A STUDY ON THE RELIABILITY OF CLASS III POWER SUPPLY SYSTEM OF A SODIUM COOLED FAST BREEDER REACTOR UNDER INTERNAL AND EXTERNAL EVENTS

by

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DECLARATION

I, hereby declare that the investigation presented in the thesis has been carried out by me. The work is original and has not been submitted earlier as a whole or in part for a degree / diploma at this or any other Institution / University.

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This work is dedicated to my kids

'Veenita and Keshav'

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ABSTRACT

The Class III power supply system of Prototype Fast Breeder Reactor (PFBR) has been taken as test case and internal and external events probabilistic safety assessment has been performed. Impact of support system on reliability models has been incorporated. Results of importance analysis and sensitivity study are used to identify significant contributors to unavailability. DG uncertainty analysis has been carried out through Monte Carlo simulations. Failure frequency contribution of Class III power supply due to internal events is 1.73E-8/ry. Seismic Probabilistic Safety Analysis model has been developed for assessment of the seismic fragility of various safety systems, structures and components and integration of seismic hazard with fragility information through appropriate logic models. Total frequency for class III power supply failure due to seismic events is 1.36E-06/ry. The failure frequency estimation of class III power supply system of PFBR due to external flood. The hazard analysis is performed for storm surge, rainfall and tsunami. Total frequency for class III power supply failure due to flooding events is 2.22 E-09/ry. In addition, wind events, aircraft crash hazard assessment, lightning, and missile protection has been included in modeling of safety system. None of these events are contributing for failure of DG. Finally, the total failure frequency of class III power supply system under internal and external events is summation of above mentioned failure frequency, which is 1.38E-06/ry. The major outcome of the thesis is the development of a methodology to perform risk assessment of a safety system for fast reactors under internal and external events. This study helped to develop hazard curves for Kalpakkam site for applications to present and future fast breeder reactors. Another important outcome of this study is the characterization of rainfall intensity variation across India, which will be useful in site selection and elevation selection for finished floor level for upcoming nuclear power plants.

CONTENTS

ABBREVIATION

CB	Circuit Breaker
BARC	Bhabha Atomic Research Centre
CCCG	Common Cause Component Group
CCF	Common Cause Failure
CD	Core Damage
CDF	Core Damage Frequency
DBFL	Design Basis Flood Level
DG	Diesel Generators
IGCAR	Indira Gandhi Centre for Atomic Research
EF	Error Factor
EFPSA	External Hazard Probabilistic Safety Assessment
EL	Elevation
EMTR	Emergency Mean Transfer Relay
ET	Event Tree
FBTR	Fast Breeder Test Reactor
FTA	Fault Tree Analysis
HEP	Human Error Probability
IE	Initiating Event
LT	Low Tension
MRI	Mean Recurrence Interval
MSL	Mean Sea Level
MTTR	Mean Time to Repair

NICB	Nuclear Island Connected Building
PCC	Power Control Centres
PFD	Probability of Failure on Demand
PFBR	Prototype Fast Breeder Reactor
PHA	Probabilistic Hazard Analysis
PGA	Peak Ground Acceleration
PSA	Probabilistic Safety Assessment
RCB	Reactor Containment Building
RP	Return Period
SBO	Station Blackout
SPSA	Seismic Probabilistic Safety Analysis
SSC	Systems, Structures and Components
SSWS	Safety Related Service Water System

SYNOPSIS

Probabilistic safety assessment (PSA) is increasingly important in the safe design and operation of nuclear power plants [1]. One of the objectives during the design of nuclear power plants is to minimize the risk to public and environment due to their operation. The term 'risk' implies accident consequences both in terms of the magnitude of the possible harm and its likelihood. There have been three major reactor accidents in the history of civil nuclear power – Three Mile Island [2], Chernobyl [3] and Fukushima [4, 5]. One was contained without any harm to anyone, the second involved an intense fire without provision for containment, and the third severely tested the containment, allowing some release of radioactivity. These are the only major accidents occurred in over 16,000 cumulative reactor-years of commercial nuclear power operation in 33 countries [6]. The risk of accidents in nuclear power plants is low and declining, but it has significant effect on public morale and government policy. Post Fukushima accident, many reactors have been shut down, approval of several new reactors has been put on hold and Germany has decided to completely switch over to conventional power [7]. Therefore it is imperative to do safety studies of reactors. It is important to identify weakness in reactors safety systems and mitigate consequences of it. The importance of identifying weaknesses in the reactor safety systems and measures for mitigating their consequences, if they fail to function, has become part of safety assessment. This type of analysis has become mandatory to obtain regulatory clearance of reactor operation. Further, improved safety guidelines are being formulated for future reactors with more emphasis on external events based on the experience of Fukushima accident. Thus, a comprehensive safety assessment taking into account all possible internal and external events is necessary to address it.

The first comprehensive application of methods and techniques of probabilistic safety assessment (PSA) dates back to 1975 to the United States Nuclear Regulatory Commission's Reactor Safety Study [8]. Since that land mark study, there has been substantial methodological development, and PSA techniques have become a standard tool in safety evaluation of nuclear power plants. The main benefit of PSA is to provide insights in to plant design, performance and environmental impacts, including the identification of dominant risk contributors and the comparison of options for reducing risk.

Level-1 PSA assessment is estimation of core damage categories with associated core damage frequencies. This level is further partitioned into internal and external event categories [1]. Internal events include random component and system failures. The external events include fire, flood, seismic, storm surge and tsunami. The last two are applicable at only coastal sites. Post Fukushima accident; there is added emphasis on accidents involving external events (particularly flood) [9].

In present analysis a safety system (Class III Power supply system) of Prototype fast breeder reactor has been taken as test case and internal and external events PSA has been performed. It gives comprehensive risk estimation for a safety system under internal and external hazards. Limited results of internal hazard [10, 11, 12] and external hazard [13, 14] assessments were reported for thermal reactors, but no study on external hazard has been reported for fast reactors. It is also to be noted that the failure of support systems and its impact on overall reliability of systems are not considered in the reported results [15, 16].

The works reported in this thesis are based on an internal and external PSA analysis which is carried out first time for a Class-III safety system of a medium sized, pool-type, sodium cooled Prototype Fast Breeder Reactor (PFBR). In order to obviate concerns about "station blackout" risk [17, 18], a comprehensive reliability analysis of a Diesel Generator of PFBR, including system modeling, fault-tree analysis and common cause failure has been performed to ensure its reliability, as a part of internal PSA analysis. Though several reliability analysis of DG exists, the present study is unique for a sodium cooled pool-type fast breeder reactor, as it considers failure of various sub-systems to estimate the overall system unavailability, viz. safety related service water system, fuel oil system and circuit breaker control power supply. In addition, importance analysis and sensitivity studies are made to identity significant contributor to unavailability. Further, uncertainty analysis is carried out through Monte Carlo simulations to determine confidence bound of unavailability. The estimated unavailability is found to be 4.75E-3 for 2/4 (DG success) and 1.47E-3 for 1/4 (DG success). Statistical analysis indicates that the DG unavailability is uncertain by Error Factor 4.4 (90% confidence bound) for 2 out of 4 DG system (system success) and by Error Factor 4.1 (90% confidence bound) for 1 out of 4 DG system (system success). Common cause failures contribute significantly to the unavailability of the system. DG fails to run, DG fails to run due to CCF and DG maintenance out of service is identified as dominant and important contributors of DG unavailability. Failure of one DG during mandatory testing (0.5/year) has been taken as initiating event frequency for computation of failure frequency. Failure frequency contribution of Class III power supply due to internal events is 1.73E-8/ry.

The main focus of this thesis is on the analysis of external events, which are events originating from outside the plant, but with the potential to create a PSA initiating event at the plant [19]. The events considered are seismic, flood (storm, rainfall and tsunami) and wind. External events can occur as single events or as combinations of two or more external events. Analysis covers procedure for identification, categorization, screening, quantification and PSA modeling. Methodology exists for performing PSA [19, 20] for external hazards. However, following tasks are essential to perform external hazard analysis: i) External hazards applicable to reactor site need to be identified. ii) External events data applicable for plant sites need to be collected. iii) Hazard curves applicable to reactor sites need to be developed. iv) Plant walkdown for computation of component fragility needs to be performed. v) System models need to be developed through appropriate logic models. vi) Any other events which may potentially affect system need to be identified.

In this analysis hazard curve for Kalpakkam site (storm surge, rainfall, and wind) has been developed based on data obtained from annual maximum value and asymptotic extreme value analysis has been performed (e.g. 104 (1901-2004AD) years of rainfall data obtained from IMD, Pune, Hourly tide gauge data from Chennai for a period from 1974-1988 for storm surge and wind hazard data using 110 years (1891-2000) of observed cyclonic data, covering IMD stations around Kalpakkam). Seismic Probabilistic Safety Analysis model has been developed for assessment of the seismic fragility of various safety systems, structures and components and integration of seismic hazard with fragility information through appropriate logic models. The analysis indicates that the significant contribution to failure frequency from seismic hazard input is from 0.1 g to 0.25 g and above. The hazard region above 0.2 has large uncertainty. Total failure frequency for class III power supply due to seismic events is 1.36E-06. A new method to perform stationary analysis based on exponent variation and L-moments has been developed to develop rainfall hazard curve. Variability characterization of rainfall intensity over different nuclear power plants sites has been performed. Class III power supply failure frequency has been calculated at different elevation (30-40 EL). Total frequency for class III power supply failure due to flooding events is 2.22 E-09. Wind hazard, aircraft crash, lighting and internal missiles are not contributing to failure of DG. Finally, the total failure frequency of class III power supply system under internal and external events is 1.38E-06/ry.

This thesis work has helped to develop hazard curve for Kalpakkam site and a calculation methodology for internal and external PSA evaluation of future nuclear power plants. Another important outcome of this study is the characterization rainfall intensity variation in India. This will be useful in site selection and elevation selection for finished floor level for upcoming nuclear power plant. This thesis work has produced four research papers and two conference papers. One describes about failure frequency computation due to internal events, one about DG power supply system reliability of PFBR, and two papers about development of rainfall hazard curve applicable to Kalpakkam site.

The thesis has been partitioned into following six chapters.

Chapter 1 describes literature survey, probabilistic safety assessment levels, objective, scope and safety targets of Nuclear Power Plants. Internal and external event probabilistic safety assessment (PSA) has been described in this chapter. It also describes limitations of probabilistic safety assessment.

Chapter 2 describes Class III power supply system description, system boundary, system functions and support systems. It also describes support systems needed to perform reliability analysis.

Chapter 3 describes general methodology used in Level-1 probabilistic safety assessment. Accident sequence analysis, safety functions and success criteria have been described in this chapter.

Chapter 4 describes system modeling under internal events. It presents detailed methodology to perform system analysis using Fault tree method, where the top event of the fault tree is taken as the system failure state(s) identified by the event tree analysis. The Fault tree has been evaluated using the ISOGRAPH software [21]. Common cause failure methodology has been described. It also includes methods to perform Importance analysis, sensitivity analysis and uncertainty analysis using Monte Carlo simulation.

Chapter 5 describes system modeling under External events. External events are defined as events originating from outside the plant, but with the potential to create a PSA initiating event at the plant [19]. External events analysis covers procedure for identification, categorization, screening analysis, quantification, and PSA modeling of External events. This analysis covers Hazard analysis, Plant-system and structure response analysis, evaluation of the fragility and vulnerability of components, Plant-system and sequence analysis and consequence analysis. External hazards have been further divided into three parts. First part deals with seismic events. Second part is flooding events. It includes storm surge, rainfall and tsunami. All other events have been described in third part. The other events include Wind hazard, aircraft crash, lighting and missile protection. This chapter computes failure frequency of class III power supply system arising from external Event, which is summation of failure frequency due to seismic events, flooding events and other events. This chapter develops methodology to perform external hazard. Hazard curve for Kalpakkam Site (storm surge, rainfall, and wind) has been developed. Methodology to arrive at appropriate hazard curve (storm surge, rainfall, and wind) has been described in this chapter. Performing fragility and estimating failure frequency by fault tree and event tree models has been described in this chapter.

Chapter 6 describes summary, conclusion and future directions arising from the thesis.

To conclude, in this thesis work, a comprehensive risk assessment (internal and external events) of a safety system (Class III power supply) of a pool type, sodium cooled fast breeder reactor is carried out for the first time. Under internal events, impact of support system on Class III reliability, importance measures, sensitivity analysis, and confidence bound using Monte

Carlo simulations are systematically studied and the results are quantified. To perform external PSA analysis, hazard curve applicable for Kalpakkam site (storm surge, rainfall, tsunami and wind) is developed for the first time. A new method to perform stationary analysis of rainfall based on exponent variation and L-moments is also developed. In addition, rainfall intensity variability throughout India has been characterized. The new methodology developed, in particular for external hazard analysis, will be useful for Level-1 PSA analysis of safety systems for the existing and the upcoming nuclear power plants.

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Chapter 1
Introduction

1.1 Background

Probabilistic Safety Assessment (PSA) is an established technique for numerically quantifying the risk in Nuclear Power Plants (NPPs) [1]. In PSA, all safety related systems and components are modeled in terms of their reliability and are logically linked together to determine likelihood of core melt accident. It is a logical and deductive technique, in which an undesired top event is specified which might lead to an undesired event (e.g. core damage) and it is usually modeled with the help of Fault Trees (FT) and Event Trees (ET). The importance of identifying weaknesses in the reactor safety systems and measures for mitigating their consequences, if they fail to function, has become a part of safety assessment. One of the objectives during the design of a NPP is to minimize its risk to public and environment due to its operation. The term 'risk' implies accident consequence in terms of its magnitude of possible harm and its likelihood. Three major reactor accidents have occurred in the history of civil nuclear power – Three Mile Island [2], Chernobyl [3] and Fukushima [4, 5]. One was contained without any harm to anyone, the second involved an intense fire without provision for containment, and the third severely tested the containment, allowing some release of radioactivity. These are the only major accidents during 16,000 cumulative reactor-years of commercial nuclear power operation in 33 countries [6]. The risk of accidents in nuclear power plants is very low and it is declining due to stringent safety criteria followed. After Fukushima accident, many reactors have been shut down, approval of several new reactors has been put on hold. Germany has decided to completely switch over to conventional power [7]. Based on the lessons learnt from Fukushima accident, revised safety criteria has been formulated, which demands a comprehensive internal and external events PSA study to ensure plant safety [8].

1.2 Nuclear Safety

Nuclear power plants are complex engineering facility. Designers comply with a number of stringent regulations aimed at limiting the risks inherent in these installation, which are primarily the possible release of radioactivity. These regulations are applied from the design, construction stages, operating phases and final decommissioning. Nuclear safety embodies the principal concern of all those involved with the plant, from construction engineers to operators or regulators. Pursuing these objectives enable operators to achieve the overall goal of nuclear safety, namely to protect man and environment by limiting the release, under any circumstances, of the radioactive materials that the facility contains. It has following three objectives [9]:

- Ensure that nuclear facilities operate normally and without an excessive risk to operating staff and environment being exposed to radiation from the radioactive materials contained in the facility;
- Prevent incidents and;
- Limit the consequences of any incidents that might occur.

The important safety functions that are essential to be performed for ensuring safety of a NPP are a) control of core reactivity, b) removal of heat from the core and c) confinement of radioactive material and control of operational discharges as well as limitation of accident releases. To achieve this ‘defense in depth’ approach is followed in designing and operating of nuclear facilities which prevents and mitigates accident that releases radiations and hazardous material. This approach of creating multiple, independent and redundant layers of defense is followed to compensate for the potential human and mechanical failures, so that no single layer, no matter how robust, is exclusively relied upon. Lists of initiating event (design basis accidents) are identified during the design stages and are grouped in different categories based on their frequency of occurrence as shown in Table1.1 [9].

Table 1.1: Acceptance Criteria in Nuclear Safety

Occurrence (1/ry)	Characteristics	Plant state	Terminology	Acceptance criteria
10^{-2} - 1 (expected over the lifetime of the plant)	Expected	Anticipated operational occurrences	Anticipated transients, transients, frequent faults, incidents of moderate frequency, upset conditions, abnormal conditions	No additional fuel damage
10^{-2}-10^{-4} (chance greater than 1% over the lifetime of the plant)	Possible	Design basis accidents	Infrequent incidents, infrequent faults, limiting faults, emergency conditions	No radiological impact at all, or no radiological impact outside the exclusion area
10^{-4}-10^{-6} (chance less than 1% over the lifetime of the plant)	Unlikely	Beyond design basis accidents	Faulted conditions	Radiological consequences outside the exclusion area within limits
$>10^{-6}$ (very unlikely to happen)	Remote	Severe accidents	Faulted conditions	Emergency response needed

1.3 Probabilistic and Deterministic Safety

Both deterministic methods and probabilistic methods are required to be applied for safety studies [10]. The first comprehensive application of methods and techniques of PSA was carried out in 1975 for two NPPs, (PWR and BWR) by the United States Nuclear Regulatory Commission's Reactor Safety Study [11]. Since that landmark study, there has been substantial methodological development, and PSA techniques have become a standard tool in safety evaluation of nuclear power plants. The main benefit of PSA is to provide insights in to plant design, performance and environmental impacts, including the identification of dominant risk contributors and the comparison of options for reducing risk. The steps involved are a) accident sequence initiating event analysis, b) accident sequence analysis, c) definition of core damage, d) identification of safety functions, safety systems and success criteria, e) modeling of accident sequences (ET), f) identification of end points of accident sequences and plant damage states, g) system analysis (FT). Fault tree and Event tree uses Boolean logic. One of the approaches follow small Event tree and large Fault tree. Event tree analysis is based on binary logic in which events have occurred or not occurred. An event tree begins with an initiating event, such as component failure, reactor transients etc. The consequences of the events are followed by possible paths and each path is assigned a probability of occurrence and the probability of the various possible outcomes can be calculated. Fault tree analysis is top-down approach, which begins with a general conclusion (i.e. fault definition), then attempts to determine the specific causes of the conclusion by constructing logic diagram. There are specialized software like ISOGRAPH [12], PSAPACK [13], Risk spectrum [14] etc. available to compute fault tree and event tree. PSA can be used for a) back fitting decisions, b) identification of design and operational weaknesses, c) providing information usable in the independent process of resolving regulatory issues, d) evaluation of significant occurrences, e) reliability assurance, f) future safety goal integration and possible implementation, g) establishment of priorities for research activities and h) operator training. Probabilistic safety assessment (PSA) models developed for nuclear power plants provide valuable information and insight that can make important contributions to the process of evaluating safety issues of regulatory significance [15].

Deterministic safety analyses for a nuclear power plant predict the response to postulated initiating events [9]. Major steps are a) identification and characterization of events, b) analysis of enveloping events, c) evaluation of radiological consequence and d) verification with respect to acceptance criteria. PSA differs from traditional deterministic safety analysis in that it provides a methodological approach to identifying accident sequences that can follow from a broad range of initiating events and it includes systematic and realistic determination of accident frequencies and consequences. A major advantage of PSA is that it allows for the quantification of uncertainties in safety assessments together with the quantification of expert opinion and/or judgment for revising design.

Though all preventive and mitigate measures are considered in the design, the plant still has some residual risk for the outside world. The PSA approach allows making a better evaluation of the major contributors to the residual risk. Nowadays nuclear regulators demand detailed PSA to be performed for safety clearance requirement.

1.4 Probabilistic Safety Assessment: Levels, Objective, Scope and Safety Targets

The safety assessment considers the probability, progression and consequences of equipment failures or transients conditions to derive numerical estimate that provide a consistent measure of the safety of the plant as follows [1]:

Level-1: The assessment of plant failures leading to the determination of core damage frequency (CDF).

Level-2: The assessment of containment response, together with Level-1 results, to the determination of containment release frequencies.

Level-3: Estimation of risk to public (consequence and associated frequencies)

The assessments in each of these levels are further partitioned into internal and external event categories. Internal events include random component and system failures. External events are defined as events originating from outside the plant, but with the potential to create a PSA initiating event at the plant [16]. They are fire, flood, seismic, storm surge and tsunami. The last two are applicable at only coastal sites. The full scope PSA considers all the events mentioned above as illustrated in Fig. 1.1.

The scope of PSA depends on the application as illustrated in Fig. 1.2. It is used for conceptual design, design of safety systems and Level-1, Level-2, Level-3 full scope PSA.

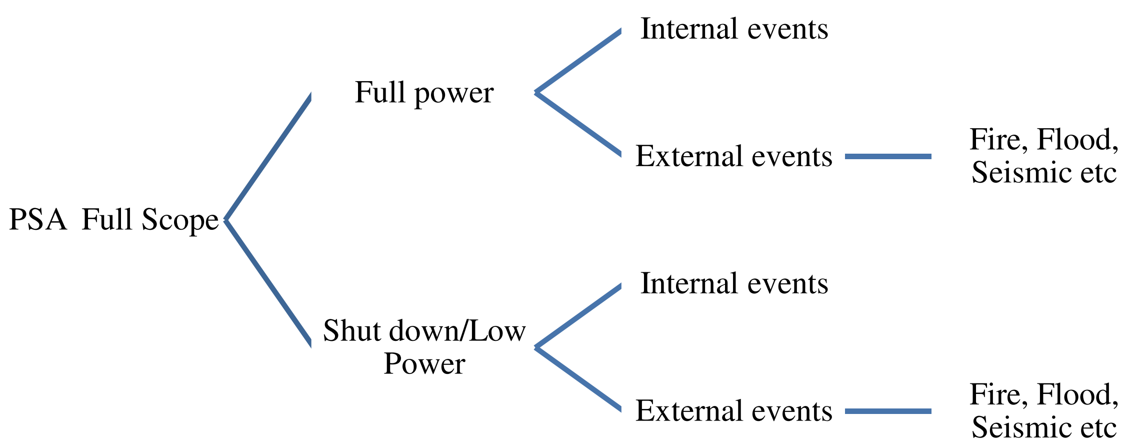

Fig.1.1: Probabilistic Safety Analysis Scope

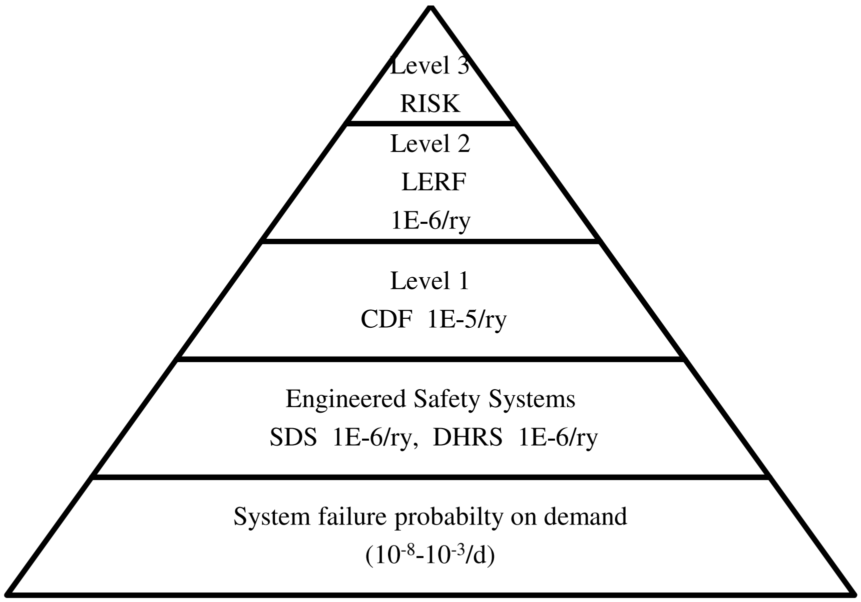

Fig. 1.2: Task Pyramid for Safety Studies

The core damage frequency is sum of all accident sequence frequencies as,

$$\text{CDF} = \sum_j \lambda_j^{IE} P_j \ , \tag{1-1}$$

where λ_j^{IE} are the initiating event frequencies (equivalently demands), P_j are the safety system failure probabilities. In general, Pj may involve product of several terms. When the results are dominated by one type of safety systems, this expression can be written in terms of the safety system failure frequencies (λ_k) as,

$$\text{CDF} = \sum_k \lambda_k \, , \tag{1-2}$$

where each of the λ_k (safety system failure frequency) has the form,

$$\lambda_k = \sum_i d_i P_i \, , \tag{1-3}$$

here, d_i is the number of demands on the system of type i, and P_i is the corresponding probability of failure on demand.

From the above discussion, it is evident that it is natural to specify system level targets in terms of failure frequencies. This is because (i) it is derivable from CDF targets (as discussed above) and ii) when expressed in frequency unit, the final number can include contribution from a range of demands on the system with appropriate mission times.

The system level failure targets are sometimes referred to as probability/year. However, system level targets when specified in terms of unavailability or probability of failure on demand (PFD), it can be used as a very approximate measure for system design. This is because when unavailability is specified, it is ambiguous with respect to the mission time and demand type. It may be noted that while no unique unavailability or PFD can be associated with a safety system, it can be ascribed a failure frequency considering the type and number of demands expected on the system.

Basis for Core Damage frequency (CDF)

Core damage is defined as local fuel temperature rises above 1204 ºC for LWR [17]. The frequency limits regarding core damage vary between 10^{-4} and 10^{-6} per year.

Basis for Large Early Release Frequency (LERF)

The release for which a numerical criterion is given is also defined in several different ways [18, 19]. The frequency limits regarding LERF vary between 10^{-5} and 10^{-7} per reactor year (/ry).

Here safety targets for CDF and LERF are presented. The safety objective of Atomic Energy Regulatory Board (AERB), India [20] and International Atomic Energy Association (IAEA) [21] are presented. There is minute variation in safety objectives of AERB and IAEA as they present for LWR, PHWR and new and existing plants respectively. AERB specifies the safety targets such as CDF and LERF in terms of plant operating years (/ry).

Radiological Safety Objective

To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is as low as reasonably achievable, economic and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents [21].

Technical Safety Objective

To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor, and to ensure that the likelihood of severe accidents with serious radio-logical consequences is extremely small.

The CDF and LERF targets of AERB and IAEA are presented in Table 1.2 and Table 1.3 respectively.

Table: 1.2: CDF and LERF Safety Targets of AERB

	LWR	PHWR
CDF	< 1E-6/ry (I); < 1E-5/ry (I+E)	< 1E-5/ry (I+E)
LERF	< 1E-7/ry	< 1E-6/ry

'I' stands for internal events and 'E' for External events.

Table: 1.3: CDF and LERF Safety Targets of IAEA

	For new plants	For existing plants
CDF	< 1E-5/ry (I+E)	< 1E-4/ry (I+E)
LERF	< 1E-6/ry	< 1E-5/ry

The safety criteria for different countries are shown in Fig. 1.3.There are 668 reactors of various types that has been constructed or under construction [22, 23, and 24]. Out of these many reactors 158 have been shut down permanently, two are long shutdown, 445 reactors are under operation and 63 reactors under construction. Based on the operating experience, the postulated CDF for an individual plant is taken as 1E-5/ry for LWR and 1E-4/ry for FBR.

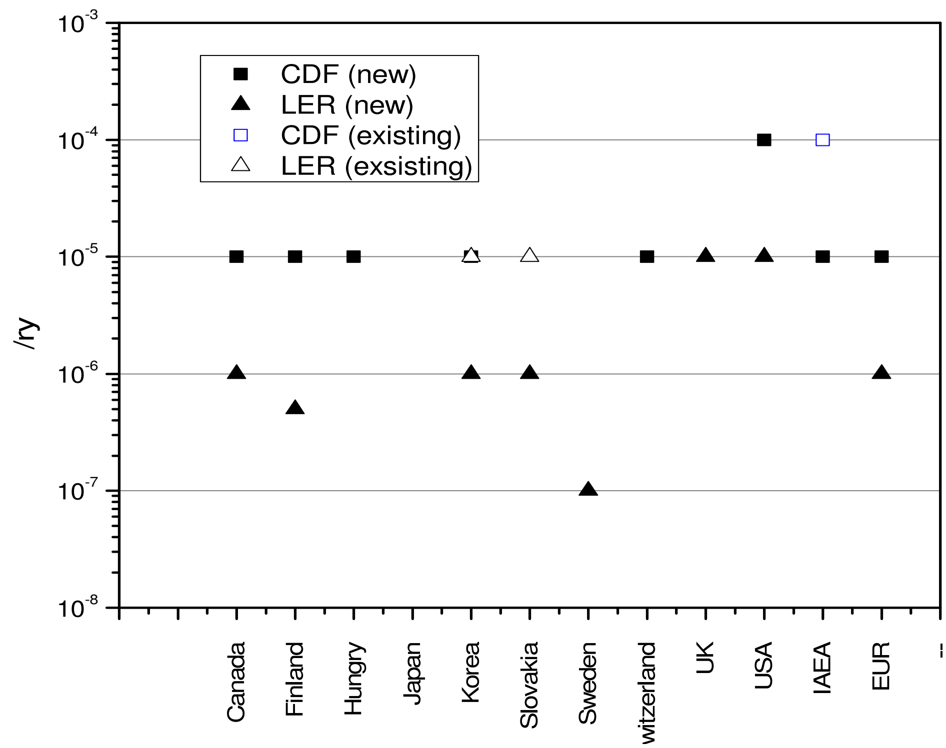

Fig. 1.3: Criteria for CDF and LERF for Different Countries

1.5 Limitations of PSA

The results of a PSA study invariably contain uncertainties arising from following three main sources [25]:

- *Completeness uncertainty:* It is impossible to demonstrate the exhaustiveness of a PSA, even when the scope of the analysis has been extended to as large a number of situations as possible in terms of various reactor operating states and potential initiating events.
- *Data uncertainty:* These uncertainties concern the reliability data for plant components, the frequency of initiating events, common-mode failures and failures resulting from

human actions. The main uncertainties are those relating to the frequency of rare initiating events.

- *Modeling uncertainty:* These uncertainties arises from those models which cannot easily be quantified, such as the resistance of certain components under accident conditions, poorly understood physical phenomena, human actions etc.

1.6 Literature Survey

The current practices and research performed using PSA with internal and external hazard in reactor safety are discussed here. United States Nuclear Regulatory Commission has published a guide to perform the Probabilistic Safety Assessment (PSA) for Nuclear Power Plants [26]. A total loss of power called ‘‘Station Blackout” (SBO) occurs as a result of complete failure of both offsite and onsite power supply [27, 28]. During loss of offsite power, Diesel Generators (DG) provides onsite electrical power (Class III). Offsite power failure is initialing event for Class III power supply system and reactor needs to be shut down. Sanyasi Rao, V.V.S. et al., (2002) has estimated frequency of offsite power failure for Kalpakkam site [29]. Assessing reliability of DG power supply is an important task for class III safety study. NUREG/CR-2989 (1983) and NUREG-CR-5500 (1996) has estimated reliability of Emergency Diesel Generator in service of various Nuclear Power Plants [30, 31]. NUREG-CR-5994 (1994) reports the unavailability of DG used in the United States in the range of 3.94E-3 to 1.77E-2 [32]. Harry F. Martz et al. (1996) has updated Empirical Bayes estimation of the reliability of Nuclear Power Plant Diesel Generators [33]. Andrija Volkanovski et al, (2009) demonstrated the reliability analysis of power supply system using Fault tree method [34]. Class III power supply frequency is a function of loss of offsite power transient. See-Meng Wong (1984) has estimated reliability analysis for the emergency power system of PWR during a loss of offsite power transient [35]. Few DGs during operation undergoes preventive maintenance and its impact on overall safety is an important aspect considered in the design. Maintenance and failure unavailability and their risk impacts on Diesel Generator had been studied and recommendations were given for preventive maintenance [32]. Another important parameter for PSA analysis is Common Cause Failures (CCF). It is predominant contributor of failure of any safety system. This aspect on overall CDF targets has been studied extensively and it is found to be very high [36, 37]. Further, uncertainty quantification is an important aspect for system reliability analysis [25]. It needs to be carried to determine confidence bound of unavailability. It is usually performed through Monte Carlo simulations [38].This method can be used to provide high confidence estimates of individual percentiles, for example an estimate of the z^{th} percentile with a 95% confidence that the true z^{th} percentile is less than the estimate. In addition, importance analysis and sensitivity analysis are needed to identity significant contributor to unavailability. Fussell-Vesely, Birnbaum, Barlow-Proschan and Sequential importance measures are usually evaluated in system reliability. M. Vander Borst et al. (2001) demonstrated procedure for importance analysis [39].

Although safety studies using Level-1 PSA for a safety system is well established, following shortcomings are identified for class III reliability analysis. Limited results of internal hazard [32, 35, and 40] assessments were reported for thermal reactors, but no study has been reported for fast reactors. It is also to be noted that the failure of support systems and its impact on overall reliability of systems are not considered in the reported results [30, 31]. However they are necessary for reliability analysis and need to be incorporated into DG internal boundary [30, 41].

Methodology exists for performing PSA [16, 42] for external hazards. It includes [42] identification of a) types of external events, b) source of events listing, c) identification of events and d) characterization of events. However, some external events identified need to be screened out based on location and layout of the plant.

Studies have been reported for probabilistic seismic assessment [43, 26 and 44]. It includes hazard assessment, seismic fragility, plant response modeling and risk quantification. Seismic hazard at a site is represented by hazard maps representing ground motion parameters such as peak ground acceleration (PGA), and its corresponding non-exceedance frequency. PGA is not a measure of the total energy of an earthquake, but rather, how hard the earth shakes in a given geographic area [44]. To estimate PGA, attenuation relation needs to be used in the analysis [45]. Earthquake occurrence is represented as a random process which follows Poisson distribution. To perform seismic hazard at reactor site, seismic data and faults applicable to site are used. Seismic fragility is the conditional failure of components based on a given earthquake. It has to perform for system and structure response analysis. Most widely used method is Zion method and Seismic Safety Margins Research Program (SSMRP) [46.]. Seismic fragility data is not available through experiments for all components. Some component fragility information is used from literature for the analysis [47]. ASME /ANS-RA-S-2008 have published seismic fragility of component [47]. The plant safety function logic fault tree models need to be developed as appropriate for seismic PSA. Although risk from seismic hazard for some reactor is reported [48], it is site dependent and seismic hazard analysis for a reactor needs to be performed.

Incidents like the storm induced flooding at Le Blayais NPP, France [49], flood events at Fort Calhoun NPP, USA [50] and the tsunami at Fukushima Daiichi, Japan [51] have pointed out the importance of external flooding as an important contributor to NPP risk. Procedure for performing flooding events has been published by International Atomic Energy Agency [52]. Similar methodology document of External Flood Probabilistic Safety Analysis has been published by AERB, India [53]. The study requires identification of hazard applicable for reactor site.

Data analysis of flooding events is usually done by extreme value analysis. Three types of asymptotic distributions viz., Type I (Gumbel), Type II (Frechet) and Type III (Weibull), are used for extreme value analysis [54]. It is also performed by using power law [55] and exponential distribution [56]. In addition, the r-largest annual maxima method [57, 58] is used. This method has been chosen as it is capable to use more than one value, thereby allowing more reliable estimates of return periods with less number of years of data.

Extreme water levels due to cyclonic storms for Kalpakkam coast has been studied [59], and has estimated that the observed storm surge value (2-3m). The analysis for storm surge is generally carried out by two different methods [57, 58]. The first method uses a physical method in which a non-linear hydrodynamic model for storm surge event and the second method is a statistical one which is based on extreme value analysis of observed maximum sea level during a storm surge event. A.K. Ghosh (2008) has performed assessment of earthquake-induced tsunami hazard at a nuclear power plant site in eastern coast of India [60]. He has estimated tsunami level applicable to Kalpakkam coast (4.5m).

Several studies have been performed for estimation of rainfall flood hazard in India on national scale [61, 62] as well as regional/cities scale [63-66]. All these studies have taken broad area (e.g. north India, east India or whole India) for analysis. But, rainfall at reactor site is necessary for realistic estimation of flood hazard. It is increasingly being recognized that a long-term change in rainfall at the regional scale can significantly affect magnitude and frequency of flood. Several studies based on hydrological data from around the world have now provided evidence of rainfall-related changes in flood activity [67-70]. The probability of detecting shifts or changes in rainfall on the decadal or century scale is greater from longer records. Several studies to investigate stationary of rainfall have been performed over India [71-74]. Detecting shifts or changes in rainfall on the decadal or century scale is an important objective to perform stationary analysis. Possible violation of stationary in climate, increases concern among designer about the currently used design estimates for civil infrastructure projects [75]. Therefore, stationary analysis taking those regions where NPPs are present is necessary for realistic hazard estimation.

In addition, other external events (like wind, lightening, aircraft crash) which have potential to affect reactor safety are also important for safety assessment. Thus, it is important to perform a comprehensive risk assessment, including the internal and external hazard, by taking into account all above discussed aspects, to estimate reliability of a safety system.

1.7 Research Objectives

It is found from the literature survey that the methodology to perform internal and external event PSA is well established for thermal reactors and many studies have been reported. But, very few studies are reported for fast reactors. The methodology to perform external PSA, in particular flood PSA, is still in the developing stage. Further, lack of reliable site-specific external events data and the corresponding hazard curves are other bottlenecks to perform such a study. Hence, a highly challenging problem of comprehensive risk assessment for a safety system in a medium sized fast reactor due to internal and external events is taken for this thesis work. The safety system chosen is the Class-III power supply system of a 500MWe, sodium cooled, pool type, mixed oxide Prototype Fast Breeder Reactor (PFBR) [76], which is in the advanced stage of commissioning at Kalpakkam, India. Important works and challenges identified are:

i) Internal events analysis has to be performed by defining a system boundary covering all the important support systems.
ii) It is essential to develop hazard curves for external events (seismic, storm surge, rainfall, tsunami and wind) applicable to Kalpakkam site.
iii) A methodology for external events PSA analysis for PFBR has to be developed.
iv) The contribution from all external events has to be included for estimating the overall failure frequency of a Class III power supply system.
v) Finally, perform a comprehensive internal and external events Level-I PSA of PFBR due to failure of Class III power supply by using the indigenously developed methodology and the Kalpakkam site specific data.

1.8 Organization of the Thesis

All mentioned research objectives have been studied and results are reported in the following six chapters:

Chapter 1 describes probabilistic safety assessment levels, objective, scope and safety targets of nuclear power plants. Internal and external event probabilistic safety assessment (PSA) has been described in this chapter. It also describes limitations of probabilistic safety assessment.

A detailed literature survey on internal and external PSA has been presented and research objectives have been identified.

Chapter 2 describes class III power supply system description, system boundary, system functions and support systems. It also describes support systems needed to perform reliability analysis.

Chapter 3 describes general methodology used in Level-1 probabilistic safety assessment. Accident sequence analysis, safety functions and success criteria have been described. Internal and external methodology has been described. External events methodology to perform seismic and flooding events has also been described in this chapter.

Chapter 4 describes system modeling under internal events. It presents detailed methodology to perform system analysis using Fault tree method, where the top event of the fault tree is taken as the system failure state(s) identified by the event tree analysis. The Fault tree has been evaluated using the ISOGRAPH software [12]. Common cause failure methodology has been described in this chapter. It also includes methods to perform importance analysis, sensitivity analysis and uncertainty analysis using Monte Carlo simulation. Results of internal events are disused in this chapter.

Chapter 5 describes system modeling under external events. External events analysis covers procedure for identification, categorization, screening analysis, quantification, and PSA modeling of external events. This analysis covers hazard analysis, plant-system and structure response analysis, evaluation of the fragility and vulnerability of components, plant-system and sequence analysis and consequence analysis. External hazards have been further divided into three parts. First part deals with seismic events. Second part is flooding events. It includes storm surge, rainfall and tsunami. All other events have been described in third part. The other events include wind hazard, aircraft crash, lighting and missile protection. This chapter computes failure frequency of class III power supply system arising from external events, which is summation of failure frequency due to seismic events, flooding events and other events. This chapter develops methodology to perform external hazard. Hazard curve for Kalpakkam Site (storm surge, rainfall, and wind) has been developed. Methodology to arrive at appropriate hazard curve (storm surge, rainfall, and wind) has been described in this chapter. Performing fragility and estimating failure frequency by fault tree and event tree models has been described in this chapter. . Results of external events are disused in this chapter.

Chapter 6 describes summary, conclusion and future directions arising from the thesis.

Chapter 2
Class III Power Supply System Description

2.1 PFBR Description

The PFBR is a 500 MWe, sodium cooled, pool type, mixed oxide (MOX) fuelled reactor having two secondary loops [77]. The reactor is located at Kalpakkam, close to the 2 × 220 MWe PHWR units of the Madras Atomic Power Station (MAPS). Kalpakkam is situated at 68 km south of Chennai on the coast of Bay of Bengal. The primary objective of the PFBR is to demonstrate techno-economic viability of fast breeder reactors on an industrial scale. The reactor power is chosen to enable adoption of a standard turbine as used in fossil power stations, to have a standardized design of reactor components resulting in further reduction of capital cost and construction time in future and compatibility with regional grids. The MOX fuel is selected on account of its proven capability of safe operation to high burnup, ease of fabrication and proven reprocessing. The main vessel is made of highly ductile AISI 316 LN material and it satisfies leak before break criteria. A two loop design has been adopted in view of its economical benefits and it meets the safety requirements.

The overall flow diagram comprising primary circuit housed in reactor assembly, secondary sodium circuit and balance of plant is shown in Fig.2.1 [78]. The nuclear heat generated in the core is removed by circulating sodium from cold pool at 670 K to the hot pool at 820 K. The sodium from hot pool after transporting its heat to four intermediate heat exchangers (IHX) mixes with the cold pool. The circulation of sodium from cold pool to hot pool is maintained by two primary sodium pumps and the flow of sodium through IHX is driven by a level difference (1.5 m of sodium) between the hot and cold pools. The heat from IHX is in turn transported to 8 steam generators (SG) by sodium flowing in the secondary circuit. Steam produced in SG is supplied to turbo-generator.

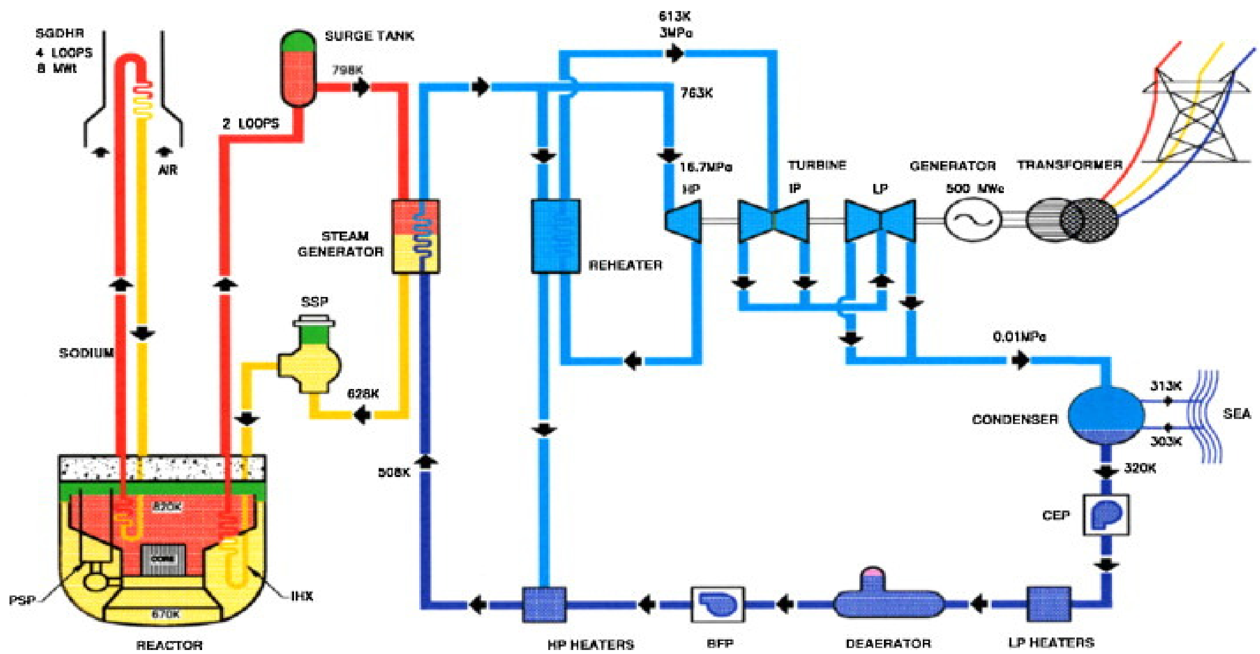

Fig 2.1: Schematics of Heat Transport System of PFBR

The main components that comprise the reactor assembly are main vessel along with thermal baffles, core support structure along with core catcher, grid plate along with primary sodium pipes, core subassemblies, inner vessel, top shield, control plug, absorber drive mechanisms, intermediate heat exchangers (IHX), primary sodium pumps, fuel handling systems and safety vessel. The reactor assembly is supported on the reactor vault, which consists of two portions made of concrete: inner wall supporting the safety vessel and outer wall supporting the top shield, which in turn supports the main vessel and its internals and primary sodium.

The main vessel houses the entire primary sodium circuit including core. The sodium is filled in the main vessel with free surfaces, blanketed by argon. The inner vessel separates the hot and cold sodium pools. The reactor core consists of 1757 subassemblies including 181 fuel subassemblies. The control plug, positioned just above the core, houses mainly 12 absorber rod drive mechanisms, thermocouples and neutron detectors and failed fuel identification modules. The top shield supports the primary sodium pumps, IHX, control plug and fuel handling systems.

2.2 Description of Electrical Power Systems of PFBR

Electrical power systems are the source of power for the reactor coolant pumps and other auxiliaries during normal operation and for protection system and engineered safety features during normal and accident conditions. Both off-site and on-site power systems are provided to cater to the needs of the station. For meeting the long term needs of power, class IV (Grid) and class III (DG) power supplies are provided. Class II (UPS) and class I (DC power) are derived from class III power sources with batteries providing the back up when needed. The batteries are designed to feed the rated load of their buses (i.e. 50% of the station loads) for a period of 30 minutes. Following this period these will be catering to the essential loads required to continue safe plant shutdown.

2.3 Class III Power Supply Description

Class III buses receive power from Class IV supply under normal conditions of operation. Diesel Generators are the source of onsite power to Class III buses under loss of Class IV power (FSAR Chapter-9, 2014). The Class III power supply system is provided with two independent divisions located in Electrical Building 1 and Electrical Building 2 respectively. Class III power supply scheme is shown in Fig. 2.2 [79]. Each division is having its own 6.6 kV bus arranged in two sections. The two sections of the division -1 receive the normal power supply (Class-IV) feeder from the unitbus-1 and station bus-1 respectively. Similarly the two sections of the division -2 receive the power supply (Class-IV) from the unit bus-2 and station bus-2. Each section is having an incomer from an independent DG. The two DG sets of a division are located in two different DG buildings. The Class III 6.6 kV busses are normally supplied from the unit busses and station buses with the bus couplers of Class III 6.6 kV buses open. When a feeder from the unit bus trips due to cable fault (feeder fault), the feeder from the station bus will feed the Class III 6.6 kV bus by auto closing the bus coupler breaker. The change over time is about 500ms. The bus change over is a normal operation in any power plant. The bus changeover is carried out through numerical check synchronizing relay. There are 4 DG sets and one each is connected to one section of Class III power supply bus at 6.6 kV level. The DGs are not designed for operation in parallel in order to have redundancy in the supply capacity [80]. Each DG is capable of taking 50% (2.25 MVA) of its capacity as first step load and subsequently DG is capable of taking 25% (1.125 MVA) load steps after every 4 sec. When, one out of two primary sodium pumps is not available for operation, the other primary sodium pump main drive motor is to be run at 40 percent of the rated speed for decay heat removal. Each DG is rated to supply emergency loads connected to the respective Class III bus and the power required to run one primary sodium pump (420KVA) at a speed of 40 % of the rated speed.

The 4 DGs are housed in two independent DG building. The two units housed in one DG building are physically segregated from one another by fire barrier wall and each DG is also functionally independent from the other. The two DG building are separated from one another. One DG building is located on the eastern side of the Nuclear Island Connected Building (NICB) and the other is located on the western side of the NICB. The cooling water systems for the two DG buildings are also independent. All the auxiliaries of a DG set are fed from the same section of the 415 V system bus to which the DG set is connected. The main and standby air compressors of all DG sets are fed from Class III. A schematic diagram of Class III power system is shown in Fig. 2.2 [79].

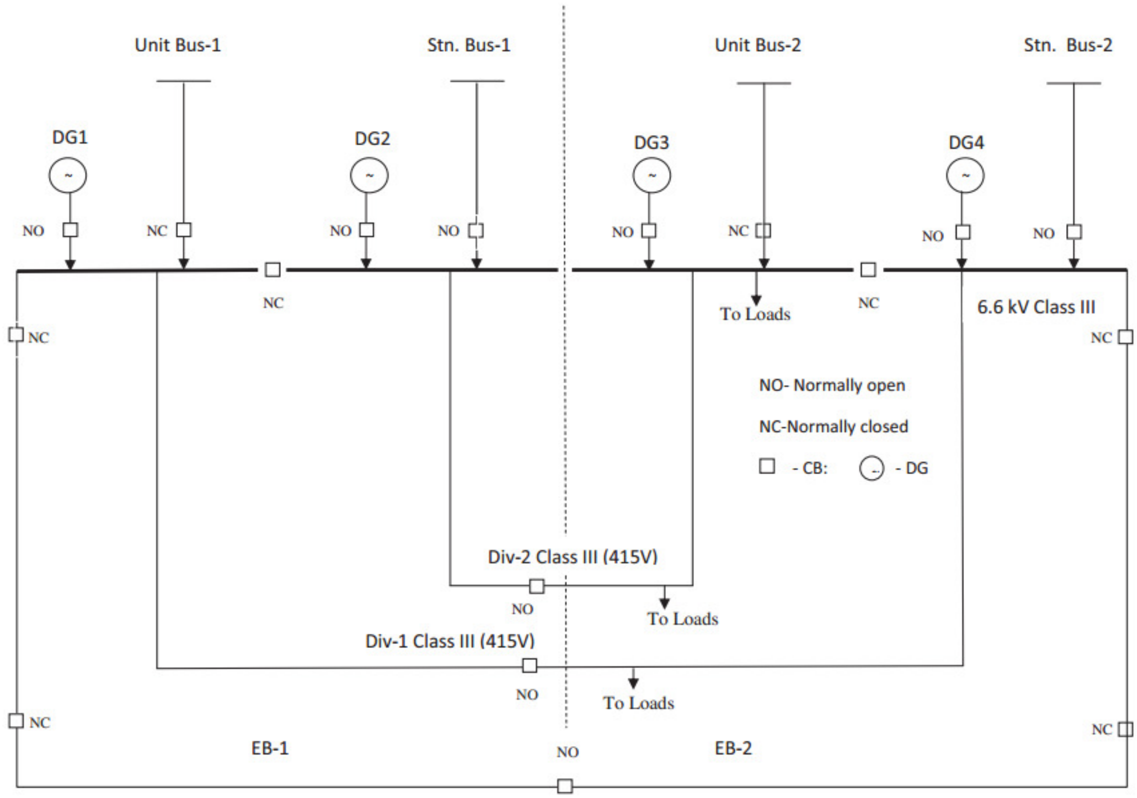

Fig.2.2: Schematic Diagram of Class- III Power Supply System

AT 6.6 kV level, there are two divisions of Class III buses with two bus sections per division with 1 DG connected to each section. Bus sections within a division are connected by inter-sectional bus-couplers (auto). There is an inter-divisional tie line and circuit breakers which can be closed manually to provide power to a bus (when its normal supply is lost) from a bus in the other division. Further details regarding Class IV and Class III power supply systems are available in Reference [79].

2.3.1 System boundary

Diesel Generator system boundary is shown in Fig. 2.3. These boundaries are consistent with the boundaries identified in similar studies [30, 41]. The boundary of the EDG includes combustion air intake and exhaust system, lubricating oil, fuel oil system (including day tank) and the starting compressed air system. DG oil storage capacity planned is adequate to meet the emergency load demand for 7 days either with 4 DGs running or 2 DGs running. In line with the location of DGs, two storage tanks are located on the eastern side of NICB and the other two on the western side. The fuel oil storage tanks and the oil transfer pump for the two DG buildings are independent. Each DG has a day tank of sufficient capacity to run DG for a period of 4 hours at rated capacity with 10 % required margin. A number of auxiliary systems have also been indicated, which includes control power supply (Class I and II), EMTR and Safety related service water system. All auxiliaries of the DG such as fuel oil pump and starting air compressor are also independent [81, 82]. The Fig. 2.4 represents the fuel oil storage and transfer system and Fig. 2.5 represents starting air system. The loads on Safety related service water system are presented in detail in Fig. 2.6.

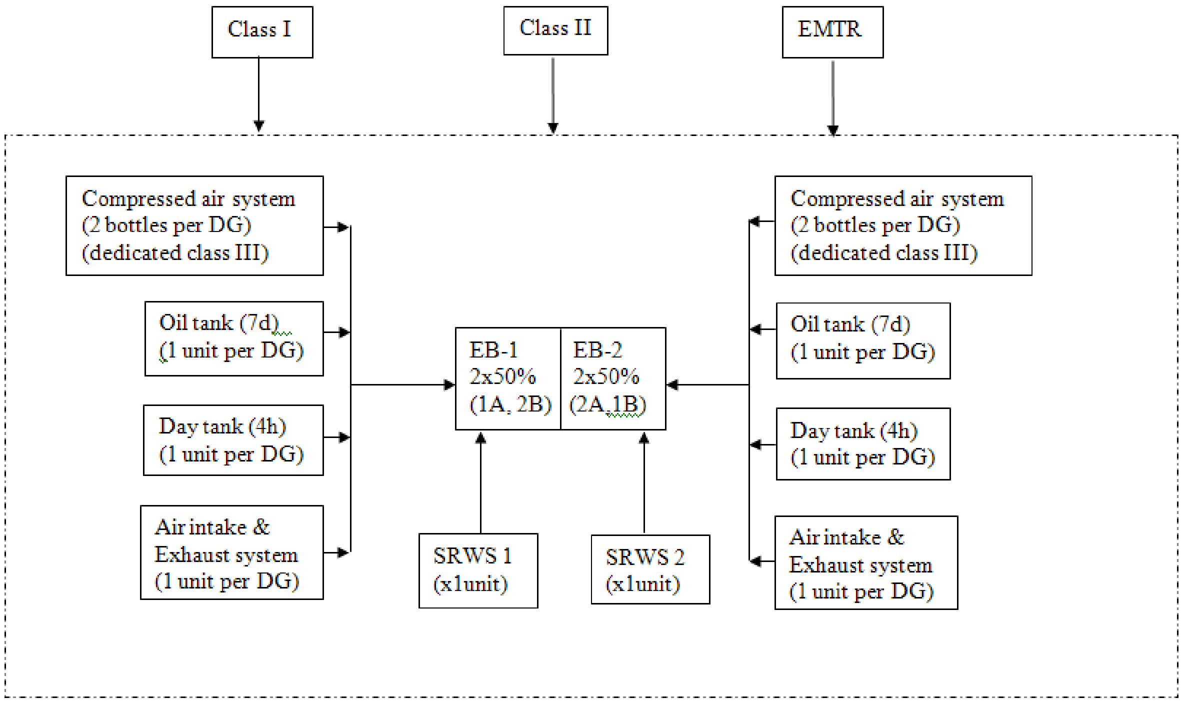

Fig 2.3: Diesel Generator Boundary

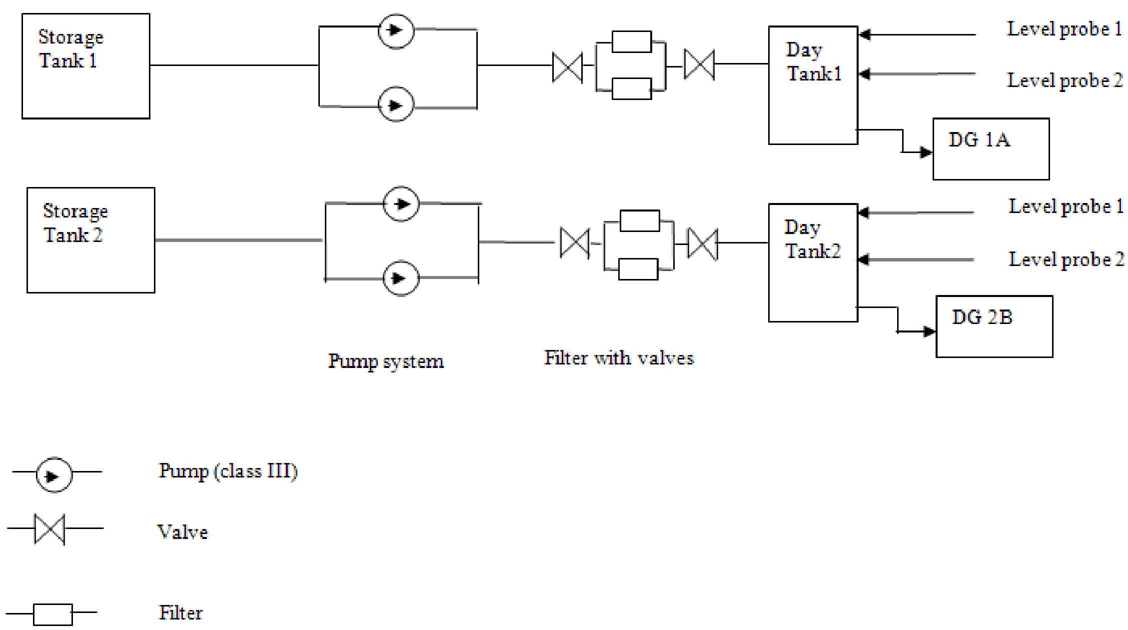

Fig 2.4: Fuel Oil Storage and Transfer System for EB1

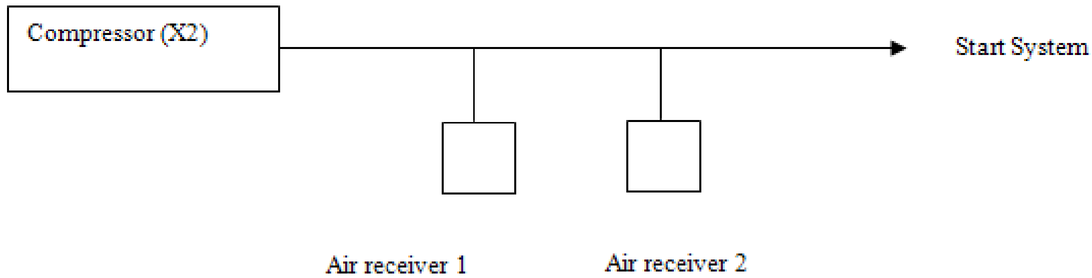

Fig.2.5: Starting Air System

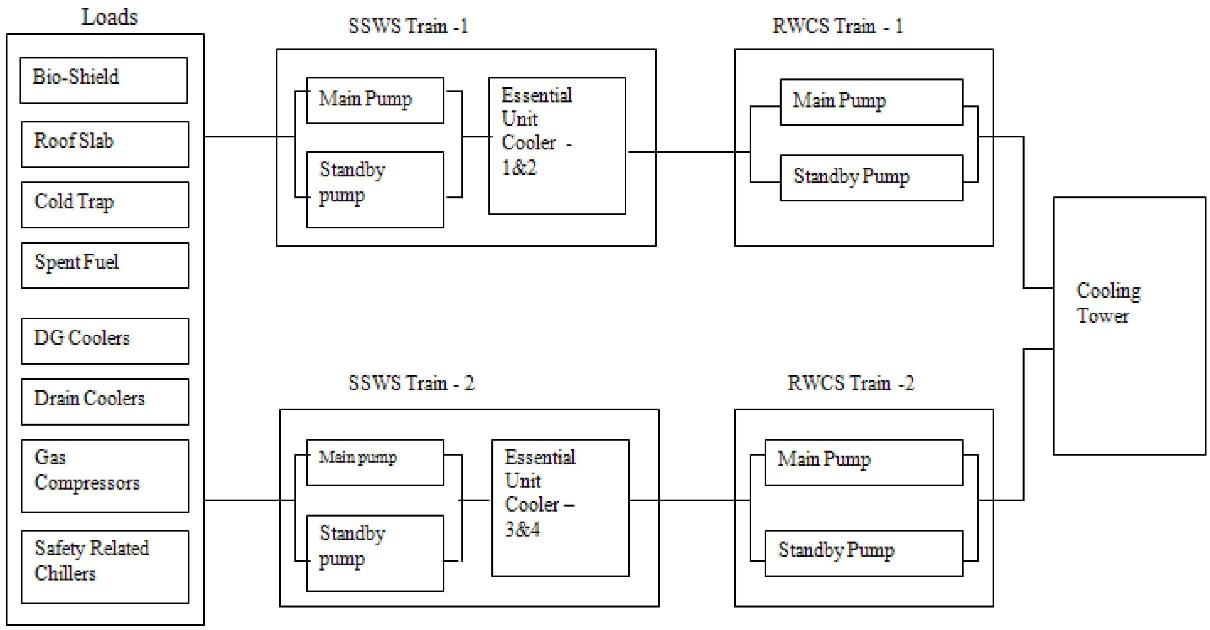

Fig 2.6: Block Diagram of Safety related service water system

2.3.2 System function

During loss of the Class IV supply condition, if one DG fails to supply a bus section in a division, the bus coupler will be closed automatically and the power will be fed from the other section of the same division which is supplied by DG. Inter divisional ties are also provided for additional flexibility and they are manually operated. Further, even if the two DGs located in a DG building go out of operation, the entire Class III loads of the station could be supplied from the two numbers of DG in operation in the other DG building and thus independence of the two divisions is maintained. This is made possible in the design by locating the DG of a division in two different DG buildings. The Class III, 415 V busses or the Power Control Centres (PCC) are fed from the 6.6 kV Class III busses through the LT auxiliary transformers. They are designed on the load centre substation concept with the primary 6.6 kV feeder being radial and the secondary selective arrangement on the 415 V side busses. Each PCC has two sections of 415 V bus and each section is normally fed by a dry type LT auxiliary transformer. A bus coupler is provided between the two sections of the PCC and it is normally kept in the open position. One LT auxiliary transformer is rated to feed the entire loads on both the sections of the PCC. Logic is provided in EMTR for closure of tie breaker between 2 sections of a division at Class III 415 V, on loss of voltage. Logic is provided such that out of the three CBs (two at the downstream of Auxiliary Transformer and one bus coupler) closing of the third breaker is prohibited if two CBs are closed.

For the present analysis of Class III power system, the parameter of interest is the unavailability of DG power at 6.6 kV and 415 V Class III bus levels. Dedicated power from standby diesel generators should be available to the Class III buses on loss of normal supply from Class IV buses. Regarding the system function, it is assumed that the coupler between the two sections in each division i.e. between 1A and 1B and between 2A and 2B is normally closed. In case of offsite power failure, the bus couplers will open and enable DGs to operate independently and feed their respective bus sections. If power supply from a DG to its respective bus section is unavailable, the bus coupler is manually closed.

2.3.3 Support systems

Safety related service water system and control power supply (Class I and II) have been included as support system for DG. SSWS is used because water cooled engines are dependent on the plant service water system. If the cooling subsystem fails then DG can run for few minutes before it overheats. This can be rectified by using air cooled engines or an engine with a dedicated water cooling. In present analysis DG engines are cooled by plant service water system only. Class II has been used in control power supply of DG and Class I has been used in circuit breaker power supply.

Chapter 3
Level-1 PSA Methodology

3.1 Introduction

Probabilistic Risk Analysis for Nuclear Power plants is carried out in convenient vertical phases designated as Level-1 concerned with estimating system reliabilities and identification of various core damage categories and its associated frequency, Level-2 concerned with assessing the containment robustness, evaluating the source term and estimating the large early release frequencies and level-3 deals with assessment of risk to the public and the environment. As the studies are centered on the concept of potential challenges to safety systems (initiators) and responses, it is once again convenient to organize the PSA in terms of the types of initiating events, such as internal events; mainly of random component and system failures, external events; chiefly from fire, flood, seismic sources and other environmental hazards. Any man made external and internal hazards are treated separately as the methodologies are different.

The PFBR design takes into account the lessons learnt from the operating experience of the fast breeder test reactor (FBTR) at Kalpakkam and others. All the reactor structures, systems and components are classified systematically depending on their safety functions and the requirements under seismic events have also been identified. The safety related and critical components are analysed in detail for all the design basis events (DBE) and it has been demonstrated that the design safety limits are met. The events with probability of occurrence ≥ 10^{-6}/ ry are considered as DBE [83].

3.2 Internal Events Methodology

General methodology of development of Level-1 PSA [1] due to internal events is described in following paragraphs. These methods described are also applicable to external hazards; however in external hazard there is added emphasis on development of hazard curves applicable to reactor sites.

3.2.1 Accident sequence initiating event analysis

To perform Level 1 PSA, the set of initiating events need to be identified first. An initiating event is an event that could lead to core damage or deviation from normal operation. It requires successful mitigation using safety or non-safety systems to prevent core damage. In this analysis initiating events are grouped based on internal and external events. In internal events IE groupings has been done based on components random failure and in external events it is based on external causes. Loss of offsite power has been termed as internal event and, in recent listings, fires and floods generated inside the plant.

3.2.2 Accident sequence analysis

Accident sequence analysis is to determine the response of the plant to each group of initiating events that requires the operation of safety systems to carry out the safety functions to prevent core damage. Such safety functions include shutting down the reactor, keeping it subcritical and removal of heat from the reactor core. The events that are identified in the accident sequences will relate to the success or failure of the safety systems and human actions taken in carrying out the safety functions required for the groups of initiating events. The end points of the accident sequence models will be a safe stable state or core damage. Safe stages are those stages, where all required safety functions have been performed successfully.

3.2.3 Definition of core damage

Core damage end state in present analysis is defined as either whole core accident or a subassembly failure. There are several other end states defined for level-1 PSA study (e.g. few pin failure, subassembly failure and whole core accident). However in this study, end states are defined as core damage (CD) or safe (Safe) states.

3.2.4 Safety functions, safety systems and success criteria

The safety functions that need to be performed to prevent core damage have been identified as

i) Detection of initiating event and reactor trip

ii) Shutdown of the reactor and maintaining subcriticality

iii) Decay Heat removal through primary flow path (OGDHRS)

iv) Decay Heat removal through secondary flow path (SGDHRS)

The Safety systems identified for this study are SDS, OGDHRS and SGDHRS. Success criteria for NPP is said to be achieved, if the above mentioned safety functions are achieved.

3.2.5 Modeling of accident sequences (ET)

Accident sequences have been modeled with Event Tree (ET) approach. The approach followed is small Event tree/Large Fault tree [1]. The events sequence models simulate response of plant in case of accident initiator or transient. It models the minimum successful response required from the various systems to arrive at safe of core damage state.

3.2.6 End points of accident sequences and plant damage states

The accident sequence analysis will identify accident sequences, which is safe state, where all the required safety functions have been carried out in a satisfactory manner so that core damage will not occur, or core damage ,where one or more of the safety functions have not been carried out so that core damage is assumed to occur.

3.2.7 Systems analysis

System failures that are identified in the accident sequence analysis are modeled in this analysis. This is usually done by means of Fault Tree (FT) analysis, where the top event of the fault tree is taken as the system failure state(s) identified by the event tree analysis. The fault tree is top-down approach, where system failure extend the analysis down to the level of individual basic events, which typically include component failures, unavailability of components under periodic maintenance/ testing, common cause failures of redundant components and human failure events.

3.2.8 Fault tree analysis

This Fault tree has been developed using the ISOGRAPH software [12]. Fault tree has been modeled using immediate cause approach. The quantification of Fault tree has been done by rare-event approximation. Since PFBR is under construction, there is no operational data available for reliability analysis. The data used in the quantification of the unavailability of various systems is taken from generic source [84].

3.3 External Events

In the context of Probabilistic Safety Analysis (PSA) of nuclear power plants (NPP), external events are defined as events originating from outside the plant, but with the potential to create a PSA initiating event at the plant [16]. They may, however, originate from within the site (e.g. local transportation accidents), or even from another plant on the same site (e.g. fire spreading between plants).

External events can occur as single events or as combinations of two or more external events. Potential combined events are two or more external events having a non-random probability of occurring simultaneously, e.g., strong winds occurring at the same time as high sea water levels. Combined events which may contribute significantly to the plant risk need to be identified during the analysis.

External events are normally grouped into natural events and man-made events. Examples of man-made external events are airplane crash and gas explosion, while coastal flooding and various extreme weather conditions are examples of natural external events. External events analysis covers procedure for identification, categorization, screening analysis, quantification, and PSA modeling of External events.

Redundancy is one of the fundamental techniques employed to achieve high level of functional reliability in a safety-systems. External events pose a definitive challenge to redundancy, due to its ability to induce common cause failures. These events also challenge offsite power, the integrity of plant structures and threaten the provided onsite mitigation measures. It is therefore essential to understand the accident progression and the impact of these external events in an NPP and identify the key components of NPP, which contribute to risk from external events in order to comprehend and protect the plant from such risks. External event PSA is an accepted tool to accomplish this task. Probabilistic safety analysis of external events requires the use of specialized methods to address the assessment of frequency of occurrence versus magnitude for external events and the modeling of component and structure failure in terms of variables that describe physical interactions [26]. The basic elements of the analysis of risk from an external event are

1. Hazard analysis
2. Plant-system and structure response analysis
3. Evaluation of the fragility and vulnerability of components
4. Plant-system and sequence analysis
5. Consequence analysis

The different elements of an external event PSA are illustrated in figure 3.1.

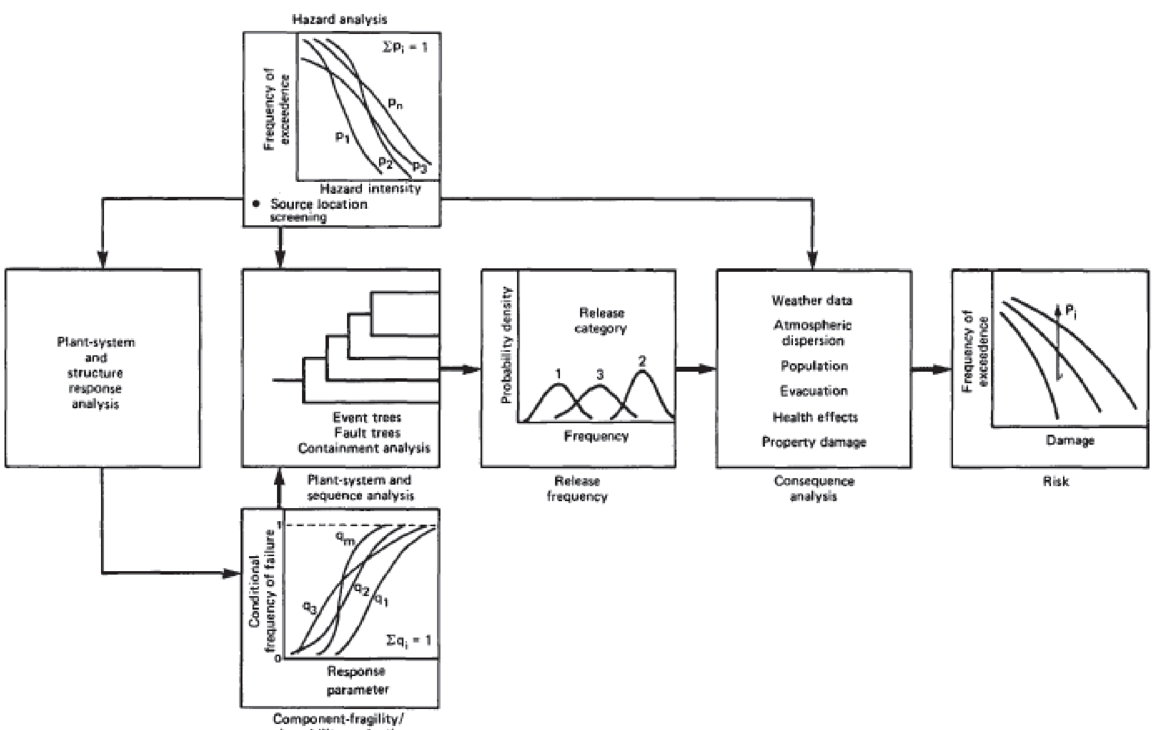

Fig 3.1: Different Elements of External Event PSA

3.3.1 List of hazard considered

External hazards have been further divided into three parts. First part deals with seismic events. Second part is flooding events. It includes storm surge, rainfall and tsunami. All other events have been described in third part. The other events include wind hazard, aircraft crash, lightning and missile protection.

3.3.2 Methodology

Seismic hazard analysis methodology

The Seismic Probabilistic Safety Analysis (SPSA) Level-1 for DG of PFBR aims in estimating earthquake initiated failure frequency for class III power supply system. The method consists of three major parts, viz., assessing the seismic hazard at the ground level, assessment of the seismic fragility of various safety systems, structures and components and integration of seismic hazard with fragility information through appropriate logic models of plant safety functions [26, 43]. The analysis tasks are depicted in Fig. 3.2 [43].

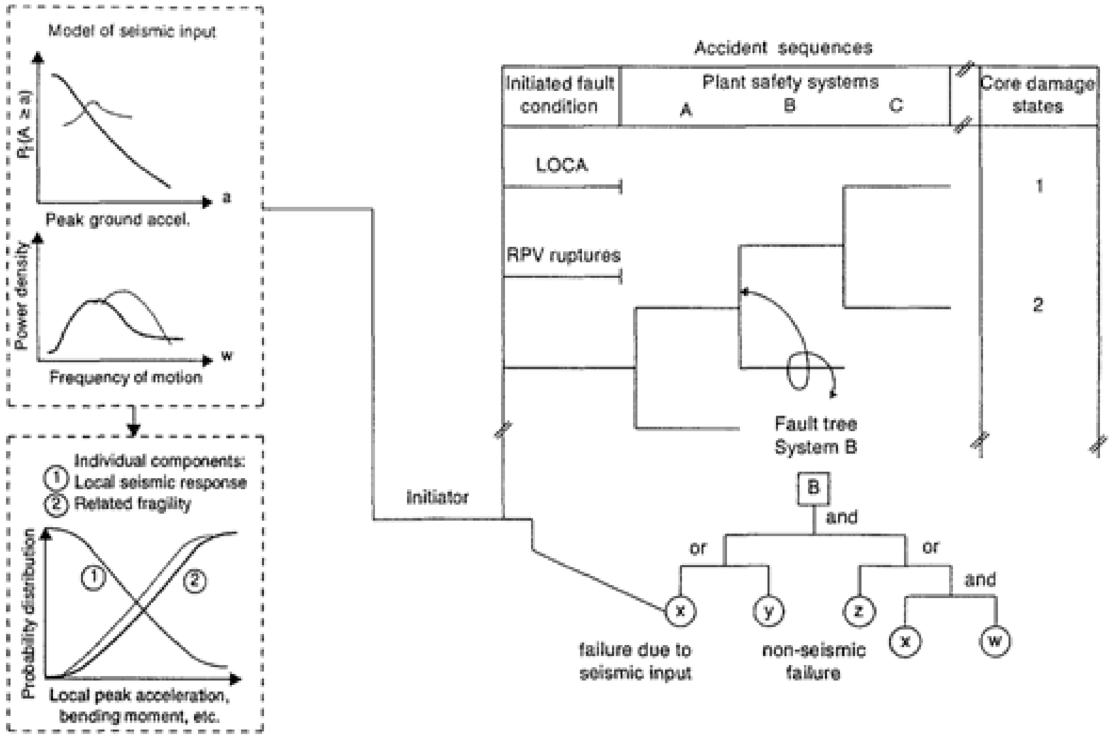

Fig. 3.2: Schematic Overview of Seismic PSA

Flooding events methodology

External food probabilistic safety analysis has been performed to understand the accident progression in an NPP initiated due to an external flood [50]. EFPSA also helps in determining the key contributors to the risk due to external flooding in NPP. EFPSA begins with a probabilistic flood hazard assessment, which is followed by postulation of initiating events that can arise out of flood impact at critical locations. For the postulated initiating events, the event trees for the desired safety objectives are to be developed. For each system function in the ET, fault trees are developed. The basic events of the fault trees required to achieve the safety functions can be determined to form the essential SSC list. The survivability of each component of SSC list for the possible effects of flooding (viz static/dynamic pressure, equipment malfunction, debris impacts, submergence etc.) needs to be assessed based on analysis and plant walk-down and the margin quantified in terms of fragilities. These fragilities are used to quantify the FT and subsequently the ETs to obtain the plant fragility, which can be convoluted with the hazard curve to obtain the failure frequency due to external flooding.

Chapter 4
Internal Events

4.1 Introduction

The initiating events are generally classified into internal IEs and hazards (internal and external). Internal IEs are hardware failures in the plant or faulty operations of plant hardware through human error or computer software deficiencies. External hazards (external events) are events that create extreme environments common to several plant systems. External hazards include earthquakes, floods, high winds and aircraft crashes. Internal hazards include internal flooding, fire and missile impact. It also includes the loss of off-site power.

4.2 Success Criteria

The system is analyzed for two success criteria namely, i) 2/4 DG Success where at least 2 DGs are needed for Class III buses, ii) 1/4 DG Success where at least 1 DG are needed. Since the DGs supply the loads through 6.6 kV buses and 415V buses, success criteria at bus level are as follows.

Case 1X: 2/4 DG (success); any one bus section failure at 6.6 kV level and any one bus section failure at 415V level.

Case 2X: 1/4 DG (success); any three bus sections failure at 6.6 kV level and 5/8 bus failures at 415V level.

4.3 Assumptions in Fault Tree Analysis

A mission time of 12 hours has been assumed for DG run failure, as most demands on DGs require a mission time of less than 12 hours (Table 4.1). A preventive maintenance of 10 days in a year has been assumed for unavailability calculation.

4.4 Scope

The scope of the present analysis is limited to risk at full power operating state due to internal events including offsite power failure events. Initiating events arising during low power and shutdown states are excluded from this analysis. External events i.e., fire (internal/external), flood (internal/external), high wind and seismic events are excluded in this study. The failures of components represent only random failures. The metric obtained is unavailability of DG power supply system, defined as sum of frequency of components failure that leads to DG power supply system failure.

4.5 Fault tree Modeling

The Fault tree has been developed for this system (Fault tree page of condition 1X mentioned in section 4.2 is shown in Fig. 4.1(Annexure)). This Fault tree has been evaluated using the ISOGRAPH software [12]. Fault tree has been modeled using immediate cause approach. Fault tree has been quantified using rare-event approximation. Since PFBR is under construction, there is no operational data available for reliability analysis. The data used in the quantification of the unavailability of Class III power system is taken from generic source [84] and is shown in Table 4.1(Annexure). This database has accounted earlier fast reactors operating data and experience. Some data like DG failure to start, failure to run have been taken from operating experience of FBTR (Fast Breeder Test Reactor). The data utilized for the quantitative assessment of system reliability or unavailability is on a point-estimate basis. A review of DG subsystem failures is performed to determine failure modes. For ease of handling, the components are codified as shown in this Table 4.1(Annexure). It can be seen from Table 4.1, that the data needed for the evaluation of unreliability of components of different types is different. The details of the failure modes of the components considered in the Fault tree for class III power system are also shown in the Table 4.1(Annexure). Support system of safety related service water system has been modeled in detail in the Fault tree. The basic events, parameters and models for safety related service water system are shown in Table 4.2(Annexure).

4.6 Common Cause Failure Analysis

A common cause failure is the failure of more than one component, sub-system or system due to the same cause. Common cause failures often make a substantial contribution to the unavailability of systems that contain redundancy.

4.6.1 Identification of common cause component group (CCCG)

Common Cause Component Group (CCCG) has been made for common attributes of similar component and for failure mechanism which can lead to common cause failures. It has been traditionally relied on common sense, engineering insight and obvious sign of dependence. Identical non-diverse components which provide redundancy are always put into a common cause group [85]. If diverse component has subcomponents which are identical and redundant, then subcomponents should be identified as CCF group. In the present analysis, CCCG has been chosen with following attributes

- Component types (e.g. DG Air Filters, Fuel Transfer Pumps)
- Component use (e.g. DG air bottle 1, DG air bottle 2)
- Component location (e.g. DG in EB1 fail to run, DG in EB2 fail to start)
- Operating mode of the component (e.g. SSWS stand by pump fail to run)

The two DGs located at a DG station are subjected to the same environment and hence experience a higher unavailability due to common cause failure. However, the chances of a common cause affecting all the four DGs is considered to be less probable than that affecting two DGs at the same station. This assumption has been made because DG stations are located at each side of the Reactor Containment Building (RCB). The data shown in Table 4.3 reflects these aspects. Common cause component groups are formed for different circuit breakers based on location. CCF due to maintenance has not been considered as only one DG is taken out for maintenance at a time. Conservatively it is assumed that the trip coils of all circuit breakers are connected to one bus section of 220 V DC. This assumption is made to address the common cause failure of trip coils in CB due to control power supply failure. Similar assumption has been made for Compressed air system, i.e. single line of Compressed air system is connected to all valves. Therefore separate air receiver is provided apart from Compressed air system for each DG starting air system to keep redundancy. Table 4.3A, 4.3B, 4.3C represents components group and CCF models used in Fault tree.

Table 4.3A: CCF models used for analysis of Class III Power supply system (Beta factor)

No,	Name	Description	Redundancy	Beta factor
1	BUS FAIL	Bus Failure	4	0.01
2	CB-INC-CCF	Incomer Circuit Breaker CCF	8	0.05
3	DG-FS-EB1-CCF	DGs in Electrical Building-1 fail to start CCF	2	0.03
4	DG-FR-EB1-CCF	DGs in Electrical Building-1	2	0.03

No,	Name	Description	Redundancy	Beta factor
		fail to run CCF		
5	CB-DG-EB1-CCF	CB of DG in EB1 Failure due to CCF	2	0.05
6	CB-DG-EB2-CCF	CB of DG in EB2 Failure due to CCF	2	0.05
7	CB-BC-D1D2-CCF	CB b/w 1A and 1B fail to change position due to CCF	2	0.05
8	CB-LD-CCF	Load Side Circuit Breaker CCF	4	0.05
9	DGCOMP-EB1-FS- CCF	Compressors for DGs in EB-1 fail to start CCF	4	0.05
10	DGAIRBOT-EB1-CCF	Air bottles for DGs in EB-1 CCF	4	0.05
11	DGCOMP-EB2-FS-CCF	Compressors for DGs in EB-2 fail to start CCF	4	0.05
12	DGAIRBOT-EB2-CCF	Air bottles for DGs in EB-2 CCF	4	0.05
13	FUPMP-EB1-FS-CCF	Fuel pumps for DGs in EB-1 fail to start CCF	4	0.05
14	FUPMP-EB2-FS-CCF	Fuel pumps for DGs in EB-2 fail to start CCF	4	0.05
15	DTLC-EB1-CCF	Day Tank Level Control for DGs in EB-1 CCF	4	0.05
16	DTLC-EB2-CCF	Day Tank Level Control for DGs in EB-1 CCF	4	0.05
17	FUPMP-EB1-FR-CCF	Fuel pumps for DGs in EB-1 fail to run CCF	4	0.05
18	FUPMP-EB2-FR-CCF	Fuel pumps for DGs in EB-2 fail to run CCF	4	0.05
19	DGCOMP-EB1-FR-CCF	Compressors for DGs in EB-1 fail to run CCF	4	0.05
20	DGCOMP-EB2-FR-CCF	Compressors for DGs in EB-2 fail to run CCF	4	0.05
21	BUS-415V-CCF	415V Bus failure due to CCF	8	0.01
22	TRANS-CCF	6.6 kV / 415V Transformers CCF	8	0.01
23	DG-FR-EB2-CCF	DGs in Electrical Building-2 fail to run CCF	2	0.03
24	DG-FS-EB2-CCF	DGs in Electrical Building-2 fail to start CCF	2	0.03

No,	Name	Description	Redundancy	Beta factor
25	DG-VALVE-CCF	DG valve CCF	8	0.10
26	DG-FILTER-CCF	DG filter CCF	8	0.10

Table 4.3B: CCF models used for analysis of SSWS (Beta factor)

No,	Name	Description	Redundancy	Beta factor
1	SSWS-HXB-F-BSCCF	CCF of HX Bio Shield	2	0.10
2	SSWS-HXB-F-SFSBCCF	CCF of HX at spent fuel storage bay	3	0.10
3	SSWS-HXP-F-ESUCCCF	CCF of ESUC with level of redundancy 4	4	0.10
4	RWCS-QBF-FS-CCF	CCF of fail to start of cooling tower fans	4	0.10
5	RWCS-QBF-FR-CCF	CCF of fail to run of cooling tower fans	4	0.10
6	SSWS-VXA-D-CCF	CCF of fail to remain in position of manually operated valves	4	0.10
7	SSWS-HXB-F-CTCCF	CCF of cold trap HX	2	0.10
8	SSWS-HXB-F-RSCCF	CCF of Roof slab HX	2	0.10
9	SSWS-HXB-F-DGCCF	CCF of DG coolers HX	4	0.10
10	SSWS-HXB-F-DRCCCF	CCF of HX of drain coolers	4	0.10
11	SSWS-HXB-F-GCCCF	CCF of HX of Gas compressors	2	0.10
12	SSWS-HXB-F-SRCCCF	CCF of HX of safety related chillers	4	0.10
13	MKUP-VXA-D-CCF	valve fail to remain in position due to CCF	2	0.10
14	RWCS-FEA-AL-CCF	CCF of Joints in RWCS circuits	2	0.10
15	SSWS-FEA-AL-CCF	CCF of joints in SSWS circuits	2	0.10
16	SSWS-HXB-F-GCCCF	CCF of HX of Gas compressors	2	0.10
17	SSWS-HXB-F-	CCF of HX of safety related	4	0.10

No,	Name	Description	Redundancy	Beta factor
	SRCCCF	chillers		
18	MKUP-VXA-D-CCF	valve fail to remain in position due to CCF	2	0.10
19	RWCS-FEA-AL-CCF	CCF of Joints in RWCS circuits	2	0.10
20	SSWS-FEA-AL-CCF	CCF of joints in SSWS circuits	2	0.10
21	MKUP-FEA-AL-CCF	CCF failure of joints in makeup water circuit	2	0.10

Table 4.3C: CCF models used for analysis of SSWS (Alpha factor)

CCF models			Alpha factors			
Name	Redundancy	Description	Alpha 1	Alpha 2	Alpha 3	Alpha 4
RWCS-PMA-FR-CCF-4	4	CCF of fail to run of RWCS pumps	0.9623	0.0161	0.0055	0.0161
RWCS-PMA-FS-CCF-2	2	CCF of Fail to start of RWCS pumps	0.9639	0.0361	0	0
RWCS-VMA-F-CCF-4	4	CCF of motor operated flow control valve	0.9753	0.0137	0.0079	0.0031
SSWS-PMA-FR-CCF-4	4	CCF Fail to run of SSWS pumps	0.9623	0.0161	0.0055	0.0161
SSWS-PMA-FS-CCF-2	2	CCF Fail to start of SSWS pumps	0.9639	0.0361	0	0
MKUP-PMA-FR-CCF-2	2	CCF of makeup water fail to run	0.9618	0.0382	0	0
MKUP-PMA-FS-CCF-2	2	CCF of makeup water pump fail to start	0.9639	0.0361	0	0

4.6.2 Common cause failure (CCF) model

The common cause failure model used in the analysis is beta factor and alpha factor model as described below.

Beta factor model

The beta factor model is based on the assumption that, if the common cause failure were to occur, all components in the CCF group would fail together [85]. The beta factor model is a single parameter model requiring only the beta factor to be specified. For example, if the event A belongs to CCF group CCF1 then the expression in Boolean algebra will be,

$$A \rightarrow A+CCF1 \qquad (4\text{-}1)$$

where A represents independent failure of A.

The unavailability values of the independent and CCF events are given by

$$Q_I = (1-\beta).Q_T \qquad (4\text{-}2)$$

$$Q_{CCF} = \beta.Q_T \qquad (4\text{-}3)$$

where β, Q_I , Q_T ,Q_{CCF} , represents beta factor, Independent unavailability, Total unavailability, and unavailability due to CCF respectively.

Beta factor CCF models for different component groups are shown in Table 3A and 3B.

Alpha Factor Model

The alpha factor model defines common cause failure probabilities from a set of failure frequency ratios and the total component failure probability Q_T [37]. In terms of the basic event probabilities, the alpha factor parameters ($\alpha 1$, $\alpha 2$, $\alpha 3$, $\alpha 4$) are defined as

$$Q_k = \frac{k}{{}^{m-1}_{k-1}C} . \frac{\alpha_k}{\alpha_T} . Q_T \qquad (4\text{-}4)$$

$$\alpha_T = \sum_{k=1}^{m} k.\alpha_k \qquad (4\text{-}5)$$

Where Q_k = unavailability of kth order CCF failure, m=group size, Q_T= Total unavailability

Alpha factor CCF models for different component groups are shown in Table 4.3C.

4.7 Human Error

Each division is connected to two DG. Failure of division will lead to failure of all loads connected to that division. Failure probability of simultaneous failure of two DG is very small. If this happens then only intra division tie circuit breaker between bus sections are used. Hence for manual closing of intra division tie circuit breaker between bus sections in a division Human Error probability has been conservatively assumed as 1.

4.8 Importance Measures

The purpose of the importance evaluation is to identify the important accident sequences, system failures and component failures with regard to unavailability of system. This section presents Fussell-Vesely, Birnbaum, Barlow-Proschan and Sequential importance measures [39]. The value of the Fussell-Vesely, Birnbaum, Barlow-Proschan and Sequential importance measures of selected few events are shown in Table 4.4. This analysis on importance measures indicates that the major components whose reliability needs to be improved are DG fail to run, DG fail to run due to CCF and DG under maintenance.

4.8.1 Fussell-Vesely importance

The Fussell-Vesely standard importance measure for system in a Fault tree indicates an event or event group’s contribution to the system unavailability. Increasing the availability of events with high importance values will have the most significant effect on system availability. The standard Fussell-Vesely unavailability importance value for an event i is given by

$$I_i^{FV} = \frac{Q_{sys} - Q_{sys}(q_i = 0)}{Q_{sys}} \tag{4-6}$$

where I_i^{FV} is the Fussell-Vesely importance for event I, Q_{sys} is the unavailability of the system and $Q_{sys}(q_i = 0)$ denotes the unavailability of the system when the ith component is not available.

4.8.2 Birnbaum importance

The Birnbaum unavailability importance is given by

$$I_i^{BB} = \frac{\partial Q_{sys}}{\partial q_i} \tag{4-7}$$

where I_i^{BB}= Birnbaum importance measure for component I, Q_{sys} = system unavailability , q_i = unavailability of component i.

4.8.3 Barlow-Proschan importance

The Barlow-Proschan event importance measure considers the sequence of event failures. It is, in effect, the probability that the system fails because a critical cut set containing the event fails, with the event failing last. The Barlow-Proschan importance measure is given by

$$IMP_{BP} = \frac{\omega_{Event} .A}{Q_{Event} .B} \tag{4-8}$$

where ω_{Event} =Event failure frequency, Q_{Event}= Event unavailability, A= Sum of unavailability of cutsets containing events, and B= system failure frequency

4.8.4 Sequential importance

The sequential importance measure for an event considers the role of the failure of component i when another component j actually causes the system to fail. The method of calculation of the measure is best illustrated by an example. Take 3 events A, B and C all occurring in the same cut set A.B.C. The contribution to the importance value for event A is given by

$$I_{SI} = Q_A Q_B \omega_C + Q_A Q_C \omega_B \tag{4-9}$$

where ω_{Event} =Event failure frequency, Q_{Event}= Event unavailability. (i.e. all contributions to the cut set failure frequency except for the term where A is the final failure). Contributions for each cut set are summated and divided by the system failure frequency.

Table 4.4: Importance Ranking for Class III Power Supply (2 out of 4 Success (1X))

No.	Event	Description	Fussell Vesely Importance	Birnbaum Importance	Barlow-Proschan Importance	Sequential Importance
1	DG1B-FR	DG-1B fail to run	1.597e-1	2.216e-2	7.777e-6	6.277e-6
2	DG1BM	DG 1B in maintenance	7.664e-2	1.332e-2	0	3.318e-6
3	DG-FR-4/4F-CCF	DG 4/4 Run Failure due to CCF	7.204e-2	1.000e+0	1.000e+0	0
4	DG1A-FS	DG-1A fail to start	4.512e-2	2.214e-2	0	1.774e-6
5	DGCOMP-EB1-FS-CCF	Compressors for DGs in EB-1 fail to start CCF	4.268e-2	1.693e-1	0	8.648e-7
6	MKUP-PMA-FS-CCF-2	CCF of makeup water pump fail to start[Main pump fail to start Standby pump fail to start]	4.245e-2	1.000e+0	0	0

No.	Event	Description	Fussell Vesely Importance	Birnbaum Importance	Barlow-Proschan Importance	Sequential Importance
7	SSWS-IAA-F-TR02	Other instrumentation failures	4.064e-2	1.645e-1	6.57e-8	8.621e-7
8	RWCS-IAA-F-TR02	Instrumentation Failure	4.064e-2	1.645e-1	6.57e-8	8.621e-7
9	DG-FR-EB1-CCF	DGs in Electrical Building-1 fail to run CCF	3.773e-2	1.693e-1	1.837e-6	7.645e-7
10	CB-INC-CCF	Incomer Circuit Breaker CCF	3.046e-2	1000e+0	0	0
11	CB-LD-CCF	Load Side Circuit Breaker CCF	3.046e-2	1.000e+0	0	0

4.9 Sensitivity Analysis

Sensitivity calculation of the basic events is carried out to find the top event unavailability responses to variations in basic input values of failure rates. The result of this analysis indicates the relative importance of individual component failure rates. Sensitivity of the top event unavailability to variation in all basic events (10% changes in the values) is analyzed and selected few are presented in Table 4.5. The result indicates DG mechanical failure to run is most sensitive part.

Table 4.5: Sensitivity Analysis of Class III Power Supply System

No.	Event Name	Event Description	Sensitivity
1	DG1B-FR	DG-1B fail to run	1.17E+00
2	DG1AM	DG 1A in maintenance	1.07E+00
3	DG-FR-4/4F-CCF	DG 4/4 Run Failure due to CCF	1.07E+00
4	DG1A-FS	DG-1A fail to start	1.04E+00
5	RWCS-IAA-F-TR02	Instrumentation Failure	1.04E+00
6	SSWS-IAA-F-TR01	Other instrumentation failures	1.04E+00

7	EMTR	Emergency transfer failure	1.02E+00
8	HE	Human Error after the auto failure	1.02E+00
9	DG-FS-4/4-CCF	DG 4/4 start failure due to CCF	1.02E+00
10	CB-DG1-1A	CB of DG 1A fails to close	1.01E+00
11	CB-LD-IA	Load side breaker fail to open and close(x2)	1.01E+00
12	CB-BC-1AB	CB b/w 1A and 1B fail to change position	1.01E+00
13	BUS1AH	Bus 1A fails	1.00E+00
14	CB-UB1-1A1	CB connecting Unit bus-1 to Bus-1A fail to open	1.00E+00
15	AIRBOT-1-FAIL	Air Bottle-1 failure	1.00E+00

4.10 Uncertainty Analysis

The most widely used technique for propagating uncertainties is Monte Carlo simulations. In general, Monte Carlo simulations consist of generating a random sample of the inputs in the model and determining the PSA output from each set of inputs in the sample [1]. This process results in a random sample of the PSA output. Quantitative measures of the uncertainty associated with the output are then derived from this random sample. Uncertainty values may be specified for selected parameters in the event data models. These uncertainty values are only used during confidence analysis. If a lognormal distribution is specified, the uncertainty value is the Error Factor and the parameter value may represent the median, mode or mean of the distribution. The expression for the lognormal distribution is given by

$$f(x) = \frac{1}{x\sigma\sqrt{2\pi}}\exp[\frac{-(lnx-\mu)^2}{2\sigma^2}] \qquad (4\text{-}10)$$

with median, mean and S.D are given as

Median=e^{μ} (4-11)

Mean=$e^{\mu+0.5\sigma^2}$ (4-12)

$(\mathrm{S.D.})^2 = e^{2\mu+\sigma^2} . (e^{\sigma^2} - 1)$ (4-13)

The percentiles are obtained by direct translation from the normal distribution: qth percentile of Y = exp (qth percentile of X). In particular

95th percentile = $e^{\mu+1.65\sigma}$ (4-14)

5th percentile = $e^{\mu-1.65\sigma}$. (4-15)

It follows that the Error Factor, defined as the ratio of the 95th percentile to the median, is

EF = $e^{1.65\sigma}$. (4-16)

For 90th percentile Error Factor is given as

EF = $e^{1.28\sigma}$ (4-17)

ISOGRAPH software allows a lognormal distribution to be specified in several ways, either μ and σ, or the mean and Error Factor, or the median and Error Factor. Any of these pairs uniquely determines the distribution. In the present analysis lognormal distribution is based on the Error Factor (EF) and median failure rate. The EF is assigned based on available information and judgment ranges from 5 to 10 (Maintenance EF=2). The sampled values are propagated to get the top event distribution. For the Diesel Generator, the top event is simulated for 10000 times and lognormal distribution is assumed for all parameters. The analysis indicates that the DG unavailability is uncertain by Error Factor 4.4 (90% confidence bound) for 2 out of 4 DG system (system success) and by Error Factor 4.1 (90% confidence bound) for 1 out of 4 DG systems (system success). Uncertainty distribution is shown in Fig. 4.2.

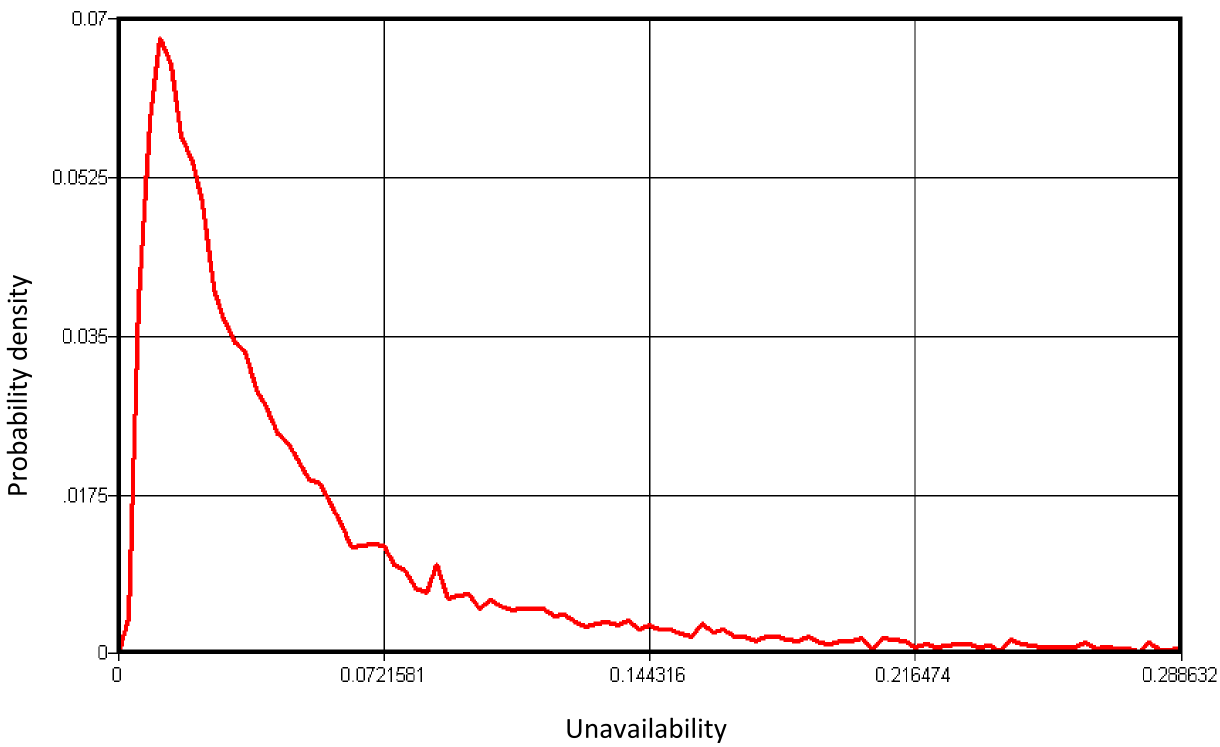

Fig. 4.2: Uncertainty Distribution for DG

4.11 Results

The minimal cut sets (combination of minimum number of components failure that lead to unavailability of the system) of Class III power system (Case 1X: 2/4 DG success) are evaluated and these are shown in Table 6. The unavailability of DG power supply system for this case is evaluated to be 4.75E-3. The first 20 minimal cut sets are shown in Table 4.6. All the minimal cut sets, except those due to common cause failures and Bus failures are at least of second order (due to two or more component failures). EMTR is single component failure. The importance measures of the components of Class III power system are shown in Table 4.4. (Importance measures determine the change in the system metric due to change in parameters of the model). Based on these importance measures, critical parameters are identified. By focusing more resources on the most critical parameters, system performance can be improved effectively. It can be seen from the Table 4.4 that DG fails to run, DG fails to run due to CCF and DG under maintenance are important basic events in Class III power system. It can be noticed from Table 4.6, common cause failures contribute significantly to the unavailability of the system.

Table 4.6: Minimal Cut sets for Class III Power Supply (2 out of 4 Success (Case 1X))

No.	Cut set	Event description	Unavailability
1	DG-FR-4/4F-CCF	DG 4/4 Run Failure due to CCF	3.43e-4
2	MKUP-PMA-FS-CCF-2	CCF of makeup water pump fail to start[Main pump fail to start Standby pump fail to start]	2.02e-4
3	CB-LD-CCF	Load Side Circuit Breaker CCF	1.45e-4
4	CB-INC-CCF	Incomer Circuit Breaker CCF	1.45e-4
5	DG-FS-4/4-CCF	DG 4/4 start failure due to CCF	1.00e-4
6	EMTR	Emergency transfer failure	1.00e-4
7	MKUP-PMA-FR-CCF-2	CCF of makeup water fail to run[Main Pump Fail to RunStandBy Pump fail to run]	7.05e-5
8	SSWS-PMA-FR-CCF-4	CCF Fail to run of SSWS pumps[Pump Fail to Run Stand By Pump Fail to run Pump Fail to Run Stand By Pump Fail to run]	5.74e-5
9	RWCS-PMA-FR-CCF-4	CCF of fail to run of RWCS pumps[Pump Fail to Run Pump Fail to run Pump Fail to Run Pump Fail to run]	5.74e-5
10	DGCOMP-EB1-FS-CCF. DG2A-FR	Compressors for DGs in EB-1 fail to start CCF DG-2A fail to run	4.12e-5
11	DG1A-FR. DGCOMP-EB2-FS-CCF	DG-1A fail to run Compressors for DGs in EB-2 fail to start CCF	4.12e-5
12	DGCOMP-EB2-FS-CCF. DG2B-FR	Compressors for DGs in EB-2 fail to start CCF DG-2B fail to run	4.12e-5
13	DGCOMP-EB1-FS-CCF. DG1B-FR	Compressors for DGs in EB-1 fail to start CCF DG-1B fail to run	4.12e-5
14	RWCS-IAA-F-TR01. DG2B-FR	Instrumentation Failure DG-2B fail to run	4.04e-5
15	DG1A-FR. RWCS-IAA-F-TR02	DG-1A fail to run Instrumentation Failure	4.04e-5
16	DG1A-FR.	DG-1A fail to run Other instrumentation failures	4.04e-5

No.	Cut set	Event description	Unavailability
	SSWS-IAA-F-TR02		
17	SSWS-IAA-F-TR01. DG2A-FR	Other instrumentation failures DG-2A fail to run	4.04e-5
18	RWCS-IAA-F-TR01. DG2A-FR	Instrumentation Failure DG-2A fail to run	4.04e-5
19	SSWS-IAA-F-TR02. DG1B-FR	Other instrumentation failures DG-1B fail to run	4.04e-5
20	RWCS-IAA-F-TR02. DG1B-FR	Instrumentation Failure DG-1B fail to run	4.04e-5

Evaluations are also carried out for the case where availability of 1 out of 4 DGs is sufficient to meet the needs of essential power supply under extended Class IV failure condition (Case 2X: 1/4 DG success). The unavailability of DG power supply system for this case is 1.47E-3.The minimal cut sets and their associated probabilities contributing to Class III power system unavailability, with this new system success criterion have been calculated. The top 10 cut sets remains same. The importance measures of the components have also been calculated. The major difference from case 1X is DG under maintenance becomes more important than, DG fails to run due to CCF and DG fails to run. This happens because a single DG has to run for required mission time.

The percentage contribution from various subsystems is depicted in Fig.4.3. It can be seen from the pie chart (Fig.4.3) that no single subsystem has predominant contribution to system unavailability. Major contributors are all DG fail to start and fail to run due to CCF (9%). Circuit breakers contribute 3%, EMTR contributes about 2% and Safety related service water system contributes 8 % to overall unavailability in operating Diesel Generator. Other failures includes instrumentation, compressor and combination of thee DG fail to run when one DG is under maintenance.

A literature survey has been done to estimate the range of DG Power Supply failures. This literature survey has been done to compare the unavailability of PFBR DG system with unavailability of other DG systems. As reported in reference [32] the unavailability of DG is in the range of 3.94E-3-1.77E-2.

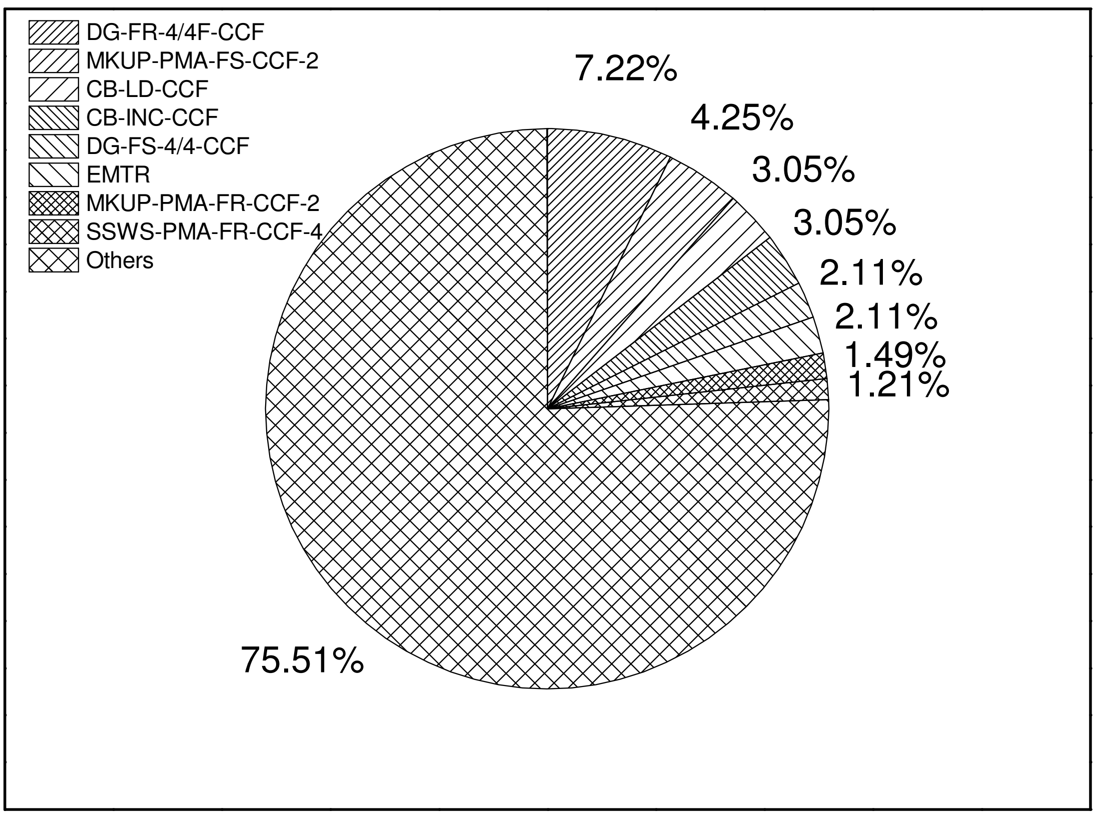

Fig.4.3: Contribution from Cut sets of Diesel Generator

4.12 Failure Frequency Contribution of Class III Power Supply from Internal Events

Initiating Events

Loss of offsite power failure is initiating event for class III power supply system. This initiating event can occur due to the failure of grid (class IV) power supply. Loss of offsite power for PFBR is defined as complete loss of power supply in both stations buses and both unit buses. Total frequency for Loss of offsite power failure is 2/year [29]. Otherwise Class III power supply is always on standby mode. For preventive maintenance one DG set is taken at a time. The event considered in this group is loss of one diesel generator during mandatory testing when the reactor is on power. A DG can fail during testing due to the failure of different sub systems associated with DG like start air system, jacket cooling system, generator, DG protection and control system. This initiating event does not initiate automatic scram. As per technical specification, the unavailability of one DG and its non-recovery within 7 days require manual shutdown of the reactor. There are no trip parameters for this event. Recovery actions will be initiated after this event occurs. In case of non-recovery within the stipulated time in technical specification, manual scram will be initiated.

Event Tree

The event tree depicting the sequence of events following this initiating event having frequency 0.5 is shown in Fig.4.4 and represented by failure of one DG during mandatory testing.

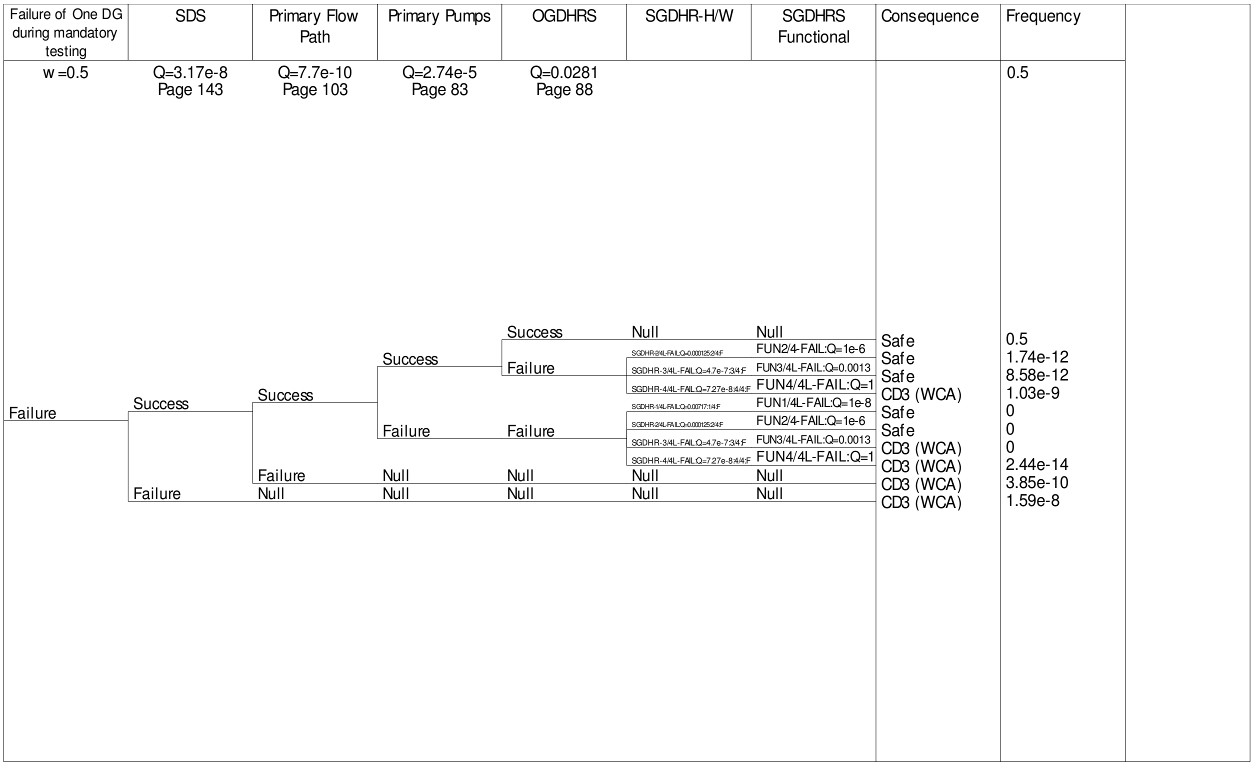

Fig: 4.4: Event tree for Class III power supply system

The failure frequency contribution from class III power supply system following this initiating event is 1.73E-8/ry. Top 20 minimal cutsets for this event tree is shown in Table 4.7.

Table 4.7: Minimal cutsets for event Tree failure of one DG during mandatory testing

No.	Cut-set	Event description	Frequency
1.	ONE-DG-FAIL. CDRCF0001	Failure of One DG during mandatory testing CCF (mechanical) of CSRs & DSRs	1.50e-8
2.	ONE-DG-FAIL. SWCCF12	Failure of One DG during mandatory testing CCF SCRAM Switches in system 1 and 2	5.00e-10
3.	ONE-DG-FAIL. PHT-IHX-CCF-BL	Failure of One DG during mandatory testing PHT-IHX-CCF-BL	3.60e-10
4.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-TBYP-VMA-F-V1	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Electrohydraulic valve of turbine bypass system fail to function	1.08e-10

No.	Cut-set	Event description	Frequency
		(FO+FRO)-3units	
5.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-TB-VMA-F-V2	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Electro hydraulic valve of turbine branch fail to function (FC+FRC)-3 units	1.08e-10
6.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-TBYP-VMA-F-013	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Motor operated valve of OGDHRS fail to function (FO+FRO)	5.42e-11
7.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-TB-VMA-F-009	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Motor operated valve of SWS branch fail to function (FC+FRC)	5.42e-11
8.	ONE-DG-FAIL. HETHC0002. HETHC0001	Failure of One DG during mandatory testing Human error in threshold setting Human Error in threshold setting	5.00e-11
9.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-DHRCS-PL-COOLANT	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Pipe line leakage	4.33e-11
10.	ONE-DG-FAIL. HETHC0002. OPACF0001	Failure of One DG during mandatory testing Human error in threshold setting P/Q computing element CCF	4.00e-11
11.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. LOSP-4H	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail Loss of off site power supply	3.25e-11
12.	ONE-DG-FAIL. STACK-CCF-LF. FUN4/4L-FAIL. SWC-TBYP-VMA-F-V1	Failure of One DG during mandatory testing STACK-CCF-LF Functional failure when 4/4 loops fail Electrohydraulic valve of turbine bypass system fail to function (FO+FRO)-3units	3.24e-11
13.	ONE-DG-FAIL. STACK-CCF-LF. FUN4/4L-FAIL. SWC-TB-VMA-F-V2	Failure of One DG during mandatory testing STACK-CCF-LF Functional failure when 4/4 loops fail Electro hydraulic valve of turbine branch fail to function (FC+FRC)-3 units	3.24e-11
14.	ONE-DG-FAIL. AIR-DMP-CCF-MECH. AIR-XDM-	Failure of One DG during mandatory testing	3.00e-11

No.	Cut-set	Event description	Frequency
	FO-XHE. FUN4/4L-FAIL. SWC-TBYP-VMA-F-V1	AIR-DMP-CCF-MECH fail to manually open damper Functional failure when 4/4 loops fail Electrohydraulic valve of turbine bypass system fail to function (FO+FRO)-3units	
15.	ONE-DG-FAIL. AIR-DMP-CCF-MECH. AIR-XDM-FO-XHE. FUN4/4L-FAIL. SWC-TB-VMA-F-V2	Failure of One DG during mandatory testing AIR-DMP-CCF-MECH fail to manually open damper Functional failure when 4/4 loops fail Electro hydraulic valve of turbine branch fail to function (FC+FRC)-3 units	3.00e-11
16.	ONE-DG-FAIL. HETHC0002. FLXCF0001	Failure of One DG during mandatory testing Human error in threshold setting Neutron sensor CCF	2.80e-11
17.	ONE-DG-FAIL. THCCF0002. HETHC0001	Failure of One DG during mandatory testing Thermocouple CCF Human Error in threshold setting	2.50e-11
18.	ONE-DG-FAIL. HETHC0002. SLFNO0001	Failure of One DG during mandatory testing Human error in threshold setting SLFIT failure	2.25e-11
19.	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL. FUN4/4L-FAIL. SWC-HP-SL-BR	Failure of One DG during mandatory testing INT-PSF-CCF-LK-ALL Functional failure when 4/4 loops fail High Pressure Steam Line Break	2.16e-11
20.	ONE-DG-FAIL. THCCF0002. OPACF0001	Failure of One DG during mandatory testing Thermocouple CCF P/Q computing element CCF	2.00e-11

4.13 Summary

In this study reliability analysis of class III power supply system has been performed. Support systems, namely safety related service water system, fuel oil system and circuit breaker control power supply dependency have been modeled. Unavailability has been calculated for two success criteria (2/4 DG success and 1/4 (DG success). Common cause failure has been estimated using beta factor and alpha factor model. Importance analysis and sensitivity study are used to identify significant contributors to unavailability. Fussell-Vesely, Birnbaum, Barlow-

Proschan and Sequential importance measures have been evaluated. The result of sensitivity analysis indicates the relative importance of individual component failure rates. Further, uncertainty analysis has been carried to determine confidence bound of unavailability. It has been performed through Monte Carlo simulations. The combination of the lognormal distribution and Monte Carlo simulations can be used to give the overall shape and high confidence bounds for particular percentiles of interest. Failure of one DG during mandatory testing (0.5/year) has been taken as initiating event frequency for computation of failure frequency. Failure frequency contribution of Class III power supply due to internal events is 1.73E-8/ry.

Chapter 5
External Events

External events analysis covers procedure for identification, categorization, screening analysis, quantification, and PSA modeling of External events. This analysis covers Hazard analysis, Plant-system and structure response analysis, evaluation of the fragility and vulnerability of components, Plant-system and sequence analysis and consequence analysis. External hazards have been further divided into three parts. First part deals with seismic events. Second part is flooding events. It includes storm surge, rainfall and tsunami. All other events have been described in third part. The other events include Wind hazard, aircraft crash, lighting and missile protection. This chapter computes failure frequency of class III power supply system arising from external Event, which is summation of failure frequency due to seismic events, flooding events and other events.

5.1 Seismic Events

The Seismic Probabilistic Safety Analysis (SPSA) Level-1 for DG of PFBR aims in estimating earthquake initiated failure frequency for class III power supply system. The method consists of three major parts, viz., (a) assessing the seismic hazard at the ground level, (b)assessment of the seismic fragility of various safety systems, structures and components and (c) integration of seismic hazard with fragility information through appropriate logic models of plant safety functions [26,43].

5.1.1 Seismic hazard

Seismic hazard at a site is represented by hazard maps representing ground motion parameters such as peak ground acceleration (PGA), velocity (PGV) or displacement (PGD) and their corresponding non-exceedance frequency. Unlike the Richter and Moment magnitude scales, PGA is not a measure of the total energy of an earthquake, but rather of how hard the earth shakes in a given geographic area [44]. The damage to structures is known to correlate with PGA for medium magnitude earthquakes and with PGV for high magnitude earthquakes. The damage also depends on the duration of oscillation. PGA is not a measure of ground shaking but is considered relevant and adequate for earthquake engineering purposes. The hazard also could be represented in terms of the more detailed measure known as spectral acceleration, i.e., acceleration at a given frequency or over a range of frequencies of interest [45]. The methods of assessing seismic hazard at a site consists of enumerating potential sources from historical records which are within a distance of direct influence (< 300 km) from historical data and seismotectonic studies and then applying attenuation characteristics of the intervening path [45].

For PFBR, Operation Base Earthquake (OBE) with PHA of 0.078 g should have the frequency less than 10^{-2}. For Safe Shutdown Earthquake (SSE), the PHA considered in the design is 0.156 g and its frequency should be less than 10^{-4} so that it is a category-4 event [86].

5.1.2 Seismic hazard modeling for Kalpakkam

Seismic hazard at a nuclear power plant site can be represented by a hazard curve, which is a plot of annual frequency of exceedance against peak ground acceleration (PGA). It is used as an input in the seismic PSA. Because of the large uncertainty in the seismic hazard analysis, a family of hazard curves is usually developed with different confidence levels, such as 5%, 50%, 95%, etc. This is called a probabilistic seismic hazard analysis (PSHA) and involves the following four steps (Jinkai Wang & Modi Lin, 2018): (i) seismic source characterization and assessment (ii) earthquake recurrence relationship modeling (iii) ground motion attenuation relationship analysis (iv) determination and presentation of hazard curves.

In the process of seismic hazard evaluation, modeling of location/size/incidence of earthquakes that may occur in the vicinity of the site is conducted using active fault data and historical earthquake data. Then the propagation of seismic ground motion caused by the earthquakes is evaluated based on a distance attenuation model or fault model in order to obtain the relationship between the strength of seismic ground motion and the exceedance probability. This analysis is based on historical earthquake reports and instrumental records as well as the geology of the region. Earthquake data for the present analysis is for the period 1504-2001 AD, which was obtained from various catalogues available as published literature (Chandra U. 1977, for an example) [87] and data from Gauribidanur Seismic Array (GBA) of Bhabha Atomic Research Centre for the period 1977-1995 AD [88]. These data have been consolidated as BARC report [89]. Earthquakes, whose moment magnitude Mw ≥3.0 alone has been considered for this analysis. Earthquakes less than this magnitude are unlikely to cause structural damage so they have not been considered. There are studies which consider only earthquakes with moment magnitude ≥4.0 [90]. There are a total of 271 earthquakes with Mw ≥3.0 reported in [91]. Out of this 64 are from global sources and 207 are from GBA.

5.1.3 Regional recurrence relation

Earthquake occurrence is represented as a random process. Even though several models are available for this purpose, the most widely used model is the Poisson process which assumes time independence of earthquake occurrence. If Poisson process is used, the aftershocks are normally removed from the catalogue. The seismic activity of a region is characterized by the Gutenberg and Richter earthquake recurrence law. This is given by equation (5-1),

$$Log_{10} N = a - bM \tag{5-1}$$

where N is the total number of earthquakes of magnitude M and above in a year, a and b are the seismic parameters of the region. The parameters a and b are calculated in various studies which are given in Table 5.1.1.

Table 5.1.1: Calculated Values of 'a' and 'b' in various studies

Sl. No	a	b	Reference
1.	2.35	0.5989	Global sources(Ghosh A.K, and Rao K.S 2009) [89]
2.	3.28	0.78	GBA(Ghosh A.K, and Rao K.S 2009) [89]
3.	4.58 (±0.4)	0.891 (±0.07)	Vipin K.S et al 2009 [91]
4.	5.30	0.81	Ram.A and Rathor H.S 1970 [92]
5.	3.25	0.70	Kaila K.L,1972 [93]
6.	4.40	0.85	Ramalingeswara Rao B and Sitapathi Rao P. 1984 [94]

From equation (5-1), it is clear that the parameter b represents the slope of the straight line fit in the semi-log graph. This represents the variation of number of earthquakes (exceeding a specified magnitude) with magnitude. This parameter plays an important role in subsequent calculations. In the present study the value of b used is 0.5989 reported in [89]. The use of this slope gives a slightly conservative estimate of number of earth quakes on the higher magnitude side. As per McGuire [44], one of the important considerations in estimating rates of occurrence and b-values from historical data are the effects of uncertainty in the estimates. . It is suggested to use a recurrence rate estimates a magnitude m* for each earthquake, that is

$$m^* = \overline{m} + 0.5\,\beta\,\sigma_m^2 \tag{5-2}$$

This accounts both for the effects of uncertainty in magnitude and for the slope of the magnitude distribution. In the present analysis these uncertainties are not considered.

5.1.4 Identification and characterization of seismic sources

The study area is 300 km radius with Kalpakkam as centre. Consideration of 300 km radius with site of interest as centre is the practice normally followed in Seismic Hazard Analysis. The earthquake data is considered for this area only. The seismic sources in the study area were identified from online seismotectonic atlas available in Geological Survey of India website (Geological Survey of India Web Portal). From this atlas a total of 28 seismic sources have been identified and used for further analysis. The approximate shortest distance of each source from Kalpakkam is measured using online tools. The length of these sources and maximum magnitude of earthquake that has occurred in the specific fault surrounding are obtained from reference [91]. Table 5.1.2. lists the approximate fault parameters for all the 28 faults that are used in this analysis.

Table 5.1.2: Details of Fault Parameters

Fault No.*	Distance (km)	M_{max}	Length (km)	Fault No.	Distance (km)	M_{max}	Length (km)
1.	72	4.4	150	15.	296	4.9	75
2.	143	5.2	176	16.	219	4.7	61
3.	149	4.5	95	17.	265	4.7	121
4.	151	4.8	592	18.	146	4.2	143
5.#	16	4.6	78	19.	252	5.2	286
6.#	76	4.5	204	20.	245	5.7	283
7.	84	4.3	385	21.	278	4.3	120
8.#	132	5.0	107	22.	251	4.8	88
9.#	215	4.6	506	23.	264	6.3	457
10.	138	4.8	114	24.#	29	4.3	120
11.	128	4.8	115	25.#	128	4.3	121
12.	166	5.6	100	26.	71	4.5	151
13.#	181	5.0	118	27.	94	5.0	143
14.	133	5.2	146	28.	132	4.9	54

*- They are user defined, which does not match with actual fault numbers used by GSI

#- Active faults as in (Geological Survey of India Web Portal [95]

Since Kalpakkam is located in peninsular India, the attenuation relation (for peak ground acceleration and spectral acceleration) developed for peninsular India [45] is used in this analysis. The attenuation relationship is given in equation (5-3) as

$$\ln(y) = c_1 + c_2(M-6) + c_3(M-6)^2 - \ln(R) - c_4R + \ln(\in) \qquad (5\text{-}3)$$

where,

y – Peak Ground Acceleration (PGA) / Spectral Acceleration (SA) (g)

M – Moment magnitude

R- Hypocentral distance

ε – Error associated with regression

c_1=1.7816; c_2=0.9205; c_3=-0.0673; c_4= 0.0035; σ(ln ε)=0.3136

This relationship calculates PGA at bedrock level. Due to variations in local site conditions, the surface level PGA could be different from bedrock level. Techniques are available for estimating surface level PGA by considering local site effects. But in the present analysis PGA at bedrock level alone are considered. In equation (5-3) ln (y) is normally distributed and y is log normally distributed [45]. Even though the depth of occurrence of an earthquake is a random variable, in the present analysis the depth is assumed to be 15 km for all sources. This is used to calculate the hypocentral distance.

For evaluating seismic hazard for Kalpakkam site based on the above mentioned details, a set of programs were developed. The general flow chart for the analysis is given in Fig. 5.1.1.

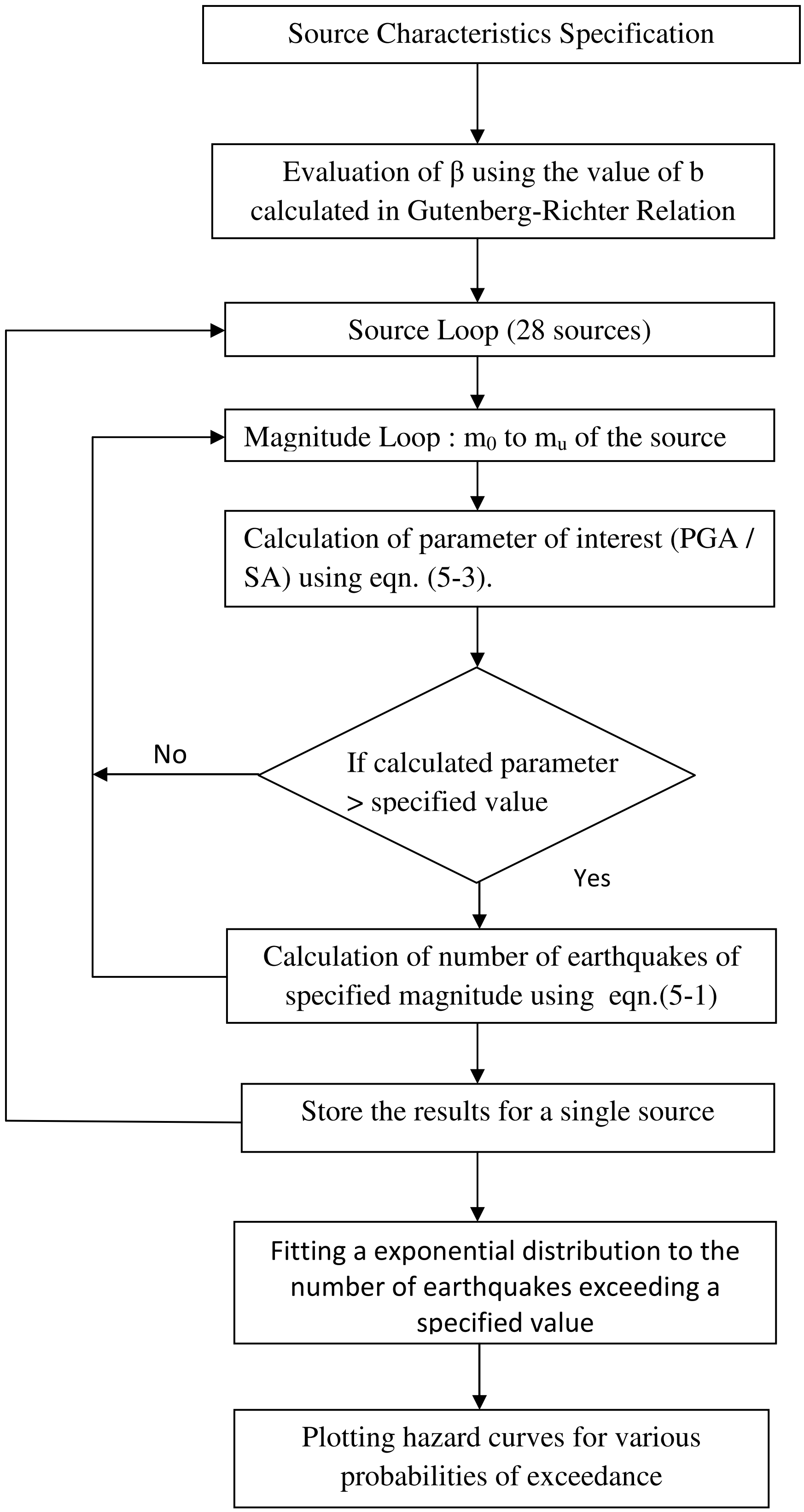

Fig. 5.1.1. General flow chart for seismic hazard analysis programs

The seismic hazard curves are generated for Kalpakkam site in terms of annual frequency of exceedance versus PGA. The three curves correspond respectively to 5%, 50% and 95% probability of exceedance in 50 years and are shown in Fig 5.1.2.

From Fig. 5.1.2, it is observed that the hazard curve is commensurate with the deterministic seismic design category levels at 5% exceedance probability value. For PFBR, Operation Base Earthquake (OBE) with PHA of 0.078g should have the frequency less than 10^{-2}. From Fig. 5.1.2 the value for that PHA is 7.0E-04 for 5% exceedance probability. For Safe Shutdown Earthquake (SSE), the PHA considered for PFBR design is 0.156g and its frequency should be less than 10^{-4} so that it is a category-4 event. From Fig. 5.1.2, at 5% probability of exceedance the frequency is ~1.0 E-04.

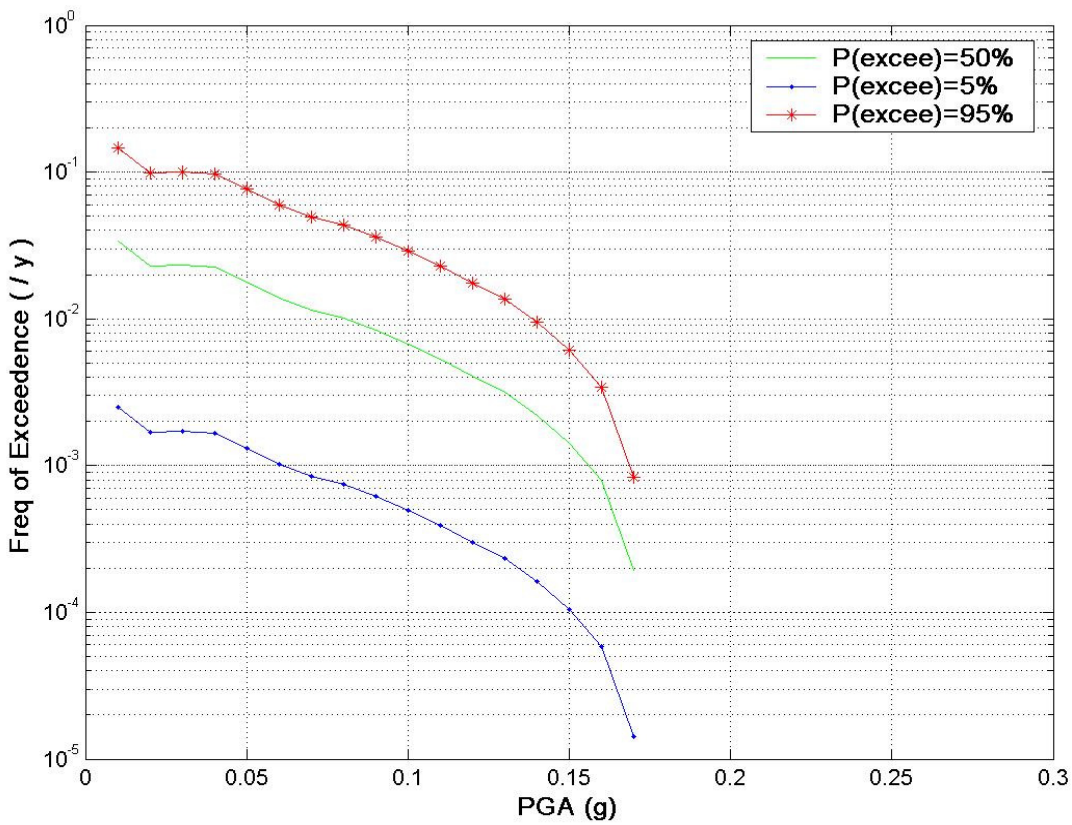

Fig. 5.1.2. PGA (g) Vs Frequency of Exceedance

5.1.5 Seismic Fragility Analysis

Component selection

Seismic fragility is the conditional failure of SSC, for a given seismic load. Seismic fragility analysis of safety relevant components (safety systems and support systems that has significant contribution to risk) is in principle to be done by experiments or by engineering analysis (with data and methods validated by experiments). As this form of precise analysis is very expensive and time consuming for hundreds of components, a number of methods starting from generic fragility data to methods that use seismic margin information and seismic test results are in use. The approach used for selecting the method is based on a number of considerations, like scope, level of detail of the study, plant specificity or fidelity and realism as enumerated in the PSA standard ASME /ANS RA-S-2008. The basic event data used for Class III power supply system and acceleration capacity is shown in Table 5.1.3.

Table 5.1.3: Basic event fragility analysis

S.No.	Name	Description	Safety Class	Am	BR	BU	Remarks	Range
1	CL3-415V-F-TRANS	6.6kv / 415V Transformer failure (Support)	4	2.505	0.32	0.46	Generic transformers data	0.3-5.80
2	CL3-415V-F-CT	415V Cable Tray failure (Support)	2	3.147	0.37	0.52	Generic Cable Trays data	1.10-5.8
3	CL3-415V-F-INPA	Instrumentation and Control Panel of 415V fails	4	0.4696	0.35	0.32	Calculated from test data	NA
4	CL3-415V-F-DB	415V Distribution Board Failure	2	4.987	0.35	0.32	Generic electric cabinets data	2.77-7.60
5	CL3-6.6KV-F-DGB	Collapse of DG Building (Shear Failure)	3	5.5	0.36	0.35	Concrete data	2.5-9.2
6	CL3-6.6KV-F-CT	6.6 KV cable tray failure (Support)	2	3.147	0.37	0.52	Generic Cable Trays data	1.10-5.8
7	CL3-6.6KV-F-DB	Distribution Board of 6.6 KV fails	2	4.987	0.35	0.32	Generic electric cabinets data	2.77-7.60
8	CL3-6.6KV-F-DG	Diesel Generator Failure	3	2.084	0.27	0.37	Generic DG data	0.70-3.89
9	CL3-6.6KV-F-FT	DG Fuel Tank Failure		2.343	0.27	0.37	Generic data for other tanks and vessels	1.07-3.91
10	CL3-6.6KV-	DG Fuel Pump Failure		3.653	0.33	0.3	Generic other pumps data	2.10-5.47

	F-FP							
11	CL3-6.6KV-F-FPIP	Fuel Pipe line failure		7.325	0.29	0.54	Generic pipe data	2.50-13.6
12	CL3-6.6KV-F-DGINPA	DG Instrumentation Panel failure		0.4696	0.35	0.32	Calculated from test data	NA
13	CL3-6.6KV-F-STCOMP	Start air compressor/air bottle failure		3.057	0.3	0.44	Generic HVAC data is used	1.10-5.58
14	CL3-6.6KV-F-JCPIP	Jacket cooling pipeline failure		3.429	0.29	0.54	med-service water generic pipe data	2.80-4.10
15	CL3-6.6KV-F-SSWSHX	SSWS Heat Exchanger failure		5.224	0.29	0.41	Generic heat exchanger data	0.3-13.0

5.1.6 Fragility Estimation

As described in the PRA procedures guide [26], there are two approaches for evaluating seismic fragilities : i) the zion method wherein the fragility is expressed as a function of a global ground motion parameter (e-g: peak ground acceleration) and ii) Seismic Safety Margins Research Program (SSMRP) method, which defines the fragility in terms of a local response parameter. In the present analysis, fragility analysis is done by zion method.

Zion method

The entire fragility family for a component corresponding to a particular failure mode can be expressed in terms of the best estimate of the median ground acceleration capacity A_m and two random variables [46]. Thus the ground acceleration capacity, A is given by:

$$A = A_m \epsilon_R \epsilon_U \tag{5-4}$$

in which ε_R and ε_U are random variables with unit medians. They represent respectively, the inherent randomness about the median and the uncertainty in the median value. In this model it is assumed that both ε_R and ε_U are lognormally distributed with logarithmic standard deviations β_R and β_U respectively. The formulation for fragility given by eqn (5-4) and the assumption of lognormal distribution enable easy development of the family of fragility curves appropriately representing their uncertainty. For the quantification of fault trees in the plant system and sequence analysis, the uncertainty in fragility needs to be expressed in terms of a range of failure frequencies for a given ground acceleration.

When only random uncertainty is present (β_R), the conditional frequency of failure f_0 for given peak ground acceleration level, a, is given by

$$f_0 = \varphi\left[\frac{\ln\left(\frac{a}{A_m}\right)}{\beta_R}\right] \tag{5-5}$$

where φ ( ) is the standard Gaussian cumulative distribution function.

The frequency of failure f' at any non-exceedence probability level Q can be derived as,

$$f = \varphi\left(\frac{\ln\left(\frac{a}{A_m}\right)+\beta_U\varphi^{-1}(Q)}{\beta_R}\right) \tag{5-6}$$

where Q=P [F<fla] - probability that the conditional frequency of failure F, is less than f for a peak ground acceleration a, and φ^{-1}( ) is the inverse of standard Gaussian cumulative distribution function. In some applications, the composite variability β_c is used which is defined by

$$\beta_c = \sqrt{\beta_R^2 + \beta_U^2} \tag{5-7}$$

The use of β_C and A_m provides a single "best estimate" fragility curve which does not explicitly separate out uncertainty from underlying randomness.

<u>Failure frequency estimation</u>

Failure frequency has been quantified for different PGA values. The conditional failure of class III power supply under seismic events for the given PGA is shown in Table 5.1.5. The failure probability vs PGA (g) for freq (PGA>a) is shown in Fig.5.1.3. The Log (failure probability) vs PGA (g) for freq (PGA>a) is shown in Fig.5.1.5.

Table 5.1.4: Failure Frequency for freq (PGA>a)

			Frequency (pga>a)	
PGA(g)	P	log(P)	F	P*F
0.05	1.79E-11	-10.747147	0.0013	2.33E-14

0.078	3.29E-07	-6.4828041	0.0008	2.63E-10
0.1	2.53E-05	-4.59687948	0.0005	1.27E-08
0.15	5.37E-03	-2.27002571	0.0001	5.37E-07
0.156	8.06E-03	-2.09366496	0.0001	8.06E-07
0.2	6.87E-02	-1.16304326	0	0.00E+00
0.25	2.55E-01	-0.59345982	0	0.00E+00
0.3	5.06E-01	-0.29584948	0	0.00E+00
0.4	8.60E-01	-0.06550155	0	0.00E+00
0.5	9.72E-01	-0.01233374	0	0.00E+00
0.6	9.95E-01	-0.00217692	0	0.00E+00
0.7	9.99E-01	-0.00043451	0	0.00E+00
0.8	1.00E+00	0	0	0.00E+00
0.9	1.00E+00	0	0	0.00E+00
1	1.00E+00	0	0	0.00E+00
			Total Freq.	**1.36E-06**

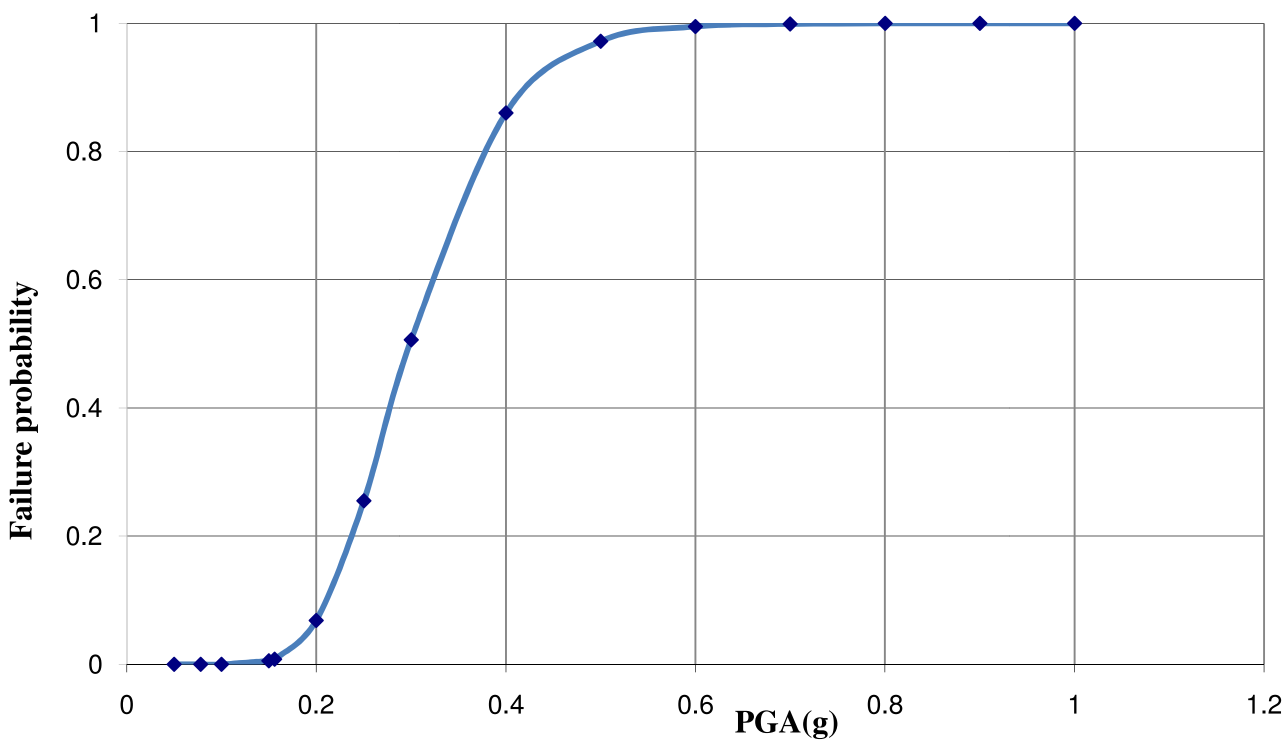

Fig. 5.1.3: Failure probability vs PGA (g) for freq (PGA>a)

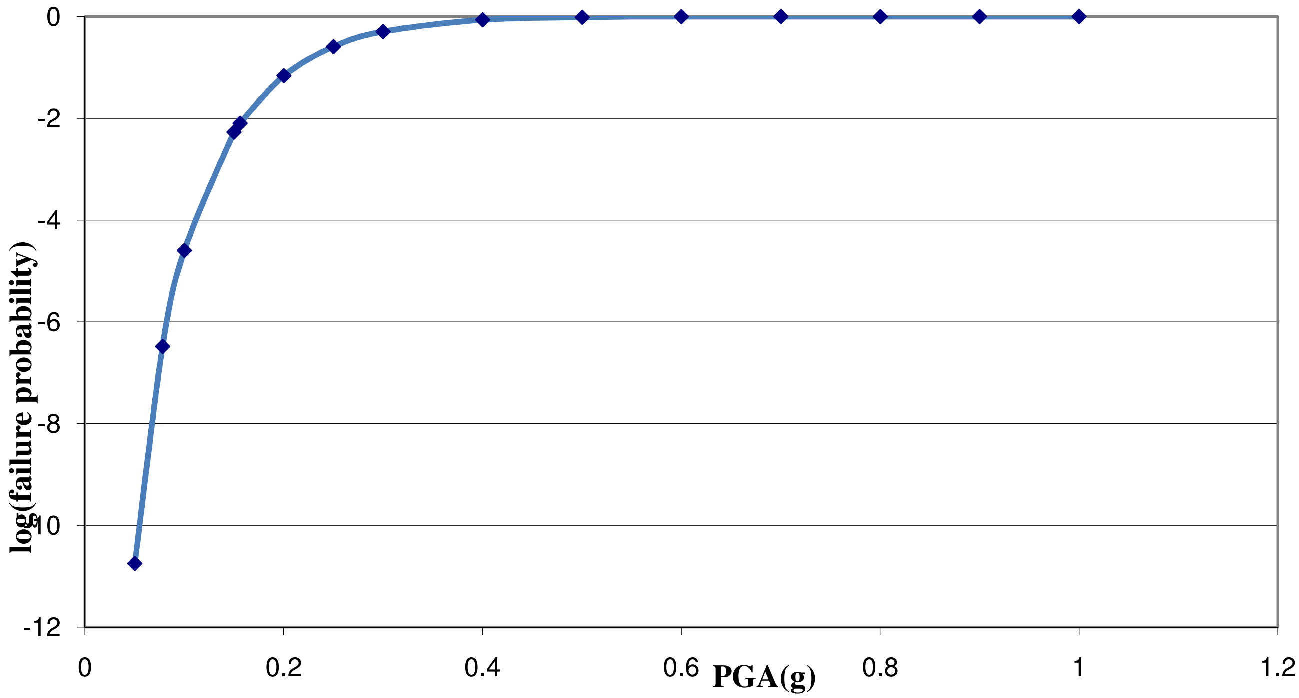

Fig 5.1.4: Log (Failure probability) vs PGA (g) for freq (PGA>a)

5.1.7 Seismic Fault Tree Analysis

The plant safety function logic is captured by fault tree and event tree models as appropriate for seismic PSA. The fault trees are system level fault tree which are different from that used in the internal event PSA, as the component failure modes are different and common cause (dependent) failures are expected to be dominant.

<u>Assumptions in Fault Tree Analysis for Seismic hazard</u>

The following assumptions are made in the development of the fault tree.

(i) The detailed modeling of systems upto the component level is not required for Seismic PSA fault tree. So the components are modeled upto the block level that is at the cabinet/ panel or subsystem level.

(ii) The redundancy at train or component level is not considered in this analysis as most of the redundant trains or components have identical acceleration capacities. If there is a seismic specific redundancy then the train or component with highest acceleration capacity will be modeled e-g: DSR of Shut Down System (SDS).

(iii) The Human Error Probability (HEP) is assumed to be one in a seismic scenario.

(iv) During a seismic event, the unavailability of offsite power is assumed to be one. This is also a conservative assumption.

(v) IFTM, Fuel Transfer Arm and inner vessel have been assumed not to contribute significantly to Core Damage Frequency (CDF).

Fault tree

The FT for seismic hazard analysis of DG is shown in Fig. 5.1.5.

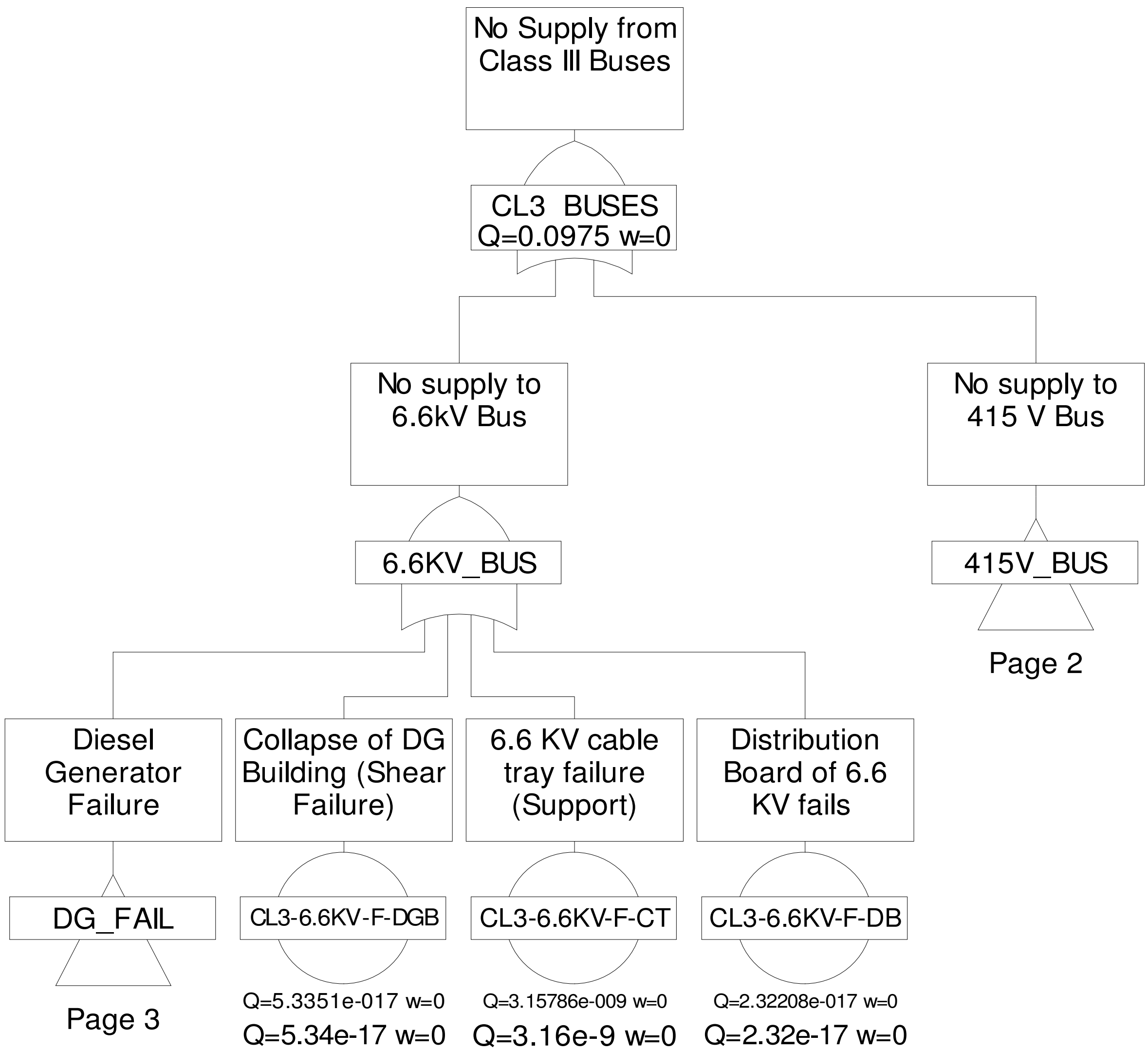

Fig. 5.1.5: Seismic FT of Class III Power Supply System

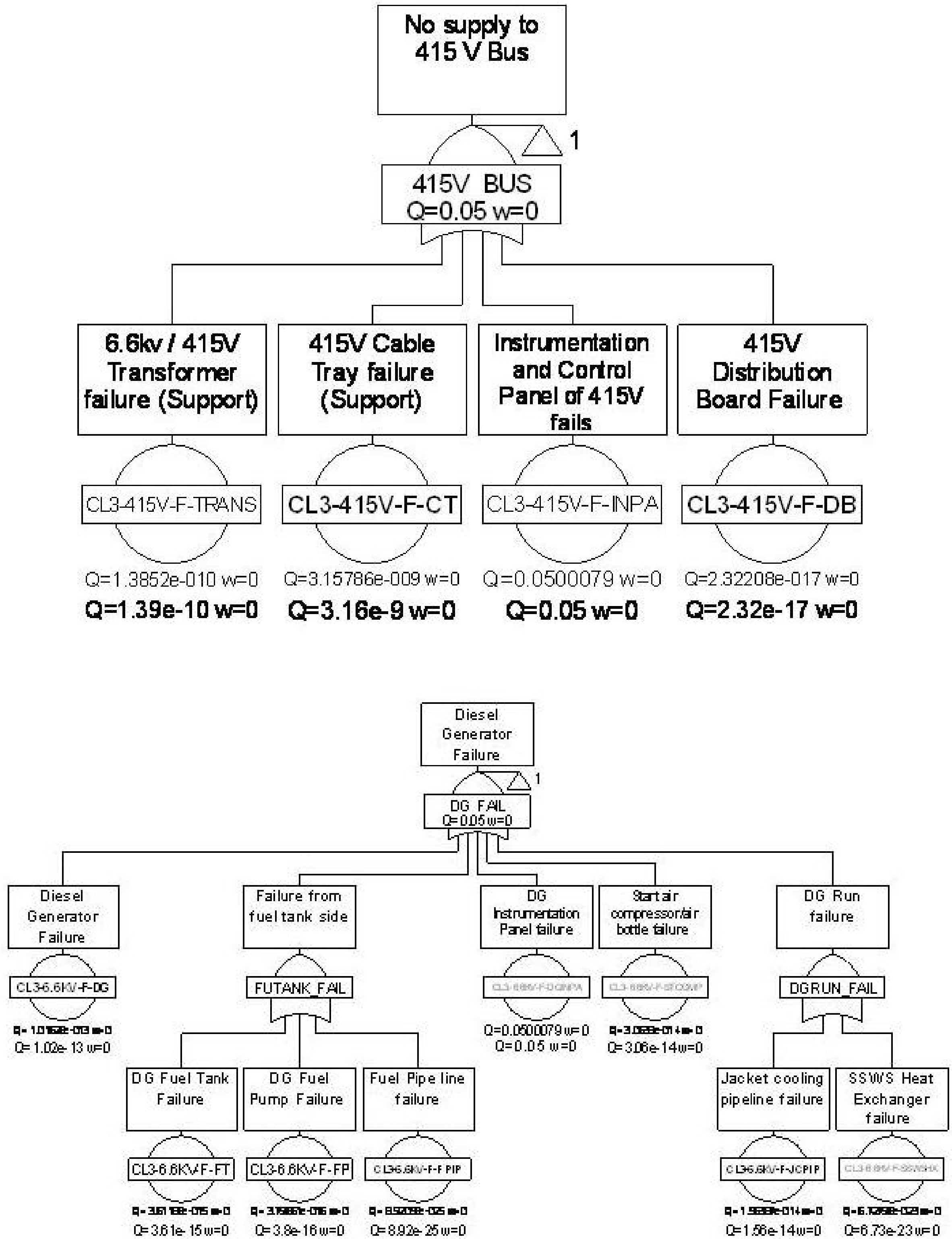

Fig. 5.1.5: Seismic FT of Class III Power Supply System (Continued.)

Fault tree cutsets

The minimal cutsets for seismic hazard is shown in Table 5.1.4.

Table 5.1.5: Cut sets for seismic events

No.	Cut set	Unavailability
1	CL3-6.6KV-F-DGINPA	5.00e-2
2	CL3-415V-F-INPA	5.00e-2
3	CL3-415V-F-CT	3.16e-9
4	CL3-6.6KV-F-CT	3.16e-9
5	CL3-415V-F-TRANS	1.39e-10
6	CL3-6.6KV-F-DG	1.02e-13
7	CL3-6.6KV-F-STCOMP	3.06e-14
8	CL3-6.6KV-F-JCPIP	1.56e-14
9	CL3-6.6KV-F-FT	3.61e-15
10	CL3-6.6KV-F-FP	3.80e-16
11	CL3-6.6KV-F-DGB	5.34e-17
12	CL3-6.6KV-F-DB	2.32e-17
13	CL3-415V-F-DB	2.32e-17
14	CL3-6.6KV-F-SSWSHX	6.73e-23
15	CL3-6.6KV-F-FPIP	8.92e-25

5.1.8 Results

The methodology of seismic risk assessment has been applied to estimate the failure frequency of class III due to seismic events. Detailed probabilistic seismic hazard analysis has been carried out for the site. Fragility analysis of various safety related equipment has been done mostly with generic median capacity. Detailed plant logic models, suitable for seismic risk analysis has been developed. The analysis indicates that the significant contribution to core damage frequency from seismic hazard input is from 0.1 g to 0.25 g and above. The hazard region above 0.2 g has large uncertainty. The obtained total frequency for class III power supply failure due to seismic events is 1.36E-06/ry.

5.1.9 Summary

The seismic PSA of Class III system of PFBR has been performed. The methodology consists of three major parts, viz., assessing the seismic hazard at the ground level, assessment of the seismic fragility of various safety systems, structures and components and integration of seismic hazard with fragility information through appropriate logic models of plant safety functions. The scope of analysis is confined to direct effects of seismic events, neglecting secondary effects. Level of detail of the analysis is restricted to capability category I as listed in ASME standard. For the first part, probabilistic seismic hazard analysis for the site, detailed study has been carried out to arrive at the peak ground acceleration at the site as a function of occurrence frequency. For fragility analysis and median seismic capacity assessment, analysis is mostly based on generic data as applicable to PFBR. For the third part, i.e., for integrating seismic hazard and fragility information detailed plant logic models in the form of Fault Trees has been developed. Apart from producing failure frequency estimates, the analysis identifies range of seismic inputs that has dominant impact on the failure frequency and identifies components for detailed engineering fragility analysis. The obtained total frequency for class III power supply failure due to seismic events is 1.36E-06/ry.

5.2: Flooding Events

In comparison to other external event PSA like earthquake, flood PSA received less attention over the years due to the prevailing perception that floods are less likely than fires and earthquake to induce accidents contributing significantly to the overall risk from a nuclear power plant. The perception has also been supported by the facts that

- The flood protection measures provided for NPP results in low frequencies of flood that cause serious damage.
- The availability of significant amount of warning time to safely shutdown the reactor before significant damage to important SSCs.
- Unlike earthquakes whose effects are felt even at low intensities, the effect of flood is perceived only after the water level crosses a particular height.

However there are several reasons to consider flood as an important risk contributor in PSA studies, an important one being large uncertainties in the estimated frequencies of external floods and in the associated plant fragilities [54]. It was considered that internal floods have a relatively greater potential to cause a reactor accident [52]. Recent flooding events have altered this thought to some extent. Incidents like the storm induced flooding at Le Blayais NPP, France [49], flood events at Fort Calhoun NPP, USA [50] and the tsunami at Fukushima Daiichi, Japan [51] have pointed to the importance of external flooding as an important contributor to NPP risk.

The Fukushima accidents highlighted the potential of external flooding to cause core damage of NPPs. Identifying weak links of a plant during a flooding event is an important step towards enhanced safety of NPPs. PSA for external events to identify cliff edge effects has been an important recommendations post Fukushima [96] Further, improved safety guidelines are being formulated for future reactors with more emphasis on external flooding events based on the experience of Fukushima accident [97]. Which are a) Seek out and act on new information about hazards b) Improve nuclear plant systems, resources, and training to enable effective ad hoc responses to severe accidents c) Strengthen capabilities for assessing risks from beyond-design-basis events d) Further incorporate modern risk concepts into nuclear safety regulations e) Examine offsite emergency response capabilities and make necessary improvements f) Improve the nuclear safety culture. Thus, a comprehensive safety assessment taking into account all possible internal and external events is necessary to address it. Considering these aspects, External Flooding PSA (EFPSA) of class III power supply system of PFBR has been performed. The external flooding phenomena that need to be considered include

- Ocean flooding from storm surge including wind induced waves
- Ocean flooding from tsunami
- Flooding from heavy precipitation

5.2.1: Storm Surge Hazard Analysis

5.2.1.1 Introduction

Storm surge is the increase in sea level due to severe cyclonic storm events. Since, these events have the potential to inundate large parts of land in coastal areas, it is necessary to study the maximum sea level during a cyclonic event in order to assess the impact of a cyclonic event on coastal nuclear plants. The storm level is based on storm surge hazard study reported for Chennai based on hourly tide gauge data [58, 59] and a specific study performed for the Kalpakkam site [57]. The objective of this analysis is to estimate the frequency of exceedance for various sea levels. This hazard is one of the inputs to evaluate the risk in addition to the contributions from hazards like tsunami and rainfall.

5.2.1.2 Methodology

The methodology for evaluating storm surge and return periods are illustrated in references [58, 59]. The analysis is generally carried out by two different methods. The first method is a physical method in which a non-linear hydrodynamic model was developed for storm surge event. The second method is a statistical one which is based on extreme value analysis of observed maximum sea level during a storm surge event. Extreme Value Analysis has been used in this study to model return periods. The risks associated with extreme events can be assessed from the estimates of return periods and return levels. The return period is defined as the inverse of the exceedance probability of an event. The methods used for the estimation of exceedance probability are i)the annual maximum method ii) Joint probability method iii) Revised joint probability method (RJPM) and iv) Exceedance probability method. In reference [58, 59], the r-largest annual maxima method is used which is an extension of the widely used classical method of annual maximum of Gumbel. This method has been chosen as it is capable to use more than one value of extreme sea levels for each year, thereby allowing more reliable estimates of return periods with less number of years of data.

5.2.1.3 Data collection

Hourly tide gauge data from Chennai for a period from 1974-1988 is used for this analysis. From the hourly data the annual maximum sea level during a storm event is arrived at after appropriate filtering of data. In the reference [58, 59], the analysis was carried out by two different methods. The first method is a physical method in which a non-linear hydrodynamic model was developed for storm surge event. The second method is a statistical one which is based on extreme value analysis of observed maximum sea level during a storm surge event. In the present analysis the statistical approach is followed. The hourly tide gauge data is not reported in any of these references. The data is given in the form of a curve in[58, 59] with corresponding probabilities. For the present analysis, the maximum sea level and probability of exceedance extracted from the curves are used. The extracted data points are given in Table 5.2.1.1.

Table 5.2.1.1: Extracted Annual Maximum Sea Level and Probability of Exceedance from Observed Data

Sl. No	Observed Maximum sea level (m)	Probability of exceedance
1	1.78	4.72e-01
2	1.88	2.57e-01
3	1.95	1.50e-01
4	2.02	9.16e-02
5	2.08	5.73e-02
6	2.14	3.50e-02
7	2.20	2.09e-02
8	2.27	1.28e-02
9	2.32	8.00e-03
10	2.39	4.78e-03
11	2.45	2.92e-03
12	2.51	1.75e-03
13	2.58	1.04e-03
14	2.64	6.37e-04
15	2.70	3.72e-04
16	2.77	2.07e-04

5.2.1.4 Data Analysis

The validity of type I extreme value distribution which is also known as Gumbel distribution, to the extracted data given in Table 5.2.1.1 is to be checked. This is done by plotting -ln(-ln(cumulative probability)) against maximum sea level. The plot is shown in Figure 5.2.1.1. The curve is a straight line which shows that the extracted data follows Gumbel distribution.

5.2.1.5 Gumbel Distribution Fit to the Data

Having confirmed the validity of Gumbel distribution to the given data, the next step is to evaluate the distribution parameters. The Gumbel distribution is a two parameter distribution. The probability density function of Gumbel distribution is given by [54],

$$f(x|\mu,\sigma) = \frac{1}{\sigma} e^{-\frac{x-\mu}{\sigma}} \exp\left(-e^{-\frac{x-\mu}{\sigma}}\right) \qquad -\infty \le x \le \infty \qquad (5.2.1\text{-}1)$$

where μ is the location parameter and σ is the scale parameter. The data reported in Table 5.2.1.1 is fitted to a gumbel distribution with location and scale parameter as given in Table 5.2.1.2.

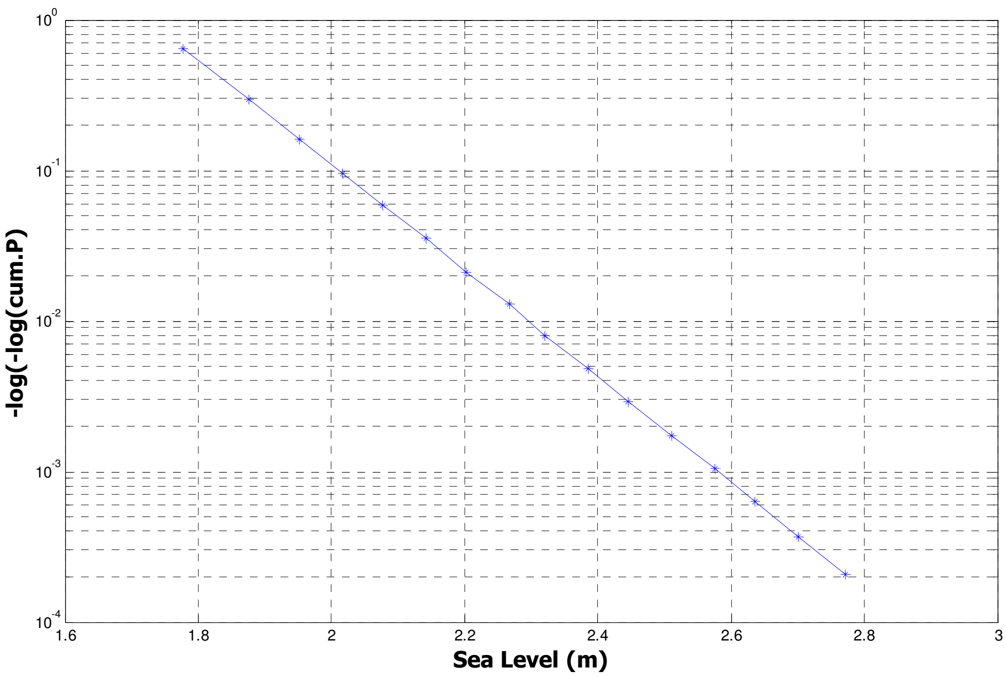

Fig 5.2.1.1: Plot of extracted sea level against -log(-log(cumulative probability))

Table 5.2.1.2: Parameters of the Gumbel fit to the data

Parameter	Mean Values	95% confidence Bounds
μ	1.72	[1.64,1.80]
σ	0.126	[0.066,0.186]

The mean values of the fitted parameters matches with the values reported by B. Sindhu and A.S.Unnikrishnan (2012) [59]. The 95% confidence bounds are estimated based on the reported error estimates given in [59]. The 95% confidence bounds are estimated using the expression

Bounds = mean of the parameter ± (2*Error reported for the parameter).

5.2.1.6 Results and Discussion

The Fig. 5.2.1.2 gives the maximum sea level using fitted distribution for various frequencies of exceedance. The 95% confidence bounds are also shown. Here the maximum sea level includes both the tide and surge levels. It is apparent from Fig. 5.2.1.2 that the fitted Gumbel distribution closely matches the observed sea level. The maximum sea level for Kalpakkam reported in [57] is 2.5 m for a 50 year return period. It is within the 95% confidence bounds. The fitted model predicts a median maximum sea level of ~2.3 m for a 100 year return period and ~2.58 m for 1000 year return period. The 95% upper bound maximum sea level predicted by the fitted model is ~2.7 m for a 100 year return period and ~3.1 m for 1000 year return period. The sea level reported is a combination of storm surge and the tide level at that time. The reported tide level at Chennai is approximately between 0.4 m and 1.0 m. All the levels indicated above are with reference to Mean Sea Level (MSL).

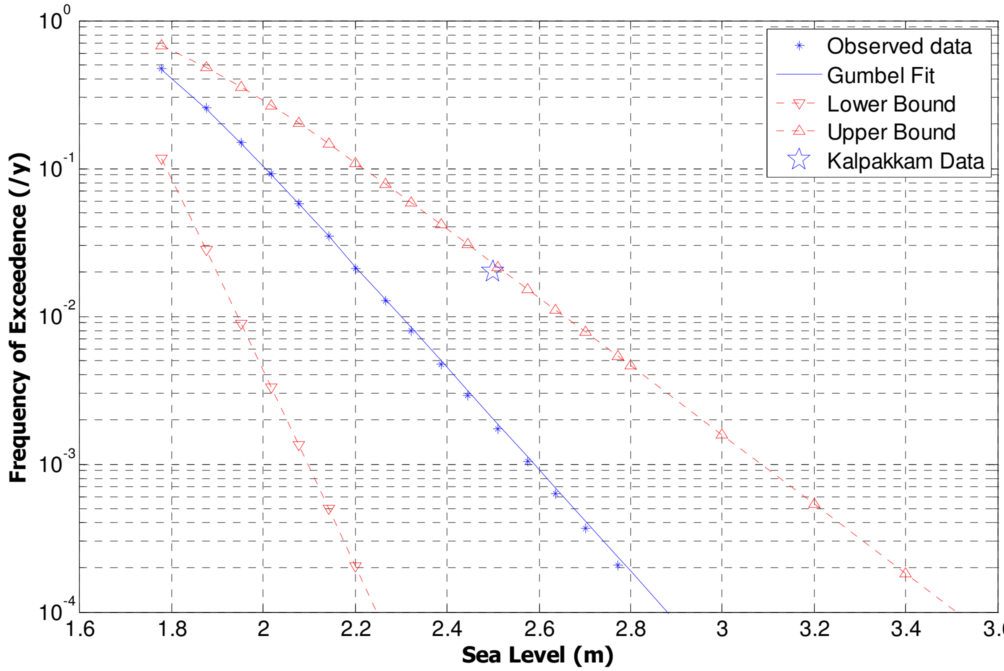

Fig 5.2.1.2 Comparison of Observed Maximum sea level and fitted distribution for Chennai

5.2.2: Rainfall hazard analysis

5.2.2.1 Introduction

Rainfall induced flooding occur time to time in India. Floods occur often in the region triggered by heavy monsoon precipitation and can cause enormous damages to lives, property, crops and infrastructure. Potential for damage to infrastructure due to flooding is great. The thresholds for flooding of built infrastructure are based mostly on the location of buildings and facilities in relation to flood extents/depths based on return period [98]. For this reason, flood hazard is assessed primarily on the micro scale location of critical buildings. Flooding depths at which buildings and infrastructure services could not continue to supply services without interruption can be predicted [99]. Here, risks on NPP due to rainfall hazard have been explored. Detecting shifts or changes in rainfall on the decadal or century scale is also an objective of rainfall hazard analysis. Based on the plant layout, catchment area and drainage capacity, flood hazard risk to NPPs has been determined, which is vital input for taking measures to mitigate its consequences.

The presence of water in many areas of the plant may be a reason for common cause failure for safety related systems, such as the emergency power supply systems or the electric switchyard, with the associated possibility of losing the external connection to the electrical power grid, the decay heat removal system and other vital systems. Considerable damage can also be caused to safety related structures, systems and components by the infiltration of water into internal areas of the plant, induced by high flood levels. Water pressure on walls and foundations may challenge their structural capacity. Deficiencies in the site drainage systems and non-waterproof structures may also contribute to flooding on the site. This has happened many times in the past, with consequent large-scale damage documented, and the possibility should be considered in the hazard evaluation and in the design of measures for site protection [52].

Flood level in an area of the site is determined by three factor namely (i) rainfall intensity, (ii) effective catchment area and (iii) effective drainage capacity. Meteorological parameters such as rainfall follow a seasonal cycle and the continuous survey of any meteorological parameter reveals annual extreme values. Projection of extreme values of environmental parameters likely to be encountered in the future using historically observed data is normally handled by extreme value statistical methods. Confidence levels of the statistically derived value depend on the size of the data as well as the data scatter with respect to fitted probability distribution function. Statistically, one can also find out the probability of non-exceedance of the value in terms of a mean recurrence interval (MRI).

The primary objective is to evaluate the rainfall flood hazard at Kalpakkam site. Stationary analysis has been performed for detecting trends in rainfall.

5.2.2.2 Methodology

Rainfall data forms an important input to estimate maximum water level at the proposed site. Rainfall intensity (annual maximum) such as rainfall rates in each hour (mm/h) and averaged over longer intervals of few days are needed for flooding analysis. The first type of data are used for designing the storm water drainage around the site while the other data are needed for generating design basis flood water level at inland sites which are often situated near a river course or dam. Although continuous recording rain gauge is preferred, in cases where the continuous measurement of rainfall data is not available, measurements carried out over discrete time intervals (i.e. one hour) is made use of to arrive at running average data for desired duration(e.g. 24 hours, if requires daily rainfall). An adjustment factor, which depends on the interval between successive measurements, will have to be applied to the observed sequential data set to arrive at the 24 hour running average rainfall data [100]. In many cases, due to non-availability of measured extreme values of 1 h data, this data is approximated with using 24hr data with empirical equations. For selection of the appropriate distribution, the measured parameters (i.e., extreme values of meteorological parameter like rainfall) are plotted on a plotting paper. In the analysis of extreme values, three types of asymptotic distribution are used. These distributions are known as Type I (Gumbel), Type II (Frechet) and Type III (Weibull) distribution [54]. The Gumbel distribution, also known as the *Extreme Value Type I* distribution, is unbounded (defined on the entire real axis), and has the cumulative density function

$$F(x) = exp\ \{-exp\ (-z)\},\ x \sim R; \qquad (5.2.2\text{-}1)$$

Where $z=(x-\mu)/\sigma$, μ is the location parameter, and σ is the distribution scale $(\sigma>0)$. The cumulative probability can also be written as (by taking log two times)

$$x=\mu-\sigma \ln [\{-\ln F(x)\}] \qquad (5.2.2\text{-}2)$$

Plotting x against -ln(-ln(F(x))) gives a straight line. This property enables a visual check to be made of the extent to which a data set fits the Gumbel distribution.

The Cumulative distribution function (CDF) of Generalized Extreme Value (GEV) distribution is

$$F(x)=\exp[-\{1+k(\frac{x-\mu}{\sigma})^{-1/k}\}] \qquad (5.2.2\text{-}3)$$

Where k is shape parameter.

The data set has also been plotted with Generalized Extreme value distribution.

Rainfall frequency is often estimated by power law fit and exponential distribution.

The cumulative distribution function of power law fit is

$$F(x)=ax^{b}, \quad x\sim R; \qquad (5.2.2\text{-}4)$$

And the cumulative distribution function of Exponential fit is

$$F(x)=ae^{bx}, \quad x\sim R; \qquad (5.2.2\text{-}5)$$

Plotting ln(x) against ln(F(x)) gives a straight line. This property enables a visual check to be made of the extent to which a data set fits the power law distribution.

Plotting x against ln(F(x)) gives a straight line. This property enables a visual check to be made of the extent to which a data set fits the exponential distribution.

5.2.2.3 Data Collection and Analysis

Rainfall data is gridded rainfall data obtained from IMD Pune for a period of 1901-2004AD (National Climate Centre, Pune) [101]. The gridded rainfall data has been collected over 1˚x1˚ longitude latitude high resolution daily rainfall data (24hrs) for the Indian Region. This corresponds to 111 km x 111 km at equator. Rainfall data is arranged in 35x33 grid points for Indian region. This represents a rectangular box in which city is located and rainfall data over various grids have been recorded. Fig.5.2.2.1 represents a grid in which Kalpakkam (*) is represented. The total number of rainfall station within this grid (Kalpakkam) is 15. From data, it is not clear that rainfall data used in the analysis is average over these rainfall station or maximum observed over a station. It is also not clear that which station in nearest to Kalpakkam. A FORTRAN program has been written to retrieve data for a particular grid. The data extracted is maximum rainfall in a single day (24 h) during entire year period. For 104 years a data set has been prepared, which is maximum daily rainfall for each year.

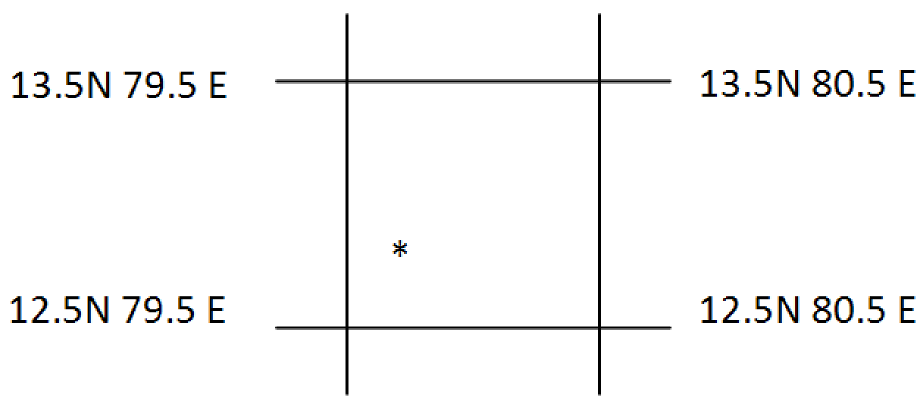

Fig 5.2.2.1: Kalpakkam (*) Latitude Longitude representation

These data points have been fit using Gumbel distribution and return period has been estimated. The Gringorten plotting position [102] has been used for the analysis. As per

Reference [54], correction factor 1.13 has been used. The parameters for Gumbel fit has been shown in Table 5.2.2.1.The frequency of exceedance for observed and predicted by the Gumbel fit for various rainfall levels are shown in Fig.5.2.2.2. The technique used for estimation of mean and standard deviation of Gumbel distribution is least squares fit.

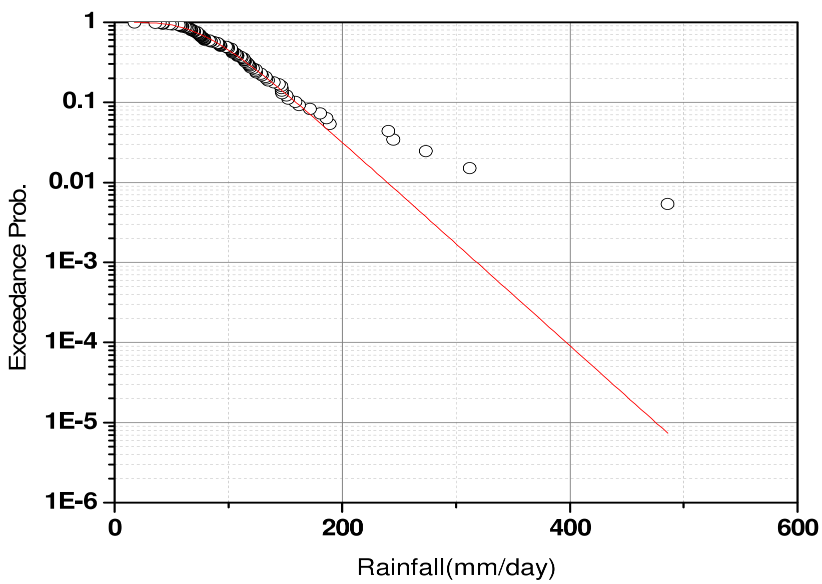

Fig. 5.2.2.2: Observed Annual Maximum Rainfall in a Day for Kalpakkam vs. Log exceedance Probability

The parameters for Gumbel fit has been shown in Table 5.2.2.1.

Table 5.2.2.1: Parameters of the Gumbel fit to the data

Parameter	Mean value	Error	95% confidence bound
μ	82.52	0.28	[83.09,81.95]
σ	34.13	0.44	[34.98,33.26]

The 95% confidence bounds are estimated using the expression

Bounds = mean of the parameter ± (2*Error estimated for the parameter).

A fit to the Generalized Extreme value distribution yields a fit very close to Gumbel distribution as shown in Fig. 5.2.2.3. The parameter for Generalized Extreme value distribution is given in Table 5.2.2.2.

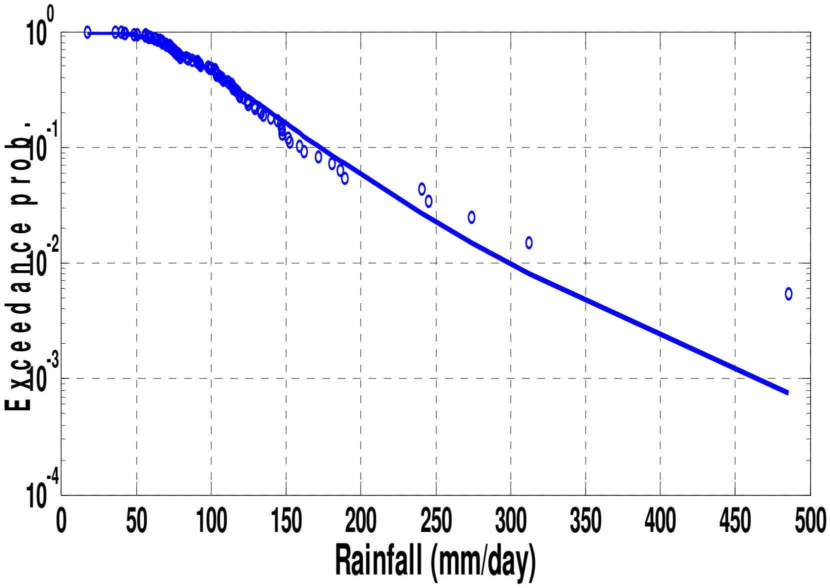

Fig. 5.2.2.3: Observed Annual Maximum Rainfall in a Day for Kalpakkam vs. Log exceedance Probability (GEV)

Table 5.2.2.2: Parameters of the Generalized Extreme value distribution to the data

Parameter	Detail	value
μ	Location parameter	81.83
σ	Scale parameter	35.01
k	Shape parameter	0.12

As seen from the Fig. 5.2.2.2, there are quite a few points which are deviating from the fitted line. Therefore power law and exponential fit are tried. For power law fit the data range considered is 100mm-500mm/day. The Fig. 5.2.2.4 represents power law fit for Log (Exccedance probability) vs Rainfall.

The entire set of rainfall data for Kalpakkam has also been fit using exponential distribution and return period has been estimated. The Fig. 5.2.2.5 represents exponential fit for Log (Exccedance probability) vs Rainfall. The exponential fit gives higher estimate of rainfall in the region than Gumbel distribution but lower estimate than power law fit.

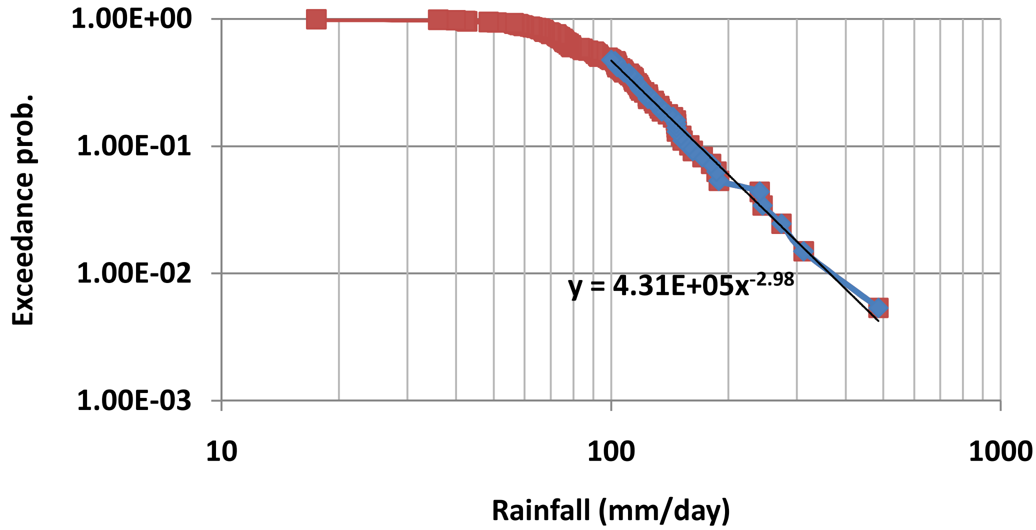

Fig. 5.2.2.4: Observed Annual Maximum Rainfall in a Day for Kalpakkam (Power Law fit)

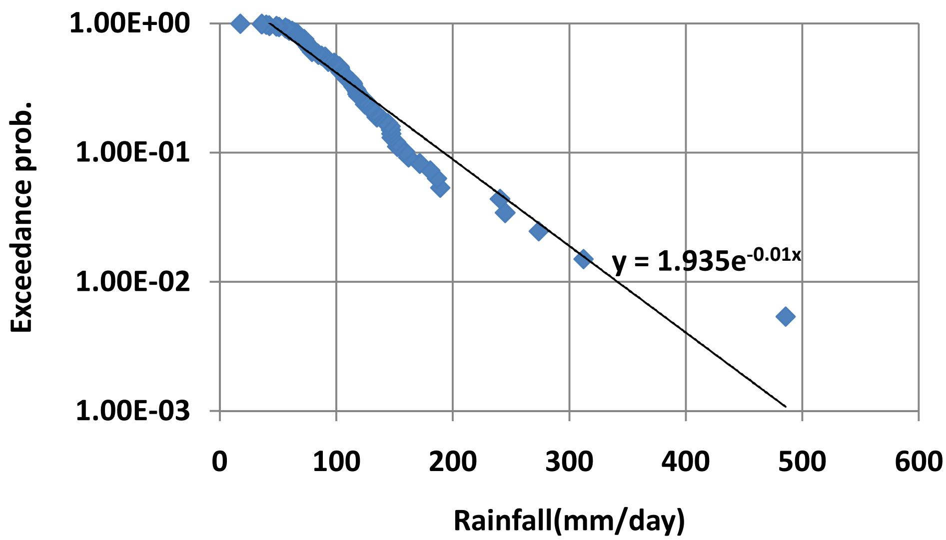

Fig. 5.2.2.5: Observed Annual Maximum Rainfall in a Day for Kalpakkam (Exponential fit)

5.2.2.4 Results

The predicted maximum rainfall for 100 years and 1000 years return period for Kalpakkam region by the Gumbel fit is shown in Table 5.2.2.3.

Table 5.2.2.3: Expected Rainfall (Gumbel fit) at Kalpakkam with upper bound

Return period (years)	Rainfall (mm/day)	
	Mean	95% UB
100	240	245
1000	320	325
10000	400	410

The predicted maximum rainfall for 100 years and 1000 years return period for Kalpakkam region by power law fit and exponential fit are shown in Table 5.2.2.4.

Table 5.2.2.4: Expected Rainfall at Kalpakkam

Return period (years)	Kalpakkam Rainfall (mm/day)			
	Gumbel fit	GEV fit	Power law	Exponential fit
100	240	280	364	344
1000	320	465	789	494
10000	400	620	1710	625

The goodness of fit for different model is shown in Table 5.2.2.5.

Table 5.2.2.5: Goodness of Fit for Sample Data

Model	Reduced χ^2	Adjusted R^2
Gumbel(0-500mm)	5.13×10^{-4}	0.994
Power law (100-500mm)	1.02×10^{-4}	0.990
Expoential(0-500mm	6.01×10^{-4}	0.929

5.2.2.5 Stationarity Analysis of Extreme rainfall Data

For performing stationary analysis several rainfall data set is required. The present study has done stationary analysis of rainfall taking those regions where NPPs are present. This study is also significant because it calculates stationarity in eastern coast, western coast and inland nuclear site in detail. Data points have been extracted for other NPP site to perform stationarity of extreme rainfall data. For this analysis the reactor sites have been divided into coastal and inland sites. The nuclear sites have been further divided into eastern and western coast. The division of nuclear sites with respect to coastal, inland, eastern and western coast gives understanding regarding variation of rainfall at different nuclear sites. The longitude latitude location of various sites used in the analysis is shown in Table 5.2.2.6.

Table 5.2.2.6: Latitude Longitude of different Nuclear Sites

Coastal Sites	Latitude	Longitude	Inland Sites	Latitude	Longitude
Kalpakkam	12.56N	80.18E	Nagpur	21.15N	79.09E
Mumbai	18.97N	73.82E	Indore	22.72N	75.86E
Visakhapatnam	17.68N	83.22E	Kaiga	14.86N.	74.44E
Kolkata	22.57N	88.37E	Kakrapara	21.24N	73.35E
Kochi	9.97N	76.28E	Kota	25.18N	75.83E
Mithi Vardi	22.15N	72.15E			

Figure 5.2.2.6 represents location of various nuclear power plants in the map of India (Map courtesy: wikihow).

Figure 5.2.2.6: Nuclear power plants location in India

It is increasingly being recognized that long term change in rainfall at regional scale can significantly affect magnitude and frequency of flood. Several studies based on hydrological data from around the world have now provided evidence of rainfall related changes in flood activity [67-70]. The probability of detecting shifts or changes in rainfall on the decadal or century scale is greater from longer records. Several studies to investigate stationarity of rainfall have been performed over India [71-74]. Recent studies by Dash S.K. et al (2007) [103] and Dash and Hunt (2007) [104] have emphasized on the decrease in the mean monsoon rainfall over India although the decreasing trend is small. Also earlier results of Srivastava H.N. et al (1992) [105] and Goswami B.N. et al (2006) [72] indicate that the change in the summer monsoon rainfall is not statistically significant.

Extreme rainfalls are the essential input to develop Intensity Duration Frequency curve, which are used to derive Design Basis Flood Level (DBFL) for a nuclear power plant. However increase in frequency and magnitude of extreme precipitation has already been observed for many regions [106]. IPCC (2007) [75] reported that the intensity and frequency of extreme rainfall events are very likely to increase in future. Increase in frequency and magnitude of extreme precipitation events question the stationarity in climate, which is the main assumption of frequency analysis of extreme rainfall. Possible violation of stationarity in climate increase concerns among designer about the currently used design rainfall estimates for civil infrastructure projects. Therefore it is imperative to do stationarity analysis of rainfall for infrastructure as critical as nuclear power plant.

Several studies had been conducted to investigate extreme rainfall trends over India [71-74]. However all of them had done study taking broad area (e.g. north India, east India or whole India) for analysis. The present study has done stationarity analysis of taking those regions where nuclear power plants are present (eleven sites has been selected and compared). This study is also significant because it calculates stationarity in eastern coast, western coast and inland nuclear site in detail.

The present analysis has been done by two methods namely i) exponent variation and ii) L-Moments ratio variation. Change in pattern of exponent is calculated using 10 data set moving window. Exponent for each data set is calculated and plotted on plotting paper. From this study it has been found that rainfall pattern is stationary over all nuclear sites in India. From the analysis it has been observed that eastern coast rainfall is more variable than western coast.

Exponent variation

Changes in the regional growth curve parameters and its slope coefficient (exponent) during the period from 1901 to 2004 have been examined using 10 data set moving window. To access the stationarity of the rainfall data, a moving window estimate of the exponent for the exponential fit and power law fit have been performed. The number of data set considered for exponent evaluation is 10 points. The evaluation is repeated ~ 94 times starting from year 1901-1910. It uses 10 data set and evaluate exponent for the set, then remove one data and add another one (e.g. 1901- 1910 one data set , then 1902-1911 other set).The obtained exponents are plotted in Fig 5.2.2.7. It is inferred from the plot that the exponent are oscillating w.r.t mean as shown in Fig 5.2.2.7. A small jump observed in the exponent set is due to the occurrence of at least one extreme value of rainfall intensity in corresponding set. Since mean value of coefficients are not changing, the rainfall pattern is stationary. Here a coastal site (Kalpakkam) and inland site (Kota) parameters for the exponential fit variation and the power law fit variation are shown in Fig 5.2.2.7.

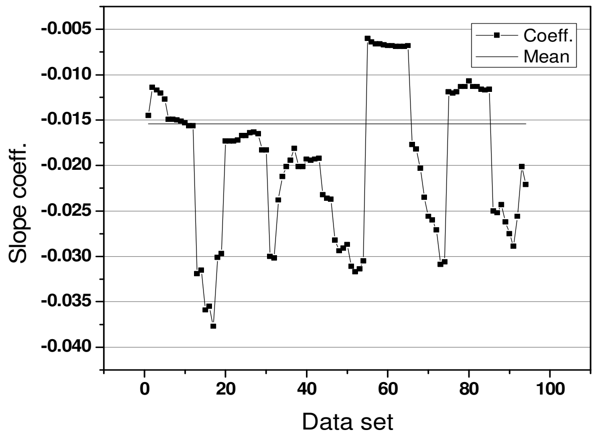

Fig. 5.2.2.7A: Exponents variation with mean for Kalpakkam (Exponential fit)

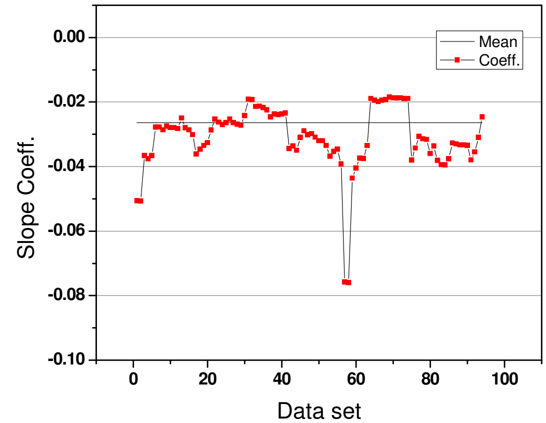

Fig. 5.2.2.7B: Exponents variation with mean for Kota (Exponential fit)

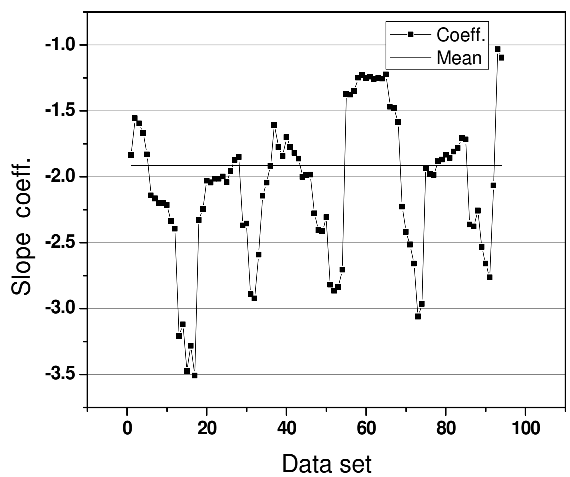

Fig. 5.2.2.7C Exponents variation with mean for Kalpakkam (power law fit)

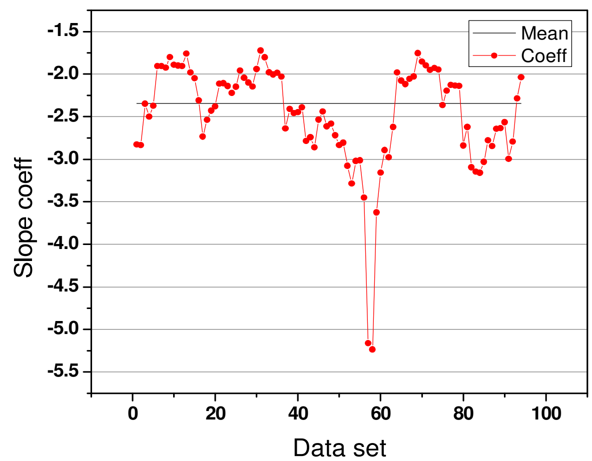

Fig. 5.2.2.7D Exponents variation with mean for Kota (power law fit)

L-moments

L-moment statistics are used for computing sample statistics for data at individual sites; for testing for homogeneity/heterogeneity of proposed groupings of sites (regions); for conducting goodness-of-fit tests for identifying a suitable probability distribution(s); and for solving for distribution parameters for the selected probability distribution [107]. L-moments obtain their name from their construction as linear combinations of order statistics.

The L-moment measure of location and L-moment ratio measures of scale, skewness and kurtosis are:

Location (Mean) = L_1 (5.2.2-6a)

Scale (L-CV) = L_2/ L_1 (5.2.2-6b)

L-Skewness = L_3/ L_2 (5.2.2-6c)

L-Kurtosis = L_4/ L_2 (5.2.2-6d)

where

$L_1 = \beta_0$ (5.2.2-6e)

$L_2 = 2\beta_1-\beta_0$ (5.2.2-6f)

$L_3 = 6\beta_2-6\beta_1+\beta_0$ (5.2.2-6g)

$L_4 = 20\beta_3-30\beta_2+12\beta_1-\beta_0$ (5.2.2-6h)

and, where the data ($x_{1:n}$) are first ranked in ascending order from 1 to n and:

$\beta_0= n^{-1}\sum_{j=1}^{n} x_j$ (5.2.2-6i)

$\beta_2= n^{-1}\sum_{j=2}^{n} x_j\,[(j-1)/(n-1)]$ (5.2.2-6j)

$\beta_3= n^{-1}\sum_{j=3}^{n} x_j\,[(j-1)(j-2)/(n-1)(n-2)]$ (5.2.2-6k)

$\beta_4= n^{-1}\sum_{j=4}^{n} x_j\,[(j-1)(j-2)(j-3)/(n-1)(n-2)(n-3)]$ (5.2.2-6l)

Changes in L-moment ratios

In this study, a regional frequency analysis approach based on L-moments [108] has been used to produce rainfall growth curve with an extreme value distribution. L- moments are used for solving distribution parameters for selected probability distribution. They are a dramatic improvement over conventional product moment statistics for characterizing the shape of a probability distribution and estimating the distribution parameters, particularly for environmental data where sample sizes are commonly small. Unlike product moments, the sampling properties for L-moments statistics are nearly unbiased, even in small samples, and are nearly normally distributed. These properties make them well suited for characterizing environmental data that commonly exhibit moderate to high skewness [107]. The three L-moment ratios L-CV, L-Skewness and L-Kurtosis have been determined for the annual maximum rainfall for each site (first moment is mean value). The L- moment ratios are direct measures of the extreme value distribution and as such provide a better illustration of change than fitted Extreme Value parameters themselves [109]. Fig. 5.2.2.8 shows the variation of L-CV against L-Skewness for various sites. In simple terms this represents a change in extreme rainfall properties, with increased variability (rising L-CV) and increased intensity (rising L-Skewness) in the region.

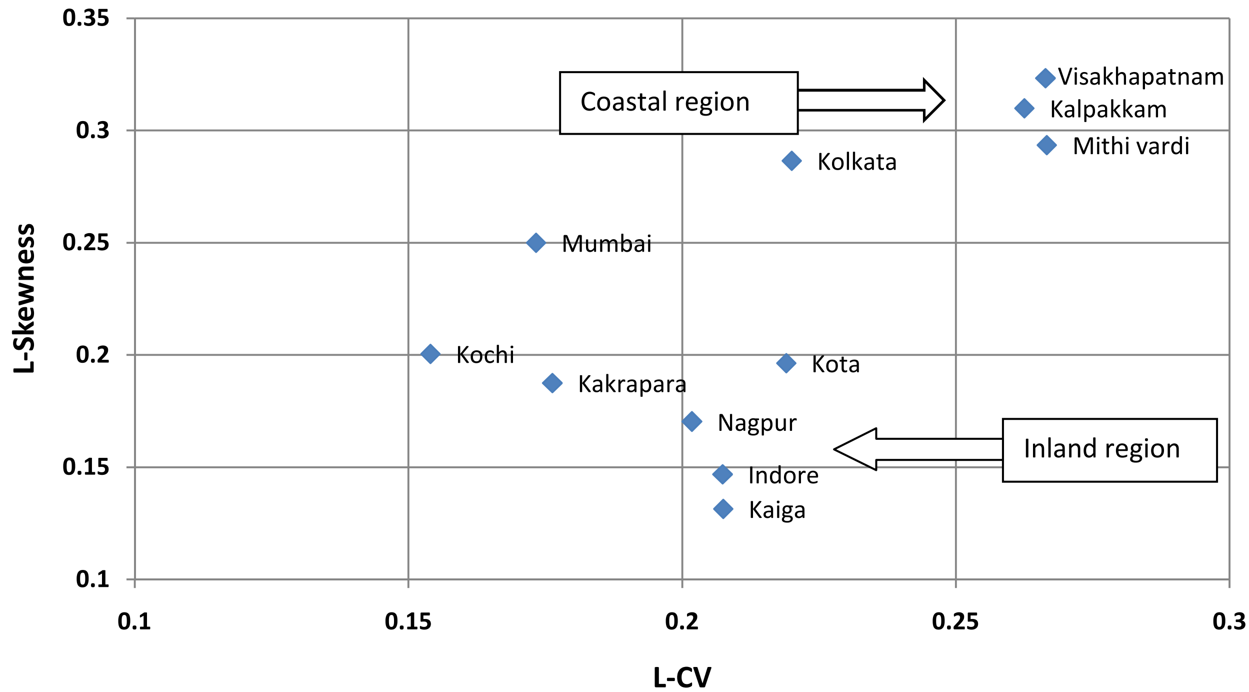

Fig. 5.2.2.8: Comparison of Mean L-CV and L-Skewness of different Nuclear Sites

It can be seen from Fig. 5.2.2.8 that coastal regions display a greater L-CV value than inland regions, suggesting higher variability in these regions. Also eastern coast display a greater L-CV value than western coast (except Mithi vardi, which is itself in bay of Khambat). Also coastal region show higher L-Skewness Value which plunges as we move inland region. In simple terms coastal regions are having intense rainfall and higher variability than inland regions.

This analysis will be helpful in keeping extra safety margin for flooding in coastal region (particularly eastern coast).

The L moments have been calculated for a coastal site (Kalpakkam) and inland site (Kota) using 10 data set moving window. The variation of L-moments is shown in Fig. 5.2.2.9.

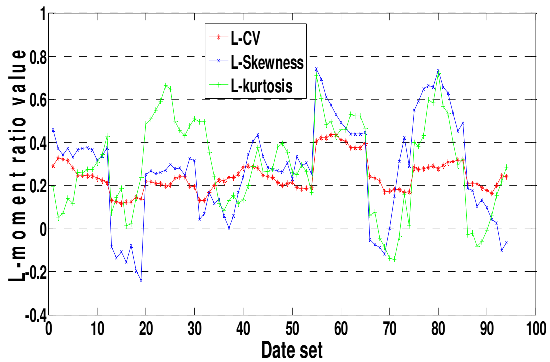

Fig. 5.2.2.9A: L moments variation (Kalpakkam)

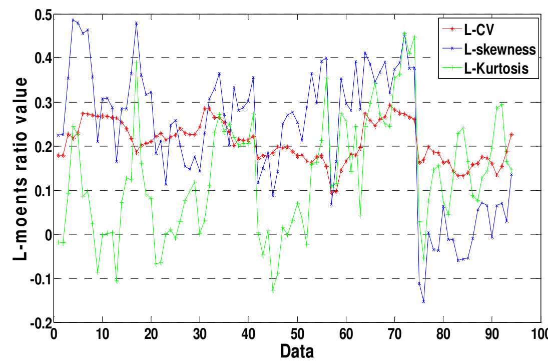

Fig. 5.2.2.9B: L moments variation (Kota)

The obtained L-moments are oscillating as shown in Fig 5.2.2.8. A small jump observed in the exponent set is due to the occurrence of at least one extreme value of rainfall intensity in corresponding decade. Otherwise the rainfall pattern is stationary. Since mean and covariance value of coefficients are not changing, the rainfall pattern is stationary.

5.2.2.6 Conversion of rainfall into flood level

The rise in water level in the plant due to the rainfall is given by the following model.

$$A.\frac{dH}{dt} = I - D \tag{5.2.2-6}$$

where, A is the area of the plant (catchment area). I is the inflow volume (m^3/h) and D is the discharge volume (m^3/h). The PFBR nuclear Island is elevated compared to adjacent areas and MAPS. So the catchment area of PFBR is equal to the area of the plant itself. The maximum rainfall predicted by power law is ~789 mm/day for 1000 year return period. Under this condition, with 20% of the drain assumed to be available, the discharge volume is greater than the inflow. Flooding in the plant due to rainfall is unlikely. The initiating event frequency contribution to flood from rainfall is negligibly small.

5.2.3: Tsunami Hazard Analysis and Failure Frequency Estimation

5.2.3.1 Tsunami hazard

The result of tsunami hazard analysis has been taken from report [60]. The flood height arising from tsunami hazard with combination of storm surge and rainfall is shown in Table 5.2.3.1. The finished floor level (FFL) of NICB (Nuclear Island Connected Building) PI (Power Island) along with 2004 tsunami level is illustrated below in Fig. 5.2.2.10 [110].

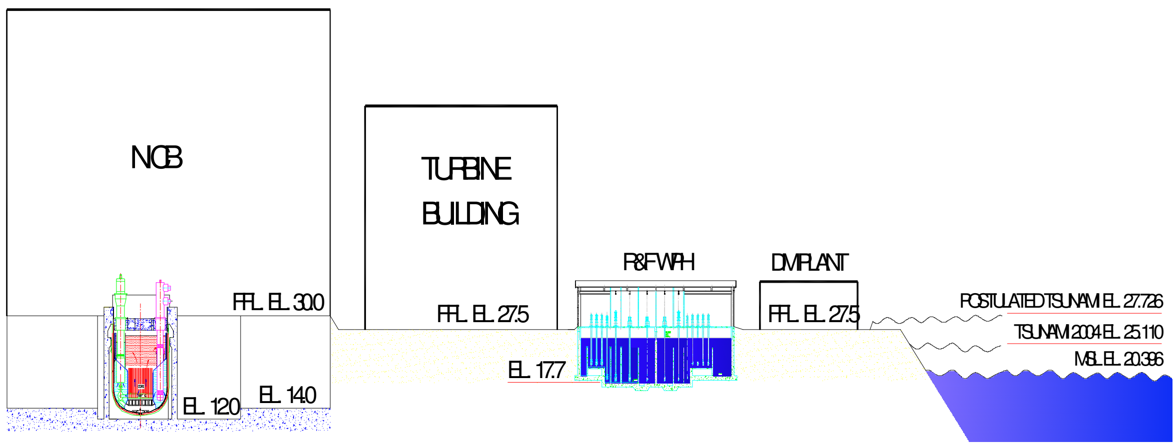

Fig. 5.2.3.1: Design Basis Flood Level And Levels During Observed And Postulated Tsunami

5.2.3.2 Plant Logic Models and Accident Sequence Models

The plant logic and accident sequence models for EFPSA are developed using fault tree and event tree techniques. The detailed system models are developed using fault tree and accident sequence models are developed using event trees. The fault tree models are developed using immediate cause approach to the possible extent. Fault trees and event trees developed for level-1 internal events PSA were used for this purpose. Even though fault trees and event trees are available from level-1 internal events PSA, the following steps were carried out to address the flood specific quantification of fault trees and event trees.

i) Definition of failure modes of components & flood Fragility

ii) External Flood Frequency occurrence computation

iii) Estimation conditional initiating event frequency for different flood levels

iv) Integration of fault tree and event tree for flood failure frequency quantification

5.2.3.3 Definition of Failure Mode and Flood Fragility

The failure mode considered in this analysis is the component failure by submergence. The submergence itself is used in a conservative way in this analysis. If the flood level is equal to a particular floor level, all components located at that elevation and components at floor levels below that are assumed to have failed. No differentiation is made between components located outside the buildings and inside the buildings.

The above definition of failure leads to a step fragility function for components. Let P(H) be the fragility of a component for a particular flood hazard level H. If h_{el} is the elevation of the component, then the fragility is defined as,

$$P\,(H) = \begin{cases} 1 \; if \; H \geq \; h_{el} \\ Q \; if \; H < h_{el} \end{cases} \tag{5.2.3-1}$$

where Q represents the unavailability of the component due to random failures. The above model of flood fragility is used in this analysis. Since fragility is function of component elevation, a plant walk down has been performed. The location and elevation of various components is shown in Table 5.2.3.1.

Table 5.2.3.1: Location and Elevation of Components of Class-III Power Supply System

Sl. No	Component ID	Description	Building	Elevation (EL)
1	EMTR	Emergency transfer failure	EB	38
2	DG1AM	DG 1A in maintenance	DGB-1	30
3	BUS1AH	Bus 1A fails	EB-1	38

Sl. No	Component ID	Description	Building	Elevation (EL)
4	BUS1BH	Bus 1B fails	EB-1	38
5	BUS2AH	Bus 2A fails	EB-2	38
6	BUS2BH	Bus 2B fails	EB-2	38
7	CB-UB1-1A1	CB connecting Unit bus 1A fail to open	EB-1	38
8	HE	Human Error after the auto failure		
9	CB-DG1-1A	CB of DG 1A fails to close	DGB-1	30
10	CB-SB1-1B1	CB connecting Unit bus 1B fail to open	EB-1	38
11	CB-DG2-1B	CB of DG 1B fails to close	DGB-1	30
12	DG2-FS	DG2 mechanical fail to start	DGB-1	30
13	DG2-FR	DG-2 mechanical fail to run	DGB-1	30
14	CB-UB2-2A1	CB connecting Unit bus 2A fail to open	EB-2	38
15	CB-DG3-2A	CB of DG 2A fails to close	DGB-2	30
16	DG3-FS	DG-3 mechanical fail to start	DGB-2	30
17	DG3-FR	DG-3 mechanical fail to run	DGB-2	30
18	CB-SB2-2B1	CB1 connecting Unit bus 2B fail to open	EB-2	38
19	DG4-FS	DG4 mechanical fail to start	DGB-2	30
20	DG4-FR	DG-4 mechanical fail to run	DGB-2	30
21	CB-DG4-2B	CB of DG 2B fails to close	DGB-2	30
22	DG1BM	DG 1B in maintenance	DGB-1	30
23	DG2AM	DG 2A in maintenance	DGB-2	30
24	DG2BM	DG 2B in maintenance	DGB-2	30
25	DG1-FR	DG1 mechanical fail to run	DGB-1	30
26	DG1-FS	DG1 mechanical fail to start	DGB-1	30
27	CB-LD-IA	Load side breaker fail to open and close(x2)	DGB-1	30
28	CB-LD-1B	load side breaker fail to open and close(x2)	DGB-1	30
29	CB-LD-2A	load side breaker fail o open and close(x2)	DGB-2	30
30	CB-LD-2B	Load side breaker fail to open and close(x2)	DGB-2	30
31	CB-UB1-1A2	CB connecting Unit bus 1A fail to open	EB-1	38
32	CB-SB1-1B2	CB connecting Unit bus 1B fail to open	EB-1	38
33	CB-UB2-2A2	CB connecting Unit bus 2A fail to open	EB-2	38
34	CB-SB2-2B2	CB2 connecting Unit bus 2B fail to open	EB-2	38
35	CB-BC-1AB	CB b/w 1A and 1B fail to change position	EB-1	38
36	CB-BC-2AB	CB b/w 1A and 1B fail to change position	EB-2	38
37	AIRBOT-1-FAIL	Air Bottle-1 failure	DGB-1	30
38	AIRBOT-2-FAIL	Air Bottle-2 Failure	DGB-1	30
39	AIRBOT-3-FAIL	Air Bottle-3 failure	DGB-1	30
40	AIRBOT-4-FAIL	Air Bottle-4 Failure	DGB-1	30

Sl. No	Component ID	Description	Building	Elevation (EL)
41	AIRBOT-5-FAIL	Air Bottle-5 failure	DGB-2	30
42	AIRBOT-6-FAIL	Air Bottle-6 Failure	DGB-2	30
43	AIRBOT-7-FAIL	Air Bottle-7 failure	DGB-2	30
44	AIRBOT-8-FAIL	Air Bottle-8 Failure	DGB-2	30
45	DG1-DTLC-FAIL	DG-1 day tank level control failure	DGB-1	30
46	DG2-DTLC-FAIL	DG-2 day tank level control failure	DGB-1	30
47	DG3-DTLC-FAIL	DG-3 day tank level control failure	DGB-2	30
48	DG4-DTLC-FAIL	DG-4 day tank level control failure	DGB-2	30
49	COMP-1-FS	Compressor-1 fail to start	DGB-1	30
50	COMP-1-FR	Compressor-1 fail to run	DGB-1	30
51	COMP-2-FS	Compressor-2 fail to start	DGB-1	30
52	COMP-2-FR	Compressor-2 fail to run	DGB-1	30
53	FUPUMP-1-FS	Fuel Pump-1 fail to start	DGB-1	30
54	FUPUMP-1-FR	Fuel Pump-1 fail to run	DGB-1	30
55	FUPUMP-2-FS	Fuel Pump-2 fail to start	DGB-1	30
56	FUPUMP-2-FR	Fuel Pump-2 fail to run	DGB-1	30
57	COMP-5-FS	Compressor-5 fail to start	DGB-2	30
58	COMP-5-FR	Compressor-5 fail to run	DGB-2	30
59	COMP-6-FS	Compressor-6 fail to start	DGB-2	30
60	COMP-6-FR	Compressor-6 fail to run	DGB-2	30
61	FUPUMP-5-FS	Fuel Pump-5 fail to start	DGB-2	30
62	FUPUMP-5-FR	Fuel Pump-5- fail to run	DGB-2	30
63	FUPUMP-6-FS	Fuel Pump-6 fail to start	DGB-2	30
64	FUPUMP-6-FR	Fuel Pump-6 fail to run	DGB-2	30
65	COMP-7-FS	Compressor-7-fail to start	DGB-2	30
66	COMP-7-FR	Compressor-7 fail to run	DGB-2	30
67	COMP-8-FS	Compressor-8 fail to start	DGB-2	30
68	COMP-8-FR	Compressor-8 fail to run	DGB-2	30
69	FUPUMP-7-FS	Fuel Pump-7 fail to start	DGB-2	30
70	FUPUMP-7-FR	Fuel Pump-7 fail to run	DGB-2	30
71	FUPUMP-8-FS	Fuel Pump-8 fail to start	DGB-2	30
72	FUPUMP-8-FR	Fuel Pump-8 fail to run	DGB-2	30
73	COMP-3-FS	Compressor-3 fail to start	DGB-1	30
74	COMP-3-FR	Compressor-3 fail to run	DGB-1	30
75	COMP-4-FS	Compressor-4 fail to start	DGB-1	30
76	COMP-4-FR	Compressor-4 fail to run	DGB-1	30
77	FUPUMP-3-FS	Fuel Pump-3 fail to start	DGB-1	30
78	FUPUMP-3-FR	Fuel Pump-3 fail to run	DGB-1	30
79	FUPUMP-4-FS	Fuel Pump-4 fail to start	DGB-1	30

Sl. No	Component ID	Description	Building	Elevation (EL)
80	FUPUMP-4-FR	Fuel Pump-4 fail to run	DGB-1	30
81	B100A-TR-F-100A	Transformer 100A (6.6kV/433V) fails	EB-1	30
82	BUS100A-CB-D-100A	Circuit Breaker 100A fails to remain in position	EB-1	38
83	B100B-TR-F-100B	Transformer 100B (6.6kV/433V) fails	EB-1	30
84	BUS100B-CB-D-100B	Circuit Breaker 100B fails to remain in position	EB-1	38
85	B100C-TR-F-100C	Transformer 100C (6.6kV/433V) fails	EB-1	30
86	BUS100C-CB-D-100C	Circuit Breaker 100C fails to remain in position	EB-1	38
87	BUS100D-CB-D-100D	Circuit Breaker 100D fails to remain in position	EB-1	38
88	B100D-TR-F-100D	Transformer 100D (6.6kV/433V) fails	EB-1	30
89	BC-100A-100B-FAIL	Bus coupler between 100A&100B fail to close	EB-1	38
90	BC-100C-100D-FAIL	Bus coupler b/w 100C&100D fail to close	EB-1	38
91	BC-200A-200B-FAIL	Bus coupler b/w 200A&200B fail to close	EB-2	38
92	BUS200A-CB-D-200A	Circuit Breaker 200A fails to remain in position	EB-2	38
93	BUS200B-CB-D-200B	Circuit Breaker 200B fails to remain in position	EB-2	38
94	B200A-TR-F-200A	Transformer 200A (6.6kV/433V) fails	EB-2	30
95	B200B-TR-F-200B	Transformer 200B (6.6kV/433V) fails	EB-2	30
96	BC-200C-200D-FAIL	Bus coupler b/w 200C&200D fail to close	EB-2	38
97	BUS200C-CB-D-200C	Circuit Breaker 200C fails to remain in position	EB-2	38
98	BUS200D-CB-D-200D	Circuit Breaker 200D fails to remain in position	EB-2	38
99	B200C-TR-F-200C	Transformer 200C (6.6kV/433V) fails	EB-2	30
100	B200D-TR-F-200D	Transformer 200D (6.6kV/433V) fails	EB-2	30

5.2.3.4 External Flood Frequency Calculation

The external flood occurrence frequency for a specific flood level is calculated by simply summing the frequency contributions from tsunami, storm surge and rainfall. Out of the three hazards considered, the flood frequency contribution from rainfall is very small due to the design of the storm water drains at PFBR site. The other two contributions are calculated from the reported hazard curves. The frequencies are estimated for flood levels corresponding to EL 30.0 (9.6 m above MSL) to EL 40.0. The tsunami hazard curves have a limiting value beyond which there is no increase. The frequency contribution for tsunami wave run up heights is computed by extrapolating the median curves over desired flood levels. The flood frequency contribution from storm surge, tsunami hazard and the total flood occurrence frequency for different flood levels are given in Table 5.2.3.2. From this table it is clear that the flood occurrence frequency is governed by frequency of occurrence of tsunami of particular wave run up height.

Table 5.2.3.2: External Flood Frequency

Sl.No	Flood Level		Frequency from Storm Surge (/y)$		Frequency from Tsunami (/y)		Total Frequency (/y)	
	Above MSL (m)	EL	Median	Upper Bound	Median*	Upper Bound#	Median	Upper Bound
1.	5.6	26.0	$<10^{-10}$	2.0E-09	1.4E-03	0.055	1.4E-03	0.055
2.	7.6	28.0	$<10^{-10}$	$<10^{-10}$	9.5E-05	7.0E-03	9.5E-05	7.0E-03
3.	9.6	30.0	$<10^{-10}$	$<10^{-10}$	4.97E-05	4.0E-03	4.97E-05	4.0E-03
4.	11.6	32.0	$<10^{-10}$	$<10^{-10}$	2.8E-07	2.5E-03	2.8E-07	2.5E-03
5.	13.6	34.0	$<10^{-10}$	$<10^{-10}$	1.9E-08	3.0E-04	1.9E-08	3.0E-04
6.	15.6	36.0	$<10^{-10}$	$<10^{-10}$	<1.0E-09	-NA-	1.0E-09	-NA-
7.	17.6	38.0	$<10^{-10}$	$<10^{-10}$	<1.0E-09	-NA-	1.0E-09	-NA-
8.	19.6	40.0	$<10^{-10}$	$<10^{-10}$	<1.0E-09	-NA-	1.0E-09	-NA-

$ - Frequencies correspond to the sea level which is a combination of storm surge and tide level. Tide level at Chennai is between 0.4m to 1.0m.
* - Median curve is extrapolated beyond 5.3m to obtain these frequencies
# - The limiting value of upper bound tsunami curve is 13.6m

5.2.3.5 Estimation of Conditional Initiating Event Frequency

The next step is to estimate the conditional initiating event frequency for different initiating events. The objective is to estimate the frequency f (IE | H) where IE is the initiating event and H is flood hazard level. f (IE | H) is the frequency of occurrence of the initiating event IE for flood level H. The conditional initiating event frequency is given by equation (5.2.3-2).

$$f\ (IE \mid H) = P(IE\ |H).f(H) \qquad (5.2.3\text{-}2)$$

P(IE|H) is the conditional probability of initiating event occurrence for a flood level H. f (H) is the frequency of occurrence of the specified flood level. P(IE|H) is zero if the initiating event is unlikely to occur for a specific flood level. If the initiating event is likely to occur for a specific flood level, then it is assumed that P(IE|H)=1 for that flood level. There are certain set of initiating events which are not likely to occur due to external flood. For those events the conditional initiating event frequency is zero that is, f (IE | H) = 0. The initiating event groups from level-1 internal events PSA is used in this study. For DG initiating event group is named as PSS2. A value of zero indicates that particular initiating event group is unlikely to occur due to flood hazard of specified level. A value of 1 indicates that those initiating events are likely to occur for the specified flood hazard level with a frequency mentioned in equation (5.2.3-2). In present case DG is likely to be affected for flood levels ≥ EL 30.

5.2.3.6 Integration of Fault Tree and Event Tree

The detailed system fault trees are integrated with event trees to quantify the failure frequency. This step requires the fragility values need to be computed for different flood hazard levels. This is achieved by using an EXCEL macro for fragility computation. This macro uses equation (5.2.3-1) to estimate the fragility of different components. This macro also performs the

CCF group adjustment. If the fragility of a component is equal to 1 for a specific hazard level, then that component will be removed from that particular CCF group. This is because CCF failure probabilities are calculated as fractions of failure probabilities of components.

Assumptions in this Study

The study has been conducted with following assumptions

a) The Reactor Containment Building (RCB) is assumed to be leak tight and the components inside RCB are not affected by flood.

b) Loss of OffSite Power (LOSP) is assumed for flood level ≥ EL 28.0.

Event Tree

The event Tree for initiating event PSS2 (One DG not available) is shown in Fig. 5.2.3.1.

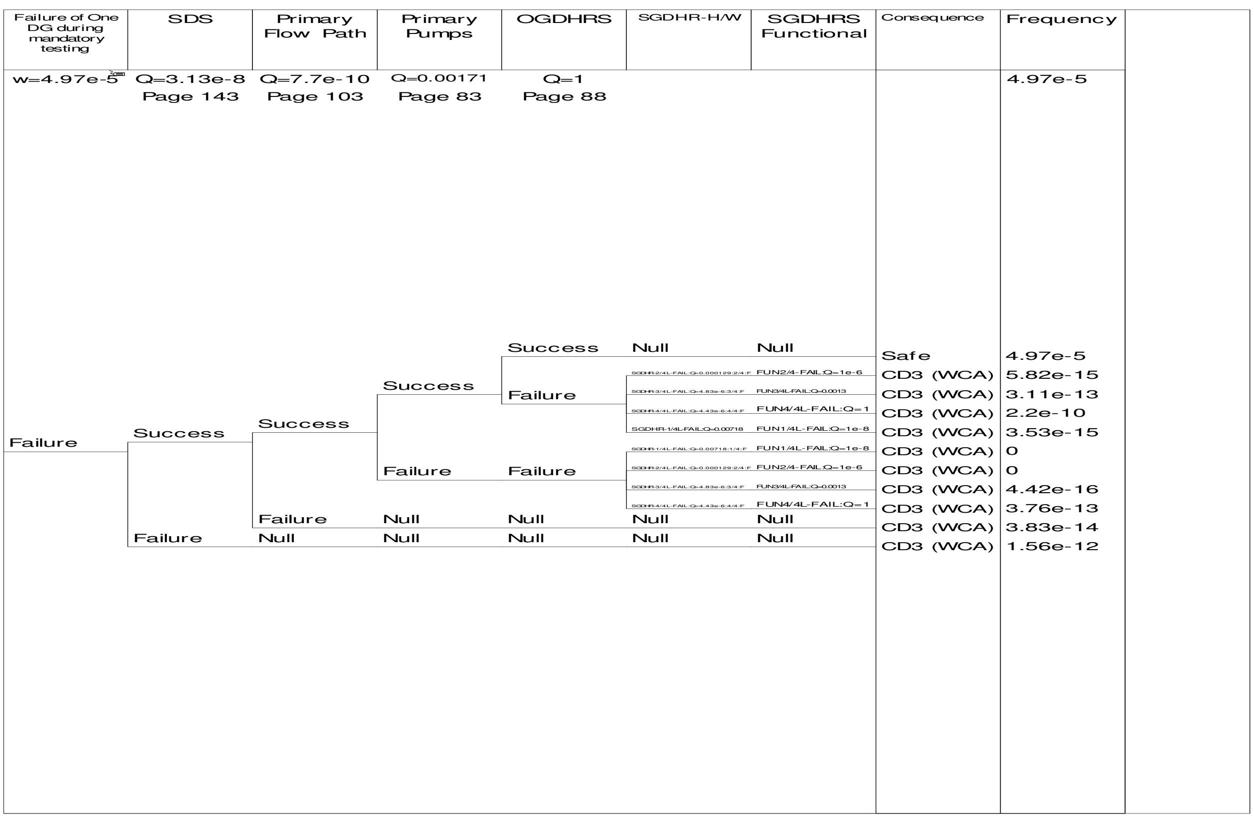

Fig. 5.2.3.2: Event Tree for PSS2

5.2.3.7 Results

The class III power supply failure due to external flood for PFBR is calculated by quantifying the event trees. The class III power supply failure frequency has been calculated at different elevation (30-40 EL).The failure frequency of DG due to flooding events is shown in Table 5.2.3.3. Total frequency for class III power supply failure due to flooding events is 2.22 E-09/ry. The contribution of failure frequency at intermediate elevation (32-36 EL) is low. This is expected as frequency of occurrence of flood level at these height decreases, however at higher elevation it increases (38-40 EL) because even though frequency of occurrence of this event is low, more and more components will be submerged at this height. The fragility dominates return period at this elevation. The minimal cutsets arising due to flooding events are shown in table 5.2.3.4.

Table 5.2.3.3: CDF for DG at Different Elevation

Sl. No	IE Group	CDF for Different Flood Levels (EL)						
		28	30	32	34	36	38	40
1.	PSS2	-NA-	2.2E-10	1.3E-12	8.5E-14	4.5E-15	1.0E-9	1.0E-9

Table 5.2.3.4: Cutsets for PSS2

No.	Cut set	Frequency
1	ONE-DG-FAIL. AIR-XDM-CCF-POS	1.67e-10
2	ONE-DG-FAIL. AIR-XDM-FO-CRXHE. AIR-XDM-FO-XHE	4.97e-11
3	ONE-DG-FAIL. INT-PSF-CCF-LK-ALL	1.79e-12
4	ONE-DG-FAIL. CDRCF0001	1.49e-12
5	ONE-DG-FAIL. STACK-CCF-LF	5.37e-13
6	ONE-DG-FAIL. AIR-DMP-CCF-MECH. AIR-XDM-FO-XHE	4.97e-13
7	ONE-DG-FAIL. AUX-ICC-CCF	1.99e-13
8	ONE-DG-FAIL. INT-LOC-CCF-INAD-ALL	1.99e-13
9	ONE-DG-FAIL. INT-TNK-CCF-LK-ALL	1.79e-13
10	ONE-DG-FAIL. INT-AHX1-CCF-LK-ALL. INT-AHX2-CCF-LK-ALL	9.31e-14
11	ONE-DG-FAIL. DMP-FREEZ-CCF	4.97e-14

12	ONE-DG-FAIL. SWCCF12	4.97e-14
13	ONE-DG-FAIL. PHT-IHX-CCF-BL	3.58e-14
14	ONE-DG-FAIL. INT-DHX1-CCF-LK-ALL. INT-AHX2-CCF-LK-ALL	1.55e-14
15	ONE-DG-FAIL. INT-AHX1-CCF-LK-ALL. INT-DHX2-CCF-LK-ALL	1.55e-14
16	ONE-DG-FAIL. HETHC0002. HETHC0001	4.97e-15
17	ONE-DG-FAIL. INT-AHX1-CCF-LK-ALL. INT-AHX-LK-AHX4. FUN3/4L-FAIL	3.02e-15
18	ONE-DG-FAIL. INT-AHX1-CCF-LK-ALL. INT-AHX-LK-AHX3. FUN3/4L-FAIL	3.02e-15
19	ONE-DG-FAIL. INT-AHX-LK-AHX2. INT-AHX2-CCF-LK-ALL. FUN3/4L-FAIL	3.02e-15
20	ONE-DG-FAIL. INT-AHX-LK-AHX1. INT-AHX2-CCF-LK-ALL. FUN3/4L-FAIL	3.02e-15

5.2.4 Summary

This chapter presents the failure frequency estimation of class III power supply system of PFBR due to external flood. The hazard analysis is performed for storm surge, rainfall and tsunami. Considering the elevation of PFBR nuclear island, various buildings, safety systems and drainage design, the hazard information together with the component / subsystem immersion fragility, failure frequency is obtained. This study is significant because it shows methodology to perform flooding hazard analysis at NPPs. This study will be beneficial in estimating the design basis flood level (DBFL) for future NPPs. The failure frequency contribution from storm surge and rainfall is very small. The analysis for storm surge hazard is generally carried out by two methods. The first method is a physical method in which a non-linear hydrodynamic model, the second method is a statistical one which is based on extreme value analysis of observed maximum sea level during a storm surge event. Extreme Value Analysis has been used in this study to model return periods. In performing stationary analysis of rainfall following models were developed. A mathematical model based on moving window estimate to check stationary *has been developed. It uses 10 year data set and evaluates its exponent (slope coefficient) for the set. It then removes one data and adds another one (e.g. 1901–1910 one data set, then 1902–1911 other set). A mathematical model to study L-moments was developed. Stationarity analysis of rainfall data shows that rainfall pattern is stationary in both coastal and inland regions. The coastal regions show intense rainfall and higher variability than inland regions. Eastern coast shows higher variability in rainfall than western coast. This analysis will be helpful in keeping extra safety margin for flooding in coastal regions (particularly eastern coast). Significant results have been established in the field of rainfall variation/pattern in India. To calculate level rise at plant site due to rainfall, mass flow continuity equation has been used. The tsunami hazard governs failure frequency contribution class III power supply system of PFBR. The component fragilities were modeled with step fragility function considering submergence mode failure and random failure of components. For computing fragility EXCEL Macro sheet has been developed. This macro uses submergence to estimate the fragility of different components. This macro also performs the CCF group adjustment. If the fragility of a component is equal to 1 for a specific hazard level, then that component will be removed from that particular CCF group. Total frequency for class III power supply failure due to flooding events is 2.22 E-09/ry.*

5.3: Other Events

5.3.1 Wind Events

5.3.1.1 Introduction

Primary purpose of wind speed analysis is choosing the appropriate basic wind velocity for the design of buildings and structures [111]. Prediction of site specific wind speed plays a key role in the determination of wind-induced response of structures [112].Traditional methods for assessing the wind climate for wind engineering purposes are based on analysis of long term surface records from a nearby meteorological station, typically measured at a height of 10 m above ground. For this assessment, hourly surface wind data is obtained from Bhartiya Vidutya Nigam (Bhavini). Extreme value analyses are then performed for the determination of wind speeds with respect to return period.

5.3.1.2 Distribution of Extreme values

The Generalized Extreme Value (GEV) distribution is a flexible three-parameter model that combines the Gumbel, Fréchet, and Weibull maximum extreme value distributions [101]. It has the following probabilistic density function (PDF):

$$f(x)=\begin{cases} \frac{1}{\sigma}\exp\left(-(1+kz)^{-1/k}\right)(1+kz)^{-1-\frac{1}{k}} & k \neq 0 \\ \frac{1}{\sigma}\exp\left(-z-\exp(-z)\right) & k = 0 \end{cases} \tag{5.3.1-1}$$

Where $z=(x-\mu)/\sigma$, and k, σ, μ are the shape, scale, and location parameters respectively. The scale must be positive (α>0), the shape and location can take on any real value. The range of definition of the GEV distribution depends on k:

$$\begin{gathered} 1+\mathrm{k}\frac{x-\mu}{\sigma}>0 \quad \text{for } \mathrm{k} \neq 0 \\ -\infty < x < +\infty \; for\; k = 0 \end{gathered} \tag{5.3.1-2}$$

The Cumulative distribution function (CDF) of Generalized Extreme Value (GEV) distribution is

$$F(x)=\exp[-\{1+k(\frac{x-\mu}{\sigma})^{-1/k}\}] \tag{5.3.1-3}$$

Extreme value distribution is often estimated by power law fit. The cumulative distribution function of power law fit is

$$F(x) = ax^{b}, \qquad x\sim R; \tag{5.3.1-4}$$

Plotting ln(x) against ln(F(x)) gives a straight line. This property enables a visual check to be made of the extent to which a data set fits the power law distribution.

5.3.1.3 Data collection and analysis

Hourly averaged wind speed data at heights such as 30 m, 60 m and 10 m are obtained from database available at Environmental Survey Laboratory, Kalpakkam and from the wind rose diagram, data which is part of report submitted by M/s Mecon Limited to BHAVINI. The same is given in the Table 5.3.1. The predominant wind direction is seen to be south and average wind speed is 12-20 km/hr. The highest instantaneous wind speed (over 1 minute period) recorded was 182 km/h on 12.11.85 when a cyclonic storm crossed the east coast near Kalpakkam. An extreme value analysis was carried out by SERC, Chennai, for PFBR, using 110 years (1891-2000) of observed cyclonic data, covering IMD stations around Kalpakkam. Of the cyclonic disturbances that have crossed the Tamil Nadu coast during this period, 15 were designated as severe storms. Seven storms of varying severity have occurred during 1990 – 2000. None crossed the Tamil Nadu coast during 2001 – 11. Based on all the available information, a 3-sec gust wind speed at 10 m level, for the terrain category 2, using IS 875 (Part3-1987), is 85 m/s (306 km/h), for a return period of 1000 years. This was used as the basic wind speed for PFBR, FBR 1 & 2.

Table 5.3.1 Wind speed data for Kalpakkam

Year	Hourly avg. wind speed (km/h)	Taken at height (m)	Hourly avg. wind speed (km/h) Normalised	Hourly avg. wind speed (km/h) Normalised	Hourly avg. wind speed (km/h) Normalised)
1970	55.5	30	65.75	78.9	98.625
1971	43	30	50.94	61.1286858	76.41086
1972	66.5	30	78.78	94.5362233	118.1703
1973	48.2	30	57.10	68.5209919	85.65124
1974	49	30	58.05	69.6582698	87.07284
1975	53	30	62.79	75.3446592	94.18082
1976	63	30	74.63	89.5606326	111.9508
1977	72.1	30	85.41	102.497168	128.1215
1978	53.5	30	63.38	76.0554579	95.06932

Year	Hourly avg. wind speed (km/h)	Taken at height (m)	Hourly avg. wind speed (km/h) Normalised	Hourly avg. wind speed (km/h) Normalised	Hourly avg. wind speed (km/h) Normalised)
1979	67.4	30	79.85	95.8156609	119.7696
1980	47.7	30	56.51	67.8101933	84.76274
1981	37.8	30	44.78	53.7363796	67.17047
1982	58.4	30	69.18	83.0212848	103.7766
1983	61.7	30	73.09	87.7125561	109.6407
1984	77	30	91.22	109.462995	136.8287
1985	52.9	30	62.67	75.2024994	94.00312
1986	50.2	30	59.47	71.3641866	89.20523
1987	61.9	30	73.33	87.9968755	109.9961
1988	51.2	30	60.65	72.785784	90.98223
1989	54.5	30	64.56	77.4770552	96.84632
1990	56.3	60	58.06	69.6753243	87.09416
1991	56.6	60	58.37	70.046596	87.55825
1992	44.5	60	45.89	55.0719704	68.83996
1993	60	60	61.88	74.2543421	92.81793
1994	64	60	66.00	79.2046315	99.00579
1995	50.8	60	52.39	62.8686763	78.58585
1996	69	60	71.16	85.3924934	106.7406
1997	64	60	66.00	79.2046315	99.00579
1998	50.8	60	52.39	62.8686763	78.58585
1999	45.3	60	46.72	56.0620283	70.07754
2000	71.1	60	73.33	87.9913953	109.9892
2001	50	60	51.57	61.8786184	77.34827
2002	45.7	60	47.13	56.5570572	70.69632
2003	39.6	10	58.44	70.1287406	87.66093
2004	51.1	10	75.41	90.4944103	113.118
2005	43	10	63.46	76.1498951	95.18737
2006	46	10	67.89	81.4626785	101.8283
2007	39	10	57.56	69.066184	86.33273
2008	50	10	73.79	88.5463897	110.683
2009	29.8	10	43.98	52.7736483	65.96706
2010	43.9	10	64.79	77.7437302	97.17966

The rose diagram for direction of wind is shown in Fig. 5.3.1 [113].

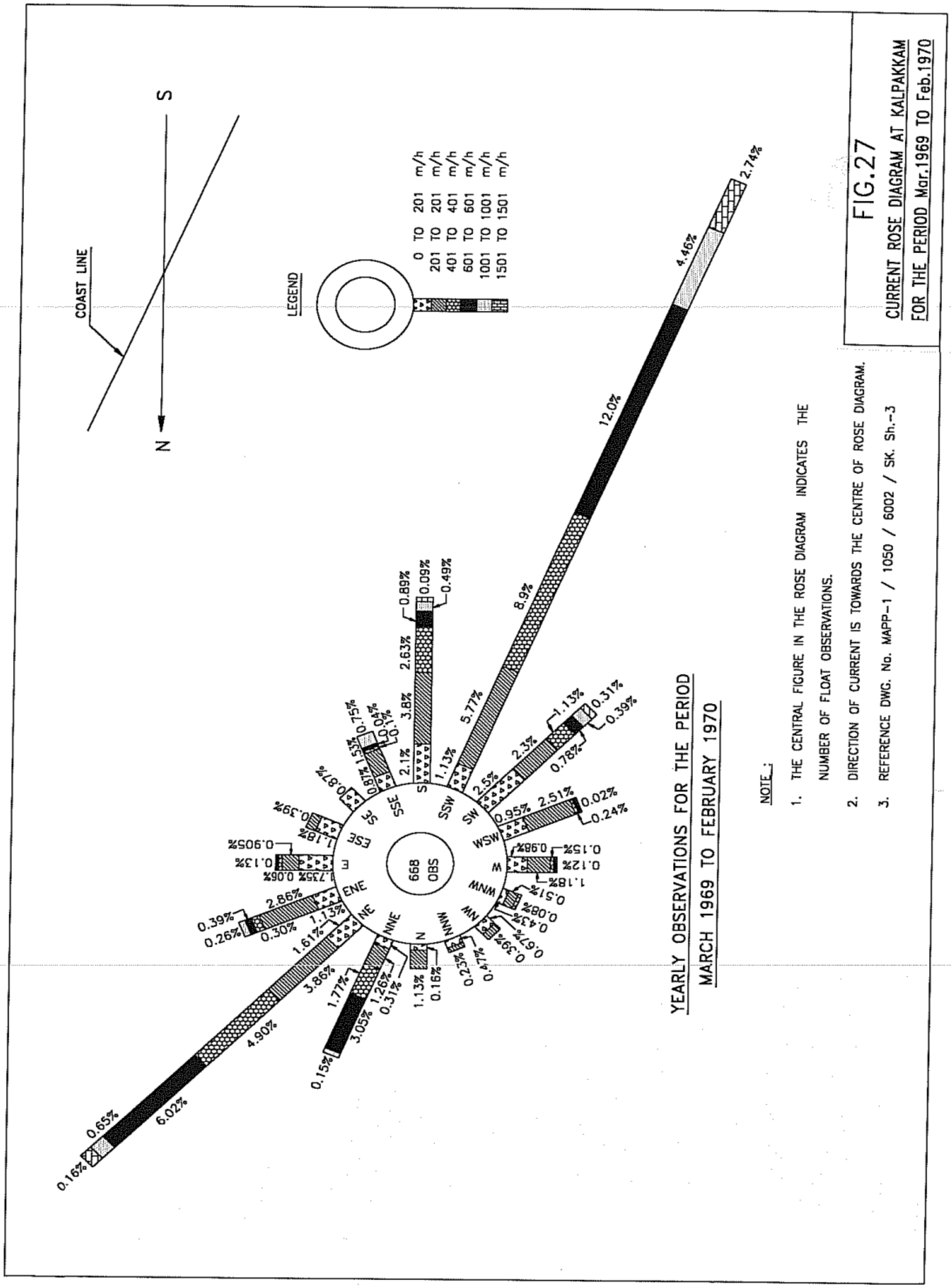

Fig.5.3.1: Rose diagram of wind for Kalpakkam

5.3.1.4 Extreme value analysis for wind

Extreme value analysis of wind speed for Kalpakkam has been performed and wind speed data has been fit using power law distribution and Generalized extreme value distribution (GEV). The Exceedance probability vs wind speed (km/h) for power law fit is shown in Fig.5.3.2. The Exceedance probability vs wind speed (km/h) for GEV fit is shown in Fig.5.3.3

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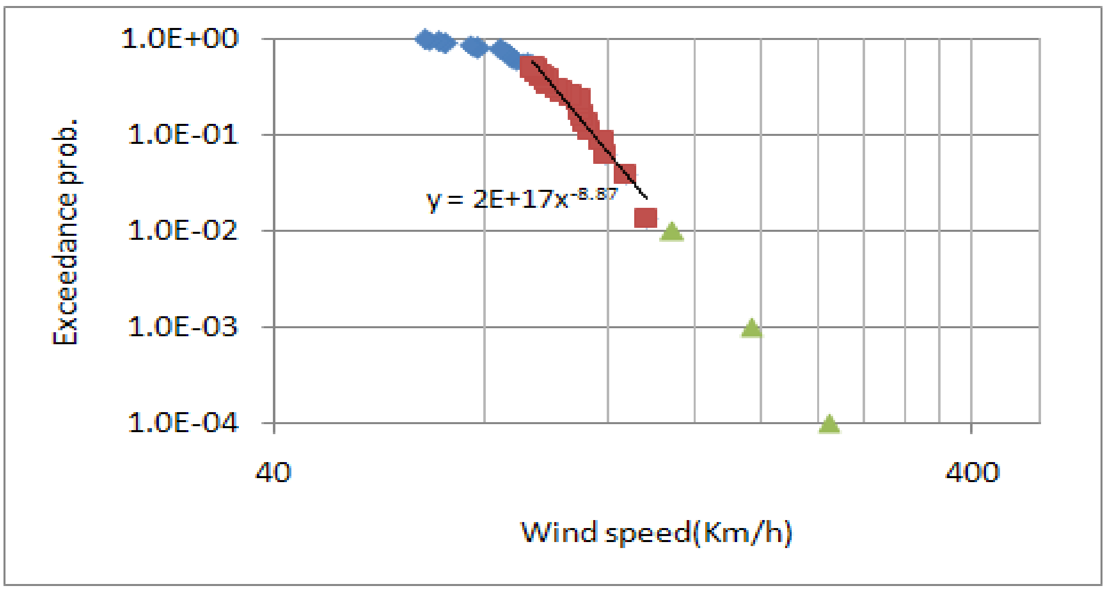

Fig. 5.3.2: Exceedance Probability vs Wind speed (Power Law fit)

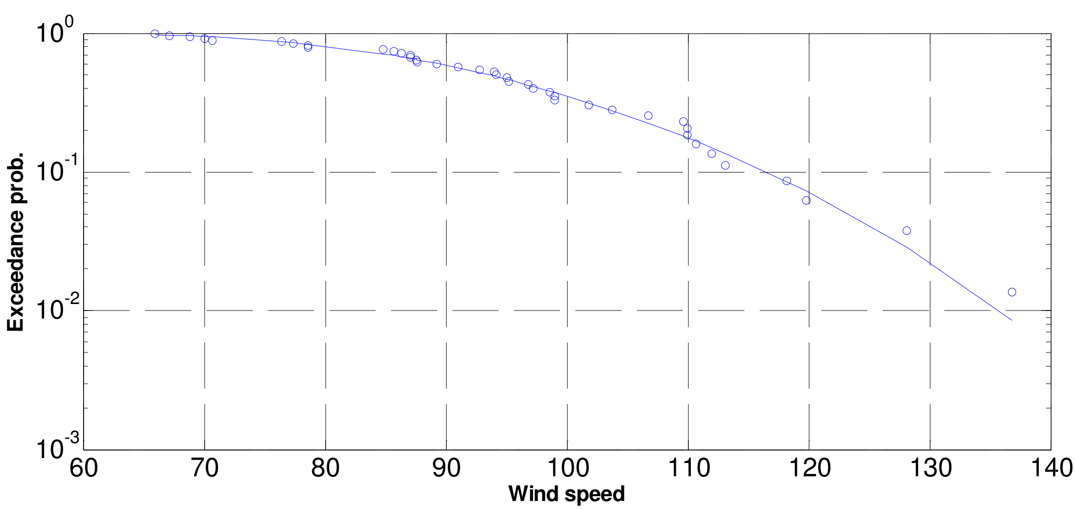

Fig. 5.3.3: GEV fit for wind data vs Exceedance probabilty

The parameters for GEV fit are shown in Table 5.3.2. The estimated wind speed from power law fit is shown in Table 5.3.3. The estimated wind speed from GEV fit is shown in Table 5.3.4.

Table 5.3.2: Parameters for GEV distribution

	Parameters for GEV		
	k	σ	μ
Colum 1	-0.1853	10.2557	58.7938
Colum 2	-0.1853	12.3072	70.5527
Colum 3	-0.1853	15.3840	88.1909

Table 5.3.3: Wind speed (Power law fit)

	Power law fit		
Return period (years)	100	1000	10000
Colum 1	98.94	121.09	149.96
Colum 2	128.27	156.97	194.41
Colum 3	149.12	193.26	250.48

Table 5.3.4: Wind speed (Gev fit)

	GEV fit		
Return period (years)	100	1000	10000
Colum 1	91	98	103
Colum 2	109	119	123
Colum 3	137	154	157

5.3.1.5 Results

The calculated wind speed data for Kalpakkam lies in the range of 137 km/h- 149 km/h for 100 years return period. The calculated wind speed data for Kalpakkam lies in the range of 154 km/h- 193 km/h for 1000 years return period. However the basic wind speed has been taken as 180 km/h for a return period of 50 years [113]. This is applicable for a structure upto 10 m in height and life of 50 years. Since no part of DG is located above 10 m surface, no study is warranted in this regard. Wind may cause snapping of transmission line to grid, but this failure probability is included in grid failure. Otherwise no component of class III is affected by wind.

The maximum wind data calculated for Chennai is 145km/h for 100 years return period and 177km/h for 1000 years return period [111].

5.3.2 Aircraft crash hazard assessment

IAEA safety series defines [114] the following two preliminary screening approaches as part of acceptance criteria

- Screening distance value approach
- Screening probability level approach

Aircraft hazard may be dismissed in the initial condition if the proposed site does not lie within the Screening distance value approach.

5.3.2.1Screening distance value approach

The potential hazard arising from aircraft are to be considered if any one of the following criteria hold true.

i) Airways or airport pass within 4 km of the site

ii) Airports are located within 10 km of site for all but the biggest airport.

iii) Military installation or air space usage such as bombing or firing range within 30 km of plant site.

For Kalpakkam site none of the above conditions are applicable due to flowing reason.

No airport within 10 km radius (nearest airport 60 km) and nearest military installation (Tambaram air force station) is 50 km from site. Therefore detailed crash hazard analysis has been screened out based on the above criteria.

5.3.3 Lightning

Protection against voltage surges during lightning strikes is made in the form of surge limiting devices, proper grounding techniques to avoid inductive and resistive coupling (common earthing point) and segregation of cable carrying safety signal from grounding conductors [115].

5.3.4 Missile protection

By identifying, locating and arranging potential missile generating equipment and components, during design stage itself, risk of items important to safety becoming primary missile targets has been minimized/eliminated.

Protection against missile for safety related systems, component and structure have been designed in accordance with IAEA safety guide 50-SG-D4, 1980 [116]. For PFBR there are two sources of missile generation. One is external, from MAPS and internally from PFBR. Between the options of protection through antimissile mode and layout protection means, layout design mode is favored for protection of nuclear safety related structures against turbine missile. Both PFBR and MAPS safety related structures are mutually outside the cone of influence of respective other Low Trajectory Turbine Missile [117].

5.3.5 Summary

Each hazard study has been included in modeling of safety system. However none of these events are contributing for failure of DG. The rationale being based on i) wind events: no vulnerable DG structure w.r.t. wind, ii) aircraft crash hazard assessment: based on screening distance and screen probability level approach, iii) lightning: based on surge limiting devices, iv) missile protection: based on cone of influence. Wind hazard, lightening and internal missiles have no contribution to failure of DG. For aircraft crash there is no contribution as per *screening distance approach. However, screening distance approach has limitations though the installations have defenses against aircraft crash. A detailed methodology analyzing the likelihood, type and magnitude of aircraft impact and the plant defenses would be helpful. There are additional safety features incorporated in nuclear power plant safety. For example, restriction of air space, layout of critical structures and inclusion of air defense equipment at reactor site. A detailed study in future could be carried out to justify this claim.*

Chapter 6

Summary and Future Directions

6.1 Summary

Nuclear power plants are designed to achieve high level of safety at all stages of lifetime, including extreme natural events like earthquake, flood, tsunami, wind etc. The design of these plants should be made such that workers, public and environment is protected from the harmful effects of radiations emerging from the plant. A "defense in depth" philosophy is followed in designing and operating of nuclear facilitates, which prevents and mitigates accidents that lead to the release of radiations to the environment. Though, all measures have been made to prevent and mitigate accidents during the design stage, some concerns of its residual risk to the public still exists. Hence, it is important to identify all weaknesses in reactors safety systems and mitigate consequences of it, if they fail. Nowadays , this type of analysis has become mandatory to obtain regulatory clearance of reactor operation. A comprehensive safety assessment due to internal and external events, using both deterministic and probabilistic (PSA) methods, is usually made to ensure that all requirements established for the design are met and are in accordance with relevant national and international codes and standards, laws and regulations. PSA approach is well recognized of safety assessment of NPPs, because it allows to make a better evaluation of the major contributors to this residual risk so that it can be further reduced by implementing proper corrective steps in the design.

Methodology is well established for the evaluation of core damage frequency (CDF) in NPPs due to internal and external events. But, very few studies are reported for fast reactors. In general, it is a reality that attention has not been paid enough for flooding events. The importance of this event was realized after Fukushima accident and regulators now demand a comprehensive PSA study using internal and external events including flooding. The external hazard analysis for a plant has challenges due to limited (a) site-specific data of external events and their hazard curves, (b) component fragility information and (c) suitable system models.

In this thesis, a comprehensive Level-I risk assessment (internal and external events) of a safety system (Class III power supply) of an Indian 500 MWe pool type sodium cooled fast breeder reactor (PFBR) at Kalpakkam site is studied for the first time. Main emphasis has been given on the analysis of external events, viz. seismic, flood (storm, rainfall and tsunami) and wind. Analysis covers procedure for identification, categorization, screening, quantification and PSA modeling. This study helped to develop methodology for internal and external hazard of a safety system for an Indian NPP. It resulted in development of hazard curve for Kalpakkam site and will be useful for upcoming NPPs at this site. Details of the analysis and the results obtained are briefly summarized below:

Internal Event PSA

The internal events modeling of Class III system has been carried out first time for sodium cooled fast breeder reactor (PFBR). Impact of support systems, viz. Safety related service water system, fuel oil system and circuit breaker control power supply, dependency have been studied. Safety related service water system contributes about 2 % failure of DG. Common cause failure has been estimated using beta factor and alpha factor model. DG fails to run due to CCF is dominant contributor of DG failure. Importance analysis is made to identify significant contributors to unavailability by estimating Fussell-Vesely, Birnbaum, Barlow-Proschan and Sequential importance measures. It is found that DG fails to run, DG fails to run due to CCF and

DG under maintenance are critical basic events for DG system. The relative importance of individual component failure has been studied by performing the sensitivity analysis and found that mechanical failure to run is most sensitive part. Monte Carlo based uncertainty analysis has been carried out to determine confidence bound for unavailability. The analysis indicates that the DG unavailability is uncertain by Error Factor 4.4 (90% confidence bound) for 2 out of 4 DG system (system success) and by Error Factor 4.1 (90% confidence bound) for 1 out of 4 DG systems (system success). This confidence bound estimation, shows that the reliability of DG is not deviating much from reported values. All this study including support system modelling, common cause failure analysis, importance, sensitivity and uncertainty analysis has helped in better estimation of class III reliability. Failure of one DG during mandatory testing (0.5/year) has been taken as initiating event frequency for computation of failure frequency. Failure frequency contribution of Class III power supply due to internal events is 1.73E-8/ry.

External Event PSA

Seismic Probabilistic Safety Analysis model has been developed for assessment of the seismic fragility of various safety systems, structures and components and integration of seismic hazard with fragility information through appropriate logic models. Detailed probabilistic seismic hazard analysis has been carried out first time for Kalpakkam site by using published earthquake data obtained from various catalogues containing several years of data. By using attenuation relation developed for peninsular India, peak ground acceleration has been determined. The PGA value has been determined for various return period (0.08 g for 100 y and 0.16 g for 1000 y return period). Fragility analysis using 'Zion method' has been carried out for various safety related equipment with generic median acceleration capacity. Detailed plant logic models, suitable for seismic risk analysis has been developed. By incorporating hazard curve, fragility data and system modelling, failure frequency of Class III power system has been estimated. Total frequency for class III power supply failure due to seismic events is 1.36E-06/ry. This study helped to develop the hazard curve for Kalpakkam site and can be used for upcoming NPPs at this site.

External Flood Probabilistic Safety Analysis (EFPSA) has been performed for Kalpakkam site incorporating lesson learnt from Fukushima accident. EFPSA consist of hazard estimation, fragility estimation and incorporating plant logic model suitable for flooding events. The external flooding phenomena that have been considered are a) ocean flooding from storm surge including wind induced waves, b) ocean flooding from tsunami and c) flooding from heavy precipitation. In this analysis, hazard curve for Kalpakkam site (storm surge, rainfall, and wind) has been developed based on data obtained from annual maximum value (e.g. 1901-2004 AD years of rainfall data obtained from IMD, Pune; Hourly tide gauge data from Chennai for a period from 1974-1988 for storm surge; and wind hazard data from 1891-2000 of observed cyclonic data covering IMD stations around Kalpakkam) based on asymptotic extreme value analysis. The hazard analysis has been carried out by using extreme value analysis, which helped to obtain hazard curve. For this, external hazard data has been fitted using Gumbel, exponential and power law fit and return period has been estimated. The analysis shows that the failure frequency contribution from storm surge and rainfall is very small. To check stationarity of rainfall at Kalpakkam and other NPPs sites in India, a mathematical model based on exponent variation and L-moments has been developed. Stationarity analysis shows that rainfall pattern is stationary in both coastal and inland regions. The coastal regions show intense rainfall and higher variability than inland regions. Eastern coast shows higher variability in rainfall than western coast. This analysis will be helpful in keeping extra safety margin for flooding in coastal regions (particularly eastern coast). To calculate level rise at plant site due to rainfall, mass flow continuity equation has been used. The study shows importance of incorporating catchment areas and plant area in flood hazard estimation. The tsunami hazard governs failure frequency contribution of class III power supply system of PFBR. The component fragilities were modeled with step fragility function considering submergence mode failure and random failure of components. The class III power supply failure due to external flood for PFBR is calculated by quantifying the event trees at different elevations (30-40 EL). Total frequency for class III power supply failure due to flooding events is 2.22 E-09/ry. In addition, wind events (based on extreme value analysis), aircraft crash hazard assessment (based on screening value analysis), lightning (based on surge limiting devices), and missile protection (based on cone of influence) has been included in modeling of safety system. None of these events are contributing for failure of DG.

Finally, the total failure frequency of class III power supply system under internal and external events is summation of above mentioned failure frequency, which is 1.38E-06/ry.

The major outcome of the thesis is the development of a methodology to perform risk assessment of a safety system for fast reactors under internal and external events by which a comprehensive risk assessment has been made possible for a class III power supply system of PFBR for the first time. This study helped to develop hazard curves for Kalpakkam site for applications to present and future fast breeder reactors. Another important outcome of this study is the characterization of rainfall intensity variation across India, which will be useful in site selection and elevation selection for finished floor level for upcoming nuclear power plants.

6.2 Future Directions

The future research areas identified during the course of present research work are

i) Fire hazard PSA study of Diesel Generators.

ii) To evaluate Living PSA of Class III power supply systems as part of total PSA of plant.

iii) To evaluate more realistic flood fragility based on design specific features i.e. beyond using submergence and step function fragility in safety systems.

iv) Detailed aircraft crash risk analysis for on nuclear power plant .

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117 Chapter 4: Plant Layout. PFBR/FSAR/R-0/Mar’10

Annexure:

Fault tree and basic event data for internal event PSA of Class III power supply system

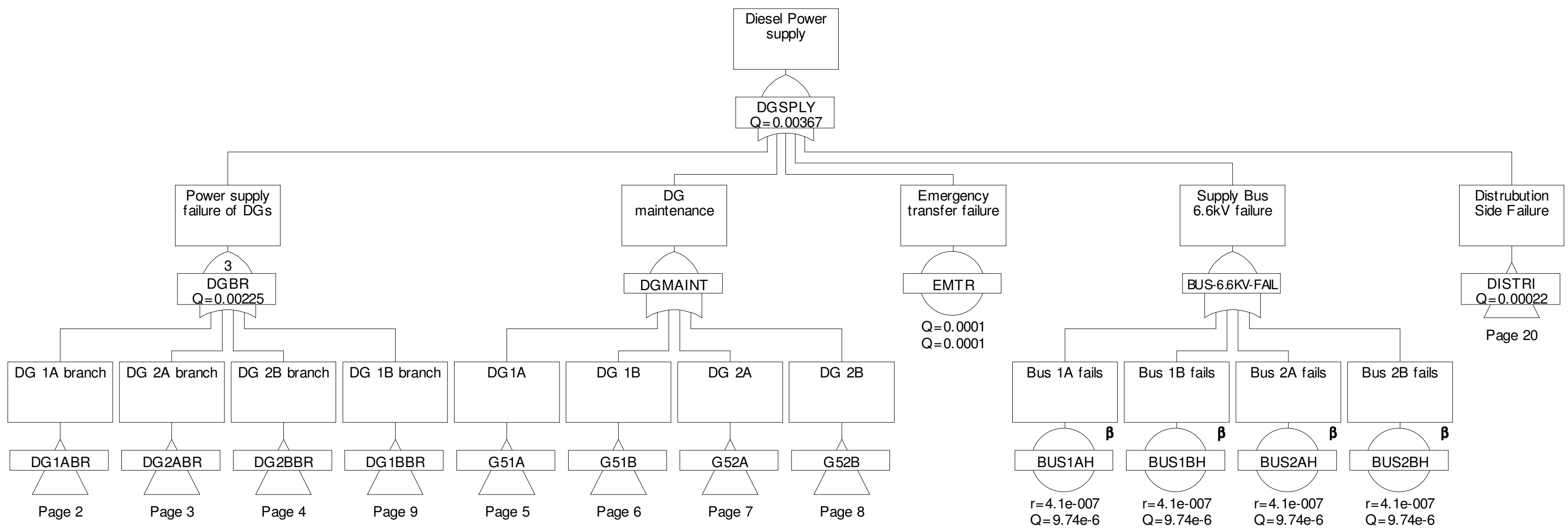


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis

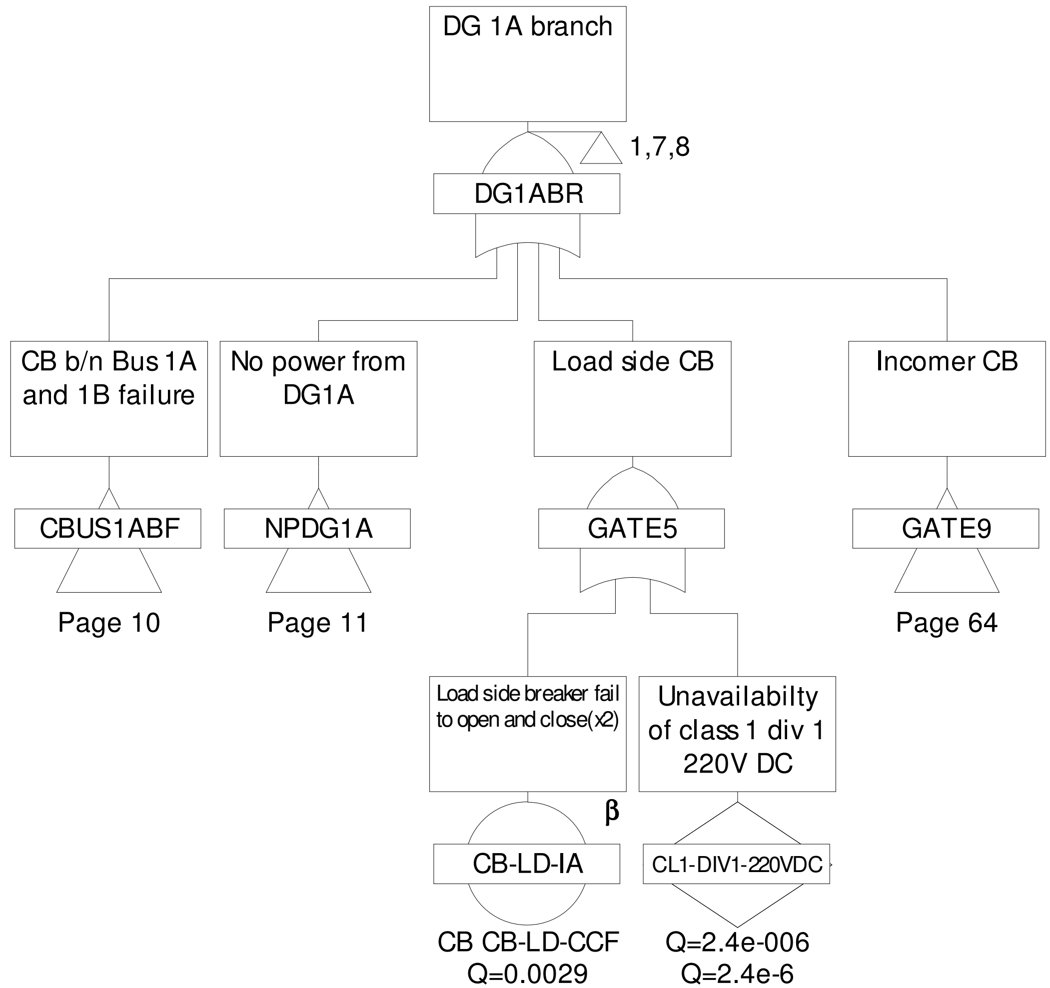


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)

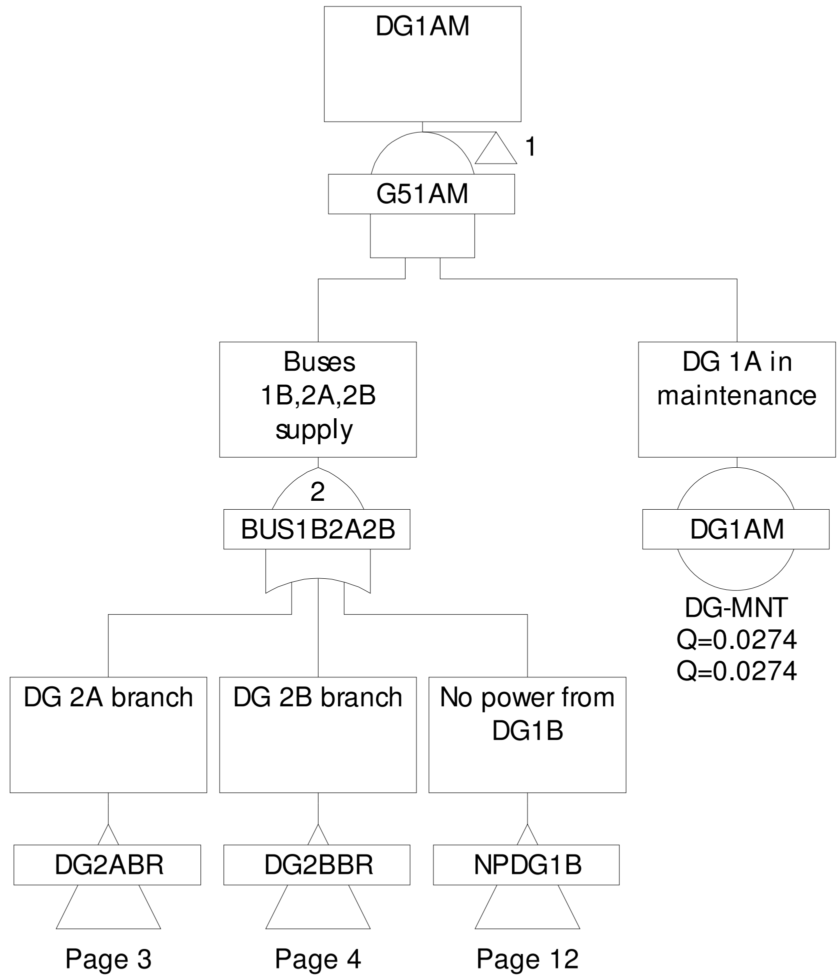


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)

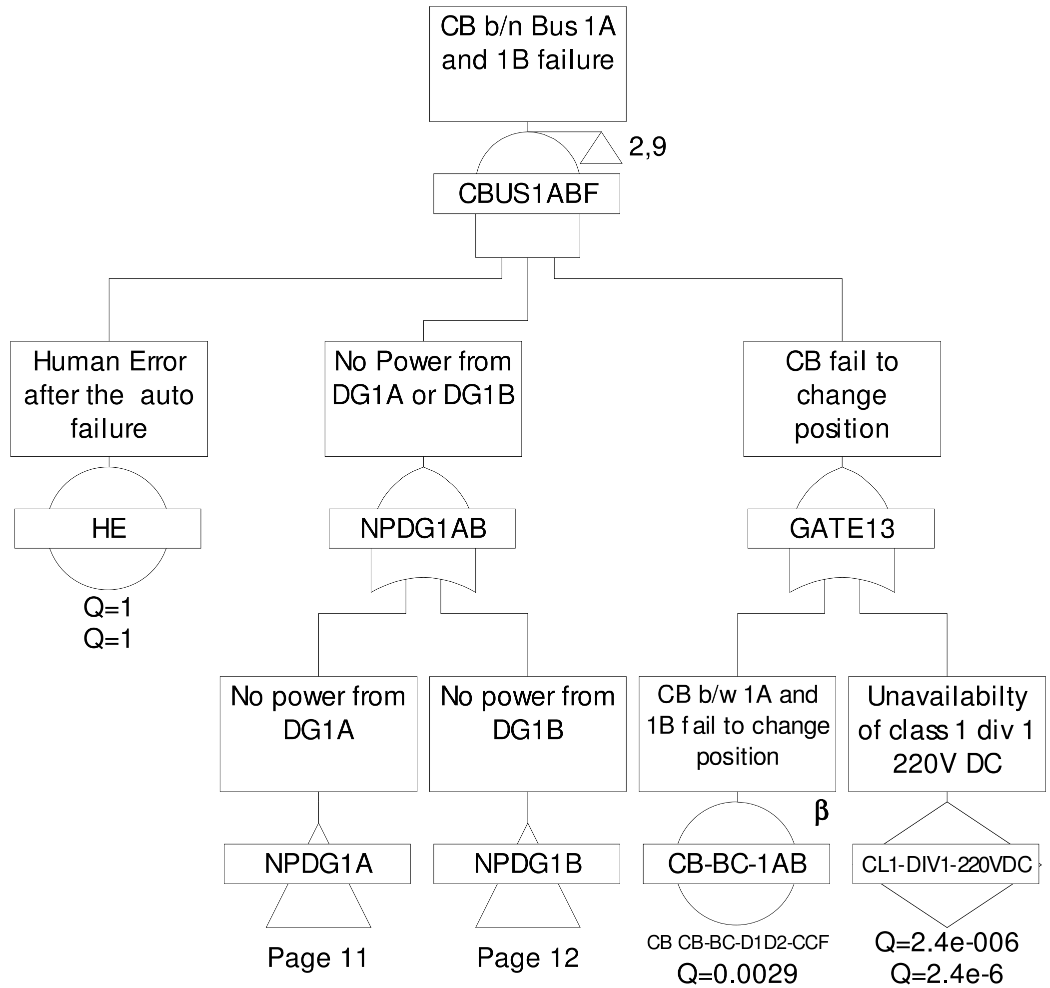
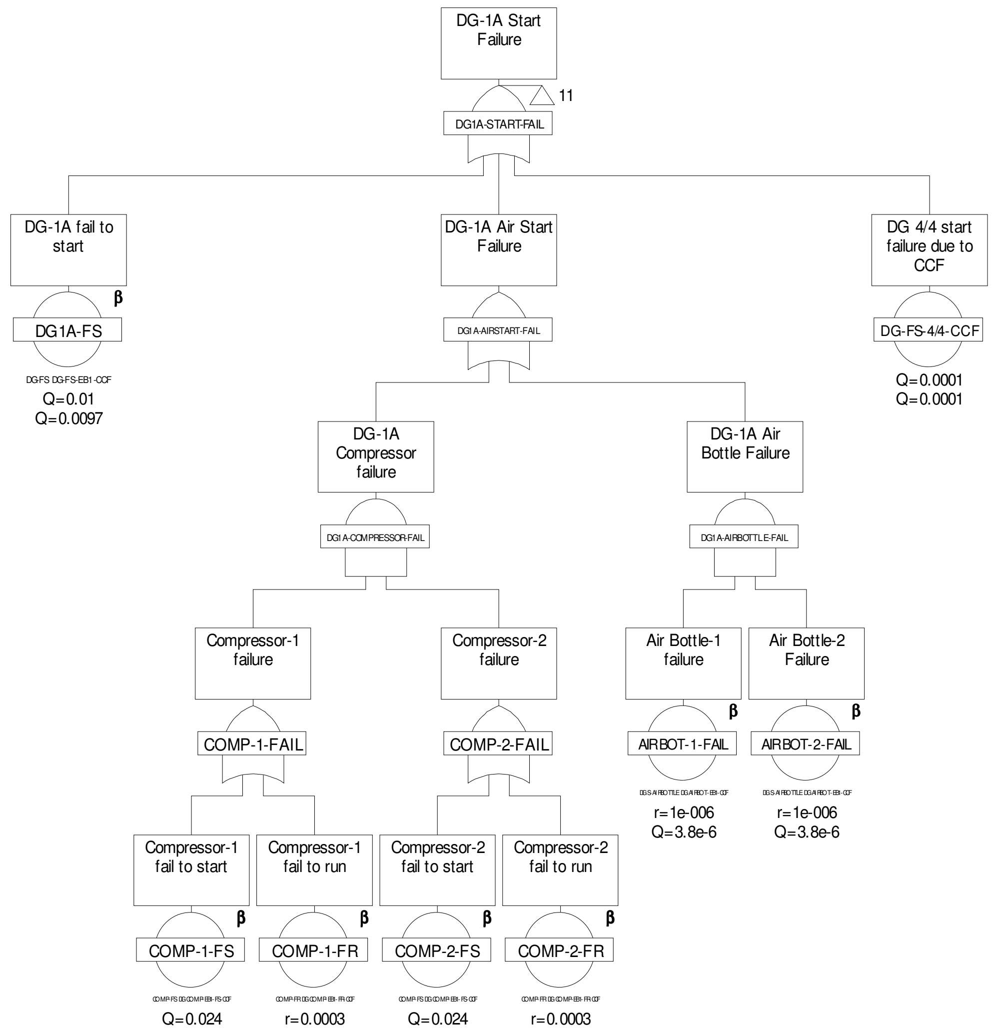


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)



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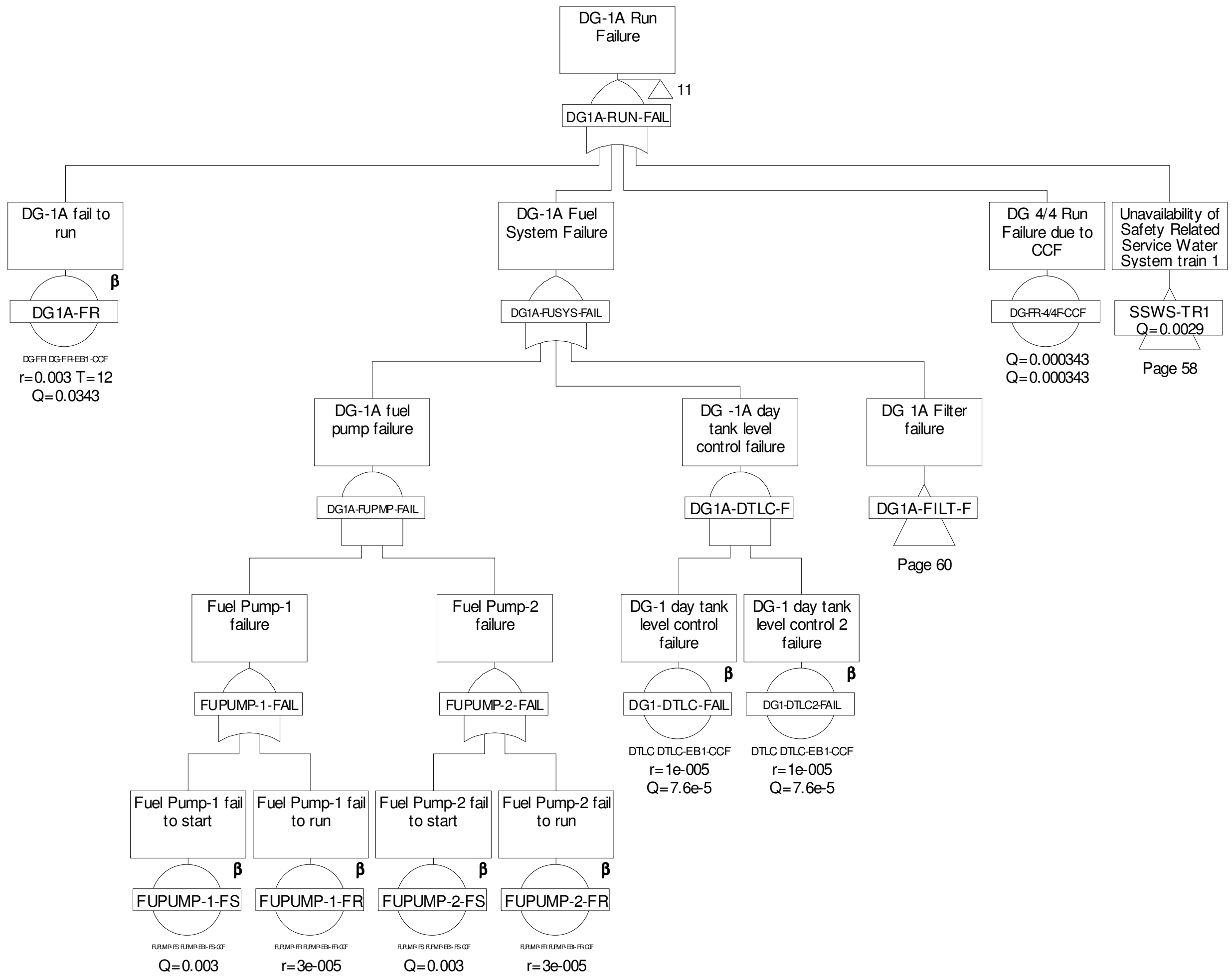

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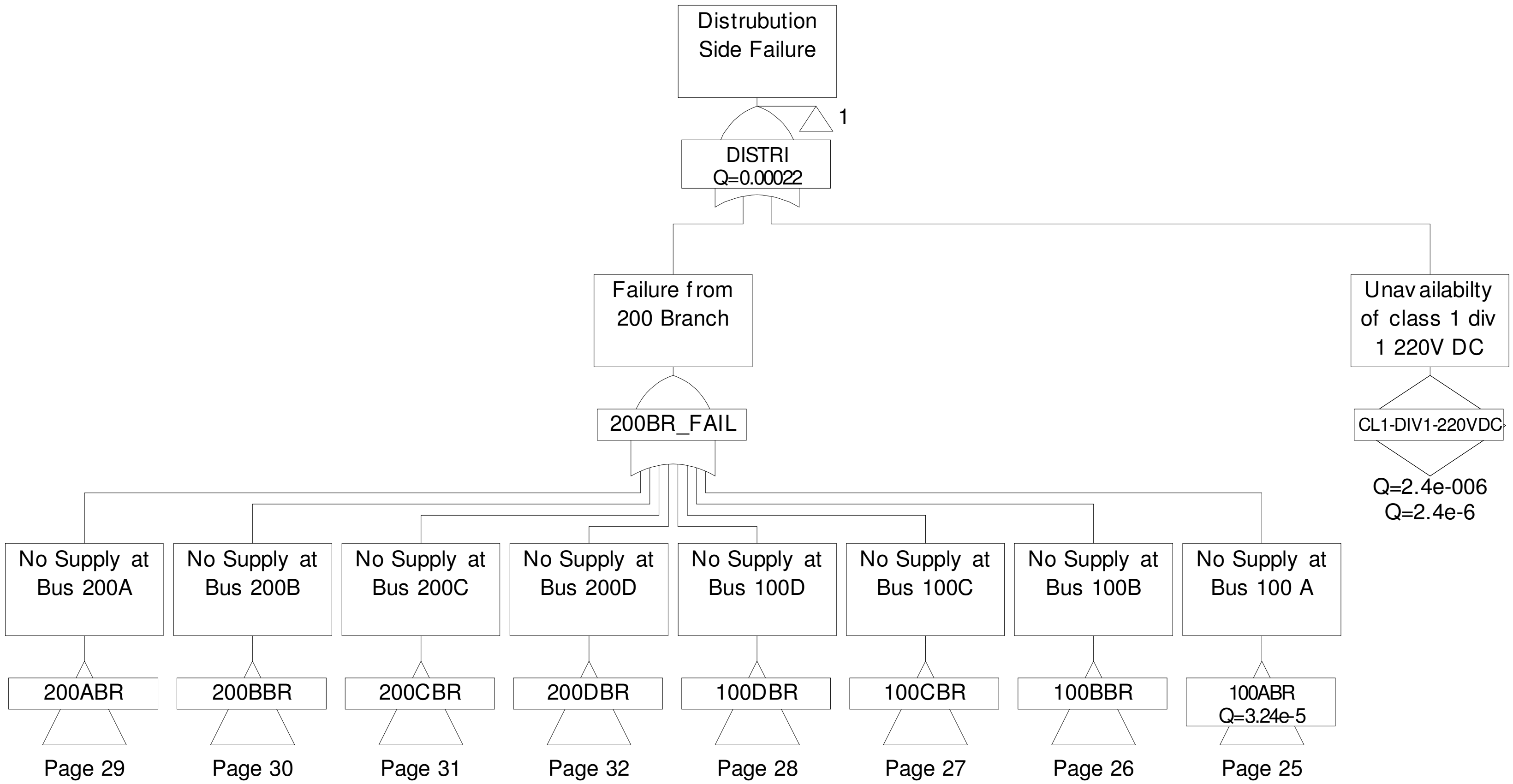

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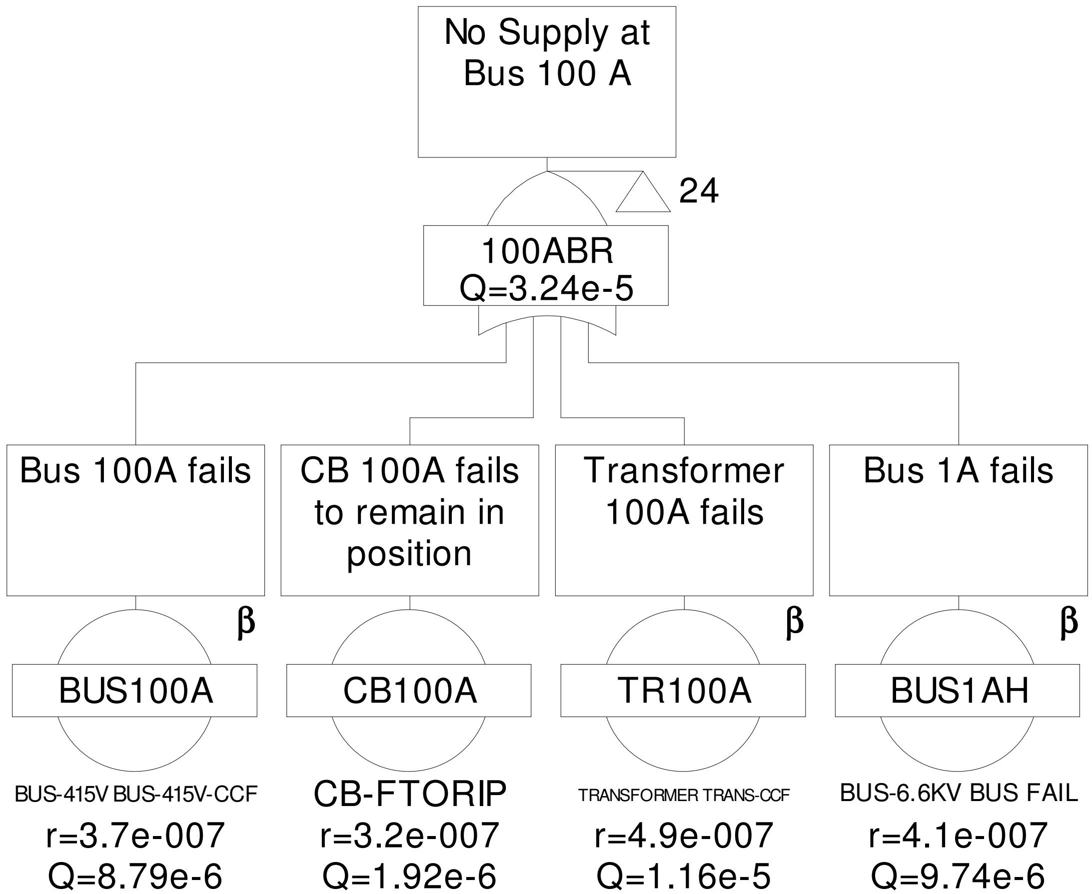

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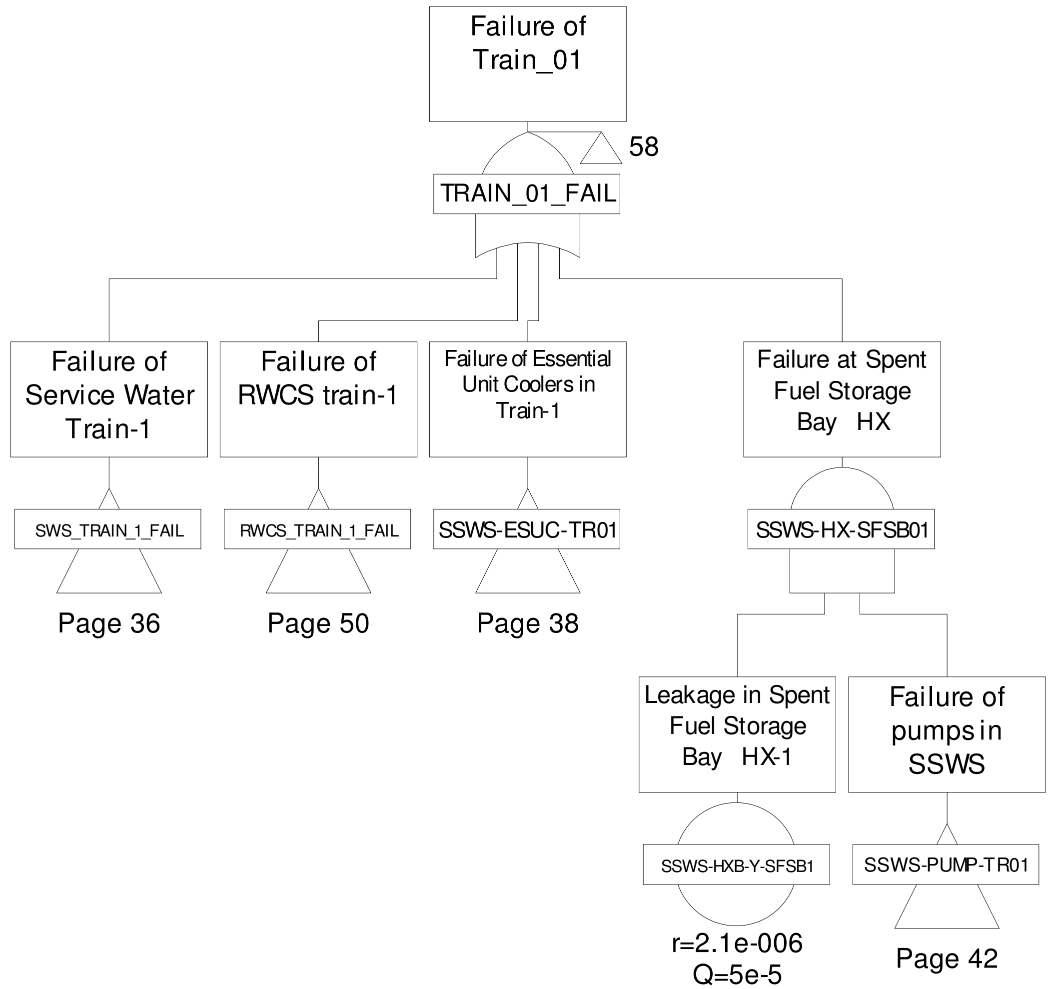
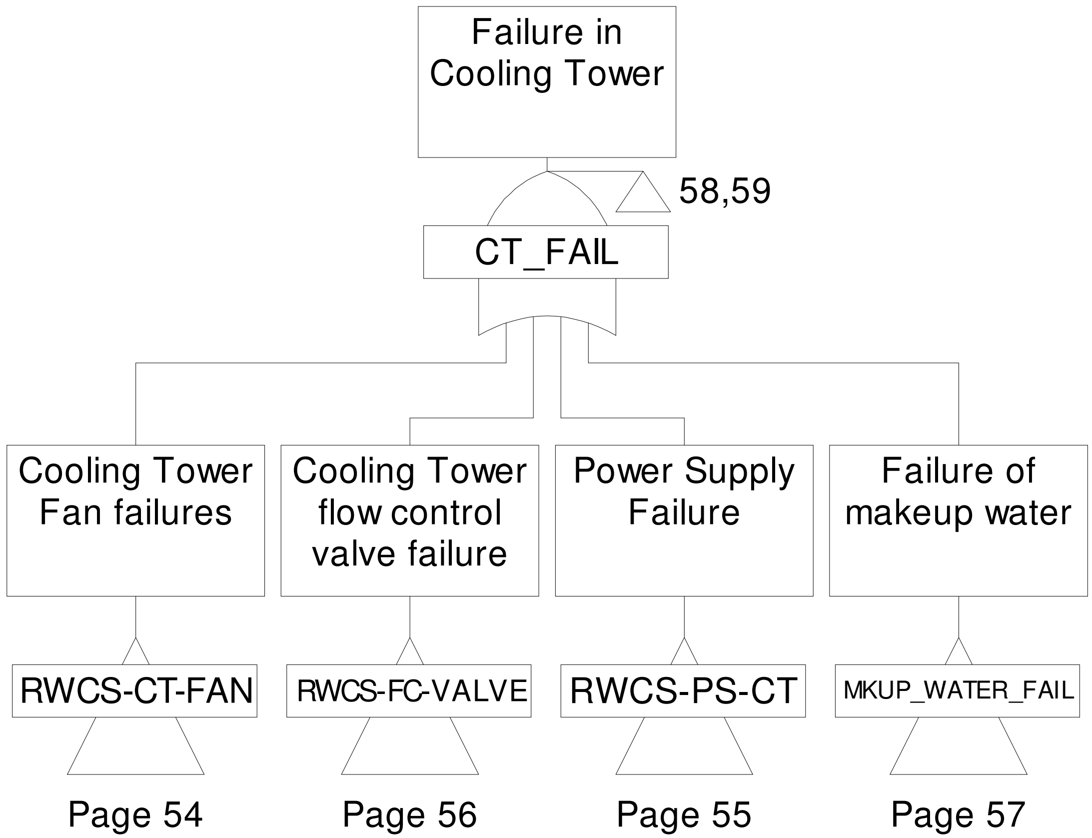


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)



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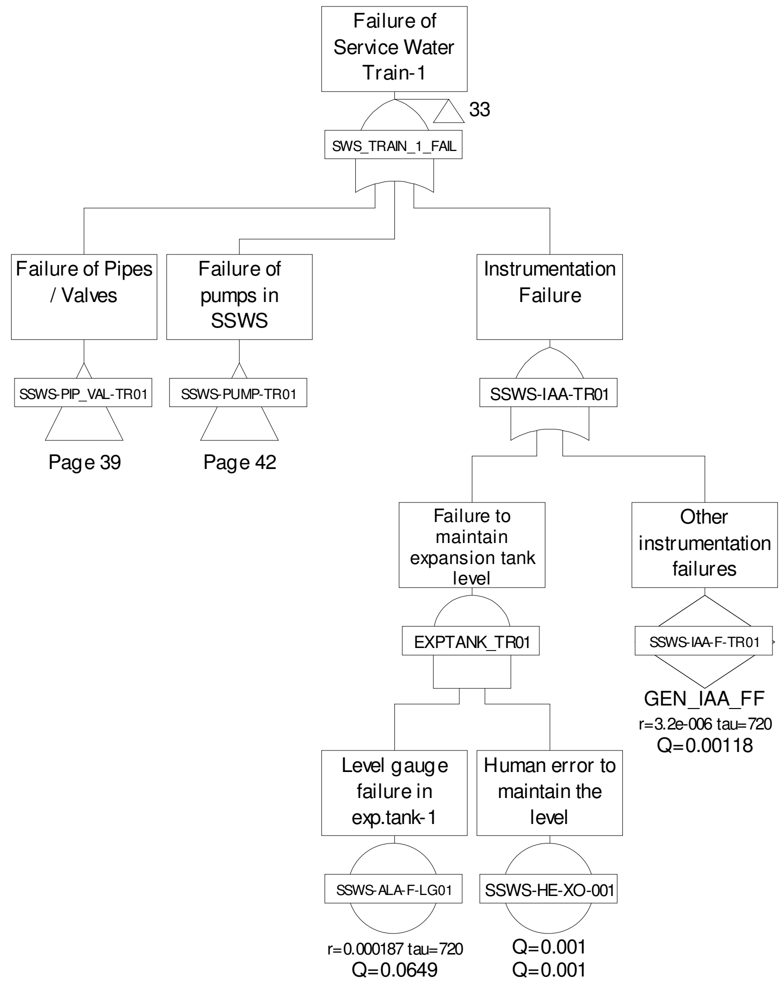

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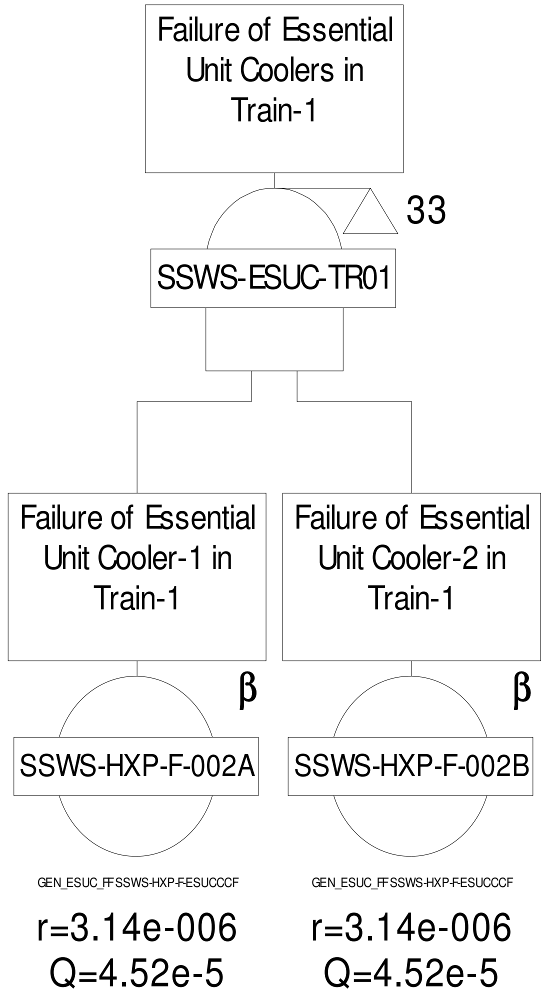

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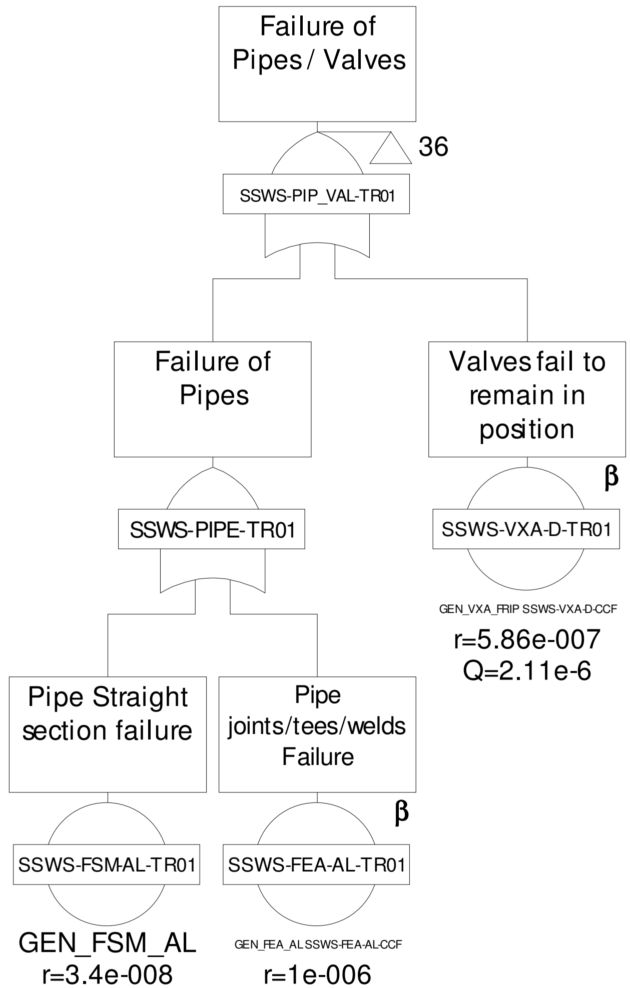
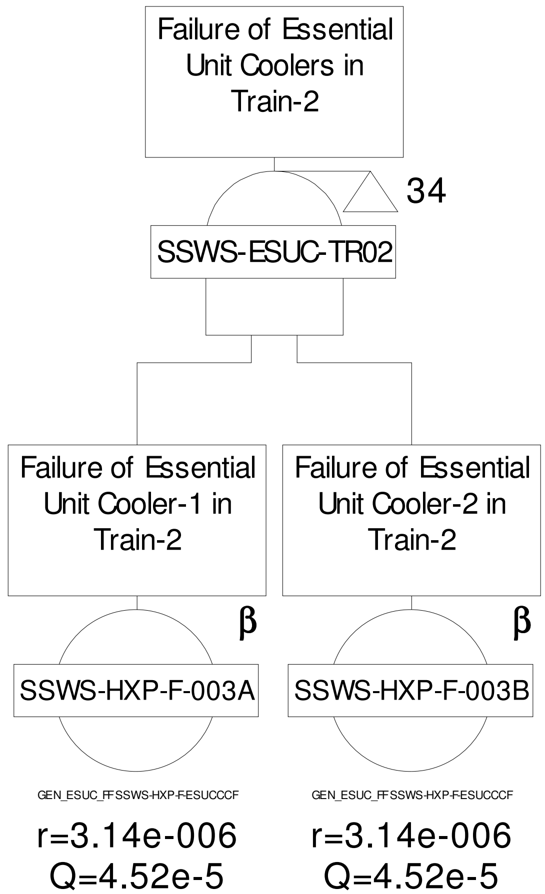


Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)



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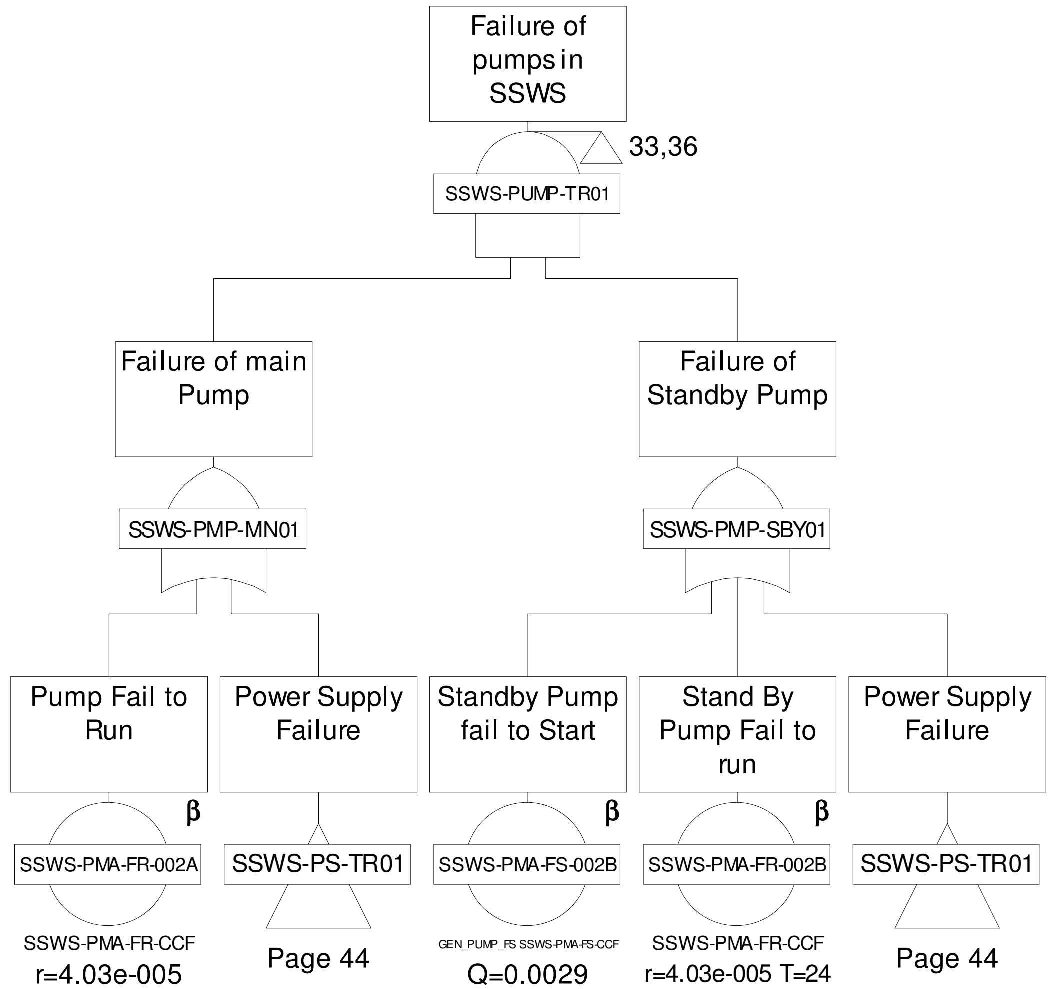

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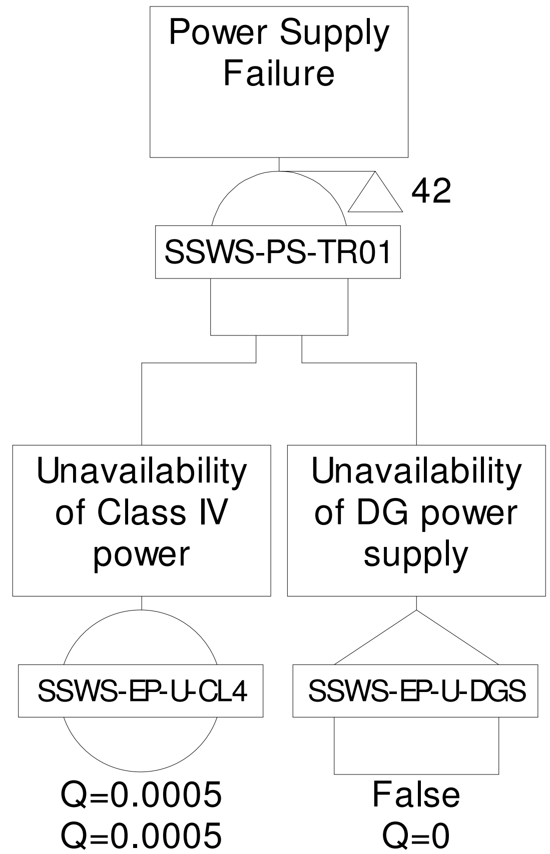

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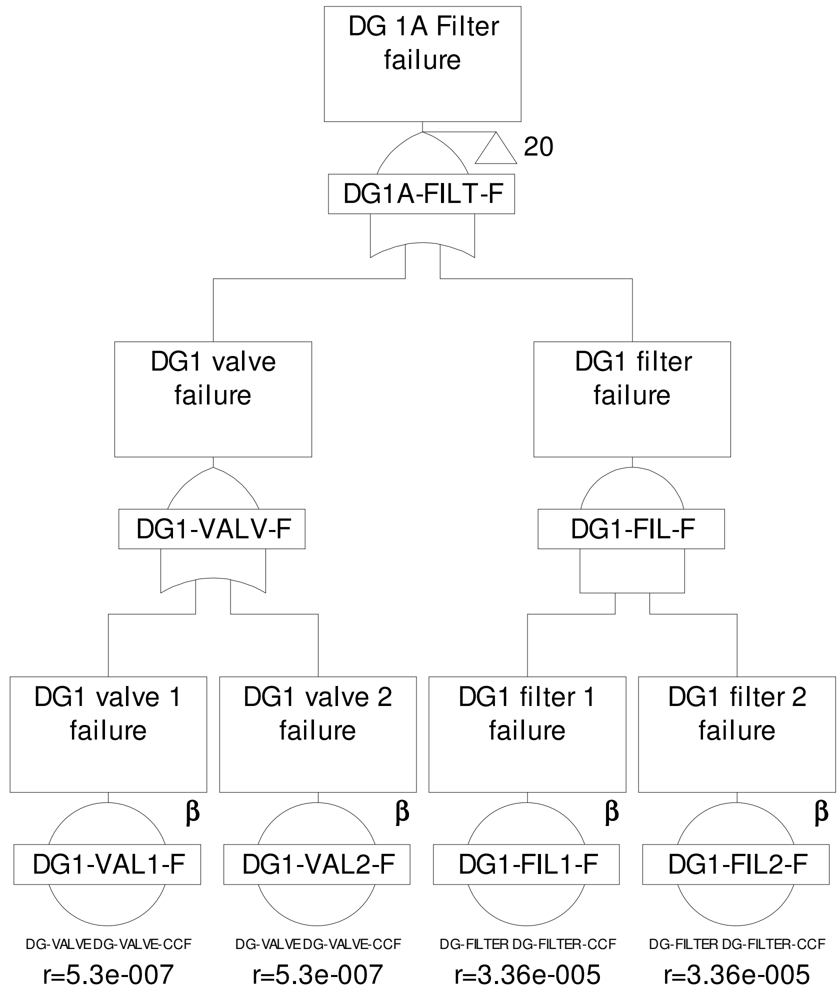

Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)

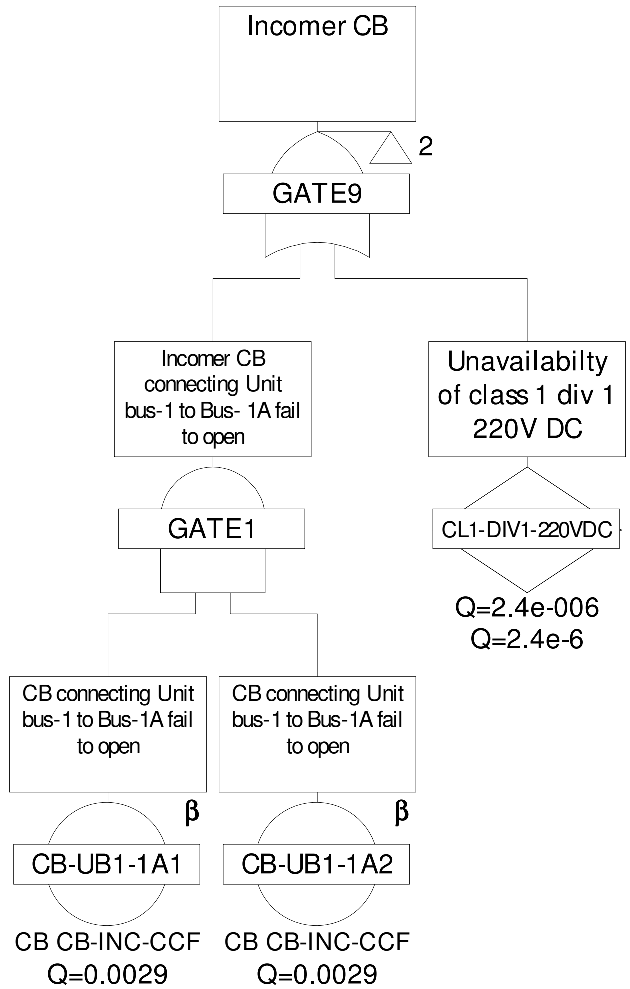

Fig 4.1: Fault Tree for Diesel Generator Reliability Analysis (Continued.)

Table: 4.1: Data Used in the Analysis of Class III Power Supply System

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
1	Bus 6.6 kV	Fail to function	BUS1AH	Rate/MTTR	4.10E-07	24		AERB/NPP/TD O-1
			BUS1BH					
			BUS2AH					
			BUS2BH					
2	Bus 415 V	Fail to function	BUS100A	Rate/MTTR	3.70E-07	24		AERB/NPP/TD O-1
			BUS100B					
			BUS100C					
			BUS100D					
3	Circuit Breaker	Fails to close/open on demand	CBIA1IE	Fixed			2.90E-03	AERB/NPP/TD O-1
			CB1B1IE					
			CB2A1IE					
			CB2B1IE					
			CBUS1					
			CBUS2					
			CBUS3					
			CBUS4					
			CB-INC-CCF					
			CB-LD-CCF					

<table>
<tr><th>No.</th><th>Name</th><th>Mode of failure</th><th>Code</th><th>Failure model type</th><th>Failure Rate (per hour)</th><th>MTTR/ Mission Time (h)</th><th>Failure prob. (/d)</th><th>References</th></tr>
<tr><td rowspan="2">4</td><td rowspan="2">Circuit Breaker (CCF)</td><td rowspan="2">Fails to close on demand (common cause)</td><td>CB100A</td><td rowspan="2"></td><td rowspan="2"></td><td rowspan="2"></td><td rowspan="2">1.45E-4 **</td><td rowspan="2">A beta factor of 0.05 is applied to CB failure data</td></tr>
<tr><td>CB100B</td></tr>
<tr><td rowspan="4">5</td><td rowspan="4">Circuit Breaker</td><td rowspan="4">Fail to remain in position</td><td>CB200A</td><td rowspan="4">Rate/MTTR</td><td rowspan="4">3.20E-07</td><td rowspan="4">6</td><td rowspan="4"></td><td rowspan="4">AERB/NPP/TD O-1</td></tr>
<tr><td>CB200B</td></tr>
<tr><td>COMP-1-FS</td></tr>
<tr><td>COMP-2-FS</td></tr>
<tr><td rowspan="4">6</td><td rowspan="4">Compressor (starting air)</td><td rowspan="4">Fail to start</td><td>COMP-3-FS</td><td rowspan="4">Fixed</td><td rowspan="7">3.00E-04</td><td rowspan="4"></td><td rowspan="4">2.40E-02</td><td rowspan="4">AERB/NPP/TD O-1</td></tr>
<tr><td>COMP-4-FS</td></tr>
<tr><td>COMP-1-FR</td></tr>
<tr><td>COMP-2-FR</td></tr>
<tr><td rowspan="3">7</td><td rowspan="3">Compressor (starting air)</td><td rowspan="3">Fail to run</td><td>COMP-3-FR</td><td rowspan="3">Rate/MTTR</td><td rowspan="3">24</td><td rowspan="3"></td><td rowspan="3">AERB/NPP/TD O-1</td></tr>
<tr><td>COMP-4-FR</td></tr>
<tr><td>DGCOMP-EB1-FS-CCF</td></tr>
</table>

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
			DGCOMP-EB2-FS-CCF					
8	Compressor (starting air) CCF	Fail to start due to CCF	DGCOMP-EB1-FR-CCF				1.20E-03	5 % of the value for Compressor fails to start on demand.
			DGCOMP-EB2-FR-CCF					
9	Compressor (starting air) CCF	Fail to run due to CCF	DG1A-FS				3.60E-04	5 % of the value for Compressor fails to run on demand.
			DG1B-FS					
10	Diesel Generator	Fails to start on demand	DG2A-FS	Fixed			1.00E-02	AERB/NPP/TD O-1
			DG2B-FS					
			DG-FS-EB1-CCF					-5.00E-03
			DG-FS-EB2-CCF					FBTR operating experience

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
11	Diesel Generator (CCF)	Fails to start on demand (common cause due to location)	DG-FS-4/4-CCF					3 % of the value for DG fails to start on demand.
			DG1A-FR				3.00E-04	
12	Diesel Generator (CCF)	Fails to start on demand (common cause due to other causes)	DG2A-FR				1.00E-04	1% of the value for DG fails to start on demand.
13	Diesel Generator	Fails to run	DG1B-FR	Time at Risk	3.00E-03	12.0(Mission Time)		AERB/NPP/TD O-1
			DG2B-FR					
			DG-FR-EB1-CCF					-3.20E-03
			DG-FR-EB2-CCF					FBTR operating experience

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
14	Diesel Generator (CCF)	Fails to run (common cause due to location)	DG-FR-4/4-CCF				1.08E-04	3 % of the value for DG fails to run
			DG1AM					
15	Diesel Generator (CCF)	Fails to run (common cause due to other causes)	DG2AM				3.60E-04	1 % of the value for DG fails to run
16	Diesel Generator	Preventive Maintenance	DG1BM	Fixed				Assumed
			DG2BM				2.74E-2*	(DG under maintenance for 10 days in a year)
			FUPUMP-1-FS					
			FUPUMP-2-FS					
17	Pump (Fuel pump)	Fail to start	FUPUMP-1-FR	Fixed			3.00E-03	AERB/NPP/TD O-1
			FUPUMP-1-FR					

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
18	Pump (Fuel pump)	Fail to run	FUPMP-EB1-FS-CCF FUPMP-EB2-FS-CCF	Rate/MTTR	3.00E-05	8		AERB/NPP/TD O-1
19	Pump (Fuel pump) CCF	Fail to start due to CCF	FUPMP-EB1-FR-CCF FUPMP-EB2-FR-CCF				1.50E-04	5 % of the value for Fuel pump fails to start
20	Pump (Fuel pump) CCF	Fail to run due to CCF	TR100A TR100B				1.20E-05	5 % of the value for Fuel pump fails to run
21	Transformer	Fail to function	TR200A TR200B TRANS-CCF EMTR	Rate/MTTR	4.90E-07	24		AERB/NPP/TD O-1

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
22	Transformer CCF	Fail to function due to CCF	DG1-FIL1-F				1.18E-07	1 % of the value for transformer fail to function
23	EMTR***	All modes	DG1-FIL2-F	Fixed			1.00E-04	Assumed(single relay failure)
24	Filter	All modes	DG-FILTER-CCF DG1-VAL1-F	Rate/MTTR	3.36E-07	8		IAEA-TECDOC-930
25	Filter CCF	Fail to function due to CCF	DG1-VAL2-F				2.68E-05	10% DG filter of fuel transfer line
26	Valve	All modes	DG-VALVE-CCF	Rate/MTTR	5.30E-07	8		NUREG/CR-1363

No.	Name	Mode of failure	Code	Failure model type	Failure Rate (per hour)	MTTR/ Mission Time (h)	Failure prob. (/d)	References
27	Valves CCF	Fail to function due to CCF					4.24E-07	10% DG valve of fuel transfer line

(de) denotes demand

** A preventive maintenance of 10 days duration per year per DG is assumed.*

*** The value is same for incomer and load side Circuit Breaker.*

**** EMTR has been assumed to be separable and value added to the top event.*

Table 4.2: List of Basic Events and Parameters for Safety Related Service Water System

No.	Name	Description	Failure Model Type	Failure rate (/hr) / Fixed Q	MTTR / Mission Time(hr)	Test Interval (hr)	Remarks
1	SSWS-ALA -F-LG01	Level gauge failure	Dormant	1.87E-4	3	1 month	Ref : BNL. Description: Float type level transducer. MTTR ref: IAEA-TECDOC-478
2	MKUP-FEA-AL-JW01	Pipes Joints/welds/tees failure in train-1	Rate	1.0E-6	8	NA	Assuming 10 joints are there in MKUP water circuit. Base-fr=9.0e-8. ref: IAEA-TECDOC-930 FR= 10* base-fr ≈ 1e-6
3	MKUP-FSM-AL-PI01	Pipe Straight section failure in train-1	Rate	3.4E-8	8	NA	Assuming 10 sections are there in MKUP water circuit. Base-fr=3.4E-9. Ref : FREDI Geometric Average FR= 10* base-fr

No.	Name	Description	Failure Model Type	Failure rate (/hr) / Fixed Q	MTTR / Mission Time(hr)	Test Interval (hr)	Remarks
4	MKUP-PMA-FR-MP	Main Pump Fail to Run	Rate	4.03E-5	24	NA	Ref : FREDI average of data whose order is occurring most no. of times
5	MKUP-PMA-FR-SP	Standby Pump fail to run	Rate	4.03E-5	24	NA	-do-
6	MKUP-VXA-D-MV1	Valves fail to remain in position in train-1	Rate	5.86E-7	4	NA	Ref: FREDI Average
7	RWCS-FEA-AL-TR01	Pipe joints/tees/welds Failure	Rate	1E-6	8	NA	Assuming 10 joints are there in RWCS train-1. Base-fr=9.0E-8.(ref : IAEA-TECDOC-930) FR= 10*base-fr
8	RWCS-HE-XO-CTF	Human error in operating the fans	Fixed	1E-3	NA	NA	Ref : IAEA-TECDEOC-592
9	RWCS-HE-XO-FCV	Human error in operating the FC valve	Fixed	1E-3	NA	NA	-do-
10	RWCS-IAA-F-TR01	Instrumentation Failure	Dormant	3.2E-6	8	1 month	Other instrumentation in train-1of RWCS. Ref : FREDI average of the data whose order is occurring most no.of times.
11	RWCS-PMA-FR-MP01	Pump Fail to Run	Rate	4.03E-5	24	NA	Ref: FREDI average of data whose order is occurring most no. of times
12	RWCS-PMA-FR-SP01	Pump Fail to run	Time At Risk	4.03E-5	24	NA	MTTR has been considered as mission time. Ref for fr : FREDI average of data whose order is occurring most no of times.
13	RWCS-PMA-FS-SP01	Standby pump fail to start	Fixed	2.90E-3/de	NA	NA	Ref : FREDI Average
14	RWCS-QBF-FR-CTF1	Cooling tower fan-1 fails to run	Rate	6.04E-5	24	NA	Ref : FREDI Average
15	RWCS-QBF-FR-CTF4	Cooling tower standby fan fails to run	Dormant	6.04E-5	24	7 days	Ref : FREDI Average
16	RWCS-QBF-FS-CTF1	Cooling tower fan-1 fails to start	Fixed	1.60E-4 / de	NA	NA	Ref: IAEA-TECDOC-930 Page-63. 8 failures in 6309 demands for 8 components

No.	Name	Description	Failure Model Type	Failure rate (/hr) / Fixed Q	MTTR / Mission Time(hr)	Test Interval (hr)	Remarks
17	RWCS-QBF-FS-CTF4	Cooling tower Standby fan fails to start	Fixed	1.60E-4 / de	NA	NA	Ref: IAEA-TECDOC-930 Page-63. 8 failures in 6309 demands for 8 components
18	RWCS-UIE-F-FCV	Status indication failure of FCV	Dormant	2.64E-6	2	1 month	Ref for FR : FREDI Average MTTR ref: IAEA-TECDOC-478
19	RWCS-UNA-F-CTA	Alarm failure for tripping of the fans	Dormant	3.78E-5	4	1 month	Ref : FREDI Average
20	RWCS-VMA-F-C1	Motor operated valve of cell-1 fail to function	Rate	2.04E-5	4	NA	Ref : FREDI Average
21	RWCS-VMA-F-SBC	MO valve of Standby cell fails to function	Dormant	2.04E-5	4	7 days	Ref : FREDI Average
22	RWCS-VXA-D-TR01	Valves fail to remain in position	Rate	5.86E-7	4	NA	Base-fr=2.93E-7/hr. Ref : FREDI Average. 4 valves with 2 in one branch in parallel with two in another branch .FR=2*base-fr
23	SSWS-DM-U-WAT	Loss of DM water	Fixed	1.00E-5	NA	NA	
24	SSWS-EP-U-DGS	Unavailability of DG power supply	Fixed	2.40E-3	NA	NA	Ref: Estimation of SBO frequency PFBR /01160/DN/1000 Rev-A, Mar-2001. DG: 2/4 mode
25	SSWS-EP-U-CL4	Unavailability of Class IV power	Fixed	1.65E-2	NA	NA	Ref : Statistics of loss of off-site power at Kalpakkam, PFBR/01160/DN/1000 Rev-A, Dec-2000
26	SSWS-FEA-AL-TR01	Pipe joints/tees/welds Failure	Rate	1.00E-6	8	NA	Assuming 10 joints are there in SSWS train-1. Base-fr=9E-8. Ref : IAEA-TECDOC-930. FR= 10*base-fr
27	SSWS-FSM-AL-COMM	Common Pipe Section Failure	Rate	3.40E-8	8	NA	Ref : FREDI Geometric Average
28	SSWS-FSM-AL-TR01	Pipe Straight section failure	Rate	3.40E-8	8	NA	Assuming 10 sections are there in SSWS train-1. Base-fr=3.4E-9. Ref : FREDI Geometric Average. FR= 10*base-fr
29	SSWS-HE-XO-001	Human error to maintain the	Fixed	1.00E-3	NA	NA	Ref : IAEA-TECDOC-592

No.	Name	Description	Failure Model Type	Failure rate (/hr) / Fixed Q	MTTR / Mission Time(hr)	Test Interval (hr)	Remarks
		level					
30	SSWS-HXB-F-BS01	Failure of BSC HX-1	Rate	3.14E-6	24	NA	Ref: FREDI Average of Generic HX. (FR of shell and tube HX : 4.8E-6 /h. IEEE-500 reported in AERB compendium of std.Generic reliability database AERB/NPP/TD/O-1)
31	SSWS-HXB-F-CT02	Failure of cold trap HX-2 (Stand By)	Dormant	3.14E-6	24	1 month	-do-
32	SSWS-HXB-Y-SFSB1	Leakage in SFSB HX-1	Rate	2.10E-6	24	NA	FR for leakage:2.1E-6/hr. FREDI Average of Data for general HX were used
33	SSWS-HXP-F-002A	Failure of Essential Unit Cooler-1 in Train-1	Rate	3.14E-6	16	NA	Ref : FREDI average of Generic HX data
34	SSWS-IAA-F-TR01	Other instrumentation failures	Dormant	3.20E-6	8	1 month	Ref : FREDI Average of the data whose order is occurring most no. of times
35	SSWS-PMA-FR-002A	Pump Fail to Run	Rate	4.03E-5	24	NA	Ref : FREDI Average of the data whose order is appearing most no.of times.
36	SSWS-PMA-FR-002B	Stand By Pump Fail to run	Time At Risk	4.03E-5	24	NA	MTTR has been assumed as the mission time for standby pumps Ref : as above
37	SSWS-PMA-FS-002B	Standby pump fail to start	Fixed	2.90E-3/de	NA	NA	Ref : FREDI Average
38	SSWS-VXA-D-TR01	Valves fail to remain in position	Rate	5.86E-7	4	NA	Base-fr=2.93e-7/hr. Ref : FREDI Average. 4 valves with 2 in one branch in parallel with two in another branch .FR=2*base-fr.