

Fire Risk Analysis for Nuclear Power Plants

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U.S. Nuclear Regulatory
Commission**

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ABSTRACT

A methodology for evaluating the frequency of severe consequences due to fires in nuclear power plants is presented. The methodology produces a list of accident scenarios and then assesses the frequency of occurrence of each. Its framework is given in six steps. In the first two steps, the accident scenarios are identified qualitatively and the potential of fires to cause initiating events is investigated. The last four steps are aimed at quantification. The frequency of fires is obtained for different compartments in nuclear power plants using Bayesian techniques. The results are compared with those of classical methods and the variation of the frequencies with time is also examined. The combined effects of fire growth, detection, and suppression on component failure are modeled. The susceptibility of cables to fire and their failure modes are discussed. Finally, the limitations of the methodology and suggestions for further research are given.

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1. INTRODUCTION AND SUMMARY

1.1 Introduction

This study presents a methodology for evaluating the frequency of consequences due to fires (fire risk) in nuclear power plants. These consequences can be defined in terms of the extent of release of radionuclides into the environment. The main source of the radionuclides is the core of the reactor [1] and only accidents can lead to large releases. For this to happen, both the reactor vessel and the containment must be breached and the core must be severely damaged. Thus, we concentrate on analyzing scenarios that involve fire incidents which can lead to core damage and containment failure.

Similar to any risk study, the methodology identifies a comprehensive list of scenarios and then assesses the frequency of occurrence of each. This process requires knowledge about almost all aspects of a fire incident (that is, ignition, progression, detection and suppression, characteristics of materials under fire conditions, etc.) as well as the plant safety functions and their behavior under accident conditions.

Fire research is a multidisciplinary effort that is being vigorously pursued in many countries around the world and which covers a large spectrum of topics (for example, physics of combustion, flame behavior in

compartments, and fire detector response characteristics). The level of sophistication of the tools used varies greatly. Very few sources have looked into the probabilistic aspects of fire incidents and, especially, the public risk stemming from these occurrences. This is true for the fire risk in nuclear power plants. Many fires have occurred in these plants [2] and concern about them as a potential common cause event has been greatly increased since the well-known Browns Ferry fire [3]. Many regulatory actions followed this incident. A special review group from the Nuclear Regulatory Commission (NRC) analyzed it in detail [4]. The Reactor Safety Study [5] also investigated the chances of that incident leading to core melt-by postulating various failure scenarios. Reference 2 used their approach in performing a parametric study and found that the conditional frequency of core melt could have been as high as 0.03. Reference 5 found that the unconditional frequency is about 1×10^{-5} incidents per reactor year.

The NRC has requested all utilities to submit a fire protection analysis report [4] which evaluates each plant with respect to fire protection guidelines [6]. Although the information given varies from plant to plant, basically they have all enumerated the administrative actions taken against fires and have

listed for each compartment the existing safety-rated items, the fuel loading (Btu/ft^2), the type of fuel, the fire protection equipment, etc. Based on these studies, changes have been recommended and safe shutdown methods are analyzed Reference 7 is an example.

This is a success-oriented analysis of the different paths to safe shutdown. It starts with the plant at full power and assures that reactivity control and core cooling are achieved and that temperature and pressure indicators are available during a fire incident. Credit has been given to manual actuators (at pump or valve locations), special cables, and special fire detection and suppression systems. These reports provide valuable information about the fire hazard in each plant.

Three studies evaluate the fire risk of nuclear power plants. The first study was a part of the Clinch River Breeder Reactor (CRBR) Risk Assessment Study [8]. The second study is part of the High Temperature Gas-Cooled Reactor [9] risk assessment study conducted by the General Atomic Company [10]. The third study is one performed at the Rensselaer Polytechnic Institute (RPI) [11 12]. In all of these studies, the critical locations for fires are identified qualitatively and the frequencies are given in terms of point estimates, although certain upper bounds are sometimes evaluated.

The CRBR study was the first attempt in this direction. It uses failure modes and effects analysis to identify critical locations and covers all types of fire (cable tray, oil etc.) including sodium fires. Event trees and fault trees are used to establish fire-initiated sequences leading to core melt.

The fire risk study for an HTGR plant [10] is a small part of a larger effort wherein the overall HTGR risk is assessed. Its inclusion was instigated by the Browns Ferry incident. The authors recognize that, except for some special aspects, the general features of risk assessment methodology are also applicable to fires. Very detailed data on individual fire incidents have been collected as part of their work. In Section A.2 of Appendix A, we review the information provided. They have used the data to obtain fire occurrence rates and analyze fire progression characteristics. Their data includes estimates on the physical size and duration of the fires. From this, they establish a distribution for the extent of fire growth.

The HTGR report also proposes a method similar to failure modes and effects analysis to identify the critical locations within a plant, called "Fire Location and Progression Analysis" (FLPA). As part of this methodology, for every area, information is

collected about the fire loading, the components and their potential failure modes (caused by fire), etc. Then, the critical locations are chosen judgmentally. Subsequently, the authors look into the sequences of events that can be caused by fires and eliminate several areas by comparing two frequencies for the same sequence of events. One frequency is due to fires and the other is due to causes other than fires. They have found that the cable spreading room poses the largest fire risk and the frequency of core heatup is 10^{-5} per reactor year.

Two reports are published from the Rensselaer Polytechnic Institute study [11,12]. The first one presents a detailed analysis of data collected from insurance companies and regulatory bodies. In Section A.1 of Appendix A, we have tabulated part of their results. The authors have looked at many different aspects of the data and have also proposed models for ranking systems, components, and fire zones. For this, they have defined importance measures which are linked to some conditional frequencies; for example, the frequency of fire occurrence, the fraction of fires of a certain type, etc.

In Reference 12, the second report from the RPI work, the author has developed a method for analyzing loss of safety functions in a boiling water reactor (BWR)

due to fires. He models the systems by success trees and the fires by event trees. The event trees are used for modeling the time-dependent characteristics of a fire. The first event (the initiating event) is ignition. The second and third events question the success of detection and suppression activities, respectively. The fourth event concerns propagation. Once again, the success of detection and suppression is questioned in the fifth and sixth events. The exiting states of the event tree are labeled as "Component lost," which denotes that several vital components have failed due to the fire. This report presents a very detailed discussion on detection and suppression. The plant locations are analyzed one at a time and a failure-modes-and-effects-analysis type of approach such as that used in the CRBR and HTGR studies is employed. The worst case fires for each fire are identified and fire scenarios are quantified using point values for the frequencies.

1.2 Summary

The main goal of the methodology presented in this study is to identify the dominant contributors to fire risk. A contributor is defined as a sequence of events (a scenario) that begins with a fire and terminates with the release of radionuclides to the environment.

There are basically two parts to the problem:

first, the identification of the scenarios and second, their quantification. The framework of the methodology is given in six steps. These are the building blocks of any sophisticated algorithm which can be developed to use more efficient methods for obtaining the dominant contributors.

The first two steps are for scenario identification and they are discussed in Chapter 2.

Step 1 - Initiating events are analyzed to see how fires can cause them (Section 2.1).

Step 2 - The mitigating functions for the initiating events of Step 1 and accident sequences are analyzed to see how the fires can affect them. The result of this step is a list of scenarios (see Section 2.2).

These two steps cover the first part (i.e., scenario identification) of the problem. Event trees and fault trees are the main tools here. Examples are cited from the Zion and Indian Point Fire Risk Studies [13,14] where the proposed methodology is implemented. We take fires as the cause of the initiating events (a perturbation in the balance of plant). We have examined the possibility of experiencing a large LOCA due to fires in a pressurized water reactor. The accident mitigating functions are modeled in a simple manner. The critical locations are identified by qualitative arguments which are based

mainly on the safety-related items that can be affected by fire in the area.

The remaining steps are aimed at quantification and are described in Chapter 3. The general model is given in Section 3.1.

Step 3 - The frequency of fire incidents in different compartments is obtained (see Section 3.2).

Bayesian methods are used in this step. This data comes mainly from insurance sources and the results are compared with classical methods. The variation of the frequencies with time is also examined.

Step 4 - Fire growth analysis is performed and conditional frequencies of affecting relevant components are obtained. The effects of detection and suppression are taken into account (see Section 3.4.1 for fire growth, 3.4.3 for detection, 3.4.4 for suppression, and 3.6 for obtaining the conditional frequency).

Step 5 - Conditional frequency of accident sequences given a fire is derived (see Section 3.5 for component failures when affected by fire and 3.6 for deriving the conditional frequency.)

Step 6 - Unconditional frequencies of accident sequences are derived (see Section 3.6).

In Section 3.3, we establish representative fire scenarios based on the components in the event sequence to limit the scope of fire growth analysis. A model is

proposed for the conditional frequency of failing a known set of components within a room. It takes into account the periods for growth of fire, detection, and suppression. These are estimated in Section 3.4. The model also includes the failure frequency of the components, given that they are exposed to fires. In Section 3.6, we show how the difference frequencies of the methodology are assembled and the unconditional frequency of some severe consequences is obtained.

In this work, we find that human error is an important part of a fire analysis because: (1) manual activation of components is possible when they become disconnected from the control room due to cable fires, and (2) when instrumentation-related components are affected by a fire, the operators may react to erroneous information on the control board. In the latter case, the question of completeness of the analysis becomes important.

It is important to note that the occurrence of fires and their effects on plant safety are very complex issues which have not attracted the attention that other parts of risk assessment have in the literature. It is natural, therefore, that assumptions, usually conservative, have to be made for the analysis to be completed. Effects of smoke, external fires, secondary fires, flooding due to water-type

extinguishers, and fires caused by earthquakes are not addressed in this study. The limitations of the proposed approach as well as suggestions for future work are discussed in Chapter 4.

The details of the fire incident data are described in Appendix A. The total number of years for different compartments in power plants under commercial operation is computed in Appendix B. The response time of the fire detectors and the suppression time are discussed in Appendices C and D, respectively. A literature survey on cable fire tests is presented in Appendix E.

2. SCENARIO IDENTIFICATION

This chapter covers the first and second steps of the algorithm described in the preceding chapter. The main goal is to identify scenarios in terms of component failure modes and their physical locations. The initiating events are analyzed first. The mitigating functions of each initiating event are analyzed next. The information obtained is then combined to define fire-related scenarios which are given in terms of components, their failure modes, and the cause of failure.

2.1 Analysis of the Initiating Events

2.1.1 General Remarks

The list of initiating events (IEs) developed for other parts of a probabilistic risk assessment should be used here [15]. Reference 15 gives a comprehensive list of these events for PWRs and BWRs. We can devide them into two broad groups: first, the loss of coolant accidents (LOCA's) where the core coolant is discharged from an opening in the cooling system, and second, the transients where the existing balance is perturbed (e.g., reactor trip). In this step (Step 1) we determine how a fire can cause an IE. Note that a fire is taken as the cause of the initiating event and not the initiating event itself. The relationship between fires and IEs can be found by constructing

fault trees for these events. This is illustrated by an example.

2.1.2 Example: LOCA in a PWR

The possibility of a large loss of coolant accident at the Zion station is studied in this example. A large LOCA is an opening larger than 6 inches in equivalent diameter in the primary side (for a PWR).

Figure 2.1 shows the primary loops of one of the units. From this figure, we conclude that there are only two ways in which a large LOCA can occur--a pipe break in one of the larger pipes, and spurious opening of the Residual Heat Removal (RHR) isolation valves. These are motor operated valves (MOVs) RH8701 and RH8702. The failure of the check valves is judged to be less likely than pipe failures because, in all cases, at least two check valves in series should fail.

Figure 2.2 shows a simplified diagram of the control circuit for opening one of the isolation valves. Parts of the circuit that do not affect the "open" signal are not shown here. The compartments or zones through which the circuit passes are also shown. The control switch and the pressure interlock could be bypassed if segments A and B of the wires in the cable spreading rooms or the MCC room touch each other, thus closing the circuit and energizing coil "O" which would open the valve. In Section 3.5 where cable failures

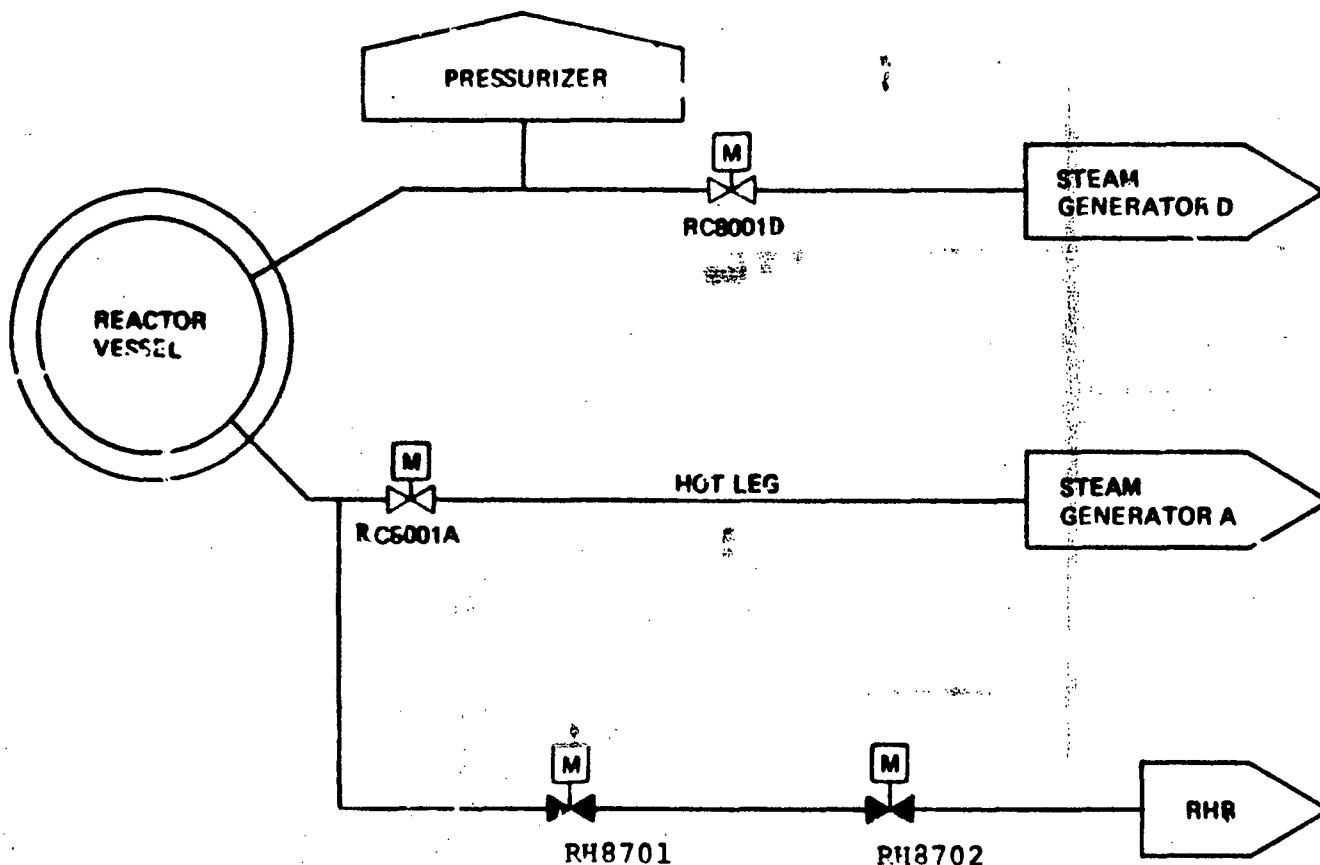


Figure 2.1a Simplified Diagram of Piping of the Main Coolant Loops

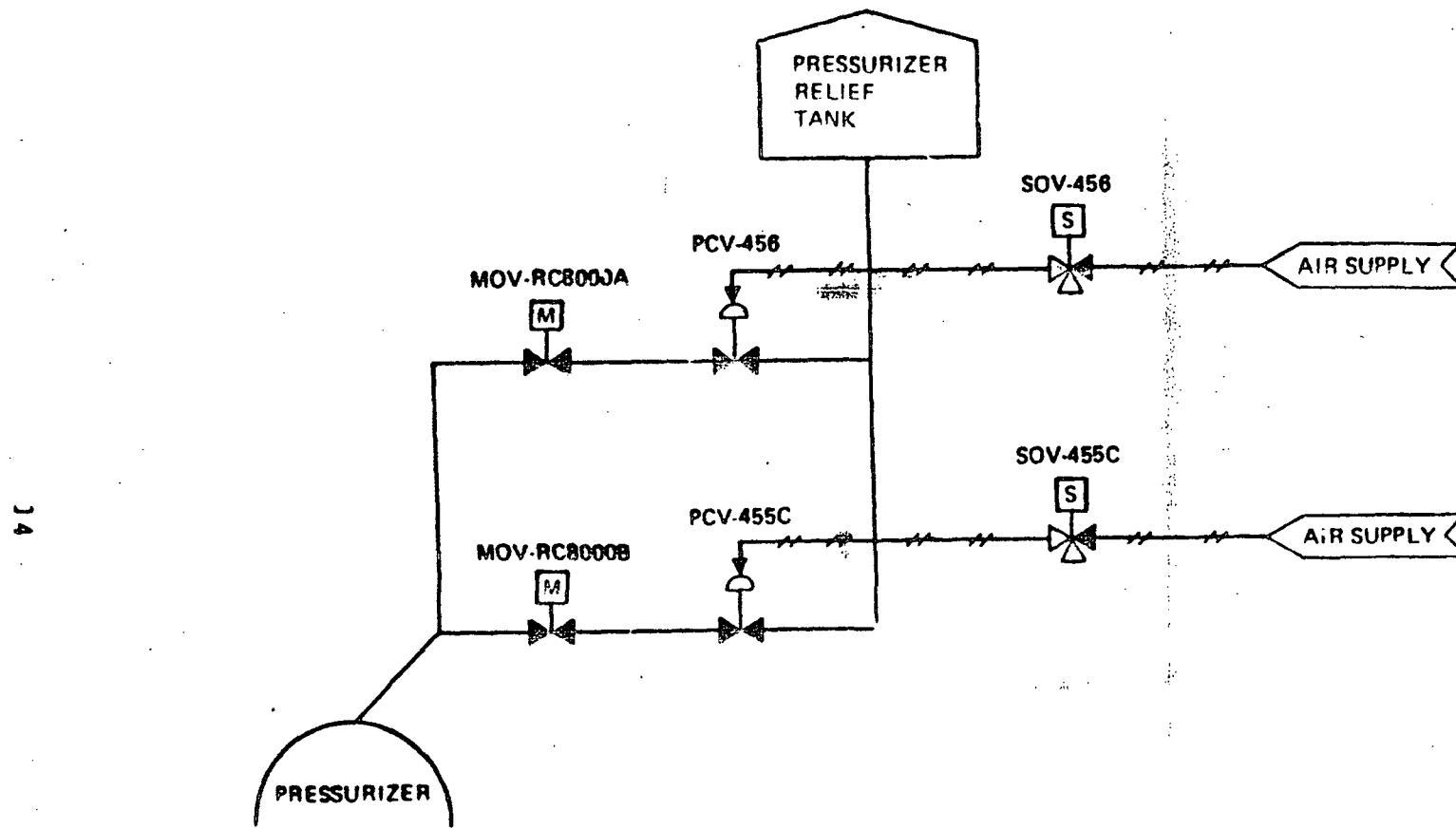


Figure 2.1b Simplified Diagram of Piping of the Main Coolant Loops

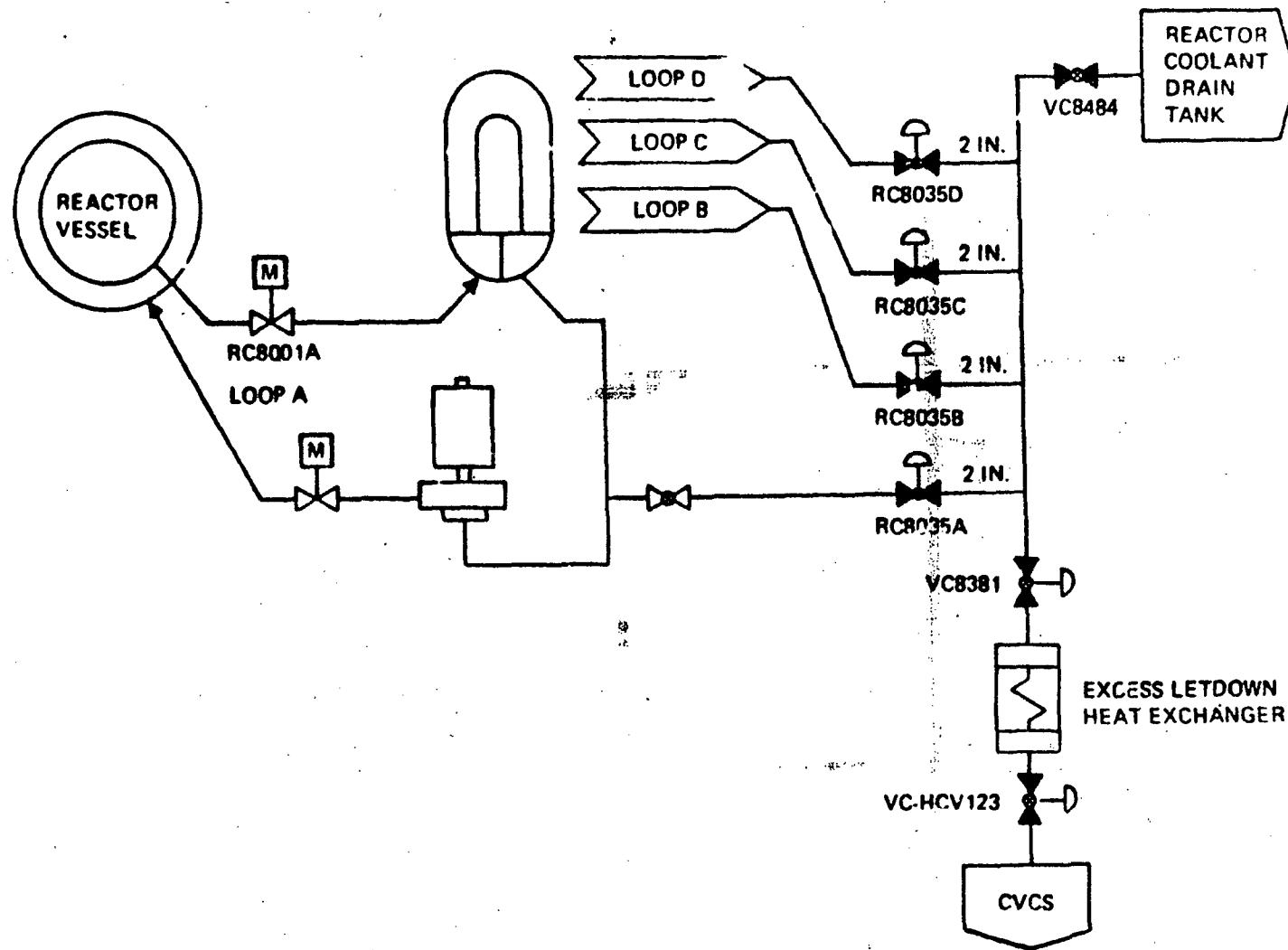


Figure 2.1c Simplified Diagram of Piping of the Main Coolant Loops

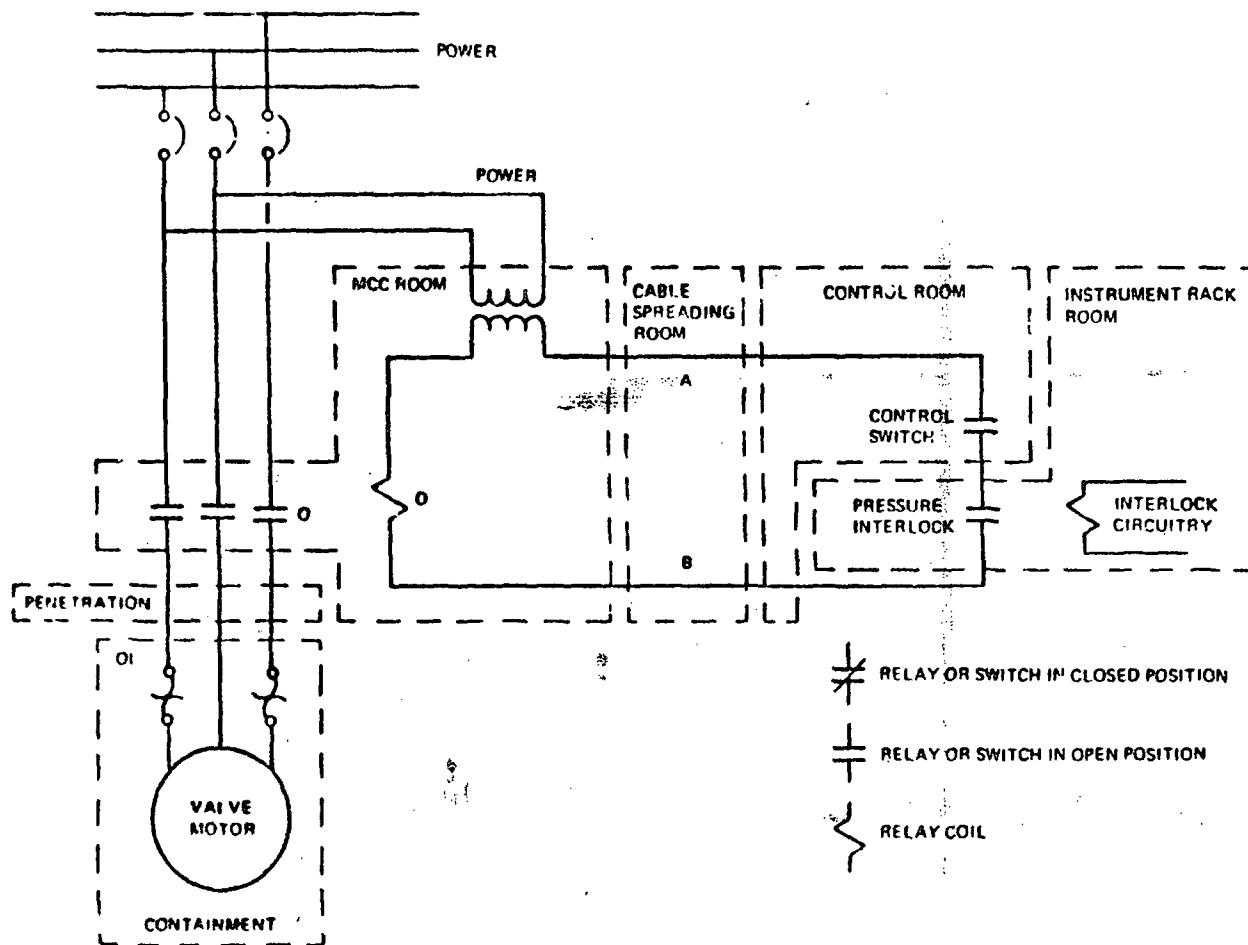


Figure 2.2 - Simplified Diagram of the Control Circuitry for Opening an RHR Isolation MOV (RH8701 or RH8702)

are discussed, we call this failure mode a "hot short." For MOV RH8701, wires A and B are in the same cable. For MOV RH8702, they are in different cables but in the same tray. The cables for the two valves are in different divisions.

Based on this information, the fault tree of Figures 2.3 and 2.4 is constructed. Note that the basic events are shown only in terms of component failure modes. Their causes were discussed separately. The locations where a fire may cause the top event (i.e., large LOCA) can now be identified. This list follows and includes the failures to which they would lead:

- Cable Spreading Room: Wires A and B of both MOVs contact each other
- MCC Room: Wires A and B of both MOVs contact each other or no. 52 breakers of both MOVs transfer closed
- Control Room and Instrument Rack Room: Control switch and pressure interlock switch of both MOVs transfer closed.

Two things should be pointed out here. First, two sets of information are given--the location of the fire (the cause of failure) and the components that it could affect. Both are necessary in the quantification process. Second, in some cases only part of the fault

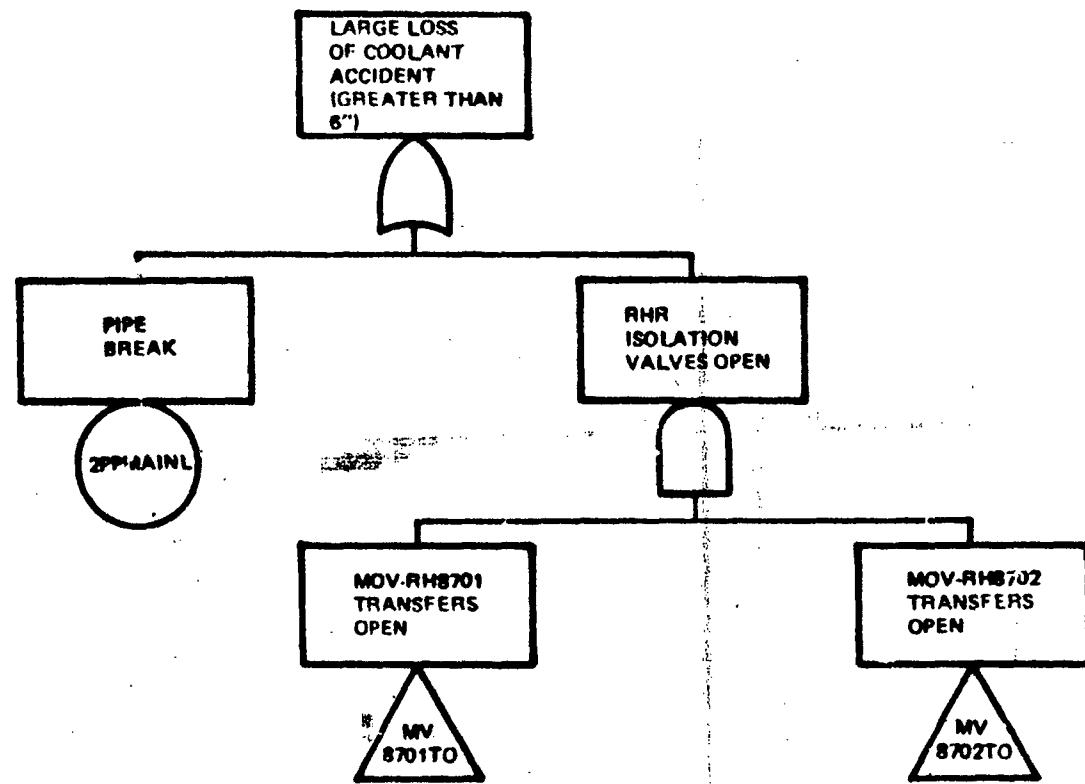


Figure 2.3 Top Structure of the Fault Tree for Large LOCA

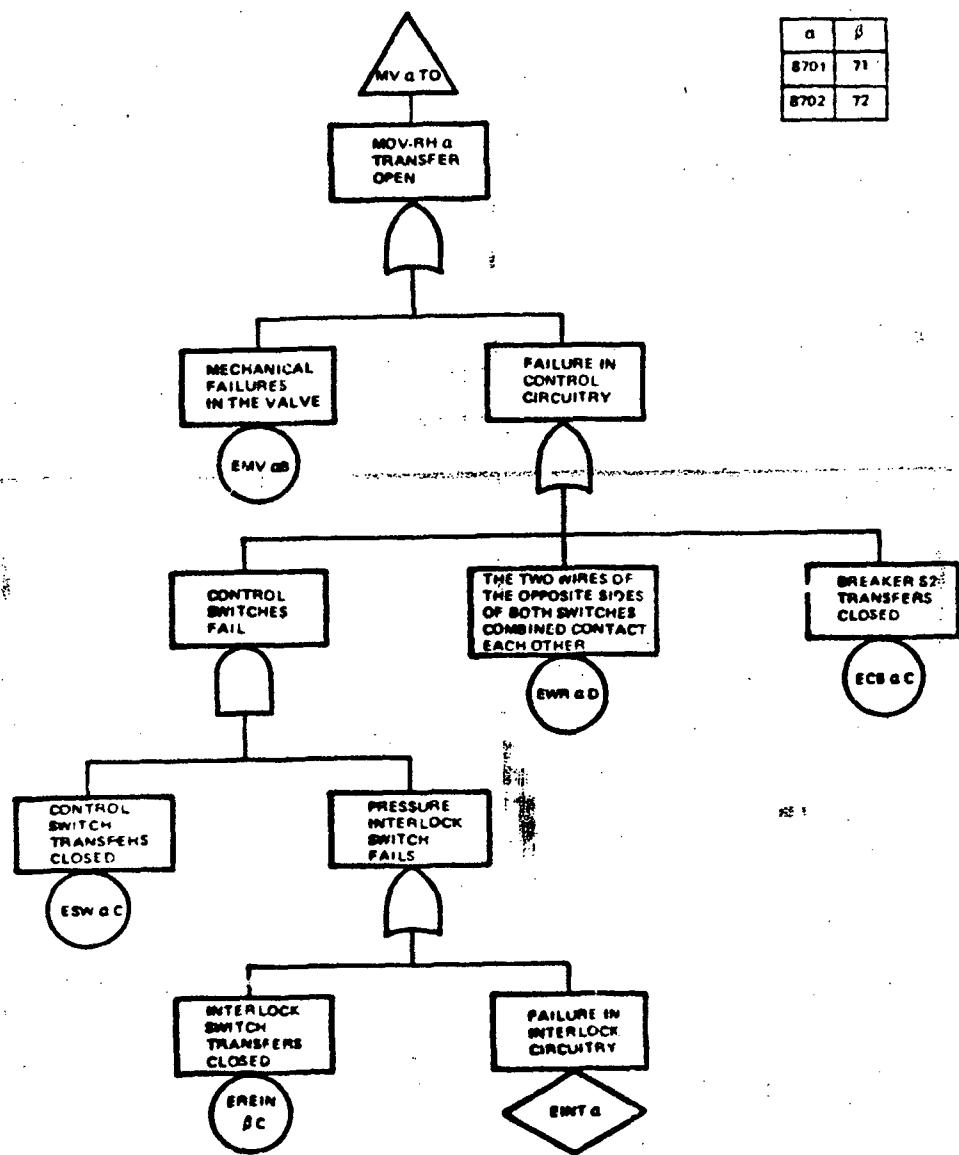


Figure 2.4 - Fault Tree for One RHR Isolation MOV Transfer
Open

tree is affected by a fire. For example, a control room fire can fail the control switches. However, failure of the pressure interlock switches due to other causes is also necessary for the top event to occur. Two conclusions can be drawn: (1) location cut sets can be defined similar to minimal cut sets, and (2) the fire location (the failure cause) should be specified when only partial failures can be achieved. The latter is very important for a sequence of events because it should be specified if the initiating event is caused by a fire.

In Section 3.5, we find that pipes and valve bodies are not susceptible to fires. Valve motors would fail as is and would not move spuriously. Breakers and relays would fail in their deenergized mode and, in this design, it is the open position. Control switches fail in their current status. Thus, the control room and instrument rack room-related failures (i.e., two relays and two switches transfer closed) cannot be caused by a fire. "Hot shorts" in the cables in the cable spreading room or MCC room are the most likely path to large LOCA. We have mentioned earlier that, for MOV.RH8702, the two wires A and B are in different cables but in the same tray. We judge it to be very unlikely that these two wires would come in contact with each other before touching any grounded conductor.

Thus, by qualitative arguments, we have reduced the number of fire locations to two. We developed this fault tree for illustrative purposes. In reality, it is very unlikely that the valves could be unseated given that the motors are energized because there is a tremendous pressure difference across the gates and the motors are underpowered by design.

2.1.3 Initiating Events in a PWR

For a simple approach, we suggest that all LOCA's be analyzed in a manner similar to that given in Section 2.1.2. For transients, one may conservatively assume that when a safety-related component is affected, reactor trip result, thus, a transient. There are areas in a plant that do not contain safety-related components, but a fire in them may lead to reactor trip. For example, the balance of plant-related items are in such areas. However, since safety-related items are not affected, safe shutdown can be achieved independently of the fire.

2.2 Analysis of the Mitigating Functions

2.2.1 The Mitigating Functions

The detailed event trees (ET) and fault trees (FT) that are constructed for risk analysis could be used here. These give the most comprehensive list of sequences that experts can envision. These trees lead us to a very large number of sequences in such a manner

that the efficiency of the methodology becomes important. Chapter 4 discusses this issue. Here we propose a simple approach.

For a PWR, the fundamental functions necessary for safe shutdown are summarized in Reference 7 as:

- "a. Maintaining a condition of negative reactivity,
- b. Removing reactor decay heat, and
- c. Monitoring and controlling the primary system coolant inventory and pressure."

The containment heat removal functions are also important [16], because they determine how the released radionuclides are contained within the containment.

The availability of these functions should be questioned along with the decay heat removal in item (b) above. In the following subsections, these mitigating functions are discussed in general terms and then they are illustrated collectively by an example.

2.2.2 Reactivity Control

Reactivity control is the first thing that should be checked for fire vulnerability in Step 2. Similar to the initiating events, we can do this by constructing a fault tree. The fire locations, the components, and their failure modes should be identified. For the two power plants that we have looked into, that is, Zion and Indian Point, the electronic and electrical components will lead to reactor trip upon deenergization.

Thus, reactor scram always occurs by at least human intervention if not due to the fire itself. The possibility of fires at the mechanical components of this system on top of the reactor vessel has not been studied.

2.2.3 Decay Heat Removal for a PWR

The systems used for mitigating a small LOCA and all of the transients are basically the same [16].

They require the availability of the scram system, high pressure primary cooling systems and secondary cooling systems. If both cooling functions fail, core melt will eventually occur. The small LOCA sequences become different from the transients when the molten core leaves the vessel. In transients, the pressure of the primary system stays very high because there is no bleeding capability except for the safety relief valves that blow steam into the containment. Thus, there is a large pressure difference between the vessel and the containment when the vessel is breached by the molten fuel. In the case of a small LOCA, this pressure difference would not be as high; therefore, the form of radiation release (release category) would be different. In both cases, containment heat removal is necessary. Small LOCAs will require containment heat removal sooner than the transient events.

In the case of a large LOCA, negative reactivity is inserted by the loss of coolant. Heat removal

should be provided almost immediately. The low pressure injection systems provide this function. Containment heat and iodine removal are necessary for safe shutdown in addition to maintaining containment integrity.

2.2.4 Coolant Inventory and Pressure

The monitoring of coolant inventory and pressure is an essential part of the safe shutdown process. The related components are typically transducers, electronic circuits, and electrical components. The essential parameters in a PWR are the pressurizer level and pressure [7].

The control of coolant inventory and pressure is achieved by the same systems as in decay heat removal.

2.2.5 Example: Accident Mitigation in a PWR

Figure 2.5 gives the event tree used in the Zion Fire Risk Analysis [13]. The event tree is based on the assumption that reactor trip has been successful. The secondary side cooling is provided by the auxiliary feedwater system (AFWS) which has three trains. Each train has one pump that can deliver adequate flow for decay heat removal. Two of the pumps are motor-driven. The third is turbine-driven and uses the steam of the main steam generators. All valves are air-operated and fail open upon loss of power to the solenoid valves.

The primary side coolant bleed and feed consists of the power operated relief valves (PORVs) on top of

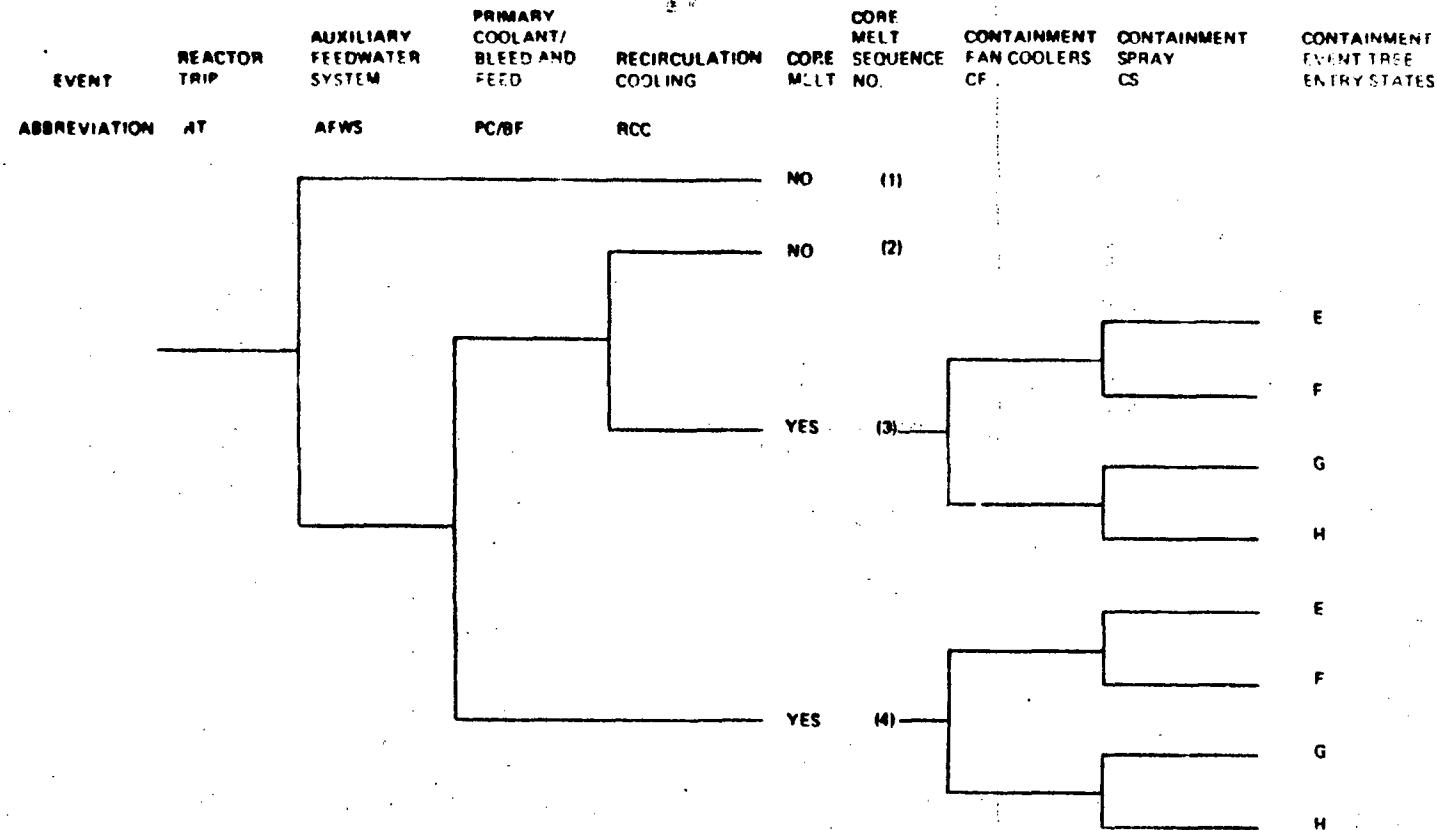


Figure 2.5 Simplified Event Tree for Transient Events

the pressurizer for bleeding, and charging or safety injection (SI) pumps for feeding. The charging pumps can inject coolant at high pressure whereas the shut-off head of the SI pumps is lower, 1,500 psig. Both sets of pumps take suction from the refueling water storage tank (RWST). The suction and injection lines of the SI pumps are normally open. However, for the charging pumps, two parallel motor operated valves isolate the suction under normal conditions. The injection side is open. Figure 2.1 shows the PORVs. They are air operated valves PCV456 and PCV455C. The MCVs upstream of these two, that is, RC8000A and RC8000B, are called PORV block valves. All four are normally closed and automatic control systems do not control them.

The containment heat removal function is provided by two systems--the containment spray system and the fan coolers. Depending on the availability of these systems, containment event tree entry states E, F, G or H result. The worst state is H where radiation release is a certainty. There are three trains in the containment spray system. Each train consists of a pump that can deliver 3,000 gpm, and several valves. Two of the pumps are motor driven and the third is diesel engine driven. The diesel fuel and batteries for startup are in the same general area as the pump. There are two

MOVs downstream from each pump. One of them is normally closed. All three pumps take suction from the RWST. The system is activated by simultaneous signals from SI and containment high-high pressure or SI and manual spray.

There are five fan coolers inside the containment. They operate at high speed under normal conditions. All shift to the low speed accident mode upon an SI signal. Containment air is drawn through the filtration plenums and cooling coils and back into the containment. The low fan speed is necessary to ensure that the fans are not overloaded by the increased mass of the containment air. The coils are cooled by the service water system.

The availability of these systems in a transient event tree is questioned only when core melt has occurred.

Sequence Number 1 - This is a success sequence where core melt does not occur. If at least one AFWS train is available, decay heat can be removed adequately. The primary makeup is not a critical function unless the heat removal rate cannot be controlled. If the system is overcooled, the primary pressure and level may drop so that the core would become uncovered. AFWS flow control is manual (from the control room).

The operators need pressurizer level and pressure indications for this purpose.

Sequence Number 2 - This is a success sequence also; however, the auxiliary feedwater system is unavailable. At least one of the four pumps (two charging and two SI) can provide adequate cooling flow into the core. The primary coolant bleed and feed PC/BF is manually controlled. The goal of this mode of operation is to depressurize the primary side without achieving saturation conditions. The pressurizer pressure and level indicators provide the necessary information.

The RWST would become exhausted in about 10 hours. At that stage, the valving should be changed to allow for recirculation cooling. This mode of operation uses the RHR pumps in addition to those in the injection phase. The RHR pumps take suction from the containment sump where the coolant that was discharged from the PORVs is collected. The coolant passes through the RHR heat exchangers where it is cooled by the component cooling water system, then it is routed to the suction side of the high pressure pumps (SI or charging). This phase of heat removal provides long-term cooling.

Sequence Number 3 - This is a core melt sequence where the APWs and Recirculation Cooling System have failed; however, the primary coolant bleed and feed is

successful. Thus, core cooling failure occurs beyond 10 hours after the accident and it takes more than 60 minutes to core damage inception. At that point, the primary system may be at low pressure. This depends on how the bleed and feed was performed. The pressure level affects the release categories. In Figure 2.5, we have conservatively assumed that the system is pressurized when core melt occurs.

The following two observations are in order. First, the timing in this sequence of events is long such that the restoration of failed systems can be a significant contributor. Second, more detailed scenarios are necessary; otherwise, only conservative measures can be considered.

Sequence Number 4 - This is a core melt sequence where both APWS and PC/BF have failed. If both fail at reactor trip, it would take about 4.5 hours for core melt to occur due to a total loss of heat removal. The system is assumed to be pressurized when the molten fuel breaches the vessel. At this point, the containment pressure would rise, the fan coolers would switch to the accident mode and depending upon whether SI signal exists, the containment spray system would be activated. Note that the timing is much shorter in this sequence of events than the previous one.

2.3 Combining Initiating Events with Mitigating Functions

2.3.1 Scenario Identification

Similar to the approach in the Fire Hazard Analysis [17], we perform our analysis one location at a time. For each location we check the following:

- (1) Can at least one of the LOCAs be caused by a fire (see Section 2.1)?
- (2) Are there any safety-related items? If so, assume that a reactor trip has occurred and safe shutdown is necessary.
- (3) Can reactor trip be defeated due to a fire?
- (4) If the answers to items (1) and (2) are affirmative, identify the systems for safe shutdown (Section 2.2.1) and sequences that will lead to core melt and radiation release (see Section 2.2.5 for an example).
- (5) Identify the components of the safety systems that are necessary for their operation and that are inside or outside the location.

In item (5), we identify a series of scenarios. Each consists of a location where a fire can occur, a sequence of events in terms of an initiating event, systems and release category or containment event tree entry state, the components (equipment, etc.) of this sequence that can be affected by the fire, and components

of this sequence that cannot be affected by the fire.

If, in item (3), it is found that reactor trip can be affected by a fire, then more detailed analysis would become necessary. It would be important to know how much negative reactivity can be inserted and what heat removal rates would be necessary. Quantification of this event may help us decide if further analysis is warranted. The approach, given in items (4) and (5) will lead us toward the desired scenarios.

2.3.2 Example: Scenarios For a Cable Spreading Room

Fire of a PWR

The cable spreading room (CSR) of the Zion station is studied in this example. There are two CSRs for each unit, called the inner and outer cable spreading rooms. The control and instrumentation cables of almost all safety-related items are routed through the inner room. There is no safety-related reactor shutdown and cooling equipment in these rooms except for cables [7,18]. The outer room contains some power cables in addition to those of the inner room. These cables are 4,160V power feeds to auxiliary feedwater pumps B and C, power cables to both centrifugal charging pumps and both safety injection pumps, 4,160V power feeds to Service Water Pumps, and 4,160V power feeds to three component cooling pumps.

Following the steps given in Section 2.3.1 we first

check for LOCA's. In Section 2.1.2, we found that a large LOCA was extremely unlikely. By inspecting Figure 1.1, we conclude that a medium LOCA is impossible because there are no openings with an equivalent diameter of 2 to 6 inches. A small LOCA is a possibility (through the PORVs); however, it is likely to be terminated within 30 minutes because the hot shorts (see Section 3.5.2.1 on Cable Failure Modes) will become open circuits. When this happens, the air-operated PORVs would close, thus terminating the LOCA. Failure of the valves to close due to other reasons may pose problems. This will be further investigated during the quantification. Therefore, we conclude that only a small LOCA may occur, and that requires an independent failure in addition to the fire.

Transients would occur because many safety-related cables could be affected. Although the assumption of reactor trip in item (2) may not hold for some areas (e.g., SI pump room), it is an appropriate one to make for the cable spreading room. Many instrumentation and control cables are linked with the balance of plant and safeguards control systems. Their failure would definitely upset the existing balance, and so a transient would be instigated.

Thus, so far, we have found that transients are highly likely and there is some chance for a small

LOCA. In both cases, reactor trip is necessary. In Section 2.2.2, we found that the latter could not be prevented by a cable spreading room fire. Now, item (4) follows. In Section 2.2.5, the mitigating functions for a transient were studied based on the event tree of Figure 2.5. A similar event tree applies to small LOCCAs. The only difference is in the containment event tree entry states. For a LOCA, the primary system may not be pressurized when the molten core leaves the vessel. Figure 2.5 shows that, for each initiating event, we have two core melt sequences.

Before the sequences are studied, we investigate the manner in which the three mitigating systems or functions can be affected by a CSR fire. The auxiliary feedwater system (Section 2.2.5) has all of its control and power cables routed through this room. All the closed valves are air operated and of the fail-open type. Therefore, they will open when their control cables fail in an open circuit mode. The two motor-driven pumps can be started manually at the pump location if their control cables are lost. However, if their power cables are affected, that pump train would be totally lost. The turbine-driven pump is an exception because the fire may start it by simply causing an open circuit in the control cable of the steam line stop valve. Furthermore, there are no power

cables to this pump; therefore, its operation is independent from the control room. In summary, both motor driven pump trains of the AFWS are susceptible to a CSR fire and the turbine-driven train can be assumed as totally independent.

The primary coolant bleed and feed consists of two parts--bleeding by the PORVs and feeding by the charging or the SI pumps. The power and control cables of all these items (with the associated valves) pass through the cable spreading room. Therefore, this mode of operation is completely susceptible to a CSR fire; moreover, if the power cables are lost, local manual action would be ineffective.

The availability of the recirculation mode of operation is questioned after bleed and feed has been performed successfully for more than 10 hours. This means that not all control functions are lost to the fire. Also, in view of the fire loading and past experience with fires in nuclear power plants (Section 3.2), it is judged to be quite unlikely that the fire would still be burning by this time. Then, the failure of recirculation cooling should be attributed to causes other than the fire. Human error at switchover may be affected; however, the long time period to any adverse situations would reduce the impact.

Only the control cables of the containment spray (CS) and containment fan cooler (CF) systems are routed through the cable spreading room. Since the power cables remain unaffected, all three CS pumps can be started manually from outside the control room. Each train has a normally closed MOV that should be opened. If their control cables are lost, they can be opened manually only at the valve location. The timing is important here because the containment spray becomes essential only when the molten core has left the vessel. At that point, the containment high+high pressure signal would be initiated and, if an SI signal already exists, the containment spray start signal would also be generated. Again, human intervention becomes important because the SI signal and even the manual containment spray signal can be initiated by the operators based on their judgment about accident progression.

The fan coolers are located inside the containment. Any damage to their control cables would only fail them as they are; that is, they would not switch to low speed. Under normal conditions, they are running at high speed. If they fail to switch to the accident mode, that is, low speed, they may eventually fail due to high load caused by the steam in the air. If the control cables are lost, the operators cannot intervene in their operation.

Now we have enough information to develop scenarios. We start with the transients, sequence number 3, and containment event tree entry state E (see Figure 2.5). The failed systems are AFWS and recirculation cooling. For AFWS, we found earlier in this section that two motor-driven pump trains could be affected by a CSR fire and the turbine-driven pump was totally independent. The recirculation cooling was also found to be totally independent from the fire. The remaining three systems (or functions) are assumed available. The first line of Table 2.1 depicts this scenario.

The remaining scenarios can be identified in a similar manner. Table 2.1 shows some of them. The first column of that table is simply a number assigned to each scenario. In the second column, the causes of failure are given. In this example, we have listed the fire in the cable spreading room, human error due to the fire and other causes. The latter covers a broad gamut of failure causes including human errors that are not affected by the fire. The remaining entries are aligned such that they are in the same line as the corresponding cause of failure. Note that fires are listed as one of the causes. The remaining columns correspond to the events in the event tree of Figure 2.5 and they show the failure mode or state of each event. The core melt sequence number and

TABLE 2.1 - Fire Initiated Scenarios for a PWR

SCENARIO NO.	CAUSE OF FAILURE	TRANSIENT INITIATING EVENT	AUXILIARY FLOW SYSTEM	PRIMARY COOLANT RISER & FEED	RECIRCULATION COOLING	CORE MELT SEQUENCE NO.	CONTAINMENT SPRAY	CONTAINMENT FAN COOLERS	CONTAINMENT EVENT TREE ENTRY STATE
1	Fire in Cable Spreading Room	Occurs	Failure of both motor driven pumps	Success	The whole system fails	3	Success	Success	E
	Other Causes		Failure of turbine driven train						
2	Fire in Cable Spreading Room	Occurs	Failure of both motor driven pumps	Success	The whole system fails	3	Loss of control to all pumps and MOV's	Success	F
	Other Causes		Failure of turbine driven train						
	Human Error						Failure to manually activate at least 1 train		
3	Fire in Cable Spreading Room	Occurs	Failure of both motor driven pumps	Success	The whole system fails	3	Success	Failure to shift to Accident Mode	G
	Other Causes		Failure of turbine driven train						
	Human Error	Occurs	Failure of both motor driven pumps	Success	The whole system fails	3	Loss of control to all pumps and MOV's	Failure to shift to Accident Mode	H
	Other Causes		Failure of turbine driven train						

TABLE 2.1 - Continued

containment entry state are given to make it easier to trace the sequence back to the event tree.

2.4 On the Definition of Fire Locations

In the preceding sections of this chapter, the location of a fire or compartments where fires can occur are mentioned without formally defining them. On the other hand, in all our examples, we have not considered the possibility of fires propagating from one compartment to another. This assumption of nonpropagation is very important in simplifying the methodology. The impact is obvious. For example, in Table 2.1 we basically have two causes: first, those related to the fire, i.e., the fire location and the human error; and second, all other causes. Actually, the human error is linked to the fire because it is related to manual operation of equipment failed by the fire.

A fire location should be enclosed by distinct fire barriers. More precisely, the boundaries of a location should be chosen such that the frequency of surpassing its threshold fire resistance would be very low. Furthermore, the frequency of loss of penetration seals during commercial power generation should be very low. The requirement of power operation is important because accident analysis is mainly focused on this phase and, during other phases (e.g., refueling), some penetration seals may be removed in the course of implementing changes.

Our judgment is that a 1-hour or better fire barrier is adequate in view of the typically low fire loadings in safety-related compartments. The possibility of smoke propagation and water ingress should be taken into account. One might think that these restrictions would lead to a small number of absurdly large locations. This should be avoided by judiciously choosing boundaries that do not satisfy some of the aforementioned conditions.

Example - The inner cable spreading room that was chosen as an example in Section 2.3.2 has the following characteristics [18].

- The floor is a 6-inch thick, structurally, reinforced concrete slab on unprotected steel beams. It is the roof of the laboratory area. Fire can only propagate from below to the cable spreading room. Such a fire would be very large and its frequency should be very low. We did not include the laboratory area as part of the cable spreading room.
- The east wall is 24-inch structurally reinforced concrete with solidly imbedded steel columns. This wall is shared with the turbine building.
- The other three walls are of 11-5/8 inch hollow concrete blocks with the holes filled solidly with mortar. They are shared with the outer

cable spreading room and the stairwell. All of the beams are also protected.

- The roof is 6 inches thick, structurally reinforced concrete, and is supported by steel beams which are covered by 2 inches of concrete or gypsum. It is 12 feet above the floor and is the floor slab of the control room.
 - Both doors (south and north walls) have at least a 1-1/2 hour fire rating. They are closed almost all of the time.
 - The electrical penetrations are sealed with inorganic fiber insulating material and covered with flamemastic.
 - Fire dampers are installed in the ducts penetrating the walls. These are 1-1/2 hour fire rated steel, activated by fusible links at 160°F
- Reference 18 gives more detailed information.

Smoke or extinguishing agents (such as CO₂) may propagate to other areas until the dampers close due to a rise in temperature. The control room ventilation system is independent from this area. Water ingress to the cable room has only one source--the control room. The chances of using water are small because the primary extinguishing agents available in this area are CO₂ and dry chemicals. The penetration seals would also act as a barrier.

Thus, we can consider the inner cable spreading room as a fire location. That is, we can assume that cable failures due to fire in this room are independent from other component failures. However, when smoke propagation or fires in the laboratory areas are considered, the validity of this assumption should be double-checked.

2.5 On Fires as Causes of the Initiating Events

In Sections 2.1 and 2.2, the given examples are geared toward fires that cause an initiating event and, at the same time, affect the mitigating systems. This is an adequate approach if the IE has a small frequency of occurrence due to other causes. This may not always be the case. For example, the loss of offsite power to the Indian Point Power Station has a median of 0.14 per year (95th percentile is 0.6 per year). It takes 4.5 hours for core melt to occur in case of total blackout and loss of turbine-driven auxiliary feed-water pump. There are three diesel generators that receive automatic start signals upon loss of offsite power. There are also three gas turbine generators near the site that can be started manually.

The diesel generators are housed in the same building. They are divided by 1/8-inch aluminum partitions which are erected as oil splash shields. At one end of this building, the control cabinets of

all three diesels are located. A 14-foot high concrete wall separates these cabinets from the diesels. Simultaneous failure of all three diesels may be caused by a single fire, either in the engine area or control board area. It is judged that the latter is more likely because it requires a much smaller fire than that of the engine area. Failure of the gas turbines and delay in restoring the offsite power should be due to causes other than that fire.

Thus, the simultaneous occurrence of a diesel generator building fire (not an initiating event) and independent occurrence of a loss of offsite power (an initiating event) would lead to station blackout. There are more than 4.5 hours available to power back, either by restoring the offsite power or starting one of the three gas turbine generators.

2.6 On the Details of the Scenarios

In Section 2.3, we showed how to identify scenarios but we did not discuss the level of detail that should be sought. For example, in Scenario Number 1 of Table 2.1, we point out the possibility of losing two AFWS trains to a single fire, but we have not elaborated in terms of all possible combinations of components. Obviously, more detail entails more work. Our judgment is that, for a simple approach, the scenarios should be stated in gross terms--trains of components,

supercomponents, or even whole systems. When these are quantified, bounding methods should be used. Based on those numbers, one can then judge if more detailed work is warranted.

3. QUANTIFICATION

3.1 Introduction

In this chapter, we describe a method for quantifying the scenarios obtained in Chapter 2. The simplest and very conservative approach would be to take the frequency of fires at a physical location as the frequency of failure of all components within that location. For some areas, this would result in unreasonably high core melt or radionuclide release frequencies. Therefore, a more detailed model is warranted.

In Chapter 1, we identified the major steps for quantification as part of the general methodology (Steps 3 through 6). Figure 3.1 shows a block diagram based on these steps and gives an overall picture of the quantification process. It references the related sections within this chapter where detailed descriptions are given. There is some dependence among the different blocks in the diagram that is not shown in Figure 3.1. For example, at "the representative cases for fire growth analysis," we need some knowledge about the fire growth history (i.e., growth, detection, and suppression). One can use iterative methods to further refine certain parts of the quantification process. Such methods would certainly depend on the specifics of the problem.

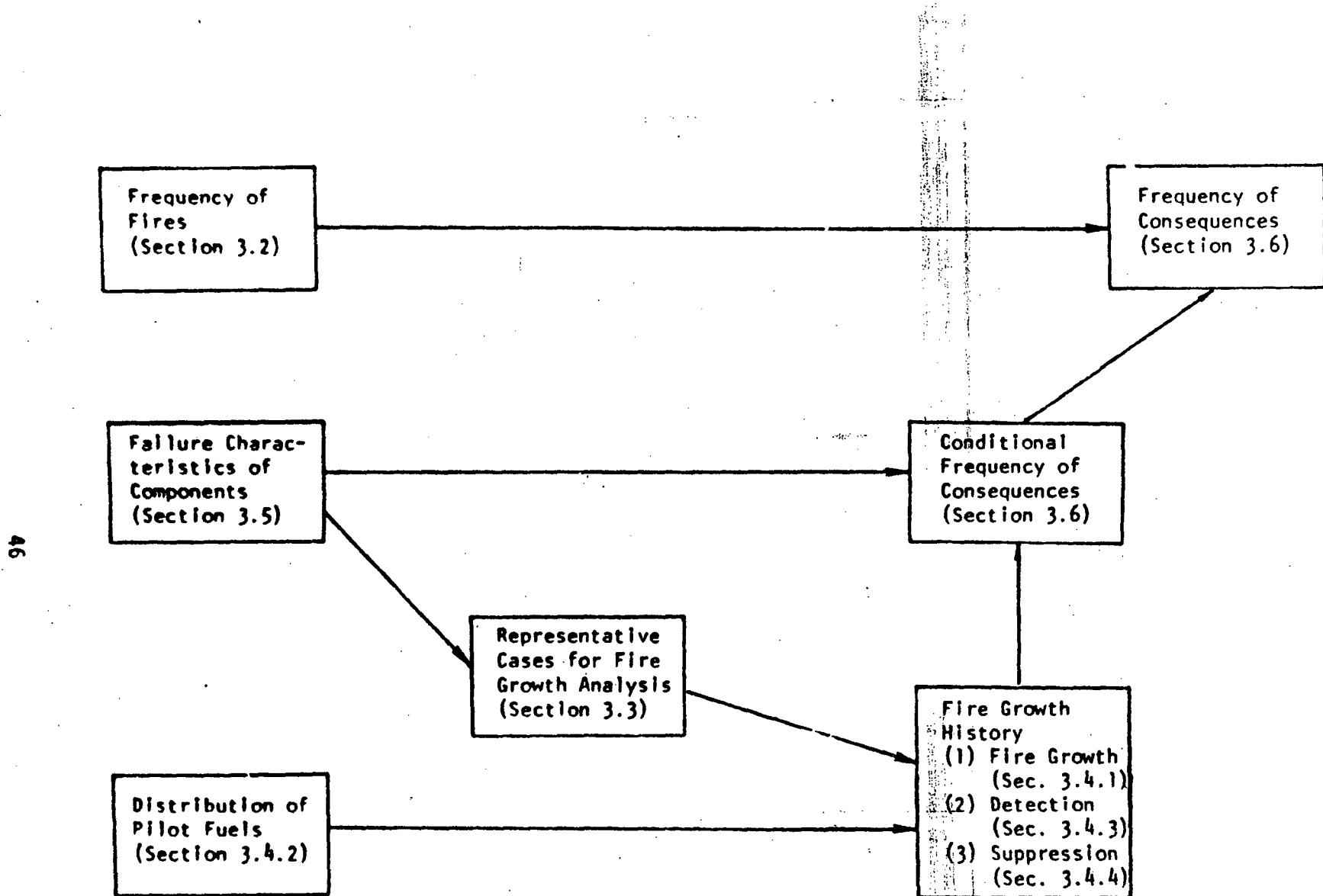


FIGURE 3.1 Block Diagram for Quantifying Fire Scenarios

References 10 and 12 have developed probabilistic models for the frequency of core melt due to fires. There have also been two other studies but with much narrower scope where the Browns Ferry fire incidents were analyzed [1,2].

Reference 10 focuses mainly on the cable spreading room fires. The minimal cut sets that contain cables passing through this room are identified by fault tree and event tree analysis. The layout of these cables is drawn to see what distances should be considered for fire growth modeling. Geometric fractions are combined with a growth model and the conditional frequency of core melt is obtained. By using geometric fractions, it is assumed that fire occurrence is uniformly distributed across the floor of the cable spreading room. For the growth model, it is assumed that: (1) all cables below or above a burning cable are also burning, and (2) the maximum radii of the base of the fires are exponentially distributed. The mean maximum radius is obtained from the fire incident data in nuclear power plants.

In Reference 12, the basic model is similar to the one which we propose here; only the differences are highlighted. Frequentist methods are used to assess the mean and the bounds of the frequencies. The frequency of ignition of sustained fires is attributed only to

the fuel type. An extensive model is developed for the effects of fire detection and suppression. Some of the results of this reference are used in this study. The fragility of the components has not been addressed. Instead, total failure is assumed given that the fire has engulfed an item.

3.2 The Frequency of Fires

3.2.1 Introduction

Distributions for the frequency of fires in nuclear power plant compartments are assessed in this section. These distributions will be used as inputs to fire risk analysis which will analyze the effects of these fires on the accident sequences that may lead to core meltdown.

The analysis is Bayesian [19, 20]. The frequency of fires is treated as an unknown quantity and its distribution expresses our current state of knowledge about the values of that frequency. An important factor that shapes our state of knowledge is the observed frequencies in the past. Thus, a significant part of the work is to investigate the available statistical experience and to decide what information it contains. We then use Bayes' theorem to formally incorporate this experience in our body of knowledge. Estimates of the frequency of fires are also derived using frequentist methods and the results are compared

with those of the Bayesian methods.

The data is described in Section 3.2.2. That section also gives the reasons behind our choice of compartments. Appendices A and B give a detailed account of the data used. Section 3.2.3 describes the Bayesian calculations. The prior distribution is gamma and the likelihood is Poisson. Table 3.2 gives the results. The uncertainties in the frequencies are of the state-of-knowledge-type. Section 3.2.4 shows that a lognormal prior has minimal impact on the final result. In Section 3.2.5, we find that, after the Browns Ferry fire incident, the overall frequency of fires has increased. The frequentist (classical approach) methods for uncertainty analysis give comparable results in Section 3.2.6. The magnitude of fires represented by the frequencies of Table 3.2 are discussed in Section 3.2.7. Finally, in Section 3.2.8, the type of uncertainties covered by the distributions are clarified.

3.2.2 Data

Data on fires in Light Water Reactors (LWRs) have been analyzed in several studies [1,10,11,21,22,23]. Although they have been done independently, they have some common aspects, e.g., some of the sources of data are the same. For example, almost all studies have used data from the Nuclear Regulatory Commission. Some have

also used data from the insurance industry. All have reported the overall frequency of fires within a small range of 0.11 per reactor year. These studies give tables of data on various features of fire incidents, e.g., causes of fires, components involved, systems affected, location of fires, etc. Reference 22 gives the most detailed tabulation. Reference 10 has included data on the size, shape, and duration of the fires, and it also discusses the methods used for detection and extinction.

There are two kinds of information needed: (1) the number of fire incidents that have occurred in specific compartments during commercial operation, and (2) the number of compartment years that the nuclear power industry has accumulated. A compartment year is defined as 1 calendar year of use of a specific compartment in commercial operation. Reference 23 is our main source for the first part. Most of its data comes from reports of insurance inspectors to American Nuclear Insurers (ANI), although other sources are also used, e.g., the U.S. Nuclear Regulatory Commission. While the NRC requires the reporting of fires that, in some way, affect the safety of the plant, ANI has more stringent requirements in the sense that all fire events must be reported [23]. It is still not clear, however, whether all the potentially significant events are reported and

what constitutes an insignificant fire. In Reference 23, incidents in all nuclear facilities are classified in several ways, e.g., according to the location of occurrence, the mode of suppression, the cause of fire, etc.

These tables do not provide data readily applicable to our model (see Section 3.2.3). This is because those tables for the incidents during commercial operation cover all types of facilities including educational reactors, reprocessing plants, etc. Furthermore, the tables on specific facility types cover all phases of plant life (i.e., construction, operation, etc). The number of incidents is derived by comparing several tables. Appendix A gives a detailed account of this derivation. The results are given in the first column of Table 3.1.

The time period covered by the ANI data starts in January 1955 (which is essentially the beginning of the nuclear power industry in the U.S.) and ends on May 31, 1978. Thus the compartment years are computed by adding the age of all compartments (within a certain category of compartments) of units that were in commercial operation by the end of May 1978. The age is defined as the time between first commercial operation and the end of May 1978 (or date of decommissioning).

Reference 24 and the Final Safety Analysis Reports (FSARS) [25] are consulted for the dates of commercial

TABLE 3.1. Statistical Evidence of Fires in LWRs

(As of May 1978)

Area	Number of Fires r	Number of Compartment Years T
Control Room	1	288.5
Cable Spreading Room	2	301.3
Diesel Generator	10	593.0
Containment	5	337
Turbine Building	9	295.3
Auxiliary Building	10	303.3

operation and the number of each compartment type per unit. Appendix B gives a detailed listing of the number of compartment types in each plant and their ages. The resulting compartment years are given in the second column of Table 3.1.

The choice of different classes of compartments is partly dictated by the data available and partly by how typical a given compartment is. The latter is an important factor because power plants do not have similar layouts. This is particularly true when PWRs and BWRs are compared. We have identified six areas, typically found in nuclear power plants. These are: the control room, the cable spreading room, the diesel generator, the containment, the turbine building and the auxiliary building. By diesel generator, we mean a unit comprised of a diesel engine and an attached generator.

In most plants, the first three areas are single compartments. However, the remaining three are typically large buildings within which are many compartments. Table 3.1 gives the cumulative age of these areas. The differences in age are mainly due to the fact that the units in some multiunit plants share some of these compartments. The only area that the units do not share is the containment; therefore, the containment years (i.e., 337 years) are equal to the reactor

years. The large experience years for the diesels is expected because almost all plants have at least two diesel generators.

3.2.3 Bayesian Calculations

We must now construct the distributions of the frequency of fires in the various areas that we have identified. The fundamental tool that enables us to incorporate the statistical evidence that we have assembled into our state of knowledge is Bayes' theorem which we write as

$$\pi'(\lambda/E) = \frac{\pi(\lambda)L(E/\lambda)}{\int_0^{\infty} d\lambda \pi(\lambda)L(E/\lambda)} \quad (3.1)$$

where

$\pi'(\lambda/E)$: probability density function of λ given evidence E (posterior distribution)

$\pi(\lambda)$: probability density function of λ prior to having evidence E (prior distribution)

$L(E/\lambda)$: probability of the evidence given λ (likelihood function).

A model for the occurrence of fires is the Poisson distribution (see Section 3.2.5), i.e., the likelihood function is

$$L(E/\lambda) = e^{-\lambda T} \frac{(\lambda T)^r}{r!} \quad (3.2)$$

where r and T are given in Table 3-1.

The prior distributions should reflect our state of knowledge prior to obtaining the evidence contained in Table 3.1. That knowledge, we feel, is vague. While we know that the frequency of fires in reactor compartments cannot be large, say 10 per compartment year, we are unable to say with high confidence what the values of this frequency are. Therefore, the prior distributions will be diffuse over a wide range of possible values of λ . At this point, there is no compelling reason for us to choose a particular family of prior distributions except that we would like them to be of a standard type because they can be easily visualized via their parameters and they are less complicated to manipulate. Also, they should be skewed to the left because, in nuclear power plants, the quality of fire protection is good.

Lognormal and gamma families of distributions comply with our requirements. The former fits our state of knowledge better because the bulk of the distribution is around the median and, for given 95th and 50th percentiles, the 5th percentile is not unreasonably low as in the case for gamma distribution. However, to facilitate the calculation of the integral in Bayes' theorem, we choose the gamma family of distributions which is conjugate with respect to the Poisson

distribution, i.e., the posterior distribution is also a gamma distribution. In Section 3.2.4, we will see that this choice does not have significant impact on the posterior distribution. The gamma distribution is

$$\pi(\lambda) = \frac{\beta^\alpha}{\Gamma(\alpha)} \lambda^{\alpha-1} \exp(-\beta\lambda) \quad (3.3)$$

where α and β are the two parameters of the distribution. A consequence of the conjugate property is that $\pi'(\lambda/E)$ is also of the form of Equation (3.3) with parameters

$$\alpha' = \alpha + r \quad (3.4)$$

and

$$\beta' = \beta + T. \quad (3.5)$$

The prior knowledge is represented by the pair (α, β) and the evidence by (r, T) . The greatest possible ignorance is represented by the values $\alpha = 0$ and $\beta = 0$ [19] in which $\pi(\lambda)$ is proportional to λ^{-1} (this is equivalent to saying that $\ln \lambda$ is uniformly distributed over the whole real line). For our purposes, we feel that the distribution of complete ignorance does not give appropriate weight to values of λ in the neighborhood of 1 per compartment year; therefore, we will use slightly more conservative prior distributions.

For the control and cable spreading rooms, we take

$\alpha = 0.182$ and $\beta = 0.96$ which yield a gamma distribution with characteristic values

$$\lambda_{05} = 5 \times 10^{-8}; \lambda_{50} = 1.5 \times 10^{-2}; \lambda_{95} = 1.0;$$

$$\langle \lambda \rangle = 0.21 \text{ (mean value)}$$

For the other areas, we wish to give more weight to higher values of λ and we choose $\alpha = 0.32$ and $\beta = 0.29$. The prior distribution has characteristic values

$$\lambda_{05} = 2.1 \times 10^{-4}; \lambda_{50} = 0.30; \lambda_{95} = 5; \langle \lambda \rangle = 1.11.$$

These distributions cover a wide range of values for the frequency of fires, thus expressing our vague prior knowledge.

We can now use Bayes' theorem to derive the posterior distributions for each area using the evidence of Table 3.1 and Equations (3.4) and (3.5). The results are shown in Table 3.2.

We observe that the evidence reduces the dispersion of the prior distributions significantly. For example, even for the weakest evidence (Control Room, $r=1$, $T=288.5$) the 90% interval of the posterior distribution is $(3.1 \times 10^{-4}, 1.2 \times 10^{-2})$ while that of the prior distribution is $(5 \times 10^{-8}, 1.0)$. Figures 3.2 and 3.3 show the prior and posterior distributions.

Reference 26 is a summary of this section. The results

TABLE 3.2 - Distribution of the Frequency of Fires (Events Per Room Year)

Area	a	B	λ_{05}	λ_{50}	λ_{95}	$\langle \lambda \rangle$	
Control Room	Prior	0.182	0.96	5×10^{-8}	1.5×10^{-2}	1.0	0.21
	Posterior	1.182	289.46	3.1×10^{-4}	3.0×10^{-3}	1.2×10^{-2}	4.1×10^{-3}
Cable Spreading Room	Prior	0.182	0.96	5×10^{-8}	1.5×10^{-2}	1.0	0.21
	Posterior	2.182	302.26	1.6×10^{-3}	6.2×10^{-3}	1.7×10^{-2}	7.2×10^{-3}
Diesel Generator	Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
	Posterior	10.32	593.29	9.5×10^{-3}	1.7×10^{-2}	2.7×10^{-2}	1.7×10^{-2}
Containment	Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
	Posterior	5.32	32.29	6.2×10^{-3}	1.4×10^{-2}	2.8×10^{-2}	1.6×10^{-2}
Turbine Building	Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
	Posterior	9.32	295.59	1.7×10^{-2}	3.6×10^{-2}	5.0×10^{-2}	3.2×10^{-2}
Auxiliary Building	Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
	Posterior	10.32	303.59	1.9×10^{-2}	3.3×10^{-2}	5.3×10^{-2}	3.4×10^{-2}

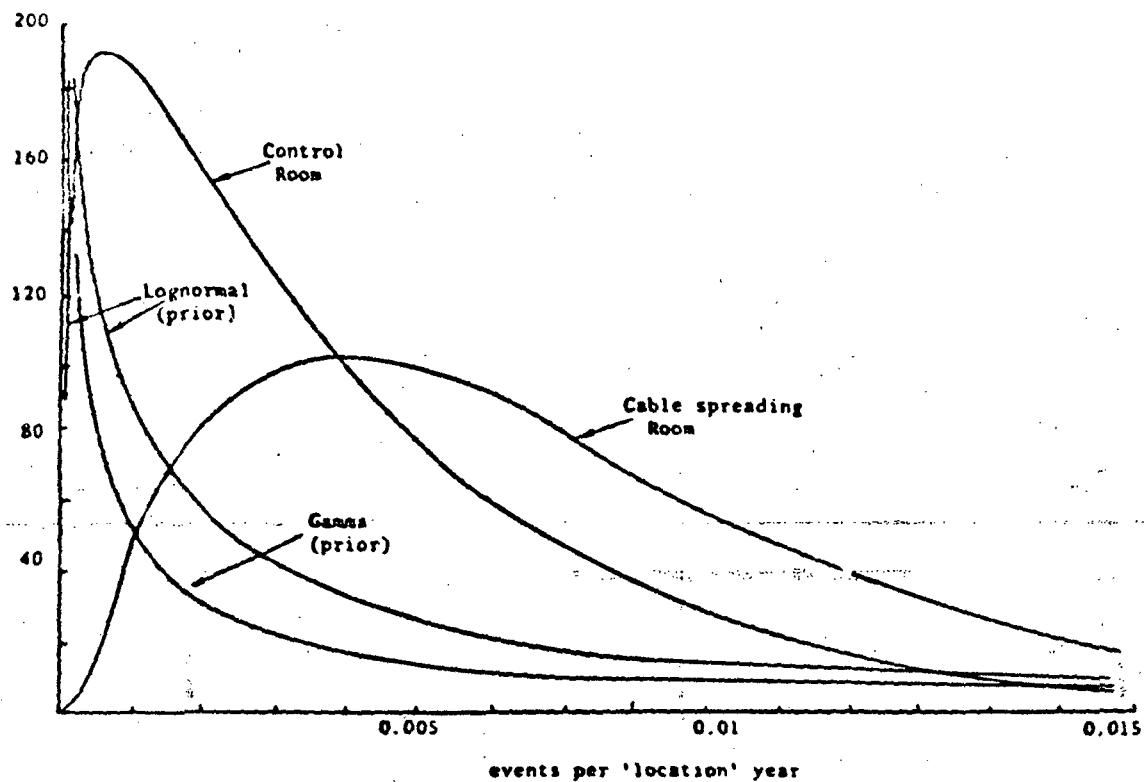


Figure 3.2 The prior and posterior densities for the control room and cable spreading room.

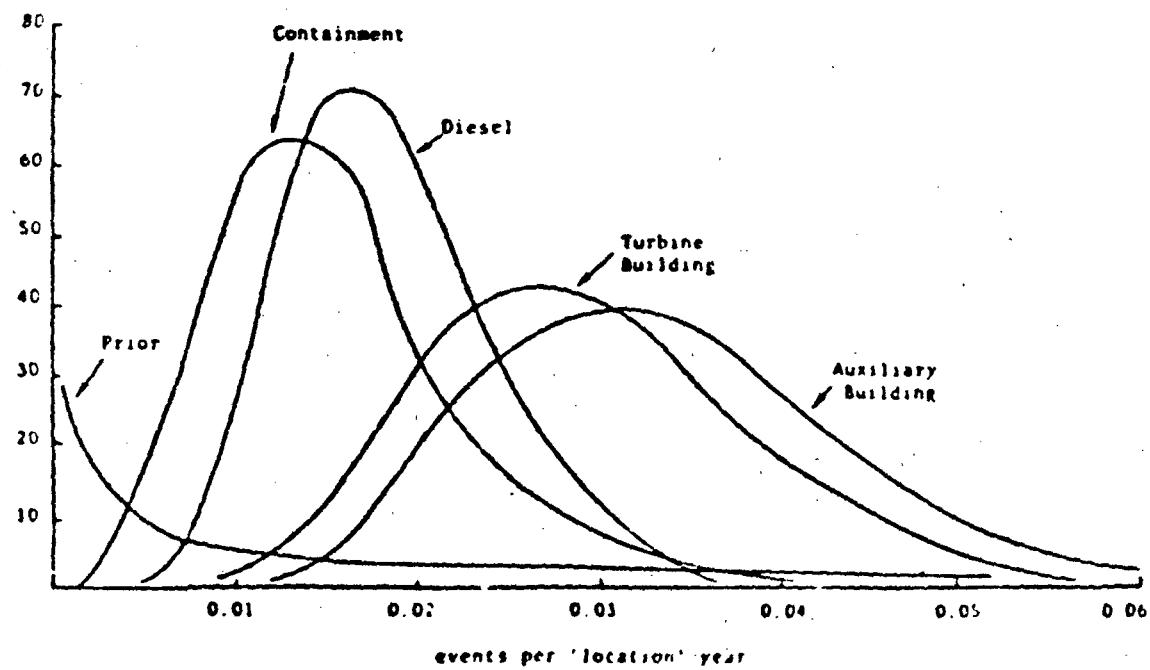


Figure 3.3 The prior and posterior densities for the containment, diesel, turbine building and auxiliary building.

for diesels given in Table 3.2 have been modified since the publication of that reference.

The special computer program MDGAMMA [27] was used to calculate the percentiles of Table 3.2. There are also approximate methods that can be used for the same purpose. Depending on the values of α and β , the approach of Reference 28, or chi-square distribution, can be used.

For $\alpha < 3$, Reference 28 gives the following equations:

$$\lambda_{05} = \frac{1}{\beta y(\alpha)} \quad (3.6)$$

$$y(\alpha) = [0.05\Gamma(\alpha+1)]^{-1/\alpha} - \frac{1}{\alpha+1} \quad (3.7)$$

$$\lambda_{50} = \lambda_{05} C(\alpha, 50) \quad (3.8)$$

$$\lambda_{95} = \lambda_{05} C(\alpha, 95) \quad (3.9)$$

where $C(\alpha, 1)$ is a function of α , the shape parameter in the gamma distribution and the percentile. It is tabulated in Reference 28. $\Gamma(\alpha+1)$ is the gamma

function which can be obtained from Sterling's asymptotic series [28] for positive values greater than 1.0:

$$\Gamma(x) = x^x e^{-x} \sqrt{\frac{2\pi}{x}} \left[1 + \frac{1}{12x} + \frac{1}{288x^2} - \frac{1}{51840x^3} - \frac{571}{2488320x^4} + \dots \right] \quad 1 \leq x. \quad (3.10)$$

Example - For the control room we have:

$$\alpha = 1.182 \text{ and } \beta = 289.46.$$

Both satisfy the above given conditions. From Equations (3.10), (3.7), and (3.6) we obtain:

$$\Gamma(1.182) = 1.091$$

$$y(1.182) = 11.254$$

$$\lambda_{05} = 3.07 \times 10^{-4} \text{ ry}^{-1}.$$

By interpolating the data given in Reference 27, we obtain:

$$C(1.182, 50) = 10.011$$

and from Equation (3.8)

$$\lambda_{50} = 3.07 \times 10^{-3} \text{ ry}^{-1}.$$

Similarly,

$$C(1.182, 95) = 38.808$$

and

$$\lambda_{95} = 1.19 \times 10^{-3} \text{ ry}^{-1}.$$

Note that these approximate percentiles are very close to those in Table 3.2.

For $\alpha > 3$, the chi-square approximation can be used. A gamma distribution with α and β parameters can be approximated by a chi-square distribution [30] where the random variable is $\chi^2 = 2\lambda\beta$ and the number of degrees of freedom (ν) is equal to the integer closest to 2α .

Example - For the auxiliary building we have

$$\alpha = 10.32 \text{ and } \beta = 303.59$$

The number of degrees of freedom is:

$$\nu \approx 2\alpha = 20.64 \approx 21.$$

The chi-square percentiles with 21 degrees of freedom are found in Reference [29]:

$$\chi^2_{0.05} (21) = 11.6$$

$$\chi^2_{0.50} (21) = 20.3$$

$$\chi^2_{0.95} (21) = 32.7.$$

Using

$$\lambda_I = \frac{\chi^2_{I}(\nu)}{2\beta} \quad (3.11)$$

we find the percentiles of the frequency of fires to be

$$\begin{aligned}\lambda_{05} &= 1.91 \times 10^{-2} \text{ ry}^{-1} \\ \lambda_{50} &= 3.34 \times 10^{-2} \text{ ry}^{-1} \\ \lambda_{95} &= 5.35 \times 10^{-2} \text{ ry}^{-1}.\end{aligned}$$

Note that these approximate percentiles are very close to those in Table 3.2.

If chi-square tables are not available, Poisson tables may be used [30]. The closest integers of α and β are the number of events (k) and units of time (τ), and for the I th percentile we have

$$I = \sum_{n=k}^{\infty} e^{-\lambda_I \tau} (\lambda_I \tau)^n . \quad (3.12)$$

Reference 29 also tabulates this summation. The mode, mean, and variance of a gamma distribution can be found from:

$$\text{Mode} = \frac{\alpha-1}{\beta} \quad \text{for } 1 < \alpha \quad (3.13)$$

$$\text{Mean} = \frac{\alpha}{\beta} \quad (3.14)$$

$$\text{Variance} = \frac{\alpha}{\beta^2} \quad (3.15)$$

3.2.4 Sensitivity Analysis

To test the effect of the particular form of the prior distribution that we have used on the posterior distributions, we repeat the calculations with log-normal prior distributions. The lognormal distribution is

$$\pi(\lambda) = \frac{1}{\lambda \sqrt{2\pi}\sigma} \exp\left[-\frac{(\ln\lambda - \mu)^2}{2\sigma^2}\right] \quad (3.16)$$

where μ and σ are its parameters.

For the cable spreading and control rooms, the new prior distribution is lognormal with $\mu = -4.2$ and $\sigma = 2.55$ having the following characteristic values:

$$\lambda_{95} = 2.2 \times 10^{-4}; \lambda_{50} = 2.5 \times 10^{-2}; \lambda_{95} = 1.0$$

$$\langle \lambda \rangle = 0.39,$$

i.e., the distribution has the same λ_{50} and λ_{95} as the original gamma prior distribution but it is less dispersed. Its shape is shown in Figure 3.3.

The Bayesian calculations must now be carried out numerically because the lognormal distribution is not conjugate with respect to the Poisson distribution which still serves as the likelihood function.

Table 3.3 compares the posterior distributions.

We observe that the use of a lognormal prior distribution does not significantly affect the posterior distributions. This is to be expected because both prior distributions represent a fairly vague prior knowledge, thus the posterior distributions are dominated by the statistical evidence.

For the other rooms, the lognormal prior distribution is chosen with parameters $\mu = -1.20$ and $\sigma = 1.71$. Its characteristic values are:

Table 3.3. Comparison of Posterior Distributions
for Gamma and Lognormal Prior Distributions

Area		λ_{05}	λ_{50}	λ_{95}	$\langle \lambda \rangle$
Control Room	Gamma	3.1×10^{-4}	3.0×10^{-3}	1.2×10^{-2}	4.3×10^{-3}
	Lognormal	5.2×10^{-4}	3.3×10^{-3}	2.3×10^{-2}	4.3×10^{-3}
Cable Spreading Room	Gamma	1.4×10^{-3}	6.2×10^{-3}	1.7×10^{-2}	7.2×10^{-3}
	Lognormal	1.6×10^{-3}	6.1×10^{-3}	1.6×10^{-2}	7.1×10^{-3}

$$\lambda_{05} = 1.8 \times 10^{-2}; \quad \lambda_{50} = 0.30;$$

$$\lambda_{95} = 5.0; \quad \langle \lambda \rangle = 1.3.$$

Again, the posterior distributions are insensitive to the new prior distribution. For example, for the turbine building we get

$$\lambda_{05} = 1.7 \times 10^{-2}; \quad \lambda_{50} = 3.0 \times 10^{-2};$$

$$\lambda_{95} = 5.0 \times 10^{-2}; \quad \langle \lambda \rangle = 3.2 \times 10^{-2}.$$

3.2.5 The Choice of a Poisson Likelihood

The choice of a Poisson likelihood function implies that the frequency of fire occurrences is constant in time. It can be argued, however, that earlier fire incidents have offered valuable lessons and have resulted in major improvements in the safety of the plants from the fire standpoint. This is particularly true for the Browns Ferry incident [4] which is considered as one of the most significant safety-related incidents in the history of the nuclear industry. In reaction to that, the NRC has enforced a detailed plan for fire protection evaluation and updating based on the lessons learned [31]. As a result, in many power plants, the fire protection provisions have been upgraded which prompts us to think that the frequencies could be decreasing with time, an effect that would be similar to the "burn-in" region of the bathtub curve.

A simple test is performed to check this notion.

The time period is divided into two parts--one starts on the first of January 1968 and ends on the end of December 1975, the other starts on the first of January 1976 and ends on end of May 1978. The distribution of the frequency of fires for the overall plant is computed using Bayesian methods described in Section 3.2.3.

Noninformative prior distributions (i.e., $\alpha = 0$, $\beta = 0$) and the Poisson likelihood are used. The incidence data comes from Reference 10 (see Section A.2 of Appendix A for more detail) and the containment years of Table B.1 of Appendix B are utilized as unit years. The results are shown in Table 3.4.

Contrary to our expectations, we find that the overall frequency has slightly increased in recent years. We believe that this is due to the fact that, since the Browns Ferry incident, the fire reporting criteria have become more stringent. Consequently, some of the fires that are being reported today would have gone unnoticed prior to that incident because their impacts were minimal. In any case, Table 3.4 shows that the increase in frequency is very small.

3.2.6 Comparison With Frequentist Results

We demonstrate here that, since our posterior distributions are dominated by the statistical evidence, the methods of frequentist statistics give

Table 3.4 The Number of Fires, the Reactor Years
and the Frequency of Fire Incidents
in the Overall Plant for Two Time Periods

Period	1/68 - 12/75	1/76 - 4/78	1/68 - 4/78
No. of Incidents	24	25	49
No. of Reactor Yrs.	159.6	131.3	290.9
Frequency (per reactor year)			
5th Percentile	0.104	0.132	0.131
Mode	0.144	0.183	0.165
Median	0.148	0.188	0.168
Mean	0.150	0.190	0.168
95th Percentile	0.204	0.257	0.210

results that are numerically close to those of the Bayesian calculations. We use as examples the control room (weakest evidence) and the turbine building.

The maximum likelihood estimate of the frequency of fires is

$$\hat{\lambda} = \frac{r}{T} \quad (3.17)$$

and it is to be compared with the mean value of the posterior gamma distribution

$$\langle \lambda \rangle = \frac{\alpha'}{\beta'} = \frac{\alpha + r}{\beta + T} \quad (3.18)$$

Equation (3.18) shows that, when the evidence (r, T) dominates the prior knowledge (α, β) , the two estimates of Equations (3.17) and (3.18) are approximately the same. Numerically, we get

Control Room

$$\hat{\lambda} = \frac{1}{288.5} = 3.47 \times 10^{-3}; \quad \langle \lambda \rangle = \frac{1.182}{289.46} = 4.08 \times 10^{-3}$$

Turbine Building

$$\hat{\lambda} = \frac{9}{295.3} = 3.05 \times 10^{-2}; \quad \langle \lambda \rangle = \frac{9.32}{295.59} = 3.15 \times 10^{-2}$$

As expected, the agreement in the case of the turbine building (stronger evidence) is better although, for the purposes of risk analysis, the agreement of the estimates in both areas is satisfactory.

We next compare confidence intervals. The classical 90% confidence interval for λ is [32].

$$\frac{\chi^2_{I/2}(2r)}{2T} < \lambda < \frac{\chi^2_{1-I/2}(2r+2)}{2T} \quad (3.19)$$

where $\chi^2_I(2r)$ is the 100Ith percentile of a chi-square distribution with $2r$ degrees of freedom. In the present case, $I=0.10$ and we get.

Control Room

$$\chi^2_{0.05}(2) = 0.103, \chi^2_{0.95}(2) = 9.43;$$

$$1.8 \times 10^{-4} < \lambda < 1.6 \times 10^{-2}$$

Bayesian interval: $3.1 \times 10^{-4} - 1.2 \times 10^{-2}$

Turbine Building

$$\chi^2_{0.05}(18) = 9.39, \chi^2_{0.95}(20) = 31.4$$
$$1.6 \times 10^{-2} < \lambda < 5.3 \times 10^{-2}.$$

Bayesian interval: $1.7 \times 10^{-2} - 5.0 \times 10^{-2}$

The agreement is, again, better for the turbine building. Even though the frequentist and Bayesian intervals are approximately the same, their interpretation are fundamentally different. In the Bayesian approach, the value of the frequency λ that will be revealed after many years of experience is unknown and the probability that this value is in the stated

interval is 0.90. In the frequentist approach, λ has a fixed but unknown value, and the interval itself is random. If we envision many samples over the same period T and calculate the confidence intervals, then 90% of these intervals will contain the unknown value of λ .

The reason why the two intervals are approximately the same can be seen when we establish the relation between the gamma distribution and the chi-square distribution. It can be shown [30] that, if $\lambda\alpha'$ is a positive integer, then the variable $2\lambda\beta'$ has a chi-square distribution with $2\alpha'$ degrees of freedom. In the case of the control room, $2\alpha'=2.364 \approx 2=2r$ and $\beta'=290.26 \approx T$; therefore, the Bayesian confidence interval can be calculated by

$$\frac{\chi^2_{1/2}(2r)}{2T} < \lambda < \frac{\chi^2_{1-\alpha/2}(2r)}{2T} \quad (3.20)$$

then

$$1.8 \times 10^{-4} < \lambda < 1.0 \times 10^{-2}$$

which is the frequentist result (Equation 3.19), except that the upper bound here is evaluated for $2r$ degrees of freedom. The use of $2r+2$ degrees of freedom in Equation 3.19 stems from the notion that an additional fire incident may have occurred right at the termination

of the time period. This gives larger upper and lower bounds. It is normal practice [32] to choose the widest interval, thus Equation (3.19) results. For the case of smallest r (i.e., $r = 1$) the difference is not large (1.0×10^{-2} versus 1.6×10^{-2}). For larger values, it would be smaller. Of course, if the prior beliefs were significant, the parameters α and β of the prior distribution would be comparable to r and T , thus invalidating the approximations $\alpha' \approx r$ and $\beta' \approx T$. In this case, the Bayesian and frequentist results would be numerically different.

3.2.7 Normal Approximation

The normal distribution can be used as an approximation of the posterior distribution. It can be shown [19] that, for strong evidence, the posterior distribution is approximately normal with mean equal to the maximum likelihood estimate and variance given by

$$\beta^{-2} = -\left. \frac{\partial^2 \ln \pi}{\partial \lambda^2} \right|_{\lambda=\frac{r}{T}} = -\left. \frac{\partial^2}{\partial \lambda^2} \ln \left[e^{-\lambda T} \frac{(\lambda T)^r}{r!} \right] \right|_{\lambda=\frac{r}{T}} = \frac{T^2}{r} . \quad (3.21)$$

Thus, the variance is:

$$\beta^2 = \frac{r}{T^2} \quad (3.22)$$

This must be compared with the variance of the gamma

posterior distribution, i.e.,

$$\text{Var}(\lambda) = \frac{\alpha + r}{(\beta + T)^2} \quad (3.23)$$

Similar to the mean value, Equation (3.18), Equation (3.23) shows that when the evidence (r, T) dominates the prior knowledge (α, β) , the two estimates i.e., Equations (3.21) and (3.23) are approximately equal. Again, we use the control room and the turbine building as numerical examples.

Control Room

The mean and standard deviation (square root of variance) are:

$$\langle \lambda \rangle = \frac{1}{288.5} = 3.5 \times 10^{-2} \text{ ry}^{-1}, \quad \beta = \frac{\sqrt{1}}{288.5} = 3.5 \times 10^{-3}$$

The percentiles are found using:

$$\lambda_I = \langle \lambda \rangle + z_I \beta \quad (3.24)$$

where z_I is the I th percentile of standard normal distribution.

$$\lambda_{05} = 3.5 \times 10^{-3} - 1.645 \times 3.5 \times 10^{-3} = -2.23 \times 10^{-3}$$

$$\lambda_{95} = 3.5 \times 10^{-3} + 1.645 \times 3.5 \times 10^{-3} = 9.1 \times 10^{-3}$$

Bayesian interval: $3.1 \times 10^{-4} - 1.2 \times 10^{-4}$

Turbine Building

$$\langle \lambda \rangle = \frac{9}{295.3} = 3.0 \times 10^{-2}, \quad \beta = \frac{\sqrt{9}}{295.3} = 1.02 \times 10^{-2}$$

$$\lambda_{05} = 3.0 \times 10^{-2} - 1.645 \times 1.02 \times 10^{-2} = 1.37 \times 10^{-2}$$

$$\lambda_{95} = 3.0 \times 10^{-2} + 1.645 \times 1.02 \times 10^{-2} = 4.70 \times 10^{-2}$$

Bayesian Interval = $1.7 \times 10^{-2} - 5.0 \times 10^{-2}$.

Obviously, the lower bound for the control room is unacceptable. This means that the evidence (one fire incident) is not sufficiently strong and the shape of this posterior cannot be approximated by this normal distribution. On the other hand, for the turbine building, the evidence (nine fires in 195.3 years) is rather strong and the approximate bounds are quite close to the exact values.

3.2.8 Frequency-Magnitude Relationship

Fire is a frequency-magnitude phenomenon like earthquakes, tornadoes, etc. By this, we mean that, to describe the events accurately, we need their frequency of occurrence and a measure of their magnitude (severity). For example, for earthquakes, a well-defined scale for the magnitude is the Richter scale. Unfortunately, such a scale has not yet been established for fires. This lack of a measure for the magnitude of fires necessarily introduces a certain degree of fuzziness into the analysis.

The data covers a large spectrum of fires. Our judgment is that there have been many small fires that have not had any significant impact, and consequently, have not been reported. Thus, a lower bound should be envisioned for fire severities represented by the frequencies of Section 3.2.3. The whereabouts of this

bound (physically) cannot be easily determined. It will be further discussed when these frequencies are applied.

3.2.9 Uncertainties in the Frequency of Fires

Ideally, in this chapter, we should have assessed the distribution of the annual rate of fire occurrence at every point in a given power plant. In the literature, we usually find that the frequency of fires for the overall plant is assessed using industry-wide data. One such frequency is given in Table 3.4 (the last column). There, we have pooled the fire data on the different plants into one piece of evidence (r, T). Ideally, the evidence should be in terms of (r_{ij}, T_{ij}) where i specifies the power plant and j specifies the fire location. Obviously (r, T) are the sum over all i and j of (r_{ij}, T_{ij}) . This pooling of evidence into one implies that all nuclear power plants are believed to have the same fire behavior.

The resulting distribution depicts our state of knowledge uncertainty and does not give any information on the plant-to-plant or the within-the-plant variability. Note that, for larger (r, T) , the resulting distribution becomes narrower (compare Columns 1 and 2 with 3 in Table 3.4). This is consistent with our interpretation because, with more evidence, our uncertainty about the exact value of the frequency becomes smaller.

In this study, we have broken down the mentioned summation into specific locations, i.e., we have summed over all i's only, and groups of j's that represent the desired areas (e.g., turbine building). The same implications can be drawn here, i.e., all similar locations in all power plants have the same fire behavior. The results are shown in Table 3.2.

Again, the distributions depict our state-of-knowledge uncertainty. They do not show the plant-to-plant variability. That is, one cannot take the lower portions of the distributions as representing the better quality (from a fire hazard standpoint) power plants. We interpret this distribution as representing our uncertainty about the mean frequency of fires in those compartments.

3.2.10 On the Extrapolation of the Results to Other Areas

Only a limited number of areas are studied in this chapter. The extrapolation of the results to compartments not covered in Table 3.2 can be done using judgment. We illustrate this by two examples.

Example 1 - Auxiliary Electrical Equipment Room at the Zion Station

The auxiliary electrical equipment (AEE) room contains the logic circuits, small transformers, and relays for automatic control systems, interlocks, and

instrumentation. These are installed in metal cabinets. The room also contains the battery chargers and inverters. The cable trays are near the ceiling. There are no power cables in the room.

This room is not typical of all nuclear plants. Consequently, data on fire incidents is not available. The fire loading in the room is very low (5,300 lbs. of cable insulation in the cable trays and 5,200 lbs. in the cabinets and panels). It is a controlled area in that permission of the shift supervisor is required to enter the room, which happens fairly often (on the average once every 2 days). All of the components except for those which are battery-related are typically installed in the control rooms. Thus, we judge that the room is fairly similar to the control room; therefore, we can use the distribution for the frequency of fires in those rooms.

**Example 2 - Primary Auxiliary Building, Fan House,
and Control Building at the Indian
Point Station.**

The primary auxiliary building, fan house and control building house equipment that is typically found in the auxiliary building of other plants. Thus, we can assign one distribution (that of the auxiliary buildings) to the collection of these three buildings. The fraction of auxiliary building fires that would

occur in each area is estimated judgmentally based on the following observations:

- Both portions of the cable tunnel have identical characteristics except that, in one of them, a sprinkler system is installed.
- In the switchgear room, equipment with moving parts (air compressors) and breakers for electric power are installed.
- There are more than 25 compartments in these three buildings that have characteristics similar to these three fire zones.
- The most conservative situation is when all auxiliary building fires are assigned to these three fire zones.

We take the most conservative case as the upper bound (the 95th percentile). We believe that fires are more likely to occur in the switchgear room than the cable tunnels. Thus we chose 0.5 for the upper bound for the switchgear room and 0.25 for each portion of the tunnel. Note that their sum equals to 1.0.

For the lower bound we use

$$\frac{1}{25} = 0.04$$

which implies that all compartments experience identical fire occurrence rates. It may be conservative for the cable tunnels and optimistic for the

switchgear room. In both cases, we believe that the real fractions are closer to the lower bound than the upper bound. Therefore, a lognormal distribution is chosen to describe our state of knowledge uncertainty about these fractions. For the switchgear room, it has the following characteristic values:

$$Q_{05} = 0.04, Q_{50} = 0.14, Q_{95} = 0.5$$

$$\langle Q \rangle = 0.19$$

For each portion of the Cable Tunnel, it has the following characteristic values:

$$Q_{05} = 0.04, Q_{50} = 0.10, Q_{95} = 0.25$$

$$\langle Q \rangle = 0.12$$

3.3 Representative Cases for Fire Growth Analysis

Ideally, after the components of a scenario are identified, we should perform fire growth analysis for all points in the fire location to see which fires would lead to component failure. The results should then be combined with the frequency distribution of fires igniting at those points. Also, if there is more than one set of components attributed to the same accident sequence, special precautions should be taken to avoid double-counting. Obviously, this would entail large amounts of computation. We suggest that it be replaced by a bounding approach where only a few cases are chosen, such that the distance between fire locations and component locations is minimized. Ease of

failure of the components should be also considered. For example, pipes are not as susceptible to fires as cables or switchgear. Thus, their contribution to the sequence can be dropped from the analysis. Our idea of a bounding analysis may be better understood through the following two examples.

**Example 1. - Inner and Outer Cable Spreading Rooms
in the Zion Station.**

Figure 3.4 shows the routing of some of the cables relevant to the scenarios described in Section 2.3.2 and Table 2.1. It does not show the vertical separation of the cable trays. The design criteria in the Zion station [18] requires that: (1) power, control, and instrumentation cables be placed in separate trays; and (2) the redundant divisions be separated by at least 4 feet vertically and 3 feet horizontally. The latter has not been followed in the cable spreading rooms. There, the vertical separation between two trays could be smaller. However, as part of the separation criteria, solid covers (or barriers) are provided for power cable trays wherever they run below a control or instrumentation tray or the separation is less than that stated above. A similar barrier is provided for control and instrumentation cables wherever the separation of redundant channels becomes less than 18 inches.

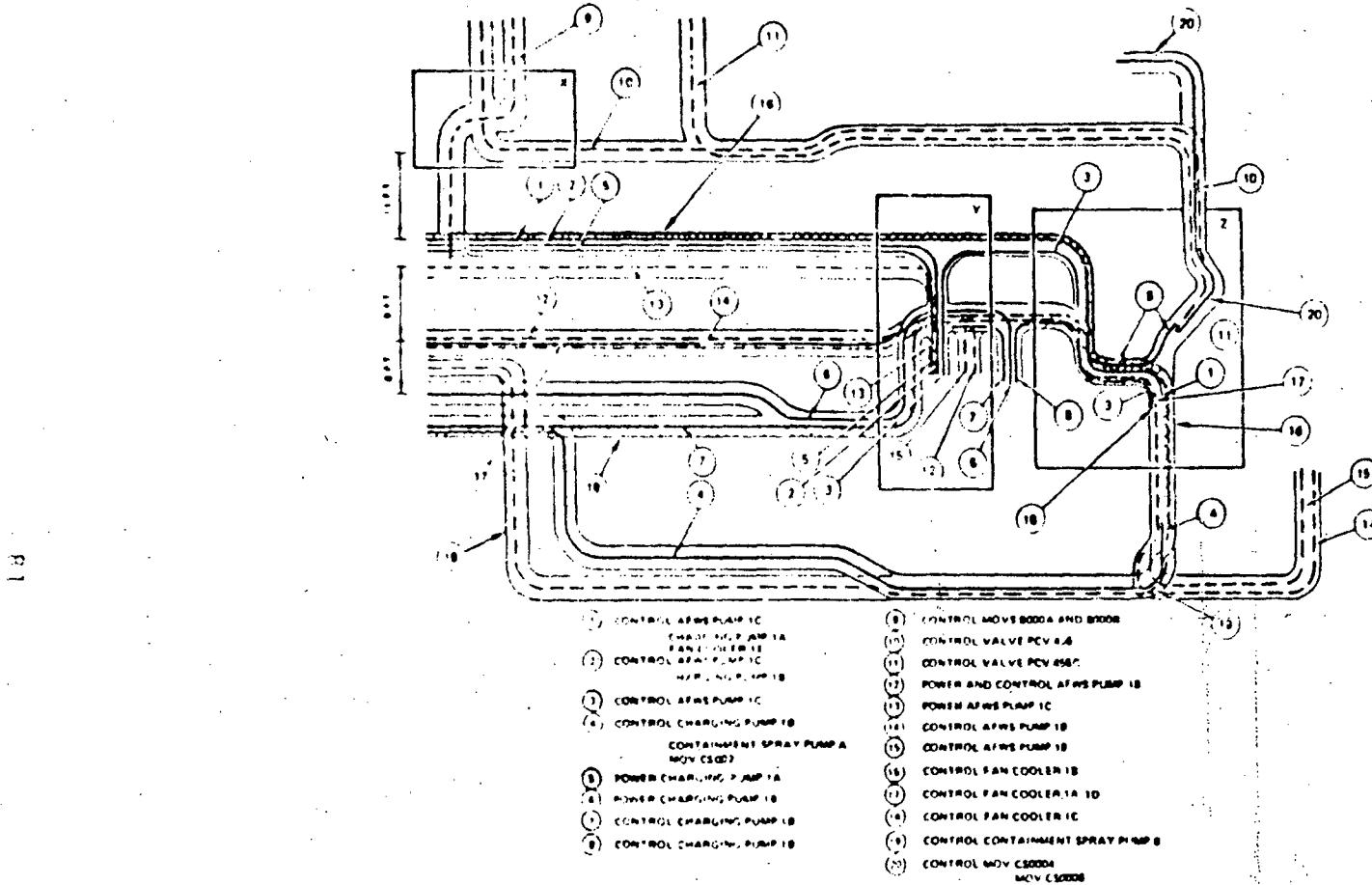


Figure 3.4 - Cable Routing of Some Specific Cables Inside the Outer Cable Spreading Room

Ideally, we should postulate fires at different points in the cable spreading rooms and study the fire growth and, consequently, cable failures from that point. As mentioned earlier, we replace this with a bounding analysis. Obviously, areas where many cables cross one another are the prime candidates here. We have identified three major zones (see the rectangles in Figure 3.4 designated as X, Y, and Z). If a fire occurs in area X, a small LOCA may occur. However, most of the cables relevant to small LOCA mitigation are far from that fire zone.

If the fire occurs in areas Y or Z, a transient initiating event is imminent. The power cables to both chargin pumps and both motor-driven auxiliary feed-water pumps can be affected. Also, the control cables to all fan coolers and two containment spray pump trains can be affected. The PORVs, safety injection pumps, turbine driven auxiliary feedwater pump, and one containment spray pump may not be affected. The unavailability of this equipment may also increase significantly because we have not identified all the control and instrumentation cables in the area. The effect of these cables on the systems, especially via the operator in the control room, is not known to us.

We see that many scenarios can be identified within one fire zone. The notion of scenario here is

slightly different than that of Chapter 2. There, it included a list of components, their failure modes, and the causes of the failures. In this section, we add the time history of the fire growth as part of the scenario. To simplify the matter, we judgmentally define a representative case for fire growth as:

1. Transient fuels ignite and form a 12x12-inch (base measurements) pilot fire on top of the cables in a horizontal cable tray (Tray 1 of Figure 3.5).
2. There is another tray parallel to, and 4 feet above, the first one (see Figure 3.5).
3. The time for the ignition of the upper cables due to the fire growth is assessed (for the reason on this choice of growth time, see the following sections).
4. When the cables of the upper tray ignite, then all the cables of the marked area (i.e., X, Y or Z of Figure 3.4) are assumed to have failed. This is because the fire is rather large when the second cable tray ignites and the additional time to a third tray or beyond is very short.

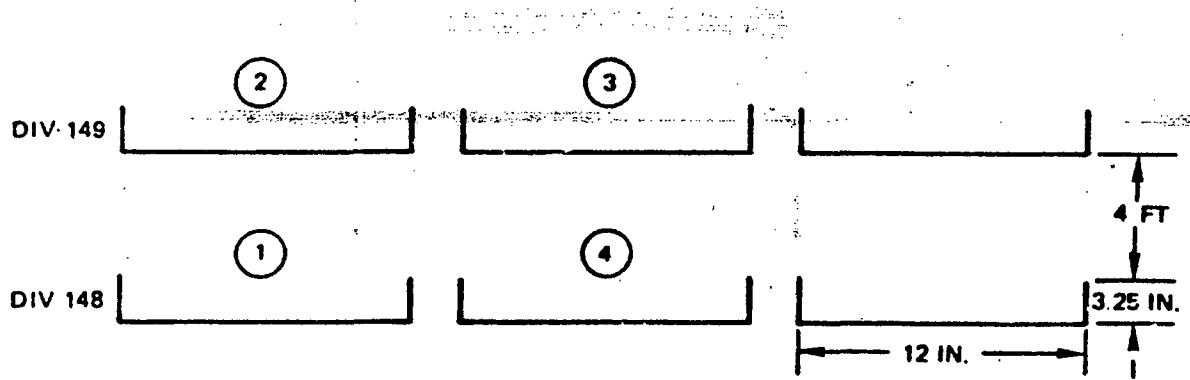


Figure 3.5 Cable Tray Configuration in the Zion
Cable Spreading Room

**Example 2. - Auxiliary Electrical Equipment Room
in the Zion Station**

In Section 3.2.10, we described the type of equipment that exists in the Auxiliary Electrical Equipment (AEE) room. Figure 3.6 shows the position of the metal cabinets and battery-related equipment. Only a partial list of the circuits and their corresponding cabinets is provided in that figure. Reference 2 describes how fires in the AEE room may lead to severe consequences. It is found that fires in areas A and B (circled in Figure 3.6) would inflict the most serious damage to the safe shutdown process by failing the circuits inside the cabinets on the two sides of the aisle.

Using judgment, we define the representative case for fire growth as:

1. Transient fuel ignites in one of the noted areas (the amount of burning fuel is discussed in Section 3.4.)
2. The fuel is on the floor.
3. The position of the center of the fire is on the centerline of the two facing cabinets.
4. The position of the center of the fire is uniformly distributed in the aisle portion and the fire does not occur inside a cabinet.

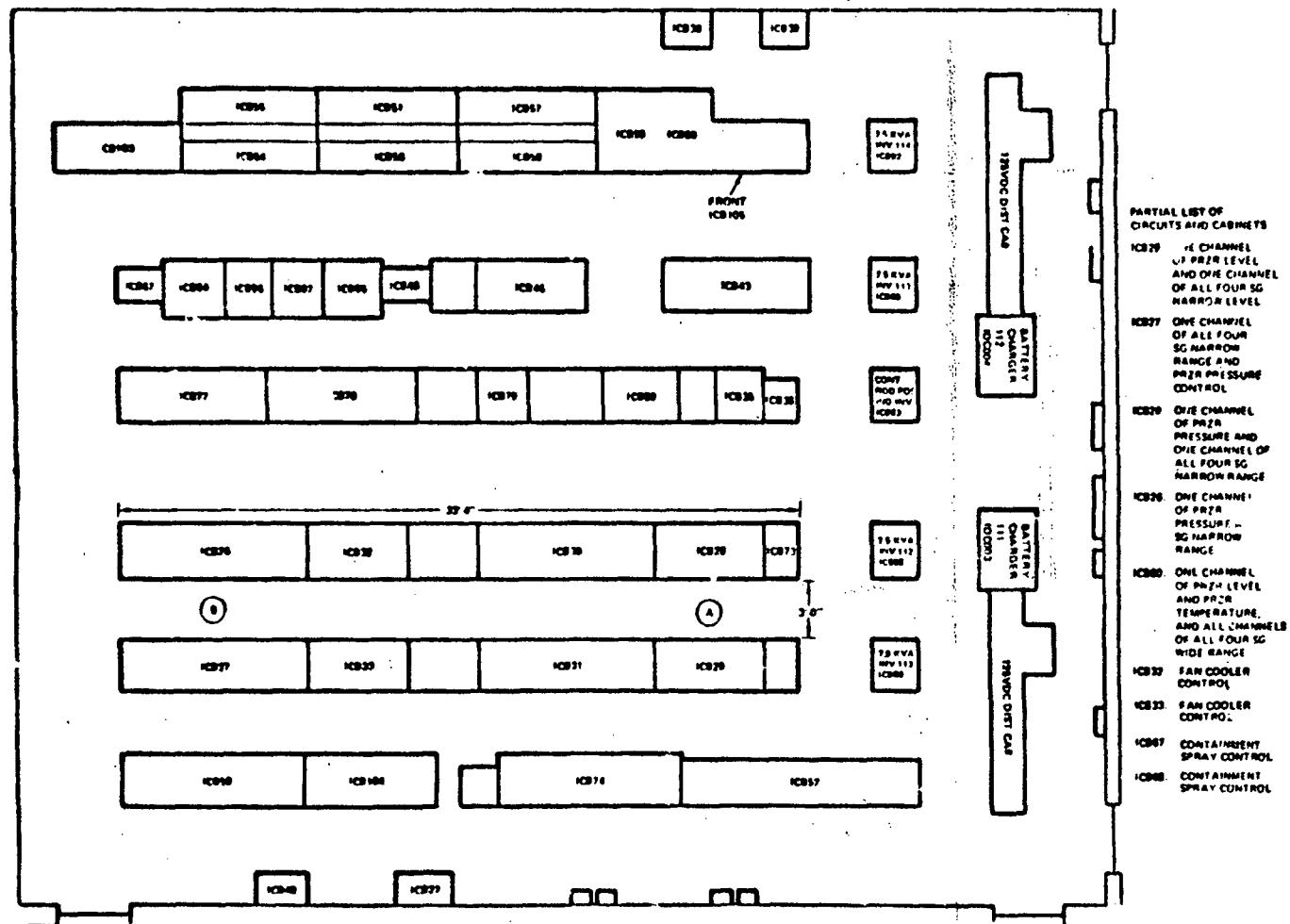


Figure 3.6 - Cabinet Positions in Auxiliary Electrical Equipment Room and a Partial List of the Contents

5. The air temperature inside the far cabinet (farthest from the flames) is of interest. In Section 3.5, we find that, for temperatures higher than 260°F, circuit failure becomes very likely. We conservatively assume that the cabinet air temperature is the same as that of the circuits.

3.4 Fire Growth Analysis

There are three parts to the problem of fire growth: the fire growth itself, detection, and suppression. Ideally, these processes should be analyzed together because growth and suppression interact with one another very closely. As a fire grows, the chances for applying some extinguishing agent become greater. As more suppressant is applied, the fire would grow less vigorously and eventually would start to decrease in size.

A simple and conservative model is proposed in this study. Fire growth analysis is performed separately from the suppression, analysis (see Section 3.4.1). We assume that fires are not affected by the extinguishing agents until the time of successful extinguishment. The goal of growth analysis is to find the probability distribution of the growth period which is defined by the representative cases of Section 3.3. The result is combined with the time to

complete suppression and the frequency of the scenario (the fire related part) is obtained. The time to complete suppression is the sum of two time periods--detection and suppression (given detection). Some preliminary thoughts are given about the characteristics of these two time periods (i.e., detection and suppression) in Sections 3.4.3 and 3.4.4. Statistical data is used in assessing their sum.

3.4.1 Growth Analysis

The goal of this part is to establish the probability distribution of the growth period without considering the effects of suppression activities. A physical model for fire growth has been developed in Reference 33. The input to this model includes the fuel type and its distribution within the room among other compartment parameters. The output is a multitude of response surfaces which give the growth time as a function of input parameters and the heat content of the pilot fuels. The different curves represent our state of knowledge uncertainty about the input parameters. The pilot fuels, the fuels upon which the fire burns initially, are emphasized here because they are used to model ignition as noted in Reference 33. The following example illustrates the approach. It is summarized from that reference.

**Example - Cable Spreading Room fires in the Zion
Station**

In the first example of the previous section (Section 3.3), we defined a representative case for fire growth in the cable spreading rooms of the Zion station. Figure 3.5 shows the cable tray configuration that is analyzed. As part of the representative case, we assume that a 12x12-inch pilot fire is established on top of the cables in Tray 1. The objective is to assess the time for the fire to propagate to Trays 2 and 4. Figure 3.7 shows the result [33]. The probability is a measure of our state of knowledge uncertainty about the input parameters and modeling of fire propagation. Note the large uncertainty band.

The distribution of pilot fuel heat content (Q_p) used in this example is as follows:

Q_p Btu	Probability
400	0.10
2,000	0.44
10,000	0.44
40,000	0.02

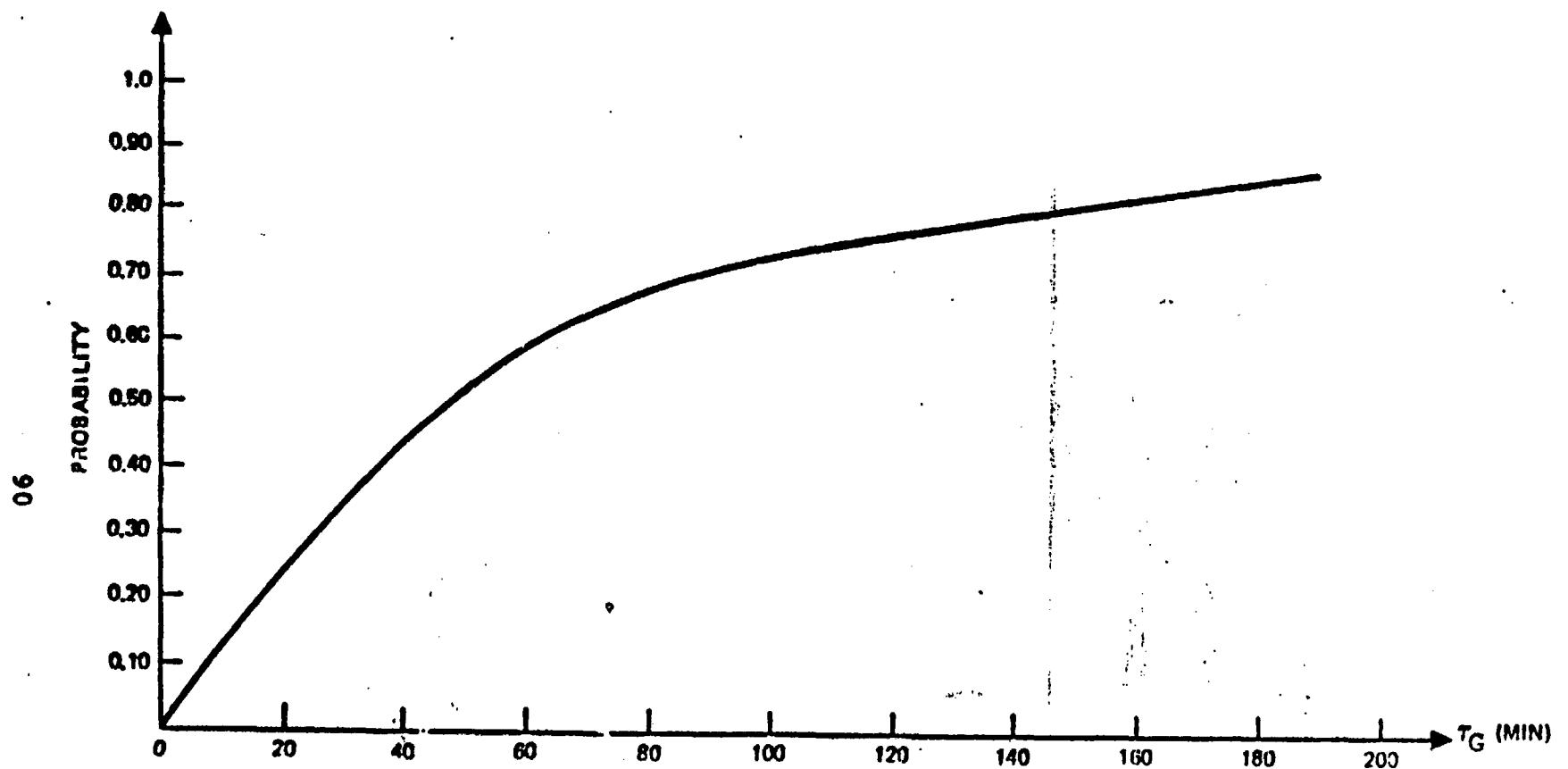


Figure 3.7 Distribution of τ_G Which Includes Both Parameter and Modeling Uncertainties

3.4.2 On Ignition and Pilot Fires

In Reference 33, the process leading to initial fire is not modeled because the frequency of fires can be estimated from the available data discussed in Section 3.2. In that reference, it is assumed that a small, established fire, which we call a pilot fire, already exists in the area of interest. For example, in the cable spreading rooms of the Zion station, we assumed it to be on top of the cables in Tray 1 of Figure 3.5. Two pieces of information are necessary here: (1) the type, and (2) the amount of the combustibles in the pilot fires. From the Fire Protection Reports [17], we can find the installed fuel contents of a compartment. In addition to these, there might be transient fuels whose quantities are subjectively estimated because little statistical data exists about them.

Before further discussion, we consider what types of material and what quantities constitute the transient fuels. We cite the following quotations from different sources. For cable trays, Reference 34 mentions "a review of past cable tray fires indicates that trays are a collection place for combustible waste materials such as lunch wrappers or oily wiping cloths." The heat contents of these items are estimated to be about 200 Btu and 5,000 Btu, respectively.

In the Fire Protection Evaluation Reports [17], the amount of transient fuels is also estimated. For a general area in the plant, the Zion Fire Protection Report [18] states (page 2.1-2): "transient fire loads are unidentified combustibles, defined as being equivalent in Btu content to the fire load that would be contributed by the combustion of a 55-gallon drum of lubricating oil or greater as defined in the fire area analysis." However, the analysis becomes more specific when individual areas are analyzed. For example, for cable spreading rooms, it is stated that (p. 2.3-16): "transient loads are not considered in this room, since it is a controlled area."

The Indian Point Fire Protection Report [35] defines three levels of transient fuel loading defined in terms of type and amount of the fuels. For example, Type 1 transient fuels include 5 pounds of wood, 2 pounds of paper, and 3 pounds of plastic with a total heat content of 76,000 Btu. The most severe case is called Type 3 which contains 15 pounds of wood, 8 pounds of paper, 8 pounds of plastic, 5 pounds of oil, and 10 pounds of grease; the total heat content is 433,000 Btu. It is interesting to note that, for the cable spreading room, it is judged that Type 3 transient fuel loading applies. The lubricating oil may be present because the motor-generator sets of the

Reactor Protection System are installed in that area. In both cases (Zion and Indian Point), the fire loading is given for a compartment as a whole and its density per unit floor area is not specified.

From this information, we judge that, in areas where machinery with lubricating oil is installed, the predominant fuel for the pilot fires is oil (e.g. the diesel generator areas or pump areas). In other locations, the transient fuels are the main contributors to the pilot fires. Among such areas, we can mention the cable spreading rooms in Zion where the combustibles (cables) are not readily flammable and the auxiliary electrical equipment room in Zion which is kept very clean and to which access is strictly controlled.

For the models of Reference 33, the pilot fires are expressed in terms of a probability distribution of their heat content and the type of fuel. One such distribution is given in the example of the previous section. The first two lowest values (i.e., 400 and 2,000 Btu) are judged to be fueled by cellulosic materials such as wood and paper. The higher values are attributed to lubricating oil.

3.4.3 Detection Time

The objective of this section is to assess the time it takes after fire ignition for the plant personnel to become aware of the fire. A fire can be

detected either by the personnel or by automatic detection systems. In References 10 and 11, we find that about 75 percent of the fires that occurred during the operational phase of the nuclear power plants were detected by humans. This is an average fraction and specific cases could be quite different. For example, in areas where access is normally restricted, this fraction may be much smaller. Unfortunately, the data available to us is not classified by the combination of detection mode and location.

The time for fire detection by humans would be almost instantaneous in the incidents where they are the cause of the fire or when the area has significant through-traffic. The detection time may be longer (if they are not started by humans) for very small fires, smoldering fires, or restricted areas such as the cable spreading room.

The remaining 25 percent of the fires were detected by automatic systems. There are many types of fire detectors and detecting systems. These are described in References 36 through 40. The detectors commonly used in nuclear power plants are sensitive to smoke. Reference 41 develops an expression for heat sensing devices. In Appendix C, some test results for the response times of smoke and heat detectors are quoted. These tests were performed in residential and

hospital environments. Power plant environments could be very different [40] because the compartments in nuclear power plants are generally large and air flow patterns are very different.

From those tests, we can see that there is a finite likelihood that the detectors would not respond at all. This, of course, depends on a multitude of parameters such as the air flow characteristics of the compartment, the size of released particles, detector type, and availability of the detector itself. It should be noted that, in about 50 cases which References 10 and 11 have studied, all fires were detected before self-extinguishment. Obviously, the frequency of such events would also depend on the magnitude of the fires. A large fire should have a vanishingly small frequency of remaining undetected.

Due to lack of data, there is no clear-cut approach in assessing fire detection time. The information given in this section and in Appendix C should be used judgmentally for more specific cases.

Example - Cable Spreading Rooms in the Zion Station

The cable spreading rooms are controlled access areas where work is not done under normal operating conditions. The rooms are provided with ionization-type smoke detectors that annunciate in the control

room. They are positioned in a square lattice that measures about 20 feet on each side.

We judge (conservatively) that detection is solely dependent on the automatic system and credit is not given to human presence. It is very likely that the fire would be detected within the first 5 minutes. We do not agree with the large probability of no response, found in Appendix C. We believe that it is vanishingly small because fires relevant to this study (which means fires of rather significant severity) would eventually fail some cables such that they would disturb the indicators on the control boards in the control room, thus prompting the operators who would be searching for the cause of the disturbance. We believe that, within 60 minutes, it is extremely likely that the fire would be detected.

3.4.4 Suppression Time

The suppression time is loosely defined as the time period between fire detection and total extinguishment. Many parameters influence this time period. For example, the extinguishing agent and system, severity of fire, nature of fire, accessibility of the fire zone, etc. Similar to detection, fires can be extinguished either manually or automatically.

In the data given in References 10 and 11, we find that about 80% to 90% of the fires in nuclear power

plants were extinguished manually. It should be added here also that suppression time is a function of fire severity.

Very few sources have studied fire suppression time. Thus, in deriving the frequency distribution for suppression time, subjective judgment should be extensively employed. Appendix D provides information that can help us to assess these distributions. In the first part of that appendix, we find some estimates on suppression time. The second part shows that the uncertainties must be very large, and in the last one, failure to extinguish is discussed.

In Section D.1 of that appendix, we find two distributions for the time between detection and putting the fire under control. One distribution is for the overall plant and the other is for electrical fires. Both are based on data given in Reference 10 which are in most cases, estimates by experts. These data did not specify the type of extinguishing agent or system used and, in all cases, the fire is eventually extinguished.

The two longest suppression times occurred at the Browns Ferry power plant. The last one (24-hour duration) is the charcoal adsorber fire incident of the off gas system on July 18, 1977 [42]. The other suppression time (7 hours) belongs to the well-known cable

spreading room fire [3]. In that incident, the main reason for delay was the operator judgment about not applying water to an electrical fire.

In Section D.2, fire suppression test results are given. From the data of that section we conclude that: (1) there should be a large variation in suppression time; and (2) even in the cases of large fires, the physical process of extinguishment can be short given that the necessary equipment and personnel accesses are readily available.

Several sources [12,36,43-45] have studied the effectiveness of the extinguishment systems. In all cases, it has been expressed in terms of a fraction; however, the definitions are not uniform. We interpret them as the complementary frequency of failure to extinguish. For sprinkler systems, a sufficiently large data base exists. The overall effectiveness of all types of sprinkler systems and all types of conditions varies between 88 percent (the Factory Mutual experience, see Reference 43) and 99 percent (the Australian experience, see Reference 44).

For extinguishing systems other than sprinklers, lower effectiveness fractions are generally reported and the data is very sketchy. For example, for carbon dioxide type extinguishers, the effectiveness fraction is about 50 percent. In Reference 12, an extensive

treatise on these types of extinguishing systems and agents is given.

Example - Cable Spreading Rooms in the Zion Station

In the examples of the previous chapters and sections, we have analyzed some of the different aspects of a fire incident in the inner and outer cable spreading rooms of the Zion station. Fire suppression is examined here.

Portable extinguishers are located in these rooms. A manual hose station is in the outer cable spreading room and has access to the inner room. A Halon primary suppression system with a CO₂ backing is also installed.

In Section 3.3, we find that areas Y and Z (Y is in the outer room and Z in the inner) of Figure 3.4 are the most damaging locations for fires. Both are easily accessible because they are close to the entry doors from the stairwell and cable densities are rather low. In a cable spreading room fire, the main burning fuel would be the cables; therefore, the growth rate would be relatively slow. Thus, it is very likely that the fire would not be large at the time the personnel would attempt to extinguish it. However, on the other hand, due to burning cables the fire would be producing large amounts of smoke and impede the accessibility of in fire area.

3.4.5 Fire Duration

We define fire duration as the time between ignition and complete suppression. It is the sum of two time periods, detection and suppression. Ideally, we should define a third period in between these two which would represent the time it takes after detection to start applying an extinguishing agent. For example, this would be the time between fire alarms sounding in the control room and operators applying the CO₂ onto a fire. In the previous section, suppression time is defined such that it covers this middle period also. The suppression time and detection time are not independent random variables. Thus, in deriving the probability distribution of the fire duration, one should use the joint distributions of the two random variables. In the following example, we use the statistical data given in Appendix D for this purpose.

Example - Cable Spreading Rooms, Zion Station

In the examples of Section 3.4.3, and 3.4.4, we describe the detection and suppression mechanisms in case of a fire in the inner or outer cable spreading rooms. In Section 3.4.3, we have found that the majority of fires in nuclear power plants are detected manually and also, if the detection is not manual, a good majority of them would be detected automatically within a few minutes (see Appendix C). In Table D.1 of

Appendix D, we find that 54 percent of the fires that have occurred in nuclear power plants involving electrical insulation were suppressed within 18 minutes. Furthermore, in the example of Section 3.4.4, we have found that the critical areas of these cable spreading rooms are easily accessible. Based on these arguments, we chose 40 and 30 percent for the average fire duration of 5 and 15 minutes, respectively. We believe that the former is more likely than the latter because there are good chances that the fires are caused by the personnel themselves.

In Table D.1 of Appendix D, we find that 29 percent of fires involving electrical insulation took about 1/2 hour to be brought under control and, similarly, for 16 percent of those fires, it took more than 1 hour. Thus, this table suggests a slowly decreasing upper tail for the distribution of the fire duration. We believe that, for the cable spreading rooms of the Zion station, it (the tail portion) decreases rather rapidly because the data base for Table D.1 includes the Browns Ferry incident [3]. The fire had stopped propagating in the earlier stages in that incident and further damage was not being inflicted. Furthermore, we do not believe that such a long delay (7 hours) in suppression is likely to occur today in light of the lessons learned from that incident. Therefore, we chose 20 percent and 10 percent for the

average fire durations of 1/2 and 1 hour, respectively. This histogram is shown in Table 3.5. It gives our state-of-knowledge uncertainty about the average fire duration.

3.5 Component Failure Under Fire Conditions

3.5.1 Introduction

The effects of fires on components typically found in power plants are discussed in this section.

Exposure to heat is our main concern. Effects of smoke or extinguishing agents are discussed for only some limited cases.

In the following section, the failure of electrical cables is discussed; their failure modes are identified (Section 3.5.2.1) and frequencies of failure are analyzed (Section 3.5.2.7 through 3.5.2.9).

In Section 3.5.3, the failures of other components are analyzed and, in the last two (Sections 3.5.4 and 3.5.5), the effects of extinguishing agents and smoke are discussed.

3.5.2 Electrical Cables

3.5.2.1 Failure Modes of Electrical Cables

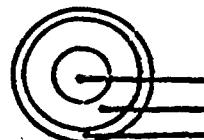
Many different types of cables are generally used in a nuclear power plant. In most of the parts of a plant, they are laid horizontally in cable trays and, in some areas, they may be routed through metal conduits. The power cables are usually of a single

**Table 3.5 Fire Duration in the Inner or Outer
Cable Spreading Rooms in Zion**

Average Fire Duration (Minutes)	Probability	Cumulative Probability
5	0.4	0.4
15	0.3	0.7
30	0.2	0.9
60	0.1	1.0

conductor-type with large core (the copper conductor) to outer-diameter ratio. The control or instrumentation cables are generally smaller in diameter and may have two or more conductors (wires). Figure 3.8 shows several cable cross sections. Several failure modes may occur depending on cable construction, circuit design, and the adjacent cables. Two classes of failure modes are identified. The first class (or failure mode) which we call "open circuit" consists of failures that lead to loss of functionality. That is, the signal (or power) does not reach its destination; for example, a grounded power cable or a shorted control cable. The second group, called "hot short," consists of failures that cause an inadvertent action; for example, the large LOCA occurrence described in the example of Section 2.1.2. Figure 3.9 is another example. Here, if wires A and B contact each other, the solenoid valve would open and the air operated valve would change position. These wires could be in the same cable (see Figure 3.8).

We believe that these two failure modes (hot short and open circuit) are important in fire risk analysis. Other failure modes lead to one or the other. For example, circuit disjunction, when a conductor (wire) experiences severe damage (e.g., melting) and its continuity is lost, is an open circuit. In the case of

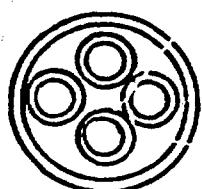


Conductor
Insulation
Jacket

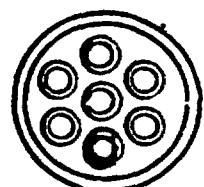
Single
Conductor



2-Conductor



4-Conductor



7-Conductor

Figure 3.8 A Sample of Cable Cross Sections

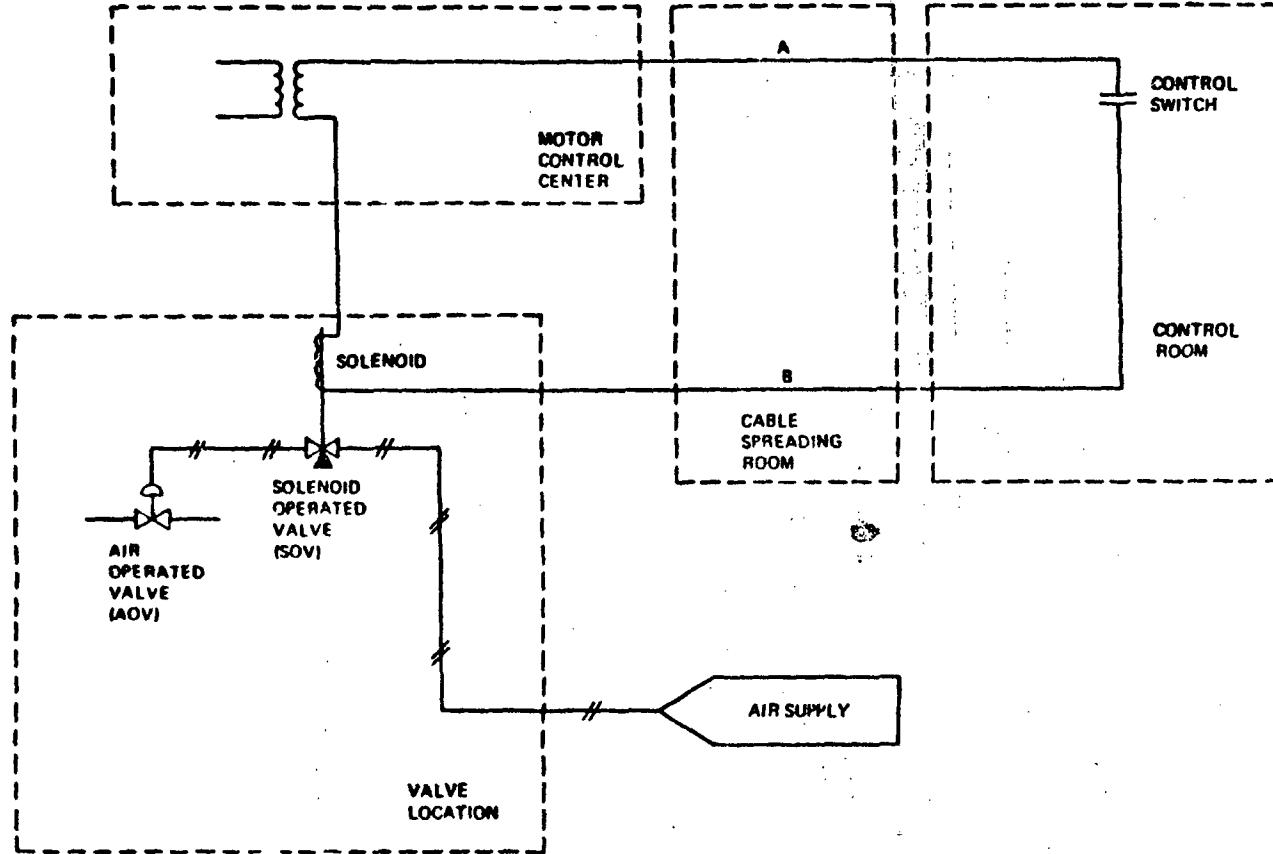


Figure 3.9 - Simplified Schematic Diagram of the Control Circuit of a Solenoid Valve That Operates an Air-Operated-Valve

a grounding, when a conductor contacts another grounded conductor, the outcome is the same as circuit disjunction. When two not-grounded wires contact each other, the outcome would depend on their position in the circuit. The possibility of secondary fires is a different situation and may also be important. It is discussed further in Chapter 4.

"Open Circuit" is the dominant failure mode because all conductors will eventually contact the grounded cable tray. The "Hot Short" cannot exist with grounded wires. Therefore, it will eventually lead to "Open Circuit" as more insulation decompose or melt away. However, the outcome of a "hot short," (e.g., energization of the motor operator of a valve) may not be reversed because, some circuits are designed such that the command signal is locked into the circuit and another signal is necessary to reverse the action.

3.5.2.2 Cable Failure Frequency

Fire exposure can be translated as heat flux to the cable by radiation and convection. Depending on its position with respect to the fire, one or the other becomes important. In both cases, some heat flux would be impinging on the cable surface and its overall temperature would be rising from its preaccident level. Certainly, the temperature distribution within the cable would not be uniform. The outer surface would be

very hot and the inner section would be cooler because the copper conductor would conduct the heat away to the cooler parts of the cable. Beyond a certain temperature level, the materials will change state or decompose chemically. Thus, as certain sections of the cable reach this threshold temperature, they will experience drastic physical changes. Cable damage starts at this point. However, circuit failure would be delayed, depending on the shape and characteristics of the cables. For example, the jacket may be damaged but the insulator would still be in functional form. The exact modeling of these processes involves very complex formulations and its solution could be extremely time consuming if not impossible.

Several parameters are very important in these processes; these are: impinging heat flux, heat losses, threshold temperature for failure, materials behavior at high temperatures, and time (to allow the temperature to rise due to the heat flux). The cable failure time is a function of these parameters and the direction of dependence is quite obvious. For example, under higher heat fluxes, time to failure should become shorter. We call these parameters impact parameters.

In the preceding subsection, we identified two failure modes--open circuit and hot short--and we concluded that the latter eventually leads to the

former if the cables continue deteriorating. These arguments imply that there should be three frequencies: (1) the frequency of cable failure given that it is exposed to flames, (2) the fraction of these failures that lead to a certain failure mode, and (3) the duration of the failure mode. In this subsection, we concentrate on the first frequency.

Many cable fire tests have been performed by the manufacturers, utilities, and regulatory bodies. Their results are summarized in Appendix E. In almost all tests, cable failure time is recorded and very few have reported the impinging heat flux or cable temperature. The cable failure time can be easily incorporated into our model for fire growth and component failure. The following observations summarize our state of knowledge about cable failure time:

- In reported cable fire tests, the failure time ranged from 1.0 to 20.0 minutes and, in some cases, failure did not occur. In Appendix E, we give further details on these test results.
- Most of the cables failed between 5 and 10 minutes.
- In many instances, different fire tests (not necessarily cable fire tests) have resulted in contradictory conclusions.

- The cable fire tests performed so far have only modeled limited variations of fire incidence characteristics.

The last two observations imply that simple statistical manipulations of the data may yield incorrect results because, they may well be representing only a limited set of fire scenarios. Furthermore, the time to failure is measured from the moment the cables were immersed in the flames. However, in the case of a growing fire, a cable may be partially deteriorated due to the impinging heat fluxes before the flames have reached it. This introduces a degree of fuzziness in the exact definition of failure time because, ultimately, the time between component failure and fire ignition should be obtained. Because of these two limitations, we choose to use the growth period to represent the time between ignition and cable failure. The result is certainly conservative.

3.5.2.3 Relative Frequencies of the Failure Modes

"Hot shorts" are very specific failure modes because they require very specific sets of events. For example, we saw that, in Figure 3.9, wires A and B should contact each other to cause this type of failure. Also, frequency of this event depends on their location relative to each other. They could be in separate

cables or in a multiconductor cable. The number of wires in the latter is important also (see Figure 3.8). For example, the relative frequency should be large for a two-conductor cable when compared to a seven-conductor one because, for the latter, conductors from other circuits may cause other effects.

Data on "hot shorts" are very sketchy. In the Browns Ferry fire incident [3], some systems (mainly ECCS) started spuriously in the first half hour. However, the fraction of cables that led to this is not known. Another source is the cable fire tests (see Appendix E for a discussion on these tests). In some bonfire tests, the cables were hung over a burning bucket of oil (as compared to laying them on a cable tray). Shorting of the conductors was observed extensively; however, circuit disjointment rarely occurred. This shows that there is a significant frequency (on the order of 0.1 or larger) that wires in a multiconductor cable would contact one another before touching the grounded tray. However, since only specific wires can form a "hot short," then its frequency must be lower and must depend on the relative position of the wires concerned. For a multiconductor cable that contains both of the wires, we judge that the frequency is less than 0.2. We express our state of knowledge

uncertainty by a lognormal distribution with the 5th and 95th percentiles at 0.01 and 0.2.

The characteristic values are:

$$Q_{05} = 0.01, Q_{50} = 0.045, Q_{95} = 0.20$$
$$\alpha_0 = 0.068.$$

The relative frequency of open circuit is the complement of the frequency of "hot short." The characteristic values are:

$$Q_{05} = 0.80, Q_{50} = 0.955, Q_{95} = 0.99$$
$$\alpha_0 = 0.932.$$

3.5.2.4 Duration of a Failure Mode

As mentioned earlier, "open circuit" is the dominant failure mode and, when occurred, would not change (most probably) into other failures. However, further deterioration of the insulating materials would cause the hot shorts to become open circuits. In the Browns Ferry incident, the spurious signals occurred mainly in the first half hour. We believe that the time for "hot short" leading to "open circuit" is distributed normally with 5th and 95th percentiles at 5 and 35 minutes.

Example - Termination of Small LOCA

In Section 2.2.7, it is found that a small LOCA may occur due to the spurious opening of two air-operated valves. These valves are designed to fail closed. This means that upon loss of air pressure,

they transfer to the closed position. This is also the deenergized position of the solenoid valve. The latter would eventually occur due to "hot shorts" turning into "open circuits." Thus, the small LOCA will terminate (almost certainly) within 40 minutes of its occurrence. No severe consequences can occur within this time period, even if all of the cooling systems become unavailable.

3.5.3 Other Components

Instrumentation and control circuits usually consist of electronic components (such as amplifiers and bistables), small transformers, and relays. They are grouped together because they are normally found in the same location inside metal cabinets. The behavior of the electronic components changes with temperature. Some circuits may be designed such that these changes are compensated. Thus, the effect of temperature on solid state devices depends on their components and circuit design. The typical operating temperature is about 38°C (100°F) and they can withstand a temperature rise of up to about 125°C (260°F) [46]. It has been observed that their failure rate increases dramatically with temperature [47] and the instrumentation circuits drift considerably, thus giving erroneous information to the control room operators.

The effect of fire on this type of electrical

components can be modeled by component temperature level. The time element of the heating process is not included because, in Reference 33, it is found that, for components inside a metal cabinet and a fire outside the cabinet, the component temperature rises to an equilibrium.

Unlike electrical cables, we do not have any fire test results for this type of electrical components. In the first paragraph of this subsection, we have summarized the available information.

By electrical equipment we mean such items as pumps and valves that need electrical power to perform their function. Their failure mechanism is very similar to cables because it involves insulation deterioration. The power cables to this equipment are its weakest part. These cables are usually run in metal conduits when they leave a cable tray and meet the component. The power cable would fail before the internals of the component because the conduit would act like an oven and expedite the failure process.

Pipes are generally filled with water; therefore, they would not fail due to a fire because of the large heat sink that water provides. Dry pipes are likely to fail if they become exposed to flames for a long time.

3.5.4 Effects of Extinguishing Agents on Component

Availability

The only extinguishing agent that may interfere with the operation of a component is water. Its electrical conductivity is the main source of trouble. Halon and CO₂ are in gas form and most likely would not interfere with component operation except for such equipment as the diesel generators that need air for normal operation.

Application of water on cable fires has been discussed extensively in Reference 48. Based on the information given in that source, we make the following conservative judgments:

- (1) If a cable is affected by the flames, then cable failure due to water application is a certainty.
- (2) The relative frequency of experiencing a "hot short" or an "open circuit" is the same as that given in Section 3.5.2.3
- (3) The duration of failure is infinite for both failure modes. That is, "hot shorts" would not turn into "open circuits" because the insulation would stop deteriorating after extinguishment.

(4) If a cable is intact (i.e., the flames have not affected it yet), water would not have any impact on its operation. Splices may be susceptible to water but this failure mode would not have a significant contribution because only bad splices would fail. Usually, there are very few splices along a cable tray.

Susceptibility of the electrical equipment to water spray depends on its design. Electrical motors can fail; however, they could be encased in a manner such that water cannot reach them. Transformers are similar to motors. Switchgear and motor control centers are highly susceptible to water. The cabinets in which they are located usually have waterproof features.

3.5.5 Smoke Damage

The extent of smoke damage depends on the burning material. For example, burning PVC cables emit chloride acid which is extremely corrosive. In one fire incident [49], a cable fire caused extensive damage to relays and switchgear about 40 feet away from the combustion zone. We believe that the impact of smoke would not fail components within the first few hours after the fire incident. Thus, they do not pose an immediate threat to plant safety. However, its impact

on operator effectiveness is crucial and should be addressed separately.

3.6 Frequency of a Sequence of Events

In this section, we propose formulations for the frequency of severe consequences due to fires. A model is suggested for evaluating the fraction of fires that could propagate and fail components. Some parameters in both models have not been quantified yet. We have included them for the sake of completeness and to show how they fit into the overall picture.

The frequency of a sequence of events, ϕ_S , can be written as the product of three frequencies: the frequency of fire occurrence, ϕ_F ; the conditional frequency of fires failing part of the sequence given that a fire has occurred, $\phi_{S/F}$; and the frequency of occurrence of the remaining events of the sequence due to other causes, ϕ_o . The mathematical expression would be

$$\phi_S = \phi_F \phi_{S/F} \phi_o . \quad (3.26)$$

The quantification of the latter (i.e., ϕ_o) is extensively discussed in studies such as References 13 and 14. The first frequency (i.e., ϕ_F) was discussed in Section 3.2 and the second one (i.e., $\phi_{S/F}$) is analyzed here.

Two competing phenomena determine the fraction (frequency) of fires that would cause a sequence of

events (or part of it). These are, on the one hand, fire growth and component failure processes and, on the other hand, fire detection and suppression. We call the latter fire duration. If the first one occurs sooner than the other one, then the sequence would materialize. Thus, we model these processes in terms of their governing time periods, i.e., component failure period, detection period, etc. In the scenarios of Chapter 2, we found that a fire may cause some portion or the whole sequence of events. In the rest of this section, we refer to both as "sequence of events" or "sequence."

We define the component failure time, T_{GF}^i , as the time after ignition when component i fails and T_{GF} as the time after ignition that the sequence occurs. The latter occurs when the last component of the sequence fails. We can write:

$$T_{GF} = \max_{\text{all components } i \text{ of the sequence}} (T_{GF}^i) \quad (3.26)$$

Define T_F as the time to failure measured from the time the flames reach the component. The growth period is given in terms of the time it takes for flames to reach component i , T_F^i , after ignition. Then T_{GF}^i can be written as:

$$T_{GF}^i = T_{GF} + T_F^i. \quad (3.27)$$

The fire duration, T_{FD} , is the time from ignition to final extinguishment and is the sum of three time periods: detection (T_D), application (T_A), and suppression (T_S). Detection period is the time that it takes for the personnel to become aware of the fire. The application period is defined as the time between detection and inception of extinguishment efforts. The suppression period is obviously the time that it takes to put out the fire after exposing it to an extinguishing agent. Thus, we can write:

$$T_{FD} = T_D + T_A + T_S. \quad (3.28)$$

Certainly, these are dependent random variables and have the fire growth characteristics as the common denominator.

The sequence of events only exists when $T_{GF} < T_{FD}$. Thus, the frequency that we are interested in:

$$\Phi_{S/F} = fr \left(\begin{array}{l} \text{sequence} \\ \text{occurrence} \end{array} \middle| \text{Fire} \right) = \\ fr (T_{GF} - T_{FD} < 0). \quad (3.29)$$

To quantify this frequency, we need to have detailed information about each variable. That is, their statistical variations should be known. From the available data in most parts, we get the mean values for these time periods and we assume that the fire

duration is distributed exponentially with mean τ_{FD} . This implies that the frequency of extinguishing the fire within the time period $t' - t$, given that it is not extinguished by the time t , is independent of t . The available data (see Section 3.4.4) does not contradict with this assumption.

The fraction of fires that propagate and lead to the sequence is equal to the frequency of fire duration exceeding the time to sequence occurrence. For the latter, we use the mean time, τ_{GF} , thus ignoring the statistical variations in the component failure and fire growth processes. We can write:

$$\phi_{S/F} = \exp \left(- \frac{\tau_{GF}}{\tau_{FD}} \right) \quad (3.30)$$

Equation (3.30) is derived for components such as cables whose failure is modeled in terms of mean time to failure. In the following example, we illustrate an application of a simplified version of the proposed model.

Example - Inner and Outer Cable Spreading Rooms of the Zion Station

In the example of Section 3.3, we found a representative case for fire growth analysis. According to that case, if two cable trays in two certain areas (areas Y and Z of Figure 3.4) within these rooms become engulfed in a fire, a transient would occur and the

power cables to both charging pumps, both motor-driven auxiliary feedwater pumps, the control cables of all containment fan coolers, and two of the three containment spray pump trains would be affected. In Table 2.1, we find several scenarios for the cable spreading rooms. We can see that part of Scenario No. 8 agrees with the above list. The turbine-driven auxiliary feedwater pump, the PORVs, the SI pumps, and one containment spray train may remain unaffected.

However, operator action during the course of the transient is very important because the information on the control board may become severely affected by the fire. In our scenario analysis, we did not look into the instrumentation and control cables. Their failure may cause conflicting or erroneous information on the control board. We conclude that the unavailability of these components, physically unaffected by the fire, may be dominated by human error. The only exception here is the turbine-driven auxiliary feedwater pump that would certainly start upon cable failure, even if the operators had interfered with its operation originally. For this, we conclude that for core melt to occur the turbine-driven pump should be unavailable during the fire incident. If the unaffected containment spray pump train remains available, then the containment event tree entry state G would result

because all of the fan coolers would be failed by the fire. Then, Equation 3.25 should be modified to

$$\phi_G = \phi_F \phi_{S/F} \phi_{HE} \phi_{TD} \quad (3.31)$$

where

ϕ_{HE} = the unavailability of the unaffected components due to other causes, except for the turbine-driven auxiliary feedwater pump,

ϕ_{TD} = the unavailability of the turbine-driven auxiliary feedwater pump due to all causes.

Equation (3.30) should be used for $\phi_{S/F}$. First, we evaluate τ_{GF} from Equation (3.27). There is only one growth period, τ_G , for all of the cables because of the way the representative case is defined. Then we define

$$\tau_G = \tau_G^i \quad (3.32)$$

and its distribution is plotted in Figure 3.7.

Table 3.6 gives the histogram that has been derived from this distribution. For mean time to cable failure, we conservatively assume that

$$\tau_F^i = 0, \quad \text{for all } i. \quad (3.33)$$

**Table 3.6. Histogram Derived from Figure 3.7
for Fire Growth to Second Division**

τ_G (Minutes)	Probability	Cumulative Probability
10	0.235	0.235
30	0.207	0.442
50	0.168	0.610
70	0.095	0.705
90	0.069	0.774
110	0.062	0.836
130	0.048	0.884
>140	0.116	1.00

Then,

$$\tau_{GF} = \tau_G. \quad (3.34)$$

The average fire duration (τ_{FD}) is evaluated in the example of Section 3.4.5 and it is presented in Table 3.5.

Now we have all the information necessary for Equation (3.30). The histogram of Table 3.7 is the result of applying the two histograms of Table 3.5 and 3.6 to that equation. It depicts our uncertainty about $\phi_{S/F}$, the fraction of fires that would cause a transient and fail the power cables to both charging pumps, both motor-driven auxiliary feedwater pumps, the control cables to all fan coolers, and two out of the three containment spray pump trains.

The availability of the remaining mitigating components due to human error (as a consequence to the control board response to the cable fire) is judgmentally assessed. We believe that ϕ_{HE} is log-normally distributed with a medium of 10^{-2} and an error factor of 10. The unavailability of the turbine-driven pump, ϕ_{TD} , is found in Reference 2. It is also lognormally distributed with a mean of 5.8×10^{-2} and variance of 7.3×10^{-5} . The frequency of cable spreading room fires, ϕ_p , is given in Table 3.2. We multiply these four histograms in two steps. First, we multiply the first three to get the conditional

**Table 3.7 Histogram for the Frequency
of Failing Part of a Sequence of Events
due to Fire in the Cable Spreading Rooms**

Frequency ($\phi_{S/F}$)	Probability	Cumulative Probability
$< 10^{-6}$	0.156	0.156
4.5×10^{-5}	0.102	0.258
1.9×10^{-3}	0.116	0.374
9.7×10^{-3}	0.094	0.468
5.9×10^{-2}	0.100	0.568
0.14	0.094	0.662
0.17	0.118	0.780
0.48	0.149	0.930
0.76	0.070	1.00

frequency of sequence occurrence given that a fire has occurred. The result is shown in Table 3.8. This is then multiplied with the frequency of fires and the unconditional frequency of sequence occurrence is obtained. The result is presented by Table 3.9.

**Table 3.8 Histogram for the Conditional Frequency
of (Containment Event Tree) Entry State G
Given a Fire in the Cable Spreading Rooms**

Frequency	Probability	Cumulative Probability
$< 10^{-10}$	0.050	0.050
3.4×10^{-10}	0.103	0.153
5.6×10^{-8}	0.115	0.268
1.2×10^{-6}	0.101	0.369
7.8×10^{-6}	0.119	0.488
2.8×10^{-5}	0.103	0.591
7.9×10^{-5}	0.107	0.698
1.8×10^{-4}	0.118	0.815
5.1×10^{-4}	0.112	0.927
2.3×10^{-3}	0.073	1.000

**Table 3.9 Histogram for the Frequency of State G
Due to a Fire in the Cable Spreading Rooms**

Frequency (ry ⁻¹)	Probability	Cumulative Probability
< 10^{-10}	0.160	0.160
4.3×10^{-10}	0.108	0.268
8.3×10^{-9}	0.109	0.377
4.5×10^{-8}	0.101	0.478
1.5×10^{-7}	0.111	0.589
4.6×10^{-7}	0.119	0.708
1.2×10^{-6}	0.107	0.815
3.5×10^{-6}	0.111	0.926
1.7×10^{-5}	0.074	1.00

4. CONCLUDING REMARKS

A method for assessing the frequency of radionuclide release from nuclear power plants due to fires is described in this report. Special attention is given to simplicity and expressing the uncertainties in all parts of the methodology. Similar to all other methodologies used in probabilistic risk assessment studies, we first identify a list of accident scenarios and then quantify them. A scenario in this case includes the location of the fire, the time history of the fire (i.e., growth, detection, and suppression), the components that the fire can affect, and other components such that their simultaneous failure would cause an initiating event and lead to radionuclide release.

In Section 1.2, we have summarized the methodology in six steps and we have further expanded them in Sections 2.3.1 and 3.1. The objective in defining these steps was to show how the different parts of the methodology relate to each other. Certainly, not all paths of interaction are identified. There is a potential for improving on our suggested steps and defining an algorithm of a more mechanistic nature. In this chapter, we first present our conclusions from this study and then give a general discussions about the methodology. The factors leading to conservatisms and

nonconservatisms, and areas for future work are highlighted.

The element of time is an important factor in several parts of this methodology. In one example where system failures were considered, the recirculation system is called upon several hours after fire ignition. Thus, it is very likely that the fire would be out by then and the system would be restored (if it is affected by the fire). In the quantification of fire-induced failures, time is an integral part of the model. The component failures and fire suppression are two competing, time-dependent phenomena. If the suppression time surpasses the failure time, then component failure is imminent.

A fire risk study is extremely plant specific. For example, in the cable spreading room of the Zion station, we concentrated on the exact location of certain cables (Figure 3.4). This plant specificity can be partially attributed to the fact that a fire risk study is a location-dependent event analysis. Although there is a large similarity among the same type of plants (e.g., PWRs) there are still large differences in plant layouts.

In the examples that we have discussed -- this study, human error plays an important role. In some scenarios, human actions at the component location are

crucial and in some other scenarios, some component failures are attributed to human error from the control room due to inaccurate information on the control board (caused by the fire).

In Chapter 2 we have used simplified event trees for event sequences and have analyzed the systems related to each event in terms of their major components. This is in contrast with other parts of probabilistic risk assessment studies where detailed event trees and fault trees are used. Several computer codes have been developed to analyze fault trees in the context of common cause events [50]. They stop short of identifying multiple common cause events and carrying the results further into the event tree sequences. A simple extrapolation of these algorithms can give us our desired results. However, the volume and time necessary to carry out this task may become limiting factors. We certainly need the detailed event trees and fault trees to clearly understand the workings of the power plant as a whole and the individual systems. However, the merits of directly using these trees in the fire risk analysis needs examination.

In identifying the scenarios, we inspected each compartment in the plant to see how a fire there could lead to radionuclide release. To limit the scope of the task, we chose for inspection those compartments

that contained safety-related items. The other areas were not inspected based on the premise that components necessary for safe shutdown are not affected. If this assumption is not valid, it certainly would introduce nonconservatism to the results. This point needs further investigation.

Another source of nonconservatism lies in the definition of a compartment. We did not investigate the possibility of failing the barriers that enclose a compartment. These barriers are penetration seals, walls, doors, etc.

There are several other sources of non-conservatism related to fire incidents that we have not included in this study. These are: smoke propagation, extremely large fires, secondary fires, flooding due to water type fire extinguishing systems, and fires due to earthquakes. The impact of smoke on the availability of equipment may not have short term impact. However, it can certainly reduce the effectiveness of the operators in fire fighting in manipulating components locally and, more importantly, in the control room.

The only source for extremely large fires is oil stored in large amounts. In most cases, the oil is located in areas where safety-related items are not installed. However, it may affect those items by

failing fire barriers, a subject that has not been addressed here. Another source is an aircraft crash or other large external fires.

The secondary fires may be caused by shorts in electrical circuits. For example, if two wires of a circuit short across a load, then the resulting high current may cause overheating of the other loads in other locations. Flooding can occur only due to water type suppression systems and inadequate drainage systems. Earthquakes may cause fires through electrical components or flammable liquid. This aspect of fire analysis should be performed within the context of an earthquake risk analysis.

The quantification process of a scenario is divided into three basic elements: the frequency of fire occurrence in a compartment, the conditional frequency of component failures given the fire, and the frequency of component failure due to causes other than the direct impact of the fire. The first frequency has been assessed, based on statistics on fire incidents, independently from the other two conditional frequencies. Furthermore, these frequencies of fire occurrence are derived from evidence collected from all United States nuclear power generating stations. They are average frequencies and do not necessarily reflect the conditions of a specific power plant. For example, it is

debatable whether the inclusion of some actual fire occurrences, e.g., the Browns Ferry fire, is appropriate when the cable spreading room of a specific plant is being analyzed.

The lack of a quantitative measure for the magnitude of fire is another limiting factor in any fire risk analysis. It has raised the most difficulties in the development of the quantification parts of this methodology. We have used qualitative arguments to relay our understanding about the spectrum of fires that are covered by the frequencies derived in Section 3.2. If such a measure could be defined, the exponential model that was used in Section 3.6 could be replaced with a more sophisticated model. Then, an integral equation could be defined wherein the above defined product of the three frequencies could be integrated over all fire severities.

The evaluation of the conditional frequency of component failures given that a fire has occurred involved fire propagation, detection, suppression, and component failure. Our treatment of these factors is very crude. We modeled them by their duration and combined these time periods in an exponential distribution to obtain the desired conditional frequency. It is not clear if the choice of an exponential distribution is conservative in this case. However, we took

However, we took other conservative measures such as the assumption of cable failure upon ignition. Certainly, further research is needed concerning all four factors (i.e., propagation, detection, etc.). More information about the first three (i.e., propagation, detection, and suppression) will allow us to better model the interaction between propagation and suppression.

The failure modes of some components may be different when exposed to fires than in normal usage. For relays and switchgear, their susceptibility and failure modes when exposed to fires is not clear. In some cases similar to cables, their failure mode may depend on their specific applications and circuit configuration. We have found it very important to clearly define the behavior of some of the components. For example, in the cable spreading room fire, we assume that the control board indicators misinform the operators because of cable failures. However, we do not elaborate on exactly how this misinformation is produced.

The examples used in this study have focused only on PWRs. There are differences among different facility types. However, we believe that the methodology has general applicability.

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APPENDIX A - FIRE INCIDENCE DATA

The details of the data used in different parts of this study are given in this appendix. The two main sources are References A.1 and A.2 which are discussed individually in the following sections. Both references offer more information than what is discussed here. In both cases, only the U.S. nuclear experience is collected.

A.1 Data from American Nuclear Insurers

The American Nuclear Insurers (ANI), in conjunction with the Rensselaer Polytechnic Institute, have organized a data file system where information about fire incidents in nuclear power plants is stored. Most of their data come from reports of insurance inspectors which contain location, cause, combustibles, means of detection, and extinguishment of fire. The financial loss and plant status are also reported. Their other sources are the U.S. Nuclear Regulatory Commission and local government regulatory bodies. The insurance reports contain more information because they also include less severe fires than the regulatory records.

The period of January 1955 through May 1978 is covered by the data. A total of 214 incidents in all types of facilities (including nonpower generating

facilities) are reported. Of these, 158 occurred in commercial nuclear power plants.

Our main interest is in fire incidents during commercial operation of light water reactors. Table 13 of Reference A.1 is very close to what we need. It gives statistics on different aspects of fire incidents during the operational phases of all types of nuclear facilities. Table A.2 shows one part of Table 13 of that report. Table A.1 provides the number of fires during different operational modes and facility types. The first column covers all facility types and comes from Table 13 of the ANI report. The next three columns come from Tables 14 and 15 of the same report. In these tables (similar to Table 13), fire incident statistics are given for the two types of LWRs, that is, pressurized water reactors and boiling water reactors. However, in both cases, all phases of plant life are covered.

The data may have some omissions. The authors of that report acknowledge this. We quote (page 8): "it must be indicated that the data ... contains certain errors or omissions."

In the following discussion, the data used in Section 3.2 is derived from Table A.2. All facility types are included in that table whereas, in Section 3.2, we present the results as though they are

Table A.1 - Number of Fires and Plant Status

	All Types of Facilities	Light Water Nuclear Generating Plants	Pressurized Water Reactors	Boiling Water Reactors
Normal Operation	42	40	21	19
Hot Shutdown	2	2	2	0
Cold Shutdown	1	0	0	0
Refueling or Extended Outage	3	1	0	1
TOTAL	48	43	23	20

**Table A.2 - Number of Fires and Location
of Fires in Nuclear Facilities of all Types**

Location	No. of Fires
Containment	1
Reactor Building	6
Auxiliary Building	10
Turbine Building	9
Diesel Generator Room	10
Control Room	1
Cable Spreading Room	2
Relay Room	2
Radwaste Building	1
Switch yard	1
Warehouses	1
Temporary Building	1
Yard	1
Outside Structure	1
Offsite	<u>4</u>
TOTAL	51

specific to LWRs. Our judgment is that this discrepancy does not have a large impact and, in all cases, it results in larger frequencies which are conservative. The following observations support this conclusion:

- The total number of fires in Table A.2 is 51 and in Table A.1, we find that the total number of LWR fires during the operational phase is 43. Thus, the difference is not large and most of the fires occurred at some LWR.
- Other types of power plants (high temperature gas-cooled reactors HTGRs and fast breeder reactors, PBRs) have a small contribution to the data. For all phases of plant life, there are two and four cases for HTGRs and PBRs, respectively.
- The areas for which frequencies are computed in Section 3.2 are typical of power plants.

The number of fires used in Section 3.2 is derived here by comparing those of Table A.2 with Tables 14 through 20 of the ANI report. Note that the latter tables cover all phases of plant life, that is, they include the construction phase.

Control Room - only one case is found in Table A.2. There are zero cases in all nonLWR facilities. Therefore, this one case definitely occurred in an LWR.

However, it is not clear which type of LWR it was because one is reported for each type (BWR and PWR) in Tables 14 and 15 of the ANI report.

Cable Spreading Room - two cases are found in Table A.2. Both occurred at LWRs because zero incidents are reported for all other facilities. One of the two occurred at a BWR. We believe that this one is the Browns Ferry fire incident of March 1975. The other occurred at a PWR.

Diesel Generator Room - Ten cases are found in Table A.2. There are zero cases in all nonLWR facilities. Six cases occurred at BWRs and four cases at PWRs. All LWR incidents are during commercial operation. This is expected because most of the diesel fires occur during testing [A.3] which is periodically performed during commercial operation.

Containment - Table A.2 gives one case in the containment and six cases in the reactor building. We judge that both mean the same thing, which is an area where the reactor and some equipment necessary for core heat removal during normal operation are located. Thus, we conclude that there were seven incidents in containments (or reactor buildings). Not all of them occurred at LWRs. The following table shows the number of incidents in different facility types at all phases of plant life.

NUMBER OF FIRES

Facility Type	Total All Phases	Total Commercial Operation	Containment	Reactor Bldg.
Educational/Research	27	---	0	2
BWR	62	20	4	7
PWR	96	23	10	10
HTGR	2	0	1	1
FBR	4	3	0	2
Other	23	---	---	---

For the educational or research facilities, the evidence is not conclusive because the number of fires during the operational phase is not given. For the HTGRs, we conclude that the two incidents did not occur during commercial operation. For the FBR, at least one of the two had occurred during this phase and there is a good chance that the other one was also during commercial operation. In light of the uncertainties about the two incidents in the educational facilities and the second incident in the PBR, we judge that two of the seven fires occurred in nonLWR plants. Thus, five fires have occurred in containments of LWRs.

Turbine Building - Table A.2 gives nine cases in the turbine buildings. There were no fires reported in

facilities other than the LWRs, except for one case in educational or research facilities. It is not clear at which stage (construction or operation) this fire occurred. Therefore, we use nine incidents in our frequency calculations.

Auxiliary Building - In Table A.2 we find ten fires reported in auxiliary buildings. There are no fires reported in auxiliary buildings at facilities other than LWRs. Thus, we use ten incidents in our frequency calculations.

A.2 Data from the HTGR Study

In a study of the risk to the public from an HTGR, General Atomic Company collected data on fire incidents in nuclear power plants and used it in assessing the contribution of these events to the overall risk. The data came from Reference A.4 and is tabulated in Reference A.2. We used their data in Section 3.2.5 to check how the fire occurrence rate varies with time.

The data covers all LWR incidents during commercial operation, except for hydrogen explosions in the off-gas systems. That is because these events are typical of LWRs and would not occur in the HTGRs. The time period that is covered by this data ends in May 1978. The beginning of this time period is not well defined in Reference A.2. There are no data points for the years before 1968. We judge that it starts with the inception of nuclear power production in the United States.

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APPENDIX B - CALCULATION OF COMPARTMENT YEARS

This appendix shows in detail how we obtain the number of years for the control rooms, the cable spreading rooms, the diesel generators, the containments, the turbine buildings, and auxiliary buildings in the light water nuclear generating plants of the United States that were operating commercially by the end of May 1978. The number of compartment years is defined as the time period between the first day of commercial operation and the end of May 1978 (or date of decommissioning). Table B.1 gives a detailed account of the compartment years and number of the above listed areas or compartments for every power plant. Reference B.1 gives the date of first commercial operation. The number of compartment types for each plant is found in its Final Safety Analysis Report (FSARs--see Reference B.2).

The available information is not complete. Therefore, the following assumptions are used during the construction of Table B.1.

- In Reference B.1, the date of commercial operation is given in terms of the month and the year; for example, 2/75 (i.e., February 1975). We assume that operation began on the last day of that month. Thus, the compartment years listed in Table B.1 are

smaller than the actual ones. The final result is slightly conservative because it increases the computed frequency of fires.

- In almost all multiunit plants, the operating time of the units is not the same. This is simply because they start commercial operation at different dates. For shared facilities (e.g., control room, turbine building) the largest number of years is assigned assuming that the facility was completed when the first unit went into commercial operation.
- Again, for multiunit plants, the age of the oldest unit is given to the first two diesel generator rooms.
- Only two plants, Indian Point Unit 1 and Humboldt Bay, were decommissioned before May 1978. In both cases, the date of decommissioning is assumed to be the last day of the last month in which they generated electricity. Gray Books [B.3] were used for this purpose.
- Plants other than LWRs are not included in the list. Consequently, the Shipping Port plant (a light water breeder reactor) and Hanford-N (a light water-cooled, graphite moderated reactor) are not in Table B.1.

- In cases of inadequate information, two diesel generators are assigned to the single unit plants.

References

- B.1 Nuclear News, 23, 80-85 (1980)
- B.2 Final Safety Analysis Report, a report filed at the U.S. Nuclear Regulatory Commission by utilities for each nuclear power plant.
- B.3 "Operating Units Status Report," NUREG-0020, U. S. Nuclear Regulatory Commission.

TABLE B.1. OPERATING YEARS OR SOME AREAS OF COMPARTMENTS IN LIGHT WATER REACTORS IN THE UNITED STATES AT THE END OF MAY, 1978.

PLANT NO.	PLANT NAME & UNIT NUMBER	DATE OF COMMERCIAL OPERATION (DECOMMISSION)	NUMBER AND OPERATING YEARS OF THE AREAS											
			CONTROL ROOM		CABLE SPREADING ROOM		DIESEL GENER.		CONTAINMENT		TURBINE BLDG.		AUXILIARY BLDG.	
			NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO
1	Beaver Valley 1	4/77	1	1-1	1	1-1	2	2x(1-1)	1	1-1	1	1-1	1	1-1
2	Big Rock Point	12/62	1	15-5	1	15-5	2	2x(15-5)	1	15-5	1	15-5	1	15-5
3	Browns Ferry 1	8/74	1	3-9	1	3-9	2	2x(3-9)	1	3-9	1	3-9	1	3-9
4	Browns Ferry 2	3/75	1	3-2	1	3-2	2	2x(3-2)	1	3-2	1	3-2	1	3-2
5	Browns Ferry 3	3/77	1	1-2	1	1-2	2	2x(1-2)	1	1-2	1	1-2	1	1-2
6	Brunswick 1	3/77	shares	w/#2	1	1-2	2	2x(1-2)	1	1-2	1	1-2	shares	w/#2
7	Brunswick 2	11/75	1	2-6	1	2-6	2	2x(2-6)	1	2-6	1	2-6	1	2-6
8	Calvert Cliffs 1	5/75	1	3-0	1	3-0	2	2x(3-0)	1	3-0	1	3-0	1	3-0
9	Calvert Cliffs 2	4/77	shares	w/#1	1	1-1	1	1-1	1	1-1	shares	w/#1	shares	w/#1
10	Cooper	7/74	1	3-10	1	3-10	2	2x(3-10)	1	3-10	1	3-10	1	3-10
11	Crystal River 3	3/77	1	1-2	1	1-2	2	2x(1-2)	1	1-2	1	1-2	1	1-2
12	Davis Besse	11/77	1	0-6	1	0-6	2	2x(0-6)	1	0-6	1	0-6	1	0-6
13	Donald C. Cook	8/75	1	2-9	1	2-9	2	2x(2-9)	1	2-9	1	2-9	1	2-9
14	Dresden 1	8/60	1	17-9	1	17-9	2	2x(17-9)	1	17-9	1	17-9	1	17-9
15	Dresden 2	8/70	1	7-9	1	7-9	2	2x(7-9)	1	7-9	1	7-9	1	7-9
16	Dresden 3	10/71	shares	w/#2	shares	w/#2	1	6-7	1	6-7	shares	w/#2	shares	w/#2
(Subtotal)				63-10		66-1		137-8		72-8		65-0		63-10

TABLE B.1 (CONTINUED)

PLANT NO.	PLANT NAME & UNIT NUMBER	DATE OF COMMERC'L (DECOMMISSION)	NUMBER AND OPERATING YEARS OF THE AREAS											
			CONTROL ROOM		CABLE SPREADING ROOM		DIESEL GENER.		CONTAINMENT		TURBINE BLDG.		AUXILIARY BLDG.	
NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	
17	Duane Arnold	5/74	1	4-0	1	4-0	2	2x(4-0)	1	4-0	1	4-0	1	4-0
18	Edwin I. Hatch 1	12/75	1	2-5	1	2-5	2	2x(2-5)	1	2-5	1	2-5	1	2-5
19	Fort Calhoun	6/73	1	4-11	1	4-11	2	2x(4-11)	1	4-11	1	4-11	1	4-11
20	Haddam Neck	1/68	1	10-4	1	10-4	2	2x(10-4)	1	10-4	1	10-4	1	10-4
21	Humboldt Bay 3	8/63 (7/76)	1	12-11	1	12-11	2	2x(12x11)	1	12-11	1	12-11	1	12-11
22	Indian Point 1	10/62 (10/74)	1	12-0	1	12-0	2	2x(12-0)	1	12-0	1	12-0	1	12-0
23	Indian Point 2	7/74	1	3-10	1	3-10	1	3-10	1	3-10	1	3-10	1	3-10
24	Indian Point 3	8/76	1	1-9	1	1-9	3	1x(1-9)	1	1-9	1	1-9	1	1-9
25	James Fitzpatrick	7/75	1	2-10	1	2-10	2	2x(2-10)	1	2-10	1	2-10	1	2-10
26	Joseph H. Forney 1	12/77	1	0-5	1	0-5	2	2x(0-5)	1	0-5	1	0-5	1	0-5
27	Keweenaw	6/76	1	3-11	1	3-11	2	2x(3-11)	1	3-11	1	3-11	1	3-11
28	La Crosse BWR	11/69	1	8-6	1	8-6	2	2x(8-6)	1	8-6	1	8-6	1	8-6
29	Maine Yankee	12/72	1	5-5	1	5-5	2	2x(5-5)	1	5-5	1	5-5	1	5-5
30	Millstone 1	12/70	1	7-5	1	7-5	2	2x(7-5)	1	7-5	1	7-5	1	7-5
31	Millstone 2	12/75	shares	w/#1	1	2-5	shares	w/#1	1	2-5	shares	w/#1	1	2-5
32	Monticello	7/71	1	6-10	1	6-10	2	2x(6-10)	1	6-10	1	6-10	1	6-10
(Subtotal)			87-6		89-11		172-11		89-11		87-6		89-11	

TABLE B.1 (CONTINUED)

PLANT NO.	PLANT NAME & UNIT NUMBER	DATE OF COMMERCIAL OPERATION (DECOMMISSION)	NUMBER AND OPERATING YEARS OF THE AREAS											
			CONTROL ROOM		CABLE SPREADING ROOM		DIESEL GENER.		CONTAINMENT		TURBINE BLDG.		AUXILIARY BLDG.	
			NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO
33	Mine Mile Point 1	12/69	1	8-5	1	8-5	2	2x(8-5)	1	8-5	1	8-5	1	8-5
34	Nuclear One 1	12/74	1	3-5	1	3-5	2	2x(3-5)	1	3-5	1	3-5	1	3-5
35	Oconee 1	7/73	1	4-10	1	4-10	0		1	4-10	1	4-10	1	4-10
36	Oconee 2	9/74	shares	w/#1	1	3-8	0		1	3-8	shares	w/#1	shares	w/#1
37	Oconee 3	12/74	1	3-5	1	3-5	0		1	3-5	shares	w/#1	1	3-5
39	Oyster Creek 1	12/69	1	8-5	1	8-5	2	2x(8-5)	1	8-5	1	8-5	1	8-5
40	Pallisades	12/71	1	6-5	1	6-5	2	2x(6-5)	1	6-5	1	6-5	1	6-5
40	Peach Bottom 2	7/74	1	3-10	1	3-10	2	2x(3-10)	1	3-10	1	3-10	1	3-10
41	Peach Bottom 3	12/74	shares	w/#2	shares	w/#2	2	2x(3-5)	1	3-5	shares	w/#2	1	3-5
42	Pilgrim 1	12/72	1	5-5	1	5-5	2	2x(5-5)	1	5-5	1	5-5	1	5-5
43	Point Beach 1	12/70	1	7-5	1	7-5	2	2x(7-5)	1	7-5	1	7-5	1	7-5
44	Point Beach 2	10/72	shares	w/#1	shares	w/#1	shares	w/#1	1	5-7	shares	w/#1	shares	w/#1
45	Prairie Is. 1	12/73	1	4-5	1	4-5	2	2x(4-5)	1	4-5	1	4-5	1	4-5
46	Prairie Is. 2	12/74	shares	w/#1	shares	w/#1	shares	w/#1	1	3-5	shares	w/#2	1	3-5
47	Quad - Cities 1	8/72	1	5-9	1	5-9	2	2x(5-9)	1	5-9	1	5-9	1	5-9
(Subtotal)			61 - 4		65-5		'13-10		77-10		58-4		68-7	

TABLE B.1 (CONTINUED)

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PLANT NO.	PLANT NAME & UNIT NUMBER	DATE OF COMMERC'L (DECOM-MISSION)	NUMBER AND OPERATING YEARS OF THE AREAS											
			CONTROL ROOM		CABLE SPREADING ROOM		DIESEL GENER.		CONTAINMENT		TURBINE BLDG.		AUXILIARY BLDG.	
			NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO
48	Quad - Cities 2	10/72	shares	w/#1	shares	w/#1	shares	w/#1	1	5-7	shares	w/#1	shares	w/#1
49	Rancho Seco	4/75	1	3-11	1	3-11	2	2x(3-11)	1	3-11	1	3-11	1	3-11
50	Robert E. Ginna	3/70	1	8-2	1	8-2	2	2x(8-2)	1	8-2	1	8-2	1	8-2
51	Robinson 2	3/71	1	7-2	1	7-2	2	2x(7-2)	1	7-2	1	7-2	1	7-2
52	Salem 1	12/76	1	1-5	1	1-5	2	2x(1-5)	1	1-5	1	1-5	1	1-5
53	San Onofre 1	1/68	1	10-4	1	10-4	2	2x(10-4)	1	10-4	1	10-4	1	10-4
54	St. Lucie 1	12/76	1	1-5	1	1-5	2	2x(1-5)	1	1-5	1	1-5	1	1-5
55	Surry 1	12/72	1	5-5	1	5-5	2	2x(5-5)	1	5-5	1	5-5	1	5-5
56	Surry 2	5/73	shares	w/#1	shares	w/#1	1	5-0	1	5-0	shares	w/#1	shares	w/#1
57	Three Mile Is. 1	9/74	1	3-8	1	3-8	2	2x(3-8)	1	3-8	1	3-8	1	3-8
58	Trojan	5/76	1	2-0	1	2-0	2	2x(2-0)	1	2-0	1	2-0	1	2-0
59	Turkey Point 3	12/72	1	5-5	1	5-5	2	2x(5-5)	1	5-5	1	5-5	1	5-5
60	Turkey Point 4	9/73	shares	w/#3	shares	w/#3	shares	w/#3	1	4-8	1	4-8		w/#3
61	Vermont Yankee	11/72	1	5-6	1	5-6	2	2x(5-6)	1	5-6	1	5-6	1	5-6
(Subtotal)				54-5		54-5		118-10		69-8		59-1		54-5

TABLE B.1 (CONTINUED)

PLANT NO.	PLANT NAME & UNIT NUMBER	DATE OF COMMERC'L (DEC'M-MISSION)	NUMBER AND OPERATING YEARS OF THE AREAS											
			CONTROL ROOM		CABLE SPREADING ROOM		DIESEL GENER.		CONTAINMENT		TURBINE BLDG.		AUXILIARY BLDG.	
			NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO	NO.	OPERATING YEARS YR - MO
62	Yankee	6/61	1	16-11	1	16-11	2	2x(16-11)	1	16-11	1	16-11	1	16-11
63	Zion 1	6/73	1	4-11	1	4-11	3	3x(4-11)	1	4-11	1	4-11	1	4-11
64	Zion 2	12/73	shares	w/61	1	4-5	2	2x(4-5)	1	4-5	1	4-5	1	4-5
(Subtotal)			21-10		26-3		57-5		26-3		26-3		26-3	
From previous pages			63-10		66-1		137-8		72-8		65-0		63-10	
			87-6		89-11		172-11		89-11		87-6		89-11	
			61-4		65-5		113-10		77-10		58-4		68-7	
			54-5		54-5		113-10		69-8		59-1		54-5	
TOTAL			288-5		301-3		593-0		337		295-2		802-3	

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APPENDIX C - DATA ON DETECTION TIME FOR DIFFERENT FIRE
DETECTORS

C.1 Smoke Detectors

Three sources are used in this appendix for deriving the probability distribution for the response time of smoke detectors. These are References C.1, C.2, and C.3. We quote from the first one (page 46): "Response time of the detectors under ... test conditions have varied from 16 to 55 seconds for ionization detectors, and 91 to 310 seconds for photoelectric detectors." The test conditions mentioned here simulated a residential compartment.

In Reference C.2, we find response times for different detectors of both heat and smoke-sensitive types. An average-size residential home was used in their tests. The detectors came from different manufacturers. The test fires covered a large gamut of severity. The results show that, in about 70 percent of the cases, the detectors responded within the first 5 minutes, and in about 15 percent they responded in more than 15 minutes. In about 10 percent of the cases, the detectors did not respond at all. We believe that, because of low ceiling heights, these test results are underestimates for nuclear power plant compartment conditions.

Reference C.3 gives data on test fires in hospital rooms. Also in these tests, in about 70 percent of the cases, the detectors responded within 5 minutes. However, most of the responses (about 50 percent) were in the first 1 minute. In about 25 percent of cases, the detectors did not respond at all. This shows that compartment related parameters have significant impact on detector performance.

C.2 Heat Detectors

In the study of Reference C.2, heat sensing devices were also used. The response times were much longer than those of smoke detectors. In about 30 percent of the cases, the heat detectors responded within 15 minutes; and in about 70 percent of the cases, they did not respond at all.

In Reference C.1, we find 84 to 144 seconds as the typical response times for the rate of rise heat detectors. These results correspond to a residential compartment.

Reference C.4 reports on experiments with high challenge fires. The tests simulated large fires in large compartments such as warehouses and varied the height of the ceiling and type of combustibles. One of the main observations was that the sprinkler heads (in almost all cases) opened after the flames reached the ceiling. The time for the first sprinkler head opening

has been measured. It varies between 50 seconds to 10 minutes.

C.3 Nuclear Plant Experience

In the data given in the HTGR fire risk study [C.5], a column titled "Interval Prior to Detection" is shown where only five cases are given. These cases are described below:

- (1) Detection time 20 minutes; mode of detection manual. An electrical insulation fire with electrical origin had occurred in some cabinets at Peach Bottom Unit 3, April 1977. Fire was put under control in 6 minutes.
- (2) Detection time 30 minutes; mode of detection manual. It occurred on May 1976 in Browns Ferry Unit 1 and combustible solids were ignited. Inplant fire fighters put out this fire in 20 minutes.
- (3) Detection time 1 hour; detection mode automatic. This is the charcoal adsorber explosion incident of July 1977 at Browns Ferry Unit 3 [C.6]. Fire was burning inside the charcoal adsorber beds for 2 hours until it was self-extinguished.

- (4) Detection time 2 hours; detection mode manual. Electrical insulation caught fire due to electrical causes in July 1972 at Quad Cities Unit 2. Fire self-extinguished in a few minutes after discovery.
- (5) Detection time 3 hours; detection mode manual. Expansion joints caught fire at Robinson Unit 2 in April 1974. Fire was extinguished in 15 minutes by the in plant fire fighters.

References

- C.1 Jensen, "Talking Fire Protection Systems," Fire Journal, January 1975.
- C.2 "Home Fire Alarm Tests," Fire Journal, July 1971.
- C.3 Waterman, T. E., J. G. Degenkolb, and C. W. Stickney, "A Study - Early Warning Fire Detection Performance in the Hospital Patient Room," Fire Technology 10, No. 4, (1974).
- C.4 Troup, E. W. J., "The High-Challenge Fireball - A Trademark of the Seventies," Fire Tech, Vol. 6, No. 3, August 1970.

C.5 Fleming, K. N., W. T. Houghton, and F. P. Scaletta,
"A Methodology for Risk Assessment of Major
Fires and its Application to an HTGR Plant,"
GA-A15402, General Atomic Company (1979).

C.6 Mays, G. T., "Browns Ferry Charcoal Adsorber
Incident," Nuclear Safety, Vol. 20, No. 1,
pp. 78-82, January-February 1979.

APPENDIX D DATA ON FIRE SUPPRESSION

There are very few sources that have investigated the suppression time for fires. In most cases, the overall effectiveness of the extinguishing system has been questioned. In the following subsections, we discuss the data given by these few references.

D.1 Data from the HTGR Study

In Section A.2 of Appendix A, we discussed the fire data used in the HTGR Study by General Atomic Company [D.1]. The data includes a column entitled "Time to Bring Fire Under Control." Based on the numbers of that column, we have derived the two frequency distributions of Table D.1; one uses all the cases that occurred when under commercial operation and the other uses those cases that involved electrical insulation (also under commercial operation.) We believe that the numbers in that column represent the time between fire detection and fire growth inhibition. It is after the detection time because the adjacent column is entitled "Interval Prior to Detection" and, in some cases, it contains numbers larger than the previous column. Thus, the distributions of Table D.1 depict a period that is certainly smaller than fire duration. Also, it should be noted that the time periods given in Reference D.1 are, in most cases, estimates by experts and not a result of actual time measurements during the fire incident.

TABLE D.1 FREQUENCY DISTRIBUTION
 OF "TIME TO BRING FIRE UNDER CONTROL"
 FROM REFERENCE D.1 FOR THE COMMERCIAL OPERATION PHASE

Time	All Cases		Cases Where Electrical Insulation was Involved	
	Frequency (%)	Cumulative Frequency (%)	Frequency (%)	Cumulative Frequency (%)
~ 0	10	10	12	12
3 min	8	18	24	36
6 min	25	43	12	48
12 min	17	60	6	54
18 min	9	69	6	75
30 min	11	80	23	77
42 min	4	84	6	83
1 hour	4	88	6	89
1-3 hours	4	92	6	95
1-5 hours	2	94		
2 hours	2	96		
7 hours	2	98	5	100
24 hours	2	100		

D.2 Fire Suppression Test Results

Reference D.2 presents the results of several tests that were conducted in two small compartments to measure the minimum water requirements for suppression. They allowed the fire to reach flashover and attacked it about 30 to 120 seconds after that. They concluded that the amount of water used strongly depends on the type of furnishings involved in the fire and the techniques used by the fire fighters. They have tabulated the flow rate of the water, total water used to control the fire, and also total water used to complete extinguishment. By complete extinguishment, they meant suppression of all visible flames (smoldering may still have been in progress). From their data, we have computed the time to bring the fire under control, which varies from between 15 seconds to 2 minutes. Also, we have computed the time to total extinguishment and this varies from between 1/2 to 4 minutes. The following observations are in order:

- (1) Even though the tests were in a controlled environment and the variations in the methods used were limited, the suppression time has a significantly large variation (about one order of magnitude).

(2) The time periods are short because adequate extinguishing equipment was available at all times of the experiment.

References

- D.1 Fleming, K. N., W. T. Houghton, and F. P. Scaletta,
"A Methodology for Risk Assessment of Major
Fires and its Application to an HTGR Plant,"
GA-A15402, General Atomic Company (1979).
- D.2 Salzberg, F., F. J. Vodvar, and G. L. Maatman,
"Minimum Water Requirements for Suppression of
Room Fires," Fire Tech., Vol. 6, No. 1, pp. 22-28,
February 1970.

APPENDIX E DATA ON CABLE FAILURE TIME

E.1 Introduction

The data on failure time comes from sources that have conducted cable fire tests. The conditions of these tests in some instances have been quite different. Therefore, we first discuss their history very briefly (Section E.2) and then summarize the relevant results (Section E.3).

E.2 Fire Tests

Many fire tests have been developed in recent years by various organizations to measure the different aspects of a fire incident. Most of these measure only one phenomenon, e.g., ignition temperature, flammability, fire spread, etc. Reference E.1 provides a review of these tests and classifies them as those that test for: (a) ease of ignition, (b) surface flame spread, (c) heat release, (d) fire endurance, (e) smoke evolution, and (f) combustion products. Most of these tests are designed for small samples. In many instances, it has been proven that fuel geometry, quantity, and ambient conditions have severe effects on the fire test results [E.2, E.3]. Full scale tests have been devised also to observe these effects. Reference E.1 lists facilities that perform such tests. It has been found that reproducing the same test result is very difficult. This is attributed to the large number of

parameters that have significant impact on the time history of a fire. For example, the ignition process is only partly controlled by the fuel area being exposed to heat, uniformity of exposure, drafts in the room, duration of exposure, and the heating rate.

The same types of problems exist in cable fire tests. In recent years some tests have been specialized for cables only. The nuclear industry has been the main motive behind these developments.

Reference E.4 summarizes some of these tests. Bench-type fire tests using short lengths of single cables have been used to determine the flammability of the cables. Also, large scale tests have been devised to better simulate real fire conditions and overcome the erroneous conclusions stemming from the small scale tests.

Independent tests have been carried out by the different factions of the industry: the manufacturers [E.5], the utilities [E.6], and the regulatory bodies [E.7]. From these tests, standards have emerged [E.4]. The most important standard for cable testing is the IEEE Std 383-1974 [E.8] where a vertical set of cables is exposed to a 70,000 Btu/hr heat source with a flame temperature of 1500°F. Reference E.4 describes the activities that helped to develop this IEEE standard.

By reviewing these tests, their results, and some relevant literature, we have learned that:

- (1) Grouped cables pose a special fire hazard [E.9,E.10].
- (2) The position of the cables is important. Cables in vertical trays pose a larger hazard for propagation [E.11].
- (3) Tray construction plays a role too. In totally closed trays (top and bottom closed) and conduits, circuit failure occurs sooner than in cables in open-type trays. Although cable ignition is delayed, combustion is more severe [E.4]. This is because the closed tray or conduit acts like an oven.
- (4) The density of cables (number of cables per unit width of tray) in horizontal, open bottom trays affects their flammability. This has to do with volatile gases being able to pass through the trays and burn at upper layers.
- (5) Deep seated fires may occur [E.12,E.13]. These are hard to put out. This is especially true for nonwater extinguishing agents [E.14].

(6) The heat release rate of the donor fire (the ignition source) may make a difference.

Reference E.13 quotes experiments where a 70,000 Btu/hr donor fire did not reproduce the real incident of fire propagation, whereas a 210,000 Btu/hr fire did so.

(7) Corrosion effects may also become important.

In a cable fire incident described in Reference E.13, the corrosive gases given off by the cable insulation or jacket (occurs at high temperature) corroded electrical contacts and relays some 40 feet away.

(8) Finally, the currently existing standards and test procedures still cannot accurately determine the degree of fire retardancy of the cables.

E.3 Data

In this section, we summarize the results available to us. Most of the tests referenced here are designed for investigating flammability. However, what we are interested in is the time to circuit failure. Fortunately, many experimenters have recognized the need for this piece of information and have recorded it in addition to flammability-related information. In the main text, Section 3.5.2.2, the time to cable

failure is defined as the time between flames engulfing the cable and its failure (the failure modes are discussed separately). In all the tests, the cables were immersed into flames. Also, all the conductors were subjected to voltage and circuit integrity was checked with some light bulbs. Failure time was measured from the ignition of the donor flames until the light went out.

The Boston Insulated Wire Company has tested its own cables according to different standards and procedures [E.5]. In Reference E.5, seven types of cables are tested by exposing single cables to a Bunsen burner for 5 minutes with a flame temperature of about 1600°F. Two of the cables shorted in 2 and 6 minutes; the others experienced extensive damage but did not short out. In another test, oil-soaked burlap was burned next to several runs of the same cable on a vertical tray. PVC insulated cables shorted in 3-1/2 and 4-1/4 minutes. Bostrad 7 (a brand name) cable did not short for the 19-1/2 minutes that the flames existed. Flame temperature is unknown. In controlled bonfire tests, a bundle of cables is suspended over a bonfire for 5 minutes. Cross linked, polyethylene EPR nylon jacket and Bostrad 7 shorted in 1-1/2 to 3 minutes. Bostrad 7S (a brand name) did not

short in these 5 minutes. Radiation effects are also investigated in this reference.

Reference E.6 summarizes the results of the tests that were performed by Baltimore Gas and Electric Company to choose appropriate cables for their Calvert Cliffs Nuclear Power Plant. In their tests, the cable trays were loaded with one layer of test cables, allowing 1/2 diameter of space between cables, and no more than 15 cables per tray. Transil oil in a 5 gallon can was burned for 5 minutes. Flame temperature was about 1800°F. Forty-five types of control cables and twelve types of power cables were tested in this way. The majority of the cables shorted after the oil can was removed. The results are shown in terms of the following histogram:

<u>Time to Short (Minutes)</u>	<u>Fraction Percent</u>	<u>Cumulative Fraction</u>
1-5	14	14
5-10	46	60
10-15	26	86
15-20	3	89
> 20	11	100

From these tests, it was concluded that silicon rubber-insulated control cables with glass-braid fillers and asbestos-braid jackets have better flame

retardance characteristics than the others. Six such cables were subjected to this test and only one failed (at about 15 minutes). Also, two other control cables did not fail in 20 minutes. Both were silicone insulated and the jackets were of PVC (poly vinyl chloride) and Neoprene.

Okonite Company has performed a series of cable fire tests that helped the development of the IEEE Std. 383-1974 [E.8]. They are discussed in Reference E.4. Only control cables were subjected to those tests. Some of their results are given in the following:

- Neoprene cables in a vertical tray are subjected to burning burlap soaked in oil. It burned for about 20 minutes. The flame temperature was about 1200°F and shorts occurred between 7 and 16 minutes.
- The same neoprene cables in a vertical tray are subjected to burning propane gas. They burned the propane for 20 minutes. The flame temperature was about 1500°F and shorts occurred between 6 and 8 minutes.
- A set of control cables was subjected to two levels of burner heat release rates (70,000 Btu/hr and 210,000 Btu/hr). The cables shorted between 10 and 21 minutes. For the

second case (210,000 Btu/hr) they shorted between 6 and 11 minutes.

- Similar to Reference E.6, they found that silicone insulated cables with glass braid jacket do not short. However, the jacket was charred and the core damaged. They had applied the flames for 20 minutes.

The scientists at the Sandia Laboratories have conducted some cable fire tests to investigate the effects of fire-retardant coatings [E.15]. These are special compounds that are sprayed (or applied by trowels) on top of the cables. These tests are part of a larger study entitled "Fire Protection Research Program." Five different coatings were used on a single horizontal cable tray. Propane burners were installed underneath the tray. It was applied in cycles of 10 minutes, 5 minutes on and 5 minutes off, for six times. In seven out of ten tests, the cables had not shorted at the end of the sixth cycle. In the remaining three cases, they shorted between 15 and 26 minutes. In the same setup, they made three additional tests without applying any coating. In two cases the cables had passed the IEEE 383 test and shorted in 5 and 9 minutes. In the third case, the cable had not passed the IEEE test and shorted in 6 minutes.

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16. ABSTRACT (200 words or less) A methodology for evaluating the frequency of severe consequences due to fires in nuclear power plants is presented. The methodology produces a list of accident scenarios and then assesses the frequency of occurrence of each. Its framework is given in six steps. In the first two steps, the accident scenarios are identified qualitatively and the potential of fires to cause initiating events is investigated. The last four steps are aimed at quantification. The frequency of fires is obtained for different compartments in nuclear power plants using Bayesian techniques. The results are compared with those of classical methods and the variation of the frequencies with time is also examined. The combined effects of fire growth, detection, and suppression on component failure are modeled. The susceptibility of cables to fire and their failure modes are discussed. Finally, the limitations of the methodology and suggestions for further research are given.		
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