Reliability-Based Nuclear Power Plant Maintenance Resource Allocation in order to Reduce Unfavorable Operational Publicity

By

Kenichi Tezuka

M.E., Mechanical Engineering (1988)

Waseda University

Submitted to the Department of Nuclear Engineering
In Partial Fulfillment of the Requirements for the Degree of
Master of Science in Nuclear Engineering

at the

Massachusetts Institute of Technology

June 1999

Signature of Author ....
Department of Nuclear Engineering
May 19, 1999

Certified by ....
Michael W. Golay
Professor of Nuclear Engineering
Thesis Supervisor

Certified by ....
Neil E. Todreas
Professor of Nuclear Engineering
Thesis Reader

Accepted by ....
Sow-Hsin Chen
Chairman, Department of Committee on Graduate Students

Reliability-Based Nuclear Power Plant Maintenance Resource Allocation in order to Reduce Unfavorable Operational Publicity

By

Kenichi Tezuka

Submitted to the Department of Nuclear Engineering
On May 19, 1999 in partial fulfillment of the
Requirements for the Degree of Master of Science in
Nuclear Engineering

Abstract

Investigations of unplanned shutdowns at the Seabrook and the Pilgrim Nuclear Power Stations were carried out to improve the understanding of the cause of those events, which had been reported in Boston newspapers. Failures of turbine-generator systems and nuclear steam supply systems reduced plant reliability, while failures of safety-related systems did not. Risk analysis with fault tree modeling was carried out to quantify risk of major components in turbine-generator systems, and in nuclear steam supply systems. The risk was evaluated in terms of the frequency of unplanned shutdowns in order to identify which components are important to plant reliability. As a result, feedwater pumps, major valves installed on main steam lines, the chemical and volume control system, and switchyards are identified as risk significant components. Other equipment was also ranked according to their respective risk significance. Uncertainty analysis for failure rates of components was carried out using Monte Carlo approach to compare the result of fault tree analyses with historical data. Taking uncertainty into consideration, the analytical data and the historical data showed good agreement in regard to reliability over different time spans of operation. Maintenance practices for major components at the Seabrook Nuclear Power Station was investigated in order to correlate this practice and the components' risk significance. Some components, such as Feedwater Pumps and Main Steam Isolation Valves, are maintained by the condition-based methods; however, because they have high risk significance in terms of plant shutdowns, preventive maintenance is preferable for these components.

Thesis Supervisor : Michael W. Golay
Title : Professor of Nuclear Engineering

Acknowledgements

I wish to express my deep appreciation to my thesis advisor, Professor Michael W. Golay, for his constant encouragement and valuable suggestions throughout the course of this work. I would also like to thank Professor Neil E. Todreas. His insightful advice has made a great contribution to this thesis. I am also thankful to my colleagues. I enjoyed discussion and interaction with them. Special thanks are due to Tokyo Electric Power Company (TEPCO) for thesis financial support. TEPCO allowed me to have an outstanding opportunity to pursue nuclear engineering at MIT.

Table of Contents

Chapter 3. Investigation of Events

Chapter 4. Targeted Systems

Chapter 5. Development of Risk Model

Chapter 6 Failure Data

List of Figures

Number	Caption	Page

List of Tables

Nomenclature

AC	Alternating Current
AP	The Associated Press
APRM	Average Power Range Monitor
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CFR	The Code of Federal Regulations
CIV	Combined Intercept Valve
CRD	Control Rod Drive
CS	Chemical and Volume Control System
CV	Control Valve
CW	Circulating Water System
D/G	Diesel Generator
DC	Direct Current
ECCS	Emergency Core Cooling System
EHC	Electro Hydraulic Control System
EPIX	The Equipment Performance and Information System
EPRI	The Electric Power Research Institute
ESF	Engineering Safety Feature
F&MR	Failure and Maintenance Report
FCV	Flow Control Valve
FW	Feed Water System
GSC	Generator Stator Cooling System
HP	High Pressure
HPCI	High Pressure Core Injection System
IAEA	The International Atomic Energy Agency
INPO	The Institute of Nuclear Plant Operation
IPE	Individual Plant Examination
IRM	Intermediate Range Monitor
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LPRM	Local Power Range Monitor
M/S	Moisture Separator/Reheaters

MCS	Minimal Cut Set
MG	Motor-Generator
MGL	Multi Greek Model
MHC	Mechanical Hydraulic Control System
MOV	Motor Operated Valve
MR	Maintenance Request
MSIV	Main Steam Isolation Valve
NI	Neutron Flux Instrumentation
NI	Nuclear Instrumentation
NPRDS	Nuclear Plant Reliability Data system
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulatory Commission Guideline
O&M	Operation & Maintenance
OPDT	Over Power Delta T
OTSV	Oil Trip Solenoid Valve
PB	Pilgrim Station BOP System
PCV	Primary Containment Vessel
PCV	Pressure Control Valve
PDF	Probability Distribution Function
PN	Pilgrim Station NSSS System
PNPS	The Pilgrim Nuclear Power Station
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWM	Rod Worth Monitor

SAPHIRE	The Systems Analysis Programs for Hands-on Integrated Reliability
SB	Seabrook Plant BOP System
SG	Steam Generator
SJAE	Steam Jet Air Ejector
SN	Seabrook Plant NSSS System
SNPS	The Seabrook Nuclear Power Station
SRM	Source Range Monitor
SSPS	Solid State Protection System
SSW	Salt Service Water System
SWC	Stator Cooling System
TBCCW	Turbine Building Closed Cooling Water System
TCV	Temperature Control Valve
TIPS	The Traversing In-core Probe System
TSV	Turbine Stop Valve
WH	Westinghouse

1. Introduction

1.1 Background and Motivation

The licensees of the U.S. nuclear power plants have been trying to achieve high availability of nuclear power generation in order to survive economical competition with other energy resources. Achieving high availability of nuclear power plants at all times requires reducing the frequency of unplanned shutdowns, shortening outage periods, and achieving longer fuel cycles. According to the NUREG-1187 *Performance Indicators for Operating Commercial Nuclear Power Reactors Data Through September 1998* (Ref.1), the forced outage rate is still around eight percent annually, while the number of automatic reactor scrams has been decreasing over the past ten years. In addition to those technical perspectives, utilities are concerned about the reliability of those systems which are more likely to result in unfavorable publicity or to embarrass regulators. Effective maintenance practices are supposed to play an important role in order in achieving high availability and reliability. Evaluating the respective importance of system components from the perspective of the reliability of nuclear power stations is an essential ingredient in rationalizing maintenance resource allocation, and in reducing unfavorable publicity and unplanned shutdowns.

1.2 Objective

The reliability of components is one of the major maintenance indicators used to develop effective maintenance programs. Identifying reliability-critical items in terms of both shutdowns and unfavorable publicity is necessary in addition to doing this in terms of core damage frequencies. The objective of our work is to improve the understanding of how to avoid unfavorable publicity of nuclear plants as well as keeping the plants economically competitive by rationalization of maintenance resources with the resource allocation.

1.3 Overview

Chapter 2 shows the overall probabilistic methodology for reliability analysis performed in order to avoid unfavorable publicity and to rationalize maintenance resource allocation.

Chapter 3 shows the results of our investigation regarding unplanned plant shutdowns at the Seabrook and the Pilgrim Nuclear Power Stations. This information is obtained from the articles appearing in the media related to events occurring in nuclear power station, and the Licensee Event Reports (LERs).

Chapter 4 shows which systems we should focus on in terms of the risk of plant shutdowns. In order to prioritize which systems are more likely to receive unfavorable publicity, we must first categorize all reported events into major systems, such as Balance of Plant System and the Nuclear Steam Supply System.

Chapter 5 shows the application of risk analysis in the Pilgrim and the Seabrook Nuclear Power Stations, where fault trees are employed in order to describe the propagation of failures of basic events into plant shutdowns. In our work, we used the probabilistic risk assessment (PRA) code developed by Idaho National Engineering Laboratory, called The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE)(Ref.2).

Chapter 6 shows how to treat failure data which is necessary as an input of risk analysis. These data have been obtained from generic databases, such as the Institute of Nuclear Power Operations (INPO) and the U.S. Nuclear Regulatory Commission (NRC). Also, plant specific data from the Independent Plant

Examinations (IPEs; Ref.3, Ref.4) conducted in each plant.

Chapter 7 shows the result of the fault tree analyses, indicating the risk significant components whose performance can have large effects on the risk of plant shutdowns.

Chapter 8 suggests some improvements of maintenance resource allocations for the major components. In our work, systems and components which are considered to be risk-important to plant shutdown and/or unfavorable publicity are identified using risk importance measures. Taking these risk importance measures and current maintenance practice into considerations, some recommendations for improving the maintenance practice are developed.

2. Probabilistic Methodology for Reliability Analysis

2.1 Introduction

In order to avoid unfavorable publicity and optimize resource allocation, risk modeling analysis is a suitable method for power plant analysis for the following reasons.

(1) Identification of potential events leading to media coverage

An event tree and/or a fault tree can involve all plant failure events which may lead to unfavorable publicity. A complete set of events can be described in the event/fault trees, thus avoiding the omission of events of significance.

(2) Quantification of risk

All events which appear in the event/fault trees can be evaluated in the form of a minimal cut set, each with its respective quantified risk importance, such as the Risk Achievement Worth (RWA). This risk significance ranking enables analysts to prioritize the components according to their significance in causing events.

In this chapter, a probabilistic methodology to avoid unfavorable nuclear power plant publicity is described as follows:

2.2 Investigation of Events

In order to understand events and relevant public concern, we must improve our understanding of those events which have been reported in the past. This investigation consists of following steps.

(1) Selection of reference plants

One Boiling Water Reactor (BWR) and one Pressurized Water Reactor (PWR) selected for study.

(2) Surveys of past nuclear power plant publicity

Articles appearing in the media related to events occurring in nuclear power station are selected for study.

(3) Detailed description of the examined events
Descriptions of events taken from relevant reports, such as the Licensee Event Reports (LERs), are investigated to develop an accurate risk model.

The application of such investigations to the Pilgrim and Seabrook nuclear power stations is described in detail in Chapter 3.

2.3 Targeted Systems

In order to prioritize which systems are more likely to receive unfavorable publicity, we must first categorize all reported events, and take into account plant status, including power level, as one important key factor in determining whether the events occurred under specific conditions. Because a vast number of articles have been written about nuclear power stations, we selected those articles which dealt with the unplanned shutdowns of the nuclear reactors, not those which investigated related health issues or public concerns. The events examined can be categorized in terms of systems involved: Balance of Plant System (BOP) and Nuclear Steam Supply System (NSSS). The application of the categories to the Pilgrim and Seabrook nuclear power stations is described in detail in Chapter 4.

2.4 Development of Risk Model

2.4.1 Objective

One of the major concerns of utility companies is to understand how to avoid unfavorable publicity concerning nuclear plants; this understanding may also keep nuclear power generation economically competitive by reducing maintenance costs through more rational allocation of resources. In order to avoid unfavorable publicity and to optimize allocation of resources we employ a risk modeling analysis. This analysis includes the following items:

(1) Quantification of risk

All components and events can be prioritized according to their risk significance for initiating plant shutdowns. By paying attention to the most risk-significant factors, (i.e., those which are likely to initiate shutdowns), we can use limited resources more efficiently so that negative newspaper coverage can be avoided.

(2) Identification of potential events

Not all events which may lead to unfavorable publicity have occurred in power stations. Risk modeling can also identify some unrealized events which have a serious potential to cause unfavorable newspaper stories.

(3) Treatment of key factors

Key factors which affect events leading to unfavorable publicity can be quantified and evaluated in the risk model. Since some key factors, such an editor's judgement and the constraint of available newspaper space, are not physical events which occur in power stations, their quantification is one of the most challenging problems of our work.

The application of risk analysis to the Pilgrim and Seabrook Stations is described in Chapter 5.

2.4.2 Risk Model

Fault trees and/or event trees are usually used to model how basic events propagate into serious, plant-wide consequences in nuclear stations. A fault tree is used to model component-based failures, such as a pump failure or heat exchanger leakage. An event tree is used to model plant-wide events, such as reactor core damage. Fault tree models developed in this paper are described in chapter 5.

2.4.3 Analysis Code

Creating fault trees and event trees and quantifying the risk significance require the use of risk analysis codes because of the complexity of the tree structure. The analysis code must be able to do the following:

- work on microcomputers or workstations
- create fault trees and event trees easily
- calculate several parameters, such as the probability of initiating top events and minimal cut sets, and risk importance measures.

In our work we used the probabilistic risk assessment (PRA) code developed by Idaho National Engineering Laboratory, called *The Systems Analysis Programs for Hands-on Integrated F iability Evaluations (SAPHIRE)* (Ref.2). This code is a set of microcomputers programs that were developed by in order to create and analyze PRAs, primarily for nuclear plants.

The application of risk analysis code to the Pilgrim and Seabrook stations is described in Chapter 5.

2.5 Failure Data

Failure data is necessary as an input of risk analysis. Many sources are available for risk analysis in PRA fields. The following are typical considerations for choosing failure data.

(1) Type of data

- Component-based data, Event-based data
- Data on modes of failure
- Data on common cause failures and human errors
- Point estimates, probability density function(pdf)

(2) Sources of data

- Database of Institute of Nuclear Plant Operation(INPO)
- In-house data of individual plants
- The U.S. Nuclear Regulatory Commission

(3) Selection

A refined set of data is the key to successful and reliable risk analysis. However, obtaining a refined set of data is sometimes difficult for reasons, such as insufficient data tracking or reluctance to disclose information about events.

(4) Key event factors

Quantifying the key event factors that are described in Section 3.4 is the most difficult task in this work because these factors are not quantified in any kind of database.

Detail application for the Pilgrim and Seabrook nuclear power stations is described in chapter 6.

2.5.1 Generic Data

The Institute of Nuclear Plant Operation (INPO) has been accumulating plant information including failure data. The information is systematically computerized as generic data concerning the reactors in the United States. The U.S. Nuclear Regulatory Commission (NRC) also records generic failure data for typical components in nuclear power plants (Ref.5).

2.5.2 Plant Specific Data

Each U.S. plant has already conducted an Independent Plant Examination (IPE), in which risk analysis is the essential part of the evaluation of plant safety. Some kinds of failure data are available as in-house data in each station. These data are more reliable than that of INPO and NRC because they reflect the specific plant environments.

2.5.3 Finalization of Data Set

Careful examination of data is necessary to finalize the data set for the risk analysis. Examination criteria include the following:

- Availability of data
- Accuracy of data
- Point estimate or probability density function (pdf) format

2.6 Evaluation of Risk Importance

After developing fault tree models and finalizing data sets, we calculated the probability of top events of interest and the risk significance of each basic event using the risk analysis code. Risk significance is a classification applied to basic events whose performance has an effect on plant risk (i.e., as measured by the relative probability of reactor shutdown in this paper). In the Maintenance Rule (Ref.6), risk significance is usually evaluated with following criteria:

1) Risk Achievement Worth importance measure
2) Risk Reduction Worth importance measure
3) Fussell-Vesely importance measure
4) A cut sets percentage of top events risk

In our work, these criteria are used to evaluate the risk significance of each event. Detail application for the Pilgrim and Seabrook nuclear power stations is described in chapter 7.

2.7 Strategy for the Prospective Improvement

All utility companies have limited resources for operating and maintaining (O&M) nuclear power stations. Therefore, the methodology to rationalize maintenance resource allocation is necessary for nuclear power generation to be economically competitive. In our work, systems and components which are considered to be risk important to plant shutdown and/or unfavorable publicity can be identified using risk importance measures. Taking those risk importance measures and current maintenance practice into consideration, some recommendations for improving the maintenance practice are developed. Detail application for the Pilgrim and Seabrook nuclear power stations is described in chapter 8.

3. Investigation of Events

In order to avoid unfavorable publicity, it is necessary to improve our understanding of those events which have been reported in the past. This investigation is focused on the Pilgrim and the Seabrook Nuclear Power Plants as reference nuclear power plants for Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR). The articles in the Boston Globe and the Licensee Event Reports (LERs) submitted to the U.S. Nuclear Regulatory Commission (NRC) by plant owners are two major sources of information concerning past plant events. In our work, we are focusing upon unplanned shutdowns at the Seabrook and the Pilgrim Nuclear Power Stations.

3.1 Surveys of Newspaper Articles

The archives of the Boston Globe contain many stories about these nuclear power stations. The Boston Globe has an on-line archive of its articles via its Web site. The summaries of the set of articles appearing between 1989 and 1997 are shown in Table 3.1.1.

Table 3.1.1 Boston Globe Articles relevant to the Pilgrim and the Seabrook Power Stations appearing between 1989 and 1997

Power Plant	Number of articles	Number of unplanned shutdowns reported in articles	Number of actual unplanned shutdowns	Percentage of unplanned shutdowns reported (%)
Pilgrim	96	8	25	32
Seabrook	106	8	24	33
Total	202	16	49	33

Many articles were written about the Seabrook and the Pilgrim nuclear power stations in late `80s and early `90s since both plants started or restarted commercial operations in 1990. This was a publicity-sensitive period for nuclear power generation because many people were concerned about the safety of both power stations. Sixteen of the forty-nine unexpected shutdowns that occurred during the past ten years were featured in approximately 33 % of the total number of newspaper stories that were written about the plants(the reminder involved non-shutdown events or planned shutdowns). Table 3.1.2 and Table 3.1.3 show the summaries of topics of news stories about the Pilgrim Station. Table 3.1.4 and Table 3.1.5 show the summaries of topics of news stories about the Seabrook Station. The trend of the number of the articles are shown in Fig. 3.1.1 and Fig. 3.1.2.

3.2 Surveys of Licensee Event Reports

Licensees of the U.S. nuclear power plants are required to report their events to the U.S. Nuclear Regulatory Commission (NRC) under Title 10 of the Code of Federal Regulations, Part 50, Sections 50.72 and 50.73 (10 CFR 50.72 and 50.73) within 30 days of occurrence. These reports, Licensee Event Reports (LERs), include event descriptions, root cause analyses and corrective actions taken. Most unexpected plant shutdowns are reported by plant owners to the NRC.

Since there are tens of LERs issued by both plants every year, the investigation in our work was conducted for the LERs relevant only to unplanned plant shutdowns. Forty nine unplanned shutdowns were recorded in the two plants between 1989 and 1997. In order to prioritize which systems are more likely to be involved in unexpected shutdowns, the categorization of all events into six systems is necessary as follows: stand-by safety-related

Table 3.1.2 Summaries of Topics of News Stories about the Pilgrim Nuclear Power Station in the Boston Globe betweem 1989 and 1998

Year	Unplanned shutdowns	Planned shutdowns	NRC comments	Technical problems	Health environment.	Waste disposal	Critics statement	Management finance	Other plants	General energy	Other	Annual Total
1998			1		1						1	3
1997								2				2
1996			1		1			1				3
1995	1		1					1			1	4
1994	0		1			1					2	5
1993	2	2	2				1					6
1992			4	1	2		1	2			1	11
1991	1	2	2	1	1		1				1	9
1990	1		3		11	1	3	3	1		3	26
1989	5		9	2	1		4	3		2	1	27
Total	10	4	24	4	17	2	10	12	1	2	10	96

Unplanned shutdowns	:	Unplanned shutdown and related issues
Planned shutdowns	:	Planned shutdown and related issues
NRC comments	:	Comments, warning issued by regulation(NRC)
Technical problem	:	Technical issues not leading to plant shutdown
Health environment.	:	Health and environmental concern for the public
Waste disposal	:	Issues related to waste
Critics statement	:	Statement and activities by critics and public
Management finance	:	Company management and financing
Other plants	:	Issues from other nuclear plants
General energy	:	General issue of energy resources
Other	:	Other issues

Table 3.1.3 Unplanned Shutdowns at the Pilgrim Nuclear Power Station in the Boston Globe News Stories between 1989 and 1998

Event Descriptions	Date
High temperature in cooling water associated with generator	(03/24/95)
Malfunction of a small transformer in turbine building SW/GR room	(08/31/93)(*)
Plant operator did misread a gauge of cooling water level during start up	(11/08/93)
Seal on one of two PLRpp was leaking at the rate of 2.5 Gal/min.	(04/29/91)
Malfunction of water valve in reactor cooling system and the failure of back up system	(09/02/90)
Pressure change in a small line did lead to scram during a routine test	(12/08/89)
Troubles with electrical transformer causing Troubles with a voltage regulator in T/G	(08/30/89)
Unplanned repair of electrical transformer to the main steam line to cut down electrical noise	(06/11/89)(*)
Equipment trouble	(05/03/89)
Pressure drop in steam line(two problems)	(03/04/89)

(*) These two events are not found in the Licensee Event Reports (LERs).

Table 3.1.4 Summaries of Topics of News Stories about the Seabrook Nuclear Power Station in the Boston Globe between 1989 and 1998

Year	Unplanned shutdowns	Planned shutdowns	NRC comments	Technical problems	Health environment.	Waste disposal	Critics statement	Management finance	Other plants	General energy	Other	Annual Total
1998					1							1
1997	1	1						1				3
1996			1									1
1995								3				3
1994	1										1	2
1993	4		1					1			1	7
1992			5	1				1			1	8
1991	3	1	6	1				2		2	3	18
1990	1	3	6	5		1		1	2		6	25
1989			8	9		1	5	5	3		7	38
Total	10	5	27	16	1	2	5	14	5	2	19	106

Unplanned shutdowns : Unplanned shutdown and related issues
Planned shutdowns : Planned shutdown and related issues
NRC comments : Comments, warning issued by regulation(NRC)
Technical problems : Technical issues not leading to plant shutdown
Health environment. : Health and environmental concern for the public
Waste disposal : Issues related to waste
Critics statement : Statement and activities by critics and public
Management finance : Company management and financing
Other plants : Issues from other nuclear plants
General energy : General issue of energy resources
Other : Other issues

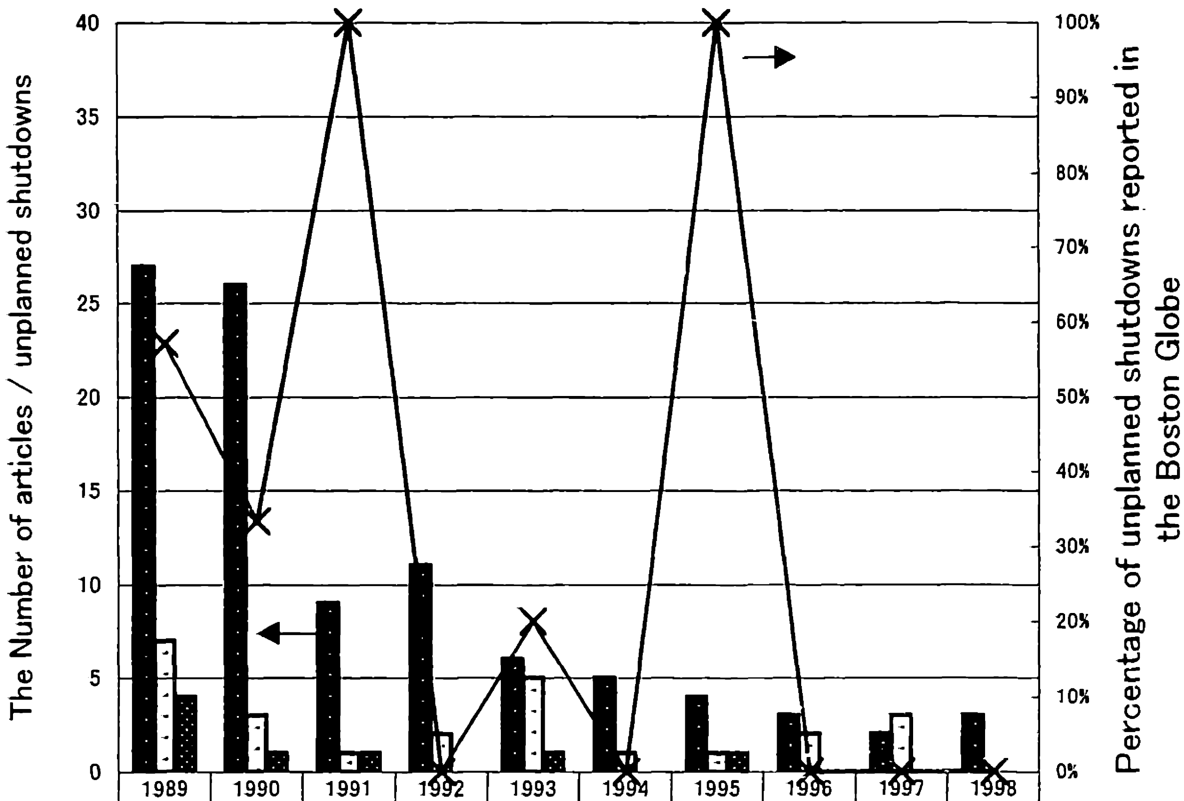
Table 3.1.5 Unplanned Shutdowns at the Seabrook Nuclear Power Station in the Boston Globe News Stories between 1989 and 1998

Event Descriptions	Date
Leaking valves in a backup cooling system	(12/05/97)(*)
Malfunction of a main steam isolation valve when it closed too far	(01/25/94)
A short-circuit in a test light circuit, a protection sys. monitored electrical prob.	(07/27/93)
A turbine valve malfunction during a test : 10% close test -> full closed	(05/20/93)
A short-circuit : 3rd time in a month	(01/14/93)
Excessive sulfuric acid got into the system	(12/27/91)(*)
An instrument line began leaking thousands of gallons of radioactive steam	(07/25/91)
A switch signaled the plants turbine control valves to close	(06/02/91)
Electricity that powers valves controlling the steam flow was interrupted, causing the valve to close	(02/12/91)
Fix a grounded circuit in the rotating portion of the generator	(06/20/90)

Full Power in July, Commercial Operation in August, 1990

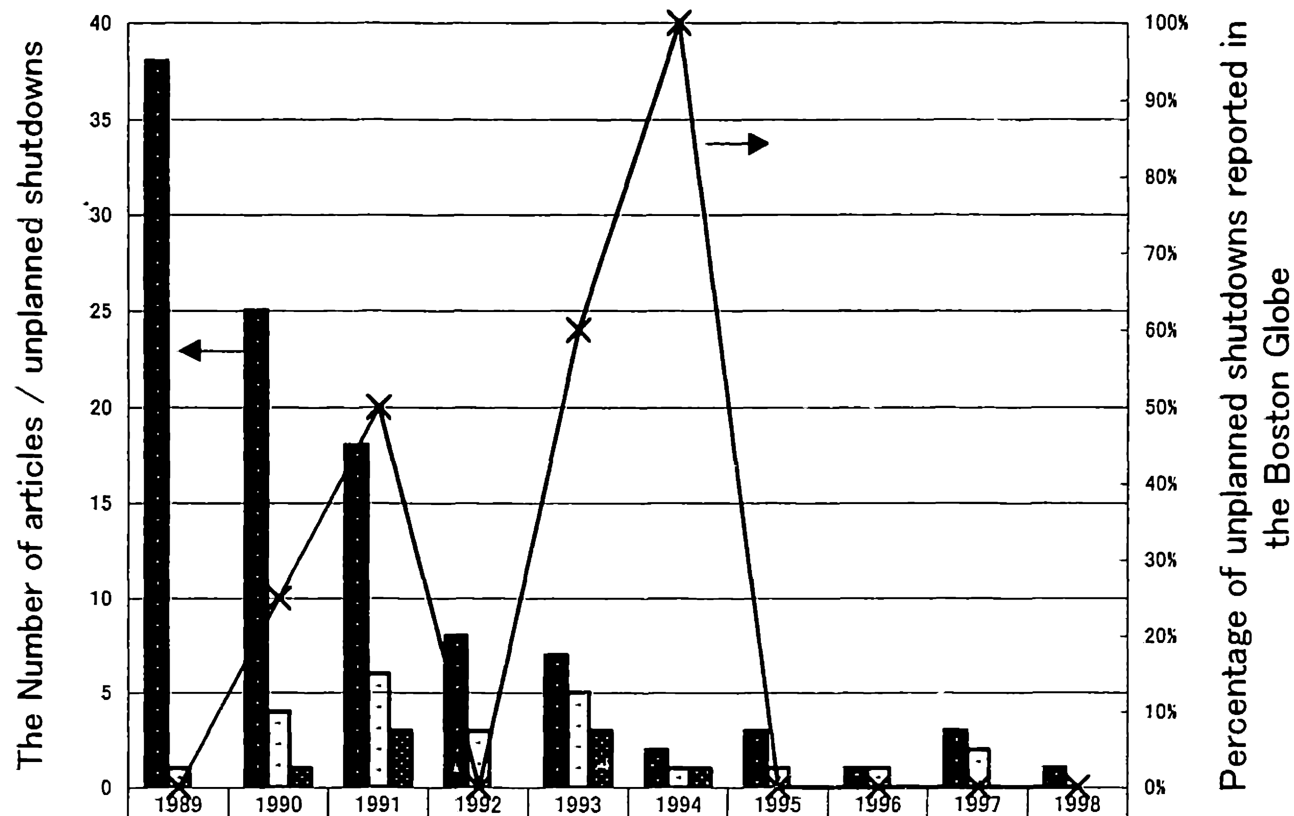
(*) These two events are not found in the Licensee Event Reports (LERs).

Fig. 3.1.1 The Number of Articles about Unplanned Shutdowns at the Pilgrim Nuclear Station in the Boston Globe during 1989 and 1998



	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998
The number of articles	27	26	9	11	6	5	4	3	2	3
The number of actual unplanned shutdowns	7	3	1	2	5	1	1	2	3	0
The number of shutdowns repoted in the newspaper	4	1	1	0	1	0	1	0	0	0
Percentage of shutdowns reported in thenewspaper	57%	33%	100%	0%	20%	0%	100%	0%	0%	0%

Fig. 3.1.2 The Number of Articles about Unplanned Shutdowns at the Seabrook Nuclear Station in the Boston Globe during 1989 and 1998



	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998
The number of articles	38	25	18	8	7	2	3	1	3	1
The number of actual unplanned shutdowns	1	4	6	3	5	1	1	1	2	0
The number of shutdowns repoted in the newspaper	0	1	3	0	3	1	0	0	0	0
Percentage of shutdowns reported in thenewspaper	0%	25%	50%	0%	60%	100%	0%	0%	0%	0%

(ECCS and ESF); balance of plant (BOP); nuclear steam supply system (NSSS); fuel handling; containment (main steam isolation valves); and external events. Dividing events into six categories makes it clear that more than 80% of the shutdowns were caused by the NSSS or the BOP system. Plant operators and maintenance engineers are often involved in the maneuver which caused the unplanned shutdown. According to the investigation, 37% of the events involves some human maneuver, and human error account for 22 % of all root causes. Table 3.2.1 and Table 3.2.2 show the summary of the investigation of LERs.

3.3 Discussion With Newspaper Editors

Through a discussion with an editor of the Boston Globe, five key factors are identified which may affect newspaper coverage in regard to the incidents which occur in nuclear power plants. These are discussed as follows.

1) Reason for shutdown

The reason nuclear power plants shut down unexpectedly is the main factor in determining whether the shutdown will be reported/discussed in a newspaper or not. If the reason is trivial, such as a trip of a non-safety related pump, the event is less likely to appear in the newspaper. On the other hand, if the event involves the failure of safety related components or radioactive release, it is more likely to be reported. There are no written criteria which determine whether an event is trivial or major. It all depends on the judgement of the editors. Table 3.3 summarizes the reason for unplanned shutdown of nuclear power plants.

Table 3.2.1 Summaries of LERs related to the Unplanned Shutdowns issued by the Seabrook and the Pilgrim Nuclear Power Staiotns during 1989 and 1997 (page 1 / 4)

Facility (a)	Event Date	LER Title	Reactor Power Level	Reported in the Boston Globe	Similar Events Y : Similar events occurred in the past	System (b)	Plant Status (C)	Human Involvment (d)	Root Cause Y: Human Error
S	6/22/89	Manual Reactor Trip during Natural Circulation Test	3			B	N/A	V	
S	6/20/90	Turbine Trip with Reactor Trip Due to Ground Fault Relay Actuation	30	Yes		B	A		
S	7/5/90	Reactor Trip Due to a Low Electrohydraulic Control Oil Pressure Signal	75			B	A		
S	8/22/90	Reactor Trip Due to Loss of Voltage on the Elecrohydraulic 24 Volt DC Bus	100		Y	B	R	M	Y
S	11/9/90	Reactor Trip Due to Steam Generator Low-Low Level Signal	100			B	R		
S	2/12/91	Reactor Trip Due to Loss of Electrohydraulic Control System Pressure	100	Yes		B	R	M	Y
S	3/30/91	Manual Reactor Trip Due to Loss of Vital Bus	100			B	R		
S	6/2/91	Reactor Trip Due to an Inadvertent Actuation of the Turbine Mechanical Overspeed Protection System	100	Yes		B	R	V	
S	6/27/91	Turbine Trip with Reactor Trip Due to an Inadvertent Actuation of Switchyard Circuit Breakers	100			B	R	M	
S	7/4/91	Reactor Trip Due to Reactor Coolant System Low Flow	100			N	R		
S	7/25/91	Reactor Coolant System Unidentified Leakage	100	Yes		N	R		
S	9/7/92	Reactor Trip Due to Steam Generator Low-Low Level	12		Y	B	D	H	Y
S	11/27/92	Reactor Trip Resulting from Spurious OPDT Signal	100			N	R		
S	12/13/92	Manual Reactor Trip Due to Unavailability of Two Circulating Water System Pumps	100			B	R		
S	1/3/93	Manual Reactor Trip Due to Loss of Feedwater	100		Y	B	R	M(H)	Y

Table 3.2.1 Summaries of LERs related to the Unplanned Shutdowns issued by the Seabrook and the Pilgrim Nuclear Power Staiotns during 1989 and 1997 (page 2 / 4)

Facility (a)	Event Date	LER Title	Reactor Power Level	Reported in the Boston Globe	Similar Events Y : Similar events occurred in the past	System (b)	Plant Status (C)	Human Involvment (d)	Root Cause Y: Human Error
S	1/14/93	Automatic Reactor Trip Due to a Phase to Ground Fault on the 25kv System	100	Yes	Y	B	R		
S	5/20/93	Manual Reactor Trip Due to Inadvertent MSIV Closure	100	Yes	Y	C	R	V	
S	7/27/93	Reactor Trip Due to Electrical Fault in Solid State Protection System Cabinet	100	Yes		N	R		
S	9/22/93	Automatic Reactor Trip Due to Main Generator Exciter Brush Failure	100			B	R		
S	1/25/94	Reactor Trip and Safety Injection Due to Inadvertent MSIV Closure	100	Yes	Y	C	R	V	
S	6/18/95	Manual Reactor Trip Due to Loss of Turbine Electro-hydraulic Control Pumps	100		Y	B	R	M	
S	1/27/96	Automatic Reactor Trip	100		Y	B	R		
S	5/10/97	Automatic Reactor Trip and Feedwater Isolation	8			N	D	H	Y
S	6/13/97	Reactor Trip and Engineering Safety Feature Actuation on LO-LO Steam Generator Level	0			B	N/A (Outage)	M(H)	Y
P	1/10/89	Completion of a Shutdown Due to a Potential Problem with the Air Supply for Two Primary Containment Air Operated Valves	1		Y	C	A		
P	3/4/89	Automatic Closing of Main Steam Isolation Valves and Subsequent Reactor Scram Due to Automatic Cycling of Turbine Bypass Valves	10	Yes	Y	B	D		
P	4/12/89	Inadvertent Over Pressurization of the RCIC System Suction Piping during Surveillance Testing and Subsequent Completion of a Shutdown	25		Y	S	N/A	V	Y
P	5/3/89	Automatic Turbine Trip, Generator Trip and Reactor Scram Due to High Reactor Vessel Water Level	24	Yes	Y	N	N/A	M	Y
P	7/18/89	Decreasing Main Condenser Vacuum Resulting in the Initiation of a Manual Scram Due to Procedure Inadequacy	35		Y	B	N/A		Y(Mainte. Err.)
P	8/30/89	Automatic Scram Resulting from a Turbine Runback Due to Failure of Potential Transformer and Voltage Balance Relay Wiring Error	65	Yes		B	N/A		

Table 3.2.1 Summaries of LERs related to the Unplanned Shutdowns issued by the Seabrook and the Pilgrim Nuclear Power Staiotns during 1989 and 1997 (page 3 / 4)

Facility (a)	Event Date	LER Title	Reactor Power Level	Reported in the Boston Globe	Similar Events Y : Similar events occurred in the past	System (b)	Plant Status (C)	Human Involvment (d)	Root Cause Y: Human Error
P	12/8/89	Automatic Scram at 95 Percent Power during Surveillance Testing Due to False Low Water Level Signal	95	Yes	Y	N	R	V	
P	5/13/90	Automatic Scram Resulting From Load Rejection at Full Power	100		Y	E	R		
P	7/3/90	Completion of a Shutdown Due to One Inoperable Recirculation System Loop	100		Y	N	R		
P	9/2/90	Manual Reactor Scram due to Lockup of the Feedwater Regulatory Valves	60	Yes	Y	N	N/A		
P	4/29/91	Completion of a Shutdown Due to Drywell Floor Sump Leakage Late and Subsequent Scram Signal While Shutdown	48	Yes	Y	N	N/A		
P	12/13/92	Automatic Scram Resulting From Load Rejection at 48 Percent Reactor Power	48		Y	B	D	V (COND BW)	
P	12/20/92	Reactor Scram and Closing of Main Steam Isolation Valves Due to Trip Setting of Main Steam Radiation Monitor	75		Y	C	A		Y(Mainte. Err.)
P	3/13/93	Automatic Scram Resulting From Loading Rejection at 100% Reactor Power	100		Y	B	R		
P	5/31/93	Automatic Scram Resulting From Operation of Auxiliary Transformer Differential Relay During Power Accession	24		Y	B	A		
P	7/22/93	Completion of Shutdown Due to Reactor Coolant Pressure Boundary Leakage	5		Y	N	A		
P	9/10/93	Loss of Preferred Offsite Power and Automatic Scram Resulting from Load Rejection at 100% power	100		Y	E(lightening)	R		
P	11/8/93	Low Reactor Vessel Water Level While Shutdown Resulting in Automatic Scram Signal and Containment System Isolation	0	Yes	Y	N	N/A (Outage)	H	Y
P	8/29/94	Automatic Scram Resulting From Load Rejection at 100 Percent power	100		Y	B	R		
P	3/24/95	Manual Scram Due to Main Generator Stator Cooling Coil Water Temperature Controlling Valve Controller Failure	60	Yes	Y	B	D		
P	4/19/96	Automatic Scram Due to Turbine Vibration during Planned Power Reduction	22		Y	B	D		

Table 3.2.1 Summaries of LERs related to the Unplanned Shutdowns issued by the Seabrook and the Pilgrim Nuclear Power Staiotns during 1989 and 1997 (page 4 / 4)

Facility (a)	Event Date	LER Title	Reactor Power Level	Reported in the Boston Globe	Similar Events Y : Similar events occurred in the past	System (b)	Plant Status (C)	Human Involvment (d)	Root Cause Y: Human Error
P	9/18/96	Reactor Building Closed Cooling Water System Heat Exchanger Leak – Technical Specification Required Shutdown	100		Y	N	R		
P	2/15/97	Manual Scram Due to Increasing Reactor Water Level During Power Reduction for Refueling Outage	20		Y	N	D		
P	11/23/97	Completion of a Required Shutdown due to Two Inoperable MSIVs	50			C	D	S	
P	12/6/97	Automatic Scram Due to High Reactor Water Level During Power Accession	75		Y	N	A		

NOTATION

(a) S : the Seabrook Nuclear Power Staion
P : the Pilgrim Nuclear Power Staion

(b) S: Stand-by Safety related(ECCS)
B: Power Distribution(BOP)
N: NSSS
F: Fuel handling
C: C
E: External Event

(C) R: Rated Power
A: Power Ascending
D: Power Descending

(d) Human Involvement
V: Surveillance
M: Maintenance
H: Operators Action

Table 3.2.2 Summaries of Failures of Systems resulted in Unplanned Shutdowns at the Pilgrim and the Seabrook Nuclear Power Stations during 1989 and 1997

Failed System	Reactor Status						Human Action Involvement					Human Error Root Cause	
	Rated Power	Power Ascending	Power Descending	N/A	Total	% total LERs	Surveillance	Maintenance	Operators Action	Total	% total LERs	Total	% total LERs
Stand-by Safety related(ECCS)				1	1	2%	1			1	2%	1	2%
Power Distribution (BOP)	15	3	5	3	26	53%	3	6	1	10	20%	6	12%
NSSS	6	2	2	5	15	31%	1	1	2	4	8%	3	6%
External Event	2				2	4%				0	0%		0%
Containment	1	3	1		5	10%	2		1	3	6%	1	2%
Fuel handling					0	0%				0	0%		0%
Total	24	8	8	9	49	100%	7	7	4	18	37%	11	22%
% total LERs	49%	16%	16%	18%	100%	/	14%	14%	8%	37%	/	22%	/

Table 3.3 Summary of the Reasons of Reactor Unplanned Shutdowns of Nuclear Power Plants

Categorizations of reasons	Examples
Considered to be trivial	- No safety significance - Short time power failure in the power grid - No impact on availability of sufficient power to the grid
Considered to be major	- Safety concerns/implications (Cooling system failure, radioactive leak) - Potentially serious consequences, Abnormal operation - Certain level of incident - Unknown cause

However, those categorizations are not written as criteria. Experience of editors and reporters concerning past history of events at power stations is important. Editors often assess events by utilizing background information gained from experts, such as the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), technical institute, political pressure group.

2) News sensitive environment (period) for nuclear power

In 1989 and 1990, many people might have been concerned with nuclear power generation in Massachusetts because the Seabrook nuclear power station started trial operation in 1989, and the Pilgrim nuclear power station restarted operation also in 1989. Given this situation, even if an event was trivial, many events could become subject to newspaper coverage due to a sensitive environment surrounding nuclear power. This factor is a completely non-technical issue in nuclear operation, but greatly affects the occurrence of related newspaper stories. The environment can be sensitized by the following factors:

-Recurrence of incidents, frequent plant shutdowns
-Public concern : Public activism, protest movements
-Serious problems in other nuclear power plants

Also, an attitude of trying to hide the incident may make media suspicious and aggressive in investigating events in a nuclear power station.

3) Timing of occurrence

If the event occurred late at night, it cannot appear in the morning paper because it occurred too late. Timing of the occurrence may be also one of the factors for newspaper coverage. In this case, the event may appear in the newspaper later if it is considered to be an important issue.

4) Competition for space in paper

In practice, there is only limited space in a newspaper. Therefore, the events may not be reported in the newspaper if other big news stories exist at the same time. A minimum of two editors are in charge of each page, and five editors are in charge of the front page. They prioritize the news stories and decide which articles should appear in the paper. In fact, a single isolated shutdown is less likely get onto the front page. It will be more likely to appear on the Metro/Region or Business page.

5) Accidental reason

Some accidental reasons make it impossible for editors to report the events that occurred in power stations. For example, the crucial editor and reporter assigned to regular coverage being away, or the fax machine being down, so that they do not learn of a plant event.

Pubic relations officers in nuclear power stations usually report events to the Associated Press(AP) and local papers. The AP distributes articles to all relevant mass media. If editors in the Boston Globe think it an important event, reporters in the Boston Globe may directly investigate the event later.

3.4 Verification of Key Factors

In order to verify that the hypothesis of five key factors is correct, all forty nine events reported have been investigated according to criteria shown in Table 3.4.1.

Table 3.4.1 Verifications of Five Key Factors

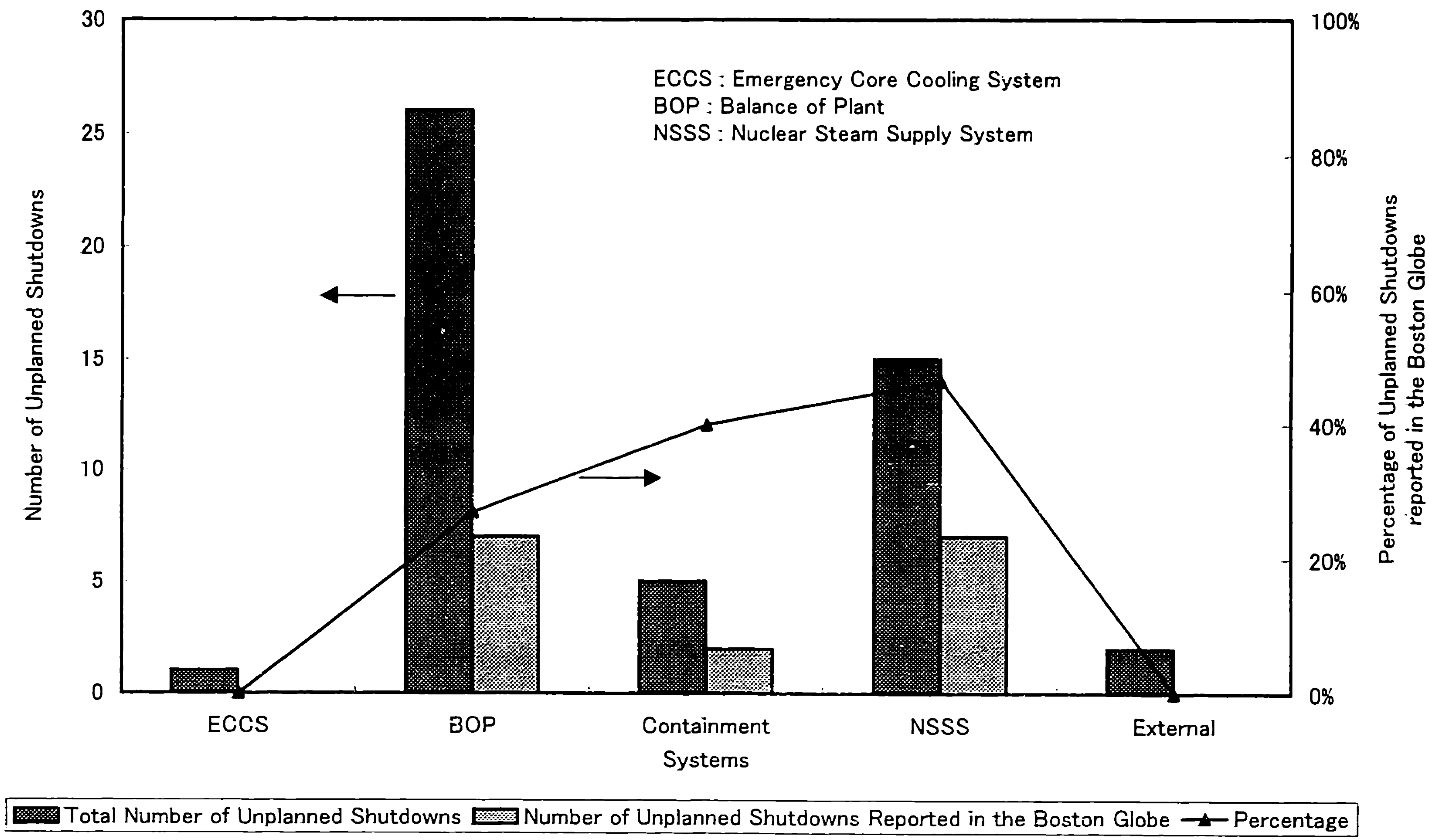
No	Five Key Factors	Major Items to be investigated
1	Reason of Shutdown	- Failed Systems - ECCS/ESF Failures - ECCS/ESF Actuation - Coolant Leakage - Plant Status
2	Sensitive Environment	- Nuclear Incidents reported to IAEA - Nuclear Testing - Frequent Occurrences
3	Timing of Occurrence	- Day of the week and Time
4	Space Limitations	- World Wide Events
5	Accidental Reason	N/A

(1) Reasons of Shutdown

No event posed a threat to the public health and safety during the interval between 1989 and 1997. Figure 3.4.1 shows which systems were most likely to have failed and be reported in articles. The BOP and NSSS have failed mostly, however, they were not always reported in articles.

Emergency Core Cooling System (ECCS) and Engineering Safety Features (ESF) are considered to be safety related systems. One

Fig. 3.4.1 Number of System Failures Arranged According to System category which Resulted in Unplanned Shutdowns at the Pilgrim and the Seabrook Nuclear Power Stations between 1989 and 1997



failure of Reactor Core Isolation Cooling System (RCIC), which is one of the safety related systems, was recorded but not reported in the articles. One ECCS and One ESF system had actuated due to the specific plant unplanned shutdowns. ECCS actuation included RCIC and High Pressure Core Injection (HPCI) system, and ESF actuation included the Emergency Diesel Generator (D/G). The actuation of D/G was reported, and ECCS was not. Three leakages inside containment vessels also occurred, and two of them were reported.

The number of reported events does not show strong dependency on plant status, such as power level. In some events, longer outage periods are required to correct the causes of plant shutdown. The duration of a plant shutdown may indicate the magnitude of the failure. However, events with long shutdown duration, such as more than one month, are not always reported. There is no correlation between the duration of shutdown and the number of reported events. No correlation has been observed among reactor power level, shutdown duration and newspaper coverage.

(2) Sensitive Environment

The public concern is assumed to be sensitized by the following items:

- Frequent occurrences of unexpected plant shutdowns
- Incidents reported to the International Atomic Energy Agency (IAEA)
- Nuclear testing in the world

If a plant frequently shuts down, most people, including the media, become sensitized to events occurring at power stations even if the root cause is considered to be trivial. However, an unplanned shutdown which occurred within 14 days of a previous unexpected shutdown did not always result in newspaper coverage. The number of reported unplanned shutdowns and the days from the previous unexpected shutdown do not have strong correlation.

The number of reported unplanned shutdowns occurring in the Seabrook and the Pilgrim Nuclear Power Stations do not show strong correlation with the number of days from reporting to the IAEA of the most recent incident.

Nuclear tests may often sensitize public concern around the world. However, France conducted nuclear tests in 1995, followed by 10 nuclear tests conducted by China and the U.S. between 1995 and 1997. Although five unexpected shutdowns occurred one month before and after these nuclear tests, none of five were reported in the articles.

(3) Timing of Occurrence

The day of the week and the time events occurred are identified as two of the important contributors for the newspaper coverage. Fig. 3.4.2 and Fig. 3.4.3 shows the numbers of events and the timing of their occurrence. Fig. 3.4.2 shows that fewer events were reported during the weekend than during weekdays. For example, if an event occurs on Tuesday, it is more likely to be reported in the article than an event which occurs on Sunday.

(4) Space Limitations

Most newspaper readers are interested in world-wide events and incidents. Therefore space competition in the newspaper is an important factor. Although defining major news may be subjective, the following fourteen world news events were selected as major events which may have caused space competition in the Boston Globe during 1989 and 1997:

Fig. 3.4.2 Number of Unplanned Shutdowns and Newspaper Stories According to the Day of the Week of Unplanned Shutdown at the Pilgrim and the Seabrook Nuclear Power Stations between 1989 and 1997

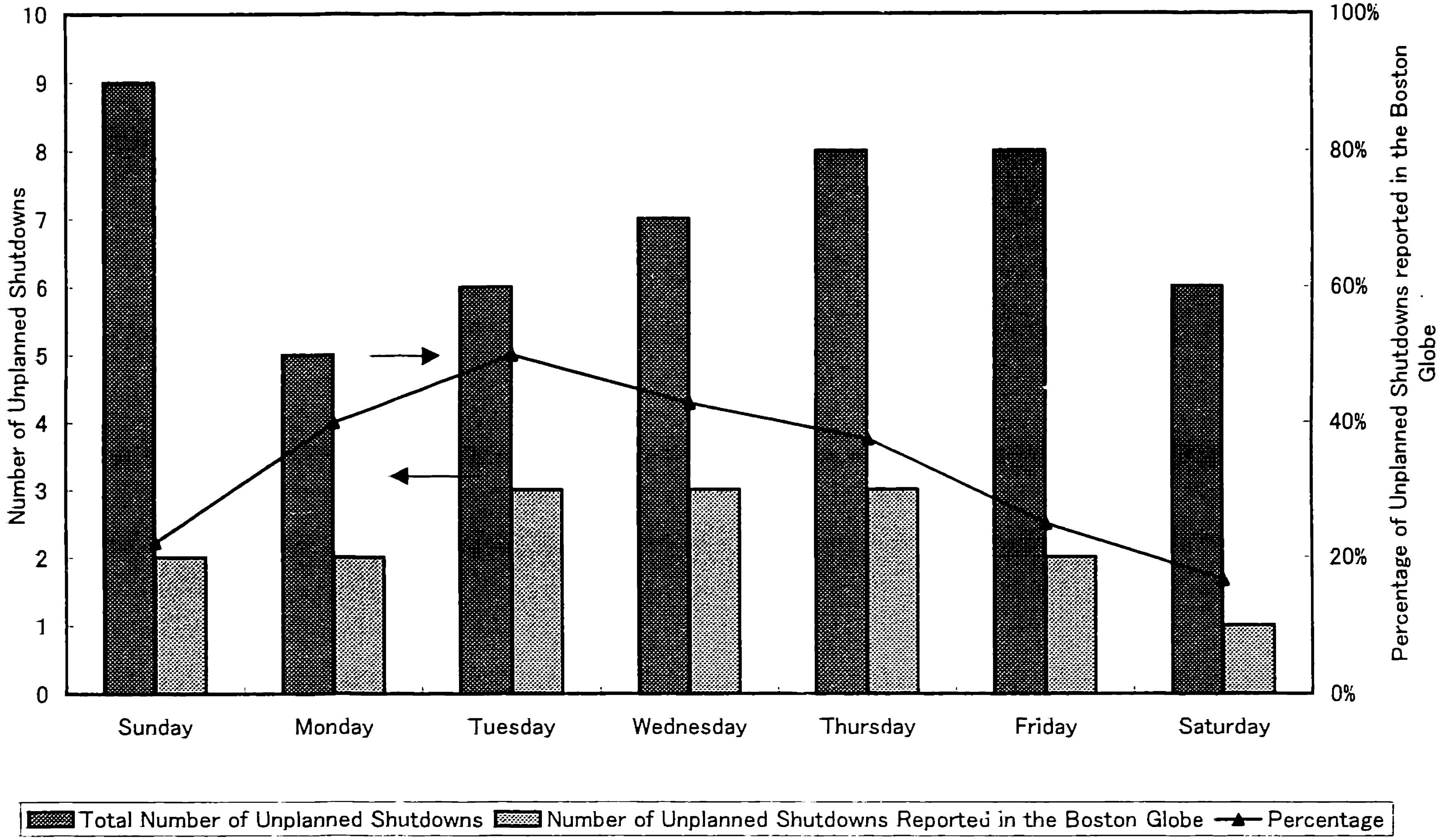
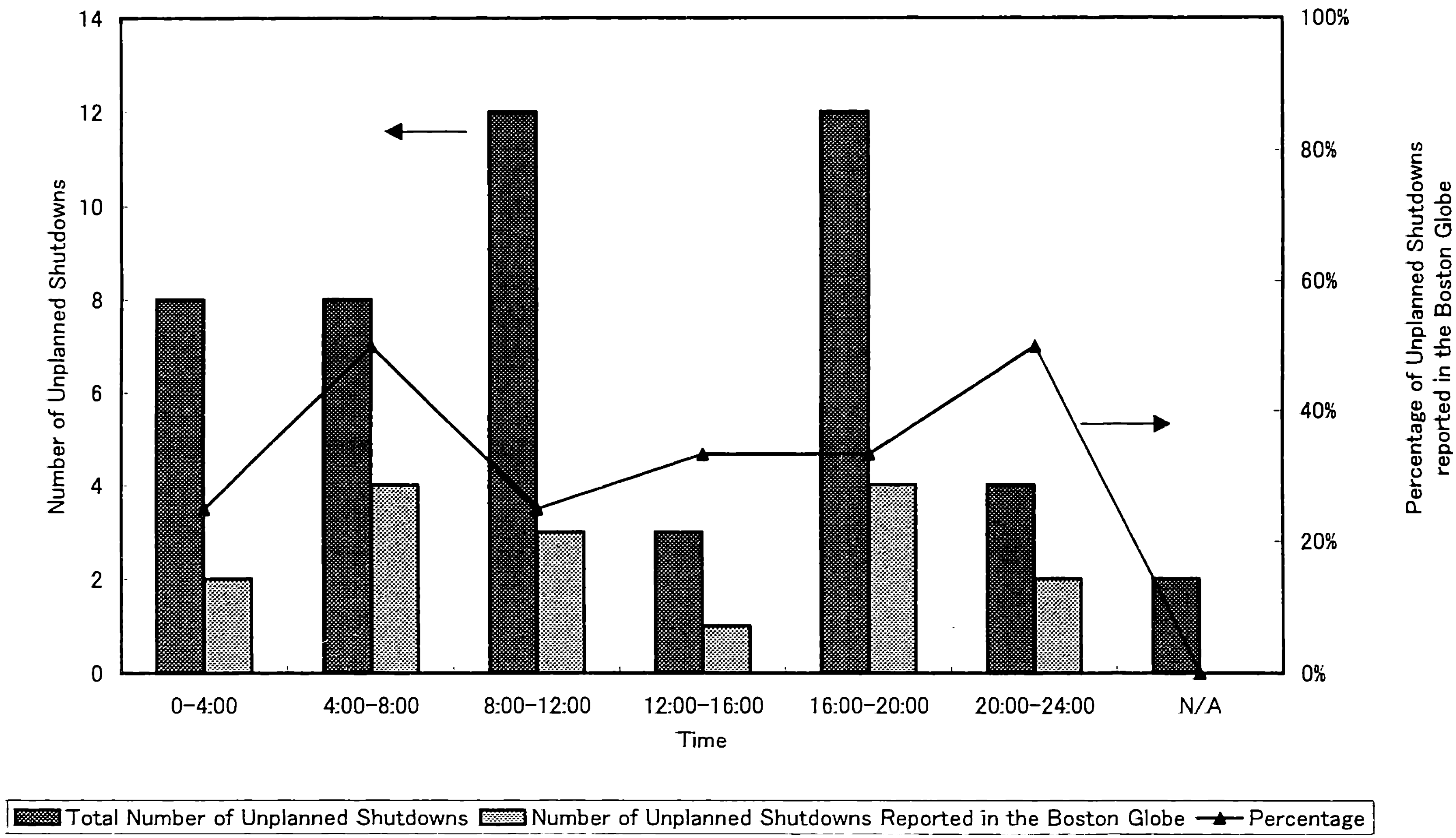


Fig. 3.4.3 Number of Unplanned Shutdowns Arranged According to the Time of Occurrence at the Pilgrim and the Seabrook Nuclear Power Stations between 1989 and 1997



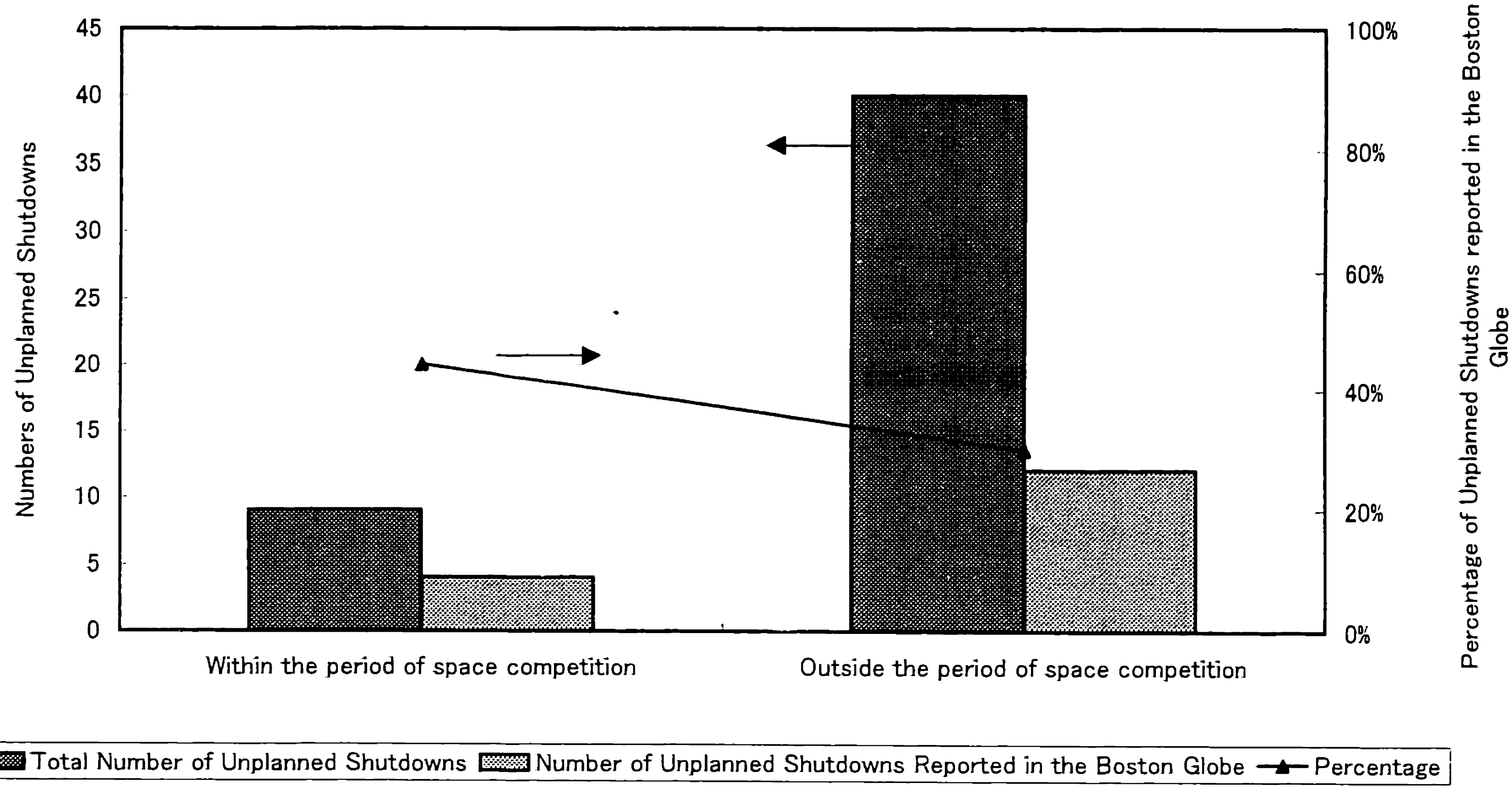
- Tearing down of Berlin wall (8/30/89)
- Iraqi invasion of Kuwait (7/5/90)
- German Reunification (9/2/90)
- Persian Gulf War (1/17/91)
- Dissolution of Soviet Union (7/25/91)
- Olympics : Albertville (2/8/92)
- Olympics : Barcelona (7/26/92)
- Election of Clinton Administration (11/3/92)
- Olympics : Lillehammer (2/12/95)
- Federal building bombing at Oklahoma city (4/15/95)
- Olympics : Atlanta (7/19/96)
- Reelection of Clinton Administration (11/97)
- Return of Hong-Kong (7/1/97)
- Princess Diana fatal accident (9/1/97)

The investigation has been conducted concerning events which occurred in power stations in the period during which people may be concerned with the news events (i.e., one month before, one month after, or during the news events). Fig. 3.4.4 shows that nine events occurred in nuclear power stations during this period, and four out of nine events were reported in articles. This indicates there is no correlation between space competition and events at the nuclear power stations. However, this may be misleading, since shutdowns are generally not reported on the front page, where big news events are placed, but on the metro/region or business pages.

(5) Other Factors

Accidental factors, such as an editor being away or a fax machine being down, cannot be investigated in our work. Other factors, however, such as human error or the fact that a similar event may recently have occurred at a power station, are investigated. None of these shows a correlation with the

Fig. 3.4.4 Effect of Space Competition in the Newspaper upon the Frequency of Reported Incidents at the Pilgrim and the Seabrook Nuclear Power Stations between 1989 and 1997



appearance of articles in the paper.

3.5 Conclusion of Key Factors

According to the investigation, we can conclude that of the five possible factors shown in table 3.5.1., only one shows a high correlation.

Table 3.5.1 Five Hypothesis and the Result

Key Event Factors	Result
Reason of Shutdown	No Correlation Observed
Sensitive Environment	No Correlation Observed
Timing of Occurrence	Incidents which occurred during the weekend are less likely to be reported in the newspaper.
Space Competition	No Correlation Observed
Accidental Reason	No Correlation Observed

Table 3.5.2 shows the summary of the investigation on the five key factors.

Table 3.5.2 Summary of Unplanned Shutdowns Reported in Newspaper at the Seabrook and the Pilgrim Nuclear Power Station between 1989 and 1997

No.	Five Key Event Factors	Criteria	Categorizations	Number of Shutdowns	Number of Reported Shutdowns	Percentage (*)	Comments
1	Reason of shutdown	System Involved	ECCS	1	0	0%	
			BOP	26	7	27%	
			Containment	5	2	40%	
			NSSS	15	7	47%	
			External	2	0	0%	
		Safety Related	ECCS/ESF Failures	1	0	0%	ONE RCIC failure recorded, which resulted in unplanned shutdown
			ECCS/ESF Actuation	2	1	50%	TWO unplanned shutdowns recorded, in which safety related systems actuated 1) RCIC and HPCI actuated 2) Diesel Generator (D/G) actuated Actuation of D/G appeared in the Boston Globe
		Leakage	Coolant Leakage	3	2	67%	]
		Power Level	0-20%	9	2	22%	
			20-40%	6	2	33%	
			40-60%	3	1	33%	
			60-80%	4	1	25%	
			80-100%	27	10	37%	
		Power Changing	Ascending	7	1	14%	
			Descending	7	1	14%	
			Rated	27	10	37%	
			N/A	8	4	50%	
		Shutdown Duration	Within 1 week	22	5	23%	
			1W-2Weeks	8	3	38%	
			2W-1Month	7	3	43%	
			More Than 1 Month	3	1	33%	
			Outage	9	4	44%	
		Human Maneuver	With Human Maneuver	18	7	39%	
			No Human Maneuver	31	9	29%	
2	Sensitive Environment	After Incidents Reported to the IAEA	0-7days	2	1	50%	1992 and later
			8-14days	5	0	0%	
			15-30days	5	1	20%	
			31days and more	15	4	27%	
		Interval from Previous Unplanned Shutdown	0-14days	4	1	25%	
			15-30days	6	2	33%	
			30-60days	11	3	27%	
			60-90days	8	3	38%	
			90-180days	7	4	57%	

Table 3.5.2 Summary of Unplanned Shutdowns Reported in Newspaper at the Seabrook and the Pilgrim Nuclear Power Station between 1989 and 1997

No.	Five Key Event Factors	Criteria	Categorizations	Number of Shutdowns	Number of Reported Shutdowns	Percentage (*)	Comments
2		Interval from Previous Unplanned Shutdown	180-365days	6	3	**50%**	
			366days and more	6	0	0%	
			N/A	1	0	0%	
		Similar Events	Similar Events Recorded	17	6	**35%**	
			No Similar Events Recorded	32	10	31%	
3	Timing of occurrence	Date	Sunday	9	2	22%	
			Monday	5	2	**40%**	
			Tuesday	6	3	**50%**	
			Wednesday	7	3	**43%**	
			Thursday	8	3	**38%**	
			Friday	8	2	25%	
			Saturday	6	1	17%	
		Time	0-4:00	8	2	25%	
			4:00-8:00	8	4	**50%**	
			8:00-12:00	12	3	25%	
			12:00-16:00	3	1	**33%**	
			16:00-20:00	12	4	**33%**	
			20:00-24:00	4	2	**50%**	
			N/A	2	0	0%	
		Nuclear Testing	ithin One month (before/after	4	0	0%	1995 and later : 15 testing
4	Space limitation	Big News	ithin One month (before/after	9	4	**44%**	14 big news
5	Accidental reason	No Investigation has been conducted					

* Bold and underlined numbers are above the average value.

- Total Number of Shutdowns 49
- Total Number of Shutdowns Reported in the Newspaper 16
- Ratio of the Number of Shutdowns Reported in Newspaper to the Total Number of Shutd 33%

4. Targeted Systems

In order to understand events and relevant public concern, we must improve our understanding of those events which have been reported in the past. The Licensee Events Reports (LERs) at the Seabrook and the Pilgrim Nuclear Power Stations were investigated in order to develop fault tree models of these plants because the LER may reveal failure modes that may not be evident from inspection of a system description.

4.1 Criteria of System Selection

(1) Relevant articles

The Boston Globe has issued 202 articles relevant to the Seabrook and Pilgrim power stations between 1989 and 1998. The following topics have emerged in the newspaper coverage:

- Unplanned shutdown
- Planned shutdown (Refueling outage)
- NRC statement
- Health and environmental issue
- Critiques statement
- General energy issue
- Politics

In light of the objective, events which have led to the articles about unplanned shutdowns are taken into consideration in our work.

(2) System involved

Failures of various systems have led to newspaper coverage. The following are the six categories which are considered to be major systems in nuclear power stations:

- Nuclear Steam Supply System (NSSS)
- Balance of Plant (BOP: power conversion)
- Stand-by safety related (ECCS)
- Containment
- Fuel handling
- External events

The root causes of the reactor shutdown due to failures in BOP, NSSS, Main steam Isolation Valves in Containment System, and lightening strike on power grid in External Events have accounted for more than 95 % of total root causes of unexpected shutdowns. Therefore, we focused on these events in order to develop fault tree models in chapter 5.

(3) Plant status

Unplanned shutdown have occurred when the plant was at various levels of power -- rated power, ascending power and descending power. As a result of the investigation, no strong correlation was observed between the power of the reactors and the appearance of newspaper stories.

4.2 System Descriptions

In this section, system description for the reference nuclear power station is summarized to develop risk models in the following chapters. The U.S. Nuclear Regulatory Commission has the information of system description about all U.S. reactors.

4.2.1 General System Descriptions at PNPS

(1) General Information

Plant Name:	Pilgrim Unit 1
Owner/Operator:	Boston Edison Company
Other Plants On Site:	None
Reactor Type:	BWR 3

Facility Licensed Thermal Power Limit:	1998 MWt
Main Turbine Generator Rated Output:	687 MWe
Ultimate Heat Sink:	Cape Cod Bay
NSSS Vendor:	General Electric
Turbine Generator:	General Electric
Architecture Engineering:	Bechtel
Constructor:	Bechtel
Date of Operating License Issuance:	09/15/72
Date of Commercial Operation:	12/01/72

(2) Plant summary

<u>Primary Containment</u>

Containment Type:	Mark I
Maximum Internal Design Pressure:	62 psig
Torus/Pressure Suppression Chamber	

<u>Fuel</u>

No. of Fuel Assemblies:	580
No. of Rods per Assembly:	62

<u>Reactor Core Isolation Cooling System (RCIC)</u>

Pump TS Flow Rate and related	
Discharge Head range:	400 gpm@150-1000 psig

<u>High Pressure Coolant Injection System (HPCI)</u>

Pump Tech Flow Rate and related	
Discharge Head range:	4250 gpm@150-1140 psig

<u>Low Pressure Coolant Injection (LPCI)</u>

No. of Pumps:	4
Pump TS Flow Rate and related	
Discharge Head:	4,800 gpm @ 165 psig
Pump Shutoff Head:	268 psig

Core Spray

No. of Pumps:	2
Pump TS Flow Rate and related Discharge Head:	3,300 gpm @ 104 psig

Feedwater System

Condensate

No. of Pumps:	3
Pump Shutoff Head:	448 psig

Reactor Feed

No. of Pumps:	3 (Motor Driven)

Class 1E Emergency Power

No. of Class-1E Diesel Generators (EDG) onsite: 2

Blackstart Diesel Generator is also located at the Pilgrim Site.

4.2.2 General System Descriptions at SNPS

(1) General Information

Plant Name:	Seabrook Unit 1
Owner/Operator:	North Atlantic Energy
Facility Licensed Thermal Power Limit:	3411 MWt
Main Turbine Generator Rated Output:	1198 MWe
Ultimate Heat Sink:	Atlantic Ocean Cooling Tower
PWR Type:	4-loop
NSSS Vendor:	Westinghouse
Turbine Generator:	General Electric
Date of Operating License Issuance:	03/15/90
Date of Commercial Operation:	08/19/90

(2) PLANT SUMMARY

Containment

Containment Type:	Large Dry
Design Pressure:	52 psig

Steam Generator

Steam Generator Model:	Westinghouse F series
Number of Steam Generators:	4
Number of Main Steam Isolation Valves:	4

Fuel

No. of Fuel Assemblies:	193

Chemical and Volume Control System

No. of Centrifugal Pumps:	2
No. of Positive Displacement Pumps:	1

Pump Design Flow Rate: 150 gpm @ 5800 ft (centrifugal pumps)
98 gpm @ 5800 ft (PDP pump)

High Head (Safety) Injection System

No. of Pumps:	2
Pump Design Flow Rate:	150 gpm @ 5800 ft
Pump Shutoff Head:	6200 ft

Intermediate Head (Safety) Injection System

No. of Pumps:	2
Pump Design Flow Rate:	425 gpm @ 2700 ft
Pump Shutoff Head:	3545 ft

Low Head (Safety) Injection System (LHI)

No. of Pumps:	2
Pump Shutoff Head:	460 ft

Residual Heat Removal System

No. of Pumps:	2
Pump Shutoff Head:	460 ft

Main Feedwater

Condensate No. of Pumps:	3
Feedwater No. of Pumps:	2 (steam-driven)

Startup Feedwater

No. of Pumps:	1

Auxiliary Feedwater System

Number of Motor Driven Pumps:	1
Capacity:	710 gpm @ 1320 psig
Number of Turbine Driven Pumps:	1
Capacity:	710 gpm @ 1320 psig

Class 1E Emergency Power

No. Class-1E Diesel Generators (EDG):	2
No. Dedicated per Unit:	2

4.3 System Descriptions of BOP and NSSS at PNPS

Following are brief descriptions of the major subsystems in Balance of Plant (BOP) and Nuclear Steam Supply System (NSSS) at the Pilgrim Nuclear Power Station (PNPS).

(1)RBCCW: Reactor Building Closed Cooling Water

The RBCCW Systems provides cooling water to numerous heat exchangers associated with other equipment, including Residual Heat Removal System (RHR) heat exchangers, fuel pool exchangers, cleanup nonregenerative heat exchanger, recirculation pump cooling system, Control Rod Drive (CRD) pump oil, and recirculation motor lube oil coolers. Two independent closed loops make up the RBCCW system. Each loop contains a surge tank, chemical addition tank, three pumps in parallel, and one cooling water heat exchanger. Both loops can be interconnected through two 12-inch cross-tie headers by using four valves. A 500-gallon head tank is located at the highest point of each loop to accommodate system volume changes, maintain static pressure, and detect gross leaks in the system. Five out of six pumps must fail for system failure to occur (system operation is possible with two pumps) . Both RBCCW heat exchangers are required, but system operation is possible with one surge tank and one chemical addition tank. Major components are as follows:

RBCCW Pumps/Motors A-F
- Heat exchangers
- Surge tanks
- Chemical addition tanks
- Butterfly valves
- Valve TV3835/3836

(2)TBCCW: Turbine Building Closed Cooling Water

The TBCCW System provides cooling water to various equipment in the turbine building and to the station air conditioning systems. The TBCCW System provides cooling water to numerous heat exchangers associated with other equipment, including air compressor after coolers and water jackets, control and computer room air conditioning, control area air conditioning, reactor feed pump seal water, lube oil coolers, and generator and condensate pump cooling equipment. The TBCCW System consists of a single closed loop with two centrifugal pumps in parallel taking suction from two TBCCW heat exchangers arranged in parallel and delivering cooling water to major equipment. A 500-gallon head tank is located at the highest position of the loop to accommodate system volume changes. This tank also maintains static pressure on the pump, detect gross leaks in the TBCCW System, and provide a means for adding make-up water. The TBCCW System normally operates with one or both pumps running depending on cooling demands. Both heat exchangers are normally in service, and system temperature is maintained between 70-90 F by the heat exchanger bypass valve. Major components are as follows:

TBCCW Pumps/Motors A/B
Heat exchangers
Surge tank
Chemical addition tank
Valves

(3) Neutron Monitoring

The Neutron Monitoring System consists of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides an indication of neutron flux, which is correlated with thermal power level, for the entire range of flux in the core. The source range monitors (SRM) and the intermediate range monitors (IRM) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRM) and average power range monitors (APRM) allow assessment of local and overall flux conditions during power range operation. Rod block monitors (RBM) are provided to prevent rod withdrawal when reactor power should not be increased at the existing reactor coolant flow rate. The Traversing Incore Probe System (TIPS) provides a means of calibrating the individual neutron monitoring sensors. The purpose of the Neutron Monitoring Systems is to monitor reactor power continuously for unusual or emergency conditions. Several conditions may exist in a reactor core, such as sudden pressure increase, rod drop accident. As a direct result of this type of power deviation, fuel damage and release of radioactive material to the environment could occur.

(4) RPS: Reactor Protection System

The Reactor Protection System (RPS) initiates a rapid automatic shutdown (SCRAM) of the reactor. This action is taken in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operation transients. Under normal operation, RPS is a passive, energized system. It is composed of 2 trip systems, A and B. Each system contains 2 redundant automatic trip channels and a manual trip channel resulting in a *1-out-of-3* scram logic. Both A an B must trip to cause a reactor trip, therefore, RPS is said to have *1-out-of-3 twice* logic. All 145 control rods are rapidly inserted into a reactor by the RPS signals. Major components are as follows:

RPS Trip logic channels/relays
Scram pilot valves
RPS M-G set

(5) Reactor Pressure Vessel (RPV) & Internals

The Reactor Pressure Vessel (RPV) is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The normal operating pressure ranges from 1020 to 1050 psia in the steam space above the separators. The vessel is fabricated of carbon steel and has internal stainless steel cladding. RPV contains the core and supporting structures; the steam separator and dryers; the jet pumps, the control rod guide tubes; core spray, core shroud, and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, feedwater lines, control rod drive housing. In our work, miscellaneous leakage from RPV and its associated lines is only considered as a failure of RPV.

(6) Reactor Recirculation System

The Reactor Recirculation System pumps reactor coolant into the core to remove the energy generated in the fuel. This is accomplished by two recirculation loops external to the reactor vessel. The two recirculation loops have a cross-connect line with two normally closed valves. The recirculation pumps are single-stage variable speed centrifugal pumps vertically mounted in the recirculation line. Mechanical pump seals control the recirculation pump shaft seal leakage. The recirculation pump breakdown bushing limits leakage flow, with both mechanical seals failed, to about 20gpm. The recirculation pump speed can be varied to control reactor power level through the effects of coolant flow rate on moderator void content. Motor Generator set (MG set) provides variable frequency voltage regulation power to the recirculation pump motor. The MG set has three major components; drive motor, fluid coupler and generator, and two supporting systems : oil system and ventilation system. Major components are as follows:

Reactor Recirculation System
- Recirculation pump/motor
- MG set motor/generator
- Fluid coupling
- Oil system
- Pump inlet/outlet valves MO-5A/B

(7) Main Turbine

The main turbine is a tandem compound, dual flow, non-reheat impulse unit designed to run at 1800 RPM. Dry saturated steam at approx. 970 psi is admitted to the High Pressure (HP) Turbine via 4 sets of main stop and control valves. The steam enters the center of the turbine and expands outward to each end. Two exhausted lines on each end of the HP Turbine direct the steam to the moisture separators. Steam passes through a set of Combined Intermediate Valves (CIV) and enters the Low Pressure (LP) Turbine. The steam expands toward each end of the LP Turbine and exhausts to each Condenser. Bypass valves allow for passing 25% rated steam flow directly to the condenser. Major components are as follows:

Main Turbine

- Main Stop Valves (TSV)
- Main Control Valves (CV)
- High Pressure Turbine (HP)
- Low Pressure Turbine (LP)
- Moisture separator (M/S)
- Cross around relief valves
- Shaft coupling
- Journal/Thrust bearings
- Bypass valves
- Mechanical Hydraulic Control system (MHC)
- Lubrication system

(8) Main Generator

The main generator converts turbine rotational energy to electrical energy (24kV) which is fed into the 345 kV ring bus via the main output transformer. Main generator excitation and relaying systems provide main generator output regulation and control as well as automatic generator protection on an electrical fault. Frequency is maintained constant (60Hz) by maintaining constant speed (1900 rpm). The main generator is synchronized to the 345 kV system via the main transformer. The automatic voltage regulator maintains an acceptable voltage. The unit auxiliary transformer provides normal AC power to the station whenever the main generator is on-line. Station auxiliary loads are automatically transferred to the start-up transformer on a generator trip or any time the unit auxiliary transformer is lost. The generator gas control system provides hydrogen gas to cool the main generator. It also purges the generator of H_2, CO_2 and air. If partial H_2 pressure is lost, generator output must be reduced. If all H_2 gas is lost, generator should be shut down. The Hydrogen Seal Oil system supplies oil to generator seals at

8 psig above hydrogen gas pressure to prevent hydrogen leakage from the generator. Losing the main oil seal pump causes hydrogen purity in the generator to decrease. Losing all seal oil pumps causes seal oil pressure drop, allowing hydrogen to blow out past the seals. The Stator cooling water system provides low conductivity water to cool stator windings and field rectifier assemblies. Losing stator cooling water causes the generator to run back load to protect the generator and field rectifiers from overheating and extensive damage. Stator cooling water trips on a generator lockout.

(9) Main condenser

The main condenser is of the single-pass, divided, water-deaerating type. It consists of two shells, one for each low pressure turbine cylinder. Each half-capacity condenser has two feedwater heaters located in its neck. The hotwells of each condenser are designed to provide a minimum condensate retention time of two minutes, permitting decay of short-lived radioactive isotopes. Deaeration is provided in the condensers for removal of dissolved gases from the condenser. The main condenser vacuum system consists of the air removal system and off gas system. One twin-element (100 percent spare capacity) two-stage Steam Jet Air Ejector (SJAE) equipped with inter and after condenser is provided for evacuating gases from the turbine and main condenser. One mechanical vacuum pump removes gases from the main condenser during startup and shutdown, when steam is not available for air ejectors. Condenser low vacuum (>23 in. Hg) initiates the reactor trip

(10) Condensate Water system

The condensate water system consists of condensate pumps and a condensate demineralizer system. Three condensate pumps take suction from the hotwell and discharge through a header to the demineralizers. Each condensate pump has a capacity of 37 percent of full power plant output. The condensate demineralizer system consists of seven parallel circuit purifying vessels (one as standby) to maintain required reactor feedwater purity. Ionic and particulate materials from the upstream are removed from the condensate.

(11) CW: Circulating Water Pump System

Two circulating water pumps located in the intake structure provide a continuous supply of condenser cooling water. The water is pumped from and returned to Cape Cod Bay. Trash racks and travelling water screens protect the circulating pumps from debris. Each pump can support 100 percent plant output.

(12) SSW: Salt Service Water System

The Salt Service Water System supplies coolant to the secondary sides of the heat exchanger of RBCCW and TBCCW to remove heat produced during normal operation, shutdown and accident conditions. Cooling water is taken at the intake structure by the five service water pumps and discharged with the condenser circulating water. Three pumps are normally operating and two pumps are in the stand-by position.

(13) Feedwater System

Three-motor driven reactor feedwater pumps are used to pressurize the feedwater to the reactor pressure level. Each of the feedwater pumps has a capacity of 40 percent of full power plant output. Three feedwater pumps are required to operate for the full power operation. The flow rate is controlled by the two regulating valves which are installed at the header of two pump discharge lines. Two high pressure heater and 3 low pressure heaters are installed in train A and train B in order to accomplish the required heating of the feedwater at 360 F. Each train is capable of passing enough water to maintain the unit at 50 percent power. There are bypass lines around the low and high pressure heaters that are capable of passing water equal to 50 percent.

(14) Steam supply to high pressure turbine

The main steam directs high pressure steam from RPV to the main turbine through four 20-inch main steam lines. Two safety valves and four relief valves are mounted on the main steam lines to protect RPV from overpressure during abnormal conditions. The Main Steam Isolation Valves (MSIVs) are installed in each main steam line, one valve inside the Primary Containment Vessel (PCV) and another valve outside the PCV. Steam lines pass through the main steam line tunnel to the turbine building, where steam can be either directed to the main turbine or bypassed directly to the main condenser. A turbine stop valve (TSV) and control valve (CV) are mounted in each main steam line. Steam flow through the turbine bypass line is regulated by three bypass valves.

4.4 System Descriptions of BOP and NSSS at SNPS

Following are brief descriptions of the major subsystems in BOP and NSSS at the Seabrook Nuclear Power Station (SNPS). Since many systems at the Seabrook Nuclear Power Station are similar to those at the Pilgrim Nuclear Power Station, only a few specific systems at the Seabrook Nuclear Power Station are described in this section.

(1) RCS : Reactor Coolant System

The RCS consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop has a steam generator and a reactor coolant pump (RCP). The system also interfaces, via interconnecting piping, with the pressurizer and pressure relief tank. All of the systems' mechanical components are located inside the reactor containment building. When the system is operating, the RCPs move the pressurized water (coolant) through the vessel and the loops. The water, which cools the core and moderates the nuclear reaction, picks up the heat of fission (thermal energy) as it passes through the core. The thermal energy in the water is carried to the steam generators, where it is transferred to the secondary side. The coolant is then returned to the core inlet by the RCPs to continue the energy removal process. The pressurizer establishes and maintains pressure in the RCS. The pressurizer is partially filled with water that is kept saturated by means of electric immersion heaters. The saturated conditions in the pressurizer mitigate pressure transients. Excessive system pressure is relieved through two power-operated relief valves and three spring-loaded safety valves connected at the top of the pressurizer. These valves discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

(2) RCP: Reactor Coolant Pumps

The RCPs are vertical, single-speed, centrifugal pumps powered by air-cooled, three-phase 7000hp motors located above the pumps. Each RCP provides 100,600 gallons of flow per minute at a head of 289 feet (127 psid). Pump suction is at the bottom of the casing, while discharge is radial. Most pump parts that contact the reactor coolant are stainless steel except for the bearings and seals. Leakage along the shafts is controlled by three shaft seals. High pressure seal water, supplied by the chemical and volume control (CS) system, is injected through the pump's thermal barrier flange.

(3) Pressurizer

The pressurizer is a vertical, cylindrical vessel made of carbon steel. All parts of the pressurizer in contact with reactor coolant have austenitic stainless steel cladding. The bottom of the pressurizer is connected by a surge line to the hot leg of reactor coolant loop 3. Electrical immersion heaters are installed through the pressurizer bottom head. The pressurizer vessel is partially filled with water (25% at zero power). Above the water space is a steam bubble formed by the operation of the immersion heaters. A spray nozzle is located at the top of the vessel. The spray nozzle has three potential sources of supply. The two normal sources are the cold legs of reactor coolant loops 1 and 3. The alternate source is from the discharge of the charging pumps in the CS system. Two power-operated relief valves (PORVs) and three safety valves, which are located at the top of the pressurizer, relieve to a pressurizer relief tank.

(4) CS: Chemical and Volume Control System

The functions of the CS are to:

- Maintain the programmed water level in the pressurizer
- Maintain the seal water injection flow to No.1 seals of RCPs
- Control the reactor water chemistry, radioactivity levels, and concentration of soluble chemical neutron absorber.
- Provide emergency core cooling
- Provide a means for filling, draining, and pressure testing of the RCS.

Also, the CS can be divided into four major subsystems

- Letdown system
- Purification and chemistry control system
- Seal water system
- Charging system

The letdown system is used to remove coolant from the RCS at a flow rate selected by the operator. The letdown system lowers the temperature of the coolant by means of heat exchangers and reduces the pressure of the letdown flow by means of flow control valves and a pressure control valve. The purification and chemistry control system is used to remove impurities from the letdown flow and to add chemicals as necessary to maintain the reactor coolant chemistry parameters within specifications. The charging system takes this purified, chemically treated water and charges the water back into the RCS. The seal water system supplies water to the RCP No.1 seals and collects and recycles the leakoff from the No. 1 seals.

(5) SG: Steam Generators

There are four SGs in the reactor coolant system, one in each loop. Each SG consists of a primary section (tube side) and secondary side (shell side). The major components are as follows:

- Hemispherical channel head with a divider plate
- Tube sheet
- U-tubes
- Shell
- Tube bundle wrapper
- Moisture separators

(6) NI: Excore Nuclear Instrumentation

The NI system monitors leakage neutron flux associated with reactor operation from 10^{-10}% to 200% of full power. This represents a neutron flux leakage range of over 12 decades from 10^{-2} to over 10^{10} neutrons/cm^2/sec. The Westinghouse NI system, which provides for automatic plant protection and control, consists of three ranges and a total of eight channels in order to cover the whole neutron flux range. Two redundant channels are provided for the source range, two redundant channels are provided for the intermediate range, and four redundant channels are available in the power range.

(7) Feedwater System

The feedwater system is required to maintain proper steam generator water level inventory during power operation. Major components of the feedwater system include the following:

- Steam generator feed pumps (two)
- High pressure feedwater heaters (two)
- A Startup feed pump
- Steam generator recirculation and wet layup pump
- Valves

(8) CW: Circulating Water System

The primary function of the CW is to provide cooling water to the main condensers for removal of heat rejected from the turbine steam cycle. The CW consists of the following major components:

- Offshore structures, tunnels, and transition structures
- Screen wash water pumps (three)
- Traveling screens (three)
- CW pumps (three)
- Valves

(9) Main Steam System

The major function of the main steam system is to transfer the steam generated in the four steam generators to the turbine system for conversion to electric power. The main steam system utilizes the following major components:

- Atmospheric steam dump valves (4)
- Steam generator safety valves (20)
- Main steam isolation valves (4)
- Main steam dump valves (12)

5. Development of Risk Model

5.1 Objective

In order to avoid unfavorable publicity and optimize allocation of resources, a risk modeling analysis is a suitable method for power plant analysis for the following reasons:

(1) Identification of potential events leading to media coverage

An event tree and/or a fault tree can involve all plant failure events which may lead to unfavorable publicity. A complete set of events can be described in the event/fault trees, thus avoiding the omission of events of significance.

(2) Quantification of risk

All events which appear in the event/fault trees can be evaluated in the form of a minimal cut set, each with its respective quantified risk importance, such as the Risk Achievement Worth (RWA). This risk significance ranking enables analysts to prioritize the components according to their significance in causing events.

(3) Treatment of key factors

The feasible level of engineering analysis is limited to events leading to an unplanned shutdown. Subsequent events can affect whether the shutdown becomes the subject of a newspaper story. Key factors which may affect events leading to unfavorable publicity can be quantified and evaluated in the risk model. Some key factors, such editor's judgement and constraint of newspaper space, are not physical events which occur in power stations and therefore cannot be included in the risk models.

As described in chapter 2, fault tree models are used in our work.

5.2 Fault Trees

5.2.1 General Considerations

(1) Top event

Fault tree models are completed by identifying the components within each subsystems that could cause subsystem failure by failing individually or in conjunction with other components. Some events eventually result in the reactor trip and some have led to unfavorable publicity. In our work, a top event is defined as a unexpected reactor shutdown. We calculated the probability of that event based upon occurrence caused by the system failure of BOP or NSSS within one year of plant operation from startup. For purpose of illustrating the major contributors to plant unreliability, this treatment is sufficient.

(2) Reactor shutdown and primary systems

The event of a reactor shutdown is defined as the unexpected reactor shutdown due to the failure of subsystems within one year of operation from start up. A reactor may trip due to several reasons. The investigation in Chapter 3 shows that more than 95% of reactor trips were caused by the failure of the BOP and the NSSS systems, including failures of Main Steam Isolation Valves (MSIVs). Therefore, in our work, a nuclear power plant system can be rationally modeled by two primary systems, BOP and NSSS. Both BOP and NSSS employ fault trees composed of subsystems whose failures eventually result in a reactor shutdown. Failures of MSIVs are categorized into these systems in fault tree models.

(3) Subsystems

Both BOP and NSSS consist of major subsystems with functions whose failures eventually result in a reactor shutdown. Each subsystem employs fault trees and consists of basic events. Basic events are defined as the lowest level of components/events for which data are available. Components can be defined as an aggregate of discrete elements, such as system piping and supports, as long as the reliable component data can be obtained to represent this system.

To a greater or lesser extent, any basic event (i.e., failure of components) in nuclear power stations may affect plant availability/shutdown. However, it is impossible to take all events into account in the fault trees. Therefore, we chose subsystems and components whose failures are more likely to affect plant shutdowns. In some cases, failures of components do not immediately result in plant shutdown, although, plant shutdown is required in order to repair them. In this case, failure of these components is also considered to have resulted in a shutdown, and is modeled as a basic event.

(4) Control and instrument system

Instrument and control subsystems consist of so many small elements that their failures are described as a single basic event in our work. For example, failures of the turbine control systems and neutron monitoring system failures in the Pilgrim Nuclear Power Station are treated as basic events (i.e., no more detailed fault tree analysis is necessary) for the following three reasons: 1) there were few failures recorded relevant to those systems in the Pilgrim Nuclear Power Station; 2) useful failure data for those subsystems can be obtained from generic databases; and 3) those systems consist of so many small elements that their overall failure should be represented as a single event in order to simplify the model.

(5) Support system

Failures of such support systems as instrumental air or electrical supply may affect the performance of other subsystems. For example, loss of condenser vacuum, which is one of the subsystems in BOP, is caused by the failure of an air-operated valve. This air-operated valve may fail to operate due to the loss of instrumental air. However, the loss of instrumental air simultaneously affects the failure of other subsystems. Accurately modeling the dependency of those subsystem failures requires a great deal of knowledge of plant behavior. Therefore, in our work, only Turbine Building Closed Cooling Water system (TBCCW) and Reactor Building Closed Cooling Water system (RBCCW) at the Pilgrim Station are modeled as subsystems in the fault tree models to represent support systems.

(6) Repair rate

Many systems have redundancy in order to back up the system failure (i.e., they are equipped with stand-by components). A stand-by component starts operating upon demand when a normally operating component fails to run. While a stand- by components is operating, a failed component may go under repair. Therefore, probability of system failure during mission time also depends on the repair rate of a failed component in the case that the mission time is somewhat longer. However, a repair rate is not available in the IPE of both nuclear power plants. In our work, we do not take into account failures resulting from stand-by components failing to operate after successful start upon demand .

(7) Power reduction

Failures of components or systems sometimes result in the reduction of reactor power. For example, condenser vacuum decreases if one of three circulation pumps fails to run during the rated power. This failure eventually causes a reactor power reduction, and obviously affects a capacity factor of electric power generation. In order to take power reductions into consideration in our fault tree models, a great deal of investigation is necessary because the number of components whose failures cause power reductions are far more than that of components whose failures directly cause the plant shutdown. Therefore, power reductions are considered to be within normal operation in our work. In the case of the circulation pump failure mentioned above, two out of three circulation pumps must fail to operate to initiate reactor shutdown. The criteria for plant operators to shut down a reactor if some components fail is not always explicitly stated in documents. To develop fault tree models in detail, we need to speak with systems engineers to determine the criteria for reactor shutdown.

(8) Criteria to develop fault trees

Each subsystem will be designed to employ fault trees if they satisfy following criteria:

- Components within the subsystem have caused plant shutdown between 1989 and 1997.
- IPE have taken them into consideration and made fault tree models.
- They are not I&C systems.

5.2.2 Subsystems of NSSS at PNPS

The Nuclear Steam Supply System (NSSS) at the Pilgrim Nuclear Power Station(PNPS) consists of the following seven subsystems:

(1) Reactor recirculation system
- Recirculation pumps/motors
- M-G sets
- Inlet/Discharge valves

(2) RPV and Internals
- Miscellaneous leakage
- Internal structures
- Fuel assemblies
- Instruments

(3) Feedwater system

- Loss of Feedwater

(4) Main steam line valves

- Main steam isolation valves (MSIV)
- Safety relief valves (SRV)
- Safety valves (SV)
- Turbine stop valves (STV)
- Turbine bypass valves (BPV)
- Turbine control valves (CV)

(5) RPS

- PRS devices failure
- Human errors

(6) Neutron monitoring

- Failure of APRMs, LPRMs and other instruments

(7) RBCCW

- Loss of RBCCW

5.2.3 Subsystems of BOP at PNPS

The Balance of Plant systems (BOP) at the Pilgrim Nuclear Power station consists of the following seven subsystems;

(1) Generator

- Transformers/switchyard
- Stator cooling water system (SWC)
- Generator hydrogen cooler system
- Hydrogen oil sealing system
- Exciter
- Main generator/stator/voltage regulator

(2) Turbine

- Combined Intercept Valves (CIV)
- Steam seal supply systems
- Bearings
- Moisture separators
- Mechanical hydraulic control system

- Turbine lubrication system
- High/Low pressure turbines
- Turbine control/supervisory system

(3) Condenser
- Loss of condenser vacuum

(4) Condensate water
- Loss of condensate water

(5) TBCCW
- Loss of TBCCW

5.2.4 Subsystems of NSSS at SNPS

The NSSS at the Seabrook Nuclear Power Station (SNPS) consists of the following four subsystems:

(1) Reactor Coolant System (RCS)
- RCS pumps/motors
- Pressurizer and Pressurizer Relief Tank
- Pressurizer Power Operated Relief Valves (PORVs)
- Pressurizer Safety Valves
- Pressurizer Spray Valves

(2) RPV and Internals
- Miscellaneous leakage
- Internal structures
- Fuels
- Instruments

(3) RPS
- PRS M-G sets
- RPS devices failures (breakers, relays)
- Spurious signals

(4) Nuclear Instrument
- Intermediate Range Monitors (IR)
- Source Range Monitors (SR)
- Power Range Monitors (PR)

5.2.5 Subsystems of BOP at SNPS

The BOP at the Seabrook Nuclear Power Station (SNPS) consists of the following eight subsystems:

(1) Steam Generators (SG)

- Steam generators

(2) Circulating Water System(CW)

- Circulation pumps
- Traveling screens
- Traveling screens wash pumps
- Butterfly valves
- Intake structure

(3) Turbine

- Main steam line valves
- Electro Hydraulic Control System (EHC)
- Moisture Separator/Reheaters (MS/R)
- Steam seal supply system
- Main turbine bearings
- Turbine lubrication system
- High/Low pressure turbines
- Turbine control/supervisory system

(4) Generator

- Transformer/switchyard
- Stator cooling water system (SWC)
- Generator hydrogen cooler system
- Hydrogen oil sealing system
- Exciter
- Main generator/stator/voltage regulator

(5) Feedwater System (FW)

- Feedwater pumps
- Steam Generator inlet valves and regulating valves
- High pressure heaters
- Operator error for the FW control

(6) Condensate water

(6) Condensate water
  - Condensate pumps
  - Low pressure heaters
  - Heater drain system
  - Steam packing exhausting system

(7) Condenser
  - Loss of condenser vacuum

(8) US-52 Failure
  - Unit power station failure

5.3 Analysis Code

Creating fault trees and quantifying the risk requires the use of a risk analysis code because of the complexity of the structure. *The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations(SAPHIRE)* (Ref. 2) is a set of microcomputer programs that were developed by the Idaho National Engineering Laboratory in order to create and analyze probabilistic risk assessments(PRAs), primarily for nuclear plants. SAPHIRE enables users to create fault trees and event trees easily to calculate several important parameters, such as the probability of initiating top events and minimal cut sets. In our work, analysis has been done using the risk analysis code *SAPHIRE*. The fault trees of the Seabrook and the Pilgrim Nuclear Power Stations developed by *SAPHIRE* are attached in Appendixes.

Probabilities of basic events are described in Chapter 6.

6 Failure Data

6.1 Failure Mode

In order to calculate the probability of a top event, failure rate data are obtained from several databases and assigned to basic events in fault trees. Failure rates due to component failures are mainly categorized according to the following three failure modes:

(1) Failure to operate

This type of failure is time-dependant, and probabilities are given as functions of the failure rate (frequency of failure) and the mission time of the components. This type of failure includes events in which a pump fails to run, a valve fails to remain open, or lightning strikes on the power grid within a three-year period.

(2) Failure upon demand

This type of failures occurs when a component fails upon demand. For example, an event in which a pump fails to start upon demand or a valve fails to close upon demand is categorized as this type of failure.

(3) Failure due to maintenance unavailability

Failures which occur as a result of a component being under maintenance are categorized in this type of failure mode. Unavailability due to maintenance does not imply a failure of a component. The unavailability value is usually estimated as the product of the frequency and the duration of the maintenance being performed. All components have some maintenance unavailability, mainly due to corrective and predictive maintenance. Maintenance practices differ from one plant to another; therefore, in our work, maintenance unavailability is considered only if the data are explicitly described in reference documents.

In addition to these three failure modes, two other important failure causes exist: human errors and common cause failure.

6.2 Human Errors

A great deal of research is being conducted in order to evaluate the probabilities of human errors. Human errors can be evaluated using a methodology such as Technique for Human Error Rate Prediction (THERP) (Ref.7). Convenient tables to quantify the rates of human errors are prepared in this report. The investigations on the Licensee Event Reports (LERs) of the Seabrook and Pilgrim Nuclear Power Station show that human errors accounted for about 20 % of all root causes which have contributed to plant shutdowns. This report gives guidelines for estimating human error rates under specific situations. In our work, human errors are considered only if an event occurred due to a human error, or the data of human error is explicitly described in reference documents. In some cases, assigning numerical value to the probability of human error is so difficult that the probabilities are assumed to be 1E-3 in our work.

6.3 Common Cause Failure (CCF)

During the process of developing system fault trees, potential dependencies that could compromise system reliability are identified and explicitly treated in a PRA. The first step of dependency analysis is to identify the component groups for each fault tree that could be exposed to common cause failure (CCF). CCF resulting from such factors as design, manufacturing, or installation errors can be one of the major contributors in the initiation of events. However, CCF cannot be always explicitly treated. Therefore, they are addressed by common cause failure analysis, such as Beta Factor Model and Multiple Greek Model (Ref.8). In our work, CCF is considered only if the data of CCF in the fault tree is explicitly described in reference documents.

The IPE of the Pilgrim Nuclear Power Station employs the Multi Greek Letter Model (MGL) to evaluate common cause failures. MGL is one of the parametric models in which the total failure probability of a component is divided into independent failure probability and dependent failure probability. Common cause probabilities obtained in such a way are used as one of the basic events in the fault trees. Following are examples of the analysis of MGL models used in the IPE of the Pilgrim Nuclear Power Station.

Suppose we have a system composed of three components A, B and C. Suppose this system may fail when two out of three components fail. In this case, we have seven Minimal Cut Sets (MCS) as follows:

MCS = {A,B}, {A,C}, {B,C}, {Cab}, {Cac}, {Cbc}, {Cabc}

where A : failure of component A from independent causes
Cab : failure of component A and B from common cause
Cabc : failure of component A, B and C from common cause
*Same notation in component B and C

then, we define the Basic Parameter Model as follows:

Q_k: the probability of a event involving k specific component

Using Q_k values, the system failure probability, Qs, and the total failure probability of component A, Q_a, can be written as follows;

$$Qs = 3Q_1^2 + 3Q_2 + Q_3$$
$$Qa = Q_1 + 2Q_2 + Q_3$$

If we could obtain each Q_k value from data, no further probabilistic analysis is necessary. However, the data required to estimate Q_k values directly are not normally available. Therefore, we employ such parametric models as the Beta Factor Model and MGL model in order to estimate Q_k values. Following are the key characteristics of the Beta factor model and Multi Greek Letter model for CCF quantification.

Beta Factor Model

$$Qi = (1-\beta)^* Qa$$

$$Qc = \beta^* Qa$$

where

$$\beta = \left(\sum_{k=2}^{m} kn_k\right) / \left(\sum_{k=1}^{m} kn_k\right)$$

Qi : Independent failure of a component

Qc : Global CCF which includes all CCF in one value

Qa : Total failure probability of a component

m : Number of component

n_k : Number of events involving k componetns

Multi Greek Letter Model (MGL) for three components

$$Q_1 = (1-\beta)^* Qa$$

$$Q_2 = 1/2^* \beta^* (1-\gamma)^* Qa$$

$$Q_3 = \beta^* \gamma^* Qa$$

where

$$\beta = \left(\sum_{k=2}^{m} kn_k\right) / \left(\sum_{k=1}^{m} kn_k\right)$$

$$\gamma = \left(\sum_{k=3}^{m} kn_k\right) / \left(\sum_{k=2}^{m} kn_k\right)$$

$\vdots$ for four components $\delta = \left(\sum_{k=4}^{m} kn_k\right) / \left(\sum_{k=3}^{m} kn_k\right)$

Usually, the amount of plant specific data for common cause failure analysis is very limited. Therefore, in most cases, we need to use data from the industry experience and a variety of sources to make statistical inferences about the frequency of CCF. In the IPE, the Qa value (i.e., total failure probability of component A) was simply evaluated by dividing the total number of failures by the total number of demands or total operation time. In order to estimate n_k values, a great deal of data from the Electric Power Research Institute (EPRI) data base was collected. Once n_k is determined from data, all parameters used in MGL can be calculated and CCF can be quantified. For example, the data of TBCCW pumps failing to run is as follows:

Number of components failed to run	Number of events (n_k)
1	71(+36*0.1)
2	1(+3*0.1)
3	0
4	0(+1*0.1)
5	0
6	0

Note that potential failures are added to actual number of failures in the parenthesis. The number of potential failures is multiplied by a weight factor determined as 0.1. The number of events were obtained by counting the number of failures without employing an impact vector. Using the equation above, we finally obtained the MGL parameter and the associated probabilities, as follows:

$$\beta = 0.039 \qquad \gamma = 0.13 \qquad \delta = \varepsilon = \phi = 1$$

$$Q_1 = 4.2E-6 \qquad Q_2 = 2.9E-8 \qquad Q_3 = Q_4 = Q_5 = 0 \qquad Q_6 = 2.2E-8$$

6.4 Uncertainty

6.4.1 Point Estimate

In the Individual Plant Examinations (IPE), point estimate values are used which give us exact numerical values of probability without any uncertainties. Most available data mentioned in this chapter are obtained in the form of point estimates. Therefore, point estimate values are used to evaluate failure rate and failure probability of components in the Pilgrim Nuclear Stations. Since some failure rates are given with the failure rate distribution in the IPE of the Seabrook Nuclear Power Station, we can take those uncertainties into consideration to evaluate the probability of the plant shutdown of the Seabrook Nuclear Power Station.

6.4.2 Log Normal Approximations

In practice, failure rates of components include uncertainties for several reasons. These uncertainties in failure rates affect the probabilities of component failures . In the risk analysis of nuclear power plants, the Log Normal distribution is commonly used to describe the failure rate distribution(i.e., probability density function). The failure rates used in the IPE of the Seabrook Nuclear Power Station are given by the probability distribution with the mean value, the 5^{th} and the 95^{th} percentile, and the mean value. Although they are not always Log normally distributed, the approximation by Log Normal Distribution is assumed to be satisfactory. In our work, the distributions of failure rate in the IPE of the Seabrook Nuclear Power Station are assumed to be the Log Normally Distribution in order to calculate the probability of top events.

6.4.3 Bayesian Parameter Estimation

In order to estimate the unavailability of components (i.e., failure probability) the Bayesian Estimation is commonly used in the PRA field (Ref.9). The IPE of the Seabrook Nuclear Power Station also employs the Bayesian Estimation in order to estimate the failure rates of the basic events. In the Bayesian Theorem, the probability density function of the unknown parameter ϕ (i.e., failure rate) takes the following form:

$$\pi_1(\phi/E) = \frac{L(E/\phi)}{\int_\phi L(E/\phi) * \pi_0(\phi)\, d\phi} * \pi_0(\phi)$$

where $\pi_1(\phi/E)$: Posterior probability density function of parameter ϕ

$\pi_0(\phi)$: Prior probability density function of parameter ϕ

$L(E/\phi)$: Likelihood function

E : New evidence

The left hand side of this equation is the posterior probability density function of an unknown parameter (i.e., failure rate) which is obtained by updating the prior probability distribution, given the evidence E. The Likelihood function, L, can take several forms, such as the Binomial distribution, Poisson distribution and exponential distribution in accordance with the failure mode of components. The evidence, E, is conventionally written in the form of such empirical data as the number of failures upon demand.

One of the potential problems with the conventional applications of Bayesian Estimation is that although data are typically collected from a variety of sources (i.e., different plants with different underlying parameters of interest), they are treated as coming from a single source. The result is that and the resulting posterior distribution tends to be too narrow. Therefore, the two-stage Bayesian Estimation is used in the IPE of the Seabrook Nuclear Power Station in order to address plant-to-plant variability explicitly. The general procedure in the two stage Bayesian Estimation is to create a prior distribution of the unknown parameter (i.e., failure rate) based on generic data(the first stage), and then to update this prior distribution using plant specific data. This results in a plant specific posterior distribution for failure rate. In the first stage, the parameters of failure rate are updated using generic plant data, assuming that a plant examined is not necessarily much better or much worse than the rest of the plants. Suppose a failure rate ϕ is a Beta distribution with parameters α and β. Given the evidence E from the general data, the distribution of ϕ, g(ϕ /E), is written as follows;

$$g(\phi / E) = \iint g(\phi / \alpha, \beta)\, \pi_1(\alpha, \beta/\mathrm{E})\, \mathrm{d}\alpha \mathrm{d}\beta$$

where parameters α and β are assumed to be distributed as $\pi(\alpha, \beta/\mathrm{E})$. These parameters are updated using the following equations;

$$\pi_1(\alpha, \beta/\mathrm{E}) = \frac{\mathrm{L}(\mathrm{E}/\alpha, \beta)}{\iint\limits_{\alpha,\beta} \mathrm{L}(\mathrm{E}/\alpha, \beta) * \pi_0(\alpha, \beta)\, \mathrm{d}\alpha \mathrm{d}\beta} * \pi_0(\alpha, \beta)$$

wherer $\pi_1(\alpha, \beta/\mathrm{E})$; Posterior probability density function of parameter α, β

$\pi_0(\alpha, \beta)$; Prior probability density function of parameter α, β

$\mathrm{L}(\mathrm{E}/\alpha, \beta)$; Likelihood function

E ; New evidence from generic deta

The likelihood functions are described as follows;

$$L(E/\alpha,\beta) = \prod_{i=1,\neq k}^{m} L(E/\alpha,\beta)$$

$$L(E_i/\alpha,\beta) = \int_0^1 P(r_i / n_i, \phi_i) g(\phi_i / \alpha, \beta) d\phi_i$$

Where E_i indicates the generic data of i^{th} plant, and k^{th} is the plant examined.

6.5 Source of Data

Three sources are used in this thesis.

- Databases of the Institute of Nuclear Plant Operation (INPO)
- The U.S. Nuclear Regulatory Commission (NRC)
- Plant specific data

6.5.1 The Institute of Nuclear Plant Operation (INPO)

The Institute of Nuclear Plant Operation (INPO) has been accumulating plant information including failure data. The information is systematically computerized; this is called the Equipment Performance and Information Exchange (EPIX, Ref.10). EPIX succeeded the former data management system, called Nuclear Plant Reliability Data System (NPRDS, Ref.11). NPRDS contains the failure data until 12/31/96, while EPIX contains the data since then. Since NPRDS has more failure data than EPIX, we mainly used NPRDS to obtain more accurate failure rates. EPIX is used only if some specific data cannot be obtained in NPRDS. For example, failure data of the main transformer are not available in NPRDS; therefore, we obtained them from EPIX. These data can be used to estimate the probability of each basic event only if in-house data are not available for some components. We set up the evaluation period between 1/1/90 and 12/31/96 because this period gives the latest seven-year data in NPRDS. The failure rate can be approximately estimated as follows:

$$\lambda = \frac{N}{R * Y * C * U}$$

where

λ: failure rate (1/year)

N : Number of failures of a component/system during evaluation period Y

R : Number of units (BWR : 34, PWR : 32 WH - 4Loops)

Y : Evaluation period

NPRDS : 7 years 1/1/90 - 12/31/96

EPIX : 2 years 1/1/97 - 12/31/98

C : Approximate Capacity Factor 0.7

U : Number of components per unit

The failure rate of each event is shown in section 6.7.

6.5.2 The U.S. Nuclear Regulatory Commission (NRC)

The U.S. NRC issued the report NUREG/CR-4550 *Analysis of Core Damage Frequency: Internal Events Methodology* (Ref.5), which summarizes failure rate distributions of typical components as a form of Log Normal Distribution in nuclear power stations. However, these data are so generic that we used this report only if neither the INPO nor plant specific data were available.

6.5.3 Plant Specific Data

Plant specific data are the most favorable data source because they reflect plant specific conditions, such as environment and maintenance practice. Each plant has already conducted an Independent Plant Examination (IPE), in which risk analysis is the essential element for the evaluation of plant safety. Most failure data in our work are available as plant specific data.

If available, plant specific data should be the primary source of failure data, with the INPO and the U.S. NRC data used as secondary sources.

6.5.3.1 Plant Specific Data at PNPS

The Individual Plant Examination (IPE) of the Pilgrim Nuclear Power Station(PNPS) was completed in 1992. The data which were used in IEP of PNPS included components operation history and failure history during the period between 01/01/81 and 09/30/89. These dates are selected because

1) The data analysis began in June, 1990, and the 9/30/89 end data allowed a minimum of nine months for Pilgrim component failures to be recorded and become available for access on microfilm.

2) The start date of 1/1/81 was selected based on the practice of including the equivalent of at least five full years of plant operating history in the database. PNPS had its license suspended between 1986 and 1989. Data prior to 1981 were excluded because Pilgrim's early operating history is less well documented, and is not as representative of current/future performance as more recent data is.

The data in the IPE is divided into three categories.

- Component Failure Rates (1/year, 1/demand)
- Component Maintenance Frequency and Duration (%)
- Initiating Event Frequency (1/year)
- Common cause failure
- Human error

If there were no historical data available, the data from NUREG or EPRI were substituted. Initiating event frequency, which is used for the event tree model, were simply obtained by the historical data in PNPS. The following sources were used to compile the data.

- Maintenance Request (MR)
- Failure and Maintenance Report (F&MR)
- Licensee Event Report (LER)
- Nuclear Plant Reliability Data System (NPRDS)
- Limiting Condition of Operation (LCO) Log

Point Estimate values were used for all calculations without taking the uncertainties into consideration. Following are the examples of the data used in the IPE of PNPS.

- Loss of offsite accident : 1.42E-01/y
- Manual Scram : 3.89/y
- Failure of Motor operated valve : 2.63E-3/demand

Key assumptions used in the IPE are as follows:

- Reactor runs at rated power.
- If no failures were experienced, 0.5 failures during the evaluation period were assumed to occur as a measure of conservatively estimating the failure frequency and of retaining all potential basic events in the analysis. This was done in order to avoid setting the probability of the event equal to zero, which would result in the basic events logically becoming absent from the analysis. No clear reason exists why 0.5 failures rather than some different values were assumed to occur during the period, however, this is a conservative assumption used to evaluate the risks.

- A minimum mission time of 24 hours was used for a time dependent component failure rate, except for those components which are required to satisfy their functions during a certain period. The probability of failure can be calculated by the Poisson Distribution with constant failure rate and required mission time.

In our work, the probability of top events is defined as the probability that a reactor will experience at least one shutdown between time 0 and one year. Therefore, a modification of failure rate in IPE is necessary because failure rates are given according to a 24-hour operation. Table 6.5.1 and Table 6.5.2 shows a full set of data used in the analysis of PNPS in this paper.

6.5.3.2 Plant Specific Data at SNPS

The Individual Plant Examination (IPE) of the Seabrook Nuclear Power Station makes the same assumptions as the IPE of the Pilgrim Nuclear Power Station does. Table 6.5.3 and Table 6.5.4 shows a full set of data used in the analysis of SNPS in this paper.

6.6 Key Assumptions

(1) Mission time

The top event of the fault tree is defined as the reactor shutdown within one year operation from start up due to the failure of the BOP system. Therefore, the mission time of components is one year (8760hrs).

(2) Constant failure rate

Failure rates of components/events can be obtained from the IPE data, the U.S. NRC reports, or INPO database. In our work, failure rates are assumed to be constant in order to simplify the calculation of probabilities.

(3) Exponential distribution

The probabilities of time-dependant basic events, such as failure to run or failure to remain open, are calculated by Exponential Distribution with constant failure rates as follows:

$$p = 1 - e^{-\lambda t}$$

where

p: probability of initiating basic events during [0, t]

λ: failure rate [1/year, 1/hour]

t: mission time [8760hrs]

The probability of other types of basic events are as follows:

- Failure on Demand
- Human error
- Common Cause Failure

These values are directly taken from the IPE Report because the calculations consume so much time.

(4) Point estimate

Probabilities of basic events are point estimated values in this paper because it is difficult to obtain sufficient pdf for all basic events.

(5) Evaluation period

Before 1986, the Pilgrim Nuclear Power Station (PNPS) had a poor operation history; 110 scrams were recorded within twelve operational years. PNPS had its license suspended for three years, but since 1989 it has improved its performance remarkably. Therefore, the frequency of failures will be calculated from the operation data between 1989 and 1997. The Seabrook Nuclear Power Station also started its trial operation in 1989. Therefore, the frequency of failures are also calculated during the same period.

6.7 Recorded Events at PNPS

6.7.1 Recorded Events in BOP at PNPS

Nine events recorded during the evaluation period in the BOP system at PNPS have led to unplanned shutdown. Following are the descriptions of those events and failure rate estimates. (note that PB represents Pilgrim BOP and PN represents Pilgrim NSSS)

(1) PB1

The high vibration of two bearings of the main turbine caused a manual turbine trip. The root cause was unknown. The direct cause of the reactor trip was Main Steam Isolation Valve (MSIV) closure due to the pressure fluctuation of the main steam. This event is simply described as a turbine trip event. NPRDS shows 5 turbines tripped due to the failure of bearings, and gives us the rate of occurrence of a main turbine trip due to bearing failure as 2.33E-3.

(2) PB2

The loss of condenser vacuum due to the inadequate configuration of Steam Jet Air Ejector (SJAE) required operators to scram the reactor manually. The root cause was an improper procedure used to configure SJAEs, which was described in the guideline. This event is classified as a human error. The probability of this event is assumed to be 1E-3.

(3) PB3

Potential transformer tripped due to an incorrect description in the wiring diagram. This event resulted in a generator trip and a turbine runback. The direct signal which initiated the reactor scram was the high pressure of the reactor vessel. The failure data of a potential transformer is not available on NPRDS nor EPIX. Therefore, I will substitute the failure rate of the unit auxiliary transformer in NPRDS & EPIX, 1.4E-2 (per year), for that of the potential transformer.

(4) PB4, PB5

The 345kV switchyard tripped due to flashover which was caused by salt deposition(PB5) and snow deposition(PB6). These events were evaluated in the IPE. From the historical data, the frequency of the 345kV switchyard trip is assumed to be 0.475 per year in the IPE. The deposition of snow and salt resulted in 6 out of 10 actual trips. The other 4 trips were caused by lightning. Therefore, the frequency of a 345 kV trip due to the deposition of snow and salt is assumed to be 0.285 (per year).

(5) PB6

The auxiliary transformer failed due to an unknown root cause. As mentioned in PB3, NPRDS & EPIX give the frequency of transformer failure as 1.4E-2 (per year).

(6) PB7, PB8

The main generator tripped due to the failure of the stator cooling tube(PB8) and the failure of the air-operated temperature control valve (TCV) of the cooling water system(PB9). NUREG gives the failure rate of cooling coil as 8.76E-3 (per year), and IPE of PNPS gives the failure rate of air-operated valve failure as 8.76E-4(per year).

(7) PB9

A new low pressure turbine tripped due to high vibration. This vibration was caused by a design error. The probability of design error is assumed to be 1E-3 as a human error.

In addition, lightening strikes on power grids are evaluated as failures in BOP in fault tree models.

6.7.2 Recorded Events in NSSS at PNPS

Eight events recorded during the evaluation period in the NSSS system at PNPS have led to unplanned shutdown. Following are the descriptions of those events and failure rate estimates.

(1) PN1

The feed water regulator valve (FCV) in train B failed to operate properly. This event resulted in a high level of reactor water and a manual reactor shutdown. IPE gives the probability of failure of FCV as 8.32E-4.

(2) PN2

Careless maintenance maneuver initiated spurious signals of low reactor level which resulted in an auto scram. The root cause was human error, and 1E-3 is assigned for the probability of this human error.

(3) PN3

The reactor recirculation pump failed to run due to the failure of a M-G set generator. The root cause was generator failure. NPRDS shows 47 generators failed to run during the period between 1/1/90 and 12/31/97 (7 years). Thus, the frequency of this failure is estimated as 1.3E-1.

(4) PN4, PN9, PN10

A feed water regulator valve (FCV) in train A failed to operate properly. This event resulted in a manual reactor shutdown. IPE gives the probability of failure of FCV as 8.32E-4.

(5) PN5

The leakage of the mechanical seal of a recirculation pump at PNPS resulted in a manual shutdown. The NPRDS database shows the frequency of seal leakage to be 8.27E-2.

(6) PN6

Unidentified leakage inside the Primary Containment Vessel(PCV) required the plant shutdown described in Technical Specification of PNPS. The frequency of this failure can be obtained from NPRDS as an external leakage of RPV as 5.5E-3.

(7) PN7

The plant operator misread the water level of the reactor during power reduction. This human error resulted in an auto scram and is assigned the probability of 1E-3.

(8) PN8

The Reactor Building Closed Cooling Water system (RBCCW) became inoperable due to the leakage of a heat exchanger in train A. The Technical Specifications of PNPS require operators to shut down the plant. Assuming each plant has two RBCCW heat exchangers, NPRDS gives the probability of failure of RBCCW heat exchanger as 6E-3.

In addition, failures of Main Steam Isolation Valves are evaluated as failures in NSSS in fault tree models.

6.8 Recorded Events at SNPS

6.8.1 Recorded Events in BOP at SNPS

Seventeen events recorded during the evaluation period in the BOP system at SNPS have led to plant shutdown. Following are the descriptions of those events and failure rate estimates.

(1) SB1

During the natural circulation test procedure, a condenser steam dump valve being used to control temperature failed to remain closed. This open valve caused an increased steam demand which initiated an unplanned plant cooldown. The cause of the steam dump valve failing to the full open position was the positioner feedback linkage which became disconnected during the test. The IPE gives the probability of a loss of condenser vacuum as 1.1E-1.

(2) SB2

At 30% reactor power and increasing, a turbine-generator trip with reactor trip occurred. This trip was initiated by the actuation of a main generator ground fault relay designed to protect generator windings from a ground fault. The root cause of this event was undetermined. The NPRDS gives the probability of a main generator failure as 5.13E-2

(3) SB3

A reactor tripped due to failures of the Electrohydraulic Control (EHC) oil pressure switches satisfying the two out of three logic for a reactor trip on low EHC oil pressure. The root cause of this event was excessive vibration of the EHC pressure switches due to their mounting location. This vibration caused switches to close even though adequate EHC oil pressure existed. The IPE gives the probability of a pressure switch failure as 2.69E-4.

(4) SB4

A reactor trip was initiated by a loss of voltage on the EHC 24 volt DC bus during corrective maintenance activities. The root cause for the loss of voltage on the EHC 24 volt DC bus was not conclusively determined, although a contributing factor was the troubleshooting activity associated with the EHC circuit. The IPE gives the probability of a loss of AC power, which supplies DC power to the EHC, as 1.39E-1.

(5) SB5

A reactor trip with turbine-generator trip occurred, which was initiated by a steam generator low-low narrow range level signal. The initiating event was caused by the fatigue failure of the control air pipe nipple of the feedwater flow control valve, 1-FW-FCV-520. The root cause was a vibration-induced fatigue failure of the pipe fitting due to an improper design. The IPE gives the probability of a motor operated valve failure as 8.1E-4.

(6) SB6

A reactor trip occurred due to the loss of the EHC system pressure. Prior to this event, 480 volt AC unit substation ED-US-14 was cross connected to unit substation ED-US-21 for maintenance tasks on the primary breaker, secondary breaker and transformer for ED-US-14. During the maintenance tasks, the secondary breaker for ED-US-21 tripped due to two large cyclic loads. This trip resulted in the loss of power to both EHC pumps, causing a loss of EHC system pressure. The root cause was an inadequate procedure of maintenance task. The IPE gives the probability of AC failure as 1.39E-1.

(7) SB7

A manual reactor trip was initiated while the plant was at approximately 50% power due to a turbine runback coincident with a loss of the condenser steam dump valves. While the plant was at 100% power, an electrical fault occurred in the transformer section of 480 volt AC unit substation EDE-US-52, resulting in the loss of all loads powered from this bus. Consequently, a turbine runback was initiated by the loss of Generator Stator Cooling (GSC) system control power. Additionally, control power to the condenser steam dump valves was lost. In response to the loss of the condenser steam dump valves, all four atmospheric steam dump valves and several steam generator safety valves automatically actuated. The root cause for the electrical fault was the failure of the AC transformers primary winding. The IPE gives the probability of a transformer failure as 4.35E-3.

(8) SB8

A reactor trip occurred while the plant was at 100% power due to an inadvertent actuation of the turbine mechanical overspeed protection system. During a weekly turbine mechanical overspeed trip test, the oil trip solenoid valve (OTSV) did not return to its original position. While maintenance personnel removed one of the solenoid valve housing covers to inspect the actual position of the OTSV and its limit switch, the limit switch changed state causing the mechanical lockout solenoid valve to reset and a turbine trip. The cause of the oil trip solenoid valve failure was the corrosion products inside the valve body. The NPRDS gives the probability of turbine control system failure as 1.96E-2.

(9) SB9

A reactor trip occurred while the plant was at 100% power due to the two switchyard 345kV circuit breakers tripping open. The event occurred during the performance of a preventive maintenance activity on a breaker failure relay for 345kV circuit breaker. As the relay was being returned to service, momentary arcing occurred across the contacts. The root cause was a manufacturing error in the relay housing contact block assembly for the breaker failure relay. The IPE gives the probability of loss of 345 kV as 4.72E-2.

(10) SB10

During a routine shutdown to begin a refueling outage, a steam generator (SG) level oscillations occurred in one SG. Subsequent level oscillations resulted in one SG being overfed. This caused the isolation of feedwater to all four SGs and a turbine trip.

One of four SGs reached the low-low level setpoint and initiated a reactor trip. The root cause for this event was incomplete communication to control feedwater level. The probability of human error is assumed to be 1E-3.

(11) SB11

A manual reactor trip was initiated from full-power due to the trip of two out of three Circulating Water System (CW) pumps. The CW pumps tripped due to a high differential water level across the traveling screens located upstream of the pumps suction. The high differential water level was caused by debris (primarily seaweed) brought into the CW intake during a storm. During this event, the traveling screens tripped on thermal overload allowing debris to increase the differential water level and to trip the CW pumps. The root cause for this event was the thermal overload of traveling screens. The IPE gives the probability of a motor failure as 8.06E-4 for 24 hours mission time assumed.

(12) SB12

A manual reactor trip was initiated from 100% power after both main feedwater pumps shut down due to low suction pressure. Prior to this event, a failure of the tube side relief valve for a condensate heater required isolation of the heater. Subsequent condensate and feedwater level oscillations caused the other condensate heater strings to isolate and the heater bypass valve to open. An incorrect restoration of these condensate heater strings isolated condensate flow to the main feed water pumps which caused both pumps to shut down due to low suction pressure. The primary root cause was incomplete communications and a secondary root cause was lack of procedural guidance. For purposes of illustration, I have assigned the plausible value of 1E-3 for the probability of a human error controlling feed water level.

(13) SB13

An automatic reactor trip from 100 percent power occurred, which was caused by a ground fault on the 25 kV main generator isolated phase bus. The ground fault occurred when the blade of a fan became detached from the damper frame and was carried in the air stream, eventually making contact with the bus and an air duct. The damper blade failure was attributed to an inadequate damper blade pivot pin design. IPE gives the probability of damper failure as 8.12E-4.

(14) SB14

An automatic reactor trip from 100 percent power was initiated by a reactor coolant pump (RCP) undervoltage reactor trip signal. This signal was caused by degradation of the main generator rotor field. The voltage and current used to develop the magnetic field on the main generator rotor are supplied by an exciter. The degradation of the exciter output voltage due to an arcing at the exciter brush/collector ring interface had led to a degrading exciter field. The degrading exciter field resulted in a degrading main generator field and the Station undervoltage conditions. The NPRDS gives the failure probability of an exciter as 6.37E-2.

(15) SB15

A reactor was manually tripped after both EHC pumps lost their electric power supply. Prior to the reactor trip, Unit Substation US-14 was cross-tied to Unit Substation US-21 in order to restore power to two motor control centers, after the primary feeder breaker on US-21 failure. The US-14 transformer tripped, while cross-tied to US-21, due to an unrelated end-of-life fault. This resulted in the loss of power to the EHC pumps. The loss of power to buses US-14 and US-21 complicated the secondary plant trip response. The root cause was an inadequate design of the transformer. The IPE gives the probability of loss of AC power as 1.39E-1.

(16) SB16

An automatic reactor trip occurred at 100% due to high Pressurizer pressure. This events was caused by turbine combined intercept valve (CIV) closure and turbine control valve (CV) closure. The resulting turbine load rejection caused the Reactor Coolant System temperature and pressure to increase. The turbine combined intercept valve and control valve closure was caused by a failure of a circuit card in the turbine speed control circuit in the Electro-Hydraulic Control system. The root cause was the random failure of the EHC speed control circuit card. The EPRDS gives the probability of an EHC circuit failure as 4.5E-2.

(17) SB17

The "C" Steam Generator was inadvertently drained from 38% Narrow Range (NR) level to 13% NR level causing SG Lo-Lo Reactor Protection System. The draining occurred when the "C" SG feedwater isolation valve was stroked open for testing and the SG water drained through a normally closed drain valve. The drain valve was previously opened for maintenance. After stroking open the feedwater isolation valve, control room operators noticed the decreasing "C" SG level and re-closed the isolation valve. The "C" SG reached 13% NR level and caused reactor trip. For the moment, I have assigned 1E-3 for the probability of human error controlling feed water level.

In addition, failures of Main Steam Isolation Valves, which are on the secondary side, and lightening strikes on power grids are evaluated as failures in BOP in fault tree models.

6.8.2 Recorded Events in NSSS at SNPS

Five events recorded during the evaluation period in the NSSS system at SNPS have led to plant shutdown. Following are the descriptions of those events and failure rate estimates.

(1) SN1

A reactor tripped while the plant was at 100% power due to the low flow in Reactor Coolant System (RCS), loop 3. The initiating event was caused by an electrical fault in a bus bar located in the electrical terminal connector box for Reactor Coolant Pump "C". As a result of the electrical fault, RCP "C" tripped due to protective relaying. The root cause for the electrical fault was a quarter-inch bolt that was missing from the top access cover plate of the electrical terminal connector box. Additionally, the gasket for this cover plate had pulled away from the connector box. The missing bolt and loose gasket created a direct path for dirt and moisture to enter the electrical terminal connector box which caused the electrical fault. The NPRDS gives the probability of a RCS pump motor failure as 2.28E-2.

(2) SN2

The Reactor Coolant System (RCS) unidentified leakage rate exceeded the 1.0 gpm Technical Specification limit while at 100% reactor power. The leak was found to be in the vicinity of the pressurizer, but the exact location could not be determined; therefore a controlled reactor shutdown was initiated. Following plant shutdown, the leakage source was identified as a failed tube coupling of the pressurizer gas space sample line. The IPE gives the probability of an small leakage inside a containment vessel as 3.37E-3.

(3) SN3

A reactor trip occurred from 100% power due to a spurious signal in the Reactor Coolant System (RC) Overpower Delta T (OPDT) protection circuit. The NUREG gives the probability of the sensor failure due to a spurious signal as 2.28E-2.

(4) SN4

An automatic reactor trip from 100 percent power occurred due to the electrical fault in the Train "A" Solid State Protection System (SSPS) cabinet. The failure of SSPS cabinet resulted in a loss of power to the undervoltage coil for the Train "A" reactor trip breaker. The root cause for this event was a faulty lamp test assembly in the logic test panel of the Train A SSPS cabinet. The IPE gives the probability of a loss of AC bus as 4.35E-3

(5) SN5

An automatic reactor trip occurred at 8% power during a routine shutdown to begin the refueling outage. The reactor trip was initiated by the high flux signal from one of two Intermediate Range Neutron Flux Instrumentation (IR NI) channels. The causes of the high flux were: 1) the inadequate monitoring and trending of the intermediate range channels power supplies, 2) a collective lack of knowledge about the effects that changing IR NI detector currents have on IR NI reactor trip and reset setpoint, and 3) inadequate procedural guidance in the plant shutdown procedure. The NPRDS gives the probability of an intermediate range monitor as 6.56E-3.

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
1	1TEF	Initiating events Full loss of offsite power	T	1.42E-01	1.32E-01	IPE	Loss of Offsite Power 345 & 24kV
2	1TEP	Initiating events partial loss of offsite power	T	4.75E-01	3.78E-01	IPE	Loss of Offsite Power 345kV
3	BHX209AXXU	HxA maintenance	M		2.66E-02	IPE	
4	BHX209BXXU	HxB maintenance	M		2.66E-02	IPE	
5	BPM5XXXCCR	CCF of 5 or more pumps failure to run	T	1.93E-04	1.93E-04	IPE	
6	BPM5XXXCCS	CCF 5 or more FTS	D		4.80E-05	IPE	
7	BPMABCXCCR	A,B,C pumps CCF to Run	T	0.00E+00	0.00E+00	IPE	
8	BPMABCXCCS	A,B,C CCF FTS	D		0.00E+00	IPE	
9	BPMABXXCCR	Pump A,B CCF to run	T	2.54E-04	2.54E-04	IPE	
10	BPMABXXCCS	A,B CCF FTS	D		1.40E-05	IPE	
11	BPMACXXCCR	Pump A,C CCF to run	T	2.54E-04	2.54E-04	IPE	
12	BPMACXXCCS	A,C CCF FTS	D		1.40E-05	IPE	
13	BPMAXXXXXR	202A FTR	T	3.69E-02	3.62E-02	IPE	
14	BPMAXXXXXS	202A FTS	D		6.30E-04	IPE	
15	BPMBCXXCCR	Pump B, C CCF to run	T	2.54E-04	2.54E-04	IPE	
16	BPMBCXXCCS	B,C CCF FTS	D		1.40E-05	IPE	
17	BPMBXXXXXR	202B FTR	T	3.69E-02	3.62E-02	IPE	
18	BPMBXXXXXS	202B FTS	D		6.30E-04	IPE	
19	BPMBXXXXXU	202B corrective Maintenance	M		1.09E-02	IPE	
20	BPMCXXXXXR	202C FTR	T	3.69E-02	3.62E-02	IPE	
21	BPMCXXXXXS	202C FTS	D		5.30E-04	IPE	
22	BPMCXXXXXU	202C corrective Maintenance	M		1.09E-02	IPE	
23	BPMDEFXCCR	D,E,F pumps CCF to Run	T	0.00E+00	0.00E+00	IPE	
24	BPMDEFXCCS	D,E,F CCF FTS	D		0.00E+00	IPE	
25	BPMDEXXCCR	Pump D,E CCF to run	T	2.54E-04	2.54E-04	IPE	
26	BPMDEXXCCS	D,E CCF FTS	D		1.40E-05	IPE	

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
27	BPMDFXXCCR	Pump D,F CCF to run	T	2.54E-04	2.54E-04	IPE	
28	BPMDFXXCCS	D,F CCF FTS	D		1.40E-05	IPE	
29	BPMDXXXXXR	202D FTR	T	3.69E-02	3.62E-02	IPE	
30	BPMDXXXXXS	202D FTS	D		6.30E-04	IPE	
31	BPMEFXXCCR	Pump E,F CCF to run	T	2.54E-04	2.54E-04	IPE	
32	BPMEFXXCCS	E,F CCF FTS	D		1.40E-05	IPE	
33	BPMEXXXXXR	202E FTR	T	3.69E-02	3.62E-02	IPE	
34	BPMEXXXXXS	202E FTS	D		6.30E-04	IPE	
35	BPMEXXXXXU	202E corrective Maintenance	M		1.09E-02	IPE	
36	BPMFXXXXXR	202F FTR	T	3.69E-02	3.62E-02	IPE	
37	BPMFXXXXXS	202F FTS	D		6.30E-04	IPE	
38	BPMFXXXXXU	202F corrective Maintenance	M		1.09E-02	IPE	
39	BPV1	BPV1 FTRO	T	4.50E-03	4.49E-03	NPRDS	
40	BPV2	BPV2 FTRO	T	4.50E-03	4.49E-03	NPRDS	
41	BPV3	BPV3 FTRO	T	4.50E-03	4.49E-03	NPRDS	
42	BSP4008XXE	PS4008 FT energize	D		1.00E-05	IPE	PS fails to energize
43	BSP4058XXE	PS4058 FT energize	D		1.00E-05	IPE	PS fails to energize
44	BVC419XXXN	CheckValve 30ck419 FTO	D		1.00E-04	IPE	Pump discharge check valve
45	BVC420XXXN	CheckValve 30ck420 FTO	D		.00E-04	IPE	Pump discharge check valve
46	BVC421XXXN	Check valve 30ck421 FTO	D		1.00E-04	IPE	Pump discharge check valve
47	BVC422XXXN	Check valve 30ck422 FTO	D		1.00E-04	IPE	Pump discharge check valve
48	BVC423XXXN	CheckValve 30ck423 FTO	D		1.00E-04	IPE	Pump discharge check valve
49	BVC424XXXN	CheckValve 30ck424 FTO	D		1.00E-04	IPE	Pump discharge check valve
50	BVH1XXXXXF	HO-1 FTRO	T	8.76E-04	8.76E-04	IPE	HO
51	BVH2XXXXXF	HO-2 FTRO	T	8.76E-04	8.76E-04	IPE	HO
52	BVH3837XXF	HO-3837 FTRO	T	8.76E-04	8.76E-04	IPE	HO

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
53	BVH3842XXF	HO-3842 FTRO	T	8.76E-04	8.76E-04	IPE	HO
54	BVH5XXXXXF	HO-5 FTRO	T	8.76E-04	8.76E-04	IPE	HO
55	BVH6XXXXXF	HO-6 FTRO	T	8.76E-04	8.76E-04	IPE	HO
56	BVM3800XXF	MO-3800 FTRO	T	8.76E-04	8.76E-04	IPE	HO
57	BVM3801XXC	MO-3801 FTC	D		1.65E-03	IPE	MO
58	BVM3805XXC	MO-3805 FTC	D		1.65E-03	IPE	MO
59	BVM3806XXF	MO-3806 FTRO	T	8.76E-04	8.76E-04	IPE	MO
60	CCF_FCV642	CCF FCV642AB	T	4.38E-05	4.38E-05	IPE	
61	CV1	CV-1 FTRO	T	1.05E-02	1.04E-02	NPRDS	
62	CV2	CV-2 FTRO	T	1.05E-02	1.04E-02	NPRDS	
63	CV3	CV-3 FTRO	T	1.05E-02	1.04E-02	NPRDS	
64	CV4	CV-4 FTRO	T	1.05E-02	1.04E-02	NPRDS	
65	DRNCOOLERA	Drain cooler A failure	T	6.00E-04	6.00E-04	NPRDS	Same value as feedwater heater failure
66	DRNCOOLERB	Drain cooler B failure	T	6.00E-04	6.00E-04	NPRDS	Same value as feedwater heater failure
67	EXPJOINTHTR	Expansion joint failure	T	1.81E-04	1.81E-04	IPE	Turbine expansion joints
68	FCV643FTO	FCV643 FTO	D		2.00E-03	IPE	
69	FCV643OPER	Operator FTO 643	D		5.00E-03	IPE	
70	FLUIDCOUPLING(	Fluid Coupling(A) failure	T	6.34E-02	6.14E-02	NPRDS	
71	FLUIDCOUPLING(	Fluid Coupling(B) failure	T	6.34E-02	6.14E-02	NPRDS	
72	FUEL	Fuel failure		0.00E+00			P=0
73	FV3435	Rec valve 3435 FTRC	T	4.38E-03	4.37E-03	IPE	
74	FV3436	Rec valve 3436 FTRC	T	4.38E-03	4.37E-03	IPE	
75	FV3437	Rec valve 3437 FTRC	T	4.38E-03	4.37E-03	IPE	
76	FWPAP	FWP A pump FTR	T	2.98E-01	2.58E-01	IPE	FTR
77	FWPBP	FWP B pump FTR	T	2.98E-01	2.58E-01	IPE	FTR
78	FWPCP	FWP C pump FTR	T	2.98E-01	2.58E-01	IPE	FTR

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
79	HTR1A	HTR 1A failure	T	6.00E-04	6.00E-04	NPRDS	Feedwater Heater failure
80	HTR1B	HTR 1B failure	T	6.00E-04	6.00E-04	NPRDS	
81	HTR2A	HTR 2A failure	T	6.00E-04	6.00E-04	NPRDS	
82	HTR2B	HTR 2B failure	T	6.00E-04	6.00E-04	NPRDS	
83	HTR3A	HTR 3A failure	T	6.00E-04	6.00E-04	NPRDS	
84	HTR3B	HTR 3B failure	T	6.00E-04	6.00E-04	NPRDS	
85	HTR4A	HTR 4A failure	T	6.00E-04	6.00E-04	NPRDS	
86	HTR4B	HTR 4B failure	T	6.00E-04	6.00E-04	NPRDS	
87	HTR5A	HTR 5A failure	T	6.00E-04	6.00E-04	NPRDS	
88	HTR5B	HTR 5B failure	T	6.00E-04	6.00E-04	NPRDS	
89	HTRBPV	HTR BPV FTRC	T	4.38E-03	4.38E-03	IPE	value from FCV Recirculation Valve
90	HXBLEAKAGE	HxB Leakage	T	6.00E-03	5.98E-03	NPRDS	RBCCW Hx failure: PN8
91	Internal	Internal Structure		0.00E+00			Core internal : P=0
92	MGGENERATOR(E	MGset generator failure	T	1.30E-01	1.22E-01	NPRDS	PN3
93	MGMOTOR(A)	MGset Motor Failure	T	3.86E-02	3.79E-02	NPRDS	
94	MGMOTOR(B)	MGset Motor Failure	T	3.86E-02	3.79E-02	NPRDS	
95	MO3428	MO3428 FTRO	T	4.38E-03	4.37E-03	IPE	FCV
96	MO3472	MO3472 FTRO	T	4.38E-03	4.37E-03	IPE	FCV
97	MO4A	Inlet MO 4A FTRO	D		2.63E-03	IPE	IPE : MOV/Demand
98	MO4B	Inlet MO 4B FTRO	D		2.63E-03	IPE	IPE : MOV/Demand
99	MO5A	Disch MO5A FTRO	D		2.63E-03	IPE	IPE : MOV/Demand
100	MO5B	Disch MO5B FTRO	D		2.63E-03	IPE	IPE : MOV/Demand
101	MOTOR(A)FAIL	Motor(A) fail to run	T	3.86E-02	3.79E-02	NPRDS	
102	MOTOR(B)FAIL	Motor(B) fail to run	T	3.86E-02	3.79E-02	NPRDS	
103	MSIVA1	MSIV A1 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
104	MSIVA2	MSIV A2 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
105	MSIVB1	MSIV B1 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
106	MSIVB2	MSIV B2 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
107	MSIVC1	MSIV C1 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
108	MSIVC2	MSIV C2 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
109	MSIVD1	MSIV D1 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
110	MSIVD2	MSIV D2 FTRO	T	8.76E-04	8.76E-04	IPE	FTRO
111	NEUTRON	Neutron monitoring (APRM, LPRM,,,)	T	1.20E-01	1.13E-01	NPRDS	RPS-Neutron monitor
112	PLROIL(A)	Lube oil system failure	T	8.27E-03	8.24E-03	NPRDS	
113	PLROIL(B)	Lube oil system failure	T	8.27E-03	8.24E-03	NPRDS	
114	PLRPP(A)FAIL	Pump A failure	T	3.59E-02	3.53E-02	NPRDS	
115	PLRPP(B)FAIL	Pump B failure	T	3.59E-02	3.53E-02	NPRDS	
116	PLRSEAL(A)	Seal Leakage (A)	T	8.27E-02	7.94E-02	NPRDS	PN5
117	PN1	FCV642B failure	T	8.32E-04	8.32E-04	IPE	FCV B failure (PN 4,9,10)
118	PN3	MGset generator failure	T	1.30E-01	1.22E-01	NPRDS	MG A generator (83)
119	PN4_9 10	FCV642A failure	T	8.32E-04	8.32E-04	IPE	FCV A failure (PN1)
120	PN5	Seal Leakage (B)	T	8.27E-02	7.94E-02	NPRDS	PLR(B) Seal leak
121	PN6	Miscellaneous leakage (RPV Drain line Leakage)	T	5.50E-03	5.48E-03	NPRDS	RPV drain line leak
122	PN7_PN2	Human Error	H		1.00E-03		Preliminary: Misreading level & mant error
123	PN8	RBCCW Hx(A) Leakage	T	6.00E-03	5.98E-03	NPRDS	RBCCW Hx A leak (81)
124	RPS_DEVICE	RPS Device Failure	T	9.00E-02	8.61E-02	NPRDS	RPS failure result in shutdown
125	RPVINST	Instrumentation	T	1.10E-03	1.09E-03	NUREG	Instruments fail to operate
126	SRV A	SRV A Leak	T	1.02E-01	9.70E-02	NPRDS	Two leakage recorded
127	SRV B	SRV B Leak	T	1.02E-01	9.70E-02	NPRDS	1/1/90-12/31/96, CF=0.7
128	SRV C	SRV C Leak	T	1.02E-01	9.70E-02	NPRDS	
129	SRV D	SRV D Leak	T	1.02E-01	9.70E-02	NPRDS	
130	STV1	STV1 FTRO	T	1.95E-02	1.93E-02	NPRDS	

Table 6.5.1 Datasheet of Basic Events for NSSS System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
131	STV2	STV2 FTRO	T	1.95E-02	1.93E-02	NPRDS	
132	STV3	STV3 FTRO	T	1.95E-02	1.93E-02	NPRDS	
133	STV4	STV4 FTRO	T	1.95E-02	1.93E-02	NPRDS	
134	SVA	SV A Leakage	T	1.02E-01	9.70E-02	NPRDS	values form SRV
135	SVB	SV B Leakage	T	1.02E-01	9.70E-02	NPRDS	values form SRV
136	TB14	Loss of Power from B14	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
137	TB15	Loss of Power from B15	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
138	TSSWA	No SSW Loop A	T	3.36E-03	3.35E-03	IPE	
139	TSSWB	No SSW Loop B	T	3.36E-03	3.35E-03	IPE	
140	TY3	Loss of 120 V AC Panel Y3	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
141	TY4	Loss of 120 V AC Panel Y4	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss

Note
(a) T: Time dependant failure mode (I.e., Fail to operate/run/transfer closed)
D : Failure on Demand
M: Maintenance unavailability
(b) IPE : The Individual Plant Examination
NPRDS :The INPO database
EPIX : The INPO database

Table 6.5.2 Datasheet of Basic Events for BOP System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
1	3703FTRE	3703FTR energized	T	8.76E-04	8.76E-04	IPE	
2	3703FTRO	3703FTRO	T	8.76E-04	8.76E-04	IPE	
3	3704FTRE	3704FTR energized	T	8.76E-04	8.76E-04	IPE	
4	3704FTRO	3704FTRO	T	8.76E-04	8.76E-04	IPE	
5	3710FTRE	3710FTR energized	T	8.76E-04	8.76E-04	IPE	
6	3710FTRO	3710FTRO	T	8.76E-04	8.76E-04	IPE	
7	3711FTRE	Solenoid valve 3711 FTR energized	T	8.76E-04	8.76E-04	IPE	
8	3711FTRO	3711FTRO	T	8.76E-04	8.76E-04	IPE	
9	9212FTRE	Solenoid valve 9212 FTR energized	T	8.76E-04	8.76E-04	IPE	
10	9212FTRO	9212FTRO	T	8.76E-04	8.76E-04	IPE	
11	9213FTRE	Solenoid valve 9213 FTR energized	T	8.76E-04	8.76E-04	IPE	
12	9213FTRO	9213FTRO	T	8.76E-04	8.76E-04	IPE	
13	9222FTRE	Solenoid valve 9222 FTR energized	T	8.76E-04	8.76E-04	IPE	
14	9222FTRO	9222FTRO	T	8.76E-04	8.76E-04	IPE	
15	9251FTRE	Solenoid valve 9251 FTR energized	T	8.76E-04	8.76E-04	IPE	
16	9251FTRO	9251FTRO	T	8.76E-04	8.76E-04	IPE	
17	9252FTRE	Solenoid valve 9252 FTR energized	T	8.76E-04	8.76E-04	IPE	
18	9252FTRO	9252FTRO	T	8.76E-04	8.76E-04	IPE	
19	9255FTRE	Solenoid valve 9255 FTR energized	T	8.76E-04	8.76E-04	IPE	
20	9255FTRO	9255FTRO	T	8.76E-04	8.76E-04	IPE	
21	9259FTRE	Solenoid valve 9259 FTR energized	T	8.76E-04	8.76E-04	IPE	
22	9259FTRO	9259FTRO	T	8.76E-04	8.76E-04	IPE	
23	9269FTRE	Solenoid valve 9269 FTR energized	T	8.76E-04	8.76E-04	IPE	
24	9269FTRO	9269FTRO	T	8.76E-04	8.76E-04	IPE	-
25	A3	No power available from A3	T	3.00E-03	3.00E-03		Same Value as DC loss
26	A4	No power available from A4	T	3.00E-03	3.00E-03		Same Value as DC loss
27	AIRLOSS	loss of essential air	T	2.00E-04	2.00E-04	IPE	Initiating Event
28	AO3707	AO3707FTRC	T	4.38E-03	4.37E-03	IPE	
29	AO3708	AO3708FTRC	T	4.38E-03	4.37E-03	IPE	
30	AO3751FTRO	AO3751FTRO	T	8.76E-04	8.76E-04	IPE	
31	AO9238	AO9238FTRO	T	8.76E-04	8.76E-04	IPE	
32	AO9239	AO9239FTRO	T	8.76E-04	8.76E-04	IPE	
33	AOG B	Operator fails to align AOG train B	H		1.50E-02	IPE	
34	B1	No power available from B1	T	3.00E-03	3.00E-03		Same Value as DC loss
35	B2	No power from B2	T	3.00E-03	3.00E-03		Same Value as DC loss
36	B22	Transfer- No power available from B22	T	3.00E-03	3.00E-03		Same Value as DC loss
37	B23	No power Available from B23	T	3.00E-03	3.00E-03		Same Value as DC loss
38	B30	No-power available from B30	T	3.00E-03	3.00E-03		Same Value as DC loss
39	B31	No power Available from B31	T	3.00E-03	3.00E-03		Same Value as DC loss
40	BANKS PULGGED	Demineralizer resin banks plugged	T	4.99E-02	4.87E-02	IPE	
41	BCORRETMAIN	Train B out for Corrective maintenance	M		3.00E-03	IPE	
42	CCFFTR130-A B	CCF FTR Pump 130-A B	T	9.64E-03	9.59E-03	IPE	
43	CCFFTS130-A B	CCF FTS Pump 130-A B	D		4.00E-04	IPE	
44	CCFP105AB	CCF P105A/B to run	D		2.26E-05	IPE	
45	CIV1	CIV #1	T	6.30E-02	6.11E-02	EPIX	
46	CIV2	CIV #2	T	6.30E-02	6.11E-02	EPIX	

Table 6.5.2 Datasheet of Basic Events for BOP System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency ·er year)	Probability	Data Source (b)	Comments
47	CIV3	CIV #3	T	6.30E-02	6.11E-02	EPIX	
48	CIV4	CIV #4	T	6.30E-02	6.11E-02	EPIX	
49	D4	loss of 125 DC From D-4	T	3.00E-03	3.00E-03	IPE	Initiating Event
50	D5	Loss of 125 V DC from D5	T	3.00E-03	3.00E-03	IPE	Initiating Event
51	DEIONIZER	Deionizer failure (Plugged)	T	4.99E-02	4.87E-02	IPE	NUREG : BANKS PULGGED
52	EAST-JOINT	East Joint	T	1.81E-04	1.81E-04	IPE	
53	F3301FTRO	Makeup Valve AO-3301 FTRO	T	8.76E-04	8.76E-04	IPE	
54	FCPA	Condensate pump A FTR	T	2.98E-01	2.58E-01	IPE	
55	FCPB	Condensate pump B FTR	T	2.98E-01	2.58E-01	IPE	
56	FCPC	Condensate pump C FTR	T	2.98E-01	2.58E-01	IPE	
57	FINITHWMUC	Hotwell makeup capacity loss	T	1.17E-02	1.17E-02		
58	FLPHTRBU	LP HTR Train B unavailable	M		1.44E-03	IPE	
59	FLPLPHTRAU	LP HTR TRAIN A out for maint.	M		1.44E-03	IPE	
60	FVHH0110XF	Makeup Valve 26-HO-110 FTRO	T	8.76E-04	8.76E-04	IPE	
61	FVHH0111XF	Makeup Valve 26-HO-111 FTRO	T	8.76E-04	8.76E-04	IPE	
62	FVM3371XXF	MO3371 FTRO	T	8.76E-04	8.76E-04	IPE	
63	FVM3372XXF	MO3372 FTRO	T	8.76E-04	8.76E-04	IPE	
64	FVM3427XXF	MO3427 FTRO	T	8.76E-04	8.76E-04	IPE	
65	FVM3428XXF	MO3428 FTRO	T	8.76E-04	8.76E-04	IPE	
66	GENEXICITER	Generator Exciter failure	T	4.67E-02	4.56E-02	N,E	NPRDS & EPIX
67	GENHX FAILURE	Hx failure (Plugged)	T	4.99E-02	4.87E-02	IPE	NUREG : BANKS PULGGED
68	GENHYDOGEN	Generator hydrogen cooler failure	T	9.34E-03	9.29E-03	N,E	NPRDS & EPIX
69	GENSTATOR	Main Generator/stator/ voltage regulator failure	T	6.54E-02	6.33E-02	N,E	NPRDS & EPIX
70	GENTRIP	GENTRIP			1.00E+00		
71	HYDROGENOILSE	Hydrogen Oil seal failure	T	9.34E-03	9.29E-03	N,E	NPRDS & EPIX
72	MAINTRANS	Main Transformer failure	T	8.40E-02	8.06E-02	N,E	NPRDS & EPIX
73	MO9206	MO9206FTRO	T	8.76E-04	8.76E-04	IPE	
74	MO9207	MO9207FTRO	T	8.76E-04	8.76E-04	IPE	
75	MO9271FTRO	Coolant Discrg valve MO9271 FTRO	T	8.76E-04	8.76E-04	IPE	
76	MOVS2	MOVS2FTO	D		1.65E-03	IPE	
77	MOVS 2FTRO	MOVS 2FTRO	T	8.76E-04	8.76E-04	IPE	
78	MOV B	Unloaded valve MOV B fail to operate	D		1.65E-03	IPE	
79	MOV D 1	Damper MOV D 1 FTRO	T	8.76E-04	8.76E-04	IPE	
80	MOV D 2	Damper MOV D 2 FTO	D		1.65E-03	IPE	
81	OPEEXTAB	Operator fails to switch to exhaust Train B	H		1.50E-02	IPE	
82	OPEFAIL	Operator Fail select AOG low flow bypass	H		5.00E-03	IPE	
83	OPEMOV S 2	Operator fails to open Bypass valve MOV-s-2	H		5.00E-03	IPE	
84	P105AFTR	P105A fail to run	T	2.63E-01	2.30E-01	IPE	
85	P105BFTR	P105B fail to run	T	2.63E-01	2.30E-01	IPE	
86	P130AFTR	P130AFTR	T	2.63E-01	2.30E-01	IPE	
87	P130AFTS	P130AFTS	D		2.60E-03	IPE	
88	P130BFTR	P130BFTR	T	2.63E-01	2.30E-01	IPE	
89	P130BFTS	P130BFTS	D		2.60E-03	IPE	
90	P136A FTR	P136A FTR	T	2.63E-01	2.30E-01	NUREG	
91	P136B FTS	P136B FTS	D		3.00E-03	NUREG	
92	PB1	Bearing Failure	T	2.33E-03	2.33E-03	NRPDS	Bearing failure

Table 6.5.2 Datasheet of Basic Events for BOP System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
93	PB2	Human Error SJAE	H		1.00E-03		SJAE Preliminary
94	PB3 PB6	Unit Auxiliary Trans failure	T	1.40E-02	1.39E-02	N,E	Unit Trans
95	PB4 PB5	SALT SNOW Deposit	T	2.85E-01	2.48E-01	IPE	Failure rate=0.475*6/10 (1/y)
96	PB7	Cooling coil failure	T	8.76E-03	8.72E-02	NUREG	Gen cooling coil
97	PB8	TCV Y07 FTRO	T	8.76E-04	8.76E-04	IPE	AOV
98	PB9	HP/LP turbine failure	T	3.00E-03	3.00E-03	NPRDS	LP high vibration
99	PCV3701	PCV3701FTRO	T	8.76E-04	8.76E-04	IPE	
100	PCV Y 63	PCV Y63 FTRO	T	8.76E-04	8.76E-04	IPE	
101	STARTRANS	Startup Trans failure	T	2.33E-02	2.31E-02	N,E	NPRDS & EPIX
102	SWTHUNDER	Switchyard fails due to thunder strike	T	1.90E-01	1.73E-01	IPE	Failure rate=0.475*4/10(1/y)
103	TBCONT	Control system failure	T	1.20E-02	1.19E-02	NPRDS	Governor
104	TBLUB	Lubrication system failure	T	8.40E-02	8.06E-02	EPIX	
105	TBMHC	MHC failure	T	9.00E-02	8.61E-02	NPRDS	
106	TBMS	Moisture separator	T	1.50E-03	1.50E-03	NPRDS	
107	TS3717A	Spurious Actuation TS3717A	T	1.93E-02	1.92E-02	IPE	
108	TS3717B	Spurious Actuation TS3717B	T	1.93E-02	1.92E-02	IPE	
109	TS3718A	Spurious Actuation TS3718A	T	1.93E-02	1.92E-02	IPE	
110	TS3718B	Spurious Actuation TS3718B	T	1.93E-02	1.92E-02	IPE	
111	TSSS	Steam Seal Supply System	T	1.80E-02	1.78E-02	NPRDS	
112	TSSWA	No SSW Loop A	T	3.36E-03	3.36E-03	IPE	
113	TSSWB	NO SSW LOOP B	T	3.36E-03	3.36E-03	IPE	
114	TVA3351XXL	Recirculation Valve FV3351 FTRC	T	4.38E-03	4.37E-03	IPE	
115	WEST-JOINT	West Joint	T	1.81E-04	1.81E-04	IPE	
116	WHX122AXXF	Hx 122A Plugs	T	4.99E-02	4.87E-02	IPE	
117	WHX122AXXU	Hx E122A out for Maintenance	M		2.80E-05	IPE	
118	WHX122BXXF	Hx E122B Plugs	T	4.99E-02	4.87E-02	IPE	
119	WHX122BXXU	Hx E122B out for maintenance	M		2.80E-05	IPE	
120	WPM110AMXS	110A Mechanical FTS	D		6.50E-04	IPE	
121	WPM110AXXR	110A FTR	T	3.68E-02	3.62E-02	IPE	
122	WPM110BMXS	110B Mechanical FTS	D		6.50E-04	IPE	
123	WPM110BXXR	110B FTR	T	3.68E-02	3.62E-02	IPE	
124	WPM110BXXU	TBCCW Pump B train out for Maintenance	M		3.00E-03	IPE	
125	WPM110XCCR	110A, B CC FTR	T	1.40E-03	1.40E-03	IPE	
126	WPM110XCCS	110A, B CC FTS	D		9.90E-05	IPE	
127	WSP4176AXR	PS4176A Fails to Pick up	D		1.00E-05	IPE	
128	WSP4176BXR	PS4176B Fails to Pick up	D		1.00E-05	IPE	
129	WSP4176CCR	PS4176A,B CC Fail to pick up	D		5.04E-04	IPE	
130	WTE4161XXR	Temperature Element TE 4161 Fails High	T	3.68E-02	3.58E-02	IPE	
131	WTI62ABCCR	62A, 62B CC FTOOP	D		1.03E-03	IPE	
132	WTI62AXXXR	Timer 62A fails to operate	D		9.27E-03	IPE	
133	WTI62BXXXR	Timer 62B Fails to operate	D		9.27E-03	IPE	
134	WVC961XCCN	CK-961, 962 CC FTO	D		1.00E-04	IPE	
135	WVC961XXXN	CK-961 FTO	D		1.00E-04	IPE	
136	WVC962XXXN	CK-962 FTO	D		1.00E-04	IPE	
137	WVH665XXXF	HO-665 FTRO	T	8.76E-04	8.76E-04	IPE	
138	WVH667XXXF	HO-667 FTRO	T	8.76E-04	8.76E-04	IPE	

Table 6.5.2 Datasheet of Basic Events for BOP System at the Pilgrim Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability	Data Source (b)	Comments
139	WVH668XXXF	HO-668 FTRO	T	8.76E-04	8.76E-04	IPE	
140	WVH670XXXF	HO-670 FTRO	T	8.76E-04	8.76E-04	IPE	
141	WVM380110F	MO 3801 FTR 10% Open	T	8.76E-03	8.72E-03	IPE	
142	WVM3801XXF	MO3801 FTRO	T	8.76E-04	8.76E-04	IPE	
143	WVM380510F	MO-3805 FTR 10% OPEN	T	8.76E-03	8.72E-03	IPE	
144	WVM3805XXF	MO-3805 FTRO	T	8.76E-04	2.40E-06	IPE	
145	WVT4161XXC	Temperature Valve TV 4161 FTC	D		2.00E-03	IPE	
146	X102	Regulator X102 fail to operate	T	3.65E-02	3.58E-02	IPE	
147	Y1	No power Available from Y1	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
148	Y2	No power Available from Y2	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
149	Y3	Loss of 12V AC from Y-3	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss
150	Y4	Loss of 120 V AC From Y4	T	3.00E-03	3.00E-03	IPE	Same Value as DC loss

Note

(a) T: Time dependant failure mode (I.e.. Fail to operate/run/transfer closed)
D : Failure on Demand
M: Maintenance unavailability
(b) IPE : The Individual Plant Examination
NPRDS :The INPO database
EPIX : The INPO database

Table 6.5.3 Datasheet of Basic Events for NSSS System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
1	FUEL	Fuel		0.00E+00						Not taken into consideration	
2	INTERNAL	Internal		0.00E+00						Not taken into consideration	
3	IRA	IR Channel A Detector Failure SN5	T	6.58E-03	6.58E-04	3.29E-03	6.56E-03	1.31E-02	NPRDS	SN5	
4	IRB	IR Channel B Detector Failure	T	6.58E-03	6.58E-04	3.29E-03	6.56E-03	1.31E-02	NPRDS		
5	PRN41A	PR N41 Detector A Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
6	PRN41B	PR N41 Detector B Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
7	PRN42A	PR N42 Detector A Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
8	PRN42B	PR N42 Detector B Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
9	PRN43A	PR N43 Detector A Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
10	PRN43B	PR N43 Detector B Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
11	PRN44A	PR N44 Detector A Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
12	PRN44B	PR N44 Detector B Failure	T	8.23E-04	8.23E-05	4.11E-04	8.23E-04	1.64E-03	NPRDS		
13	RCHEM	Chemical and Volume Control Sys Failure	T	7.24E-02	7.22E-03	3.56E-02	6.99E-02	1.35E-01	NPRDS		
14	RCSAELE	RCS Pump A Electrical Failure SN1	T	2.30E-02	2.30E-03	1.15E-02	2.28E-02	4.50E-02	NPRDS	SN1	
15	RCSBELE	RCS Pump B Electrical Failure	T	2.30E-02	2.30E-03	1.15E-02	2.28E-02	4.50E-02	NPRDS		
16	RCSCELE	RCS Pump C Electrical Failure	T	2.30E-02	2.30E-03	1.15E-02	2.28E-02	4.50E-02	NPRDS		
17	RCSDELE	RCS Pump D Electrical Failure	T	2.30E-02	2.30E-03	1.15E-02	2.28E-02	4.50E-02	NPRDS		
18	RCSPA	RCS Pump A Mechanical Failure	T	1.65E-02	1.64E-03	8.20E-03	1.63E-02	3.24E-02	NPRDS		
19	RCSPB	RCS Pump B Mechanical Failure	T	1.65E-02	1.64E-03	8.20E-03	1.63E-02	3.24E-02	NPRDS		
20	RCSPC	RCS Pump C Mechanical Failure	T	1.65E-02	1.64E-03	8.20E-03	1.63E-02	3.24E-02	NPRDS		
21	RCSPD	RCS Pump D Mechanical Failure	T	1.65E-02	1.64E-03	8.20E-03	1.63E-02	3.24E-02	NPRDS		
22	RPCV455A	Pressurizer Spray Valve PCV-455A Failure	T	8.78E-03	8.77E-04	4.38E-03	8.74E-03	1.74E-02	NPRDS	Pressurizer Spray Valve : from Safety Valve	
23	RPCV455B	Pressurizer Spray Valve PCV-455B Failure	T	8.78E-03	8.77E-04	4.38E-03	8.74E-03	1.74E-02	NPRDS	Pressurizer Spray Valve : from Safety Valve	
24	RPORV456A	PROV456A Failure	T	3.29E-03	3.29E-04	1.64E-03	3.29E-03	6.56E-03	NPRDS		
25	RPORV456B	PROV456B Failure	T	3.29E-03	3.29E-04	1.64E-03	3.29E-03	6.56E-03	NPRDS		
26	RPRESSRELIEFT	Pressurizer Relief Tank External Leakage	T	2.33E-04	2.33E-05	1.17E-04	2.33E-04	4.66E-04	IPE	Tank Rapture	7.34
27	RPRESSURE	Pressurizer Failure	T	3.29E-03	3.29E-04	1.64E-03	3.29E-03	6.56E-03	NPRDS		
28	RPSMGAFTR	RPS MG A FTR	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE	M/D PUMP	5.49
29	RPSMGBFTS	RPS MG B FTS	D				2.35E-03		IPE	M/D PUMP	4.46
30	RPSMGGENBRKA	RPS Generator Breaker A FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46
31	RPSMGGENBRKB	RPS Generator Breaker B FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46
32	RPSPWRA	Train A 120V AC Power Loss SN4	T	4.36E-03	4.36E-04	2.18E-03	4.35E-03	8.69E-03	IPE	SN4 BUS failure	3.48
33	RPSPWRB	Train B 120V AC Power Loss	T	4.36E-03	4.36E-04	2.18E-03	4.35E-03	8.69E-03	IPE		3.48
34	RPSTRIPBRKRAF	RPS Trip Breaker A FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46
35	RPSTRIPBRKRBF	RPS Trip Breaker B FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46
36	RPSTRIPBRKRBY	RPS Trip Breaker Bypass A FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46

Table 6.5.3 Datasheet of Basic Events for NSSS System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
37	RPSTRIPBRKRBY	RPS Trip Breaker Bypass B FTRC	T	2.35E-03	2.35E-04	1.17E-03	2.34E-03	4.68E-03	IPE		6.46
38	RPSTRNRLAYA	RPS Train A Relays Failure	T	3.68E-03	3.68E-04	1.84E-03	3.67E-03	7.33E-03	IPE		7.42
39	RPSTRNRLAYB	RPS Train B Relays Failure	T	3.68E-03	3.68E-04	1.84E-03	3.67E-03	7.33E-03	IPE		7.42
40	RPVINSTR	RV Instrument Failure	T	5.92E-02	5.91E-03	2.92E-02	5.75E-02	1.12E-01	NPRDS		
41	RPVLEAK	Miscellaneous Leakage SN2	T	3.40E-02	3.40E-03	1.69E-02	3.34E-02	6.58E-02	IPE	SN2 Large leak inside Containment	10.97
42	RRCV115	Pressurizer Safety Valve RC-V-115 Failure	T	8.78E-03	8.77E-04	4.38E-03	8.74E-03	1.74E-02	NPRDS		
43	RRCV116	Pressurizer Safety Valve RC-V-116 Failure	T	8.78E-03	8.77E-04	4.38E-03	8.74E-03	1.74E-02	NPRDS		
44	RRCV117	Pressurizer Safety Valve RC-V-117 Failure	T	8.78E-03	8.77E-04	4.38E-03	8.74E-03	1.74E-02	NPRDS		
45	SPURIOUSA1	RPS Train A1 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	SN3 NUREG : Sensor failure	
46	SPURIOUSA2	RPS Train A2 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	SN3 NUREG : Sensor failure	
47	SPURIOUSA3	RPS Train A3 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
48	SPURIOUSA4	RPS Train A4 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
49	SPURIOUSB1	RPS Train B1 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
50	SPURIOUSB2	RPS Train B2 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
51	SPURIOUSB3	RPS Train B3 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
52	SPURIOUSB4	RPS Train B4 Spurious Signal	T	2.63E-02	2.62E-03	1.31E-02	2.59E-02	5.12E-02	NUREG	NUREG : Sensor failure	
53	SRA	SR Channel A Detector Failure	T	6.58E-03	6.58E-04	3.29E-03	6.56E-03	1.31E-02	NPRDS		
54	SRB	SR Channel B Detector Failure	T	6.58E-03	6.58E-04	3.29E-03	6.56E-03	1.31E-02	NPRDS		

Note

(a) T: Time dependant failure mode (I.e., Fail to operate/run/transfer closed)
D : Failure on Demand
M: Maintenance unavailability

(b) IPE : The Individual Plant Examination
NPRDS :The INPO database
EPIX : The INPO database

Table 6.5.4 Datasheet of Basic Events for BOP System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
1	COE21A	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
2	COE21B	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
3	COE21C	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
4	COE22A	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
5	COE22B	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
6	COE22C	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
7	COE23A	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
8	COE23B	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
9	COE23C	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
10	COE24A	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
11	COE24B	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
12	COE24C	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
13	COE25A	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
14	COE25B	Condensate Heater	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE		3.91
15	COHDPA	Heater Drain Pump A FTR	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
16	COHDPB	Heater Drain Pump B FTR	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
17	COHDTK	Heater Drain Tank Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	HD tank : from Hx	3.91
18	CONDENSER	Condenser failure	T	1.18E-01	1.17E-02	5.73E-02	1.11E-01	2.10E-01	IPE	SB1 : Initiating Event	N/A
19	COPP(A)	Condensate Pump P30A Failure	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
20	COPP(B)	Condensate Pump P30B Failure	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
21	COPP(C)	Condensate Pump P30C FTS	D	2.35E-03					IPE	Stand-by start	4.46
22	COSPACK	Steam Packing Exhausting Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Steam exhaust pack : from Hx	3.91
23	CWBFYVA	Butterfly Valve A Failure	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MOV	4.72
24	CWBFYVB	Butterfly Valve B Failure	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MOV	4.72
25	CWPP(A)	CWpp(A)Trip	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
26	CWPP(B)	CWpp(B)Trip	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
27	CWPP(C)	CWpp(C)Trip	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
28	CWSCRMOTA	Traveling Screen A Failure	T	8.06E-04	8.06E-05	4.03E-04	8.06E-04	1.61E-03	IPE	SB 11	5.49
29	CWSCRMOTB	Traveling Screen B Failure	T	8.06E-04	8.06E-05	4.03E-04	8.06E-04	1.61E-03	IPE	SB 11	5.49
30	CWSCRMOTC	Traveling Screen C Failure	T	8.06E-04	8.06E-05	4.03E-04	8.06E-04	1.61E-03	IPE	Screen Mot FTR	5.49
31	CWSTRCT	Intake Structure Failure								not taken into consideration	
32	CWSWHPA	Wash Pump A FTR	D	2.33E-03					IPE	Wash pump FTS on demand	4.46
33	CWSWHPB	Wash Pump B FTR	D	2.33E-03					IPE	Wash pump FTS on demand	4.46
34	CWSWHPC	Wash Pump C FTR	D	2.33E-03					IPE	Wash pump FTS on demand	4.46
35	EARTHEFILA	Earthfilter A Failure	T	2.64E-01	2.60E-02	1.24E-01	2.32E-01	4.10E-01	IPE	Filter	9.91
36	EARTHEFILB	Earth Filter B Failure	T	2.64E-01	2.60E-02	1.24E-01	2.32E-01	4.10E-01	IPE	Filter	9.91
37	EHC14	US-14 Failure	T	6.02E-03	6.02E-04	3.00E-03	6.00E-03	1.20E-02	IPE	Station Service Transformer	1.78
38	EHC1421COM	US-14and21 CCF	D	1.00E-03						Maintenance Error SB4 6 15	
39	EHC21	US-21 Failure	T	6.02E-03	6.02E-04	3.00E-03	6.00E-03	1.20E-02	IPE	Station Service Transformer	1.78
40	EHCACTUATINGSUPPLY	EHC Actuating Supply Line	T	7.61E-05	7.61E-06	3.81E-05	7.61E-05	1.52E-04	IPE	Rapture	11.22

Table 6.5.4 Datasheet of Basic Events for BOP System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
41	EHCCIRCUIT	EHC Electronic Circuitry Failure	T	4.61E-02	4.60E-03	2.28E-02	4.50E-02	8.80E-02	NPRDS	SB16	
42	EHCCOOLERA	EHC Cooler A Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
43	EHCCOOLERB	EHC Cooler B Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
44	EHCHEATERA	EHC Heater A Failure	T	7.45E-02	7.42E-03	3.65E-02	7.18E-02	1.38E-01	NUREG		
45	EHCHEATERB	EHC Heater B Failure	T	7.45E-02	7.42E-03	3.65E-02	7.18E-02	1.38E-01	NUREG		
46	EHCJETSUPPLY	EHC Jet Supply Line	T	7.61E-05	7.61E-06	3.81E-05	7.61E-05	1.52E-04	IPE	Rapture	11.22
47	EHCMECHFILT	Mechanical Filter Failure	T	2.64E-01	2.60E-02	1.24E-01	2.32E-01	4.10E-01	IPE	Filter	9.91
48	EHCNTRA	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
49	EHCNTRB	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
50	EHCNTRC	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
51	EHCNTRD	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
52	EHCNTRE	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
53	EHCNTRF	EHC Nitrogen Accumulator	T	1.10E-03	1.10E-04	5.48E-04	1.10E-03	2.19E-03	NPRDS		
54	EHCPPA	Pump A FTR	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE		5.49
55	EHCPPB	Pump B FTS	D	2.35E-03					IPE	B FTS ON DEMAND	4.46
56	EHCPSWASB3	SW A Failure	D	2.69E-04					IPE	SB3	6.15
57	EHCPSWBSB3	SW B Failure	D	2.69E-04					IPE	SB3	6.15
58	EHCPSWC	SW C Failure	D	2.69E-04					IPE		6.15
59	EHCRECIRCTK	EHC Recalculating Tank Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
60	EHCRES	Reservoir Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
61	EHCTRIPSUPPLY	EHC Trip Supply Line	T	7.61E-05	7.61E-06	3.81E-05	7.61E-05	1.52E-04	IPE	Rapture	11.22
62	FWE26A	FW HTR E26 A Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	MO	3.91
63	FWE26B	FW HTR E26 B Failure	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	MO	3.91
64	FWFCV510	FW FCV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
65	FWFCV520	FW FCV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO SB5	4.72
66	FWFCV530	FW FCV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
67	FWFCV540	FW FCV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
68	FWHUMAN	Operator Error SB10 SB12 SB17	H	1.00E-03						SB10/12/17 Human error	
69	FWLV4210	FW Level CV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
70	FWLV4220	FW Level CV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
71	FWLV4230	FW Level CV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
72	FWLV4240	FW Level CV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
73	FWPUMP(A)	FWPump P32A Failure	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE	PP	5.49
74	FWPUMP(B)	FW Pump P32B Failure	T	2.94E-01	2.90E-02	1.37E-01	2.55E-01	4.45E-01	IPE	PP	5.49
75	FWV28	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
76	FWV29	FW Check Valve	T	9.11E-05	9.11E-06	4.56E-05	9.11E-05	1.82E-04	IPE	CHK VALVE	2.81
77	FWV30	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
78	FWV37	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
79	FWV38	FW Check Valve	T	9.11E-05	9.11E-06	4.56E-05	9.11E-05	1.82E-04	IPE	CHK VALVE	2.81
80	FWV39	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72

Table 6.5.4 Datasheet of Basic Events for BOP System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
81	FWV46	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
82	FWV47	FW Check Valve	T	9.11E-05	9.11E-06	4.56E-05	9.11E-05	1.82E-04	IPE	CHK VALVE	2.81
83	FWV48	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
84	FWV55	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
85	FWV56	FW Check Valve	T	9.11E-05	9.11E-06	4.56E-05	9.11E-05	1.82E-04	IPE	CHK VALVE	2.81
86	FWV57	FW MOV	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
87	GEXCITE	Exciter Failure SB14	T	6.58E-02	6.56E-03	3.24E-02	6.37E-02	1.23E-01	NPRDS	SB14 Exciter	
88	GFANSA	IPB Fan A FTR	T	6.91E-02	6.89E-03	3.40E-02	6.68E-02	1.29E-01	IPE	IPB FAN A	2.47
89	GFANSB	IPB Fan B FTS	D	4.84E-04					IPE	IPB FAN B	4.38
90	GHYDRO	Hydrogen Cooling System Failure	T	6.58E-03	6.58E-04	3.29E-03	6.56E-03	1.31E-02	NPRDS	Gen H2 Cooler	
91	GIPBDAM	IPB Dampers Failure SB13	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	SB13 Damper	4.72
92	GMAIN	Main Generator Failure SB2	T	5.27E-02	5.25E-03	2.60E-02	5.13E-02	1.00E-01	NPRDS	SB2 Main generator	
93	GRAT	Start Up transformer Failure	T	1.37E-02	1.37E-03	6.81E-03	1.36E-02	2.70E-02	IPE	Transformer	2.87
94	GSEAL	Oil Sealing System Failure	T	3.29E-02	3.29E-03	1.63E-02	3.24E-02	6.37E-02	NPRDS	Gen oil seal	
95	GST	Main Transformer Failure	T	1.37E-02	1.37E-03	6.81E-03	1.36E-02	2.70E-02	IPE	Transformer	2.87
96	GSWCSB7	Stator Winding Cooling Failure	T	1.32E-02	1.32E-03	6.56E-03	1.31E-02	2.60E-02	NPRDS	Gen Stator Cooling SB7	
97	GSWY	345 kV Switchyard Failure SB9	T	4.84E-02	4.83E-03	2.39E-02	4.72E-02	9.23E-02	IPE	SB9 LOOP	N/A
98	GUATA	Unit transformer A Failure	T	1.37E-02	1.37E-03	6.81E-03	1.36E-02	2.70E-02	IPE	Transformer	2.87
99	GUATB	Unit transformer B Failure	T	1.37E-02	1.37E-03	6.81E-03	1.36E-02	2.70E-02	IPE	Transformer	2.87
100	GVOLTRG	Voltage Regulator Failure	T	3.29E-02	3.29E-03	1.63E-02	3.24E-02	6.37E-02	NPRDS		
101	MSASDVAFTRC	Atmospheric Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
102	MSASDVBFTRC	Atmospheric Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
103	MSASDVCFTRC	Atmospheric Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
104	MSASDVDFTRC	Atmospheric Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
105	MSIVAFTRO	MSIV FTRO	T	1.75E-02	1.75E-03	8.71E-03	1.73E-02	3.44E-02	IPE	Initiating events Test & Maintenance	N/A
106	MSIVBFTRO	MSIV FTRO	T	1.75E-02	1.75E-03	8.71E-03	1.73E-02	3.44E-02	IPE	Initiating events Test & Maintenance	N/A
107	MSIVCFTRO	MSIV FTRO	T	1.75E-02	1.75E-03	8.71E-03	1.73E-02	3.44E-02	IPE	Initiating events Test & Maintenance	N/A
108	MSIVDFTRO	MSIV FTRO	T	1.75E-02	1.75E-03	8.71E-03	1.73E-02	3.44E-02	IPE	Initiating events Test & Maintenance	N/A
109	MSSDVA1	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
110	MSSDVA2	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
111	MSSDVB1	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
112	MSSDVB2	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
113	MSSDVC1	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
114	MSSDVC2	Condenser Steam Dump Valve FTRC	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
115	MSSG1SRVA	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
116	MSSG1SRVB	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
117	MSSG1SRVC	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
118	MSSG1SRVD	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
119	MSSG1SRVE	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
120	MSSG2SRVA	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A

Table 6.5.4 Datasheet of Basic Events for BOP System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
121	MSSG2SRVB	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
122	MSSG2SRVC	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	6.34E-03	IPE	Initiating events	N/A
123	MSSG2SRVD	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
124	MSSG2SRVE	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
125	MSSG3SRVA	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
126	MSSG3SRVB	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
127	MSSG3SRVC	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
128	MSSG3SRVD	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
129	MSSG3SRVE	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
130	MSSG4SRVA	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
131	MSSG4SRVB	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
132	MSSG4SRVC	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
133	MSSG4SRVD	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
134	MSSG4SRVE	SG SRV Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	Initiating events	N/A
135	SG(A)	SG(A)	T	2.14E-02	2.14E-03	1.06E-02	2.12E-02	4.19E-02	NPRDS		
136	SG(B)	SG(B)	T	2.14E-02	2.14E-03	1.06E-02	2.12E-02	4.19E-02	NPRDS		
137	SG(C)	SG(C)	T	2.14E-02	2.14E-03	1.06E-02	2.12E-02	4.19E-02	NPRDS		
138	SG(D)	SG(D)	T	2.14E-02	2.14E-03	1.06E-02	2.12E-02	4.19E-02	NPRDS		
139	TMSR1	MS/R	T	1.65E-03	1.65E-04	8.23E-04	1.64E-03	3.29E-03	NPRDS	MS/R	
140	TMSR2	MS/R	T	1.65E-03	1.65E-04	8.23E-04	1.64E-03	3.29E-03	NPRDS	MS/R	
141	TMSR3	MS/R	T	1.65E-03	1.65E-04	8.23E-04	1.64E-03	3.29E-03	NPRDS	MS/R	
142	TMSR4	MS/R	T	1.65E-03	1.65E-04	8.23E-04	1.64E-03	3.29E-03	NPRDS	MS/R	
143	TMSRDTA	MS/R Drain Tank	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
144	TMSRDTB	MS/R Drain Tank	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
145	TMSRDTC	MS/R Drain Tank	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
146	TMSRDTD	MS/R Drain Tank	T	1.71E-02	1.71E-03	8.50E-03	1.69E-02	3.36E-02	IPE	Hx	3.91
147	TMSRDV36	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
148	TMSRDV37	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
149	TMSRDV38	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
150	TMSRDV39	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
151	TMSRDV40	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
152	TMSRDV41	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
153	TMSRDV42	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
154	TMSRDV43	MS/R Drain Valve	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
155	TMSRRVA	MS/R Relief Valve Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	MSR Rv: from SG Rv	N/A
156	TMSRRVB	MS/R Relief Valve Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	MSR Rv: from SG Rv	N/A
157	TMSRRVC	MS/R Relief Valve Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	MSR Rv: from SG Rv	N/A
158	TMSRRVD	MS/R Relief Valve Leak	T	4.19E-03	4.19E-04	2.09E-03	4.18E-03	8.34E-03	IPE	MSR Rv: from SG Rv	N/A
159	TMSRSLVA	MS/R Steam Load Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
160	TMSRSLVB	MS/R Steam Load Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72

Table 6.5.4 Datasheet of Basic Events for BOP System at the Seabrook Nuclear Power Station

Event Number	Name of events	Event Description	Type of Failure (a)	Frequency (per year)	Probability 0.1Y	Probability 0.5Y	Probability 1Y	Probability 2Y	Data Source (b)	Comments	Error Factors in the IPE
161	TMSRSLVC	MS/R Steam Load Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
162	TMSRSLVD	MS/R Steam Load Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
163	TMSRSTVA	MS/R Steam Load Bypass Valve FTO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
164	TMSRSTVB	MS/R Steam Load Bypass Valve FTO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
165	TMSRSTVC	MS/R Steam Load Bypass Valve FTO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
166	TMSRSTVD	MS/R Steam Load Bypass Valve FTO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
167	TNSRSLVVA	MS/R Stop Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
168	TNSRSLVVB	MS/R Stop Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
169	TNSRSLVVC	MS/R Stop Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
170	TNSRSLVVD	MS/R Stop Valve FTRO	T	8.12E-04	8.12E-05	4.06E-04	8.12E-04	1.62E-03	IPE	MO	4.72
171	TTESTMAN	Turbine Trip Test & Maint. (Valv. Cntrl.)	T	1.65E-01	1.64E-02	7.92E-02	1.52E-01	2.81E-01	IPE	SB8	N/A

Note

(a) T: Time dependant failure mode (i.e., Fail to operate/run/transfer closed)
D : Failure on Demand
M: Maintenance unavailability
(b) IPE : The Individual Plant Examination
NPRDS :The INPO database
EPIX : The INPO database

7. Risk Importance and Fault Tree Analysis

7.1 Risk Importance Measures

7.1.1 Minimal Cut Set(MCS)

Before evaluating the risk significance, we must know the minimal cut set in fault trees. For a given fault tree, a system failure mode is clearly defined by a *cut set*, which is a collection of basic events; if all basic events occur, the top event is guaranteed to occur. A Minimal Cut Set (MCS) is one in which the removal of any basic event results in the remaining events no longer constituting a cut set. A cut set that includes some other set is not a minimal cut set. The minimal cut set concept enables analysts to reduce the number of cut sets and the number of basic events involved in each cut set.

7.1.2 Risk Importance

In order to rank the events with respect to risk-significance, three *importance measures* are commonly used for ranking elements in PRA, Fussell-Vesely Importance, Risk Reduction Worth and Risk Achievement Worth (Ref.12). Events that occurred in the Seabrook and Pilgrim Nuclear Power Stations are evaluated using these importance measures.

(1) Fussell-Vesely Importance Measure (FV)

The Fussell-Vesely Importance measure (FV) is a measure of the fractional contribution of the model elements to the overall model risk when the element probability is changed from its base value to zero. In other words, it is the sum of the cut sets involving the components divided by the sum of all cut sets. Suppose we have a system which consists of three components, A, B and C, and this system fails when two out of three components fail. The minimal cut sets of this system are, AB, AC and BC. The FV of component C is written as follows:

$$\mathrm{FV(C)} = \frac{AC + BC}{AB + AC + BC}$$

In general form, FV can be written as follows.

$$\mathrm{FV_i} = \frac{\mathrm{R_0} - \mathrm{R_i^-}}{\mathrm{R_0}} = 1 - \frac{\mathrm{R_i^-}}{\mathrm{R_0}}$$

where :

$\mathrm{R_0}$: base(reference) case overall risk

$\mathrm{R_i^+}$: Overall risk with the probability of element(basicevent) i set to 1

$\mathrm{R_i^-}$: Overall risk with the probability of element(basicevent) i set to 0

This is a measure that assesses the contribution of the group in such a way that, any cut set that has a contribution from any one member of the group is included. Since this measure does not involve assessing changes, but is a simple ratio of contributors, this is an appropriate measure of group importance.

(2) Risk Reduction Worth(RRW)

The Risk Reduction Worth (RRW) is the ratio of the base case model risk to the base case model risk with the probability of element i set equal to 0 (the event is impossible or is totally reliable). Supposing we have a same system described in the section on Fussell-Vesely, the RRW(C) can be written as follows:

$$\mathrm{RRW(C)} = \frac{AB + AC + BC}{AB}$$

Using the same notation as Fussell-Vesely measure importance, RRW can be written in general form as follows:

$$\mathrm{RRW_i} = \frac{\mathrm{R_0}}{\mathrm{R_i^-}}$$

This value represents the maximum decrease in risk for an improvement to the element. This measure is particularly useful for identifying an improvement in the reliability of elements which can most reduce risk.

(3) Risk Achievement Worth(RAW)

The Risk Achievement Worth (RAW) (sometimes called Risk Increase Worth) is the ratio of the model risk with the probability of element i set equal to 1 (the event has occurred or the equipment has failed) to the base case model risk. Again, supposing we have a same system described in the section on Fussell-Vesely, the RRW(C) can be written as follows:

$$RAW(C) = \frac{AB + A + B}{AB + AC + BC}$$

Using the same notation as Fussell-Vesely measure importance, RRW can be written in general form as follows:

$$RAW_i = \frac{R_i^+}{R_0}$$

The RAW represents a measure of the worth of the element in achieving the present level of risk and indicates the importance of maintaining the current level of reliability for the elements.

(4) Minimal Cut Set Upper Bound

The minimal cut set upper bound calculation is used in the risk analysis code *SAPHIRE* (Ref.2) to obtain the approximate probability of the top events. The equation for the minimal cut set upper bound is as follows:

$$S = 1 - \prod_{i=1}^{m}(1 - C_i) \qquad (7-6)$$

Where

S = minimal cut set upper bound for system unavailability

C_i = probability of the ith cut set, and

m = number of minimal cut set in the fault tree

The minimal cut set upper bound is always less than or equal to 1 and gives an upper bound to the exact probability of the top events. Rare Event Approximation is another common approach used to calculate the probability for a top event, where the cut set probability is given by Equation (7-7).

$$S = \sum_{i-1}^{m} C_i \qquad (7-7)$$

This is a good approximation when the cut set probabilities are small. In screening analyses, when relatively large screening values are used to bound the component failure probabilities, the rare event approximation can exceed 1.

(5) Minimal Cut Set with different Cut set Sizes

In order to evaluate which kinds of minimal cut sets (MCSs) with different cut set sizes contribute to the system unreliability (i.e., probabilities of top events), we introduce cut set probabilities Pi given by the following equation:

$$\mathrm{Pi} = 1 - \prod_{k=1}^{m}(1 - \mathrm{MCSi})_k$$

where Pi : Total Unavailab ility of top events due to MCS with size i

m : Number of MCS with size i (MCSi)

Suppose that we have a system composed of eleven components A through K, and which has the following MCS.

$$MCS = \{A\},\ \{B\},\ \{C\},\ \{D,E\},\ \{F,G,H\},\ \{I,J,K\}$$

System unreliability P_{sys} can be obtained by the MCS upper bound approximation as follows:

$$\begin{aligned} Psys &= 1 - \prod_{all} (1 - MCS)_k \\ &= 1 - (1 - A)(1 - B)(1 - C)(1 - D * E)(1 - F * G * H)(1 - I * J * K) \end{aligned}$$

Cut set probabilities are given as follows.

$$\begin{aligned} P_1 &= 1 - (1 - A)(1 - B)(1 - C) \\ P_2 &= 1 - (1 - D * E) \\ P_3 &= 1 - (1 - F * G * H)(1 - I * J * K) \end{aligned}$$

By comparing these three values, P_1, P_2 and P_3, we can evaluate which MCSs most contribute to the system unreliability. For example, if P_1 ends up with much a larger value than P_2 and P_3, failures of single components dominates the unreliability of this system. These values are calculated by the risk analysis code SAPHIRE (Ref. 2)

7.2 Fault Tree Analysis at PNPS

7.2.1 The Probabilities of Top Events

The probabilities of top events in both BOP and NSSS systems are relatively high and close to unity at one year operation. According to the result of fault tree analysis, failures of single components dominate system unreliability. The contributions of failures of the BOP and the NSSS to the probabilities of top events, Pr, are summarized in Table 7.2.1.

Table 7.2.1 Summary of the Probabilities of Top Events Occurring due to the System Failures at PNPS

System	Probabilities of top events obtained from fault tree analysis	Probabilities of top events Obtained from historical data
BOP	0.85	0.74
NSSS	0.95	0.80

The probabilities of top events obtained from historical data are evaluated based on the plant performance between 1989 and 1997 (9 years). During this period, the Pilgrim Nuclear Power Station experienced 25 unplanned shutdowns. Failures of the BOP and lightening strikes on power grid resulted in 11 out of 25 unplanned shutdowns. Failures of the NSSS and main steam isolation valves resulted in 13 unplanned shutdowns. Assuming that the average availability of the plant was 0.9 during this period, we can obtain the point estimate values of the failure rates as follows:

Failure rate due to the BOP = 11/9/0.9 = 1.36 (per year)
Failure rate due to the NSSS = 13/9/0.9 = 1.60 (per year)

Assuming the constant failure rates, we can obtain the probabilities of top events by employing the exponential distribution as follows:

$$Pr = 1-Exp(-1.36*1) = 0.74 \quad \text{due to the BOP}$$
$$Pr = 1-Exp(-1.60*1) = 0.80 \quad \text{due to the NSSS}$$

The probabilities obtained from fault tree analysis are greater than the probabilities obtained from historical data because of incomplete fault tree models, inaccurate failure data and other uncertainties. The uncertainties are described in Section 7.3.3 in detail. Table 7.2.2 shows fault tree models at the Pilgrim

Nuclear Power Station in detail. Numbers of cut set sizes differ from systems; however, failures of single components (i.e., cut set size 1) dominate the system unreliability more than 95%.

7.2.2 Risk Significant Components

Fig. 7.2.1 and Fig. 7.2.2 show the risk important components sorted by Fussell-Vesely values. Failures of three feedwater pumps dominate system unreliability in the BOP, and the failure of 345 kV switchyard dominates system unreliability in the NSSS. Table 7.2.3 shows the numbers of important components with respect to other important criteria used in the IPE and the Maintenance Rule. The probabilities of top event are already so high (i.e., close to unity) that RAW can not be greater than 2, which is one of the criteria used in the Maintenance Rule. Table 7.2.4 and Table 7.2.5 show the risk importance measures for all basic events.

Table7.2.2 Summary of Fault Tree Models and Minimal Cut Sets at the Pilgrim Nuclear Power Station

System	Probability of Shutdown Within One Year	Number of Basic Events	Number of Minimal Cut Sets and Probabilities				
			Cutset Size #1	Cutset Size #2	Cutset Size #3	Cutset Size #4	Total
NSSS	9.50E-01	141	107	78	64	0	249
	Cut Sets Probabilities P_i in One Year Operation		9.50E-01	1.15E-02	7.12E-05	-	-
	Pi / Total Probability		99.94%	0.01%	0.00%	-	-
BOP	8.48E-01	162	79	116	18	24	237
	Cut Sets Probabilities P_i in One Year Operation		8.36E-01	5.81E-02	1.72E-02	8.84E-07	-
	Pi / Total Probability		98.57%	0.04%	0.02%	0.00%	-

Table7.2.3 The Number of Risk Important Components at the Pilgrim Nuclear Power Station

System	Number of Minimal Cut Sets	Number of Risk Significant Components			
		Probability > 90% (1)	F.V. > 0.5%	RRW > 1.005	RAW > 2
NSSS	249	4	12	15	0
BOP	237	6	20	20	0

Note : (1) basic events which appear in cut sets making up the top 90 % of initiating the top event

Fig. 7.2.1 Risk Significant Components/Basic Event Frequencies in NSSS System at the Pilgrim Nuclear Power Station

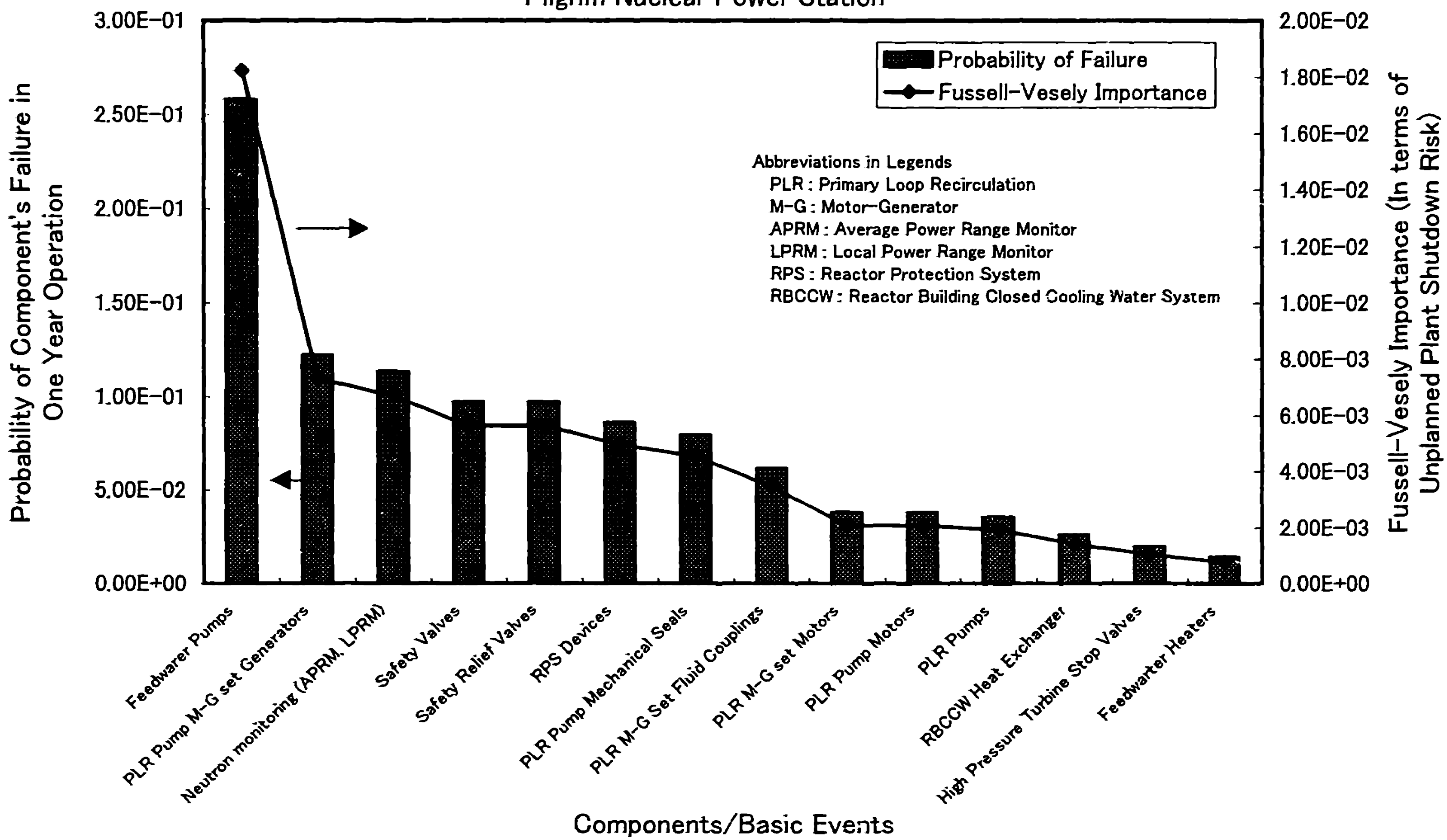


Fig. 7.2.2 Risk Significant Components/Basic Event Frequencies in BOP System at the Pilgrim Nuclear Power Station

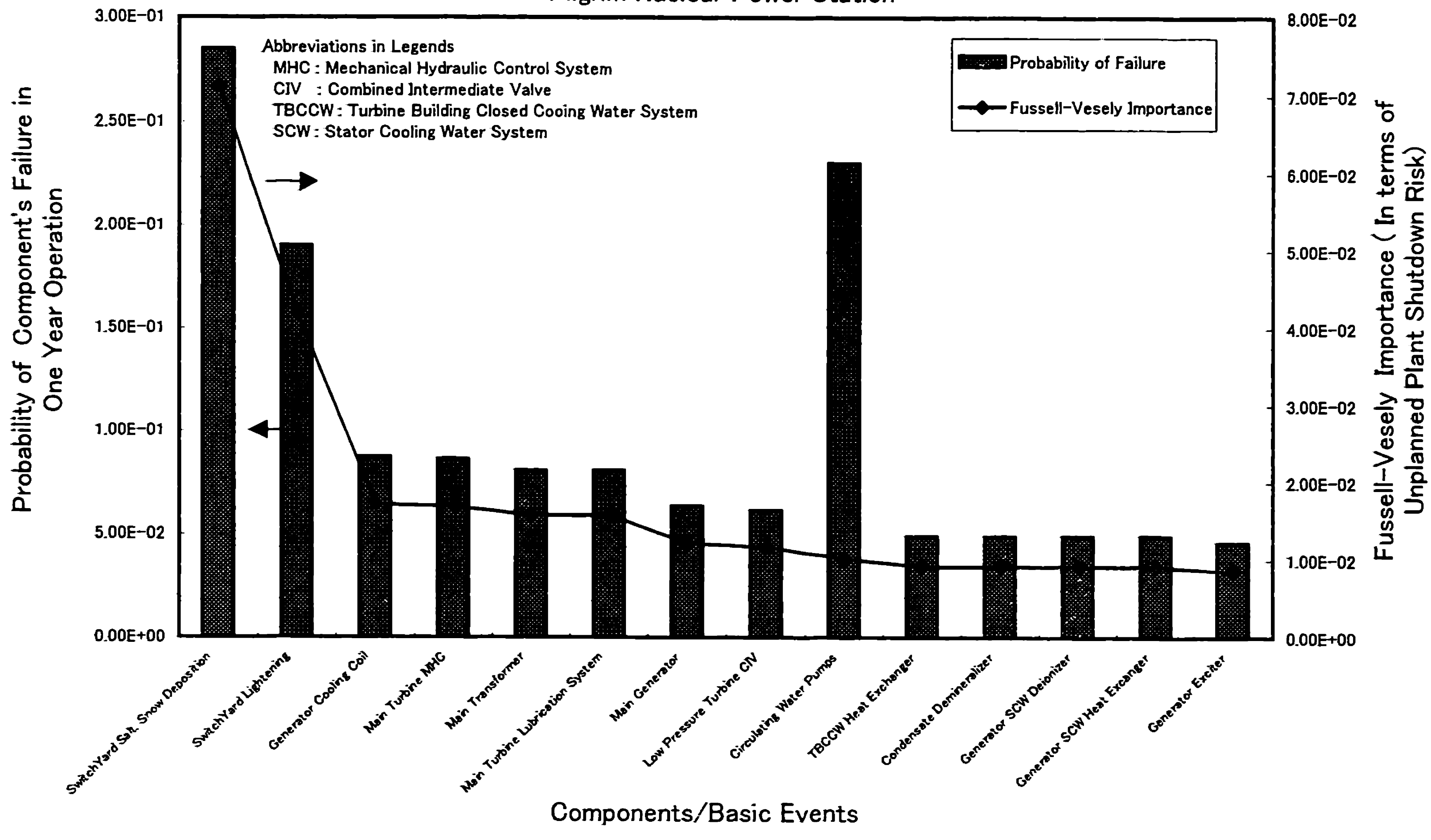


Table 7.2.4 Risk Significant Components/Basic events in NSSS System at the Pilgrim Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RAW
FWPBP	FWP B pump FTR	2,58E-01	1.82E-02	1,02E+00	1,05E+00
FWPCP	FWP C pump FTR	2,58E-01	1.82E-02	1,02E+00	1,05E+00
FWPAP	FWP A pump FTR	2,58E-01	1.82E-02	1,02E+00	1,05E+00
PN3	MGset generator failure	1,22E-01	7.29E-03	1,01E+00	1,05E+00
MGGENERATOR(B)	MGset generator failure	1,22E-01	7.29E-03	1,01E+00	1,05E+00
NEUTRON	Neutron monitoring (APRM, LPRM...)	1,13E-01	6,68E-03	1,01E+00	1,05E+00
SVB	SV B Leakage	9,70E-02	5,64E-03	1,01E+00	1,05E+00
SVA	SV A Leakage	9,70E-02	5,64E-03	1,01E+00	1,05E+00
SRVC	SRV C Leak	9,70E-02	5,64E-03	1,01E+00	1,05E+00
SRVA	SRV A Leak	9,70E-02	5,64E-03	1,01E+00	1,05E+00
SRVD	SRV D Leak	9,70E-02	5,64E-03	1,01E+00	1,05E+00
SRVB	SRV B Leak	9,70E-02	5,64E-03	1,01E+00	1,05E+00
RPS_DEVICE	RPS Device Failure	8,61E-02	4,94E-03	1,01E+00	1,05E+00
PN5	SealLeakage (B)	7,94E-02	4,52E-03	1,01E+00	1,05E+00
PLRSEAL(A)	SealLeakage (A)	7,94E-02	4,52E-03	1,01E+00	1,05E+00
FLUIDCOUPLING(A)	FluidCoupling(A) failure	6,14E-02	3,43E-03	1,00E+00	1,05E+00
FLUIDCOUPLING(B)	FluidCoupling(B) failure	6,14E-02	3,43E-03	1,00E+00	1,05E+00
MGMOTOR(B)	MGset Motor Failure	3,79E-02	2,07E-03	1,00E+00	1,05E+00
MGMOTOR(A)	MGset Motor Failure	3,79E-02	2,07E-03	1,00E+00	1,05E+00
MOTOR(B)FAIL	Motor(B) fail to run	3,79E-02	2,07E-03	1,00E+00	1,05E+00
MOTOR(A)FAIL	Motor(A) fail to run	3,79E-02	2,07E-03	1,00E+00	1,05E+00
PLRPP(B)FAIL	Pump B failure	3,53E-02	1,92E-03	1,00E+00	1,05E+00
PLRPP(A)FAIL	Pump A failure	3,53E-02	1,92E-03	1,00E+00	1,05E+00
BHX209BXXU	HxB maintenance	2,66E-02	1,43E-03	1,00E+00	1,05E+00
BHX209AXXU	HxA maintenance	2,66E-02	1,43E-03	1,00E+00	1,05E+00
HXBLEAKAGE	HxB Leakage	2,60E-02	1,40E-03	1,00E+00	1,05E+00
PN8	RBCCW Hx(A) Leakage	2,60E-02	1,40E-03	1,00E+00	1,05E+00
STV1	STV1 FTRO	1,95E-02	1,04E-03	1,00E+00	1,05E+00
STV4	STV4 FTRO	1,95E-02	1,04E-03	1,00E+00	1,05E+00
STV3	STV3 FTRO	1,95E-02	1,04E-03	1,00E+00	1,05E+00
STV2	STV2 FTRO	1,95E-02	1,04E-03	1,00E+00	1,05E+00
HTR2A	HTR 2A failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR3A	HTR 3A failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR4A	HTR 4A failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR1A	HTR 1A failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR4B	HTR 4B failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR2B	HTR 2B failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR3B	HTR 3B failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR5A	HTR 5A failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR5B	HTR 5B failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
HTR1B	HTR 1B failure	1,41E-02	7,50E-04	1,00E+00	1,05E+00
CV2	CV-2 FTRO	1,04E-02	5,51E-04	1,00E+00	1,05E+00
CV1	CV-1 FTRO	1,04E-02	5,51E-04	1,00E+00	1,05E+00
CV3	CV-3 FTRO	1,04E-02	5,51E-04	1,00E+00	1,05E+00
CV4	CV-4 FTRO	1,04E-02	5,51E-04	1,00E+00	1,05E+00
PLROIL(B)	Lub oil system failure	8,24E-03	4,36E-04	1,00E+00	1,05E+00
PLROIL(A)	Lub oil system failure	8,24E-03	4,36E-04	1,00E+00	1,05E+00
PN6	Miscellaneous leakage (RPV Drainline Leakage)	5,48E-03	2,89E-04	1,00E+00	1,05E+00
FCV643OPER	Operator FTO 643	5,00E-03	2,64E-04	1,00E+00	1,05E+00
BPV3	BPV3 FTRO	4,49E-03	2,37E-04	1,00E+00	1,05E+00
BPV1	BPV1 FTRO	4,49E-03	2,37E-04	1,00E+00	1,05E+00
BPV2	BPV2 FTRO	4,49E-03	2,37E-04	1,00E+00	1,05E+00
FV3437	Rec valve 3437 FTRC	4,38E-03	2,31E-04	1,00E+00	1,05E+00
FV3436	Rec valve 3436 FTRC	4,38E-03	2,31E-04	1,00E+00	1,05E+00
FV3435	Rec valve 3435 FTRC	4,38E-03	2,31E-04	1,00E+00	1,05E+00
HTRBPV	HTR BPV FTRC	4,38E-03	2,31E-04	1,00E+00	1,05E+00
MO3428	MO3428 FTRO	4,38E-03	2,31E-04	1,00E+00	1,05E+00
MO3472	MO3472 FTRO	4,38E-03	2,31E-04	1,00E+00	1,05E+00
BPMDXXXXXR	202D FTR	3,62E-02	1,82E-04	1,00E+00	1,01E+00
BPMAXXXXXR	202A FTR	3,62E-02	1,82E-04	1,00E+00	1,01E+00
TSSWB	No SSW Loop B	3,35E-03	1,76E-04	1,00E+00	1,05E+00
TSSWA	No SSW Loop A	3,35E-03	1,76E-04	1,00E+00	1,05E+00
BPMCXXXXXR	202C FTR	3,62E-02	1,61E-04	1,00E+00	1,00E+00
BPMFXXXXXR	202F FTR	3,62E-02	1,61E-04	1,00E+00	1,00E+00
BPMEXXXXXR	202E FTR	3,62E-02	1,61E-04	1,00E+00	1,00E+00
BPMBXXXXXR	202B FTR	3,62E-02	1,61E-04	1,00E+00	1,00E+00
TY4	Loss of 120 V AC Panel Y4	3,00E-03	1,58E-04	1,00E+00	1,05E+00
TB14	Loss of Power from B14	3,00E-03	1,58E-04	1,00E+00	1,05E+00
TY3	Loss of 120 V AC Panel Y3	3,00E-03	1,58E-04	1,00E+00	1,05E+00
TB15	Loss of Power from B15	3,00E-03	1,58E-04	1,00E+00	1,05E+00

Table 7.2.4 Risk Significant Components/Basic events in NSSS System at the Pilgrim Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RAW
MO5B	Disch MO5B FTRO	2.63E-03	1.38E-04	1.00E+00	1.05E+00
MO4B	Inlet MO 4B FTRO	2.63E-03	1.38E-04	1.00E+00	1.05E+00
MO4A	Inlet MO 4A FTRO	2.63E-03	1.38E-04	1.00E+00	1.05E+00
MO5A	Disch MO5A FTRO	2.63E-03	1.38E-04	1.00E+00	1.05E+00
FCV643FTO	FCV643 FTO	2.00E-03	1.05E-04	1.00E+00	1.05E+00
BVM3801XXC	MO-3801 FTC	1.65E-03	8.67E-05	1.00E+00	1.05E+00
BVM3805XXC	MO-3805 FTC	1.65E-03	8.67E-05	1.00E+00	1.05E+00
RPVINST	Instrumentation	1.10E-03	5.78E-05	1.00E+00	1.05E+00
PN7_PN2	Human Error	1.00E-03	5.25E-05	1.00E+00	1.05E+00
BPMFXXXXXU	202F corrective Maintenance	1.09E-02	4.83E-05	1.00E+00	1.00E+00
BPMCXXXXXU	202C corrective Maintenance	1.09E-02	4.83E-05	1.00E+00	1.00E+00
BPMEXXXXXU	202E corrective Maintenance	1.09E-02	4.83E-05	1.00E+00	1.00E+00
BPMBXXXXXU	202B corrective Maintenance	1.09E-02	4.82E-05	1.00E+00	1.00E+00
BVH3837XXF	HO-3837 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVH5XXXXXF	HO-5 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVH6XXXXXF	HO-6 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVM3806XXF	MO-3806 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVH2XXXXXF	HO-2 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVA1	MSIV A1 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVB1	MSIV B1 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVH3842XXF	HO-3842 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVB2	MSIV B2 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVM3800XXF	MO-3800 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVC1	MSIV C1 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVD2	MSIV D2 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVD1	MSIV D1 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVC2	MSIV C2 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
BVH1XXXXXF	HO-1 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
MSIVA2	MSIV A2 FTRO	8.76E-04	4.60E-05	1.00E+00	1.05E+00
PN1	FCV642B failure	8.32E-04	4.37E-05	1.00E+00	1.05E+00
PN4_9_10	FCV642A failure	8.32E-04	4.37E-05	1.00E+00	1.05E+00
DRNCOOLERA	Drain cooler A failure	6.00E-04	3.15E-05	1.00E+00	1.05E+00
DRNCOOLERB	Drain cooler B failure	6.00E-04	3.15E-05	1.00E+00	1.05E+00
BPMABXXCCR	A,B,C pumps CCF to Run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPMEFXXCCR	Pump E,F CCF to run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPMDEXXCCR	Pump D,E CCF to run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPMDFXXCCR	Pump D,F CCF to run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPMACXXCCR	Pump A,C CCF to run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPMBCXXCCR	Pump B, C CCF to run	2.54E-04	1.33E-05	1.00E+00	1.05E+00
BPM5XXXCCR	CCF of 5 or more pumps failure to run	1.93E-04	1.01E-05	1.00E+00	1.05E+00
EXPJOINTHTR	Expansion joint failure	1.81E-04	9.50E-06	1.00E+00	1.05E+00
1TEP	Initiating events partial loss of offsite power	3.78E-01	3.88E-06	1.00E+00	1.00E+00
BPMEXXXXXS	202E FTS	6.30E-04	2.79E-06	1.00E+00	1.00E+00
BPMFXXXXXS	202F FTS	6.30E-04	2.79E-06	1.00E+00	1.00E+00
BPMBXXXXXS	202B FTS	6.30E-04	2.79E-06	1.00E+00	1.00E+00
BPM5XXXCCS	CCF 5 or more FTS	4.80E-05	2.52E-06	1.00E+00	1.05E+00
BPMCXXXXXS	202C FTS	5.30E-04	2.35E-06	1.00E+00	1.00E+00
CCF_FCV642	CCF FCV642AB	4.38E-05	2.30E-06	1.00E+00	1.05E+00
BPMDXXXXXS	202D FTS	6.30E-04	1.61E-06	1.00E+00	1.00E+00
BPMAXXXXXS	202A FTS	6.30E-04	1.61E-06	1.00E+00	1.00E+00
1TEF	Initiating events Full loss of offsite power	1.32E-01	1.35E-06	1.00E+00	1.00E+00
BPMEFXXCCS	E,F CCF FTS	1.40E-05	7.34E-07	1.00E+00	1.05E+00
BPMBCXXCCS	B,C CCF FTS	1.40E-05	7.34E-07	1.00E+00	1.05E+00
BSP4008XXE	PS4008 FT energize	1.00E-05	5.25E-07	1.00E+00	1.05E+00
BSP4058XXE	PS4058 FT energize	1.00E-05	5.25E-07	1.00E+00	1.05E+00
BVC424XXXN	CheckValve 30ck424 FTO	1.00E-04	4.43E-07	1.00E+00	1.00E+00
BVC419XXXN	CheckValve 30ck419 FTO	1.00E-04	4.43E-07	1.00E+00	1.00E+00
BVC423XXXN	CheckValve 30ck423 FTO	1.00E-04	4.43E-07	1.00E+00	1.00E+00
BVC420XXXN	CheckValve 30ck420 FTO	1.00E-04	4.42E-07	1.00E+00	1.00E+00
BPMACXXCCS	A,C CCF FTS	1.40E-05	4.36E-07	1.00E+00	1.03E+00
BPMDEXXCCS	D,E CCF FTS	1.40E-05	4.36E-07	1.00E+00	1.03E+00
BPMDFXXCCS	D,F CCF FTS	1.40E-05	4.36E-07	1.00E+00	1.03E+00
BPMABXXCCS	A,B,C CCF FTS	1.40E-05	4.36E-07	1.00E+00	1.03E+00
BVC422XXXN	Check valve 30ck422 FTO	1.00E-04	2.56E-07	1.00E+00	1.00E+00
BVC421XXXN	Check valve 30ck421 FTO	1.00E-04	2.56E-07	1.00E+00	1.00E+00

Note (a) All risk importance measure including the probability of basic events are evaluated based on one year operation.
Components are sorted by Fussell-Vesely.

Table 7.2.5 Risk Significant Components/Basic Events in BOP System at the Pilgrim Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RIW
PB4_PB5	SALT SNOW Deposit	2.85E-01	7.13E-02	1.08E+00	1.18E+00
SWTHUNDER	Switchyard fails due to thunder strike	1.90E-01	4.19E-02	1.04E+00	1.18E+00
PB7	Cooling coil failure	8.72E-02	1.71E-02	1.02E+00	1.18E+00
TBMHC	MHC failure	8.61E-02	1.68E-02	1.02E+00	1.18E+00
MAINTRANS	Main Transformer failure	8.06E-02	1.57E-02	1.02E+00	1.18E+00
TBLUB	Lubrication system failure	8.06E-02	1.57E-02	1.02E+00	1.18E+00
GENSTATOR	Main Generator/stator/ voltage regulator failure	6.33E-02	1.21E-02	1.01E+00	1.18E+00
CIV3	CIV #3	6.11E-02	1.16E-02	1.01E+00	1.18E+00
CIV4	CIV #4	6.11E-02	1.16E-02	1.01E+00	1.18E+00
CIV1	CIV #1	6.11E-02	1.16E-02	1.01E+00	1.18E+00
CIV2	CIV #2	6.11E-02	1.16E-02	1.01E+00	1.18E+00
P105BFTR	CWP P105B fail to run	2.30E-01	1.01E-02	1.01E+00	1.03E+00
P105AFTR	CWP P105A fail to run	2.30E-01	1.01E-02	1.01E+00	1.03E+00
WHX122AXXF	TBCCW Hx 122A Plugs	4.87E-02	9.15E-03	1.01E+00	1.18E+00
WHX122BXXF	TBCCW Hx E122B Plugs	4.87E-02	9.15E-03	1.01E+00	1.18E+00
BANKS_PULGGED	Demineralizer resin banks plugged	4.87E-02	9.15E-03	1.01E+00	1.18E+00
DEIONIZER	Deionizer failure (Plugged)	4.87E-02	9.15E-03	1.01E+00	1.18E+00
GENHX_FAILURE	Hx failure (Plugged)	4.87E-02	9.15E-03	1.01E+00	1.18E+00
GENEXICITER	Generator Exciter failure	4.56E-02	8.54E-03	1.01E+00	1.18E+00
WTE4161XXR	Temperature Element TE 4161 Fails High	3.58E-02	6.64E-03	1.01E+00	1.18E+00
STARTRANS	Startup Transformer failure	2.31E-02	4.23E-03	1.00E+00	1.18E+00
TS3717A	Spurious Actuation TS3717A	1.92E-02	3.50E-03	1.00E+00	1.18E+00
TS3718A	Spurious Actuation TS3718A	1.92E-02	3.50E-03	1.00E+00	1.18E+00
TS3717B	Spurious Actuation TS3717B	1.92E-02	3.50E-03	1.00E+00	1.18E+00
TS3718B	Spurious Actuation TS3718B	1.92E-02	3.50E-03	1.00E+00	1.18E+00
TBCONT	Control system failure	1.90E-02	3.46E-03	1.00E+00	1.18E+00
TSSS	Steam Seal Supply System	1.78E-02	3.24E-03	1.00E+00	1.18E+00
FCPC	Condensate pump C FTR	2.58E-01	3.12E-03	1.00E+00	1.01E+00
FCPB	Condensate pump B FTR	2.58E-01	3.12E-03	1.00E+00	1.01E+00
FCPA	Condensate pump A FTR	2.58E-01	3.12E-03	1.00E+00	1.01E+00
PB3_PB6	Unit Auxiliary Trans failure	1.39E-02	2.52E-03	1.00E+00	1.18E+00
FINITHWMUC	Hotwell makeup capacity loss	1.17E-02	2.12E-03	1.00E+00	1.18E+00
GENHYDOGEN	Generator hydrogen cooler failure	9.29E-03	1.68E-03	1.00E+00	1.18E+00
HYDROGENOILSE	Hydrogen Oil seal failure	9.29E-03	1.68E-03	1.00E+00	1.18E+00
WVM380110F	MO 3801 FTR 10% Open	8.72E-03	1.57E-03	1.00E+00	1.18E+00
WVM380510F	MO-3805 FTR 10% OPEN	8.72E-03	1.57E-03	1.00E+00	1.18E+00
OPEFAIL	OperatorFail select AOG low flow bypass	5.00E-03	8.98E-04	1.00E+00	1.18E+00
TVA3351XXL	Recirculation Valve FV3351 FTRC	4.37E-03	7.85E-04	1.00E+00	1.18E+00
TSSWA	No SSW Loop A	3.36E-03	6.03E-04	1.00E+00	1.18E+00
TSSWB	NO SSW LOOP B	3.36E-03	6.03E-04	1.00E+00	1.18E+00
Y1	No Power Available from Y1	3.00E-03	5.38E-04	1.00E+00	1.18E+00
PB9	HP/LP turbine failure	3.00E-03	5.38E-04	1.00E+00	1.18E+00
B31	No power Available from B31	3.00E-03	5.38E-04	1.00E+00	1.18E+00
B30	No-power available from B30	3.00E-03	5.38E-04	1.00E+00	1.18E+00
Y2	No power Available from Y2	3.00E-03	5.38E-04	1.00E+00	1.18E+00
PB1	Bearing Failure	2.33E-03	4.18E-04	1.00E+00	1.18E+00
WPM110AXXR	110A FTR	3.62E-02	3.77E-04	1.00E+00	1.01E+00
WVT4161XXC	Temperature Valve TV 4161 FTC	2.00E-03	3.58E-04	1.00E+00	1.18E+00
WPM110BXXR	110B FTR	3.62E-02	3.58E-04	1.00E+00	1.01E+00
TBMS	Moisture separator	1.50E-03	2.69E-04	1.00E+00	1.18E+00
FLPLPHTRAU	LP HTR TRAIN A out for maintenance.	1.44E-03	2.58E-04	1.00E+00	1.18E+00
FLPHTRBU	LP HTR Train B unavailable	1.44E-03	2.58E-04	1.00E+00	1.18E+00
WPM110XCCR	110A, B CC FTR	1.40E-03	2.51E-04	1.00E+00	1.18E+00
WTI62ABCCR	62A, 62B CC FTOOP	1.03E-03	1.84E-04	1.00E+00	1.18E+00
PB2	Human Error SJAE	1.00E-03	1.79E-04	1.00E+00	1.18E+00
PB8	TCV Y07 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
9222FTRO	9222FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
9259FTRO	9259FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
9259FTRE	Solenoid valve 9259 FTR energized	8.76E-04	1.57E-04	1.00E+00	1.18E+00
WVH667XXXF	HO-667 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
WVM3801XXF	MO3801 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
WVH665XXXF	HO-665 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
WVH670XXXF	HO-670 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
FVM3372XXF	MO3372 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
FVHH0111XF	MakeupValve 26-HO-111 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
FVM3428XXF	MO3428 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00

Table 7.2.5 Risk Significant Components/Basic Events in BOP System at the Pilgrim Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RIW
FVM3371XXF	MO3371 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
PCV3701	PCV3701FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
FVM3427XXF	MO3427 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
PCV_Y_63	PCV Y63 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
FVHH0110XF	MakeupValve 26-HO-110 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
AO3751FTRO	AO3751FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
9222FTRE	Solenoid valve 9222 FTR energized	8.76E-04	1.57E-04	1.00E+00	1.18E+00
WVH668XXXF	HO-668 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
MO9271FTRO	Coolant Discrg valve MO9271 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
F3301FTRO	MakeupValve AO-3301 FTRO	8.76E-04	1.57E-04	1.00E+00	1.18E+00
A3	No power available from A3	3.00E-03	1.25E-04	1.00E+00	1.04E+00
A4	No power available from A4	3.00E-03	1.25E-04	1.00E+00	1.04E+00
P136B_FTS	P136B FTS	3.00E-03	1.23E-04	1.00E+00	1.04E+00
P136A_FTR	P136A FTR	2.30E-01	1.23E-04	1.00E+00	1.00E+00
WTI62AXXXR	Timer 62A fails to operate	9.27E-03	9.65E-05	1.00E+00	1.01E+00
WTI62BXXXR	Timer 62B Fails to operate	9.27E-03	9.16E-05	1.00E+00	1.01E+00
WSP4176CCR	PS4176A,B CC Fail to pick up	5.04E-04	9.01E-05	1.00E+00	1.18E+00
AIRLOSS	Loss of essential air	2.00E-04	3.58E-05	1.00E+00	1.18E+00
EAST-JOINT	East Joint	1.81E-04	3.24E-05	1.00E+00	1.18E+00
WEST-JOINT	West Joint	1.81E-04	3.24E-05	1.00E+00	1.18E+00
D4	loss of 125 DC From D-4	3.00E-03	3.12E-05	1.00E+00	1.01E+00
B1	No power available from B1	3.00E-03	3.12E-05	1.00E+00	1.01E+00
Y3	Loss of 12V AC from Y-3	3.00E-03	3.12E-05	1.00E+00	1.01E+00
WPM110BXXU	TBCCW Pump B train out for Maintenance	3.00E-03	2.96E-05	1.00E+00	1.01E+00
Y4	Loss of 120 V AC From Y4	3.00E-03	2.96E-05	1.00E+00	1.01E+00
D5	Loss of 125 V DC from D5	3.00E-03	2.96E-05	1.00E+00	1.01E+00
B2	No power from B2	3.00E-03	2.96E-05	1.00E+00	1.01E+00
AOG_B	Operator fails to align AOG train B	1.50E-02	2.82E-05	1.00E+00	1.00E+00
WVC961XCCN		1.00E-04	1.79E-05	1.00E+00	1.18E+00
WPM110XCCS	110A, B CC FTS	9.90E-05	1.77E-05	1.00E+00	1.18E+00
WPM110AMXS	110A Mechanical FTS	6.50E-04	6.77E-06	1.00E+00	1.01E+00
WPM110BMXS	110B Mechanical FTS	6.50E-04	6.42E-06	1.00E+00	1.01E+00
BCORRETMAIN	Train B out for Corrective maintenance	3.00E-03	5.64E-06	1.00E+00	1.00E+00
WHX122AXXU	Hx E122A out for Maintenance	2.80E-05	5.01E-06	1.00E+00	1.18E+00
WHX122BXXU	Hx E122B out for maintenance	2.80E-05	5.01E-06	1.00E+00	1.18E+00
CCFP105AB	CCF P105A/B to run	2.26E-05	4.04E-06	1.00E+00	1.18E+00
AO3708	AO3708FTRC	4.37E-03	3.41E-06	1.00E+00	1.00E+00
AO3707	AO3707FTRC	4.37E-03	3.41E-06	1.00E+00	1.00E+00
MOV_D_1	Damper MOV D 1 FTRO	8.76E-04	3.08E-06	1.00E+00	1.00E+00
9269FTRE	Solenoid valve 9269 FTR energized	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9255FTRE	Solenoid valve 9255 FTR energized	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9251FTRO	9251FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9251FTRE	Solenoid valve 9251 FTR energized	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9252FTRO	9252FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9269FTRO	9269FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
AO9238	AO9238FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
AO9239	AO9239FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9252FTRE	Solenoid valve 9252 FTR energized	8.76E-04	2.82E-06	1.00E+00	1.00E+00
9255FTRO	9255FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
MO9206	MO9206FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
MO9207	MO9207FTRO	8.76E-04	2.82E-06	1.00E+00	1.00E+00
OPEEXTAB	Operator fails to switch to exhaust Train B	1.50E-02	2.35E-06	1.00E+00	1.00E+00
MOVS_2FTRO	MOVS 2FTRO	8.76E-04	1.51E-06	1.00E+00	1.00E+00
WVC962XXXN	CK-962 FTO	1.00E-04	1.04E-06	1.00E+00	1.01E+00
WVC961XXXN	CK-961 FTO	1.00E-04	9.87E-07	1.00E+00	1.01E+00
OPEMOV_S_2	Operator fails to open Bypass valve MOV-s-2	5.00E-03	9.32E-07	1.00E+00	1.00E+00
B23	No power Available from B23	3.00E-03	5.59E-07	1.00E+00	1.00E+00
B22	Transfer- No power available from B22	3.00E-03	4.70E-07	1.00E+00	1.00E+00
WVM3805XXF	MO-3805 FTRO	2.40E-06	4.29E-07	1.00E+00	1.18E+00
MOVS2	MOVS2FTO	1.65E-03	3.08E-07	1.00E+00	1.00E+00
MOV_B	Unloaded valve MOV B fail to operate	1.65E-03	2.88E-07	1.00E+00	1.00E+00
3704FTRE	3704FTR energized	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3704FTRO	3704FTRO	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3711FTRE	Solenoid valve 3711 FTR energized	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3711FTRO	3711FTRO	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3710FTRE	3710FTR energized	8.76E-04	2.74E-07	1.00E+00	1.00E+00

Table 7.2.5 Risk Significant Components/Basic Events in BOP System at the Pilgrim Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RIW
3703FTRE	3703FTR energized	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3710FTRO	3710FTRO	8.76E-04	2.74E-07	1.00E+00	1.00E+00
3703FTRO	3703FTRO	8.76E-04	2.74E-07	1.00E+00	1.00E+00
MOV_D_2	Damper MOV D 2 FTO	1.65E-03	2.58E-07	1.00E+00	1.00E+00
P130AFTR	P130AFTR	2.30E-01	1.54E-07	1.00E+00	1.00E+00
P130BFTR	P130BFTR	2.30E-01	1.54E-07	1.00E+00	1.00E+00
WSP4176AXR	PS4176A Fails to Pick up	1.00E-05	1.04E-07	1.00E+00	1.01E+00
X102	Regulator X102 fail to operate	3.58E-02	1.02E-07	1.00E+00	1.00E+00
WSP4176BXR	PS4176B Fails to Pick up	1.00E-05	9.87E-08	1.00E+00	1.01E+00
CCFFTR130-A_B	CCF FTR Pump 130-A B	9.59E-03	2.73E-08	1.00E+00	1.00E+00
9213FTRO	9213FTRO	8.76E-04	4.94E-09	1.00E+00	1.00E+00
9213FTRE	Solenoid valve 9213 FTR energized	8.76E-04	4.94E-09	1.00E+00	1.00E+00
9212FTRO	9212FTRO	8.76E-04	4.94E-09	1.00E+00	1.00E+00
9212FTRE	Solenoid valve 9212 FTR energized	8.76E-04	4.94E-09	1.00E+00	1.00E+00
P130AFTS	P130AFTS	2.60E-03	1.74E-09	1.00E+00	1.00E+00
P130BFTS	P130BFTS	2.60E-03	1.74E-09	1.00E+00	1.00E+00
CCFFTS130-A_B	CCF FTS Pump 130-A B	4.00E-04	1.14E-09	1.00E+00	1.00E+00

Note (a) All risk importance measure including the probability of basic events are evaluated based on one year operation.
Components are sorted by Fussell-Vesely.

7.3 Fault Tree Analysis at SNPS

7.3.1 The Probabilities of Top Events

The probability of top event (i.e., unreliability of system) in the NSSS is relatively high and close to unity at one year operation. According to the results of fault tree analysis, failures of single components in both BOP and NSSS systems dominate system unreliability. The contributions of the failures of the BOP and the NSSS to the probabilities of top events, Pr, are summarized in Table 7.3.1.

Table 7.3.1 Summary of the Probability of Top Events due to the System Failures at SNPS

System	Probabilities of top events Obtained from fault tree analysis	Probabilities of top events Obtained from historical data
BOP	0.86	0.90
NSSS	0.35	0.46

The probabilities of top events obtained from historical data are estimated based on plant performance between 1989 and 1997 (9 years). During this period, the Seabrook Nuclear Power Station experienced 24 unplanned shutdowns. The failures of the BOP system and main steam isolation valves, which are installed in secondary side, resulted in 19 out of 24 unplanned shutdowns. and the NSSS resulted in 5 unplanned shutdowns. Assuming that the average availability of the plant was 0.9 during this period, we can obtain the point estimate values of the failure rates of the plant shutdowns as follows:

Failure rate due to the BOP = 19/9/0.9 = 2.35 (per year)
Failure rate due to the NSSS = 5/9/0.9 = 0.62 (per year)

Assuming the constant rate, we can obtain the probabilities of the top events by employing the exponential distribution as follows:

$$Pr = 1-Exp(-2.35*1) = 0.90 \quad \text{due to the BOP}$$
$$Pr = 1-Exp(-0.62*1) = 0.46 \quad \text{due to the NSSS}$$

These historical data show good agreement with analyzed data.

Table 7.3.2 shows the comparison of the probability of top events due to the system failures at the Pilgrim and Seabrook Nuclear Power Stations.

Table 7.3.2 Comparison of the Probability of Top Events due to the System Failures at SNPS and PNPS

System	The pilgrim Nuclear Power Station		The Seabrook Nuclear Power Station	
	Fault Tree Analysis	Historical Data	Fault Tree Analysis	Historical Data
BOP	0.85	0.74	0.86	0.90
NSSS	0.95	0.80	0.35	0.46

In our work, the point-estimated probabilities of top events obtained from unplanned shutdown records were less than the probabilities of top events obtained from fault tree analyses for the Pilgrim Nuclear Power Station. On the other hand, for the Seabrook Nuclear Power Station, the point-estimated probabilities of top events obtained from unplanned shutdown records were greater than the mean probabilities of top events obtained from fault tree analyses.

Since both data contained some uncertainties, it is difficult to explain why the data obtained from the fault tree analyses became greater or smaller than the data obtained from the historical records. Also, mean values of the shutdown probabilities obtained from uncertainty analyses and point-estimated probabilities of unplanned shutdowns obtained from historical records were close to each other; therefore, the tendencies concerning which value became greater or smaller were very subtle.

In addition, common cause factors, human errors and maintenance unavailability were considered as basic events in fault tree models only at the PNPS; therefore, the practices used in fault tree analyses at the PNPS were more likely to yield larger results than were those at the SNPS.

The following are factors that may affect to the both of analytical data and historical data.

- Data accuracy and uncertainty
- Accuracy of fault tree models

Table 7.3.3 show fault tree models at the Seabrook Nuclear Power Station in detail. Failures of single components dominate the system unreliability in the Seabrook Power Station as well as the Pilgrim Power Station.

7.3.2 Risk Significant Components

Fig. 7.3.1 and Fig. 7.3.2 show the risk important components sorted by the Fussell-Vesely values. Failures of two feedwater pumps dominate system unreliability in the BOP, and the failure of chemical and volume control system and main steam line valves dominate system unreliability in the NSSS. The failure rates of these valves are so high that they are most likely to initiate the top event. Table 7.3.4 shows the numbers of important components in respect to the other important criteria used in the IPE and the Maintenance Rule. The probability of top event in the BOP system is already so high (i.e., close to unity) that RAW cannot be greater than 2. Table 7.3.5 and Table 7.3.6 show the risk importance measures for all basic events.

7.3.3 Uncertainty Evaluation

(1) Uncertainties in failure rates

We have evaluated the uncertainties of top events using the uncertainty data of failure rates which are available in the Individual Plant Examination(IPE) of the Seabrook Nuclear Power Station. However, the IPE does not have failure rates of all basic events in fault trees, and assigning uncertainties to failure rates is limited within some components. Therefore, some data are calculated by point estimation using the INPO database. In the IPE reports, failure rates are given by mean values, median values, 5^{th} and 95^{th} percentile data. Although probability density functions are not always log normally distributed, the log normal distribution approximation used in the IPE reports is suitable to describe those probability density functions of failure rates of components using mean values and error factors.

In order to evaluate how different mission time contributes to the system unreliability, we have chosen four different mission time: 0.1 year, half year, one year and two years. Since the Seabrook Nuclear Power Station is allowed to operate for two years at most according to the Technical Specifications, the largest mission time is set to be two years. Rates of time-dependent failures (i.e., failure to operate) have been assumed to be constant, such that we can employ exponential reliability distributions.

In most PRA models, the effects of uncertainty of the top events cannot be analytically estimated. We employed the Monte Carlo approach with Latin Hyper Cube sampling in order to calculate numerically the probability distribution of top events. The number of samples calculated is chosen as 99999, which is the largest values treated in the *SAPHIRE* code (Ref.2).

Fig. 7.3.3 and Fig.7.3.4 show the reliability of BOP and NSSS with uncertainty bounds (5^{th} and 95^{th}) as functions of the plant operation period (mission time). As the operation period increase, we can observe that the reliabilities decrease and uncertainties increase. Figs. 7.3.5 and 7.3.6 show the probability density function of an unplanned shutdown as functions of the four different mission times: 0.1 year, 0.5 year, 1 year and 2 year. In the case of the BOP system, the top event probability is close to unity; therefore, the probability density functions converges around unity at long times. Consequently, the associated uncertainties become reduced.

(2) Uncertainties in historical data

In order to compare the results of our fault tree analyses with the historical data, we present a graph of historical data on the same figures. The historical data are also assumed to be exponentially distributed with constant failure rates. We can see a good agreement between those data even though the fault tree models do not completely describe the entire picture of the power station.

In fact, failure rates derived from historical data also contain uncertainties with some corresponding confidence intervals. Confidence intervals of those failure rates can be estimated by censoring analysis. Since no termination time (Type 2 censoring) nor numbers of failures (Type 1 censoring) are determined, reactor trips (i.e., failure) can be classified as a random censoring mechanism (Ref.13). In other words, a reactor trips randomly and sometimes succeeds to operate until occurrence of outages (randomly censored). In this case the total operation time of a reactor can be written as follows:

$$\sum X_i = \sum t_i + \sum C_i \cong Y * A$$

where $\sum X_i$: Total Operation Time (Total time on test)

$\sum t_i$: Total time until unplanned shutdown

$\sum C_i$: Total time of successful operations (no unplanned shutdown until outage)

Y : Calendor years

A : Average availabili ty during the period Y

Maximum likelihood estimation for failure rates can be obtained as follows:

$$\hat{\lambda} = \frac{r}{\sum X_i}$$

where $\hat{\lambda}$: Failure rate

r : Number of reactor trips (Failues)

The confidence interval of failure rate can be derived by using the result:

$$2\lambda \sum X_i \sim \chi^2(2r)$$

where χ^2 : chi - square distribution

r : Degrees of freedoms in chi - square distribution

This result uses the chi-square distribution assumption with 2r degrees of freedom; this result is satisfied exactly in the Type 2 censoring case. Therefore, a confidence interval estimate can be written as follows:

$$\frac{\hat{\lambda}\chi^2_{2r,1-\alpha/2}}{2r} < \lambda < \frac{\hat{\lambda}\chi^2_{2r,\alpha/2}}{2r}$$

where $(1-\alpha)$ is equal to the confidence interval
(e.g. For $\alpha = 0.05$, confidence interval = 90%) (Ref.13)

Table 7.3.7 shows the summary of confidence intervals of failure rates obtained from the historical data of unplanned reactor shutdowns in each system at the Seabrook and Pilgrim Nuclear Power Stations. We can observe that the failure rates for unplanned reactor shutdowns have large confidence intervals (i.e., uncertainties). Taking these uncertainties into account, we can see good agreement obtains between the results of fault tree analysis and from historical data.

Table7.3.3 Summary of Fault Tree Models and Minimal Cut Sets at the Seabrook Nuclear Power Station

System	Probability of Shutdown Within One Year	Number of Basic Events	Number of Minimal Cut Sets				
			Cutset Size #1	Cutset Size #2	Cutset Size #3	Cutset Size #4	Total
NSSS	3.49E-01	54	36	16	0	0	52
	Cut Sets Probabilities P_i in One Year Operation		3.47E-01	8.92E-03	-	-	-
	Pi / Total Probability		99.40%	0.02%	-	-	-
BOP	8.62E-01	171	104	65	21	0	190
	Cut Sets Probabilities P_i		8.15E-01	2.46E-01	1.25E-02	-	-
	Pi / Total Probability		94.51%	0.14%	0.01%	-	-

Table7.3.4 The Number of Risk Important Components at the Seabrook Nuclear Power Station

System	Number of Minimal Cut Sets	Number of Risk Significant Components			
		Probability > 90% (1)	F.V. > 0.5%	RRW > 1.005	RAW > 2
NSSS	52	11	27	27	27
BOP	190	5	15	14	0

Note : (1) basic events which appear in cut sets making up the top 90 % of initiating the top event

Fig. 7.3.1 Risk Measures of Risk Significant Components/Basic Event Frequencies in NSSS System at the Seabrook Nuclear Power Station

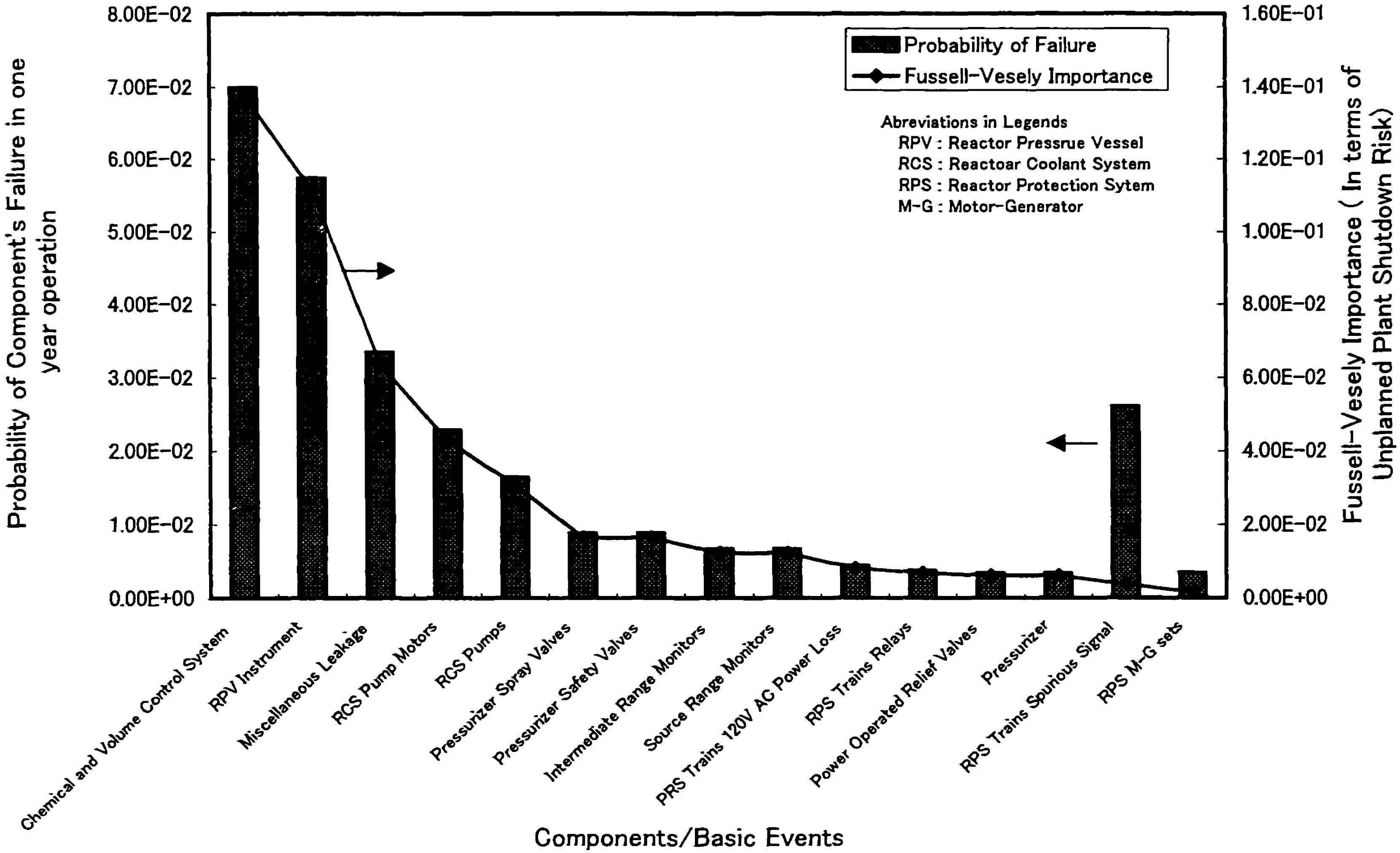


Fig. 7.3.2 Risk Measures of Risk Significant Components/Basic Event Frequencies in BOP System at the Seabrook Nuclear Power Station

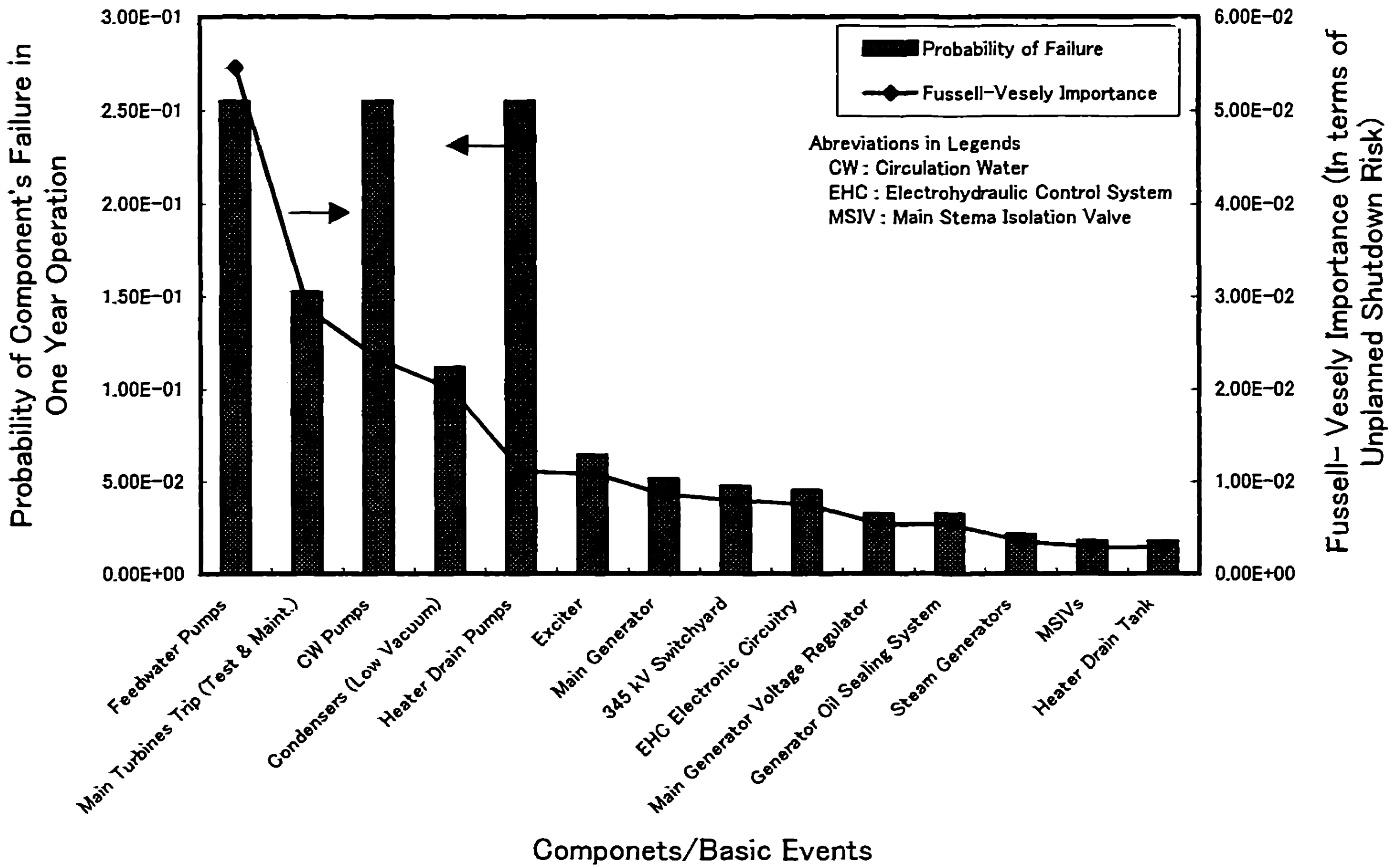


Table 7.3.5 Risk Significant Components/Basic Events in NSSS System at the Seabrook Nuclear Power Station

Events Name	Description of Basic Events	Probability (a)	Fussell-Vesely	RRW	RAW
RCHEM	Chemical and Volume Control Sys Failure	6.98E-02	1.38E-01	1.16E+00	2.83E+00
RPVINSTR	RPV Instrument Failure	5.73E-02	1.11E-01	1.13E+00	2.83E+00
RPVLEAK	Miscellaneous Leakage SN2	3.34E-02	6.34E-02	1.07E+00	2.83E+00
RCSCELE	RCS Pump A Electrical Failure SN1	2.27E-02	4.27E-02	1.05E+00	2.83E+00
RCSBELE	RCS Pump B Electrical Failure	2.27E-02	4.27E-02	1.05E+00	2.83E+00
RCSDELE	RCS Pump C Electrical Failure	2.27E-02	4.27E-02	1.05E+00	2.83E+00
RCSAELE	RCS Pump D Electrical Failure	2.27E-02	4.27E-02	1.05E+00	2.83E+00
RCSPC	RCS Pump C Mechanical Failure	1.64E-02	3.05E-02	1.03E+00	2.83E+00
RCSPA	RCS Pump A Mechanical Failure	1.64E-02	3.05E-02	1.03E+00	2.83E+00
RCSPB	RCS Pump B Mechanical Failure	1.64E-02	3.05E-02	1.03E+00	2.83E+00
RCSPD	RCS Pump D Mechanical Failure	1.64E-02	3.05E-02	1.03E+00	2.83E+00
RRCV116	Pressurizer Safety Valve RC-V-116 Failure	8.74E-03	1.62E-02	1.02E+00	2.83E+00
RPCV455A	Pressurizer Spray Valve PCV-455A Failure	8.74E-03	1.62E-02	1.02E+00	2.83E+00
RPCV455B	Pressurizer Spray Valve PCV-455B Failure	8.74E-03	1.62E-02	1.02E+00	2.83E+00
RRCV115	Pressurizer Safety Valve RC-V-115 Failure	8.74E-03	1.62E-02	1.02E+00	2.83E+00
RRCV117	Pressurizer Safety Valve RC-V-117 Failure	8.74E-03	1.62E-02	1.02E+00	2.83E+00
IRA	IR Channel A Detector Failure SN5	6.56E-03	1.21E-02	1.01E+00	2.83E+00
IRB	IR Channel B Detector Failure	6.56E-03	1.21E-02	1.01E+00	2.83E+00
SRA	SR Channel A Detector Failure	6.56E-03	1.21E-02	1.01E+00	2.83E+00
SRB	SR Channel B Detector Failure	6.56E-03	1.21E-02	1.01E+00	2.83E+00
RPSPWRB	Train B 120V AC Power Loss	4.35E-03	8.01E-03	1.01E+00	2.83E+00
RPSPWRA	Train A 120V AC Power Loss SN4	4.35E-03	8.01E-03	1.01E+00	2.83E+00
RPSTRNRLAYA	RPS Train A Relays Failure	3.67E-03	6.76E-03	1.01E+00	2.83E+00
RPSTRNRLAYB	RPS Train B Relays Failure	3.67E-03	6.76E-03	1.01E+00	2.83E+00
RPORV456A	PROV456A Failure	3.29E-03	6.04E-03	1.01E+00	2.83E+00
RPORV456B	PROV456B Failure	3.29E-03	6.04E-03	1.01E+00	2.83E+00
RPRESSURE	Pressurizer Failure	3.29E-03	6.04E-03	1.01E+00	2.83E+00
SPURIOUSA4	RPS Train A4 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSA3	RPS Train A3 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSB1	RPS Train B1 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSA1	RPS Train A1 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSB2	RPS Train B2 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSA2	RPS Train A2 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSB3	RPS Train B3 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
SPURIOUSB4	RPS Train B4 Spurious Signal	2.60E-02	3.71E-03	1.00E+00	1.14E+00
RPSMGBFTS	RPS MG B FTS	3.35E-03	1.57E-03	1.00E+00	1.47E+00
RPSMGAFTR	RPS MG A FTR	2.55E-01	1.57E-03	1.00E+00	1.01E+00
PRN43A	PR N43 Detector A Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN43B	PR N43 Detector B Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN42A	PR N42 Detector A Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN42B	PR N42 Detector B Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN44A	PR N44 Detector A Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN44B	PR N44 Detector B Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN41A	PR N41 Detector A Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
PRN41B	PR N41 Detector B Failure	8.23E-04	1.51E-03	1.00E+00	2.83E+00
RPRESSRELIEFTK	Pressurizer Relief Tank External Leakage	2.33E-04	4.27E-04	1.00E+00	2.83E+00
RPSTRIPBRKRBYPA	RPS Trip Breaker Bypass A FTRC	2.53E-03	1.09E-05	1.00E+00	1.00E+00
RPSTRIPBRKRBFTRC	RPS Trip Breaker B FTRC	2.35E-03	1.09E-05	1.00E+00	1.01E+00
RPSMGGENBRKA	RPS Generator Breaker A FTRC	2.35E-03	1.01E-05	1.00E+00	1.00E+00
RPSTRIPBRKRAFTRC	RPS Trip Breaker A FTRC	2.35E-03	1.01E-05	1.00E+00	1.00E+00
RPSMGGENBRKB	RPS Generator Breaker B FTRC	2.35E-03	1.01E-05	1.00E+00	1.00E+00
RPSTRIPBRKRBYPB	RPS Trip Breaker Bypass B FTRC	2.35E-03	1.01E-05	1.00E+00	1.00E+00

Note (a) All risk importance measure including the probability of basic events are evaluated based on one year operation.
Components are sorted by Fussell-Vesely.

Table 7.3.6 Risk Significant Components/Basic Events in BOP System at the Seabrook Nuclear Power Station

Events name	Description of Basic events	Probability (a)	Fussell-Vesely	RRW	RIW
FWPUMP(A)	FWPump P32A Failure	2.55E-01	5.46E-02	1.06E+00	1.16E+00
FWPUMP(B)	FW Pump P32B Failure	2.55E-01	5.46E-02	1.06E+00	1.16E+00
TTESTMAN	Turbine Trip Test & Maint. (Valv . Cntrl.)	1.52E-01	2.87E-02	1.03E+00	1.16E+00
CWPP(B)	CWpp(B)Trip	2.55E-01	2.32E-02	1.02E+00	1.06E+00
CWPP(C)	CWpp(C)Trip	2.55E-01	2.32E-02	1.02E+00	1.06E+00
CWPP(A)	CWpp(A)Trip	2.55E-01	2.32E-02	1.02E+00	1.06E+00
CONDENSER	Condenser failure	1.11E-01	2.00E-02	1.02E+00	1.16E+00
COHDPA	Heater Drain Pump A FTR	2.55E-01	1.11E-02	1.01E+00	1.03E+00
COHDPB	Heater Drain Pump B FTR	2.55E-01	1.11E-02	1.01E+00	1.03E+00
GEXCITE	Exciter Failure SB14	6.37E-02	1.09E-02	1.01E+00	1.16E+00
GMAIN	Main Generator Failure SB2	5.13E-02	8.64E-03	1.01E+00	1.16E+00
GSWY	345 kV Switchyard Failure SB9	4.73E-02	7.92E-03	1.01E+00	1.16E+00
EHCCIRCUIT	EHC Electronic Circuitry Failure	4.51E-02	7.53E-03	1.01E+00	1.16E+00
GVOLTRG	Voltage Regulator Failure	3.24E-02	5.34E-03	1.01E+00	1.16E+00
GSEAL	Oil Sealing System Failure	3.24E-02	5.34E-03	1.01E+00	1.16E+00
SG(C)	SG(C)	2.12E-02	3.45E-03	1.00E+00	1.16E+00
SG(A)	SG(A)	2.12E-02	3.45E-03	1.00E+00	1.16E+00
SG(D)	SG(D)	2.12E-02	3.45E-03	1.00E+00	1.16E+00
SG(B)	SG(B)	2.12E-02	3.45E-03	1.00E+00	1.16E+00
MSIVDFTRO	MSIV FTRO	1.74E-02	2.82E-03	1.00E+00	1.16E+00
MSIVCFTRO	MSIV FTRO	1.74E-02	2.82E-03	1.00E+00	1.16E+00
MSIVAFTRO	MSIV FTRO	1.74E-02	2.82E-03	1.00E+00	1.16E+00
MSIVBFTRO	MSIV FTRO	1.74E-02	2.82E-03	1.00E+00	1.16E+00
COHDTK	Heater Drain Tank Failure	1.71E-02	2.77E-03	1.00E+00	1.16E+00
TMSRDTB	MS/R Drain Tank	1.70E-02	2.75E-03	1.00E+00	1.16E+00
TMSRDTA	MS/R Drain Tank	1.70E-02	2.75E-03	1.00E+00	1.16E+00
EHCRES	Reservoir Failure	1.70E-02	2.75E-03	1.00E+00	1.16E+00
TMSRDTC	MS/R Drain Tank	1.70E-02	2.75E-03	1.00E+00	1.16E+00
COSPACK	Steam Packing Exhausting Failure	1.70E-02	2.75E-03	1.00E+00	1.16E+00
TMSRDTD	MS/R Drain Tank	1.70E-02	2.75E-03	1.00E+00	1.16E+00
EHCRECIRCTK	EHC Recalculating Tank Failure	1.70E-02	2.75E-03	1.00E+00	1.16E+00
FWE26B	FW HTR E26 B Failure	1.70E-02	2.75E-03	1.00E+00	1.16E+00
FWE26A	FW HTR E26 A Failure	1.70E-02	2.75E-03	1.00E+00	1.16E+00
GUATA	Unit transformer A Failure	1.36E-02	2.20E-03	1.00E+00	1.16E+00
GST	Main Transformer Failure	1.36E-02	2.20E-03	1.00E+00	1.16E+00
GRAT	Start Up transformer Failure	1.36E-02	2.20E-03	1.00E+00	1.16E+00
GUATB	Unit transformer B Failure	1.36E-02	2.20E-03	1.00E+00	1.16E+00
GSWCSB7	Stator Winding Cooling Failure	1.31E-02	2.12E-03	1.00E+00	1.16E+00
EHCMECHFILT	Mechanical Filter Failure	2.32E-01	2.02E-03	1.00E+00	1.01E+00
EARTHEFILA	Earthfilter A Failure	2.32E-01	2.02E-03	1.00E+00	1.01E+00
EARTHEFILB	Earth Filter B Failure	2.32E-01	2.02E-03	1.00E+00	1.01E+00
GHYDRO	Hydrogen Cooling System Failure	6.56E-03	1.05E-03	1.00E+00	1.16E+00
EHCHEATERB	EHC Heater B Failure	7.18E-02	8.27E-04	1.00E+00	1.01E+00
EHCHEATERA	EHC Heater A Failure	7.18E-02	8.27E-04	1.00E+00	1.01E+00
MSSG4SRVD	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG1SRVB	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
TMSRRVB	MS/R Relief Valve Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG4SRVE	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG1SRVA	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG2SRVD	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
TMSRRVC	MS/R Relief Valve Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG2SRVA	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG2SRVC	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG2SRVB	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
TMSRRVD	MS/R Relief Valve Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
TMSRRVA	MS/R Relief Valve Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG4SRVA	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG4SRVB	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG3SRVE	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG3SRVD	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG3SRVA	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG3SRVC	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG2SRVE	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG1SRVD	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG4SRVC	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG1SRVE	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00

Table 7.3.6 Risk Significant Components/Basic Events in BOP System at the Seabrook Nuclear Power Station

Events name	Description of Basic events	Probability (a)	Fussell-Vesely	RRW	RIW
MSSG3SRVB	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
MSSG1SRVC	SG SRV Leak	4.18E-03	6.71E-04	1.00E+00	1.16E+00
TMSR2	MS/R	1.65E-03	2.64E-04	1.00E+00	1.16E+00
TMSR4	MS/R	1.65E-03	2.64E-04	1.00E+00	1.16E+00
TMSR3	MS/R	1.65E-03	2.64E-04	1.00E+00	1.16E+00
TMSR1	MS/R	1.65E-03	2.64E-04	1.00E+00	1.16E+00
CWSWHPC	Wash Pump C FTR	2.33E-03	1.92E-04	1.00E+00	1.07E+00
CWSWHPA	Wash Pump A FTR	2.33E-03	1.92E-04	1.00E+00	1.07E+00
CWSWHPB	Wash Pump B FTR	2.33E-03	1.92E-04	1.00E+00	1.07E+00
COPP(C)	Condensate Pump P30C FTS	2.35E-03	1.91E-04	1.00E+00	1.07E+00
FWHUMAN	Operator Error SB10 SB12 SB17	1.00E-03	1.60E-04	1.00E+00	1.16E+00
EHC1421COM	US-14and21 CCF	1.00E-03	1.60E-04	1.00E+00	1.16E+00
FWV48	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSASDVBFTRC	Atmospheric Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV28	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSASDVAFTRC	Atmospheric Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWLV4240	FW Level CV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV55	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWFCV530	FW FCV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWLV4210	FW Level CV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
TMSRSTVA	MS/R Stop Valve FTRO	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWFCV540	FW FCV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWLV4230	FW Level CV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVC2	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV39	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV37	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV30	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWLV4220	FW Level CV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVC1	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
TMSRSTVB	MS/R Stop Valve FTRO	8.12E-04	1.30E-04	1.00E+00	1.16E+00
TMSRSTVD	MS/R Stop Valve FTRO	8.12E-04	1.30E-04	1.00E+00	1.16E+00
TMSRSTVC	MS/R Stop Valve FTRO	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV46	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSASDVDFTRC	Atmospheric Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVA1	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVB1	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVB2	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSSDVA2	Condenser Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
MSASDVCFTRC	Atmospheric Steam Dump Valve FTRC	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWV57	FW MOV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
CWBFYVA	Butterfly Valve A Failure	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWFCV510	FW FCV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
CWBFYVB	Butterfly Valve B Failure	8.12E-04	1.30E-04	1.00E+00	1.16E+00
GIPBDAM	IPB Dampers Failure SB13	8.12E-04	1.30E-04	1.00E+00	1.16E+00
FWFCV520	FW FCV	8.12E-04	1.30E-04	1.00E+00	1.16E+00
COPP(A)	Condensate Pump P30A Failure	2.55E-01	9.56E-05	1.00E+00	1.00E+00
COPP(B)	Condensate Pump P30B Failure	2.55E-01	9.56E-05	1.00E+00	1.00E+00
EHCPPA	Pump A FTR	2.55E-01	9.56E-05	1.00E+00	1.00E+00
EHCPPB	Pump B FTS	2.35E-03	9.56E-05	1.00E+00	1.04E+00
CWSCRMOTA	Traveling Screen A Failure	8.06E-04	6.64E-05	1.00E+00	1.07E+00
CWSCRMOTC	Traveling Screen C Failure	8.06E-04	6.64E-05	1.00E+00	1.07E+00
CWSCRMOTB	Traveling Screen B Failure	8.06E-04	6.64E-05	1.00E+00	1.07E+00
COE21A	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE25B	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE21B	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE21C	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE24B	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE22A	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE23A	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE25A	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE23C	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE23B	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE22B	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE24A	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE24C	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
EHCCOOLERA	EHC Cooler A Failure	1.70E-02	4.59E-05	1.00E+00	1.00E+00

Table 7.3.6 Risk Significant Components/Basic Events in BOP System at the Seabrook Nuclear Power Station

Events name	Description of Basic events	Probability (a)	Fussell-Vesely	RRW	RIW
EHCCOOLERB	EHC Cooler B Failure	1.70E-02	4.59E-05	1.00E+00	1.00E+00
COE22C	Condensate Heater	1.70E-02	4.59E-05	1.00E+00	1.00E+00
FWV29	FW Check Valve	9.11E-05	1.46E-05	1.00E+00	1.16E+00
FWV38	FW Check Valve	9.11E-05	1.46E-05	1.00E+00	1.16E+00
FWV47	FW Check Valve	9.11E-05	1.46E-05	1.00E+00	1.16E+00
FWV56	FW Check Valve	9.11E-05	1.46E-05	1.00E+00	1.16E+00
EHCJETSUPPLY	EHC Jet Supply Line	7.61E-05	1.22E-05	1.00E+00	1.16E+00
EHCTRIPSUPPLY	EHC Trip Supply Line	7.61E-05	1.22E-05	1.00E+00	1.16E+00
EHCACTUATINGSUPPLY	EHC Actuating Supply Line	7.61E-05	1.22E-05	1.00E+00	1.16E+00
EHC14	US-14 Failure	6.00E-03	5.75E-06	1.00E+00	1.00E+00
EHC21	US-21 Failure	6.00E-03	5.75E-06	1.00E+00	1.00E+00
GFANSB	IPB Fan B FTS	4.84E-04	5.16E-06	1.00E+00	1.01E+00
GFANSA	IPB Fan A FTR	6.68E-02	5.16E-06	1.00E+00	1.00E+00
TMSRSLVC	MS/R Steam Load Valve FTRO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVB	MS/R Steam Load Valve FTRO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVD	MS/R Steam Load Valve FTRO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVVD	MS/R Steam Load Bypass Valve FTO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVVA	MS/R Steam Load Bypass Valve FTO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVA	MS/R Steam Load Valve FTRO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVVB	MS/R Steam Load Bypass Valve FTO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRSLVVC	MS/R Steam Load Bypass Valve FTO	8.12E-04	4.21E-07	1.00E+00	1.00E+00
TMSRDV42	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV41	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV40	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV43	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV37	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV36	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV38	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
TMSRDV39	MS/R Drain Valve	8.12E-04	1.05E-07	1.00E+00	1.00E+00
EHCPSWBSB3	SW B Failure	2.69E-04	2.31E-08	1.00E+00	1.00E+00
EHCPSWASB3	SW A Failure	2.69E-04	2.31E-08	1.00E+00	1.00E+00
EHCPSWC	SW C Failure	2.69E-04	2.31E-08	1.00E+00	1.00E+00
EHCNTRC	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
EHCNTRD	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
EHCNTRB	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
EHCNTRE	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
EHCNTRA	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
EHCNTRF	EHC Nitrogen Accumulator	1.10E-03	2.12E-09	1.00E+00	1.00E+00
TMSRDV40	MS/R Drain Valve	8.12E-04	3.79E-09	1.00E+00	1.00E+00
TMSRDV42	MS/R Drain Valve	8.12E-04	3.79E-09	1.00E+00	1.00E+00
TMSRDV36	MS/R Drain Valve	8.12E-04	3.79E-09	1.00E+00	1.00E+00
TMSRDV37	MS/R Drain Valve	8.12E-04	3.79E-09	1.00E+00	1.00E+00
TMSRDV39	MS/R Drain Valve	8.12E-04	3.79E-09	1.00E+00	1.00E+00
EHCPSWB	SW B Failure	2.69E-04	8.32E-10	1.00E+00	1.00E+00
EHCPSWC	SW C Failure	2.69E-04	8.32E-10	1.00E+00	1.00E+00
EHCPSWA	SW A Failure	2.69E-04	8.32E-10	1.00E+00	1.00E+00
EHCNTRA	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00
EHCNTRD	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00
EHCNTRC	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00
EHCNTRB	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00
EHCNTRE	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00
EHCNTRF	EHC Nitrogen Accumulator	1.10E-03	2.29E-11	1.00E+00	1.00E+00

Note (a) All risk importance measure including the probability of basic events are evaluated based on one year operation.
Components are sorted by Fussell-Vesely.

Fig. 7.3.3 Reliability of NSSS of the Seabrook Nuclear Power Station

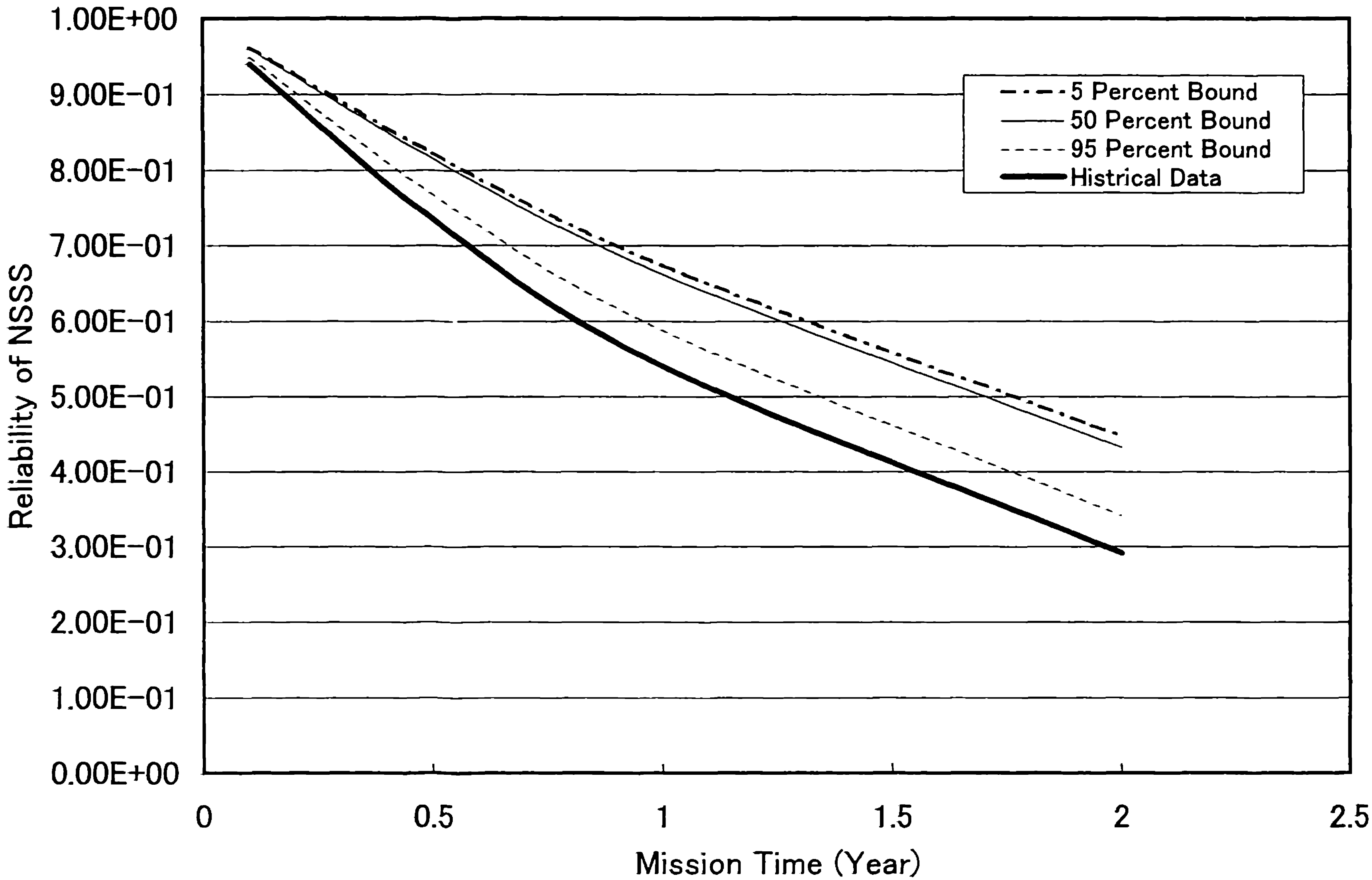


Fig. 7.3.4 Reliability of BOP of Seabrook Nuclear Power Station

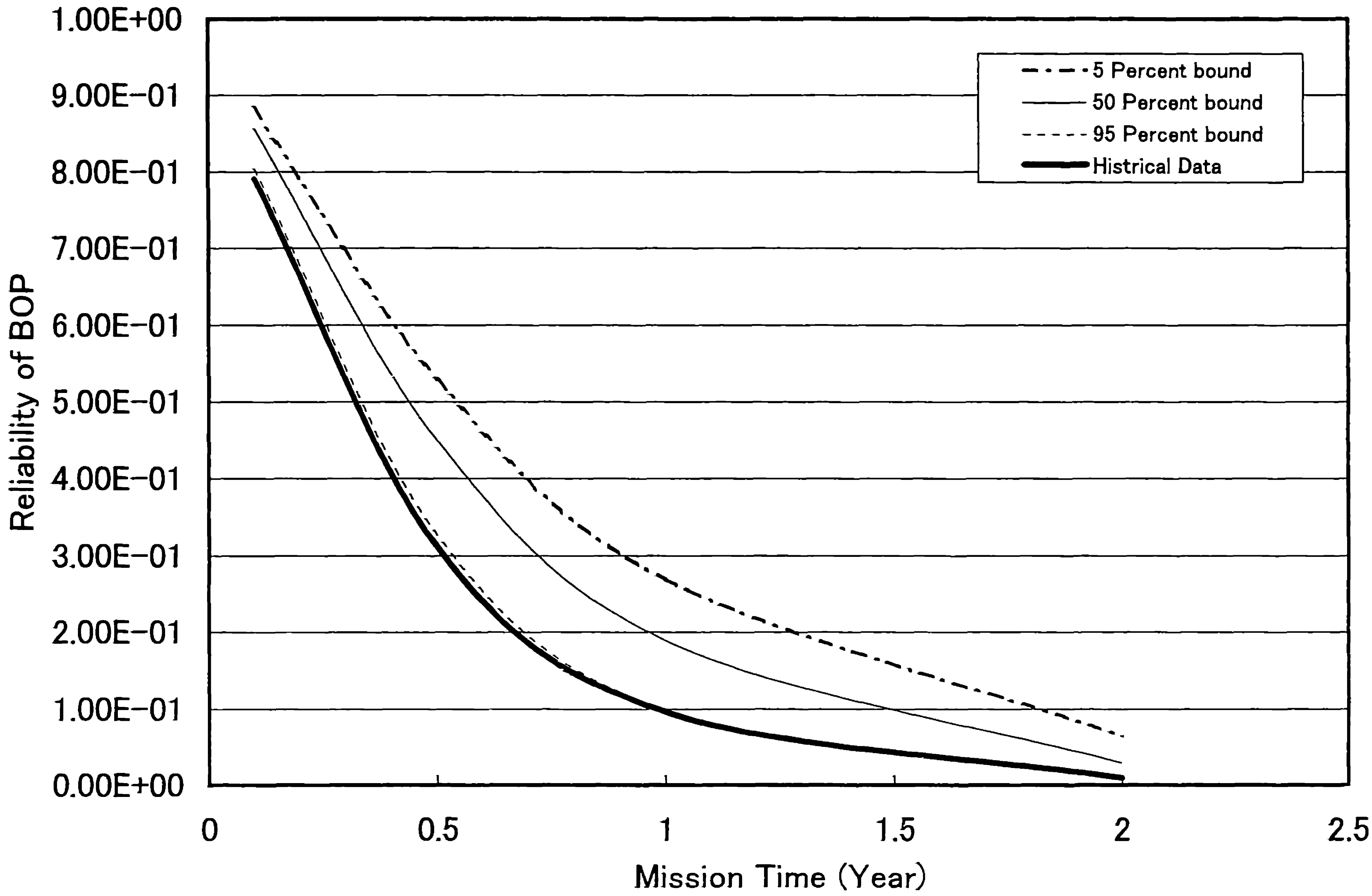


Fig. 7.3.5 Probability Distribution of Plant Shutdown due to the Failure of NSSS over Different Mission Times at the Seabrook Nuclear Power Station

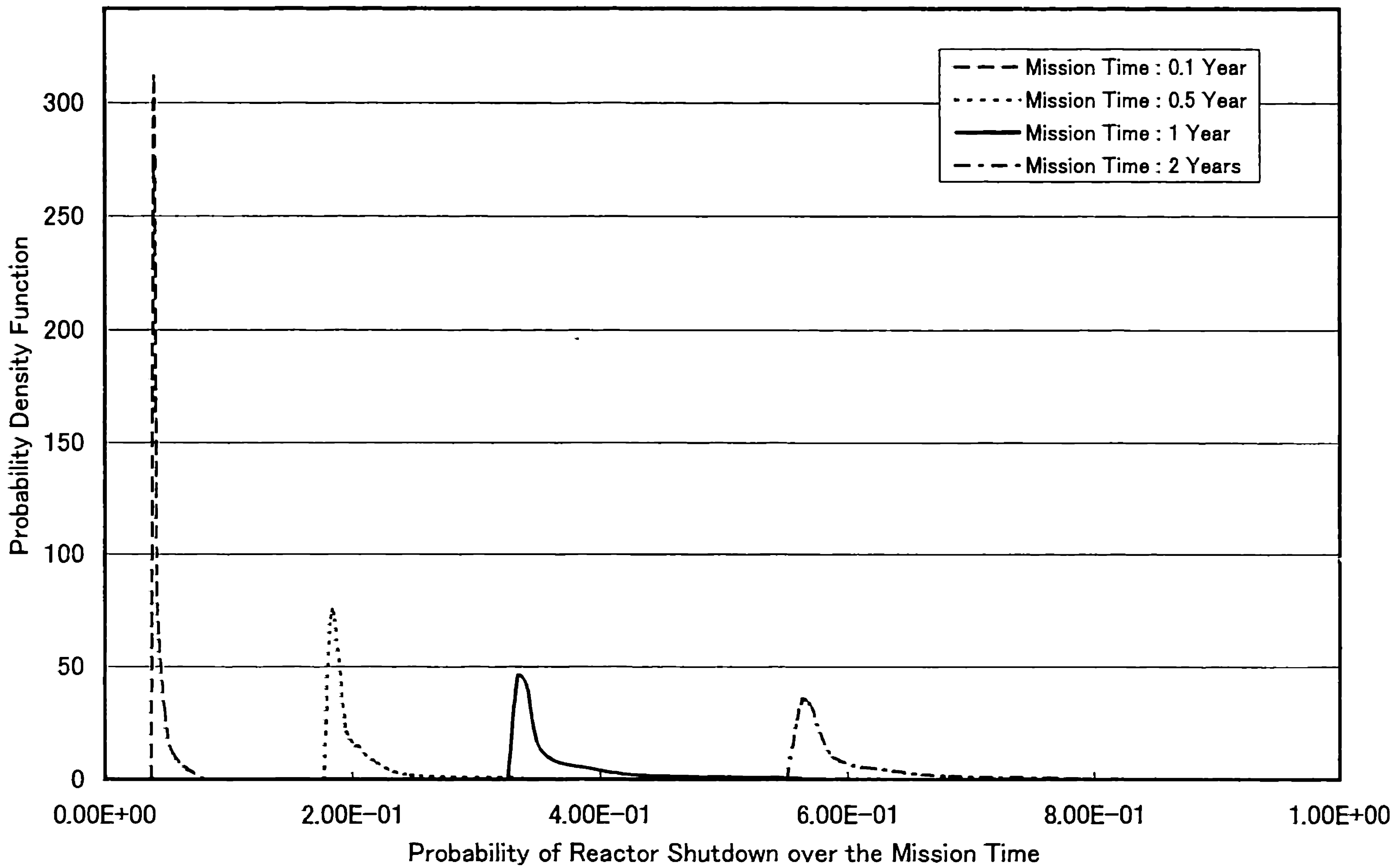


Fig.7.3.6 Probability Distribution of Plant Shutdown due to the Failure of BOP over Different Mission Times at the Seabrook Nuclear Power Station

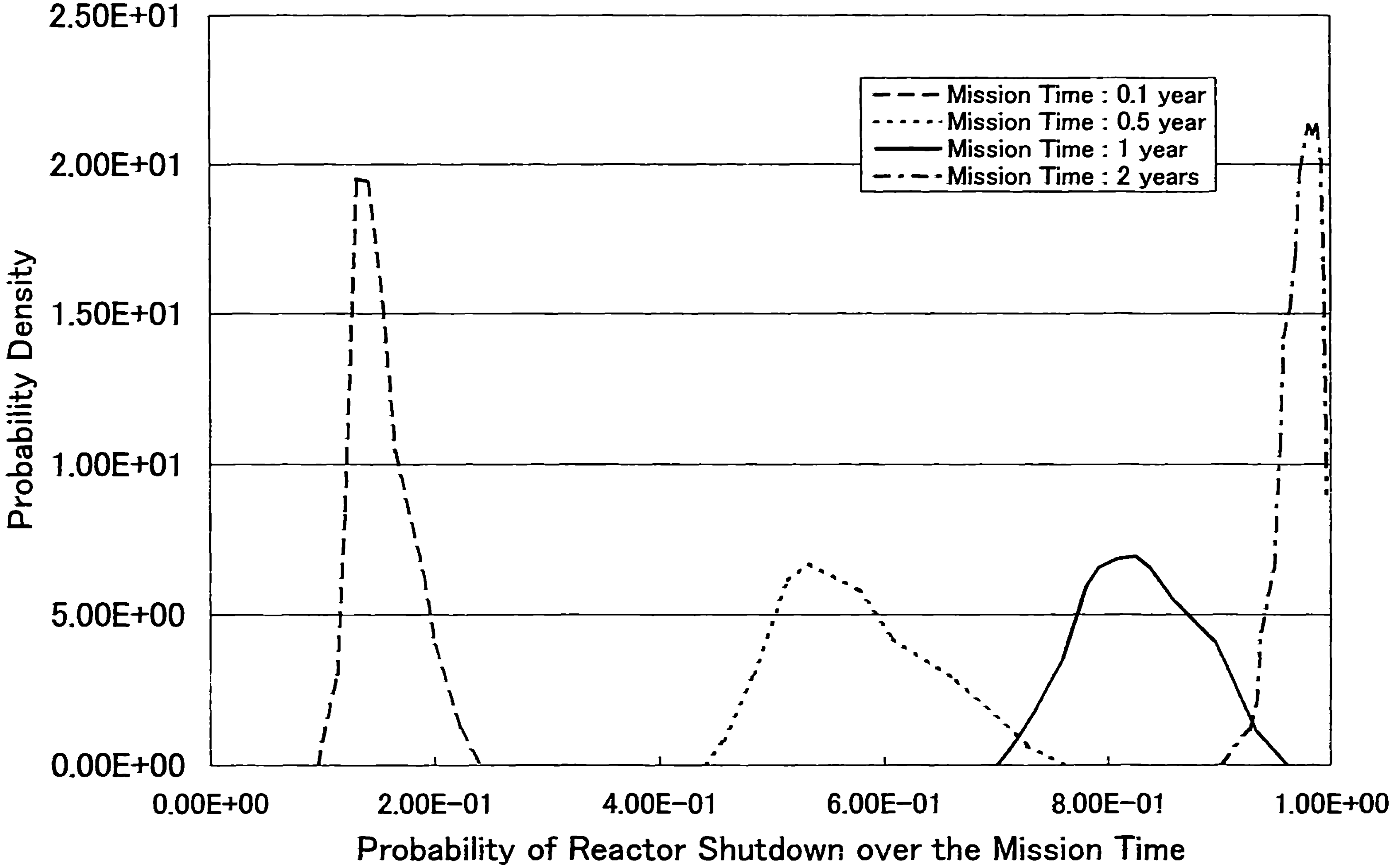


Table 7.3.7 Summary of Confidence Intervals of Frequencies of Shutdowns

Uncertainty Parameters		Seabrook		Pilgrim	
		BOP	NSSS	BOP	NSSS
Total operation time (years)		8.1			
Number of failures		19	5	11	13
Point Failure Rate Estimator (Year)		2.35	0.617	1.36	1.6
Confidence Intervals (Year)	5%	1.41	0.2	0.68	0.85
	95%	3.51	1.26	2.27	2.59

8. Prospective Improvement in Maintenance Practice

8.1 Current Maintenance Practice

One of the possible application of probabilistic risk analysis is to improve the maintenance resource allocation in respect to the risk significance of components. In general, maintenance practices in nuclear power stations are categorized into the following three areas:

- Corrective maintenance
- Condition based maintenance
- Preventative maintenance

Corrective maintenance is maintenance performed by the maintenance group to correct equipment after it fails. This method is perfectly suited for simple and reliable equipment. However, this maintenance approach leads to economic difficulties as soon as the equipment become more important and complex, which is the case of most nuclear power station systems. Thus, in a well run plant the fraction of maintenance work that is corrective should be small

Condition-based maintenance involves more than merely keeping the equipment running; it involves keeping it at a pre-determined reliability level(above a threshold value). Extensive engineering skills and experience are required to be able to evaluate current and future component performance properly. Detailed information, such as vibration analysis or oil sampling, is necessary to evaluate the real condition of equipment and to respond to it intelligently.

Preventive maintenance is mainly applied for major components in nuclear power stations. Certain components follow simple rules of deterioration and their failure age can easily be determined through careful observation. Periodic replacement of these components can avoid undesirable incidents, while keeping a certain degree of flexibility in scheduling preventive maintenance outages. This approach, preventive maintenance, reduces component unavailability and direct maintenance costs. Such a maintenance policy is easy to implement on the basis of experience and common sense, and /or by equipment analysis.

8.2 Establishment of Maintenance Program

Establishing an optimal maintenance program is not easy, and requires strong engineering capabilities and experience. The following are the four major maintenance indicators which must be taken into account in developing the maintenance program:

- Plant Availability and Core Damage Frequency (CDF)
- Component Reliability
- Direct Maintenance Costs
- Impact of failure upon the public concerns

Taking these indicators into account, we can establish a maintenance program by taking the following steps.

- Develop a system-driven critical equipment list, depending upon the specific maintenance indicator
- Specify which maintenance method is to be selected according to the failure mode/rate
- Develop the maintenance program for each significant failure mechanism of critical equipment

However, developing a maintenance program is a time-consuming process and requires deep plant-specific knowledge and a strong engineering background. Therefore, we have evaluated only component and system reliabilities in our work. Furthermore, in order to simplify the maintenance planning process, preventive maintenance is assumed to be the suitable maintenance method for risk significant components based on the following two assumptions:

Assumption 1 ;

Corrective maintenance is not suitable for major components because failures of major components are more likely to result in unplanned shutdowns. Failures of major components cannot be allowed if one is to achieve high reliability and availability of nuclear power plants.

Asummption2 ;

Monitoring and diagnosis techniques, which are necessary to conduct condition based maintenance, have not been developed reliably enough to permit one to evaluate and predict the condition of equipment.

In practice, estimating the burden (i.e., intervals and man-hours) of corrective and condition-based maintenance is difficult because there no specific intervals and man-hours for these types of maintenance have been determined. Therefore, preventive maintenance is considered to be easily improved in terms of maintenance resource allocation. Surveillance tests for components are also important in case of longer fuel cycles; however, they are less important in terms of maintenance burden. Thus, preventive maintenance is the only type of maintenance considered in our work. Due to the difficulties in obtaining necessary information concerning maintenance at the Pilgrim Nuclear Power Station, we have evaluated maintenance practices for major components only at the Seabrook Nuclear Power Station (see Section 8.3).

8.3 Maintenance Tasks and Risk Significance at the Seabrook Nuclear Power Station

Table 8.3.1 shows the maintenance tasks for the risk significant components in terms of plant shutdowns. These data are obtained from the maintenance database at the Seabrook Nuclear Power Station. Although the maintenance tasks are not explicitly described in the database, we can roughly evaluate the maintenance practices at the Seabrook Power Station.

First of all, preventive maintenance is not always applied to the risk significant components. By comparing risk importance in terms of shutdowns and maintenance tasks, we can identify equipment for which preventive maintenance may be preferable in improving plant availability as follows:

- Feedwater Pumps
- Main Turbine Control Valves and Stop Valves
- Main Steam Isolation Valves
- Chemical and Volume Control System
- Main Generator Oil Sealing System

Many major valves installed on the main steam lines are maintained by condition-monitoring even though they are critical in regard to plant availability. Certain amounts of failures occur during the surveillance testing of these valves. Limited resources make it difficult to apply preventive maintenance to an availability risk significant component.

As an example of the application of risk assessment in terms of core damage frequency, motor operated valves are ranked according to their respective Fussell-Vesely importance and Risk Achievement Worth values at the Seabrook Nuclear Power Station. Although risk importance is not the only criterion used to develop the maintenance program, Seabrook's maintenance program takes the risk significance into consideration. DuBord, et al., have suggested a methodology to optimize the maintenance resources using RAW and RRW values (Ref.14). In order to improve the maintenance resource allocation, they take the ratio of burden to risk importance, such as RAW and RRW, in each component. If the maintenance resources are properly allocated, the ratio should be constant for different components. This methodology is suitable for evaluating maintenance resource allocation if risk is expressed in terms of core damage frequency. However, it is not suitable for our work because the risk is expressed in terms of plant shutdowns and the probabilities of top events (i.e., unplanned shutdowns) are very high (close to unity). In addition, since failures of single components dominate the top events, RAW values are almost constant among the high risk components in terms of plant unavailability. In other words, there is no room for RAW to increase the failure probability because it is already too high. Therefore, ranking the components by RAW and RRW values is of limited value for the risk-dominant events in our work.

In general, many resources are spent for the safety related equipment such as Emergency Core Cooling System (ECCS). The investigation of unplanned plant shutdowns shows that only one out of forty nine shutdowns was caused by the failure of the ECCS at the Seabrook and Pilgrim Nuclear Power Stations between 1989 and 1997. Core damage must be taken into account when deciding how to rationalize maintenance resources; however, primarily one needs to look at plant availability.

8.4 Summary

The following are the conclusions drawn from the investigation at the Seabrook and the Pilgrim Nuclear Power Stations and risk analysis of the data.

1) Data availability and quality

The annual probabilities of plant unplanned shutdowns due to the failures of major systems are relatively high and close to unity. Although failure data used in fault tree analyses are not always accurate, the annual probabilities of plant unplanned shutdowns obtained from fault tree analyses are consistent with the failure probabilities obtained from the historical data. It is impossible to obtain perfect data, however, we can evaluate the system reliability at useful levels of accuracy using existing failure data with satisfactory accuracy.

2) Fault tree results

Failures of turbine-generator systems and nuclear steam supply systems compromise plant reliability, and failure of those systems have sometimes led to unfavorable publicity. Although many people may believe that some specific reasons, such as failures of safety-related systems, strongly affect the creation of newspaper stories concerning events which have occurred at nuclear power stations, no single significant key factors were observed to explain which events were reported in the articles of the Boston Globe. Failures of single components account for almost all unplanned plant shutdowns. Redundancies of major systems within a plant do not contribute greatly to a reduction in the probabilities of a plant shutdown, since most shutdowns involve failure of only a single component, particularly involving BOP components.

3) Uncertainty implications

The annual probabilities of unplanned shutdowns contain some uncertainties due to the variability of failure data for basic events. The range of plant unreliability (i.e., the ratio of the five percent confidence bound values to the ninety-five percent confidence bound values) is small enough to permit one to evaluate the plant reliability at useful level of accuracy.

4) Maintenance

Preventive maintenance is not typically applied to risk significant components in proportion to their respective risk importance values. In order to improve plant reliability and to avoid unfavorable publicity, we conclude that one should allocate most maintenance resources to major failure-prone, single components in the BOP and NSSS, such as feedwater pumps.

Table 8.3.1 Maintenance Tasks for Risk Significant Components at the Seabrook Nuclear Power Station

System	Risk Significant Components/Events sorted by Fussell-Vesely Importance (In terms of Shutdowns)	Probability of Failures	F-V	Maintenance Program (1)	PM Interval (2)
BOP	Feedwater Pumps	2.5E-01	5.5E-02	CB	
	Main Turbines Trip (Test & Maintenance)	1.5E-01	2.9E-02	CB	
	Circulating Water Pumps (CW Pumps)	2.5E-01	2.3E-02	PM	RF03
	Condensers (Low Vacuum)	1.1E-01	2.0E-02	CB	
	Heater Drain Pumps	2.5E-01	1.1E-02	PM	RF02
	Exciter	6.4E-02	1.1E-02	PM	RF04
	Main Generator	5.1E-02	8.6E-03	PM	RF04
	345 kV Switchyard	4.7E-02	7.9E-03	PM	RF01
	Electro Hydraulic Control System (EHC) Circuitry	4.5E-02	7.5E-03	PM	RF01
	Main Generator Voltage Regulator	3.2E-02	5.3E-03	PM	04WK
	Generator Oil Sealing System	3.2E-02	5.3E-03	CB	
	Steam Generators	2.1E-02	3.5E-03	PM	01YR
	Main Steam Isolation Valves (MSIVs)	1.7E-02	2.8E-03	CB	
	Heater Drain Tank	1.7E-02	2.8E-03	PM	01YR
	Moisture Separator/Reheater Drain Tanks	1.7E-02	2.8E-03	PM	01YR
	EHC Oil Reservoir	1.7E-02	2.8E-03	PM	RF01
NSSS	Chemical and Volume Control System	7.0E-02	1.4E-01	CB	
	Pressure Vessel Instrument	5.7E-02	1.1E-01	PM	RF01
	Miscellaneous Leakage	3.3E-02	6.3E-02	CB	
	Reactor Coolant System (RCS) Pump Motors	2.3E-02	4.3E-02	PM	02YR
	RCS Pumps	1.6E-02	3.1E-02	PM	RF01
	Pressurizer Spray Valves	8.7E-03	1.6E-02	PM	RF02
	Pressurizer Safety Valves	8.7E-03	1.6E-02	CM	
	Intermediate Range Monitors	6.6E-03	1.2E-02	PM	18MN
	Source Range Monitors	6.6E-03	1.2E-02	PM	18MN
	Reactor Protection System (RPS) 120V AC Power Loss	4.4E-03	8.0E-03	N/A	N/A
	RPS Trains Relays	3.7E-03	6.8E-03	N/A	N/A
	Power Operated Relief Valves	3.3E-03	6.0E-03	PM	RF01
	Pressurizer	3.3E-03	6.0E-03	PM	RF01
	RPS Trains Spurious Signal	2.6E-02	3.7E-03	N/A	N/A
	RPS Motor-Generator (M-G) sets	3.4E-03	1.6E-03	N/A	N/A
	Power Range Monitors	8.2E-04	1.5E-03	PM	18MN

(1) PM : Preventive Maintenance
CM : Condition Based Maintenance and Corrective Maintenance
N/A : No information available in maintenance database
(2) RFOX : Every X Refueling Outage
XWK : Every X week
XMN : Every X month

9. Conclusion

9.1 Results of This Study

The licensees of the U.S. nuclear power plants are concerned about the reliability of those systems which are more likely to result in unfavorable publicity or to embarrass safety regulators. In order to improve our understanding of how to avoid unfavorable publicity surrounding nuclear plants, we have investigated all newspaper stories about the Pilgrim and Seabrook Nuclear Power Plants which appeared in the Boston Globe between 1989 and 1997. The investigation showed us that two hundred and two articles were written about both power stations. We divided these into eleven categories according to the aspects that they dealt with -- unplanned shutdowns, planned shutdowns, and other technical and non-technical issues. We focused only on unplanned shutdowns. Both nuclear power stations experienced forty-nine unplanned shutdowns between 1989 and 1997, of which sixteen unplanned shutdowns, approximately 33 % of the total number of shutdowns, were written about in articles. In order to identify what factors made these sixteen unplanned shutdowns result in newspaper coverage, we spoke with an editor of the Boston Globe. Through the discussion with the editor, the following five key factors were identified: which might affect whether an event was reported in the newspaper

- Reason for Shutdown
- Sensitivity of Public Interest Environment Surrounding Nuclear Power
- Timing of Occurrence of Incidents
- Space Limitations in Newspaper
- Reasons of chance concerning the capability of the newspaper to focus attention on the event

In order to verify our hypotheses that these five key factors were indeed the most significant, all forty nine unplanned shutdowns have been investigated according to our established criteria. The investigation showed us that only the timing of the occurrence (i.e., the day of the week) was an important determinant of newspaper coverage of an event. However, this may be misleading because, in our work, we were focusing only upon unplanned shutdowns. Coverage of other types of events (planned shutdowns, and technical and non-technical issues) may correlate with these other key factors.

Further investigations were carried out in order to identify which systems were most likely to have been involved in unplanned shutdowns and reported in articles. We first investigated the Licensee Event Reports (LERs) and categorized all reported events according to the major systems that they involved, for example, the Balance of Plant System and the Nuclear Steam Supply System. It was necessary to identify which items were critical in terms both of leading to shutdowns and resulting in unfavorable publicity; therefore, risk analysis with fault tree modeling was carried out to quantify the respective unplanned shutdown risks of the major components in turbine-generator systems and in nuclear steam supply systems. The risk was evaluated in terms of the annual frequency of unplanned shutdowns.

As a result, feedwater pumps, major valves installed on the main steam lines, the chemical and volume control system, and switchyards were identified as risk significant components. Other equipment was also ranked according to its respective risk significance. In general, failures of single components in turbine-generator systems and nuclear steam supply systems reduced plant reliability, while failures of safety-related systems did not.

Uncertainty analysis for failure rates of components was carried out using the Monte Carlo approach to compare the results of fault tree analyses with historical data. Taking uncertainty into consideration, the analytical data and the historical data showed good agreement with regard to reliability over different time spans of operation.

Maintenance practices for major components at the Seabrook Nuclear Power Station was investigated in order to correlate these practices and the components' risk significance. Some components, such as Feedwater Pumps and Main Steam Isolation Valves, are maintained by the condition-based methods; however, because they have high risk significance in terms of plant shutdowns, preventive maintenance would be preferable for these components.

9.2 Future Work

The probabilistic risk analysis approach can be applied to reliability improvement by rationalizing use of maintenance resources. Components are ranked in accordance with their risk significance in terms of their potential to cause plant shutdowns. In order to develop appropriate maintenance programs, the following issues must be considered for future work.

Failure rates of components used in the IPE are not always accurate enough to be used to calculate the probability of plant shutdowns because failures of single components dominate the unexpected reactor shutdown risk. The IPE sometimes sets high failure rates on major components, which makes the estimated probability of shutdowns very high due to the lack of redundancies to prevent shutdowns. Longer mission times (i.e., one year) in our work compared to those used in the analysis of core damage frequencies (i.e., 24 hours) have also made the probability of shutdowns very high.

Appropriate modeling including entire systems and subsystems is essential to estimate the probability of shutdowns.

Since availability is more important than reliability in terms of power generation, repair rate estimation for both safety-related components and non-safety related components are necessary (although they may be difficult to obtain).

Knowledge of core damage frequencies, maintenance costs and recovery rates need to be integrated in order to establish the appropriate maintenance program.

However, the current state of reliability knowledge is sufficient to permit important changes in maintenance to be made now. The suggested future work would render such changes still more accurate and confident.

References List

1. The U.S. Nuclear Regulatory Commission, "Performance Indicators for Operating Commercial Nuclear Power Reactors : Data through September 1998", NUREG/CR-1187, Vol. 2, 1998
2. The Idaho National Engineering Laboratory, "Systems Analysis Programs for Hand-on Integrated Reliability Evaluations (SAPHIRE)", Version 6, February 1998
3. Pilgrim Nuclear Power Station, "Individual Plant Examination", Rev. 0, September 1992
4. Seabrook Nuclear Power Station, "Individual Plant Examination", Rev. 0, September 1993
5. The U.S. Nuclear Regulatory Commission, "Analysis of Core Damage Frequency: Internal Events Methodology", NUREG/CR-4550, Vol. 1, Rev.1, 1990
6. Pilgrim Nuclear Power Station, "Procedure No. 16.03 10CFR50.65 NRC Maintenance Rule", August 1997
7. The U.S. Nuclear Regulatory Commission, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report", NUREG/CR-1278-F, SAND80-200, RX, AN, August 1983
8. Ali Mosleh, "Common Cause Failure: An Analysis Methodology and Examples", Reliability Engineering and Systems Safety 34, 249-292, 1991
9. Nathan O. Siu and Dana L. Kelly; "Bayesian Parameter Estimation in Probabilistic Risk Assessment", Reliability Engineering and System Safety, Vol. 62, 1997
10. The Institute of Nuclear Plant Operations, " The Equipment Performance and Information Exchange"
11. The Institute of Nuclear Plant Operations, " The Nuclear Power Plant Reliability Data System"
12. Michael C. Cheok, Gareth W. Parry and Richard R. Sherry, "Use of Importance Measure in Risk-Informed Regulatory

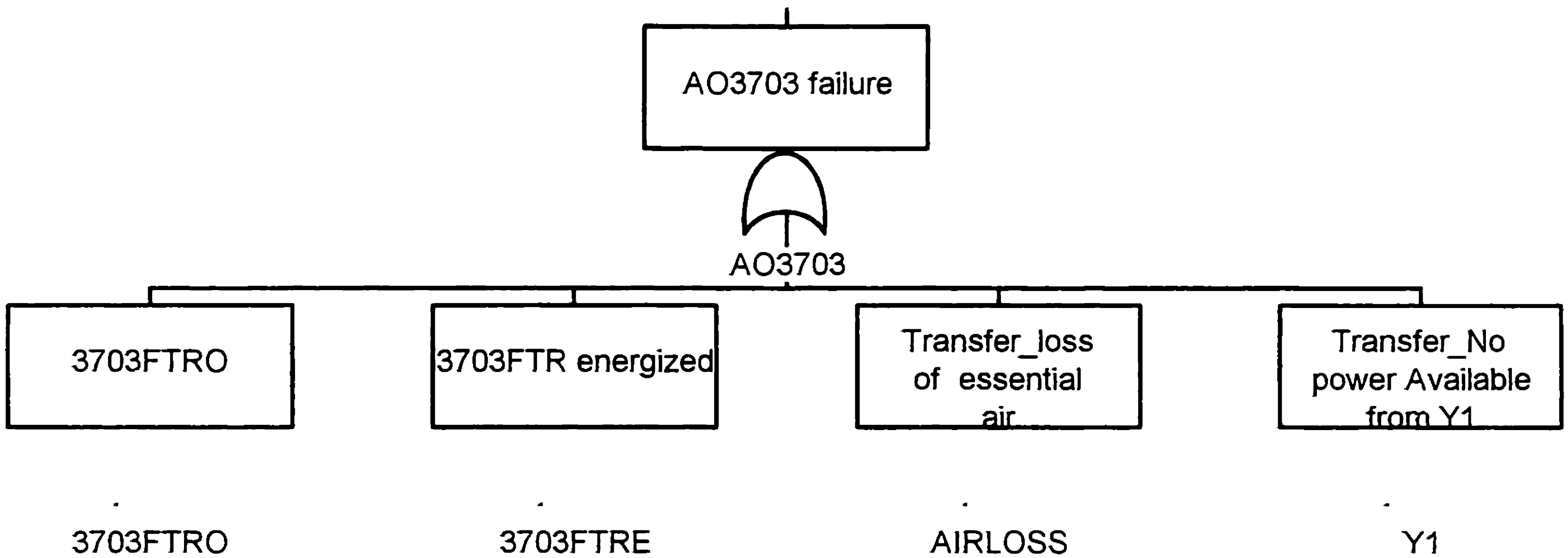
Applications"' Reliability Engineering and System Safety, Vol. 60, 1998

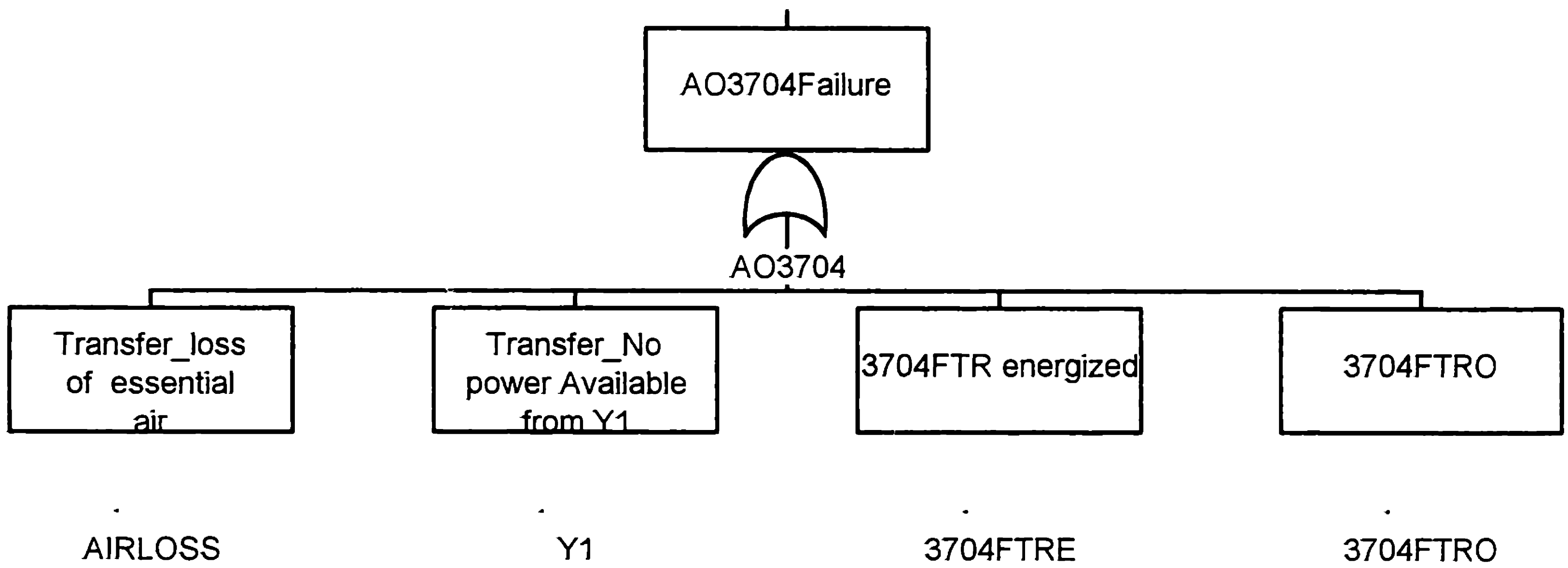
13. Lawrence M. Leemis, "Reliability; Probabilistic Models and Statistical Methods", Prentice-Hall, 1995

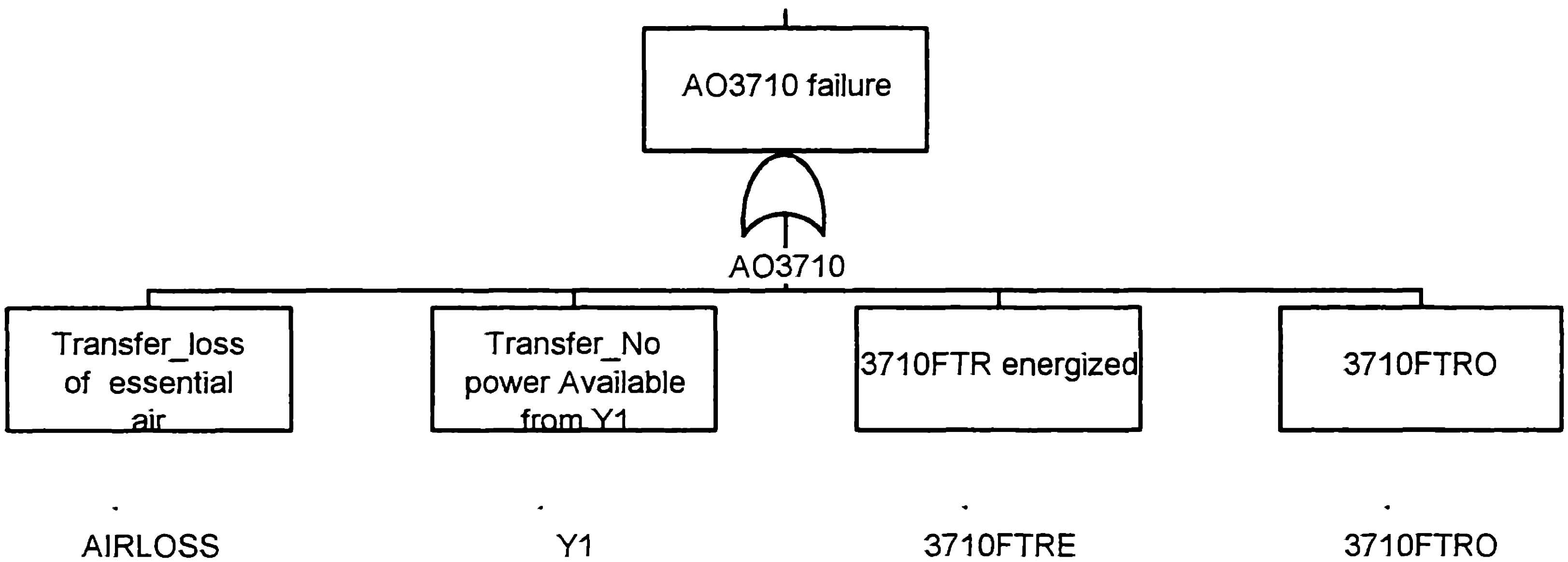
14. Renee M. Dubord, Michael W. Golay and Norman C. Rasmussen, "A Probabilistic Risk Assessment-Related Methodology to Support Performance-Based Regulation within the Nuclear Power Industry", Nuclear Technology Vol. 114, May. 1996

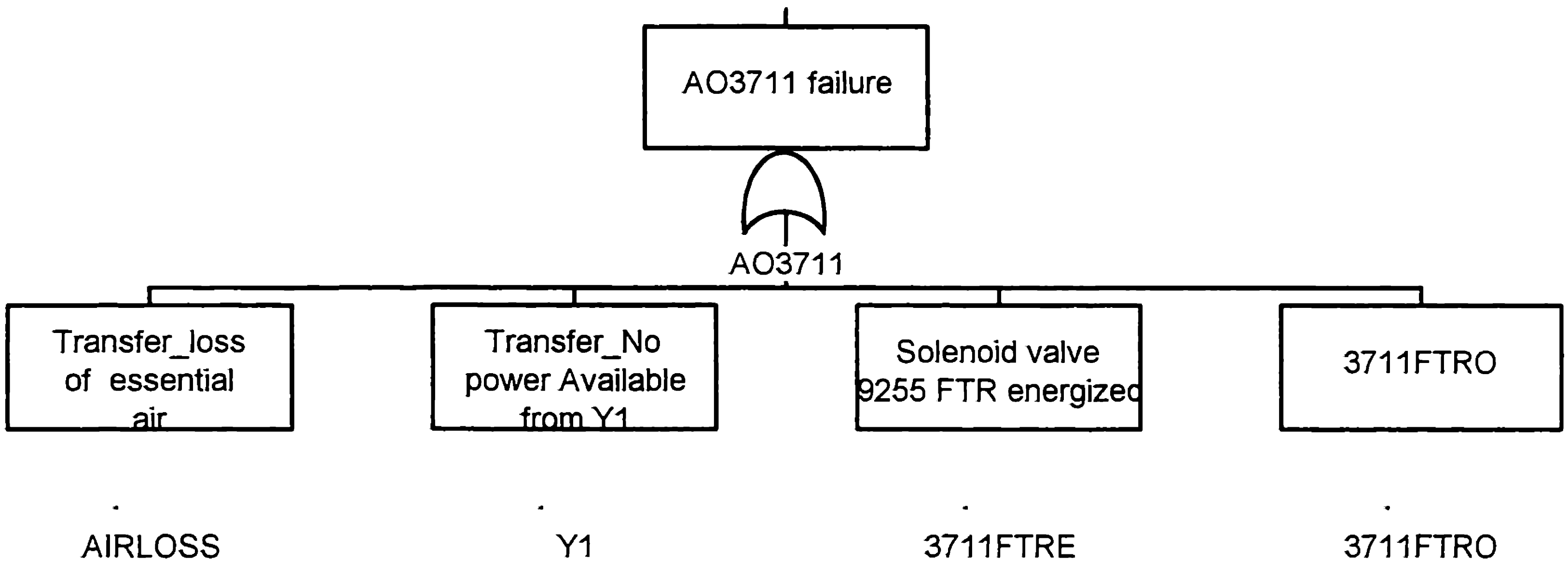
Appendix A

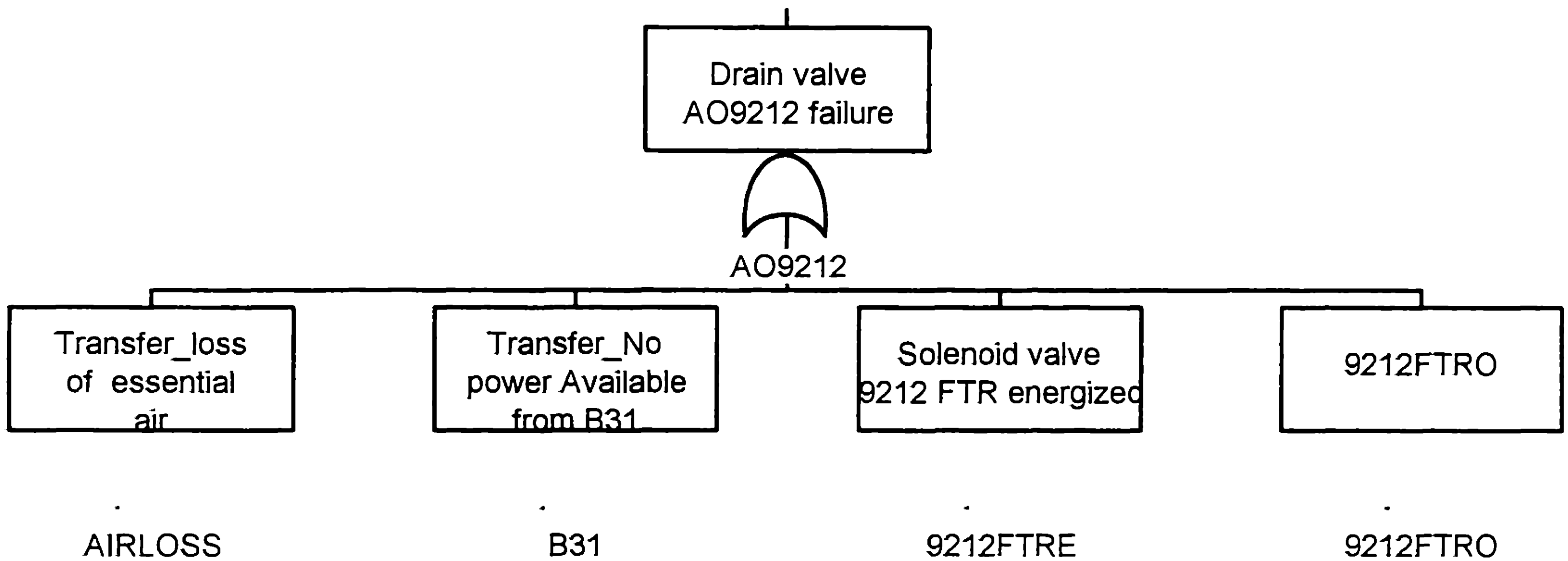
Fault Tree Models of the Balance of Plant System at the Pilgrim Nuclear Power Station

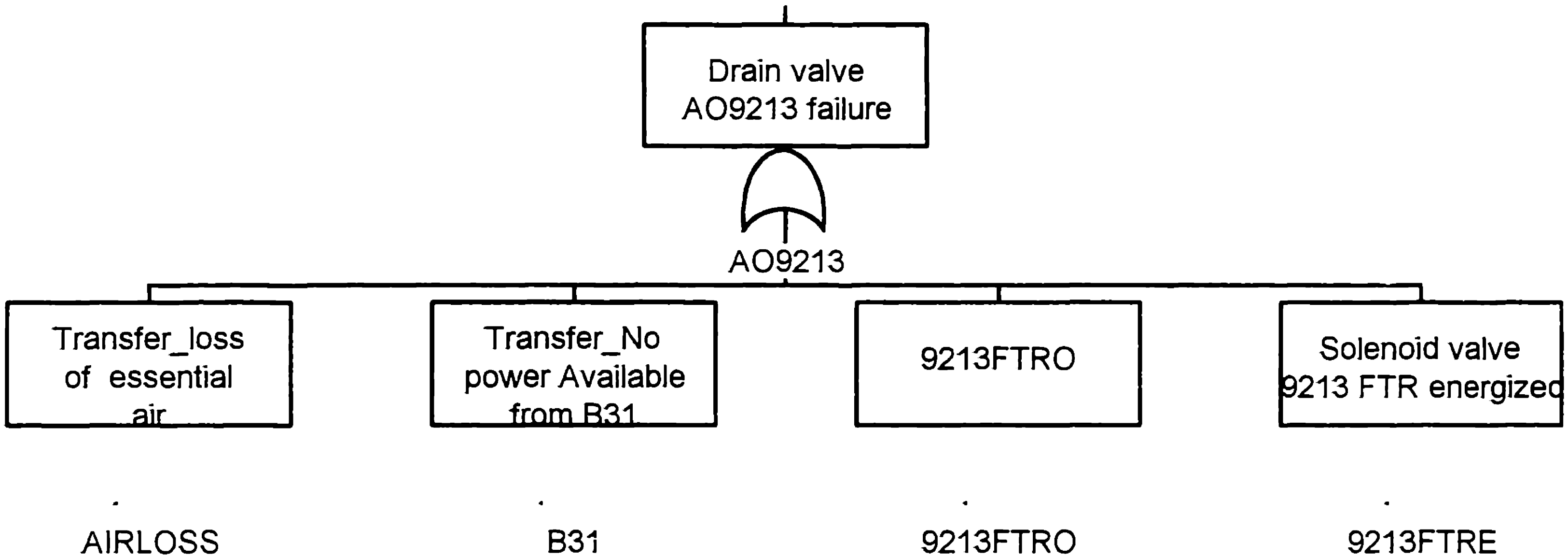

AO3703 failure
AO3703
3703FTRO
3703FTR energized
Transfer_loss of essential air
Transfer_No power Available from Y1
3703FTRO
3703FTRE
AIRLOSS
Y1

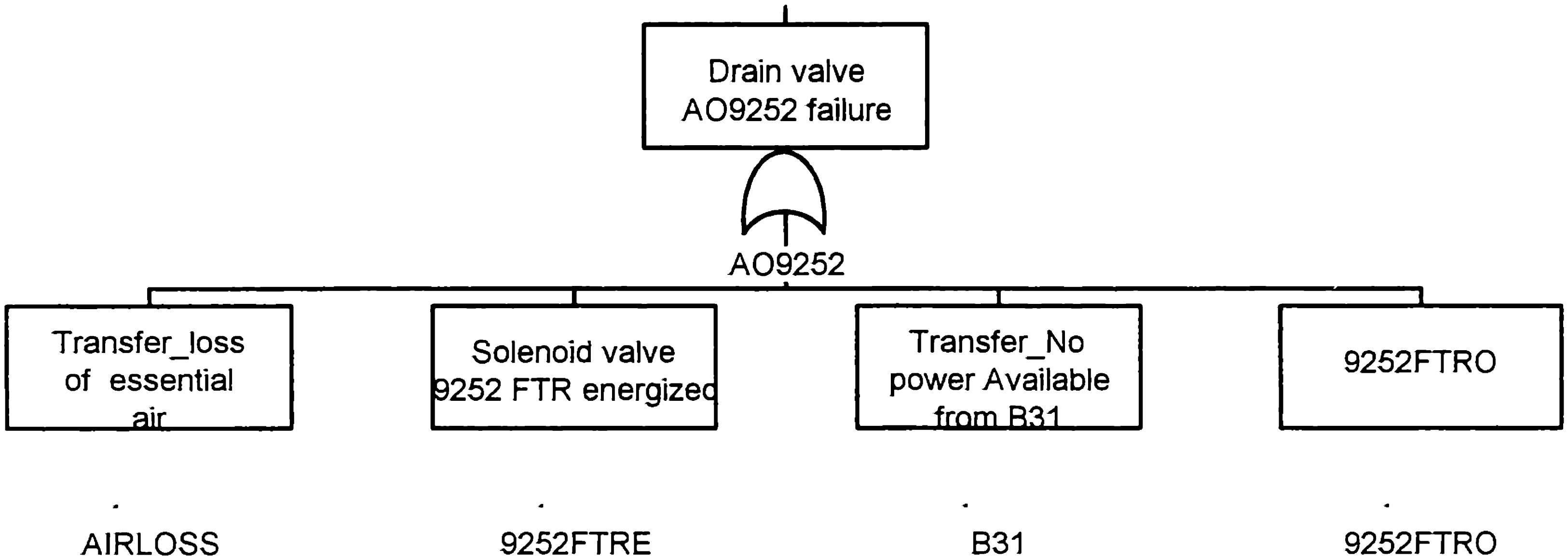

AO3704Failure
AO3704
Transfer_loss of essential air
Transfer_No power Available from Y1
3704FTR energized
3704FTRO
AIRLOSS
Y1
3704FTRE
3704FTRO


AO3710 failure
AO3710
Transfer_loss of essential air
Transfer_No power Available from Y1
3710FTR energized
3710FTRO
AIRLOSS
Y1
3710FTRE
3710FTRO

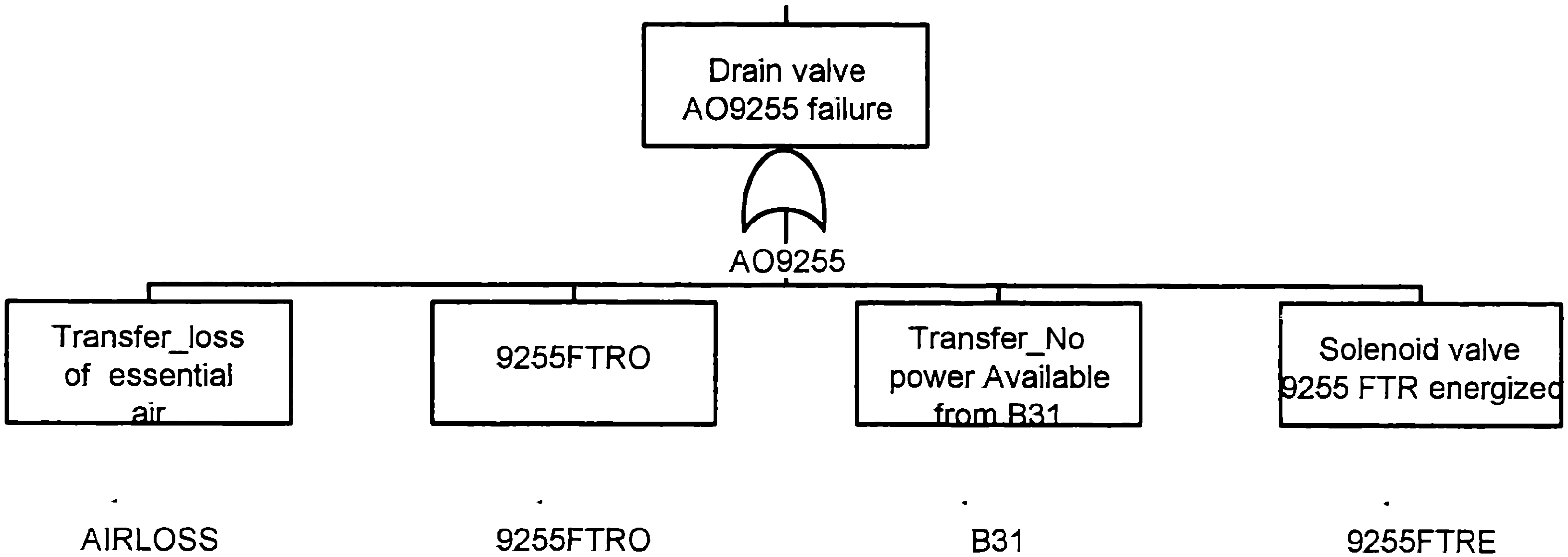

AO3711 failure
AO3711
Transfer_loss of essential air
Transfer_No power Available from Y1
Solenoid valve 9255 FTR energized
3711FTRO
AIRLOSS
Y1
3711FTRE
3711FTRO

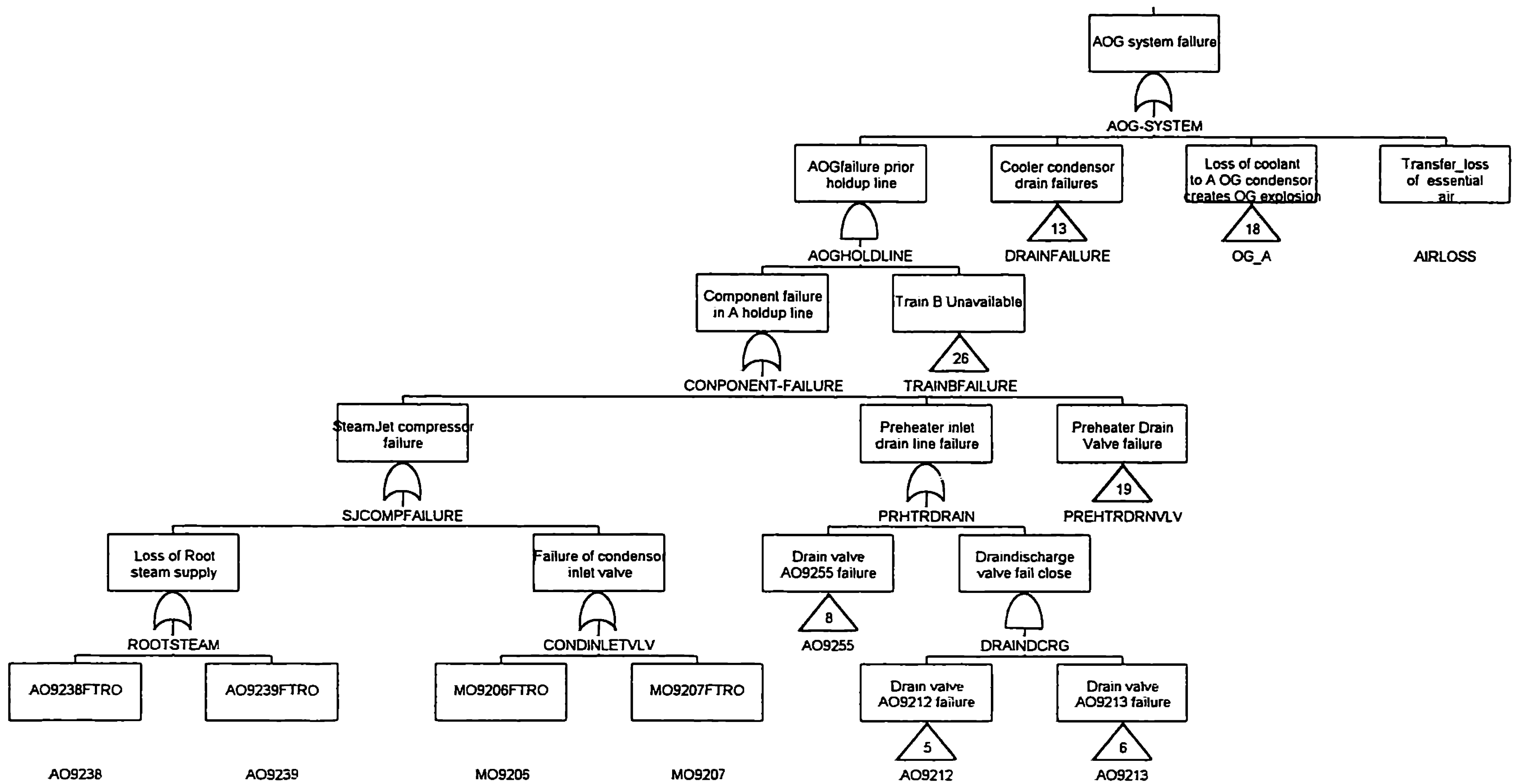

Drain valve
AO9212 failure
AO9212
Transfer_loss
of essential
air
Transfer_No
power Available
from B31
Solenoid valve
9212 FTR energized
9212FTRO
AIRLOSS
B31
9212FTRE
9212FTRO


Drain valve
AO9213 failure
AO9213
Transfer_loss
of essential
air
AIRLOSS
Transfer_No
power Available
from B31
B31
9213FTRO
9213FTRO
Solenoid valve
9213 FTR energized
9213FTRE

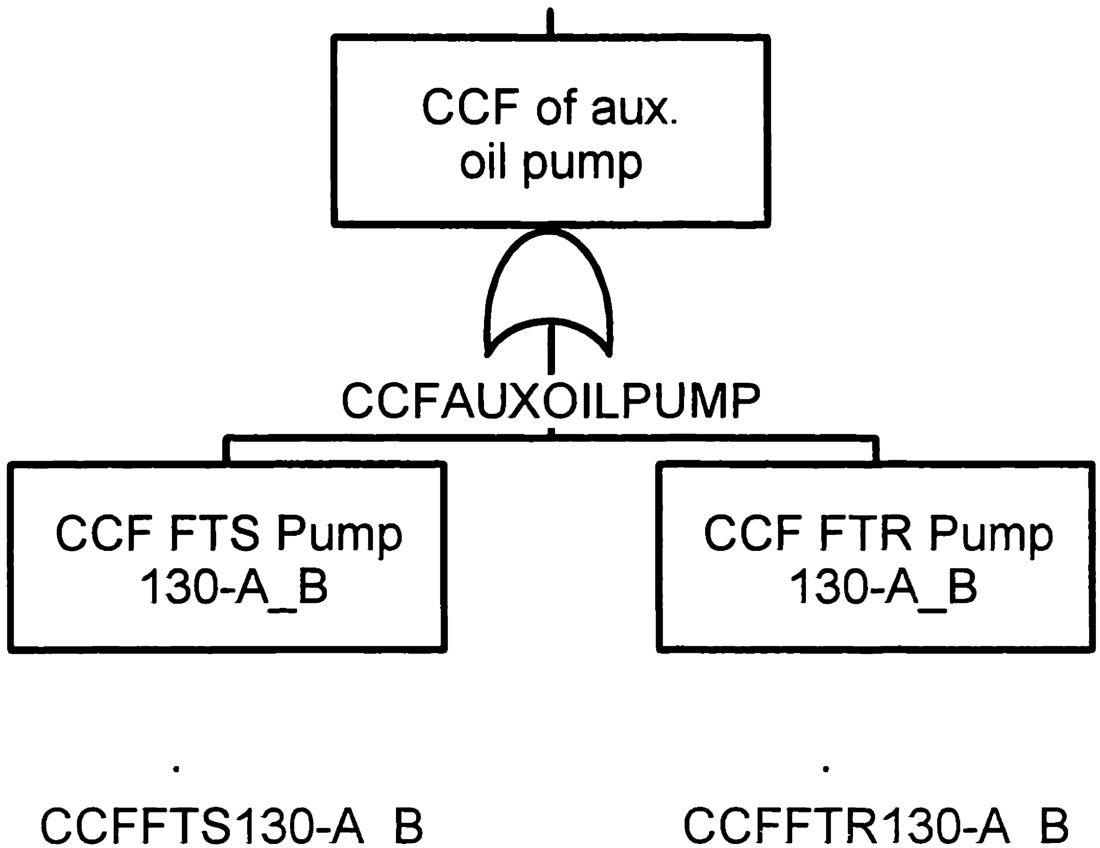


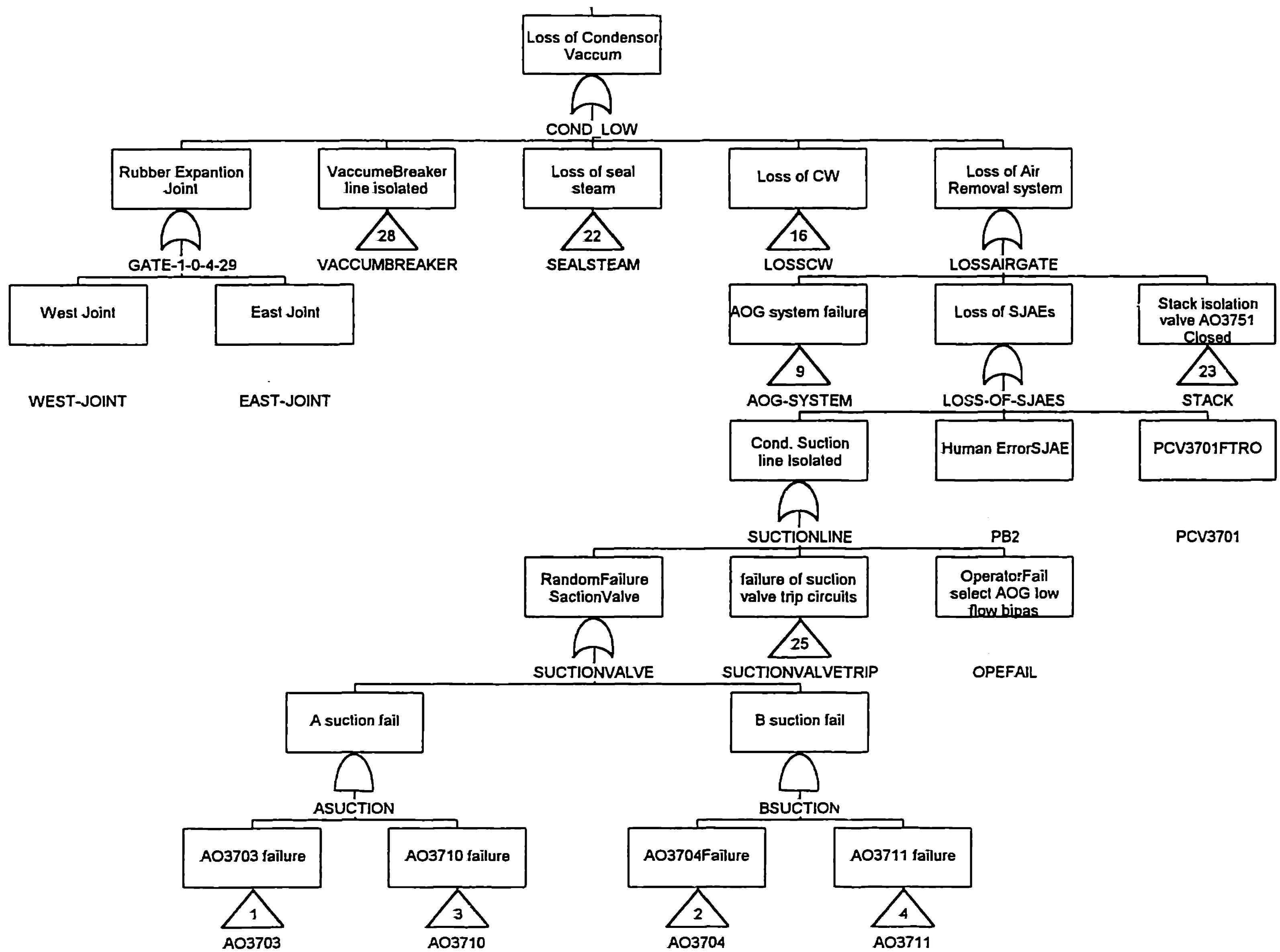
AO9252 - Drain valve AO9252 failure 1999/04/12 Page 7


Drain valve AO9255 failure
AO9255
Transfer_loss of essential air
AIRLOSS
9255FTRO
9255FTRO
Transfer_No power Available from B31
B31
Solenoid valve 9255 FTR energized
9255FTRE

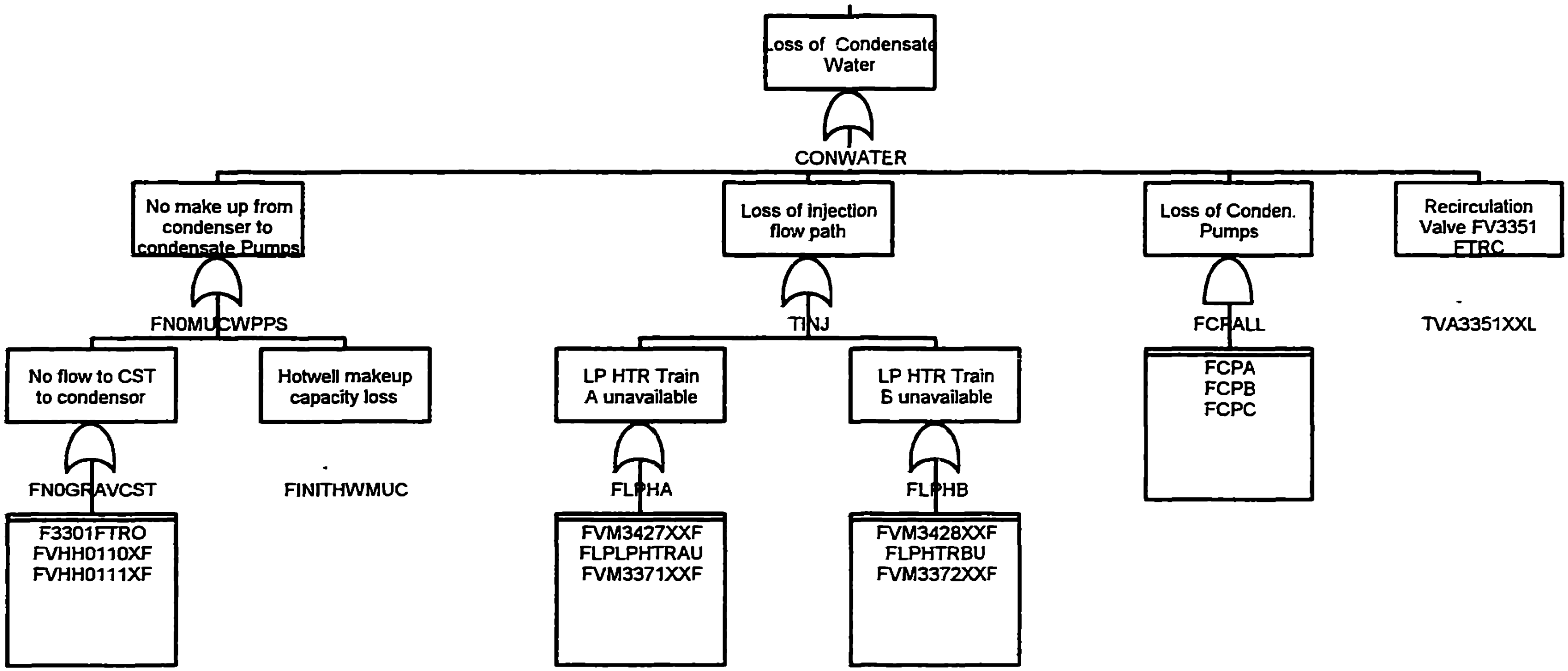


AOG-SYSTEM - AOG system failure 1999/04/12 Page 9

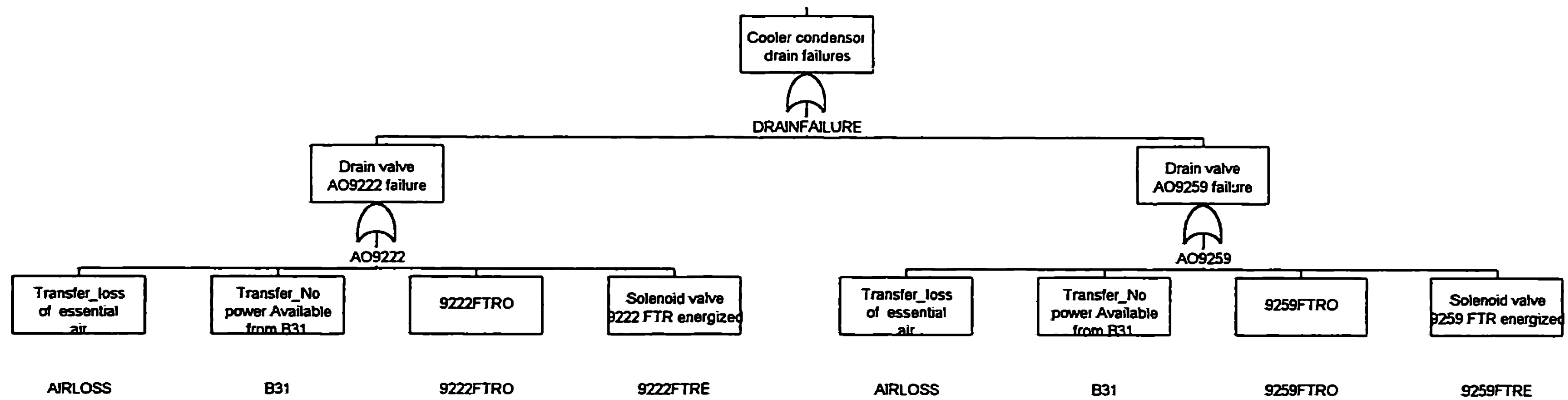

CCF of aux.
oil pump
CCFAUXOILPUMP
CCF FTS Pump
130-A_B
CCF FTR Pump
130-A_B
CCFFTS130-A_B
CCFFTR130-A_B



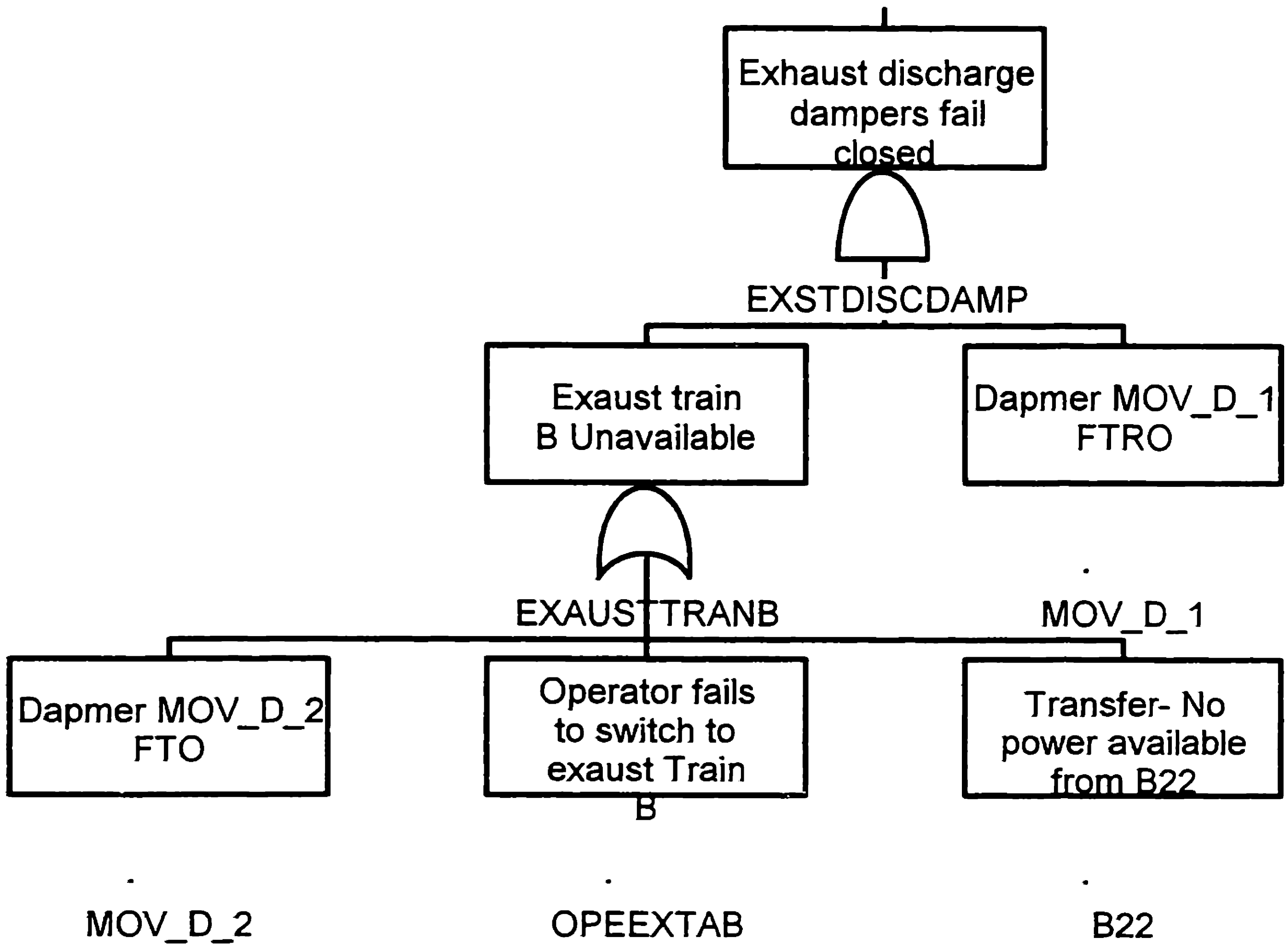
COND_LOW - Condensor Low_Vacuum 1993/02/21 Page 11

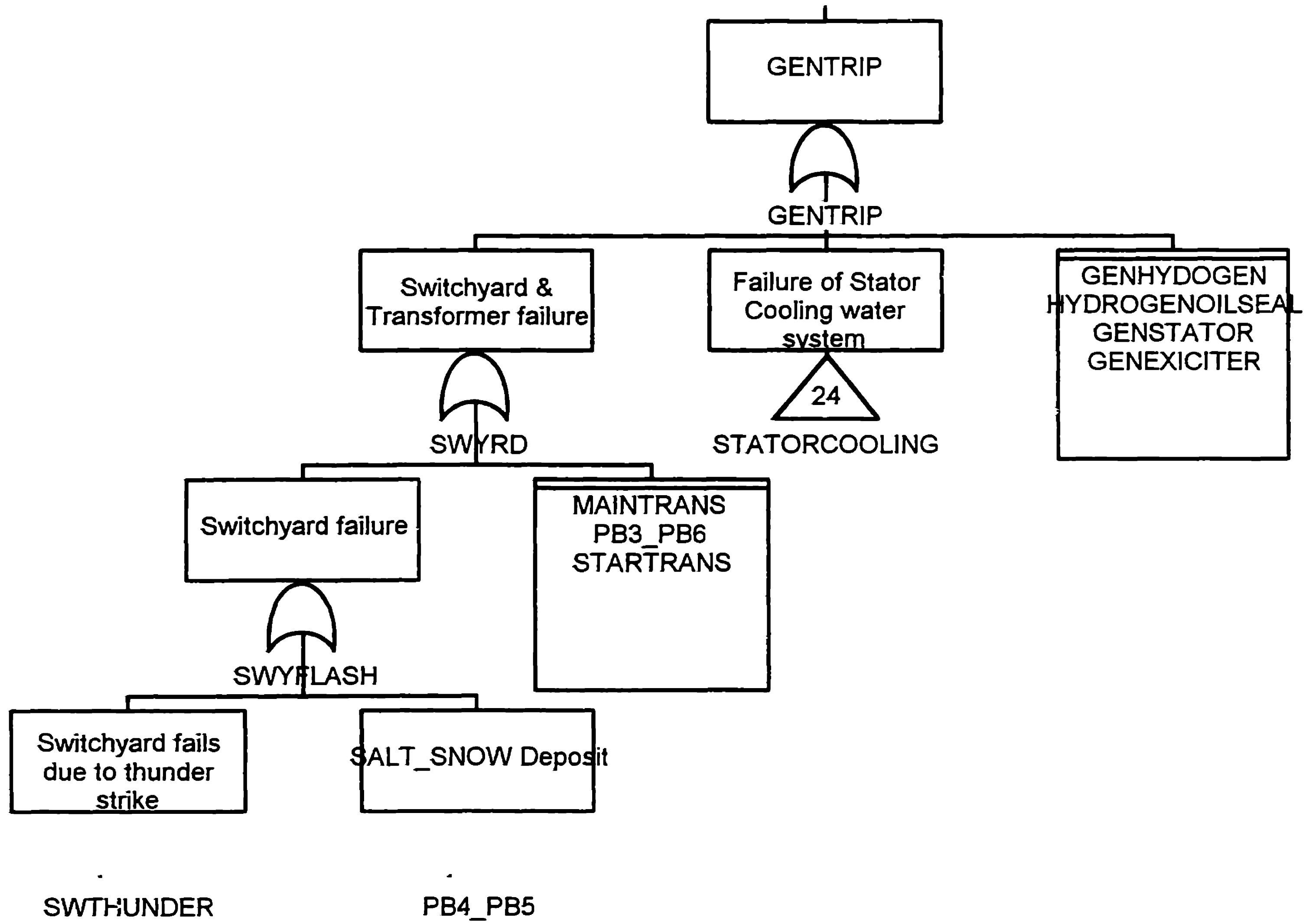


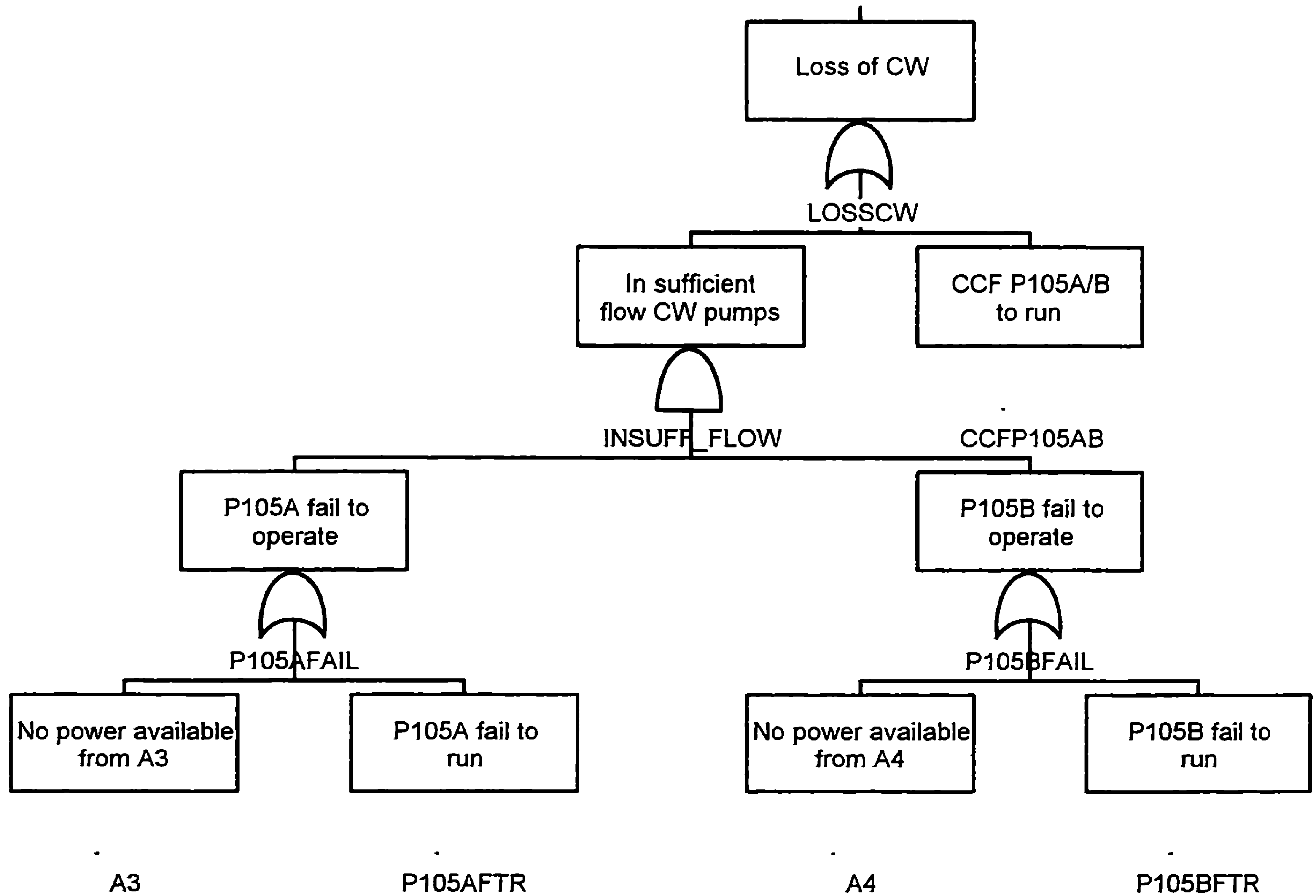
CONWATER - Condensate water system failure 1999/02/21 Page 12



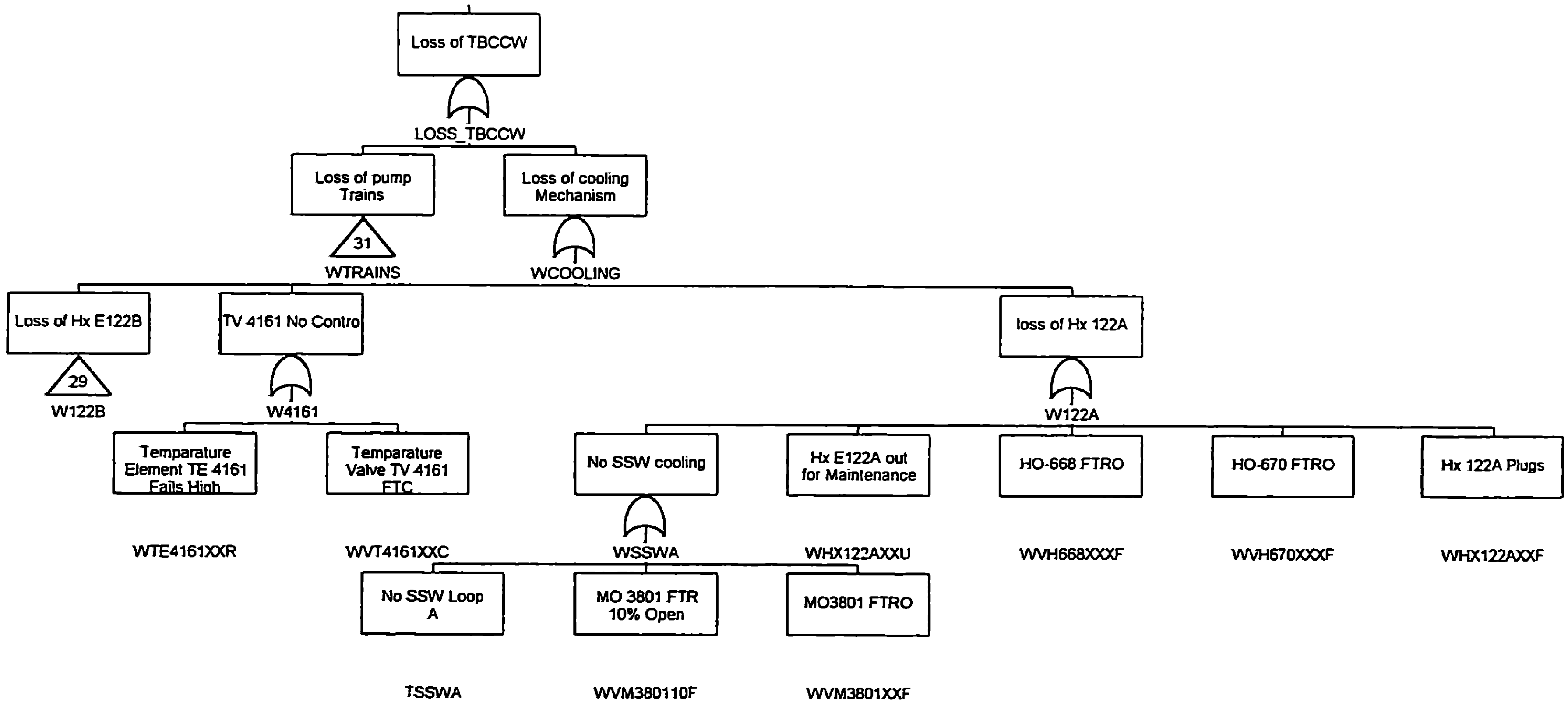
DRAINFAILURE - Cooler condensor drain failure 1999/02/21 Page 13

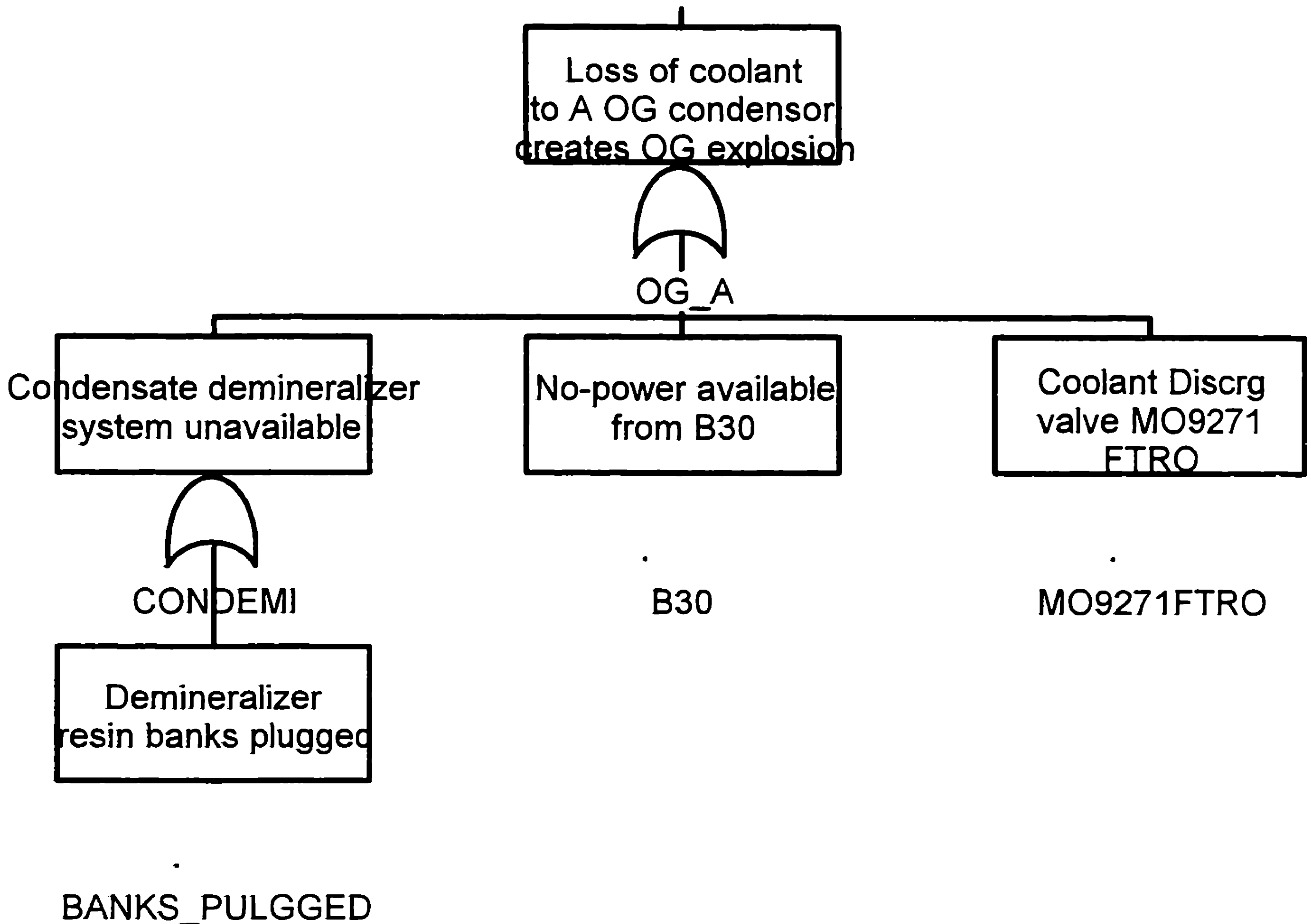

Exhaust discharge dampers fail closed
EXSTDISCDAMP
Exaust train B Unavailable
Dapmer MOV_D_1 FTRO
EXAUSTTRANB
MOV_D_1
Dapmer MOV_D_2 FTO
Operator fails to switch to exaust Train B
Transfer- No power available from B22
MOV_D_2
OPEEXTAB
B22

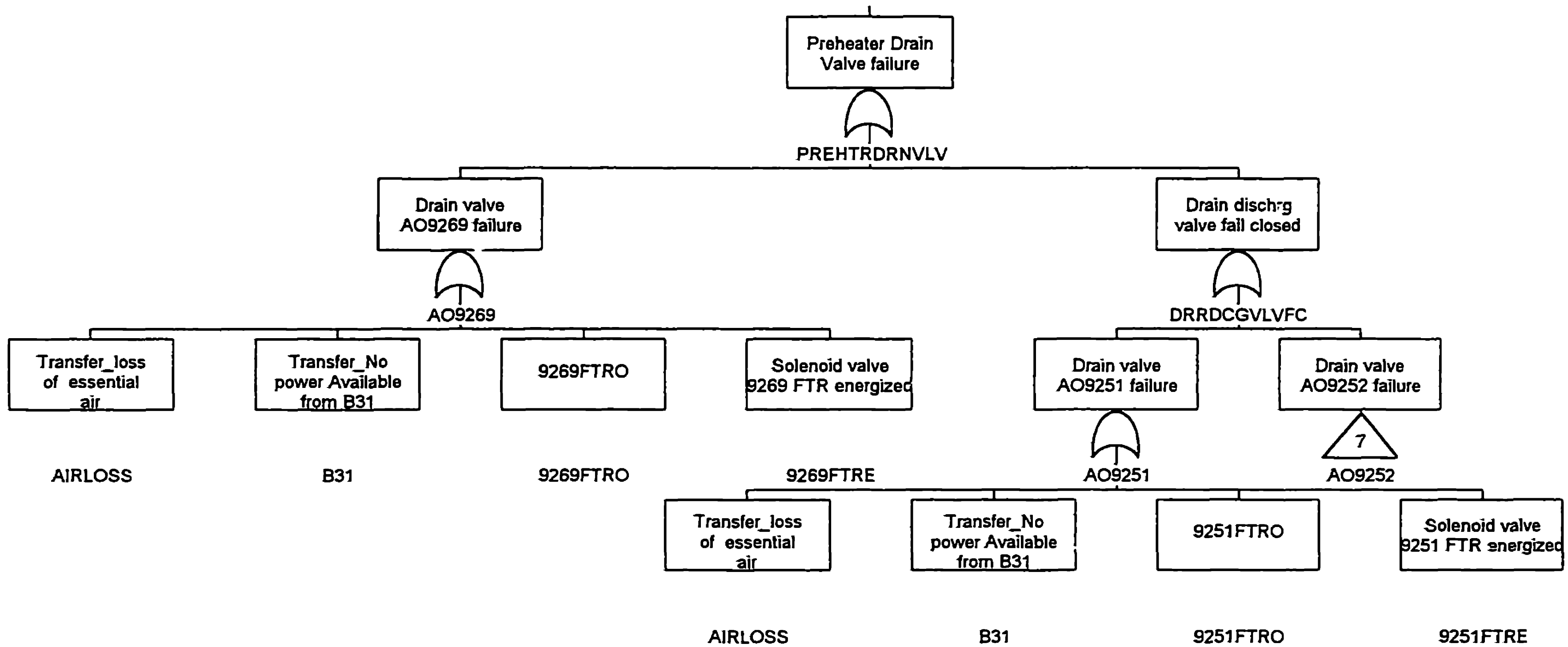

GENTRIP
GENTRIP
Switchyard & Transformer failure
Failure of Stator Cooling water system
GENHYDOGEN
HYDROGENOILSEAL
GENSTATOR
GENEXICITER
24
STATORCOOLING
SWYRD
Switchyard failure
MAINTRANS
PB3_PB6
STARTRANS
SWYFLASH
Switchyard fails due to thunder strike
SALT_SNOW Deposit
SWTHUNDER
PB4_PB5



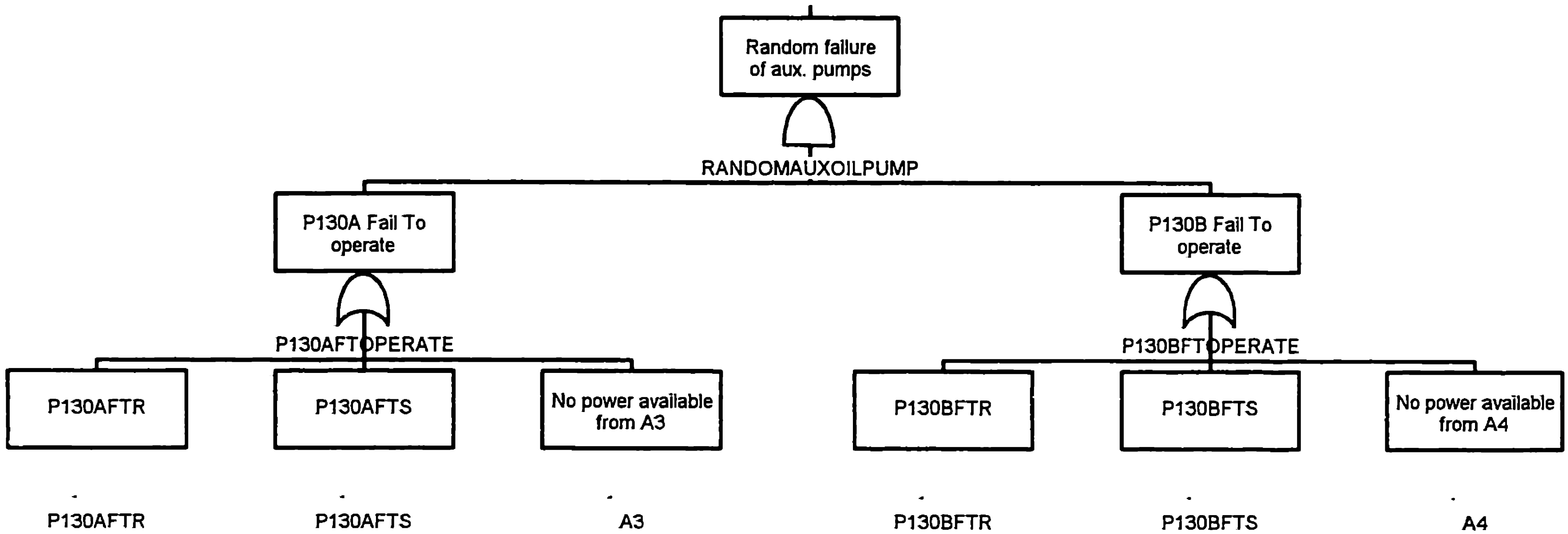
LOSSCW - Loss of CW 1999/02/21 Page 16


Loss of TBCCW
LOSS_TBCCW
Loss of pump Trains
31
WTRAINS
Loss of cooling Mechanism
WCOOLING
Loss of Hx E122B
29
W122B
TV 4161 No Contro
W4161
loss of Hx 122A
W122A
Temparature Element TE 4161 Fails High
WTE4161XXR
Temparature Valve TV 4161 FTC
WVT4161XXC
No SSW cooling
WSSWA
Hx E122A out for Maintenance
WHX122AXXU
HO-668 FTRO
WVH668XXXF
HO-670 FTRO
WVH670XXXF
Hx 122A Plugs
WHX122AXXF
No SSW Loop A
TSSWA
MO 3801 FTR 10% Open
WVM380110F
MO3801 FTRO
WVM3801XXF

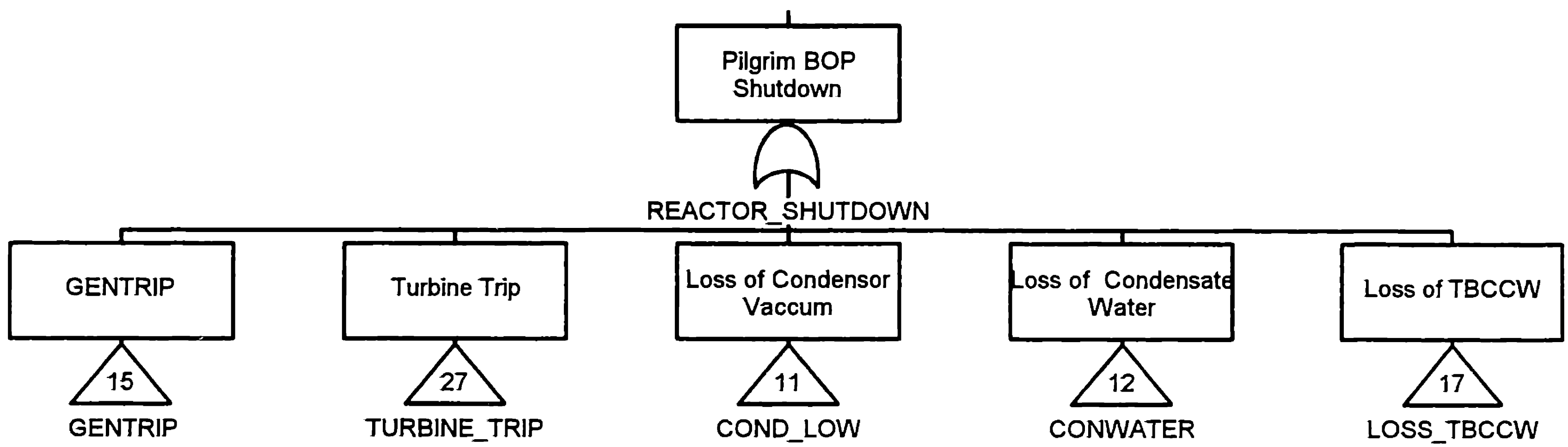

Loss of coolant to A OG condensor creates OG explosion
OG_A
Condensate demineralizer system unavailable
No-power available from B30
Coolant Discrg valve MO9271 FTRO
CONDEMI
B30
MO9271FTRO
Demineralizer resin banks plugged
BANKS_PULGGED



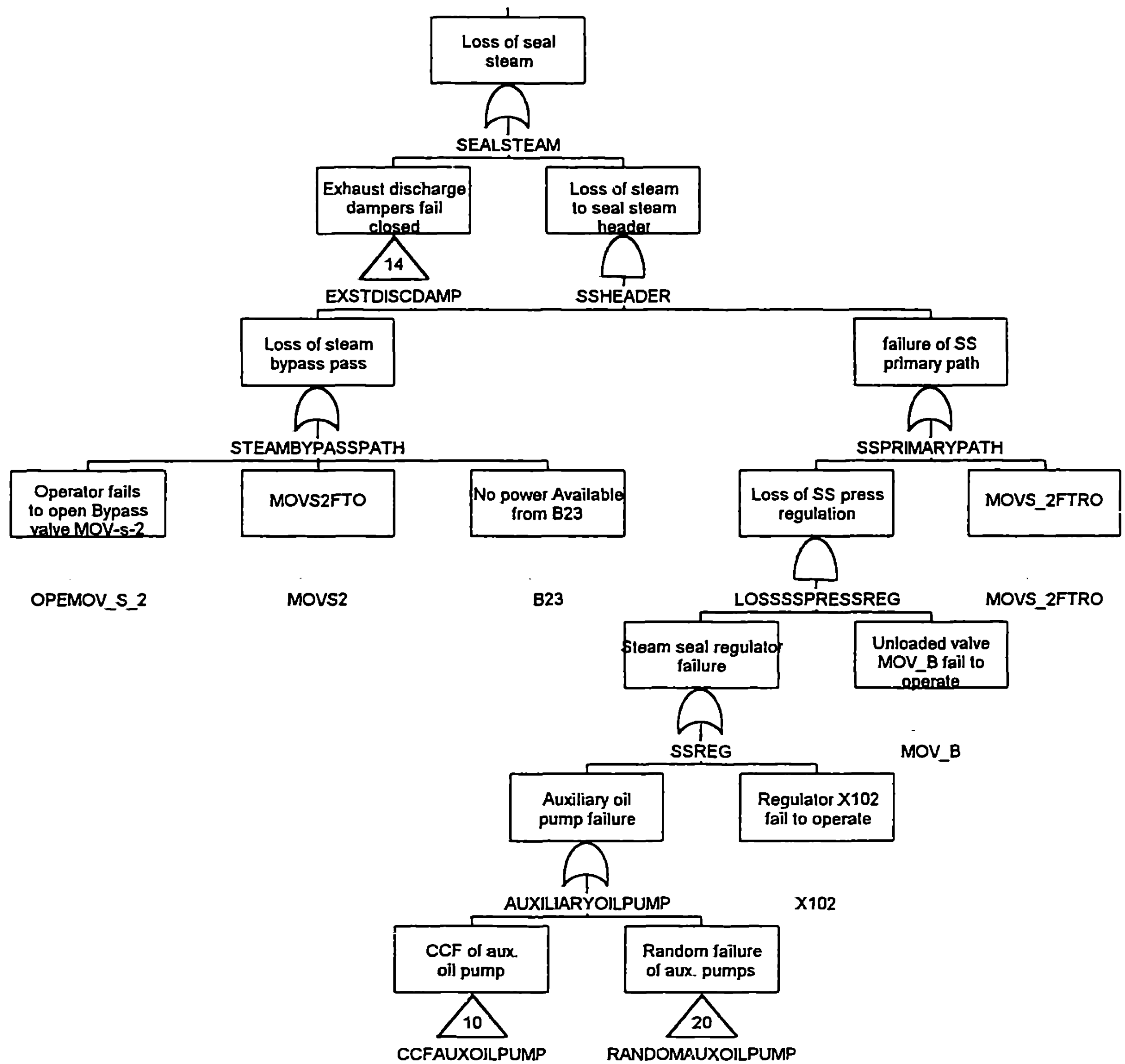
PREHTRDRNVLV - Preheater inlet drain valve failure 1999/02/21 Page 19

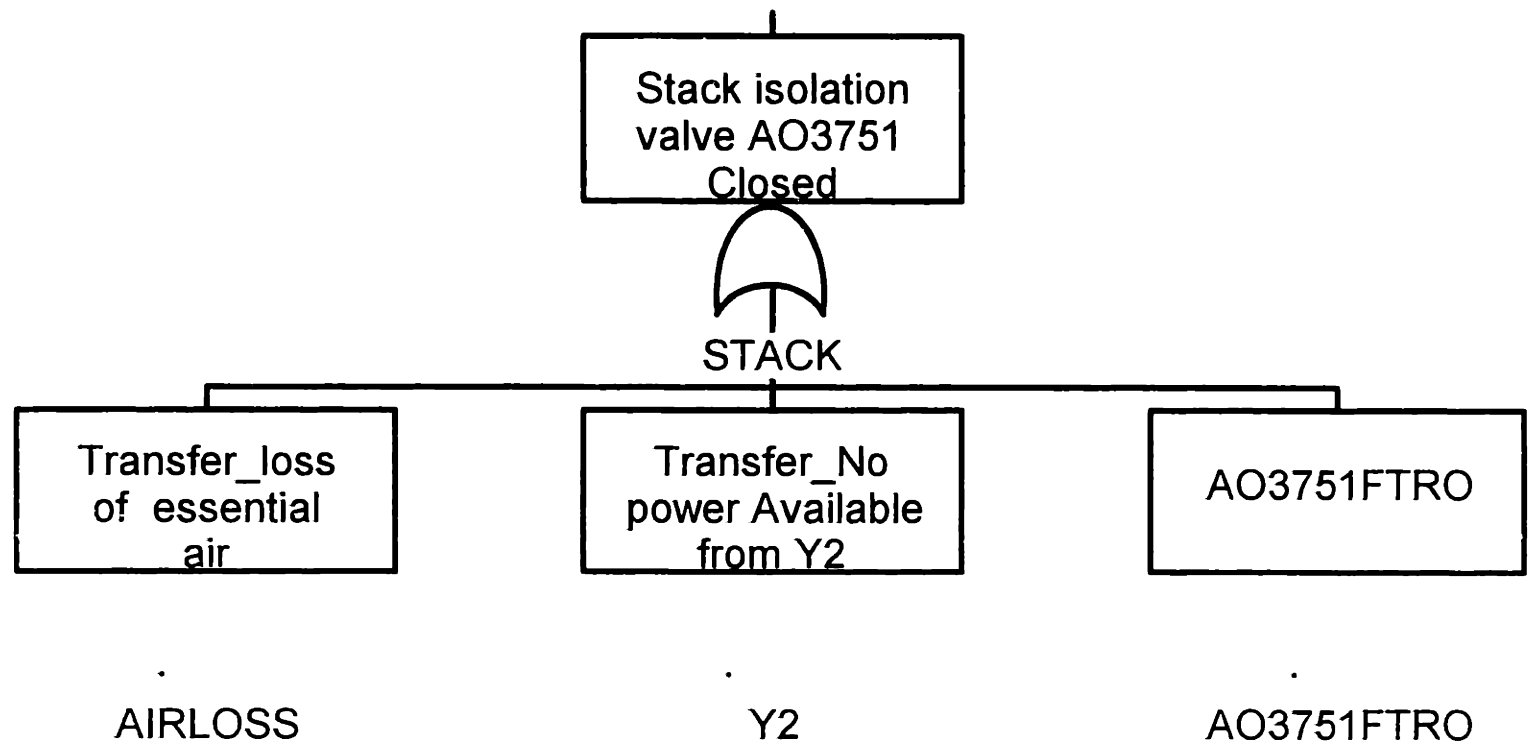


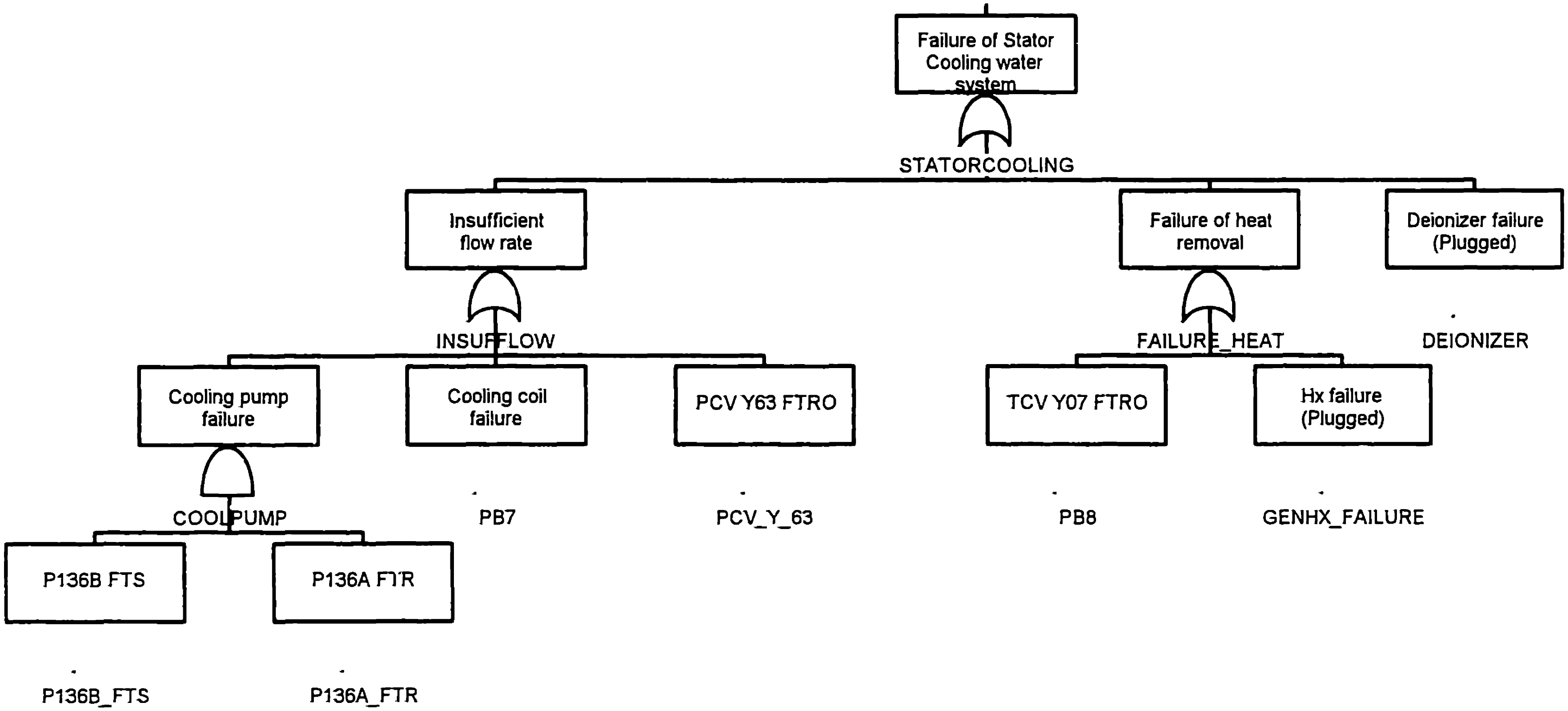
RANDOMAUXOILPUMP - Random failure of aux oil pump 1999/02/21 Page 20

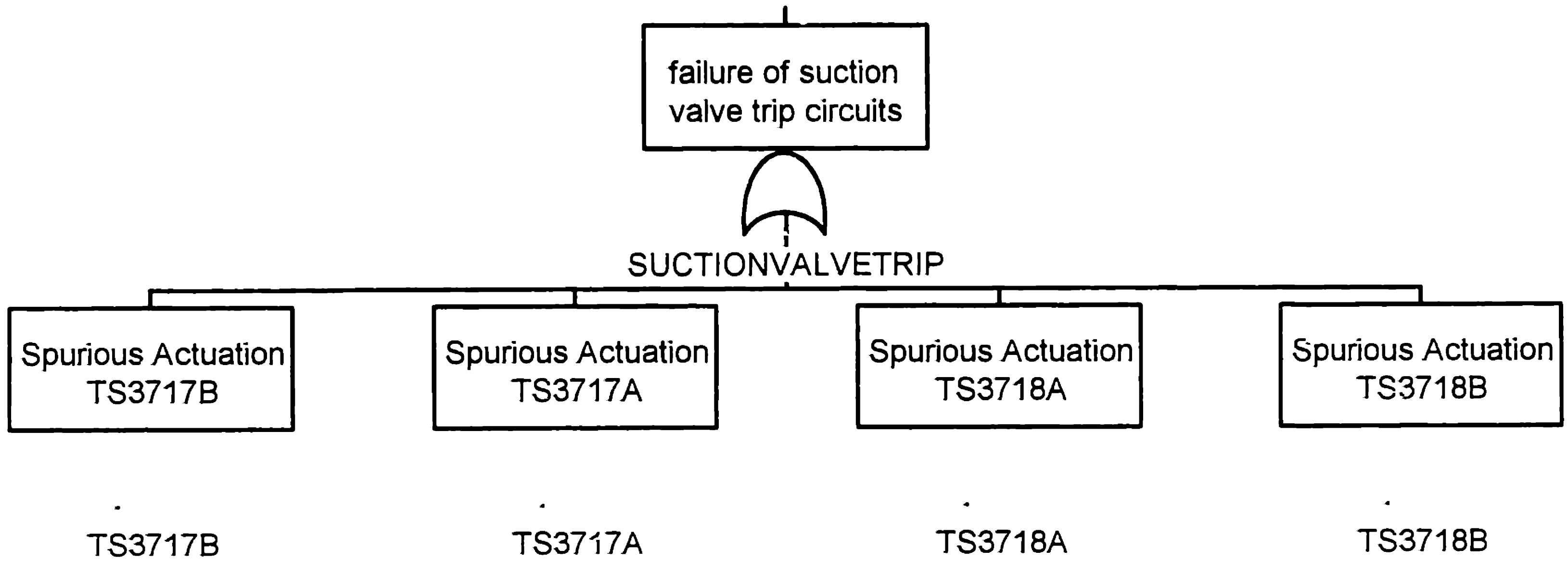


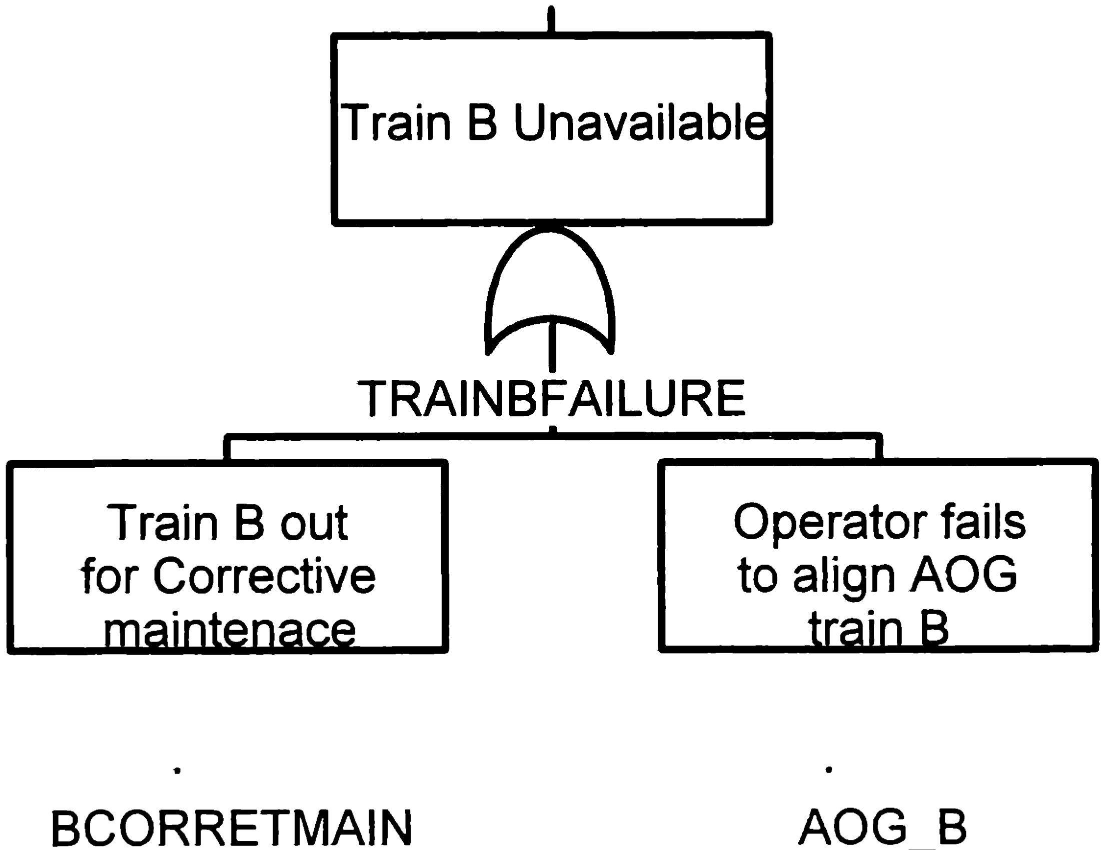
REACTOR_SHUTDOWN - Reactor_Shutdown 1999/04/12 Page 21

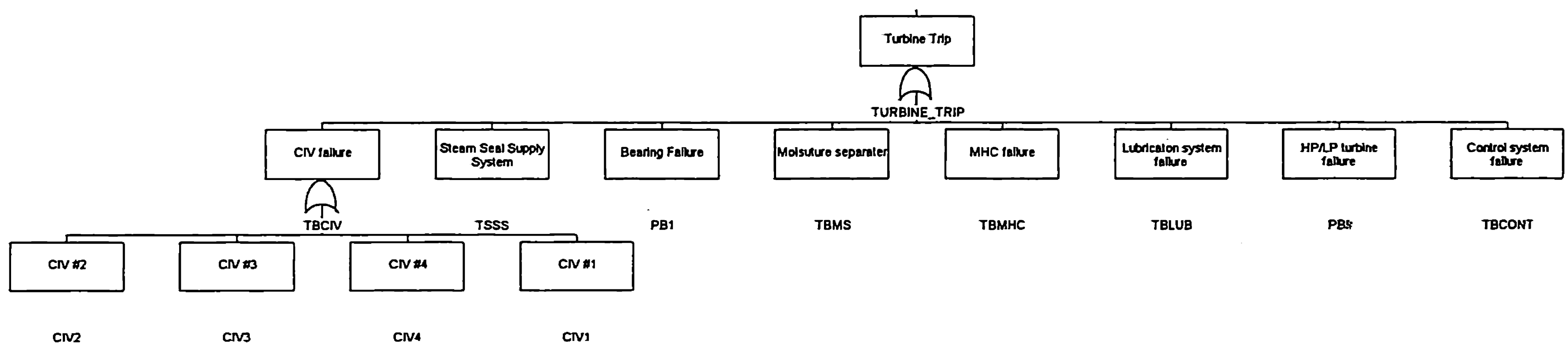

Loss of seal steam
SEALSTEAM
Exhaust discharge dampers fail closed
14
EXSTDISCDAMP
Loss of steam to seal steam header
SSHEADER
Loss of steam bypass pass
STEAMBYPASSPATH
Operator fails to open Bypass valve MOV-s-2
OPEMOV_S_2
MOVS2FTO
MOVS2
No power Available from B23
B23
failure of SS primary path
SSPRIMARYPATH
Loss of SS press regulation
LOSSSSPRESSREG
MOVS_2FTRO
MOVS_2FTRO
Steam seal regulator failure
SSREG
Unloaded valve MOV_B fail to operate
MOV_B
Auxiliary oil pump failure
AUXILIARYOILPUMP
Regulator X102 fail to operate
X102
CCF of aux. oil pump
10
CCFAUXOILPUMP
Random failure of aux. pumps
20
RANDOMAUXOILPUMP


Stack isolation valve AO3751 Closed
STACK
Transfer_loss of essential air
AIRLOSS
Transfer_No power Available from Y2
Y2
AO3751FTRO
AO3751FTRO

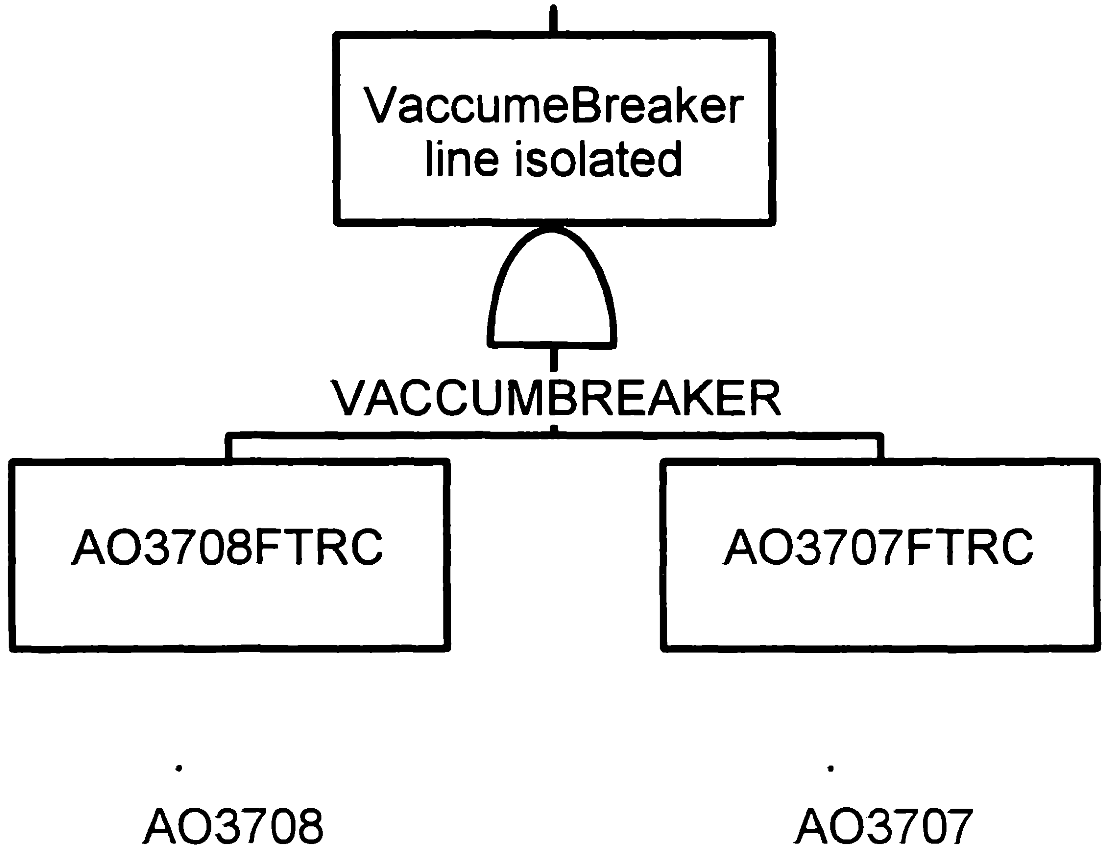

Failure of Stator Cooling water system
STATORCOOLING
Insufficient flow rate
Failure of heat removal
Deionizer failure (Plugged)
INSUFFLOW
FAILURE_HEAT
DEIONIZER
Cooling pump failure
Cooling coil failure
PCV Y63 FTRO
TCV Y07 FTRO
Hx failure (Plugged)
COOLPUMP
PB7
PCV_Y_63
PB8
GENHX_FAILURE
P136B FTS
P136A FTR
P136B_FTS
P136A_FTR

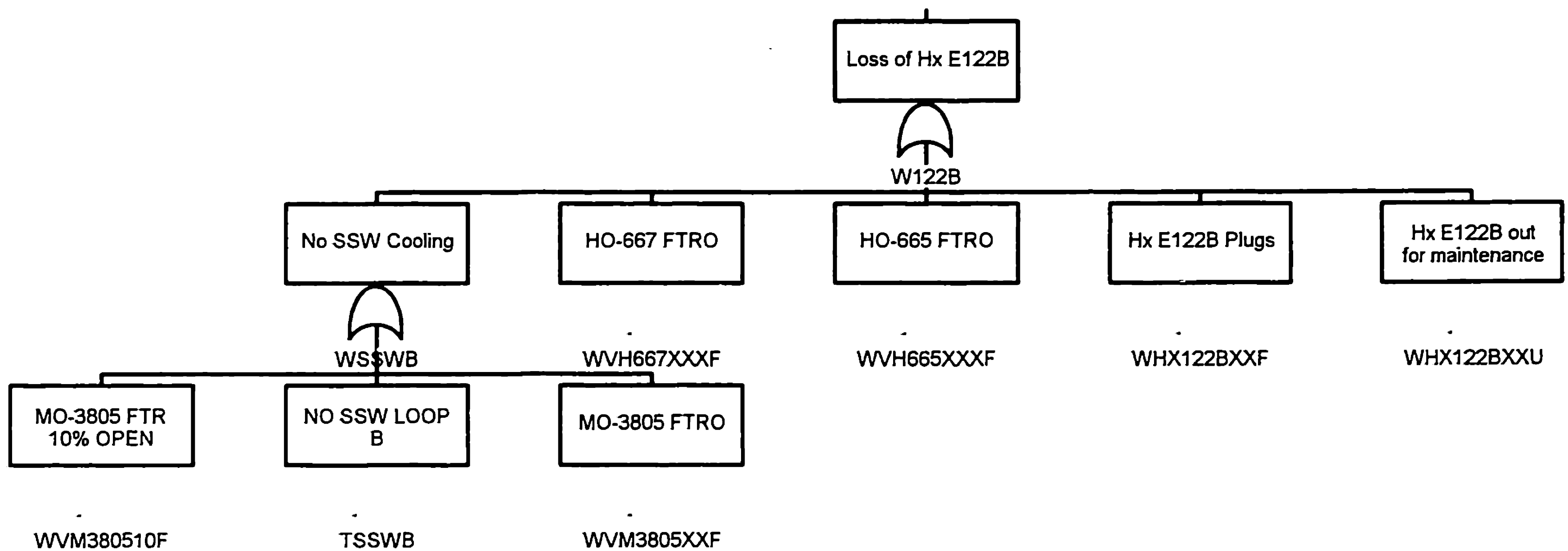

failure of suction valve trip circuits
SUCTIONVALVETRIP
Spurious Actuation TS3717B
Spurious Actuation TS3717A
Spurious Actuation TS3718A
Spurious Actuation TS3718B
TS3717B
TS3717A
TS3718A
TS3718B

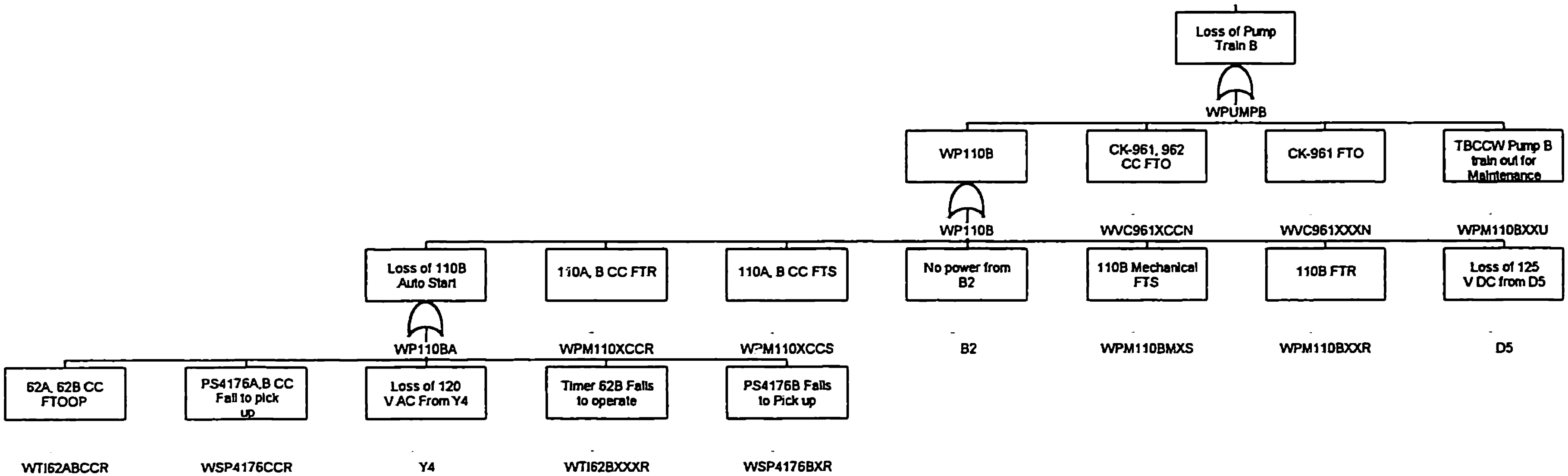

Train B Unavailable
TRAINBFAILURE
Train B out for Corrective maintenace
Operator fails to align AOG train B
BCORRETMAIN
AOG_B

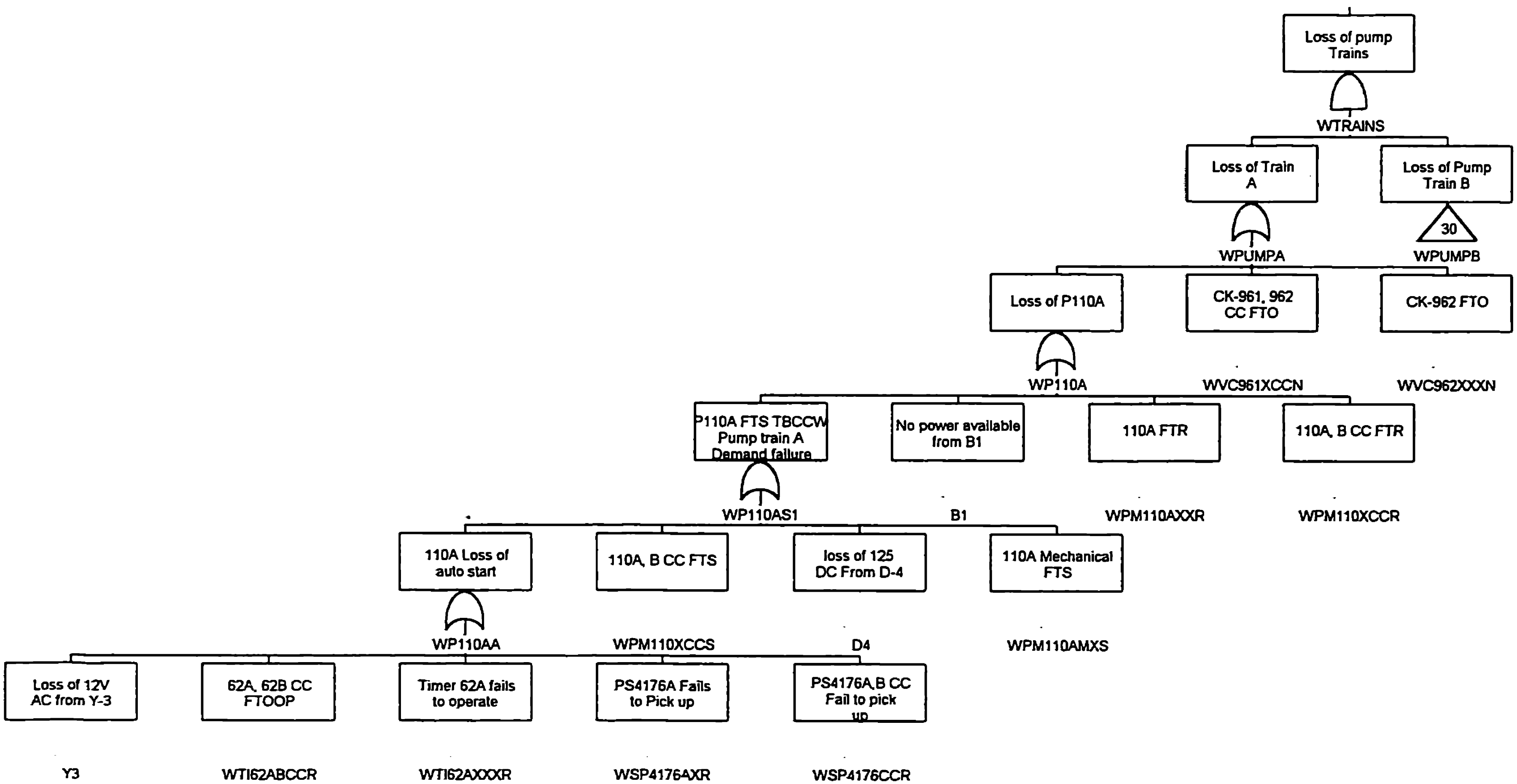


TURBINE_TRIP - Turbine Trip TCV closure 1999/02/21 Page 27


VaccumeBreaker line isolated
VACCUMBREAKER
AO3708FTRC
AO3707FTRC
AO3708
AO3707


Loss of Hx E122B
W122B
No SSW Cooling
HO-667 FTRO
HO-665 FTRO
Hx E122B Plugs
Hx E122B out for maintenance
WSSWB
WVH667XXXF
WVH665XXXF
WHX122BXXF
WHX122BXXU
MO-3805 FTR 10% OPEN
NO SSW LOOP B
MO-3805 FTRO
WVM380510F
TSSWB
WVM3805XXF

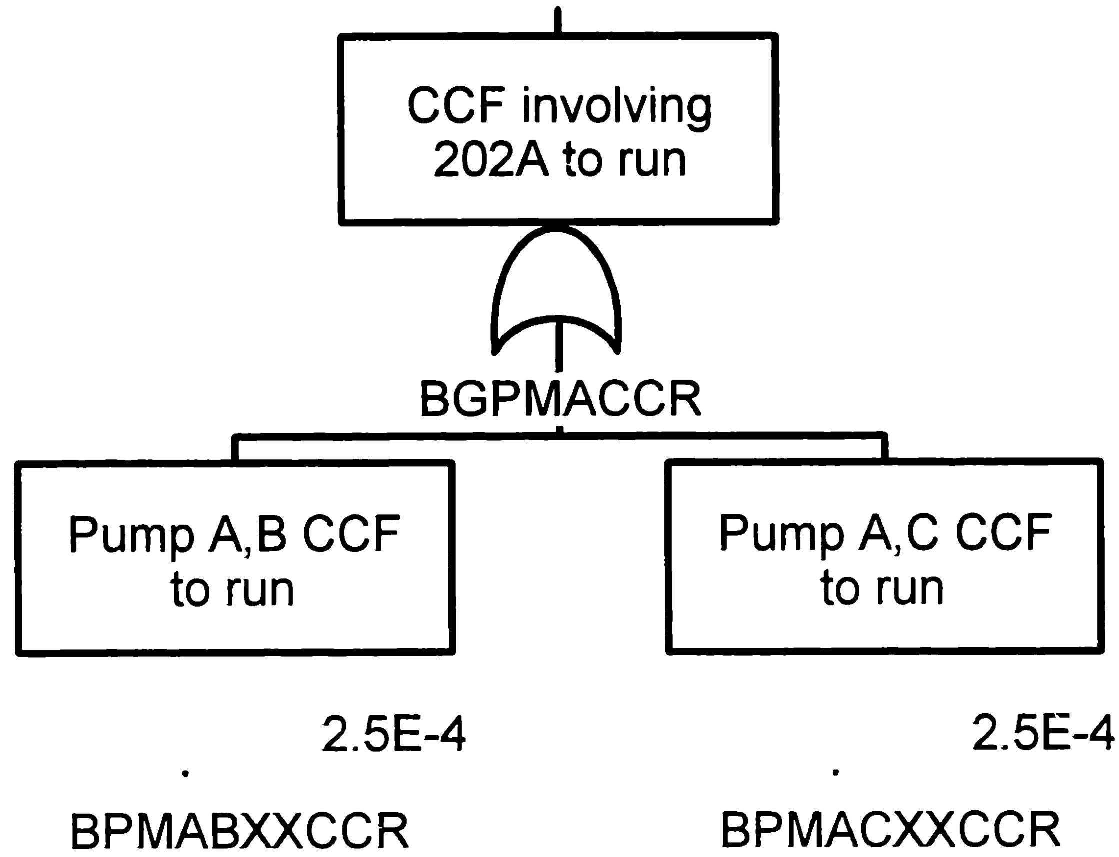

Loss of Pump Train B
WPUMPB
WP110B
CK-961, 962 CC FTO
CK-961 FTO
TBCCW Pump B train out for Maintenance
WP110B
WVC961XCCN
WVC961XXXN
WPM110BXXU
Loss of 110B Auto Start
110A, B CC FTR
110A, B CC FTS
No power from B2
110B Mechanical FTS
110B FTR
Loss of 125 V DC from D5
WP110BA
WPM110XCCR
WPM110XCCS
B2
WPM110BMXS
WPM110BXXR
D5
62A, 62B CC FTOOP
PS4176A,B CC Fail to pick up
Loss of 120 V AC From Y4
Timer 62B Fails to operate
PS4176B Fails to Pick up
WTI62ABCCR
WSP4176CCR
Y4
WTI62BXXXR
WSP4176BXR



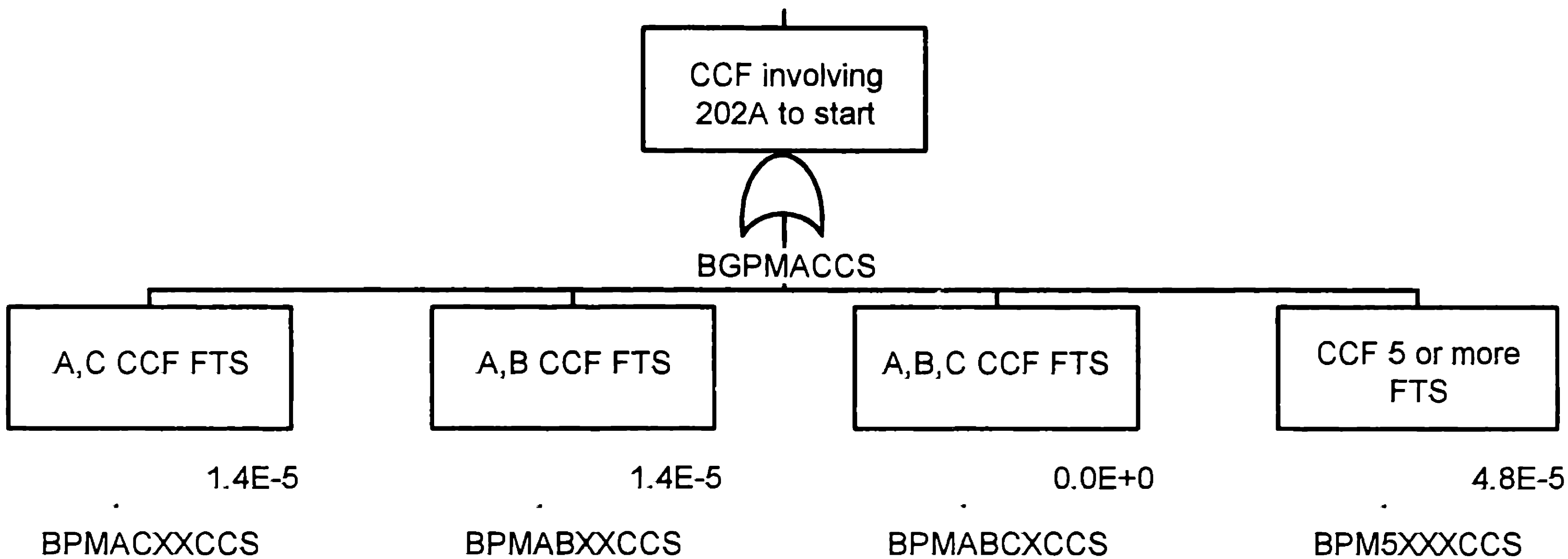
WTRAINS - Loss of pump trains 1999/02/21 Page 31

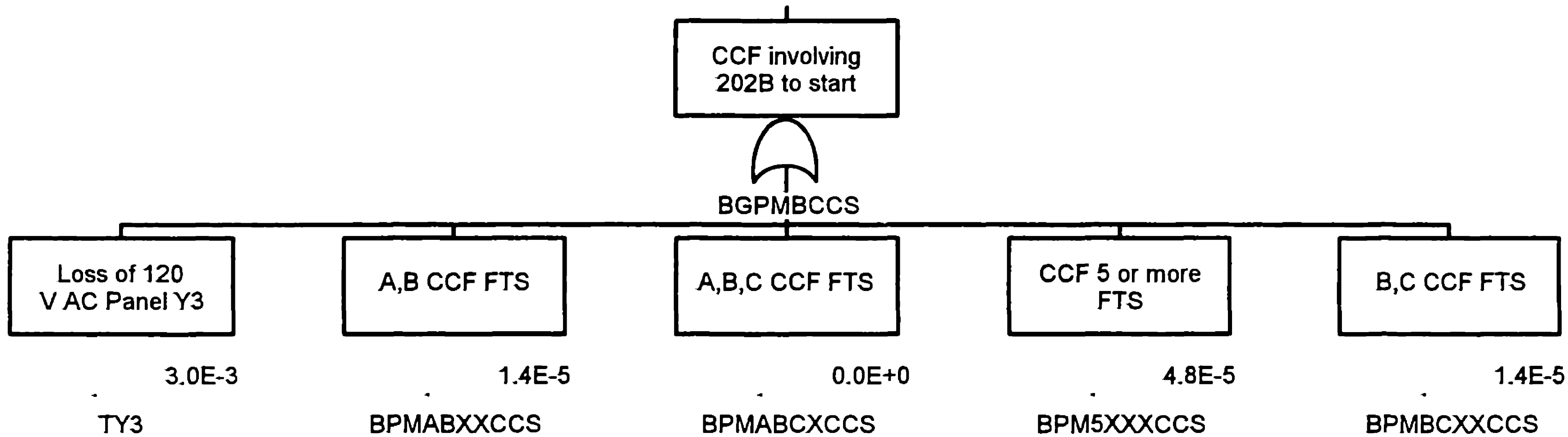
Appendix B

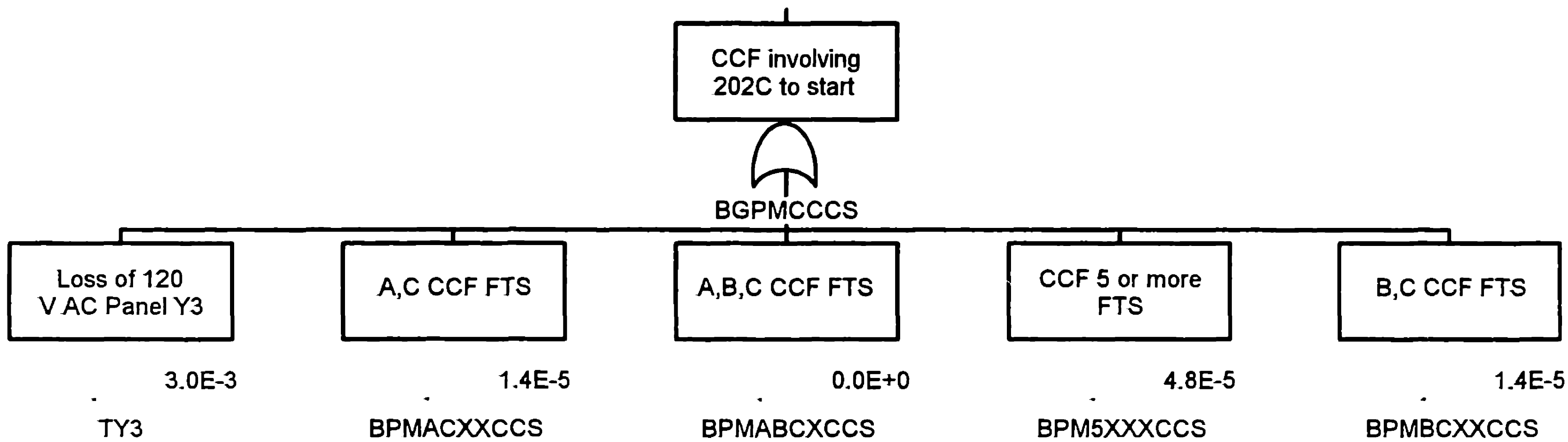
Fault Tree Models of the Nuclear Steam Supply System at the Pilgrim Nuclear Power Station



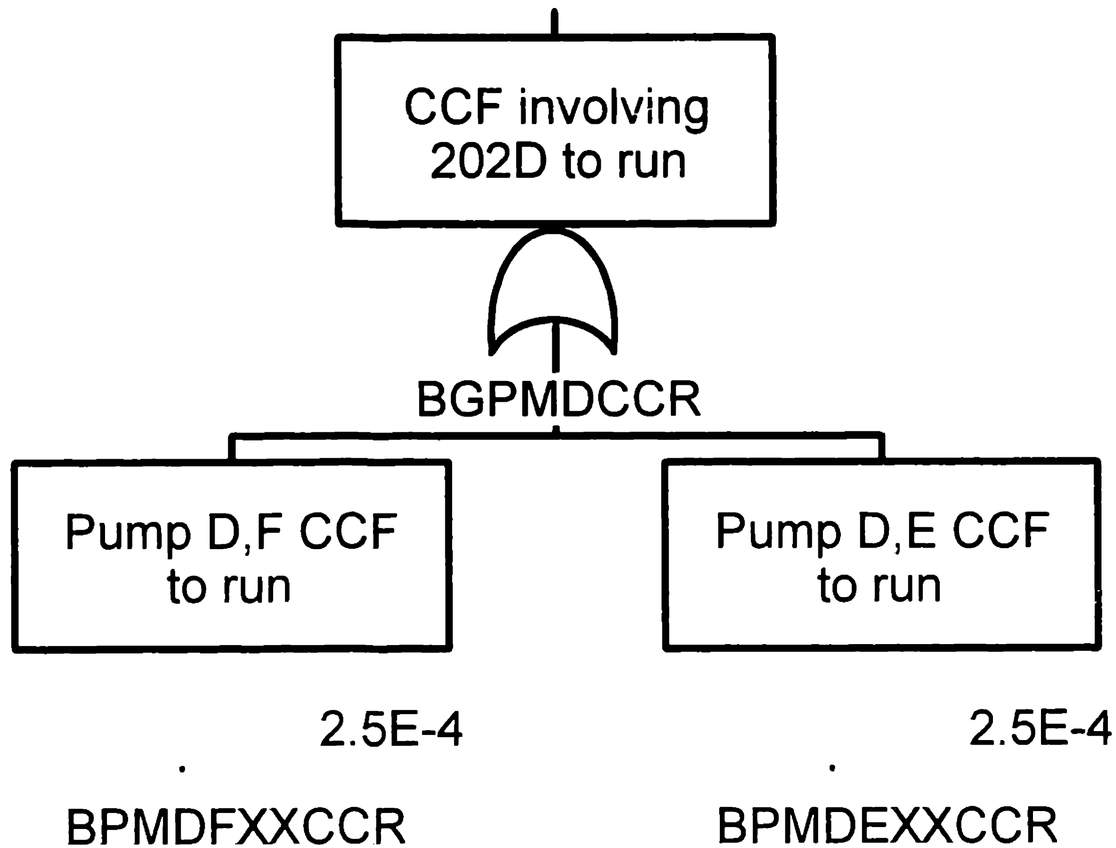
BGPMACCR - CCF involving 202A to Run 1999/04/12 Page 1

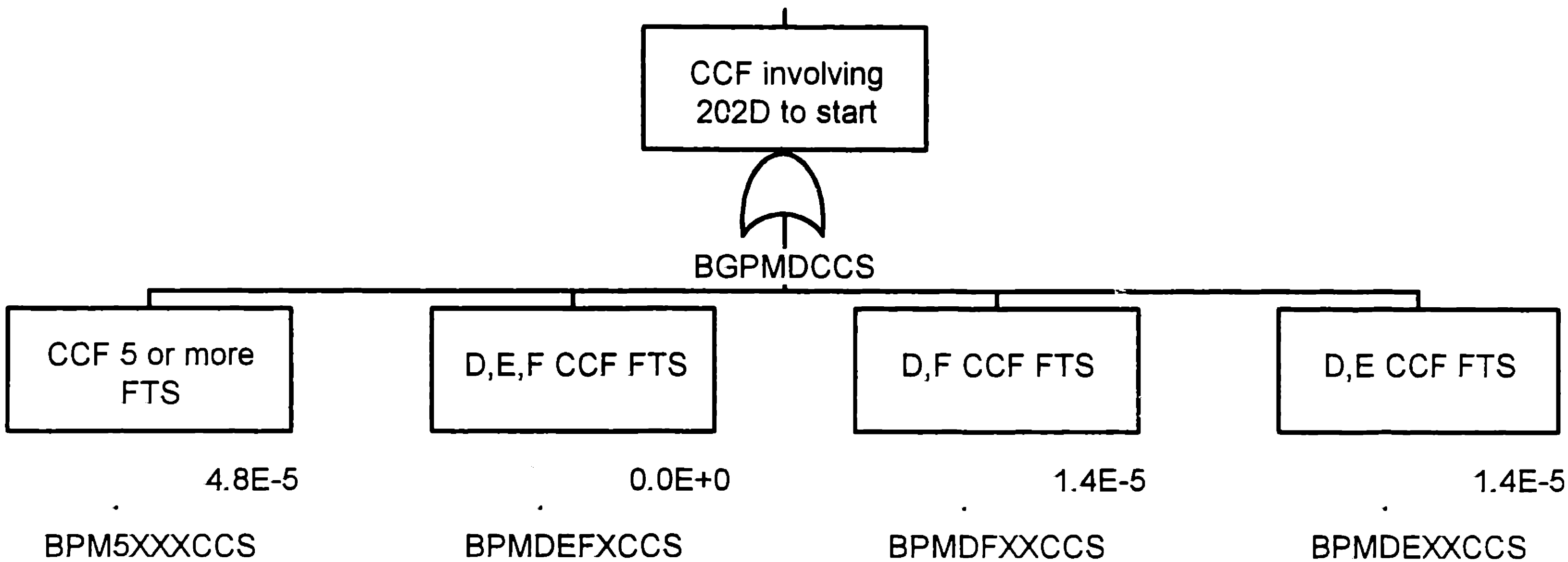

CCF involving 202A to start
BGPMACCS
A,C CCF FTS
1.4E-5
BPMACXXCCS
A,B CCF FTS
1.4E-5
BPMABXXCCS
A,B,C CCF FTS
0.0E+0
BPMABCXCCS
CCF 5 or more FTS
4.8E-5
BPM5XXXCCS

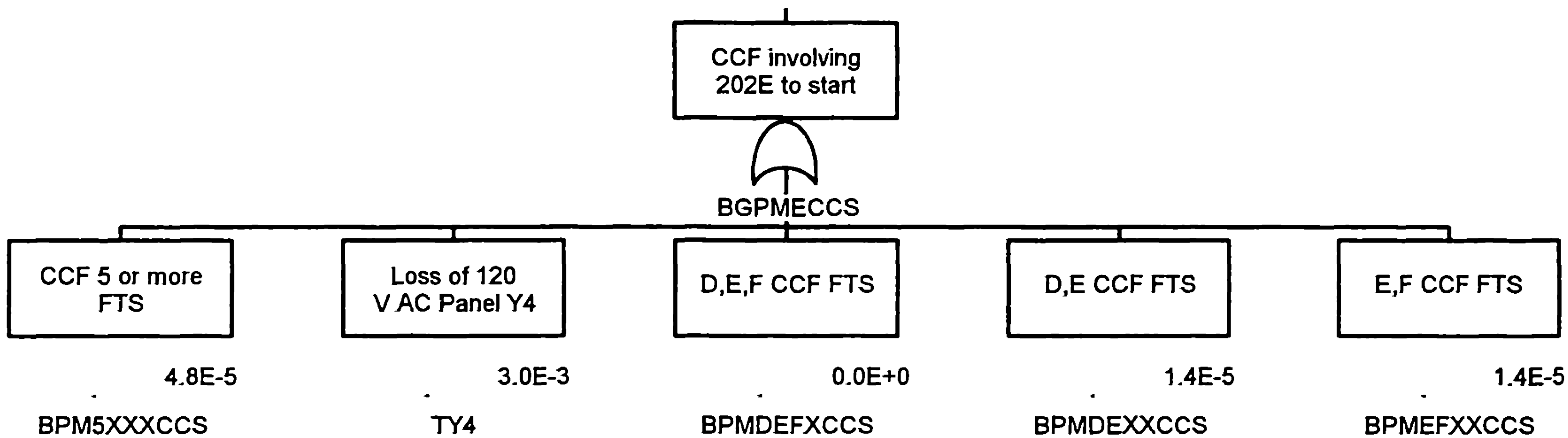

CCF involving
202B to start
BGPMBCCS
Loss of 120
V AC Panel Y3
3.0E-3
TY3
A,B CCF FTS
1.4E-5
BPMABXXCCS
A,B,C CCF FTS
0.0E+0
BPMABCXCCS
CCF 5 or more
FTS
4.8E-5
BPM5XXXCCS
B,C CCF FTS
1.4E-5
BPMBCXXCCS



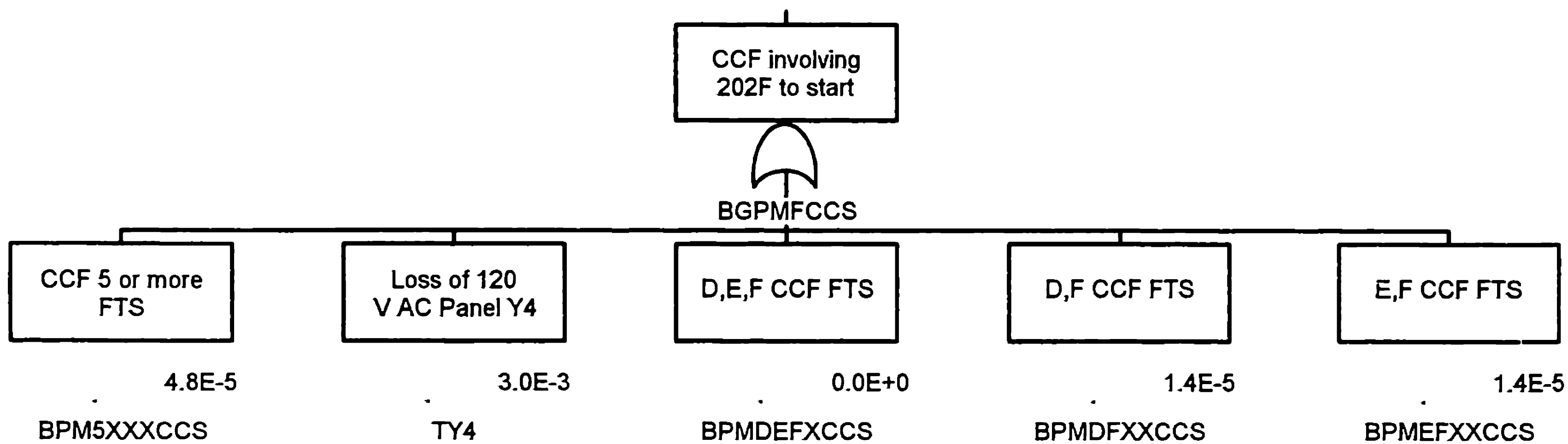
BGPMCCCS - CCF involving 202C to start 1999/02/09 Page 4

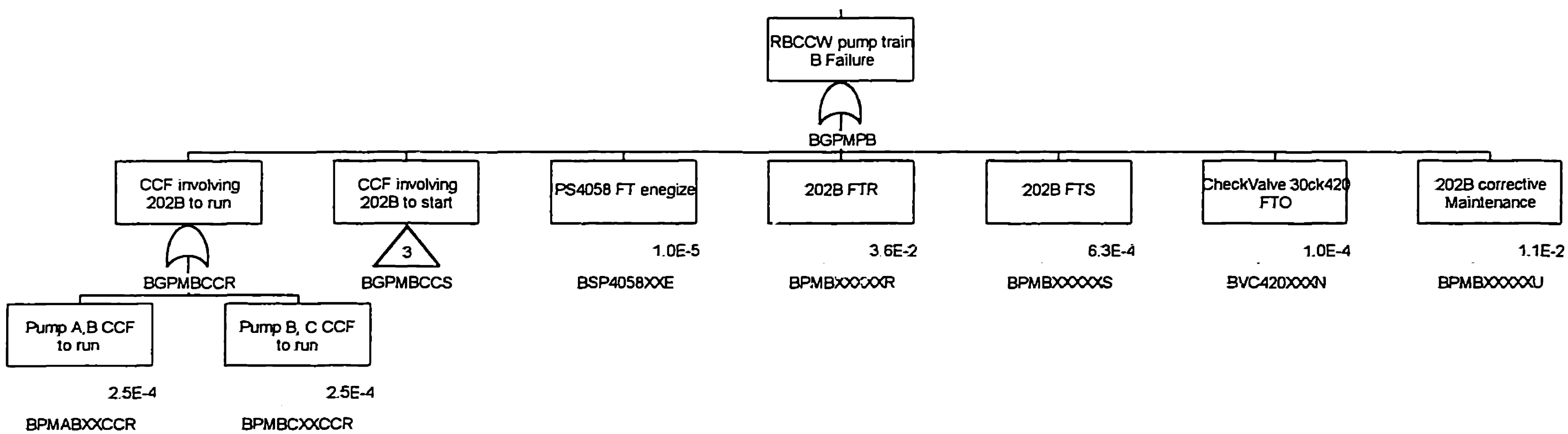

CCF involving
202D to run
BGPMDCCR
Pump D,F CCF
to run
2.5E-4
BPMDFXXCCR
Pump D,E CCF
to run
2.5E-4
BPMDEXXCCR


CCF involving 202D to start
BGPMDCCS
CCF 5 or more FTS
4.8E-5
BPM5XXXCCS
D,E,F CCF FTS
0.0E+0
BPMDEFXCCS
D,F CCF FTS
1.4E-5
BPMDFXXCCS
D,E CCF FTS
1.4E-5
BPMDEXXCCS

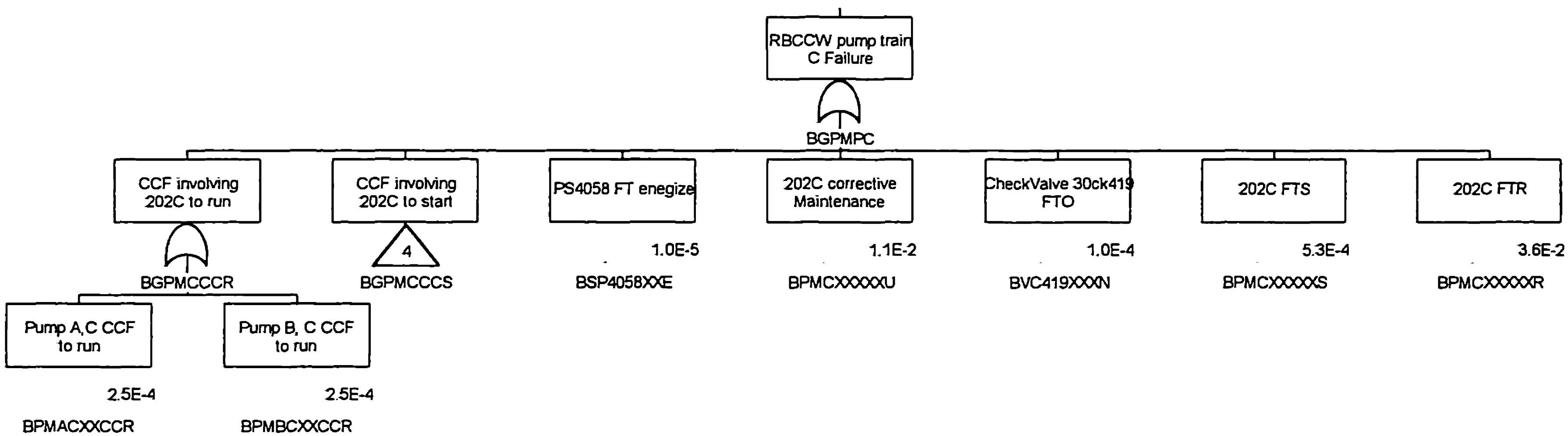


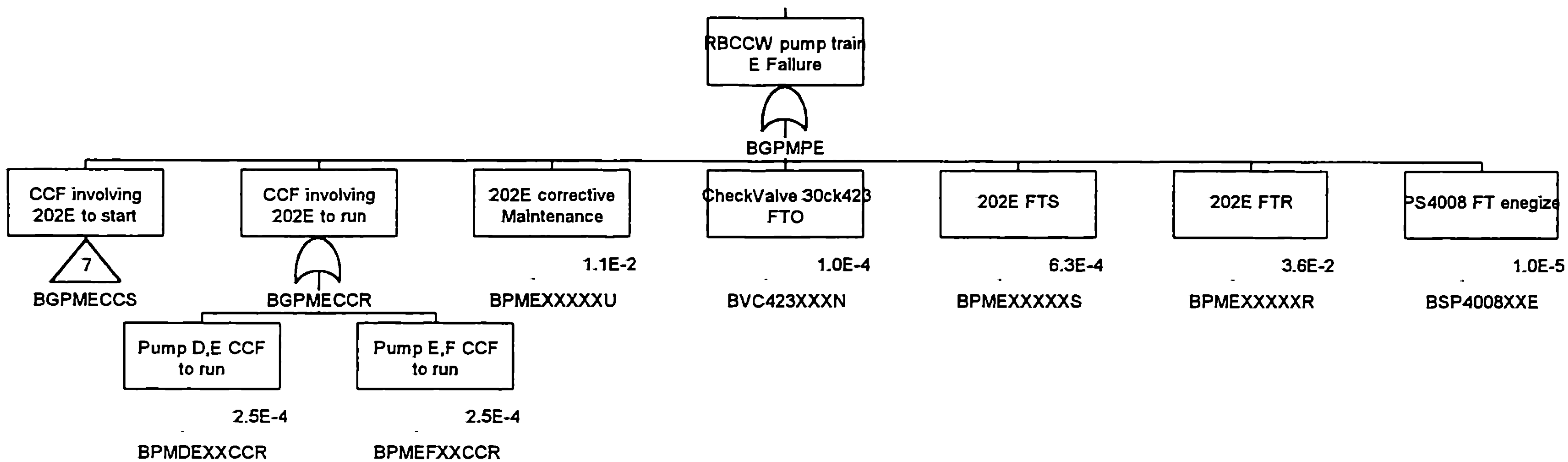
CCF involving 202E to start
BGPMECCS
CCF 5 or more FTS
4.8E-5
BPM5XXXCCS
Loss of 120 V AC Panel Y4
3.0E-3
TY4
D,E,F CCF FTS
0.0E+0
BPMDEFXCCS
D,E CCF FTS
1.4E-5
BPMDEXXCCS
E,F CCF FTS
1.4E-5
BPMEFXXCCS

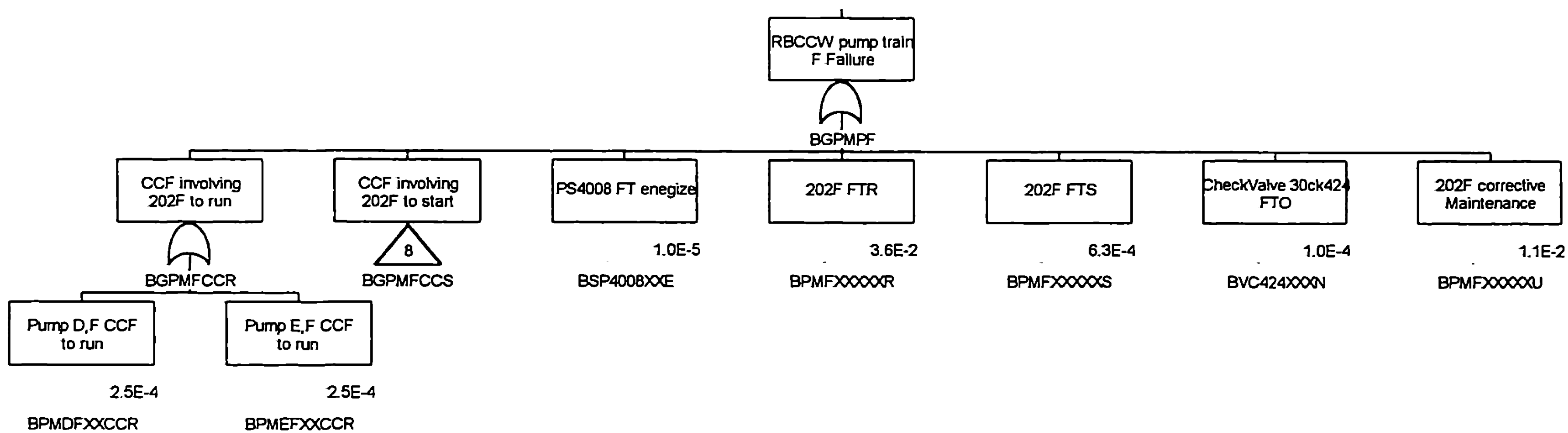

CCF involving
202F to start
BGPMFCCS
CCF 5 or more
FTS
4.8E-5
BPM5XXXCCS
Loss of 120
V AC Panel Y4
3.0E-3
TY4
D,E,F CCF FTS
0.0E+0
BPMDEFXCCS
D,F CCF FTS
1.4E-5
BPMDFXXCCS
E,F CCF FTS
1.4E-5
BPMEFXXCCS



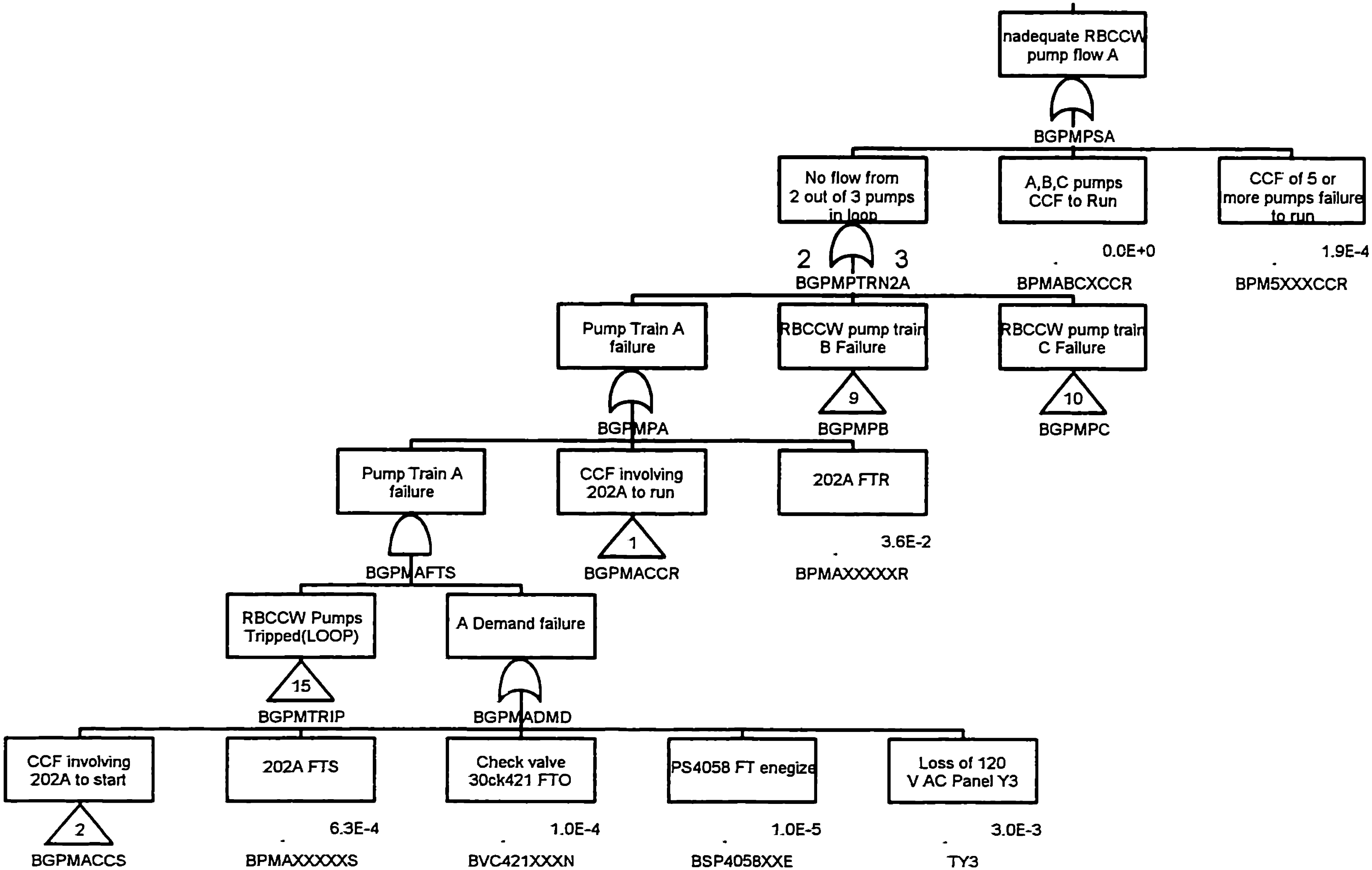
BGPMPB - Pump Train B failure 1999/02/09 Page 9

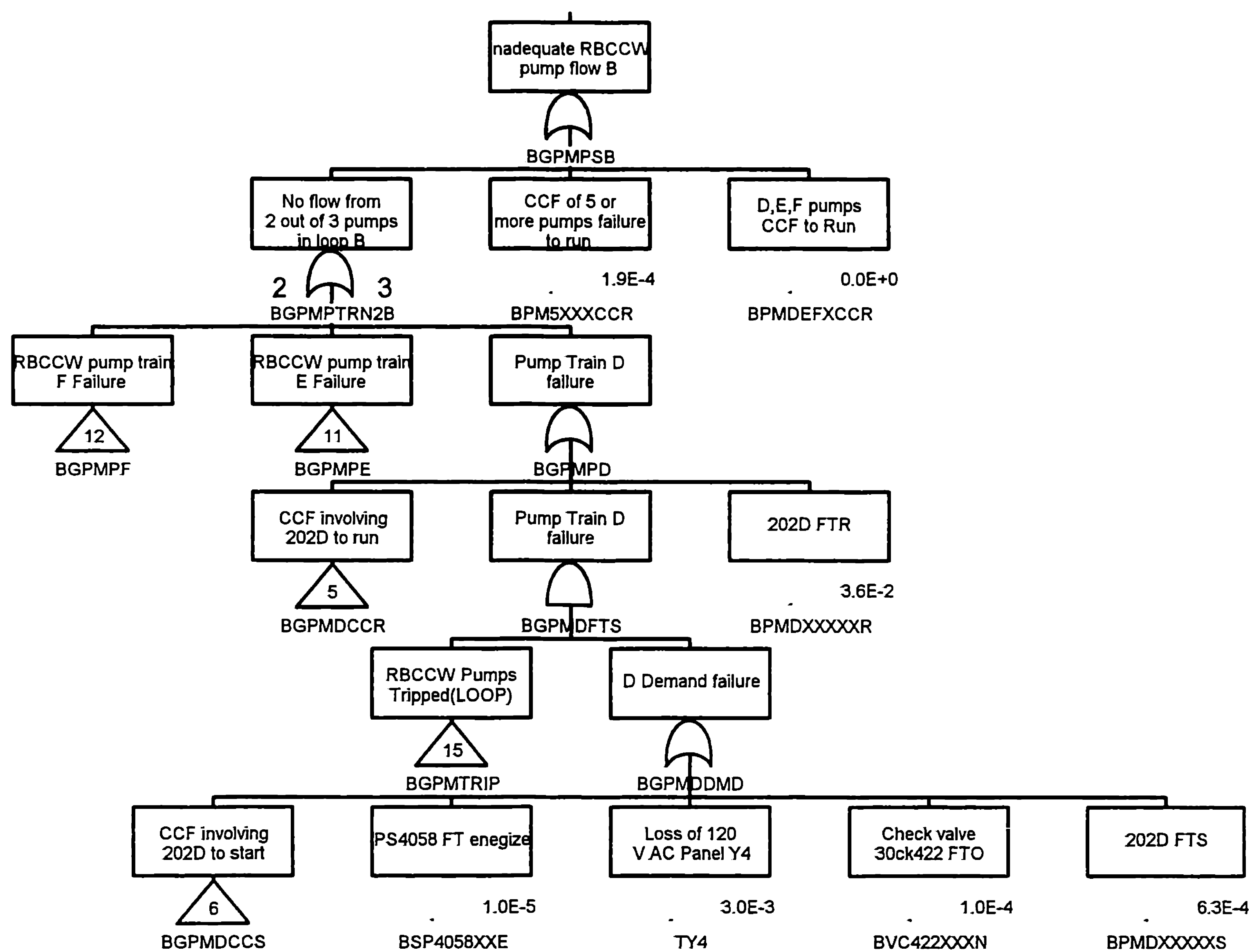

RBCCW pump train C Failure
BGPMPC
CCF involving 202C to run
BGPMCCCR
Pump A,C CCF to run
2.5E-4
BPMACXXCCR
Pump B, C CCF to run
2.5E-4
BPMBCXXCCR
CCF involving 202C to start
4
BGPMCCCS
PS4058 FT enegize
1.0E-5
BSP4058XXE
202C corrective Maintenance
1.1E-2
BPMCXXXXXU
CheckValve 30ck419 FTO
1.0E-4
BVC419XXXN
202C FTS
5.3E-4
BPMCXXXXXS
202C FTR
3.6E-2
BPMCXXXXXR


RBCCW pump train E Failure
BGPMPE
CCF involving 202E to start
7
BGPMECCS
CCF involving 202E to run
BGPMECCR
Pump D,E CCF to run
2.5E-4
BPMDEXXCCR
Pump E,F CCF to run
2.5E-4
BPMEFXXCCR
202E corrective Maintenance
1.1E-2
BPMEXXXXXU
CheckValve 30ck423 FTO
1.0E-4
BVC423XXXN
202E FTS
6.3E-4
BPMEXXXXXS
202E FTR
3.6E-2
BPMEXXXXXR
PS4008 FT enegize
1.0E-5
BSP4008XXE

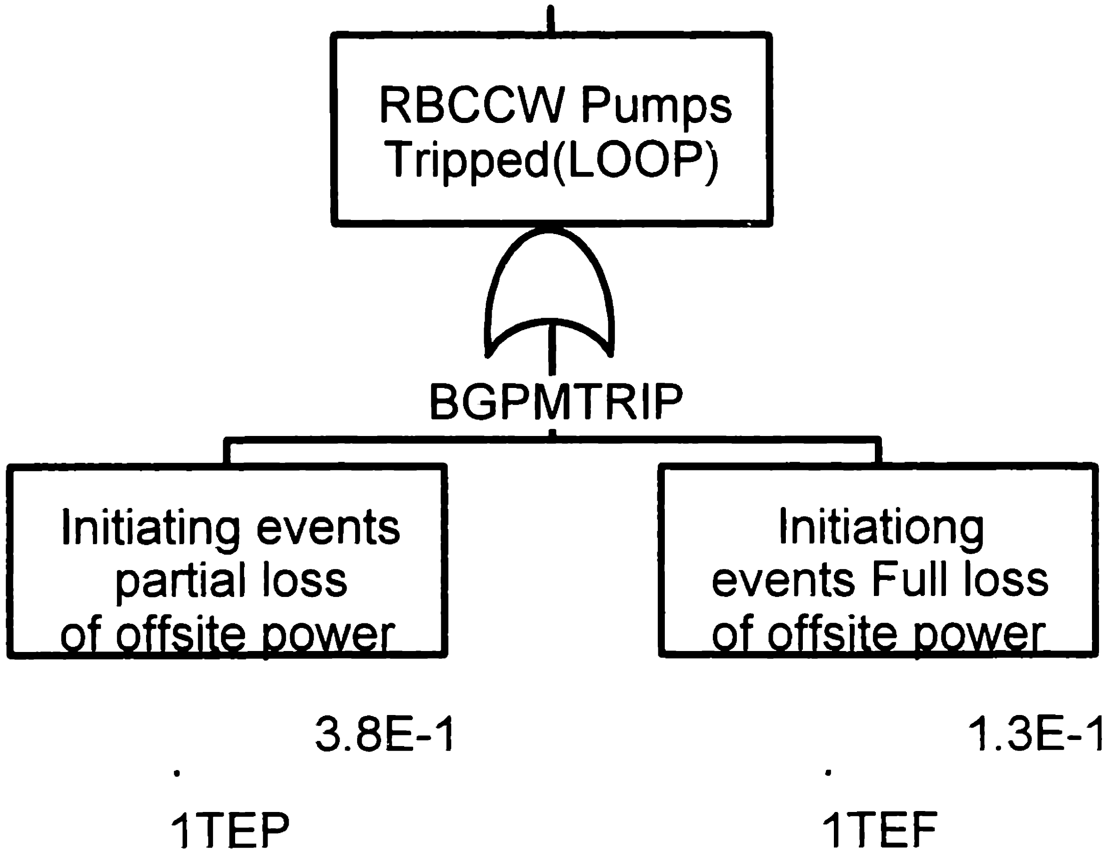


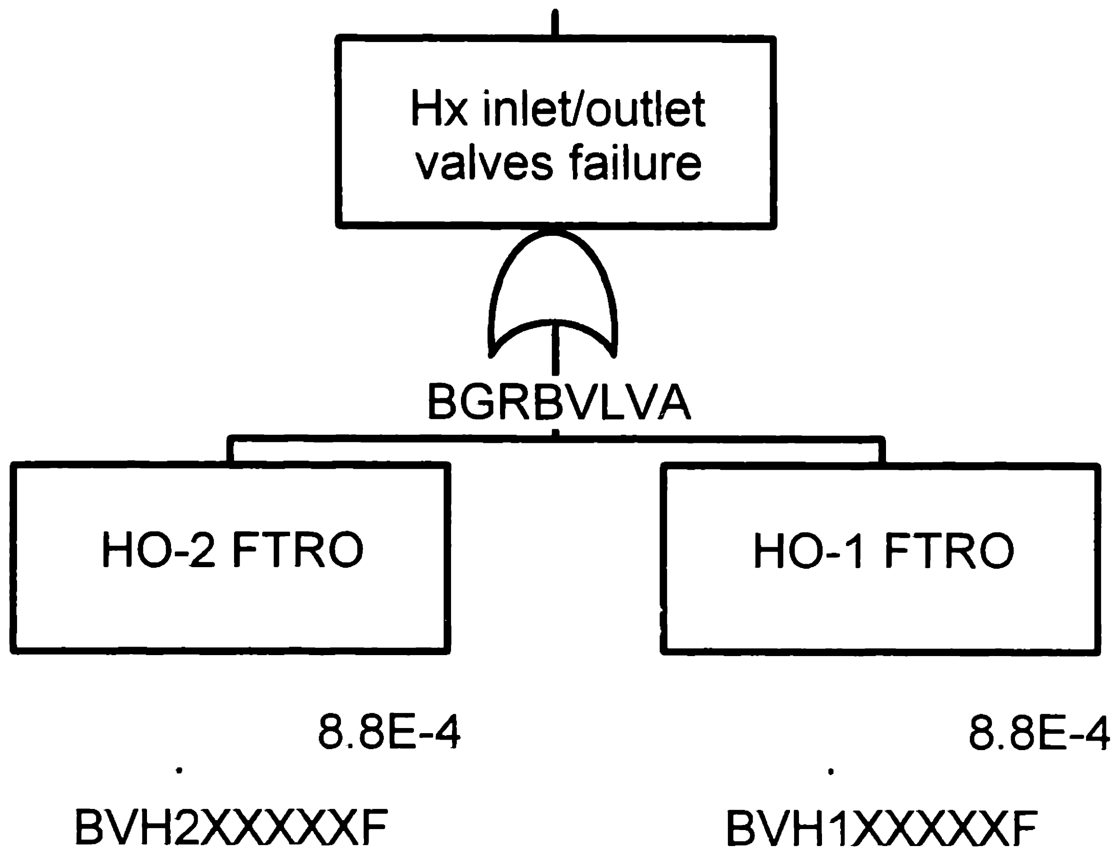
BGPMPF - RBCCW pump train F Failure 1999/02/09 Page 12

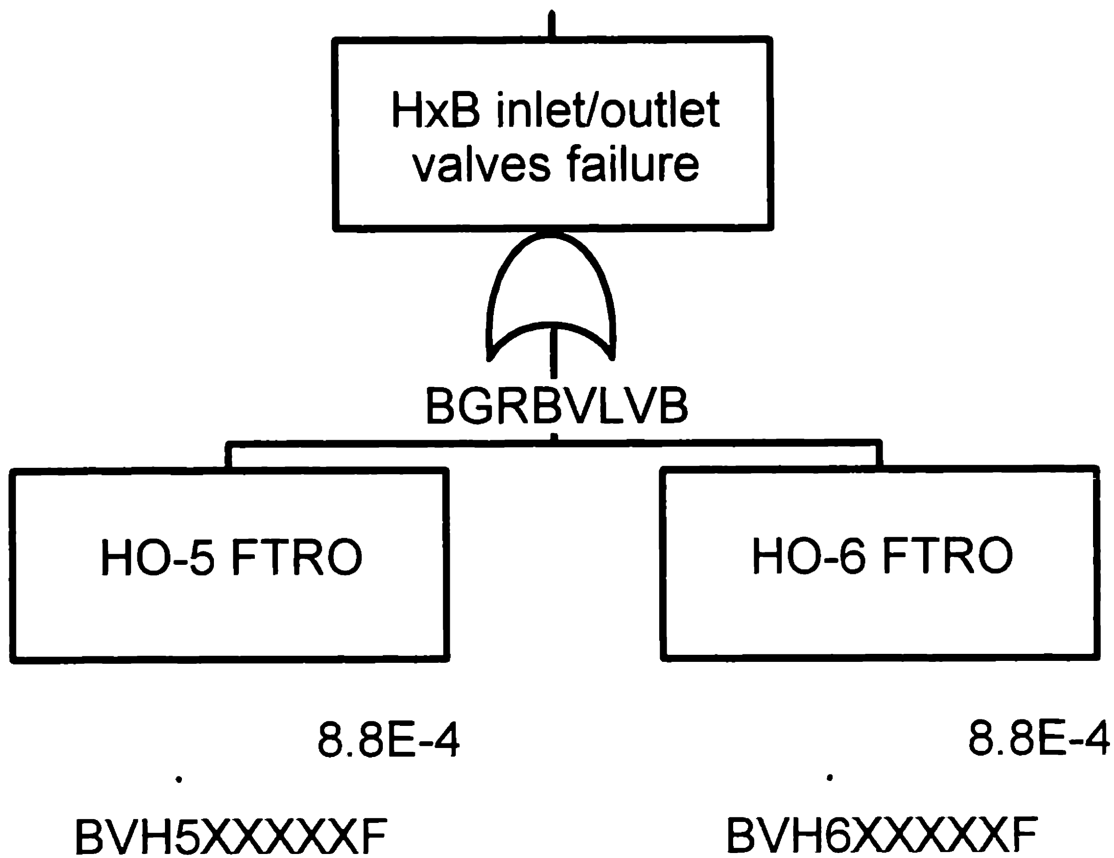

nadequate RBCCW pump flow A
BGPMPSA
No flow from 2 out of 3 pumps in loop
A,B,C pumps CCF to Run
CCF of 5 or more pumps failure to run
2
3
BGPMPTRN2A
0.0E+0
BPMABCXCCR
1.9E-4
BPM5XXXCCR
Pump Train A failure
RBCCW pump train B Failure
RBCCW pump train C Failure
9
10
BGPMPA
BGPMPB
BGPMPC
Pump Train A failure
CCF involving 202A to run
202A FTR
1
3.6E-2
BGPMAFTS
BGPMACCR
BPMAXXXXXR
RBCCW Pumps Tripped(LOOP)
A Demand failure
15
BGPMTRIP
BGPMADMD
CCF involving 202A to start
202A FTS
Check valve 30ck421 FTO
PS4058 FT enegize
Loss of 120 V AC Panel Y3
2
6.3E-4
1.0E-4
1.0E-5
3.0E-3
BGPMACCS
BPMAXXXXXS
BVC421XXXN
BSP4058XXE
TY3

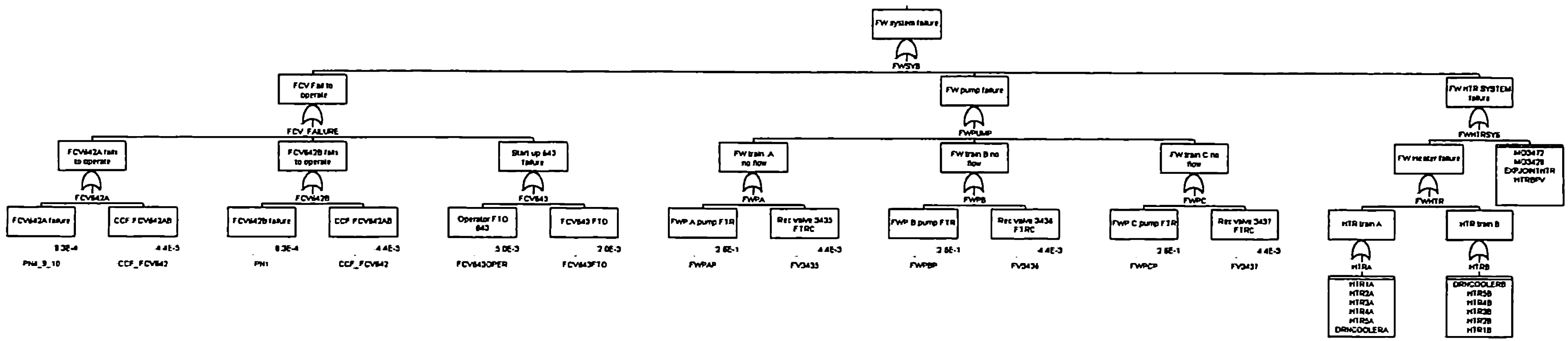


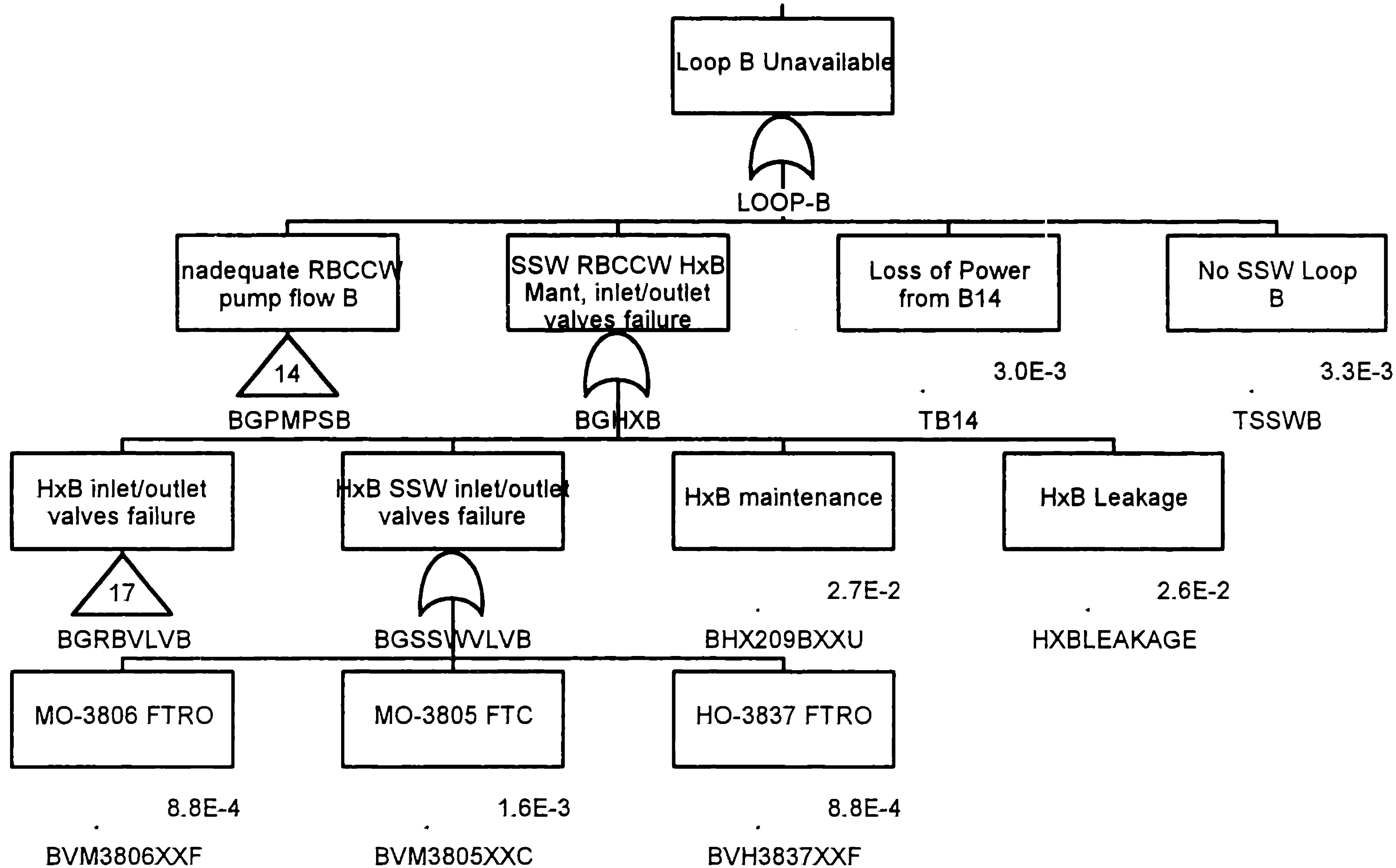
BGPMPSB - Inadequate RBCCW pump flow B 1999/02/09 Page 14

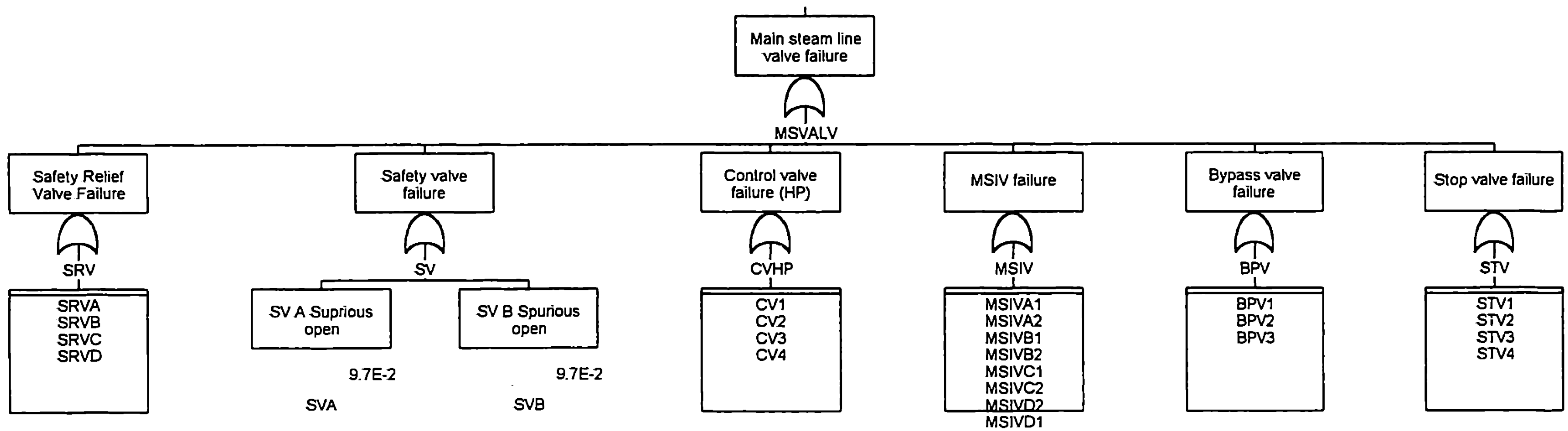

RBCCW Pumps Tripped(LOOP)
BGPMTRIP
Initiating events partial loss of offsite power
3.8E-1
1TEP
Initiationg events Full loss of offsite power
1.3E-1
1TEF


Hx inlet/outlet valves failure
BGRBVLVA
HO-2 FTRO
8.8E-4
BVH2XXXXXF
HO-1 FTRO
8.8E-4
BVH1XXXXXF

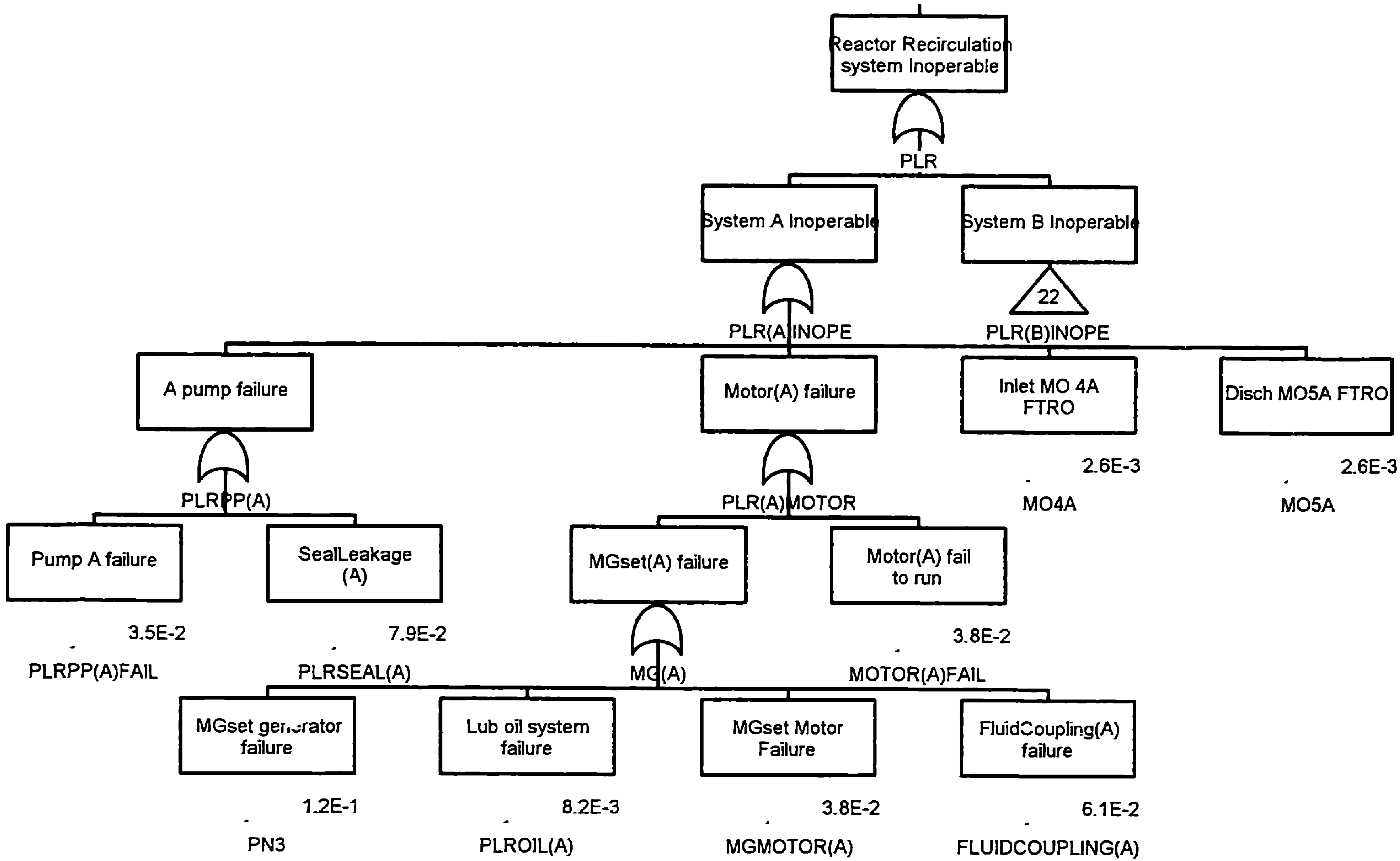

HxB inlet/outlet valves failure
BGRBVLVB
HO-5 FTRO
HO-6 FTRO
8.8E-4
BVH5XXXXXF
8.8E-4
BVH6XXXXXF


FW system failure
FWSYS
FCV Fail to operate
FCV_FAILURE
FCV642A fails to operate
FCV642A
FCV642A failure
8.3E-4
PN4_9_10
CCF FCV642AB
4.4E-3
CCF_FCV642
FCV642B fails to operate
FCV642B
FCV642B failure
8.3E-4
PN1
CCF FCV642AB
4.4E-3
CCF_FCV642
Start up 643 failure
FCV643
Operator FTO 643
3.0E-3
FCV643OPER
FCV643 FTO
2.0E-3
FCV643FTO
FW pump failure
FWPUMP
FW train A no flow
FWPA
FWP A pump FTR
2.6E-1
FWPAP
Rec valve 3435 FTRC
4.4E-3
FV3435
FW train B no flow
FWPB
FWP B pump FTR
2.6E-1
FWPBP
Rec valve 3436 FTRC
4.4E-3
FV3436
FW train C no flow
FWPC
FWP C pump FTR
2.6E-1
FWPCP
Rec valve 3437 FTRC
4.4E-3
FV3437
FW HTR SYSTEM failure
FWHTRSYS
FW Heater failure
FWHTR
MO3472
MO3478
EXPJOINTHTR
HTRBPV
HTR train A
HTRA
HTR1A
HTR2A
HTR3A
HTR4A
HTR5A
DRNCOOLERA
HTR train B
HTRB
DRNCOOLERB
HTR5B
HTR4B
HTR3B
HTR2B
HTR1B

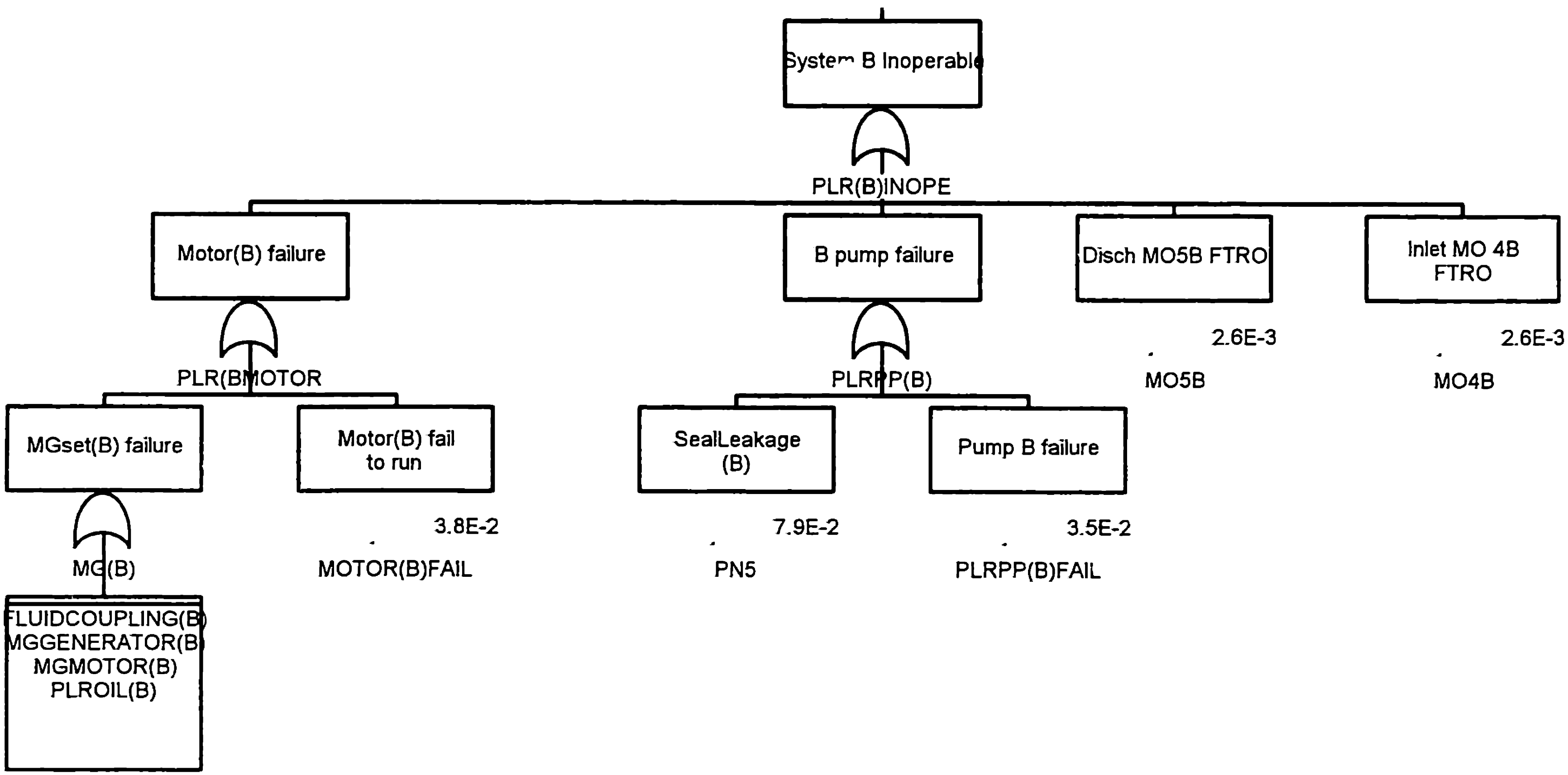

Loop B Unavailable
LOOP-B
nadequate RBCCW pump flow B
14
BGPMPSB
SSW RBCCW HxB Mant, inlet/outlet valves failure
BGHXB
Loss of Power from B14
3.0E-3
TB14
No SSW Loop B
3.3E-3
TSSWB
HxB inlet/outlet valves failure
17
BGRBVLVB
HxB SSW inlet/outlet valves failure
BGSSWVLVB
HxB maintenance
2.7E-2
BHX209BXXU
HxB Leakage
2.6E-2
HXBLEAKAGE
MO-3806 FTRO
8.8E-4
BVM3806XXF
MO-3805 FTC
1.6E-3
BVM3805XXC
HO-3837 FTRO
8.8E-4
BVH3837XXF

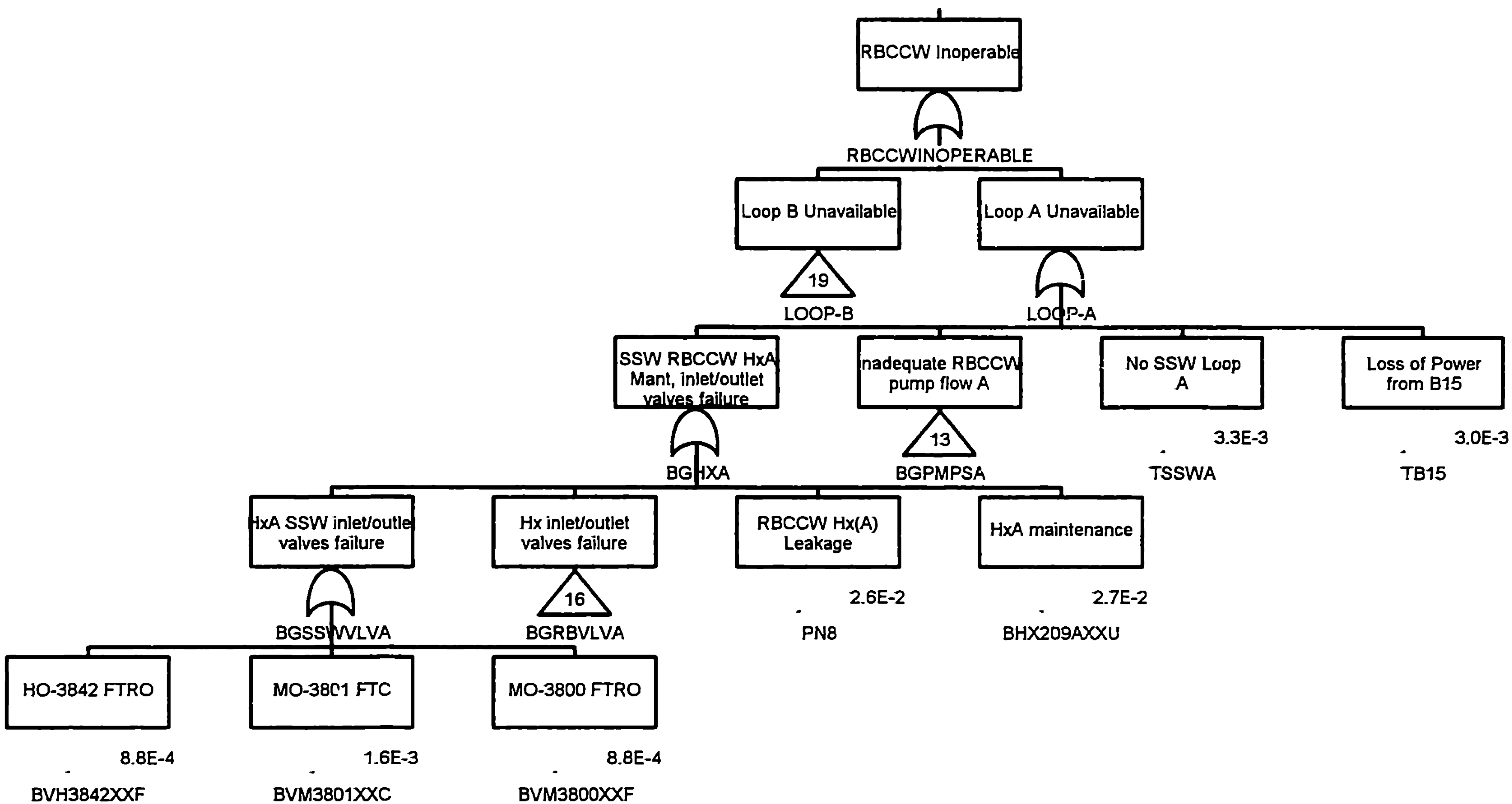


MSVALV - MS line valves failure 1999/02/18 Page 20

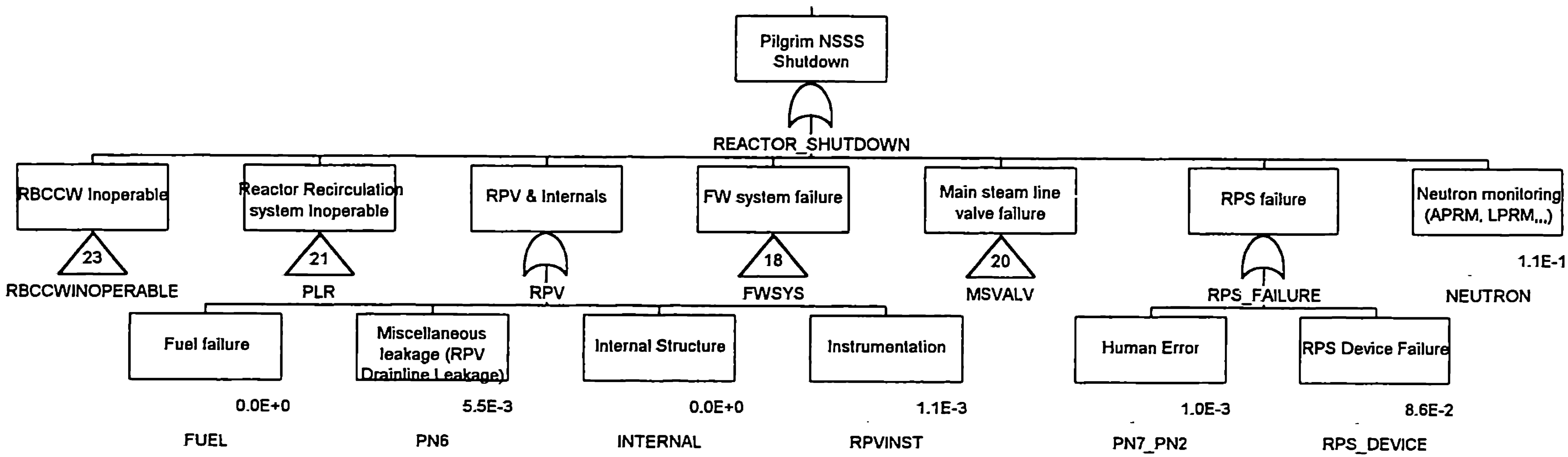


PLR - PLR system failure 1999/02/18 Page 21


System B Inoperable
PLR(B)INOPE
Motor(B) failure
B pump failure
Disch MO5B FTRO
Inlet MO 4B FTRO
2.6E-3
MO5B
2.6E-3
MO4B
PLR(BMOTOR
PLRPP(B)
MGset(B) failure
Motor(B) fail to run
SealLeakage (B)
Pump B failure
3.8E-2
7.9E-2
3.5E-2
MG(B)
MOTOR(B)FAIL
PN5
PLRPP(B)FAIL
FLUIDCOUPLING(B)
MGGENERATOR(B)
MGMOTOR(B)
PLROIL(B)

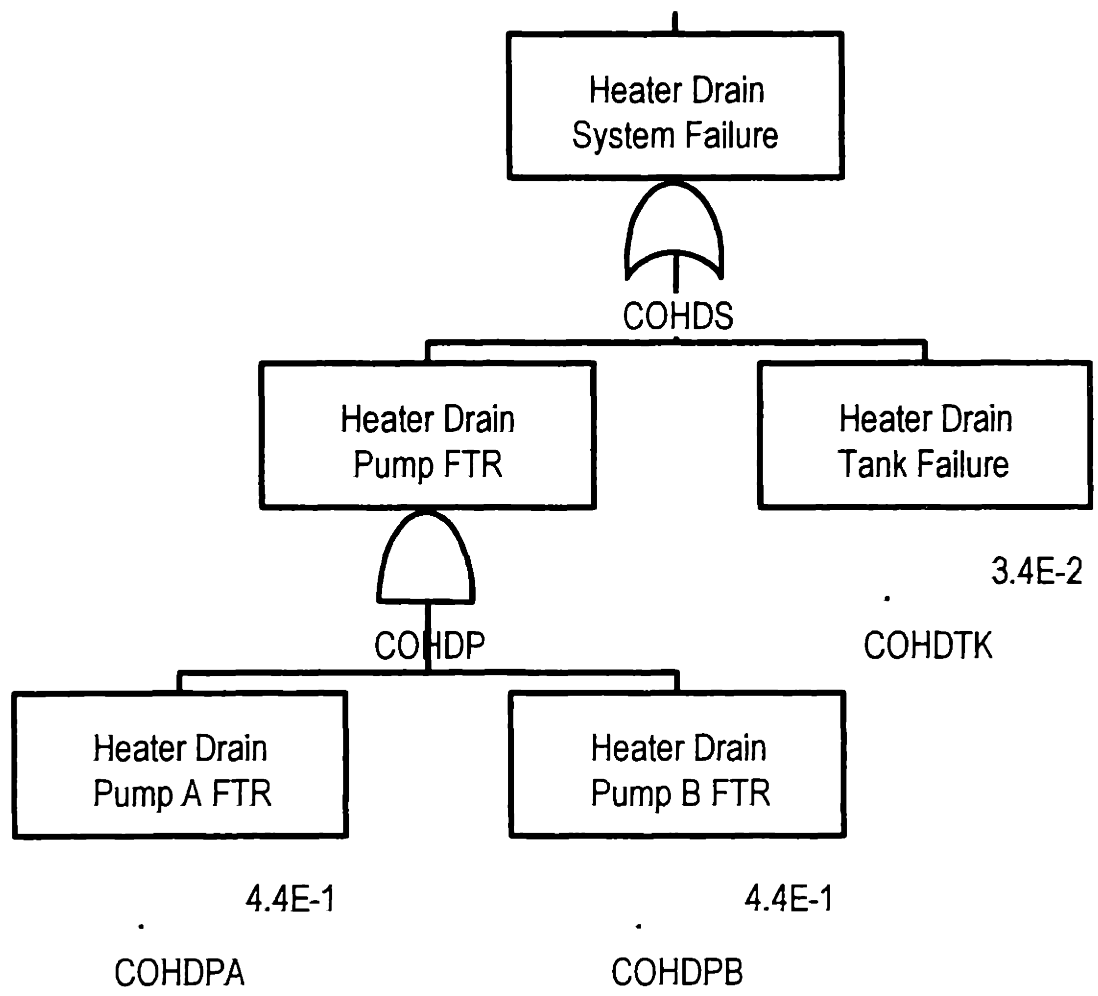


RBCCW Inoperable
RBCCWINOPERABLE
Loop B Unavailable
Loop A Unavailable
19
LOOP-B
LOOP-A
SSW RBCCW HxA Mant, inlet/outlet valves failure
nadequate RBCCW pump flow A
No SSW Loop A
Loss of Power from B15
13
3.3E-3
3.0E-3
TSSWA
TB15
BGHXA
BGPMPSA
HxA SSW inlet/outlet valves failure
Hx inlet/outlet valves failure
RBCCW Hx(A) Leakage
HxA maintenance
16
2.6E-2
2.7E-2
BGSSWVLVA
BGRBVLVA
PN8
BHX209AXXU
HO-3842 FTRO
MO-3801 FTC
MO-3800 FTRO
8.8E-4
1.6E-3
8.8E-4
BVH3842XXF
BVM3801XXC
BVM3800XXF

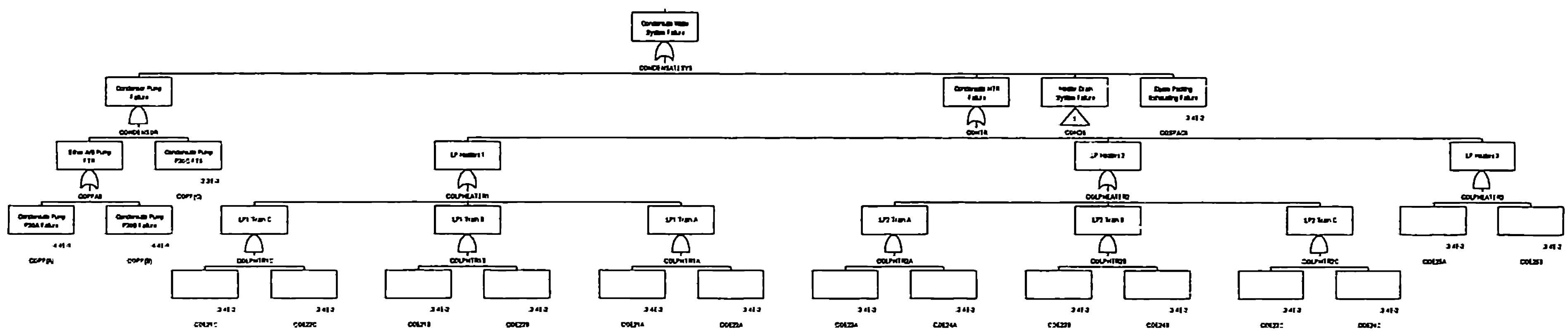

Pilgrim NSSS Shutdown
REACTOR_SHUTDOWN
RBCCW Inoperable
23
RBCCWINOPERABLE
Reactor Recirculation system Inoperable
21
PLR
RPV & Internals
RPV
FW system failure
18
FWSYS
Main steam line valve failure
20
MSVALV
RPS failure
RPS_FAILURE
Neutron monitoring (APRM, LPRM,..)
1.1E-1
NEUTRON
Fuel failure
0.0E+0
FUEL
Miscellaneous leakage (RPV Drainline Leakage)
5.5E-3
PN6
Internal Structure
0.0E+0
INTERNAL
Instrumentation
1.1E-3
RPVINST
Human Error
1.0E-3
PN7_PN2
RPS Device Failure
8.6E-2
RPS_DEVICE

Appendix C

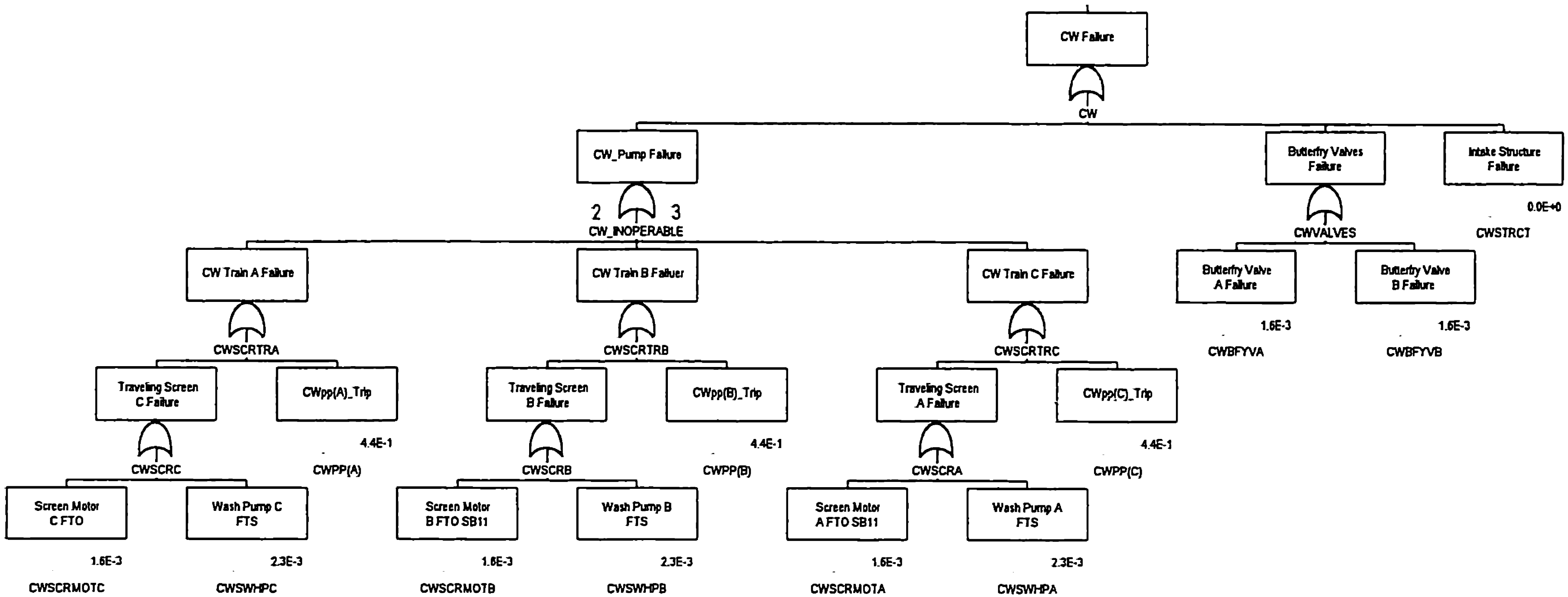
Fault Tree Models of the Balance of Plant System at the Seabrook Nuclear Power Station

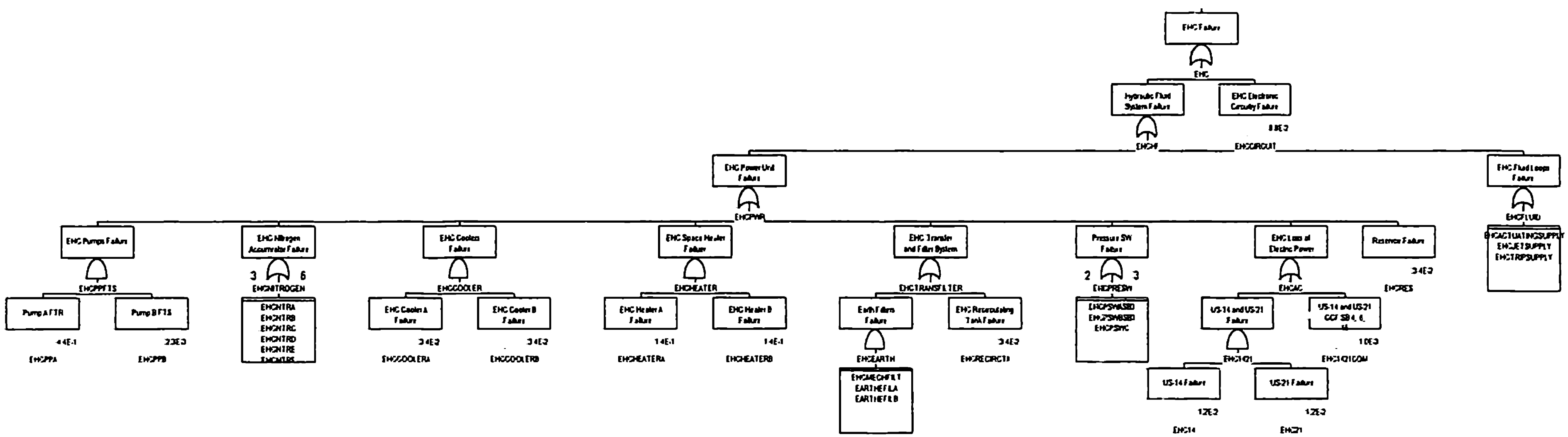


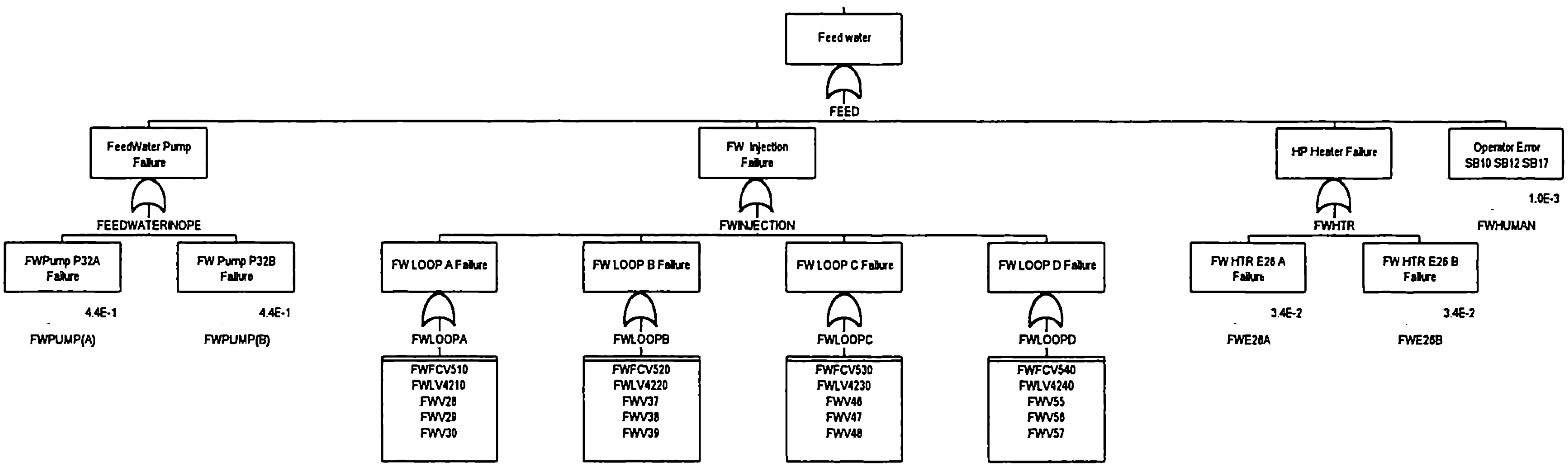
COHDS - Heater Drain System Failure 1999/05/14 Page 1

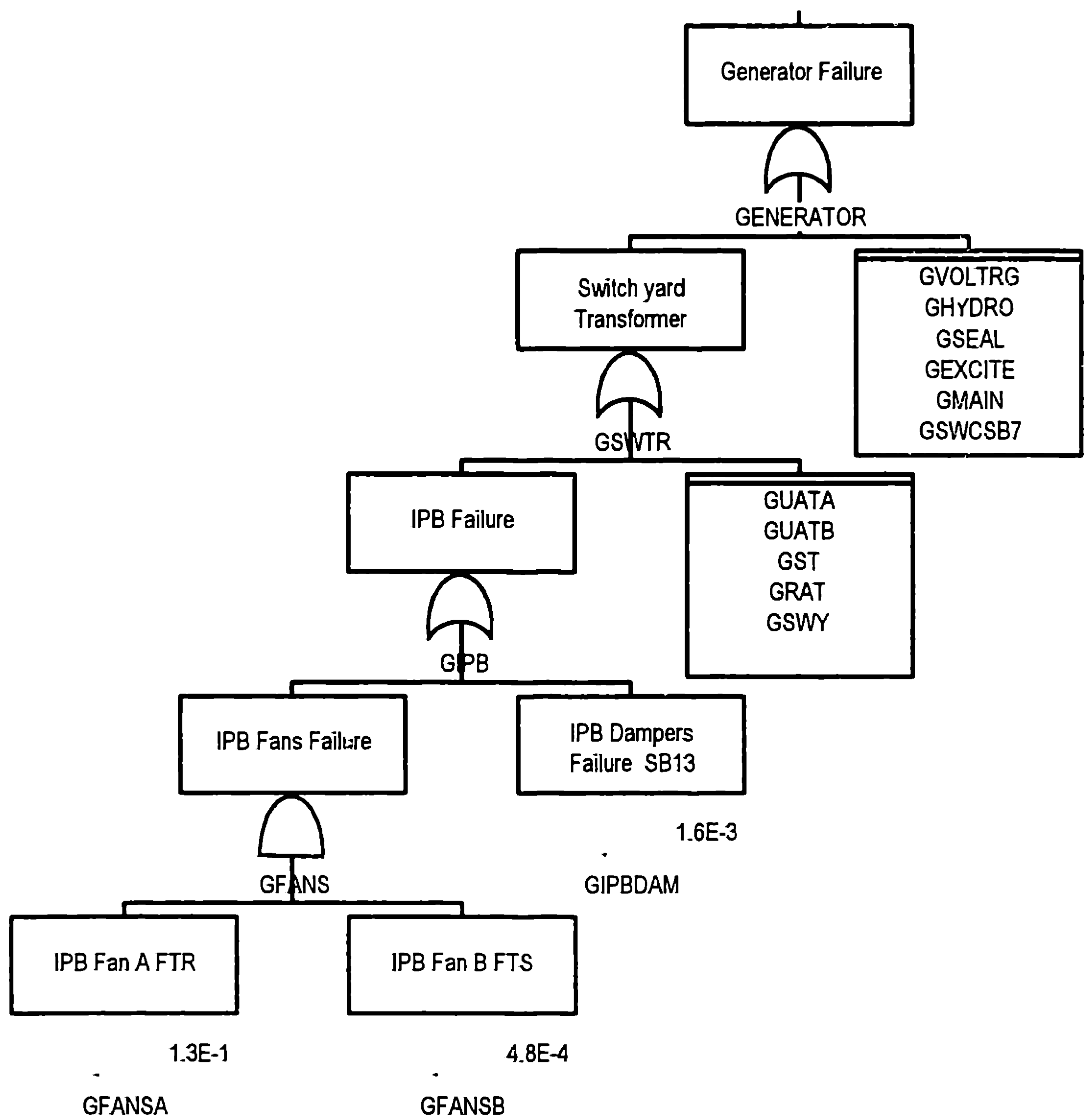


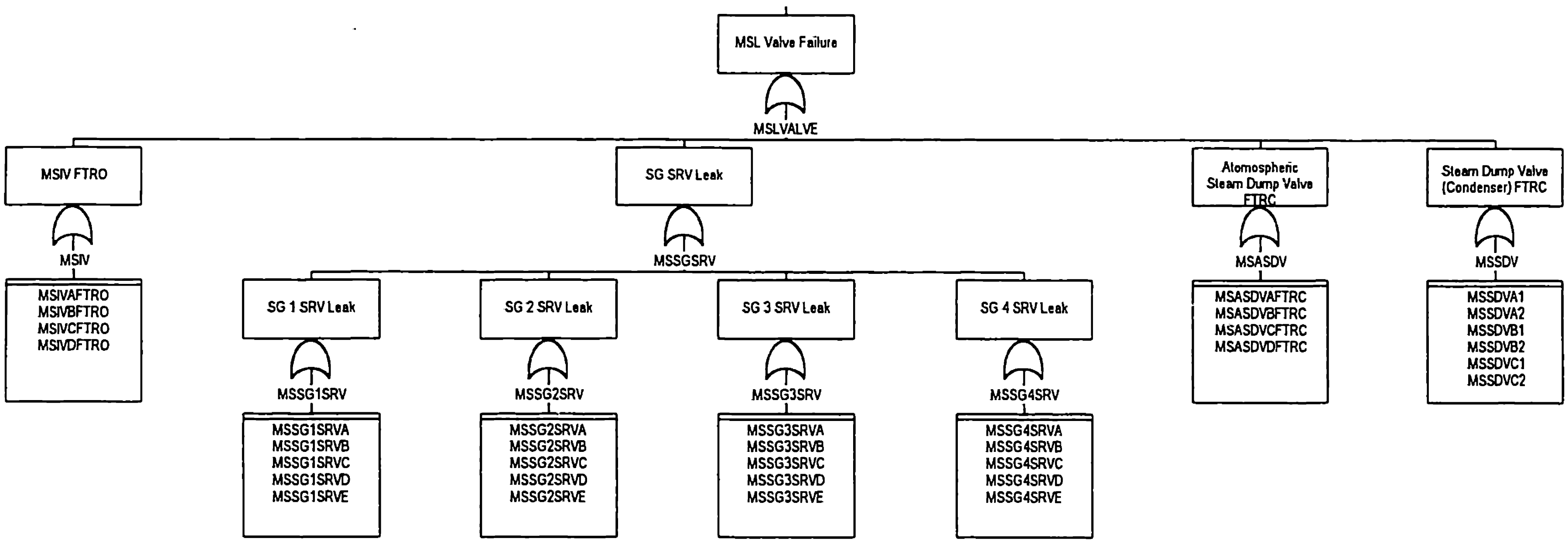
CONDENSATESYS - Condensate Water System Failure 1999/05/14 Page 2

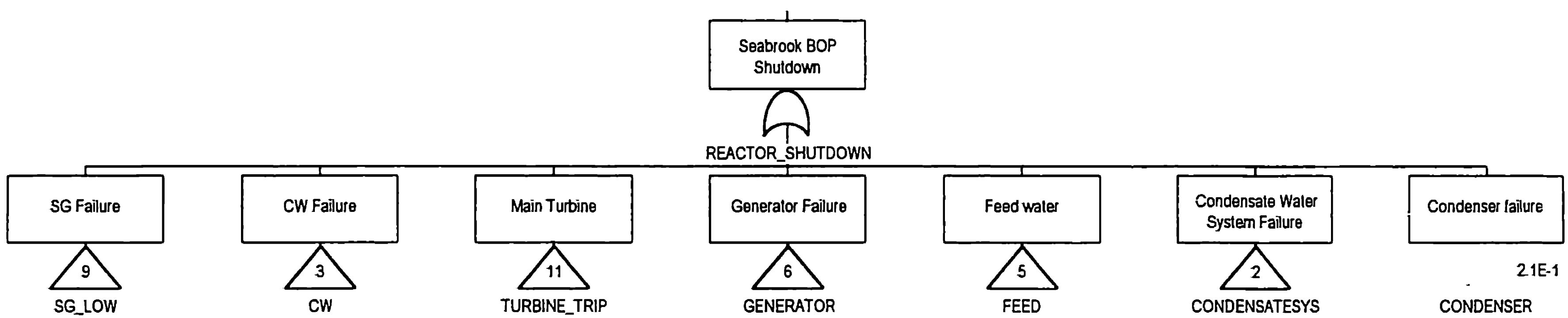

CW Failure
CW
CW_Pump Failure
2
3
CW_INOPERABLE
Butterfly Valves Failure
CWVALVES
Intake Structure Failure
0.0E+0
CWSTRCT
CW Train A Failure
CWSCRTRA
CW Train B Failuer
CWSCRTRB
CW Train C Failure
CWSCRTRC
Butterfly Valve A Failure
1.6E-3
CWBFYVA
Butterfly Valve B Failure
1.6E-3
CWBFYVB
Traveling Screen C Failure
CWSCRC
CWpp(A)_Trip
4.4E-1
CWPP(A)
Traveling Screen B Failure
CWSCRB
CWpp(B)_Trip
4.4E-1
CWPP(B)
Traveling Screen A Failure
CWSCRA
CWpp(C)_Trip
4.4E-1
CWPP(C)
Screen Motor C FTO
1.6E-3
CWSCRMOTC
Wash Pump C FTS
2.3E-3
CWSWHPC
Screen Motor B FTO SB11
1.6E-3
CWSCRMOTB
Wash Pump B FTS
2.3E-3
CWSWHPB
Screen Motor A FTO SB11
1.6E-3
CWSCRMOTA
Wash Pump A FTS
2.3E-3
CWSWHPA

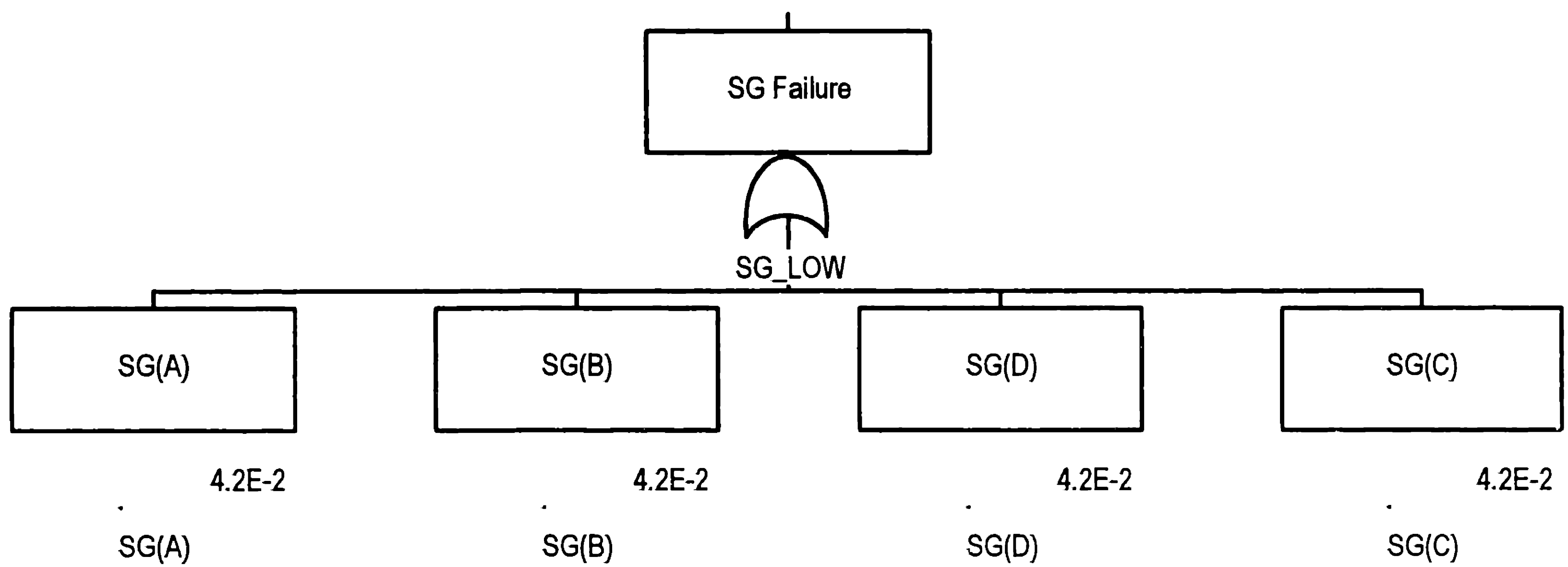

EHC
EHCCIRCUIT
EHC Power Unit Failure
EHCPWR
EHC Pumps Failure
EHCPPFTS
EHC Nitrogen Accumulator Failure
3
6
EHCNITROGEN
EHCNTRA
EHCNTRB
EHCNTRC
EHCNTRD
EHCNTRE
EHC Coolers Failure
EHCCOOLER
EHC Cooler A Failure
EHC Cooler B Failure
EHCCOOLERB
EHC Space Heater Failure
EHCHEATER
EHC Heater A Failure
EHC Heater B Failure
EHCHEATERA
EHCHEATERB
EHCTRANSFILTER
Earth Filters Failure
EHCEARTH
EARTHEFILA
EARTHEFILB
Pressure SW Failure
2
3
EHCPRESW
EHCPSWASBD
EHCPSWBSBD
EHCPSWC
EHCAC
US-14 and US-21 Failure
US-14 Failure
US-21 Failure
EHC14
EHC21
Reservoir Failure
EHCRES
EHC Fluid Loops Failure
EHCFLUID
EHCACTUATINGSUPPLY
EHCJETSUPPLY
EHCTRIPSUPPLY

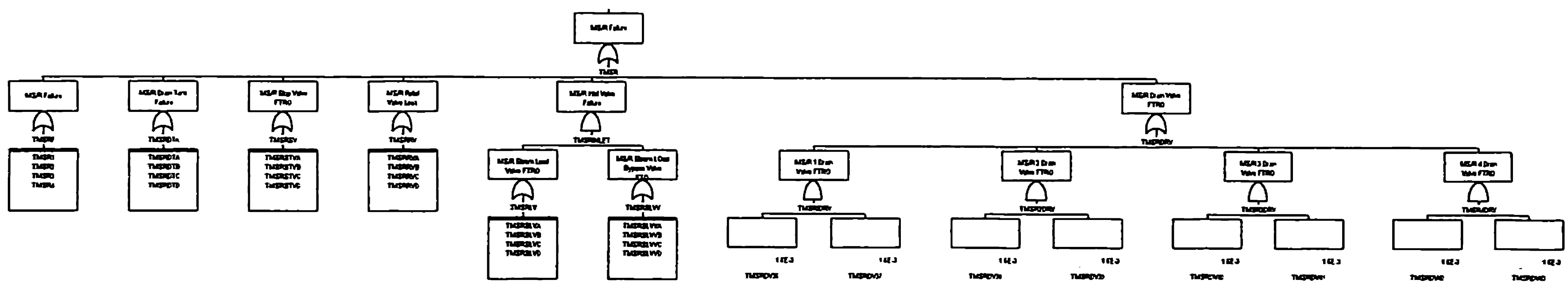

Feed water
FEED
FeedWater Pump Failure
FEEDWATERINOPE
FWPump P32A Failure
4.4E-1
FWPUMP(A)
FW Pump P32B Failure
4.4E-1
FWPUMP(B)
FW Injection Failure
FWINJECTION
FW LOOP A Failure
FWLOOPA
FWFCV510
FWLV4210
FWV28
FWV29
FWV30
FW LOOP B Failure
FWLOOPB
FWFCV520
FWLV4220
FWV37
FWV38
FWV39
FW LOOP C Failure
FWLOOPC
FWFCV530
FWLV4230
FWV46
FWV47
FWV48
FW LOOP D Failure
FWLOOPD
FWFCV540
FWLV4240
FWV55
FWV56
FWV57
HP Heater Failure
FWHTR
FW HTR E26 A Failure
3.4E-2
FWE26A
FW HTR E26 B Failure
3.4E-2
FWE26B
Operator Error SB10 SB12 SB17
1.0E-3
FWHUMAN


Generator Failure
GENERATOR
Switch yard Transformer
GVOLTRG
GHYDRO
GSEAL
GEXCITE
GMAIN
GSWCSB7
GSWTR
IPB Failure
GUATA
GUATB
GST
GRAT
GSWY
GIPB
IPB Fans Failure
IPB Dampers Failure SB13
1.6E-3
GIPBDAM
GFANS
IPB Fan A FTR
IPB Fan B FTS
1.3E-1
GFANSA
4.8E-4
GFANSB

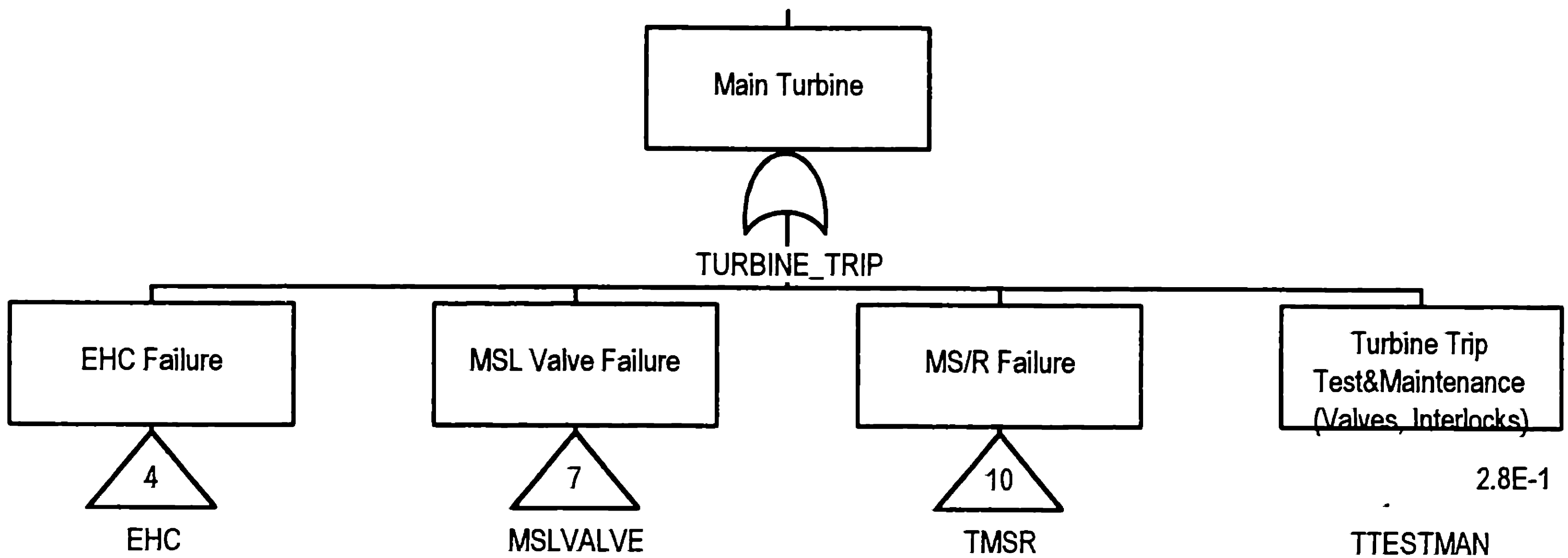

MSL Valve Failure
MSLVALVE
MSIV FTRO
MSIV
MSIVAFTRO
MSIVBFTRO
MSIVCFTRO
MSIVDFTRO
SG SRV Leak
MSSGSRV
SG 1 SRV Leak
MSSG1SRV
MSSG1SRVA
MSSG1SRVB
MSSG1SRVC
MSSG1SRVD
MSSG1SRVE
SG 2 SRV Leak
MSSG2SRV
MSSG2SRVA
MSSG2SRVB
MSSG2SRVC
MSSG2SRVD
MSSG2SRVE
SG 3 SRV Leak
MSSG3SRV
MSSG3SRVA
MSSG3SRVB
MSSG3SRVC
MSSG3SRVD
MSSG3SRVE
SG 4 SRV Leak
MSSG4SRV
MSSG4SRVA
MSSG4SRVB
MSSG4SRVC
MSSG4SRVD
MSSG4SRVE
Atomospheric Steam Dump Valve FTRC
MSASDV
MSASDVAFTRC
MSASDVBFTRC
MSASDVCFTRC
MSASDVDFTRC
Steam Dump Valve (Condenser) FTRC
MSSDV
MSSDVA1
MSSDVA2
MSSDVB1
MSSDVB2
MSSDVC1
MSSDVC2


Seabrook BOP Shutdown
REACTOR_SHUTDOWN
SG Failure
9
SG_LOW
CW Failure
3
CW
Main Turbine
11
TURBINE_TRIP
Generator Failure
6
GENERATOR
Feed water
5
FEED
Condensate Water System Failure
2
CONDENSATESYS
Condenser failure
2.1E-1
CONDENSER


SG Failure
SG_LOW
SG(A)
SG(B)
SG(D)
SG(C)
4.2E-2
SG(A)
4.2E-2
SG(B)
4.2E-2
SG(D)
4.2E-2
SG(C)

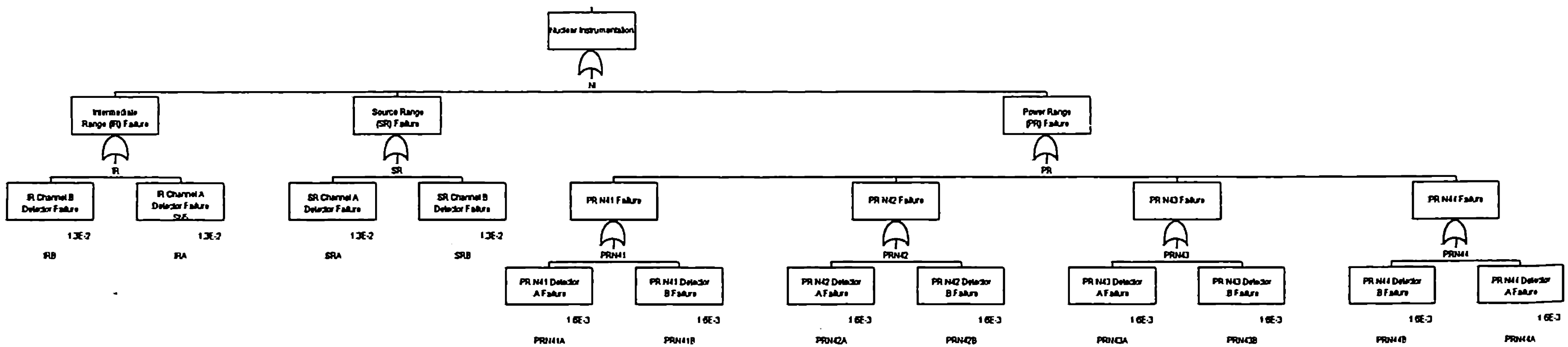


TMSR - Moisuture Separator/ Reheater Failure 1999/05/14 Page 10

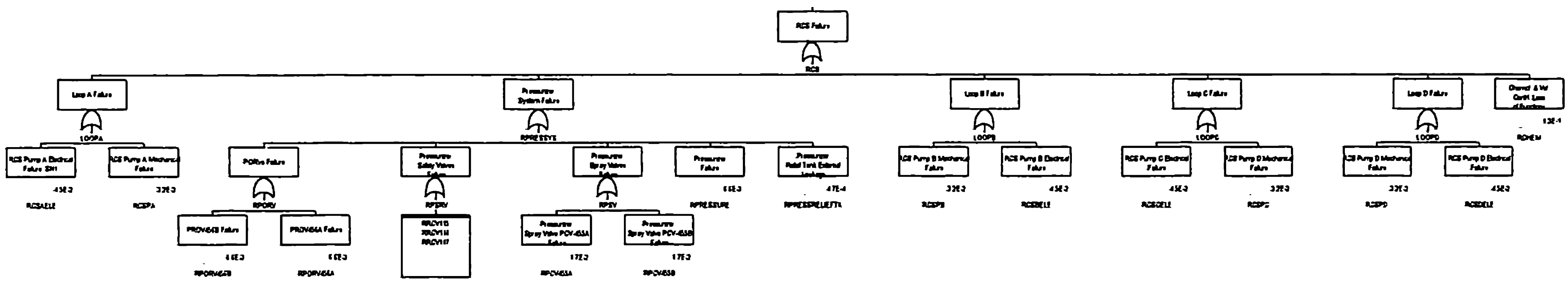

Main Turbine
TURBINE_TRIP
EHC Failure
4
EHC
MSL Valve Failure
7
MSLVALVE
MS/R Failure
10
TMSR
Turbine Trip Test&Maintenance (Valves, Interlocks)
2.8E-1
TTESTMAN

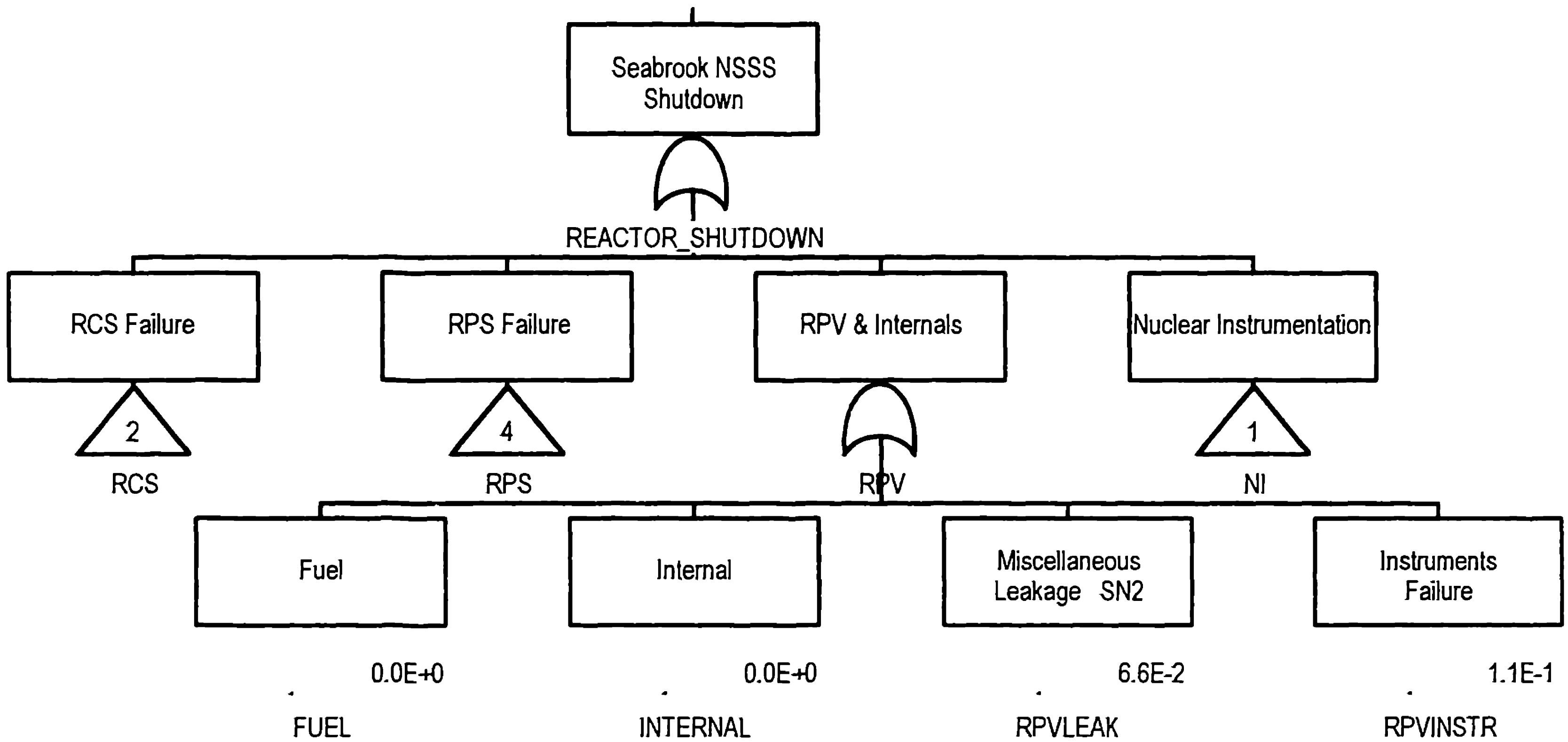
Appendix D

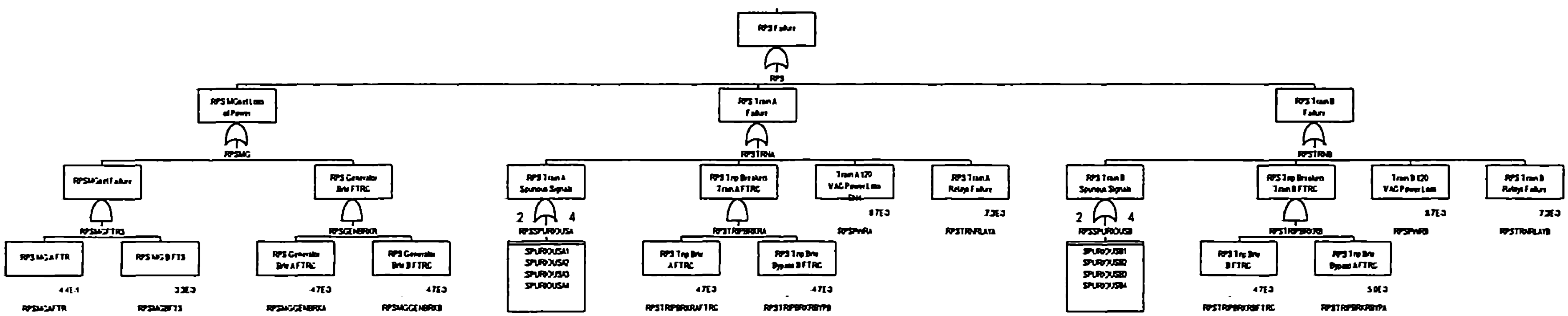
Fault Tree Models of the Nuclear Steam Supply System at the Seabrook Nuclear Power Station



Nuclear Instrumentation
NI
Intermediate Range (IR) Failure
IR
IR Channel B Detector Failure
1.3E-2
IRB
IR Channel A Detector Failure
1.3E-2
IRA
Source Range (SR) Failure
SR
SR Channel A Detector Failure
1.3E-2
SRA
SR Channel B Detector Failure
1.3E-2
SRB
Power Range (PR) Failure
PR
PR N41 Failure
PRN41
PR N41 Detector A Failure
1.6E-3
PRN41A
PR N41 Detector B Failure
1.6E-3
PRN41B
PR N42 Failure
PRN42
PR N42 Detector A Failure
1.6E-3
PRN42A
PR N42 Detector B Failure
1.6E-3
PRN42B
PR N43 Failure
PRN43
PR N43 Detector A Failure
1.6E-3
PRN43A
PR N43 Detector B Failure
1.6E-3
PRN43B
PR N44 Failure
PRN44
PR N44 Detector B Failure
1.6E-3
PRN44B
PR N44 Detector A Failure
1.6E-3
PRN44A




Seabrook NSSS Shutdown
REACTOR_SHUTDOWN
RCS Failure
2
RCS
RPS Failure
4
RPS
RPV & Internals
RPV
Nuclear Instrumentation
1
NI
Fuel
0.0E+0
FUEL
Internal
0.0E+0
INTERNAL
Miscellaneous Leakage SN2
6.6E-2
RPVLEAK
Instruments Failure
1.1E-1
RPVINSTR


RPS Failure
RPS
RPS Train A Failure
RPSTRNA
RPS Train B Failure
RPSTRNB
RPSMG
RPS Generator Brkr FTRC
RPSGENBRKR
RPS Train A Spurious Signals
RPS Trip Breakers Train A FTRC
Train A 120 VAC Power Loss
RPS Train A Relays Failure
RPS Train B Spurious Signals
RPS Trip Breakers Train B FTRC
Train B 120 VAC Power Loss
RPS Train B Relays Failure
2
4
RPSSPURIOUSA
RPSSPURIOUSB
RPS Trip Brkr A FTRC
RPS Trip Brkr Bypass B FTRC
RPS Trip Brkr B FTRC
RPS Trip Brkr Bypass A FTRC
RPS Generator Brkr A FTRC
RPS Generator Brkr B FTRC
RPSSPWRA
RPSSPWRB
RPSTRNRLAYA
RPSTRNRLAYB
4.4E-1
3.3E-3
4.7E-3
8.7E-3
7.3E-3
5.0E-3