

Safety Reports Series

No.10



TREATMENT OF
INTERNAL FIRES
IN
PROBABILISTIC
SAFETY
ASSESSMENT
FOR NUCLEAR
POWER PLANTS

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SAFETY ASSESSMENT
FOR NUCLEAR POWER PLANTS

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IN PROBABILISTIC
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FOR NUCLEAR POWER PLANTS

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FOREWORD

In 1974, the IAEA established a special Nuclear Safety Standards (NUSS) programme under which Codes (requirements) and a number of Safety Guides have been produced in the areas of governmental organization, siting, design, operation and quality assurance. The NUSS Codes and Guides are a collection of basic and derived requirements for the safety of nuclear power plants with thermal neutron reactors. They have been developed with the broadest possible international consensus.

This broad consensus is one of the reasons for the relatively general wording of the main principles and requirements which may need further elaboration and guidance for application to specific nuclear power plants. In many areas, national regulations and technical standards are available, but often even these do not answer all questions and only the practice adopted in applying certain rules fully reflects the outcome of the detailed consideration given to solving individual cases.

To present further details on the application and interpretation and on the limitation of individual concepts in the NUSS Codes (requirements) and Safety Guides, a series of publications that detail good practices has been initiated. It is hoped that many Member States will benefit from the experience presented in these publications.

The present report provides information on good practices in conducting probabilistic safety assessment (PSA) for fires in land based nuclear power plants, and is intended for the professional staff who manage or perform PSAs. It is applicable to both new and existing plants.

This Safety Report has been developed within the framework of the IAEA programme on fire safety in response to the increasing attention being given to the risk based approach, both in general safety assessment and in relation to a fire in nuclear power plants. It supplements existing guidelines on this topic.

This publication has been prepared with the help of experts from engineering and scientific organizations, regulators and plant operators, all with practical experience in the field of fire safety and fire protection in nuclear power plants. The IAEA is grateful to all the experts who helped in the drafting and reviewing of this publication.

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1. INTRODUCTION

1.1. BACKGROUND

Considerable attention has recently been devoted to the topic of fire safety at nuclear power plants, in particular to those plants that have been designed and constructed according to earlier fire protection standards. It is important that a comprehensive fire safety assessment is performed for these plants at the earliest opportunity in order to document that the existing fire safety measures are adequate to ensure plant safety. Periodic updating of fire safety analyses has to be continued throughout the life of a plant to reflect all the changes made, as well as the current requirements of and the experience gained on fire safety.

The IAEA is endeavouring to promote an exchange of information on fire safety between different countries, as well as the use of various fire safety assessment techniques. The main objectives are to achieve a better understanding of the current situation, to identify those areas that need further development, and to promote the most effective and reliable techniques. Considerable effort has been made to develop guidelines for the preparation and evaluation of a fire safety analysis for nuclear power plants.

Systematic assessment of a fire hazard is one of the important elements in implementing fire protection in plants. When applied at the plant design stage, it permits integration of the proper protection concept into the design and ensures that, throughout all stages of design, construction and commissioning, problems are identified and resolved. For plants in operation it is possible, through a systematic fire hazard assessment, to identify the existing deficiencies in fire protection and to implement practicable and worthwhile improvements in fire safety.

Deterministic and probabilistic techniques are used to assess a fire hazard. The deterministic fire hazard analysis, typically carried out first, is normally required by licensing authorities and other safety assessors. It is usually developed early in the design of new plants, updated before initial loading of the reactor fuel, and then periodically or when relevant operational or plant modifications are proposed. Probabilistic safety assessment (PSA) for fire is undertaken globally to supplement the deterministic fire hazard analysis.¹ It should be noted that a fire PSA is recognized as a tool that can provide valuable insights into plant design and operation, including

¹ Throughout these guidelines the term fire hazard analysis is applied exclusively to the deterministic assessment of fires, while the expressions fire risk assessment and fire PSA are used for the probabilistic safety assessment of fires.

identification of the dominant risk contributors, comparison of the options for risk reduction and consideration of the cost versus risk benefit.

Two publications devoted to the fire hazard analysis for nuclear power plants have been developed as part of the IAEA Nuclear Safety Standards (NUSS) programme: Preparation of Fire Hazard Analyses for Nuclear Power Plants [1] and Evaluation of Fire Hazard Analyses for Nuclear Power Plants [2]. These publications supplement Safety Series No. 50-SG-D2 (Rev. 1), Fire Protection in Nuclear Power Plants [3], by providing detailed information on the preparation and evaluation of a fire hazard analysis at a nuclear power plant. They address a systematic approach based on the deterministic technique.

No detailed information on conducting a PSA for fire in nuclear power plants is provided in other PSA related IAEA publications: Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1) [4] addresses PSA for internal events, and Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants [5] outlines the general treatment of those hazards external to a plant that are encountered and analysed most frequently: earthquakes, high winds, floods and person induced events. Since internal fire events have a localized effect on plant safety systems, no specific recommendations are given in Ref. [5] on the treatment of internal fire hazards.

The present report has been developed in response to the increased attention being given to PSA worldwide. It is intended to facilitate implementation of the risk based approach to fire safety assessment for both new and operating nuclear power plants, and supplements existing IAEA publications on fire safety assessment.

1.2. OBJECTIVES

This Safety Report provides information on good practices in conducting an internal fire PSA for land based nuclear power plants, as well as assistance in integrating the threat of a fire into an existing internal events PSA. It is intended for the professional staff who manage or perform PSAs.

Specific details of various aspects of a PSA for fire are limited globally. The report concentrates on the procedural steps for a fire PSA, but the tools needed to implement these steps remain the choice of the analyst; the references cited should not be taken as complete or authoritative.

This publication can be used to assist in implementing a PSA for fire in nuclear power plants on the basis of the current practical experience gained in this area. A particular aim is to promote a standardized framework, terminology and form of documentation for PSAs that will facilitate an external review of the results of such studies.

The methods and approaches addressed reflect the practices most widely used to date. This Safety Report is not intended to pre-empt the use of new or alternative methods; on the contrary, the promotion and use of all methods that achieve the objectives of a fire PSA are encouraged.

1.3. SCOPE

This Safety Report supplements Safety Series No. 50-9-4 [4], which deals with internal events. As such, it addresses only those specific issues that are related to fire events. The reader should also refer to Ref. [4] for information on general PSA topics, e.g. plant system modelling, methods of quantification and PSA project organization and management.

The information provided on good practices applies to land based nuclear power plants with thermal reactors of commercial use such as those of the light water, heavy water or gas cooled type. However, this material may also be of use in preparing a PSA for fire for other nuclear reactor installations, including research reactors.

The main emphasis of this publication is placed on assessing the potential risk of core damage states initiated by fires (PSA Level 1, as defined in Ref. [4]).

Some additional information is provided on the probabilistic modelling of fire induced releases from other plant systems and compartments that contain radioactive materials for the purpose of PSA Level 2, as defined in Ref. [4], if required (e.g. in-plant waste storage).

The practices addressed focus on fire events initiated under operation at full power. However, the information provided on the methodological approach is fully applicable to other operational states of the plant, including shutdown. However, in applying this approach to other plant operational states, the analyst should be aware of the specific conditions of the plant, which may differ substantially from those under normal power operation.

There is no limitation to the application of this methodological approach for any stage of the plant life cycle, including the conceptual or final design stage and the operational stage.

1.4. STRUCTURE

Section 2 provides an overview of fire PSA. It briefly highlights the differences and similarities between the probabilistic and deterministic approaches. The general methodological assumptions adopted in these guidelines are listed. Some

organizational aspects are also addressed, including the objectives and scope of the fire PSA project, the expertise of the PSA team and the quality assurance (QA) programme. The main tasks of the fire PSA are briefly surveyed and the interrelation between these tasks highlighted. Sections 3–9 discuss in detail the individual tasks of the fire PSA.

Section 3 covers collection and assessment of the data required for fire PSA and explains how the entire plant area should be subdivided into smaller parts to provide an organizational framework for data collection and to facilitate the analytical work. It further addresses familiarization with the internal events PSA, describing the requirements for the internal events PSA model and explaining how this model should be adopted and extended to create an integral fire PSA. Guidance on identification of the equipment and cables relevant to fire risk is an essential part of this description. The section also refers to preparation of a plant location oriented database for the relevant equipment and cables, as identified in the course of familiarization with the internal events PSA.

Section 4 explains how to minimize the analytical effort by screening out non-essential fire scenarios. The techniques and assumptions applied at various stages of the screening process are described. Two stages of screening are addressed: screening by impact for single and multicompartment fire scenarios, and screening by frequency.

Section 5 addresses the detailed analysis of fire risk applied to those fire scenarios that were not eliminated by screening. Indications are given of the possible refinements of the fire PSA model that can be incorporated into this stage of the analysis to reduce conservatism. Some of the techniques used in the detailed analysis of fire propagation are also discussed.

Section 6 contains information that supplements the general information given in Sections 3–5. It focuses on issues that have been found to be important to the proper execution of PSA for fire methods and that differ significantly from those discussed in Sections 3–5, either because of specific features associated with location (the main control room, cable spreading room, switchgear rooms) or because of some additional aspects that should be taken into account in the analysis (environmental survival of equipment, control system interactions, containment integrity). Conducting PSA for fire in the event of incomplete information is also addressed.

Section 7 deals with qualitative and quantitative analyses of the results, and discusses the sources and quantitative measures of uncertainty in the PSA for fire. Guidance on sensitivity and importance analyses of the PSA for fire is also given.

Section 8 provides guidance on documentation of the analysis, as well as final presentation and interpretation of the results.

Section 9 discusses the treatment of ex-core radioactive releases as a result of fire events, a modelling issue that requires a slightly different approach to that of core related risk.

2. GENERAL OVERVIEW OF A FIRE PSA PROJECT

2.1. METHODS

Fire PSA is the probabilistic analysis of fire events and their potential impact on the nuclear safety of a plant. Using probabilistic models, fire PSA takes into account the possibility of a fire at specific plant locations; the propagation, detection and suppression of the fire; the effect of the fire on safety related cables and equipment; the possibility of damage to these cables and equipment, and in severe fires the structural integrity of the walls, columns, roof beams, etc.; and assessment of the impact on plant safety. Since the physical separation between redundant safety trains can limit the extent of fire damage, quantification of the damage frequency calculations generally includes those equipment failure probabilities that are not affected by the fire, e.g random failure probabilities, and the likelihood of a maintenance outage.

Many elements of a fire PSA are the same as those used in the deterministic fire hazard analysis (as described in Ref. [1]). It should be noted, however, that the probabilistic approach includes some new aspects of modelling and applies different acceptance criteria for the evaluation of fire safety. This section discusses the specific aspects of a fire PSA, highlighting the differences and similarities between the deterministic and probabilistic approaches.

The fire risk assessment methods introduce the likelihood of a fire in each plant location, the effect of the fire on equipment and cables, and the impact of equipment failures and human actions coincident with the fire. New elements of the model specific to the risk based approach include factors such as the probability and effect of plant damage beyond individual fire compartment boundaries (as a result of barrier elements being ineffective or inoperable) and random failure of the mitigation systems. The probabilistic criteria used in fire PSA are based on the risk concept. Core damage frequency is a typical criterion used for PSA Level 1.

Fire PSA relies on the plant response model developed for the internal initiating events. The availability of a plant model that logically examines the contributions to core damage, plant damage, etc. is a prerequisite for a fire PSA. An internal events PSA Level 1 is highly desirable; however, a partial PSA Level 1 (for selecting the initiating events) or another logic model equivalent to PSA Level 1 may be an adequate substitute.

It should be pointed out that expanding an internal events PSA to a fire PSA requires a considerable amount of plant specific data, e.g. the location of cable routes in plant compartments. This information will be readily available if a

comprehensive deterministic fire hazard analysis has already been performed for the plant.²

In the same way as the deterministic method, the PSA approach is based on systematic examination of all plant locations. To facilitate this examination, the plant is subdivided into distinct fire locations, which are then scrutinized individually. It is essential to demonstrate that significant fire scenarios have not been overlooked. However, a theoretically complete and exhaustive examination would be both impractical, because of the large number of possible scenarios, and unnecessary, because there are many fires that are unlikely to pose any significant risk. Therefore, an effective screening process is essential to limit the level of effort made for the fire PSA.

It is advisable to perform the screening process in stages, starting with relatively simple, conservative models and progressing to more realistic representation of the fire scenarios at subsequent stages. Application of complex models that involve detailed investigation of the evolution of the fire and its impact on safety equipment, as well as the effect of the fire mitigation features, is limited to a relatively small number of fire scenarios, therefore the overall analytical effort is reduced substantially. This part of the PSA relies on physical fire growth models that are similar to those used in the deterministic fire hazard analysis.

Compared with the deterministic approach, the PSA model introduces some new elements that involve statistical data; as a result, further contributors to uncertainty in the final evaluation of fire safety are added. This aspect should be taken into account when applying PSA techniques to fires in nuclear power plants. In this case, sensitivity and uncertainty analyses are essential if interpretation of the results is to be correct. It should be emphasized that the main advantages of a PSA are that it can identify a number of uncertainties, and quantify and describe most of them.

2.2. MAIN ASSUMPTIONS

The fire PSA discussed in these guidelines is intended to reflect the current status of the plant using a best estimate assessment, but it does not address compliance of the plant with the fire protection codes, standards and regulations actually in force at the particular plant.

In general, with regard to the combination of events and the scope of the analysis, the assumptions recommended are consistent with those usually applied to an internal events PSA.

² This applies to most existing nuclear power plants for which the deterministic fire hazard analysis has been performed prior to a fire PSA. This is not the case for new plants, where deterministic and probabilistic analyses of fire hazards may be carried out in parallel at the early stage of design or construction.

Only a single, independent fire is assumed to occur in any plant location. The spread of this fire to adjacent fire locations is taken into account, unless it can be justified that the fire is contained in the original fire location. It should also be noted that only in very rare cases can multicompartment fires be ignited concurrently in several locations by a single initiator (e.g. an overheating cable).

For multiple reactor sites, simultaneous fires in more than one reactor plant are not postulated. However, it should be taken into consideration that a single fire in facilities shared by reactors can affect more than one reactor (addressing the worst case of system interdependence).

The most severe natural phenomena, e.g. tornadoes, flooding or earthquakes, are not assumed to occur concurrently with a fire. Internal initiating events (e.g. LOCA) are also not considered to be concurrent with a fire, unless they are a consequence of that fire.

Fires induced by other initiating events (e.g. earthquakes, sabotage) are not considered to be within the scope of these guidelines, nor is the risk associated with the spurious activation of fire protection equipment (and potential flooding). The potential for such activation is usually examined as part of the internal flooding analysis. However, secondary effects caused by the operation of fire protection systems during a fire are taken into account in a fire PSA.

2.3. PROJECT ORGANIZATION AND MANAGEMENT

The actions and activities necessary for the organization and management of a fire PSA are similar to those of an internal events PSA, including definition of the objectives and scope of the project, establishment of a project management scheme, selection of the methods and procedures, organization and composition of the project team, training of the team and establishment of the QA programme. The general guidelines for these activities, as outlined in Ref. [4], are applicable also to a fire PSA project. Some issues specific to a fire are highlighted below.

It is essential that the objectives and use of the results of a fire PSA are precisely defined at the early stages of a PSA project. In turn, these will determine the scope of the analysis, and the necessary methods and procedures. More detailed information on the general objectives of PSA, and various implications specific to the selected objectives, are given in Ref. [4].

The objectives and scope of the fire PSA are usually co-ordinated with those defined in the existing internal events PSA. This is important in order to ensure that interpretation and application of the existing internal events PSA model are correct and that any misuse of the results is avoided.

The expertise needed to conduct a fire PSA must combine several disciplines. Thorough knowledge is required of the plant design and operation, the PSA

techniques (essential to the preparation of an internal events PSA), fire science, as well as the design and operational aspects of the fire protection systems (including their interaction with the nuclear safety systems).

It is essential that the fire PSA team includes specialists who are capable of evaluating the fire damage effects (including smoke and gases, as appropriate) on those structures, systems and components that are important to safety, and of assessing fire induced failures of the power, control and instrumentation circuits. The ability to evaluate the adequacy and likely performance of the installed fire detection and suppression systems is also of importance, especially regarding the timing of system actuation compared with the timing of component failures, where such timing is used/claimed in the analysis.

The size of the workforce and the amount of time required to complete a fire PSA depend on the scope of the PSA and on the expertise available in the PSA team. Quite a large workforce is required to collect plant specific information. However, compared with an internal events PSA the number of personnel involved in systems analysis in a fire PSA is much lower.

Quality assurance of the PSA project should be viewed and established as an integral part of the PSA procedures that control all PSA activities. The specific aspects of the QA procedures applied in the organization, technical work and documentation of a PSA project are discussed in Ref. [4]. Establishment of the appropriate QA programme in a PSA project is even more important for a fire PSA. Some specific QA related aspects of a fire PSA are discussed below.

Much of the plant specific information required for a fire PSA is not easily retrievable from existing plant documentation (e.g. the cable routes). Such data collection requires that considerable attention be given to the quality of information and that systematic, disciplined QA measures be taken. As a fire PSA requires a highly specialized team, including fire related experts, co-ordination of activities within the team, particularly at the interfaces between the different disciplines, may be more complicated. Therefore, QA verification of the results of the team's work is very important. General guidance on conducting an independent peer review of a PSA (given in Refs [4, 6]) is also applicable to a fire PSA project.

2.4. MAJOR PROCEDURAL TASKS

The major procedural tasks in a fire PSA and the general flow of information between these tasks are shown in Fig. 1. It should be noted that the flow is not always sequential; some iterative loops exist between various tasks not shown on this simplified flow chart. Certain interrelated tasks are carried out throughout the entire modelling process (e.g. data collection and assessment, and documentation of the analysis).

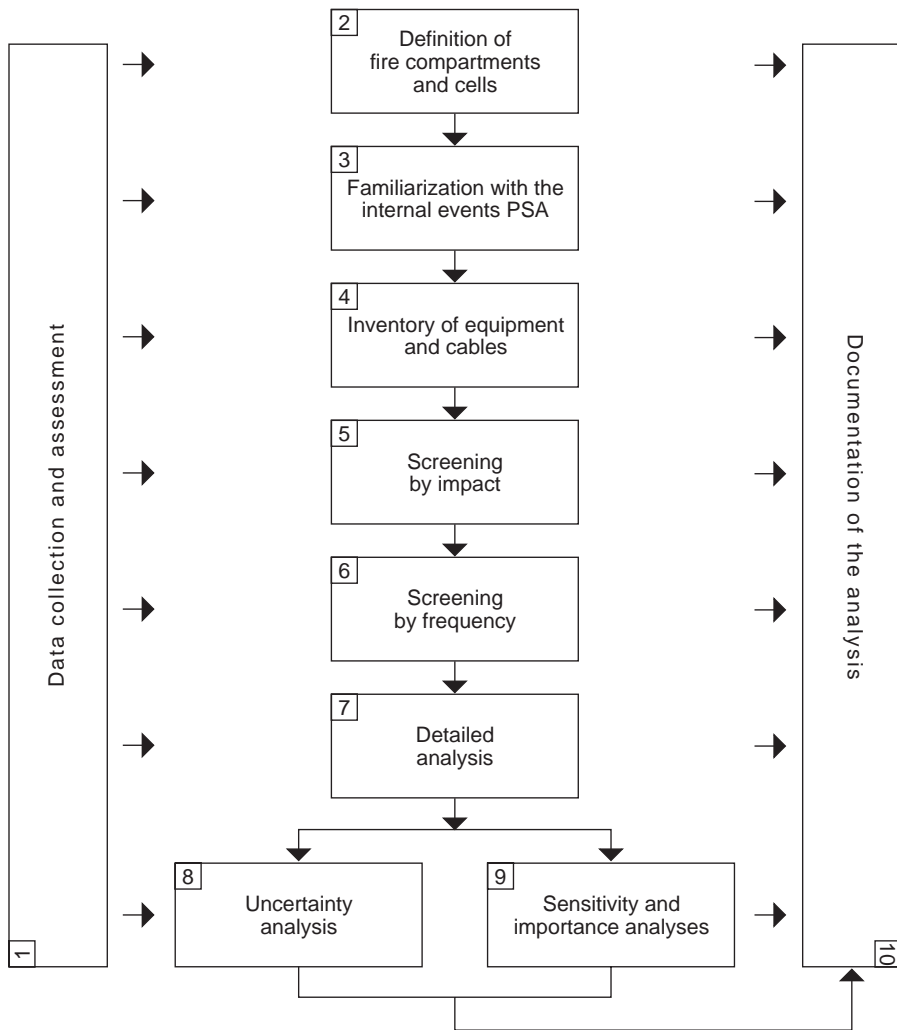


FIG. 1. Major procedural tasks in a fire PSA.

It should be pointed out that the first procedural step of a PSA, i.e. dealing with the organization and management of a PSA project, is not shown in the figure. However, the information provided in Ref. [4] on related tasks is equally applicable to a fire PSA.

Data collection and assessment (task 1) is the initial task in the fire PSA procedure. Preparation of the necessary data is a major part of a fire PSA project, and

is a very time consuming task. It concentrates on collection of the plant specific data required for fire risk modelling; however, some data used in the internal events PSA model also have to be re-assessed to account for fire induced conditions.

This task begins at the early stage of a fire PSA and continues for almost the entire duration of the project. Data collection is a plant location oriented process that proceeds in parallel with task 2, which deals with subdivision of the plant area into individual fire locations (fire compartments and cells).

It should be noted that detailed information is not needed for all plant locations. Most of the data are required for the analysis of compartments when more sophisticated models are introduced (typically those that involve higher risk). Therefore, the data collection process needs to be well co-ordinated with the analytical tasks in order to avoid the collection and assessment of data that are unnecessary for the analysis. The screening analysis can be conducted with a smaller amount of plant specific information, which can be extended at the later stages of modelling as the preliminary results are obtained, and as the models require.

Definition of fire compartments and cells (task 2) is established at the initial stage of analysis. It is aimed at the division of all plant buildings and structures into distinct fire compartments and cells³, which are scrutinized individually at the later stages of analysis. All plant buildings and structures are systematically examined. Some plant locations that do not contain any plant equipment (e.g. administrative buildings and offices) can be eliminated from further consideration at the very early stage of analysis on the basis of qualitative judgement. However, prior to elimination it has to be shown that a fire in one of these zones cannot spread to an adjacent zone that houses safety related equipment. Further elimination is carried out later on the basis of more formal screening procedures. Sometimes, redefinition of the fire compartments and cells is needed at the later stages of analysis, when more sophisticated models are introduced. The results of this task include a set of fire compartment and cell drawings and specification of all the surrounding boundaries.

Familiarization with the internal events PSA (task 3) is an important task in that it establishes a link between the existing internal events PSA models and the fire related models. It starts with examination of the internal events logic models (e.g. fault trees and event trees), and their applicability to fire risk modelling. Sometimes, these models have to be extended in order to achieve the required level of detail and completeness. This task also identifies those plant systems and equipment, and all the

³ A definition of fire compartments and cells is provided in Safety Series No. 50-SG-D2 (Rev.1) [3]. The terms fire compartment and fire cell are essentially analogous to the fire area and fire zone terms used in codes, standards and guidance documents in the United States of America.

related elements of the model, that are important to fire PSA. Identification of all the related cables and circuits is an integral part of this examination.

As a result of this task, a logic model suitable for calculating the conditional unavailability of the required safety functions (task 6) is made available. Another outcome is a list of the PSA related equipment and cables.⁴ Basic, component specific information is usually collected for certain PSA related items (e.g. for the required component functions, and the electrical and control supplies). This information may be further extended at the later stages of analysis.

The next task (*task 4*) is preparation of an *inventory of equipment and cables* (as identified in task 3). A list of PSA related items is prepared for each fire compartment and cell (defined in task 2). A plant walkdown is important in order to establish correct localization of the equipment and cables in the fire compartments and cells. During the initial stages of the screening analysis, listing components by plant fire compartment or cell is sufficient. At the later stages of detailed analysis it will be necessary to determine more accurately the component locations within the fire compartment or cell.

Screening by impact (task 5) is aimed at eliminating non-essential fire scenarios on the basis of impact oriented criteria (mostly, but not exclusively, qualitative). It starts with definition of the critical fire locations, followed by definition of the possible single and multicompartment fire scenarios. The impact oriented criteria used for screening out the individual fire scenarios take into account the characteristics of those fire compartments that are involved in the scenario considered. The result of this task is a list of fire scenarios that can be significant contributors to risk.

Screening by frequency (task 6) is aimed at the further elimination of those fire scenarios that are retained after the first stage of screening (task 5). Screening is performed on the basis of a simple, conservative estimate of damage frequency (e.g. core damage frequency). The conditional unavailability of the required safety functions (e.g. safe shutdown) because of a fire is calculated from the existing internal events PSA model. Conservative assumptions are made of the effect of a fire on equipment, and the related human actions. As a result, the number of risk significant fire scenarios is further reduced. For each of the remaining fire scenarios, a quantitative PSA model is available for further analysis.

⁴ The term PSA related equipment is systematically used in this report to describe all the items credited in the internal events PSA model. For a PSA that is limited to in-core radioactive releases, this term is equivalent to the term safe shutdown equipment, as outlined in Section 3.3.3. In a PSA that considers other sources of radioactive release, the PSA related equipment also includes those items that are related to the safety functions which need to be performed in order to prevent releases from ex-core radioactive sources, including the necessary support features.

The *detailed analysis (task 7)* is aimed at reducing the level of conservatism in the fire scenarios identified in task 6. The effect of intracompartment barriers and other fire protection measures, the location of equipment in the fire compartment or cell and other factors are taken into account. More realistic models are applied for assessing human actions, fire propagation, the effects of a fire on the equipment and cables, etc.

The analyst may select any of the above mentioned approaches (one or several at a time) to revise the risk estimates calculated in task 6 for each of the risk significant fire scenarios. More fire scenarios are screened out in the course of this process, and refined risk estimates determined for the remaining scenarios.

The *uncertainty analysis (task 8)* is aimed at identifying the sources of uncertainty, and their evaluation. Qualitative and quantitative assessments are carried out. Quantifiable uncertainties are investigated through formal uncertainty analyses using the fire risk model developed in tasks 6 and 7.

Sensitivity and importance analyses (task 9) are aimed at identifying the risk significant elements of the fire PSA model. Sensitivity studies are performed for the important assumptions, and the relative importance of various contributors to the calculated results determined. The fire risk model developed in tasks 6 and 7 is used in this task.

Documentation of the analysis (task 10) is one of the tasks that continues for almost the entire duration of a PSA project. The results of analysis of each task are thoroughly documented and the final documentation is prepared. The results of the PSA are displayed and interpreted in line with the objectives of the PSA.

3. PREPARATION PHASE (TASKS 1–4)

3.1. DATA COLLECTION AND ASSESSMENT (TASK 1)

A fire PSA relies on the availability of plant information (both qualitative and quantitative). As previously mentioned, data collection is a major, time consuming task that extends over several analytical tasks of a fire PSA. Two major types of plant specific data are obtained: internal events PSA related data and fire related data.

The fire PSA is strongly dependent on the internal events PSA. A large portion of the model is retained and used in the fire PSA, but a number of aspects will have to be reviewed, and in some cases developed. The information that needs to be gathered for the fire PSA can be categorized as a list of initiating events; the PSA logic models; the basic events of the model; and human actions.

Fire related data require a considerable amount of plant specific information. These data can be classified into the following major groups: the physical characteristics of the fire compartments, and their inventory; fire occurrence data; reliability estimates of the fire detection and suppression systems; human actions and human error probabilities; and fire induced equipment failure modes and damage criteria.

The first group may be available from a fire hazard analysis based on the deterministic approach (as described in Ref. [1]), which typically is performed for the plant prior to a fire PSA. This group of data includes the following categories: the safety system inventory; the fire compartment inventory; the combustibles inventory; the ignition sources inventory; the passive fire protection features; the fire detection and alarm systems; the fire extinguishing systems; the emergency lighting; the communication systems; the smoke and heat removal systems; and the manual fire fighting arrangements. Detailed specification of the scope of these categories can be found in Ref. [1].

The required plant specific information can be acquired from various design sources, as well as from plant walkdowns, where in situ information is gathered and verified.

Collating design information from plant documentation is usually the first phase of data collection. The recommended sources of plant specific information include a description of the systems, equipment lists, design drawings, plant procedures and other similar items. Several types of drawings should be mentioned in this context:

- (1) General arrangement drawings that display the current configuration of plant areas and the location of major equipment;
- (2) Fire barrier drawings that show the fire compartment walls and the location of fire doors;
- (3) Piping and instrumentation diagrams for the relevant systems;
- (4) Electrical distribution drawings and electrical logic diagrams;
- (5) Electrical drawings that show the connection of the power and control circuits for the systems, the arrangement of the motor control centres, and the cable and conduit routing;
- (6) Ventilation path drawings.

All the information obtained from plant documentation has to be verified by visually inspecting each fire compartment throughout the entire plant. This is essential in order to ensure that the data represent the actual and current condition of the plant.

It should be noted that a plant walkdown also plays a very important role in familiarizing the analyst with the fire specific features of the plant. The purpose of

such a walkdown is to determine or verify the equipment locations and to gather information on the physical condition of the compartment or cell, and on the fire related features, some of which may not be easily identified from other data sources (e.g. the nature of the openings in the fire barriers or the existence of louvres on cabinets).

Several plant walkdowns are necessary during different phases of the study. A plant familiarization walkdown is usually performed during the initial stages of analysis. A second, detailed walkdown may be performed when the screening analysis is nearing completion in order to confirm the information used and to gather data on those specific compartments that will require detailed analysis. Additional walkdowns, confined to compartments undergoing detailed analysis, may be required to confirm and collect additional data, and to examine corrective actions with a view to reducing potential vulnerabilities, if required (task 7).

Plant specific fire occurrence data are collected for the source of ignition, the materials involved in the fire, and the damage to equipment and cables. It is advisable that, in addition to the fire events, the analyst collects generic data on the fire initiation frequencies which are available in the literature and which are drawn from nuclear power plant operating experience. Sources of such data include Refs [7–11].

Reliability data for the fire protection features include data for active fire protection equipment (fire detection and fire extinguishing systems) and for intercompartment fire barriers (dampers, doors, curtains, penetration seals, etc.). These data can be derived from plant operational experience (event records, test and maintenance records, etc.) or, using the available plant specific data, extrapolated from generic sources of information.

Plant operating procedures, particularly those concerned with operator actions following indication of a fire or other initiating event, form the basis of the operator actions that can be included in the fire PSA. A number of the operator actions in the internal events PSA model, including certain important recovery actions, will have to be reviewed and, in some cases, requantified. This is discussed further in Sections 3.3.4 and 5.4.

Where an action local to a plant is required it is necessary to consider all those factors that may prevent plant personnel from carrying out this action, e.g. the unavailability of emergency lighting. The routes within the plant by which access is achieved have to be determined. The fire situation may involve the normal or the most direct route, but it will be necessary to identify and consider a viable alternative. Walkdowns play an important role in gathering and verifying information of this type.

In some cases, a walkdown may determine that credit should not be taken for any operator actions within a fire compartment until well after the fire has been suppressed. This would be true, for example, if the action required that the operator traverse a significant portion of the compartment, perform complicated or multiple

control actions, or remain in the compartment for any significant fraction of time. Such actions should also not be credited in scenarios involving larger fires.

The fire analyst will need to gain knowledge of the susceptibility of various types of equipment to the different phenomena that may be experienced in a fire event. The analyst will also have to establish a list of equipment types within the plant, and to specify their damage mechanisms (e.g. heat, flame, smoke and water) and failure modes. For example, passive components, such as pipes, check valves and manual valves, are generally not expected to fail in a fire. A motor (e.g. for a pump) may fail from heat but not from exposure to smoke, while an electronic device may fail from heat and smoke.

Since this data collection process produces a significant amount of interrelated data it is recommended that the information be arranged in a systematic way (e.g. tables), preferably in a computerized database. This greatly facilitates the retrieval and processing of data. It is advisable that comprehensive and well organized data sheets be used during plant walkdowns. Some examples of such data sheets can be found in Refs [1, 10, 12].

Care in the use of generic data should always be exercised. For example, regarding fire occurrence frequencies, definition of a recorded fire will vary and be influenced by the fire detection and suppression measures taken. This applies also to equipment failure rate data.

Even under the best of circumstances some gaps in the information base will remain unfilled; this issue is discussed further in Section 6.7. It is important for the analyst to recognize and acknowledge where such information gaps have occurred, and to describe in the analysis how these gaps were overcome.

3.2. DEFINITION OF FIRE COMPARTMENTS AND CELLS (TASK 2)

The division of all plant buildings and structures into distinct fire locations (fire compartments and cells), which are scrutinized individually at the later stages of analysis, is an important task that permits systematic and definable evaluation of fire events.

Physical separation between safety relevant systems and equipment is an effective fire safety feature. Such separation can be achieved through distinct fire compartments, which are plant areas completely surrounded by fire barriers.⁵ The fire compartments are designed to prevent the spread of the effects of a fire to or from other plant compartments. The fire resistance (fire rating) of the compartment barriers

⁵ A definition of fire barriers and fire barrier rating is provided in Safety Series No. 50-SG-D2 (Rev. 1) [3].

may be sufficient to contain fires initiated in that compartment (design approach based on fire containment), or may require additional fire protection measures to limit fire spread (fire influence approach).

Formal definition of the fire compartments and the fire resistance rating of the related barriers may not be readily available for the PSA. In such cases it will be necessary to undertake a review of the major construction elements of the plant in order to derive the appropriate fire compartments and the fire rating of barriers. Guidance is given in Ref. [1].

Some flexibility should be exercised by the analyst in defining fire compartments for PSA use. For instance, the analyst may prefer to consider several fire compartments as one compartment, if this facilitates the analysis. It is strongly advisable to avoid unnecessary division of the plant into a large number of small locations, at least at the early stage of analysis.

The fire resistance rating of the walls and ceilings may be determined analytically or be evaluated by engineering judgement according to simplified state of the art methods that involve the thickness and material of the wall (such as graphs or tables published in the literature).

Fire barrier elements, e.g. doors or dampers that are installed in the walls, are included in this process. The fire resistance rating of each fire compartment barrier is determined by the weakest (lowest fire rated) element of that barrier.

Where a fire rating cannot be established and justified, the barrier cannot be considered as being fire rated, and it is necessary to consider larger areas of the plant as a single fire compartment. In such a situation, the fire compartment may be subdivided along logical lines such as rooms, functional areas or areas with clearly defined spatial separation. Such areas are called the fire cells of the fire compartment. Some examples of the definition of fire compartments and cells in a nuclear power plant can be found in Ref. [3].

Typically, a set of fire compartment and cell drawings, and specification of all the surrounding boundaries, are generated in this task. Use of a comprehensive and flexible numbering system for fire compartment and cell identification is advisable; numbers are usually assigned to the fire compartments after the first stage of screening.

3.3. FAMILIARIZATION WITH THE INTERNAL EVENTS PSA (TASK 3)

A fire PSA can only be conducted if some form of plant model exists that logically examines contributions to core damage, plant damage states, etc.

This task covers examination and interpretation of the existing internal events PSA (or its equivalent) to determine the plant systems and equipment, as well as all those related elements of the model that are important to the fire PSA. Section 3.3.1

discusses the general requirements that should be fulfilled to make an internal events PSA suitable for fire risk assessment. A number of issues specific to fires also have to be considered; these are discussed in Sections 3.3.2 to 3.3.6.

3.3.1. Requirements of the internal events PSA

A fire PSA is normally performed either as an integral (later) part of a full scope PSA or as a discrete task following completion of an internal events PSA. To minimize potential errors or misuse of the PSA results, the objectives, limitations and assumptions used in the internal events PSA should be understood.

The unavailability of an internal events PSA creates a number of difficulties for fire PSA in terms of a full representation of the logical safety features of the plant. Without such models it will not be possible to estimate the relative importance of fires in a quantitative manner, and thereby provide results for informed decision making.

The information given below assumes that an internal events PSA model exists and that it is comprehensive. A comprehensive model should comply with the guidance given in Ref. [4]. Where such a PSA model is not available, the analyst may be able to adapt or tailor existing logic models to meet the minimum intent below.

The extent of the internal events PSA will also influence selection of the screening criteria to be used in task 6, depending on whether the core damage frequency, plant damage states or other ex-core releases of radioactivity are considered.

The internal events PSA is used to provide information on the initiating events⁶ and systems (including support systems) that are used in the mitigation of such events.

The internal events PSA should contain information on the possible causes of initiating events, as well as details of those initiating events that have been grouped in a particular event tree model and those that have not been modelled because of some form of qualitative screening. A review of initiating events is given in Section 3.3.2.

The internal events PSA should also contain information on those systems and components whose failure to function correctly in response to an initiating event may lead to an undesired consequence. Such equipment includes safety related frontline and support systems, and non-safety related systems such as main feedwater and off-site power. For a fire PSA, information relating to those components that use or provide an external power source (electric, pneumatic, hydraulic) is required.

A comprehensive internal events PSA should already include the failure modes of interest in the fire PSA for such components. The required detail, and its use, are described in Ref. [4].

⁶ In the context of these guidelines only, the following definition of an initiating event is applied and taken to be exclusive: An initiating event is a change in the hardware state of any equipment that leads to a perturbation in the normal heat production–removal balance of the plant.

In this context it is expected that the internal events PSA will be developed to the component level in order to identify explicitly those items that provide the safety functions required to mitigate an internal initiating event (Section 3.3.2). These include: the pumps; the motor, with pneumatically or hydraulically operated valves; electrical supply equipment, including transformers and breakers; instrumentation and control (I&C) signals and related hardware; and pipework and structures.

Typically, this involves systems that provide the following safety functions: control of reactivity; controlled removal of the core decay heat and stored heat; maintenance of the integrity of the reactor coolant boundary (pressure control); maintenance of the reactor coolant inventory; protection of containment integrity (isolation, overpressure); and scrubbing of radioactive materials from the containment atmosphere.

If the available logic models do not provide this level of information, the analyst, with extreme diligence and care, may be able to meet the intent of the PSA model. However, such an approach must be adopted with this provision in mind and in recognition of the potential problems that may arise in the quality and usefulness of the final product.

3.3.2. Review of initiating events

This task determines the list of components used or implied in the PSA in terms of their potential, as a result of a fire, to cause an initiating event that requires some form of control or mitigating action, either manual or automatic.

This task starts with a review of the initiating events considered in the PSA. A number of techniques for identifying the initiating events of an internal events PSA are described in Safety Series No. 50-P-4 [4]. In turn, each initiating event has to be reviewed in order to determine whether it can be induced by a fire.

It should be recognized that such effects include failure of the power supply to the equipment from the main electrical bus(es) and actuation signals for equipment operation (e.g. start, stop, open, close) from the control room and the control relay cabinets. In terms of completeness, the analysis could also be extended to cover identification of those instrument sensors and signals, including any processing, that may adversely affect operation of the equipment.

The initiating events identified should be the same as those already included in the internal events PSA. However, based on low probability, some analysts may have chosen to exclude certain initiating events from the internal events PSA. In such cases, the fire PSA analyst must bear in mind that the fire may cause more severe faults than those considered or modelled previously, thus necessitating the creation of a new event sequence model for evaluation or allocation to an equivalent bounding initiating event.

An example of this is the possibility that the fire may lead to multiple opening of the steam generator power operated relief valves, whereas the internal events model may be limited to spurious opening of a single relief valve. Also of concern is multiple loss of the electric power supply to the safety related components. In such cases, either a new logic model for the multiple event will have to be prepared, or the fire PSA analysis may have to adopt a representative (but conservative) equivalent event such as a steam line rupture of the steam generator in the first instance.

It is important to note that the review of potential fire initiating events should also include the support system effects on those systems that are involved in the normal operation of the reactor.

The initiating events that arise from this review can generally be categorized as one of the following:

- (1) Events leading to controlled reactor shutdown;
- (2) A reactor trip (scram) initiated by the operator;
- (3) Transients leading to an automatic scram, e.g. tripping of the turbine, loss of feed, loss of the electric power supply, loss of off-site power, opening of a steam generator relief valve;
- (4) LOCA from the primary circuit, e.g. failure of a pump seal (because of loss of seal cooling), opening of a pressurizer relief valve, interfacing system LOCA;
- (5) Events resulting in releases of ex-core radioactivity.

The level of work involved in this task varies according to the level of information already included in the PSA modelling. It is important to take into account all those items of equipment that can influence the PSA modelled function. This will necessarily extend the analysis to a detailed understanding of the operation of the system or subsystem in terms of motive, and the control power cables and signals that operate the system.

3.3.3. Identification of the PSA related systems and equipment

This task determines the list of those components that are credited in the internal events PSA in terms of their use in the control or mitigation of a fire caused by an initiating event (the term PSA related equipment is systematically used in this report to describe these components). In the case of a PSA that is limited to in-core radioactive releases, this list relates to the frontline and support systems that provide safe shutdown of the reactor and adequate heat removal from the reactor core (safe shutdown equipment); it includes alternative and dedicated systems. In a PSA that considers other sources of radioactive release, the list includes also those items that are related to the safety functions which need to be performed in order to prevent releases from ex-core radioactive sources. However, it does not include fire protection

TABLE I. WORKSHEET USED IN THE REVIEW OF THOSE COMPONENTS THAT ARE IMPORTANT TO A FIRE PSA

Component identifier
 Component description

Component location
 Building:
 Room:

Potential to cause an initiating event because of a fire? Yes / No
 Initiating event:

Components belong to the initiating event mitigating system? Yes / No
 System/redundant train No.:
 Fault tree identifier:

Electric (motive) power details
 Supply source identifier:
 Supply source location:
 (repeat as required for stand-by or alternative power supplies)

Control power details
 Control power source identifier:
 Control power source location:
 (repeat as required)

Control sensor details
 Sensor identifier:
 Sensor location:
 (repeat as required)

Normal and failed position of a component by operating mode

Plant mode	Normal position/state	Position/state on loss of supply	Comments

Note: The contents of this table can easily be expanded to include information on the cable pathways between the component and the sources of electric power/signals (see Appendix I).

systems, which are considered separately as an element of the fire propagation model (Section 5.5).

For each initiating event that has the potential to be caused by a fire event, as discussed in Section 3.3.2, it is necessary to determine the systems credited in the PSA with controlling and mitigating the effects of that initiating event. This is achieved in the first instance by inspection of the PSA logic models (event and/or fault trees). In turn, a list is developed of the equipment that provides the required safety function. Appendix I gives an example of the items to be listed, and the additional information required beyond that which may be included in the internal events PSA.

In deriving such information it is not sufficient to rely solely on a fire hazard analysis (e.g. Ref. [1]), since the PSA may include non-safety related systems, e.g. main feedwater systems and 'normal' electrical supplies.

It is useful and recommended that the list of equipment derived from a review of the PSA, and expanded where necessary to include complete information on the electric power and control supplies, be incorporated into a fire PSA database, which is then further extended in task 4. Table I illustrates the type and organization of component related data.

3.3.4. Identification of the human error contribution

This subtask determines the list of operator actions in the internal events PSA for which estimates of the probability of human error may differ as a result of a fire. It also provides guidance on the error probabilities to be used in the screening stages of the fire PSA.

Human actions are typically an integral part of event sequences in an internal events PSA. The failure probabilities for these human actions are evaluated as part of an internal events PSA effort, assuming a normal working environment. The same human actions may be addressed in modelling the impact of a fire on plant safety. However, the failure probabilities may have to be adjusted to take into account the unusual environmental conditions (e.g. smoke) imposed by the fire event.

The internal events PSA model normally includes a number of operator actions that contribute to the unavailability of systems. The actions are generally of two types: (a) those that occur prior to the occurrence of the initiating event, and (b) those that are required to be performed after the occurrence of the initiating event.

Failure to reinstate power supplies to a motor operated valve following maintenance is an action of type (a). Failure to recognize the need for a particular action (cognitive error) or failure to perform a particular action within a given timeframe (error of omission) is an action of type (b).

Values for the human error probabilities (HEPs) relating to the unavailability of components prior to the occurrence of fire initiating event (a) will not require re-evaluation. The HEP values assigned to actions of type (b) in the internal events

PSA are determined for particular conditions associated with the initiating event, and thus may not be applicable to the fire case. There may also be post-fire operator actions that the internal events PSA does not model; these have to be addressed in the fire PSA.

The actions required in response to a fire event may involve physical and psychological conditions that differ from those in internal events modelling, particularly for those actions that are undertaken outside the control room. For example, because of the fire effects on equipment and access routes, these actions may take longer than originally specified. For this reason it is recommended, at this stage, that each post-fault HEP be set to 1.0 to ensure that the fire related influences are not omitted from the screening analysis. Task 7 describes revision of the HEP values in cases where this assumption leads to unacceptably high consequences. It may be one of the first steps considered in undertaking detailed analysis (task 7).

Historically, one issue that has not been widely modelled is errors of commission in response to a fire (as defined in Ref. [4]). Indeed, the internal events PSA may not have explicitly modelled such errors. For this reason it is not possible in this report to provide specific guidance on this developing issue. However, it should be recognized that the probability of these errors may increase after a fire. The decision to include these errors currently remains with the analyst or with specific requirements on the scope of the fire PSA.

3.3.5. Identification of the PSA related cables

For the components identified in Section 3.3.3 it is necessary to determine which cables and circuits are required so that each particular component can perform its safety related function. The following circuits should be analysed: the motive power supply circuits, the control power supply circuits, and the instrumentation and control circuits.

Electric motive power supply circuits provide the power for operating electrically driven components (motors and valves). The control power supply circuits provide the electric power to I&C equipment; in turn, this equipment provides signals from the plant for processing, and also to the plant for the remote control of components. Process monitoring and component control are the main functions performed by I&C. Component control also includes permissive and/or interlock functions, i.e. to permit (or prevent) operation of the component when either a required condition exists or a certain signal is required.

All these functions can be provided by a dedicated electrical circuit or by electronic signals. The information can be transmitted in analog, digital or processed digital (multiplexed) form using electrical or fibre optic cables.

Each cable should be evaluated to determine the effect of its failure on the operation of the required PSA related components. It is important that all possible

failures are identified. The following failure modes, or a combination thereof, should be considered for cables:

- (1) *Open circuit*: A circuit failure that causes loss of the electrical continuity of a conductor or loss of the transmitting capability of a fibre optic cable.
- (2) *Short to ground*: A circuit failure that results in the cable conductors becoming connected to a grounded item (e.g. cable tray).
- (3) *Short circuit*: A circuit failure that results in the cable conductors coming into contact with each other.
- (4) *Hot short*: A short circuit failure in which a de-energized conductor comes into contact with an energized conductor such that the de-energized circuit becomes energized. Two types of hot short should be distinguished:
 - (a) *An intracable hot short*, for conductor to conductor shorts within a multiconductor cable;
 - (b) *An intercable hot short*, for a non-energized cable that comes into contact with a separate energized cable.

The most likely fault mode for a single conductor cable is a short to ground. Failures of this type can lead to deactivation of the electrical circuits, either by tripping a circuit breaker, causing a fuse to open, or by melting open the wire or cable. In control circuits this fault leads to loss of the control function; in instrument circuits, this fault causes either a loss of signal or a false signal at the high or low end of the range, depending on the circuit. An open circuit fault generally occurs because of collapse of the cable support structure, failure of the circuit protection devices to trip in the event of a sustained short to ground, or prolonged severe fire exposures. These two types of fault (short to ground and open circuit) can be treated similarly in terms of their anticipated system impact in the fire PSA.

For a multiconductor cable, the most likely initial fault mode is an intracable hot short, i.e. conductor to conductor faults within the cable. Faults of this type can simulate the actions of a manual control circuit switch, circuit breaker or solenoid switch. This might lead to undesirable effects, such as the reconfiguration of valves in an operating system and the opening of solenoid operated safety relief valves, e.g. on the pressurizer of a PWR, or to actuation of an inactive system. These faults can also lead to false readings on a sensor circuit. In the longer term, multiconductor cables are expected to short to ground as the fire damage progresses. The timing of this transition from an intracable hot short to a short to ground remains a point of uncertainty. In severe fire exposures, rapid transition is anticipated (within minutes or even seconds). In more moderate exposures, or when rapid intervention of severe fire exposure is postulated, a sustained intracable hot short is possible. The impact of both short term and sustained intracable hot shorts in multiconductor cables should be considered in the analysis, especially for the control cables.

Intercable hot shorts can occur in any system that includes energized power cables. Faults of this type can lead to actuation of the non-energized circuits, or to application of destructive voltages to lower voltage systems. The likelihood and duration of such intercable hot shorts also remain areas of uncertainty and debate. Depending on the plant and the fire scenarios, this may lead to serious initiating events or to additional contributors to system unavailability, and should not be overlooked.

To properly reflect this potential it is recommended that all the remotely operated PSA related components, particularly valves, be reviewed for potential problems from spurious actuation as a result of short circuit or hot short failures. The listed information (database) should include the normal or expected position of the valve and the worst case position as a result of the fire. This will necessarily involve a review of the control and operation of the valve.

It should be noted that fire induced failures in the process monitoring I&C do not directly affect operation of the required equipment, but may prevent the operator from taking the appropriate action associated with the required safety functions. The PSA analyst may decide to model such appropriate effects directly, or to include them in the modelling of human actions.

The following observations are made regarding analysis of the spurious actuation of electrical or I&C circuits.

Credit is not taken for the proper functioning of any electrical or I&C circuit that has not been completely analysed. For example, automatic signals that position components to the states required for safe plant shutdown are not taken into account unless it has been demonstrated that the fire will not affect those circuits that generate the automatic signals.

Where permissive or interlock functions (based on analog or digital technology) are used to control the PSA related equipment, all the cables associated with interlocking functions are included in the spurious analysis.

It is important that the analysis also addresses fire damage to those circuits and cables that are not directly related to the successful operation of PSA related components but may indirectly affect the operability of the required systems. A more detailed discussion of this problem is given in Section 3.3.6.

3.3.6. Identification of the associated circuits of concern

Components and cables of PSA related systems may share certain physical or electrical characteristics with non-essential systems (i.e. those not considered in the internal events PSA). Because of this interrelation, fire induced damage to non-essential systems can, in certain cases, negatively affect the operability of PSA related equipment. Associated systems of concern include any circuit whose fire induced damage could prevent operation or cause malfunctioning of the PSA related equipment.

The common power supply and common enclosure are the most important interrelation factors that should be evaluated in identifying the associated systems of concern.

Several possible failure modes have been identified for PSA related circuits that share a common power supply with non-essential circuits. These failures involve the possibility of low or high impedance shorting [12, 13], creating conditions that may not be covered or bounded by the failure modes considered in Section 3.3.5. Example situations are described in Appendix II.

Other non-essential circuits of concern are those associated through common enclosure (conduit, tray, junction box, panel, etc.). Two failure mechanisms are possible in this case: the physical propagation of a fire outside the immediate area, and the secondary ignition of essential cables under overcurrent conditions caused by the effects of a fire [12]. More detailed information is provided in Appendix II.

If the particular plant design is known or believed to be sensitive to such interactions, then the fire PSA may need to be expanded in order to ensure that the potential is properly assessed. The analyst should investigate all those factors that may affect the propagation of the above mentioned faults, such as the characteristics of the protective devices and cables, the protection of non-essential cables of concern (e.g. fire barrier wraps, physical separation), and the existence of written procedures (e.g. to shed non-essential plant loads from potentially affected power supplies). In principle, this task is part of the fire hazard analysis.

3.4. INVENTORY EQUIPMENT AND CABLES (TASK 4)

In this task, the location of components and the routing of cables, as identified in task 3 (Section 3.3), are specified. It should be noted that relevant components and cables include those that, when affected by a fire, may induce an initiated event, as well as those that are relevant to its mitigation. Basic design information for all the PSA related items gathered in PSA familiarization (task 3) is organized and documented.

It is advisable that this information includes identification of the component (identification number, system, train, basic event identifier); location of the component (building, elevation, compartment, cell); a brief description of the motor operated valve, charging pump and cable; function of the component and its position during normal operation and reactor shutdown (e.g. open, closed, on, off, energized, de-energized); the power supply; the control power; and the signals. The worksheet shown in Table I is an example of the format applicable to this task.

For cables, the information needs to be extended to include identification of the components to which the cables are connected, identification of the tray to which they belong, and the position of the tray in the compartment. Some of this information

may be difficult to obtain, or may not be needed to complete initial screening analyses. Gathering information on cable end points is important and should not be deferred until detailed analysis is performed. For further evaluation it may also be useful to specify which cables are already fire rated or protected by fire resistant coatings and fire rated wraps.

Where a deterministic fire hazard analysis has been performed, a list of the equipment expected to be damaged by a fire in each plant fire compartment can be obtained with relative ease. However, these deterministic studies usually do not identify all the accident mitigating equipment that is typically modelled in the internal events PSA model. This equipment may include components associated with the off-site power, feedwater, condensate and containment functions, and has to be identified on an 'as needed basis' in order to demonstrate that the screening criteria have been met, or to perform realistic detailed analyses in a later task.

It is essential that the method used to identify and locate the required components, as well as the instruments applied to achieve a given shutdown function, complies with that adopted in the plant's deterministic hazard analysis study. If such an analysis has not been performed it is necessary to collect the required data from the available plant documentation, in combination with plant walkdowns. Guidance for performing such activities is given in Ref. [1].

In all cases, this information is verified by performing plant walkdowns throughout the plant to review whether or not the collected data are actually supported by the physical conditions that exist in the plant. Some specialized methods such as signal injection techniques can be applied, if required, in addition to visual inspection to trace the actual cable routing.

4. SCREENING PHASE (TASKS 5 AND 6)

4.1. SCREENING BY IMPACT (TASK 5)

4.1.1. Overview

Screening by impact is the first stage of a systematic screening analysis which focuses on defining those fire scenarios that may be significant risk contributors.

Selection of potentially significant fire scenarios is made using very simple impact based screening criteria. Several factors, both qualitative and quantitative, are taken into account at this stage of the screening process, including the existence of

safety relevant (PSA related) equipment and cables in the compartments considered in a fire scenario, the compartment fire loads and the effectiveness of the barriers between the fire compartments.

At this stage of evaluation, all the equipment and cables exposed to the fire are assumed to have failed, i.e. the assumption is made that the detection devices and extinguishing systems are ineffective; fire shields or coatings are not taken into account.

Two fire damage situations are investigated: simple situations that involve fire initiation and growth within a single compartment (single compartment fire damage), and fire damage situations that involve a fire in more than one compartment (multicompartment damage).

A fire compartment cannot be screened out if:

- (1) After a fire event in the compartment there is a demand for safe shutdown functions because the plant cannot maintain normal operation, including the requirements of technical specification;
- (2) The compartment contains PSA related accident mitigating components or cabling, including the associated essential support features⁷ (as outlined in Section 3.3.3).

4.1.2. Single compartment fires

In the first step of impact based screening all those single compartment fires that may be significant risk contributors are identified.

A systematic evaluation is performed for all plant locations to identify those which satisfy the screening criteria outlined in Section 4.1.1. At this stage of analysis, evaluation is usually limited to the fire compartment level; fire cells are used in a more detailed analysis carried out at the later stages of fire PSA.

A list of compartments that cannot be screened out is retained for subsequent screening stages.

It should be noted that, for some fire compartments containing PSA related mitigating equipment, an unscheduled plant shutdown may not be necessary or enforced by technical specifications when a fire occurs in one of these compartments. Single compartment fires that involve such compartments are not potentially significant risk contributors. However, such compartments should not be eliminated from further evaluation, since they may be involved in risk significant multicompartment fires (Section 4.1.3).

⁷ This includes all those items related to the safety functions that have to be performed in order to prevent releases from in-core and ex-core radioactive sources (in the case where the PSA considers such releases).

4.1.3. Multicompartment fires

The objective of this step is to identify the potential risk significant fire damage situations that involve more than one compartment. It is assumed that the fire may spread from one compartment to another by way of the shared barriers or via the ventilation ducts that link the compartments. In the latter case, the compartments involved in the fire scenario may not be adjacent to each other. In addition, the effects of heat transfer through fire barriers are considered.

For each fire compartment retained for further evaluation, multicompartment complexes are defined by adding to that compartment all the surrounding compartments (in all directions) and all those compartments that share ventilation with this compartment. Then all possible combinations of the compartments are investigated with regard to the spread of combustion products and/or the transfer of heat to adjacent (or connected) compartments.

It is generally accepted that fire spread between more than two compartments is not considered. However, this assumption should not be applied automatically. It should be revised for multicompartment fires in which a fire starts in a compartment with a very high fire occurrence frequency, or for those scenarios that involve compartments separated by barriers with a high failure probability (especially those that require an active response to maintain fire barrier integrity) and for those compartments with significant loadings of highly hazardous flammable materials such as liquid fuel storage tanks and oil filled transformers.

Two factors are considered in selecting significant multicompartment cases: the fire load in the compartment in which the fire starts, and the effectiveness of the fire barriers that separate the fire compartments involved in the fire scenario. These two aspects are discussed in Sections 4.1.3.1 and 4.1.3.2, respectively. Guidance for screening multicompartment fire damage situations is given in Section 4.1.3.3.

4.1.3.1. *Compartment fire load*

The compartment fire load [1] is estimated on the basis of a mass inventory of all the combustibles inside the compartment, including any transient materials estimated realistically for the compartment under plant operational conditions. The fire load is calculated from the conservatively estimated mass of all the combustibles, and their specific combustion heat. The fire load density is defined as the total combustible energy per unit of compartment floor area.

On the basis of this parameter, a conservative estimate of the fire severity, e.g. the fire duration, can be made using standard time–temperature curves and/or analytical calculations. This estimate can be used to evaluate the effectiveness of the barriers that separate the compartments under consideration.

A compartment may be eliminated from the multicompartment damage analysis if the fire load in that compartment is less than a prespecified threshold value. This value can be related to a fire severity that would not challenge the minimum fire resistance assigned to any of the fire compartment boundaries. It is essential that, in calculating the fire severity, account is taken of the ventilation regime in the compartments.

4.1.3.2. Fire barriers

The effectiveness of a fire barrier in preventing the propagation of fire effects to adjacent compartments is determined by the physical performance of the elements of the barrier and the discrete effects associated with the transmission of fire effects through an intact barrier.

All the boundaries (walls, floor and ceiling) of a fire compartment credited as fire barriers, and their associated elements, have to be evaluated carefully. The elements of a boundary may include doors, ventilation dampers, shutters and penetration seals.

It is essential that the fire resistance of these elements is confirmed, taking into account the severity of the fire in the compartment under consideration. In compartments designed according to the fire containment approach (as referred to in Ref. [3]) it is generally accepted that the passive elements of the fire related barriers will not be breached by a fire initiated within the compartment. However, confirmation is needed that there are no concentrations of combustible material in close proximity to the fire barriers which may invalidate this intent, and that the elements have been properly maintained and remain fully intact.

In the case of fire cells designed according to the fire influence approach (see Ref. [3]), a passive barrier alone may not be sufficient to contain the effects of all the fires originating within the cell. At the screening process stage of the fire PSA, when active fire protection measures are not considered and credited, the assumption that the fire barriers are adequate may not be justified and should not be applied without adequate substantiation and confirmation by the analyst.

In both the fire containment and fire influence approach it is possible that the fire barrier is effective in its designated manner but that the fire effects in one compartment may affect sensitive equipment in another, adjacent compartment, e.g. by heat transfer across a shared fire barrier. In particular, this is more applicable to older plants designed according to earlier standards, where such sensitive equipment (e.g. electronic equipment or cabinets) may be located on or closely adjacent to a fire barrier.

In addition to the fire resistance qualification aspect, other possibilities of barrier failure have to be evaluated, including damper failures, doors left open, seal

failures and the existence of other openings. Particular attention should be given to those barrier elements that require an active response to maintain fire barrier integrity.

A fire spreading between two compartments is screened out at this stage if the compartments are separated by qualified and reliable barriers (e.g. a concrete wall with no openings). If any of the above mentioned failures are likely to occur, the scenario is retained for further analysis. It should be noted that definition of a qualified and reliable barrier may depend, to some extent, on the analyst. It may also be country specific because of the existence of specific requirements or regulations.

4.1.3.3. Screening

The screening criteria defined in Sections 4.1.3.1 and 4.1.3.2 are applied to each multicompartment complex. A list of those complexes that do not satisfy the criteria is created for use in the later stages of screening.

4.2. SCREENING BY FREQUENCY (TASK 6)

The compartments and multicompartment complexes that could not be screened out in the preceding task may be subjected to frequency based screening. The frequency of core damage can be used for this purpose.⁸ It can be expressed as:

$$F_{\text{core damage}} = F_{\text{damage}} \times \text{CCDP}$$

where CCDP is the conditional core damage probability determined using the internal events PSA logic model (Section 4.2.3), and F_{damage} is the frequency of fire occurrence and eventual damage to the equipment and cables.

In this step it is still assumed that a fire event affects all the equipment and cables within the compartment or multicompartment complex, and leads to their damage. Also, since this is a screening step, for single compartment cases, F_{damage} is taken to be the fire initiation frequency of the compartment, and for multicompartment complexes, F_{damage} is taken to be:

$$F_{\text{damage}} = F_{\text{fire initiation}} \times P_{\text{barrier failure}}$$

where $P_{\text{barrier failure}}$ is the probability of failure of the barriers between compartments that leads to a multicompartment fire situation.

⁸ For fire damage situations that involve ex-core radioactive sources, the term core damage should be replaced by the more general term plant damage, which includes other undesirable plant end states that are related to ex-core radioactive releases.

4.2.1. Frequency of a fire occurrence in a compartment

Fire occurrence frequencies are established for each of the fire locations. These estimates are usually derived on the basis of generic data and specific information on the fire location. Where reliable fire frequency data are available, particularly for older plants or plants with older sister stations, this should be strongly preferred to generic data. However, care needs to be taken when using plant records, particularly those that go back over many years. Plant design, protection or operation may have changed such that some events are no longer relevant.

For a non-typical compartment, a more judgemental approach must be adopted to determine the fraction of the overall nuclear power plant fire frequency that is applicable to a given location. Some guidance on this subject is given in Refs [10, 11, 14].

One of the possible approaches is to associate the fire frequency with the individual ignition sources, e.g. high voltage equipment; electric wires; electric power supplies of all types; electrical cabinets; engines, pumps and ventilators of all types; maintenance and human intervention; and ignition sources of all other types.

The frequency of fire in the compartment under consideration may be associated with the ignition sources and calculated as the sum of all the contributing factors. This frequency may be weighted by the quantity and type of combustibles present.

Whatever the method used to calculate the fire frequency of a compartment it is important to bear in mind that this frequency will never be equal to zero, since transient ignition sources (e.g. maintenance) and transient combustibles may always be present.

For single compartment fires (identified in task 5), the frequency of fire occurrence is directly used to calculate the core damage frequency. For multicompartment fires, the probability of fire spread is taken into account. This subject is discussed further in Section 4.2.2.

4.2.2. Frequency of multicompartment fires

For each multicompartment fire damage situation identified in task 5, the following procedures are usually followed:

- (1) For every fire barrier involved in this fire, estimate the probability of failure (e.g. for a door, the probability of its being open). If there is only one barrier element, the conditional probability of a fire spreading between compartments is equal to the probability of its failure. If the barrier includes more than one element (several in parallel), the conditional probability of a fire spreading (for this barrier) is calculated as the logical sum of the failure probabilities of the

barrier elements. For instance, if the barrier is a concrete wall with a door and cable penetration, the probability is calculated taking into account failure of the wall, the door and the penetration.

- (2) Calculate the frequency of the multicompartment fire damage as the product of the frequency of ignition in one compartment and the conditional probability of fire spreading to another compartment.

4.2.3. Conditional probability of core damage

At this stage of analysis, the internal events PSA model (adopted in task 3) is used (as described in Section 3.3).

For each of the fire damage situations selected in task 5, the conditional unavailability of safe shutdown is calculated taking into account the effect of a fire on the PSA related equipment and on the cables located in the compartments involved in the scenario. The internal events PSA model is used for these calculations. The following assumptions are made at this stage:

- (1) All the PSA related equipment involved in the fire damage situation is unavailable (the probability of damage caused by a fire is equal to unity);
- (2) All the post-fault human error probabilities required in the internal events PSA model for this fire damage situation are set to unity; no recovery actions are taken into account.

4.2.4. Screening

The fire damage situation should be screened out if the product of its overall fire frequency (see Sections 4.2.1 and 4.2.2) and the conditional probability of core damage (Section 4.2.3) are under a given threshold. This threshold may be defined as a specific value (e.g. 10^6 per reactor-year) or be given in relative terms (e.g. 0.1% of the core damage state frequency arising from the internal events).

One of the factors taken into account in selecting the screening criteria applied in this task is the level of detail of the PSA model. A smaller threshold is usually applied if the number of fire scenarios considered in the PSA is high; in consequence, the risk associated with each individual fire damage situation may be relatively low.

It should be noted that selection of the screening criteria is not a simple task. There are no general guidelines or ‘accepted’ numerical values that can be broadly applied. The selected value should be low enough to ensure that the screening scenarios are truly insignificant to the total plant risk. The screening threshold must be high enough to facilitate a practical analysis and should be insensitive to future refinements in the PSA sequence models, system analyses and data.

If a final damage situation is screened out on the basis of having a very low frequency but potentially still has a high conditional probability of core damage it is advisable that the fire frequency be subjected to uncertainty analysis at this stage. This analysis may indicate that careful re-examination of all assumptions is needed and that more realistic models for this fire damage situation should be developed (as described in Section 5).

5. DETAILED ANALYSIS (TASK 7)

5.1. BACKGROUND

The purpose of this fire analysis step is to reduce the level of conservatism in those fire damage situations that were not screened out in the preceding steps, and to obtain a realistic estimation of the fire risk. Details of the separation or protection provided within the plant are introduced into the analysis. The level of conservatism can be reduced in several ways by incorporating measures such as the protection provided by barriers, analysis of human actions for accident sequence quantification, and analysis of fire propagation and equipment fragility.

The exact steps taken to reduce the levels of conservatism may depend on the characteristics of the fire, the adopted methodology and the analyst's approach. Therefore, in this section a series of topics is discussed that can be used in conducting the detailed analysis.

It is important to define the concept of a fire scenario at this stage. Reference [10] states that: "A fire scenario starts with a fire source, defines the pattern of propagation, detection and suppression, and defines the equipment (target set) damage and human response. The fire scenario contains sufficient information to allow the analyst to quantify the scenario".

In the preceding sections, the discussions concentrated on fire compartments. However, in effect fire scenarios were addressed. Single compartment discussions are those fire scenarios that engulf the entire compartment, while multicompartment complexes are those fire scenarios that start from the exposing compartment and propagate to the exposed compartments. Simplified assumptions were used to represent these scenarios. It was assumed that, upon ignition, the fire affects every item in the compartment. Similarly, for multicompartment complexes it is assumed

that, upon fire effects entering a compartment, every item in that compartment is affected.

The purpose of detailed analysis is to refine the fire scenarios. The fire scenarios used in the screening stages actually represent those large spectrum scenarios that may potentially occur in the compartment. In detailed analysis, the analyst, in graduating levels of detail, redefines the fire scenario in terms of details such as fire propagation, suppression system actuation and effectiveness, and operator actions.

In plants where the PSA related systems are duplicated in segregated trains located in similar/identical fire compartments or cells, detailed analysis may be restricted to one of each type of compartment or cell. The results can then be reviewed for the remaining cells in order to identify any differences. In terms of evaluation of the frequency of developed fires, the results for worst cases can be used as those bounding frequencies for similar compartments or cells that are judged to be less vulnerable to fires.

In this step, the analyst selects one fire scenario at a time (using some form of ranking scheme), identifies the key elements of that scenario, and decides on how to proceed in order to reduce the conservatism in the plant damage frequency associated with each fire scenario. The detailed analysis techniques are heavily dependent on the specifics of the fire scenario and on the characteristics of the compartment in which the fire is postulated.

5.2. EFFECTS OF PASSIVE INTRACOMPARTMENTAL PROTECTION FEATURES

The analyst may chose to refine the frequency of equipment and cable damage by crediting those location specific features of a compartment that may provide protection against the combustion products generated from a fire of a certain severity. For example, within a compartment there may be several cells defined by non-rated fire barriers, or a group of cables within a compartment may be protected by a special coating or enclosing device.

If it can be shown that a location specific feature (or a combination of several location specific features) of a compartment will withstand a fire event of a certain severity, a fire resistance rating may be assigned to the credited feature(s). Then the analyst may reduce the frequency of the fire induced damage by establishing a conditional probability for experiencing fires with a severity that is greater than the rating assigned to the credited feature(s). It should be noted, that the possible benefit afforded by active fire protection measures is considered in Section 5.6 rather than here. This conditional probability may be based on generic fire experience data or on

the analyst's judgement, which is based on the experience gained with the occurrence of fire in generic industries.

5.3. RELATIVE LOCATION OF EQUIPMENT AND CABLES

Another method for reducing conservatism is based on the fact that the fire impact is often localized to a small area or to a specific set of equipment and cables within a room. This observation is supported by existing plant operational experience. For instance, a review of fire experience data in the United States of America, based on reports of licensee events [15], reveals that less than 10% of these fires caused damage outside the immediate area.

In general, widespread damage within a fire compartment involves two heat transfer mechanisms: radiation and convection. The radiative heat from a fire may be so strong that all the target materials are affected, and their temperatures surpass the damage threshold value. Even in the absence of very strong radiation effects, a hot gas layer may form and result in damage to all the items engulfed in that layer. The analyst may evaluate the specific features of the fire compartment in order to determine if such widespread damage is likely. The cluster of fuel loading engulfed in the fire and the fire heat release rate beyond a certain threshold are the most important factors to be considered.

For cases not necessarily susceptible to widespread damage, the analyst may choose to use a 'geometric' or a 'severity' factor to reduce the frequency of the risk significant fires considered for this compartment. These factors are conditional probabilities (given that a fire has occurred) that the fire is at a specific location within a compartment and/or of a specific severity.

This step includes evaluation of the exposure fires resulting from each potential ignition source in a compartment. The ignition sources in a compartment can be divided into two groups: fixed and transient.

Through use of fire modelling and/or industry and experimental data, the potential for a fixed ignition source to cause plant damage of a particular type is evaluated. In many cases, fire damage calculations may demonstrate that the postulated damage is not credible prior to fire self-extinguishing, and the scenario is screened out.

In the case of transient ignition sources, fire modelling is used to predict the critical distance between the exposure fire and the target beyond which no damage (or ignition) will occur, and the critical combustible load required to cause damage.

Using the results of this evaluation, the frequency of fire damage may be reduced on the basis of the probability that the postulated transient fire occurs within the critical distance and/or the probability that a compartment contains the evaluated

critical combustible loading. For in situ combustibles, the geometric factor can be based on the distribution of ignition sources in the cell.

Appendix III provides an example in which a fire damage situation that has been found to be a significant core damage contributor is broken down into more specific fire occurrence scenarios.

5.4. PROBABILITY OF A HOT SHORT

Several significant hot shorts may have been identified for the systems under study. Typically, this applies to motor and other automatically operated valves that have to remain in their prefire position. For example, a hot short may occur in the control circuit of a valve, thus simulating control switch closure. Depending on the location of the fire or the special wiring configuration, a hot short may either be impossible or have a low probability of occurrence. The probability of a hot short, as seen in the example provided in Appendix III, can be evaluated by analysing the specific characteristics of the affected circuit. Information on the probability of a hot short can be found in Ref. [16].

5.5. HUMAN ACTIONS

In the preceding steps it is stipulated that a proper screening approach should give no credit to the possibility of human actions (after the occurrence of the initiating event) in the frequency evaluation. It should be noted that the human actions addressed in this section are limited to those that are related to the core cooling and containment functions and do not include those that are needed to control or extinguish the fire (covered in Section 5.6).

It should be realized that the probability of human error needs to be evaluated from an integrated view of the entire chain of events, i.e. from ignition of the fire to equipment and cable damage, to additional failures and, finally, to manifestation of plant damage. The analyst should take into account the time window available for the operator to complete the task, the possibility of smoke or other products of combustion that may hinder his/her actions, the cues (alarms, instrumentation, etc.), the availability of lighting, the adequacy of post-fault procedures, the availability of access pathways, etc.

During the screening phase, the potential for reducing the probability of plant damage by human actions is not taken into account. The following assumptions are typical:

- (1) Increasing the failure probability of those operator actions that are included in the internal events PSA model to account for the additional stress placed on operators because of the postulated fire event, as well as the potential existence of contradictory signals induced by fire induced damage to instrumentation and indication equipment and cables;
- (2) Neglecting the potential for recovering the postulated damaged components by operators;
- (3) Neglecting any actions that require operators to enter or pass through the affected fire area, or delaying the timing of such actions until after the fire is conservatively assumed to have been fully suppressed.

At this stage, operator actions can be examined to assign a more realistic failure probability to the actions credited in the internal events model, and to include recovery of a failed component, if such recovery is possible.

This stepwise decrease in the HEP modification factor is deemed justifiable on the basis of the fact that, as the time available for manual action is increased, the potential stress on and distractions for the operators, and the accessibility of the local control panels, will decrease.

An example of additional human actions is that if a certain valve should remain open to allow component cooling water to enter the reactor coolant pump seal heat exchanger, a fire, via hot shorts, may close the valve inadvertently. To overcome this failure, a plant operator may have to walk to the valve and try to open it manually. To be able to do so, the control room operator must first recognize that the valve has indeed closed because of the fire before summoning a plant operator to manually open the valve. The plant operator has to find his/her way to the valve without being affected by combustion products. All this has to be done within the available time window, as dictated by the core damage and plant damage conditions.

Using well established human error probability methods (e.g. Ref. [17]), the fire analyst has to establish the performance shaping factors, and thence the human error probabilities.

5.6. ACTIVE FIRE PROTECTION MEASURES

Nuclear power plants are often equipped with sophisticated active fire protection equipment and supported by a well trained and dedicated fire brigade. These defences reduce the possibility of a fire continuing to burn for an extended period, or spreading to other combustibles or to other fire compartments or cells. In the preceding screening steps it is assumed that these systems are unavailable and that a fire is capable of damaging everything within a compartment.

It should be noted that best estimate modelling of active fire protection measures is an important element of the detailed analysis. Conservative assumptions made at the screening stage may be unacceptable, especially where the redundant safety trains are located in one fire compartment. Another extreme assumption sometimes made by analysts, i.e. that the active fire protection measures fulfil their function, may lead to considerable underestimation of the core damage frequency.

To use a more realistic approach that includes the effects of fire protection and fire fighting systems, the analyst may follow a methodology which quantifies the growth and suppression response to fires. References [10, 18–21] provide a spectrum of different approaches. The key element of these methods is estimation of the probability of failure to suppress the fire before damage to a critical set of equipment and cables.

The analyses include two competing elements: the time for a fire to grow from a ‘pilot’ size to a fully developed fire (i.e. one that is capable of damaging a critical set of equipment and cables), and the time to detect and control the fire.

Estimating the time required for damage to occur in principle requires use of numerical models that simulate fire propagation in compartments and cells. The existing analytical tools include models for fire ignition, flame growth and thermal effects within a fire enclosure. A number of computer codes are available for such analyses; selection of the code is at the discretion of the analyst. Examples of codes that have been used to support fire PSA studies can be found in Ref. [22].

The time needed to detect and control a fire is estimated from industrial experience data augmented by the room characteristics in terms of the types of fire detector available, the occupancy level (i.e. the possibility of detection by personnel), the type of fire extinguishing equipment and the accessibility of the location to the fire brigade. The reliability of the fire protection equipment is addressed as an integral part of this analysis.

It is important that the adequacy and anticipated performance of the installed fire detection and suppression systems are assessed and factored into the timing analysis; plant walkdowns need to be carried out for this purpose. For example, placement of the fire detectors relative to the postulated fire source(s), any displacement of the detectors from the compartment ceiling, and obstructions that might have an impact on early smoke and heat flow behaviour (e.g. large beams or changes in the ceiling level) should all be noted, because each of these factors could have a significant impact on the timing and reliability of fire detection system actuation. This review should include considerations of detector cross-zoning, which will require more than one detection signal before a fixed suppression system is actuated.

Example approaches are provided in Refs [11, 18–21] and involve time dependent models to describe fire detection and suppression in probabilistic terms. A

statistical fire suppression model is used to predict the probability of failure to extinguish the fire within the time required for it to cause specific damage to a piece of equipment. In this approach, the initiation time for fire detection and suppression is expressed in terms of a probability distribution that takes into consideration various plant characteristics specific to the fire location.

5.7. FIRE PROPAGATION EVENT TREE ANALYSIS

The fire propagation event tree is a useful tool for describing more realistically the risk associated with fire growth within a fire location. The event tree model introduces a discrete set of fire growth stages, each of which corresponds to a certain state of system damage and the associated plant risk. The event tree headings are composed of events that model the probability of achieving specific fire growth stages. An example of a simplified fire event tree is given in Appendix IV.

This model is useful for splitting a fire damage situation that originates in a specific location into a set of fire damage states of differing plant damage severity, and for calculating the contributing plant risk for each of the fire damage states. When quantified, the event tree predicts the risk from each level of fire growth.

Quantification of such a model requires that a probability be estimated for each of the fire growth stages. The fire propagation modelling techniques mentioned in Section 5.6 may be used for this purpose. The position of the equipment and cables in relation to the potential fire sources can be taken into account in this assessment (e.g. by introducing a reduction factor that accounts for the fraction of fires originating in the fire location which would lead to the specific fire growth stage). The conditional probabilities of core damage, given the corresponding fire growth stages, are determined from the existing internal events PSA model (setting the probability of failure of the damaged components to unity).

6. SPECIAL ISSUES

6.1. INTRODUCTION

The protocol for fire PSA described in this report has been written in very general terms and, with the exception of a few instances, no details are provided. It is intended to apply to all parts of a nuclear power plant. However, several issues have been found to be important to the proper execution of fire PSA methodologies and to differ significantly from those discussed in Sections 3–5, either because of the

specific features associated with the location or because of some additional aspects that should be taken into consideration.

6.2. ANALYSIS OF THE CONTROL ROOM

It is essential that the fire PSA model for the main control room accounts for the specific features associated with this location. The main control room differs from other plant areas in several aspects.

In the event of a fire in the main control room, the potential impact on redundant safety trains is higher than that of all other plant areas. The potential also exists for the operator to receive contradictory information and for there to be an impact on operator habitability.

However, it should be noted that the main control room is continuously occupied and therefore the likelihood of fire detection in this area may be greater than that in all other plant areas.

Regardless of the level of damage that is actually sustained as a result of a fire, production of smoke may necessitate evacuation of the control room. It should be noted that the time taken for smoke to fill the room is relatively short, around 10–15 minutes, depending on the tests carried out (as shown by control room console fire tests). Therefore, the control room should be evacuated quickly when a fire cannot be extinguished. Often, a second shutdown panel is available. Under such circumstances, the operators should isolate the main control room and shut down the plant using the alternate shutdown capability.

Regarding provisions for operator remote shutdown actions, it is essential to check whether the operators have proper training, equipment and procedures. Any resulting degradation in operator performance has to be accounted for in the PSA model.

6.3. CABLE SPREADING ROOMS AND OTHER SENSITIVE PLANT LOCATIONS

The cable spreading rooms, switchgear rooms and other control equipment rooms tend to become natural centres of convergence for equipment and wiring. These compartments contain electrical equipment and cables that may belong to more than one safety system train. Therefore, the potential impact of a fire on redundant safe shutdown and other PSA related equipment is likely to be higher than that of other plant locations. There is also a higher probability for spurious actuation because of electrical fire induced shorts in these locations. In general, these compartments cannot be screened out during the initial stages of analysis (tasks 5 and 6).

Typically, the PSA model for these compartments will have to include some quantification of the growth and effects of fires in order to justify that there is no, or only a very low probability of, loss of redundant safety items. The layout of these rooms in plants may make such justification extremely difficult. In such cases, the method of analysis does not differ from the guidance provided in Sections 3–5. In turn, these locations could be important fire risk contributors, and therefore are expected to feature prominently in the documentation and findings of the analysis.

6.4. ENVIRONMENTAL SURVIVAL OF EQUIPMENT

One of the important issues to be considered in a fire PSA is the potential damage to equipment caused by combustion products or by fire suppression agents.

With regard to the impact of smoke damage, experts recognize the scarcity of information on the vulnerability of components and the limited conclusions that can be drawn in a PSA study. Therefore, the analyst may treat this issue with discretion, depending on the special features of the plant under study.

Regarding the potential damage caused by actuation of fire suppression, it should be verified that the PSA model takes this into consideration.

6.5. FIRE INDUCED EXPLOSIONS

During the screening process or the detailed analysis, the potential for fire sequences that leads to a consequential explosion may be identified. These risks may arise through a range of potential causes:

- (1) Leakages that release flammable fluids, establishing an explosive atmosphere in confined or unconfined locations;
- (2) Leakages of high pressure hydraulic oil, leading to an oil mist explosion;
- (3) Fires that spread to containers of highly flammable liquids, leading to vessel breach and resulting in explosions of boiling liquid vapour;
- (4) Fires that spread to pressurized containers of flammable or non-flammable fluids, leading to vessel disruption and missile generation.

It may be outside the scope of a fire PSA to provide a best estimate assessment of the consequences of explosions arising from fire sequences because the damage spread mechanisms from explosions (including blast effects and/or missiles) require that different methodologies be applied. However, it is important that these potential hazards are listed and attached to the report of the fire PSA for completeness. This list

provides input to the plant's overall fault schedule, for inclusion under the relevant analysis topics.

6.6. INTERACTION OF THE CONTROL SYSTEMS

The potential for physical dependence between the control room and the remote shutdown capability (e.g. because of common components located in the control room, inadequate co-ordination of fuses, or the specific configuration of the thermal overload protection devices) is an important issue to be considered in the analysis.

Appendix V provides an example of the potential for loss of the alternate (remote) shutdown capability during a control room fire.

The deterministic fire safety analysis has to be reviewed to ensure that the safe shutdown circuits have been located in areas that are physically independent of, or can be isolated from, the control room for an exposure fire that causes loss of control from the control room. Any dependencies that result in a potential degradation in the remote shutdown capability in the event of a control room fire should be accounted for in the PSA model.

6.7. INTEGRITY OF THE CONTAINMENT

Also of importance, although not immediately obvious in the context of a fire PSA, is consideration of the integrity of the containment boundary. The special issues relating to containment aspects are threefold:

- (1) Prevention of containment bypass sequences via high pressure–low pressure interfaces, together with the potential degradation in the redundancy related to isolation via hot shorts.
- (2) Failure of the active containment isolation provisions that may be required to operate, prevent or mitigate the release of radioactivity from the containment, which forms the last barrier to release. It should be noted that in some plants it may be necessary to critically examine the ability of the containment seals and penetrations to withstand the challenge of the fire scenario.
- (3) Fire induced degradation in the active systems used to sustain containment performance during design basis and beyond design basis accidents, e.g. decay heat removal and containment spray systems, and the hydrogen suppression systems in PWRs. These systems feature in most modern Level 1 plus and Level 2 PSAs. It is important that the fire PSA analyses and confirms the performance of these systems in the same way as for other PSA related systems and components.

6.8. CONDUCTING A FIRE PSA IN THE EVENT OF INCOMPLETE INFORMATION

In practically all cases, the fire PSA has to be conducted with some level of incomplete information. For those elements of the analysis for which the required information is unavailable, the analyst has to make conservative assumptions. It should be noted that in some plants certain elements of the analysis require a large amount of data that are difficult to collect, e.g. older generation plants may not have detailed records on cable routing. For such cases, the fire analyst can use information on equipment location and cable routing patterns, and make very conservative assumptions for the potential location of PSA related cables.

In all cases where assumptions are made to address incomplete data it is very important that the assumptions and their basis/rationale are properly recorded, for possible future confirmation and re-examination.

7. ANALYSIS OF THE RESULTS (TASKS 8 AND 9)

7.1. UNCERTAINTY ANALYSIS (TASK 8)

The objective of this task is to provide qualitative discussion and quantitative measures of the uncertainties in the results of the fire PSA. This effort focuses on the assessment of various fire related risk measures, e.g. the frequency of core damage arising from a fire, the frequency of the fire induced accident sequence categories associated with the specific plant damage states, and the dominant fire induced accident sequences.

As in an internal events PSA, there are three major categories of uncertainty in a fire PSA model: lack of completeness of the model, inadequacy of the model and uncertainties in the model input parameters.

Safety Series No. 50-P-4 [4] provides more detailed discussion on the potential sources of uncertainty in a PSA model and on the methods used to propagate uncertainties through the model. This discussion is fully applicable to a fire PSA. Some additional points specific to a fire PSA are pointed out in the following subsections.

7.1.1. Completeness of the model

Some of the most significant potential sources of uncertainty arising from lack of completeness of the fire PSA model are: screening of fire scenarios, simplified

treatment of multicompartment fire scenarios, fire induced faults in electrical and control circuits, human errors of commission and completeness of the internal PSA model.

The uncertainties that belong to this category are difficult to assess or quantify. It is essential that qualitative discussion is provided in the PSA on potential concerns and that an analytical approach is applied to minimize the impact of these uncertainties.

7.1.2. Modelling adequacy

Even for those scenarios that have been identified, the event sequence and system logic models do not precisely represent reality. Simplifications are made to achieve a manageable model or to compensate for lack of knowledge. Uncertainties are also introduced by the relative inadequacy of the mathematical models, the numerical approximations and the computational limits. Regarding uncertainties in the fire PSA model, several modelling issues specific to a fire should be mentioned.

One of the relevant sources of uncertainty in the fire PSA model is the quality of the computational models and codes currently used for the physical modelling of fire growth and for assessing the fire induced damage to equipment and the fire resistance of barriers. An additional source of uncertainty is the lack of quality of plant specific information relevant to the fire modelling, or even the lack of required plant specific information (the exact routing of cables, the location of ignition sources, etc.).

For this category, sensitivity studies are performed to assess the importance of these uncertainties (see Section 7.2.1). It is also possible to quantify some of these uncertainties through formal uncertainty analyses [23].

7.1.3. Input parameter uncertainties

The parameters of the various models used in a fire PSA are not exactly known. Two sources of uncertainty are: lack of knowledge of the exact value of the particular parameter (because of a scarcity or lack of operational/experimental data), and the variability in the parameter within the population of plants or components.

The relatively high contributors to an uncertainty of this type include the input parameters used for the physical modelling of fire growth and for assessing the fire impact on equipment and cables (fire severity data and equipment vulnerability data), the fire occurrence frequency, the reliability rates of the equipment used in the fire detection and fire extinguishing systems and the probability of human errors related to post-fire actions.

Quantitative analyses of these parameters, based on formal uncertainty propagation techniques [4], are recommended. The guidance provided on this subject in Ref. [4] is fully applicable to a fire PSA.

7.2. SENSITIVITY AND IMPORTANCE ANALYSES (TASK 9)

A sensitivity analysis determines the sensitivity of the PSA results to the input assumptions, models and data. An importance analysis determines the importance of contributors to the core damage frequency, the plant damage state, the accident sequence frequency and the system unavailability. Such analyses are useful for interpreting the results of the fire PSA analysis, and should be regarded as complementary to an uncertainty analysis rather than as a replacement for it.

7.2.1. Sensitivity analysis

A sensitivity analysis is performed whenever an issue or parameter that affects the outcome of an analysis cannot be treated in a fully satisfactory way within the main analysis. The principal task for the analyst is to select those parameters or aspects of the analysis that require study.

The typical issues that can be explored in a sensitivity analysis are the model used in the analysis, the data upon which the quantification is based, the screening criteria used to identify the significant risk areas, and the approximations and assumptions introduced. Sometimes all these issues can be explored in the same sensitivity analysis, but it is preferable to explore each issue separately. In relation to fire sequences screened out against frequency criteria, particular attention needs to be given to the sensitivity of low frequency–high consequence fires to data changes.

The exact character of the sensitivity studies will depend on the details of the base case analysis, and are not specified further here. It should be emphasized that it is not necessary to employ complex or mathematically rigorous methods which may obscure the real issues or cloak them in false precision. What should be stressed is that a properly constructed sensitivity analysis can be a good way of illuminating which poorly understood issues are important, which are unimportant, and why.

7.2.2. Importance analysis

An importance analysis is used to measure the relative contribution of individual fire related and random failure events to core damage frequency, plant damage state, etc. By this means, analysis can focus on those failures that are the primary contributors to risk associated with particular fire scenarios and to the overall plant risk associated with a fire or internal events.

Ranking by importance is used to identify the significant accident sequences, the fire induced initiating events, the system failures, the component failures and the human actions. The most important failures can then be examined to ascertain whether it is possible or practicable to improve the modelled frequency of the fire induced initiating event or the unavailability of system components or operator

actions. In this way, the results of the fire PSA are either accepted (the documented analysis provides justification that the plant design is acceptable) or the analysis model will have to be revised and/or additional features implemented to reduce the frequency of core damage, etc. This will involve returning to some or all of the topics described in Section 5.

8. DOCUMENTATION OF THE ANALYSIS (TASK 10)

8.1. OBJECTIVES

Documentation of a fire PSA provides comprehensive and systematic presentation of the complete analysis and the main results of the study. The prime objective of this documentation is to fulfil the user requirements in a manner consistent with the PSA objectives. However, PSA documentation plays an important role in the analytical process itself. Clear and traceable documentation of all the intermediate tasks facilitates analysis and is one of the important attributes of the QA programme for a PSA study.

8.2. GENERAL PRINCIPLES

Reference [4] provides details of the results of the internal events PSA; many of the general principles apply to documentation of the fire PSA. The documentation should be well structured, clear and easy to follow, review and update. Explicit presentation of the assumptions, conclusions and limitations of extending and updating the PSA is also of critical importance. Reference [4] recommends that the conclusions be distinct and that they reflect not only the main, overall results but also the contributing analyses, and that emphasis should be given to analysis of the uncertainties in the data and to the sensitivity analysis, where the effects of assumptions, limitations and conservatism in the methods and modelling are clearly demonstrated.

8.3. ORGANIZATION OF DOCUMENTATION

Documentation of a fire PSA includes both unpublished internal documentation and external documentation that displays and interprets the final results of the PSA in line with the objectives of the study. It is important that the documentation provides,

within the report or by reference to available material, all the information needed to reconstruct the results of the study.

The final report on a fire PSA can be a separate document (if the fire PSA is an extension of the existing internal events PSA) or an individual part/volume of the overall PSA report. It is recommended that the detailed information used in the study be presented in appendices, which could contain major parts of the inventory of fire compartments, important assumptions, and detailed models and data.

All the intermediate analyses, calculations or assumptions not published in any external reports should be retained as working material, notes or computer outputs. This documentation has to be suitably organized to ensure that the information required for the future reconstruction and updating of each detail of the analysis is easily retrievable.

Organization of the fire PSA documentation should be governed by two general principles:

- (1) *Traceability*: To review and update the analysis it should be possible to trace any information with the minimum of effort;
- (2) *Sequentiality*: The order of analysis in the final documentation should follow as far as possible that of actual performance, addressing the tasks described in these guidelines.

The final report of the fire PSA usually includes the background, objectives and scope of the study, and the main assumptions and methods used in the analysis. It also provides an overview of the major tasks of the study, as well as the contents and organization of the documentation.

The main body of the report should correspond to the major tasks of the analysis, as described in this report: identification of the fire compartments and cells; the potential fire induced initiating events; the equipment used to mitigate the fire initiating events; the inventory of fire PSA equipment; the screening processes (impact, frequency); multicompartment considerations; quantification; the factors that influence a reduction in the calculated fire risk; and the uncertainty and sensitivity analyses.

Regardless of the specific characteristics of each PSA task, the report discusses the inputs and products of each task, as well as the assumptions and methods used.

The report could include a separate section devoted to the display and interpretation of results. In this section, the results obtained at each major step of the analysis (and discussed in the preceding sections of the report) are integrated and displayed in detail, together with the important engineering insights gained from the analysis.

The report concentrates on identifying and describing the most risk significant fire sequences. An assessment of the uncertainty, importance and sensitivity analyses

is made and all the key factors that affect the credibility of the results are discussed. Finally, more general conclusions and considerations are outlined.

These considerations may address the overall fire safety assessment of the plant and any modifications made to the design, procedures, training and licensing of safety issues. Typically, such considerations concentrate on the implementation of risk reducing measures, e.g. development of procedures and installation of additional automatic fire detection or extinguishing systems. However, the fire PSA may also identify areas where certain fire safety measures are marginal to nuclear safety.

9. EX-CORE SOURCES OF RADIOACTIVE MATERIAL

The preceding sections addressed prediction of the frequency of reactor core related events. The reactor site also includes other (ex-core) radioactive materials that could be affected by an internal fire. An outline is given of the approach used to assess the importance of fire related releases, where these are considered significant or are required within the overall reactor PSA.

The PSA Procedures Guide for Level 1 PSA [4] recommends (in task 10) that “A list should be made of all the sources of radioactive releases (including content and form) from which accidental releases could be postulated. For example, for an LWR this list should include the reactor core, the refuelling pool, spent fuel handling facilities and waste storage tanks. If any of these sources are excluded from the PSA, the exclusion should be justified in the study...”.

A fire may affect these sources in a manner not considered in the internal events PSA, either indirectly by the affect on the availability of the process systems, or directly by the influence of the energy released by the fire on stored materials. The nature of the ex-core sources of radioactive material at nuclear power plants is such that the reliance and interactions between systems are less onerous and complex than sustenance of the reactor heat removal processes. This does not infer that safety provisions and procedural controls are any less important.

Reference [24] examines use of PSA for nuclear installations with a large inventory of radioactive material and describes the method used to assess potential hazards. The method includes a number of progressive steps, some of which explicitly require consideration of a fire. In general terms, the processes involved in the control of radioactive material in such facilities are more appropriate to power plant ex-core sources. Hence, a general approach can be applied, provided that due recognition is given to the effects of a fire. The general approach can be summarized as follows:

- (1) *Preliminary hazard screening*, to screen/identify the potential consequences arising from the ex-core sources, taking into account the possible accident conditions;
- (2) *A review of the protection barriers and safety systems*, which is similar to tasks 2 and 3, as described in Section 3;
- (3) *Hazard identification techniques*, in which an internal fire and its effects are covered by the assessment;
- (4) *Accident sequence modelling*, to evaluate the progression of accident sequences and to permit quantification of the fire event frequencies and the probability of undesirable outcomes or consequences;
- (5) *Consequence analysis* (for the event sequence end states determined in item (4)), which is specific to a fire PSA and has to take into consideration the effects of the fire on the generation, composition and release of radioactive material within and external to the plant.

Many of the features of automatic and operator response to the potential fires that affect the reactor core are also relevant to analysis of the fires that affect the ex-core sources of radioactivity. In particular, during initial screening no credit is given to post-accident operator actions or fire protection or mitigation, as included in tasks 2–6. The topics discussed in task 7 may be used to reduce conservatism in the approach.

Appendix I

INFORMATION REQUIRED FROM AND IN ADDITION TO THE INTERNAL EVENTS PSA MODEL

I.1. INFORMATION IN THE INTERNAL EVENTS PSA MODEL

A simplified diagram of the electrical and control circuit is given in Fig. I.1, and shows the power circuit that provides the motive power to a pump motor, e.g. an emergency coolant injection pump or an auxiliary feedwater pump. The motive power to a pump is derived from one of the plant's electric power buses, perhaps via a transformer. Activation of a pump is controlled by the switchgear, usually located in an individual bus cabinet. In turn, this switchgear requires low voltage power for operation, in addition to control signalling. The latter is obtained via a control relay, which responds to demand signals from either the control room or the automatic actuation system.

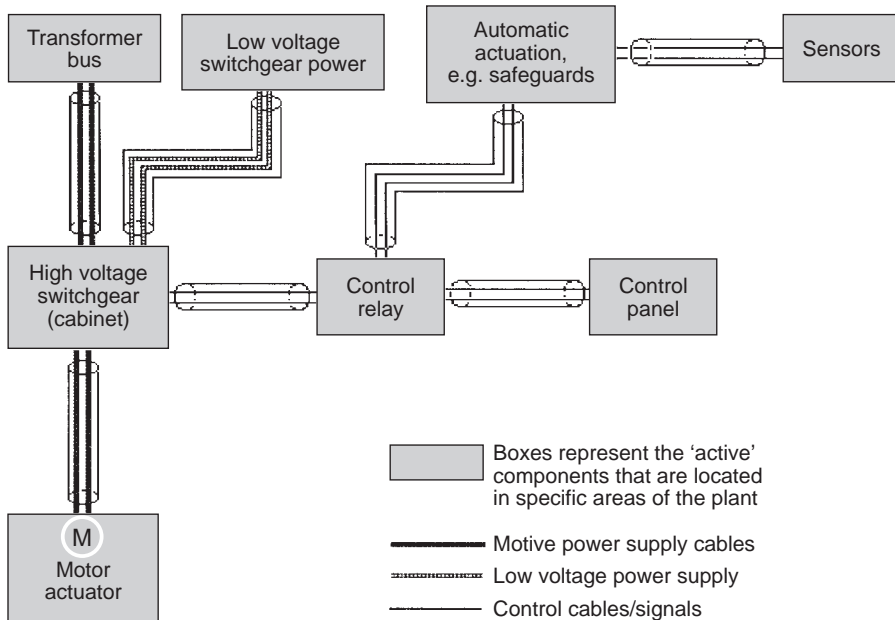


FIG. I.1. The electrical and control circuit that provides motive power.

Each of these discrete components may be located in a different area of the plant. The internal events PSA model should have taken into consideration the individual failure modes of these components, either explicitly or, in some cases, implicitly. For example, the failure of a pump to start on demand may be caused by inherent pump motor failures or no electric power supply. The causes of such an electric power supply failure can be modelled as follows:

- (1) Loss of the power supply from the bus, or switchgear failure (to provide the power supply when given the demand signal), or loss of the low voltage power supply to the switchgear;

or in a basic simplified form:

- (2) No electric power is supplied to the pump motor,

which includes the three failure modes specified in item (1).

Similarly, the PSA models may have considered the details of control system operation, or have treated the failure modes in a simplistic manner.

The internal events PSA does not usually have specifically modelled cable failures, i.e. the electrical links between the ‘active’ components in the example.

I.2. FIRE PSA NEEDS

In addition to the active components that are to be found in the internal events model, shown by boxes in Fig. I.1, details of the cable paths between components are critical to the analysis. A fire induced failure in any of the cable routes shown has the potential to affect the successful operation of components, which together support operation of the motor. Similar principles apply to actuated valves.

It should be recognized that the cable routes shown will almost certainly involve a number of additional areas of the plant, including those rooms or tunnels that contain only cables. Thus, an analysis based solely on the location of the ‘active’ components in the PSA will be incomplete and not reflect the real fire risk.

I.3. HOT SHORT POTENTIAL

In Section 3.3.5, the possibility of hot shorts in control cables is identified. Thus, referring to the schematic diagram, hot shorts occurring in any of the control circuits that provide an actuation signal to the switchgear, which supplies the component with motive electric power, should be considered in the analysis.

Deleterious hot shorts in the power supply circuits (high or low voltage), or in circuits that are disabled (i.e. not 'active' or 'live'), e.g. because of removal of a fuse, are not considered possible, and are therefore excluded.

I.4. OUTPUT

The inventory of cable paths produced under task 3 of the fire PSA may thus be developed by one of two alternative means:

- (1) For each 'active' component established as being fire PSA related, identify the cable paths between the component and its electric power and control sources (the cables shown in Fig. I.1). The analysis will thus be restricted to those compartments and cells that contain fire PSA related components.
- (2) For each fire compartment or cell, identify the PSA related system components (including cables) that are located in the compartment or cell. Some of this information may already be available from an existing fire hazard analysis.

It is recommended that the results of the analysis be incorporated into a database (or a number of linked databases) that will be used in subsequent tasks and in reporting the fire PSA.

Appendix II

ESSENTIAL EQUIPMENT MALFUNCTIONS RESULTING FROM FIRE INDUCED DAMAGE TO NON-ESSENTIAL CIRCUITS

*The following examples are extracted from guidelines developed
by the United States Department of Energy [12]*

II.1. INTRODUCTION

The examples provided in this Appendix illustrate the indirect effects of a fire resulting from fire induced damage to the associated cabling in non-essential circuits. Sections II.2 and II.3 show the effects of short circuit overcurrent protection in the event of low and high impedance faults, respectively, in non-essential circuits. Section II.4 gives an example of the physical propagation of a fire between non-essential and essential circuits, and Section II.5 illustrates the effects of secondary ignition caused by fire induced overcurrent conditions.

II.2. LOW IMPEDANCE FAULTS

In the event of a low impedance fault of an associated non-essential circuit, the circuit comes into contact with the ground, which results in a high short circuit current and leads to interruption of the faulted circuit by a protective device. If the protective devices used at various levels of the power supply circuit are not properly co-ordinated (to provide selective tripping), the power supply to the required (PSA related) equipment may be interrupted. In a properly co-ordinated circuit, fire initiated faults are rapidly isolated by the protective device located nearest to the fault before the fault current propagates, causing the tripping of any protective device upstream of the required power supply.

An example situation is illustrated in Fig. II.1. A fire in a single fire compartment (II) damages essential (PSA related) pump B and non-essential pump X, and their cabling. Since non-essential pump X is not redundant, nuclear safety appears to be maintained. However, if circuit breakers 1 and 2 are not properly co-ordinated, the main breaker 1 may trip on a short circuit overcurrent prior to the tripping of breaker 2, interrupting the power supply to the required (PSA related) pump A.

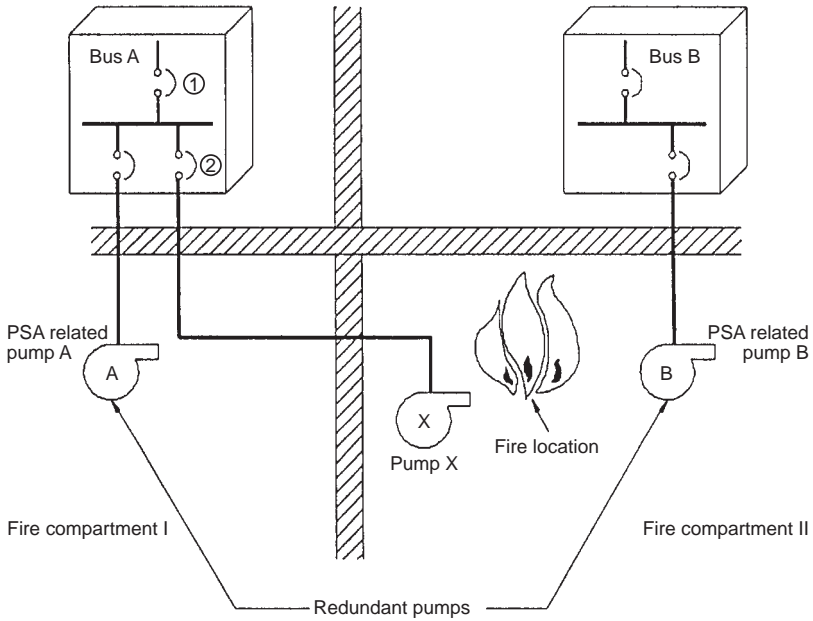


FIG. II.1. Indirect effects of fire induced damage in a non-essential circuit (low impedance faults).

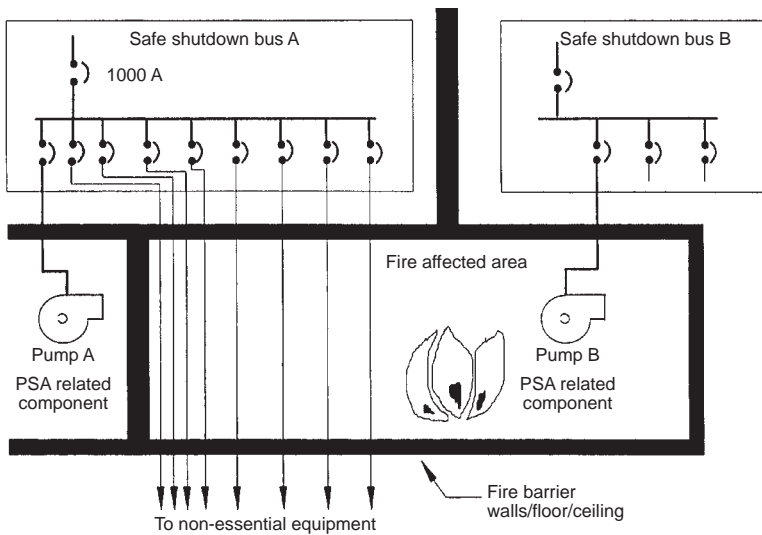


FIG. II.2. Indirect effects of fire induced damage in a non-essential circuit (high impedance faults).

II.3. HIGH IMPEDANCE FAULTS

In the event of a high impedance fault (with the fault contact erratic or not firm), the fault current is low and may not be of sufficient magnitude to trip the individual protective device. A co-ordination problem exists if a certain number of non-essential cables fail simultaneously under high impedance conditions (multiple high impedance faults). In this case, the upstream feed breaker to the required power supply may trip on an overcurrent prior to tripping of the individual load protective devices, interrupting the power supply to the required (PSA related) equipment.

An example situation is shown in Fig. II.2. A fire that originates in a single fire compartment is assumed to damage (PSA related) pump B and to cause multiple high impedance faults of the cabling related to non-essential equipment. These faults lead to tripping of the main circuit breaker (1000 A), interrupting the power supply to essential (PSA related) pump A.

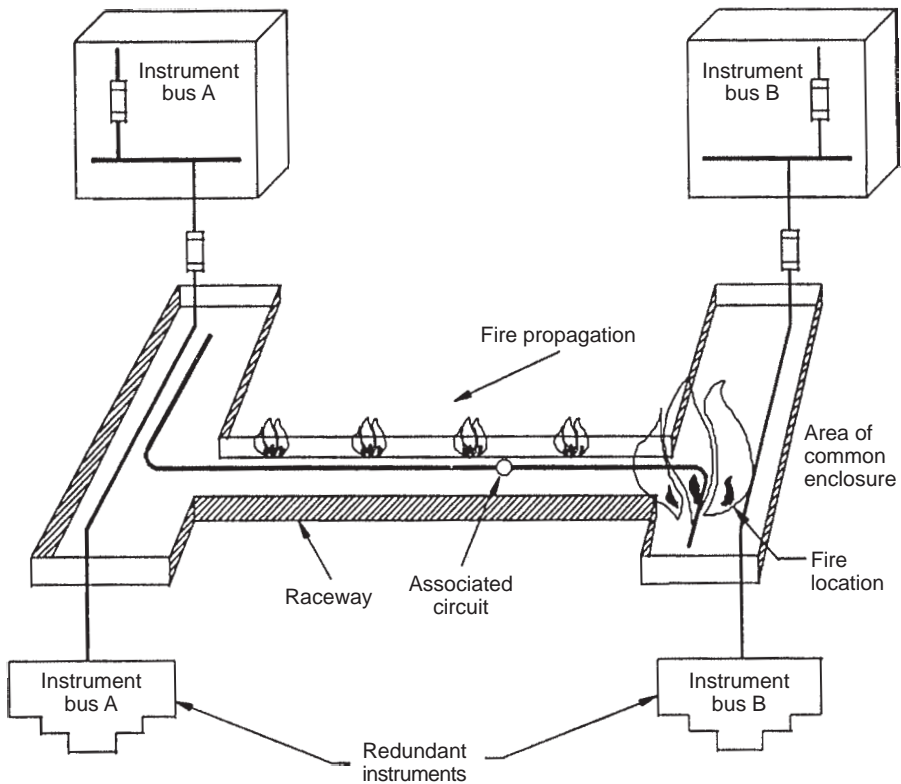


FIG. II.3. Physical propagation of a fire between non-essential and essential equipment.

II.4. PHYSICAL PROPAGATION OF A FIRE

Physical propagation of a fire from one location to another may occur where materials are present that can burn. The fire may spread from a non-critical to a critical fire area through exposed cables that contain non-retardant material and are not protected by fire stops. An example situation is shown in Fig. II.3.

II.5. SECONDARY IGNITION

Fire initiated electrical faults in inadequately protected non-essential cables may lead to overcurrent conditions, resulting in secondary ignition. A secondary fire will damage those essential cables that share a common enclosure. An example is shown in Fig. II.4.

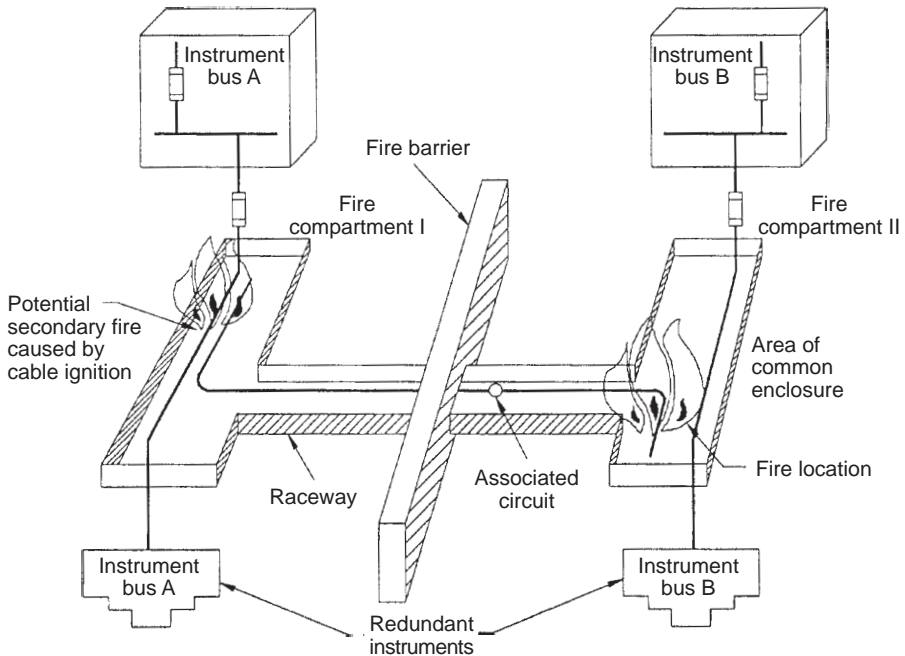


FIG. II.4. Secondary ignition of a fire caused by a short circuit overcurrent in non-essential equipment.

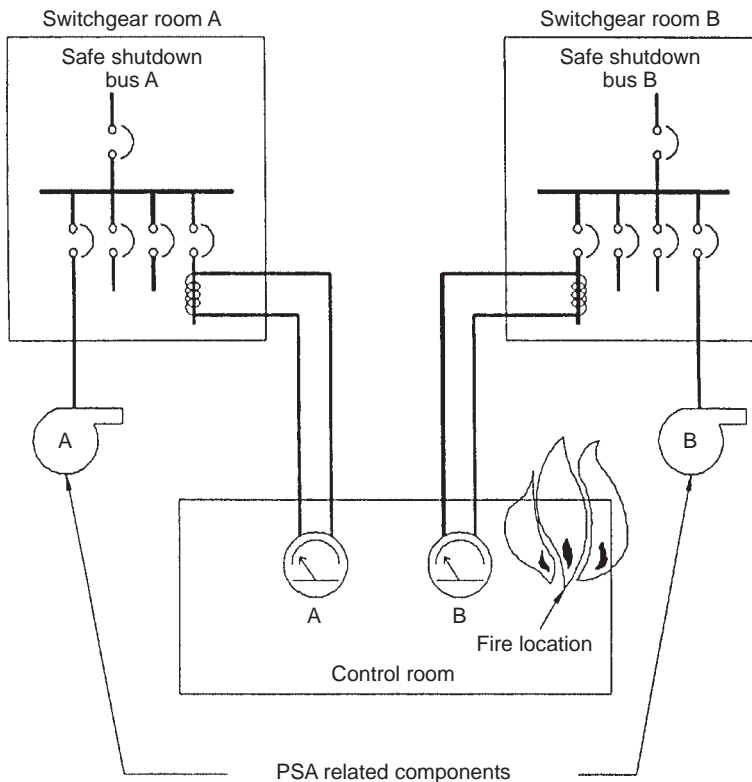


FIG. II.5. Secondary ignition of a fire caused by open current transformer secondary windings.

Secondary ignition may also occur in secondary winding of the control power transformers because of an open circuit caused by the effects of a fire. The current power transformers used throughout the electrical distribution system to monitor bus current are designed to transform high primary current into low secondary current. Because of the high primary to secondary current ratio, an opening in the secondary circuit produces excessively high voltages and may result in ignition of the transformer materials. An example situation is shown in Fig. II.5.

Appendix III

USE OF DETAILED ANALYSIS METHODS

III.1. INTRODUCTION

In this Appendix, a series of examples are provided that address the topics described in Section 5 of this report. These topics show the different methods used to reduce the conservatism applied in screening different fire scenarios. It should be noted that the numerical values used in this Appendix are provided as examples; none should be considered generic.

III.2. DESCRIPTION OF PLANT EXAMPLES

The examples are based on a postulated PWR. In this reactor, core damage may occur only after simultaneous failure of the auxiliary feedwater (AFW) system and the bleed and feed function. The AFW system consists of two trains: train A with a motor operated pump, and train B with a steam driven pump; the latter does not require electric power. However, a hot short in the control circuit of the isolation motor operated valves (MOVs) in this pump train may lead to pump train failure.

Bleed and feed is activated by control room operators. They open the power operated relief valve (PORV) to bleed steam from the reactor vessel and use one of two high pressure charging pumps to replenish the reactor coolant system with cold water.

The cable tray loading for this hypothetical plant is as follows:

<i>Cable tray coding</i>	<i>PSA related equipment</i>
1-1A	AFW electrical pump (train A)
1-1B	Isolation MOV in the steam driven AFW pump (train B)
1-2A	Charging pump (train A)
1-2B	Charging pump (train B)

III.3. FIRE SEVERITY AND GEOMETRIC FACTORS

In a fire analysis for a nuclear power plant it is assumed that fire compartment A-3 has been identified, that there are no fire cells within the compartment and that

the compartment is defined by 3 hour rated fire barriers. In task 5 of the analysis, identification is made of the fire scenario (designated as F.A-3) that occurs in this compartment, that engulfs every PSA related cable and equipment item in the room, and that is confined to the compartment boundaries. In task 6 of the fire analysis it is concluded that if all four cable trays of this compartment (Fig. III.1) are damaged, then the resulting core damage frequency is a significant contributor to the overall core damage frequency.

Close inspection of the room reveals a cable tray set-up, as depicted in Fig. III.1. The last letter of the tray designator shown in the figure represents the train assignment. Several combinations of cable failure may occur in this cell. Failure of cables in trays 1-1A and 1-2A leads to a series of failures in train A of various systems. Similarly, failure of cables in trays 1-1B and 1-2B leads to a series of failures in train B of various systems. The following possibilities are found to have different plant impacts: simultaneous failure of the cables in cable trays 1-1A and 1-2B; simultaneous failure of the cables in trays 1-1B and 1-2A; and simultaneous failure of the cables in all four cable trays.

Thus, the overall fire scenario considered in task 6 (designated as F.A-3) can be redefined (broken down) into more detailed scenarios. Considering the equipment associated with each cable tray, the following potential fire scenarios are identified: fire scenario F.A-3-1-A: a fire that affects all four cable trays; fire scenario F.A-3-2-A: a fire that affects only cable trays 1-1A and 1-2B; fire scenario F.A-3-3-A: a fire that affects only the stacked cable trays in the corner of the room (i.e. cable trays 1-1B and 1-2A).

Assuming that the fire frequency for fire scenario F.A-3 considered in task 6 is $3.0 \times 10^{-3}/a$ and that the conditional probability of core damage is 1.0, then the overall core damage frequencies for the above defined fire scenarios can be calculated as follows:

- (1) *Fire scenario F.A-3-1.* The fire has to be strong enough to damage the cables in those areas that are outside its immediate region. The conditional probability of such a fire may be taken as 0.10, based on conservative analysis of experience data in the USA. Therefore, the core damage frequency for scenario F.A-3-1 becomes $3.0 \times 10^{-3} \times 0.10 \times 1.0 = 3.0 \times 10^{-4}$ per reactor-year.
- (2) *Fire scenario F.A-3-2.* The fire has to occur in the northeast corner of the room to damage both trays at the same time. This assumption is justified by the fact that rapid propagation of a fire in a horizontal direction over a cable tray is unlikely. One can conservatively assume that the fraction of the fires that occur in the northeast corner of the room is 0.15. Also, consider that the steam driven AFW pump train will be operable if only trays 1-1A and 1-2B are damaged. Assuming that the operators initiate feed and bleed and that the failure probability of the steam driven AFW pump from causes independent of the fire

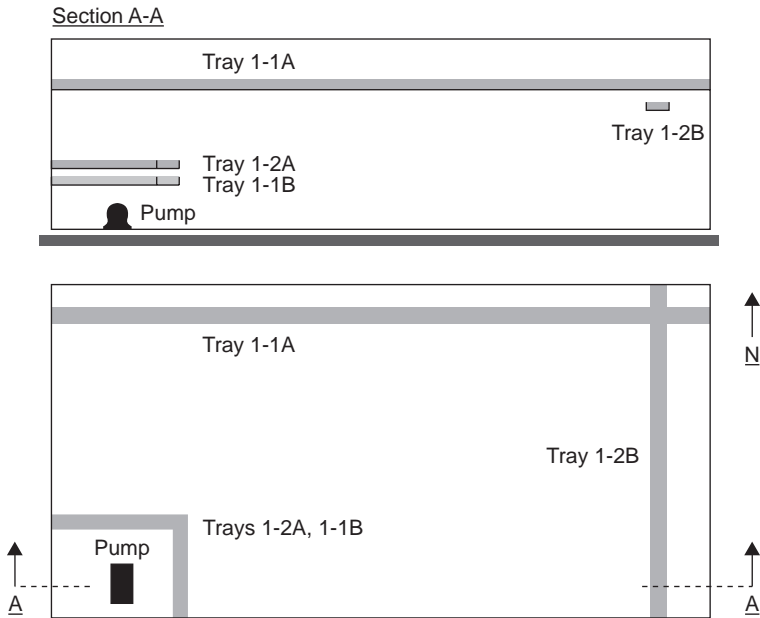


FIG. III.1. Fire compartment A-3 (for details of the cable tray codings, see Section III.2).

is 0.05, then the frequency of core damage becomes $3.0 \times 10^{-3} \times 0.15 \times 0.05 \times 1.0 = 2.3 \times 10^{-5}$ per reactor-year.

- (3) *Fire scenario FA-3-3.* To damage trays 1-1B and 1-2A at the same time, the fire has to occur in the southwest corner of the room. A pump fire may damage both trays. On the basis of fire experience data it is concluded that, in a room such as A-3, 50% of the fires can be attributed to the pump. If the two trays (1-1B and 1-2A) contain valve control related circuits for the steam driven AFW pump (train B) and the control cables for the charging pump (train A), core cooling will rely on the motor driven AFW pump (train A) and the charging pump (train B). The turbine driven isolation valve is assumed to fail from a hot short in its control circuit (with a probability that is equal to 1). Assuming that no operator actions are needed for the charging pump (train B) and that the failure probability for the motor driven AFW pump (train A) is 0.02, then the probability of core damage becomes $3.0 \times 10^{-3} \times 0.50 \times 0.02 \times 1.0 = 3.0 \times 10^{-5}$ per reactor-year.

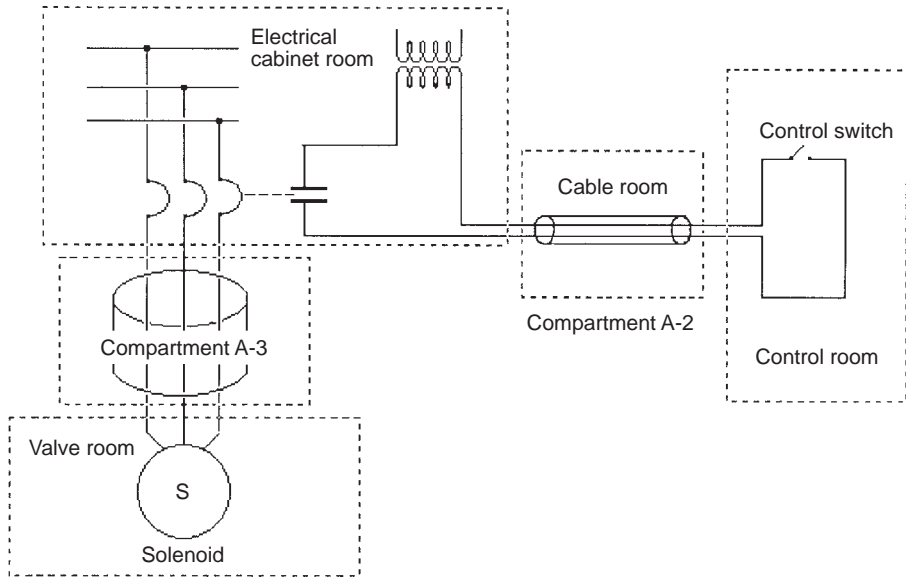


FIG. III.2. Valve control circuit.

III.4. HOT SHORT

Consider the simplified valve control circuit diagram outlined in Fig. III.2. A fire in compartment A-3 is very unlikely to lead to valve failure, since only a three phase power cable is present in that compartment. Thus, in computing the core damage frequency for compartment A-3, a hot short in the turbine driven AFW pump isolation valve can be set at a probability that is equal to zero. The steam turbine driven pump failure probability used in F.A-3-2 can then be used in fire scenarios F.A-3-1 and F.A-3-3: for fire scenario F.A-3-1, the core damage frequency becomes $3.0 \times 10^{-3} \times 0.10 \times 0.05 \times 1.0 = 1.5 \times 10^{-5}$ per reactor-year; and for fire scenario F.A-3-3, the core damage frequency becomes $3.0 \times 10^{-3} \times 0.50 \times 0.02 \times 0.05 \times 1.0 = 1.5 \times 10^{-6}$ per reactor-year.

As another example, again using Fig. III.2, a fire in compartment A-2 may lead to an intracable hot short, i.e. the action of the control switch may be simulated if two wires inside the cable come into contact with one another. Figure III.3 shows the possible positions of the cable where these wires are located in a cable tray, and a cross-section of a multiwire cable. In this case, the cable of concern contains many individual conductors, of which only two are of interest to the fire PSA. A risk

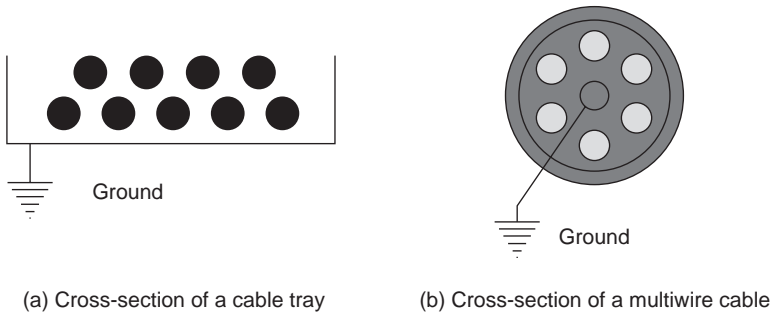


FIG. III.3. Cable tray and wires within the cable.

significant intracable hot short is possible only when the two specific ('live') wires touch one another and not the grounded tray. This, of course, will depend on the chronology of wires touching other wires or the ground. Thus, it can be concluded that under special circumstances the above mentioned intracable hot short is a rare event.

An analyst may chose to take credit for these specific circumstances and use the conditional probability of an intracable hot short (given fire damage to the cable). For our example (given the set-up inside the cable), for an intracable hot short the analyst can assign a probability that is as low as 0.1 and as high as 1.0, with a logarithmic mid-point of 0.3.

III-5. FIRE DETECTION AND EXTINGUISHING

For fire scenario F.A-3-1 it is assumed that the compartment is easily accessible from the fire brigade dress-up area, that there is at least one fire brigade team on the site at any given time, that the room is equipped with several smoke detectors which signal alarm in the control room and that there is a fire hose station outside the door of compartment A-3. Also, regarding fire propagation it is assumed that, aside from the small pump in the southwest corner of the room, there are no other combustibles present in the room and that the door is always closed.

From analysis of the time taken to propagate a fire that damages all four cable trays and the time taken to detect and control this fire it is concluded that the probability of failure to suppress the fire (non-suppression factor) is $P_{NS} = 0.3$; the core damage frequency then becomes $3.0 \times 10^{-3} \times 0.10 \times 0.3 \times 0.05 \times 1.0 = 4.5 \times 10^{-6}$ per reactor-year.

Appendix IV

FIRE PROPAGATION EVENT TREE

One of the techniques that can be used in fire PSA is event tree analysis. Selection of this approach, and the depth of its application, are dependent on the overall scale of the fire PSA exercise and the design of the plant being analysed. The likely benefits of using an event tree approach must be taken into account. For example, a plant that incorporates effective passive barriers between redundant safety related systems in separate fire compartments may not require fire propagation analysis between compartments. In plants where there is a lack of barriers it could be appropriate to use fire event tree analysis to represent fire spread; however, alternative approaches may be preferred.

The following example illustrates the comparatively complex processes involved in undertaking a detailed fire event tree analysis. It will help the prospective fire PSA analyst, or specification author, to select the appropriate approach and depth of analysis required for the plant to be analysed. The example represents the potential

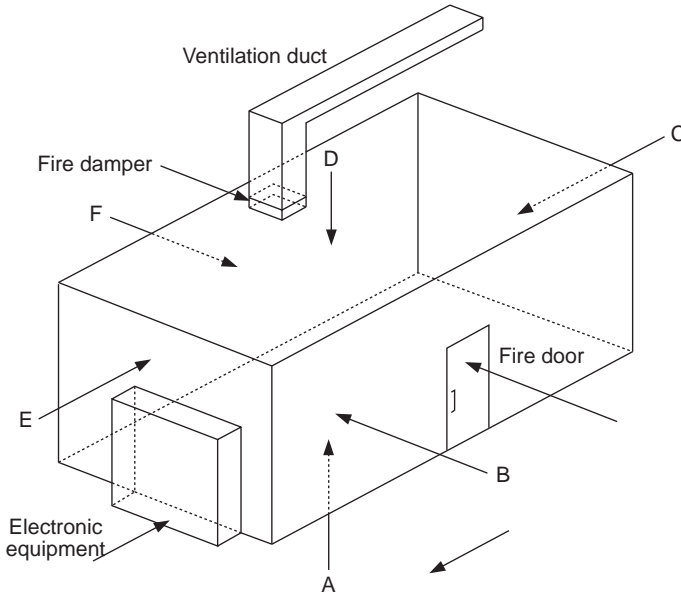


FIG. IV.1. Simple fire cell (A–F denotes the six fire barriers, details of which are given in the text).

routes through which a fire initiated within the simple fire cell (Fig. IV.1) can develop. The fire cell has six fire barriers (A–F) with the following features: barriers A, C and F are fully fire rated and will not fail, even if the fire in the cell is not suppressed; barriers B, D and F have only a limited fire rating and may eventually fail if the fire in the cell is not suppressed; barrier B incorporates a fire rated door (which could be left open); barrier D incorporates a fire damper (which could fail to close); and barrier E has sensitive electronic equipment mounted on it in the adjacent cell.

Depending on the successful action of the fire suppression systems, the fire initiated in the cell progresses from a pilot fire to a developed fire and results in a series of outcomes, each of which is provided with a specific path number. Other cells can be affected by the fire as a result of fire spread through a failed or open barrier, fire or smoke spread through a ventilation duct or open barrier, and heat transfer through an intact barrier. The event tree shown in Fig. IV.2 represents possible combinations of events (scenarios).

More detailed information on the scenarios defined by the event tree is provided in Table IV.I for each specific path (1–38). For scenarios in which other cells are affected by the spread of hazard, information is provided on the barrier(s) breached (B, D, E) and the type of hazard being spread.

With regard to the effects of a fire (smoke, heat) on the adjacent cells, the following assumptions are made for this example:

- B_s = Door open — early suppression fails — smoke spread;
- B_f = Door open — late suppression fails — fire spread through barrier B only;
- D_s = Damper open — early suppression fails — smoke spread through the ventilation duct;
- D_g = Damper open — late suppression fails — fire spread through the ventilation duct;
- E_h = Door closed — all suppression fails — heat transfer through intact barrier;
- B_f, D_f, E_f = All suppression fails and barrier fails — fire spread.

Two groups of scenarios are distinguished: those in which the fire is contained in the cell, and those in which the fire spreads to other cells. The first group also includes cases where cold smoke spreads through an open barrier and heat transfer through an intact barrier. For fires contained in the cell, the effects on the plant can be identified (at this stage of analysis) for each of the consequential damage categories (designated i–vii). In cases where other cells are affected by hot gas or fire spread through a ventilation duct or a failed barrier, the total damage state cannot be evaluated at this stage. It is assessed when the consequences of a fire spread into the affected cell(s) have been evaluated.

Figure IV.2 and Table IV.I illustrate that the effect on other cells can be the same for several fire spread sequences. In this case, there are three fire spread scenarios

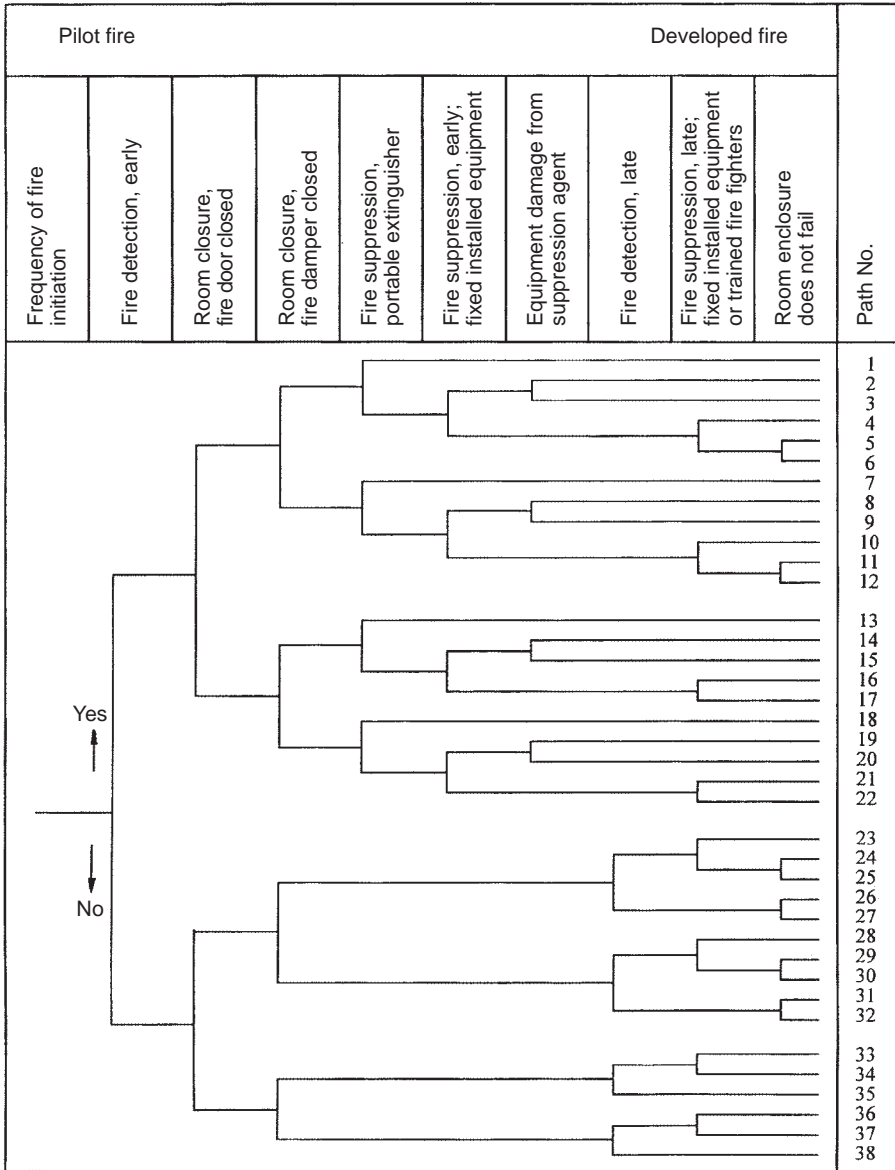


FIG. IV.2. Fire propagation event tree.

(1–3 in the fourth column of Table IV.I). The associated frequencies can therefore be summed to provide a reduced number of sequences for analysis of a fire spread into the adjacent cells.

TABLE IV.I. CONSEQUENTIAL DAMAGE CATEGORIES FOR INDIVIDUAL SCENARIOS DEFINED BY THE EXAMPLE EVENT TREE

Path No.	Other cells affected	Damage category	Fire spread sequence	Sequence frequency (1/a)			
				Contained sequences	D _g E _h	B _f	B _f D _f E _f
1		i	–	*			
2		iii	–	*			
3		ii	–	*			
4		iv	–	*			
5	E _h	v	–	*			
6	B _f D _f E _f		1				*
7		i	–	*			
8		iii	–	*			
9		ii	–	*			
10	D _s	vi	–	*			
11	D _g E _h		2		*		
12	B _f D _f E _f		1				*
13		i	–	*			
14		iii	–	*			
15		ii	–	*			
16	B _s	vii	–	*			
17	B _f		3			*	
18		i	–	*			
19		iii	–	*			
20		ii	–	*			
21	D _s	vi	–	*			
22	B _f D _f E _f		1				*
23		iv	–	*			
24	E _h	v	–	*			
25	B _f D _f E _f		1			*	*
26	E _h	v	–	*			
27	B _f D _f E _f		1			*	*
28	D _s	vi	–	*			
29	D _g E _h		2		*		
30	B _f D _f E _f		1			*	*
31	D _g E _h		2		*		
32	B _f D _f E _f		1			*	*
33	B _s	vii	–	*			
34	B _f		3			*	
35	B _f		3			*	
36	D _s	vi	–	*			
37	B _f		3			*	
38	B _f		3			*	
				Σ*	Σ*	Σ*	Σ*

Note 1: (i)–(vii) = different damage states (for fires contained in the cell).

Note 2: * = sequence frequencies to be inserted.

Note 3: Subscripts = s: cold smoke; h: heat transfer through an intact barrier; g: fire spread through a ventilation duct; and f: fire spread through a failed (or open) barrier.

Assessment of a fire spread through an adjacent cell (or possibly a remote cell if spread through a ventilation duct) is carried out in a similar manner, using sequence frequency as the frequency for the 'initiating' pilot fire in the adjacent cell. Care must be exercised by considering cases where the fire may have developed beyond the pilot stage (e.g. because of fire barrier failure).

Account should be taken of the frequency of individual path sequences as they progress through one or more cells. If the path sequence frequency falls below a predefined limit, the sequence should be terminated and discounted as being not credible.

Appendix V

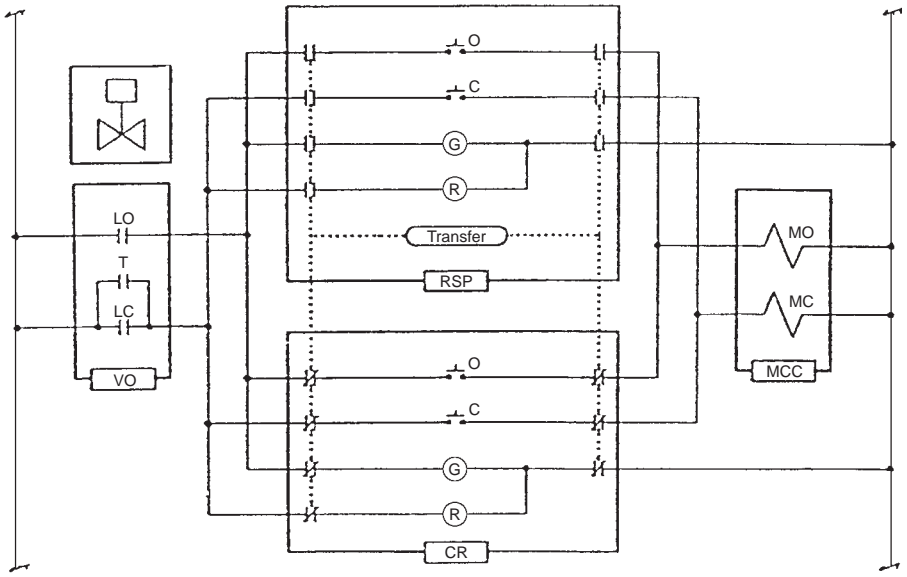
POTENTIAL FOR LOSS OF THE ALTERNATE SHUTDOWN CAPABILITY DURING A CONTROL ROOM FIRE

*The following example is extracted from
United States Nuclear Regulatory Commission Information Notice 92-18 [25]*

The example provided in this Appendix describes the conditions that could result in the loss of capability to maintain the reactor in a safe shutdown condition in the event of a control room fire. It is shown that certain MOVs needed to shut down the reactor and to maintain it in a safe shutdown condition may, as a result of hot shorts combined with the bypass of thermal overload protection, be damaged before the operator shifts control of the valves to the alternate shutdown panel.

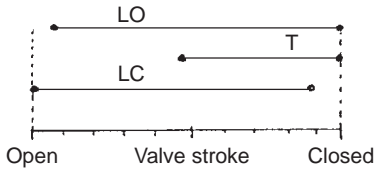
Fig. V.1 is a conceptual diagram of the control circuitry for MOVs and includes the relay coils that operate the contactors in the power circuitry for the motors. It should be noted that the torque and limit switches in the valve operator are located in a manner such that they can be bypassed in the event of a hot short.

Fig. V.2 provides an example of the manner in which the motor of a closed MOV can be energized and damaged by a hot short if its overload protection is bypassed. The hot short bypasses the push button normally used to close the MOV, thus providing the power to drive the motor in a closed direction. The power is not disconnected from the motor, although it has stalled; the current and torque are abnormally high, possibly causing electrical failure to the motor windings and mechanical damage to the valve. Such mechanical damage may be sufficient to prevent reactor operators from manually operating the valve. A similar problem can occur for open MOVs (see Fig. V.3). Shorts to other sources of power also cause failure of the MOVs.



- CR = Control room
- RSP = Remote shutdown panel
- MCC = Motor control centre
- VO = Valve operator
- MC = Relay coil – close valve
- MO = Relay coil – open valve
- C = Push button – close valve
- O = Push button – open valve
- LC = Limit switch – close valve
- LO = Limit switch – open valve
- T = Torque switch
- G = Green lamp
- R = Red lamp

Contact actuation points



The lines indicate that the switch contacts are closed. The points indicate the valve positions where the switch contacts open and close. For the torque switch, the contacts are actuated by the position of the valve disc at mid-stroke and by the preset torque at the end of the closing stroke.

FIG. V.1. Control circuitry for the motor operated valves (MOVs).

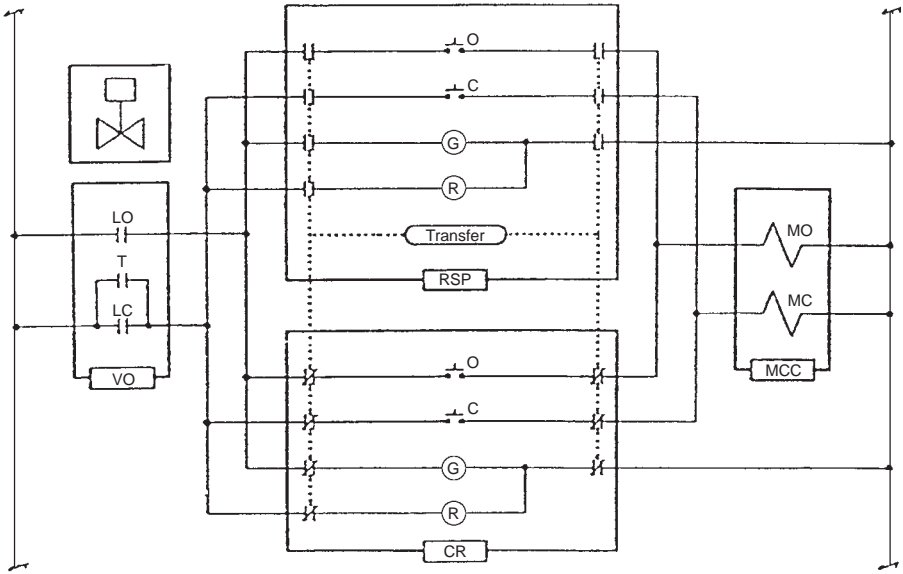


FIG. V-2. Postulated hot short that occurs while a MOV is closed (see Fig. V.1 for details of abbreviations).

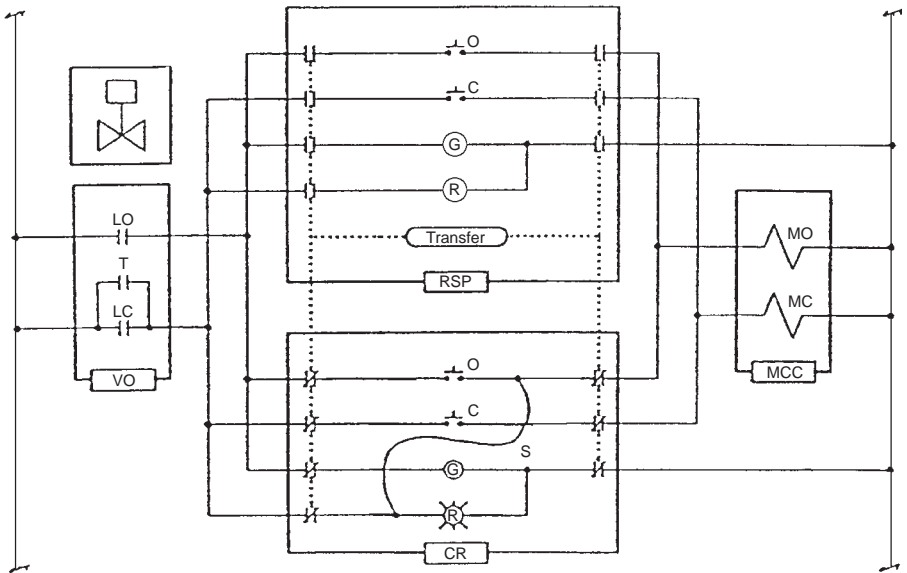


FIG. V.3. Postulated hot short that occurs while a MOV is open (see Fig. V.1 for details of abbreviations).

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