Fire Probabilistic Risk Assessment applied to the Pebble Bed Modular Reactor

Matoma P. Pooe

2001

University of North-West

P B M R

Masters in Applied Radiation Science and Technology

University of North West, Mmabatho

Fire Probabilistic Risk Assessment

Fire Probabilistic Risk Assessment Applied to the Pebble Bed Modular Reactor

Matoma P Pooe

2001

Mr. M Magugumela & Mr. B Blignaut

Supervisors

Masters in Applied Radiation Science and Technology

University of North West, Center of Applied Radiation Science and Technology, Mmabatho

Pebble Bed Modular Reactor, Centurion

Abstract

A methodology for assessing the frequencies of fire occurrences in the PBMR nuclear power plant is presented. Fire is a concern because it may lead to loss of support systems, which could result in the failure of dependent systems. Ultimately this may lead to the release of radioactive material into the atmosphere.

The methodology used is probabilistic in nature. The methodology will be applied to the Pebble Bed Modular Reactor to determine the impact of fire on safety of plant personnel and the public. The methodology described here utilises both a qualitative and quantitative assessment in order to quantify the risk posed by fires in the PBMR reactor. The qualitative steps identify areas where fire is likely to be started. Only fire events that may result in the loss of a safety function are of safety concern. Thus, this process is based on identification of critical locations where there is potential for fire initiation resulting in loss of safety function.

The quantitative analysis utilizes an event tree approach to quantify the frequency of fire initiation. An event tree is constructed based on the mitigating actions of the fire protection system in each critical area.

This study is meant to be a preliminary assessment of Fire PRA for the PBMR. This task is made difficult by the fact that the

plant is still in the design phase. Traditional Fire PRA's have been done on existing power stations with operational histories.

To demonstrate the methodology in the PBMR, data from Light Water Reactors (LWR) and Boiling Water Reactors (BWR) was used as a baseline for the Pebble Bed Modular Reactor (PBMR) since there is no historical information on the plant because one has not been built yet. When the plant is built and is in operation the assessment will need to be updated.

A brief summary of the study, the conclusion and recommendations for this study are also presented in the last section of the document.

Table Of contents

List of Figures

List of Tables

LIST OF ABBREVIATIONS

Abbreviation or acronym	Description
ACS	Active Cooling System
AVR	Arbeitsgemeinschaft Versuchsreaktor (German for Jointly-operated Prototype Reactor)
BWR	Boilling Water Reactor
CCS	Core Conditioning System
EPS	Equipment Protection System
FHSS	Fuel Handling and Storage System
FMEA	Failure Mode and Effects Analysis
FMECA	Failure Mode, Effects and Criticality Analysis
HAZOP	Hazards and Operability
HICS	Helium Inventory Control System
HMS	Helium Make-up System
HPT	High Pressure Turbine
HTGR	High Temperature Gas Cooled Reactor
HTR	High Temperature Reactor
HVAC	Heating, Ventilation and Air-conditioning
IAEA	International Atomic Energy Agency
IE	Initiating Event
LBE	Licensing Basis Event
LOCA	Loss-of-coolant accident
LOFC	Loss of Forced Cooling
LPT	Low-pressure Turbine
LWR	Light Water Reactor
MPS	Main Power System
MW	Megawatt
NNR	National Nuclear Regulator
NNS	Non-nuclear Safety
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NUREG	Nuclear Regulations (from NRC)
OCS	Operational Control System
PBMR	Pebble Bed Modular Reactor
PCU	Power Conversion Unit
PES	Primary Loop Initial Clean-up System Evacuation Sub-system

PFOD	Probability of Failure on Demand
PLICS	Primary Loop Initial Clean-up System
PLOFC	Pressurised Loss of Forced Cooling
PPB	Primary Pressure Boundary
PPSB	Power Plant Services Building
PRA	Probabilistic Risk Assessment
PT	Power Turbine
PWR	Pressurized Water Reactor
PyC	Pyrolytic Carbon
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
RCSS	Reactivity Control and Shutdown System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RPVCS	Reactor Pressure Vessel Conditioning System
RSS	Reserve Shutdown System
SAR	Safety Analysis Report
SAS	Small Absorber Sphere
SBS	Start-up Blower System
SiC	Silicon Carbide
TRISO	Triple Coated Particle
WMPS	Water Make-up Purification System

CHAPTER 1

INTRODUCTION

1.1 Introduction

Nuclear energy has both benefits and risks. One of its many benefits is power generation. Because of the inherent risks, to generate power from nuclear energy, utilities must show that the reactor will operate safely without unduly exposing the public and the workers to health risks. One of the techniques used to ensure safety in nuclear installations is probabilistic risk assessment (PRA). This is also variably known as probabilistic safety assessment (PSA). While regulators across the world are increasingly recognising PRA in licensing nuclear reactors, in South Africa nuclear power plants have been licensed based on PRA for years.

1.2 Background

PRA was first used to assess the safety of nuclear reactors in 1975 as part of the Reactor Safety Study commissioned by the United States Nuclear Regulatory Commission [1]. It has since gained use as a fundamental tool for assessing safety of installations worldwide. For example, PRA is used in such varied industries as the chemical and aviation.

The PBMR (PTY) LTD has just completed a Phase 2 PRA to assess the safety of the design [2]. Phase 2 PRA only assesses the safety impact of internal events. External events have not been adequately analysed in the current design. It is standard practice to also determine the impact of external events on the safe operation of the plant. Examples of external events include fire, earthquakes, flooding, aircraft crash, and tornadoes, etc. This study focuses on the fire assessment. The safety assessment of other external events will need to be performed at some stage.

1.3 Overview of PRA

The main objective of the PRA is to identify all possible ways that might result in plant personnel and the public being exposed to radioactive material and then determining their frequencies and consequences. Frequencies and consequences are then combined to determine the risk due to the various accidents. In pressurised water reactors (PWR), it is sufficient to determine all scenarios, which might result in the reactor core melting. This is because substantial amounts of radioactivity are assumed to be released only after large scale destruction of the nuclear fuel. In the PBMR, however, core melt is highly unlikely because of the fact that PBMR is inherently safe (this means that the occurrence of any event would not result in exposure of the public to the radioactive material), the negative temperature coefficient and the temperature tolerance of the fuel are such that sub-criticality without forced cooling can be achieved and maintained without active engineered systems or operator action for many hours. However, there are scenarios that may result in radioactivity reaching plant personnel and the public [2]. In performing PRA, it is important to ensure that all possible ways in which this can happen are identified. The procedure for conducting a PRA can be found in Chapter 2.

The probabilistic risk assessment method was introduced to evaluate the interaction between potential system failure and the resulting health impact. It is an analytical technique integrating a wide spectrum of design and operational aspects of a specific facility [9,10].

1.3.1 Why is fire of concern

Historically, fire has been found to contribute significantly to risk of nuclear power. Events [8] have shown that fire in nuclear power plants although rare, could increase the risk due to their operation. The special nuclear boundary conditions (like, the absolute confinement of radioactivity) impose severe requirements on the fire protection and the fire fighting measures. Since the Browns Ferry Fire major efforts have been made to further reduce the risk of fires in nuclear power plants [19].

Fire also tends to induce multiple failures. Nuclear power plants must demonstrate that the releases of radioactivity will still be within the limits, given a fire in any area of a plant. Although fire can occur anywhere within the plant, the risk will be greatest in certain critical fire areas [5].

1.3.2 Historical Evidence

There have been a few incidences of fire in nuclear power plants. Although some of them were minor, others threatened the safe operation of the plants. A few examples are discussed here.

The Browns Ferry Fire accident is one example of why fire is a major risk [8]. The accident occurred during maintenance when one of the maintenance personnel lit a candle in the cable room. The cables caught fire. From these accidents it also became evident that critical fire areas in power plants apart from areas containing the safety equipments are areas containing electrical equipments such as the cables, motors, transformers, switches, batteries, generators, blowers, electrical busbars and breakers and electrical cabinets.

1.3.3 Causes of fire

Experience has shown that maintenance activities could lead to fire accidents as explained above by the Browns Ferry Fire. Electrical equipments can also cause fire. Electrical fires can be explained by Ohm's law, which is expressed as:

$(V = IR)$ where,

V is the voltage drop in volts,

I is the current in amperes, and

R is the resistance in ohms

The Power (P) (in Watts), in an electrical wire can be expressed as:

$P = VI$, but, since $V = IR$

Then $P = I^2R$

From this equation it follows that when current increases and resistance also increases by the equation:

$P = I^2R$,

The power in the cables increases, and this then results in electrical cables becoming hot and there after they start burning. This is how fire occurs in electrical equipments.

At this stage in the PBMR PRA, the impact of fire on the risk to the public due to the operation of the plant has not been investigated. It is therefore very important that a fire analysis be incorporated into the PBMR's risk model.

1.4 Background

The importance of fire as a potential initiator of multiple-system failures took on a new perspective after the cable tray fire at Browns Ferry plant in 1975, where a fire was ignited during maintenance as a result of one of the maintenance personnel lighting a candle in the cable room [8].

Fire analysis involves quantifying the frequency of core melt or the ability of a plant to reach a safe reactor shut down, and containment failure given a fire incidence anywhere in a plant. The frequencies and consequences of fire in each area of a plant and the impact of these consequences on the safe shut down capability of the plant are assessed [5].

Fire analysis process requires knowledge of several important aspects of a fire incident like: ignition, progression, detection, and suppression, and characteristics of various materials under fire conditions as well as the plant safety functions and their behaviour under accident conditions [12].

Fire research is a multidisciplinary effort, which covers a large spectrum of topics like: the physics of combustion, the behaviour of flames in compartments, and the characteristics of fire detector responses [8].

To perform an assessment of fire impact on risk knowledge of the plant is required. The description of the plant is provided in the following section.

1.4.1 The Pebble Bed Modular Reactor Power plant

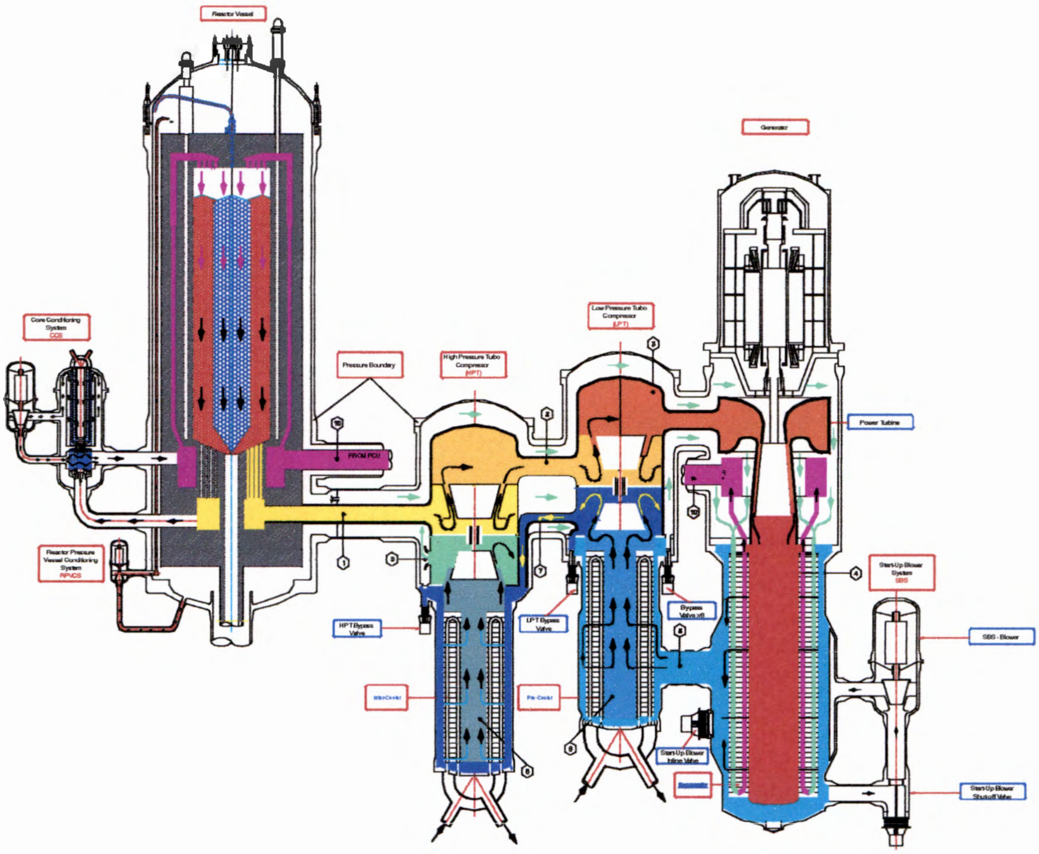
The Pebble Bed Modular Reactor (PBMR) plant is different from ordinary Light Water Reactors (LWR) systems in that:

- It is inherently safe, meaning that even a major accident such as a Depressurized Loss of Forced Cooling (DLOFC) the reactor will still be able to sustain itself without any operator actions by natural convection where heat will be transferred from the fuel in the core to the reflectors and from the reflectors and all other systems to the reactor wall and then to the ultimate heat sink.

- The negative temperature coefficient and the temperature tolerance of the fuel are such that sub-criticality without forced cooling can be achieved and maintained without active engineered systems or operator action for many hours.

The PBMR is a graphite moderated helium cooled reactor, which uses a Brayton direct gas turbine cycle to convert the heat, which is generated in the core by nuclear fission. The heat generated in the core by the fission process is then transferred to the coolant gas. The gas is then transported to the turbine where it is converted into electrical energy. Figure 1-1 General layout of the Reactor Unit and the Power Conversion Unit (October 2000). The use of Helium eliminates the possibility of a change of phase of the coolant (i.e. liquid to vapour) under accident conditions. Accident analysis is greatly simplified and uncertainties reduced by the absence of two-phase flow.

Figure 1-1 General layout of the Reactor Unit and the Power Conversion Unit (October 2000).



The basic reactor power generation system consists of:

- Reactor Unit; consisting of the Reactor Pressure Vessel, Core Structures, and Reactivity Control and Shutdown Systems.

- Power Conversion Unit; consisting of High Pressure Turbo-units, Low Pressure Turbo-units, Power Turbine, Generator System, Recuperator, Coolers, Hot Pipe and Valve System.

- Main Support Systems; consisting of Fuel Handling and Storage System, Helium Inventory Control System and Active Heat Removal System [PBMR SAR-Section 3, 2001].

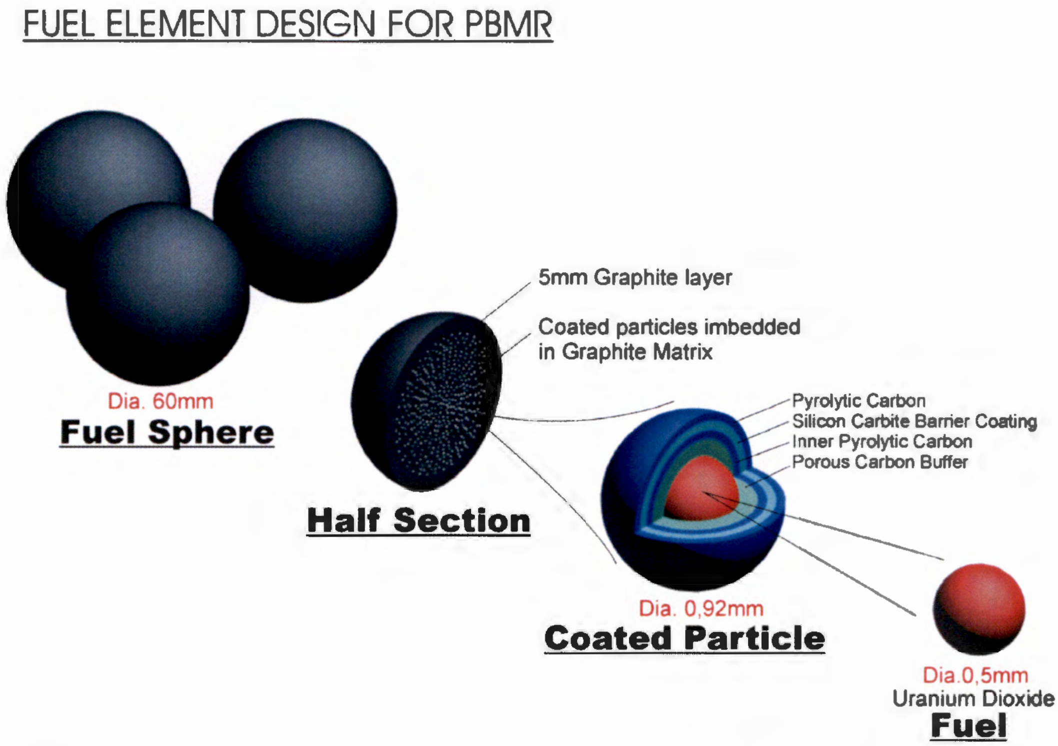
There are a number of safety systems by which the reactor activities can be controlled. These include activities such as the heat removal, and the control of heat generation. For heat removal systems such as the Reactor Cavity Cooling system (RCCS), Start-up Blower System (SBS), and the Core Conditioning System (CCS) are used. For the control of heat generation systems such as the Reactivity Control System (RCS), and Reserved Shutdown system (RSS) are used.

The function of the RCCS is to remove heat from the reactor during normal operation and shutdown. The system also removes decay heat during the loss of the heat transfer functions of the Power Conversion Unit (PCU). The objective is to prevent the reactor vessel, Reactor Pressure Vessel (RPV) supports, instrumentation and concrete walls from exceeding from exceeding their design temperature limits.

The CCS maintains the RPV at desired temperatures (280 °C to 300 °C) during normal power operation, it also removes core heat during maintenance operations when the SBS cannot be used and it can be used to recover the core from very hot conditions (up to 1200 °C) in the event of a PCU trip followed by failure of the SBS blowers to operate. During maintenance it is used to maintain the core at the required temperature although the SBS blowers will be used to initially cool the core down to low temperature maintenance conditions.

The RCS consists of a control rod system, and the RSS consists of the shutdown rods and the Small Absorber System (SAS). Both the RCS and the RSS are independently able to transfer the reactor to a sub-critical state, and keep the reactor subcritical for an indefinite period at normal operating temperature. Both systems combined must allow the reactor to be in a sub-critical state below 50 °C indefinitely.

Figure 1-2 The Pebble Bed Modular Reactor fuel elements



The PBMR uses spherical fuel elements. The PBMR reactor consists of graphite spheres situated at the centre of the reactor core surrounded by fuel spheres which also contain a layer of graphite and fuel particles known as Triso particles because of the three layers covering them (i.e. the inner pyrolytic carbon layer, the pyrolytic silicon carbide layer, and the outer pyrolytic carbon layer).

1.5 Purpose of study

Fire PRA is important for identifying the possible consequences of fire, which could prevent the safe, shut down of a nuclear power plant [8]. The purpose is also to investigate the methodology and further apply it to quantify the frequency of fire in the PBMR.

The Regulatory guide [3] mandates fire prevention and protection programs. It also requires that fire contribution to the radiological consequences be included in the probabilistic assessment. At this stage in the PBMR PRA development, fire will need to be incorporated into the PBMR's probabilistic risk model.

The importance of this study lies in using the probabilistic methods to calculate the frequency of reaching a safe reactor shutdown given that fire occurred anywhere in the PBMR power plant.

1.6 Objectives

The objectives of the study are:

I. To present the methodology of the study.

II. To use the methodology to quantify the frequency of the safe shutdown capability of the plant in case of a fire accident.

III. To evaluate the impact of fire on the safety of the PBMR power plant.

1.7 Outline of chapters

Literature survey on fire and general fire analysis methodologies, their ignition, how they spread, their detection, suppression and mitigation are discussed in Chapter 2.

Chapter 3 focuses on the step-by-step process of a fire analysis. The end result is the frequency of reaching a safe shutdown in case of fire. Uncertainty calculation is not in the scope of the study.

Chapter 4 discusses the results of the study. The summary, conclusions and recommendations are discussed in Chapter 5.

CHAPTER 2

LITERATURE SURVEY

2.1 Review of Literature

1. The book NFPA 805 [20], specifies the minimum fire protection requirements for existing light water nuclear power plants during all phases of plant operation, including shut-down, degraded conditions, and decommissioning. The book also provides information on methods of maximising defence in depth during a plant fire, goals and objectives of nuclear safety.

The fire protection standard is based on the concept of defence in depth. Protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount to this standard. Defence in depth is achieved when an adequate balance of each of the following elements is provided; (i) preventing fires from starting, (ii) rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage, (iii) providing adequate level of fire protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

The safety goals include (i) the provision of reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition; (ii) to provide reasonable assurance that a fire will not result in radiological release that adversely affects the public, plant personnel, or the environment; (iii) to provide reasonable assurance that loss of life in the event of fire will be prevented for facility occupants; (iv) to provide reasonable assurance that the risks of fire are acceptable with regard to potential economic consequences.

The nuclear safety objectives during a fire event are: (i) capability of rapidly achieving and maintaining subcritical conditions, removing decay heat and inventory control functions, preventing fuel clad damage so that the primary containment boundary is not challenged; (ii) maintenance of the containment integrity and the limitation of the source term; (iii) to protect occupants not intimate with the initial fire development from loss of life and improve the survivability of those who are intimate with the fire development as well as to provide adequate protection for essential and emergency personnel; (iv) to limit the potential property damage due to fire to acceptable levels, also the potential business damage due to fire is also limited to acceptable levels as determined by the operator.

The book also outlines a performance criteria by which fire protection features shall be capable of providing reasonable assurance that, in the event of fire the plant is not placed in an unrecoverable condition. This shall be demonstrated by: (i) insertion of the negative reactivity by the reactivity control to achieve and maintain sub-critical conditions such that fuel design limits are not exceeded; (ii) capability of inventory and pressure control to control coolant levels such that subcooling is maintained for PWR and the capability of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR such that fuel clad damage as a result of a fire is prevented; (iii)removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition; (iv) capability of vital auxiliaries to provide necessary auxiliary support equipment and systems to assure that the systems required under (i), (ii), (iii), and (v) are capable of performing their required nuclear safety functions; (v) process monitoring for providing necessary indication to assure the criteria addressed in (i) through to (iv) have been achieved and are being maintained.

2. Lees F.P, in his book [12] defines fire in the following way:

"Fire or combustion, is a chemical reaction in which a substance combines with oxygen and heat is released. Fire occurs when a source of heat comes in contact with a combustible material.

The development of fire depends on the situation of the fuel. That is, the stacking of materials in closely spaced vertical piles tends to encourage fire spread.

Fire normally grows and spreads by direct burning, which results from impingement of the flame on combustible materials, by heat transfer or by travel of burning material.

The three main modes of heat transfer from fires are: convection, conduction, and radiation. Conduction is particularly important in allowing heat to pass through a solid barrier and ignite material on the other side. Radiated heat is transferred directly to nearby objects and does not go preferentially upwards and crosses open spaces. Convection heat transfer involves the transfer of heat from one material to the next different material. Most of the heat transfer from fires is by convection and radiation.

Electrical equipment is widely used in power plants and may be the source of ignition unless close control is exercised. The basis for the control of electrical equipment to prevent its acting as a source of ignition is the classification of the plant according to the degree of hazard."

This information is important in that it provides knowledge about how fires occur, the propagation of fire from one place to another and also some advice on how to place materials or equipments in power plants (not placing materials in stacks or piles aids in minimizing fire growth). It further provides information on areas that could be classed as fire areas or areas in which fires are expected in power plants.

3. The paper (Schneider U.) [24]. Outlines how difficult it is to prevent fires in nuclear power plants because of the combustible materials, which must be used for principle reasons, that is, power plants cannot avoid the application of such materials or media in certain areas or components of the plants. According to the paper, a detailed study on the fire occurrence rate in Pressurized Water Reactors (PWR) indicated that the occurrence rates of American and German reactor facilities were of comparable size. Also the paper states that it is indicated that the 95% fractile of the occurrence rate per year and facility is of the order of 3 to $4x10^{-1}$, the appropriate mean value is of the order of $1.1x10^{-1}$. This means that one can assume that in average a fire in a nuclear power plant occurs every ten years. With respect to the type of materials involved these observations were made: roughly 1/3 of all fires occurred with oil or hydrogen, 1/3 was related to other materials. These data indicate the great importance of cable fires and the strong need for fire prevention.

4. Fischer et al. [7] present the following views regarding fire:

" In the design process the fire protection engineer needs to frequently work closely with the plant design team from the beginning. The conceptual influence of fire protection considerations starts with the plant layout; the separation as far as possible of unavoidable fire loads and safety systems; good accessibility; separation between safety systems; and the installation of effective barriers.

The need for the reduction of fire loads is generally acknowledged, but in many instances it is a question of material properties in which the designer has only limited choice. The removal of unnecessary combustible materials is generally practiced."

5. The paper (Gregory et al) [9] describes how important a combustion analysis of an entire facility is, in determining the transient pressure and heating loads on the containment or ventilation structures and safety related equipment, to plan emergency evacuation, to validate existing fire protection programs or suggest changes, and to integrate fire safety with other safety analysis requirements.

It then concludes that fires in reactor systems may be initiated and then propagate during transient releases of heat, carbon monoxide and hydrogen and therefore nuclear reactor criticality transients may lead to severe accidents.

6.The paper (Himanen et al) [8] describes what the fire analysis method required for the TVO units I and II, which is common in all other methods. It required that all 1400 rooms of the plant be visited, and it was conducted in connection with the annual fire inspection made during and immediately after its refuelling outage in 1988. The paper further outlines the probability of ignition in a room depends on the presence of persons and on the amount of electrical and mechanical equipment in the room, and that the propagation of fire is a function of numerous factors for example, the amount and location of combustible material, ventilation and many others. The total fire frequency of the plant was estimated on the basis of the experience gained in the two units of the TVO plant and similar plants in Sweden.

7. According to this book (Safety Series) [22] a Safety Analysis Report of any power plant should provide a description and a safety analysis of the fire protection system in that particular plant including information on procedures and maintenance activities. The section on the Design for Internal Fire Protection should describe the design requirements for fire protection inside the facility. According to this book this should also include passive features, such as isolation, separation, selection of materials, the building layout and zoning, the location of fire barriers and the safety system layout and protection including the separation of safety related redundant systems.

This is a book which is accepted and used world wide by plants with hazardous materials and it is recommended by the nuclear regulators internationally that organizations/plants dealing with hazardous materials perform up to the standards set by this book.

8. Licensee Event Reports (LER), Computer Aided Regulatory Library (CARL)

Power plants are supposed to report all types of events big or small, occurring during any mode of operation. The reason for this is so that power plants should learn from each other's experiences and to avoid any events that may lead to accidents. The following are examples of the Licensee Event Reports (LER) from the Computer Aided Regulatory Library (CARL). In these reports all types of events that occurred in different power plants were reported with the date of occurrence and the causes of these events and also the possible solutions to these events for future references.

I. **ANO 1 - Event Date: 10/17/1996 - LER Number: 1996-009 - Cracked Weld in an Oil Line on a Reactor Coolant Pump:** A fire was discovered in insulation around the Main Feed water nozzle ring on "B" Once Through Steam Generator (OTSG) during heat up of the Reactor Coolant System (RCS). A weld located in the discharge line of a Reactor Coolant Pump (RCP) motor oil lift pump had cracked due to a fabrication defect. The failure, believed to have occurred at the start of the outage, resulted in oil being introduced onto the insulation. Oil on the insulation allowed a wicking effect that reduced the auto-ignition point of the oil to a value lower than the documented value. The fire originated when the RCS temperature was approximately 439 degrees. Application of a light water fog from a fire hose extinguished the fire approximately 16 minutes after it was discovered. A Notification of Unusual Event was declared when the fire was not extinguished within 10 minutes. The plant returned to cold shutdown conditions to evaluate damage. Other than some minor damage to insulation, the fire did not damage any systems or components. Enhancements were made to the oil collection systems of all RCP motors, and damaged insulation was repaired or replaced prior to the subsequent heat up.

II. **BEAVER VALLEY 2 - Event Date: 04/02/1998 - LER Number: 1998-005 - Inadequate Fire Protection Safe Shutdown Analysis for Boric Acid to Boric Acid Blender Valve, 2CHS*FCV113A:** On 4/2/98, a fire protection safe shutdown review for a proposed modification to the Boric Acid to Boric Acid Blender Valve, 2CHS*FCV113A, revealed that the fire protection safe shutdown analysis for the valve did not identify all cables and supporting equipment for operation of the valve. As discovered, errors exist in the valve control circuit analysis and the description in the Beaver Valley Power Station Unit 2 (BVPS-2) Updated Final Safety Analysis

Report (UFSAR) of equipment operability following a fire in fire areas CP1, CV3, PA3, PA4, PA5, SB1, SB3, SB4, and TB1. Due to these errors, required actions for boration via the boric acid tanks specified in the analysis and the plant operating manual are not adequate for fire areas CV3 and PA5. As such, following a fire event, the valve and the required boration flow path may not be available. This condition does not comply with the fire protection evaluation, as described in Section 9.5A of the BVPS-2 UFSAR. As understood, this condition does not apply to BVPS-1. This event resulted from oversight of the missing support cables and equipment for the valve while developing the list of electrical cables in specific designated fire areas for the fire protection safe shutdown analysis. Prior to entry into Mode 4 from the current Unit 2 shutdown, the control circuit of 2CHS*FCV113A will be modified so that the valve will fail to the safe (open) position, in the event of fire damage where the redundant boration flow path is affected. By 6/30/98, the valve control circuit analysis will be documented. An extent of condition review for Unit 2 of this occurrence will be completed prior to Unit 2 entry into Mode 4. A revised Unit 2 Fire Protection Safe Shutdown Report will be issued, by 10/31/98. By 10/31/98, the condition report of this event will be made part of the required reading and distribution to Nuclear Engineering Design personnel.

III. **BEAVER VALLEY 1- Inadequate Pre-Fire Plan Procedure Guidance Resulting in Potential Damage to Emergency Diesel Generators:** On April 9, 1999, with Unit 1 in Mode 1 at approximately 98% power, an ongoing Engineering Fire Protection Safe Shutdown self-assessment identified that operator actions in certain Unit 1 pre-fire plan and alternate safe shutdown procedures were inappropriate as they would place the Unit outside the design basis of the Fire Protection Program for Appendix R Safe Shutdown capability. These actions would de-energize power to the Emergency Diesel Generator (EDG) Building Area Ventilation System Discharge Dampers 1VS-D-22-1A and 1B, and could have prevented proper ventilation for the EDGs and the fulfilment of their safety function to satisfy the Appendix R Safe Shutdown requirements. Non-emergency one-hour notification of this discovery was made to the NRC, at approximately 1846 hours on April 9, 1999, pursuant to 10 CFR 50.72(b)(1). These procedural inadequacies are not applicable to the corresponding Unit 2 procedures. The apparent cause of this event is personnel error attributed to failure to identify that the manual actions specified in the procedures would result in closure of the EDG ventilation exhaust dampers. A contributor to this personnel error was inadequate knowledge of the design of the subject ventilation dampers, which are spring return to close on loss of power. Following event determination, operating shift briefings were conducted to ensure awareness of the subject procedural inadequacies pending revision of the procedures. Unit 1 subsequently entered Mode 5, at approximately 2235 hours on April 13, 1999. Appropriate revisions to the affected procedures have been implemented to ensure that power to 1VS-D-22-1A and 1B is not de-energized by the procedures.

2.2 General fire analysis methodology in power plants

Fire analysis in nuclear power plants presently involves following a number of procedures. It also requires knowledge about fires, material properties, combustion, plant design and layouts, cable locations, and the location of safety equipments [9].

The purpose of the analytical method developed is to identify a list of the dominant accident sequences that are initiated by fire and then to assess the frequency of occurrence for each. This process requires information about several important aspects of a fire (ignition, progression, detection, and suppression, characteristics of materials under fire conditions) as well as the plant safety functions and their behaviour under accident conditions. Considerable uncertainties exist in the analysis because of gaps in the required knowledge [13].

The process of fire analysis can be divided into three main steps. The three steps are:

I. A hazard analysis,

II. A fire propagation analysis, which is somewhat analogous to a component-fragility analysis, and

III. A plant and system analysis

The hazard analysis develops the frequency and magnitude of the externally imposed stress, where "stress" is in terms of potential fire induced accident sequences.

The propagation analysis investigates the resistance of the plant to fire damage by studying the propagation of the fire and the effectiveness and timing of suppression.

The last analysis evaluates the response of plant systems to the accident sequence triggered by the fire; it considers core damage [13].

CHAPTER 3

METHODOLOGY

The methodology to be followed in fire risk assessment involves three steps. These steps were briefly introduced in the previous section and are discussed in detail in this section. The steps are:

I. Fire hazard analysis,

II. Fire propagation analysis, and

III. Plant and system analysis

3.1 Fire Hazard analysis

The purpose of the fire-hazard analysis is to identify the critical plant locations that are important to the fire-risk analysis. This is analysis is carried out in two steps. The first step is a location screening exercise, which identifies the critical plant areas. The second step involves the estimation of the fire frequency.

3.1.1 Location Screening

Theoretically, one must determine all areas with the potential for fire starting. However, this is practically impossible and is not necessary as some areas may not be important. This step of the procedure allows for unimportant areas to be screened out while retaining areas that contribute significantly to risk. The locations containing the fire vulnerable safety equipment are identified. The loss of all equipment at that location is then postulated. If it is found that an initiating event (Loss of Coolant Accident (LOCA) or transient) will not occur, the location is eliminated from consideration. Given a LOCA or transient, a number of safety functions are required for safe shutdown. If the loss of all equipment in the location of interest prohibits the performance of any or all required functions, the location is tabbed for further analysis.

This screening method, described by Kazarians and Apostolakis (1981) was employed in the fire risk portions of the Zion (Commonwealth Edison Company, 1981) and Indian Point studies (PASNY, 1982). In these analyses, the fire induced loss of control of safety systems is judged to dominate fire induced hardware losses, and therefore the critical locations that are investigated contain electrical cables and/or switchgear for many safety and non safety systems.

Given this assumption, and a fire that has caused an initiating event, it is then important to determine whether the same fire can induce failures that will prevent:

1. Reaching and maintaining a condition of negative reactivity.

2. Removing decay heat.

3. Monitoring and controlling the inventory and pressure of the reactor-coolant system (RCS).

Because of the fail-safe logic of the reactor scram circuitry, it is then assumed that the reactor trip is successful. Therefore critical fires will affect the implementation of actions 2 and 3. If no one fire can do this, the fires that can prevent either action 2 or action 3 are considered. However these fires are less likely to be significant contributors to risk, because at least one independent failure in an unaffected safety system is required for (core damage and) radionuclide release to occur.

3.1.2 Fire Occurrence Frequency

Once the locations where a fire has the potential to initiate an accident sequence leading to a release of radionuclides have been identified, it should be determined how often these fires occur. The rate of occurrence can be established from the historical record. Only the plant operating histories are suitable for assessing the risk from plants at power.

Apostolakis and Kazarians (1980) model the frequency of fires for various compartments, using a probability of frequency framework to consistently treat the uncertainties. Starting with broad prior distributions to model their weak state of knowledge they employ the statistical evidence of fires in a specific type of reactor (e.g. light water reactor) and use Bayes' Theorem, to derive the fire frequencies in specific rooms of the building.

The procedure can also be used to update the given distributions for fire frequencies, when further reactor experience is accumulated.

3.1.3 Fire Propagation Analysis

Event trees are used to divide the fire model into four elements: Ignition, Detection, Suppression, and Propagation. Each element will head a column of the event tree. The fire is assumed to start in one component and potentially propagate to the next.

3.1.4 Plant System Analysis

Once the frequencies of fire-induced component losses are assessed, it is possible to estimate the frequency of fire initiated accident sequences leading to core damage (which is not the case for the PBMR since core damage is almost impossible because of the fact that the PBMR has a negative temperature effect ensuring that all core structures never reach temperatures at which they start losing their integrity including graphite).

Like other initiating events, separate event trees are constructed for fires because the operator, rather than automatic actions, may be responsible for shutting down the plant in response to a fire. This is often done by simply modifying the front end of existing event trees for other initiating events to specialize them for fires. The postulated fire can be assumed to coincide with another initiating event, in which case the original event tree structure would be retained. (NB. However, if fires are to be treated as a separate event, care should be taken that data from which basic component-failure rates are determined do not double-count these failures from fires).

Disadvantages of the Method

This method however detailed and like many others has disadvantages. One of the disadvantages is the potential for omitting fire sequences leading partway to core meltdown but requiring additional component failures to result in the top event. Another disadvantage is that this method does not account for the possibility that the fire-caused simultaneous failures of many instruments and/or nonsafety systems may initiate accident sequences.

Comments

This method was chosen because compared to the other methods it is more detailed and more explanatory than other methods though it requires a lot of details about the plant, but it contains more familiar information than the other methods. It may not be possible to do all that this method requires, the reason being that we do not have a built plant yet and everything about the plant keeps changing so there is missing information and some of the things haven't been done yet.

3.2 Application of the methodology

3.2.1 Fire Hazard Analysis

This methodology has been mostly for Light Water Reactors (LWR), Pressurized Water Reactors (PWR), and Boiling Water Reactors (BWR). The methodology does not include some of the steps done here, but the methodology had to be altered because as mentioned previously the PBMR plant is still in the design phase.

3.2.1.1 Identification of Safety Functions and their Equipments

Safety functions were identified in order to make a list of related systems so as to know which systems will be more vulnerable and where to locate them in the building layout.

Safety functions are processes by which power plants control the reactor activities to ensure that there are no accidental releases of radioactivity to the environment.

These are activities such as: heat removal/cooling, control of heat generation (that is the control of the fission chain reaction).

There are systems involved in fulfilling the safety functions and these systems also have sub-systems that act as their support systems.

Table 1 Safety Systems and their Support Systems contains the safety functions that are important for reaching a safe reactor shutdown in case of a fire accident; it also contains the related safety systems and their support systems.

Table 1 Safety Systems and their Support Systems

Safety Function	Safety System	Support System
Heat removal	SBS	Electricals, Water circuits, Back-up cooling tower (for RCCS & CCS), Compressed air (for CCS).
	CCS	
	RCCS	
Reactivity Control	RCSS: RCS	Electricals
	RSS	

Fire tends to induce/introduce inter-system dependencies in power plants.

Table 2 Safety System Components and their Support Systems (System Dependencies) below show the safety system components and their inter-system dependencies.

Table 2 Safety System Components and their Support Systems

(System Dependencies)

Safety System components	**Support Systems**					
1.Core Fuel-RCS 18 Control Rod Systems which consist of the following (per system):	**RPS**	**EPS**	**Power plant electrical system**			
Control rod	✔	✔				
Control Rod Drive Mechanism (CRDM) with position indicator	✔	✔	✔			
Rod Emergency Shock Absorber						
2.Core Fuel-RSS 17 Mutually Independent SAS Systems which consist of the following (per system):						
Gas blower (shared)			✔			
Storage container						
Discharge container						
Storage container valve	✔	✔	✔			

3.Core Fuel-CCS	**ACS-Auxiliary closed circuit**	**ACS-Back-up cooling tower closed circuit**	**WMPS**	**EPS**	**OCS**	**Power plant electrical system**
Gas mixing chamber						
Gas-to-gas heat exchanger (Recuperator)						
Gas-to-water heat exchanger (stainless steel plates)						
Buffer circuit heat exchanger (stainless steel plates)	✔	✔				
Buffer circuit expansion tank			✔			
Two buffer circuit pumps				✔	✔	✔
Buffer circuit pump outlet NRVs						
3 x 50% blowers				✔	✔	✔
3 x blower inlet motor operated valves				✔	✔	✔
3 x blower return line motor operated valves				✔	✔	✔
Recuperator by-pass motor operated valve				✔	✔	✔
4.SBS	**EPS**	**OCS**	**Power plant**			

			electrical system			
2 x 100% blowers	✔	✔	✔			
2 x blower outlet motor operated valves	✔	✔	✔			
5.Core Fuel-RCCS, Train A	**ACS-Open circuit**	**ACS-Back-up cooling tower closed circuit**	**WMPS**	**EPS**	**OCS**	**Power plant electrical system**
Heat exchanger (titanium plates)	✔					
Expansion tank			✔			
Expansion tank WMPS motor operated valve				✔	✔	✔
Pump				✔	✔	✔
Pump outlet NRV						
Standpipes	✔	✔	✔			
Anti-siphoning devices						
Rupture discs						
Temperature, pressure and level indicators				✔	✔	✔
Back-up cooling tower supply motor operated valve				✔	✔	✔
Back-up cooling tower return motor operated valve				✔	✔	✔

Standpipe isolation motor operated valves				✔	✔	✔
6.Core Fuel-RCCS, Train B						
Heat exchanger (titanium plates)	✔					
Expansion tank			✔			
Expansion tank WMPS motor operated valves				✔	✔	✔
Pump				✔	✔	✔
Pump outlet NRV						
Standpipes	✔	✔	✔			
Anti-siphoning devices						
Rupture discs						
Temperature, pressure and level indicators				✔	✔	✔
Back-up cooling tower supply motor operated valve				✔	✔	✔
Back-up cooling tower return motor operated valve				✔	✔	✔
Standpipe isolation motor operated valves				✔	✔	✔
7.Core Fuel-RCCS, Train C						

Heat exchanger (titanium plates)	✔					
Expansion tank			✔			
Expansion tank WMPS motor operated valves				✔	✔	✔
Pump				✔	✔	✔
Pump outlet NRV						
Standpipes	✔	✔	✔			
Anti-siphoning devices						
Rupture discs						
Temperature, pressure and level indicators				✔	✔	✔
Back-up cooling tower supply motor operated valve				✔	✔	✔
Back-up cooling tower return motor operated valve				✔	✔	✔
Standpipe isolation motor operated valves				✔	✔	✔

3.2.2 Location Screening

The following locations were screened according to their containment of the fire vulnerable safety equipment. All other areas that were not included in the table below were screened out since they did not contain any fire vulnerable safety equipments.

The table below is a description of the building layout. The first number represents the level of the floor, the negative sign means that the particular level is below the ground level, zero is the ground level, and the levels with no signs mean that those levels are above the ground level, the last numbers represent the room numbers. This is a list showing all the safety equipment and it also shows what these equipments are housed with in all the different rooms of the plant. The cable locations have been included, electrical switchgear rooms, generator room, control rooms, battery rooms, and the power electronics room because according to literature, these areas are areas, which have proved to be likely fire areas.

Table 3 Building Layout

Area no (Level no and room no)	**Area Description**	**List of equipment in the area**	**Maintenance tasks in the area**
-18:32	Cable Duct		
	Cable Duct	Cables	None
-18:33	Cable Duct		
-18:34	Cable Duct		
-18:35	Cable Duct		
-18:29	PCU Cavity	PCU floor area: Pre & Intercooler.	
		Waste Water Sump.	Check sump pump system functioning
	Intercooler	Intercooler	Inspection of outer casing for cracks, leaks
	Pre-cooler	Pre-cooler	Inspection of outer casing for cracks, leaks
	Recuperator	Recuperator	Inspection of outer casing for cracks, leaks
	Recuperator	SBS	
		SBS valves	
		Recuperator Bypass valves	
	PCU Area	HPT	
		LPT	

		Power Turbine	
-14:45	Core unloading Device	Core unloading device (x2)	Replace drive
			Replace gearbox
			Replace shaft penetration
			Test flange seals
			Replace spindle with bearings
			Empty scrap collection casks
			Repair bad electrical connection
	FHSS valves	Isolation valves (x40)	Leak test on flange seals
			Replace valve drive
			Replace shaft penetration
			Replace inserts
			Service shaft penetration
		Counters (x2)	Leak test on flange seals
			Repair faulty electrical connection
			Replace inserts

		Sphere riser pipes	None
		Valve blocks	None
		Gas evacuation systems He shut off valves (x44)	Replace service kit
		Fork Lifts	None
		Valve maintenance tool	None
		Laydown pallets	None
		Core unloading device tool	None
		Scrap cask emptying device	None
	RSS valves		
-14:51	RSS blowers	RSS blowers	
-9:46	Bottom of the RPV	Bottom of RPV	NDT the bottom of the RPV for cracks, deterioration of material, creep
	RPV	RPV	
		Hot & cold manifold pipes	
		RCCS	
-9:47	Working floor area CCS		
-4:53	CCS	CCS	
		RPVCS	
-4:54	Electrical switchgear	11 kv switchgear	Inspect Equipment
		400 kv switchgear	Replace components

		Dry transformers	
		Circuit breaker trolleys	
		Replace components	
-4:104	Electrical switchgear	11 kv switchgear	Inspect Equipment
		400 kv switchgear	Replace components
		Dry transformers	
		Circuit breaker trolleys	
		Replace components	
0:63	Generator switchgear	Generator braking equipment	Test brake system
		Lighting board	
		Excitation control panel	Calibrate and check
		Generator main/back-up protection panel	
		Generator measurement panel	
0:60	Control room	Operator desks	Maintain panels and VDUs
		Video monitors	
		PEI displays	
		RPS displays	
		Printers	Check printers
0:59	Control equipment room	PLCs & control equipment	Replace cards

	room	Engineering control station	Plant optimization
			Configuration updates on plant
4:75	Electrical busbars & breakers	Generator busbars	Test transformer
		Generator circuit breakers	Check circuit breakers
		Brake air-cooled transformer	Check contacts
		Brake circuit breaker	
		Earthing transformer & resistor	
		Voltage transformer	
		Excitation transformer	
		Earthing switch	
		Generator earthing equipment	
4:111	Battery room 1	110 VDC electrical protection batteries (2 sets)	Inspect batteries
			Top-up electrolyte in batteries
			Replace individual battery cells
			Replace all battery cells (end of life)

		Nicad batteries	Inspect batteries
			Top-up electrolyte in batteries
			Replace individual battery cells
			Replace all battery cells (end of life)
		Ventilation fans	
4:110	Battery room 2	110 VDC electrical protection batteries (2 sets)	Inspect batteries
			Top-up electrolyte in batteries
			Replace individual battery cells
			Replace all battery cells (end of life)
		Nicad batteries	Inspect batteries
			Top-up electrolyte in batteries
			Replace individual battery cells
			Replace all battery cells (end of life)
		Ventilation fans	

4:71	Power electronics room	DC chargers	Maintain this equipment
		UPS	
		EMB cabinets, including test equipment	
		PLC equipment	
		RCS & PEI	
4:73	RCS room		
11:80	RPV	RPV	No access or maintenance in this area
		RCCS	
		RSS	
		RCS	
21:87	Working floor	RCSS maintenance trolleys	
		RCCS water level sensors	Maintain switches/senso rs
	Controlled ventilation exhaust air plant	Block filters	Replace filters
		Centrifugal fans	Replace fans
		Dampers and motors	Inspect and calibrate dampers
		Maintain and replace motors	1 yearly
		Controls	Inspect and maintain controllers and actuators
General	All rooms	A/C ducting	

		Cooling water supply lines	
		Electrical cabling to individual equipment	

3.2.3 Fire Occurrence Frequency

To compute the fire occurrence frequency for different specified compartments of the plant, statistical data and historical experiences are used, that is the number of fires that occurred previously in each room of interest in all plants of a similar type plus their number of years of operation are combined and from that, the fire frequencies are computed.

The Pebble Bed Modular Reactor (PBMR) is the first reactor of its kind and it has not been built yet, therefore there is not enough data and no experience to do the required calculations, like computing the fire frequency. There are however other HTGRs and HTRs that have been operating for a number of years, but there hasn't been any historical fire experience data that has been reported yet from them. In this regard statistical data and historical experiences available from reactors such as the LWRs, PWRs and BWRs (NUREG-1150 Volume 1, NUREG/CR-2300 Volume 2, NUREG/CR-4840 SAND 88-3102) were adopted for the PBMR.

This data is suitable in a sense that, all cables, control rooms, batteries, generators, and electrical switchgears, in all power plants irrespective of the type of plant are the same.

Table 4 Statistical Evidence of Fires in Light Water Reactors and the related fire frequencies.

Area	**Number of fires**	**Number of compartment years**	**Fire Frequency (per room year)**
Control Room	3	681.0	4.1E-3
Cable spreading room	2	747.3	7.2E-3
Diesel generator room	37	1600.0	1.9E-2
Reactor building	22	847.5	5.0E-2
Auxiliary building	43	673.2	3.4E-2
Electrical switchgear room	4	1346.4	1.3E-2
Battery room	4	1346.4	2.9E-3

The fire frequencies for the different rooms of the PBMR plant were stipulated in the following manner:

Generic subdivisions like the diesel generator room, cable rooms, control rooms, electrical switchgear rooms, and the battery rooms were considered to be common to all types of plants and so the fire event data were used directly with the number of years of operating experience per subdivision to determine the frequency.

Plant specific subdivisions like the reactor building, and the auxiliary building were considered on a room-by-room basis. For these rooms the frequency was obtained by dividing the frequency from data amongst all systems contained in the specific subdivision.

The fire frequencies for the reactor building, and the auxiliary building were divided for each room in a particular building according to the number of systems (especially systems that can burn) in each room respectively. And so the sum of all system frequencies in a room constituted the frequency of the particular room.

Table 3-5 Fire frequencies for different rooms in the PBMR plant

Area no (Level no & room no)	**Description of location**	**Area Description**	**List of equipment in the area**	**Fire frequency (per room year)**
-18:32	Cable rooms	Cable Duct		2.88E-3
		Cable Duct	Cables	
-18:33		Cable Duct		1.44E-3
-18:34		Cable Duct		1.44E-3
-18:35		Cable Duct		1.44E-3
0:63	Generator room	Generator switchgear	Generator braking equipment	1.9E-2
			Lighting board	
			Excitation control panel	
			Generator main/back-up protection panel	
			Generator measurement panel	
0:60	Control rooms	Control room	Operator desks	2.05E-3
			Video monitors	
			PEI displays	
			RPS displays	

			Printers	
0:59		Control equipment room	PLCs & control equipment	2.05E-3
			Engineering control station	
4:111	Battery rooms	Battery room 1	110 VDC electrical protection batteries (2 sets)	1.45E-3
			Nicad batteries	
			Ventilation fans	
4:110		Battery room 2	110 VDC electrical protection batteries (2 sets)	1.45E-3
			Nicad batteries	
			Ventilation fans	
-4:54	Electrical switchgear rooms	Electrical switchgear	11 kv switchgear	6.5E-3
			400 kv switchgear	
			Dry transformers	
			Circuit breaker trolleys	
			Replace components	
-4:104		Electrical switchgear	11 kv switchgear	6.5E-3

			400 kv switchgear	
			Dry transformers	
			Circuit breaker trolleys	
			Replace components	
-18:29	Auxiliary building	PCU Cavity	PCU floor area: Pre & Intercooler.	
			Waste Water Sump.	
		Intercooler	Intercooler	
		Pre-cooler	Pre-cooler	
		Recuperator	Recuperator	
		Recuperator	SBS	3.4E-2
			SBS valves	
			Recuperator Bypass valves	
		PCU Area	HPT	
			LPT	
			Power Turbine	
-14:45	Reactor building			5.55E-3
		FHSS valves	Isolation valves (x40)	

			Gas evacuation systems He shut off valves (x44)	
		RSS valves		
-14:51	Reactor building	RSS blowers	RSS blowers	1.85E-3
-9:46	Reactor building	Bottom of the RPV		3.70E-3
		RPV	RPV	
			RCCS	
-9:47	Reactor building	Working floor area CCS		1.85E-3
-4:53	Reactor building	CCS	CCS	3.70E-3
			RPVCS	
4:75	Reactor building	Electrical busbars & breakers	Generator busbars	1.67E-2
			Generator circuit breakers	
			Brake air-cooled transformer	

			Brake circuit breaker	
			Earthing transformer & resistor	
			Voltage transformer	
			Excitation transformer	
			Earthing switch	
			Generator earthing equipment	
4:71	Reactor building	Power electronics room	DC chargers	9.25E-3
			UPS	
			EMB cabinets, including test equipment	
			PLC equipment	
			RCS & PEI	
4:73	Reactor building	RCS room		1.85E-3
11:80	Reactor building	RPV	RPV	7.40E-3
			RCCS	
			RSS	
			RCS	

3.2.4 Fire Propagation Analysis

Two stage event trees for two redundant components were used to divide the fire model into four elements, Ignition, Detection, Suppression and Propagation. It was assumed that once fire started in one component it would potentially propagate to the next. The loss of all equipments in a room frequency was obtained using the Risk Spectrum code. The input data for the failure to suppress and detect were from literature [5].

The plant was divided into seven subsections and an event tree was constructed for each subsection, namely, the cable room, the control room, the battery room, the generator room, the electrical switchgear room, the auxiliary building, and the reactor building.

Figure 3-1 Event tree for the control room

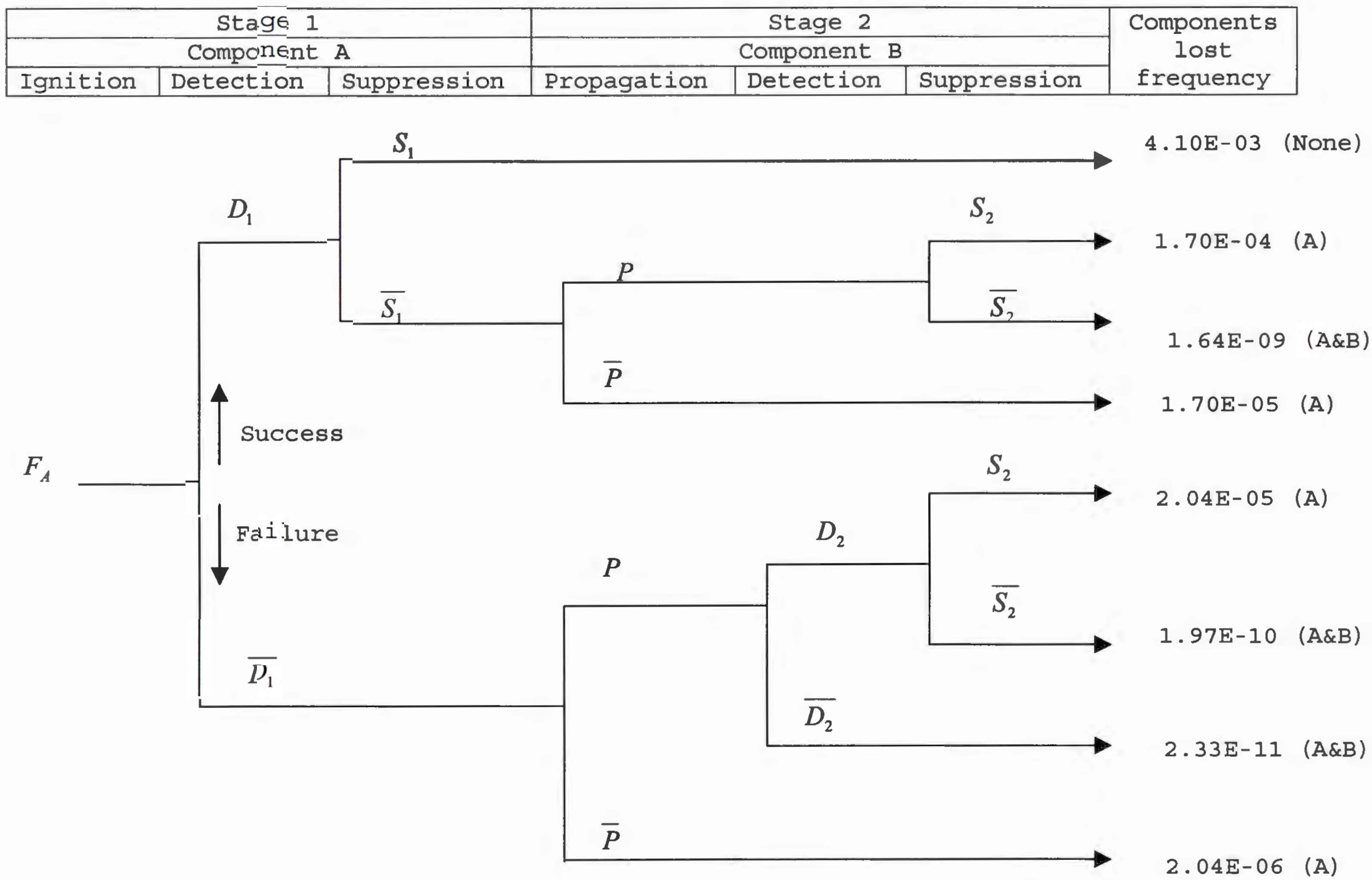


Figure 3-2 Event tree for the cable room

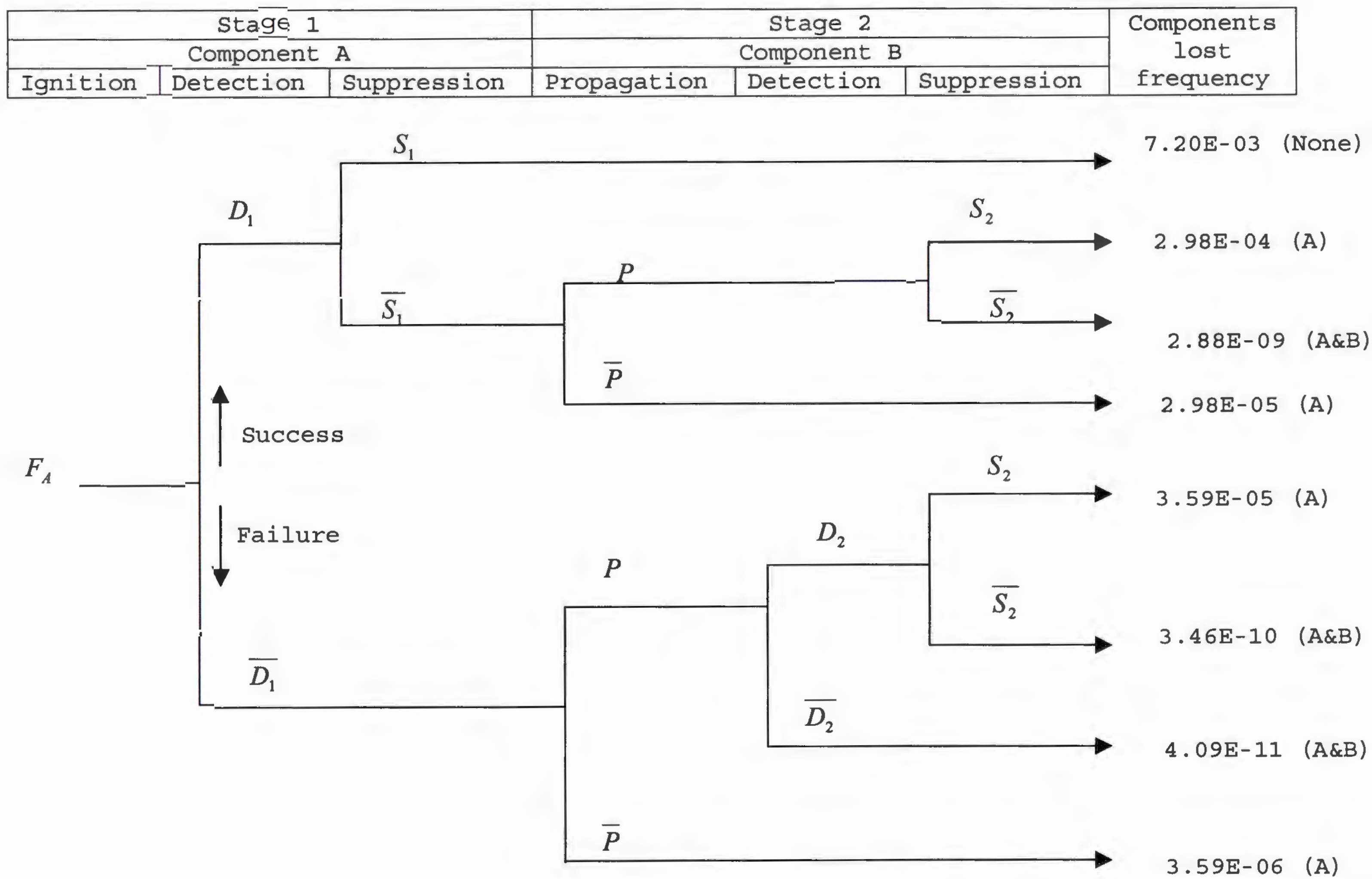


Figure 3-3 Event tree for the electrical switchgear room

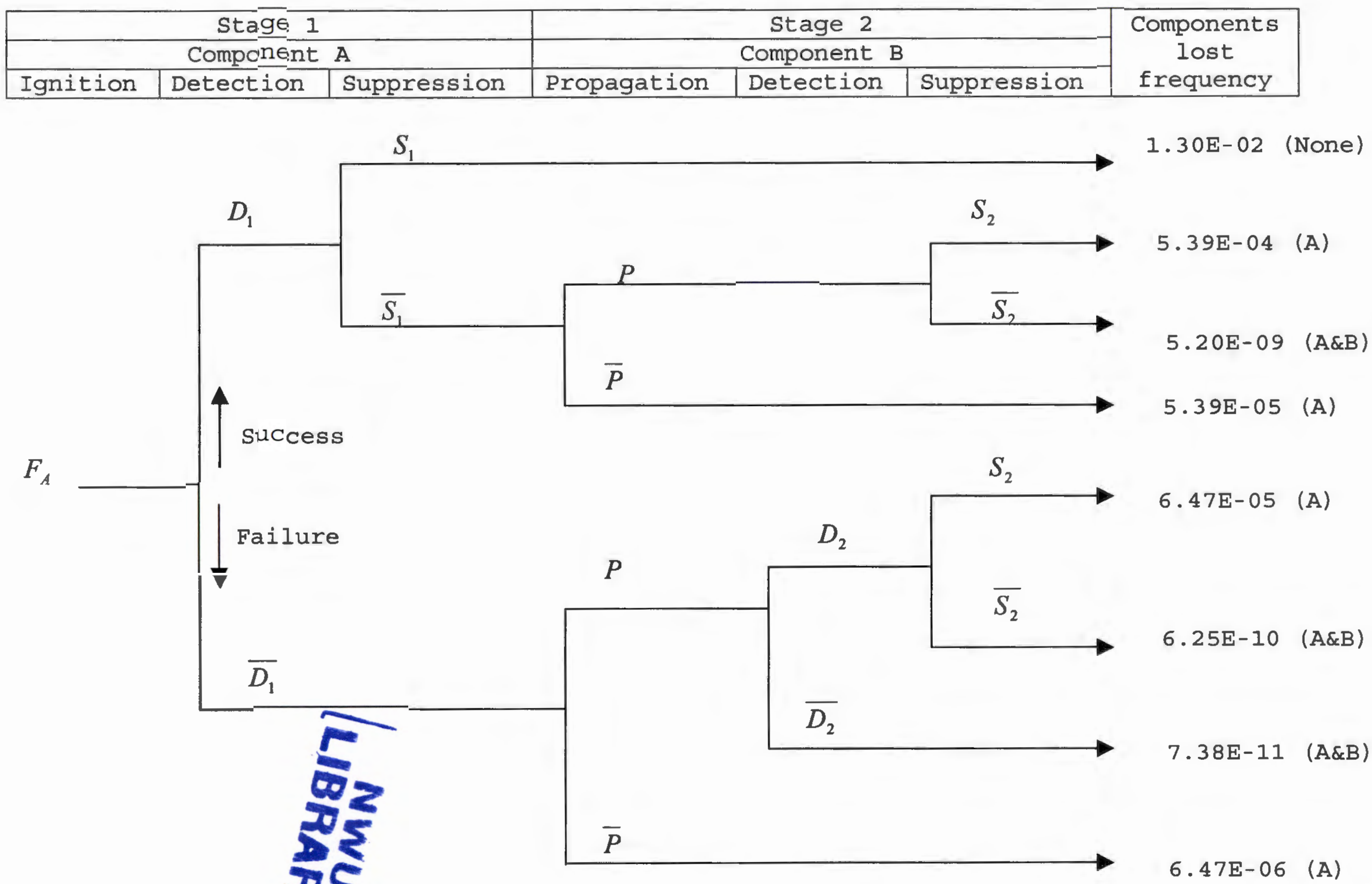


Figure 3-4 Event tree for the battery room

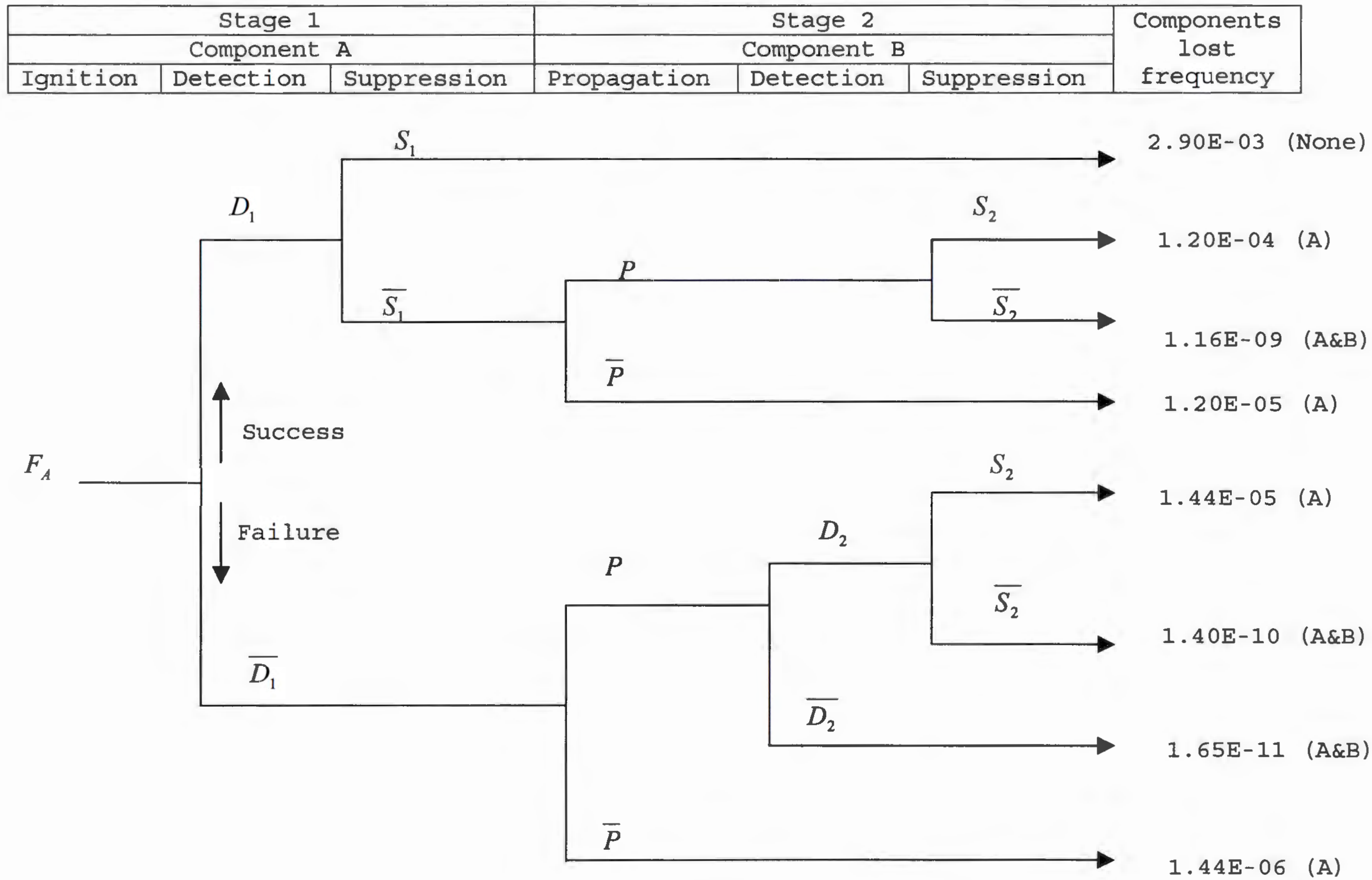


Figure 3-5 Event tree for the generator switchgear room

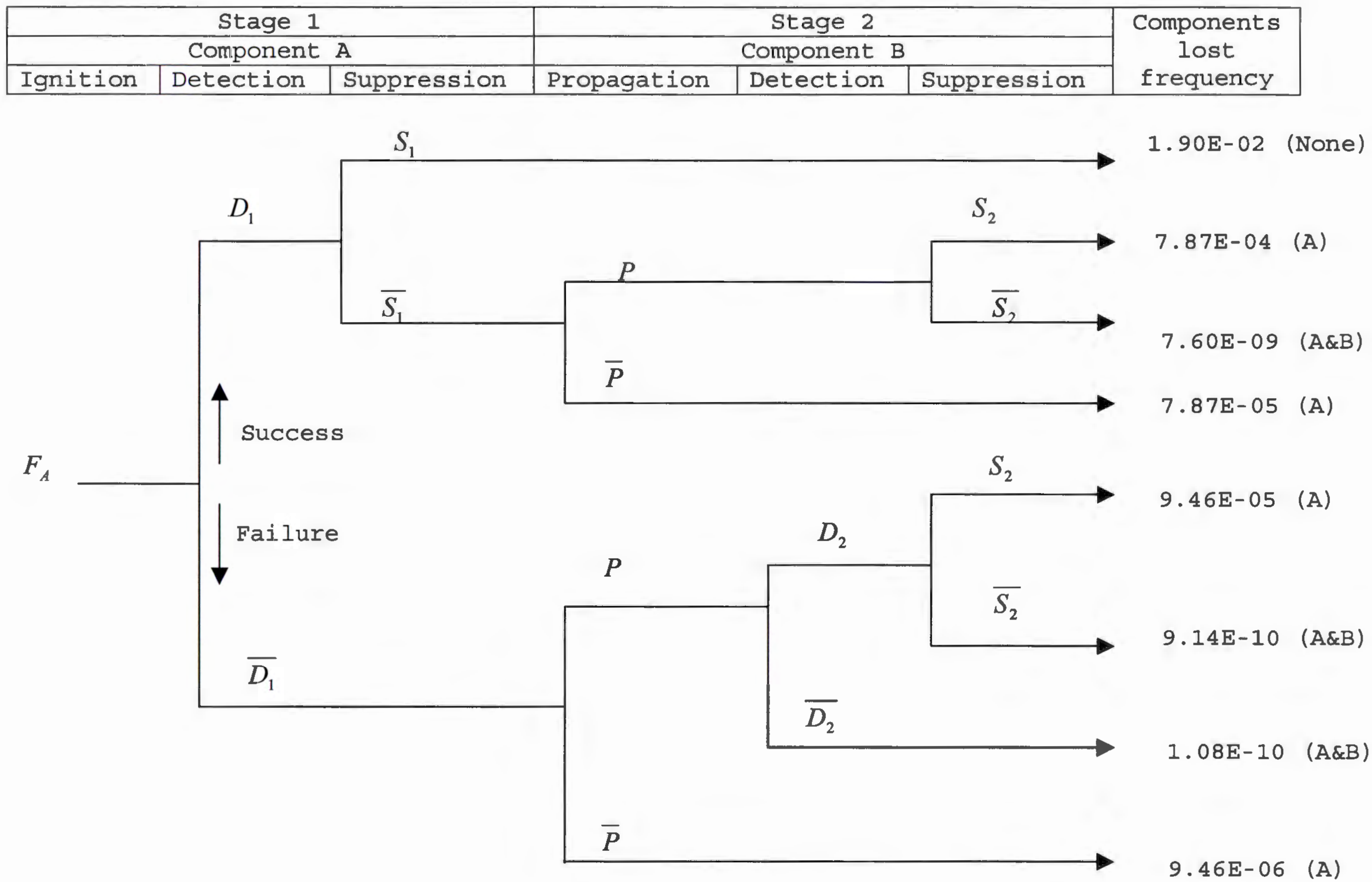


Figure 3-6 Event tree for the reactor building

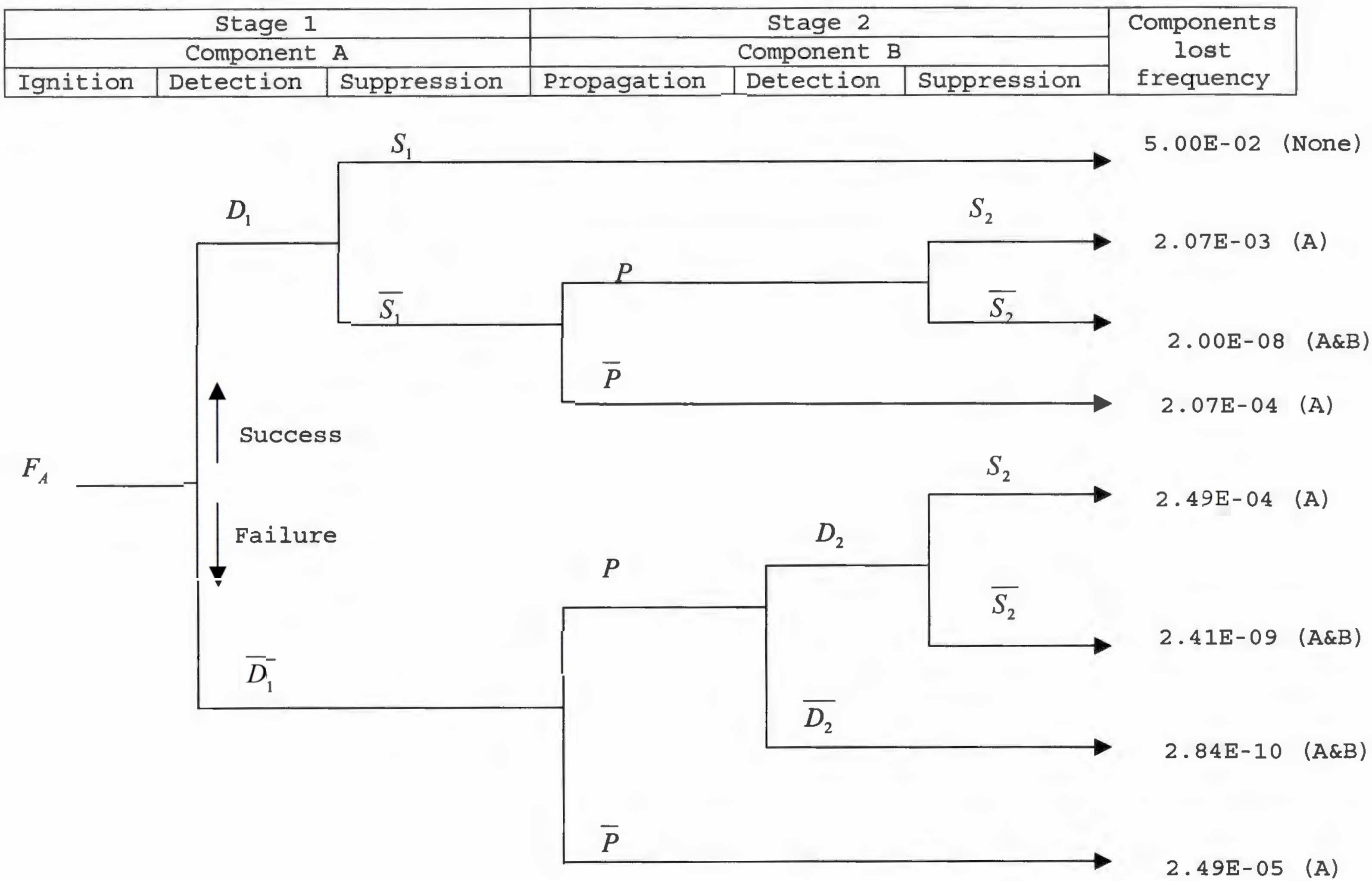
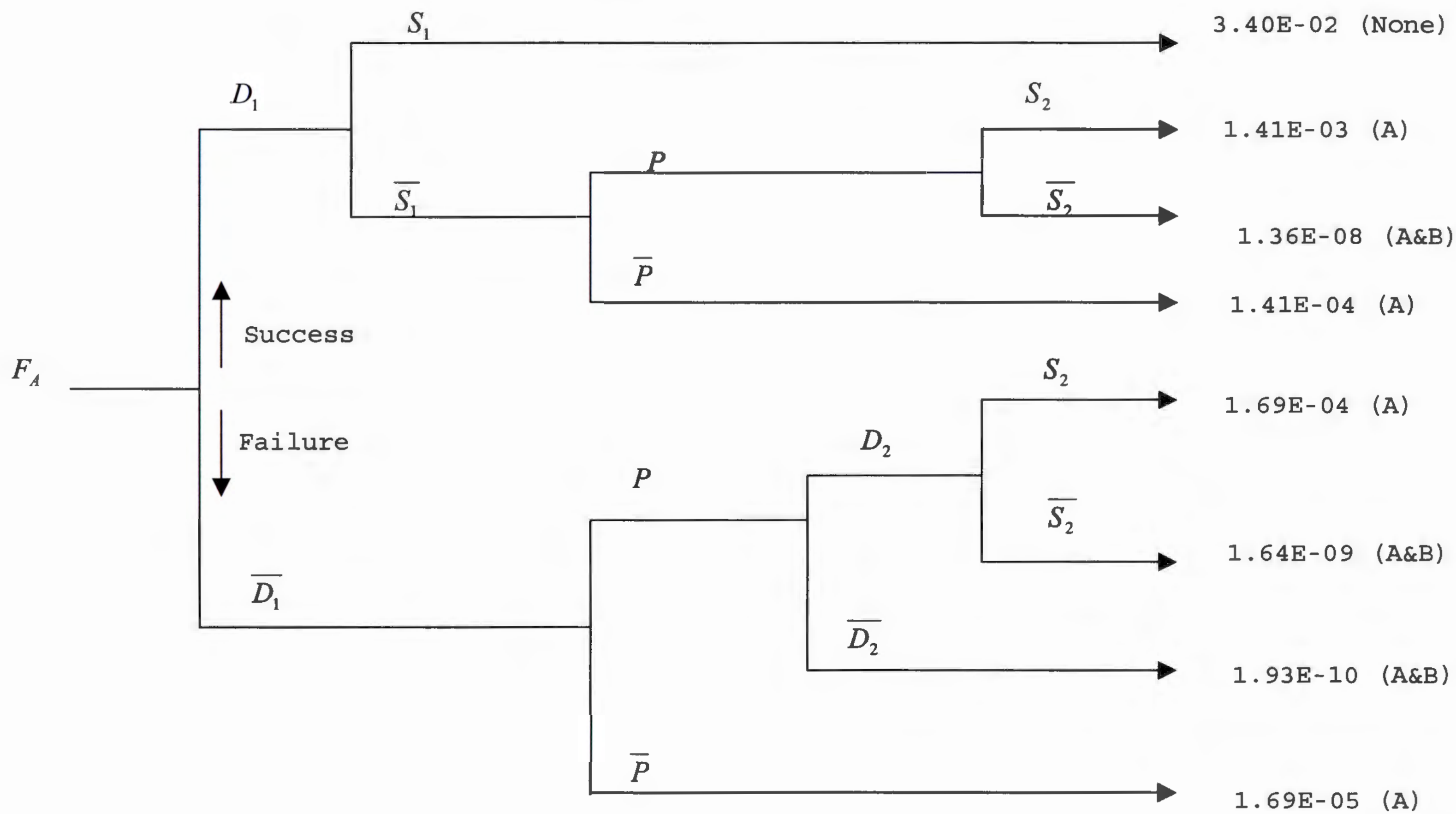


Figure 3-7 Event tree for the auxiliary building

Stage 1			Stage 2			Components lost frequency
Component A			Component B			
Ignition	Detection	Suppression	Propagation	Detection	Suppression	



3.2.5 Plant system analysis

The system analysis has been done for the PBMR plant in the PBMR PRA Report (Chapter 5 Data Analysis). For this project only the safety systems and their support systems were considered in the plant system analysis for fire accidents.

As mentioned earlier in the document, fire is unpredictable, it can cause a short circuit instead of an open circuit, fire also tends to induce multiple system inter dependencies. It is therefore important to note all the support systems for different components and systems of the plant.

As shown earlier (Table 2 Safety System Components and their Support Systems (System Dependencies)) all safety systems have a number of components that assist them in functioning. These components also have support systems, which aid them in functioning. By Table 2 Safety System Components and their Support Systems (System Dependencies):

<u>RCS</u>: Some of the RCS components are supported by the RPS and the EPS, and in some also the power plant electrical system is included.

<u>RSS:</u> Some components of the RSS are supported by the RPS and the EPS, and in others also the power plant electrical system is included.

CCS: Some of the CCS components are supported by the ACS-auxiliary closed circuit and the ACS-back up cooling tower closed circuit, and some are supported by the WMPS, whilst others are supported by the EPS, the OCS, and the power plant electrical system.

SBS: The components of the SBS are supported by the EPS, OCS, and the power plant electrical system.

RCCS: Some components of the RCCS are supported by the ACS-open circuit, and some by the ACS-back up cooling tower closed circuit, some by the WMPS, and others by the EPS, OCS, and the power plant electrical system.

Support Systems

Support systems also have components, which in turn also have their support systems.

RPS: Its components are supported by the power plant electrical system.

EPS: Its components are supported by the power plant electrical system.

OCS: Components are supported by the power plant electrical system.

Power Plant Electrical System: The components are supported by the EPS and the OCS.

ACS-auxiliary closed circuit: Some of this system's components are supported by the ACS-open circuit, and some by the WMPS, and others by the EPS, OCS, and the power plant electrical system.

ACS-back up cooling tower closed circuit: Some of its components are supported by the WMPS, and others by the EPS, OCS, and the power plant electrical system.

WMPS: Components are supported by the EPS, OCS, and the power plant electrical system.

CHAPTER 4

RESULTS AND HYPOTHESIS

This section presents the results of probabilistic fire risk assessment performed for the PBMR. The results are presented in terms of the fire frequency in the different fire compartments. The frequency estimates obtained here are a result of quantifying the detection and suppression failure fault and event trees. Both the initiating event frequencies of fire and fire event tree sequence frequencies are presented. Critical safety equipment and their support systems, including cables and power, were identified in each compartment.

Once the sequence frequencies have been calculated, one determines whether the fire in a particular compartment could result in an internal initiating event. That is, whether the fire will cause a pipe break, reactivity excursion, or loss of the main power system. If an internal initiating event could be caused as a result of fire in a particular room, then the frequency is compared to a predetermined value of 10^{-2} per year. This means that only the contribution of fire to the release of radionuclide greater than 1% is considered. If the resulting fire sequence frequency is greater than this value, then the particular frequency is retained. This is then multiplied by the internal initiating event frequency to determine the overall risk.

4.1 Results

4.1.1 Fire Initiating Event Frequencies for the different rooms of the PBMR power plant.

Table 4.1 below presents the calculated fire initiating event frequencies per identified room. These frequencies were determined using the methodology presented in Chapter 3. The table provides initiating event frequencies of fire in various compartments in the reactor, namely, control room, cable spreading room, diesel generator room, switchgear room, battery room, auxiliary building, and reactor building. The frequencies from Tables were obtained from Nureg 1150 [3] of which the values are exactly the same as the values in Table 3.4 since the PBMR plant does not have its own statistical data that can be used to stipulate the frequencies. For simplicity, it is assumed that once the fire starts, it spreads. This means that the mitigation system response has failed. Thus, the failure frequency is 1. Thus, the total sequences frequency is the same as the initiating event frequency of fire.

Table 4.1 Fire Initiating Event Frequencies

Plant Location	Fire Frequency (per reactor year)
Control room	4.1E-03
Cable spreading room	7.2E-03
Diesel Generator room	1.9E-02
Switchgear room	1.3E-02
Battery room	2.9E-03
Auxiliary building	3.4E-02
Reactor building	7.2E-03

From the table, only three rooms may potentially contribute to internal events and these are diesel generator room, switchgear room and auxiliary building. However, loss of equipment in these rooms will not lead to any radiological release. However, this is a point, which requires further studies to determine if there is any release as a result of fire.

4.2 Hypothesis

4.2.1 Fire occurrence frequencies

The occurrence of fire in the PBMR power plant is not expected to contribute significantly to the total risk to the public and plant personnel. This is a result of the inherent safety of the plant. Also, The fire frequencies for the Pebble Bed Modular Reactor plant are expected to be much less than the ones from the Light Water Reactors, Pressurized Water Reactors, and the Boiling Water Reactors, once it has gathered its own empirical data, because of its inherent safety. Another factor that may add to the fire frequencies being low is the unique design of the plant in which very small numbers of combustible materials are used where their use could not be avoided; the cable routings take different directions so that at least one cable route is available if the other one is knocked out by fire during a fire accident; all major pumps are surrounded by fire barriers to minimize the spread of fire from one channel to the next during fire accidents.

4.2.2 Loss of equipment frequencies (Fire Propagation)

The obtained frequencies seem to follow a similar pattern for all the different rooms in the plant. According to these results it seems highly unlikely that fire will propagate from one equipment to the next in a room, this is shown by very low frequencies (close to impossible) that a fire accident may destroy more than one system in a room without being noticed or without something being done.

When using the Risk Spectrum code it was noticed that as the time between detector tests (during maintenance of the detectors) increased, the loss of equipment frequencies also increased but, the more frequent the tests are done the lower the loss of equipment frequencies become (close to impossible). The disadvantage however of more frequent tests is that it is very expensive.

CHAPTER 5

SUMMARY, CONCLUSION, AND RECOMMENDATIONS

5.1 Summary

In this report the Probabilistic Risk Assessment methodology was used to assess the frequencies of occurrence of fire. To do this, the safety functions of the plant were determined. Then the safety systems that would be needed to perform these safety functions were identified together with their support systems. The safety functions considered are: control of heat generation, heat removal.

To determine the frequency of fire occurrence, areas were identified where the various safety system are located. The plant's building layout was used for this purpose. The various areas were further broken down into smaller compartments for ease of representation. Since there is no operational history for the PBMR, data from the Light Water Reactors, Pressurized Water Reactors, and Boiling Water Reactors was used to derive the fire occurrence frequencies for the different locations in the plant. To obtain frequencies for each compartment in the PBMR plant, the frequencies for different areas from the LWR and PWR plants were broken down according to the number of systems present in each compartment in the PBMR plant.

Sequence frequencies are generally determined by combining the initiating event frequencies with the failure frequencies of various systems considered in the fire mitigation systems. This part models the propagation of fire from one system to another prior to detection and suppression failures in a room, and the resulting loss of equipment. To simplify the analysis, the conditional probability of the sequence was assumed to be unity. The total fire frequency was then the equivalent to the initiating event frequency.

The Fire PRA analysis was done to investigate the impact of fire on the safety of the PBMR plant and also on the existing PBMR PRA risk frequency. This analysis was also done to determine whether the PBMR plant would be able to reach a safe reactor shut down in case of a fire accident, and most importantly to see if the total risk due to the operation of the plant is still within the risk limit specified by the National Nuclear Regulator.

5.2 Conclusion

From the frequencies presented in Table 4.1, it is clear that fire is not expected to contribute significantly to the overall risk of the PBMR. The initiating event frequencies for the various locations were below a predetermined cut-off frequency.

Thus, the impact of fire on the existing PBMR PRA risk frequency is negligibly small.

5.3 Recommendations

This study was a preliminary assessment of the impact of fire in the PBMR. The study was limited as a result of the fact that the data used as input was generic. Therefore, when new and relevant operational data becomes available, the study will have to be updated. Also, this study did not involve sensitivity and uncertainty analysis. This is essential in any PRA study. Future studies should include these assessments.

Glossary of terms

TERM	DEFINITION
Abnormal Condition	An unplanned event or sequence of events that results in undesirable consequences. An incident with specific safety consequences or impacts such as damage to one or several barriers, thus leading to the release of radioactive material.
Active Cooling System (ACS)	A system that has specifically designed controlled components to cool a system or component
Compressed Air System (CAS)	A system to provide compressed air of suitable quality to: Supply maintenance outlets. Supply pneumatic valve actuators. Pressurize header tanks of the ACS.
Core Conditioning System (CCS)	A system for the removal of decay heat when the Brayton cycle is not operating. It also provides for helium flow through the reactor core heat-up during start-up operations
Cooling Water (CW)	It provides heat removal at pressures lower than the main operating system pressure at all times during operation.
Event	A condition that deviates from normal operation
Event Sequence	A specific unplanned series of events composed of an initiating event and intermediate events that may lead to a deviation from normal operation
External Events	External conditions that cause deviations from normal operation. The examples are:

	Floods, earthquakes, volcano, terrorism, fires etc.
Failure Modes and Effects Analysis (FMEA)	A qualitative method of system analysis which involves the study of the failure modes that can exist in every component of the system, and the determination of the causes and of the effects of each failure mode
Failure Modes, Effects and Criticality Analysis (FMECA)	A variation of FMEA that includes a quantitative estimate of the significance of the consequence of a failure mode
Failure	The termination of the ability of an entity to perform a required function.
Failure Mode	The effect by which a failure is observed
Fire Protection System (FPS)	A system that provides early warning of fire in the module, PPSB and Auxiliary building, suppresses fire, is used for fire extinguishing purposes and prevents the spread of fire.
Fuel Handling and Storage System (FHSS)	A system for loading and removal of fuel and graphite spheres from the reactor
Heating, Ventilation and Air-conditioning (HVAC) System	A system that maintains specified environmental parameters of temperature and (where required) humidity and minimizes the internal building contamination by air renewal, maintaining direction of airflow and removing radioactive particles by purging and filtering.
Helium Inventory Control System (HICS)	A system for regulating the quantity and quality of helium in the PCU.
Initiating Event	The first event in an event sequence.
Internal Events	Internal conditions that cause deviations from normal

	operation, and the examples are: loss of coolant, small/medium/large pipe break, control rod failure, accidental drop of the SAS etc.
Main Power System (MPS)	The Main Power System physically combines the functions of the Reactor Unit (RU) and Power Conversion Unit (PCU) to convert nuclear energy into electricity
Maintainability	The ability of an item under given conditions of use, to be restored to a state in which it can perform its intended function, when maintenance is performed under given conditions and using stated procedures and resources.
Module	A single, stand-alone PBMR, containing all systems for operation.
Operational Control System (OCS)	A system that controls and monitors major plant, auxiliary systems and support systems. Start-up, power generation and shutdown are controlled through the OCS. The OCS also stores process data and makes data available for other external systems
Pebble Bed Modular Reactor (PBMR)	A high-temperature, direct cycle, helium-cooled nuclear reactor with fully ceramic spherical fuel elements.
Plant Mode	Manoeuvring or changes in process variables within a specific state
Plant State	Plant conditions or parameters that are fundamentally different from each other. The plant can continue 'indefinitely' in any specific state
Plant Transition	Changing from one plant state to another. The plant can only be in transition for a specific

	time period, and has defined initial and end conditions.
Power Conversion Unit (PCU)	An assembly of equipment, comprising the high-and low-pressure turbo-units, power turbine generator, a recuperator and coolers, combining to convert the heat from the helium into electricity
Power Turbine Generator (PTG)	An assembly to convert fluidic helium energy into electricity
Reactivity Control and Shutdown System (RCSS)	The system responsible for controlling the reactor during normal operation, effecting a fast transition to a sub-critical state on demand, and maintaining the reactor sub-critical indefinitely in a cold condition (100 °C)
Reactor Cavity Cooling System (RCCS)	A system to transfer the heat from the reactor cavity during all modes of reactor operation to the ultimate heat sink (sea or to the atmosphere, when the sea is not available).
Reactor Pressure Vessel (RPV)	The vessel, which supports the reactor core structures and forms part of the MPS pressure boundary.
Reactor Pressure Vessel Conditioning System (RPVCS)	A system for maintaining the RPV at a homogeneous temperature, in order to maintain operating temperatures in the range for which the materials have been qualified.
Reactor Protection System (RPS)	A system that shuts the reactor down when certain values as specified in the General Operating Rules (GOR) and/or technical specifications have been exceeded. The RPS also prevents restart of the reactor until normal operating conditions have been restored.
Recuperator	A device for returning heat in the gas downstream of the

	turbines to the gas returning to the reactor, thus improving cycle efficiency.
Spent Fuel	Fuel that has reached its maximum design burn-up.
Start-up Blower System (SBS)	A system that circulates helium gas through the main gas circuit of the MPS in order to: Start the Brayton thermodynamic cycle or Cool the core intentionally if the Brayton cycle is not operational
Power Pant Control Room	The control room in the Services Building from where all modules will be controlled.
Used Fuel	Fuel that has undergone a partial fission reaction, but has not yet reached its maximum design burn-up.

References

1. Apostolakis G, Bley D.C, Kazarians M; 1982; Nuclear Safety, Accident Analysis, Methodology for Assessing the Risk from Cable Fires; Volume 23.

2. Baschwitz R., Kohout R., Marek J., Richter P.I., Slate S.C. (eds.); 1993; Proceedings of the 1993 international Conference on nuclear waste management and environmental remediation, Volume 3: Environmental remediation and environmental management issues; New York, NY (United States); American Society of Mechanical Engineers.

3. Basic Licensing Requirements for the Pebble Bed Modular Reactor, Rev. 2 - Document No.: LG-1037.

4. Beard A.N, Bishop S.R, Drysdale D.D, Holborn P.G; 1993; Appl.Math.Modelling, Dynamic Modeling of Building Fires; Volume 17; Centre for Nonlinear Dynamics and its Applications; Butterworth-Heinemann; London.

5. Burns N.L, Cassidy B.G, Hitchler M.J; Fire Analysis Probabilistic Risk Assessment Methodology; Westinghouse Electric Corporation; Pittsburgh.

6. Centre for chemical process safety of the American Institute of Chemical Engineers; 1989; Guidelines for Process Equipment Reliability Data with data tables; New York.

7. Fischer Jorg; 1991; Nuclear Engineering and Design 125 (1991) 411-415, Review of the IAEA Fire Symposium; Elsevier Science Publishers B.V; North Holland.

8. Himanen R, Jantunen V.M, Taivainen K; 1990; Probabilistic Fire Analysis For TVO Units I and II, IAEA-SM-305/27; Teollisuuden Voima Oy, Olkiluoto; Finland.

9. Gregory W.S, Krause F.R, Travis J.R, 1991; Nuclear Engineering and Design 125(1991) 411-415, Los Alamos Combustion Modelling for Nuclear Systems; Elsevier Science Publishers B.V; North-Holland.

10. International Atomic Energy Agency; 1992; Procedures for conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1); A Safety Practice; Vienna.

11. Lamarsh John R.; 1983; Introduction to Nuclear Engineering; 2nd Edition; Addison-Wesley Publishers; Canada.

12. Lees Frank.P; 1980; Loss Prevention in the Process Industries; Hazard Identification, Assessment and Control; Volume 1; Butterworth-Heinemann Ltd; Oxford.

13. NUREG/CR-2300; PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants; Volume 2; Final Report; The American Nuclear Society and The Institute of Electrical and Electronics Engineers.

14. NUREG/CR-4229; 1985; Evaluation of Current Methodology Employed in Probabilistic Risk Assessment (PRA) of Fire Events at Nuclear Power Plants; Brookhaven National Lab; New York.

15. NUREG/CR-4550 SAND86-2084; Volume 1, Rev 1; 1990; Analysis of Core Damage Frequency: Internal Events Methodology; Sandia National Laboratories; Albuquerque NM 87185.

16. NUREG/CR-4840 SAND88-3102; 1990; Procedures for the External Event Core Damage Frequency, Analyses for NUREG-1150; Sandia National Laboratories; Albuquerque NM 87185.

17. NUREG/CR-5497; 1998; Common Cause Failure Parameter Estimations.

18. NUREG-75/014 WASH-1400; 1975; Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants; United States Nuclear Regulatory Commission.

19. PBMR; 2001; Safety Analysis Report-Section 3; Doc.No.:001929-207sec3. (Proprietary).

20. NFPA 805; 2001 Edition; Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.

21. Risk Spectrum; 1998; Professional Risk and Reliability Software; RELCON AB; Wex Tech Systems, Inc. and Microsoft Word 7.0.

22. Safety Series No.35-G1; 1994; Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report; International Atomic Energy Agency; Vienna.

23. Safety Series No.50-SG-D2 (Rev.1); 1992; Fire Protection in Nuclear Power Plants, A Safety Guide; International Atomic Energy Agency; Vienna.

24. Schneider U; 1991; Nuclear Engineering and Design 125(1991) 289-295, Introduction to Fire Safety in Nuclear Power Plants; Elsevier Science Publishers B.V; North-Holland.

25. The American Nuclear Society, and The Institute of Electrical and Electronics Engineers; 1983; A guide to the Performance of probabilistic Risk Assessments for Nuclear Power Plants, PRA Procedures Guide; Final Report; Volume 2; New York, NY 10017.