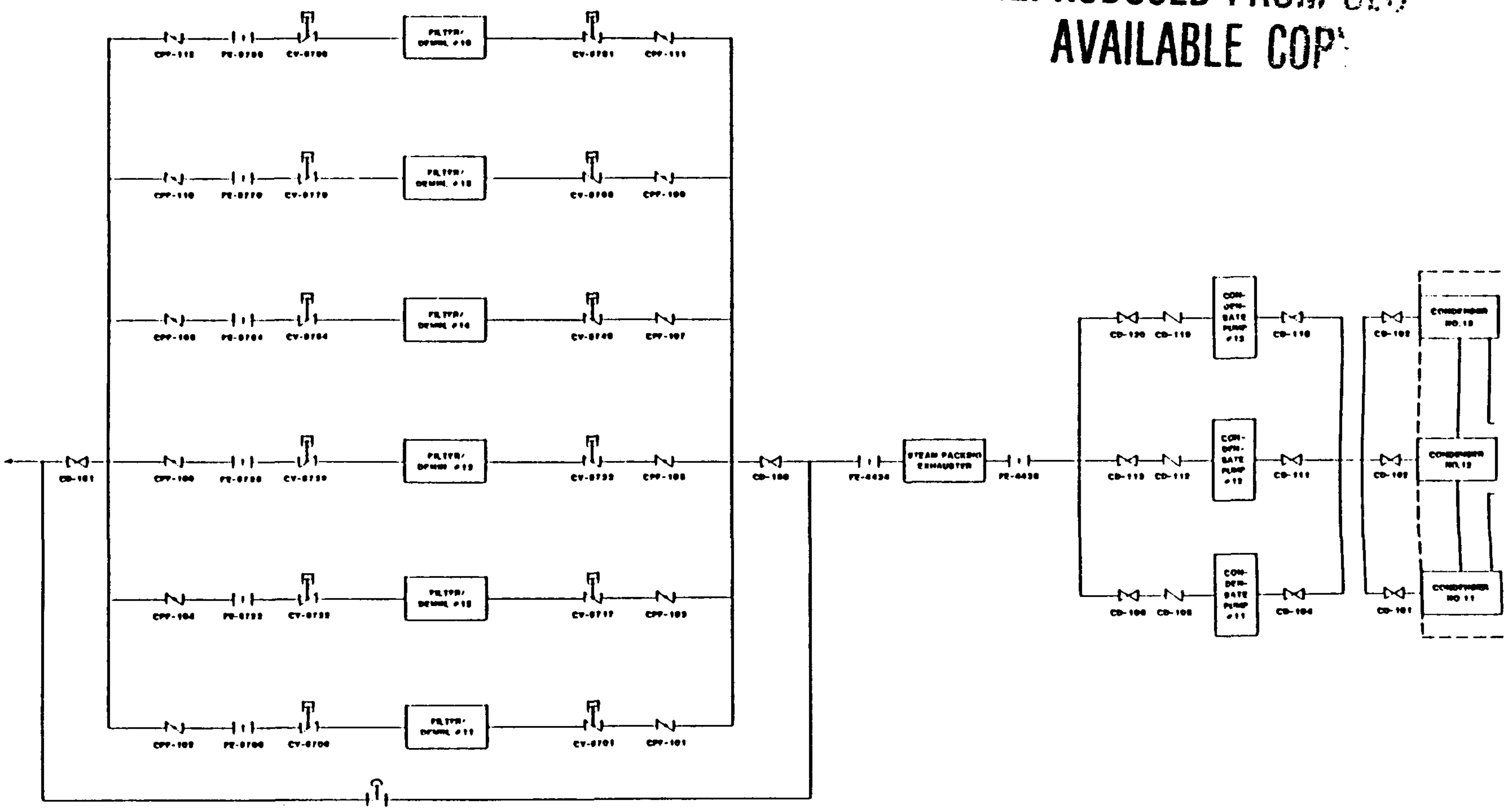


FIGURE 4.7. (Sheet 1) Simplified System Drawing of MFWCS

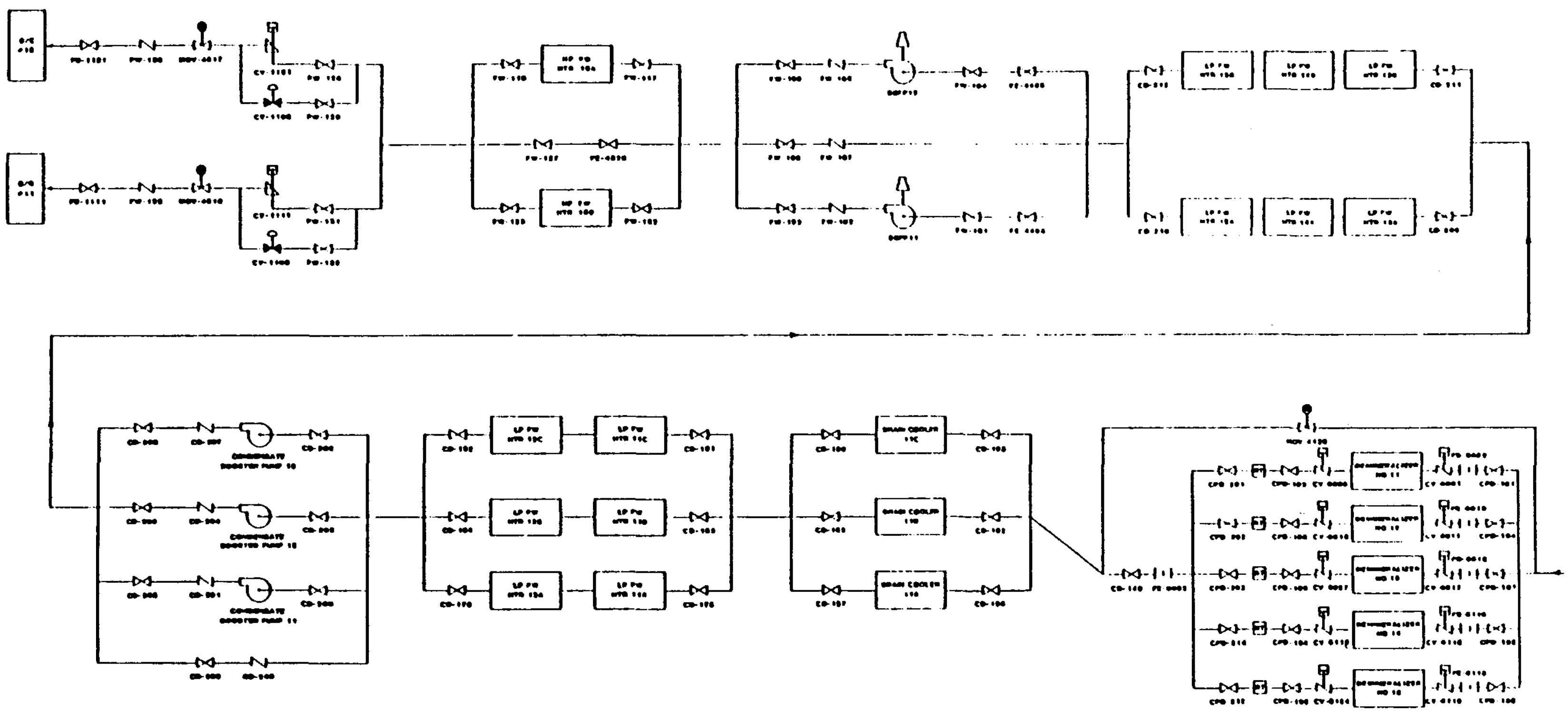


FIGURE 4.7. (Sheet 2) Simplified System Drawing of MFWCS

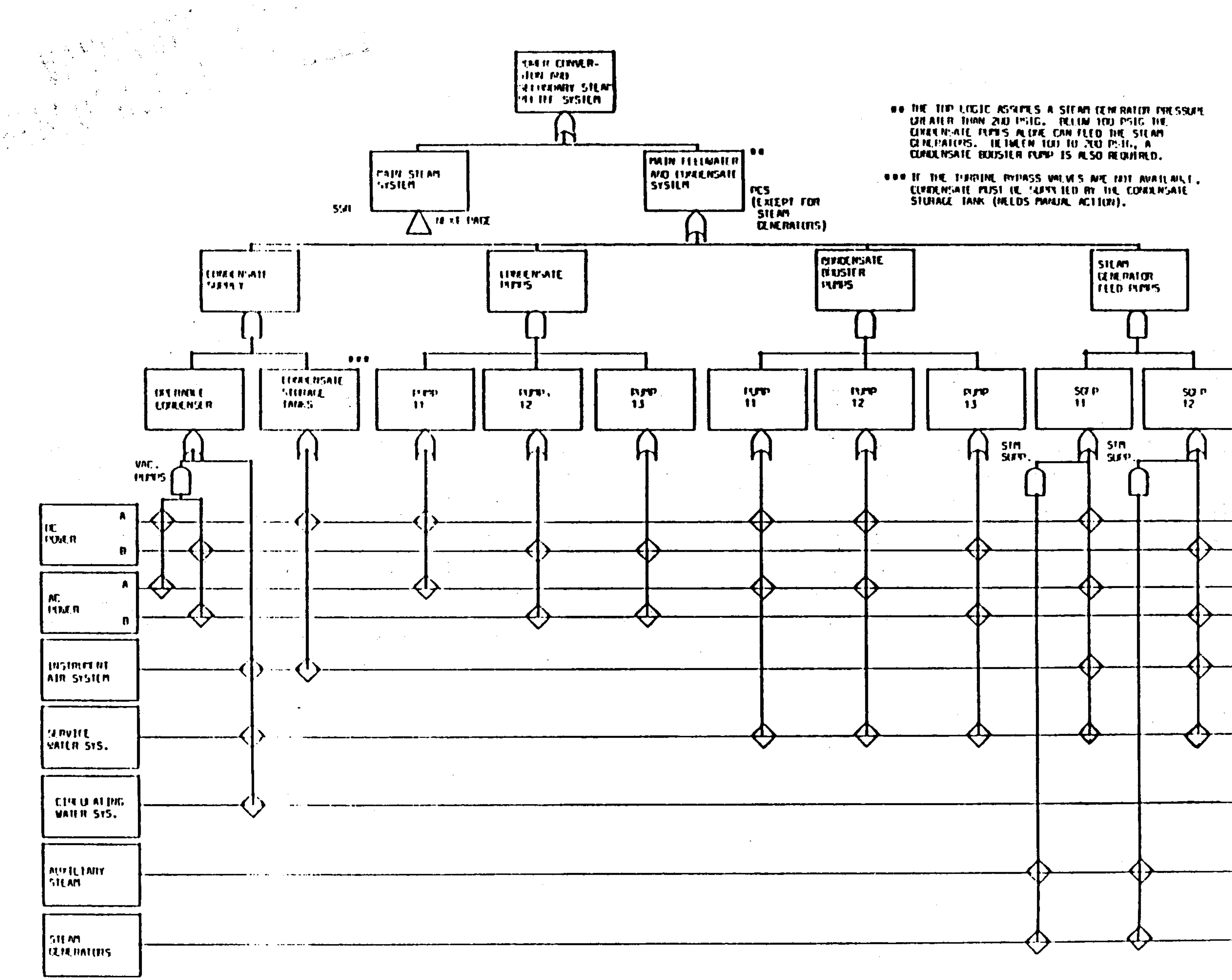


FIGURE 4.8. PCS Support System Dependency Diagram

4.6 HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM

TABLE 4.6A. HPSI SYSTEM FAILURE MODE IDENTIFICATION

The primary purpose of the High Pressure Safety Injection (HPSI) System is to inject borated water from the Refueling Water Tank (RWT) into the Reactor Coolant System (RCS) to prevent the uncovering of the core for small reactor coolant pipe breaks, to delay the uncovering of the core for intermediate-sized pipe breaks, and to guarantee sub-criticality of the core.

Once the RWT reaches a low level, the HPSI pumps are realigned to take suction from the containment sump for the recirculation mode. In this recirculation mode, the HPSI system maintains a borated water cover over the core for extended periods of time following a LOCA.

The HPSI System is a two-train, three-pump system which draws borated water from the RWT and injects it into the four RCS cold legs via four injection headers. The success criteria for the HPSI system is defined as 1 of 3 pumps providing flow through 1 of 4 injection headers.

Conditions that Lead to Failure

---

1. Containment Sump Values SI-4144 and 4145 Fail to Open On Demand

   After accident (e.g., LOCA), once the RWT reaches a low level, the HPSI pumps are automatically realigned to take suction from the containment sump for the recirculation mode through the sump isolation valves SI-4144 or 4145. Failure of both sump valves to open when required could lead to inadequate core cooling and eventual core-melt if not recovered. The dominant contributors to this failure mode are operator failure to manually open containment sump valves or failure to realign the HPSI pumps from the RWT to the containment sump. The secondary contributors are hardware or electrical failures of the valves. Operator training and awareness, verification and review of the Emergency Operating Procedures and check-off lists, surveillance, and testing of these valves should be reviewed and observed to minimize the probability of failure.

2. Injection Valves Fail to Operate

   There are eight normally closed HPSI injection valves which connect the two HPSI trains to four injection headers. These valves are required to open upon receiving a Safety Injection Actuation Signal (SIAS). Simultaneous failure of these eight MOVs (SI-616,-617,-626,-627,-636,-637,-646,-647) to open on demand leads to system failure. The dominant contributor to this failure mode is a loss of AC power from both 480V supply buses. The power supply to these valves should be reviewed to ensure reliability. In addition, operator training and awareness regarding emergency operating procedures should be reviewed to maintain adequacy.

TABLE 4.6A (contd)

3. <u>Motor-operated Valves SI-659, -660 Fail to Remain Open</u>

Failure closed of either of the two valves in the minimum flow recirculation line could cause the HPSI pumps to fail. This is because a slow drop in primary system pressure could result in pump heat up and failure due to pumping against dead head for a significant period of time. These valves are common to all HPSI, LPSI, and CSS pumps. The dominant contributor to this failure mode is valve plugging due to random hardware failures. The periodic testing and maintenance procedures for these valves should be reviewed to maintain reliability.

4. <u>HPSI Pump 11, 12, or 13 Fail to Start or Run</u>

Failure of either of these pumps to start or run degrades the HPSI system redundancy. Failure of any one of these pumps, in conjuction with failure of the remaining pumps to provide flow to the injection headers, can result in system failure. The dominant contributor to this failure mode is a loss of electric power from the 4KV bus to the pump. Secondary contributors to this failure mode are hardware failures which cause the pump to fail to start on demand. Periodic testing, maintenance, and surveillance of these pumps should be reviewed and observed to maintain high availability.

5. <u>Motor-Operated Valves SI-654, -656 Fail to Remain Open</u>

Failure of either MOV SI-654 or SI-656 prevents flow through one train of the HPSI system. Failures of either of these trains, in combination with failure of the 480V bus which supplies power to the remaining train, results in system failure. The dominant contributors to this failure mode are hardware failures. A secondary contributor to this failure mode is valve unavailability due to maintenance. The periodic testing and maintenance (scheduled and unscheduled) procedures for these valves should be reviewed to help maintain maximum availability.

6. <u>Check Valves SI-410, -4146 Fail to Operate</u>

Failure of either of these valves to remain open results in an obstruction of flow to HPSI pumps 11 and 12. This failure, in combination with failure of the 4KV bus, which supplies power to pump 13, results in system failure. The dominant contributors to this failure mode are hardware failures. The periodic testing and maintenance procedures for these valves should be reviewed to help maintain high availability.

TABLE 4.6A (contd)

7. <u>Motor-Operated Valves SI-653, -655 Fail to Remain Open</u>

Failure of these cross-tie valves to remain open degrades system redundancy and, in conjunction with failures of other components, could lead to system failure. The dominant contributors to this failure mode are hardware failures. Review of the periodic testing, maintenance and surveillance of these valves, as prescribed by the Technical Specifications, should help maintain valve availability.

TABLE 4.6B. IE MODULES FOR HPSI SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61725	Surveillance Testing and Calibration Program	Sump Valves SI-4144, -4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valve SI-653,-655	7
61726	Monthly Surveillance Observation	Sump Values SI-4144, - 4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valves SI-410,-4146	6
		Valve SI-653,-655	7
62700	Maintenance Program	Sump Values SI-4145, -4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valves SI-410,-4146	6
		Valve SI-653,-655	7
62703	Monthly Maintenance Observation	Sump Valves SI-4144, -4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valves SI-410,-4146	6
		Valve SI-653,-655	7

TABLE 4.6B (contd)

Module	Title	Components	Failure[a] Mode
71707	Operational Safety Verification	Sump Valves SI-4144, -4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valve SI-653,-655	7
71710	ESF System Walkdown	Sump Valves SI-4144, -4145	1
		Valves SI-616,-617 -626,-627,-636, -637,-646,-647	2
		Valves SI-659,-660	3
		HPSI Pump 11,12,13	4
		Breaker 11A,12B,13B	4
		Valves SI-654,-656	5
		Valve SI-653,-655	7

(a) See Table 4.6A for failure mode identification.

TABLE 4.6C. MODIFIED HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
1108	HPSI Pump 11 Circuit Breaker	27'Switch-gear Rm	Racked in/ Closed	____
1408	HPSI Pump 12 Circuit Breaker	45' Switch-gear Rm	Racked in/ Closed	____
1410	HPSI Pump 12 Control Room Switch	CR	PTL	____
1410	HPSI Pump 13 Circuit Breaker	45' Switch-gear Rm	Racked in/ Closed	____
1410	HPSI Pump 13 Disconnect	45' Switch-gear Rm	Closed	____
SI-616	Motor-operated Discharge Valve Breaker		Closed	____
SI-617	Motor-operated Discharge Valve Breaker		Closed	____
SI-626	Motor-operated Discharge Valve Breaker		Closed	____
SI-627	Motor-operated Discharge Valve Breaker		Closed	____
SI-636	Motor-operated Discharge Valve Breaker		Closed	____
SI-637	Motor-operated Discharge Valve Breaker		Closed	____
SI-646	Motor-operated Discharge Valve Breaker		Closed	____
SI-647	Motor-operated Discharge Valve Breaker		Closed	____

TABLE 4.6C (contd)

Component Number	Component Name	Location	Required Position	Actual Position
	Valves			
SI-659	Minimum flow recirculation valve		Open	____
SI-660	Minimum flow recirculation valve		Open	____
SI-654	Motor-operated valve		Open	____
SI-656	Motor-operated valve		Open	____
SI-4144	Containment Sump recirculation valve	45' Aux. Bldg.	Closed	____
SI-4145	Containment Sump recirculation valve	45' Aux. Bldg.	Closed	____
SI-616	Motor-operated Discharge Valve		Closed	____
SI-617	Motor-operated Discharge Valve		Closed	____
SI-626	Motor-operated Discharge Valve			
SI-627	Motor-operated Discharge Valve		Closed	____
SI-636	Motor-operated Discharge Valve		Closed	____
SI-637	Motor-operated Discharge Valve		Closed	____
SI-646	Motor-operated Discharge Valve		Closed	____
SI-647	Motor-operated Discharge Valve		Closed	____
SI-653	Motor-operated Cross-Tie Valve		Closed(a)	____
SI-655	Motor-operated Cross-Tie Valve		Open	____

(a) Normal valve position is opposite to that given in the Calvert Cliffs PRA report. This valve is now closed to prevent runout of either running pump if the other one should trip.

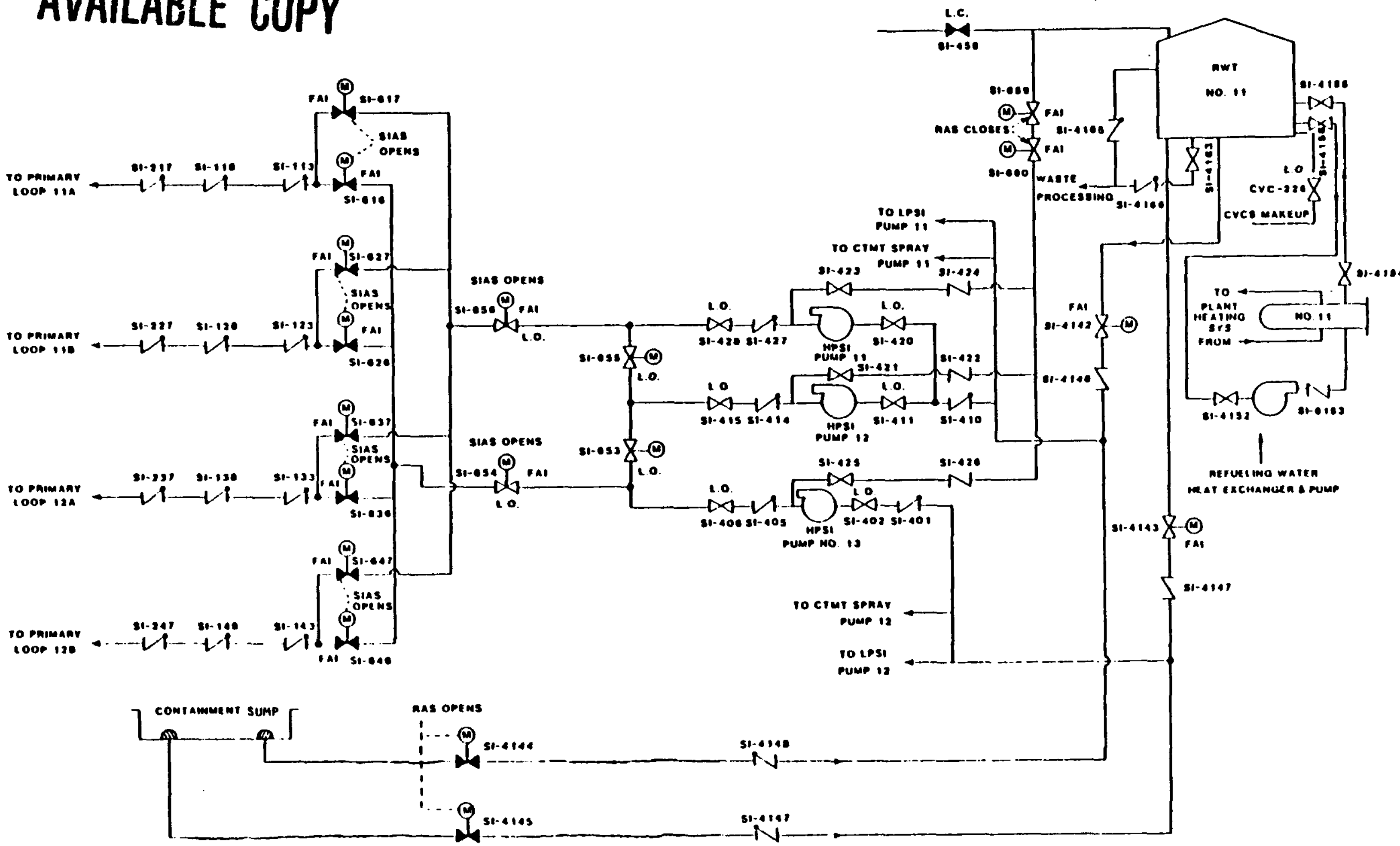


FIGURE 4.9. Simplified System Drawing of HPSI/R

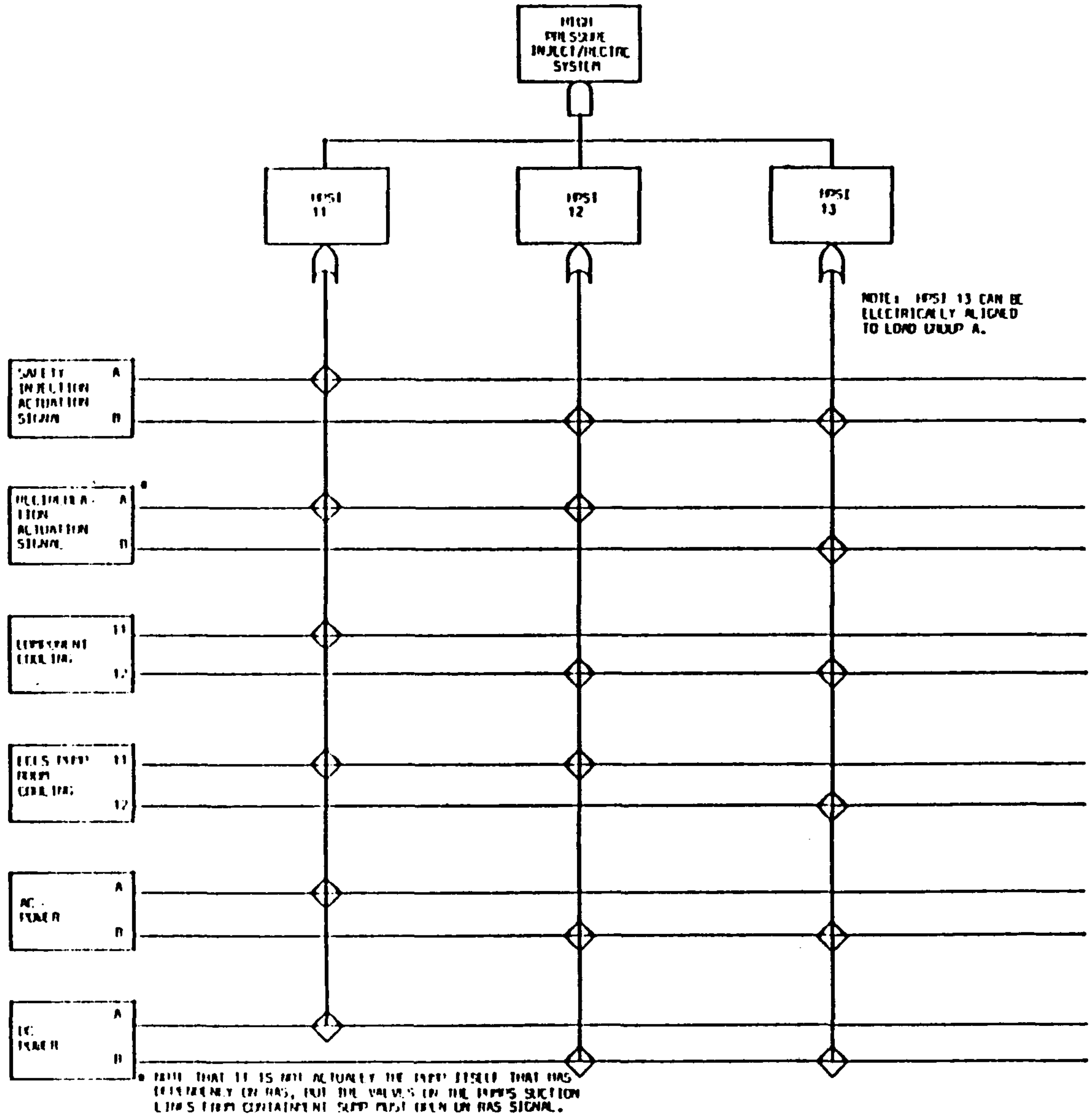


FIGURE 4.10. HPSI/R Support System Dependency Diagram

4.7 EMERGENCY CORE COOLING (ECCS) PUMP ROOM COOLING SYSTEM

TABLE 4.7A. ECCS PUMP ROOM COOLING SYSTEM FAILURE MODE IDENTIFICATION

The purpose of the ECCS pump room cooling system is to prevent the room temperature from exceeding a normal limit of 110°F or an absolute limit of 120°F so that the equipment can operate in a design basis environment. The components most susceptible to high temperatures are the air-cooled shaft seals on the containment spray pumps.

Conditions that Lead to Failure

1. Control Switches HS-5404 and 5404A Auto-Start Circuits Fail to Operate

   The ECCS room coolers will normally be on standby. Each unit has an individual (OFF-AUTO-ON) control switch located in the control room. The normal position of the control switches is AUTO so that the fans start and the saltwater cooling valves open automatically. Failure of the auto-start circuitry could prevent sufficient cooling-air flow to the ECCS room cooling units. The failure causes are hardware failures or loss of actuation signals. Testing, maintenance, surveillance, and verification of these switches should be reviewed or observed. Operator training and awareness of Operating Procedures will enhance the system reliability.

2. Air-Cooling Units' Fans Fail to Start or Run

   The system consists of two air-cooling units, one located in each ECCS compartment with the nominal capacities of 884,000 and 704,000 Btu/hr. The larger capacity cooling unit will be located in the east ECCS compartment, having two HPSI/R pumps. Each unit consists of direct-drive fans driven motors and the associated hardware and electrical equipment. Failure of fans to start or run when required will prevent air flow to the air-cooling units. The failure causes are hardware or electrical failures of the fans. Periodic testing, maintenance, and surveillance of the fans in accordance with the Technical Specifications, should reduce the probability of failure.

3. Temperature Controller Elements TE-5404 and TE-5405 Fail to Operate

   The system control parameter is room temperature, measured by temperature controller elements TE-5404 and 5405. The normal range of operation is between 95°F and 104°F. As the room temperature increases to 104°F, the temperature control element actuates the pressure switch, turns on the fans and opens the saltwater inlet and outlet valves. Failure of the temperature controller elements to operate either due to hardware or

TABLE 4.7A (contd)

electrical bus failures will prevent the cooling water from being supplied to the cooling units. Testing, surveillance, and maintenance, in accordance with Technical Specifications, should maintain the system availability.

4. Failure to Restore Air-Cooling Units' Fan Controls Following Maintenance

This failure mode is dominated by the human failure to restore the fan controls at the end of a test or following maintenance. This error, if undetected, could lead to system failure. Verification and review of the system Operating Procedures and Check-off Lists should reduce the probability of failure.

TABLE 4.7B. IE MODULES FOR ECCS PUMP ROOM COOLING SYSTEM INSPECTION

Module	Title	Components	Failure[(a)] Mode
41700	Training	Control Switches HS-5404, 5404A	1
61726	Monthly Surveillance Observation	Control Switches HS-5404, 5404A	1
		Air-Cooling Units' Fans	2,4
		Temperature Controller Elements TE-5404,5405	3
62700	Maintenance	Control Switches HS-5404, 5404A	1
		Air-Cooling Units' Fans	2,4
		Temperature Controller Elements TE-5404,5405	3
71707	Operational Safety Verification	Control Switches HS-5404, 5404A	1
		Air-Cooling Units' Fans	2,4
		Temperature Controller Elements TE-5404,5405	3
71710	ESF System Walkdown	Control Switches HS-5404, 5404A	1
		Air Cooling Units' Fans	2,4

(a) See Table 4.7A for failure identification.

TABLE 4.7C. ECCS PUMP ROOM COOLING SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
HS-5404	Air-Cooling Unit #11 Control Switch Circuit Breaker		Closed	______
HS-5404A	Air-Cooling Unit #12 Control Switch Circuit Breaker		Closed	______
Fans	Air-Cooling Units' Fans Breakers		Closed	______
TE-5404	Air-Cooling Unit #11 Temperature Control Element Circuit Breaker		Closed	______
TE-5404	Air-Cooling Unit #12 Temperature Control Element Circuit Breaker		Closed	______
	Components			
HS-5404	Air-Cooling Unit #11 Control Switch		Auto	______
HS-5404A	Air-Cooling Unit #12 Control Switch		Auto	______

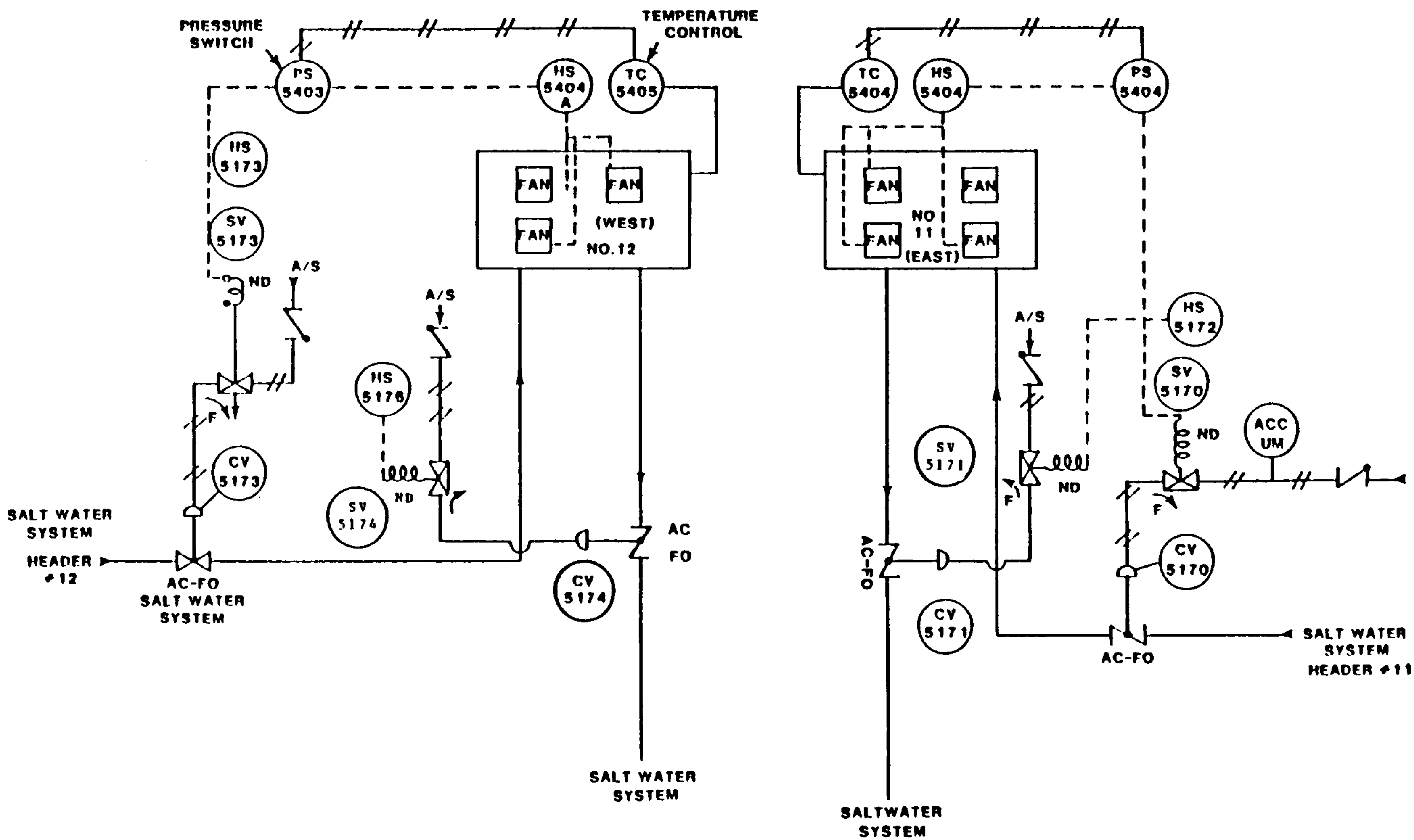


FIGURE 4.11. (Sheet 2) Simplified System Drawing of ECCS Pump Room Cooling System

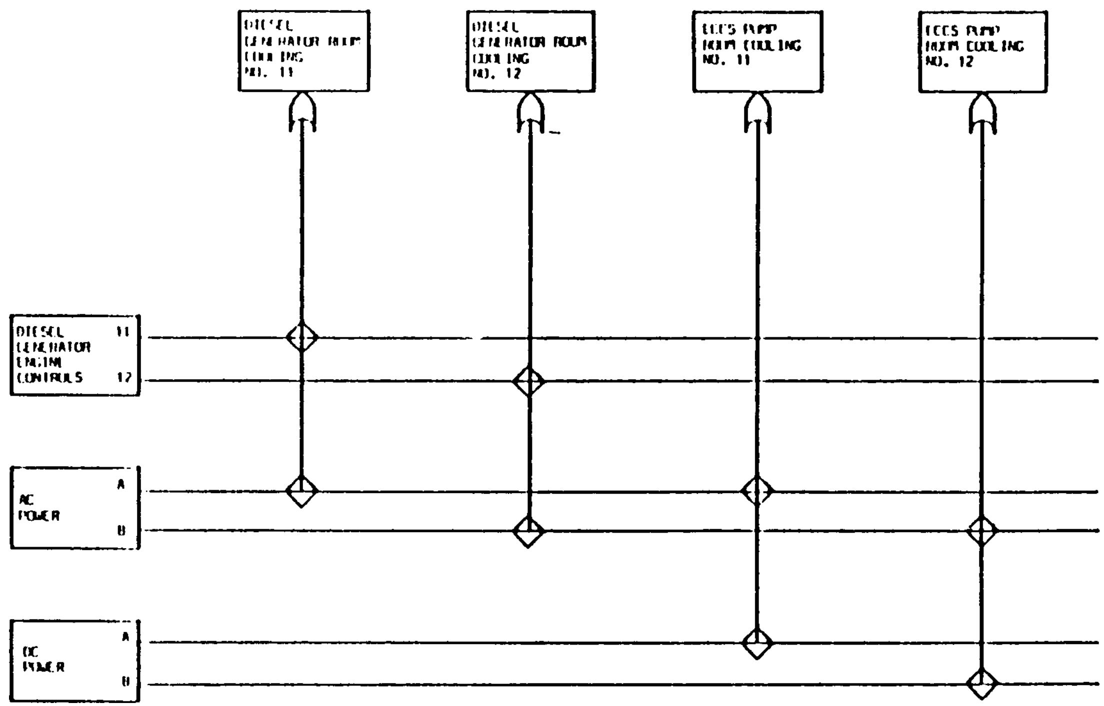


FIGURE 4.12. ECCS Pump Room Cooling System Dependency Diagram

4.8 SALT WATER SYSTEM

TABLE 4.8A. SALT WATER SYSTEM FAILURE MODE IDENTIFICATION

The purpose of the Salt Water System (SWS) is to provide cooling water to the vital equipment, (e.g., ECCS pump room air coolers, Service Water System (SRWS) heat exchangers) during both injection and recirculation phases of emergency core cooling after a LOCA. It also provides cooling water to Component Cooling Water (CCW) heat exchangers during the recirculation phase. Two of the three SWS pumps are required to supply the designated nuclear headers during the normal operation and following an initiating event.

Conditions that Lead to Failure

1. SRW Heat Exchangers' Outlet Throttle Valve SWS-5210 Fails to Operate

   During the recirculation phase, failure of the SRW heat exchangers outlet valve SWS-5210 to close (i.e., throttle) could cause the failure of cooling CCW heat exchanger 11 due to the system overload. The dominant failure causes are random electrical failures or valve control circuit fails. A contributing failure cause is human error failure to regulate the flow of salt water to meet the SRW cooling requirements. Operator training and awareness, surveillance and lineup for emergency operation should be reviewed and observed to minimize the probability of failure.

2. Failure of CCW Heat Exchangers' Inlet or Outlet Valves

   Both loops of the SWS operate during normal operation. Upon receipt of a safety injection actuation signal, CCW heat exchangers 11 and 12 are isolated by the automatic closing of inlet and outlet valves SWS-5160, SWS-5162 and SWS-5206, SWS-5208 and SWS-5163, respectively. These valves will reopen to allow the cooling flow to the CCW heat exchangers upon receipt of a recirculation actuation signal to initiate the recirculation phase. Failure to close or open these valves on demand may cause system overload. The important failure contributors are hardware or electrical failures or human failure to manually actuate these valves. Operator training and awareness, together with surveillance and testing of these valves in accordance with Technical Specification Surveillance Requirements should reduce the probability of failure.

3. SRW Heat Exchangers or ECCS Pump Room Coolers' Inlet or Outlet Valves Fail to Operate

   The valves are SRW heat exchangers or ECCS pump room coolers' inlet and outlet pneumatic valves SWS-5150, SWS-5152 and SWS-5153, or SWS-5170, SWS-5173 and SWS-5171, SWS-5174, SWS-5175, respectively. Failure of any of these valves to close or open when required could result in SWS cooling failure. The important failure contributors are electrical or hardware failures. Surveillance and testing of these valves, in accordance with Technical Specification Surveillance Requirements, should reduce the probability of failure.

4. <u>Operating Pumps 11 and 12 Fail to Run and Non-Operating Pump 13 Fails to Start and Run</u>

   Failure of any combination of two out of three pumps will prevent sufficient SWS flow from being provided to the designated headers. Testing of the pump which is not in use, according to the Technical Specifications, should reduce the probability of failure.

TABLE 4.8B. IE MODULES FOR SALT WATER SYSTEM INSPECTION

Module	Title	Components	Failure[a] Mode
41700	Training	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
61725	Surveillance Testing and Calibration Program	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
		SWS-5150,5152,5153, 5171,5173,5174,5175	3
		Pumps 11,12,13	4
61726	Monthly Surveillance Observation	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
		SWS-5150,5152,5153, 5171,5173,5174,5175	3
		Pumps 11,12,13	4
62700	Maintenance	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
		SWS-5150,5152,5153, 5171,5173,5174,5175	3
		Pumps 11,12,13	4
71707	Operational Safety Verification	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
		SWS-5150,5152,5153, 5171,5173,5174,5175	3
		Pumps 11,12,13	4
71710	ESF System Walkdown	SWS-5210,	1
		SWS-5206,5208,5160, 5162,5163	2
		SWS-5150,5152,5153, 5171,5173,5174,5175	3
		Pumps 11,12,13	4

(a) See Table 4.8A for failure identification.

TABLE 4.8C. MODIFIED SALT WATER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
SWS-5210	SWS Heat Exchanger 11 Pneumatic Outlet Valves Control Circuit Breaker		Closed	______
SWS-5206	CCW Heat Exchanger 11 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
SWS-5208	CCW Heat Exchanger 12 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
SWS-5160	CCW Heat Exchanger 11 Pneumatic Suction Valve Control Circuit Breaker		Closed	______
SWS-5162	CCW Heat Exchanger 12 Pneumatic Suction Valve Control Circuit Breaker		Closed	______
SWS-5163	CCW Heat Exchanger 12 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
SWS-5150	SWR Heat Exchanger 11 Pneumatic Suction Valve Control Circuit Breaker		Closed	______
SWS-5152	SWR Heat Exchanger 12 Pneumatic Suction Valve Control Circuit Breaker		Closed	______
SWS-5153	SWR Heat Exchanger 11 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
SWS-5171	ECCS Pump Room Cooler 11 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
SWS-5173	ECCS Pump Room Cooler 12 Pneumatic Suction Valve Control Circuit Breaker		Closed	______

TABLE 4.8C (contd)

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
SWS-5174 SWS-5175	ECCS Pump Room Cooler 12 Pneumatic Outlet Valve Control Circuit Breaker		Closed	______
1105	SWS Pump 11 Breaker	27' Switch-gear Rm	Racked in/ Closed	______
1405	SWS Pump 12 Breaker	45' Switch-gear Rm	Racked in/ Closed	______
1112/1412	SWS Pump 13 Breaker to Busses 11/14		Racked in/ Closed	______
1112/1412	SWS Pump 13 Disconnect to either Bus 11/14		Closed/ Open	______
	Air			
SWS-5210	SWS Heat Exchanger 11 Pneumatic Outlet Valve Air Supply	Under 11 HX	Open	______
SWS-5206	CCW Heat Exchanger 11 Pneumatic Outlet Valve Air Supply	E End 11 CC HX	Open	______
SWS-5208	CCW Heat Exchanger 12 Pneumatic Outlet Valve Air Supply	E End 12 CC HX	Open	______
SWS-5160	CCW Heat Exchanger 11 Pneumatic Suction Valve Air Supply	W of 11 CC HX	Open	______
SWS-5162	CCW Heat Exchanger 12 Pneumatic Suction Valve Air Supply	SW of 12 CC HX	Open	______
SWS-5163	CCW Heat Exchanger 12 Pneumatic Outlet Valve Air Supply	E End 12 CC HX	Open	______
SWS-5150	SRW Heat Exchanger 11 Pneumatic Suction Valve Air Supply	E Side SRW RM	Open	______
SWS-5152	SRW Heat Exchanger 12 Pneumatic Suction Valve Air Supply	N of 11 SRW PP	Open	______
SWS-5153	SRW Heat Exchanger 11 Pneumatic Outlet Valve Air Supply	N Wall 5' SRW RM	Open	______

TABLE 4.8C (contd)

Component Number	Component Name	Location	Required Position	Actual Position
	Air			
SWS-5171	ECCS Pump Room 11 Pneumatic Outlet Valve Air Supply	W of 11 CC HX	Open	______
SWS-5173	ECCS Pump Room 12 Pneumatic Suction Valve Air Supply	S Side CC Rm	Open	______
SWS-5174	ECCS Pump Room 12 Pneumatic Outlet Valve Air Supply	S Side CC Rm	Open	______
SWS-5175	ECCS Pump Room 12 Pneumatic Outlet Valve Air Supply	S Side CC Rm2	Open	______
	Valves			
SWS-5210	SWS Heat Exchanger 11 Pneumatic Outlet Valve	Under 11 HX	Throttle[a]	______
SWS-5206	CCW Heat Exchanger 11 Pneumatic Outlet Valve	E End 11 CC HX	Open[b]	______
SWS-5208	CCW Heat Exchanger 12 Pneumatic Outlet Valve	E End 12 CC HX	Open[b]	______
SWS-5160	CCW Heat Exchanger 11 Pneumatic Suction Valve	W of 11 CC HX	Open[b]	______
SWS-5162	CCW Heat Exchanger 12 Pneumatic Suction Valve	SW of 12 CC HX	Open[b]	______
SWS-5163	CCW Heat Exchanger 12 Pneumatic Outlet Valve	E End 12 CC HX	Open[b]	______
SWS-5150	SRW Heat Exchanger 11 Pneumatic Suction Valve	E Side SRW Rm	Open	______
SWS-5152	SRW Heat Exchanger 12 Pneumatic Suction Valve	N of 11 SRW PP	Open	______
SWS-5153	SRW Heat Exchanger 11 Pneumatic Outlet Valve	N Wall 5' SRW Rm	Open	______
SWS-5171	ECCS Pump Room 11 Pneumatic Outlet Valve	W of 11 CC HX	Closed	______

TABLE 4.8C (contd)

Component Number	Component Name	Location	Required Position	Actual Position
	Valves			
SWS-5173	ECCS Pump Room 12 Pneumatic Suction Valve	S Side CC Rm	Closed	______
SWS-5174	ECCS Pump Room 12 Pneumatic Outlet Valve	S Side CC Rm	Open[c]	______
SWS-5175	ECCS Pump Room 12 Pneumatic Outlet Valve	S Side CC Rm	Open[c]	______

(a) Valve is opened to its maximum upon receipt of SIAS signal, and returned to the throttled position upon receipt of RAS signal.
(b) Valves are closed upon receipt of SIAS signal, and reopened upon receipt of RAS signal.
(c) Normal operating position of this valve has been changed since the PRA. Position no longer changes on a SIAS or RAS signal. Consequently, the risk importance of this valve is reduced from the value determined from the PRA.

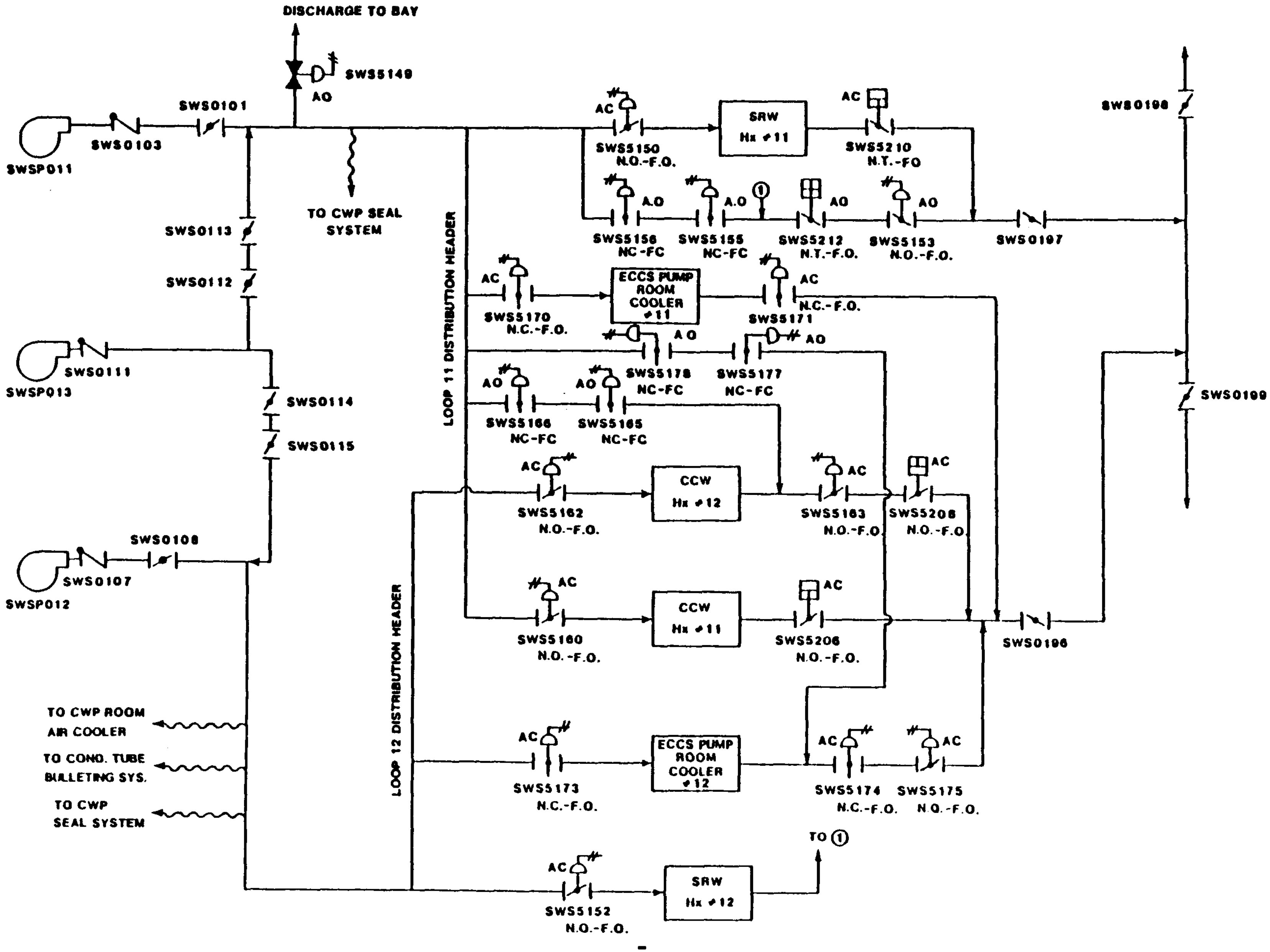


FIGURE 4.13. Simplified System Drawing of SWS

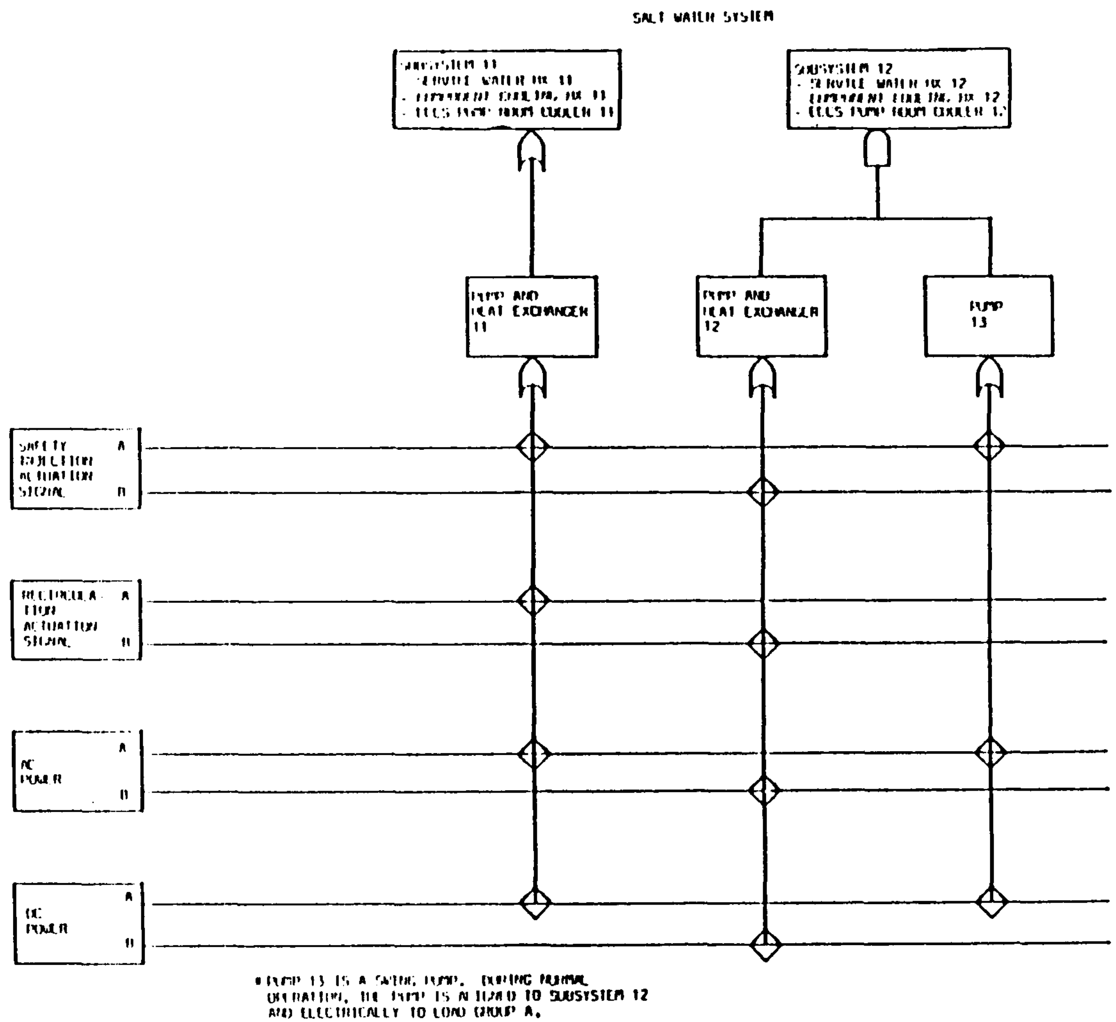


FIGURE 4.14. SWS Support System Dependency Diagram

4.9 CODE SAFETY VALVES (SRVs)

TABLE 4.9A. CODE SAFETY VALVES SYSTEM FAILURE MODE IDENTIFICATION

CC-1 is equipped with two code safety valves on the pressurizer. These valves are entirely mechanical devices. Their set points are 2500 and 2565 psia. At least one valve must be operable when the plant is at power. Under abnormal conditions, the code safety valves on the pressurizer are the means of external pressure relief for the reactor-coolant system.

Conditions That Lead to Failure

---

1. Code Safety Valves

   For those conditions where the code safety valves either are necessary to relieve system pressure or are only incidentally demanded, the valves must reclose or a transient-induced LOCA will result. The failure mode is random hardware failures of these valves. Testing and maintenance of these valves should be reviewed or observed to maintain the reliability.

TABLE 4.9B. IE MODULES FOR CODE SAFETY VALVES SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61701	Surveillance (Complex)		1
61725	Surveillance Testing and Calibration Program		1
62700	Maintenance		1
71707	Operational Safety Verification		1

(a) See Table 4.9A for failure identification.

TABLE 4.9C. MODIFIED CODE SAFETY VALVE SYSTEM WALKDOWN

Component Number	Component Name	Location	Required[a] Position	Actual Position
Walkdown is ineffective against failure of the code safety valves to reseat.				

4.10 CHEMICAL AND VOLUME CONTROL SYSTEM

TABLE 4.10A. CHEMICAL AND VOLUME CONTROL SYSTEM FAILURE MODE IDENTIFICATION

The Chemical and Volume Control System (CVCS) provides several major functions during startup, normal operation, emergency condition, and shutdown of the reactor. The reactor coolant system boron concentration is normally controlled by the makeup portion of the CVCS; however, there are occasions when it is necessary to borate at a rate that exceeds the normal, maximum capability of the makeup system. In this situation, the CVCS is initiated either by a Safety Injection Actuation System (SIAS) or manually to rapidly inject concentrated boric acid into the reactor coolant system.

Conditions That Lead to Failure

---

1. Operator Fails to Actuate the CVCS

   During a transient where SIAS and RPS actions are not initiated, operation of the CVCS in boron injection mode is completely dependent upon operator action. For example, following a slow and uncontrolled cooldown where SIAS or RPS actions are not initiated, immediate actions to be taken by the operator are: switch the make-up stop valve handswitch to the "SHUT" position; open the charging pump suction valve; switch the make-up mode selector switch to the "BORATE" position; verify that the boric acid and charging pumps are running, etc. Failure to perform any of the above actions could prevent the boric acid flow to the reactor coolant system. Operator training and awareness, verification of the Emergency Operating Procedures Check-off Lists should minimize the probability of failure.

2. Operating Charging Pumps Fail to Run and Standby Pump in Maintenance

   Two of the three charging pumps are required to supply the designated nuclear headers following an initiating event. Standby charging pump unavailable due to maintenance is significant in conjunction with hardware or electrical failures of the operating pumps. Periodic testing, surveillance, and review of the practices associated with scheduled and unscheduled maintenance of the pump should be performed.

3. Motor-Operated Valve CVC-514 Fails to Open on Demand

   For emergency boration, a boric acid direct feed valve CVC-514 is provided. This is a motor-operated valve located at the common boric acid pump discharge which supplies concentrated boric acid directly to the charging pump headers. Failure of this valve in the closed position following an initiating event will prevent boric acid flow to the reactor coolant system. The failure causes are random electrical failures or loss of SIAS. Periodic testing, maintenance, and surveillance of this valve according to the Technical Specifications should reduce the probability of failure.

4. Boric Acid Pumps 11, 12 Fail to Start and Run

   At least one of the two boric acid pumps must be operable to provide concentrated boric acid to two out of the three charging pumps for injection into the reactor coolant system when the requirement arises due to a transient. Failure of the pumps to start and run will prevent boric acid flow to the reactor coolant system cold legs. Periodic testing, maintenance, and surveillance of these pumps should be reviewed and observed to maintain reliability.

5. Check Valve CVC-235 Fails to Operate

   This is charging pump suction check valve for CVCS. Failure of this valve in the closed position will prevent boric acid fow to the designated cold legs. Testing and maintenance of this valve, in accordance with Technical Specifications, should reduce the probability of failure.

TABLE 4.10B. MODIFIED CHEMICAL AND VOLUME CONTROL SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Make-up Stop Valve,	1
		Charging Pumps 11,12,13	2
		MOV CVC-514,	3
		BA Pumps 11,12	4
61725	Surveillance Testing Program	Charging Pumps 11,12,13	2
		MOV CVC-514	3
		BA Pumps 11,12	4
		Check Value CVC-235	5
61726	Monthly Surveillance Observation	Charging Pumps 11,12,13	2
		MOV CVC-514	3
		BA Pumps 11,12	4
62700	Maintenance	Charging Pumps 11,12,13	2
		MOV CVC-514	4
		BA Pumps 11,12	3
		Check Value CVC-235	5
71707	Operational Safety Verification	Charging Pumps 11,12,13	2
		Check Value CVC-514	3
		BA Pumps 11,12	4
71710	ESF System Walkdown	Charging Pumps 11,12,13	2
		Check Value CVC-514	3
		BA Pumps 11,12	4

(a) See Table 4.10A for failure identification.

TABLE 4.10C. MODIFIED CHEMICAL AND VOLUME CONTROL SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
BA Pump 11	Boric Acid Pump 11 Breaker	MCC-114R	Closed	________
BA Pump 12	Boric Acid Pump 12 Breaker	MCC-104R	Closed	________
1115	Charging Pump 11 Breaker	27' Switch-gear Rm	Racked in/ Closed	________
1415	Charging Pump 12 Breaker	45' Switch-gear Rm	Racked in/ Closed	________
1104/1404	CP 13 Breaker from Bus 11/14		Racked in/ Closed	________
1104/1404	CP 13 Disconnect from Bus 11/14		Closed	________
CVC-514	Boric Acid Direct Feed Motor-Operated Valve Breaker	E Wall BAST Rm	Closed	________
	Valves			
CVC-514	Boric Acid Direct Feed Motor-Operated Valve	BAST Rm	Closed	________

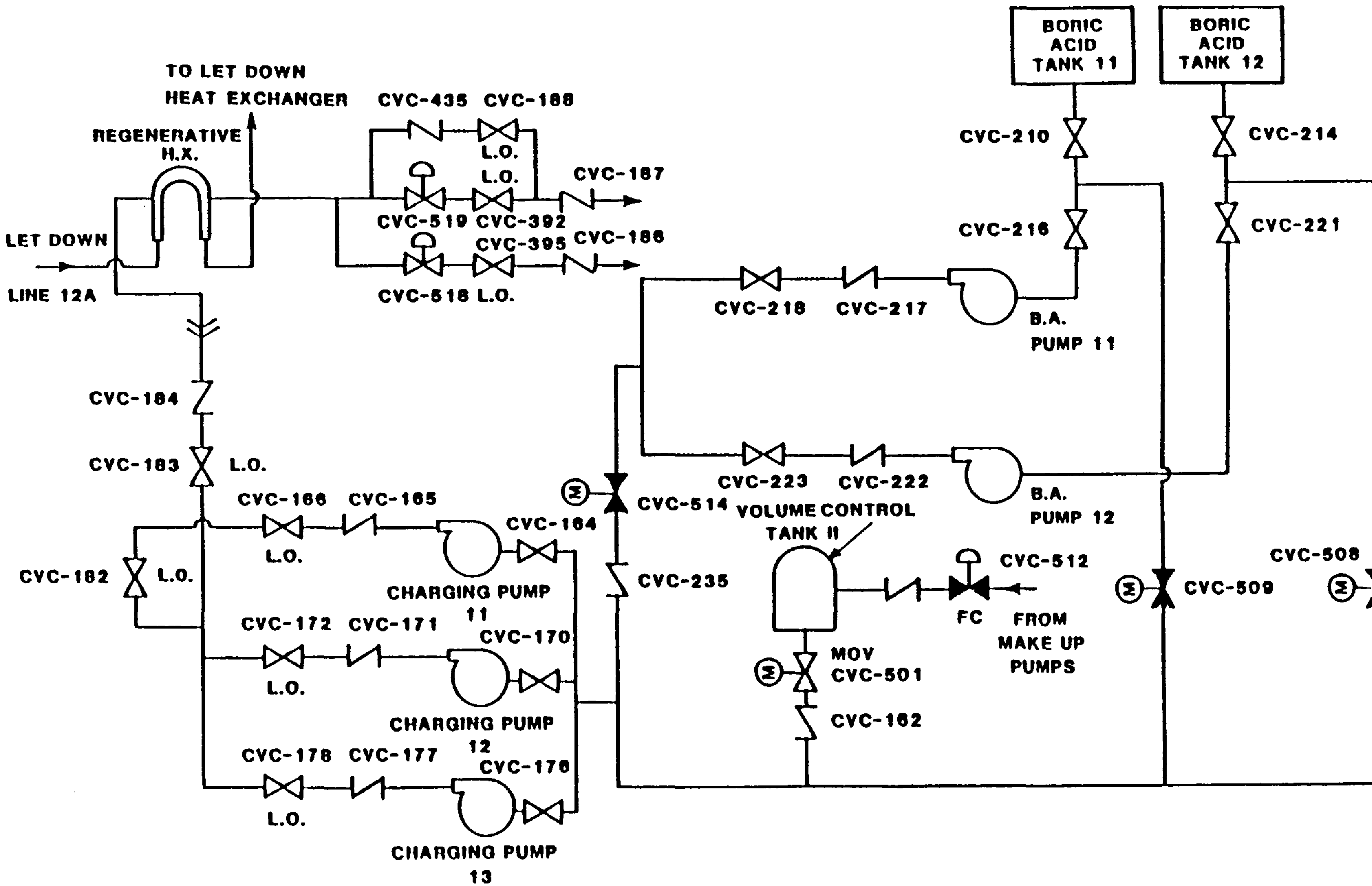
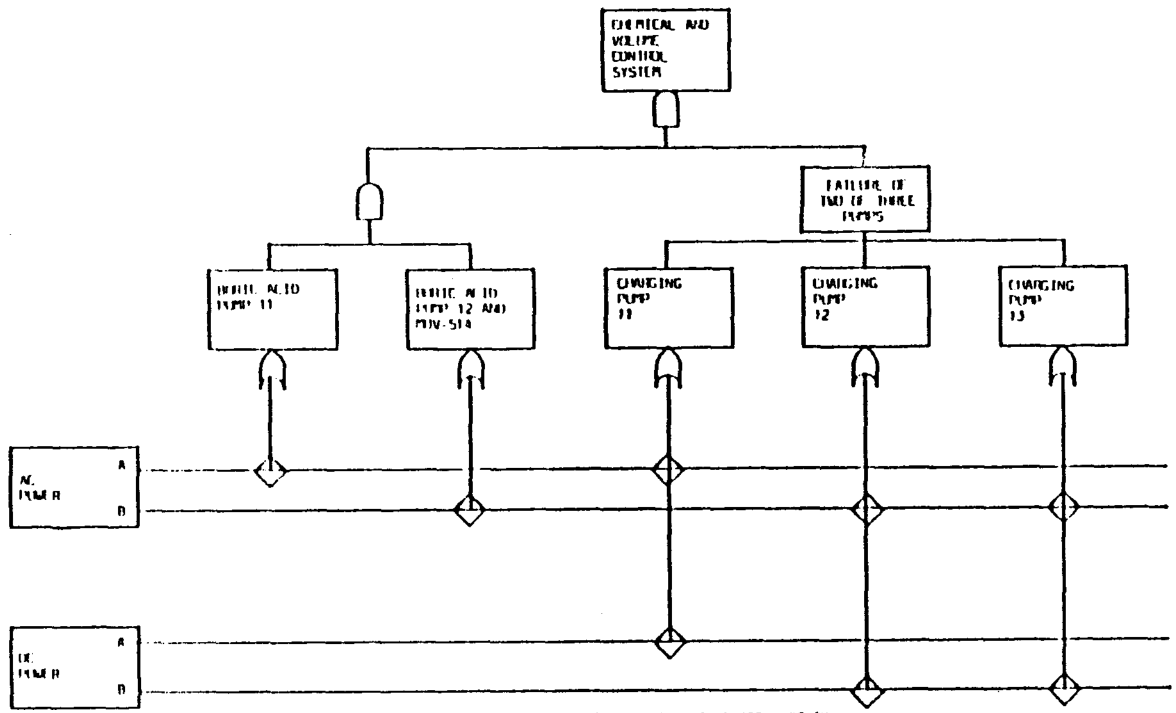


FIGURE 4.15. Simplified System Drawing of CVCS



NOTES 1. OPERATOR ACTION REQUIRED TO PLACE CVCS IN EMERGENCY BORATE MODE.

2. CHARGING PUMP 13 CAN ALSO BE ALIGNED TO LOAD GROUP A.

FIGURE 4.16. CVCS Support System Dependency Diagram

TABLE 4.11. PLANT OPERATIONS INSPECTION GUIDANCE

Recognizing that the normal system lineup is important for any given standby safety system, the following human errors are specially identified as important to risk.

System	Failure	Discussion
Reactor Protection System	Switchover/Recovery Failure	Table 4.1A, Item 1
Auxiliary Feedwater System	Improper Alignment/Recovery Failure	Table 4.2A, Item 3
Emergency AC Power System	Improper Alignment/Recovery Failure	Table 4.4A, Item 1
Power Conversion System	Switchover/Recovery Failure	Table 4.5A, Item 1
High Pressure Safety Injection System	Switchover/Recovery Failure	Table 4.6A, Item 1
ECCS Pump Room Cooling System	Switchover/Recovery Failure	Table 4.7A, Item 1
	Post-Maintenance/Testing Lineup Failure	Table 4.7A, Item 4
Salt Water System	Feed and Bleed Control Failure	Table 4.8A, Item 1
	Switchover/Recovery Failure	Table 4.8A, Item 2
Chemical and Volume Control System	Switchover/Recovery Failure	Table 4.10A, Item 1

TABLE 4.12. SURVEILLANCE INSPECTION GUIDANCE

The listed components are the risk significant components for which proper surveillance should minimize failure.

System	Component	Discussion
Reactor Protection	Reactor Trip Breakers	Table 4.1A, Item 1
	CEDM Power Buses	Table 4.1A, Item 3
Auxiliary Feedwater System	AFW Pumps 11,12,13[(a)]	Table 4.2A, Item 1,2
	Valve AFW-103,-904	Table 4.2A, Item 3
	Valve AFW-161	Table 4.2A, Item 5
	Valves AFW-3987,-5903	Table 4.2A, Item 6
	Valves AFW-4522,-4532	Table 4.2A, Item 7
Emergency DC Power System	Inverters 11,12,13,14	Table 4.3A, Item 1
	Power Cables 11,12,13,14	Table 4.3A, Item 2
	Circuit Breaker 11,12,13,14	Table 4.3A, Item 3
	Power Cables 20,26,31,23	Table 4.3A, Item 4
	Batteries 11,12,13,14	Table 4.3A, Item 5
	Circuit Breakers 20,26,31,23	Table 4.3A, Item 6
Emergency AC Power System	DG-11,DG-12	Table 4.4A, Item 2,3
	Circuit Breakers 1103,1406	Table 4.4A, Item 4
Power Conversion System	MFW Pumps 11,12	Table 4.5A, Item 1
	Condensate Booster Pumps 11,12,13	Table 4.5A, Item 2
	Condensate MD Pumps 11,12,13	Table 4.5A, Item 3
High Pressure Safety Injection System	Valves SI-616,617,626,627,636, 637,646,647	Table 4.6A, Item 1
		Table 4.6A, Item 2
	Valves SI-659,-660	Table 4.6A, Item 3
	HPSI Pumps 11,12,13	Table 4.6A, Item 4
	Valves SI-654,-660	Table 4.6A, Item 5
	Valves SI-410,-4146	Table 4.6A, Item 6
	Valves SI-653,-655	Table 4.6A, Item 7
ECCS Pump Room Cooling System	Control Switches HS-5404,5404A	Table 4.7A, Item 1
	Air Cooling Units' Fans	Table 4.7A, Item 2
	Temperature Controller Elements TE-5404, 5405	Table 4.7A, Item 3
Salt Water System	Valve SWS-5210,	Table 4.8A, Item 1
	Valves SWS-5206,5208,5160,5162, 5163	Table 4.8A, Item 2
		Table 4.8A, Item 3
	Valves SWS-5150,5152,5153,5171. 5173,5174,5175	
	SWS Pumps 11,12,13	Table 4.8A, Item 4

TABLE 4.12 (contd)

System	Component	Discussion
Code Safety Valves System	Code Safety Valves	Table 4.9A, Item 1
Chemical and Control System	Charging Pumps 11,12,13	Table 4.10A, Item 2
	Valve CVC-514	Table 4.10A, Item 3
	BA Pumps 11,12	Table 4.10A, Item 4

(a) Although credit was not given for AFW Pump #12 in the PRA (due to its "locked out" position, proper surveillance should maintain readiness.

TABLE 4.13. MAINTENANCE INSPECTION GUIDANCE

The components listed here are significant to risk because of unavailability for maintenance or testing. The dominant contributors are usually frequency of maintenance and duration of maintenance, with some contribution due to improperly performed maintenance.

System	Component	Discussion
Reactor Protection System	Reactor Trip Breakers	Table 4.1A, Item 1
	CEDM Power Buses	Table 4.1A, Item 3
	Control Rod Element Assemblies	Table 4.1A, Item 2
Auxiliary Feedwater System	AFW Pump 13	Table 4.2A, Item 1,4
	AFW Pump 11	Table 4.2A, Item 2,4
	Valves AFW-103,-904	Table 4.2A, Item 3
	Valve AFW-161	Table 4.2A, Item 5
	Valves AFW-3987,-S903	Table 4.2A, Item 6
	Valves AFW-4522,-4532	Table 4.2A, Item 7
Emergency DC Power System	Inverters 11,12,13,14	Table 4.3A, Item 1
	Power Cables 11,12,13,14	Table 4.3A, Item 2
	Circuit Breakers 11,12,13,14	Table 4.3A, Item 3
	Power Cables 20,26,31,23	Table 4.3A, Item 4
	Batteries 11,12,13,14	Table 4.3A, Item 5
	Circuit Breakers 20,26,31,23	Table 4.3A, Item 6
Emergency AC Power System	DG-11,12	Table 4.4A, Item 2,3
	Breakers 1103,1406	Table 4.4A, Item 4
Power Conversion System	MFW Pumps 11,12	Table 4.5A, Item 1
	Condensate Booster Pumps 11,12,13	Table 4.5A, Item 2
	Condensate MD Pumps 11,12,13	Table 4.5A, Item 3
High Pressure Safety Injection System	Valves SI-616,-617,-626,-627,-636 -637,-646,-647	Table 4.6A, Item 2
	Valves SI-659,660	Table 4.6A, Item 3
	HPSI Pumps 11,12,13	Table 4.6A, Item 4
	Valves SI-654,-656	Table 4.6A, Item 5
	Valves SI-410,-4146	Table 4.6A, Item 6
	Valves SI-653,-655	Table 4.6A, Item 7
ECCS Pump Room Cooling System	Control Switches HS-5404,5404A	Table 4.7A, Item 1
	Air Cooling Units' Fans	Table 4.7A, Item 2
	Temperature Controller Elements TE-5404,5405	Table 4.7A, Item 3

TABLE 4.13 (contd)

System	Component	Discussion
Salt Water System	Valve SWS-5210	Table 4.8A, Item 1
	Valves 5206,5208,5160,5162,5163	Table 4.8A, Item 2
	Valves 5150,5152,5153,5171,5173, 5174,5175	Table 4.8A, Item 3
	Pumps 11,12,13	Table 4.8A, Item 4
Code Safety Valves System	Code Safety Valves	
Chemical and Volume Control System	Charging Pumps 11,12,13	Table 4.10A, Item 2
	Valve CVC-514	Table 4.10A, Item 4
	BA Pumps 11,12	Table 4.10A, Item 3
	Valve CVC-235	Table 4.10A, Item 5

TABLE 4.14. QUALITY ASSURANCE/ADMINISTRATIVE CONTROL INSPECTION GUIDANCE

The failures listed here are the ones which the QA/Administrative staff can affect. For example, QA should ensure that both regular and post-maintenance surveillance actually test for failure mode of concern for significant equipment. Also, in the case of equipment unavailabilities, administrative control should work to minimize the plant risk.

System	Component	Discussion
Reactor Protection System	Reactor Trip Breakers	Table 4.1A, Item 1
	CEDM Power Buses	Table 4.1A, Item 3
	Control Rod Element Assemblies	Table 4.1A, Item 2
Auxiliary Feedwater System	AFW Pump 13	Table 4.2A, Item 1,4
	AFW Pump 11	Table 4.2A, Item 2,4
	Valves AFW-103,-904	Table 4.2A, Item 3
	Valve AFW-161	Table 4.2A, Item 5
	Valves AFW-3987,-S903	Table 4.2A, Item 6
	Valves AFW-4522,-4532	Table 4.2A, Item 7
Emergency DC Power System	Inverters 11,12,13,14	Table 4.3A, Item 1
	Power Cables 11,12,13,14	Table 4.3A, Item 2
	Circuit Breakers 11,12,13,14	Table 4.3A, Item 3
	Power Cables 20,26,31,23	Table 4.3A, Item 4
	Batteries 11,12,13,14	Table 4.3A, Item 5
	Circuit Breakers 20,26,31,23	Table 4.3A, Item 6
Emergency AC Power System	DG-11,12	Table 4.4A, Item 2,3
	Breakers 1103A,1406B	Table 4.4A, Item 4
Power Conversion System	MFW Pumps 11,12	Table 4.5A, Item 1
	Condensate Booster Pumps 11,12,13	Table 4.5A, Item 2
	Condensate MD Pumps 11,12,13	Table 4.5A, Item 3
High Pressure Safety Injection	Sump Valves SI-4144,-4145	Table 4.6A, Item 1
	Valves SI-616,-617,-626,-627,-636 -637,-646,-647	Table 4.6A, Item 2
	Valves SI-659,660	Table 4.6A, Item 3
	HPSI Pumps 11,12,13	Table 4.6A, Item 4
	Valves SI-654,-656	Table 4.6A, Item 5
	Valves SI-410,-4146	Table 4.6A, Item 6
	Valves SI-653,-655	Table 4.6A, Item 7
ECCS Pump Room Cooling System	Control Switches HS-5404,5404A	Table 4.7A, Item 1

TABLE 4.14 (contd)

System	Component	Discussion
Salt Water System	Valve SWS-5210 Valves SWS-5206,5208,5160,5162, 5163 Pumps 11,12,13	Table 4.8A, Item 1 Table 4.8A, Item 2 Table 4.8A, Item 4
Code Safety Valves System	Code Safety Valves	
Chemical and Volume Control System	Charging Pumps 11,12,13 Valve CVC-514 BA Pumps 11,12	Table 4.10A, Item 2 Table 4.10A, Item 3 Table 4.10A, Item 4

5.0 CONTAINMENT PROTECTION SYSTEMS AT CC-1

In the event of a core melt accident, the public risk due to radiation release is minimized by the containment building. This analysis in this report has not addressed public risk, except through the probability of core melt, because the PRA which was analyzed is a "level 1" analysis and includes only a cursory analysis of release quantities and their effects.

If the containment functions as designed, the public risk resulting from a core melt will be small (e.g., TMI-2 accident), compared to the risk when containment fails with gross releases of radioactivity to the environment. During severe accidents, the containment is protected by two systems, the Containment Spray System (CSS) and the Containment Air Recirculation and Cooling System (CARC). They limit the temperature and pressure of steam and air in the containment, and reduce the airborne radioactivity by entraining it in water spray.

In the analysis of the Oconee-3 level 3 PRA (Gore, Vo, and Harris 1987), where systems were prioritized on the basis of public risk, the most risk-important systems were found to be the containment spray and the containment air cooling (e.g., CSS and CARC) systems. This is because event sequences leading to significant radioactivity releases almost always involved failure of one or both of these systems, which then led to failure of the containment.

In this section, we identify the components of the containment spray and air cooling systems which were found to be important in the Oconee PRA, and their dominant failure modes. It is reasonable to expect that these components and failure modes are important at Calvert Cliffs also. In each case, the modes identified contributed to 95% or more of the failure probability of the system. The importance of these systems and components to public risk should be kept in mind during inspection planning at Calvert Cliffs.

5.1 CONTAINMENT SPRAY SYSTEM

Conditions Leading to Failure in Oconee PRA

1. Human Error - System operation inhibited or failure to restore valves or pump switchgear after testing.

   Operator failure to restore correct system lineup for automatic pump start and flow to spray nozzles is the most important failure in the Oconee PRA.

2. Spray Pump Failure to Start or Run

   Pump hardware or control circuit failures are important at Oconee, as are human errors in the associated procedures for surveillance or maintenance.

3. Failure of Motor-Operated Discharge Valve to Open

   (Calvert Cliffs Valves CSS-4150 and CSS-4151.) The dominant failure mode at Oconee is hardware failure, with human failure to manually actuate these valves when necessary being a contributing mode.

4. Pump Trains Unavailable Due to Maintenance and Testing

   Both scheduled and unscheduled activities are included. Minimization of this time and conformance to Technical Specifications requirements are important at Oconee.

5. Pump Suction Valves Fail to Open or Check Valves Stick Closed

   (Calvert Cliffs Valves SIS-4142 and SIS-4143.) The dominant Oconee failure modes are human error, electrical failure, or hardware failures. Lineup for standby operation and proper surveillance and maintenance are important.

   Support system dependencies for the CSS are shown in Figures 5.1 and 5.2.

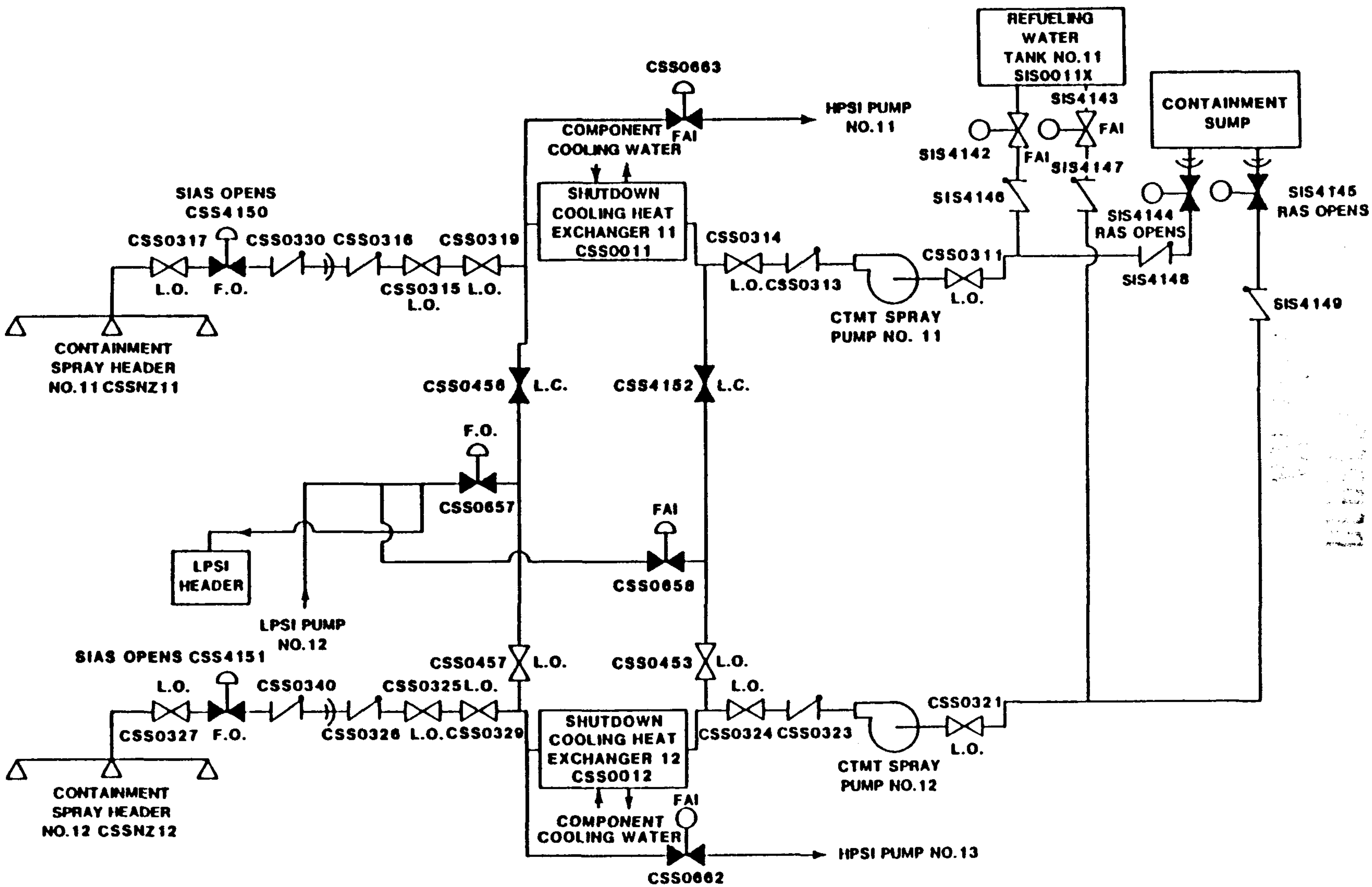


FIGURE 5.1. Simplified System Drawing of CSS SDHX

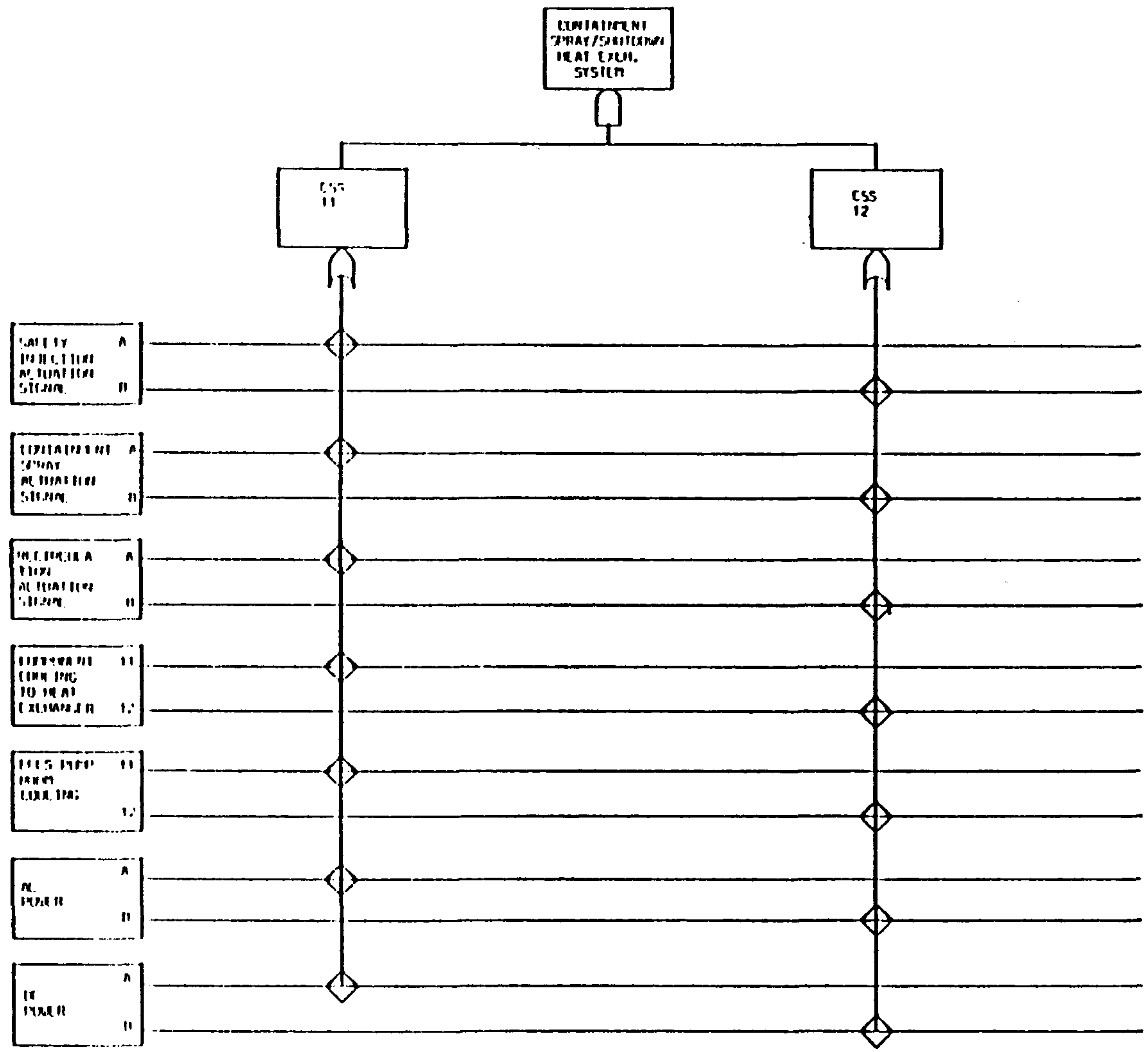


FIGURE 5.2. CSS/SDHX Support System Dependency Diagram

5.2 CONTAINMENT AIR RECIRCULATION AND COOLING SYSTEM

Conditions that Lead to Failure

1. Operating Fans Fail to Run and Non-Operating Fan Fails to Start and Run
   Fan failure due to hardware failure is the dominant system failure mode at Oconee.

2. Operating Fans Fail to Run and Non-Operating Fan in Maintenance

   At Oconee, system failure due to fan maintenance unavailability in combination with hardware failures is a significant system failure mode.

3. Motor-Operated Damper to Common Duct Header Fails to Open

   Damper misoperation is a significant failure mode at Oconee.

4. Dropout Plates Fail to Drop

   Failure of fusible dropout plates to drop and open ductwork bypasses in a post-LOCA environment is a significant Oconee failure mode.

5. Start Switches Improperly Positioned

   Human error in positioning control switches, preventing proper automatic system operation, is also an important failure mode at Oconee.

   Support system dependencies for the CARCS are shown in Figures 5.3 and 5.4.

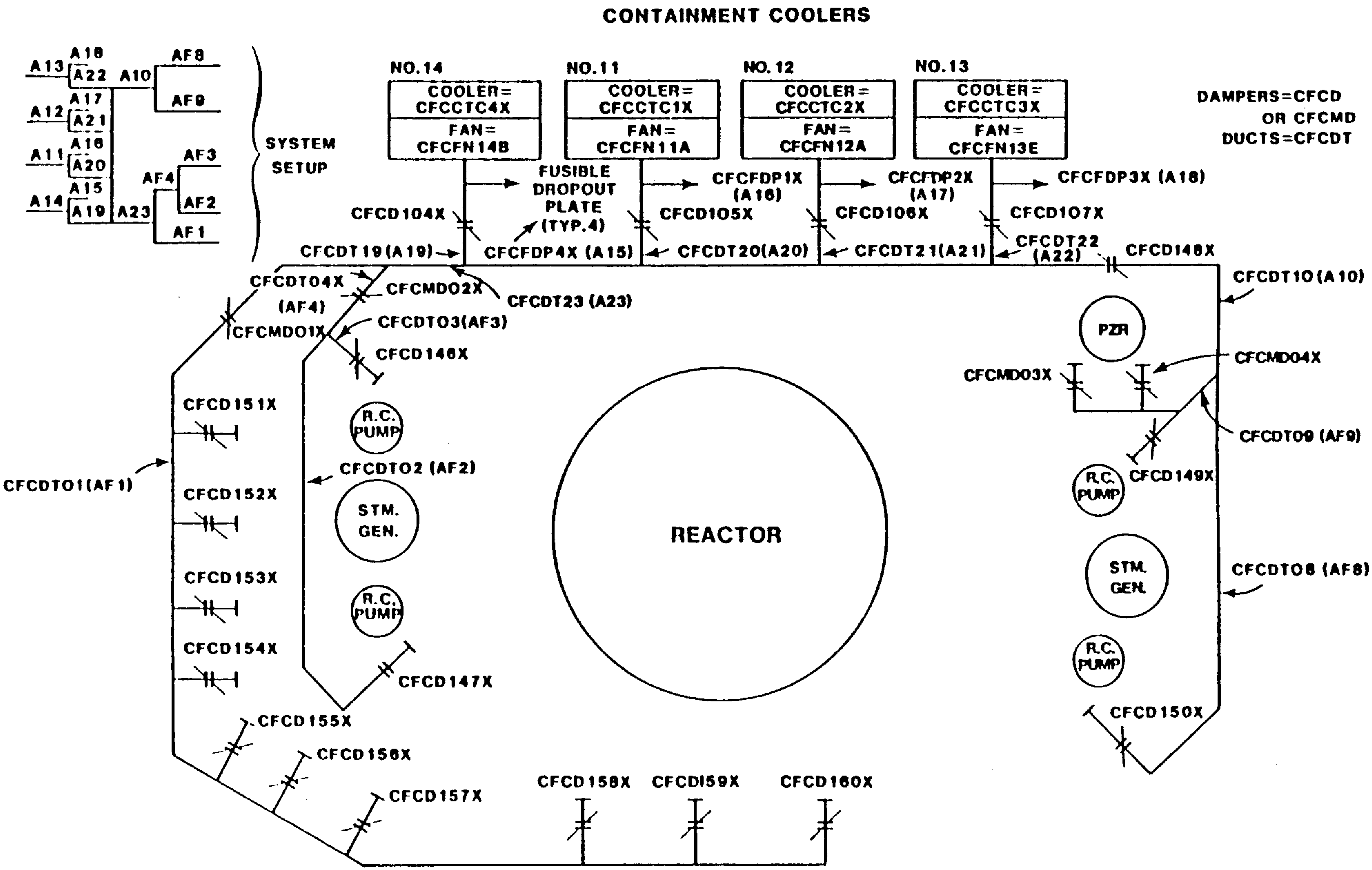
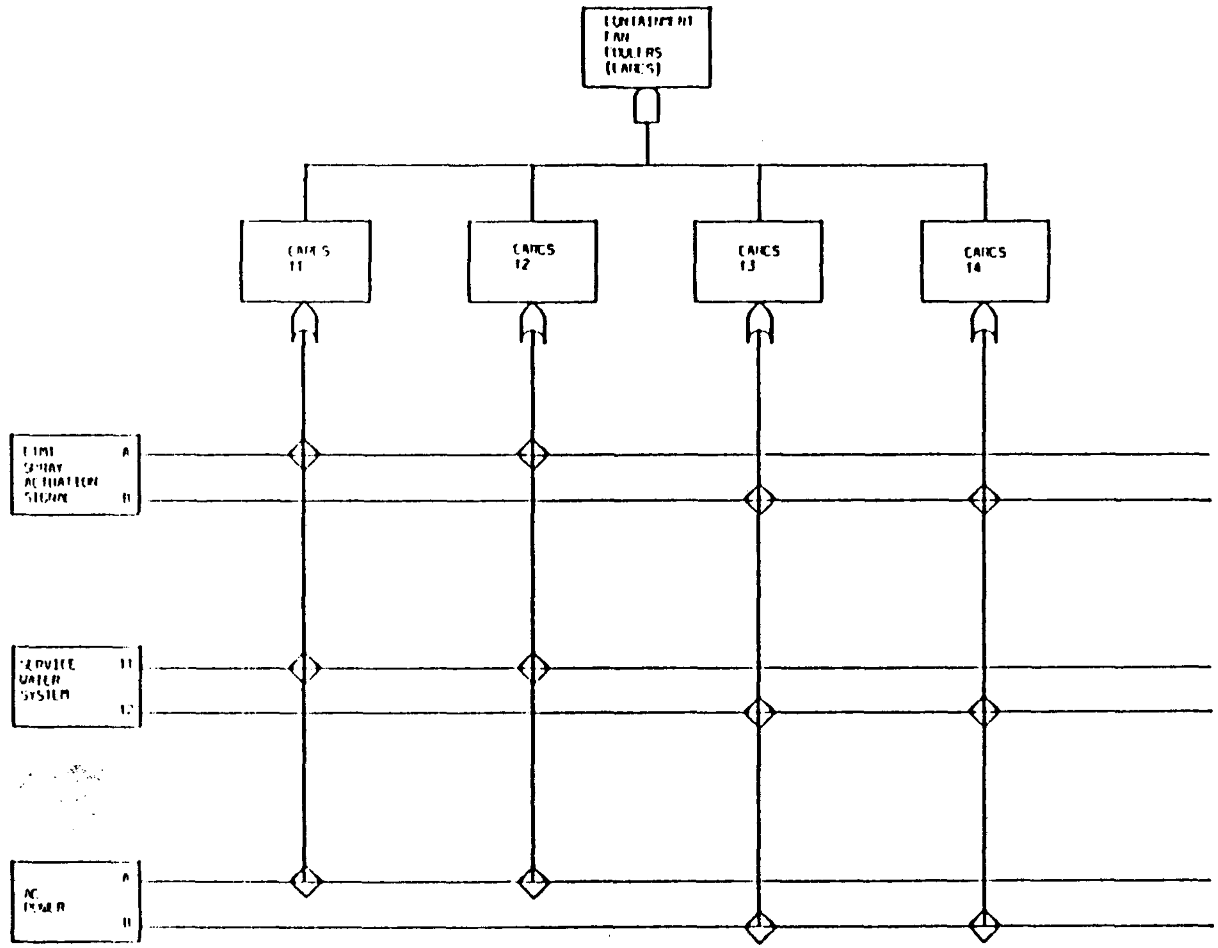


FIGURE 5.3. Simplified System Drawing of CARCS



<u>FIGURE 5.4.</u> CARCS Support Dependency Diagram

TABLE OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater System
ATWS	Anticipated Transient Without Scram
BA	Boric acid
BAST	Boric Acid Storage Tank
BWR	Boiling Water Reactor
CARC	Containment Air Recirculation and Cooling
CC	
CC-1	Calvert Cliffs 1 nuclear plant
CCW	Component Cooling Water
CCW	Component Cooling Water
CCW	Component Cooling Water system
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CSS	Containment Spray System
CSS/SDHX	Containment Spray System Shutdown Heat Exchangers
CTS	Condensate Storage Tank
CVCS	Chemical Volume Control System
CVCS,CVC	Chemical and Volume Control System
DC	Direct Current
DG	Diesel Generator
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
F-V	Fussell-Vesely (Importance)
HPSI	High-pressure Safety Injection
HPSI/R	High-pressure Safety Injection/Recirculation
HPSR	High-pressure Safety Recirculation
HS	Handswitch
HX	Heat Exchanger
IBM-PC	IBM Personal Computer
IE	Inspection and Enforcement
INEL	Idaho National Engineering Laboratory
IRRAS	Integrated Reliability and Risk Analysis (computer code)
KV	Kilo-volt
LOCA	Loss of Coolant Accident
LPSI	Low-pressure Safety Injection
MD	Motor-driven
MFW	Main Feedwater

MFWCS	Main Feedwater and Condensate System
MOV	Motor-operated valve
MS	Main Steam
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCS	Power Conversion System
PNL	Pacific Northwest Laboratory
PORV	Pressure-operated Relief Valve
PP	
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAS	
RCS	Reactor Coolant System
RPS	Reactor Protection System
RWT	Refueling Water Tank
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SRV	Safety Relief Valve
SRW	Service Water
SRWS	Service Water System
SWS	Salt Water System
TE	Temperature Controller Element
TMI-2	Three Mile Island Unit 2 nuclear plant
USNRC	Nuclear Regulatory Commission
V	Volt(s)

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11. ABSTRACT *(200 words or less)*

The level one probabilistic risk assessment (PRA) for Calvert Cliffs Unit 1 (CC-1) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency, and to identify the primary failure modes of these components. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement Manual. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance are: Reactor Protection, Auxiliary Feedwater, DC Power, AC Power, Power Conversion, High Pressure Injection, Room Cooling, Salt Water, Safety Relief Valves, and Chemical and Volume Control Systems. This ranking is based on the Fussel-Vesely measure of risk importance, i.e., the fraction of the total core melt frequency which involves failures of the system of interest.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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