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Risk Evaluation for a B&W Pressurized Water Reactor, Effects of Fire Protection System Actuation on Safety-Related Equipment

Evaluation of Generic Issue 57

Prepared by
J. Lambright, J. Lynch, S. Ross, E. Klamerus, S. Daniel

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ABSTRACT

Nuclear power plants have experienced inadvertent actuations of fire protection systems (FPS) under conditions for which these systems were not intended to actuate. They have also experienced advertent actuations with the presence of a fire. These actuations have often damaged plant equipment.

A review of the impact of past occurrences of both types of such events on nuclear power plant safety has been performed. Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. These scenarios ranged from inadvertent actuation caused by human error to hardware failure and includes seismic root causes and seismic/fire interaction. A quantification of these thirteen scenarios, where applicable, was performed on a Babcock and Wilcox Pressurized Water Reactor (lowered loop design). This report estimates the contribution of FPS actuations to core damage frequency and to risk.

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EXECUTIVE SUMMARY

In recent years, fire protection systems (FPS) in nuclear power plants have actuated at times and under conditions for which they were not intended to actuate, as well as actuating in the presence of a fire, and have often affected and even damaged plant equipment. To quantify the risk due to such events, a study was performed which involved (a) a review of pertinent Licensee Event Reports of industry experience with FPS actuations, (b) a review of Navy experience with FPS actuations, and (c) a quantification of potential scenarios for selected commercial nuclear power plants including a set of generically applicable scenarios.

In the quantification portion of the study, thirteen different root causes were identified that could result in FPS actuations. A set of criteria were developed for identifying such accident scenarios that could lead to core damage. These criteria can be applied to the probabilistic risk assessment (PRA) vital area analysis for any particular plant in question to identify those accident sequences and cut sets that could lead to core damage (assuming the FPS actuation damages critical equipment in the fire zone affected).

Inasmuch as these scenarios are plant-specific in regard to plant layout and types of FPSs present, three plants were selected for the quantification. The criteria developed were applied to two commercial pressurized water reactor (PWRs) and one commercial boiling water reactor (BWR). These plants were selected because each had a detailed PRA and supporting analyses available. This report presents the application of the methodology to a lowered-loop Babcock and Wilcox PWR.

Using the complete set of accident sequences developed in a previous PRA for the plant, a full set of scenarios based on FPS actuations was analyzed. For each accident sequence identified, values for the various parameters involved were chosen, and an estimate of the impact on core damage and risk due to FPS actuation was made. Although an effort was made to use parameter estimates from existing data bases where available, some simplifying assumptions were required due to lack of data.

The risk calculations were performed employing a methodology similar to that used in WASH-1400 and USI-45. An uncertainty analysis was performed for the core damage frequency and risk calculations.

The results of the quantification found a total mean contribution to annual core damage frequency of 5.6E-5/yr and total dose of 100 Person-REM for the B&W PWR analyzed.

1.0 INTRODUCTION

1.1 Scope

Experience in recent years has shown that FPSs in nuclear power plants have actuated at times and under conditions for which they were not intended to actuate, as well as when intended in the presence of a fire. Since these FPSs typically are located near the critical equipment they are designed to protect, these actuations have often affected and even caused damage to this critical plant equipment. On some occasions, the damage has been to safety-related equipment; that is, equipment required to ensure the capability to safely shutdown the plant. On other occasions, the damage has been to equipment required for the normal operation of the plant and the reactor was subsequently shutdown. As a consequence, the actuation of fire protection systems was judged by the USNRC to represent a potentially important safety issue requiring further study.

In the recently completed Fire Risk Scoping Study (Ref. 1.1), the inadvertent actuation of fire protection systems in commercial United States nuclear power plants was briefly reviewed. Seventy-one events resulting in submission of a Licensee Event Report (LER) were identified during the period from April 1, 1980 to July 14, 1987. The average frequency of occurrence of these inadvertent actuation events was found to be approximately 10 per year for the industry as a whole.

The Fire Risk Scoping Study was limited in scope and did not attempt to quantify the attendant contribution to core damage frequency (CDF) resulting from the inadvertent actuation of FPS's, primarily because the impact of inadvertent fire protection system actuations was found to be very plant specific. It was concluded that such events could significantly impact the risk at a specific plant only if multiple safety systems could be affected by the inadvertent fire protection system actuation event.

As a follow-on to the Fire Risk Scoping Study, a preliminary study including a scoping quantification of risk due to inadvertent FPS actuation was performed (Ref. 1.2). This study quantified the core damage frequency and risk at one generic PWR. This analysis indicated that the increase in core damage frequency due to inadvertent FPS actuations could range from 10-5 to 10-4 per reactor year.

The current study, of which this report is part, is entitled "Effects of Fire Protection System Actuation on Safety-Related Equipment" was begun in 1990. In this study, six main potential categories of root causes of inadvertent and advertent actuations of fire protection systems have been identified, as shown in Table 1.1. For the general cases of random and seismically induced actuations, several potential root causes are also shown.

Table 1.1

Causes of Potential FPS Actuation

A. Random causes of inadvertent actuation

   Human error (Root Cause 4)

   Hardware failure (Root Cause 6)

   Unknown (Root Cause 13)

B. Actuation induced by fire or by steam pipe break in an adjacent area and smoke/steam spread

   Fire in adjacent zone causing FPS actuation (Root Cause 1)

   Fire-induced FPS actuation (due to fire in adjacent zone) preventing random failure recovery action (Root Cause 2)

   Fire-induced FPS actuation (due to fire in adjacent zone) preventing access for manual fire suppression (Root Cause 3)

   FPS actuation caused by steam release (Root Cause 5)

C. Seismic induced inadvertent actuation

   Dust actuating smoke detectors (Root Cause 7)

   Failure of FPS (e.g., failure of wet pipes, sprinkler heads, etc.) (Root Cause 9)

   Actuation caused by FPS control system relay chatter (Root Cause 8)

D. Seismic induced failure of the FPS, diverting suppression agent from an area where a fire is present (Root Cause 12)

E. Fire external to plant (smoke via ventilation system) (Root Cause 10)

F. Fire present where the FPS is located (Root Cause 11)

The objective of this study was to provide a probabilistic basis on which to evaluate the impact on plant core damage frequency and risk of FPS actuations. This objective was accomplished by first reviewing past events involving fire protection system actuations. The actuations were then categorized in order to draw some useful conclusions about the causes and effects of these actuations. A quantification of the impacts of such events including sensitivity and uncertainty studies was performed both in terms of reduction in core damage frequency and risk for the scenarios identified.

1.2 Methodology

Chapter 3 of NUREG/CR-5580[a] presents an overall methodology that is used to evaluate the effects of FPS actuations on nuclear power plant risk. The objective of the analysis presented in this report is to apply this general methodology to one of a representative set of nuclear plants. In this case, the plant selected is a B&W Pressurized Water Reactor (lowered loop). Using data from industry experience and parametric values used in prior applicable PRA studies, a quantitative assessment of the incremental contribution to core damage frequency due to FPS actuations was performed.

The analysis of the thirteen root causes introduced in NUREG/CR-5580[a] is being applied on a site-specific basis. The actual site being studied is unimportant and will not be named. As the safety significance of FPS actuations is highly plant-specific and is dependent on system interdependencies derived from plant event tree and fault tree models, it follows that those models available for the specific plant in question must be used in the analysis. In this case, the system models developed as part of the USI-45 study (Ref. 1.3) augmented by site visits were used as the basis for quantification in this report.

1.3 Organization of the Report

A description of the plant systems and general plant characteristics is provided in Chapter 2. The system descriptions include simplified schematics which depict major system components.

The base case analysis (best estimate) of core damage frequency due to FPS actuations is described in Chapter 3[a]. This analysis addresses all of the root causes that apply to this nuclear power plant. This chapter also contains a description of where vital equipment is located throughout the plant, plant fire protection system locations, and an application of the methodology including results in terms of core damage frequency by root cause and by fire zone.

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a. J. A. Lambright et al, Evaluation of Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment. Root Cause Development and Summary," NUREG/CR-5580, SAND90-1507, Sandia National National Laboratories, Albuquerque, NM, September, 1992.

Chapter 4 describes the sensitivity analyses performed and the overall effect on the base case results. These studies are very plant specific, but the issues considered would likely apply to any "typical" B&W PWR. In Chapter 5 a description of the methodology to compute risk is given as are the results of the analysis.

1.4 References

1.1 J. A. Lambright et al, Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk Including Previously Unaddressed Issues, NUREG/CR-5088, SAND88-0177, Sandia National Laboratories, Albuquerque, NM, November 1988.

1.2 J. C. Romig, et al, Scoping Study of the Potential Impacts of Inadvertent Fire Suppression System Actuations in Commercial Nuclear Power Plants (Letter Report), Sandia National Laboratories, Albuquerque, NM, May 1990.

1.3 W. R. Cramond, et al, Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor-Case Study, NUREG/CR-4713, SAND86-1832, Sandia National Laboratories, Albuquerque, NM, March 1987.

2.0 PLANT DESCRIPTION

2.1 Plant Site and General Characteristics

The plant site is adjacent to a reservoir on a major river. Physically, the site is centrally situated on a "peninsula," about two miles wide and two miles long, that extends into the reservoir. On three sides, the site is surrounded by reservoir water; the shortest stretch of which is approximately one mile. Ground surface in the immediate vicinity of the plant site is predominantly meadow. Cooling water for the unit is drawn from and returned to the 36,000-acre reservoir.

The unit is one of two units on the site. Unit One, the unit under study, is a Babcock and Wilcox lowered loop closed cycle pressurized water nuclear steam supply system. Unit One is designed to operate at core power levels up to 2568 MWt with a net output of 836 Mwe. It uses chemical shims and control rods for reactivity control and generates steam with a small amount of superheat in once-through steam generators. The reactor building is a fully continuous reinforced concrete structure in the shape of a cylinder on a flat foundation slab with a shallow domed roof. The cylindrical portion is stressed by a post tensioning system consisting of horizontal and vertical tendons. The foundation slab is reinforced with high-strength reinforcing steel. A welded steel liner is attached to the inside face of the concrete shell.

The reactor building completely encloses the entire reactor and reactor coolant system including the steam generators and portions of the engineered safeguards systems. The auxiliary building houses the safety related systems including the high and low pressure safety injection, reactor building spray system, reactor building cooling system, emergency feedwater system, emergency diesel generators, electrical switchgear rooms, battery rooms, and the control room. The turbine building encloses the unit one and unit two power conversion systems and related equipment. The condensate storage tanks, borated water storage tanks, and diesel fuel oil tanks are located outside, west of the auxiliary building.

2.2 Description of Plant Systems

2.2.1 Introduction

This section discusses the system descriptions and system models of the major frontline and support systems identified as important to safety. The models used in this analysis are based on those developed as a part of The Interim Reliability Evaluation Program (Ref. 2.1) and refined in The Shutdown Decay Heat Removal Analysis (Ref. 2.2). The discussion of each of the systems that follow includes:

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

b. Safety-related success criteria that were applied to the system.

c. Interfaces and safety actuation provisions between the frontline systems and the support systems.

2.2.2 Emergency Feedwater System (EFWS)

The purpose of the EFWS is to backup the main feedwater system (MFWS) in removing post-shutdown decay heat from the reactor coolant system via the steam generators. During normal shutdowns the MFWS is throttled down to a level appropriate for the level of decay heat and the EFWS is not utilized. However, if the plant shutdown is caused by a loss of the MFWS or the reactor coolant pumps, or if the MFWS is lost subsequent to the plant shutdown, then the EFWS is put into operation.

The EFWS consists of two interconnected trains, each capable of supplying emergency feedwater (EFW) to either or both steam generators (SGs) from either of two water sources under automatic or manual initiation and control. A simplified piping diagram is shown in Figure 2.1. The system pumps take suction from either the condensate storage tank (CST) or from the service water system and discharge to the SGs. In the flow path between the EFW pumps and the SGs there are isolation valves, check valves, control valves, flow instrumentation, and pressure instrumentation to control the flow of EFW to the SGs. The EFW system is designed to provide a minimum of 500 gpm of EFW to the SGs at 1050 psig within 50 seconds of a system initiation signal.

Train A contains a motor-driven pump while the train B pump is turbine-driven. Except for electromotive and control power and actuation signals, the pumps, pump motor, and turbine are self-contained, without support system dependencies. If AC power is not available, the B train can still provide complete system function relying solely on DC power.

It is assumed that the EFW pumps would fail on activation if the CST source was unavailable before the operator could realign the suctions of the pump to service water. The success criteria of the system is to remove reactor coolant system decay heat from one of two steam generators. Either pump can supply sufficient feedwater for this purpose to either steam generator.

2.2.3 High Pressure Injection And Recirculation Systems (HPIS and HPRS)

The high pressure (HP) system is utilized during those Loss of Coolant Accidents (LOCAs) where the reactor coolant pressure remains high (i.e., above about 1500 psig, where the low pressure (LP) pumps have insufficient discharge head to inject water into the system). This condition will typically exist during small breaks and during the early stages of medium breaks. The high pressure injection system (HPIS) is, like most other engineered safeguards (ES) systems, actuated upon

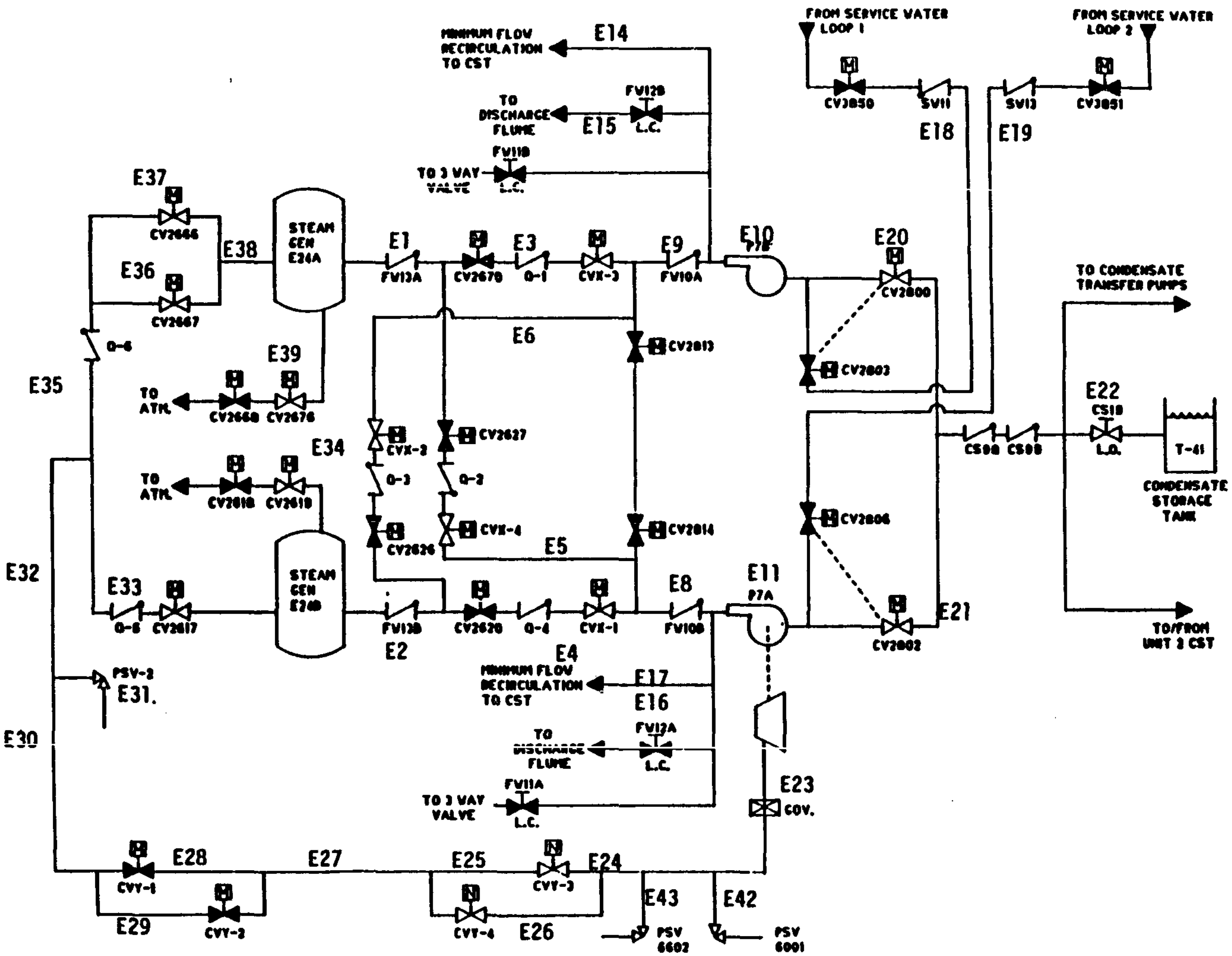

Figure 2.1. Pipe Segments for the Emergency Feedwater System.

receiving an engineered safeguards actuation system (ESAS) signal which signifies either RCS pressure decreasing below 1500 psig, or reactor building pressure increasing to 4 psig. During the injection mode, the HP system draws borated water from the borated water storage tank (BWST) via a common tank outlet header shared with the Low Pressure Injection System (LPIS) and reactor building spray (RBS) system. When the HPIS/HPRS is switched to the recirculation mode (which requires manual operator actions) the water is drawn from the reactor building sump by the LP pumps, then through the decay heat coolers, and then to the HP pump suction. The water is then injected into the reactor vessel. Figures 2.2 and 2.3 are a simplified schematic of the HP system (the discharges of the decay heat coolers to the HP pumps is through pipe segments DH7A and DH7B) with valve positions shown prior to injection. Room and lube oil cooling pipe segments are shown in Figure 2.4.

The HP System is a two train, three pump system which injects water into the reactor pressure vessel via four injection headers. There is one injection header for each cold leg of the RCS. The injection headers are cross connected such that each pump has an open flow path to all four RCS cold legs. During normal operation, one of the three HP pumps is kept running in order to provide normal makeup to the reactor coolant system, and cooling/lubrication for the RCS pump seals. Upon receiving an ESAS signal a second HP pump is started and the running HP pump is realigned from normal makeup to HPI. The realignment is accomplished by opening the suction of the HP pumps to the BWST, isolating the normal makeup (MU) tank, and by additionally realigning the discharge from the normal MU to the HPI piping.

Although there are three pumps in the HP system, only two can be run at any one time since there are only two electrical power source busses. Each bus can power only one pump. The A and C pumps are permanently connected to different electrical busses. Pump B can be selected (swing) to either of the two busses. During normal operation, pump B is aligned to the bus powering the normal MU pump. The like-aligned pump is then configured as an automatic backup to the operating MU pump and the opposite-aligned pump is the ES pump.

Another function of the HPIS is to provide cooling water to the reactor coolant pump seals. The service water system provides a backup method of seal cooling through the intermediate cooling water system (ICWS). The ICWS, however, is isolated upon an ESAS signal.

The success criteria for the system are dependent on the initiating event. For example, for small LOCA with feed and bleed, only one pump is necessary. An important system assumption is that the failure of the decay heat coolers to cool the recirculation water during HPR will fail the HP pumps as they are designed to pump water which is below 200^{o}F.

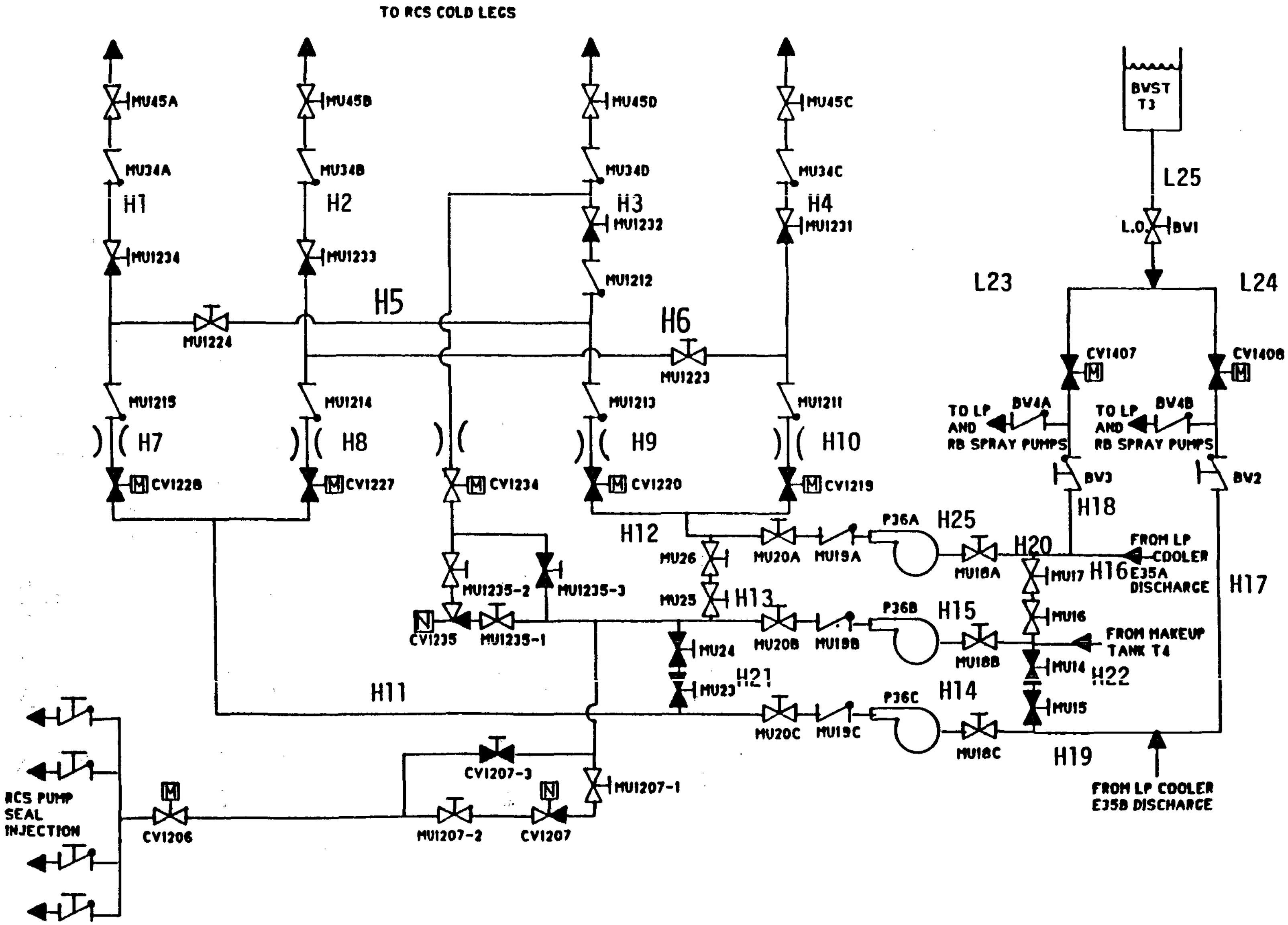

Figure 2.2. Pipe Segments of High Pressure Injection System.

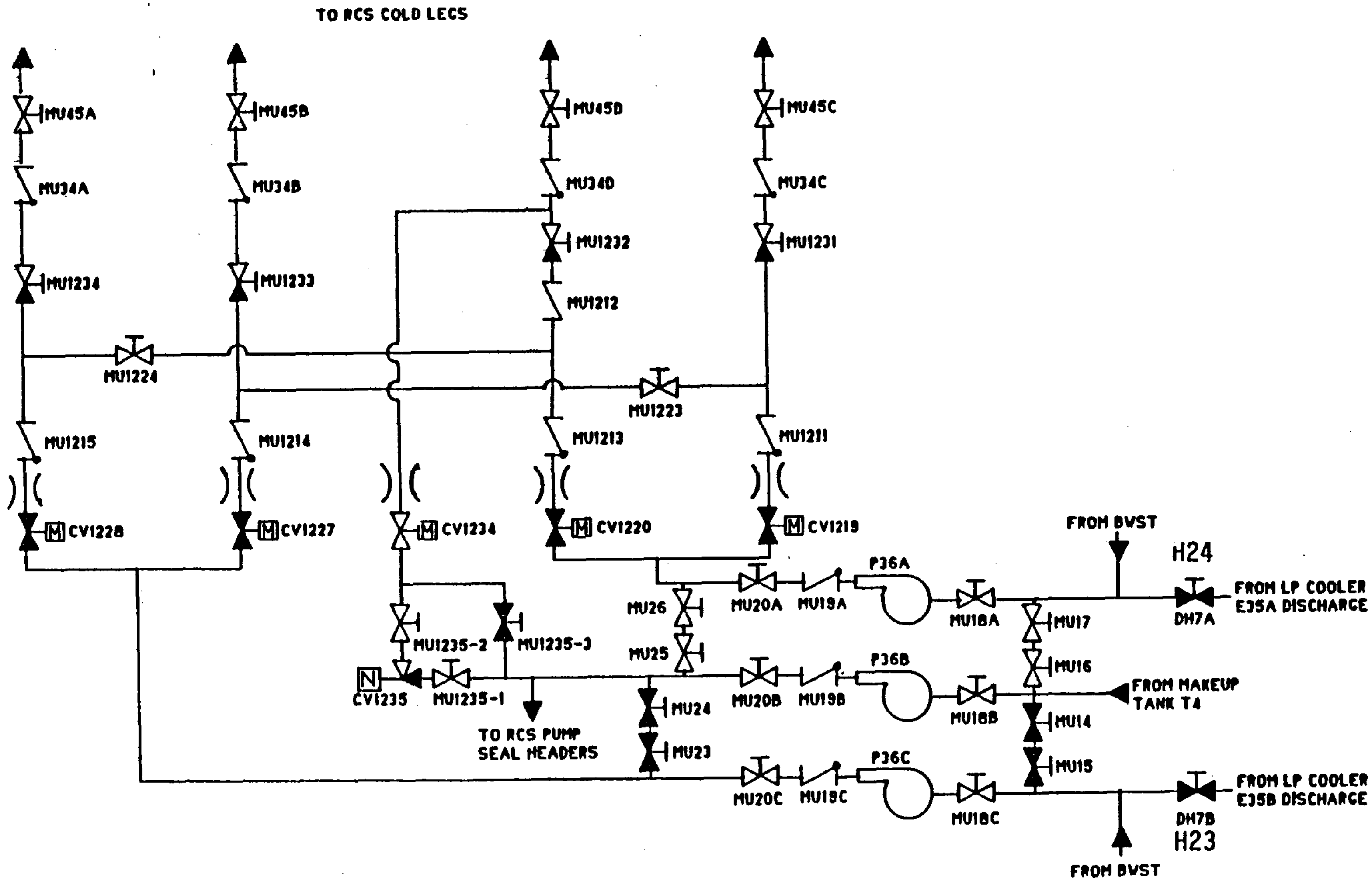

Figure 2.3. Pipe Segments of High Pressure Recirculation System.

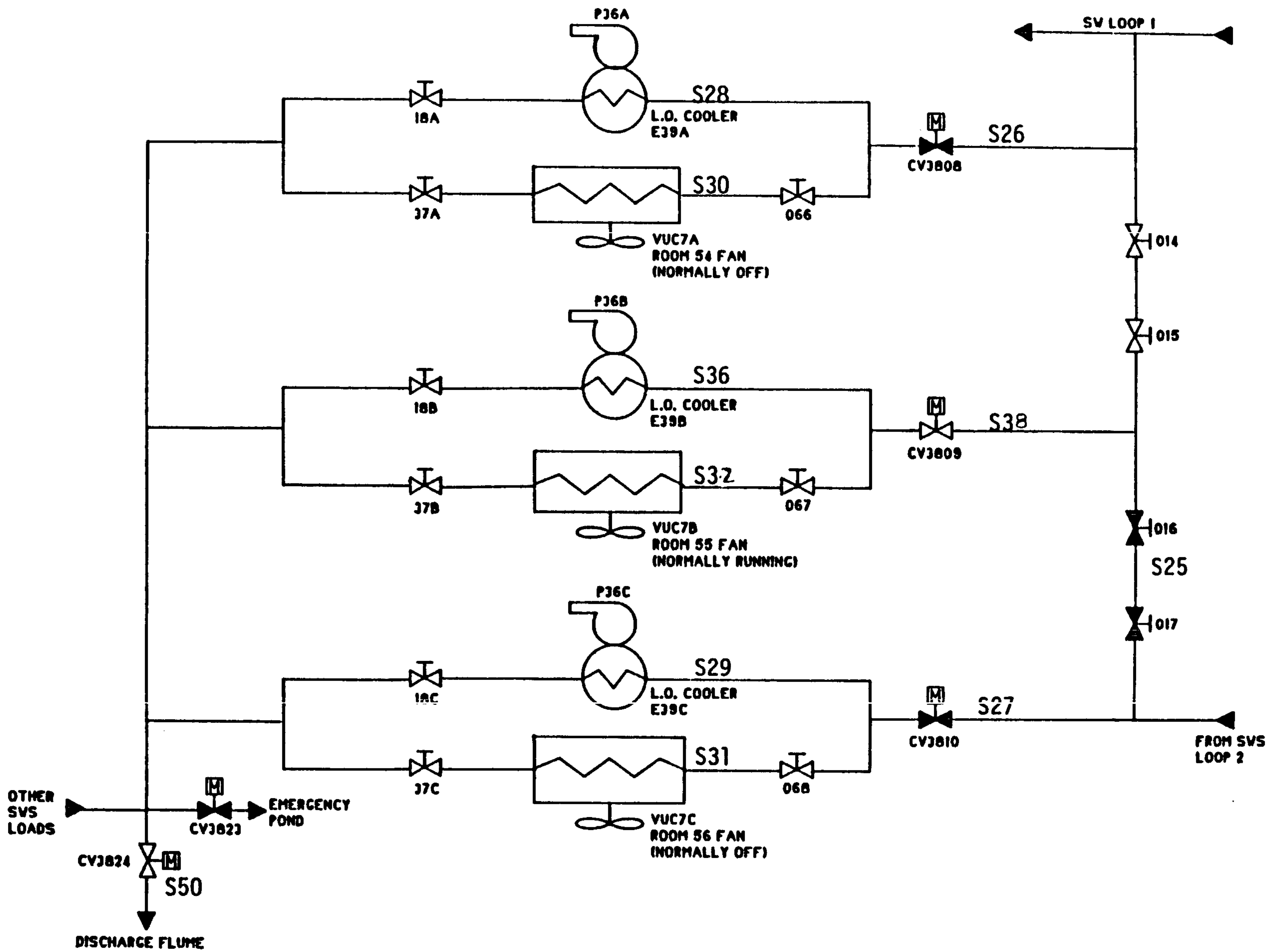

Figure 2.4. Pipe Segments of High Pressure Pumps Lube Oil and Room Cooling.

2.2.4 Low Pressure Injection And Recirculation Systems (LPI/LPR System)

The LPI/LPR system serves a number of functions during both accident conditions and normal operations. The system provides emergency core cooling (ECC) during a LOCA. During plant outages and during refueling, the system provides for decay heat removal and for filling and draining of the fuel transfer canal. During accident conditions, the system is actuated by an ESAS signal (RCS pressure decreasing below 1500 psig or reactor building pressure rising above 4 psig). However, it is not until the reactor coolant system (RCS) pressure drops to about 150 psig (as occurs during a large break LOCA) that the system is able to overcome the RCS pressure and inject borated water into the RCS. In addition to LP recirculation from the reactor sump, the system is utilized during the recirculation phase of a small break LOCA (i.e., RCS pressure remains high) in the DHRS mode of recirculation. During HPRS recirculation, the system is required to feed cooled (via the decay heatcoolers) water from the RB sump to the HP system for injection into the pressure vessel. The LPRS, in conjunction with the spray recirculation, also provides long-term cooling to the containment. Simplified drawings of the system are presented in Figures 2.5 and 2.6. The system dependencies of the low pressure system are shown in Figures 2.7, 2.8, and 2.9.

The LPI/LPR System consists of two independent trains which draw water via a common header from the BWST. This common header also supplies water to the HP and RBS systems. Each train of the LP system consists of a LP pump, a decay heat removal cooler, piping and valves. The water is injected directly into the pressure vessel through two LP injection headers which are cross-connected. The injection headers contain flow restrictors such that each train injects 50% of its flow through each injection header. These injection headers are also used by the core flood system, with the tanks being isolated from the pressure vessel by two check valves, one in the LP injection line, and one in the core flooding header.

The system is realigned for recirculation manually by the operators. This is done by opening the motor operated isolation valves, allowing flow from the RB sump to the LPI pump suction. Flow is then manually aligned from the DH cooler discharge to the HP pump suction during HPR.

The success criteria of the LP injection system requires one of two pumps. During the recirculation phase of the accident sequences, one of two pumps is also required.

2.2.5 Reactor Building Spray Injection and Recirculation Systems (RBSS/RBSRS)

For the purpose of this analysis, the Reactor Building Spray System (RBSS) performs two functions. These are: (a) to reduce the post-accident reactor building pressure to nearly atmospheric pressure during the injection phase and act in conjunction with the low pressure

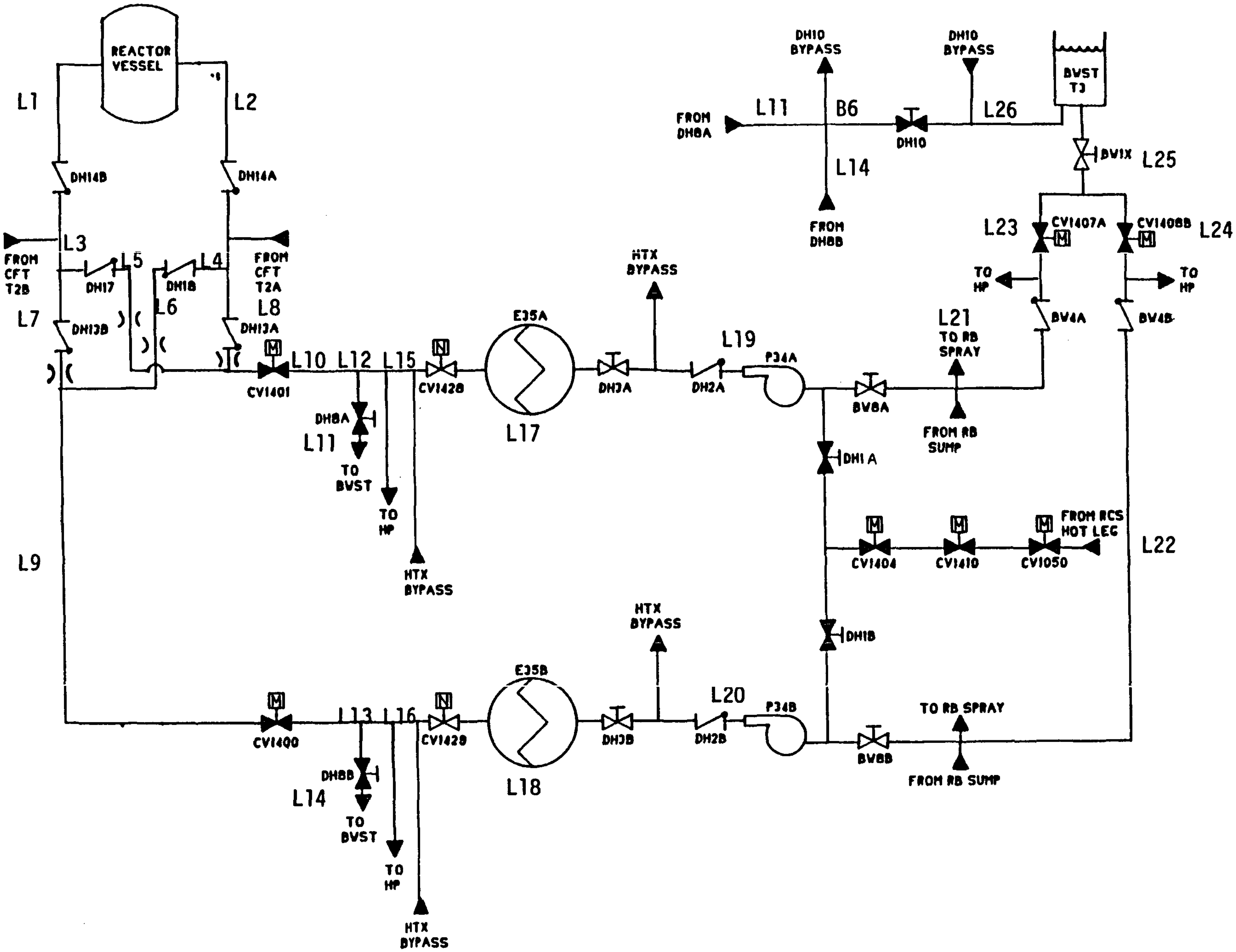

Figure 2.5. Pipe Segments for the Low Pressure Injection System.

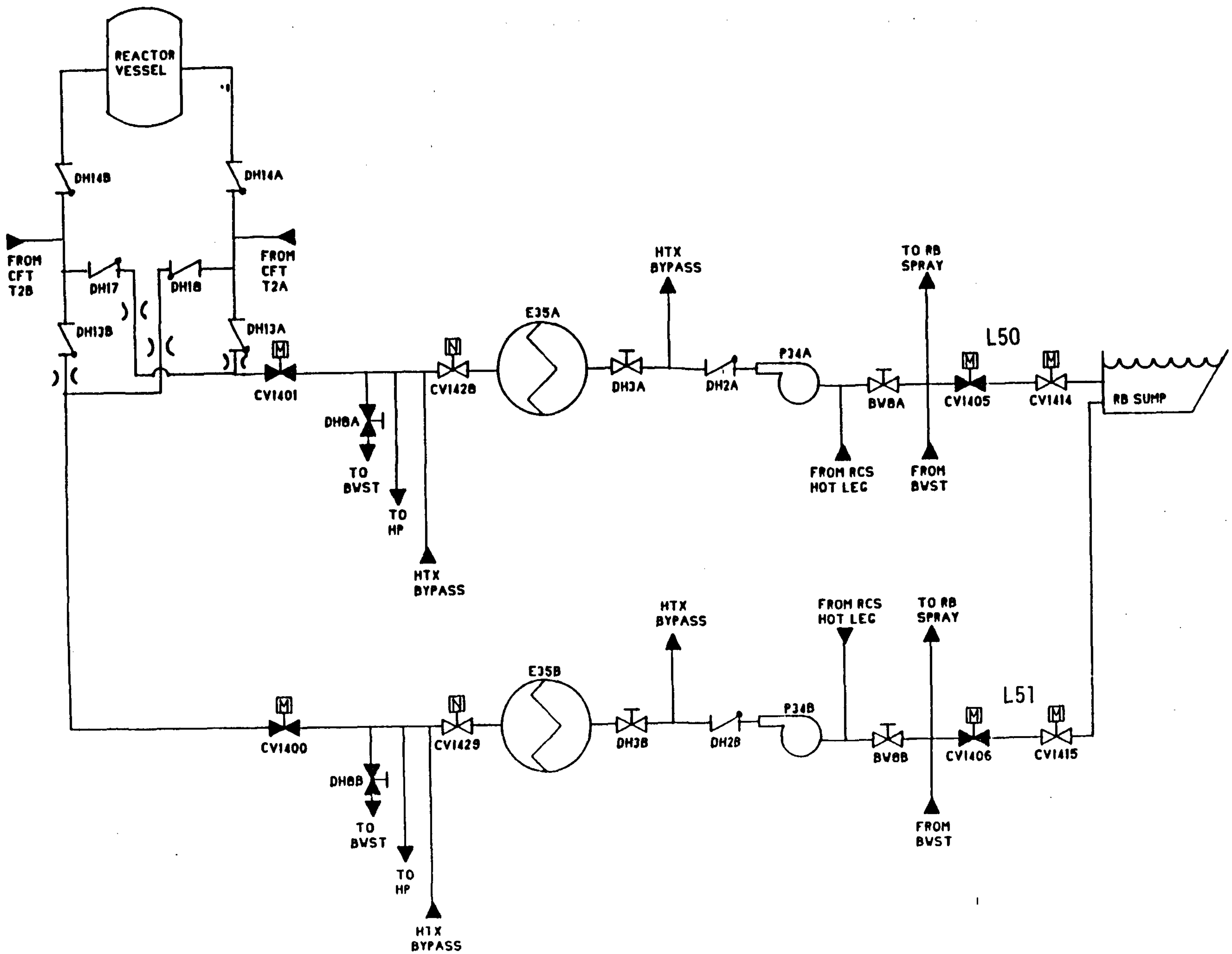

Figure 2.6. Pipe Segments for the Low Pressure Recirculation System.

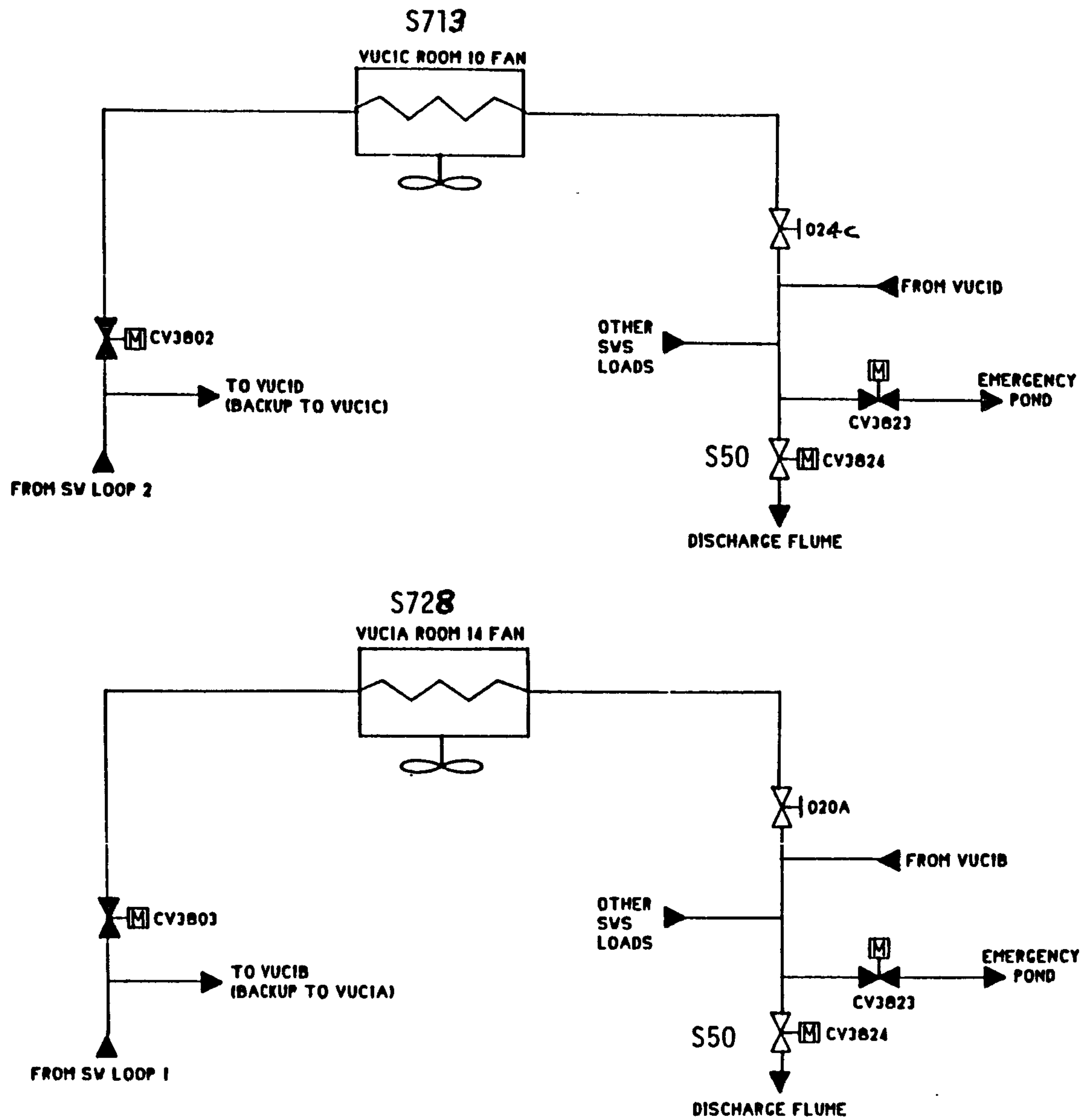

Figure 2.7. Pipe Segments for Low Pressure Pump Room Coolers.

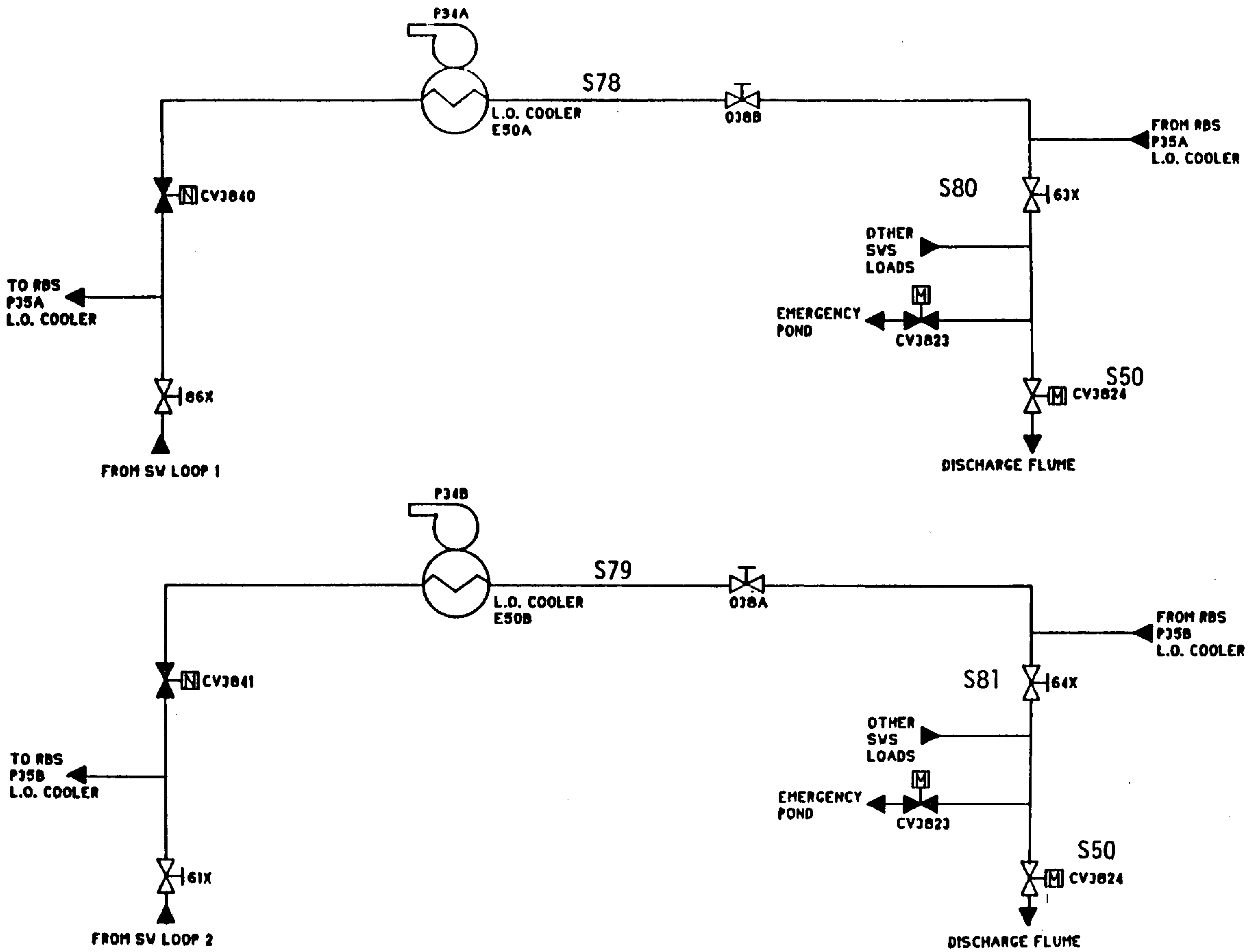

Figure 2.8. Pipe Segments for the Low Pressure Pump Lube Oil Coolers.

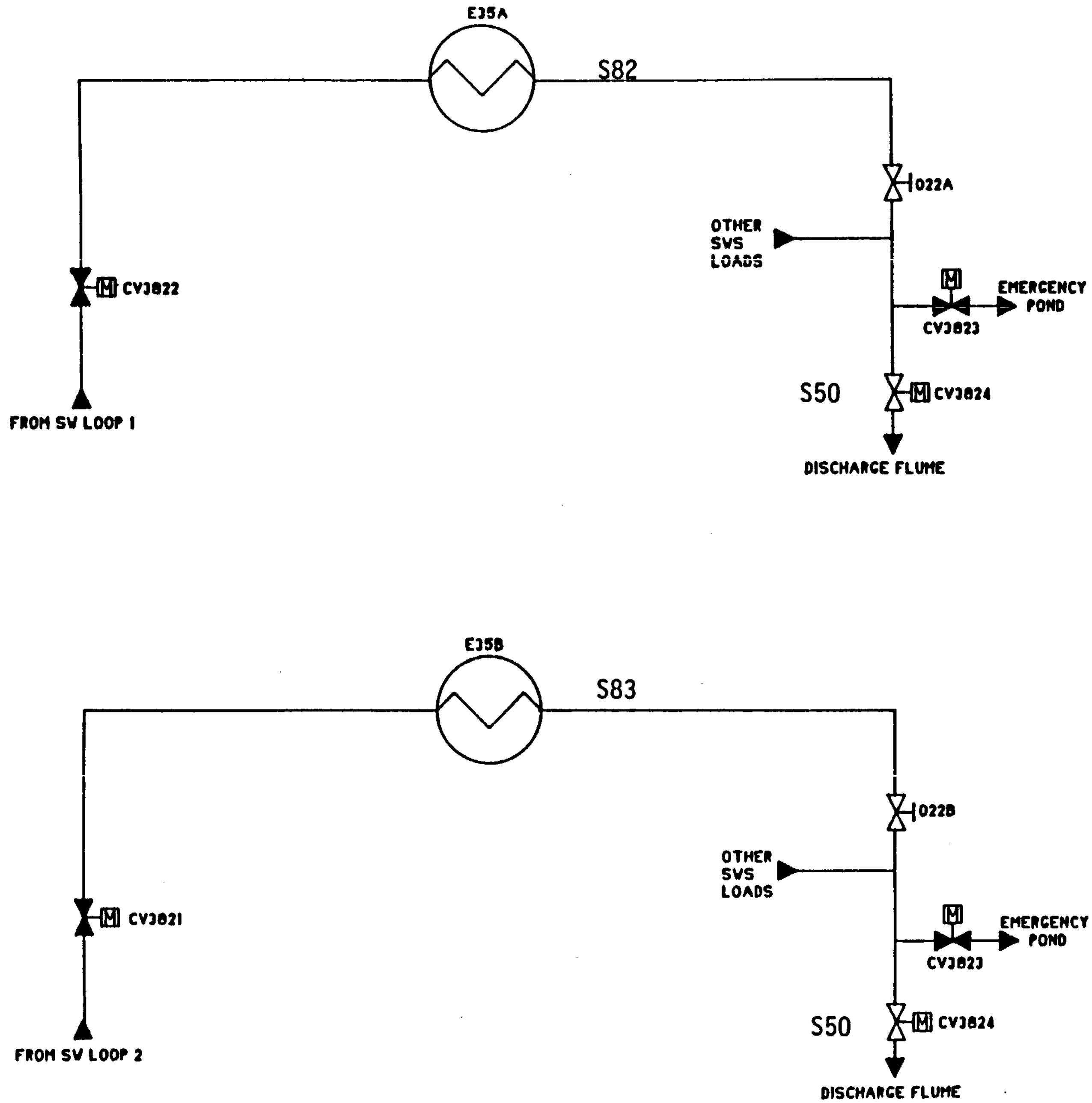

Figure 2.9. Pipe Segments for Low Pressure Decay Heat Exchangers Cooling.

recirculation system to remove heat from the containment during recirculation, and (b) to remove radioactivity from the containment atmosphere.

The RBSS serves only as an engineered safeguard system and performs no normal operating function. It consists of two independent trains. In the event of a loss-of-coolant accident (LOCA) that results in a Reactor Building pressure of 30 psig, the RBSS is actuated by the ESAS to take water from the borated water storage tank (BWST) (via the low pressure injection lines) and spray it into the Reactor Building. Once the BWST reaches a low level, the RB spray pump suction is transferred to the reactor building sump. This first phase is called Reactor Building Spray Injection (RBSI) and the second phase is called Reactor Building Spray Recirculation (RBSR). A simplified schematic of the system is shown in Figure 2.10.

The RBSS interfaces with the low pressure (LP) system in both the injection and recirculation phases. During the injection phase, in order for the water source to be available, it is necessary that the outlet manual valve (BW-1) from the BWST and the motor operated valves (MOVs) (CV1407 and CV1408) in the low pressure line be open. (The manual valve BW-1 is normally locked open; however, MOVs CV1407 and CV1408 receive an ESAS signal to open on either a 4 psig building pressure or 1500 psig coolant pressure signal. Thus, these valves are open when the RBSS requires injection water. The RBSS also requires LP valve action for recirculation.

The Reactor Building Spray Pumps require lube oil cooling. The flow paths modeled are shown in Figure 2.11.

For the two functions of the RBSS mentioned above, the success criteria is that one of the two trains must function. Two system-specific assumptions were made. First, plugging of the spray nozzles was ignored. Second, the addition of sodium hydroxide to the spray was not modeled based on the rationale that chemical addition to the spray does not significantly mitigate the offsite consequences of an accident.

2.2.6 Pressure Control and Overpressure Protection

Normal reactor coolant system pressure is maintained by the pressurizer steam bubble, and controlled by operation of the pressurizer spray and heaters. The system is protected against overpressure by reactor protection system circuits such as the high pressure trip and by pressurizer safety valves located on the top head of the pressurizer. The schematic arrangement of the safety valves is shown in Figure 2.12. Since all sources of heat in the system, i.e., core, reactor coolant pumps, and pressurizer heaters, are interconnected through the reactor coolant piping with no intervening isolation valves, all safety valves are located on the pressurizer.

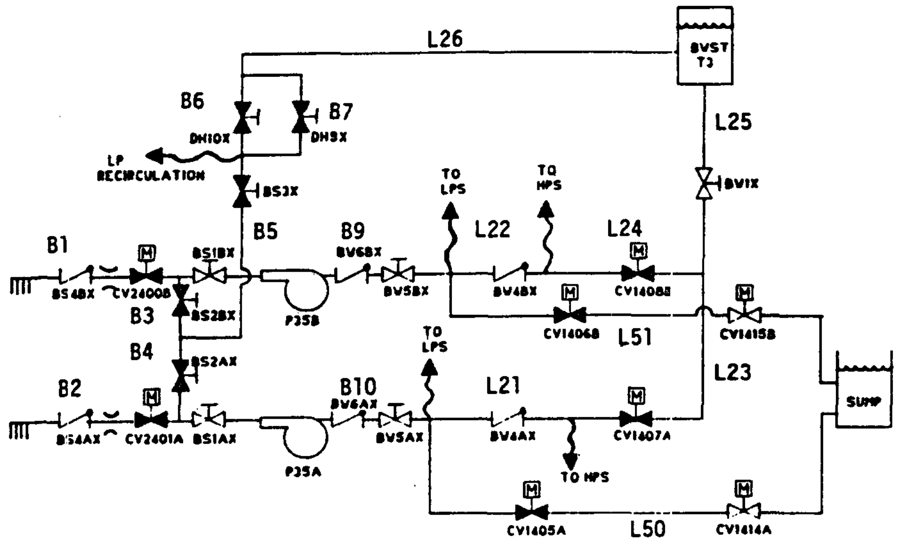

Figure 2.10. Pipe Segments for the Reactor Building Spray Injection/Recirculation.

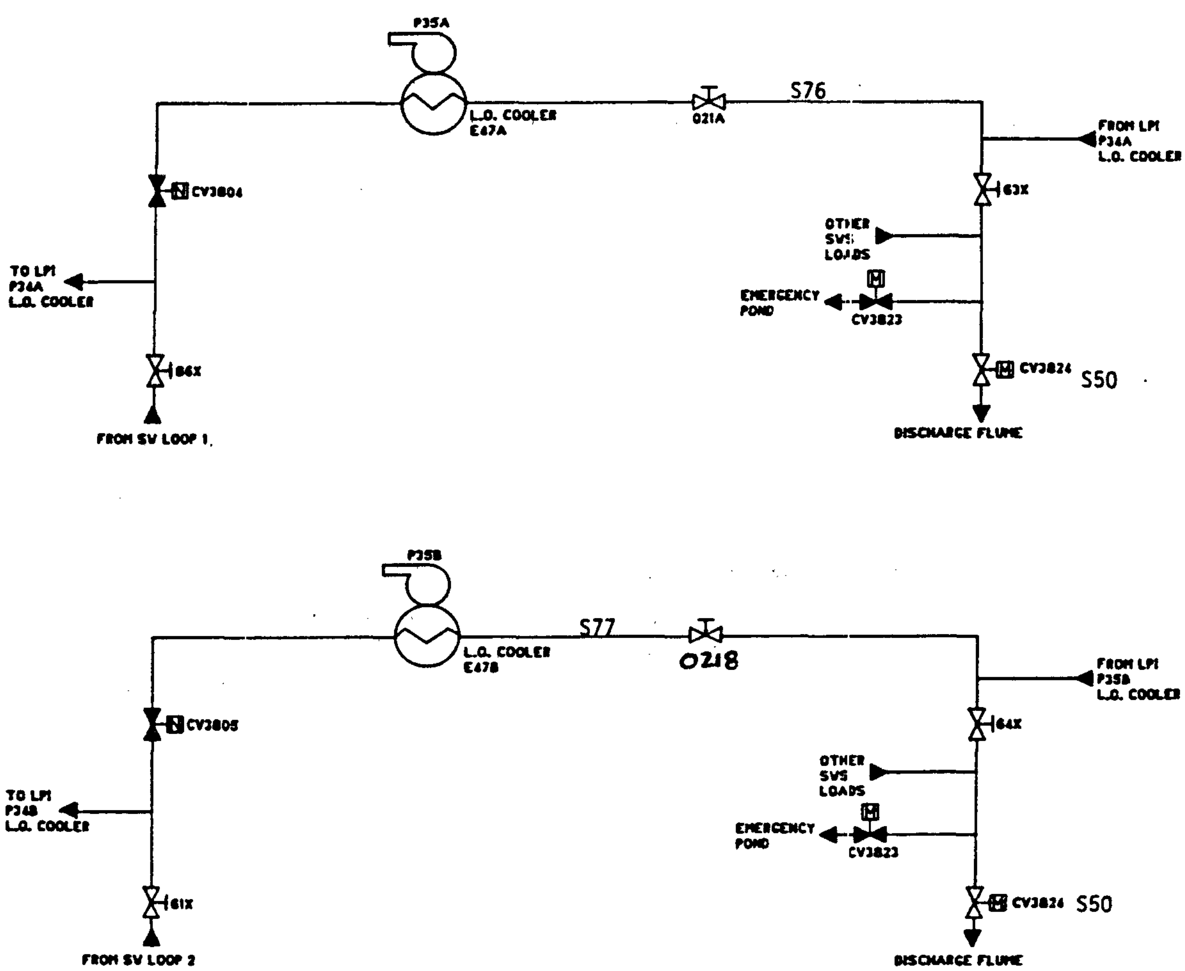

Figure 2.11. Pipe Segments for Reactor Building Spray Pumps Lube Oil Cooling.

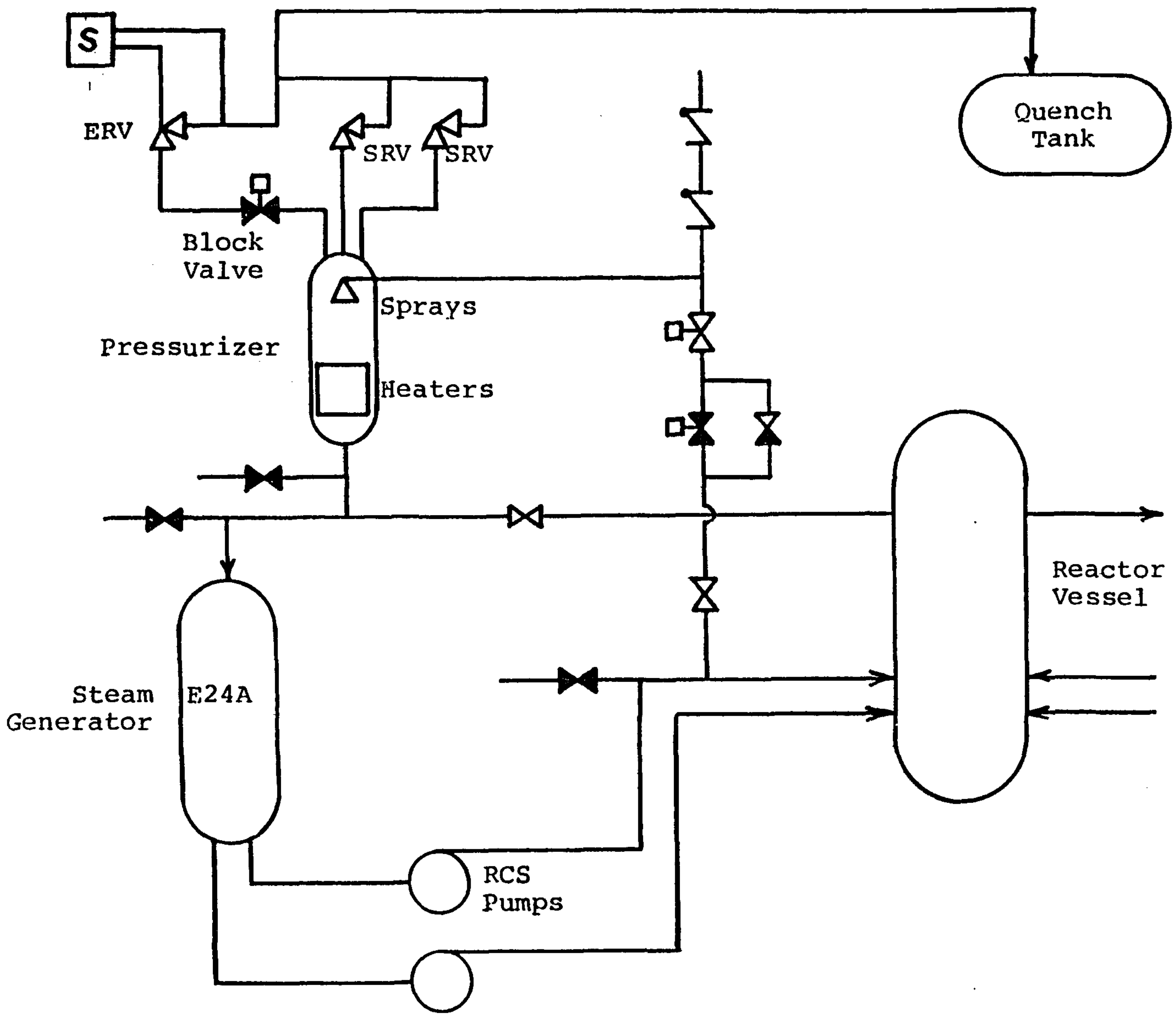

Figure 2.12. Reactor Coolant System Pressurizer.

2.2.6.1 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The nozzles are designed to transfer full discharge loads to the pressurizer. The valves have a closed bonnet with bellows and a supplementary balancing piston. The valve inlet and outlet are flanged to facilitate removal for maintenance or set point testing.

2.2.6.2 Pressurizer Electromatic Relief Valve

The pressurizer electromatic relief valve is mounted on a separate nozzle on the top head of the pressurizer. This nozzle is designed to transfer full discharge load to the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to unseat the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

2.2.6.3 Pressurizer Spray

The pressurizer spray line originates at the discharge of the reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by an electric motor operated valve using on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray valve provides a backup means of securing flow in the event the spray valve should stick open.

2.2.6.4 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during reactor coolant system heatup from a cooled down condition, and restore system pressure following transients. The heaters are arranged in 14 groups and are controlled by the pressure controller. Two groups utilize modulated control, and will normally operate at partial capacity to replace heat lost, thus maintaining pressure in the operating band. On-off control is used for the remaining 12 groups. A low water level interlock prevents the heaters from being energized with the heaters uncovered.

2.2.6.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank, which condenses and collects the relief valve effluent. This is shown schematically in Figure 2.12. The tank contents are cooled by dilution with the required minimum water volume in the tank.

The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank is vented to the gaseous radioactive waste disposal system.

2.2.7 Reactor Building Cooling System (RBCS)

The RBCS is provided to limit post-accident reactor building pressure to the design value during steam evolution within the building due to an accident. Emergency and normal cooling of the reactor building are performed by the same cooler units. Each unit contains normal and emergency cooling coils and a single speed fan. During normal plant operation, chilled water from the plant main water chillers is circulated through the normal cooling coils of the units selected for operation. Under post-accident emergency cooling operations, all units are on line with service water circulating through the emergency cooling coils for heat removal. ESAS-actuated dampers open to alter the flow from the normal to the emergency path.

The schematic flow diagram of the RBCS and its associated instrumentation is shown in Figure 2.13. The Reactor Building atmosphere enters each of the fan coolers at the fan locations. All four fans discharge into a supply air plenum which distributes cooled air throughout the Reactor Building. During emergency operations, one pair of units (VSFM1A and VSFM1B) is cooled by service water loop 1, while the other pair (VSFM1C and VSFM1D) is cooled by service water loop 2. The reactor building is normally isolated from the service water system by two ESAS-actuated pneumatically operated valves and two ESAS-actuated motor operated valves. A safeguards actuating signal, generated by an increase to 4 psig pressure in the reactor building, will cause the valves to open. Service water flows out of the coolers, exits the reactor building, and is then monitored for high radiation. High radiation in the service water closes the service water isolation valves and overrides any existing ESAS signal.

As with the service water and actuation signals, the four fans are divided with regard to motive power. The A and B fans are supplied from the odd emergency AC train, and the C and D fans from the even. Similarly, the associated valving is powered from the odd or even AC train, respectively, or the odd or even DC power train. The odd means that supplied from #1 Diesel Generator, and the even means that supplied from #2 Diesel Generator. The success criterion of the RBCS is that one of the four fan coolers must operate.

2.2.8 Engineered Safeguards Actuation System (ESAS)

The engineered safeguards actuation system (ESAS) monitors parameters associated with a major loss of reactor coolant accident and initiates operation of the proper engineered safeguards systems, i.e., emergency core cooling, reactor building isolation and cooling, and reactor building spray. Operations initiated by the ESAS are dependent on the severity of the accident.

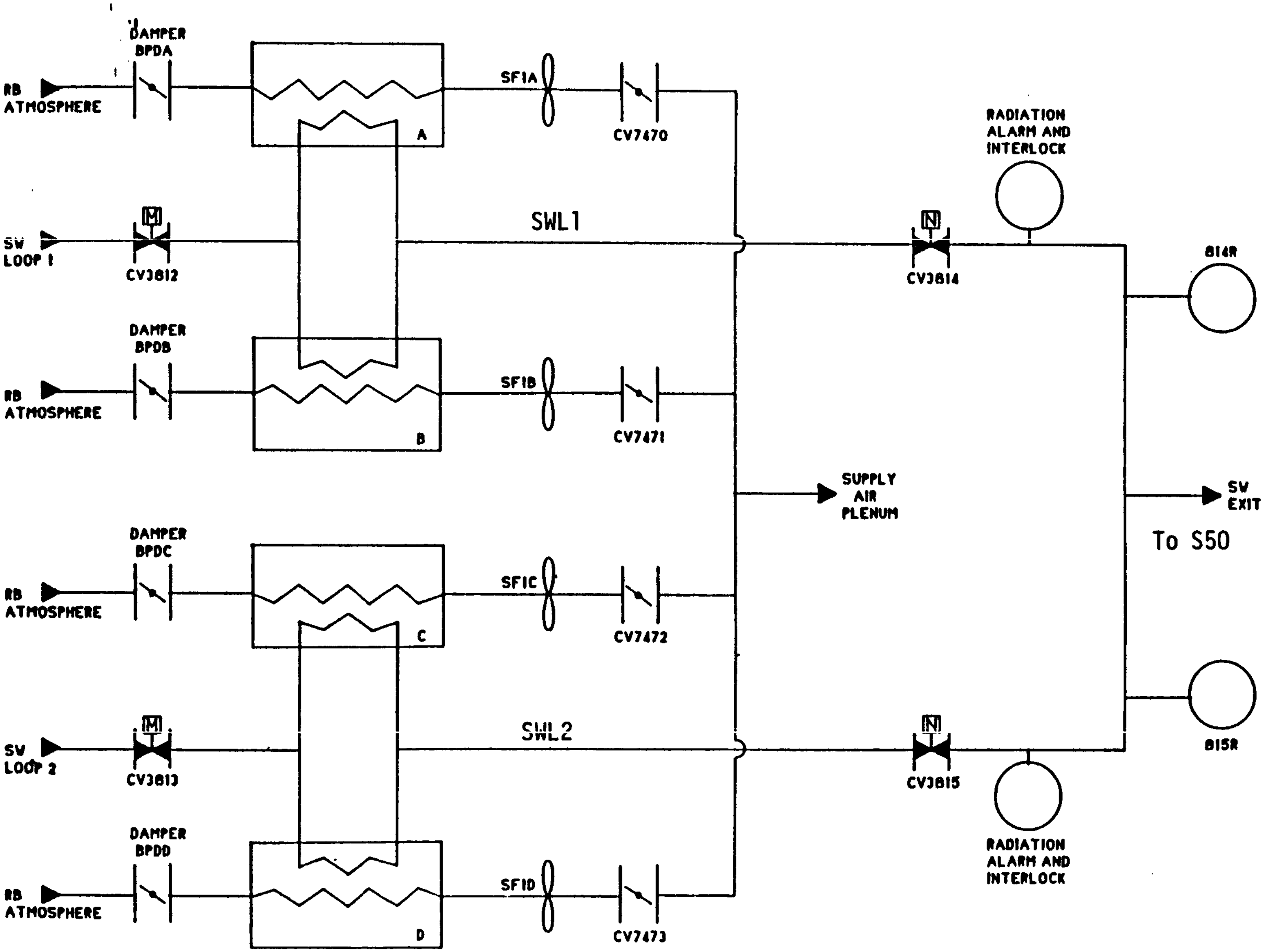

Figure 2.13. Pipe Segments for the Reactor Building Cooling System.

In addition, ESAS starts the emergency diesel generators. However, the ESAS does not automatically connect the output of the diesel generators to the emergency AC electrical system.

ESAS is composed of three redundant analog subsystems and two redundant digital subsystems as shown in Figure 2.14. Each analog subsystem contains two channels which monitor reactor coolant pressure and reactor building pressure. Each of the two digital subsystems contains five logic channels for initiation of safety action when two of three analog subsystems indicate such action is required. The components actuated by the odd and even digital subsystems are generally different but complementary, e.g., the pump in an odd system train is actuated by the odd ESAS train and the even by the even.

Each pressure sensor in the analog trains is connected to a buffer amplifier, which in turn feeds at least two bistables. Trip points for the sensors are a decrease of reactor coolant pressure to 1500 psig, an increase of reactor building pressure to 4 psig (high), or an increase of reactor building pressure to 30 psig (high-high), depending on which digital train is to be tripped. The analog components fail safe, i.e., removal of, or loss of power to a module causes the train output to initiate, which results in a 1 of 2 logic configuration remaining. During normal shutdown the low reactor coolant pressure trip can be manually bypassed while operating in the cooldown pressure band of 1500 to 1750 psig.

Each channel can be tripped by 1 of 2 methods: 2 of 3 analog subsystems trip, or manual action. Removal of, or loss of power to a digital component module does not cause an actuation output signal. The modules in the digital portion can be tested on-line. The success criteria of the eight logic trains is that each must send its signal to its actuated components when the specific accident sequence requires it. More detailed system block diagrams are given in Figure 2.15 and 2.16 for the eight actuation channels used in the analysis.

2.2.9 Power Conversion System (PCS)

The power conversion system (PCS) is designed to provide feedwater to the secondary side of the steam generators which, in turn, transfer energy generated in the reactor to the turbine generator system. Following a reactor trip, the PCS is also capable of delivering feedwater to the steam generators at a reduced rate to provide for decay heat removal. This is accomplished by throttling the PCS feedwater flow to a level commensurate with decay heat and allowing this water to boil off to the condenser or atmosphere.

Figure 2.17 shows a simplified schematic of the PCS. The feedwater portion of the PCS consists of two pump trains. Three low pressure motor driven condensate pumps feed two high pressure steam driven main feedwater pumps and one high pressure motor driven auxiliary feedwater pump. These latter three pumps in turn feed both steam generators via two injection lines. The injection lines are cross-connected by a line which contains a motor operated valve. Each injection line also

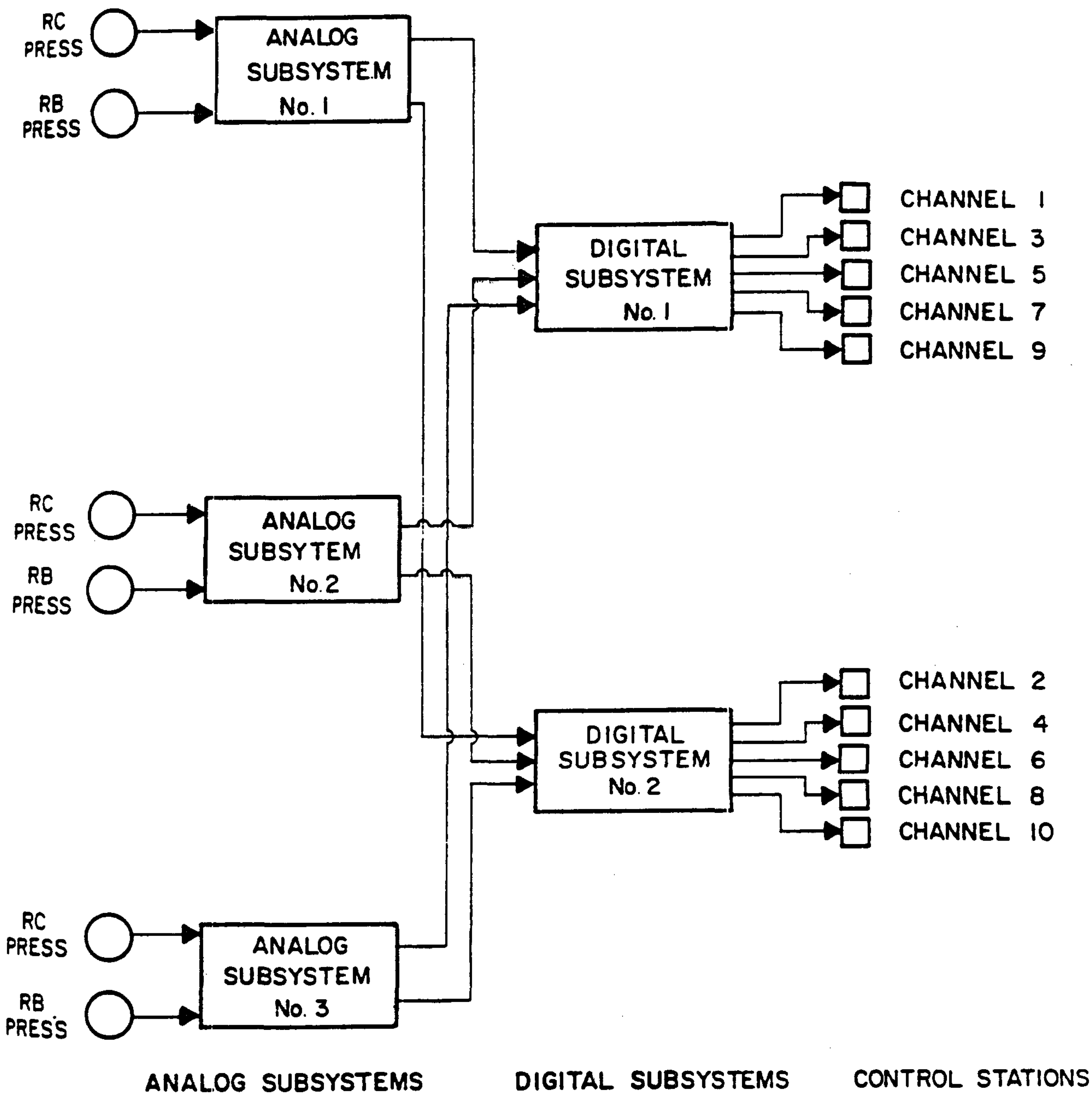

Figure 2.14. Simplified Schematic of Engineered Safeguards Actuation System.

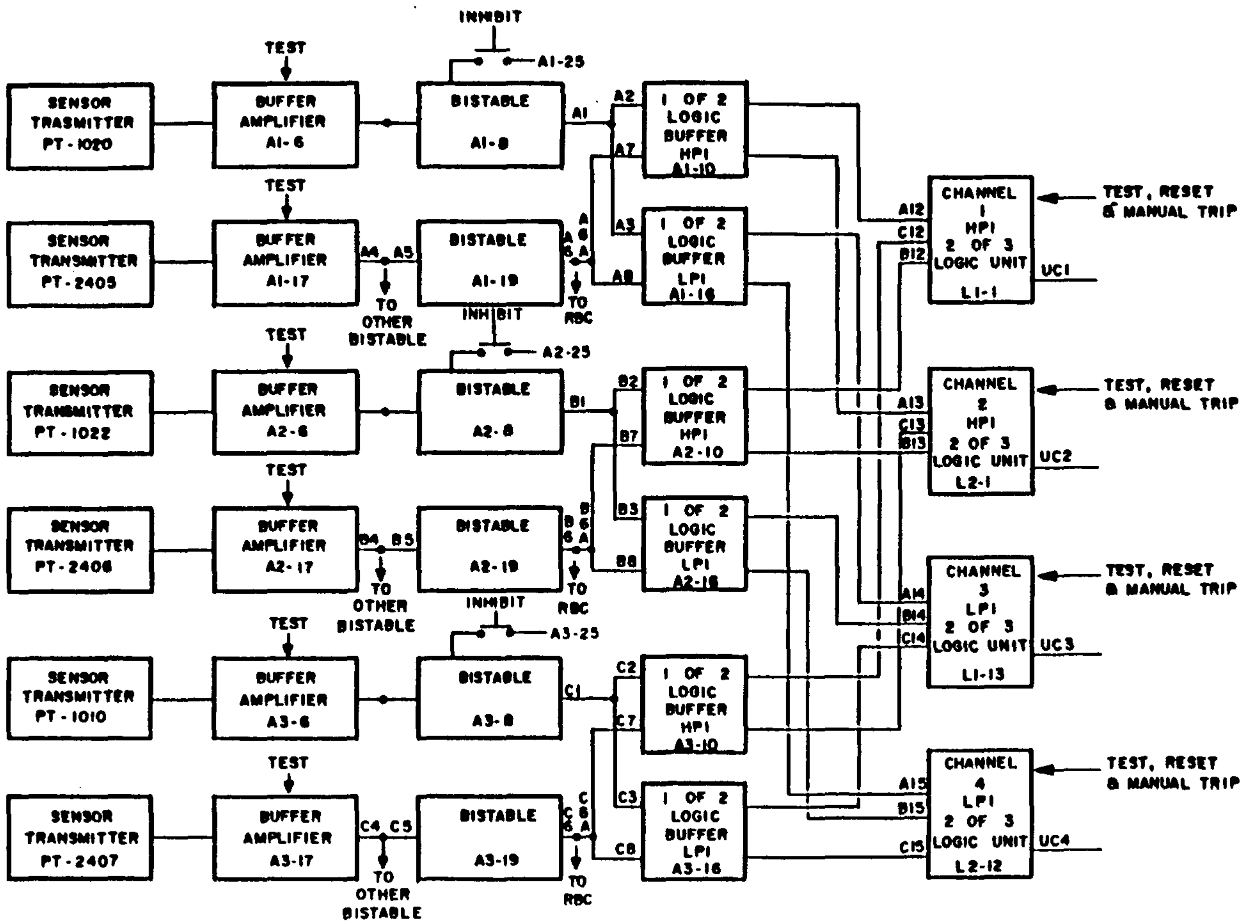

Figure 2.15. Wiring Segments of Channels 1-4 of the Engineered Safeguards Actuation System.

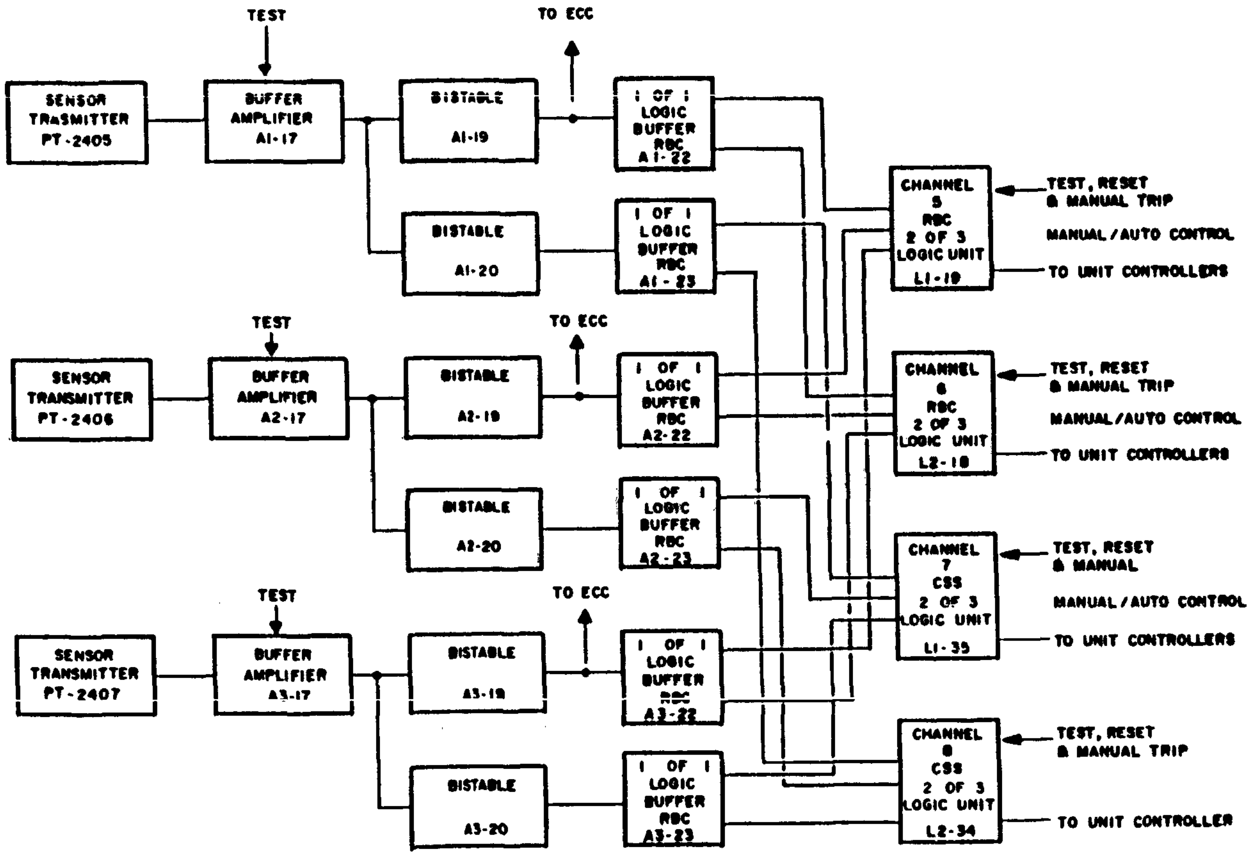

Figure 2.16. Block Diagram of Channels 5-8 of the Engineered Safeguards Actuation System.

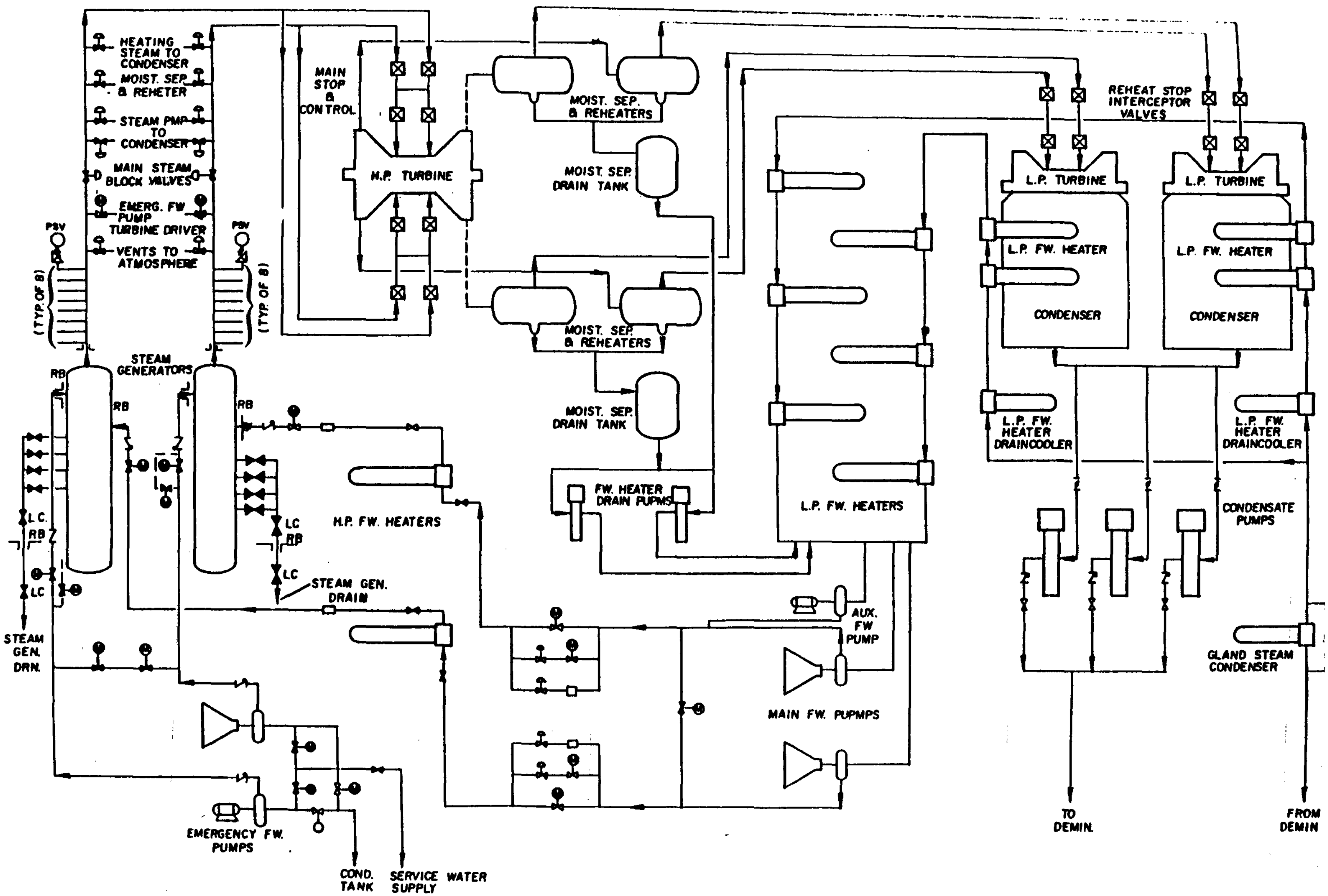

Figure 2.17. Schematic of Power Conversion System.

includes three parallel lines, two with control valves for startup and low load regulation and one with a motorized gate valve for full load; one high pressure feedwater heater; one flow-tube type flow measuring device; isolation valves; and one check valve (containment isolation valve) just outside the reactor containment.

For this analysis, a detailed fault tree of the PCS was not constructed. Instead, industry data were used which represent the PCS unavailability due to independent causes not associated with the initiating events or systems modeled in the fault trees.

2.2.10 Service Water System (SWS)

The purpose of the service water system (SWS) is to provide water to the following equipment during emergency conditions:

a. Reactor building cooling system cooling coils,

b. Diesel generator jacket heat exchangers,

c. High pressure pump oil coolers,

d. High pressure pump room coolers,

e. Circulating water pump bearing lubrication,

f. Low pressure/building spray pump room coolers,

g. Low pressure pump bearing coolers,

h. Low pressure system decay heat exchangers,

i. Building spray pump oil coolers, and,

j. Emergency feedwater system water source.

The SWS consists of two redundant loops as shown in Figure 2.18. Normal cooling is supplied from the reservoir; however, an emergency pond is available in case of loss of flow from the reservoir. The service water is normally discharged back to the reservoir via the circulating water discharge flume. If the reservoir source is lost, the service water would be discharged back to the emergency pond.

There are three SW pumps. During normal operation, two of them are in use with the third pump in standby. All of the crossover valves in the common-pump-discharge header are open, but they close upon ESAS actuation. The ESAS will only send an actuation signal to the two SW pumps that were already running in the normal mode. Success for the SWS is that all components requiring cooling in a specific accident sequence receive sufficient SWS flow for that cooling. The success criteria of the SWS in this analysis are two-fold: (a) with an ESAS signal present, and (b) without an ESAS signal present. With an ESAS signal

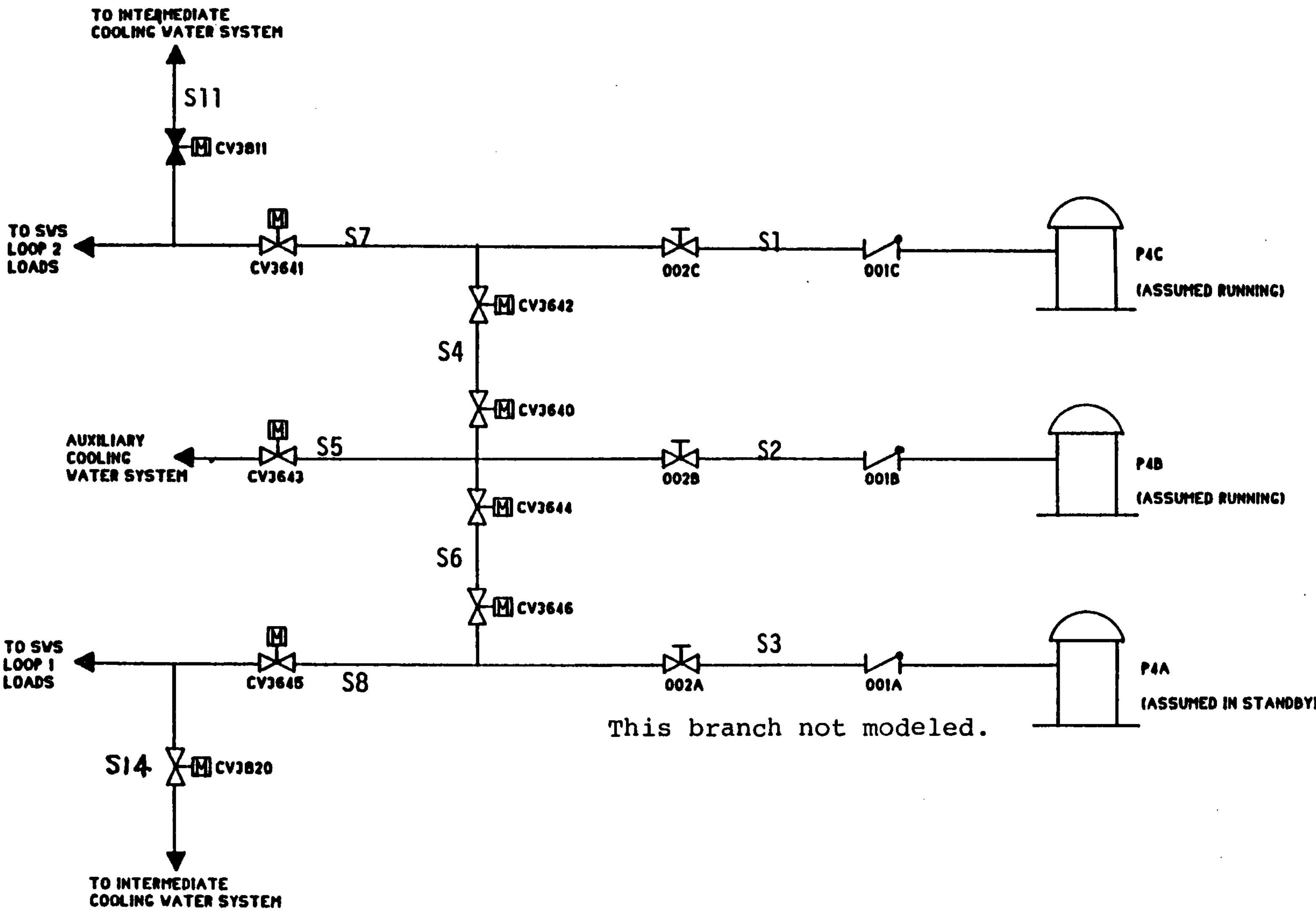

Figure 2.18. Pipe Segments for Service Water System Pump Loops.

present, no credit is allowed for one loop backing up the other. That is, the loops are designed to isolate on an ESAS signal, and if they do not, the operator is trained to isolate them. In addition, any diversion from a loop subsequent to the ESAS signal is assumed to fail that loop. For the case without an ESAS condition (or prior to one being initiated), credit is given for one loop backing up the other, and diversions to normal plant loads do not fail the SWS because the pre-ESAS heat loads are not as large.

2.2.11 Emergency Power System (EPS)

The emergency power system consists of both AC and DC sections. The AC portion is powered by emergency diesel generators. The DC portion is powered by batteries. These systems work together to energize vital control, instrumentation, and power requirements when off station power is lost in whole or in part.

2.2.11.1 Emergency AC Electrical Systems

The emergency AC electrical system (EACS) provides electric power to the ESF equipment of the mitigating systems. The EACS is composed of two trains (see Figure 2.19), each consisting of a diesel generator, 4160 V switchgear, 480 V load centers and motor control centers, 120V instrumentation panels, and associated transformers and circuit breakers. The normal power supply to the two trains is offsite power with each train having its own offsite connection at the 4160 V switchgear (A3 or A4).

There are three independent undervoltage sensing circuits per train. Two are connected to A3 and A4, and one to the 480 V load centers B5 and B6. Upon sensing an undervoltage condition, any of them will transmit a signal to open the 309 breaker for A3 (409 for A4), start the diesel in the train, and close breaker 308 (408 for A4).

Each diesel generator is equipped with a permanent magnet for field flash generation, and if the air start can crank the engine to 300 rpm, the generator will develop normal output voltage without additional external field flashing (available from the DC power system). The diesels have a maximum rated starting time of 15 seconds, measured from the admission of starting air to achieving normal output frequency and voltage. DC power is required, however, to open the solenoid valves in the lines connecting the air start tanks to the diesel. They cannot currently be manually opened. There are two independent compressed starting air tanks for each diesel. Subsequent to the start, DC power is also required for diesel generator and circuit breaker control power. SWS cooling is also required (see Figure 2.20).

As can be seen in Figure 2.19, A3 and A4 (and B5 and B6) are capable of being cross-connected. The cross-tie breaker sets are designed so that only 3 of the 4 can be closed at any time, and an undervoltage signal opens the tie if it is shut. Normal practice is not to cross-tie the two trains.

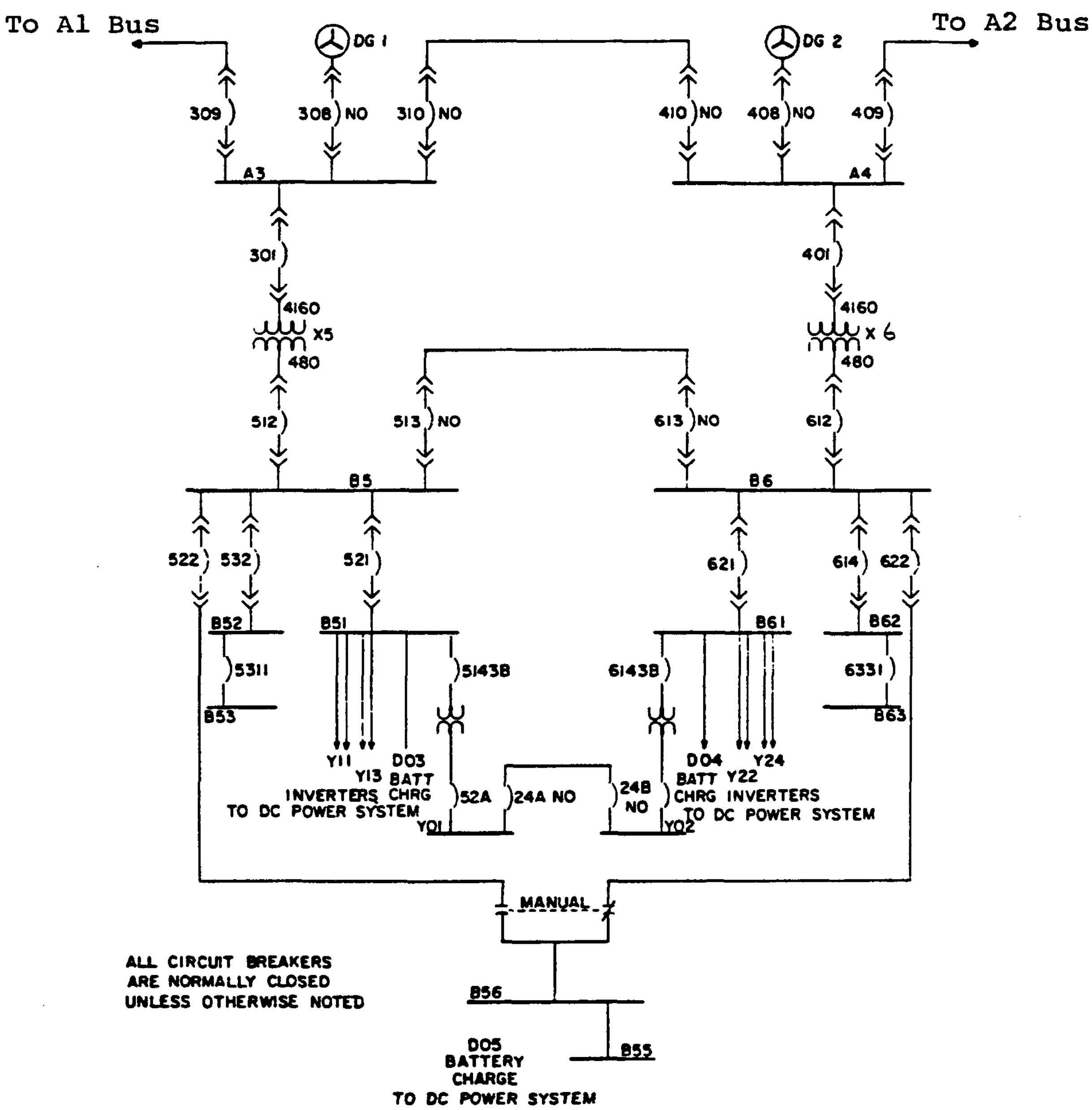

Figure 2.19. Emergency AC Electrical System.

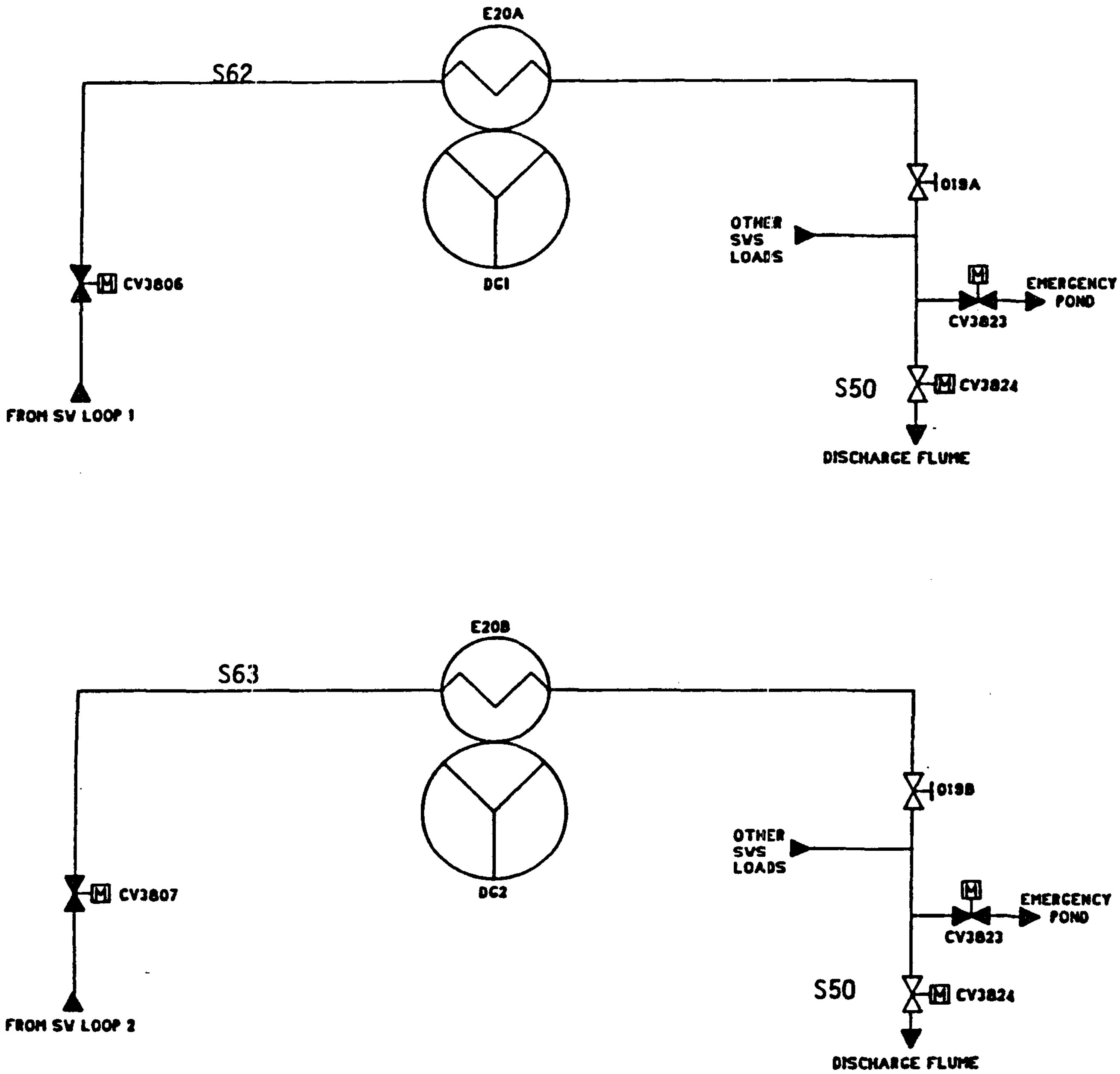

Figure 2.20. Pipe Segments for Service Water Cooling of Diesel Generators.

There is no installed integrated system to sequence loads. Instead, the ESAS sends out start signals to all components simultaneously, and the control circuitry for each major load component contains a time delay relay. Conservatively, for this analysis, it is assumed that a failure in these relays will cause the failure of the diesel which ultimately powers them. In addition to the ESAS time delay relays, each component has similar, parallel circuitry for undervoltage trips.

It is assumed that the A3 and A4 switchgear rooms will eventually require cooling. This cooling is modeled as a support system fault of the long running equipment (e.g., pumps, fans) and not for the initially acting components (valves). The reason for this approach is that these are failures of the electrical system after time has elapsed (five minutes for lack of service water, several hours for room cooling) which is not a failure mechanism for initially acting components. The success criteria for this support system is that each bus is powered when ESF equipment requires it.

2.2.11.2 DC Power System

The 125 volt DC system provides continuous power for control, instrumentation, reactor protection and engineered safeguards actuation systems, and emergency safeguard actuation control (e.g., pumps) systems. In addition, it powers the control valves in the emergency feedwater system and provides control power for the diesel generators in the emergency AC electrical system.

The DC system is composed of two separate trains, each consisting of a 125 volt battery, buswork, and control panels (see Figure 2.21). In addition to the batteries, the DC system is also supplied from the emergency AC electrical system via three battery chargers, with two in service at any given time. The charger alignment is rotated monthly.

The DC busses cannot be cross-tied. However, if maintenance is required on one train, its respective distribution panel can be manually aligned to receive power from the main bus of the other train. The emergency DC to AC inverters each have three sources of power: the respective DC bus, rectified and transformed 480 V AC from the emergency AC system, and transformer 480 V AC from the emergency AC system which does not actually pass through the inverter itself (see Figure 2.22).

Each battery is designed to carry the continuous DC and vital AC loads for a minimum of eight hours following a station blackout. This time could be extended with the manual shedding of loads per the appropriate emergency procedure. Of crucial importance to system performance is the reliability of the batteries. Discharge tests are performed at the refueling shutdown every 18 months. The rest of the time the batteries are on float charge. Other inspections include daily checks of the pilot cells, quarterly checks of all the cells and terminals, and an annual check of the cells as well as terminal cleaning.

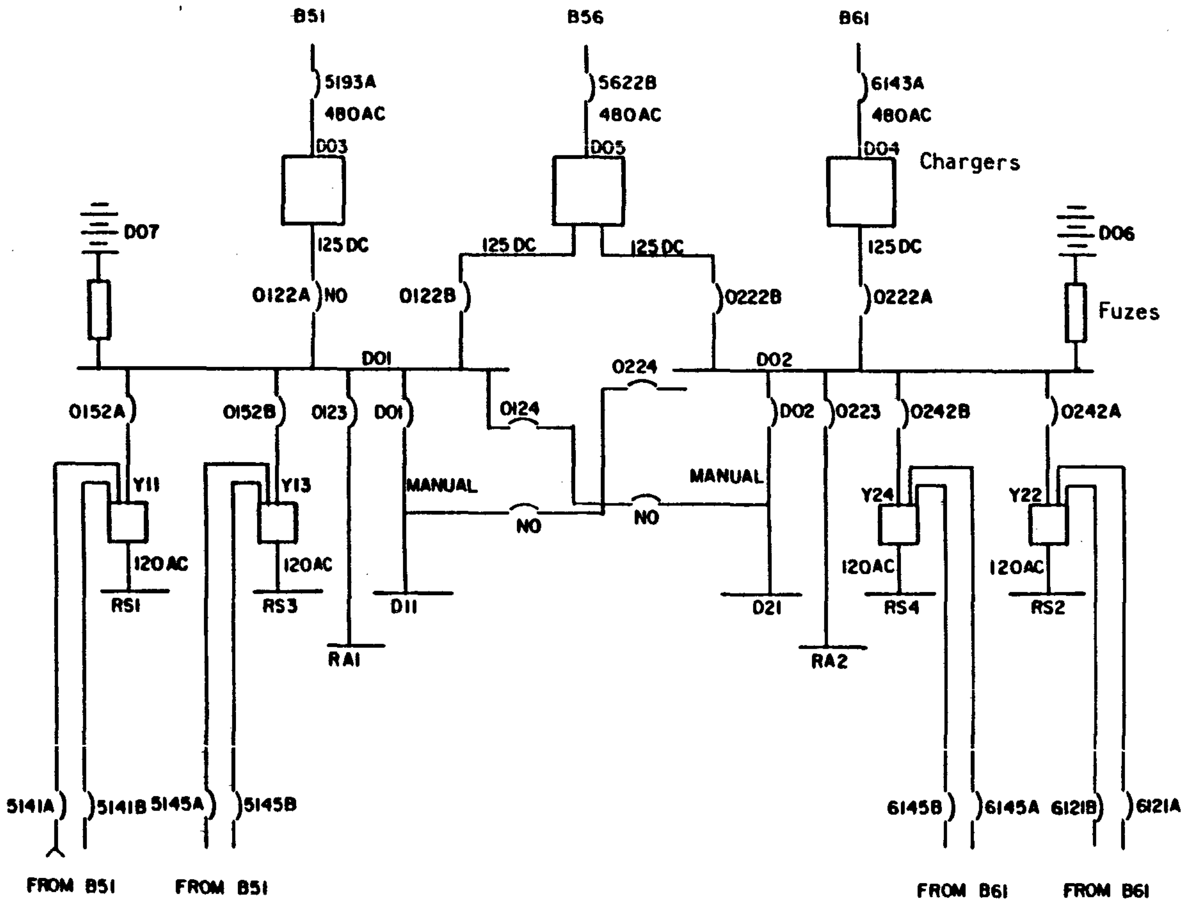

Figure 2.21. DC Power System.

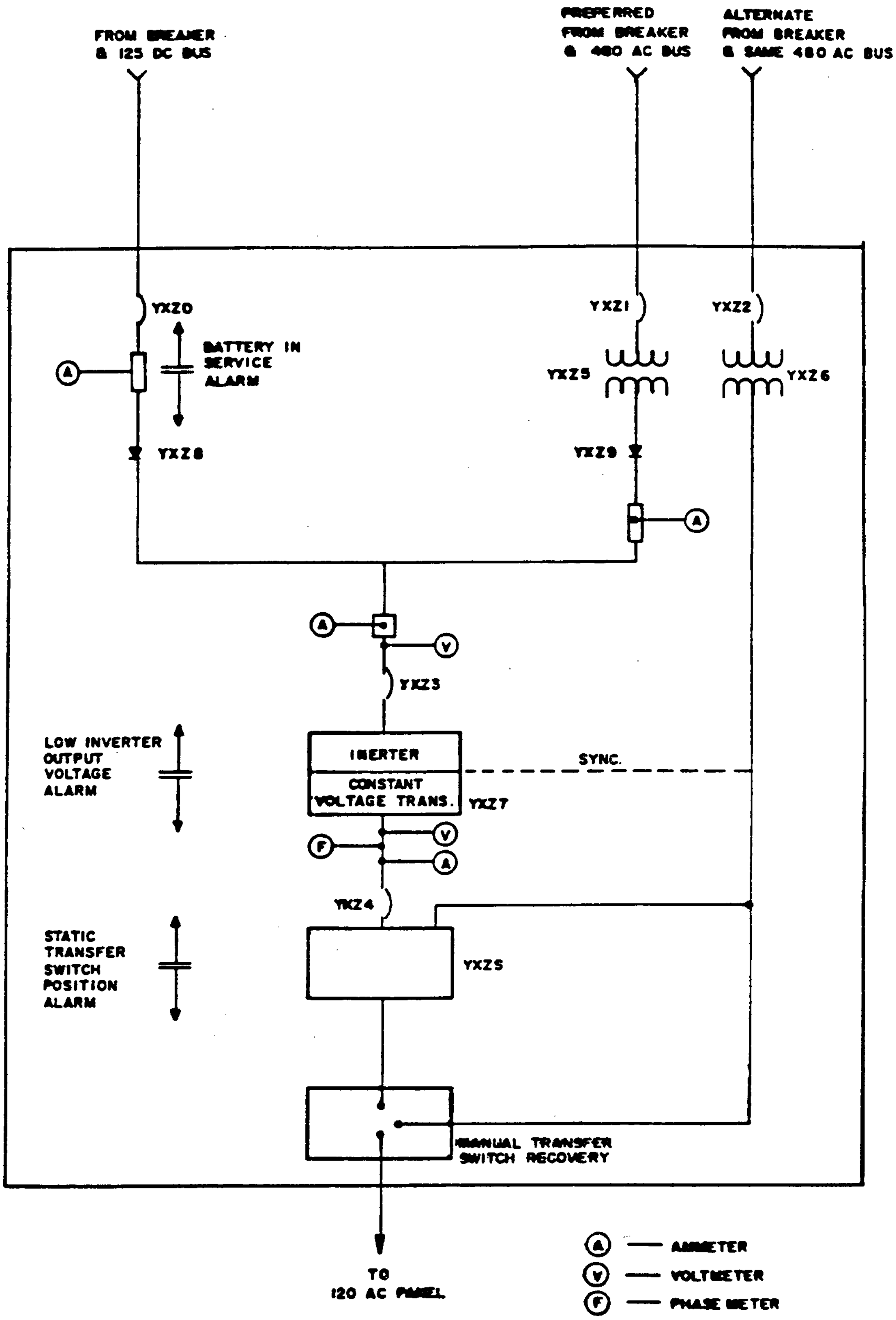

Figure 2.22. DC Power System Inverter, with Components Identified as used in the Analysis.

It is possible to have both in-service battery chargers receiving AC power from the same bus. B61 of the AC system is always fed from bus B6, and shift checklists stipulate that swing bus B56 should also be powered from there unless diesel generator 2 is undergoing maintenance (see Figure 2.18). This condition results (see Figure 2.20) in both battery chargers receiving power from B6.

It is assumed that the battery rooms will eventually require cooling. This cooling is modeled the same as that for the switchgear rooms and the diesel generator service water cooling.

The success criteria for the DC power system is that all components requiring either DC motive or control power receive it. An important assumption in this analysis is that the battery chargers can supply DC power without the batteries, i.e., they are an independent DC power source.

2.2.11.3 Battery and Switchgear Emergency Cooling System

The purpose of the battery and switchgear emergency cooling system (referred to as ECS) is to provide sufficient cooling to assure that electrical units which must operate during emergency conditions, will not fail due to excessive heat. The rooms with electrical equipment in them which could fail due to overheating are the two battery rooms and the two switchgear rooms. This conclusion is based on a utility analysis which showed that after one hour without cooling the room temperature exceeded that of the allowed duty ambient reading of the batteries.

The ECS consists of two independent identically configured chilled water trains which provide cooling to the rooms noted above, and two refrigerated air units which provide additional cooling capacity for the north and south battery rooms. Each chilled water train consists of a chilled water unit (CWU), three ventilation unit coolers (VUCs) and associated plumbing. A simplified schematic of the chilled water system is shown in Figures 2.23 and 2.24. Additional cooling for the north and south battery rooms is provided by self-contained air cooled refrigeration units which are independent of the chilled water system. A single refrigeration system is provided for each battery room. Each system consists of a condenser unit, an evaporator/blower unit and a thermostat. Each unit has sufficient capacity to accommodate 100% of the heat load in its associated battery room.

For the chilled water units, the odd ECS train cools odd components of the emergency AC and DC systems as well as being powered by them. Likewise, the even train cools and is powered by the even train. The refrigeration units, however, are both powered by the swing bus of the emergency AC system, which is normally aligned to be fed from B6. Thus, one of the cooling units for the odd train battery is normally powered by an even train component.

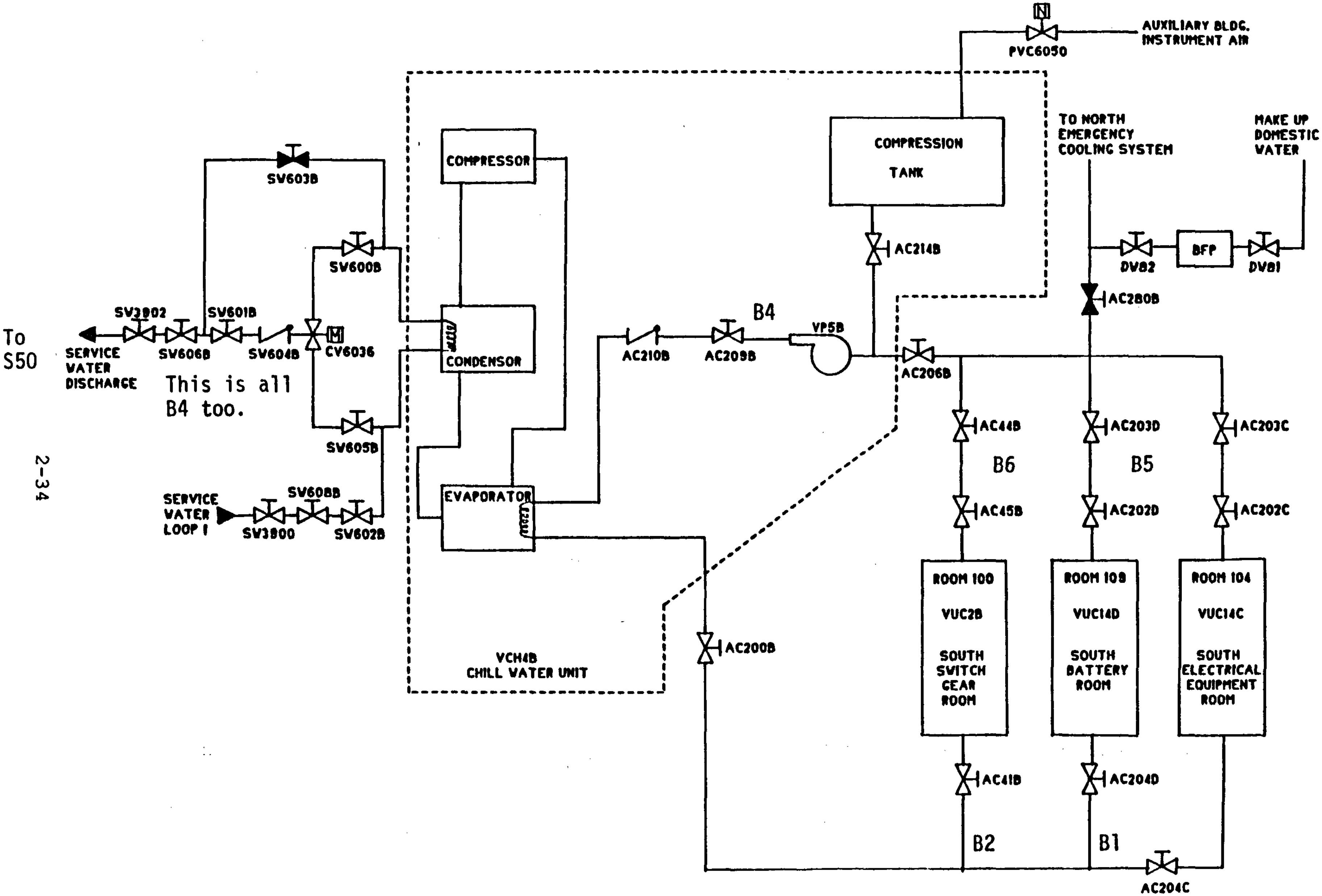

Figure 2.23. Pipe Segments for the Emergency Cooling System: South Rooms.

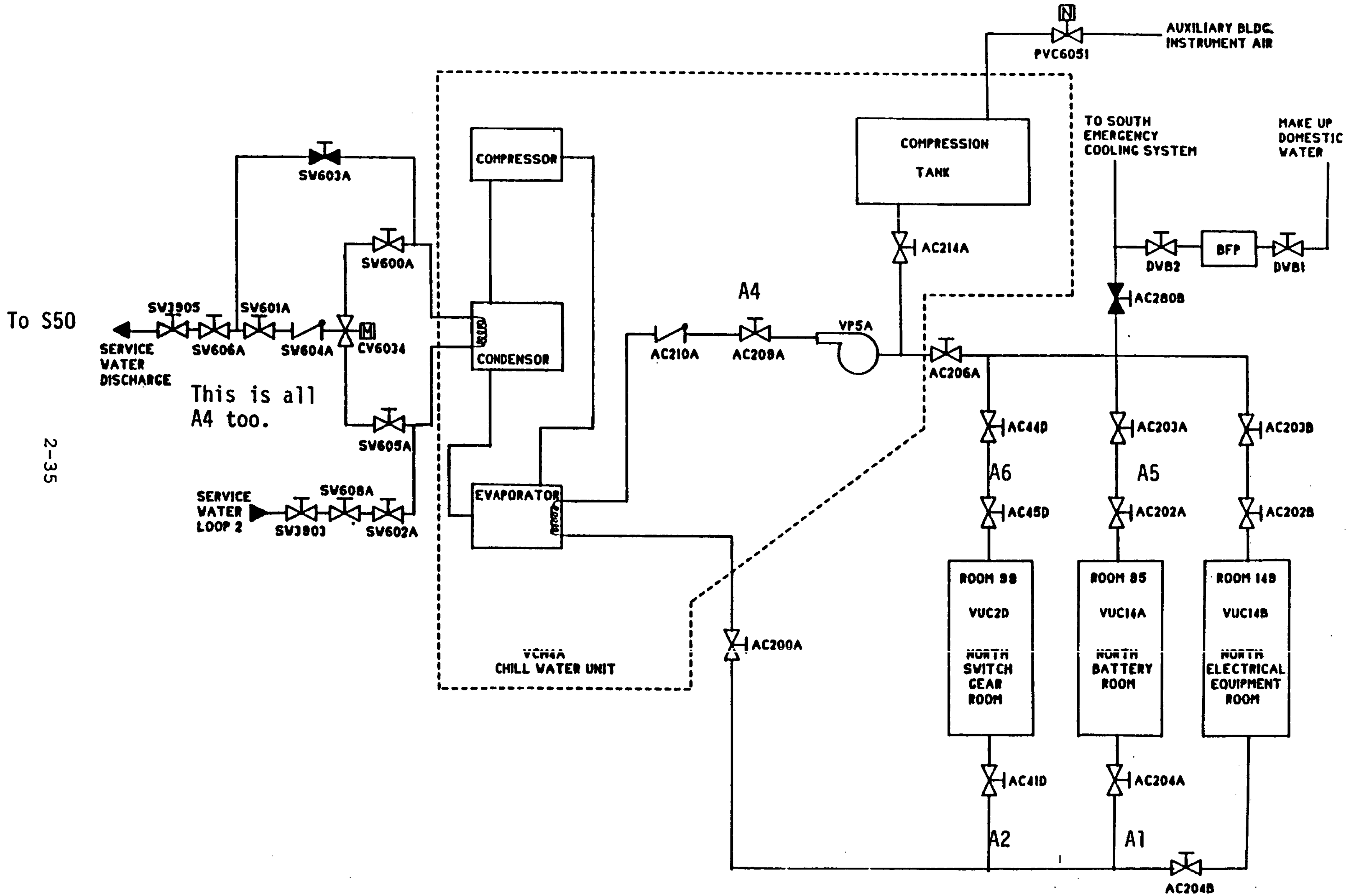

Figure 2.24. Pipe Segments for the Emergency Cooling System: North Rooms.

The success criteria of the ECS is to provide cooling to each of the four rooms mentioned above. For each of the switchgear rooms this means success of the associated chill water unit. For each of the battery rooms these criteria mean that either the associated chill water unit or the refrigeration unit must succeed.

2.3 References

2.1 G. J. Kolb, SAND82-0978, NUREG/CR-2787, Sandia National Laboratories, Albuquerque, NM, June, 1982.

2.2 W. R. Cramond et al, Shutdown Decay Heat Removal Analysis of A Babcock and Wilcox Pressurized Water Reactor, NUREG/CR-4713, SAND86-1832, Sandia National Laboratories, Albuquerque, NM, March, 1986.

3.0 ROOT CAUSE ANALYSIS

3.1 Introduction

This chapter describes the quantification and resulting contributions to core damage frequency (CDF) for each of the root cause scenarios. For the B&W PWR being studied, a detailed Decay Heat Removal PRA and supporting analyses were available as part of the NRC-sponsored Task Action Plan A-45 studies (Ref. 3.1). In the A-45 studies, plant-specific data analysis was performed as part of the internal events analysis and these results were utilized where applicable. In this study, detailed analysis of the propagation of smoke and heat within each room was performed (Refs. 3.2 through 3.4). As part of this analysis the actual location of critical equipment in the room was considered. A plant-specific evaluation of the number and type of fire barriers in each zone was made.

For this analysis, the configuration of equipment and fire protection systems at the plant were reviewed. The potential root causes of FPS actuations that could lead to core damage were identified. Based on the knowledge of the FPS configuration, a quantification of potential core damage sequences was performed.

3.2 Procedure

The initial phase of the analysis consisted of reviewing the plant configuration. This was accomplished primarily by reviewing the plant Appendix R submittal (Ref. 3.5). From this submittal, information was obtained on the overall plant layout, on the individual plant Fire Zones, and on the particular types of FPS and fire detectors installed. Reference 3.1 was used to determine the critical equipment required for safe shutdown. Location by fire zone of the critical equipment and its associated power and control cable information was used to determine those critical zones of interest for further study. Using this information, a vital area analysis was performed. A listing of all Fire Zones which resulted from the vital area analysis is given in Table 3.1. These zones contain safe shutdown equipment or circuits associated with such equipment.

Seventeen Fire Zones having automatically or manually actuated fixed Fire protection systems were identified. These zones are listed in Table 3.2 along with the type of FPS, type of detectors, FPS actuation scheme, and critical equipment in the Fire Zone. Figures 3.1 through 3.6 are simplified illustrations of these critical Fire Zones.

In several instances, the Appendix R information was supplemented by discussions with plant personnel as well as a detailed plant walkdown. Equipment and cable locations were verified during this walkdown. The Appendix R submittal was also used, along with a plant walkdown, to characterize the penetrations into each of the critical Fire Zones.

The IREP PRA (Ref. 3.6) and Decay Heat Removal PRA (Ref. 3.1) internal events reports provided additional information on the plant safe

Table 3.1

Fire Zones Containing Safe Shutdown Equipment

Fire Zone Number	Fire Zone Name
14-EE	West Decay Heat Removal Pump Room
40-Y	Pipe Area
67-U	Laboratory and Demineralizer Access
68-P	Reactor Coolant Make-up Tank Room
73-W	Condensate Demineralizer Room
75-AA	Boiler Room
76-W	Compressor Room
77-V	Upper South Piping Penetration Room
104-S	Electrical Equipment Room
128-E	Controlled Access
160-B	Computer Room
167-B	Computer Transformer Room
170-Z	Steam Piping Area
197-X	Turbine Building
20-Y	Radioactive Waste Processing Area
34-Y	Pipe Area
38-Y	Emergency Feedwater Pump Area
47-Y	Penetration Ventilation Area
53-Y	Lower North Piping Penetration Area
100-N	South Switchgear Room
97-R	Cable Spreading and Relay Room
129-7	Control Room
98-J	Uncontrolled Access Area
99-M	North Switchgear Room
112-I	Lower North Electrical Penetration
32-K	North Side Containment Building
33-K	South Side Containment Building
DFSV	Diesel Fuel Storage Vault
244	SWS Section of Intake Structure

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
20	Makeup pump rooms contain ionization type smoke detectors. Treatment makeup monitor room also contains ionization smoke detectors. Treatment makeup monitor tank room has a wet pipe sprinkler system.	EFW0P7AX Pump Cable EFW2627A Valve Cable EFW00X3A Valve HPI1219A Valve Cable HPI1220A Valve Cable HPIP36AA Pump Cable HPIP36BA Pump Cable HPIP36CB Pump Cable LPI1407A Valve Cable LPR1405A Valve Cable RBI2401A Valve Cable RBIP35BB Pump Cable SWS3803 Valve Cable SWS3808A Valve Cable SWS3809A Valve SWS3810B Valve Cable SWS3821 Valve Cable SWS3822 Valve Cable
32	Automatic preaction sprinklers are provided in cable penetration areas.	LPR1414A Valve LPR1415B Valve
38	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves.	EFW00X2B Valve EFW00X4B Valve EFW0P7AX Pump Cable EFW2627A Valve Cable EFW2800A Valve EFW2802B Valve EFW0P7BA Pump Cable HPI1219A Valve Cable HPI1220A Valve Cable LPI1401A Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
		LPI1407A Valve Cable LPR1405A Valve Cable RBI2401A Valve Cable SWS3803 Valve Cable SWS3808A Valve Cable SWS3820A Valve Cable SWS3822 Valve Cable
73	This zone is provided with smoke detectors and an automatic wet pipe sprinkler system.	EFW0P7BA Pump Cable HPIP36AA Pump Cable HPIP36BA Pump Cable HPIP36CB Pump Cable LPI0P34A Pump Cable LPI0P34B Pump Cable LPI1401A Valve Cable LPR1405A Valve Cable RBIP35AA Pump Cable RBIP35BB Pump Cable SWS0P4BA Pump Cable SWS0P4CB Pump Cable SWS3640B Valve Cable SWS3642B Valve Cable SWS3643A Valve Cable SWS3820A Valve Cable
79	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves.	EFW0P7AX Pump Cable EFW2626B Valve Cable EFW2627A Valve Cable EFW2670A Valve Cable HPI1219A Valve Cable HPI1220A Valve Cable HPI1227B Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
		HPI1228B Valve Cable LPI1400B Valve Cable LPI1401A Valve Cable LPR1406B Valve Cable RBI2400B Valve Cable RB12401A Valve Cable SWS3802 Valve Cable SWS3810B Valve Cable SWS3821 Valve Cable
86	This zone is protected by a preaction sprinkler system. Smoke and flame detectors actuate the sprinkler's water supply valves.	DG2-GEN Diesel Generator SWS3807B Valve Cable
87	This zone is protected by a preaction sprinkler system. Smoke and flame detectors actuate the sprinkler's water supply valves.	DG1-GEN Diesel Generator SWS3806A Valve Cable
97	Ionization smoke detectors are provided at ceiling level, and line heat detectors are located in cable trays. An automatic deluge system is provided that activates on cross zone detection of any heat and smoke detector.	EFW00Y1A Valve Cable EFW00Y2B Valve Cable EFW0P7AX Pump Cable EFW2620B Valve Cable EFW2626B Valve Cable EFW2627A Valve Cable EFW2670A Valve Cable EFW0P7BA Pump Cable HPI1219A Valve Cable HPI1220A Valve Cable HPI1227B Valve Cable HPI1228B Valve Cable HPIP36AA Pump Cable HPIP36BA Pump Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
		HPIP36CB Pump Cable LPI0P34A Pump Cable LPI0P34B Pump Cable LPI1400B Valve Cable LPI1401A Valve Cable LPI1401A Valve Cable LPI1407A Valve Cable LPI1408B Valve Cable LPR1405A Valve Cable LPR1406B Valve Cable RB12400B Valve Cable RB12401A Valve Cable RBIP35AA Pump Cable RBIP35BB Pump Cable SWS0P4BA Pump Cable SWSoP4CB Pump Cable SWS3640B Valve Cable SWS3642B Valve Cable SWS3643A Valve Cable SWS3802 Valve Cable SWS3803 Valve Cable SWS3806 Valve Cable SWS3807B Valve Cable SWS3808A Valve Cable SWS3810B Valve Cable SWS3802A Valve Cable SWS3821 Valve Cable SWS3822 Valve Cable
98	Ionization smoke detectors are provided at ceiling level, and line heat detectors are located in cable trays. An automatic deluge system is provided that activates on cross zone detection of any heat and smoke detector.	DC-RS1 ALT Power Supply DC-RS1ALTAC ALT Power Supply DC-RS2 ALT Power Supply DC-RS2ALTAC ALT Power Supply

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
		DC-RS3ALT Power Supply
		DC-RS3ALTAC ALT Power Supply
		DC-RS4 ALT Power Supply
		DC-RS4ALTAC ALT Power Supply
		EFW00Y2B Valve Cable
		EFW0P7AX Pump Cable
		EFW2620B Valve Cable
		EFW2626B Valve Cable
		EFW2627A Valve Cable
		EFW2670A Valve Cable
		EFW0P7BA Pump Cable
		HPI1227B Valve Cable
		HPI1228B Valve Cable
		HPIP36AA Pump Cable
		HPIP36BA Pump Cable
		HPIP36CB Pump Cable
		LPI0P34B Pump Cable
		LPI1400B Valve Cable
		LPI1408B Valve Cable
		LPR1406B Valve Cable
		RBI2400B Valve Cable
		RBIP35BB Pump Cable
		SWS0P4BA Pump Cable
		SWS0P4CB Pump Cable
		SWS3640B Valve Cable
		SWS3642 Valve Cable
		SWS3642B Valve Cable
		SWS3643A Valve Cable
		SWS3802 Valve Cable
		SWS3806A Valve Cable
		SWS3807B Valve Cable
		SWS3810B Valve Cable
		SWS3821 Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
105	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves.	EFW00Y1A Valve Cable EFW0P7AX Pump Cable EFW2620B Valve Cable EFW2627A Valve Cable LPI1401A Valve Cable SWS3803 Valve Cable SWS3806A Valve Cable SWS3808A Valve Cable
112	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves.	EFW0P7AX Pump Cable EFW2626B Valve Cable EFW2670A Valve Cable HPI1227B Valve Cable HPI1228B Valve Cable LPI1400B Valve Cable LPI1408B Valve Cable LPR1406B Valve Cable RBI2400B Valve Cable SWS3640B Valve Cable SWS3642B Valve Cable SWS3802 Valve Cable SWS3807B Valve Cable SWS3810B Valve Cable SWS3821 Valve Cable
128	Temperature monitors are installed in ventilation charcoal filters. A wet pipe sprinkler system is installed.	EFW2626B Valve Cable EFW2670A Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
129	Smoke detectors are located in the ceiling area, safety related control cabinets, and in air ducts. Thermal detectors are located in the air filter housings. An automatic Halon system is installed in the ceiling and in the raised floor. The activation of the Halon is by cross zoned smoke and thermal detection.	EFW00Y1A Valve Cable
		EFW00Y2B Valve Cable
		EFW0P7AX Pump Cable
		EFW2620B Valve Cable
		EFW2626B Valve Cable
		EFW2627A Valve Cable
		EFW2670A Valve Cable
		EFW0P7BA Pump Cable
		HPI1219A Valve Cable
		HPI1220A Valve Cable
		HPI1227B Valve Cable
		HPI1228B Valve Cable
		HPIP36AA Pump Cable
		HPIP36BA Pump Cable
		HPIP36CB Pump Cable
		LPI0P34A Pump Cable
		LPI0P34B Pump Cable
		LPI1400B Valve Cable
		LPI1401A Valve Cable
		LPI1407A Valve Cable
		LPI1408B Valve Cable
		LPR1405A Valve Cable
		LPR1406B Valve Cable
		RBI2400B Valve Cable
		RBI2401A Valve Cable
		RBIP35AA Pump Cable
		RBIP35BB Pump Cable
		SWSOP4BA Pump Cable
		SWSOP4CB Pump Cable
		SWS3640B Valve Cable
		SWS3642B Valve Cable
		SWS3643A Valve Cable
		SWS3802 Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Continued)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
		SWS3803 Valve Cable SWS3806A Valve Cable SWS3807B Valve Cable SWS3808A Valve Cable SWS3810B Valve Cable SWS3820A Valve Cable SWS3821 Valve Cable SWS3822 Valve Cable
144	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves.	EFW00Y1A Valve Cable EFW0P7AX Pump Cable
149	This zone is partially protected by a preaction sprinkler system. Smoke detectors actuate the sprinkler's water supply valves. Additionally, the hot tool room and decontamination room are protected by a wet pipe sprinkler system.	B61-480VAC Power Panel B62-480VAC Power Panel B63-480VAC Power Panel EFW00Y2B Valve Cable EFW2626B Valve Cable EFW2670A Valve Cable HPI1227B Valve Cable HPI1228B Valve Cable LPI1400B Valve Cable LPI1408B Valve Cable LPR1406B Valve Cable RBI2400B Valve Cable SWS3640B Valve Cable SWS3642B Valve Cable SWS3802 Valve Cable SWS 3807B Valve Cable SWS3810B Valve Cable SWS3821 Valve Cable

Table 3.2
Fire Protection Systems
and Safe Shutdown Equipment by Fire Zone (Concluded)

Fire Zone	Suppression Equipment	Safe Shutdown Equipment
163	This zone is protected by a wet pipe sprinkler system.	EFW00Y2B Valve Cable
197	Oil bearing systems and equipment are protected by wet pipe sprinkler or by deluge systems. Automatic wet pipe sprinklers are in the machine shop, the breathing air equipment room, sample analyzer room, and the radiochemistry room. Turbine bearings and the exciter housing have a CO_2 suppression system.	A1-4160VAC Distance Panel A2-4160VAC Distance Panel EFW00Y1A Valve Cable EFW0P7AX Pump Cable EFW2620B Valve Cable EFW2627A Valve Cable EFW0P7BA Pump Cable HPIP36BA Pump Cable LPI1401A Valve Cable LPR1405A Valve Cable SWS3820A Valve Cable

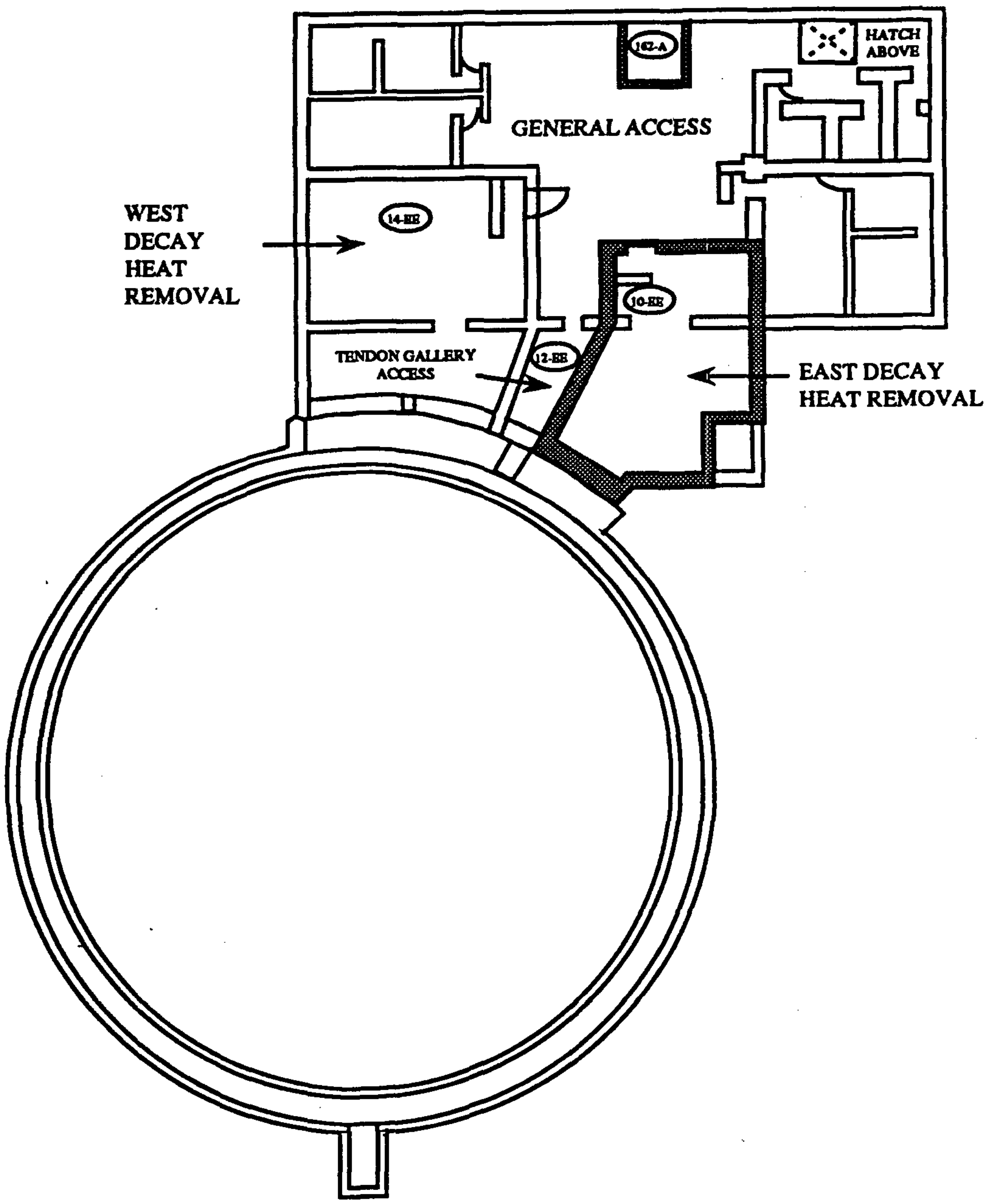

FIGURE 3.1 FIRE ZONES ON ELEVATION 317'

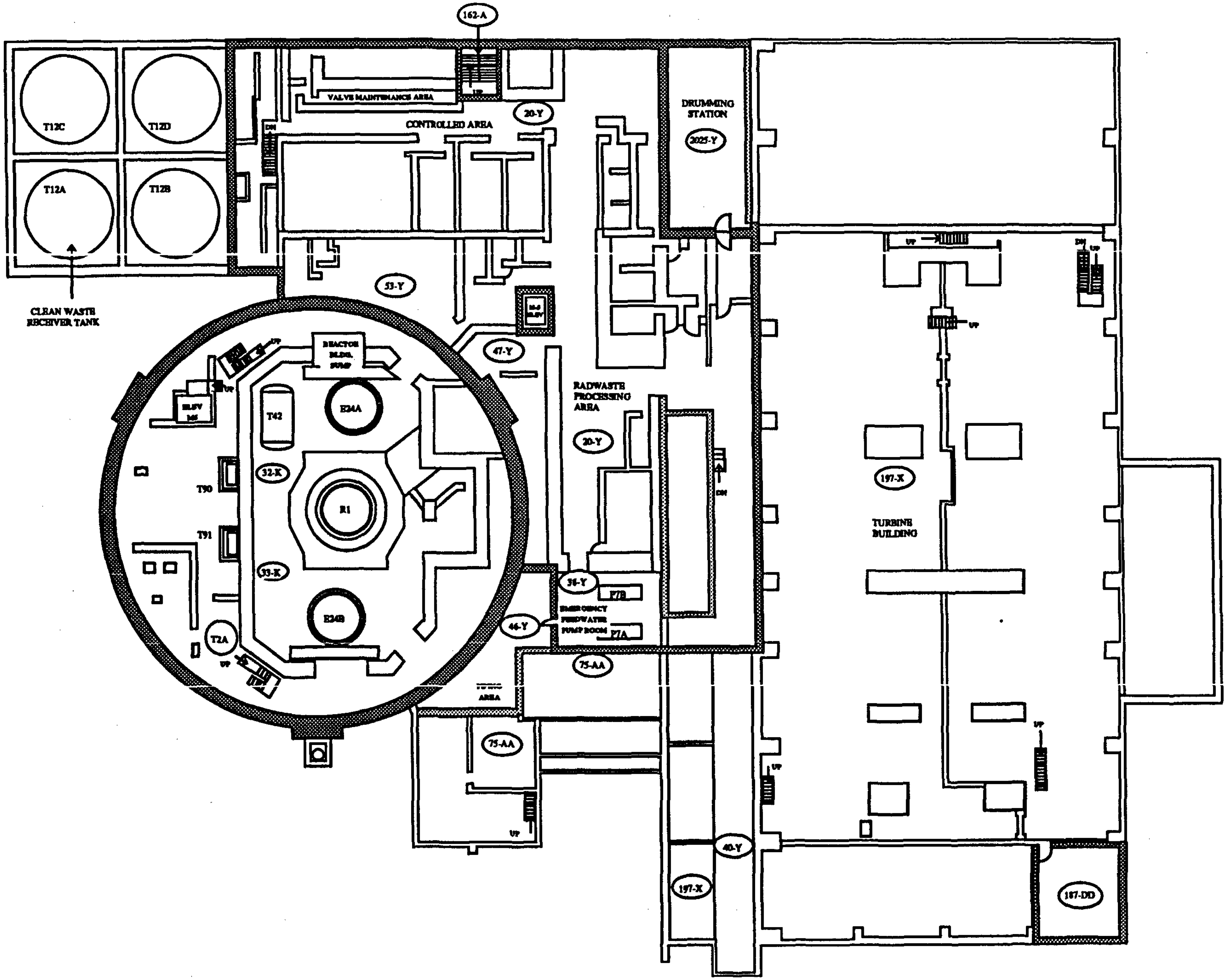

FIGURE 3.2 FIRE ZONES ON ELEVATION 335'

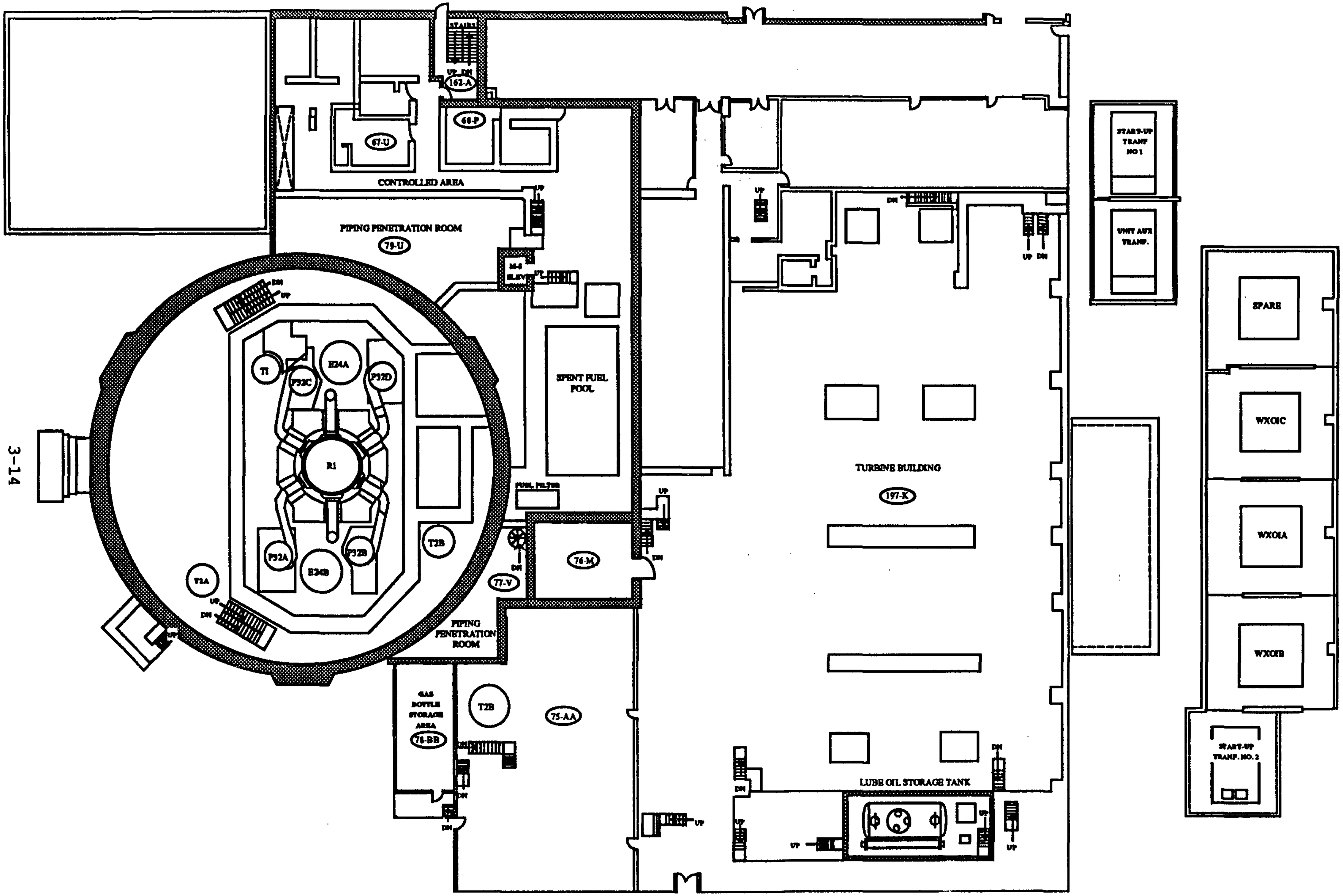

FIGURE 3.3 FIRE ZONES ON ELEVATION 354'

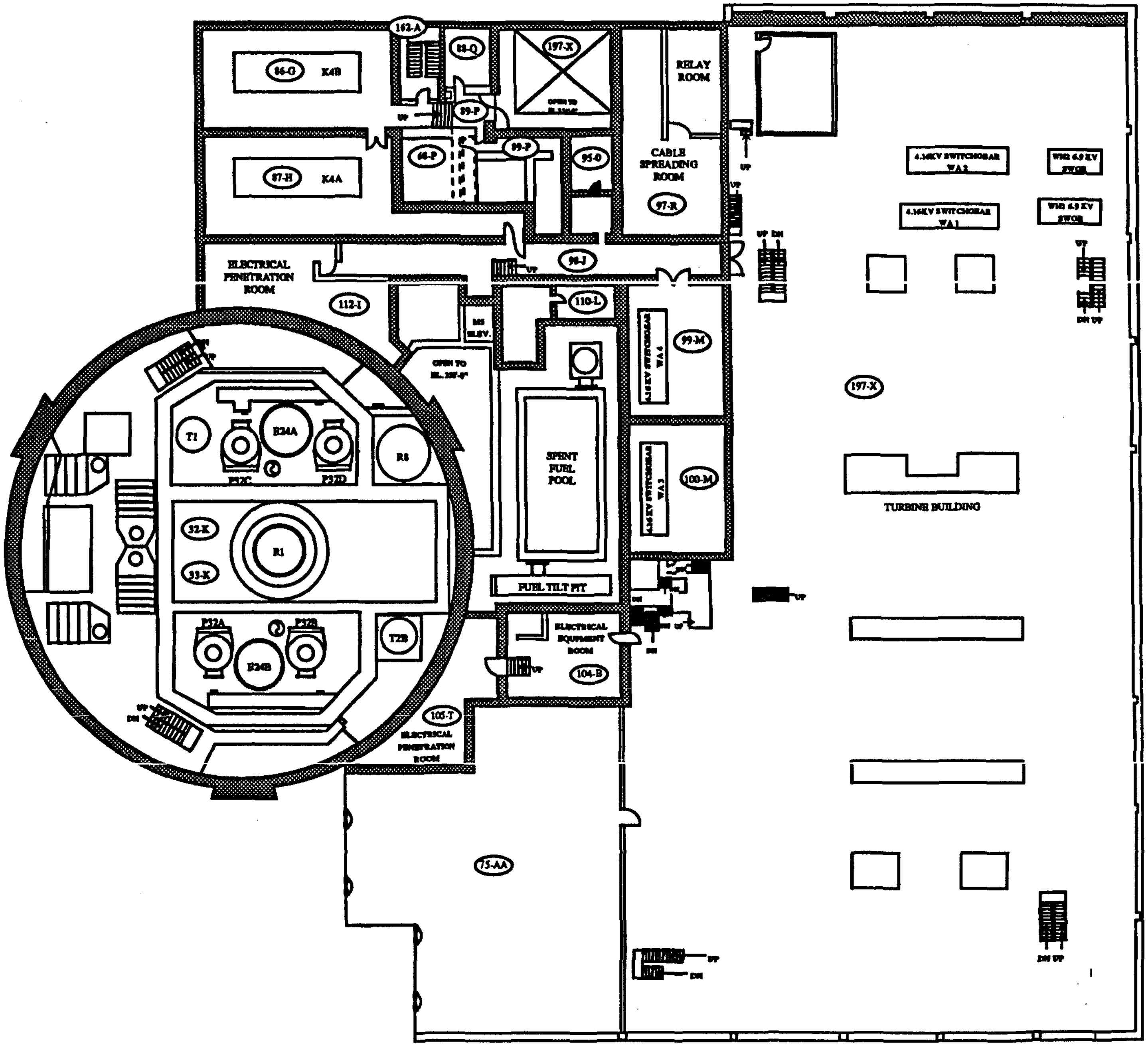

FIGURE 3.4 FIRE ZONES ON ELEVATION 368'/374'

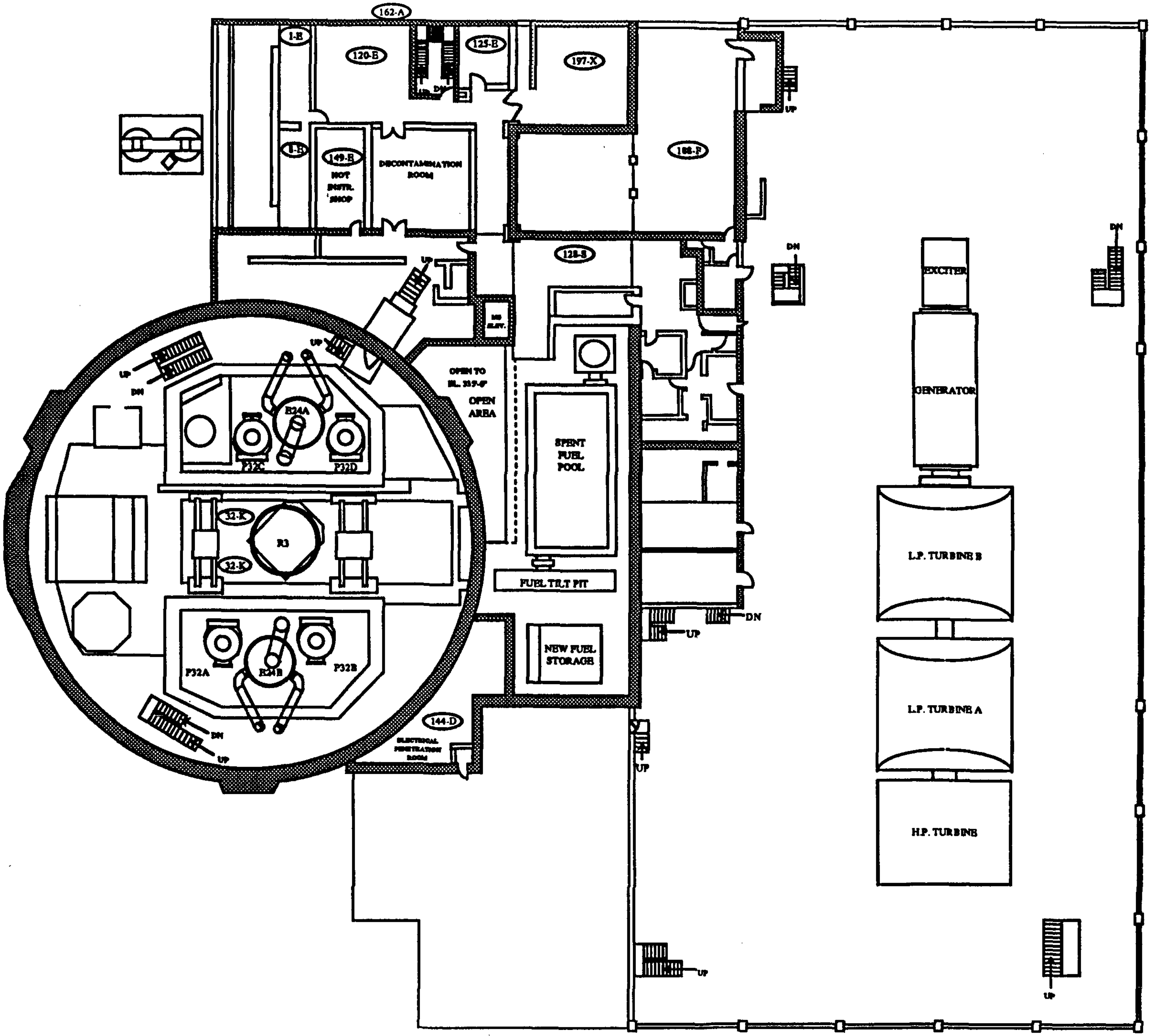

FIGURE 3.5 FIRE ZONES ON ELEVATION 386'

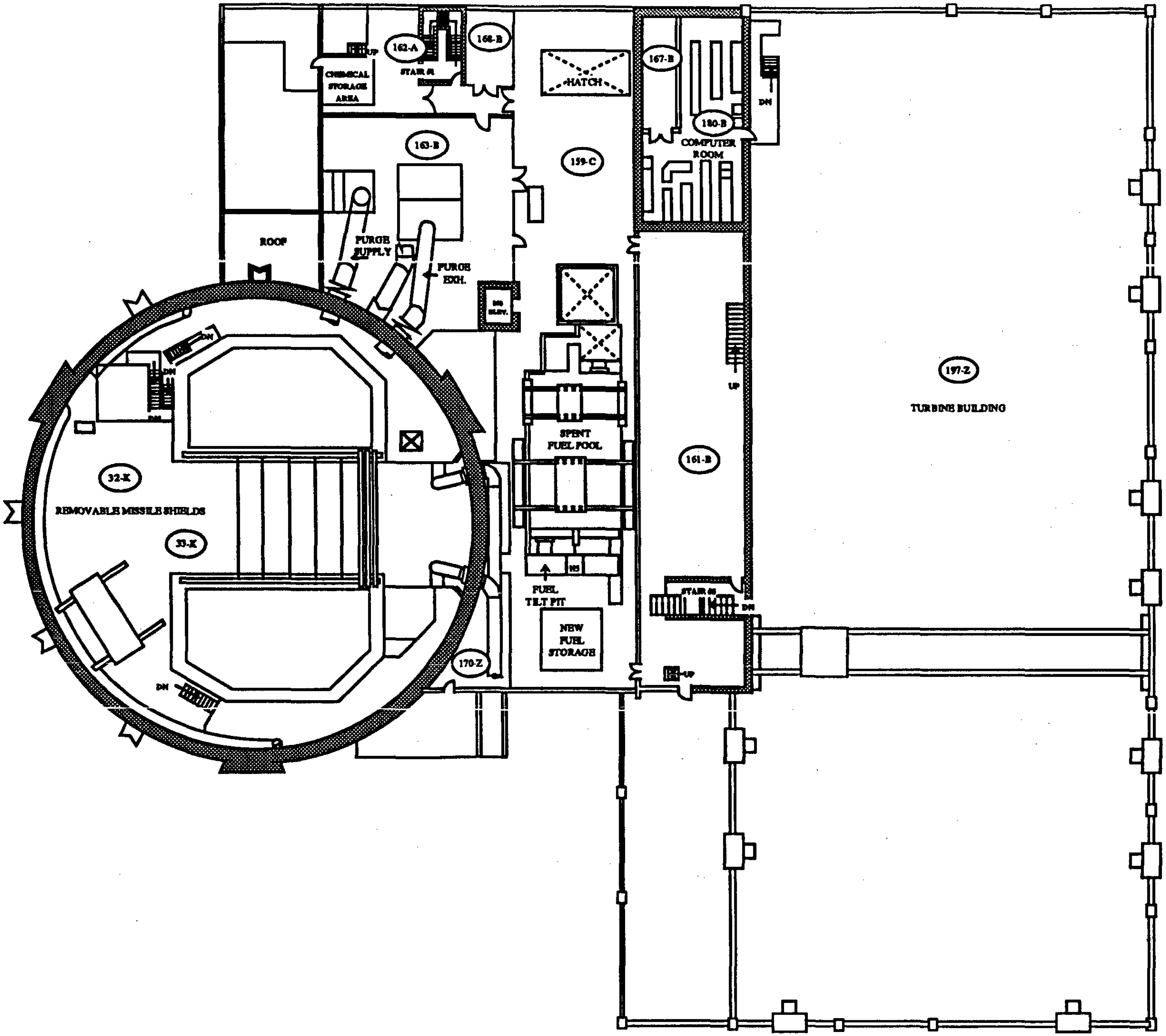

FIGURE 3.6 FIRE ZONES ON ELEVATION 404'

shutdown equipment and system models. These reports also described safety-significant recovery actions from random failures (those failures elsewhere in the plant not caused by FPS actuation or fires). These recovery actions were then analyzed for the possibility that FPS actuations could prevent them from being performed (Root Cause 2). Generic fire data (Ref. 3.7) developed in support of NUREG-1150 fire analyses provided frequencies of fires in the different areas and probabilities of Fire Zone barrier failures (smoke and heat spread).

A detailed analysis of the plant ventilation systems was performed. This analysis included a thorough review of system descriptions as well as ventilation drawings. Once this review was completed, a plant walkdown was performed to verify the review and also clear up remaining questions that resulted from the review process. For most plant areas, smoke detectors are used for indication purposes only. In those plant areas with preaction suppression systems, smoke detector signals are used to actuate valves and pressurize the piping. In Fire Zones 97 and 98, smoke detectors provide one of two signals required to actuate the deluge system. Consequently, smoke detector actuation alone will not lead to FPS agent release. Therefore, Root Causes 1, 3, 7, and 10 (fire-induced actuation due to smoke spread, fire-induced FPS actuation preventing fire-fighting access, FPS actuation due to dust in a seismic event, and external plant fires) were screened from further analyses.

It is important to note that it is often necessary to ratio the overall building fire occurrence frequency down to reflect the fact that fires in only a small subset of the building can spread smoke and heat to adjacent areas. This is called "partitioning" and is based upon analyst judgement and sensitivity calculations using a fire growth computer code. Partitioning of fire frequencies was performed for all applicable Fire Zones.

3.2.1 General Transients Caused by FPS Actuation or Fires

Using the system and event sequence models obtained from the internal events PRA (Ref. 3.1), a vital area analysis was performed incorporating critical equipment location information as well as random failures. Based on the internal events PRA, five general transient sequences that lead to core damage were identified. The general transient event tree from which these are taken is shown in Figure 3.7. No LOSP transient or pipe break LOCA caused directly by an FPS actuation or a fire alone was considered to be credible. Table 3.3 summarizes the transient sequences analyzed.

Sequence 1 is a transient with successful manual or automatic scram in which the PCS system is failed and a stuck-open relief valve leads to a S_2 LOCA (small LOCA). Sequences 2 and 4 are also stuck-open relief valve (S_2) LOCAs in which either no other systems are failed (Sequence 2) or the PCS and EFW systems fail (Sequence 4). Sequence 3 is a transient with failure of the PCS, EFW, and HPI systems. Finally, in Sequence 5 core damage occurs due to failure of the PCS and EFW systems with failure to perform feed and bleed (due to failure of relief valves to open).

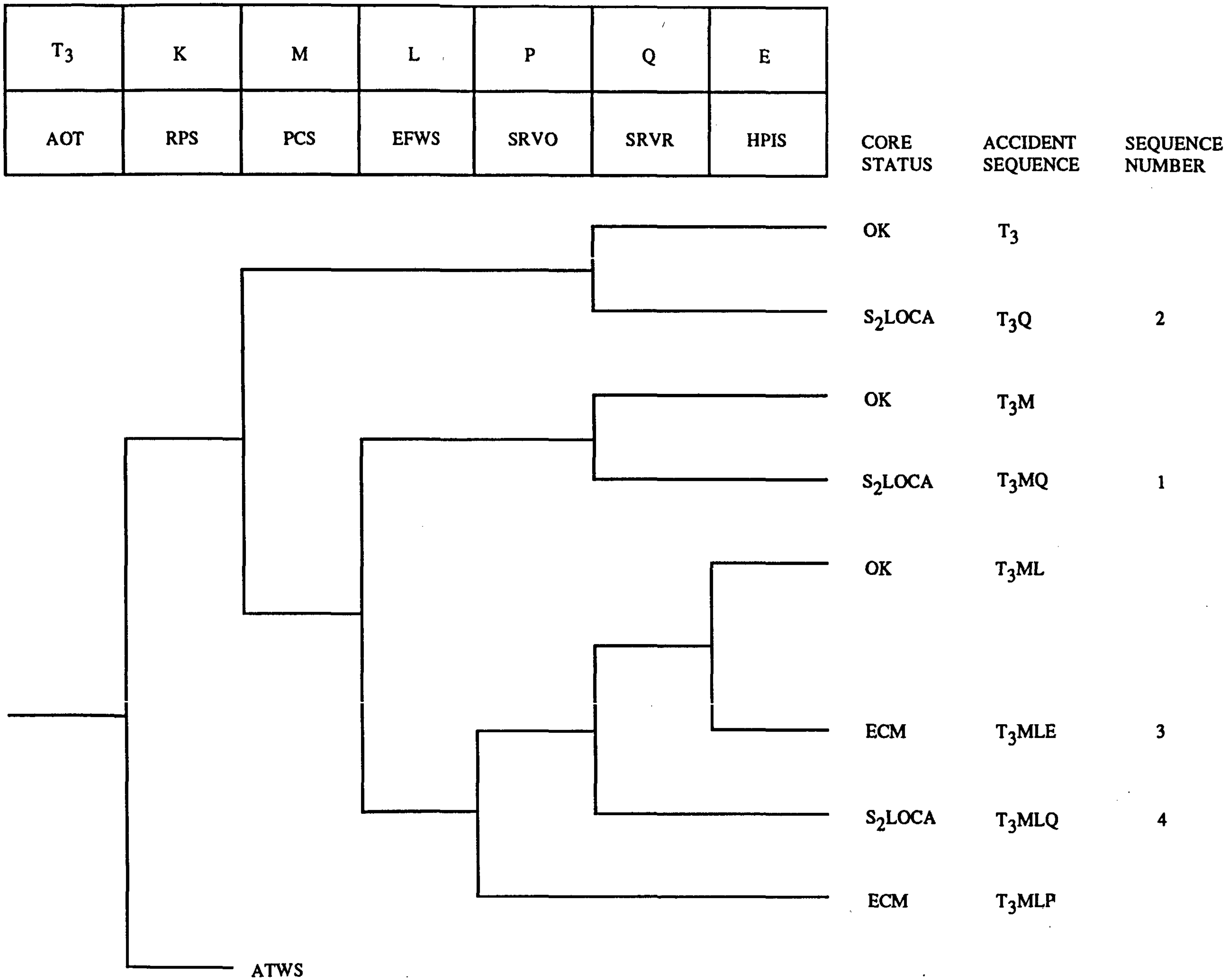

FIGURE 3.7 TRANSIENT (T_3) EVENT TREE

Table 3.3
General Transient Accident Sequences Analyzed

Sequence 1	T_3MQ (Stuck-open relief valve)
Sequence 2	T_3Q (Stuck-open relief valve)
Sequence 3	T_3MLE
Sequence 4	T_3MLQ (Stuck-open relief valve)
Sequence 5	T_3MLP
Sequence 6	T_3DI (Seal LOCA)

Safety System Nomenclature

T3: Initiating event, transient with the Power Conversion System initially available.

M: Continued success or failure of 1 of 2 trains of the Power Conversion System (PCS) system.

L: Success or failure of 1 of 2 trains of the Emergency Feedwater System where one is a motor-driven and the other is a turbine-driven pump.

P: Success or failure of 1 of 2 safety valves, assuming that the electromatic relief valve is blocked.

Q: Success or failure to reclose, of 1 of 2 safety valves that is assumed to have opened.

E: Successes or failure of 1 of 3 safety injection pump trains. In this part of the fault tree, operator initiating action is required.

I: Failure of Intermediate Cooling

In addition to these sequences, a seal LOCA was also considered. For a seal LOCA to occur, both the HPI and the intermediate cooling water systems have to fail. Since the intermediate cooling water system was never modeled in Reference 3.1, a simple model was constructed taking into account system dependencies of service water and AC electrical power.

These six sequences were analyzed for their applicability to the FPS actuation root cause scenarios using the criteria described in Section 3 of NUREG/CR-5580[a]. These criteria were applied to each cut set in the vital area analysis. In this process, several sequences were screened from further consideration. Also, all sequences were screened out except for Sequence 3 (T_3MLE). At this point in the screening analysis only the probability of random failures and operator recovery was considered. Cut sets were truncated on these two considerations at 10^{-4}. The sequences and cut sets that remained were grouped according to the thirteen root causes described in NUREG/CR-5580[a].

After the vital area analysis was completed for the six general transient sequences, the following seven plant areas remained which required a more detailed analysis:

a. Fire Zone 20 - Radwaste Processing Area

b. Fire Zone 38 - Emergency Feedwater Pump Area

c. Fire Zone 73 - Condensate Demineralizer Area

d. Fire Zone 97 - Cable Spreading Room

e. Fire Zone 98 - Uncontrolled Access Area

f. Fire Zone 129 - Control Room

g. Fire Zone 197 - Turbine Building

Some of the surviving cutsets involved damage due to FPS actuations in multiple fire zones. During the plant walkdown, any scenarios that required FPS or fire-related failures in more than one area had their physical barriers inspected. No barrier deficiencies were noted. When a generic screening barrier failure probability of 0.1 was applied, all combinations of these adjacent areas fell below the truncation probability.

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a. J. A. Lambright et al, Evaluation of Generic Issue 57: Effects of Fire Protection Systems on Safety Related Equipment. Root Cause Development and Summary Report, NUREG/CR-5580, September, 1992.

Five of the remaining seven areas could be screened based on the physical location of the critical equipment with respect to the FPS. In Fire Zone 20, the HPI pumps were in a separate room from the sprinkler system protecting the makeup tank room. In the emergency feedwater pump area (Fire Zone 38), only the turbine driven pump has any possibility of being failed by the sprinkler system. The motor-driven pump was protected from water spray. All three HPI pump cables were found to be located in Fire Zone 73. However, it was also found during the plant walkdown that cables for pumps A and C were embedded in the concrete floor. Therefore, in each of these fire areas some or all of the critical equipment (which if failed by FPS agent release would lead to core damage) could not be affected by FPS actuations.

The control room (Fire Zone 129) has its Halon system subdivided into three essentially independent systems. At least two of these systems would have to randomly actuate to expose the critical cabling. The frequency of two independent random actuations is below the screening cutoff probability. Also, these systems actuate only into the subfloor and enclosed ceiling areas. Leak tests have been performed by utility personnel in the manned areas and have found no more than 1% Halon concentration. Consequently, control room abandonment in the event of Halon release was eliminated from further consideration.

In the Turbine Building (Fire Zone 197) the CO_2 system has just a few nozzles which spray directly above the turbine bearings and nowhere else. This area is of sufficient size that CO_2 concentration anywhere else than in the turbine bearing area will be negligible. Therefore, the turbine building could also be eliminated from further consideration.

Another screening criterion was also used. In the case of Fire Zone 98, Uncontrolled Access Area, even though cable for five critical pumps was in the same area as sprinkler heads, it was found that four were enclosed in conduit. Therefore, the chance for all four to even be exposed to spray was considered to be negligible.

3.2.2 LOSP Transients Due To Seismic Events

The loss of offsite power event tree (T_1) used for this study is given in Figure 3.8. A total of four sequences leading to core damage are shown on this tree, and these sequences are listed in Table 3.4.

As was the case for the general transient sequences, an additional (seal LOCA) sequence was added which was not part of the original PRA. Sequence 1 is a loss of offsite power with failure of the power conversion system and a stuck-open relief valve leading to a small (S_2) LOCA. Sequence 2 is a transient with loss of decay heat removal due to failure of the PCS and EFW systems and failure of feed and bleed (HPI system failure). Sequence 3 is identical to Sequence 1 with the exception of the additional failure of the EFW system. Finally, Sequence 4 is identical to Sequence 2 with the exception that feed and bleed is failed by relief valves not opening.

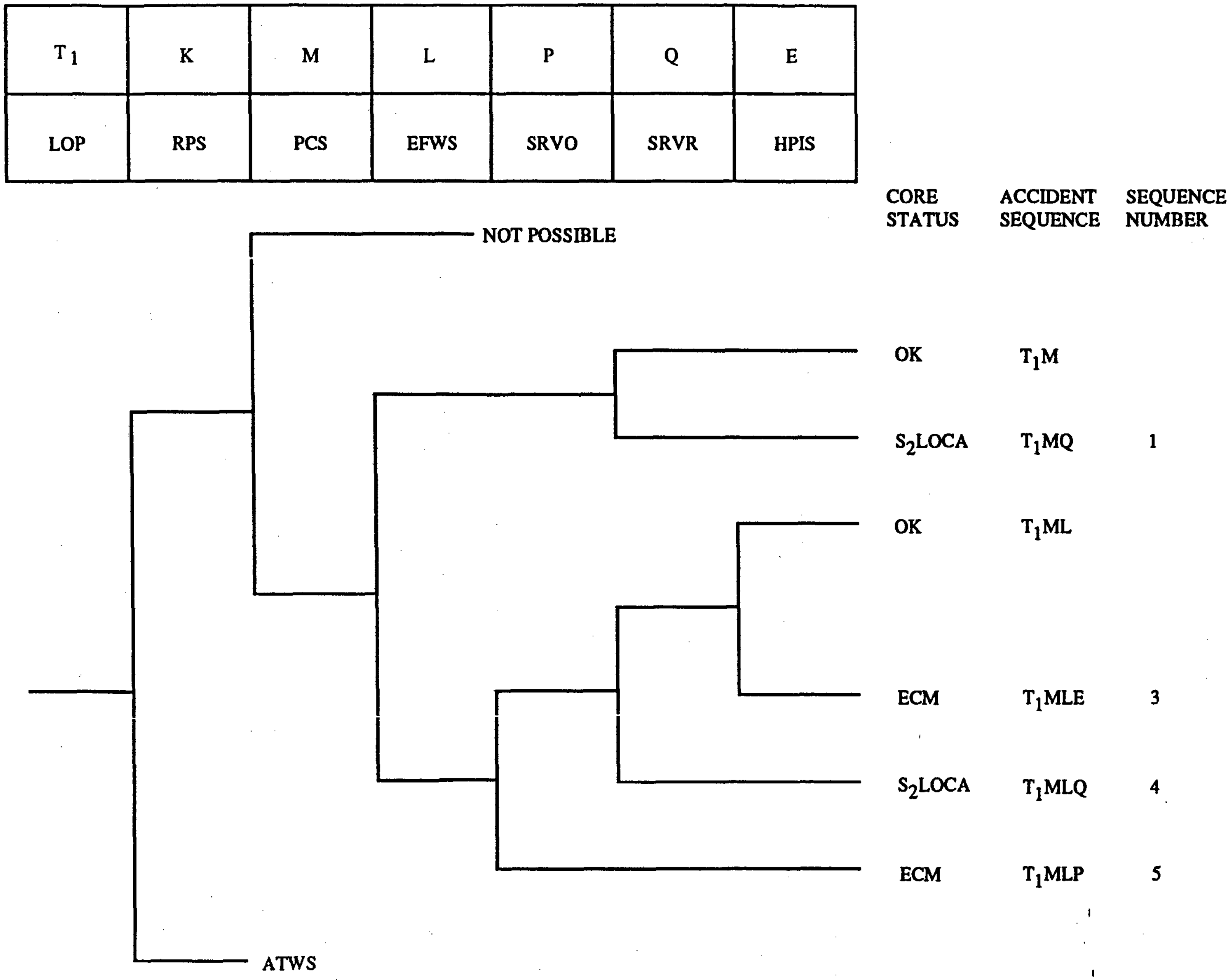

FIGURE 3.8 LOSS OF OFFSITE POWER TRANSIENT (T_1) EVENT TREE

Table 3.4

Loss of Offsite Power Transient Sequences

Sequence 1	T_1MQ	(Stuck-open relief valve)
Sequence 2	T_1MLE	
Sequence 3	T_1MLQ	(Stuck-open relief valve)
Sequence 4	T_1MLP	
Sequence 5	T_1DI	(Seal LOCA)

Refer to Table 3.3 for event descriptions.

In the vital area analysis process it was found that all sequences except for Sequence 2 (T_1MLE) were screened out based on truncation (10^{-4}) of random failure probabilities and allowance for operator recovery. The following plant fire areas remained:

a. Fire Zone 79 - Upper North Piping Penetration Area

b. Fire Zone 86/87 - Diesel Generator Rooms

c. Fire Zone 97 - Cable Spreading Room

d. Fire Zone 98 - Uncontrolled Access Area

e. Fire Zone 105 - Lower South Electrical Penetration Room

f. Fire Zone 112 - Lower North Electrical Penetration Room

Of these six, one could be screened altogether (FZ 98) because critical cables were located in conduit and not susceptible to sprinkler spray and no seismic/fire sources were located in close proximity to the critical cabling.

3.2.3 Quantification

As was noted in Section 3.2.2 one Fire Zone survived the screening process for the non-seismic root causes and five areas required detailed seismic analysis. The following subsections will discuss the quantification process for these zones.

3.2.3.1 Quantification of Random and Fire-Induced Actuation Scenarios

The occurrence of a random FPS actuation or an actuation in the presence of a fire in a nuclear power plant can result in a plant transient caused either by the operator manually tripping the plant or the plant automatically tripping as a result of the actuation itself. The purpose of this study is to quantify the impact on risk of inadvertent and advertent actuations of the FPS. The values chosen for the various parameters utilized in the calculation of the core damage frequency are best estimate values based on historical data. When little data existed, best estimate probability assignments were made based on plant walkdowns and engineering judgement. The specific equations utilized in the calculation of the core damage frequency contribution from each root cause can be found in Section 3 of NUREG/CR-5580[b]. Table 3.5 summarized the fire frequencies used for each Fire Zone. The fire frequencies were taken from Reference 3.7. Note that it is often necessary to partition the overall building fire occurrence frequency down to reflect the fact that fires in only a small subset of the building can cause the postulated heat or smoke spread to adjacent areas. This is called "partitioning" and is based on both analyst judgment and sensitivity calculations using a fire growth computer code (Refs. 3.2 and 3.3). For this study, partitioning of the fire frequencies for the larger Fire Zones was performed wherever applicable. For example, in the Uncontrolled Access Area this reduced the fire frequency by over an order of magnitude because the sources were clustered in a area less that 10% of the size of the whole zone.

3.2.3.2 Quantification of Seismically-Induced FPS Actuations

A site-specific seismic analysis was performed on the FPSs for the plant analyzed in this report. When a seismic event occurs, a loss of offsite power is highly likely due to the failure of ceramic insulators in the switchyard. Thus, the seismic sequences which must be considered are those where offsite power is assumed to be lost. Once the vital area analysis has been performed for the LOSP sequences, one can quantify them in a similar fashion as was done for the random and fire-induced FPS actuation scenarios. The one significant difference is that the accident sequences evaluated are conditional on the plant site seismic hazard curve (a function of peak ground acceleration). As such, the sequences must be integrated over the seismic hazard curve. For the base case analysis of the seismic sequences the Lawrence Livermore National Laboratory (LLNL) hazard curves were utilized. In Chapter 4, a sensitivity study was performed comparing the CDF contribution from the seismic root causes utilizing the LLNL (Ref. 3.8) and the Electric Power Research Institute (EPRI) (Ref. 3.9) hazard curves.

At the plant in question many safety related areas are protected by a water fire protection system. The source of this water is from pumps located inside the intake structure. There are three pumps located there: an electric motor driven pump, a diesel pump, and an electrical motor driven jockeypump (a small capacity pump that keeps the fire main pressurized during periods of low usage).

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b. J. A. Lambricht et al, Evaluation of Generic Issue 57: Effects of Fire Protection Systems on Safety Related Equipment. Root Cause Development and Summary Report, NUREG/CR-5580, September, 1992.

Table 3.5

Fire Frequencies Corresponding to Plant Fire Zones

Fire Zone	Fire Frequency
Upper North Piping Penetration Room (Fire Zone 79)	1.2E-3
Diesel Generator Rooms (Fire Zones 86 and 87)	2.3E-2
Cable Spreading Room (Fire Zone 97)	2.7E-3
Access Area (Fire Zone 98)	5.9E-4
Lower South Electrical Penetration Room (Fire Zone 105)	6.4E-4
Lower North Electrical Penetration Room (Fire Zone 112)	1.1E-3
Control Room (Fire Zone 129)	4.4E-4*

*Reduced by one order of magnitude to account for quick suppression of a fire in a continually occupied area.

If offsite power is lost, both electric pumps would no longer operate since they are powered from non-vital busses. Therefore, the diesel pump must operate in order to maintain system pressure.

After carefully examining the water FPS during the initial plant walkdown, it is apparent that the principal vulnerability of this system is the batteries used to start the diesel pump. There are a total of four batteries on a two level battery rack. These batteries are not anchored to the rack and there is no support on the ends of each level of the rack. As a result, during an earthquake these batteries have a high likelihood of sliding until the cable connecting the terminals breaks or until a battery falls off the ends of the rack.

In order to determine a fragility for diversion (loss of pressure) of water for the FPS, an analysis of the battery rack was performed. It was determined that if the batteries were to slip about two inches the cable connecting the terminals could disconnect. In order to estimate what size earthquake could cause that much slippage, the intake structure and the battery rack were modeled using dynamic analysis. Using the method developed by Newmark (Ref. 3.10), a rough estimate of the slippage of the batteries could be determined using peak accelerations and velocities obtained from the dynamic analysis. The resulting median fragility is 0.6g for failure of the batteries. In addition, the response ratio of the battery rack to peak ground acceleration (PGA) is a factor of 2.0. This means that an earthquake PGA of 0.3g or 1.5 times the SSE results in a 50% chance of diversion of all FPS water.

The COMPBRN fire growth code (Ref. 3.2) was used to calculate fire propagation and equipment damage for Root Cause 12 (seismic/fire interaction). COMPBRN was developed specifically for use in nuclear power plant fire PRAs. The code calculates the time required to damage critical equipment given that a fire has started. This failure time is then used in conjunction with plant specific information on fire suppression to obtain the probability that a given fire will cause equipment failure which leads to core damage before the fire can be suppressed. The latest version of the code, COMPBRN III (Ref. 3.3), with some additional modifications (Ref. 3.4), was used for the calculations.

For the plant analyzed, only two areas required detailed fire propagation modeling with COMPBRN. All the other critical Fire Zones were screened from COMPBRN modeling based on plant walkdown findings. These zones were screened because either the zone contained no seismic/fire sources, or because the seismic/fire source found would fail the critical function directly. The two areas which were analyzed with fire propagation modeling were Fire Zones 98 and 105.

Fire Zone 98, located on the 368′ elevation, is the corridor adjacent to the Cable Spreading Room and as such there are many critical cables routed through this Fire Zone. The fire scenario in this Fire Zone requires damage to five critical cables. The fire which was modeled with COMPBRN is a result of the tipping or sliding failure of energized cabinet D02 (a DC distribution panel) during a seismic event. This cabinet is located in a room with an open doorway off the main corridor leading to the battery room. The size of the room (10′ x 12′ x 11′) indicated the need for modeling the hot gas layer. However, based on COMPBRN results and engineering judgement it appears that even a large fire in this room will not lead to damage of safety-related cabling in the adjacent corridor. Although there would be heat and smoke spread, the safety-related cables would not experience a significant enough temperature increase to cause damage.

Fire Zone 105, located on the 368′ elevation, was also modeled using COMPBRN. In this Fire Zone, the fire source is the tipping or sliding failure of cabinet SCR1 during a seismic event. In this fire scenario two cables must be damaged; one in a cable tray approximately three feet from the fire source and one in conduit approximately twelve feet away from the fire. However, between the cable in conduit and the fire source there are intervening combustibles in the form of stacked cable trays approximately six feet away. The COMPBRN results predict that damage to the cable located three feet from the fire occurs in approximately 240 seconds. The intervening combustibles are damaged and ignite in 480 seconds. Damage to the cable in conduit is also indicated at the 480 second time step.

The following is a description of the seismic analysis for the remaining Fire Zones.

Fire Zone 79-Upper North Piping Penetration Area

Fire Zone 79 is protected by a preaction water sprinkler system. Based on the plant walkdown, Root Cause 12 was found to be the only applicable seismic scenario. This is due to the high likelihood of loss of pressure in the fire main (as described earlier) and because seismic/fire sources are present. The fire sources are a hydrogen analyzer cabinet and a lighting transformer box. It was determined that the lighting transformer would probably tip before the hydrogen analyzer since it is not seismically anchored. A median fragility of 0.6g was chosen for tipping of the transformer. It was assumed this would cause a fire about 20% of the time due to the low likelihood of the non-safety transformer remaining energized.

The Root Cause 12 scenario in Fire Zone 79 is as follows: an earthquake, tipping of the lighting transformer causing a fire, loss of fire main pressure, fire damage to the safety related cables within the area, and additional random failures that occur elsewhere in the plant.

Fire Zones 86 and 87-Diesel Generator Rooms

Fire Zones 86 and 87 are protected by preaction water sprinkler systems. In order to pressurize either system, both smoke detector and flame detector signals must be present. Once the system is pressurized, heat is required in order to open a fusible link sprinkler head.

Root Causes 7, 8, and 9 are all involved in the accident sequence postulated for the Diesel Generator Rooms. Root Cause 7 (smoke detector FPS actuation in seismic events) is necessary to activate the smoke detectors. This is considered highly probable during an earthquake since dust is often stirred up and enters smoke detectors causing actuation. A flame detector signal is generated due to relay chatter (Root Cause 8). The specific relays used were identified. They were manufactured by Aromat and are seismically tested to 6.0g from 10 to 55 Hz. Since no testing was performed below 10Hz, a median fragility of 4.0g was used. Root Cause 9 (FPS actuations due to seismic failures of FPS) involves breaking the FPS piping or heads and spraying water into the Diesel Generator Rooms and damaging equipment. A median fragility of 0.85g was used and was developed from data taken on FPS failures of similar systems during the Loma Prieta earthquake. Two components were found that, if sprayed, could lead directly to diesel failure. These components are the diesel excitor cabinets and the diesel control panels. Also it must be noted that in only about 25% of the sprinkler system, agent release can spray either of the two cabinets. The probability of water damaging an energized cabinet was determined from the LER database and resulted in a mean conditional probability of 0.27.

In order to fail both diesel generators the following must occur: an earthquake, actuation of the smoke detectors by dust, generation of a flame detector signal due to relay chatter, and either pipe break or sprinkler head failure. In addition, the fire main must not lose pressure.

Fire Zone 97-Cable Spreading Room

Fire Zone 97 is protected by an open head deluge water sprinkler system. In order to open the deluge valve, a smoke detector and a heat sensitive Protectowire must both activate.

Root Causes 7 and 8 are both involved in the accident sequence postulated for this zone. Root Cause 7 involves dust activating the smoke detectors. Root Cause 8 involves relay chatter activating the Protectowire control system. The relays used in the Protectowire system were made by GE and were not seismically tested. A fragility estimate, based on similar relays, was assigned a median capacity of approximately 2.0g. As was the case for the Diesel Generator Rooms, the fire main must not lose pressure. The critical equipment in the area which must be damaged is solely cabling. A mean conditional probability of cable damage of 5.0E-3 was taken from the LER data base (Ref. 3.7). As was the case for the non-seismic root cause involving the cable spreading room, credit was given for operator recovery from the remote shutdown panel.

Fire Zone 105-Lower South Electrical Penetration Room

Fire Zone 105 is protected by a preaction sprinkler system. Root Cause 12 was found to be the only applicable seismic scenario. The fire source of concern is the pressurizer heater panel SCR1. A median fragility for the pressurizer heater panel was found to be 0.8g. Fire in this zone results in loss of speed control of the turbine driven emergency feedwater pump. Procedures exist to take manual control of the pump in a transient sequence. Manual control must be taken in about 20 minutes to avoid early core damage. For the base case, a non-recovery probability of 1.0 was assigned. Recovery is considered in Chapter 4 as a sensitivity study.

The Root Cause 12 scenario in Fire Zone 105 is; an earthquake, tipping of the pressurizer heater panel causing a fire, loss of fire main pressure, fire damage to the safety related cables in the zone, and additional random failures that occur elsewhere in the plant.

Fire Zone 112-Lower North Electrical Penetration Room

Fire Zone 112 is protected by a preaction water sprinkler system. As was the case for Fire Zone 79, Root Cause 12 was found to be the only applicable seismic scenario. The only difference is in the fire sources found. These fire sources are a seismic monitoring cabinet and a source range nuclear instrument cabinet. A median fragility of 0.8g was assigned for tipping of either cabinet.

3.3 Results of Quantification

The results of quantification for the fire and random failure induced root causes and seismically induced FPS actuations are presented in Tables 3.6 through 3.10. The results presented are mean values for their associated distributions.

Appendix A presents the uncertainty calculations as well as each cut set for the seismic and non-seismic root causes. Additionally, each basic event probability value is given. The details concerning the development of these probability assignments can be found in NUREG/CR-5580[c].

3.3.1 Root Cause 1-Fire Induced FPS Actuation Due to Smoke Spread

This root cause was screened on two bases. First, a design review of the plant layout and ventilation system, confirmed by plant walkdowns, showed that in the spaces of concern, the ventilation system supply and exhaust flowrates are matched such that there is no cross flow between spaces. Ventilation supply and exhaust are from outside air only. Second, this plant has no FPSs actuated solely on smoke. Using the heat and smoke propagation code results obtained in associated report NUREG/CR-5789[d] and a detailed inspection of fire barriers performed during the plant walkdown, all Root Cause 1 scenarios could be screened from further consideration.

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c. J. A. Lambright et al, Evaluation of Generic Issue 57: Effects of Fire Protection Systems on Safety Related Equipment. Root Cause Development and Summary Report, NUREG/CR-5580, (to be published).

d. J. A. Lambright et al, Risk Evaluation for a Westinghouse Pressurized Water Reactor Effects of Fire Protection System Actuation on Safety-Related Equipment (Evaluation of Generic Issue 57), NUREG/CR-5789, (to be published).

Table 3.6

Core Damage Frequencies for Fire Zone 79
(Upper North Piping Penetration Room) (Per Reactor Year)

Root Cause	Sequence 1	Sequence 2
1	---	---
2	---	---
3	---	---
4	---	---
5	---	---
6	---	---
7	---	---
8	---	---
9	---	---
10	---	---
11	---	---
12	---	5.3E-7
13	---	---
Totals		5.3E-7

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

Table 3.7

Core Damage Frequencies for Fire Zones 86/87
(Diesel Generator Rooms) (Per Reactor Year)

Root Cause	Sequence 1	Sequence 2
1	---	---
2	---	---
3	---	---
4	---	---
5	---	---
6	---	---
7	---	---
8	---	---
9	---	3.4E-8
10	---	---
11	---	---
12	---	---
13	---	---
Totals		3.4E-8

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

Table 3.8

Core Damage Frequencies for Fire Zone 97
(Cable Spreading Room) (Per Reactor Year)

Root Cause	Sequence 1	Sequence 2	Total
1	---	---	
2	---	---	
3	---	---	
4	2.3E-6	---	
5	---	---	
6	1.4E-6	---	
7	---	---	
8	---	1.5E-6	
9	---	---	
10	---	---	
11	6.4E-7	---	
12	---	---	
13	2.9E-6	---	
Totals	7.2E-6	1.5E-6	8.7E-6

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

Table 3.9

Core Damage Frequencies for Fire Zone 105
(Lower South Electrical Penetration Room) (Per Reactor Year)

Root Cause	Sequence 1	Sequence 2
1	---	---
2	---	---
3	---	---
4	---	---
5	---	---
6	---	---
7	---	---
8	---	---
9	---	---
10	---	---
11	---	---
12	---	4.8E-5
13	---	---
Totals		4.8E-5

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

Table 3.10

Core Damage Frequencies for Fire Zone 112
(Lower South Electrical Penetration Room) (Per Reactor Year)

Root Cause	Sequence 1	Sequence 2
1	---	---
2	---	---
3	---	---
4	---	---
5	---	---
6	---	---
7	---	---
8	---	---
9	---	---
10	---	---
11	---	---
12	---	4.3E-7
13	---	---
Totals		4.3E-7

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

3.3.2 Root Cause 2-Fire Induced FPS Actuation Preventing Recovery

For this root cause, all cut sets could be screened either because the random failures were not recoverable or there was no connectivity between the zone where the FPS action occurred and the zone where the recovery action took place. Therefore, this root cause was found not to be applicable.

The criteria for allowing credit for recovery for random failures was applied consistently with the internal events analysis (Ref. 3.1). Thus, for instance, if recovery was not allowed for a mechanical failure of a check valve, it was also not considered here. Most random failures were eliminated based on this criteria. Secondly, if recovery was allowed by the internal events analysis, a determination was made in which Fire Zone(s) the recovery action(s) occurred. For the recoverable random failures it was found that none occurred in Fire Zones where FPS actuation would either hinder the action or prevent access to the zone.

3.3.3 Root Cause 3-Fire Induced FPS Actuation Preventing Fire-Fighting Access

This root cause was found not to be applicable. It was found that none of the critical Fire Zones were accessible through only one other Fire Zone, i.e. every zone had either multiple paths for firefighter access, or the singe path would not be blocked by FPS actuation.

3.3.4 Root Cause 4-FPS Actuation Caused by Human Error

Here, an incremental increase in core damage frequency of 2.3E-6/yr was found. The dominant contributors were sequence T_3MLE and the Cable Spreading Room. Credit was given for operator recovery from the remote shutdown panel.

3.3.5 Root Cause 5-FPS Actuation Caused by Pipe Break

The one area that survived the initial screening analysis was the Cable Spreading Room. A plant walkdown revealed that no steam piping was located either in the room itself or in close proximity (within 50 feet) in adjacent plant areas. Therefore, this root cause was found not to be applicable.

3.3.6 Root Cause 6-FPS Actuation Caused by Hardware Failures in FPS

The incremental increase in core damage frequency for this root cause was found to be 1.4E-6/yr. It again arises primarily due to inadvertent FPS actuations in the Cable Spreading Room leading to sequence T_3MLE. As was the case for Root Cause 4, credit was given for operator recovery from the remote shutdown panel.

3.3.7 Root Cause 7-Dust Triggered FPS Actuations in Seismic Events.

As noted earlier, the plant under consideration does not utilize automatic fire protection systems which could be actuated by dust raised during a seismic event. (Certain Fire Zones do have either ionization or smoke detectors, but they are not used to solely actuate any of the fire protection systems). Hence, this root cause was not applicable for the plant under consideration.

3.3.8 Root Cause 8-Relay Chatter FPS Actuations in Seismic Events

The incremental increase in core damage frequency was found to be 1.5E-6/yr. This arises due to relay chatter in the Cable Spreading Room as was described in section 3.2.3.2. Failure of both the HPI and EFW systems coupled with a seismically-induced LOSP leads to sequence T_1MLE.

3.3.9 Root Cause 9-FPS Actuations Due To Seismic Failures of FPS

The incremental increase in core damage frequency for this root cause was found to be 1.0E-8/yr. The dominant contributor is sequence T_1MLE within the Diesel Generator Rooms.

3.3.10 Root Cause 10-External Plant Fires Causing FPS Actuations

This Root Cause was screened from further analysis since smoke detector actuation alone could not actuate a FPS.

3.3.11 Root Cause 11-Advertent Actuation of a Suppression System

For this scenario to occur, actuation of the FPS has to be in the same Fire Zone as the fire. Critical damage must occur either as a combination of fire-related effects and FPS agent release or due to FPS agent release alone. The dominant contributor is a result of cabling failures in Fire Zone 97 leading to sequence T_3MLE. This Root Cause contributes 6.4E-7/yr to core damage frequency.

3.3.12 Root Cause 12-Seismic/Fire Interaction

This root cause was the dominant contributor to total core damage frequency. It was found to contribute 4.7E-5/yr to core damage frequency. Fire Zones 105, 79, and 112 contributed 98%, 1%, and 1% respectively to the total. Failures in those Fire Zones once again lead to sequence T_1MLE.

3.3.13 Root Cause 13-FPS Actuation Due to Unknown Causes

The incremental increase in core damage frequency was found to be 2.9E-6/yr. It again arises primarily due to inadvertent actuations in the Cable Spreading Room giving rise to sequence T_3MLE. Credit was given for operator recovery from the remote shutdown panel.

3.4 Summary

As described above, of the thirteen root cause scenarios postulated to lead to core damage resulting from actuation of this plant's fire protection systems, six were found not to be applicable (fire-induced FPS actuation due to smoke spread, FPS actuation preventing manual fire-fighting and operator recovery of random failures, FPS actuation due to pipe break, FPS actuation due to dust raised in a seismic event, and external plant fires).

The seven remaining root cause scenarios led to an increase in core damage frequency with the following distribution:

Mean	5.6E-5
Median	1.4E-5
5th%	1.9E-6
95th%	3.6E-4

The dominant contributor to this total is Root Cause 12. This scenario contributed 87 percent to the total. Root Cause 13 (unknown causes) contributed 2.9E-6/yr, while FPS actuation due to relay chatter (Root Cause 8) contributed 1.5E-6/yr. These contributions were 5.2% and 2.3% respectively.

One key factor which led to screening and significant risk reductions of many of the postulated root causes was the fact that no FPSs were configured to actuate on a smoke detector only. In all cases where automatic actuation was possible, a dual logic actuation scheme was required (smoke and heat or smoke and flame).

It must be noted that this was a plant-specific analysis. Other plants of the same type might have core damage frequency contributions from Root Causes 1, 2, 3, 5, 7, and 10 which were not applicable to this site. Also, these results are highly dependent on the plant-specific equipment and cable locations.

3.5 References

3.1 W. R. Cramond et al, Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor, NUREG/CR-4713, SAND86-1832, Sandia National Laboratories, Albuquerque, NM, March 1987.

3.2 N. O. Siu, COMPBRN - A Computer Code for Modeling Compartment Fires, University of California, UCLA-ENG-8257, NUREG/CR-3239, May 1983.

3.3 V. Ho, N. O. Siu, G. Apostolakis, COMPBRN III - A Computer Code for Modeling Compartment Fires, University of California, UCLA-ENG-8524, November 1985.

3.4 V.F. Nicolette, S.P. Nowlen and J.A. Lambright, Observations Concerning the COMPBRN III Fire Growth Code, Sandia National Laboratories, Albuquerque, NM, SAND88-2160C, presented at International Topical Meeting, Probability, Reliability, and Safety Assessment, April 1989, Pittsburg, PA..

3.5 Units 1 and 2 Fire Protection Program.

3.6 G. J. Kolb, NUREG/CR-2787, June 1982.

3.7 W. T. Wheelis, User's Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base, NUREG/CR-4586, SAND86-0300, Sandia National Laboratories, Albuquerque, NM, August, 1986.

3.8 D. L. Bernrauter et al., Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains, NUREG/CR-5250, October 1988.

3.9 Electric Power Research Institute, Seismic Hazard Methodology for the Central and Eastern United States, EPRI NP-4726, Vols. 1-10, July 1986.

3.10 N. M. Newmark, "Effects of Earthquakes on Dams and Embankments," Fifth Rankine Lecture, Geotechnique, Vol. 15, No.2 p. 139, 1965.

4.0 SENSITIVITY STUDIES

The results in Chapter 3 represent a base case analysis that uses the parameter values presented in NUREG/CR-5580[a]. As discussed there, several of the parameter value estimates are thought to be more uncertain than other estimates. In particular, the values taken for the probability of a fire given tipping or sliding failure of an energized motor control center or bus and the probability of cable damage from the FPS actuation were chosen to be best estimates but with less data for justification of assignment. This section describes sensitivity studies in which three of the more uncertain estimates are varied (i.e., the probability of a fire given tipping or sliding failure of an energized electrical cabinet, the effect of application of a non-recovery factor for the EFW system, and the probability of FPS damage to cables). In addition, a sensitivity study is presented comparing the CDF contribution from the seismic root causes utilizing the LLNL and the EPRI hazard curves. Table 4.1 summarizes the results of these studies and also presents a fifth sensitivity study which is a combination of all three sensitivity studies. Descriptions of each sensitivity study are presented below.

Calculations for the sensitivity studies of core damage frequency and risk are accomplished by the use of the top event matrix analysis code TEMAC, (Reference 4.1) and the latin hypercube sampling code (Reference 4.2).

4.1 Sensitivity Study 1--Comparison of CDF Utilizing the LLNL and EPRI Seismic Hazard Curves

At this time, both sets of hazard curves are viewed by the USNRC as being equally credible. As such, calculations of the seismic core damage frequencies can be made for both sets of hazard curves and the results viewed as a measure of methodological uncertainty in the hazard curve development process.

In the base case analysis, the LLNL seismic hazard curves were utilized to calculate the CDF contribution for each of the applicable seismic root causes (8, 9, and 12) to be consistent with the NUREG-1150 studies. As a point of comparison, the CDF contribution from the seismic root causes were also calculated using the EPRI seismic hazard curves. All other values were kept the same as in the base case study. The results are presented in Table 4.2. Figures 4.1 and 4.2 present the LLNL hazard curves and the EPRI hazard curves, respectively.

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a. J. A. Lambright, et al., Evaluation of Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," NUREG/CR-5580, (to be published).

Table 4.1

Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 1 EPRI Hazard Case	Study 2 Decrease in Probability of a Seismic/Fire
1.	Not applicable for plant under consideration.		
2.	Not applicable for plant under consideration.		
3.	Not applicable for plant under consideration.		
4.	2.3E-6	N/A*	N/A
5.	Not applicable for plant under consideration.		
6.	1.4E-6	N/A	N/A
7.	Not applicable for plant under consideration.		
8.	1.5E-6	3.1E-8	N/A
9.	<1.0E-8	<1.0E-8	N/A
10.	Not applicable for plant under consideration.		
11.	6.4E-7	N/A	N/A
12.	4.7E-5	1.8E-6	9.4E-6
13.	2.9E-6	N/A	N/A
Total	5.6E-5	9.1E-6	1.7E-5

*All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

**All entries in this table represent mean values of uncertainty analysis results given in Appendix A.

Table 4.1 (Concluded)

Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 3 EFW Recovery	Study 4 Reduced Water Damage to Cable	Study 5 All Combined
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	2.3E-6	N/A*	4.6E-7	4.6E-7
5.	Not applicable for plant under consideration.			
6.	1.4E-6	N/A	2.8E-7	2.8E-7
7.	Not applicable for plant under consideration.			
8.	1.5E-6	N/A	3.0E-7	<1.0E-8
9.	<1.0E-8	N/A	N/A*	<1.0E-8
10.	Not applicable for plant under consideration.			
11.	6.4E-7	N/A	1.3E-7	1.3E-7
12.	4.7E-5	2.4E-5	N/A	1.8E-7
13.	2.9E-6	N/A	5.8E-7	5.8E-7
Total	5.6E-5	3.1E-5	4.9E-5	1.7E-6

*All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

**All entries in this table represent mean values of uncertainty analysis results given in Appendix A.

Table 4.2

Core Damage Frequencies for Sensitivity Study 1
EPRI Seismic Hazard Curves
(Per Reactor Year)*

Root Cause	Sequence 1
1	---
2	---
3	---
4	---
5	---
6	---
7	---
8	3.1E-8
9	<1.0E-8
10	---
11	---
12	1.8E-6
13	---
Total	1.8E-6

Sequence 1 - T_1MLE

*All entries in this table are mean values.

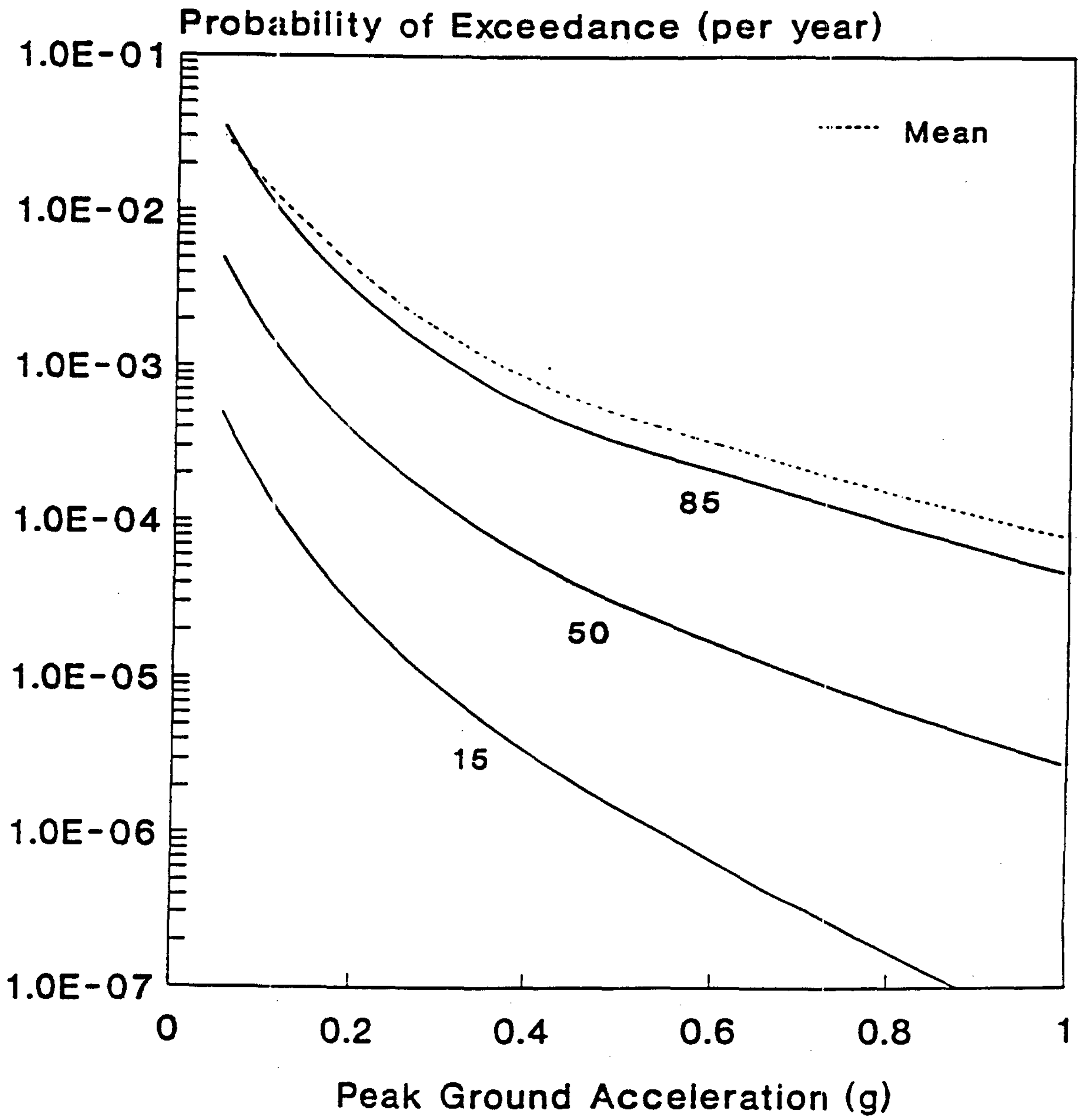

Figure 4.1 LLNL Hazard Curves: Mean, Median 85th and 15th Percentile Curves

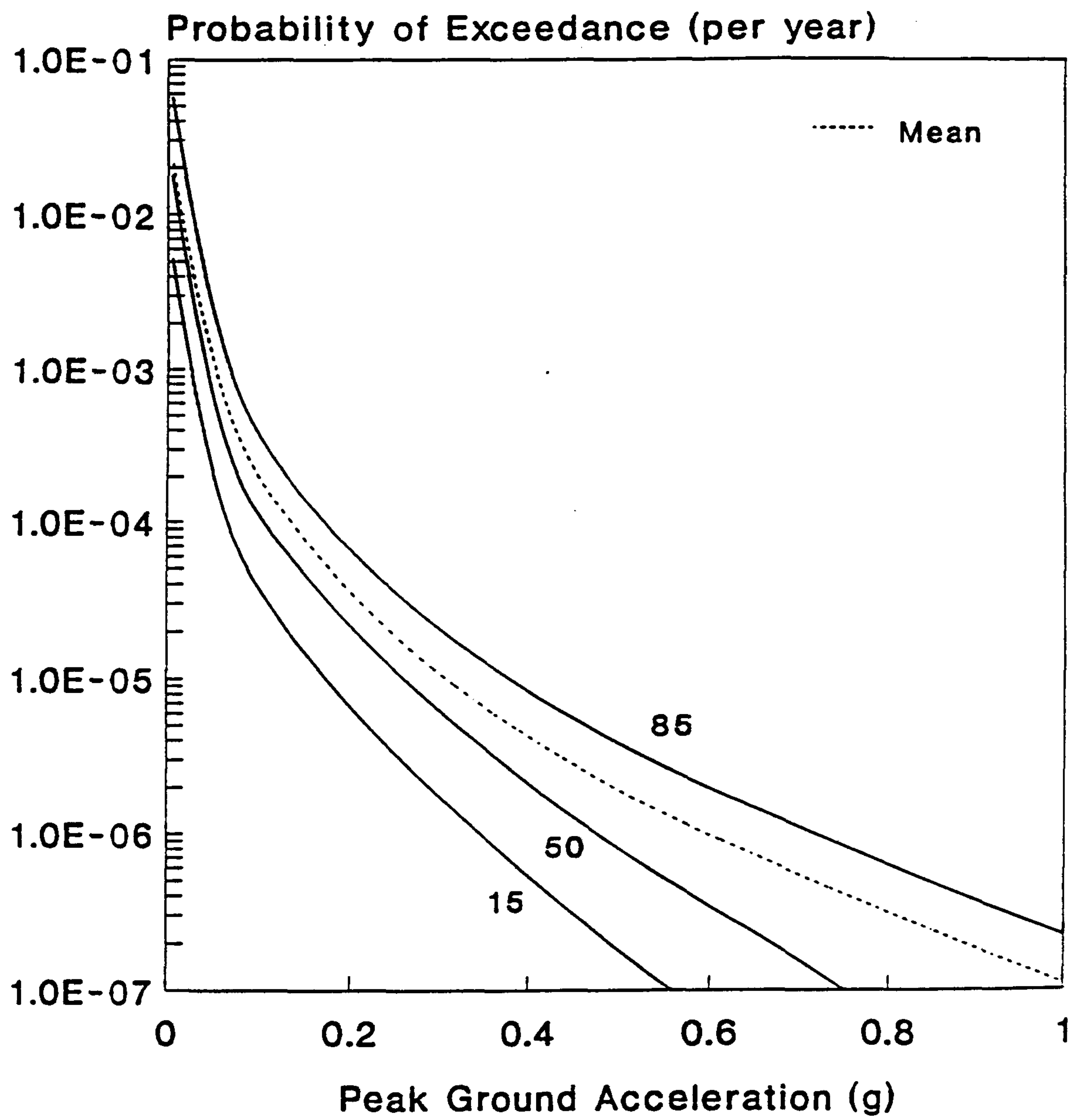

Figure 4.2 EPRI Hazard Curves: Mean, Median 85th and 15th Percentile Curves

4.2 Sensitivity Study 2--Decrease in the Probability of a Fire Given Tipping or Sliding Failure of an Energized Electrical Cabinet

For the base case analysis, the probability of a fire given the tipping or sliding failure of an electrical cabinet was assigned a value of 0.5 (with the exceptions noted in Section 3.2.3.2). This value was based on engineering judgment which takes into account industrial earthquake experiences of a similar nature (Reference 4.3). However, the actual probability may be less than the base case value. Consequently, for this sensitivity study, the probability of fire given the tipping or sliding failure of an energized electrical cabinet was reduced by a factor of 5. All other numerical values were kept the same as in the base case. The accident sequence cut sets were requantified to determine a new value of the incremental increase in core damage frequency. Since this study involves seismic/fire interaction, the only Root Cause affected is Root Cause 12. The requantified contribution to the core damage frequency is presented in Table 4.3.

4.3 Sensitivity Study 3--Recovery of EFW System following a Seismic/Fire Interaction in Fire Zone 105

In the analysis, it must be decided whether to give credit for recovery of the EFW system following a transient initiated by a seismic/fire interaction in Fire Zone 105. This seismic/fire interaction is predicted to damage cable GC1428C which provides turbine steam admission valve position indication to turbine speed control circuitry. Failure of this cable could prevent the automatic sequencing of the turbine driven pump past idle speed, and thus fail to provide feed to the steam generators. In a similar plant PRA (Ref. 4.4), core damage results from such a failure to feed within 20 minutes after transient initiation. Alternate shutdown procedures direct the Number 2 Reactor Operator to leave the control room, proceed to the Emergency Feedwater Room, and take manual control of the turbine driven pump for the plant under study in this report. The alternate shutdown procedure timeline for gaining control of the pump is 11 minutes.

For the base case, no credit was given for EFW recovery. The failure to give credit is based on the postulated overall degraded condition of the plant and its operators following a severe earthquake, loss of offsite power, and a fire precluding such action within 20 minutes. As a sensitivity study, credit for recovery is given. Again from a similar plant PRA, an EFW non-recovery factor of 5.0E-1 was applied. This is based on analysis of operator recovery of EFW within 30 minutes, given a single alternative feedwater system being available. The requantified incremental increase in core damage frequency is presented in Table 4.4.

4.4 Sensitivity Study 4--Decrease in Cable Damage From Suppressant Agent

In the base case analysis, any type of FPS actuation was assumed to damage cables with equal probability. Cable damage is assumed to occur due to inadequate seals for the cables and the possibility of erroneous signals being generated in cables exposed to an overdump of CO_2, water

Table 4.3

Core Damage Frequencies for Sensitivity Study 2
Reduced Probability of a Fire Given Tipping or Sliding Failure of an Energized Electrical Cabinet (Per Reactor Year)*

	Root Cause	Sequence 1
1	---	
2	---	
3	---	
4	---	
5	---	
6	---	
7	---	
8	---	
9	---	
10	---	
11	---	
12	9.4E-6	
13	---	
Total		9.4E-6

Sequence 1 - T_1MLE

*All entries in this table are mean values.

Table 4.4

Core Damage Frequencies for Sensitivity Study 3
Recovery of EFW System Following A Seismic/Fire
Interaction in Fire Zone 105

	Root Cause	Sequence 1
1	---	
2	---	
3	---	
4	---	
5	---	
6	---	
7	---	
8	---	
9	---	
10	---	
11	---	
12	2.4E-5	
13	---	
Total		2.4E-5

Sequence 1 - T_1MLE

*All entries in this table are mean values.

intrusion, or exposure to Halon. The probability of FPS damage to cables was treated as a sensitivity issue. In this sensitivity study, the mean probability of FPS damage to cables was lowered from 5.0E-3 to 1.0E-3.

For the plant under study, this reduced probability affects only the Cable Spreading Room. The requantified incremental increases in core damage frequency are presented in Table 4.5.

4.5 Sensitivity Study 5--Combination of Studies 1, 2, 3, and 4

For this final sensitivity study, the changes mentioned in the four previous studies were incorporated simultaneously. Specifically, the EPRI seismic hazard curves were used in place of the LLNL curves to obtain the CDF contribution for each of the seismic root causes. The probability of a fire given tipping or sliding failure of an energized cabinet and the probability of suppressant damage were both reduced by a factor of five. Recovery of the EFW system following seismic/ fire damage in Fire Zone 105 is allowed which results in a reduction for all applicable sequences of a factor of two.

The accident sequence cut sets were then requantified with all other values being kept the same as in the base case. Hence, this sensitivity study represents the most optimistic analysis--and the most optimistic results--in this report.

The resulting increments in core damage frequency are summarized in Table 4.6. The total increment for Root Cause 4 decreases from 2.3E-6/yr in the base case to 4.6E-7/yr. Sequence T_3MLE and the Cable Spreading Room remain as the dominant contributors.

For Root Cause 6, the total increment decreases from 1.4E-6/yr to 2.8E-7/yr. Sequence T_3MLE and the cable spreading room remain as the dominant contributors for this root cause.

For Root Cause 11, the total increment reduces from 6.4E-7/yr to 1.3E-7/yr. The dominant contributor are sequence T_3MLE and the Cable Spreading Room.

The total increment for Root Cause 13 contribution to core damage frequency decreased from 2.9E-6/yr to 5.8E-7/yr. Sequence T_3MLE and the Cable Spreading Room are the dominant contributors for this root cause.

The core damage frequency contribution from seismic Root Cause 8, which involves relay chatter in the Diesel Generator Rooms, decreased from 1.5E-6/yr to <1.0E-8/yr. The dominant factor in the reduction of core damage frequency is the result of utilizing the EPRI hazard curves.

For seismic Root Cause 9, which involves FPS actuation due to mechanical damage, the reduction in cable damage probability combined with utilizing the EPRI hazard curves reduced core damage frequency from 1.0E-8/yr to <1.0E-8/yr.

Table 4.5

Core Damage Frequencies for Sensitivity Study 4--Reduced Probability of Cable Damage from Water (Per Reactor Year)*

Root Cause	Sequence 1	Sequence 2	Total
1	---	---	
2	---	---	
3	---	---	
4	4.6E-7	---	
5	---	---	
6	2.8E-7	---	
7	---	---	
8	---	3.0E-7	
9	---	---	
10	---	---	
11	1.3E-7	---	
12	---	---	
13	5.8E-7	---	
Totals	1.5E-6	3.0E-7	1.8E-6

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

*All entries in this table are mean values.

Table 4.6

Core Damage Frequencies for Sensitivity Study 5-Combination of Sensitivity Studies 1, 2, 3 and 4 (Per Reactor Year)*

Root Cause	Sequence 1	Sequence 2	Total
1	---	---	
2	---	---	
3	---	---	
4	4.6E-7	---	
5	---	---	
6	2.8E-7	---	
7	---	---	
8	---	<1.0E-8	
9	---	<1.0E-8	
10	---	---	
11	1.3E-7	---	
12	---	1.8E-7	
13	5.8E-7	---	
Totals	1.5E-6	1.8E-7	1.7E-6

Sequence 1 - T_3MLE

Sequence 2 - T_1MLE

* All entries in this table are mean values.

For Root Cause 12, which is seismic/fire interaction in Fire Zones 79, 105, and 112, the reduction in the probability of fire given tipping or sliding failure of an energized cabinet, and the recovery of EFW,

combined with utilizing the EPRI hazard curves reduced core damage frequency from 4.7E-5/yr to 1.8E-7/yr. The dominant contributor in this reduction was the use of EPRI hazard curves.

The net result of this most optimistic analysis is to decrease the increments in total core damage frequency by approximately a factor of 30. Root Causes 13, 4, and 6 are now the dominant root causes.

4.6 Summary

The requantified contributions to core damage frequency are summarized in Table 4.1. The results of these sensitivity studies show that the most dominant effect is the utilization of EPRI hazard curves. This reduced the core damage frequency by 90 percent. The second most dominant effect on reduction of core damage frequency was the effect of decreasing the probability of a seismically-induced fire.

4.7 References

4.1 R.L. Iman, et al., A User's Guide for the Top Event Matrix Analysis Code (TEMAC), NUREG/CR-4598, SAND86-0960, Sandia National Laboratories, August 1986.

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4.3 S. W. Swan and S. P. Harris, Survey of Earthquake-Induced Fires in Electric Power and Industrial Facilities, EPRI NP-6989, September 1990.

4.4 Nuclear Safety Analysis Center and Electric Power Research Institute, A Probabilistic Risk Assessment of Oconee Unit 3, NSAC-60, June 1984.

5.0 OFFSITE DOSE AND RISK ASSESSMENT

This chapter provides the derivation of the offsite dose calculations for this analysis and presents the risk calculations for each of the applicable root causes. Appendix B provides the uncertainty calculations for risk as well as for each cut set for the seismic and non-seismic root causes.

5.1 Containment System Event Tree for Transient

The containment system event tree for transients is shown in Figure 5.1. Each heading in the tree relates to a system or combination of systems. Each system has success criteria and a fault tree model. The success criteria are given by:

Containment overpressure protection (COP):

Y: Success or failure of 1 of 4 Reactor Building Cooling Fan trains.

C: Success or failure of 1 of 2 Reactor Building Spray Injection trains during the injection phase.

F: Success or failure of 1 of 2 Reactor Building Recirculation Spray trains during the recirculation phase plus sump mixing with 1 of 3 HPRS and 1 of 2 LPRS with heat exchange.

Post accident radioactivity removal (PARR):

C: Success or failure of 1 of 2 Reactor Building Spray Injection trains during injection phase.

F: Success or failure of 1 of 2 Reactor Building Spray Recirculation trains during recirculation phase.

In this event tree, Z_t=YC is defined in a way to show containment overpressure protection exists when:

One containment fan unit operates, or, 1 of 2 Reactor Building Spray trains is operational.

5.2 Offsite Dose Calculations

To convert the calculated core damage frequencies to offsite dose, the methodology used was that outlined in Reference 5.1. This methodology is based on the MARCH code studies performed as part of the Calvert Cliffs RSSMAP Study. This methodology was selected because the Calvert Cliffs' containment size and design is similar to that of this plant.

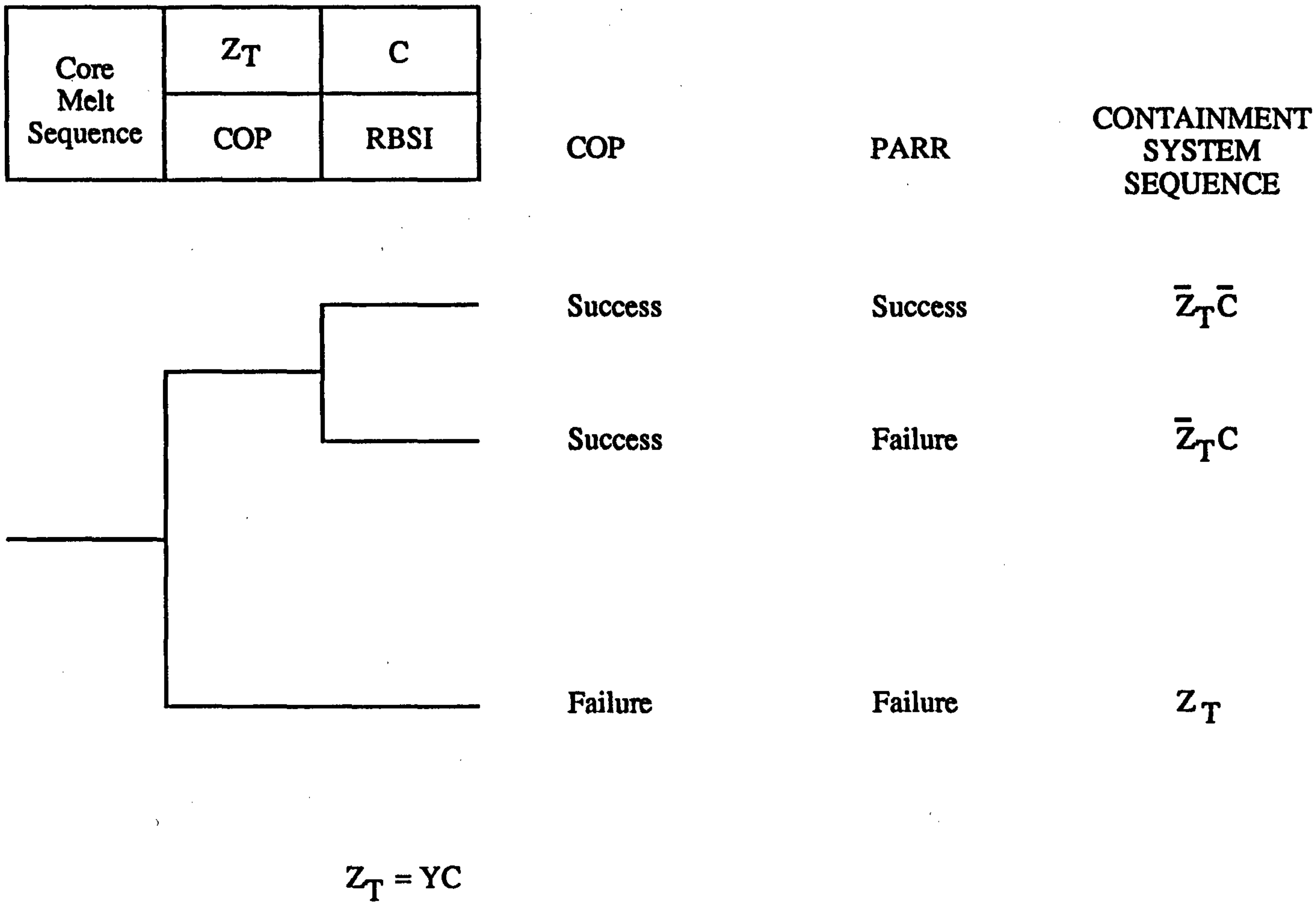

FIGURE 5.1 CONTAINMENT SYSTEM EVENT TREE FOR TRANSIENTS

There are two containment functions that are important during accidents: containment overpressure protection (COP) and post accident radioactivity removal (PARR). Successful COP is defined as successful blowdown of steam from the reactor vessel. Successful long-term COP requires that heat then be removed from the reactor building sump via the Low Pressure Recirculation system. PARR also involves the reactor building sump and is dependent on successful COP. If the reactor building sump water inventory is maintained and cooled during a core meltdown then a large fraction of the fission products released from the core should be retained in the pool. Knowing the status of COP and PARR during a severe accident is the starting point for estimating containment failure modes and accident releases. Table 5.1 provides a listing and description of the containment failure modes for each of the accident sequence types.

Using the estimate of fission product release, the potential consequences that could result from an accident were calculated. The calculations were performed using the MARCH code. The result is the radiation dose in person-REM received by the population around the plant after an accident integrated out to a distance of fifty miles. It was assumed that the remaining operational life of the plant is 20 years.

Three different sets of results were calculated as presented in Table 5.2. The first calculation is called the "upper bound" calculation. In the second calculation, the source term is reduced by a factor of seventy percent (0.3 times the upper bound). This is called the "central estimate" calculation. In the final "lower bound" calculation, the source term was reduced by ninety percent (0.1 times the upper bound values). This approach is the same taken in the analyses performed in Reference 5.2 and analyses conducted in resolution of U.S. NRC Unresolved Safety Issue 45 (USI-45), "Decay Heat Removal Requirements." These additional calculations were performed to illustrate the potential sensitivity of the results to variations in the source term. This selection of source terms should not, however, be interpreted as an endorsement of any particular set. The "real" source term may be larger or smaller.

Table 5.3 provides the results in terms of risk (person-REM) for the base case as well as the sensitivity studies described in Chapter 4 of this report. The base case total is 100 person-REM. The results of these sensitivity studies show that the most dominant effect was use of the EPRI hazard curves which reduced risk from 100 to 14 person-REM.

Table 5.1

Estimated Containment Failure Modes

Containment Sequence	Containment Failure Mode with Probability and Release Category				
	α	β	γ, δ_e	δ_1	ε
$\bar{Z}_t\ \bar{C}$	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 5	2.5E-1 7
$\bar{Z}_t\ C$	1E-4 1	2E-3 4	1.4E-2 3	1.8E-1 3	2.5E-1 6
$\bar{Z}_t$	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6

Where,

α = In-vessel steam explosion,

β = Containment leakage,

γ = Hydrogen burn overpressure,

δ_e = Ex-vessel steam spike,

δ_1 = Steam and non-condensible gas overpressure, and

ε = Base mat melt through.

Table 5.2

MARCH Code Results for Release Categories
(Person-REM/reactor-year within 50 miles of plant)

Release Category	Upper Bound	Central Estimate Case	Lower Bound
1	7.9E+5	4.7E+5	2.9E+5
2	7.5E+5	4.9E+5	3.5E+5
3	5.9E+5	3.5E+5	2.3E+5
4	3.2E+5	2.1E+5	1.4E+5
5	2.0E+5	1.2E+4	6.5E+4
6	5.4E+4	2.3E+4	1.0E+4
7	6.0E+3	2.4E+3	2.1E+3

Table 5.3

Summary of Base Case and Sensitivity Study Results in Terms of Risk (Person-REM)**

Root Cause	Base Case	EPRI Hazard Curve	Decrease in Probability of a Seismic Fire
1.	Not applicable for the plant under consideration.		
2.	Not applicable for the plant under consideration.		
3.	Not applicable for the plant under consideration.		
4.	3.4	N/A*	N/A*
5.	Not applicable for the plant under consideration.		
6.	1.9	N/A	N/A
7.	Not applicable for the plant under consideration.		
8.	2.6	.05	N/A
9.	0.02	<0.01	N/A
10.	Not applicable for the plant under consideration.		
11.	1.0	N/A	N/A
12.	87	3.4	17
13.	4.6	N/A	N/A
Total	100	14	31

*N/A reflects no modification from the base case.
**All values listed in table are mean values on the uncertainty analysis given in Appendix B.

Table 5.3 (Concluded)

Summary of Base Case and Sensitivity Study Results in Terms of Risk (Person-REM)**

Root Cause	Base Case	EFW Recovery	Reduced FPS Damage to Cable	All Combined
1.	Not applicable for the plant under consideration.			
2.	Not applicable for the plant under consideration.			
3.	Not applicable for the plant under consideration.			
4.	3.4	N/A*	0.68	0.68
5.	Not applicable for the plant under consideration.			
6.	1.9	N/A	0.38	0.38
7.	Not applicable for the plant under consideration.			
8.	2.6	N/A	0.52	0.01
9.	0.02	N/A	N/A*	<0.01
10.	Not applicable for the plant under consideration.			
11.	1.0	N/A	0.20	0.20
12.	87	44	N/A	0.34
13.	4.6	N/A	0.92	0.92
Total	100	58	90	2.5

*N/A reflects no modification from the base case.
**All values listed in table are mean values.

5.3 References

5.1 S. Hatch, et al., Reactor Safety Study Methodology Applications Program, NUREG/CR-1659, SAND80-1897, Sandia National Laboratories, Albuquerque, NM, May 1982.

5.2 U.S. Nuclear Regulatory Commission, Reactor Safety Study, Appendix VI, Calculation of Reactor Accident Consequences, NUREG-075/14, WASH-1400, U.S. NRC, 1975.

APPENDIX A

Uncertainty Analysis
Core Damage Frequency

DEFINITION OF TERMS

Fcm	- Frequency of core damage (Fire Zone, Root Cause)
L-OPW	- Frequency of operator error leading to initiation of H_2O deluc suppressant system
NRP	- Non-recovery probability from the remote shutdown panel
PDAMC	- Probability of cable damage from water suppressant
L-RAW	- Frequency of hardware failure leading to initiation of H_2O deluge suppressant system
L-UNW	- Frequency of H_2O deluge system initiation due to unknown cause
L-CSR	- Frequency of Cable Spreading Room fire
QITG	- Success probability of Cable Spreading Room deluge system
A	- Root Cause 12, Fire Zone 79
B	- Root Cause 7/8/9, Fire Zone 86 and 87
C	- Root Cause 7/8, Fire Zone 97
D	- Root Cause 12, Fire Zone 105
E	- Root Cause 12, Fire Zone 112

Top Event Matrix Analysis Code

The following printouts represent the output of the Top Event Matrix Analysis Code (TEMAC) used to quantify the uncertainty analyses for Core Damage Frequency and for Risk. TEMAC accomplishes this quantification using parameter value samples generated by the Latin Hypercube Sampling cod (LHS). LHS is a constrained Monte Carlo technique which forces all parts c distribution to be sampled. For the composite, and for each Root Cause, th following information is provided:

- Top event frequency distribution.
- Risk increases and reductions by base events sorted by risk reduction.
- Risk reduction by base event.
- Risk increase by base event.
- Cutset frequencies.
- Cutsets contributing to the Root Cause.

Definitions of key terms in the TEMAC printouts are:

• Risk reduction - For each basic event, the probability of occurance of that event is set to zero and the reduction in core damage frequency or risk is calculated.

• Risk increase - For each basic event, the probability of occurance of that event is set to 1.0 and the increase in core damage frequency or risk is calculated.

• Uncertainty importance - For each basic event, its distributi is eliminated from the overall uncertainty calculation by setting the event its mean value. The percent decrease in the logarithm of the overall uncertainty is then calculated.

Composite Uncertainty Analysis

Core Damage Frequency

ROOT CAUSE COMPOSITE RERUN 5-8-92

TOP EVENT COMPOSITE-RT-CS CONTAINS   12 EVENTS IN   9 CUT SETS

THE FREQUENCY OF TOP EVENT COMPOSITE-RT-CS IS   5.77E-05

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT COMPOSITE-RT-CS

N	1000
MEAN	5.58E-05
STD DEV	1.09E-04
LOWER 5%	1.89E-06
LOWER 25%	5.99E-06
MEDIAN	1.37E-05
UPPER 25%	3.64E-05
UPPER 5%	3.55E-04

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE,
ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   1.89E-06                                 ***LOG SCALE***                                 95% =   3.55E-04
I---------------------[----------------*-------------------]-------MN-----------------------------------I
```


NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE COMPOSITE RERUN 5-8-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
B	1	3.44E-08	(12.0)	1.00E+00	(3.0)	1.72E-12	(12.0)
D	1	4.78E-05	(8.0)	1.00E+00	(3.0)	7.49E-06	(4.0)
E	1	4.30E-07	(11.0)	1.00E+00	(3.0)	1.35E-07	(10.0)
C	1	3.05E-06	(9.0)	1.00E+00	(3.0)	6.28E-07	(9.0)
A	1	5.25E-07	(10.0)	1.00E+00	(3.0)	8.74E-08	(11.0)
PDAMC	4	3.00E-03	(6.0)	3.23E-03	(6.0)	1.57E-05	(1.5)
L-OPW	1	9.40E-03	(4.0)	3.61E-04	(8.0)	1.62E-06	(6.0)
L-UNW	1	1.30E-02	(3.0)	3.61E-04	(8.0)	1.11E-05	(3.0)
L-RAW	1	5.50E-03	(5.0)	3.61E-04	(8.0)	2.02E-06	(5.0)
L-CSR	1	2.70E-03	(7.0)	3.51E-04	(10.0)	9.88E-07	(7.5)
NRP	4	6.40E-02	(2.0)	2.13E-04	(11.0)	1.57E-05	(1.5)
QITG	1	9.50E-01	(1.0)	1.02E-06	(12.0)	9.88E-07	(7.5)

ROOT CAUSE COMPOSITE RERUN 5-8-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
B	1	3.44E-08	(12.0)	1.00E+00	(3.0)	4.61E-12	6.70E-08
D	1	4.78E-05	(8.0)	1.00E+00	(3.0)	8.86E-08	3.46E-04
E	1	4.30E-07	(11.0)	1.00E+00	(3.0)	1.95E-09	1.03E-06
C	1	3.05E-06	(9.0)	1.00E+00	(3.0)	1.12E-08	8.62E-06
A	1	5.25E-07	(10.0)	1.00E+00	(3.0)	2.54E-09	1.37E-06
PDAMC	4	3.00E-03	(6.0)	3.23E-03	(6.0)	2.39E-07	2.66E-05
L-OPW	1	9.40E-03	(4.0)	3.61E-04	(8.0)	4.26E-08	9.44E-06
L-UNW	1	1.30E-02	(3.0)	3.61E-04	(8.0)	5.67E-08	1.25E-05
L-RAW	1	5.50E-03	(5.0)	3.61E-04	(8.0)	2.47E-08	5.31E-06
L-CSR	1	2.70E-03	(7.0)	3.51E-04	(10.0)	2.43E-08	2.42E-06
NRP	4	6.40E-02	(2.0)	2.13E-04	(11.0)	2.39E-07	2.66E-05
QITG	1	9.50E-01	(1.0)	1.02E-06	(12.0)	2.43E-08	2.42E-06

ROOT CAUSE COMPOSITE RERUN 5-8-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
NRP	4	6.40E-02	(2.0)	1.57E-05	(1.5)	7.73E-06	3.17E-04
PDAMC	4	3.00E-03	(6.0)	1.57E-05	(1.5)	2.20E-04	6.24E-03
L-UNW	1	1.30E-02	(3.0)	1.11E-05	(3.0)	1.05E-05	8.26E-04
D	1	4.78E-05	(8.0)	7.49E-06	(4.0)	1.00E+00	1.00E+00
L-RAW	1	5.50E-03	(5.0)	2.02E-06	(5.0)	1.05E-05	8.32E-04
L-OPW	1	9.40E-03	(4.0)	1.62E-06	(6.0)	1.05E-05	8.26E-04
L-CSR	1	2.70E-03	(7.0)	9.88E-07	(7.5)	1.02E-05	8.04E-04
QITG	1	9.50E-01	(1.0)	9.88E-07	(7.5)	8.61E-10	1.38E-07
C	1	3.05E-06	(9.0)	6.28E-07	(9.0)	1.00E+00	1.00E+00
E	1	4.30E-07	(11.0)	1.35E-07	(10.0)	1.00E+00	1.00E+00
A	1	5.25E-07	(10.0)	8.74E-08	(11.0)	1.00E+00	1.00E+00
B	1	3.44E-08	(12.0)	1.72E-12	(12.0)	1.00E+00	1.00E+00

ROOT CAUSE COMPOSITE RERUN 5-8-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
6	1	5.53E-05	(1.0)	8.86E-08	3.46E-04
2	3	1.36E-05	(2.0)	5.67E-08	1.25E-05
8	1	3.68E-06	(3.0)	1.12E-08	8.62E-06
3	3	3.42E-06	(4.0)	4.26E-08	9.44E-06
4	3	3.07E-06	(5.0)	2.47E-08	5.31E-06
1	4	1.48E-06	(6.0)	2.43E-08	2.42E-06
5	1	6.12E-07	(7.0)	2.54E-09	1.37E-06
7	1	5.65E-07	(8.0)	1.95E-09	1.03E-06
9	1	3.44E-08	(9.0)	4.61E-12	6.70E-08

ROOT CAUSE COMPOSITE RERUN 5-8-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT COMPOSITE-RT-CS WITH TOP EVENT FREQUENCY 5.77E-05

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	6	1	5.53E-05	0.00000	D	+							
3	2	3	1.36E-05	0.00000	L-UNW	*	NRP	*	PDAMC	+			
4	8	1	3.68E-06	0.00000	C	+							
5	3	3	3.42E-06	0.00000	L-OPW	*	NRP	*	PDAMC	+			
6	4	3	3.07E-06	0.00000	L-RAW	*	NRP	*	PDAMC	+			
7	1	4	1.48E-06	0.00000	L-CSR	*	NRP	*	PDAMC	*	QITG	+	
8	5	1	6.12E-07	0.00000	A	+							
9	7	1	5.65E-07	0.00000	E	+							
10	9	1	3.44E-08	0.00000	B	.							

Root Cause 4

ROOT CAUSE 4 RERUN 5-8-92

TOP EVENT ROOT-CAUSE-4 CONTAINS 3 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-4 IS 1.80E-06

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-4

N	1000
MEAN	2.28E-06
STD DEV	5.18E-06
LOWER 5%	4.26E-08
LOWER 25%	2.13E-07
MEDIAN	7.02E-07
UPPER 25%	2.20E-06
UPPER 5%	9.44E-06

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   4.26E-08                                ***LOG SCALE***                              95% =   9.44E-06
I-------------------------------[----------------------*-----------------N---M---------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 4 RERUN 5-8-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
L-OPW	1	9.40E-03	(2.0)	3.61E-04	(1.0)	1.62E-06	(2.0)
PDAMC	1	3.00E-03	(3.0)	3.31E-04	(2.0)	1.62E-06	(2.0)
NRP	1	6.40E-02	(1.0)	2.18E-05	(3.0)	1.62E-06	(2.0)

ROOT CAUSE 4 RERUN 5-8-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
L-OPW	1	9.40E-03	(2.0)	3.61E-04	(1.0)	4.26E-08	9.44E-06
PDAMC	1	3.00E-03	(3.0)	3.31E-04	(2.0)	4.26E-08	9.44E-06
NRP	1	6.40E-02	(1.0)	2.18E-05	(3.0)	4.26E-08	9.44E-06

ROOT CAUSE 4 RERUN 5-8-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
PDAMC	1	3.00E-03	(3.0)	1.62E-06	(2.0)	3.57E-05	2.17E-03
NRP	1	6.40E-02	(1.0)	1.62E-06	(2.0)	1.34E-06	1.24E-04
L-OPW	1	9.40E-03	(2.0)	1.62E-06	(2.0)	1.05E-05	8.26E-04

ROOT CAUSE 4 RERUN 5-8-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	3	3.42E-06	(1.0)	4.26E-08	9.44E-06

ROOT CAUSE 4 RERUN 5-8-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-4   WITH TOP EVENT FREQUENCY   1.80E-06

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	1	3	3.42E-06	0.00000	L-OPW	* NRP	* PDAMC	.

Root Cause 6

ROOT CAUSE 6 RERUN 5-8-92

TOP EVENT ROOT-CAUSE-6 CONTAINS 3 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-6 IS 1.06E-06

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-6

N	1000
MEAN	1.36E-06
STD DEV	3.53E-06
LOWER 5%	2.47E-08
LOWER 25%	1.41E-07
MEDIAN	4.20E-07
UPPER 25%	1.24E-06
UPPER 5%	5.31E-06

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   2.47E-08                              ***LOG SCALE***                              95% =   5.31E-06
I-------------------------------[---------------------*-----------------N--]M-------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 6 RERUN 5-8-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
PDAMC	1	3.00E-03	(3.0)	4.13E-04	(1.0)	2.02E-06	(2.0)
L-RAW	1	5.50E-03	(2.0)	3.61E-04	(2.0)	2.02E-06	(2.0)
NRP	1	6.40E-02	(1.0)	2.73E-05	(3.0)	2.02E-06	(2.0)

ROOT CAUSE 6 RERUN 5-8-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
PDAMC	1	3.00E-03	(3.0)	4.13E-04	(1.0)	2.47E-08	5.31E-06
L-RAW	1	5.50E-03	(2.0)	3.61E-04	(2.0)	2.47E-08	5.31E-06
NRP	1	6.40E-02	(1.0)	2.73E-05	(3.0)	2.47E-08	5.31E-06

ROOT CAUSE 6 RERUN 5-8-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
NRP	1	6.40E-02	(1.0)	2.02E-06	(2.0)	8.53E-07	7.45E-05
PDAMC	1	3.00E-03	(3.0)	2.02E-06	(2.0)	1.98E-05	1.31E-03
L-RAW	1	5.50E-03	(2.0)	2.02E-06	(2.0)	1.05E-05	8.32E-04

ROOT CAUSE 6 RERUN 5-8-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	3	3.07E-06	(1.0)	2.47E-08	5.31E-06

ROOT CAUSE 6 RERUN 5-8-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-6 WITH TOP EVENT FREQUENCY 1.06E-06

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	1	3	3.07E-06	0.00000	L-RAW	* NRP	* PDAMC

Root Cause 7/8

ROOT CAUSE 7-8 RUN 2-10-92

TOP EVENT ROOT-CAUSE-7-8 CONTAINS 1 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-7-8 IS 3.05E-06

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-7-8

N	1000
MEAN	1.46E-06
STD DEV	2.65E-06
LOWER 5%	1.12E-08
LOWER 25%	6.21E-08
MEDIAN	2.29E-07
UPPER 25%	1.49E-06
UPPER 5%	8.62E-06

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =    1.12E-08                                ***LOG SCALE***                                95% =    8.62E-06
I--------------------------[---------------------*----------------------------M]----------N---------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 7-8 RUN 2-10-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
C	1	3.05E-08	(1.0)	1.00E+00	(1.0)	6.28E-07	(1.0)

ROOT CAUSE 7-8 RUN 2-10-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
C	1	3.05E-06	(1.0)	1.00E+00	(1.0)	1.12E-08	8.62E-06

ROOT CAUSE 7-8 RUN 2-10-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
C	1	3.05E-06	(1.0)	6.28E-07	(1.0)	1.00E+00	1.00E+00

ROOT CAUSE 7-8 RUN 2-10-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	1	3.68E-06	(1.0)	1.12E-08	8.62E-06

ROOT CAUSE 7-8 RUN 2-10-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-7-8   WITH TOP EVENT FREQUENCY   3.05E-06

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

```
      2       1   1  3.68E-06   0.00000      C                .
```

Root Cause 8/9

ROOT CAUSE 7-8-9 RUN 2-10-92

TOP EVENT ROOT-CAUSE-7-8-9 CONTAINS 1 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-7-8-9 IS 3.44E-08

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-7-8-9

```
N                1000
MEAN         9.59E-09
STD DEV      2.08E-08
LOWER 5%     4.67E-12
LOWER 25%    6.27E-11
MEDIAN       3.60E-10
UPPER 25%    6.95E-09
UPPER 5%     6.70E-08
```

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   4.67E-12                                ***LOG SCALE***                                95% =   6.70E-08
I---------------------------[------------------*--------------------------------]--M--------------N------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 7-8-9 RUN 2-10-92

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
B	1	3.44E-08	(1.0)	1.00E+00	(1.0)	1.72E-12	(1.0)

ROOT CAUSE 7-8-9 RUN 2-10-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
B	1	3.44E-08	(1.0)	1.00E+00	(1.0)	4.61E-12	6.70E-08

ROOT CAUSE 7-8-9 RUN 2-10-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
B	1	3.44E-08	(1.0)	1.72E-12	(1.0)	1.00E+00	1.00E+00

ROOT CAUSE 7-8-9 RUN 2-10-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	1	3.44E-08	(1.0)	4.61E-12	6.70E-08

ROOT CAUSE 7-8-9 RUN 2-10-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-7-8-9 WITH TOP EVENT FREQUENCY 3.44E-08

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	1	1	3.44E-08	0.00000	B

Root Cause 11

ROOT CAUSE 11 RUN 2-10-92

TOP EVENT ROOT-CAUSE-11 CONTAINS 4 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-11 IS 4.92E-07

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-11

N	1000
MEAN	6.44E-07
STD DEV	9.95E-07
LOWER 5%	2.43E-08
LOWER 25%	1.03E-07
MEDIAN	2.99E-07
UPPER 25%	7.46E-07
UPPER 5%	2.42E-06

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   2.43E-08                            ***LOG SCALE***                              95% =   2.42E-06
I--------------------------------[-------------------------*----------N-----M--]--------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 11 RUN 2-10-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
L-CSR	1	2.70E-03	(4.0)	3.51E-04	(1.0)	9.88E-07	(2.5)
PDAMC	1	3.00E-03	(3.0)	2.02E-04	(2.0)	9.88E-07	(2.5)
NRP	1	6.40E-02	(2.0)	1.34E-05	(3.0)	9.88E-07	(2.5)
QITG	1	9.50E-01	(1.0)	1.02E-06	(4.0)	9.88E-07	(2.5)

ROOT CAUSE 11 RUN 2-10-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
L-CSR	1	2.70E-03	(4.0)	3.51E-04	(1.0)	2.43E-08	2.42E-06
PDAMC	1	3.00E-03	(3.0)	2.02E-04	(2.0)	2.43E-08	2.42E-06
NRP	1	6.40E-02	(2.0)	1.34E-05	(3.0)	2.43E-08	2.42E-06
QITG	1	9.50E-01	(1.0)	1.02E-06	(4.0)	2.43E-08	2.42E-06

ROOT CAUSE 11 RUN 2-10-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
NRP	1	6.40E-02	(2.0)	9.88E-07	(2.5)	8.26E-07	2.56E-05
PDAMC	1	3.00E-03	(3.0)	9.88E-07	(2.5)	2.13E-05	5.28E-04
QITG	1	9.50E-01	(1.0)	9.88E-07	(2.5)	8.61E-10	1.38E-07
L-CSR	1	2.70E-03	(4.0)	9.88E-07	(2.5)	1.02E-05	8.04E-04

ROOT CAUSE 11 RUN 2-10-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	4	1.48E-06	(1.0)	2.43E-08	2.42E-06

ROOT CAUSE 11 RUN 2-10-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-11 WITH TOP EVENT FREQUENCY 4.92E-07

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	1	4	1.48E-06	0.00000	L-CSR	*	NRP	*	PDAMC	*	QITG

Root Cause 12

ROOT CAUSE 12 RUN 2-10-92

TOP EVENT ROOT-CAUSE-12 CONTAINS 3 EVENTS IN 3 CUT SETS

THE FREQUENCY OF TOP EVENT ROOT-CAUSE-12 IS 4.88E-05

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ROOT-CAUSE-12

N	1000
MEAN	4.72E-05
STD DEV	1.07E-04
LOWER 5%	2.40E-07
LOWER 25%	1.17E-06
MEDIAN	3.93E-06
UPPER 25%	2.49E-05
UPPER 5%	3.47E-04

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =    2.40E-07                                    ***LOG SCALE***                                  95% =    3.47E-04
I----------------------[-----------------*--------------------------]---------MN---------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 12 RUN 2-10-92

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
D	1	4.78E-05	(1.0)	1.00E+00	(2.0)	7.49E-06	(1.0)
E	1	4.30E-07	(3.0)	1.00E+00	(2.0)	1.35E-07	(2.0)
A	1	5.25E-07	(2.0)	1.00E+00	(2.0)	8.74E-08	(3.0)

ROOT CAUSE 12 RUN 2-10-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
D	1	4.78E-05	(1.0)	1.00E+00	(2.0)	8.86E-08	3.46E-04
E	1	4.30E-07	(3.0)	1.00E+00	(2.0)	1.95E-09	1.03E-06
A	1	5.25E-07	(2.0)	1.00E+00	(2.0)	2.54E-09	1.37E-06

ROOT CAUSE 12 RUN 2-10-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
D	1	4.78E-05	(1.0)	7.49E-06	(1.0)	1.00E+00	1.00E+00
E	1	4.30E-07	(3.0)	1.35E-07	(2.0)	1.00E+00	1.00E+00
A	1	5.25E-07	(2.0)	8.74E-08	(3.0)	1.00E+00	1.00E+00

ROOT CAUSE 12 RUN 2-10-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
2	1	5.53E-05	(1.0)	8.86E-08	3.46E-04
1	1	6.12E-07	(2.0)	2.54E-09	1.37E-06
3	1	5.65E-07	(3.0)	1.95E-09	1.03E-06

ROOT CAUSE 12 RUN 2-10-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ROOT-CAUSE-12 WITH TOP EVENT FREQUENCY 4.88E-05

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	2	1	5.53E-05	0.00000	D	+
3	1	1	6.12E-07	0.00000	A	+
4	3	1	5.65E-07	0.00000	E	.

Root Cause 13

ROOT CAUSE 13 RERUN 5-8-92

TOP EVENT ANO-RT-CAUSE-13 CONTAINS 3 EVENTS IN 1 CUT SETS

THE FREQUENCY OF TOP EVENT ANO-RT-CAUSE-13 IS 2.50E-06

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ANO-RT-CAUSE-13

N	1000
MEAN	2.93E-06
STD DEV	5.41E-06
LOWER 5%	5.67E-08
LOWER 25%	3.18E-07
MEDIAN	9.84E-07
UPPER 25%	3.07E-06
UPPER 5%	1.25E-05

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =  5.67E-08                              ***LOG SCALE***                              95% =   1.25E-05
I--------------------------------[---------------------*-----------------N--M]--------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

ROOT CAUSE 13 RERUN 6-8-92

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
PDAMC	1	3.00E-03	(3.0)	2.28E-03	(1.0)	1.11E-05	(2.0)
L-UNW	1	1.30E-02	(2.0)	3.61E-04	(2.0)	1.11E-05	(2.0)
NRP	1	6.40E-02	(1.0)	1.50E-04	(3.0)	1.11E-05	(2.0)

ROOT CAUSE 13 RERUN 5-8-92

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
PDAMC	1	3.00E-03	(3.0)	2.28E-03	(1.0)	5.67E-08	1.25E-05
L-UNW	1	1.30E-02	(2.0)	3.61E-04	(2.0)	5.67E-08	1.25E-05
NRP	1	6.40E-02	(1.0)	1.50E-04	(3.0)	5.67E-08	1.25E-05

ROOT CAUSE 13 RERUN 5-8-92

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
NRP	1	6.40E-02	(1.0)	1.11E-05	(2.0)	1.77E-06	1.67E-04
L-UNW	1	1.30E-02	(2.0)	1.11E-05	(2.0)	1.05E-05	8.26E-04
PDAMC	1	3.00E-03	(3.0)	1.11E-05	(2.0)	4.89E-05	3.03E-03

ROOT CAUSE 13 RERUN 5-8-92

CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%
1	3	1.36E-05	(1.0)	5.67E-08	1.25E-05

ROOT CAUSE 13 RERUN 5-8-92

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ANO-RT-CAUSE-13 WITH TOP EVENT FREQUENCY 2.50E-06

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	1	3	1.36E-05	0.00000	L-UNW	* NRP	* PDAMC	.

APPENDIX B

Uncertainty Analysis
Risk

Composite Uncertainty Analysis - Risk

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

TOP EVENT ALL-RSK-UNC CONTAINS 18 EVENTS IN 35 CUT SETS

THE FREQUENCY OF TOP EVENT ALL-RSK-UNC IS 8.50E+01

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT ALL-RSK-UNC

N	1000
MEAN	1.01E+02
STD DEV	1.61E+02
LOWER 5%	5.64E+00
LOWER 25%	1.39E+01
MEDIAN	2.76E+01
UPPER 25%	6.94E+01
UPPER 5%	5.01E+02

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   5.64E+00                                    ***LOG SCALE***                                  95% =   5.01E+02
I-------------------[----------------*--------------------]----N--M-------------------------------------I
```

NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-4	5	2.28E-06	(8.0)	1.80E+06	(4.0)	7.06E-01	(11.0)
RC-CM-6	5	1.36E-06	(10.0)	1.80E+06	(4.0)	1.58E+00	(9.0)
RC-CM-7-8	5	1.46E-06	(9.0)	1.80E+06	(4.0)	8.32E-01	(10.0)
RC-CM-7-8-9	5	9.59E-09	(12.0)	1.80E+06	(4.0)	1.23E-02	(12.0)
RC-CM-11	5	6.44E-07	(11.0)	1.80E+06	(4.0)	3.39E+00	(8.0)
RC-CM-12	5	4.72E-05	(6.0)	1.80E+06	(4.0)	3.59E+01	(5.0)
RC-CM-13	5	2.93E-06	(7.0)	1.80E+06	(4.0)	5.31E+00	(7.0)
FM4	7	1.80E-01	(2.0)	7.04E+01	(8.0)	3.21E+02	(3.0)
FM3	7	1.40E-02	(3.0)	7.67E+00	(9.0)	5.40E+02	(1.0)
FM5	7	2.50E-01	(1.0)	6.43E+00	(10.0)	1.93E+01	(6.0)
FM2	7	2.00E-03	(4.0)	4.69E-01	(11.0)	2.34E+02	(4.0)
FM1	7	1.00E-04	(5.0)	5.25E-02	(12.0)	5.25E+02	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	35	2.00E+01	(6.0)	8.50E+01	(1.0)
IE-R	7	2.30E+04	(5.0)	1.32E-04	(2.0)
IE-Q	7	3.50E+05	(3.0)	9.54E-05	(3.0)
IE-P	7	4.90E+05	(1.0)	7.42E-06	(4.0)
IE-N	7	2.10E+05	(4.0)	1.06E-06	(5.0)
IE-M	7	4.70E+05	(2.0)	5.30E-08	(6.0)

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-4	5	2.28E-06	(8.0)	1.80E+06	(4.0)	1.32E-01	1.37E+01
RC-CM-6	5	1.36E-06	(10.0)	1.80E+06	(4.0)	8.01E-02	7.34E+00
RC-CM-7-8	5	1.46E-06	(9.0)	1.80E+06	(4.0)	3.59E-02	1.23E+01
RC-CM-7-8-9	5	9.59E-09	(12.0)	1.80E+06	(4.0)	2.68E-05	9.27E-02
RC-CM-11	5	6.44E-07	(11.0)	1.80E+06	(4.0)	6.44E-02	3.60E+00
RC-CM-12	5	4.72E-05	(6.0)	1.80E+06	(4.0)	7.11E-01	4.86E+02
RC-CM-13	5	2.93E-06	(7.0)	1.80E+06	(4.0)	1.80E-01	1.84E+01
FM4	7	1.80E-01	(2.0)	7.04E+01	(8.0)		
FM3	7	1.40E-02	(3.0)	7.67E+00	(9.0)		
FM5	7	2.50E-01	(1.0)	6.43E+00	(10.0)		
FM2	7	2.00E-03	(4.0)	4.69E-01	(11.0)		
FM1	7	1.00E-04	(5.0)	5.25E-02	(12.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	35	2.00E+01	(6.0)	8.50E+01	(1.0)		
IE-R	7	2.30E+04	(5.0)	1.32E-04	(2.0)	4.58E-01	4.26E+01
IE-Q	7	3.50E+05	(3.0)	9.54E-05	(3.0)	4.71E+00	4.14E+02
IE-P	7	4.90E+05	(1.0)	7.42E-06	(4.0)	5.06E-01	4.17E+01
IE-N	7	2.10E+05	(4.0)	1.06E-06	(5.0)	3.07E-02	2.59E+00
IE-M	7	4.70E+05	(2.0)	5.30E-08	(6.0)	3.43E-03	2.93E-01

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	7	1.40E-02	(3.0)	5.40E+02	(1.0)		
FM1	7	1.00E-04	(5.0)	5.25E+02	(2.0)		
FM4	7	1.80E-01	(2.0)	3.21E+02	(3.0)		
FM2	7	2.00E-03	(4.0)	2.34E+02	(4.0)		
RC-CM-12	5	4.72E-05	(6.0)	3.59E+01	(5.0)	1.29E+06	2.20E+06
FM5	7	2.50E-01	(1.0)	1.93E+01	(6.0)		
RC-CM-13	5	2.93E-06	(7.0)	5.31E+00	(7.0)	1.29E+06	2.20E+06
RC-CM-11	5	6.44E-07	(11.0)	3.39E+00	(8.0)	1.29E+06	2.20E+06
RC-CM-6	5	1.36E-06	(10.0)	1.58E+00	(9.0)	1.29E+06	2.20E+06
RC-CM-7-8	5	1.46E-06	(9.0)	8.32E-01	(10.0)	1.29E+06	2.20E+06
RC-CM-4	5	2.28E-06	(8.0)	7.06E-01	(11.0)	1.29E+06	2.20E+06
RC-CM-7-8-9	5	9.59E-09	(12.0)	1.23E-02	(12.0)	1.29E+06	2.20E+06

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-4	4.0	11.0
RC-CM-6	4.0	9.0
RC-CM-7-8	4.0	10.0
RC-CM-7-8-9	4.0	12.0
RC-CM-11	4.0	8.0
RC-CM-12	4.0	5.0
RC-CM-13	4.0	7.0
FM4	8.0	3.0
FM3	9.0	1.0
FM5	10.0	6.0
FM2	11.0	4.0
FM1	12.0	2.0

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

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RISK INCR  -0.7407

             RISK RED
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THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .05 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
19	4	8.88E+01	(1.0)	5.94E-01	4.01E+02	1.0447	1.0447	0.0362	0.8159
18	4	9.61E+00	(2.0)	6.23E-02	3.99E+01	0.1131	1.1577	0.0040	0.1025
20	4	8.73E+00	(3.0)	5.66E-02	4.14E+01	0.1026	1.2604	0.0035	0.1141
24	4	8.03E+00	(4.0)	1.49E-01	1.53E+01	0.0945	1.3549	0.0012	0.5001
14	4	3.58E+00	(5.0)	5.30E-02	3.01E+00	0.0422	1.3970	0.0004	0.1606
4	4	3.45E+00	(6.0)	1.09E-01	1.12E+01	0.0406	1.4376	0.0008	0.4005
9	4	3.00E+00	(7.0)	6.52E-02	6.04E+00	0.0353	1.4729	0.0006	0.2836
29	4	2.52E+00	(8.0)	2.92E-02	1.03E+01	0.0296	1.5025	0.0002	0.3929
23	4	8.66E-01	(9.0)	1.59E-02	1.53E+00	0.0102	1.5127	0.0001	0.0546
25	4	8.25E-01	(10.0)	1.49E-02	1.61E+00	0.0097	1.5224	0.0001	0.0553
17	4	5.16E-01	(11.0)	3.75E-03	2.53E+00	0.0061	1.5285	0.0002	0.0062
15	4	3.86E-01	(12.0)	5.07E-03	3.17E-01	0.0045	1.5330	0.0000	0.0177
13	4	3.85E-01	(13.0)	6.04E-03	2.97E-01	0.0045	1.5375	0.0000	0.0180
3	4	3.74E-01	(14.0)	1.09E-02	1.13E+00	0.0044	1.5419	0.0001	0.0434
5	4	3.27E-01	(15.0)	9.32E-03	1.19E+00	0.0038	1.5458	0.0001	0.0456
8	4	3.24E-01	(16.0)	6.94E-03	6.71E-01	0.0038	1.5496	0.0001	0.0322
10	4	3.01E-01	(17.0)	6.18E-03	6.60E-01	0.0035	1.5531	0.0001	0.0310
28	4	2.73E-01	(18.0)	3.05E-03	1.02E+00	0.0032	1.5564	0.0000	0.0408
30	4	2.44E-01	(19.0)	2.67E-03	1.04E+00	0.0029	1.5592	0.0000	0.0372
16	4	6.56E-02	(20.0)	4.01E-04	2.81E-01	0.0008	1.5600	0.0000	0.0007
22	4	4.23E-02	(21.0)	9.68E-04	9.45E-02	0.0005	1.5605	0.0000	0.0034
34	4	2.21E-02	(22.0)	2.24E-05	7.43E-02	0.0003	1.5608	0.0000	0.0041
2	4	2.15E-02	(23.0)	6.92E-04	6.89E-02	0.0003	1.5610	0.0000	0.0027
12	4	1.67E-02	(24.0)	3.33E-04	1.81E-02	0.0002	1.5612	0.0000	0.0011
7	4	1.67E-02	(25.0)	4.02E-04	3.97E-02	0.0002	1.5614	0.0000	0.0020
27	4	1.50E-02	(26.0)	1.86E-04	6.56E-02	0.0002	1.5616	0.0000	0.0026
21	4	5.89E-03	(27.0)	1.11E-04	1.09E-02	0.0001	1.5617	0.0000	0.0004
11	4	2.61E-03	(28.0)	4.22E-05	2.10E-03	0.0000	1.5617	0.0000	0.0001
1	4	2.56E-03	(29.0)	7.48E-05	8.13E-03	0.0000	1.5617	0.0000	0.0003
33	4	2.39E-03	(30.0)	2.39E-06	8.47E-03	0.0000	1.5617	0.0000	0.0005
35	4	2.23E-03	(31.0)	1.97E-06	7.83E-03	0.0000	1.5618	0.0000	0.0005
6	4	2.21E-03	(32.0)	5.03E-05	4.62E-03	0.0000	1.5618	0.0000	0.0002
26	4	1.86E-03	(33.0)	2.16E-05	7.18E-03	0.0000	1.5618	0.0000	0.0003
32	4	1.22E-04	(34.0)	1.40E-07	4.77E-04	0.0000	1.5618	0.0000	0.0000
31	4	1.63E-05	(35.0)	1.61E-08	5.47E-05	0.0000	1.5618	0.0000	0.0000

RISK UNCERTAINTY ROOT CAUSE COMPOSITE RUN (5-8-92)

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT ALL-RSK-UNC WITH TOP EVENT FREQUENCY 8.50E+01

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	19	4	8.88E+01	1.04467	FM4	* IE-20	* IE-Q	* RC-CM-12	+
3	18	4	9.61E+00	1.15774	FM3	* IE-20	* IE-P	* RC-CM-12	+
4	20	4	8.73E+00	1.26038	FM5	* IE-20	* IE-R	* RC-CM-12	+
5	24	4	8.03E+00	1.35485	FM4	* IE-20	* IE-Q	* RC-CM-13	+
6	14	4	3.58E+00	1.39701	FM4	* IE-20	* IE-Q	* RC-CM-11	+
7	4	4	3.45E+00	1.43758	FM4	* IE-20	* IE-Q	* RC-CM-4	+
8	9	4	3.00E+00	1.47288	FM4	* IE-20	* IE-Q	* RC-CM-6	+
9	29	4	2.52E+00	1.50252	FM4	* IE-20	* IE-Q	* RC-CM-7-8	+
10	23	4	8.66E-01	1.51270	FM3	* IE-20	* IE-P	* RC-CM-13	+
11	25	4	8.25E-01	1.52240	FM5	* IE-20	* IE-R	* RC-CM-13	+
12	17	4	5.16E-01	1.52848	FM2	* IE-20	* IE-N	* RC-CM-12	+
13	15	4	3.86E-01	1.53301	FM5	* IE-20	* IE-R	* RC-CM-11	+
14	13	4	3.85E-01	1.53754	FM3	* IE-20	* IE-P	* RC-CM-11	+
15	3	4	3.74E-01	1.54194	FM3	* IE-20	* IE-P	* RC-CM-4	+
16	5	4	3.27E-01	1.54579	FM5	* IE-20	* IE-R	* RC-CM-4	+
17	8	4	3.24E-01	1.54960	FM3	* IE-20	* IE-P	* RC-CM-6	+
18	10	4	3.01E-01	1.55315	FM5	* IE-20	* IE-R	* RC-CM-6	+
19	28	4	2.73E-01	1.55636	FM3	* IE-20	* IE-P	* RC-CM-7-8	+
20	30	4	2.44E-01	1.55923	FM5	* IE-20	* IE-R	* RC-CM-7-8	+
21	16	4	6.56E-02	1.56000	FM1	* IE-20	* IE-M	* RC-CM-12	+
22	22	4	4.23E-02	1.56050	FM2	* IE-20	* IE-N	* RC-CM-13	+
23	34	4	2.21E-02	1.56076	FM4	* IE-20	* IE-Q	* RC-CM-7-8-9	+
24	2	4	2.15E-02	1.56101	FM2	* IE-20	* IE-N	* RC-CM-4	+
25	12	4	1.67E-02	1.56121	FM2	* IE-20	* IE-N	* RC-CM-11	+
26	7	4	1.67E-02	1.56141	FM2	* IE-20	* IE-N	* RC-CM-6	+
27	27	4	1.50E-02	1.56158	FM2	* IE-20	* IE-N	* RC-CM-7-8	+
28	21	4	5.89E-03	1.56165	FM1	* IE-20	* IE-M	* RC-CM-13	+
29	11	4	2.61E-03	1.56168	FM1	* IE-20	* IE-M	* RC-CM-11	+
30	1	4	2.56E-03	1.56171	FM1	* IE-20	* IE-M	* RC-CM-4	+
31	33	4	2.39E-03	1.56174	FM3	* IE-20	* IE-P	* RC-CM-7-8-9	+
32	35	4	2.23E-03	1.56177	FM5	* IE-20	* IE-R	* RC-CM-7-8-9	+
33	6	4	2.21E-03	1.56179	FM1	* IE-20	* IE-M	* RC-CM-6	+
34	26	4	1.86E-03	1.56181	FM1	* IE-20	* IE-M	* RC-CM-7-8	+
35	32	4	1.22E-04	1.56182	FM2	* IE-20	* IE-N	* RC-CM-7-8-9	+
36	31	4	1.63E-05	1.56182	FM1	* IE-20	* IE-M	* RC-CM-7-8-9	.

Root Cause 4

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

TOP EVENT RC4-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC4-RSK-UNC IS 3.47E+00

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC4-RSK-UNC

N	1000
MEAN	3.36E+00
STD DEV	4.40E+00
LOWER 5%	1.32E-01
LOWER 25%	3.72E-01
MEDIAN	1.28E+00
UPPER 25%	4.11E+00
UPPER 5%	1.37E+01

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

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5% =   1.32E-01                                ***LOG SCALE***                                95% =   1.37E+01
I----------------------[---------------------------*--------------------MN---]--------------------------I
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NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-4	5	2.28E-06	(6.0)	1.80E+06	(1.0)	7.06E-01	(6.0)
FM4	1	1.80E-01	(2.0)	2.87E+00	(2.0)	1.31E+01	(3.0)
FM3	1	1.40E-02	(3.0)	3.13E-01	(3.0)	2.20E+01	(1.0)
FM5	1	2.50E-01	(1.0)	2.62E-01	(4.0)	7.87E-01	(5.0)
FM2	1	2.00E-03	(4.0)	1.92E-02	(5.0)	9.56E+00	(4.0)
FM1	1	1.00E-04	(5.0)	2.14E-03	(6.0)	2.14E+01	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	3.47E+00	(1.0)
IE-R	1	2.30E+04	(5.0)	1.96E-06	(2.0)
IE-Q	1	3.50E+05	(3.0)	1.41E-06	(3.0)
IE-P	1	4.90E+05	(1.0)	1.10E-07	(4.0)
IE-N	1	2.10E+05	(4.0)	1.57E-08	(5.0)
IE-M	1	4.70E+05	(2.0)	7.83E-10	(6.0)

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-4	5	2.28E-06	(6.0)	1.80E+06	(1.0)	1.32E-01	1.37E+01
FM4	1	1.80E-01	(2.0)	2.87E+00	(2.0)		
FM3	1	1.40E-02	(3.0)	3.13E-01	(3.0)		
FM5	1	2.50E-01	(1.0)	2.62E-01	(4.0)		
FM2	1	2.00E-03	(4.0)	1.92E-02	(5.0)		
FM1	1	1.00E-04	(5.0)	2.14E-03	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	3.47E+00	(1.0)		
IE-R	1	2.30E+04	(5.0)	1.96E-06	(2.0)	9.32E-03	1.19E+00
IE-Q	1	3.50E+05	(3.0)	1.41E-06	(3.0)	1.09E-01	1.12E+01
IE-P	1	4.90E+05	(1.0)	1.10E-07	(4.0)	1.09E-02	1.13E+00
IE-N	1	2.10E+05	(4.0)	1.57E-08	(5.0)	6.92E-04	6.89E-02
IE-M	1	4.70E+05	(2.0)	7.83E-10	(6.0)	7.48E-05	8.13E-03

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	2.20E+01	(1.0)		
FM1	1	1.00E-04	(5.0)	2.14E+01	(2.0)		
FM4	1	1.80E-01	(2.0)	1.31E+01	(3.0)		
FM2	1	2.00E-03	(4.0)	9.56E+00	(4.0)		
FM5	1	2.50E-01	(1.0)	7.87E-01	(5.0)		
RC-CM-4	5	2.28E-06	(6.0)	7.06E-01	(6.0)	1.29E+06	2.20E+06

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-4	1.0	6.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	5.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR -Ø.336Ø

RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .Ø5 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	3.45E+00	(1.0)	1.09E-01	1.12E+01	0.9944	0.9944	0.7511	0.8751
3	4	3.74E-01	(2.0)	1.09E-02	1.13E+00	0.1079	1.1023	0.0609	0.1231
5	4	3.27E-01	(3.0)	9.32E-03	1.19E+00	0.0943	1.1966	0.0452	0.1396
2	4	2.15E-02	(4.0)	6.92E-04	6.89E-02	0.0062	1.2028	0.0035	0.0076
1	4	2.56E-03	(5.0)	7.48E-05	8.13E-03	0.0007	1.2035	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 4 RUN (5-8-92)

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC4-RSK-UNC WITH TOP EVENT FREQUENCY 3.47E+00

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	3.45E+00	0.99439	FM4	*	IE-20	*	IE-Q	*	RC-CM-4	+
3	3	4	3.74E-01	1.10234	FM3	*	IE-20	*	IE-P	*	RC-CM-4	+
4	5	4	3.27E-01	1.19661	FM5	*	IE-20	*	IE-R	*	RC-CM-4	+
5	2	4	2.15E-02	1.20281	FM2	*	IE-20	*	IE-N	*	RC-CM-4	+
6	1	4	2.56E-03	1.20355	FM1	*	IE-20	*	IE-M	*	RC-CM-4	.

Root Cause 6

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

TOP EVENT RC6-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC6-RSK-UNC IS 2.07E+00

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC6-RSK-UNC

N	1000
MEAN	1.91E+00
STD DEV	2.47E+00
LOWER 5%	8.01E-02
LOWER 25%	2.55E-01
MEDIAN	7.26E-01
UPPER 25%	2.24E+00
UPPER 5%	7.34E+00

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

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5% =    8.01E-02                                 ***LOG SCALE***                                  95% =    7.34E+00
I-------------------------[-----------------------*--------------------M--N-]-------------------------I
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NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-6	5	1.36E-06	(6.0)	1.80E+06	(1.0)	1.58E+00	(5.0)
FM4	1	1.80E-01	(2.0)	1.71E+00	(2.0)	7.81E+00	(3.0)
FM3	1	1.40E-02	(3.0)	1.87E-01	(3.0)	1.31E+01	(1.0)
FM5	1	2.50E-01	(1.0)	1.56E-01	(4.0)	4.69E-01	(6.0)
FM2	1	2.00E-03	(4.0)	1.14E-02	(5.0)	5.70E+00	(4.0)
FM1	1	1.00E-04	(5.0)	1.28E-03	(6.0)	1.28E+01	(2.0)

RISK REDUCTIONS BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	2.07E+00	(1.0)
IE-R	1	2.30E+04	(5.0)	4.37E-06	(2.0)
IE-Q	1	3.50E+05	(3.0)	3.15E-06	(3.0)
IE-P	1	4.90E+05	(1.0)	2.45E-07	(4.0)
IE-N	1	2.10E+05	(4.0)	3.50E-08	(5.0)
IE-M	1	4.70E+05	(2.0)	1.75E-09	(6.0)

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-6	5	1.36E-06	(6.0)	1.80E+06	(1.0)	8.01E-02	7.34E+00
FM4	1	1.80E-01	(2.0)	1.71E+00	(2.0)		
FM3	1	1.40E-02	(3.0)	1.87E-01	(3.0)		
FM5	1	2.50E-01	(1.0)	1.56E-01	(4.0)		
FM2	1	2.00E-03	(4.0)	1.14E-02	(5.0)		
FM1	1	1.00E-04	(5.0)	1.28E-03	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	2.07E+00	(1.0)		
IE-R	1	2.30E+04	(5.0)	4.37E-06	(2.0)	6.18E-03	6.60E-01
IE-Q	1	3.50E+05	(3.0)	3.15E-06	(3.0)	6.52E-02	6.04E+00
IE-P	1	4.90E+05	(1.0)	2.45E-07	(4.0)	6.94E-03	6.71E-01
IE-N	1	2.10E+05	(4.0)	3.50E-08	(5.0)	4.02E-04	3.97E-02
IE-M	1	4.70E+05	(2.0)	1.75E-09	(6.0)	5.03E-05	4.62E-03

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	1.31E+01	(1.0)		
FM1	1	1.00E-04	(5.0)	1.28E+01	(2.0)		
FM4	1	1.80E-01	(2.0)	7.81E+00	(3.0)		
FM2	1	2.00E-03	(4.0)	5.70E+00	(4.0)		
RC-CM-6	5	1.36E-06	(6.0)	1.58E+00	(5.0)	1.29E+06	2.20E+06
FM5	1	2.50E-01	(1.0)	4.69E-01	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-6	1.0	5.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR -0.2327

RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .05 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	3.00E+00	(1.0)	6.52E-02	6.04E+00	1.4507	1.4507	0.7511	0.8751
3	4	3.24E-01	(2.0)	6.94E-03	6.71E-01	0.1567	1.6074	0.0609	0.1231
5	4	3.01E-01	(3.0)	6.18E-03	6.60E-01	0.1456	1.7530	0.0452	0.1396
2	4	1.67E-02	(4.0)	4.02E-04	3.97E-02	0.0081	1.7611	0.0035	0.0077
1	4	2.21E-03	(5.0)	5.03E-05	4.62E-03	0.0011	1.7621	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 6 RUN (5-8-92)

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC6-RSK-UNC   WITH TOP EVENT FREQUENCY   2.07E+00

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	3.00E+00	1.45070	FM4	*	IE-20	*	IE-Q	*	RC-CM-6	+
3	3	4	3.24E-01	1.60743	FM3	*	IE-20	*	IE-P	*	RC-CM-6	+
4	5	4	3.01E-01	1.75299	FM5	*	IE-20	*	IE-R	*	RC-CM-6	+
5	2	4	1.67E-02	1.76106	FM2	*	IE-20	*	IE-N	*	RC-CM-6	+
6	1	4	2.21E-03	1.76212	FM1	*	IE-20	*	IE-M	*	RC-CM-6	.

Root Cause 7/8

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

TOP EVENT RC78-RSK-UNC   CONTAINS   12 EVENTS IN   5 CUT SETS

THE FREQUENCY OF TOP EVENT RC78-RSK-UNC   IS   2.22E+00

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC78-RSK-UNC

```
N                1000
MEAN         2.61E+00
STD DEV      4.03E+00
LOWER 5%     3.59E-02
LOWER 25%    1.10E-01
MEDIAN       4.55E-01
UPPER 25%    2.78E+00
UPPER 5%     1.23E+01
```

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   3.59E-02                              ***LOG SCALE***                              95% =   1.23E+01
I------------------[-------------------------*----------------------------N-M-]-------------------------I
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NOMENCLATURE:

PD   = PARTIAL DERIVATIVE

TEF   = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
       = FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

```
RISK REDUCTION = PD x EV(J)
               = TEF - TEF(EVALUATED WITH EV(J) = 0)
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2. FOR BASE EVENTS ONLY:

```
RISK INCREASE  = PD - RISK REDUCTION
               = PD x (1 - EV(J))
               = TEF(EVALUATED WITH EV(J) = 1) - TEF
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RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-7-8	5	1.46E-06	(6.0)	1.80E+06	(1.0)	8.32E-01	(5.0)
FM4	1	1.80E-01	(2.0)	1.84E+00	(2.0)	8.38E+00	(3.0)
FM3	1	1.40E-02	(3.0)	2.00E-01	(3.0)	1.41E+01	(1.0)
FM5	1	2.50E-01	(1.0)	1.68E-01	(4.0)	5.04E-01	(6.0)
FM2	1	2.00E-03	(4.0)	1.23E-02	(5.0)	6.12E+00	(4.0)
FM1	1	1.00E-04	(5.0)	1.37E-03	(6.0)	1.37E+01	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	2.22E+00	(1.0)
IE-R	1	2.30E+04	(5.0)	2.31E-06	(2.0)
IE-Q	1	3.50E+05	(3.0)	1.66E-06	(3.0)
IE-P	1	4.90E+05	(1.0)	1.29E-07	(4.0)
IE-N	1	2.10E+05	(4.0)	1.85E-08	(5.0)
IE-M	1	4.70E+05	(2.0)	9.23E-10	(6.0)

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-7-8	5	1.46E-06	(6.0)	1.80E+06	(1.0)	3.59E-02	1.23E+01
FM4	1	1.80E-01	(2.0)	1.84E+00	(2.0)		
FM3	1	1.40E-02	(3.0)	2.00E-01	(3.0)		
FM5	1	2.50E-01	(1.0)	1.68E-01	(4.0)		
FM2	1	2.00E-03	(4.0)	1.23E-02	(5.0)		
FM1	1	1.00E-04	(5.0)	1.37E-03	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	2.22E+00	(1.0)		
IE-R	1	2.30E+04	(5.0)	2.31E-06	(2.0)	2.67E-03	1.04E+00
IE-Q	1	3.50E+05	(3.0)	1.66E-06	(3.0)	2.92E-02	1.03E+01
IE-P	1	4.90E+05	(1.0)	1.29E-07	(4.0)	3.05E-03	1.02E+00
IE-N	1	2.10E+05	(4.0)	1.85E-08	(5.0)	1.86E-04	6.55E-02
IE-M	1	4.70E+05	(2.0)	9.23E-10	(6.0)	2.16E-05	7.18E-03

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	1.41E+01	(1.0)		
FM1	1	1.00E-04	(5.0)	1.37E+01	(2.0)		
FM4	1	1.80E-01	(2.0)	8.38E+00	(3.0)		
FM2	1	2.00E-03	(4.0)	6.12E+00	(4.0)		
RC-CM-7-8	5	1.46E-06	(6.0)	8.32E-01	(5.0)	1.29E+06	2.20E+06
FM5	1	2.50E-01	(1.0)	5.04E-01	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-7-8	1.0	5.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR	-Ø.2327
	RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .Ø5 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	2.52E+00	(1.0)	2.92E-02	1.03E+01	1.1342	1.1342	0.7511	0.8751
3	4	2.73E-01	(2.0)	3.05E-03	1.02E+00	0.1229	1.2571	0.0609	0.1231
5	4	2.44E-01	(3.0)	2.67E-03	1.04E+00	0.1100	1.3671	0.0452	0.1396
2	4	1.50E-02	(4.0)	1.86E-04	6.55E-02	0.0068	1.3738	0.0035	0.0077
1	4	1.86E-03	(5.0)	2.16E-05	7.18E-03	0.0008	1.3747	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 7-8 RUN

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC78-RSK-UNC   WITH TOP EVENT FREQUENCY   2.22E+00

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	2.52E+00	1.13418	FM4	*	IE-20	*	IE-Q	*	RC-CM-7-8	+
3	3	4	2.73E-01	1.25708	FM3	*	IE-20	*	IE-P	*	RC-CM-7-8	+
4	5	4	2.44E-01	1.36706	FM5	*	IE-20	*	IE-R	*	RC-CM-7-8	+
5	2	4	1.50E-02	1.37383	FM2	*	IE-20	*	IE-N	*	RC-CM-7-8	+
6	1	4	1.86E-03	1.37467	FM1	*	IE-20	*	IE-M	*	RC-CM-7-8	.

Root Cause 8/9

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

TOP EVENT RC789-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC789-RSK-UNC IS 1.46E-02

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC789-RSK-UNC

N	1000
MEAN	1.74E-02
STD DEV	3.08E-02
LOWER 5%	2.68E-05
LOWER 25%	1.20E-04
MEDIAN	7.22E-04
UPPER 25%	1.36E-02
UPPER 5%	9.27E-02

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =    2.68E-05                             ***LOG SCALE***                                  95% =    9.27E-02
I------------------[----------------------*------------------------------------N-M---------------------I
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NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-7-8-9	5	9.59E-09	(6.0)	1.80E+06	(1.0)	1.23E-02	(5.0)
FM4	1	1.80E-01	(2.0)	1.21E-02	(2.0)	5.50E-02	(3.0)
FM3	1	1.40E-02	(3.0)	1.32E-03	(3.0)	9.27E-02	(1.0)
FM5	1	2.50E-01	(1.0)	1.10E-03	(4.0)	3.31E-03	(6.0)
FM2	1	2.00E-03	(4.0)	8.06E-05	(5.0)	4.02E-02	(4.0)
FM1	1	1.00E-04	(5.0)	9.01E-06	(6.0)	9.01E-02	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	1.46E-02	(1.0)
IE-R	1	2.30E+04	(5.0)	3.41E-08	(2.0)
IE-Q	1	3.50E+05	(3.0)	2.46E-08	(3.0)
IE-P	1	4.90E+05	(1.0)	1.91E-09	(4.0)
IE-N	1	2.10E+05	(4.0)	2.73E-10	(5.0)
IE-M	1	4.70E+05	(2.0)	1.36E-11	(6.0)

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-7-8-9	5	9.59E-09	(6.0)	1.80E+06	(1.0)	2.68E-05	9.27E-02
FM4	1	1.80E-01	(2.0)	1.21E-02	(2.0)		
FM3	1	1.40E-02	(3.0)	1.32E-03	(3.0)		
FM5	1	2.50E-01	(1.0)	1.10E-03	(4.0)		
FM2	1	2.00E-03	(4.0)	8.06E-05	(5.0)		
FM1	1	1.00E-04	(5.0)	9.01E-06	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	1.46E-02	(1.0)		
IE-R	1	2.30E+04	(5.0)	3.41E-08	(2.0)	1.97E-06	7.83E-03
IE-Q	1	3.50E+05	(3.0)	2.46E-08	(3.0)	2.24E-05	7.43E-02
IE-P	1	4.90E+05	(1.0)	1.91E-09	(4.0)	2.39E-06	8.47E-03
IE-N	1	2.10E+05	(4.0)	2.73E-10	(5.0)	1.40E-07	4.77E-04
IE-M	1	4.70E+05	(2.0)	1.36E-11	(6.0)	1.61E-08	5.47E-05

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	9.27E-02	(1.0)		
FM1	1	1.00E-04	(5.0)	9.01E-02	(2.0)		
FM4	1	1.80E-01	(2.0)	5.50E-02	(3.0)		
FM2	1	2.00E-03	(4.0)	4.02E-02	(4.0)		
RC-CM-7-8-9	5	9.59E-09	(6.0)	1.23E-02	(5.0)	1.29E+06	2.20E+06
FM5	1	2.50E-01	(1.0)	3.31E-03	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-7-8-9	1.0	5.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR -0.2327

RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .05 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	2.21E-02	(1.0)	2.24E-05	7.43E-02	1.5171	1.5171	0.7511	0.8751
3	4	2.39E-03	(2.0)	2.39E-06	8.47E-03	0.1638	1.6810	0.0609	0.1231
5	4	2.23E-03	(3.0)	1.97E-06	7.83E-03	0.1530	1.8340	0.0452	0.1396
2	4	1.22E-04	(4.0)	1.40E-07	4.77E-04	0.0083	1.8423	0.0035	0.0077
1	4	1.63E-05	(5.0)	1.61E-08	5.47E-05	0.0011	1.8434	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 7-8-9 RUN

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC789-RSK-UNC WITH TOP EVENT FREQUENCY 1.46E-02

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	2.21E-02	1.51713	FM4	*	IE-20	*	IE-Q	*	RC-CM-7-8-9	+
3	3	4	2.39E-03	1.68096	FM3	*	IE-20	*	IE-P	*	RC-CM-7-8-9	+
4	5	4	2.23E-03	1.83399	FM5	*	IE-20	*	IE-R	*	RC-CM-7-8-9	+
5	2	4	1.22E-04	1.84233	FM2	*	IE-20	*	IE-N	*	RC-CM-7-8-9	+
6	1	4	1.63E-05	1.84344	FM1	*	IE-20	*	IE-M	*	RC-CM-7-8-9	.

Root Cause 11

RISK UNCERTAINTY ROOT CAUSE 11 RUN

TOP EVENT RC11-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC11-RSK-UNC IS 9.80E-01

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC11-RSK-UNC

N	1000
MEAN	1.00E+00
STD DEV	1.12E+00
LOWER 5%	6.44E-02
LOWER 25%	1.84E-01
MEDIAN	5.27E-01
UPPER 25%	1.38E+00
UPPER 5%	3.60E+00

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

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5% =   6.44E-02                              ***LOG SCALE***                              95% =   3.60E+00
I--------------------------[--------------------------*---------------NM-------]------------------------I
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NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 11 RUN

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-11	5	6.44E-07	(6.0)	1.80E+06	(1.0)	3.39E+00	(4.0)
FM4	1	1.80E-01	(2.0)	8.11E-01	(2.0)	3.70E+00	(3.0)
FM3	1	1.40E-02	(3.0)	8.84E-02	(3.0)	6.22E+00	(1.0)
FM5	1	2.50E-01	(1.0)	7.41E-02	(4.0)	2.22E-01	(6.0)
FM2	1	2.00E-03	(4.0)	5.41E-03	(5.0)	2.70E+00	(5.0)
FM1	1	1.00E-04	(5.0)	6.05E-04	(6.0)	6.05E+00	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	9.80E-01	(1.0)
IE-R	1	2.30E+04	(5.0)	9.41E-06	(2.0)
IE-Q	1	3.50E+05	(3.0)	6.78E-06	(3.0)
IE-P	1	4.90E+05	(1.0)	5.27E-07	(4.0)
IE-N	1	2.10E+05	(4.0)	7.53E-08	(5.0)
IE-M	1	4.70E+05	(2.0)	3.76E-09	(6.0)

RISK UNCERTAINTY ROOT CAUSE 11 RUN

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-11	5	6.44E-07	(6.0)	1.80E+06	(1.0)	6.44E-02	3.80E+00
FM4	1	1.80E-01	(2.0)	8.11E-01	(2.0)		
FM3	1	1.40E-02	(3.0)	8.84E-02	(3.0)		
FM5	1	2.50E-01	(1.0)	7.41E-02	(4.0)		
FM2	1	2.00E-03	(4.0)	5.41E-03	(5.0)		
FM1	1	1.00E-04	(5.0)	6.05E-04	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	9.80E-01	(1.0)		
IE-R	1	2.30E+04	(5.0)	9.41E-06	(2.0)	5.07E-03	3.17E-01
IE-Q	1	3.50E+05	(3.0)	6.78E-06	(3.0)	5.30E-02	3.01E+00
IE-P	1	4.90E+05	(1.0)	5.27E-07	(4.0)	6.04E-03	2.97E-01
IE-N	1	2.10E+05	(4.0)	7.53E-08	(5.0)	3.33E-04	1.81E-02
IE-M	1	4.70E+05	(2.0)	3.76E-09	(6.0)	4.22E-05	2.10E-03

RISK UNCERTAINTY ROOT CAUSE 11 RUN

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	6.22E+00	(1.0)		
FM1	1	1.00E-04	(5.0)	6.05E+00	(2.0)		
FM4	1	1.80E-01	(2.0)	3.70E+00	(3.0)		
RC-CM-11	5	6.44E-07	(6.0)	3.39E+00	(4.0)	1.29E+06	2.20E+06
FM2	1	2.00E-03	(4.0)	2.70E+00	(5.0)		
FM5	1	2.50E-01	(1.0)	2.22E-01	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 11 RUN

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-11	1.0	4.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	5.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 11 RUN

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR -0.0860

RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .05 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 11 RUN

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	3.58E+00	(1.0)	5.30E-02	3.01E+00	3.6583	3.6583	0.7511	0.8751
5	4	3.86E-01	(2.0)	5.07E-03	3.17E-01	0.3937	4.0520	0.0452	0.1396
3	4	3.85E-01	(3.0)	6.04E-03	2.97E-01	0.3927	4.4448	0.0609	0.1231
2	4	1.67E-02	(4.0)	3.33E-04	1.81E-02	0.0171	4.4619	0.0036	0.0077
1	4	2.61E-03	(5.0)	4.22E-05	2.10E-03	0.0027	4.4645	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 11 RUN

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC11-RSK-UNC WITH TOP EVENT FREQUENCY 9.80E-01

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	3.58E+00	3.65832	FM4	*	IE-20	*	IE-Q	*	RC-CM-11	+
3	5	4	3.86E-01	4.05203	FM5	*	IE-20	*	IE-R	*	RC-CM-11	+
4	3	4	3.85E-01	4.44477	FM3	*	IE-20	*	IE-P	*	RC-CM-11	+
5	2	4	1.67E-02	4.46185	FM2	*	IE-20	*	IE-N	*	RC-CM-11	+
6	1	4	2.61E-03	4.46451	FM1	*	IE-20	*	IE-M	*	RC-CM-11	.

Root Cause 12

RISK UNCERTAINTY ROOT CAUSE 12 RUN

TOP EVENT RC12-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC12-RSK-UNC IS 7.18E+01

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC12-RSK-UNC

N	1000
MEAN	8.73E+01
STD DEV	1.61E+02
LOWER 5%	7.11E-01
LOWER 25%	2.07E+00
MEDIAN	7.31E+00
UPPER 25%	4.73E+01
UPPER 5%	4.86E+02

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

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5% =   7.11E-01                                    ***LOG SCALE***                                   95% =   4.86E+02
I----------------[-------------------*----------------------------]------N-M--------------------------I
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NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 12 RUN

RISK REDUCTIONS   RISK INCREASES   BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-12	5	4.72E-05	(6.0)	1.80E+06	(1.0)	3.59E+01	(5.0)
FM4	1	1.80E-01	(2.0)	5.95E+01	(2.0)	2.71E+02	(3.0)
FM3	1	1.40E-02	(3.0)	6.48E+00	(3.0)	4.56E+02	(1.0)
FM5	1	2.50E-01	(1.0)	5.43E+00	(4.0)	1.63E+01	(6.0)
FM2	1	2.00E-03	(4.0)	3.96E-01	(5.0)	1.98E+02	(4.0)
FM1	1	1.00E-04	(5.0)	4.44E-02	(6.0)	4.44E+02	(2.0)

RISK REDUCTIONS   BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	7.18E+01	(1.0)
IE-R	1	2.30E+04	(5.0)	9.96E-05	(2.0)
IE-Q	1	3.50E+05	(3.0)	7.17E-05	(3.0)
IE-P	1	4.90E+05	(1.0)	5.58E-06	(4.0)
IE-N	1	2.10E+05	(4.0)	7.97E-07	(5.0)
IE-M	1	4.70E+05	(2.0)	3.98E-08	(6.0)

RISK UNCERTAINTY ROOT CAUSE 12 RUN

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-12	5	4.72E-05	(6.0)	1.80E+06	(1.0)	7.11E-01	4.86E+02
FM4	1	1.80E-01	(2.0)	5.95E+01	(2.0)		
FM3	1	1.40E-02	(3.0)	6.48E+00	(3.0)		
FM5	1	2.50E-01	(1.0)	5.43E+00	(4.0)		
FM2	1	2.00E-03	(4.0)	3.98E-01	(5.0)		
FM1	1	1.00E-04	(5.0)	4.44E-02	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	7.18E+01	(1.0)		
IE-R	1	2.30E+04	(5.0)	9.96E-05	(2.0)	5.66E-02	4.14E+01
IE-Q	1	3.50E+05	(3.0)	7.17E-05	(3.0)	5.94E-01	4.01E+02
IE-P	1	4.90E+05	(1.0)	5.58E-06	(4.0)	6.23E-02	3.99E+01
IE-N	1	2.10E+05	(4.0)	7.97E-07	(5.0)	3.75E-03	2.53E+00
IE-M	1	4.70E+05	(2.0)	3.98E-08	(6.0)	4.01E-04	2.81E-01

RISK UNCERTAINTY ROOT CAUSE 12 RUN

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	4.56E+02	(1.0)		
FM1	1	1.00E-04	(5.0)	4.44E+02	(2.0)		
FM4	1	1.80E-01	(2.0)	2.71E+02	(3.0)		
FM2	1	2.00E-03	(4.0)	1.98E+02	(4.0)		
RC-CM-12	5	4.72E-05	(6.0)	3.59E+01	(5.0)	1.29E+06	2.20E+06
FM5	1	2.50E-01	(1.0)	1.63E+01	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 12 RUN

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-12	1.0	5.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 12 RUN

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

RISK INCR -0.2327

RISK RED

THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .05 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 12 RUN

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	8.88E+01	(1.0)	5.94E-01	4.01E+02	1.2369	1.2369	0.7511	0.8781
3	4	9.61E+00	(2.0)	6.23E-02	3.99E+01	0.1339	1.3707	0.0609	0.1231
5	4	8.73E+00	(3.0)	5.66E-02	4.14E+01	0.1216	1.4923	0.0452	0.1396
2	4	5.16E-01	(4.0)	3.75E-03	2.53E+00	0.0072	1.4995	0.0035	0.0077
1	4	6.56E-02	(5.0)	4.01E-04	2.81E-01	0.0009	1.5004	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 12 RUN

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC12-RSK-UNC WITH TOP EVENT FREQUENCY 7.18E+01

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	8.88E+01	1.23686	FM4	*	IE-20	*	IE-Q	*	RC-CM-12	+
3	3	4	9.61E+00	1.37073	FM3	*	IE-20	*	IE-P	*	RC-CM-12	+
4	5	4	8.73E+00	1.49226	FM5	*	IE-20	*	IE-R	*	RC-CM-12	+
5	2	4	5.16E-01	1.49945	FM2	*	IE-20	*	IE-N	*	RC-CM-12	+
6	1	4	6.56E-02	1.50036	FM1	*	IE-20	*	IE-M	*	RC-CM-12	.

Root Cause 13

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

TOP EVENT RC13-RSK-UNC CONTAINS 12 EVENTS IN 5 CUT SETS

THE FREQUENCY OF TOP EVENT RC13-RSK-UNC IS 4.46E+00

DESCRIPTIVE STATISTICS FOR THE FREQUENCY OF TOP EVENT RC13-RSK-UNC

N	1000
MEAN	4.56E+00
STD DEV	5.91E+00
LOWER 5%	1.80E-01
LOWER 25%	5.71E-01
MEDIAN	1.75E+00
UPPER 25%	5.58E+00
UPPER 5%	1.84E+01

90% UNCERTAINTY INTERVAL FOR TOP EVENT FREQUENCY (INNERMOST BRACKETS DENOTE INTERQUARTILE RANGE, ASTERISK DENOTES MEDIAN, N DENOTES NOMINAL VALUE AND M DENOTES MEAN)

```
5% =   1.80E-01                                ***LOG SCALE***                                      95% =   1.84E+01
I------------------------[-------------------------*--------------------M----]--------------------------I
```


NOMENCLATURE:

PD = PARTIAL DERIVATIVE

TEF = FREQUENCY OF THE TOP EVENT

EV(J) = PROBABILITY OF EVENT J FOR BASE EVENTS
= FREQUENCY OF EVENT J FOR INITIATING EVENTS

MEASURES:

1. FOR BASE EVENTS AND INITIATING EVENTS:

RISK REDUCTION = PD x EV(J)
= TEF - TEF(EVALUATED WITH EV(J) = 0)

2. FOR BASE EVENTS ONLY:

RISK INCREASE = PD - RISK REDUCTION
= PD x (1 - EV(J))
= TEF(EVALUATED WITH EV(J) = 1) - TEF

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

RISK REDUCTIONS RISK INCREASES BY BASE EVENT AND SORTED BY RISK REDUCTION

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	RISK INCREASE	(RANK)
RC-CM-13	5	2.93E-06	(6.0)	1.80E+06	(1.0)	5.31E+00	(5.0)
FM4	1	1.80E-01	(2.0)	3.69E+00	(2.0)	1.68E+01	(3.0)
FM3	1	1.40E-02	(3.0)	4.02E-01	(3.0)	2.83E+01	(1.0)
FM5	1	2.50E-01	(1.0)	3.37E-01	(4.0)	1.01E+00	(6.0)
FM2	1	2.00E-03	(4.0)	2.46E-02	(5.0)	1.23E+01	(4.0)
FM1	1	1.00E-04	(5.0)	2.75E-03	(6.0)	2.75E+01	(2.0)

RISK REDUCTIONS BY INITIATING EVENT AND SORTED BY RISK REDUCTION

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)
IE-20	5	2.00E+01	(6.0)	4.46E+00	(1.0)
IE-R	1	2.30E+04	(5.0)	1.47E-05	(2.0)
IE-Q	1	3.50E+05	(3.0)	1.06E-05	(3.0)
IE-P	1	4.90E+05	(1.0)	8.25E-07	(4.0)
IE-N	1	2.10E+05	(4.0)	1.18E-07	(5.0)
IE-M	1	4.70E+05	(2.0)	5.89E-09	(6.0)

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RC-CM-13	5	2.93E-06	(6.0)	1.80E+06	(1.0)	1.80E-01	1.84E+01
FM4	1	1.80E-01	(2.0)	3.69E+00	(2.0)		
FM3	1	1.40E-02	(3.0)	4.02E-01	(3.0)		
FM5	1	2.50E-01	(1.0)	3.37E-01	(4.0)		
FM2	1	2.00E-03	(4.0)	2.46E-02	(5.0)		
FM1	1	1.00E-04	(5.0)	2.75E-03	(6.0)		

RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
IE-20	5	2.00E+01	(6.0)	4.46E+00	(1.0)		
IE-R	1	2.30E+04	(5.0)	1.47E-05	(2.0)	1.49E-02	1.61E+00
IE-Q	1	3.50E+05	(3.0)	1.06E-05	(3.0)	1.49E-01	1.53E+01
IE-P	1	4.90E+05	(1.0)	8.25E-07	(4.0)	1.59E-02	1.53E+00
IE-N	1	2.10E+05	(4.0)	1.18E-07	(5.0)	9.68E-04	9.45E-02
IE-M	1	4.70E+05	(2.0)	5.89E-09	(6.0)	1.11E-04	1.09E-02

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK INCREASE	(RANK)	LOWER 5%	UPPER 5%
FM3	1	1.40E-02	(3.0)	2.83E+01	(1.0)		
FM1	1	1.00E-04	(5.0)	2.75E+01	(2.0)		
FM4	1	1.80E-01	(2.0)	1.68E+01	(3.0)		
FM2	1	2.00E-03	(4.0)	1.23E+01	(4.0)		
RC-CM-13	5	2.93E-06	(6.0)	5.31E+00	(5.0)	1.29E+06	2.20E+06
FM5	1	2.50E-01	(1.0)	1.01E+00	(6.0)		

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

BASE EVENTS RANKED BY VARIOUS METHODS (RANK 1 = MOST IMPORTANT) AND SORTED BY RISK REDUCTION

BASE EVENT	RISK REDUCTION	RISK INCREASE
RC-CM-13	1.0	5.0
FM4	2.0	3.0
FM3	3.0	1.0
FM5	4.0	6.0
FM2	5.0	4.0
FM1	6.0	2.0

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

MATRIX OF TOP-DOWN CORRELATION COEFFICIENTS

```
RISK INCR   -Ø.2327

            RISK RED
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THERE ARE NO TOP-DOWN CORRELATIONS THAT ARE SIGNIFICANT AT THE .Ø5 LEVEL OR LESS

THE TOP-DOWN CORRELATION COEFFICIENTS ARE USED TO DETERMINE THE LEVEL OF AGREEMENT AMONG ALL PAIRS OF RANKINGS IN THE PREVIOUS TABLE. THE TOP-DOWN CORRELATION COEFFICIENT GIVES MORE WEIGHT TO THE SMALLEST RANKS. NOTE THAT THE SMALLEST RANKS ARE ASSIGNED TO THE MOST IMPORTANT BASE EVENTS. MORE DETAIL ON THIS STATISTIC CAN BE FOUND IN THE IMAN AND CONOVER (1985) REFERENCE IN THE USER'S GUIDE THAT ACCOMPANIES THIS CODE.

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

CUT SET FREQUENCIES AND NORMALIZED CUT SET FREQUENCIES (WITH ASSOCIATED UNCERTAINTY INTERVALS)

CUT SET	ORDER	CUT SET FREQUENCY	(RANK)	LOWER 5%	UPPER 5%	NORMALIZED FREQUENCY	CUMULATIVE NORMALIZED FREQUENCY	LOWER 5%	UPPER 5%
4	4	8.03E+00	(1.0)	1.49E-01	1.53E+01	1.8018	1.8018	0.7511	0.8751
3	4	8.66E-01	(2.0)	1.59E-02	1.53E+00	0.1943	1.9961	0.0609	0.1231
5	4	8.25E-01	(3.0)	1.49E-02	1.61E+00	0.1850	2.1811	0.0452	0.1396
2	4	4.23E-02	(4.0)	9.88E-04	9.45E-02	0.0095	2.1906	0.0035	0.0077
1	4	5.89E-03	(5.0)	1.11E-04	1.09E-02	0.0013	2.1919	0.0004	0.0009

RISK UNCERTAINTY ROOT CAUSE 13 RUN (5-8-92)

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT RC13-RSK-UNC WITH TOP EVENT FREQUENCY 4.46E+00

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

2	4	4	8.03E+00	1.80181	FM4	*	IE-20	*	IE-Q	*	RC-CM-13	+
3	3	4	8.66E-01	1.99608	FM3	*	IE-20	*	IE-P	*	RC-CM-13	+
4	5	4	8.25E-01	2.18111	FM5	*	IE-20	*	IE-R	*	RC-CM-13	+
5	2	4	4.23E-02	2.19061	FM2	*	IE-20	*	IE-N	*	RC-CM-13	+
6	1	4	5.89E-03	2.19193	FM1	*	IE-20	*	IE-M	*	RC-CM-13	.

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11. ABSTRACT (200 words or less)

Nuclear power plants have experienced inadvertent actuations of fire protection systems (FPSs) under conditions for which these systems were not intended to actuate, and also have experienced advertent actuations with the presence of a fire. These actuations have often damaged plant equipment.

A review of the impact of past occurrences of both types of such events on nuclear power plant safety has been performed. Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. These scenarios ranged from inadvertent actuation caused by human error to hardware failure. and includes seismic root causes and seismic/fire interaction. A quantification of these thirteen scenarios, where applicable, was performed on a Babcock and Wilcox (B & W) Pressurized Water Reactor (lowered loop design). This report estimates the contribution of FPS actuations to core damage frequency and to risk.

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