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A METHODOLOGY FOR RISK ASSESSMENT OF MAJOR FIRES AND ITS APPLICATION TO AN HTGR PLANT

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ABSTRACT

A risk assessment methodology is presented for a specific class of reactor accidents, major fires. Principal elements of the methodology include collection and interpretation of experience data, identification and screening of important fire locations, event tree/fault tree analysis, and accident consequence assessment. About 900 reactor-years experience was examined for reported fires either during construction or plant operation. Statistical analyses were performed of fire initiation and progression characteristics such as occurrence rate, distance of fire spread, and time to bring fire under control.

The methodology and experience data were applied to a specific HTGR design and plant layout. Bounding calculations were performed to eliminate all but five locations in the plant from consideration as potentially important fire locations. Qualitative analysis similar to failure modes and effects analysis (FMEA) were performed on each of the five candidate fire locations to determine that two of these had a relatively high potential for producing dominant risk accident sequences. One of these, the cable spreading room, was selected for quantitative event tree/fault tree analysis. A fire propagation model was developed to interface the experience data with quantification of the fault trees. The probability of a cable spreading room fire leading to core heatup was found to be about 1×10^{-5}/reactor-year, or about a factor of 3 less than that determined for all other initiating events analyzed previously for the high-temperature gas-cooled reactor (HTGR). At lower levels of accident probability, 1×10^{-7}/reactor-year and less, fires were found to be a dominant cause of failure of multiple, diverse systems. Hence, the results of this study have made an important contribution to the treatment of common cause failures in the HTGR risk assessment study.

PREFACE

Accident Initiation and Progression Analysis (AIPA) is an ongoing probabilistic risk assessment study of the safety of high-temperature gas-cooled reactors (HTGRs). A total of nine volumes of reports have been issued from 1974 through 1978 (see list on page ii) including the report GA-A15000 (Ref. 1), in which a comprehensive assessment of the risks to public health and safety from hypothetical HTGR accidents is presented. The term "comprehensive" is used to mean that, in the judgment of the authors, the AIPA study had progressed to the point where scientifically based conclusions could be made. In addition, it was felt that valid, order-of-magnitude comparisons could be made against the results of the Reactor Safety Study (Ref. 2) as may be required to support the licensing of future HTGRs.

The term "comprehensive" should not be confused with "complete," since there appears to be a consensus that there is no way to assure completeness in the risk assessment of low probability events such as reactor accidents (Ref. 3). Among the major areas of the risk assessment methodology (shown in Fig. 1-1) that are subject to errors of omission on the part of the assessors are those associated with the identification of accident sequences. These include the selection of initiating events and the construction of event trees. If the relevant operating experience is carefully taken into account, and if the event trees and list of initiating events are scrutinized via peer review, reasonable assurance can be given that dominant risk accident sequences are not overlooked. However in the case of extremely low accident probabilities, the unquantified element of uncertainty associated with missed accident sequences may be significant. In the judgment of the AIPA study team, there is reasonable assurance that dominant risk accident sequences in the HTGR having probabilities greater than about 10^{-7}/reactor-year have not been overlooked. For this reason, summary results of the health effects and property damage impact of HTGR accidents are not emphasized for accident probabilities less than 10^{-7}/reactor-year even though numerical calculations are made and presented as low as 10^{-9}/reactor-year.

Because of the completeness issue discussed above, the AIPA study team has taken the position that a risk assessment study of rare events cannot be considered final. This is one reason why the study has continued into its fifth year, and it is felt that there will be a continuing need to update the predictions as more operating experience is accumulated and as further methodological advances are made. In this report, the HTGR risk assessment is extended to include a class of initiating events, major fires, that was not accounted for previously in the AIPA study.

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1. INTRODUCTION

1.1. IMPORTANCE OF FIRES AS INITIATING EVENTS

The selection of fires as initiating events is an example of the need to update the risk assessment results in light of new operating experience as it is accumulated. An important incident in the U.S. reactor operating experience was the cable spreading room fire at Brown's Ferry in 1975. Although the fire resulted in no release of radioactivity or irreparable plant damage, the impact on redundant safety systems was extensive. In the context of risk assessment methodology development, the Brown's Ferry fire underscored the importance of the treatment of common cause failures* from the standpoint of both delineation of accident sequences and estimation of their probabilities.

The overall approach to the treatment of common cause failures in the AIPA risk assessment methodology is outlined in Table 1-1. Methods 1 through 4 are associated with the identification of specific causes of multiple component and multiple system failure, whereas method 5, the Beta Factor method (Ref. 4) is a general reliability assessment procedure for common cause failures in redundant systems that does not require the enumeration of specific causes. If a specifically identified cause has the potential for resulting in the failure of multiple systems important to safety, it becomes a candidate for selection as an initiating event. At the onset of the AIPA study, 17 initiating events were selected as a representative set for preliminary event tree/fault tree analysis (Ref. 5). In the Phase II study (Ref. 1) about 20 new events were added to give a

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*The term "common cause failure" is used to describe multiple failures of two or more components, subsystems, or systems caused by a single common event.

TABLE 1-1

TREATMENT OF COMMON CAUSE FAILURES IN AIPA RISK ASSESSMENT METHODOLOGY

Step in Risk Assessment Methodology	Methods for Treatment of Common Cause Failures	Examples of Common Cause Failures Treated Using Indicated Method
Initiating event selection	1. Consider events with potential to fail multiple systems and fission product barriers	Fires, earthquakes, loss of electric power
Event tree construction	2. Identify interdependencies among system failures appearing in event tree	System A fails as direct consequence of failure of system B.
Event tree quantification/ fault tree analysis	3. Consider each branch point probability in event tree as conditional on preceding steps in sequence	System A and system B both depend on common support system
	4. Identify specific causes of multiple failures within systems	Component X fails as direct consequence of failure of component Y
	5. Consider multiple failures in redundant systems always dependent, quantify coupling factor (β) for each component using same experience data used to estimate failure rates	Redundant set of components left out of service due to improper test, or share same design error

more complete coverage of core heatup accidents. Of all the initiating events that have been analyzed, at least three qualify as examples of common cause failures that can affect multiple systems important to safety. These are earthquakes, loss of all electric power, and pressure vessel failures that exceed the design bases of the reactor systems important to safety.

Although fires were not selected in the original list of initiating events, it was recognized following the Brown's Ferry fire that these events could not be ruled out as potentially important accident initiators. Hence, a risk assessment study of fires was undertaken. The purpose of this report is to present the results of this study.

The general features of the AIPA risk assessment methodology are applicable to any selected initiating event including fires. Some detailed aspects of the methodology, however, had to be developed especially for fires, including those regarding the analysis and interpretation of experience data, the screening of the plant layout to identify the most important fire locations, and the manner in which the fault trees were developed. It was felt that the methodology development itself was a useful product of the study and it is therefore presented separately in Section 2. The data involving fires that were compiled and analyzed are also presented separately (Section 3), since they may be of use in other studies.

In Sections 4 through 6, the methodology is applied to an HTGR steam-cycle plant of the same design that was analyzed in the comprehensive HTGR risk assessment study reported in GA-A15000 (Ref. 1). Details of this design of relevance to the initiation and progression of fires are described in Section 4. This particular design was selected because of a desire to make direct comparisons against the results of GA-A15000 and because of the availability of design details about the balance of plant and the layout of piping, cable trays, and plant components. Because of their dominance across most of the accident spectrum as determined in prior studies, accident sequences resulting in core heatup were emphasized. An important result obtained and described in Section 6 is that sequences initiated by fires tend to dominate the overall risk assessment at very low probabilities, below 10^{-7}/reactor-year. The dominance of fire-initiated sequences at low probabilities arises from the potential for the same fire to cause simultaneous failures of redundant and diverse systems needed to cool the reactor core and close the containment isolation valves. Some thoughts on how the results might have been different if the more stringent fire protection criteria of today had been in effect are also given in Section 6.

1.2. LITERATURE SURVEY

Since many aspects of the initiation and progression of fires are generic to nuclear plants and not peculiar to HTGRs, it was felt that much could be learned from a survey of the technical literature relevant to fires in nuclear power plants regardless of reactor type. The information obtained in this survey was useful in the development of the methodology and served as a basis of comparison for checking the results. The most important body of information obtained from the literature, the data base used to quantify the rate of occurrence and propagation characteristics of fires, is presented separately in Section 3. Some other useful aspects of the relevant scientific literature are briefly summarized in the following paragraphs.

As mentioned previously, the most serious fire that has occurred in nuclear power plants prior to the Three Mile Island accident is the electrical cable fire at the Brown's Ferry plant on March 27, 1975. It is not surprising, therefore, that much of the useful scientific literature relevant to predicting the risks of nuclear power plant fires was published after this event. The sequence of events at Brown's Ferry was described by Green (Ref. 6) and by a special NRC review group chaired by Hanauer which also made recommendations as to how to reduce the probability and consequences of such a fire (Ref. 7). These recommendations were later factored into Regulatory Guide 1.120 (draft) on fire protection.

An estimate of the probability that the Brown's Ferry fire could have progressed to the point of core melt was presented in the Reactor Safety Study (Ref. 2) final report in response to comments made on the draft version (Ref. 8). In consideration of the specific impact that fire had on the boiling water reactor (BWR) systems at Brown's Ferry, the conditional probability that the fire would have proceeded to core melt was estimated by fault tree analysis to be 3×10^{-3}. When combined with the observed frequency of such fires, the rate of occurrence of core melt accidents initiated by fires of the Brown's Ferry type was set at 1×10^{-5}/reactor-year. This result was then compared with the frequency of core melt summed over all other initiators and averaged over a mix of pressurized water reactors (PWRs) and BWRs, 5×10^{-5}/reactor-year.

The Clinch River Breeder Reactor (CRBR) Risk Assessment Study (Ref. 9) included fires in its list of accident initiators having some potential for resulting in core melt. A systematic procedure was followed to determine, qualitatively, the locations of interest for postulating the occurrence of fires. Failure modes and effects analyses were then employed to evaluate each candidate fire location with respect to the likelihood of causing damage to certain safety-related systems such as the heat removal systems and the reactor shutdown systems. In addition to the types of fires that could occur in any industrial plant, e.g., fires in cable trays, fuel oil tanks, etc., sodium fires and damage from the sodium-water reaction were considered. Fault trees and event trees were used to define representative fire sequences leading to core melt, which were then quantified. The probability of occurrence of a cable spreading room fire leading to core melt was estimated at 5 x 10^{-7}/reactor-year (upper bound). Although this result is more than a factor of 10 below that assessed in Ref. 8 for Brown's Ferry, the CRBR plant was designed to more stringent separation and fire protection criteria and utilizes redundant cable spreading rooms. Certain aspects of the methodology developed for use in this study, which is presented in Section 2, are similar to the approach followed in the CRBR study.

Kazarians and Apostalakis (Refs. 10, 11) extended the probabilistic assessment of the Brown's Ferry fire that was initiated in Ref. 8. A survey was conducted of abnormal occurrences at nuclear power plants and, on the basis of these data, estimates were made of the frequency of various types of fires. The rate of occurrence of a fire of any (reported) severity during commercial plant operation was observed to be 0.12/reactor-year based on 130 reactor-years of data from January 1, 1970 to March 30, 1975. This result is in excellent agreement with an estimate of the fire frequency obtained in this study using a larger data base (see Section 3 for details). An upper bound of the fire frequency on the cable spreading room was assessed at 0.05/reactor-year. Some aspects of fire progression, detection, and suppression were investigated. Two important conclusions are drawn from probabilistic analyses of the Brown's Ferry fire. First, the estimate of the probability of core melt during the fire is shown to be quite sensitive to assumptions regarding the repair model for the equipment damaged by the fire. Second, the risk of core melt could have been reduced if the confusion about the use of water on the fire had been eliminated, i.e., if water had been applied earlier to put out the fire.

Additional reports and papers provided useful information in more specialized areas of study, and these are called out at appropriate places in the text.

2. METHODOLOGY

2.1. OVERVIEW

One of the continuing objectives of the AIPA study has been to develop and refine the methodology for probabilistic risk assessment of nuclear reactor safety. Although the initial AIPA methodology development was carried out beginning in 1974 when the Reactor Safety Study (RSS) (Ref. 2) was in progress, its general aspects are similar to that employed in the RSS. Some of the major differences are described in Sections 2.4 and 2.5. The AIPA methodology, initially described in Ref. 12 and refined in Ref. 1, is depicted in Fig. 2-1 as it applies to any accident initiator (termed initiating event), including major fires. As explained below, certain aspects of the methodology were modified and specialized in this study in consideration of some of the unique characteristics of fires. The roles of the specialized and general elements of the methodology are presented in Fig. 2-2.

One characteristic of fires that is important in methodology development is that they can occur at practically any location in the plant, including any with combustible material of either a permanent or transient nature. Fire locations can be categorized in terms of the probability of initiation and the potential for propagation into a fire of major proportions by identifying the inventory of combustibles, ignition sources, and other factors influencing fire initiation and progression. However, for a fire of any size to be of importance to public health and safety, it must result in, or increase the probability of, the release of radioactivity to the environment. Hence, only those fires that damage or degrade radionuclide barriers or systems such as core cooling systems required to maintain barrier integrity are of interest.

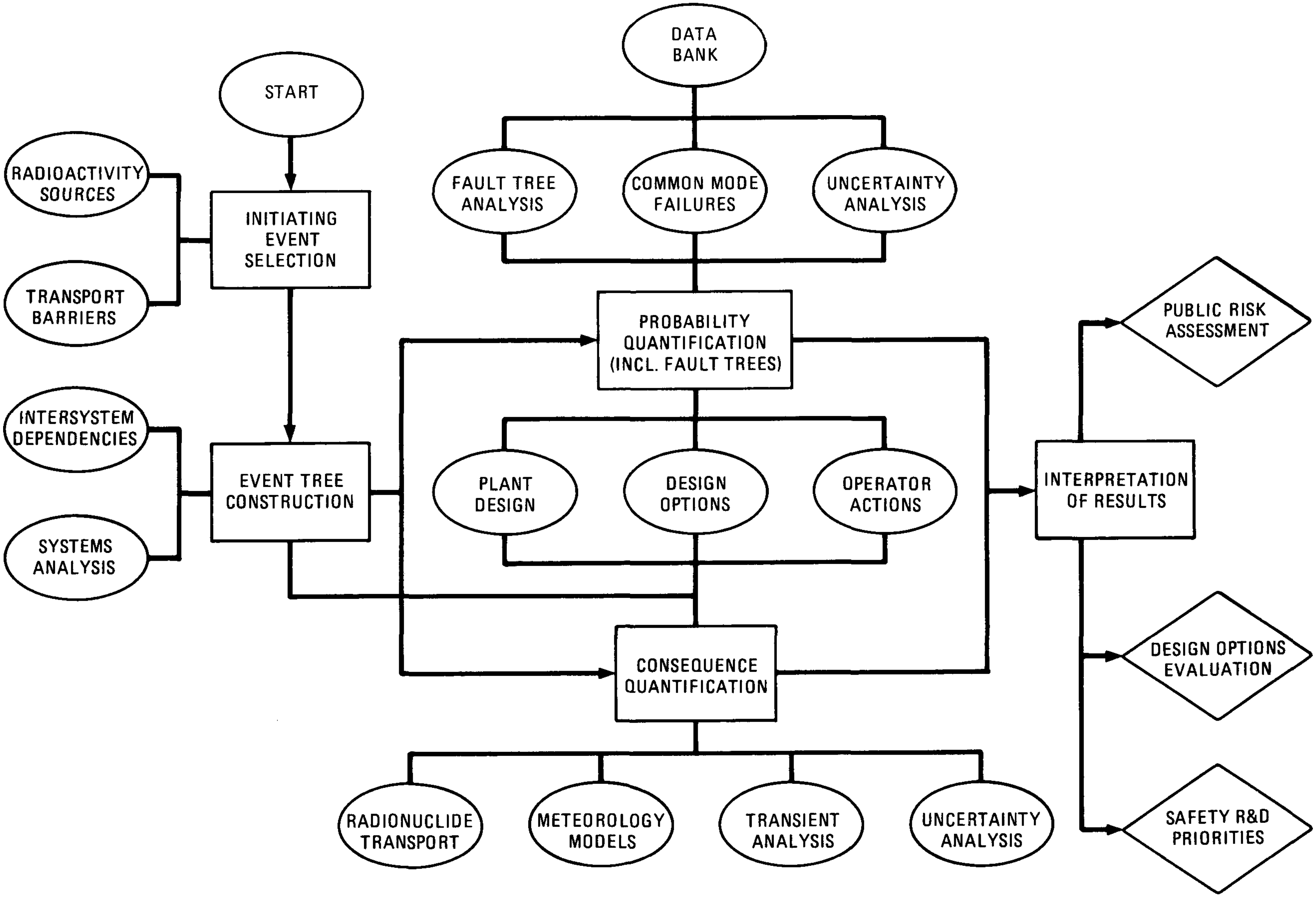


Fig. 2-1. Major elements of AIPA risk assessment methodology

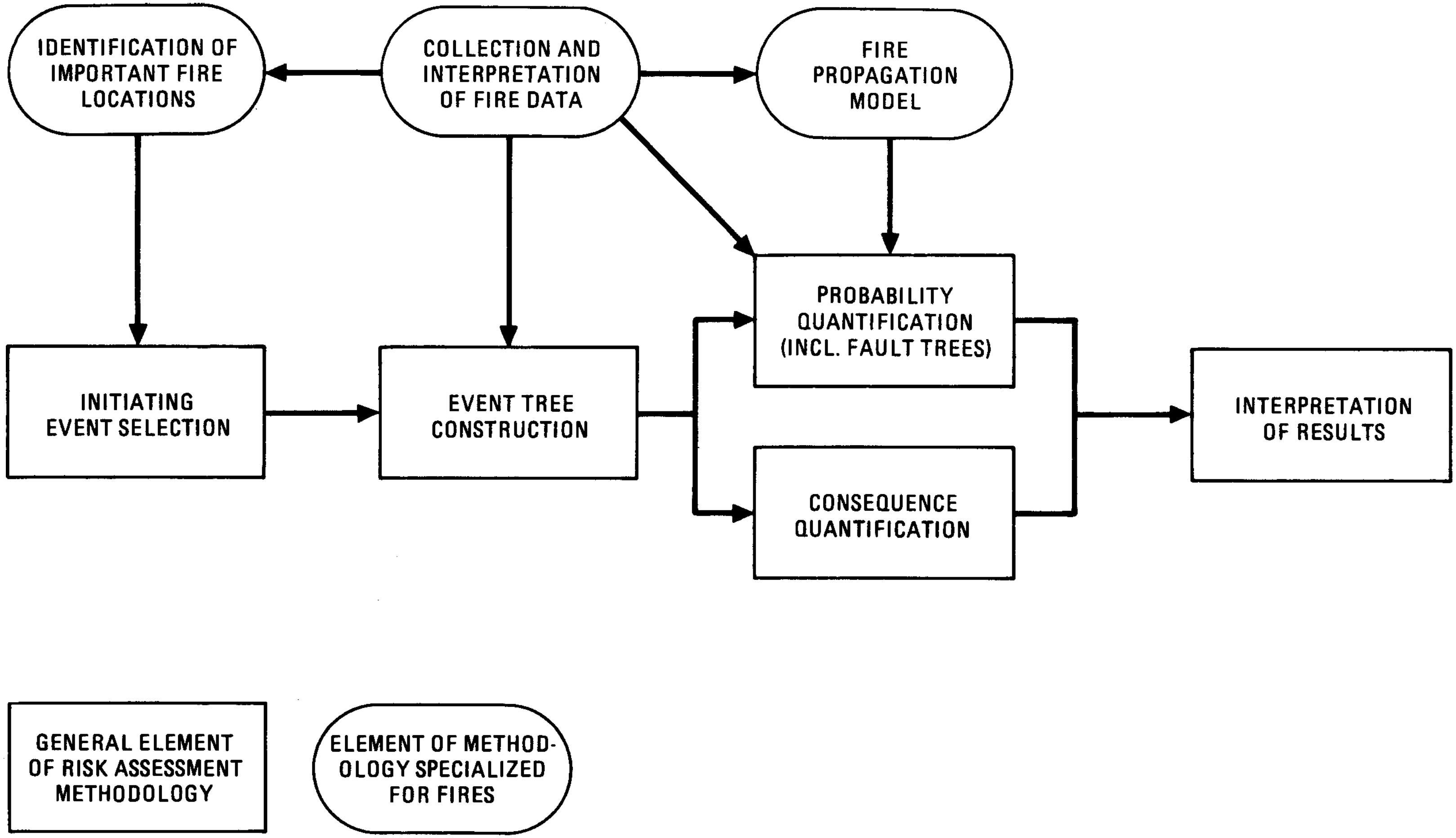


Fig. 2-2. Specialized elements of methodology for risk assessment of fires

Unfortunately, a large number of plant locations typically remain after rejection of those that do not contain components or systems related to radionuclide barriers. One of the first problems encountered in adapting the risk assessment methodology to fires is how to give adequate coverage to all potentially important locations without requiring an exorbitant amount of resource-consuming quantitative analysis. This problem was solved with the development of special methods, described in Section 2.3, for screening the plant layout and evaluating the importance of fire locations. This process results in a reduction in the number of candidate locations to a manageable number for event tree/fault tree analysis.

Unlike some of the initiating events analyzed in risk assessment studies with which there is little or no experience, there has been a fair amount of experience with nuclear power plant fires. In addition to the direct use of these experience data in the quantification of occurrence rates and probabilities, insights are obtained which are useful in the identification of important fire locations as well as the delineation of accident sequences in the event trees. Therefore, the collection and interpretation of the experience data involving fires form an integral part of the methodology, as discussed in greater detail in Section 2.2.

In any risk assessment study that seeks to estimate the risks of rare events, it is necessary to utilize models that in effect extend the data base, i.e., enable the estimation of probabilities that are well below that which can be set directly from experience data. This is precisely what is accomplished in fault tree analysis in that a logic model is used to express rare events in terms of combinations of subordinate events whose probabilities are estimated, ideally, from the data base. In addition to fault trees, it has been necessary to utilize a fire propagation model as a means of considering important characteristics of the fire not directly quantifiable from the data base. A fire propagation model developed especially for fires in a cable spreading room is described in Section 5.3.

The overall methodology for the risk assessment of fires, including the specialized and general elements, is organized into four major phases, as shown in Table 2-1. Details regarding each phase are presented below.

2.2. DATA COLLECTION AND INTERPRETATION

The objectives of this phase of the methodology are to obtain data for use in quantification of the event trees and information useful in their construction. The first step in obtaining this information is to review reports of individual incidents involving fires and to prepare brief summaries with the following information:

Plant identification data (name, location, reactor type).

Dates of fire, first electrical generation, and start of construction.

Incident report references, docket numbers, etc.

Location of fire initiation.

Cause.

Time to put out fire, and distance of fire propagation.

Impact of fire on plant systems.

Response of fire detection and suppression systems and fire-fighting activities.

Other factors relevant to initiation and progression of fire.

TABLE 2-1

ELEMENTS OF AIPA RISK ASSESSMENT METHODOLOGY SPECIALIZED FOR FIRES

Study Phase	Major Elements
I. Data collection and interpretation	Quantify fire occurrence rates. Define probability distributions for fire progression characteristics (e.g., distance, time). Obtain insights for event tree construction.
II. Identification of important locations	Examine plant layout for location of key systems and components. Employ bounding calculations to eliminate unimportant fire locations. Evaluate candidate locations and select key locations for quantitative risk analysis.
III. Event tree/fault tree analysis	Construct event trees and expand detail of existing fault trees. Employ fire propagation model. Utilize fire data and results of prior fault tree analyses in quantification of new trees.
IV. Consequence and risk assessment	Estimate consequences and uncertainties based on comparison with prior work. Construct plots of accident occurrence rate versus consequences and CCDF. Determine risk impact of fire. Recommend design changes to improve fire protection if needed.

Once the above information has been recorded for each fire incident, it is useful to organize the incidents and relevant characteristics into a single table to facilitate further data reduction. Actual data summaries using both methods of recording are given in Section 3, together with statistical analyses for use in quantification of the event trees.

2.3. IDENTIFICATION OF IMPORTANT LOCATIONS

After the data involving fires have been collected and interpreted, it is necessary to select important fire locations within the power plant under investigation having the greatest potential for producing dominant risk accident sequences. Once selected, these locations are analyzed quantitatively in the event tree/fault tree analysis phase. The objectives of the location selection phase are somewhat competing and should be balanced in a meaningful risk assessment study. The first objective is to minimize the possibility that an important location might be overlooked, and this leads to the consideration of a potentially large number of candidate locations. The second objective is to minimize the effort spent in the quantification of event trees and fault trees for fire locations that turn out to be unimportant. A proper balance of these objectives is one that results in an ideal allocation of resources and efficiency of the assessment.

For a fire location to meet the first objective, it is necessary that the probability of initiation and potential for serious fire progression be great enough to produce relatively high risk accident sequences. Some insight as to which locations possesses these characteristics may be obtained from the data base. Unfortunately, the data base alone may provide insufficient information on which to judge the importance of locations, for several reasons. First, the plants from which the data were obtained may be significantly different from the one under investigation. In addition, the data base may be too small to be the sole basis for rejecting certain locations that may produce relatively low probability but high risk accident sequences. Finally, many, if not most, of the fires in the data base, despite their occurrence, are of little importance to public health and safety, since they have no impact on fission product barriers or plant systems important to the retention of radioactivity.

To supplement the information obtained from the data base and to provide a systematic procedure for evaluating plant locations, a qualitative analysis method was developed, which is referred to as "Fire Location and Progression Analysis (FLPA)". The FLPA is a specialization of the "Failure Modes and Effects Analysis (FMEA) (Ref. 13) and consists of an evaluation of each candidate location in terms of the characteristics listed in Table 2-2. An example FLPA is given in Table 2-3. The main advantage of this approach is that it provides a systematic means of interpreting the data base in light of the particular design features and layout of the plant being analyzed. The disadvantages are that it may be somewhat tedious to apply, especially if the number of candidate locations is large, and that the ultimate judgment of importance is subjective. The application of FLPA to candidate fire locations in an HTGR plant is presented in Section 4.2

The number of candidate locations to be analyzed and compared in terms of FLPA can be reduced by taking into account quantitative information about dominant risk accident sequences associated with initiating events other than fires. Historically, risk assessments of fires have been preceded by those of other initiating events for HTGRs (Ref. 1) and LWRs (Ref. 8) in the U.S. As shown below, information about the dominant accident sequences not involving fires, together with some bounding calculations using the data base, can be used to reject many locations from consideration as candidates for important fire locations.

Suppose that risk information exists about the dominant accident sequences not involving fires. For simplicity, assume that the dominant sequences can be grouped into three release categories, 1, 2, and 3, having total occurrence rates λ_1, λ_2, and λ_3 and consequences C_1, C_2, and C_3, where

$$C_1 > C_2 > C_3 \quad . \tag{2-1}$$

TABLE 2-2
EVALUATION CHARACTERISTICS IN FIRE LOCATION AND PROGRESSION ANALYSIS (FLPA)

Inventory of combustible material

Inventory of major components, including pipes and cables

Failure modes of equipment caused by fire

Immediate systems effects

Fire barriers/adjacent fire cells

Fire detection and protection equipment

Fireman (brigade) access

Ventilation/smoke exhaust pathways

Likelihood of initiation and significant progression (qualitative)

TABLE 2-3

FLPA OF PENETRATION BUILDING ELECTRICAL CHASE

Room or Cell Identification: Penetration Building Electrical Chase (III)

Location: Penetration Building

Combustible Material	Major Components and Cables	Failure Modes	Immediate System Effects	Fire Barrier(s)/ Adjacent Cells	Automatic Fire Detection and Protection System	Fireman Access	Ventilation/ Smoke Exhaust	Remarks On Probability of Significant Progression
Cable insulation, packing material where cables are routed through two end walls to Service Building	Instrumentation and control cables, Divs. I-VII; 480-V power cables, Divs. II, III, and VII	Short to ground and open (for control and instrumentation	Trip of control using inhibit logic; fail to trip for circuits using transmission logic	Chase walls, ceiling, floor (concrete). Barriers not shown in dwgs. No fire stops to compartmentize chase shown.	Smoke and excess temperature detectors give indication in control room.	Hatches or doors at various points.	Three vent supply fans provided, manually controlled.	Horizontal runs and layout mean that probability of significant progression is less than in vertical chases.
Transient combustibles	No 4160-V cables Three vent supply fans (manually controlled)	Short to power (for control and instrumentation)	Fail to trip for inhibit logic; inadvertent operation of controls using transmission logic (see cable spreading FLPA for examples).					
		Short to ground and open (for power cable, Div. II, III	Loss of power to 480-V ESMCC and switchgear. Examples: auxiliary motor cooling water pump; RPCWS fan, CACS cooling fan; gen. H_2 seal oil pressure; CB purge sup. fan; CR sup. fan; DG start A/C.					
		Short to ground and open (for power, Div. VII)	Loss of power to 480-V BOP MCC and switchgear. Examples: RSB crane; transfer distr. panel (emerg.); He purif. train A; gen. stator cooling pump 1A; CB recirc. unit 1A; serv A/C; nitrogen recondenser.					

The first step in the bounding procedure is to determine which plant systems must fail to produce the consequences in each release category. For simplicity, suppose that the plant consists of three systems, A, B, and C, and that the relations between consequences and failed systems are given by

$$C_1 = ABC \quad , \tag{2-2}$$

$$C_2 = AB \quad , \tag{2-3}$$

$$C_3 = A \quad , \tag{2-4}$$

where ABC denotes that all three systems are failed, etc.

Consider that a fire occurs at a specific location in the plant, and assume that the fire has the maximum extent of system damage that could occur at that location. In this context, the boundaries of the location, sometimes referred to as a fire cell, coincide with major fire barriers such as concrete walls, fire doors, concrete floors, and other barriers that would be expected to contain the fire. By examination of the plant systems layout, the maximum extent of systems impact that could be caused by this fire is determined, and this impact is denoted by j. The impact might be, for example, failure of one or more of the components or subsystems in A, B, or C, or it might not involve these systems at all. Using the existing fault trees, assumed to be available for systems A, B, and C, one can estimate the conditional probability of having an accident sequence that results in each of the three release categories given that the fire has the specific impact j described above; i.e.,

$$\Pr\{C_1|j\} = \Pr\{ABC|j\} \quad , \tag{2-5}$$

$$\Pr\{C_2|j\} = \Pr\{AB|j\} \quad , \tag{2-6}$$

$$\Pr\{C_3|j\} = \Pr\{A|j\} \quad , \tag{2-7}$$

where $\Pr\{ABC|j\}$ = conditional probability that systems A, B, and C fail (thereby giving rise to consequences C_1) given that a fire occurs with specified impact j.

Now, for the postulated fire having impact j to produce an accident with a risk comparable to those analyzed previously, it is necessary that

$$\lambda_{f\to j\to i} = {}_{f\to j}\,\Pr\{C_i|j\} \geq \lambda_i \qquad i = 1, 2, 3 \quad , \tag{2-8}$$

where $\lambda_{f\to j\to i}$ ≡ occurrence rate for accident sequences initiated by fires having impact j and resulting in consequences C_i,

$\lambda_{f\to j}$ ≡ occurrence rate for fires having impact j,

and λ_i is as before. Hence, the condition on the fire occurrence rate becomes

$$\lambda_{f\to j} \geq \frac{\lambda_i}{\Pr\{C_i|j\}} \quad . \tag{2-9}$$

Although the value of $\lambda_{f\to j}$ is unknown in the absence of a quantitative fault tree analysis or its equivalent, it may be possible to establish upper bounds using the information in the data base. If it can be shown that condition 2-9 is not met with the use of bounding calculations, then the location in question can be rejected from consideration as an important risk contributor. It will be shown in Section 4 that all but 5 locations in a particular HTGR plant layout can be eliminated from consideration as important fire locations using this type of bounding procedure.

2.4. EVENT TREE/FAULT TREE ANALYSIS

The general aspects of the event tree/fault tree analysis methodology for accident sequence delineation and probability quantification are the same for fires as for other initiating events. The methodology developed and used in the AIPA study is presented in Ref. 12 and examples of its application to other initiating events can be found in Refs. 1 and 14. The most significant difference between the event tree/fault tree methods used in the AIPA study and the Reactor Safety Study (RSS) lies in the treatment of unidentified causes of multiple dependent failures, commonly referred to as common cause failures. The method developed and used in the AIPA study for the treatment of unidentified causes of multiple failures, called the Beta Factor method, is described and compared to other methods in Ref. 15. Specific causes of multiple failure that are identified explicitly are analyzed in the usual manner using standard fault tree analysis techniques (Ref. 16).

A category of specific causes of multiple failures that deserves special treatment includes fires that can directly result in failure of multiple systems and large numbers of components. Other examples include earthquakes and loss of all electric power. Since these causes of common cause failure can themselves have a great influence on the progression of an accident sequence, they are analyzed as initiating events. Once an initiating event is selected, the event tree/fault tree analysis proceeds in the usual manner. Care must be taken to insure that multiple system failures that occur along the accident sequences are treated as dependent events coupled by the fire. The procedure used here to model this type of dependency is to link together all the system failures that occur along each accident sequence with an AND gate. The same procedure was used in the RSS for treating the dependence of multiple systems on a common event or subsystem. Application of this procedure is described in Section 5.3.

Another special consideration in the event tree/fault tree analysis of fires is the effect of fire-fighting activities on efforts to safely shut down the plant. In the case of cable fires, it may be difficult to determine the status of major plant systems until the fire is put out or brought under control. Also, additional damage may be caused by water or other substances used to put out the fire. Examples of these special considerations are discussed in Section 5.

Depending on the particular fire location under investigation and the quality of information that can be obtained from the data base, it may be necessary to utilize a fire propagation model. In general, the fault trees will contain minimal cut sets S_j of the form

$$S_j = \{X_1, X_2, \ldots, X_M, Y_1, Y_2, \ldots Y_N\} \quad , \tag{2-10}$$

where X_i, $i = 1,2,\ldots,M$ are the component failures (basic events) caused by the fire and Y_k, $k = 1,2,\ldots,N$ are the component failures (basic events) independent of the fire.

In the process of fault tree quantification, terms of the form $\Pr\{Y_1 Y_2 \ldots Y_N\}$ will appear which require knowledge of the probability that the fire causes failure of specific sets of components. Since it is unlikely that such specific details can be derived directly from the data base, it may be convenient to express the fire damage probabilities in terms of something more basic, such as the distance of fire progression. The approach taken in this study is to write

$$\Pr\{Y_1 Y_2 \ldots Y_N\} = \int_0^L f(\ell) \Pr\{Y_1 Y_2 \ldots Y_N | \ell\} \, d\ell \quad , \tag{2-11}$$

where $f(\ell)\,d\ell$ is the probability that the fire progresses to distance in the interval $[\ell, \ell + d\ell]$, and $\Pr\{Y_1 Y_2 \ldots Y_N | \ell\}$ is the conditional probability that a specific set of N components is failed by a fire given that it progresses to a distance ℓ.

It may be possible to obtain the density function $f(\ell)$ from the data base and to develop simple models for $\Pr\{Y_1 Y_2 \ldots Y_N | \ell\}$ that enable the integration of Eq. 2-11. Such an approach was followed in the analysis of the cable spreading room fire in Section 4 for which the fire propagation model in Section 5 was derived.

2.5. CONSEQUENCE AND OVERALL RISK ASSESSMENT

Apart from the possibility that accident sequences initiated by fires might have some unique characteristics not shared by other initiating events, the methodology for the assessment of the consequences of fire sequences and for combining these with the probability assessments to obtain an overall risk assessment is the same as that employed for any other initiating event. The key elements of consequence assessment are the simulation of the physical processes of the accident relevant to the integrity of fission product barriers, the prediction of radionuclide transport through these barriers, and the resulting consequences to health and safety of the public as well as property damage impact. Detailed comparisons given in Ref. 1 show that the radionuclide transport and consequence models used in the AIPA methodology give results comparable to those employed in the RSS when the same radionuclide source term for environmental release is specified. The physical processes of accidents, on the other hand, are highly dependent on the unique characteristics of the reactor. Hence, accident simulation models were developed specifically for HTGRs, and these are described in Refs. 1, 17, and 18.

The consequences of a wide spectrum of initiating events and accident sequences have been analyzed extensively in the AIPA study for the HTGR. The accident sequences that have been found to result in the greatest risk as well as the greatest potential for consequences relative to those analyzed involve overheating of the reactor core. A comprehensive assessment of core heatup accidents, analyzed in terms of 20 initiating events and about 150 accident sequences, is presented in Ref. 1, the ninth volume of the AIPA report series. From this set of sequences, 22 were selected for detailed simulation of physical processes to obtain representative core heatup scenarios. A final grouping of sequences in terms of their scenarios into six release categories was then made based on similarity in the estimate of the radiological consequences measured in rem, which was made using detailed computer simulations for each scenario. A simplified analytical model was then constructed for each release category, which included the most sensitive independent variables (e.g., containment building leak rate, weather conditions) and was normalized to match the results of the detailed computer simulations. These simplified models enabled the quantification of uncertainties in the consequence assessment using a Monte Carlo simulation procedure (Ref. 19) and evaluated for a large number of trials (several thousand). The output of this procedure is a complementary cumulative distribution function (CCDF) of the consequences of reactor accidents.

The number of different sequences, scenarios, and release categories was expanded in the risk assessment of alternative HTGR containment designs, as described in Ref. 20. This information, together with that in Ref. 1, was used as a basis for the assessment of the consequences of accidents initiated by fires, as described in Section 5.

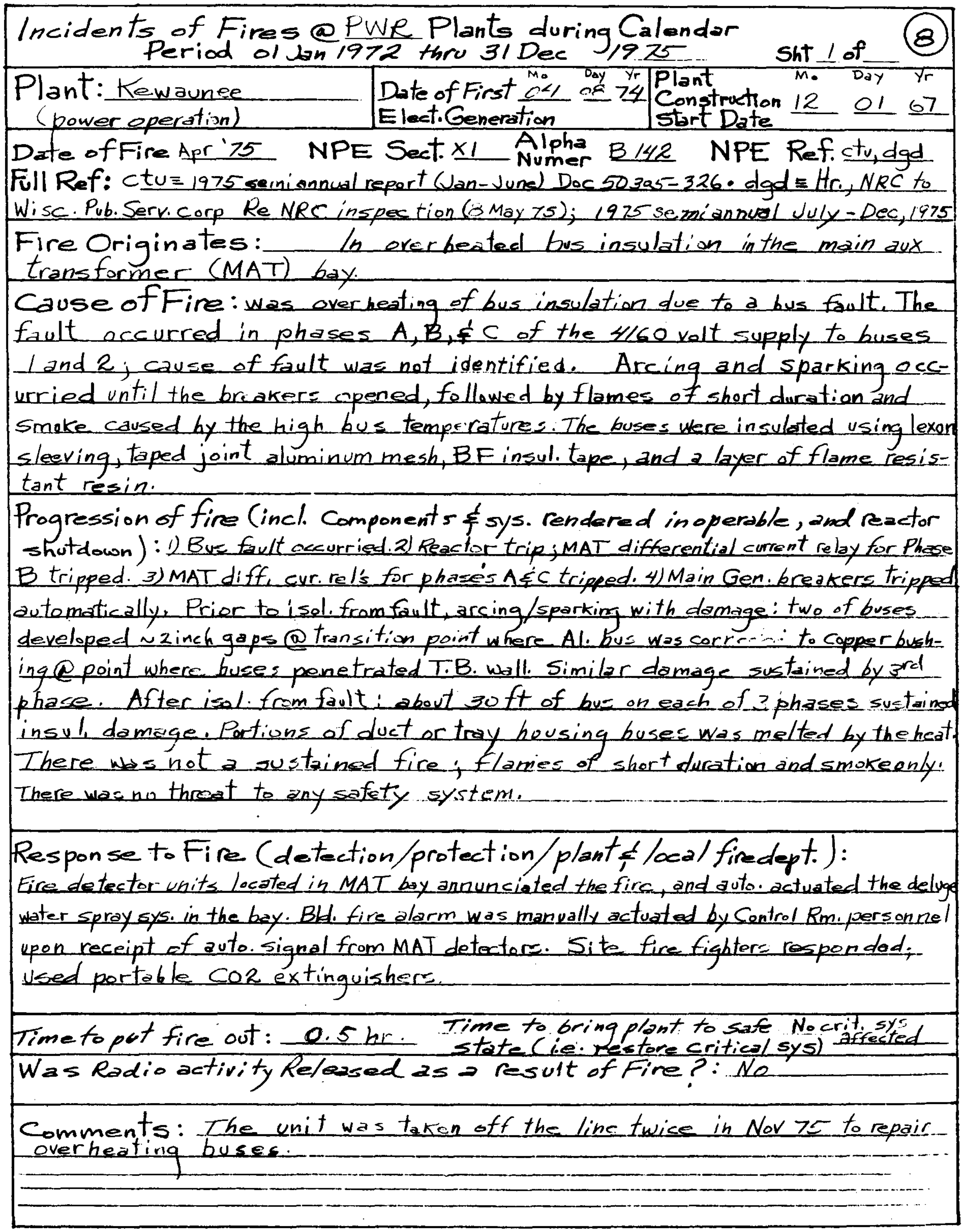
3. EXPERIENCE DATA WITH NUCLEAR PLANT FIRES

As explained in Section 2.2, the first step in applying the risk assessment methodology for fires to an actual plant is the collection and interpretation of relevant data involving fires. The data base was restricted to commercial U.S. nuclear plant experience up to the time when this phase of the study was carried out, May 1978. Among the most prominent reasons for this restriction is the relative difficulty in obtaining consistently reported events in non-nuclear plants and nuclear plants abroad. However, some extremely severe non-nuclear plant fires were included to help put the severity of the nuclear plant fires into perspective.

Although most of the U. S. experience is with PWRs and BWRs, nearly all of the experience is applicable to the HTGR because of the similarities in inventories of combustibles, fire detection and suppression systems, and separation and fire protection criteria. One related categroy of events that was excluded from the data base because of inherent differences between HTGRs and LWRs is hydrogen explosions in the off-gas systems due, in part, to radiolysis of primary coolant which does not occur with helium.

3.1. INCIDENT CATEGORIZATION

The basic sources of U.S. nuclear plant information used for obtaining fire data were the summaries of abnormal occurrences compiled by Verna (Ref. 21) and the U.S. Nuclear Regulatory Commission (Refs. 22 and 23). The first step was to compile one-page summaries of each incident involving fires in the format of the example in Fig. 3-1. The second step was to compile the information in a single expanded table, which is presented in Table 3-1. The latter table was used to help categorize the fires with



Incidents of Fires @ PWR Plants during Calendar Period 01 Jan 1972 thru 31 Dec 1975 Sht 1 of ⑧

Plant: Kewaunee (power operation)	Date of First Elect. Generation Mo 04 Day 08 Yr 74	Plant Construction Start Date Mo 12 Day 01 Yr 67

Date of Fire Apr '75 NPE Sect. XI Alpha Numer B142 NPE Ref. ctu, dgd

Full Ref: ctu = 1975 semiannual report (Jan-June) Doc 50305-326. dgd = Ltr., NRC to Wisc. Pub. Serv. corp Re NRC inspection (8 May 75); 1975 semiannual July-Dec, 1975

Fire Originates: In overheated bus insulation in the main aux transformer (MAT) bay.

Cause of Fire: was overheating of bus insulation due to a bus fault. The fault occurred in phases A, B, & C of the 4160 volt supply to buses 1 and 2; cause of fault was not identified. Arcing and sparking occurried until the breakers opened, followed by flames of short duration and smoke caused by the high bus temperatures. The buses were insulated using lexon sleeving, taped joint aluminum mesh, BF insul. tape, and a layer of flame resistant resin.

Progression of fire (incl. Components & sys. rendered inoperable, and reactor shutdown): 1) Bus fault occurried. 2) Reactor trip; MAT differential current relay for Phase B tripped. 3) MAT diff. cur. rel's for phases A & C tripped. 4) Main Gen. breakers tripped automatically. Prior to isol. from fault, arcing/sparking with damage: two of buses developed ~2 inch gaps @ transition point where Al. bus was corr[illegible] to copper bushing @ point where buses penetrated T.B. wall. Similar damage sustained by 3rd phase. After isol. from fault: about 30 ft of bus on each of 3 phases sustained insul. damage. Portions of duct or tray housing buses was melted by the heat. There was not a sustained fire; flames of short duration and smoke only. There was no threat to any safety system.

Response to Fire (detection/protection/plant & local firedept.): Fire detector units located in MAT bay annunciated the fire, and auto. actuated the deluge water spray sys. in the bay. Bld. fire alarm was manually actuated by Control Rm. personnel upon receipt of auto. signal from MAT detectors. Site fire fighters responded; used portable CO2 extinguishers.

Time to put fire out: 0.5 hr. Time to bring plant to safe state (i.e. restore critical sys) No crit. sys affected

Was Radio activity Released as a result of Fire?: No

Comments: The unit was taken off the line twice in Nov 75 to repair overheating buses.

Fig. 3-1. Fire accident summary format used in initial screening of data

TABLE 3-1
DOMESTIC LWR PLANT FIRE SUMMARY

EVENT NO.	PLANT (PWR)	DATE	NPE REF.		TYPE OF FIRE [a]												DETECT. AND ALARM [b]		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE [c]			CONFINEMENT ENVELOPE [c] a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL ADSORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
1.	Oconee-(hot functional testing)	Dec'72	V	A20					X								X				X		0.3(e)			a) 5(e)		Oil drained to secure Containers	
2.	Oconee 1-(during construction; prior to initial criticality)	Mar'73	V	A25					X								X	X			X		4(d)			a) 25(e)	smoke, heat	Depletion of fuel	False rad. alarm.
3.	Oconee 2 (hot shutdown)	Jan'75	V	A43					X									X	X		X		0.3(e)			a) 3(e)		Limited fuel	
4.	Conn. Yankee-(@100% power)	9 Sept '75	VI	G4					⊗			X						X*			X*		0.7(d)			b) 10×3 diam (e)			Oil soaked insulation.
5.	Yankee Rowe (operating at pwr.)	June '73	VI	A27					⊗			X						X*			X		1.5(e)			a) 3(e)			Oil soaked Insulation
6.	Surry 1 (prolonged shutdown)	Jan'74	VIII	B18					X								X	X			X		1(e)			b) 7×4 diam (e)			
7.	Salem 2 (Construction)	Oct'74	XI	B95	X						X							X				X	2(e)			c) 15×20×30 (e)	Strong winds		Cause not determinable
8.	Kewaunee (power operation)	Apr'75	XI	B142	X												X	X		X	X		0.5(e)			b) 30×2 diam (d)	Smoke	Elect. isog. of bus; not a sustained fire.	Smoke in turbine bld.
9.	Three Mile Isl. 1 (@ power)	Nov'74	XI	A74					X									X			X		0.1(e)			a) 2		Limited supply of fuel!	
10.	Three Mile Isl. 1 (@ power)	Dec'74	XI	A74					⊗			X						X	X		X		0.1(e)			a) 2		Limited supply of fuel	Oil soaked lagging

[a] CIRCLED X DENOTES INITIATING FIRE, WHERE IDENTIFIED.

[b] ASTERISK (*) DENOTES SYSTEM INFERRED FROM TEST.

[c] LETTERS IN PARENTHESES FOLLOWING TIME INTERVAL: (d) = DATA; (e) = ESTIMATE

TABLE 3-1
DOMESTIC LWR PLANT FIRE SUMMARY

EVENT NO.	PLANT (PWR)	DATE	NPE REF.		TYPE OF FIRE [a]												DETECT. AND ALARM[b]		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE[c]			CONFINEMENT ENVELOPE[c] a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL ADSORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
11.	Robinson 2 (@100% power)	Apr'74	XIV	B19									X					X			X		0.2(e)	3(e)		b) 4X1diam (e)			
12.	Robinson 2 (@100% power)	Apr'74	XIV	B19									X					X			X		0.3(e)			b) 4X1diam (e)			
13.	Beaver Valley 1 (construction)	Oct72	XVI	C13	⊗						X							X			X		0.2(e)			c) 4×8×4 (e)			
14.	Salem 1 (Construction)	Apr'74	XVI	C126		X												X	X		X		0.02(d)			a) 1 (e)			
15.	Pt. Beach 1 (Shut-down for maintenance)	Nov72	XVI	C25							X							X			X		0.2(e)			a) 2 (e)			
16.	Pt. Beach 1 (Shut-down for maintenance)	Dec72	XVI	C25							X							X			X		0.1(e)			a) 1 (e)			
17.	Farley 1 (construction)	Mar'73	XVI	C40							X							X			X		0.2(e)	3(e)		a) 5 (e)			
18.	Millstone 2 (during operation)	Jul 77	XVI	C507					X		X							X			X		0.2(e)			b) 4x1diam (d)		Contained by cable tray	Smoldering debris and oil ignited by welding sparks
19	Indian Pt. 2 (construction; after fuel loading, prior to initial criticality)	Nov71	XVI	C7		X					⊗							X			X	X	2(d)			c) 40x30x10 (e)	Smoke		Fire set by arson; smoke in control rooms; loss of essential pwr. and instrum.
20	Palisades (cold shutdown)	June 71	IX	H1							X						X		X				<0.02(e)			a) < 1		Fire self-exting. & contained in equip housing	

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TABLE 3-1
DOMESTIC LWR PLANT FIRE SUMMARY

EVENT NO.	PLANT (PWR)	DATE	NPE REF.		TYPE OF FIRE (a)												DETECT. AND ALARM(b)		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE(c)			CONFINEMENT ENVELOPE(c) a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/ AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL ADSORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
21.	Kewaunee (100% Power)	Sept77	XI	A203							X						X		X				<0.02(e)			a) <1		Fire self-exting, in equipment housing	
22.	St. Lucie 1 (100% power)	Apr77	VI	B21			X											X			X	X	1.3(d)			a) 5 (e)			
23.	Zion 2 (25% power)	Sept76	XI	A168	X													X	X	X	X		1.3(d)			c) 5×10×7 (e)	Plant wide loss of lighting		Ineffective fire stop; Smoke entered EES Bus Room
24.	San Onofre 1 @ power oper.	Feb68	XI	B2	X													X			X		0.5(e)			b.) 4×2 diam(e)			
25.	San Onofre 1 @ power oper.	Mar68	XI	B2	X													X			X		0.7(e)			b. 15 × 4 diam(e)			
26.	Millstone 2 (hot stand by)	Nov76	XI	B120	X													X*	X		X		0.1(e)			a.) 1.5 (e)		Fire extinguished when MCC de-energized	Critical facility equip. dependent on MCC lost.
27	Trojan (shutdown)	Mar76	XI	B123	X													X			X		0.05(d)			c) 5×7×08 (d)		Plantwide	CM elect problems poss. due to design/assy/QC
28	Three Mile Is 2 (maintenance outage)	Nov76	XI	B172	Ⓧ						X							X*			X(e)		0.5(e)			c) 20×10×8 (e)	Fire spread past insulation barriers		CM elect. problems poss. due to design/assy/QC/CM to industry
29	Indian Pt 2 (heat-up)	Dec76	V	A59					Ⓧ			X						X*			X(e)		0.2(e)			a) 3 (e)			Oil-soaked insulation
30	Millstone 2 (cold shutdown)	June77	V	A60					X									X			X		0.1(e)			a) 3 (e)			

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TABLE 3-1
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EVENT NO.	PLANT (PWR)	DATE	NPE REF.		TYPE OF FIRE [a]												DETECT. AND ALARM[b]		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE[c]			CONFINEMENT ENVELOPE[c] a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/ AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL AD-SORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
31.	Point Beach Constr. (hot functional testing)	Jul70	V	A12					⊗			X						X			X		0.3(e)			a) 3 (e)			Oil soaked insulation
32.	Conn. Yankee (@ full power)	Aug70	V	A17					⊗			X						X			X		0.5(e)			a) 3 (e)	Loop in service, remote fire fighting req.		Oil soaked insulation
33.	Millstone 2 (100% power)	Sept76	XI	A155					X	X		X						X			X		0.2(e)			a) 2 (e)			Oil and fuel soaked insulation
34.	Calvert Cliffs 1 (100% power)	Jul77	XI	A195					X								X				X		0.2(e)			a) 2(e)			
35.	Arkansas One 1 (refueling)	Mar78	XI	A235					X								X	X			X		0.3(e)			b) 20 x 0.5 diam.		Fire contained in D.G. exhaust stack	
	(BWR)																												
36.	Quad Cities 1 (Construction)	Sept70	II	1							X							X			X		0.7(e)			b) 20 x 15 diam.(e)		Confined within reactor vessel during constr.	
37.	Vermont Yankee (3 min after reactor scram)	Nov72	VI	A 17	X												X*			X	X		0.5(e)			a.) 7(e)			Incident occurred 3 min. after reactor scram
38.	Arnold (70% power)	Nov76	XI	A143						X								X			X		0.1(e)			a) 2(e)			DG fuel line fitting failure
39	Fitzpatrick (62% power)	Mar75	XI	A151					X									X	X				0.2(e)			N.A.		Fire confined in diesel eng. and exhaust system	Loss of lube oil pressure in oper. DG
40.	Peach Bottom 3 (25% power)	Apr77	IV	E289	X													X	X		X		0.1(e)	0.3(d)	48hr(d)	a) 0.5 (e)		Confined to cabinet and limited amount of fuel	Wrong materials used in retainers

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TABLE 3-1
DOMESTIC LWR PLANT FIRE SUMMARY

EVENT NO.	PLANT (BWR)	DATE	NPE REF.		TYPE OF FIRE (a)												DETECT. AND ALARM (b)		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE (c)			CONFINEMENT ENVELOPE (c)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL/FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL ADSORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)	a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)			
41.	Peach Bottom 2 (shutdown)	July77	IX	E327	X													X	X		X		0.2(e)			a) 1 (e)		Confined to cabinet & limited amount of fuel	Ckt failures in incidents & poss plantwide & industrywide
42.	Nine Mile Point 1 (start up testing)	Ap'70	XI	B13	X													X*			X(e)		0.05(e)			c) 5 x 7 x 0.8 (e)			Similar to Trojan fire ?
43	Quad Cities 2 (power escalation testing, @ 80% power)	July72	XI	B17	X													X	X		X		0.05(e)	2(d)	8days(d)	b) 5 (d) x 0.5 diam(e)		Protection devices functioned OK and self exting.	
44.	Fitzpatrick (88% power)	Apr77	XI	B137	X												X*		X				<0.05(e)		5min(d)	a.) 0.5(e)		Self-exting in cabinet	
45.	Arnold (Refueling shutdown)	May77	XI	B139	(X)						X						X*				X		<0.05(e)			b.) 0.5 x 1 diam(e)		Confined to cabinet	
46.	Oyster Creek (Refueling shutdown)	May77	XI	B142	X												X			X			0.08(d)			b) 10 x 0.5 diam (e)			Reactor scram resulted from loss of pwr to react protect system
47.	Browns Ferry 3 (47% Power)	July77	XIII	88/101				X						X			X		X				24(d)	1(d)	2days	b.) 66 x ~3 diam (d)		& explosion fire entirely contained in charcoal adsorber vessels	
48.	Monticello (plant in operation)	Fall 6(e)	XIV	B20	(X)									X				X			X		<1(d)			b) No idea		Fire entirely contained in charcoal filter housing	
49.	Hatch 1 (Construction)	Apr74	XVI	C78								X						X			X		0.1(d)	.05(d)		b) 5 x 7 diam (e)		Contained entirely within containment vessel during constr.	
50.	Quad Cities 1 (@ 90% power	Dec72	VIII	C7		X			(X)			X						X			X		0.05(e)			c) 0.5 x 1 x 0.3 (e)		Contained in pump motor case	Started by welder's sparks

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			SECT.	ALPHA NUMER	ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/ AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL AD-SORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
51.	Arnold (plant operating)	Dec75	XI	A102					Ⓧ			X					X	X			X		0.1(e)			b) 4x1 diam (e)		Limited supply of oil fueling fire	
52.	Arnold (plant operating)	Feb76	XI	A111					Ⓧ			X						X			X		0.1(e)			b) 4x1 diam (e)		Limited supply of oil fueling fire	
53.	Arnold (plant operating)	Mar76	XI	A112					X									X			X		0.1(e)			b) 4x1 diam (e)		Limited supply of oil fueling fire	
54.	Arnold (plant operating)	1day later	XI	A112					X									X			X		0.1(e)			b) 4x1 diam (e)		Limited supply of oil fueling fire	
55.	FitzPatrick (28% Power)	Oct76	XI	A158					X									X			X		0.1(e)			b) 10 x 0.5 diam		Contained in D.G. exhaust pipe	
56.	Browns Ferry 1 (Shutdown)	May76	XVI	C224							X							X			X		0.3(e)	0.5(e)		a) 4 (e)		Limited material fueling fire; ease of accessibility	
57.	Brunswick 1 (cold shutdown)	May77	XVI	C347							X							X			X		0.05(e)			a) 4 diam (e)			
58	Brown's Ferry 1, 2 (both @100% pwr)	Mar75	XIV	A15	X	X						Ⓧ					X	X		X	Ⓧ	X	T(d)			c) 30x40x12 Ht (e)	See NPE text and Fire journal article May 77	Final decision to use water	See NPE text

(a) CIRCLED X DENOTES INITIATING FIRE, WHERE IDENTIFIED.

(b) ASTERISK (*) DENOTES SYSTEM INFERRED FROM TEST.

(c) LETTERS IN PARENTHESES FOLLOWING TIME INTERVAL: (d) = DATA; (e) = ESTIMATE

TABLE 3-1
DOMESTIC LWR PLANT FIRE SUMMARY

EVENT NO.	PLANT (OTHER)	DATE	REF.	TYPE OF FIRE [(a)]												DETECT. AND ALARM [(b)]		FIRE FIGHTING				TIME INTERVALS ASSOCIATED WITH FIRE [(c)]			CONFINEMENT ENVELOPE [(c)] a) SPHERE/HEMI-SPHERE DIAM. (FT) b) CYLINDER (OR HALF CYL.) LENGTH, DIAM. (FT) c) RECTANG. SPACE DIMEN. (FT)	OBSTACLES TO ENDING INCIDENT	FACTORS WHICH FACILITATED ENDING INCIDENT	COINCIDENTAL FACTORS OF INTEREST FOR EVENT TREE/ FAULT TREE CONSTRUCTION
				ELECT. INSUL. (OF ELECT. ORIGIN)	ELECT. INSUL. (OF NONELECT. ORIGIN)	GENERATOR H_2	OFF-GAS H_2	LUBE OIL/GREASE/ HYD. FLUID	FUEL OIL-DG/AUX. BOILER/FIRE PUMP	COMBUSTIBLE SOLID	NONELECT. INSUL./FILLER MATERIAL	EXPAN. JOINT FILLER	OFF-GAS CHARCOAL AD-SORBER BED/FILTERS	LIGHTNING	SPONTANEOUS COMBUSTION	AUTOMATIC	MANUAL	SELF-EXTING.	AUTOMATIC	IN-PLANT FIRE FIGHTERS	LOCAL F.D.	TIME TO BRING FIRE UNDER CONTROL (HR)	INTERVAL PRIOR TO DETECT. (HR)	RESTORATION OF CRITICAL FUNCTIONS (HR)				
59	Telephone Exch Fire, N.Y, N.Y.	Feb 75	Fire Journal July 75	ⓧ	X					X	X					X	X				X	16	0.5(e)	48 emerg type	C_1) 20x20 X 300 C_2) 20x10x 275 C_3) 20 x 10 x 275 C_4) 2 X 10 X 275	Basement 1st floor 2nd floor 3rd floor Toxic smoke	Sprinklers in trash rm; ease of getting to fire.	Fire Stops between floors effective in retarding spread.
60	World Trade Center Fire, N.Y, N.Y.	Feb 75	Fire Journal July 75		X					ⓧ						X	X				X	12(e)	0.5(e)		C_1) 35 X 15 X 130 C_2) 50 X 15 X 60 C_3) 6 X 4 X 100		Confinement to telephone closets; Smoke control	Smoke detectors rendered ineffective by shutdown of Bldg. HVAC at night

[(a)] CIRCLED X DENOTES INITIATING FIRE, WHERE IDENTIFIED.

[(b)] ASTERISK (*) DENOTES SYSTEM INFERRED FROM TEST.

[(c)] LETTERS IN PARENTHESES FOLLOWING TIME INTERVAL: (d) = DATA; (e) = ESTIMATE respect to the type of combustible material involved, cause, means of detection, extent of damage, and other factors relevant to the initiation and progression of fires. Of the total set of 58 fires identified in this sample, most could be categorized as either involving electrical components (18) or flammable liquids (23). Nearly all of the fires were put out quickly and most did not spread appreciable distances. The section below presents some statistical analyses of the data summarized in Table 3-1.

3.2. STATISTICAL ANALYSES OF DATA

A summary of the overall experience with fires in the U.S. is given in Table 3-2, which includes 483 reactor-years of construction experience and 372 reactor-years of operation beyond first electrical generation. Estimates of the average occurrence rate during operation, about 1 fire per 10 reactor-years, are in agreement with those presented in Refs. 7, 9, and 10, which were based on a smaller data base. The surprisingly smaller rate observed during construction may be attributed to censorship in the summaries used as data sources and to nonexistance of reports of minor fires that occur before the fueling of the reactor.

In support of event tree/fault tree quantification, it was of interest to determine the extent of damage caused by the fires and the time required to bring the fires under control. In some cases, this information could be obtained directly from the incident reports; in other cases, rough estimates were made of the dimensions of the fire. This information was grouped into class intervals of approximate fire diameter and time to put out the fire, as presented in Table 3-3. As expected, the distances and times were found to be highly correlated, as evidenced by the linear regression analyses summarized in Figs. 3-2 through 3-4 for all fires, electrical fires, and flammable liquid fires, respectively. Exponential fits of the CCDF in the distance variable are given in Fig. 3-5. These were used in the cable fire propagation model in Section 5.3.

TABLE 3-2
SUMMARY OF NUCLEAR PLANT EXPERIENCE WITH FIRES

Number of reactor units in data base(a)	77
Number of units reporting fires	37
Reactor-years experience(b)	
During construction	483
During operation(c)	372
Number of fires observed	
During construction	9
During operation(c)	49
Average rate of occurrence per reactor-year	
During construction	0.02
During operation	0.1

(a) Includes 50 PWRs and 27 BWRs in the U.S.

(b) Through April 1978.

(c) Includes all data following date of first electrical generation.

TABLE 3-3
NUCLEAR PLANT FIRE DATA
[NUMBER OF FIRES (FRACTION OF DATA BASE)]

Approx. Maximum Diameter of Fire (ft)	Time to Put Out Fire (hr)			
	0-1	1-3	3-10	10-30
0-1	6 (0.1)	--	--	--
1-3	12 (0.2)	--	--	--
3-10	23 (0.4)	3 (0.05)	--	--
10-30	7 (0.1)	1 (0.02)	1 (0.02)	--
30-70	1 (0.02)	2 (0.03)	1 (0.02)	1 (0.02)

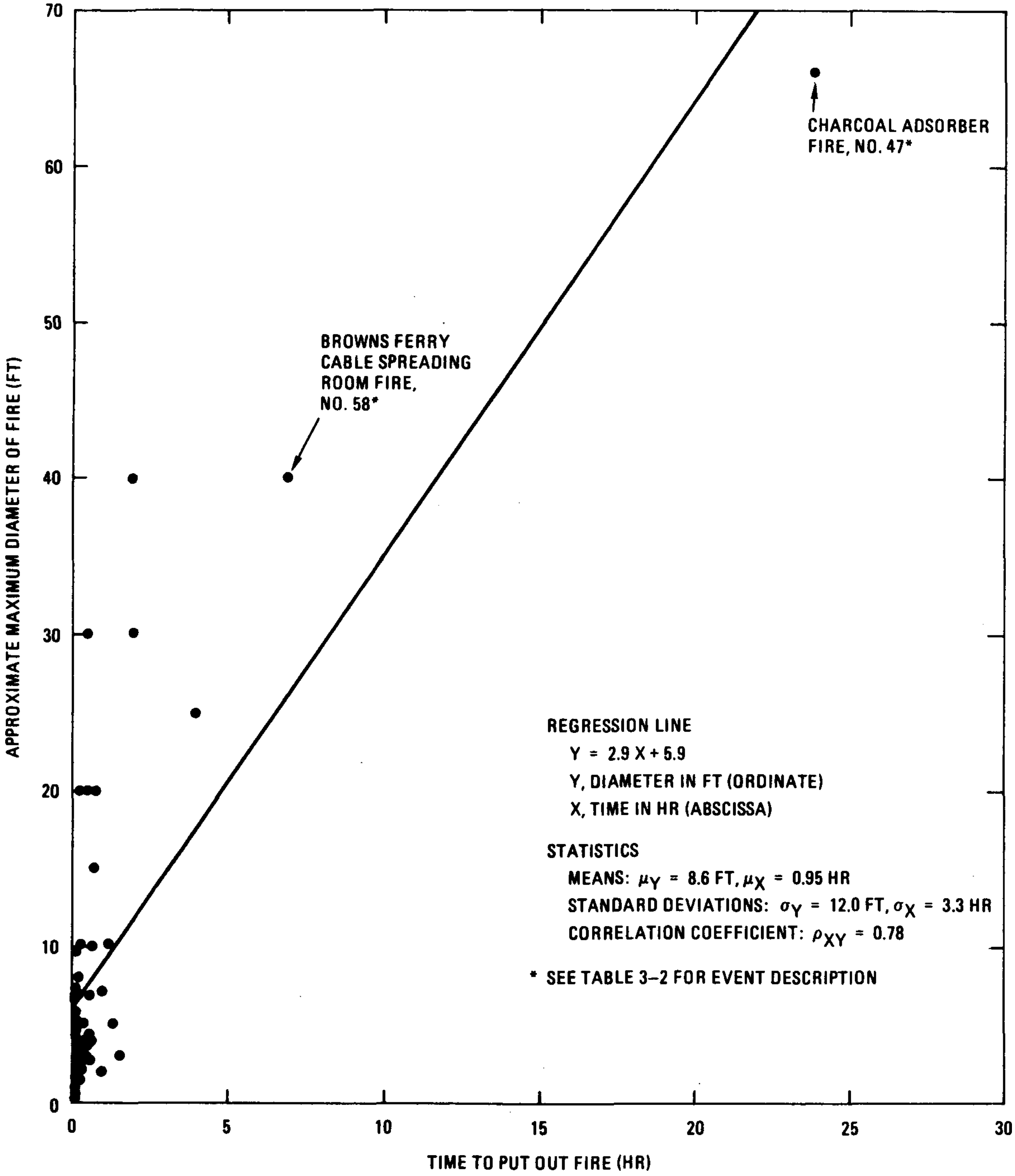

Fig. 3-2. Linear regression analysis of fire diameter versus time to put out fire -- all nuclear plant data (58 events)

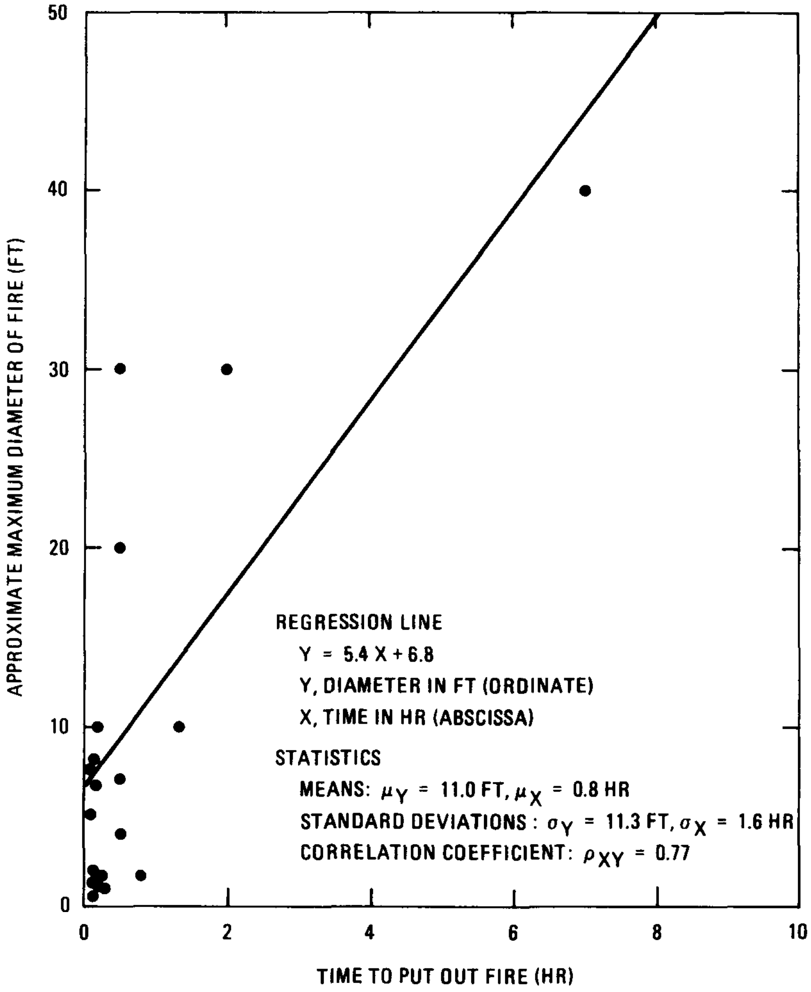

Fig. 3-3. Linear regression analysis of fire diameter versus time to put out fire -- electrical fires (18 events)

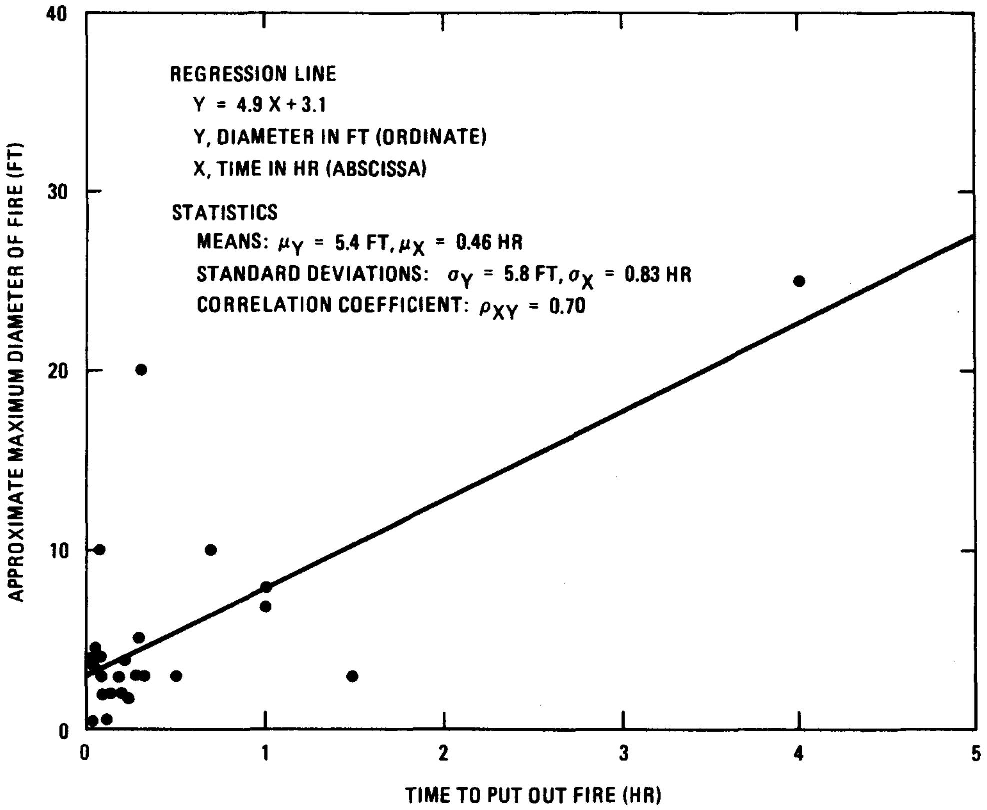

Fig. 3-4. Linear regression analysis of fire diameter versus time to put out fire -- lube oil fires (23 events)

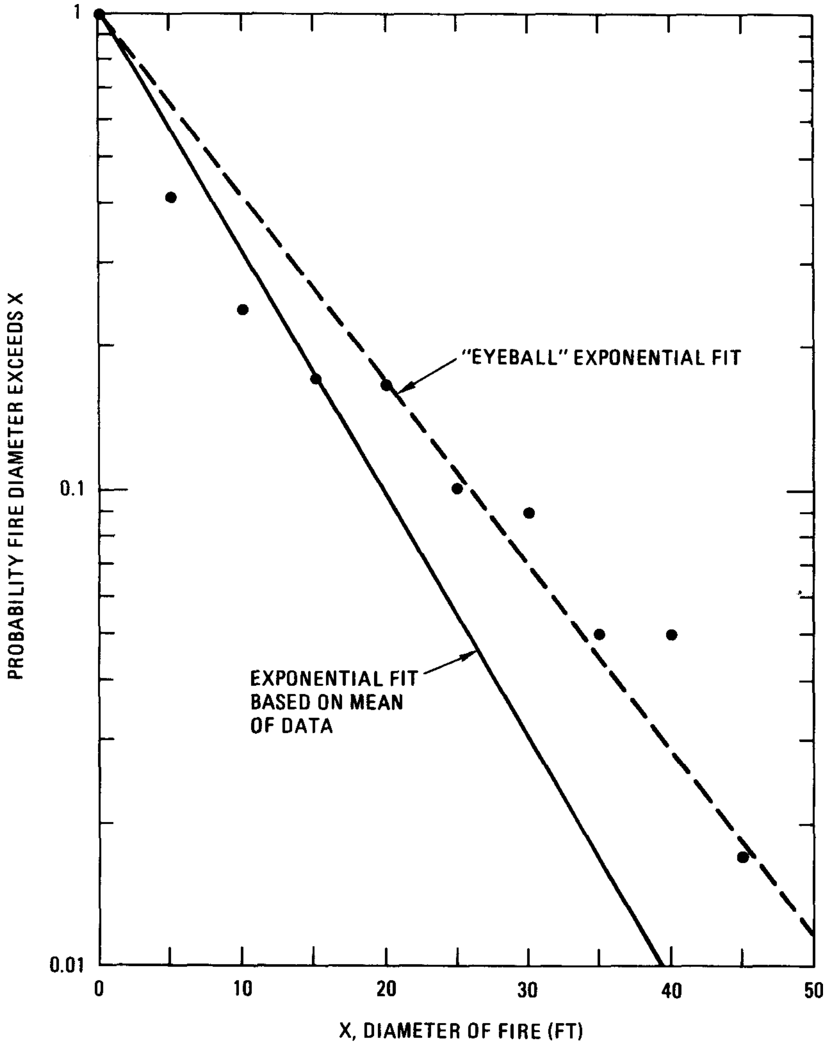

Fig. 3-5. Exponential fits of fire diameter complementary cumulative distribution function

4. IMPORTANT FIRE LOCATIONS IN AN HTGR PLANT

Insights obtained from examination of the data involving fires, summarized in the previous section, are used in this section in conjunction with the methods described in Section 2.3 to identify important potential fire locations in a specific HTGR plant. This plant is the 1975 reference HTGR steam-cycle design with a rated power output of 1160 MW(e). The nuclear steam supply system is that described in GASSAR (Ref. 24). The balance-of-plant design and layout drawings are those by Gilbert Associates (Ref. 25). In Section 4.1, bounding calculations are performed to narrow the list of candidate locations to six. A qualitative analysis procedure patterned after Failure Modes and Effects Analysis (FMEA) and referred to as Fire Location and Progression Analysis (FLPA) is used in Section 4.2 to identify the candidate locations having the greatest potential for producing risk-contributing accident sequences. The location found to be among the most important, the cable spreading room immediately below the control room, is analyzed quantitatively in Sections 5 and 6.

4.1. BOUNDING CALCULATIONS IN SCREENING LOCATIONS

Accident sequences associated with initiating events other than fires have been analyzed extensively for the HTGR in prior studies, especially the comprehensive assessment of core heatup sequences in Ref. 1. Therefore, the bounding calculations described in Section 2.3 can be used to narrow the list of candidates of important fire locations. The procedure starts by establishing the relationship between HTGR accident release categories and the sets of systems whose failure would result in the various release categories. Because of their dominance across most of the accident spectrum, accidents involving core heatup are emphasized. As explained in Section 2.4, the release categories CH-1 through CH-6 presented in Table 4-1 were obtained from a two-stage grouping process involving about 150

TABLE 4-1
RELEASE CATEGORIES FOR HTGR ACCIDENTS INVOLVING CORE HEATUP

Release Category i	Description	Failed Systems(a),(b)	Total Occurrence Rate λ_i (median estimate) (events/reactor-year)
6	Core heatup with intact containment, effective PCRV plateout	MLCS, CACS	3×10^{-5}
5	Core heatup with containment overpressurization at 12 days, effective PCRV plateout	MLCS, CACS, LCS	3×10^{-6}
4	Same as 5 with containment failure at 8 days	MLCS, CACS, LCS	2×10^{-7}
3	Same as 5 with poor PCRV plateout	MLCS, CACS, LCS, ACP(c)	3×10^{-7}
2	Same as 4 with poor PCRV plateout	MLCS, CACS, LCS, ACP(c)	2×10^{-8}
1	Same as 6 with nonisolated containment	MLCS, CACS, CIV	1×10^{-9}
0	Same as 5 with nonisolated containment	MLCS, CACS, LCS, CIV	$<1 \times 10^{-9}$

(a) MLCS = Main Loop Cooling System
CACS = Core Auxiliary Cooling System
LCS = PCRV Liner Cooling System
ACP = all ac electric power failed for ≥ 30 hr
CIV = Containment Isolation Valves

(b) Categories 4 and 5 are different because of different physical processes; same is true for categories 2 and 3.

(c) In this case, MLCS, CACS, and LCS all fail as a consequence of ACP.

accident sequences that were analyzed in Ref. 1. A seventh release category, CH-0, was encountered in a study of containment design options for the HTGR (Ref. 20). Also indicated in Table 4-1 are the failed systems and the total occurrence rate summed over all sequences that result in the indicated release category.

If there exists an accident sequence initiated by a fire and resulting in the systems impact corresponding with one or more of the release categories in Table 4-1, at an occurrence rate that approaches the current total, then the fire would make a significant contribution to the overall risk assessment. Note that it is not necessary for a fire to be the direct cause of all the failures indicated, only that the probability of the entire accident sequence including any failures independent of the fire be significant compared with the values in Table 4-1.

A preliminary screening of the systems layout drawings was performed to determine that some of the plant locations do not contain any of the components in any of the important systems in Table 4-1. A second group of locations contains components in one of the important systems, and a relatively small group of locations has components in two or more of these systems. Another important consideration about the layout is that all the listed systems except the main loop cooling system (MLCS) are redundant engineered safety systems that have been designed according to many applicable codes and standards and the General Design Criteria. In locations where two or more of the redundant legs of each of those systems are found, the layout conforms to separation requirements such as the single failure criterion and those in Regulatory Guide 1.75. Regulatory Guide 1.75 specifies separation requirements for electrical systems. Note, however, that the plant was designed in the 1973-1974 time frame and does not conform to all the fire protection criteria in Regulatory Guide 1.120 (draft), which was written after the Brown's Ferry fire in 1975.

Various levels of systems impact were postulated to occur corresponding to the maximum damage that could be expected at each group of locations described above. For example, in certain locations a fire could conceivably cause failure of the MLCS or one or more of the redundant legs of the core auxiliary cooling system (CACS), etc. The fault trees for these systems, presented in Ref. 1, were used to estimate the conditional probability of an accident resulting in each of the release categories given a specified fire impact. Equation 2-9 was then used to estimate the minimum occurrence rate of the fire necessary to result in a significant contribution to the risk assessment. The results of this procedure are presented in Table 4-2.

It was shown in Section 3 that the average occurrence rate of fires of any severity experienced in about 370 reactor-years of experience in the U.S. is 1×10^{-1}/reactor-year. This experience is with LWRs, and nearly all of the 56 fires result in minor damage with no impact on systems important to safety. Also, nearly all of the fires that have occurred are neither peculiar to nuclear plants nor unique to LWRs. Since an HTGR would be licensed with the same separation and fire protection criteria as LWRs, and since the unique characteristics of HTGRs do not appear to have any bearing on either the initiation or progression of fires,* the average fire occurrence rate of 1×10^{-1}/reactor-year is considered applicable to HTGRs as well. If all the equipment in a given location is assumed to fail because of a fire (described by impact j) and the fire occurrence rate needed to result in a significant contribution to risks is greater than 1×10^{-1}/reactor-year, that location is ruled out as an important fire location.

Locations that only contain components in one of the redundant cooling loops of the CACS ($j = 1$) can be ruled out, since a CACS component-disabling fire would have to occur at a rate of 30 times/year to produce a significant risk contribution. In a similar fashion, locations whose impact potential is limited to the PCRV liner cooling system (LCS) ($j = 6$) as well

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*An exception to this are small hydrogen explosions in off-gas systems in LWRs associated with radiolysis of primary coolant. These were not included in the data base.

TABLE 4-2

FIRE OCCURRENCE RATE AND DAMAGE IMPACT NECESSARY TO MAKE SIGNIFICANT CONTRIBUTION TO RISK

Release Category i	Systems Postulated to Fail Due to Fire j	Conditional Probability of Release Category Given $\Pr\{C_i \mid j\}$	Fire Occurrence Rate Needed to Make Significant Contribution to Risk $\lambda_{f \rightarrow j} = \frac{\lambda_i}{\Pr\{C_i \mid j\}}$
CH-6	1. 1 CACS loop	1×10^{-6}	30(a)
	2. MLCS	3×10^{-4}	1×10^{-1}(a)
	3. CACS	2×10^{-3}	2×10^{-2}
	4. MLCS, 1 CACS loop	4×10^{-3}	8×10^{-3}
	5. MLCS, CACS	1	3×10^{-5}
CH-5, 4	6. LCS	6×10^{-7}	5(a)
	7. LCS, 1 CACS loop	1×10^{-5}	3×10^{-1}(a)
	8. LCS, MLCS	3×10^{-4}	1×10^{-2}
	9. LCS, CACS	2×10^{-3}	2×10^{-3}
CH-3, 2	10. TT(b)	3×10^{-8}	10(a)
CH-1	11. CIS	6×10^{-7}	2×10^{-3}
	12. CIS, 1 CACS loop	1×10^{-5}	1×10^{-4}
	13. CIS, MLCS	3×10^{-4}	3×10^{-6}
	14. CIS, CACS	2×10^{-3}	5×10^{-7}

(a) Exceeds bounds on occurrence rate; location ruled out as important fire contributor.

(b) TT = turbine trip

as the LCS plus one CACS loop (j = 7) are ruled out because the fire occurrence rate exceeds the bound. The same is true for fires resulting in turbine trip (j = 10). The necessary fire occurrence rate when the damage potential is limited to the MLCS is equal to the bound of 1×10^{-1}/ reactor-year. However, these locations can also be ruled out since nearly all the fires that occurred in the data base would not have had any effect on operation of the MLCS.

All the remaining locations contain components in two or more of the redundant subsystems of the CACS or containment isolation system (CIS) or contain components in at least two of the systems in the set {CACS, CIS, LCS, MLCS}. This effectively eliminates all but six locations in the HTGR plant layout used in this study, and these are listed in Table 4-3. The approximate layout of these locations within the HTGR plant is indicated in Fig. 4-1. In the following section, each of these six locations is analyzed using the specialized FMEA procedure described in Section 2.3.

4.2. QUALITATIVE ANALYSIS OF IMPORTANT LOCATIONS

The qualitative Fire Location and Progression Analysis (FLPA) procedure described in Section 2.3 was applied to each of the six candidate locations listed in Table 4-3. An example is shown in Table 2-3. Of the six locations evaluated, two were found to have the greatest potential for producing dominant risk accident sequences, the cable spreading room (II) and one of the vertical equipment chases (IV). Note that, although Regulatory Guide 1.75 was followed in the cable tray layout, nearly all of the instrument and control cables that are run to the control room are routed through a single cable spreading room.

The second vertical equipment chase (III) was found to be less important than IV because it contained cables and components (primarily piping) in fewer systems and subsystems. The horizontal chase (V) was also found to be less important than IV because vertical runs of cables can propagate a fire much more rapidly than horizontal ones, and a greater distance of propagation is required in V to make a comparable level of systems impact.

TABLE 4-3
IMPORTANT LOCATIONS FOR MAJOR FIRES OBTAINED WITH BOUNDING PROCEDURE[a]

No.	Location
I	Control room
II	Cable spreading room below control room
III	Exterior vertical chase in control building
IV	Interior vertical chase between control building and reactor service building
V	Horizontal chase in penetration building (surrounds containment building)
VI	Containment building

[a] These results are highly dependent on the plant layout.

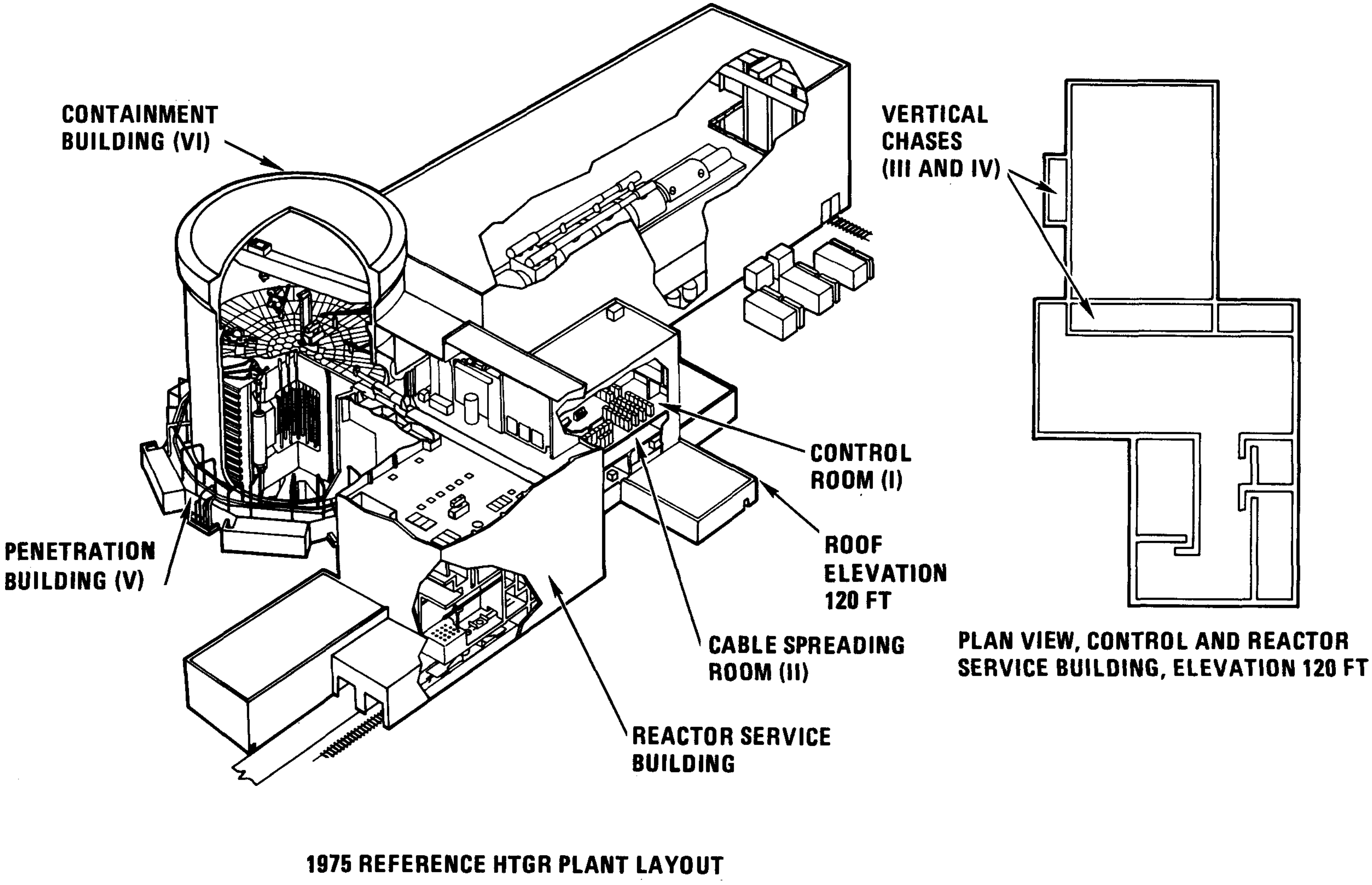


Fig. 4-1. Layout of HTGR plant and important potential fire locations

The control room has essentially the same inventory of cables as the cable spreading room in addition to the control room instruments. However, in view of the fact that the control room is always attended and the cable spreading room is rarely occupied, and in consideration of the enhanced accessibility to all areas of the control room, there was found to be a far greater potential for significant fire propagation in the cable spreading area.

The containment building was ruled out for several reasons. Although all of the systems important to safety have essential components located here, separation distances among redundant legs of the engineered safety feature (ESF) systems, e.g., CACS, CIS, are much greater here than in the selected areas (II and IV). Hence, a fire would have to progress much farther inside the containment to produce a comparable degree of impact. The second reason is that the ESF components inside the containment are designed and qualified for a set of design basis accidents that produce more severe environments than those outside the containment. Containment building fires, although not significant accident initiators, have been analyzed extensively as consequential effects of accidents involving high-temperature water/graphite reactions (Refs. 1, 24, 25).

It was not possible to determine whether the cable spreading room (II) or the interior vertical chase (IV) has the greatest potential for producing risk-contributing accident sequences. The former has a greater potential in terms of inventory of systems, whereas the latter has more potential for propagation, since the cables are run predominantly in the vertical direction. The cable spreading room was selected for the event tree/fault tree analysis presented in Sections 5 and 6. Ideally, both locations would have been analyzed. However, to demonstrate the methodology in this report, only one needed to be analyzed. Also, it was recognized that the quantitative results are dependent on the fire protection criteria imposed on the design and that significant modifications would be needed if the analyzed design were upgraded to meet more stringent fire protection criteria such as Regulatory Guide 1.120 (draft). The cable spreading room fire at Borwn's Ferry was, of course, an important consideration in the selection of this location.

5. RISK ASSESSMENT OF CABLE SPREADING ROOM FIRES

The event tree for the initiating event, "fire occurs in the cable spreading room," is presented in this section, together with the quantification of the event tree probabilities and consequences. The risk assessment gives comprehensive coverage of cable spreading room fires in that fires of all sizes up to those that cause complete failure of the room's inventory of components are considered. In the development and quantification of the event tree, component and system failures that are independent of the fire but that compound its consequences are considered in addition to those that directly result from the fire.

The development of the event tree is discussed in Section 5.1, along with the results of the event tree probability quantification. In Section 5.2, the quantification of probabilities, which includes the initiating events and the fault tree analysis, is described. The latter makes use of a cable fire propagation model, described in Section 5.3. The fault trees for a representative accident sequence are presented in the Appendix to illustrate the degree of detail that was required in this area. In Section 5.4, the physical processes and accident consequences for each release category are briefly discussed.

5.1. EVENT TREE

It was determined in prior studies that accident sequences involving core heatup tend to dominate the HTGR risk assessment across most of the accident spectrum, including the relatively high-consequence, low-probability region of the spectrum. Therefore, core heatup accidents were emphasized in the development of the cable spreading room fire event tree. Use was made of what was learned in the analysis of the physical processes and radiological consequences of a wide spectrum of initiating events and accident sequences in the Phase II core heatup study presented in Ref. 1 and briefly described in Section 2.5. Since the consequences of a large number of accident sequences have now been determined, it was possible to keep the fire event tree relatively simple so as to distinguish only among sequences that belong to different release categories. Whereas the core heatup event trees developed previously (Ref. 1) defined hundreds of sequences, sequences, the fire event tree described below contains only eight.

The qualitative information necessary to construct the event tree is presented in block diagram format in Fig. 5-1. The initiating event (1) is the occurrence of a fire of any size in the cable spreading room. Once detected by the plant operator, either directly by indication of smoke or high temperature in the ventilation system instruments or indirectly by the response of control room instruments to fire damage, the normal and desirable plant response would be an orderly plant shutdown, which includes a reduction in power and reactor scram (2a) and core cooldown using the main cooling loops (3). If the reactor operator is unable to detect the fire and it continues to develop, it is likely that an open circuit or short to ground would eventually occur in one or more of the reactor plant protection system circuits. This type of fault would simulate, in effect, a reactor scram (2b), since the reactor trip circuits operate on inhibit logic; i.e., they deenergize to actuate. A common cause failure of multiple control rod mechanisms could prevent control rod insertion, even if one of these circuits were grounded or severed. However, several hours would be available for the operator to insert the reserve shutdown system (2c) before the guide tubes for inserting the B_4C pellets into the core become damaged and while the reactor shuts itself down on the negative temperature coefficient of reactivity after about 5 minutes. Damage of a sufficient number of reactor shutdown system circuits could prevent its operation since it operates on transmission logic; i.e., it energizes to actuate.

If both core cooling systems, the MLCS (3) and CACS (4), fail subsequent to shutdown, either as the direct result of the fire, failures

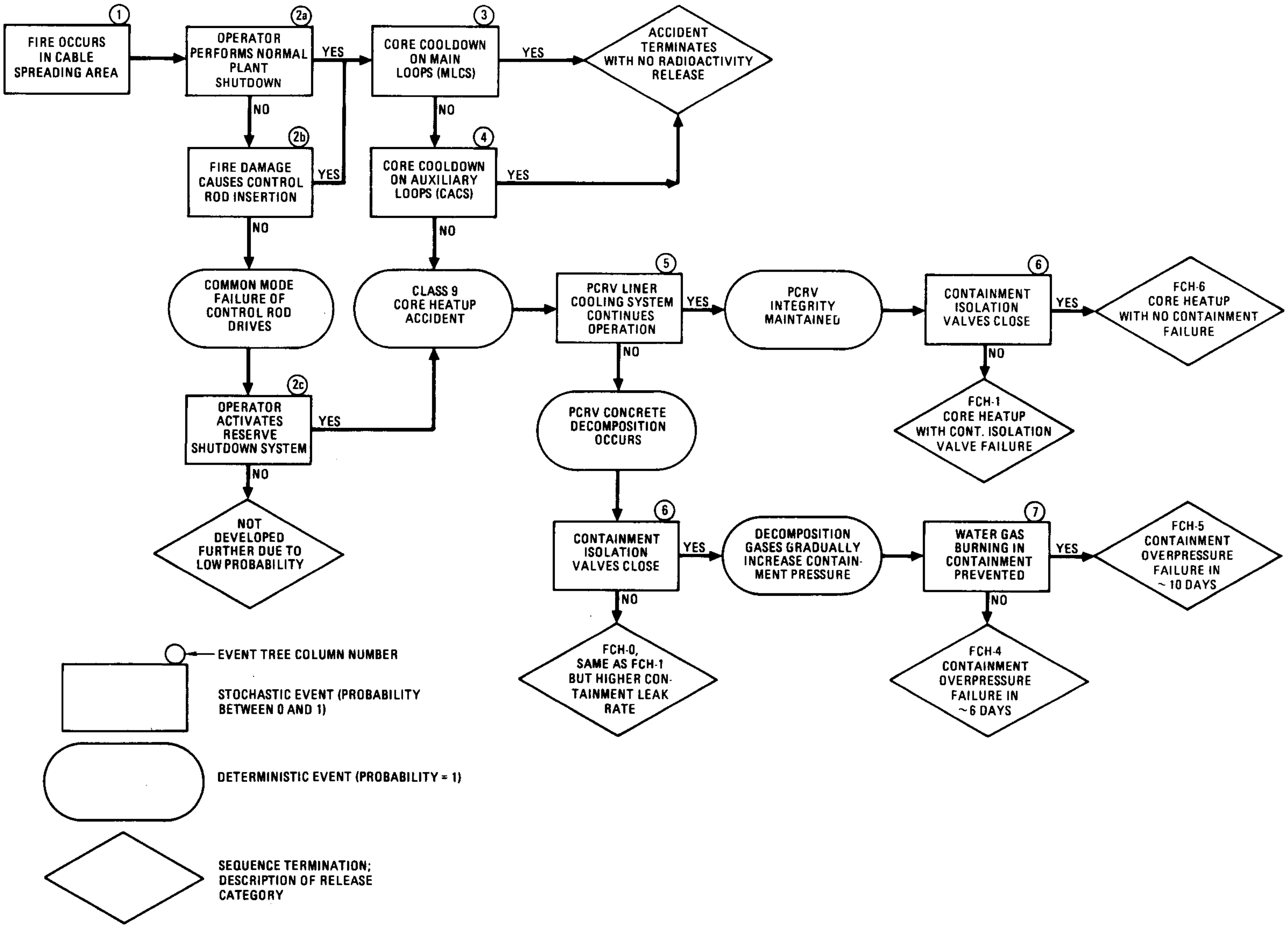


Fig. 5-1. Event sequence block diagram for fire in cable spreading area

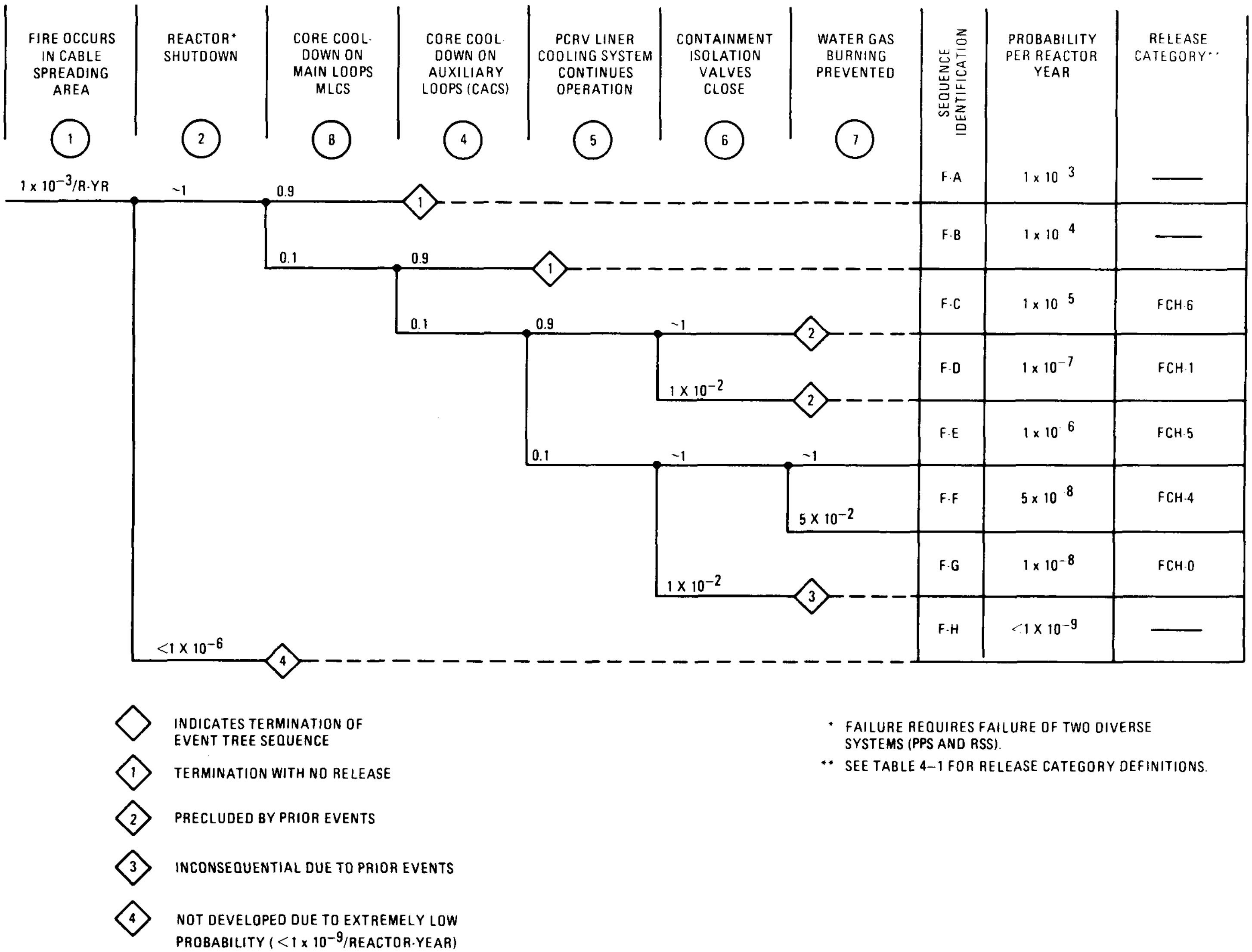
independent of the fire, or some combination of these, a Class 9 core heatup accident would result. A Class 9 accident is more severe than the design basis accidents established for the large HTGR in the licensing process. Fire-induced failures of the core cooling systems at this fire location would most likely involve opens and/or grounds in a number of the various circuits of the control and protection systems. The latter include the Overall Plant Control System (OPCS), the Operational Protection System (OPS), and the Plant Protection System (PPS).

If neither cooling system can be restored within about 3 hours, it is likely that the maximum reactor coolant temperature is too high to consider restart, and the loss of forced cooling via helium circulation becomes permanent. [In Ref. 1, this 3-hour figure is referred to as the Maximum Time to Restore Cooling (MTRC).] If the PCRV liner cooling system (LCS) survives the elevated temperatures of the heatup transient (5), it possesses a sufficient heat removal capability to terminate the increase in core temperatures and eventually cools down the reactor core. If the LCS fails, on the other hand, the concrete in the PCRV eventually begins to decompose after about 4 days, producing some noncondensible gases, primarily CO_2 from limestone and steam from dehydration of concrete. Sufficient noncondensables are produced after 12 days to overpressurize the containment if the isolation valves are closed (6). Overpressurization can occur as early as 8 days if global burning of water gas (H_2 + CO) occurs as a result of the dehydration moisture reacting with the core graphite at elevated temperature ($H_2O + C \rightarrow H_2 + CO$).

In these core heatup sequences, the radiological consequences are determined primarily by the containment leakage characteristics. In the case of FCH-6, the consequences are the lowest, since the containment atmosphere temperature and pressure transients never exceed their design limits and the containment leak rate remains low throughout the accident transient. On the other side of the consequence spectrum is FCH-0, which involves failure of the containment isolation valves to close, leaving a large opening of about 3 ft in the containment boundary. The opened containment is combined with the generation of concrete decomposition gases, which creates a driving force to sweep radioactive gases out the hole in the containment. Category FCH-1 is associated with isolation valve failure without the driving force of decomposition gases. An intermediate level of consequences is associated with categories FCH-4 and FCH-5, which involve a low leak rate for several days and a high leak rate subsequent to overpressure failure of the containment. Quantitative information about accident consequences is summarized in Section 5.4.

The event tree for the cable spreading room fire and the results of the probability quantification are illustrated in Fig. 5-2. The occurrence rate for a fire of any size in the cable spreading room was assessed at 1×10^{-3}/reactor-year, based primarily on reactor operating experience, as explained in Section 5.2. When this event tree is compared with the event trees developed in the Phase II core heatup study (Ref. 5-1), it can be seen that the fire has a significant, in fact, dominating, impact on the failure probability of the CACS (4) and containment isolation valves (6). This point is illustrated in Table 5-1, where the event probabilities along accident sequence D are compared with those associated with a similar accident sequence following the initiating event, loss of condenser function, taken from Ref. 1. The latter is the highest probability sequence analyzed in prior studies having the same consequences as sequence D.

The initiating event occurrence rate and the probability of MLCS failure are significantly less for the fire sequence than for the condenser failure sequence. The probability of successful reactor shutdown and PCRV liner cooling system operation are high for both sequences. However, the core auxiliary cooling system and containment isolation valves, both of which are redundant engineered safety systems, fail at a much higher probability following the cable fire. This illustrates dramatically the importance of fires as a common cause failure. The joint failure probability of these two systems, the CACS and CIS, is increased by a factor of more



SEQUENCE IDENTIFICATION	PROBABILITY PER REACTOR YEAR	RELEASE CATEGORY**
F-A	1×10^{-3}	——
F-B	1×10^{-4}	——
F-C	1×10^{-5}	FCH-6
F-D	1×10^{-7}	FCH-1
F-E	1×10^{-6}	FCH-5
F-F	5×10^{-8}	FCH-4
F-G	1×10^{-8}	FCH-0
F-H	$<1 \times 10^{-9}$	——

◇ INDICATES TERMINATION OF EVENT TREE SEQUENCE

◇1 TERMINATION WITH NO RELEASE

◇2 PRECLUDED BY PRIOR EVENTS

◇3 INCONSEQUENTIAL DUE TO PRIOR EVENTS

◇4 NOT DEVELOPED DUE TO EXTREMELY LOW PROBABILITY ($<1 \times 10^{-9}$/REACTOR-YEAR)

* FAILURE REQUIRES FAILURE OF TWO DIVERSE SYSTEMS (PPS AND RSS).

** SEE TABLE 4–1 FOR RELEASE CATEGORY DEFINITIONS.

Fig. 5-2. Event tree for fire in cable spreading area

TABLE 5-1
IMPACT OF FIRES ON THE FAILURE PROBABILITY OF MULTIPLE REDUNDANT SYSTEMS ALONG ACCIDENT SEQUENCE D

Events Along Accident Sequence	Loss of Condenser Function (Sequence M-II-BC)	Fire in Cable-Spreading Area (Sequence D)
1. Initiating event	3×10^{-1}/r-yr	1×10^{-3}/r-yr
2. Reactor shutdown	∿1	∿1
3. Main loop cooling system fails	∿1	10^{-1}
4. Auxiliary core cooling fails	10^{-4}	10^{-1}
5. PCRV liner cooling maintained	∿1	∿1
6. Containment isolation valves fail	3×10^{-5}	10^{-2}
Frequency of core heatup (events 1-4 only)	3×10^{-5}	1×10^{-5}
Frequency of core heatup with containment failure (events 1-6)	10^{-9}	10^{-7}

than 10^5 during the fire sequence! The reason for this result, as will be shown below, is that a single fire in the cable spreading area can cause failure of multiple systems that have cables routed in a location common to the fire. To cause this much impact, it is only necessary for the fire to encompass multiple separation categories of cables. The separation requirements in Regulatory Guide 1.75, followed in the HTGR design analyzed here, apply to the separation of safety-related divisions (redundant networks), but there is no requirement to separate among the various systems. Hence, cables that serve different systems are routed through the same cable tray.

The probability per reactor-year of a cable spreading room fire leading to core heatup (approximated well by that of sequence C) was found to be 1×10^{-5}, or about a factor of 3 less than that determined for all other initiating events that have been analyzed (Ref. 1). However, the probability per reactor-year of the relatively high consequence release categories, FCH-0 and FCH-1, was found to be much greater in the case of fires, as is apparent from examination of Table 5-2. In Section 6, the implications of this result on the overall HTGR risk assessment are discussed.

5.2. PROBABILITY QUANTIFICATION

The probability per reactor-year of a fire in the cable spreading room was assessed primarily on the basis of experience data and in consideration of the design features of the HTGR analyzed in the study. The data involving nuclear plant fires in LWRs summarized in Section 3 include several fires that involve cable trays, and only one, the Brown's Ferry fire of 1975, that was located in the cable spreading area in a total of 372 reactor-years of experience through May 1, 1978. There has been additional experience without additional cable spreading room fires in the U.S., including that with non-LWRs and that since May 1, 1978, bringing the total to about 480 reactor-years as of May 31, 1978. This gives an average observed occurrence rate for cable spreading room fires of about 2×10^{-3} events per reactor-year.

TABLE 5-2
CONTRIBUTIONS TO RELEASE CATEGORY PROBABILITIES BY FIRES AND OTHER INITIATING EVENTS

	Probability Per Reactor-Year	
Release Category	Cable Spreading Room Fire	All Other Initiating Events
CH-0	1×10^{-8}	$<1 \times 10^{-9}$
CH-1	1×10^{-7}	1×10^{-9}
CH-2	(a)	2×10^{-8}
CH-3	(a)	3×10^{-7}
CH-4	5×10^{-8}	2×10^{-7}
CH-5	1×10^{-6}	3×10^{-6}
CH-6	1×10^{-5}	3×10^{-5}

(a) No sequences identified in this release category.

The same approach was used to estimate the frequency of fires in the cable spreading room in Refs. 8 and 9. For historical reasons, the estimate was based on less experience with a correspondingly greater value of the occurrence rate. An estimate is also given in Ref. 9 for the average occurrence rate of fire in any particular location, referred to as a fire cell, with a value of 7 x 10^{-4}/cell-year. In Ref. 10, an upper bound on the cable spreading room fire occurrence rate is set at 5 x 10^{-2} considering all types of observed fires that could conceivably occur in the cable spreading room based on their causes without regard to the possibility that similar fires could occur in different locations.

The Brown's Ferry fire of 1975 is the only cable spreading room fire in the data base and was caused by an event that would not be expected to occur in the HTGR design that was analyzed. The fire was started by a candle used to check for air leakage in an electrical penetration between the reactor (containment building) and the cable spreading room. In the HTGR layout (see Fig. 4-1), the cable spreading room is not adjacent to the reactor building; hence, exactly the same event could not occur. Nonetheless, the initiation of a fire caused by ignition of transient combustibles or wiring overloads cannot be ruled out.

For historical reasons, the Brown's Ferry plant was designed to fire protection and separation criteria less stringent than those used in the design of the HTGR (Regulatory Guide 1.75). A comparison of the criteria followed at Brown's Ferry with those in Regulatory Guide 1.75 is presented in Ref. 7. In view of this comparison and the above layout considerations, the occurrence rate for fires in the HTGR cable spreading room was taken to be a factor of 2 less than the value set by experience, or 1 x 10^{-3}/ reactor-year.

In the quantification of event tree probabilities, use was made of the fault tree analysis presented in Ref. 1 for each system represented in events 2 through 7 in Fig. 5-2. It was necessary to expand the detail of the fault tree to identify specific causes of failure resulting from the fires, primarily opens and shorts in the control and instrumentation cables routed through the room and shorts caused by water used to put out the fire leaking through floors into switchgear located at lower elevations. Since the causes of system failure independent of the fire were also considered in the fault trees, the analysis includes the effects of faults that compound the consequences of the fire itself.

Since the same fire, if sufficiently developed and properly located, can cause failure of more than one system along an accident sequence, the system failure probabilities in the event tree are mutually dependent. The procedure used to account for this dependency is illustrated in Fig. 5-3 and consists of linking together all the system fault trees for the sequence with an AND gate. The Boolean reduction of the sequence fault tree, or a logically equivalent procedure, will then automatically account for the dependency introduced by the same fire, causing failure of multiple components in different systems. The fault tree for sequence D is presented in the Appendix.

Another type of interdependency encountered in the quantification of event tree probabilities is the interference between the progression of the fire and efforts to put it out and operator actions required to bring the plant under control. As the fire continues to develop, faults in the electrical cables continue to occur, making it difficult for the operator to determine the plant status and carry out the procedures to repair failed systems. It was assumed that efforts to repair failed equipment cannot commence until the fire is brought under control. Linear regression analysis of the fire data, presented in Section 3, was performed to establish the relationship between the distance of fire progression and the time to put out the fire.

The fault trees were developed to a level of detail that resolved faults in specific circuits in the instrumentation and control systems. It was necessary to determine the allocation of circuits to cable trays and

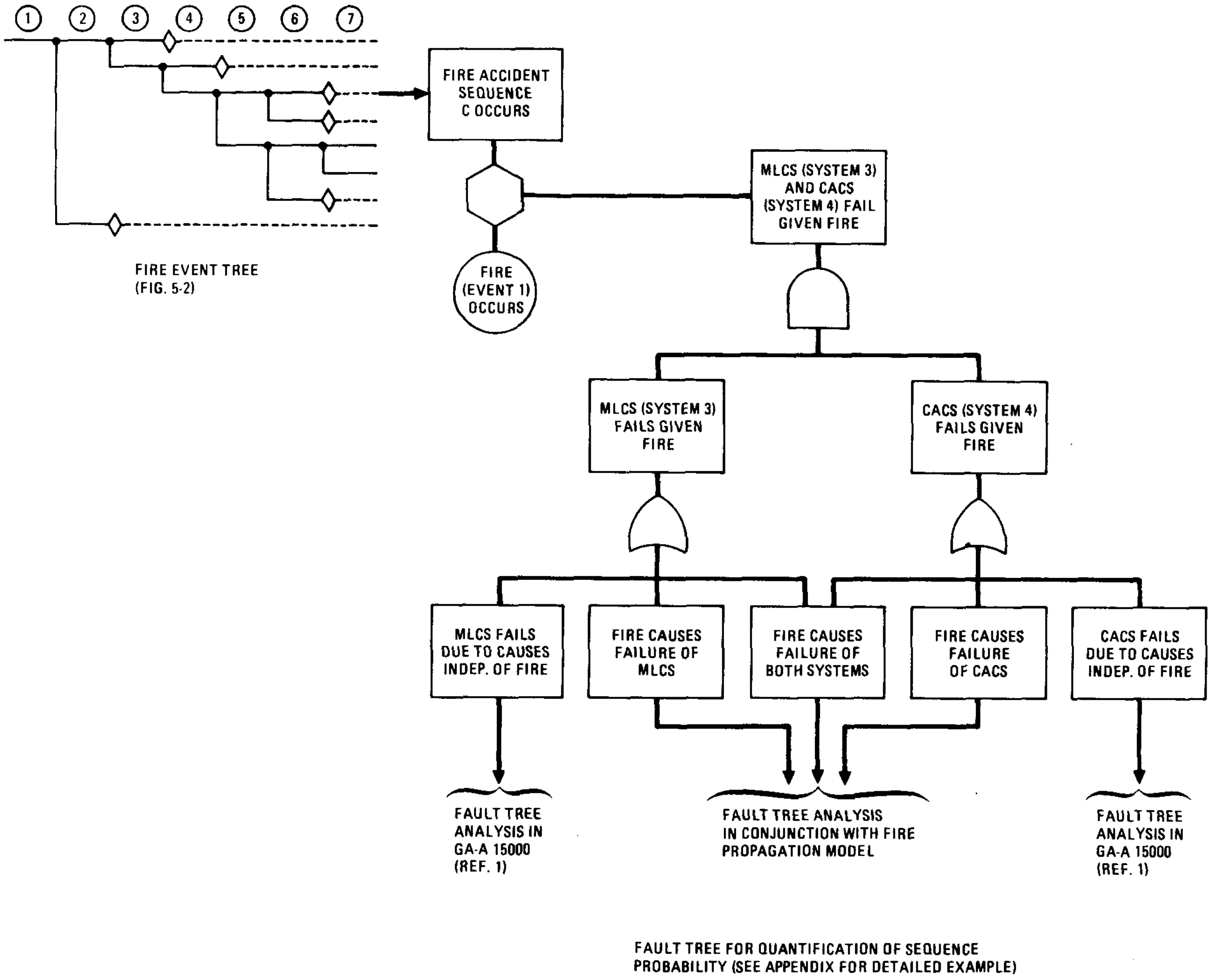


Fig. 5-3. Approach for treatment of intersystem dependencies (multisystem failures) caused by fire

the specific layout of trays within the control room. Figure 5-4 is a sketch of the general cable tray layout indicating the location of 10 sets of vertically stacked trays determined to be the most important in the fault tree analysis. The minimal cut sets in the fault trees include damage of cables in one or more of these sets indicated by letters. The cable fire propagation model described in Section 5.3 was used to determine the probability that cables in specific sets of cable trays would be damaged by the fire.

5.3. CABLE FIRE PROPAGATION MODEL

In the quantification of the fault trees, terms of the form $\Pr\{Y_1Y_2...Y_N\}$ were encountered, where $\{Y_1Y_2...Y_N\} \equiv$ specific set of N cable trays. A cable fire propagation model, described below, was developed for use with the cable tray layout in Fig. 5-4.

In the development of this model, it was assumed for simplicity that damage to cables is always complete within a set of cable trays when the fire progresses to that set. In other words, at any given time, all the cables within a given set of vertically stacked trays are either completely failed or not affected at all by the fire. It was also assumed that, in view of the possibility of transient combustibles, a fire could initiate at any random point in space within the room. Once initiated, the fire would progress a certain distance before being extinguished. In the model, the region of damage is treated as an expanding vertical cylinder in consideration of the heating of cables above the flames and the possibility of molten insulation dripping below the flames. Other considerations for the use of a cylinder model are that the cable tray sets in Fig. 5-4 are separated primarily within the same horizontal plane and that the resulting two-dimensional model is relatively simple.

The probability of failure of a specific set of N cable trays can be written as

$$\Pr\{Y_1Y_2...Y_N\} = \int f(x)\ \Pr\{Y_1Y_2...Y_N|x\}\ dx \quad , \tag{5-1}$$

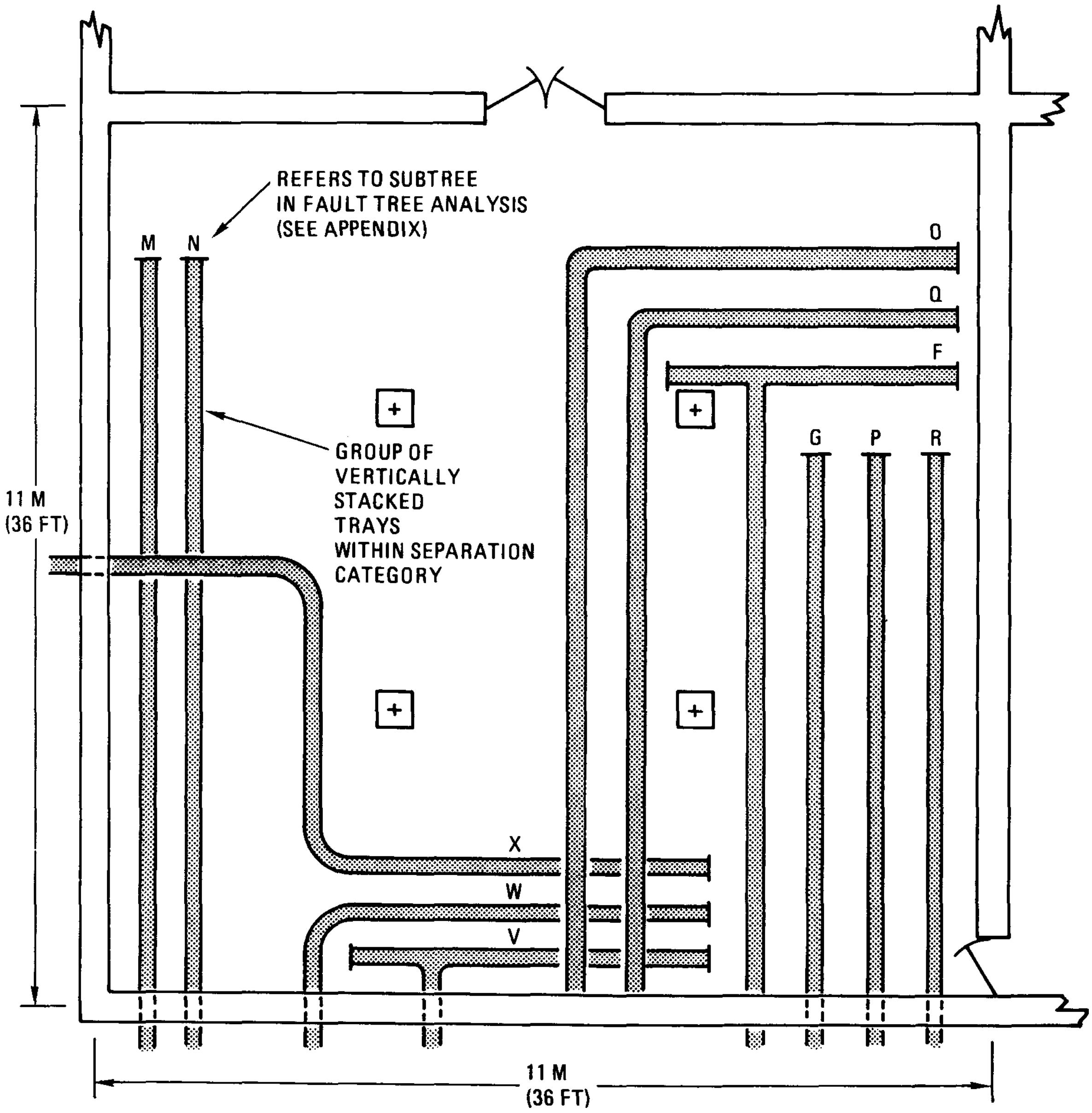


Fig. 5-4. Layout of key cable trays in cable spreading room

where $f(x)\,dx \equiv$ probability the fire progress to a distance in the interval $[x, x + dx]$,

$\Pr\{Y_1 Y_2 \ldots Y_N | x\} \equiv$ probability that a specific set of N cable trays is failed given that the fire progresses to a distance x.

In the above equation, x corresponds to the radius of the cylinder of the fire as described above. The density function of the progression distance f(x) was determined in Section 3 to be exponential, of the form

$$f(x) = \alpha e^{-\alpha x} \quad ,$$

where $\alpha = 1/\hat{x}$ and $\hat{x}$ = mean (radial) distance of fire progression.

The probability that a specific set of N cable trays will fail as a result of a fire that progresses from a random point of origin to a distance x is determined using the geometry in Fig. 5-5. A rectangle is defined having length ℓ and width w that encompasses the specific set of cable trays in question, in this case, three trays $\{Y_1, Y_2, Y_3\}$ running in parallel. The next step is to define the locus of points of origin of fires having radius x that will cause failure of all cable trays in the set $\{Y_1, Y_2, Y_3\}$. In order to cause complete failure of the set, the fire must cut across the width of the rectangle running perpendicular to the direction of cable routing. The locus, actually a rectangular box with rounded corners, is adequately approximated by the outer rectangle indicated by dotted lines. Consistent with the assumption that the fire can occur at any random point in the square room with side L, the probability of failure, $\Pr\{Y_1 Y_2 Y_3 | \ell\}$, is simply the ratio of the area of the outer rectangle to that of the room, or

$$\Pr\{Y_1 Y_2 Y_3 | x\} = \frac{\left[\ell + 2\sqrt{x^2 + (w/2)^2}\right]\left(2\,x - w\right)}{L^2} \quad . \tag{5-3}$$

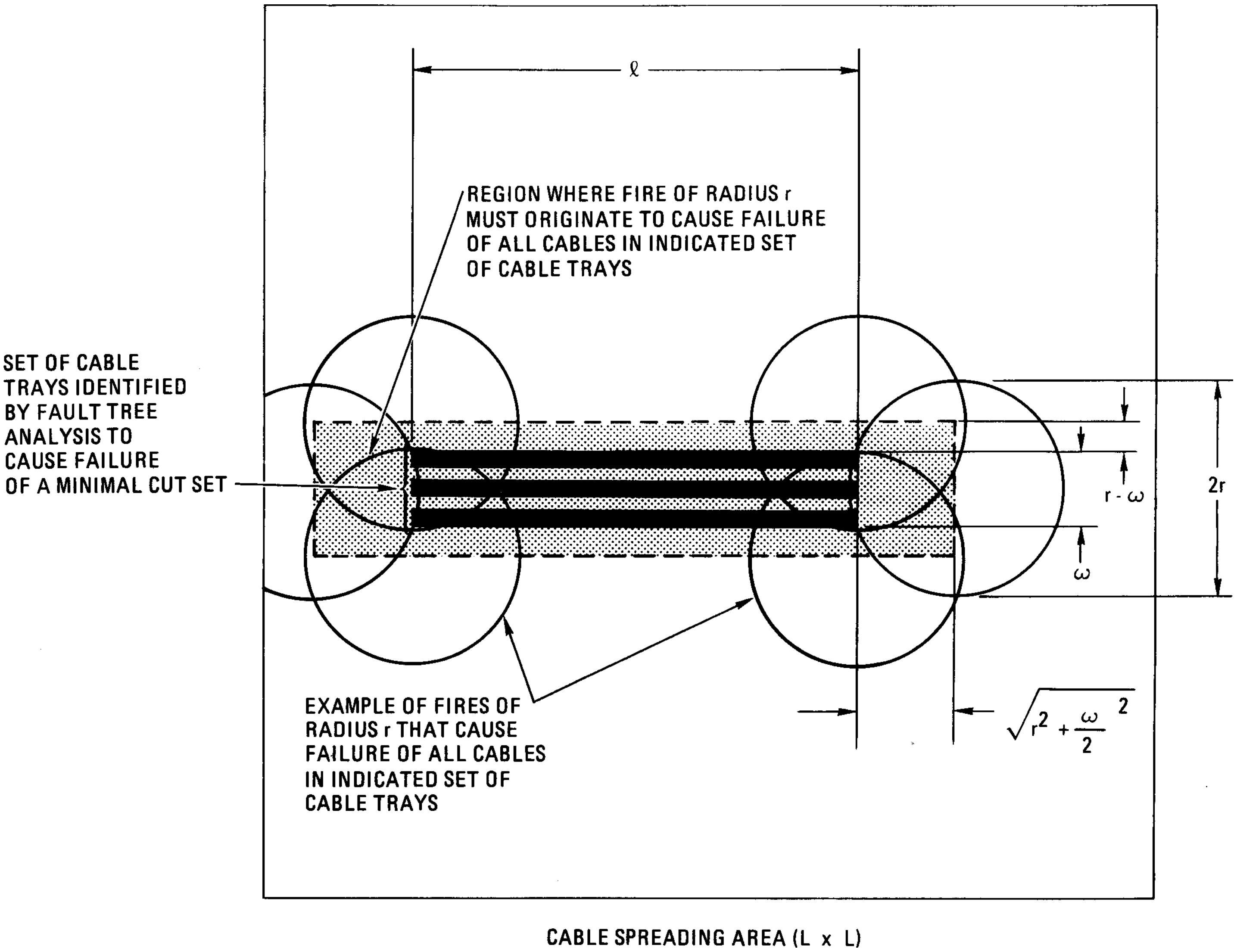


Fig. 5-5. Geometry for model for estimating probability that a fire of radius r will damage all cables in specific set of cable trays

Note that Eq. 5-3 does not hold when one or more edges of the inner rectangle are located near a wall. To simplify the integration of Eq. 5-1, Eq. 5-3 is approximated as

$$\Pr\{Y_1Y_2Y_3|\ell\} \cong \frac{(\ell + 2\ x)(2\ x - w)}{L^2} \quad . \tag{5-4}$$

Substitution of Eqs. 5-2 and 5-4 into 5-1 gives the result

$$\Pr\{Y_1Y_2Y_3\} = \int \alpha e^{\alpha x} \left[\frac{(2\ x + \ell)(2\ x - w)}{L^2}\right] dx \quad . \tag{5-5}$$

In carrying out the integration of Eq. 5-5, it is necessary to ensure that the locus of points of origin of the fire does not fall outside the room. If it is assumed in Fig. 5-5 that

$$d_1 < d_2 < d_3 < d_4 \quad , \tag{5-6}$$

then

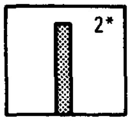
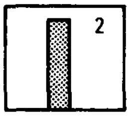
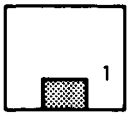
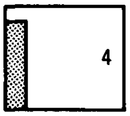
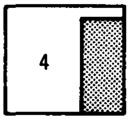
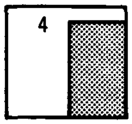
$$\begin{aligned}\Pr\{Y_1Y_2Y_3\} = &\int_w^{d_1} \alpha e^{\alpha x} \left[\frac{(2\ x + \ell)(\ 2\ x - w)}{L^2}\right] dx \\ &+ \int_{d_1}^{d_2} \alpha e^{\alpha x} \left[\frac{(x + d_1 + \ell)(2\ x - w)}{L^2}\right] \\ &+ \int_{d_2}^{d_3} \alpha e^{\alpha x} \left[\frac{2x - w}{L}\right] dx \\ &+ \int_{d_3}^{d_4} \alpha e^{\alpha x} \left(\frac{x}{L}\right) dx \quad .\end{aligned} \tag{5-7}$$

The results of the cable fire propagation model applied to key sets of cable trays determined in the fault tree analysis in the Appendix are presented in Fig. 5-6. The letters are keyed to vertically stacked sets of trays in the layout sketch of Fig. 5-4. All the cable trays (usually three) within a set indicated by a single letter belong to the same separation category. Two or more letters in a Boolean term, e.g., MN, indicate that two or more specific separation categories must be damaged by the fire. The results range from 0.04 for the case of two parallel sets of trays separated horizontally by a distance of 3 ft to 0.002 for the case of six parallel trays 17 ft apart.

It is noted that the results obtained from the cable fire propagation model are dependent on assumptions regarding the probability distribution of the distance of fire propagation. The data base regarding cable fire propagation is sparse. As more information becomes available from, for example, cable fire testing programs, these results should be reviewed. It is believed, however, that the overall treatment of fires in the model is conservative. Some of the reasons for this view are that the average (mean) fire is treated as a cylinder 6 ft in diameter running the entire height of the room, which is sufficient to cause failure of cable in two different separation categories; the damage in the fire zone is assumed to be complete; and some conservative numerical approximations were made to simplify the computations.

5.4. QUANTIFICATION OF EVENT TREE CONSEQUENCES

As explained in Section 2.5, a sufficient number of different HTGR accident sequences have been analyzed in prior studies, in particular, those reported in Refs. 1 and 20, to obviate the need for analyzing the physical processes or radiological consequences of the accident sequences in the cable fire event tree in Fig. 5-2. A key that relates the cable fire release categories to core heatup scenarios analyzed in prior studies

BOOLEAN TERM** S_i (SET OF CABLE TRAYS)	APPROXIMATE LAYOUT OF CABLE TRAY SET	DIMENSIONS OF RECTANGLE CONTAINING CABLE TRAY SET		PROBABILITY OF COMPLETE FAILURE OF CABLE TRAY SET $Pr [S_i \mid F]$	
		ℓ (FT)	ω (FT)	α = 0.06 FT^{-1}*** UPPER BOUND	α = 0.3 FT^{-1} BEST ESTIMATE
OQ, QF		32	2.5	0.3	0.04
OQF		32	5.0	0.2	0.02
VWX		15	5	0.2	0.01
MN		34	25	0.2	0.04
FGPQO, FGPQR		32	15	0.2	0.006
FGOPQR		32	17	0.1	0.002

* NUMBER DENOTES A PARTICULAR VARIATION OF THE MODEL FOR INCLUDING VARIATIONS IN GEOMETRY.

** EACH LETTER DENOTES A PARTICULAR TRAY WITHIN THE SHADED RECTANGLE DAMAGED BY THE FIRE.

*** $\alpha = \frac{1}{\hat{r}}$ WHERE $\hat{r}$ IS THE MEAN RADIUS OF FIRES IN THE CABLE SPREADING AREA.

Fig. 5-6. Results of cable tray model applied to sets of cable trays in the fault tree for sequence F-D (failure of MLCS, CACS, and CIS) judged to have similar consequences is presented in Table 5-3. The results in terms of the timing of physical processes and radiological consequences are summarized below.

The timing of key events along the accident sequences in the fire event tree is given in Table 5-4. At the level of probability of the core heatup sequences, 1 x 10^{-5}/reactor-year and below, it is likely that the fire propagates a sufficient distance to cause failure of most safety-related control and instrumentation. This causes failure of the core auxiliary cooling system, main loop cooling system, reserve shutdown system, and containment isolation valves and simulates a reactor trip signal. This scale of fire would be a minimum of 30 ft in diameter and would take at least 3 hours to bring under control according to the time-distance correlation obtained in Section 3 from the data base. Since repair efforts were assumed not to commence until after the fire is out and since restart of either cooling system must occur before 3 hours to be successful, at 3 hours the loss-of-forced-cooling situation becomes permanent, resulting in a core heatup transient with a negligible chance of repair.

Because of the low core power density, high heat capacity, and good high-temperature performance of the fuel, significant release from the core does not begin until about 7 hours and proceeds slowly thereafter. At about 11 hours, the PCRV pressure reaches the relief valve setpoint, the relief valves open and fail to close due to excessive temperature, and the primary coolant inventory is released into the containment building. Although the containment isolation valve circuits are also likely damaged by the fire, efforts to close the valves manually before PCRV blowdown are successful for all but two of the sequences, F-D and F-G.

At 20 hours, 35% of the most volatile radioactive gases are still inside the core, and not until 60 hours does significant release of less volatile elements such as cesium and strontium begin to occur. At 90 hours, the thermal barrier in the PCRV top head separating the PCRV liner from the

TABLE 5-3
KEY RELATING FIRE SEQUENCES TO SCENARIOS ANALYZED IN PRIOR STUDIES WITH SIMILAR CONSEQUENCES

Event Tree Sequence	Release Category	Core Heatup Scenario Analyzed Previously With Similar Consequences	Reference
F-G	FCH-0	Vented-B	Containment Options Study, GA-A14970 (Ref. 20)
F-D	FCH-1	V	Phase II Risk Assessment Study, GA-A15000 (Ref. 1)
F-F	FCH-4	F	Phase II Risk Assessment Study, GA-A15000 (Ref. 1)
F-E	FCH-5	G	Phase II Risk Assessment Study, GA-A15000 (Ref. 1)
F-C	FCH-6	H	Phase II Risk Assessment Study, GA-A15000 (Ref. 1)

TABLE 5-4
TIMING OF KEY EVENTS ALONG ACCIDENT SEQUENCES INITIATED BY CABLE SPREADING ROOM FIRES

Event	Time (hr)
Fire in cable spreading area occurs	0
Fire causes reactor trip; failure of MLCS, CACS, CIV, and RSS; core heatup begins	∿0
Fire brought under control; efforts to repair CACS or MLCS prevented by fire; core heatup becomes permanent	3
Significant release of volatile nuclides (e.g., Kr, Xe, I) from core begins	7
Primary coolant pressure reaches 780 psia; PCRV relief valves lift, fail to reclose due to excessive valve temperatures	10
Containment isolation valves close (except for FCH-0 and FCH-1)	10
Average fuel temperature reaches 2000°C (3600°F); most fuel particle coatings failed, ∿65% of Kr and Xe nuclides released from core	20
Significant release of Cs and Sr from core begins	60
PCRV top head thermal barrier begins to fall off; heat transfer to PCRV liner cooling system (LCS) increases	90
All events beyond here apply only to categories FCH-0, FCH-4, and FCH-5	
PCRV LCS fails due to extreme temperature environment	90
PCRV top head concrete begins to decompose	100
PCRV side wall concrete begins to decompose	130
Earliest time global water gas flammability conditions met in containment, overpressure may occur if ignition occurs.	190
Sufficient noncondensable gases generated to overpressurize containment if containment not failed previously	290

upper plenum directly above the core begins to fall off due to excessive temperatures. The removal of thermal insulation causes a sharp increase in the heat transfer to the liner cooling system. Depending on the sequence of thermal barrier components coming down, the liner cooling system would either fail as a result of water boiling in the cooling tubes behind the liner or survive the high temperatures and eventually terminate the heatup transient. More details can be found in Ref. 1.

Failure of the liner cooling system causes thermal decomposition of PCRV concrete, resulting in the production of gases including CO_2 and water vapor. Some of the water vapor reacts with the core graphite at high temperature, producing CO and H_2. Continuation of the heatup transient beyond this point would cause continued production of these noncondensable gases, which can result in overpressurization of the containment building, at times ranging from about 200 to 300 hours, depending on whether flammable gas mixtures of water gas (H_2 + CO) are ignited or not. No consideration was given in the development of these scenarios to external actions that could be taken in the time scale of days to weeks to further mitigate their consequences, including steps to prevent or filter the release after containment failure.

The methodology for simulating the physical processes of HTGR accidents and associated radionuclide transport is described in Ref. 1. Of the 187 radionuclides that are tracked in the consequence calculation, about five key nuclides were selected for use in Table 5-5 to illustrate the radioactivity transport characteristics for each of the release categories FCH-0 through FCH-6. The greatest quantities released of each key nuclide are associated with FCH-0 and the least with FCH-6. The release categories are listed in order of I-131 release, which turns out to be the same order as the radiological doses described below. The only difference among the release categories in the PCRV release fractions

TABLE 5-5
TRANSPORT AND ENVIRONMENTAL RELEASE OF KEY RADIONUCLIDES DURING FIRE-INDUCED CORE HEATUP ACCIDENTS

Release Category	Fraction of Core Release Escaping PCRV				Time of Significant Release to Environment (hr)	Fraction of Initial Inventory Released to Environment[(a)]				
	I	Te	Cs	Sr		Xe-133	I-131	Te-132	Cs-134	Sr-90
FCH-0	5×10^{-2}	1×10^{-2}	9×10^{-3}	1×10^{-3}	11-720	0.31	~0.006	6×10^{-3}	0.04	1×10^{-4}
FCH-1	3×10^{-2}	8×10^{-3}	8×10^{-4}	1×10^{-4}	11-720	0.13	5×10^{-3}	2×10^{-4}	2×10^{-6}	1×10^{-7}
FCH-4	5×10^{-2}	1×10^{-2}	9×10^{-3}	1×10^{-3}	190-720	0.36	3×10^{-4}	8×10^{-7}	6×10^{-5}	1×10^{-5}
FCH-5	5×10^{-2}	1×10^{-2}	9×10^{-3}	1×10^{-3}	290-720	0.20	1×10^{-4}	2×10^{-7}	1×10^{-7}	$<10^{-7}$
FCH-6	3×10^{-2}	8×10^{-3}	8×10^{-4}	1×10^{-4}	11-720	3×10^{-3}	5×10^{-7}	9×10^{-8}	$<10^{-7}$	$<10^{-7}$

(a) Inventories (Ci): Xe-133 = 2×10^{8}
I-131 = 8×10^{7}
Te-132 = 1×10^{8}
Cs-134 = 1×10^{7}
Sr-90 = 7×10^{6}

is determined by whether or not PCRV concrete decomposition occurs. The generation of gases associated with this process increases the PCRV egress rate.

Point estimates of the integrated radiological dose to an individual 2500 meters directly downwind of the release point are presented in Table 5-6 for each release category. Included are the external whole body gamma and inhalation doses to the thyroid, bone, and lung. Uncertainties in these estimates were quantified in terms of probability distributions that account for variations in physical processes such as atmospheric dispersion and uncertainties in input data used by the accident simulation models. These probability distributions were obtained with use of a Monte Carlo error propagation procedure in conjunction with a simplified analytical model of accident consequences that approximates a series of computer programs which model, in detail, the physical processes of the accident and the transport of radioactivity. The independent variables of the simplified model listed in Table 5-7 were assigned appropriate uncertainty distributions. Monte Carlo sampling of these distributions was performed using the STADIC (Ref. 19) code to obtain a complementary cumulative distribution function (CCDF) for the consequences of each release category. As shown in Fig. 5-7, an overall risk assessment of the fire event tree in Fig. 5-2 is obtained as a summation of the above CCDF curves weighted by the probability per year of each release category. The consequence unit of Fig. 5-7, the health effects dose, is related to the total number of latent cancer fatalities for a representative U.S. site integrated over the 10- to 40-year period following the accident:

1 heat effects rem = 0.13 latent cancer fatality.

A comparison of these results with those obtained for other accidents analyzed in prior studies is given in Section 6.

TABLE 5-6
POINT ESTIMATES OF 2500-M DOWNWIND DOSES (REM) RESULTING FROM FIRE-INDUCED CORE HEATUP ACCIDENTS

Release Category	Integrated Radiological Exposures to Hypothetical Individual 2500 M Downwind From Release (rem)			
	External Whole Body Gamma Dose	Inhalation Dose		
		Thyroid	Bone	Lung
FCH-0	16	2300	12	60
FCH-1	6.6	1700	0.2	26
FCH-4	5.4	45	2.6	2.2
FCH-5	3.1	30	0.003	0.01
FCH-6	0.05	0.12	0.0004	0.003

TABLE 5-7
KEY INDEPENDENT VARIABLES INCLUDED IN CONSEQUENCE UNCERTAINTY ASSESSMENT

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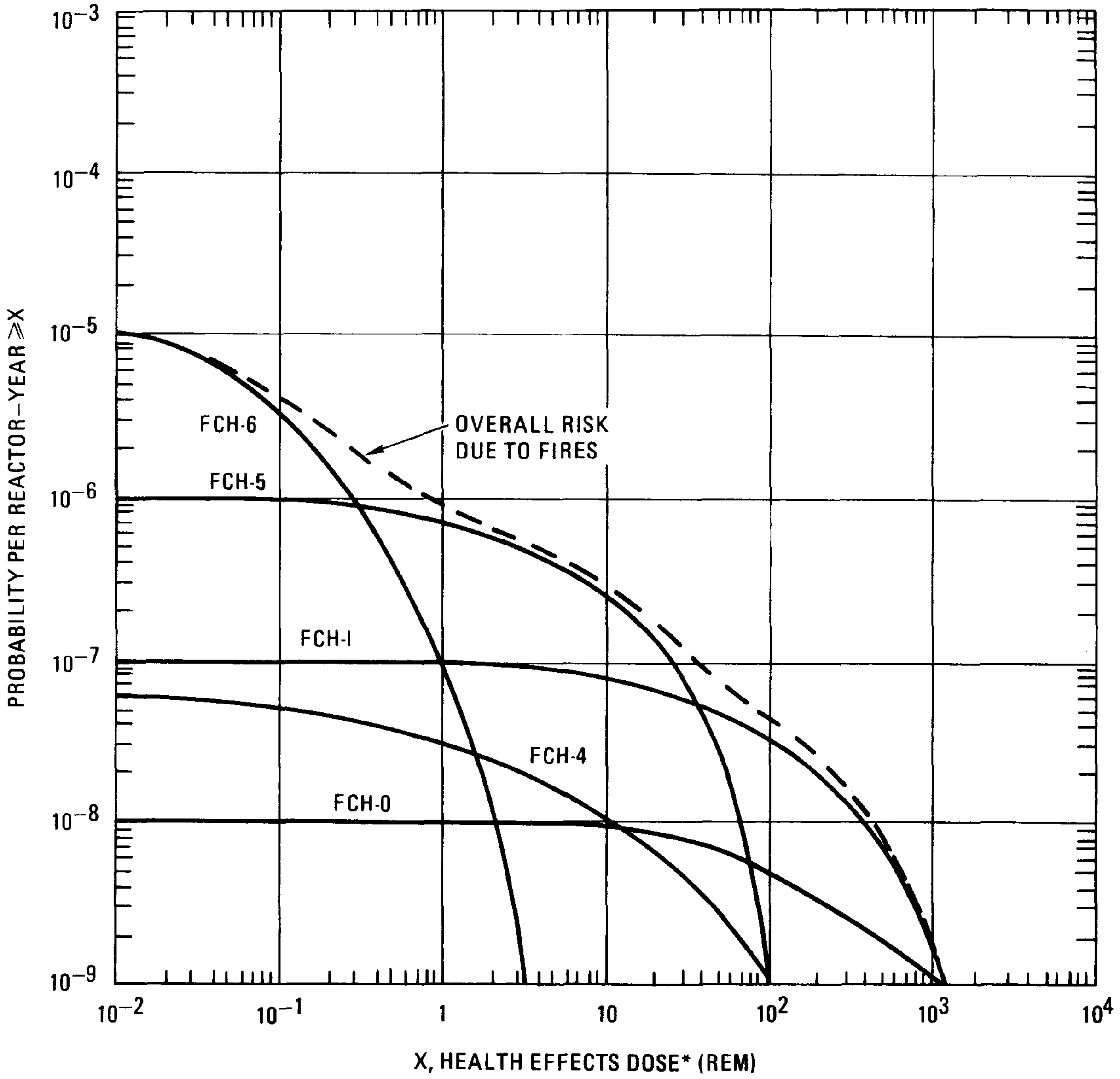
1. Times of release from the core for different fission products
2. Containment natural deposition and fallout rates
3. PCRV non-plateout escape fraction
4. a. Time of ignition of water gas (flammability sequences only)

   b. Containment failure pressure (gas accumulation sequences only)
5. PCRV egress rate before concrete degradation
6. PCRV side cavity flow rate
7. Containment leak rate before failure
8. Helium diluent addition
9. Time of start of top heat concrete degradation
10. Concrete spalling rate
11. Containment cleanup startup time
12. Fraction of containment atmosphere release at time of containment failure
13. Atmosphere dispersion factor
14. Dose effectivities



* LINEAR COMBINATION OF EXTERNAL AND INHALATION DOSES PROPORTIONAL TO THE NUMBER OF LATENT CANCER FATALITIES 1 REM = 0.13 LATENT CANCER FATALITIES FOR A REPRESENTATIVE U.S. SITE

Fig. 5-7. Risk of HTGR accidents due to fires with contributions of release categories

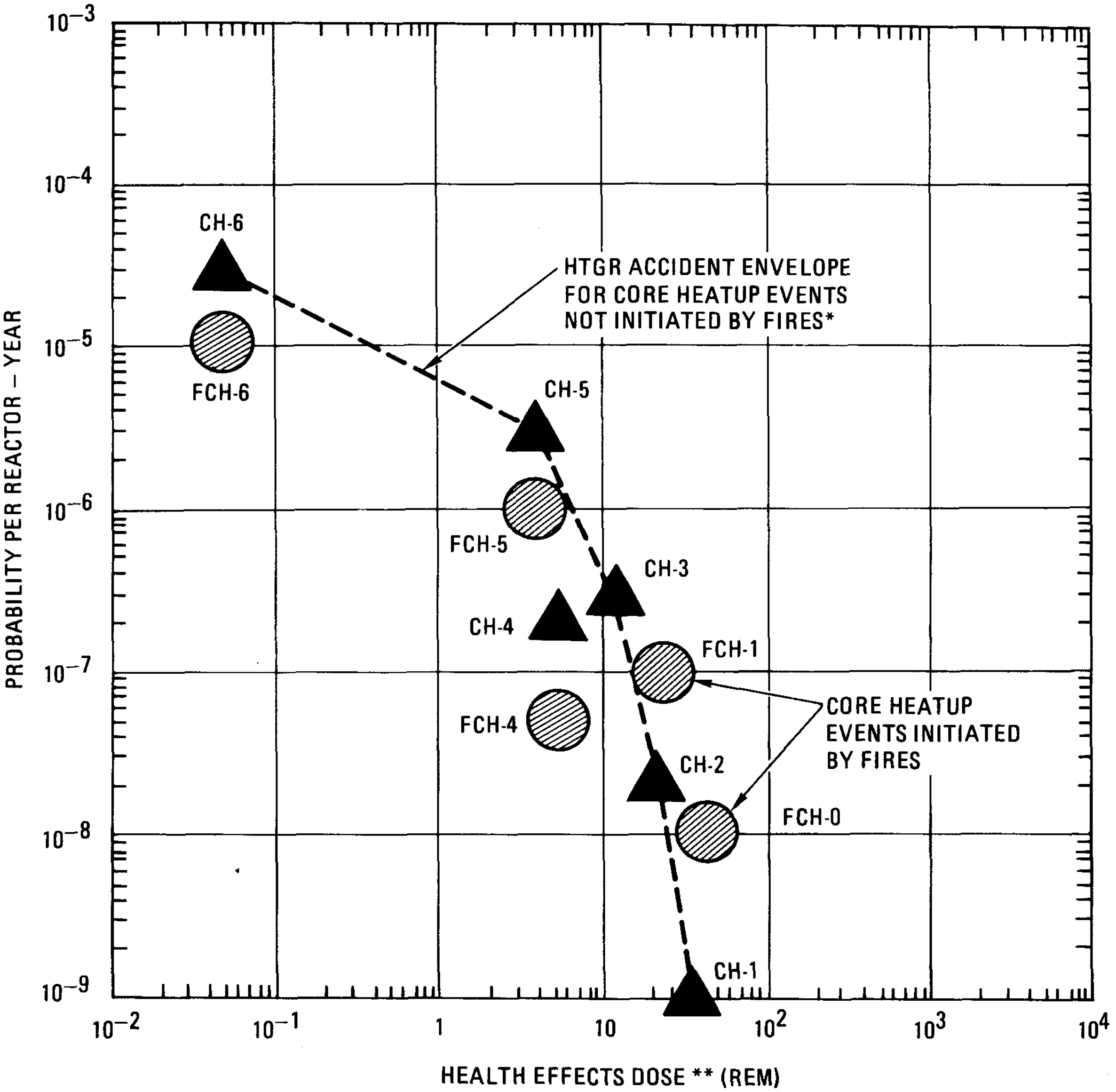
6. RESULTS AND CONCLUSIONS

In Section 6.1, the results of the cable room fire risk assessment are compared with the results from prior studies to give an indication of the relative contribution of fires to the overall HTGR risk assessment. Some comments as to how the results might have changed if more stringent separation and fire protection criteria had been imposed are given in Section 6.2. The conclusions of this study and recommendations for further work in this area are presented in Sections 6.3 and 6.4, respectively.

6.1. RISK CONTRIBUTION OF CABLE SPREADING ROOM FIRE

Point estimates of the probabilities and consequences of the fire release categories in comparison with the results for other initiating events analyzed for the HTGR (Ref. 6-1) are presented in Fig. 6-1. Release categories CH-1 through CH-6 represent the total contribution from about 20 initiating events resulting in core heatup accidents. The fire release categories FCH-4, FCH-5, and FCH-6 were found to have a probability of occurrence about a factor of 3 less than that of their counterparts having the same level of consequences. However, the remaining fire release categories, FCH-1 and FCH-0, fall outside the risk envelope established for the HTGR. Therefore, it has been determined that among those hypothetical accidents that have been analyzed for the HTGR, the cable spreading room fire results in dominant risk accident sequences in the low-probability, high-consequence region of the accident spectrum.

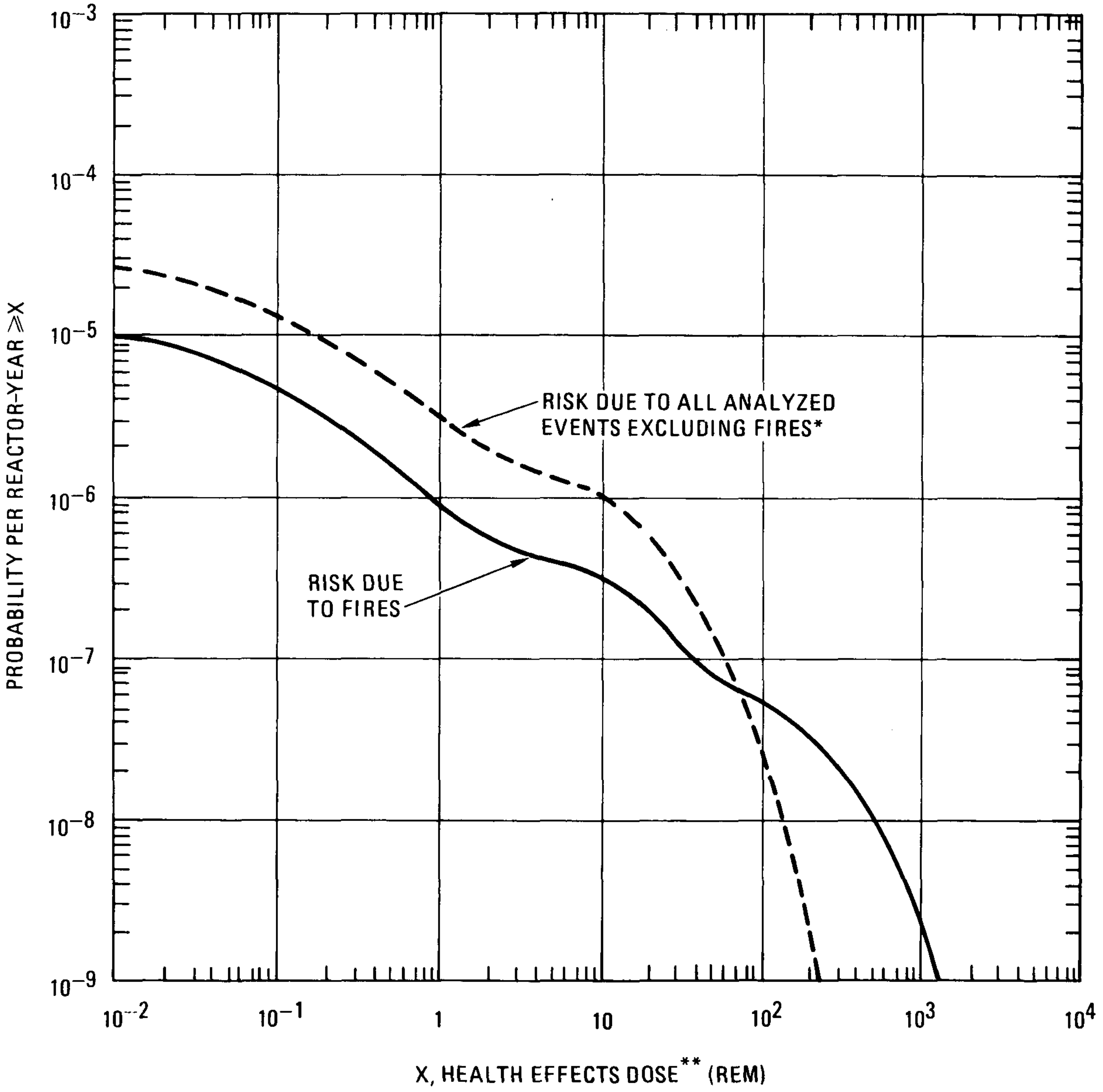
A similar result is obtained when uncertainties in the consequence predictions are taken into account in the cumulative risk curves of Fig. 6-2. At accident probabilities less than 10^{-7}/reactor-year, risks assessed



* TAKEN FROM REPORT GA-A15000 (REF. 1)

** LINEAR COMBINATION OF EXTERNAL AND INHALATION DOSES PROPORTIONAL TO THE NUMBER OF LATENT CANCER FATALITIES 1 REM = 0.13 LATENT CANCER FATALITIES FOR A REPRESENTATIVE U. S. SITE.

Fig. 6-1. Point estimates of probabilities and consequences of core heatup accidents initiated by fires and other causes



*TAKEN FROM REPORT GA-A 15000 (REF. 1)

** LINEAR COMBINATION OF EXTERNAL AND INHALATION DOSES PROPORTIONAL TO THE NUMBER OF LATENT CANCER FATALITIES 1 REM = 0.13 LATENT CANCER FATALITIES FOR A REPRESENTATIVE U.S. SITE

Fig. 6-2. Comparison of risk of HTGR accidents due to fires with that due to other initiating events for fires exceed those due to all other analyzed events even though the probability of a fire resulting in core heatup is less by a factor of 3. This result shows the importance of fires as a common cause failure of multiple components and systems, as discussed in Section 5.1. Although other initiating events have been identified that result in a core heatup accident at a higher level of probability, the cable spreading room fire was found to have a much greater probability of resulting in core heatup combined with containment failure. As a result, the contribution of the fire sequences to the overall risk assessment for the HTGR is small in the high-probability, low-consequence end of the accident spectrum and dominant at probability levels less than about 10^{-7}/reactor-year.

6.2. ROLE OF FIRE PROTECTION CRITERIA

The HTGR design analyzed in this study was completed in 1975, well after the Brown's Ferry plant was designed and licensed but before enhanced fire protection criteria for nuclear power plants were proposed. Although Regulatory Guide 1.75 on separation of electrical systems was followed in the HTGR cable tray layout, more stringent requirements embodied in Regulatory Guide 1.120 (draft) were not yet in effect. In particular, the primary means of meeting separation requirements in the HTGR cable spreading room was the provision for a minimum horizontal and vertical spacing between cable trays in different redundant divisions or separation categories. Although a greater distance of fire propagation is necessary to affect multiple separation categories in this case, in comparisons with Brown's Ferry, physical fire barriers were not provided in the cable spreading room.

By contrast, the more recent Regulatory Guide 1.120 (draft) states:

> "A separate cable spreading room should be provided for each redundant division."

It is clear that the risk associated with cable spreading room fires would have been considerably less if separate rooms had been provided for each of the safety-related separation categories of cables. It is difficult to estimate quantitatively how much the risk would have been reduced in the absence of a specific design and layout that satisfies this criterion.

Additional aspects of Regulatory Guide 1.120 lead to improvements in fire detection and suppression systems and other areas of plant design of importance to the initiation and progression of fires. Since fire initiation and progression characteristics were modeled in this study primarily on the basis of experience data from plants that were designed and built without the benefit of Regulatory Guide 1.120, it seems reasonable to assume that the probability of a given size fire should be smaller in more modern plants. Hence, if the HTGR analyzed in this study were redesigned to meet the requirements of Regulatory Guide 1.120 (draft), cable spreading room fires would have made a smaller impact on the HTGR risk assessment than that indicated in Figs. 6-1 and 6-2. Unfortunately, however, some locations remain where it is impractical to employ physical barriers among redundant divisions, for example, the control room.

6.3. CONCLUSIONS

The primary objectives of this study, to develop a methodology for the risk assessment of nuclear reactor accidents initiated by major fires and to apply these methods to an HTGR plant, have been met. The data compiled and analyzed and the methods developed may be of use in other studies such as the fire hazard analyses mentioned in Regulatory Guide 1.120 (draft).

Methods for the identification of important potential fire locations were developed and successfully applied to an HTGR plant. A quantitative method was used to screen the plant layout to determine that all but five locations could be ruled out as having potential for producing dominant risk accident sequences. A qualitative procedure similar to Failure Modes and Effects Analysis was used to evaluate the five candidate locations to determine that two were more important than the other three. One of these two locations, the cable spreading area, was selected for event tree/fault tree analysis.

The results of the risk assessment of potential cable spreading room fires have shown the importance of fires as a common cause failure of multiple components and systems. Although the probability of a fire leading to a core heatup accident was found to be somewhat less, the probability of core heatup combined with containment failure was estimated to be two orders of magnitude greater than that assessed previously for all other initiating events analyzed for the HTGR. The lesson to be learned from this result regarding the treatment of common cause failures is that, in addition to the problem of estimating system failure probabilities, for which the Beta Factor method (Ref. 15) was developed, events that have the potential for causing failure of multiple systems should be treated separately as initiating events. A separate event tree for this type of common cause failure is necessary to determine its influence on the development and progression of accident sequences as well as to obtain realistic estimates of the accident probabilities.

Hypothetical accident sequences initiated by fires in the cable spreading room of an HTGR plant were found to make a small contribution to the overall HTGR risk assessment at accident probabilities greater than about 10^{-7}/reactor-year. At extremely low probabilities, less than 10^{-7}/reactor-year, fire-induced accidents were found to dominate the risk assessment obtained previously of a wide spectrum of initiating events (Ref. 1). It is clear that if the plant were redesigned in accordance with some of the new fire protection criteria proposed in Regulatory Guide 1.120, such as the use of separate cable spreading rooms for each redundant division of cables, the estimated risk would be lower.

6.4. RECOMMENDATIONS FOR FURTHER STUDY

The numerical results obtained in the quantification of event tree probabilities are dependent on the progression characteristics assumed for cable fires. Although the cable fire propagation model utilized experience data to quantify progression characteristics, the data available with cable tray fires per se is sparse. Hence, the results should be reviewed if and when additional relevant data such as those provided by additional operating experience and tests become available.

A natural extension of this work would be the use of the methodology as a tool for evaluating the safety characteristics of alternative fire detection and suppression systems as well as alternative layouts of plant equipment. Certain aspects of the methodology would also be useful in the study of other types of location-dependent common cause failures.

ACKNOWLEDGMENTS

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APPENDIX
FAULT TREE ANALYSIS OF FIRE SEQUENCE F-D

The fire location analyzed in this study is the cable spreading area which contains, primarily, electrical cables for the numerous instrumentation and control circuits in the plant protection and control systems. The fault trees for the sets of systems identified in the fire event tree of Fig. 5-2 were developed, starting from the trees developed for the same systems in Ref. 1. Then, because of the specific location of the fire, the trees were developed to a high degree of detail so that faults in specific circuits in control and protection systems could be identified. The fault trees were developed in such a fashion that multiple failures in circuits caused by the same fire, i.e., a single fire in one or more adjacent cable trays, could be identified. This required knowledge of the details of the cable tray layout and the allocation of circuits to the cable trays. The approximate layout of cable trays is shown in Fig. 5-4.

The fault tree for one of the most important accident sequences, F-D (release category FCH-1), is shown in Figs. A-1 through A-7. This sequence was found to dominate the overall HTGR risk assessment curve at accident probability levels less than 10^{-7}/reactor-year and to involve failure of both core cooling systems, the MLCS and CACS, and the containment isolation valves. The minimal cut sets of the tree are numerous and have been omitted for brevity. All the input probability input data other than that obtained from the fire propagation model, summarized in Table 5-2, was taken directly from Ref. 1.

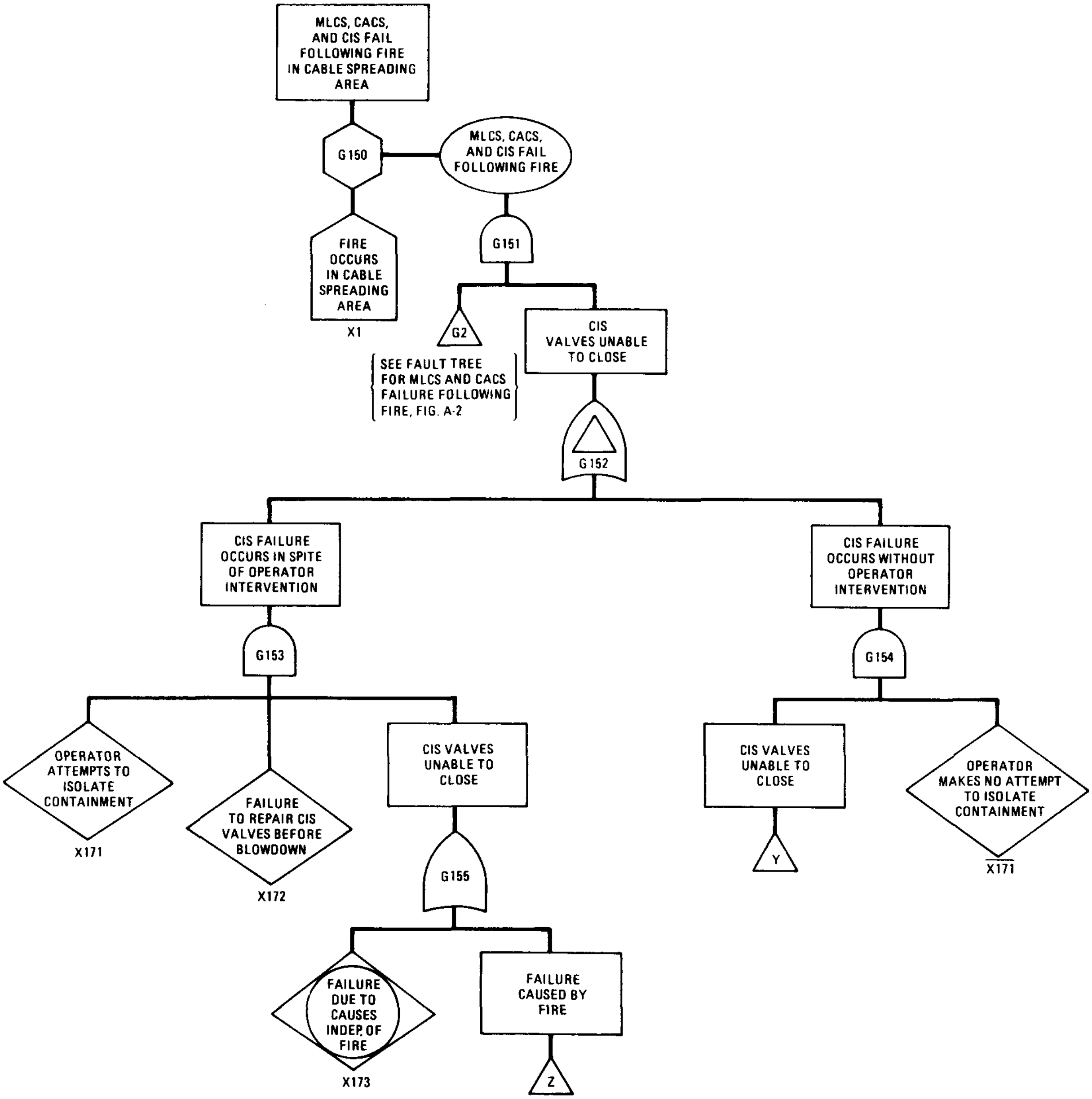


Fig. A-1. Fault tree for sequence F-D

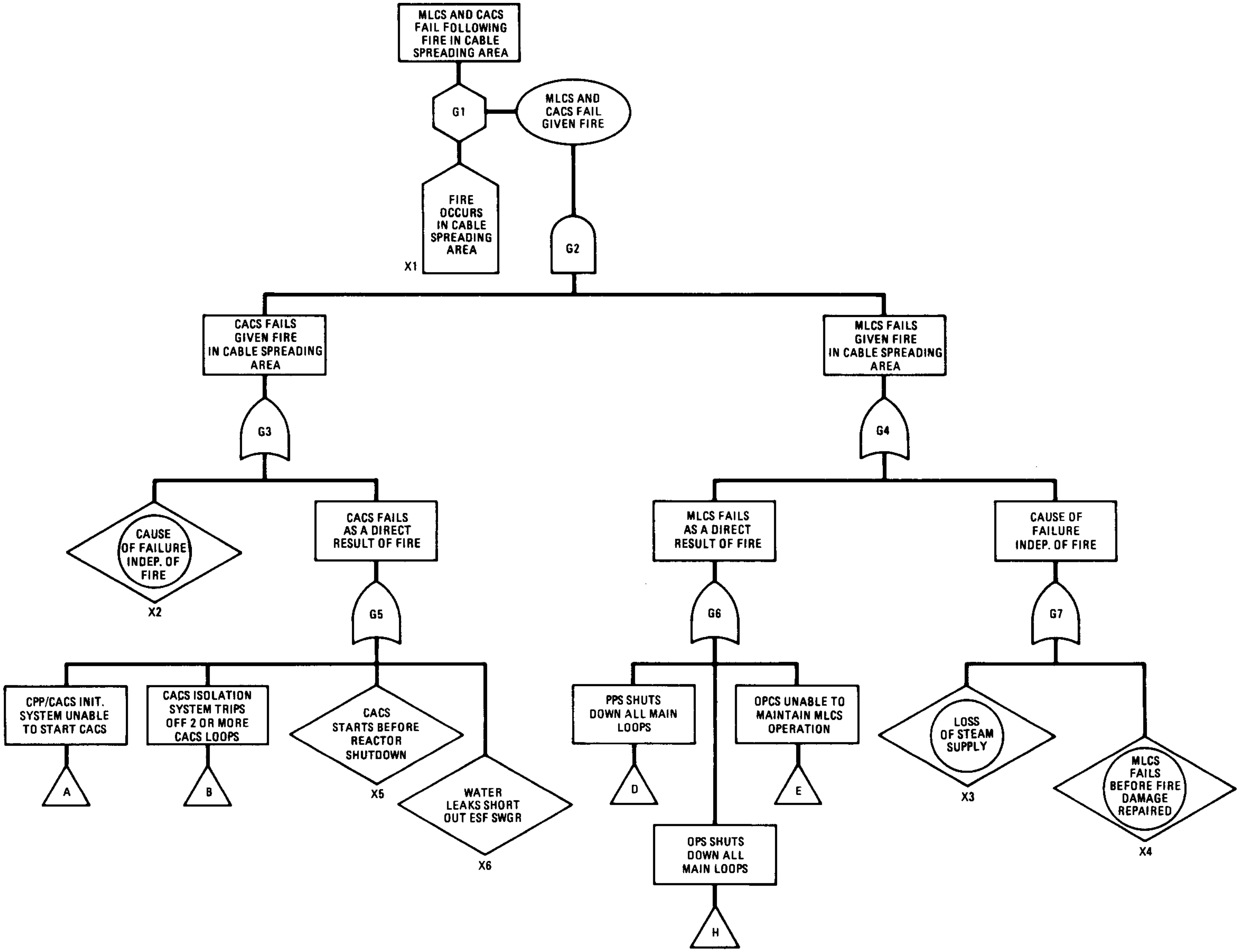


Fig. A-2. Fault tree for sequence F-C and continuation of Fig. A-1

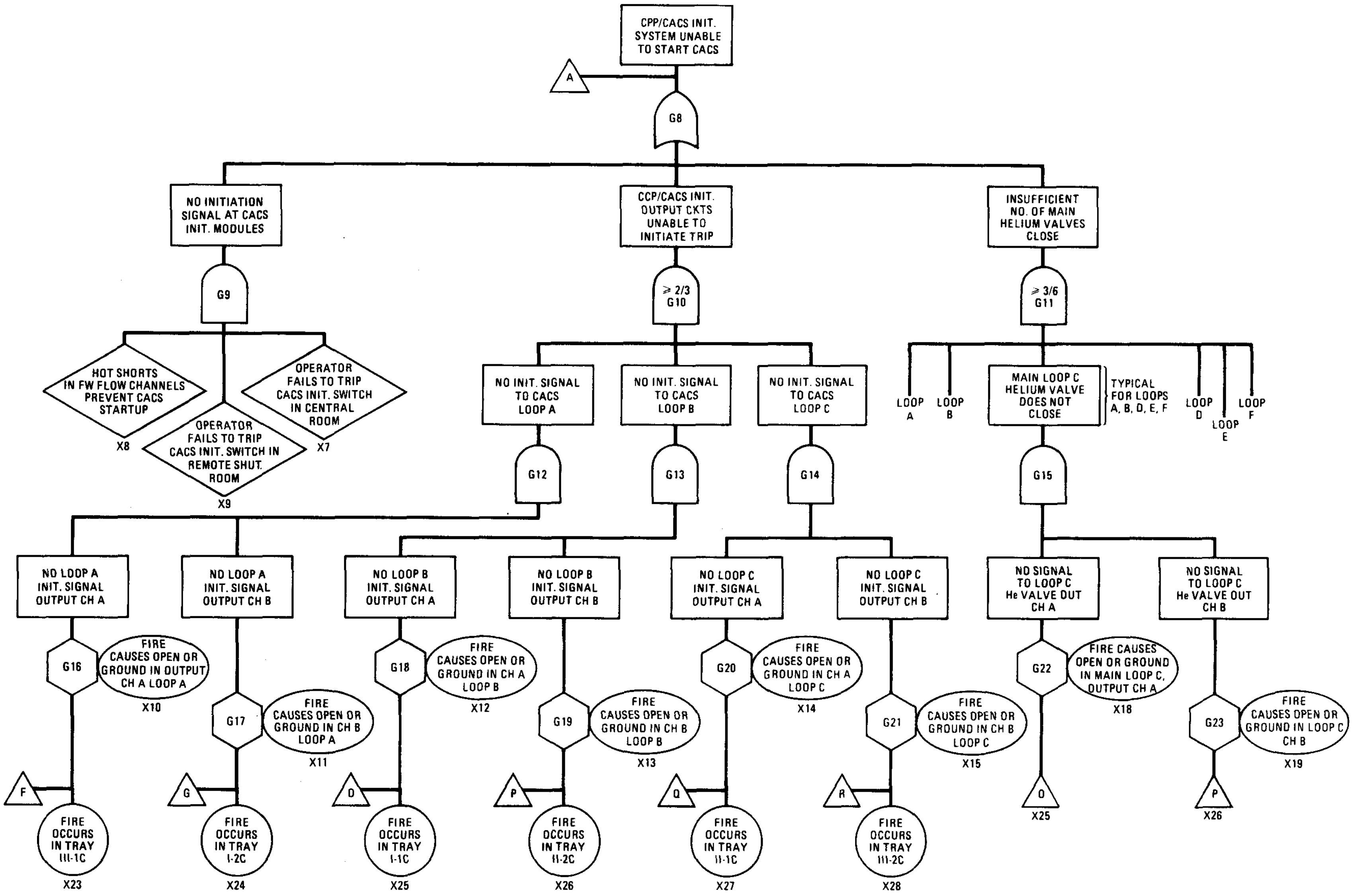


Fig. A-3. Subtree for failure of CACS initiation system

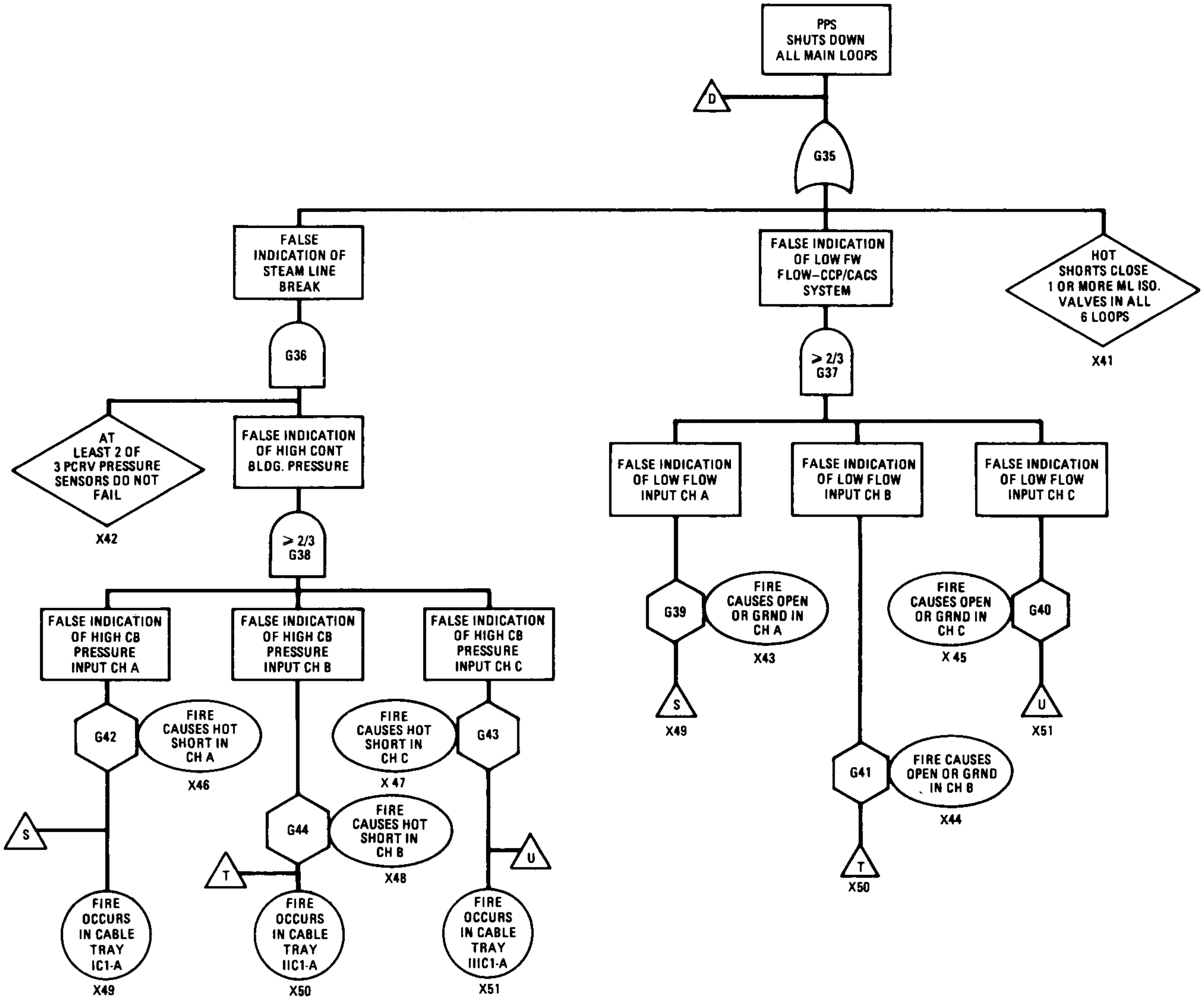
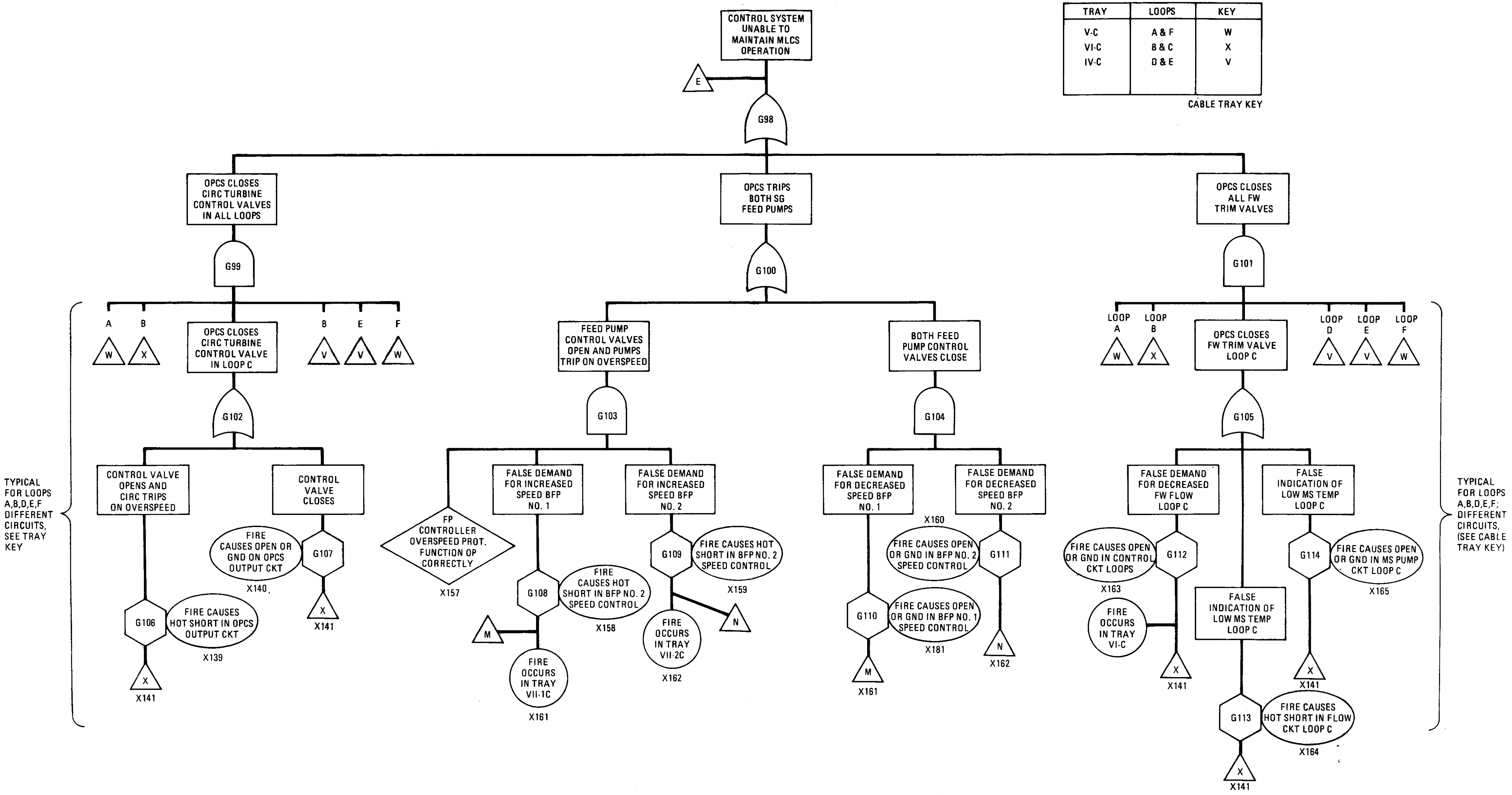


Fig. A-4. Subtree for protection system shutdown of main cooling loops



TRAY	LOOPS	KEY
V-C	A & F	W
VI-C	B & C	X
IV-C	D & E	V

CABLE TRAY KEY

Fig. A-5. Subtree for plant control system failure to keep MLCS on line

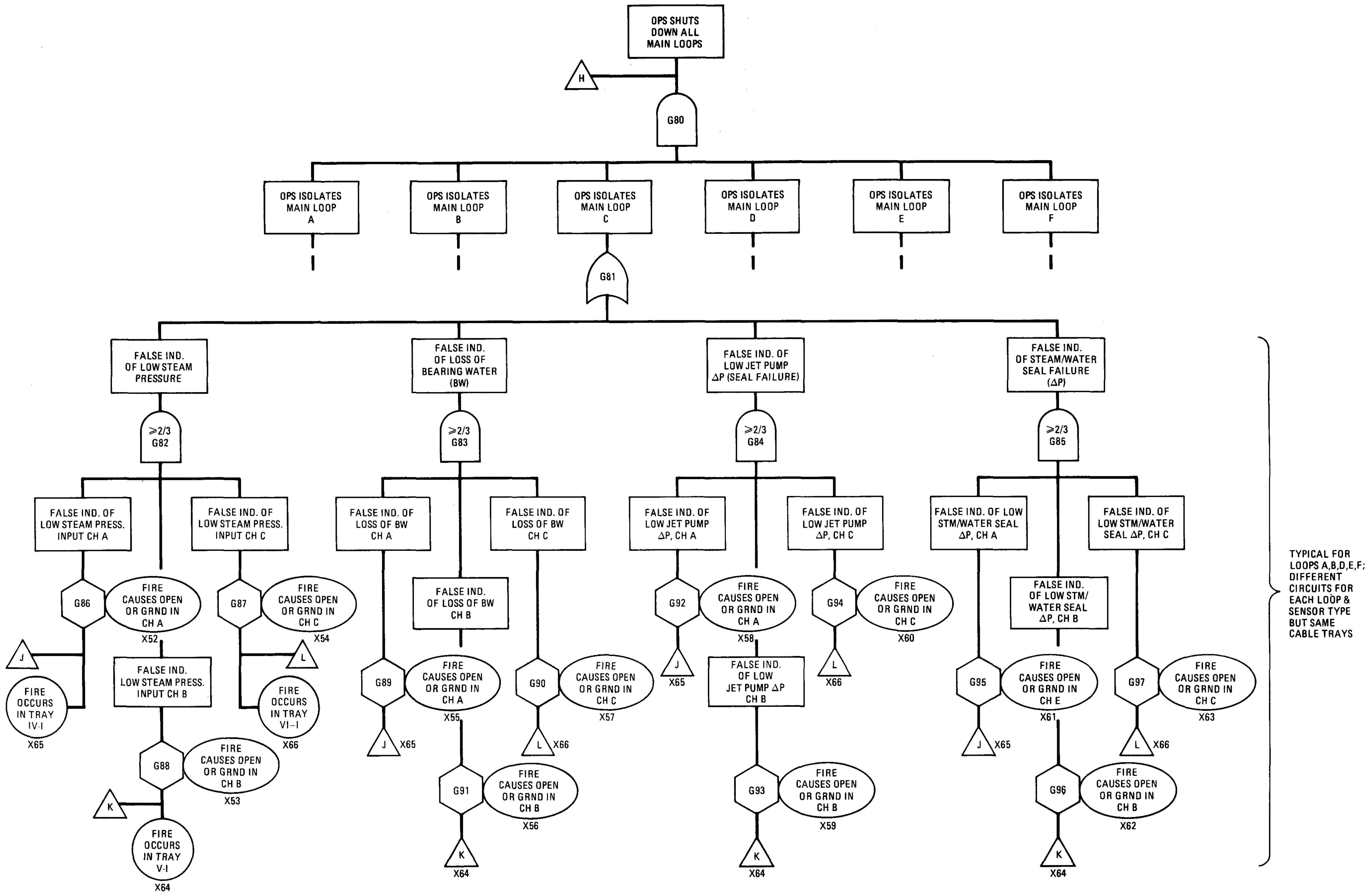


Fig. A-6. Subtree for operational protection system shutdown of MLCS

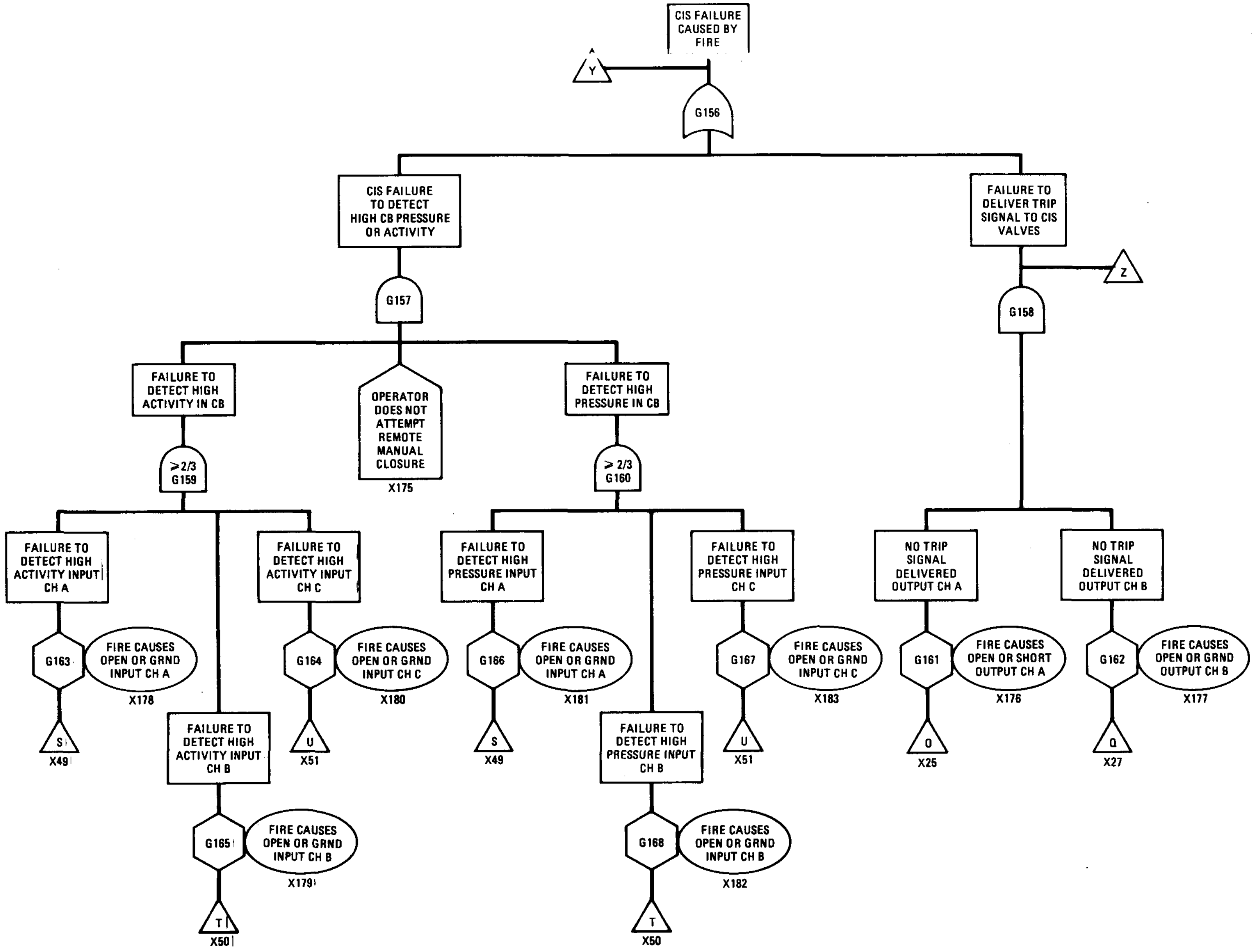


Fig. A-7. Subtree for failure of containment solution values