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Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events

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Sandia National Laboratories

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ABSTRACT

This document contains the accident sequence analysis of internally initiated events for the Peach Bottom, Unit 2 Nuclear Power Plant. This is one of the five plant analyses conducted as part of the NUREG-1150 effort for the Nuclear Regulatory Commission. The work performed and described here is an extensive reanalysis of that published in October 1986 as NUREG/CR-4550,Volume 4. It addresses comments from numerous reviewers and significant changes to the plant systems and procedures made since the first report. The uncertainty analysis and presentation of results are also much improved, and considerable effort was expended on an improved analysis of loss of offsite power. The content and detail of this report are directed toward PRA practitioners who need to know how the work was done and the details for use in further studies.

The mean core damage frequency is 4.5E-6 with 5% and 95% uncertainty bounds of 3.5E-7 and 1.3E-5, respectively. Station blackout type accidents (loss of all AC power) contributed about 46% of the core damage frequency with Anticipated Transient Without Scram (ATWS) accidents contributing another 42%. The numerical results are driven by loss of offsite power, transients with the power conversion system initially available, operator errors, and mechanical failure to scram. External events were also analyzed using the internal event fault tree and event tree models as a basis, and are reported separately in Part 3 of NUREG/CR-4550, Volume 4, Revision 1.

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FOREWORD

This is one of numerous documents that support the preparation of the NUREG-1150 document by the NRC Office of Nuclear Regulatory Research. Figure 1 illustrates the front-end documentation. There are three interfacing programs at Sandia National Laboratories performing this work: the Accident Sequence Evaluation Program (ASEP), the Severe Accident Risk Reduction Program (SARRP), and the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP). The Zion PRA was performed at Idaho National Engineering Laboratories and Brookhaven National Laboratories.

Table 1 is a list of the original primary documentation and the corresponding revised documentation. There are several items that should be noted. First, in the original NUREG/CR-4550 report, Volume 2 was to be a summary of the internal analyses. This report was deleted. In Revision 1, Volume 2 now is the expert judgment elicitation covering all plants.

Volumes 3 and 4 include external events analyses for Surry and Peach Bottom. External events for Sequoyah, Grand Gulf and Zion will be analyzed in follow-up studies after NUREG-1150 is published.

The revised NUREG/CR-4551 covers the analysis included in the original NUREG/CR-4551 and NUREG/CR-4700. However, it is different from NUREG/CR-4550 in that the results from the expert judgment elicitation are given in four parts to Volume 2 with each part covering one category of issues. The accident progression event trees are given in the appendices for each of the plant analyses.

Originally, NUREG/CR-4550 was published without the designation "Draft for Comment." Thus, the final revision of NUREG/CR-4550 is designated Revision 1. The label Revision 1 is used consistently on all volumes, including Volume 2 which was not part of the original documentation. NUREG/CR-4551 was originally published as a "Draft for Comment" so, in its final form, no Revision 1 designator is required to distinguish it from the previous documentation.

There are several other reports published in association with NUREG-1150. These are:

NUREG/CR-5032, SAND87-2428, Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-site Power Incidents at Nuclear Power Plants, R. L. Iman and S. C. Hora, Sandia National Laboratories, Albuquerque, NM, January 1988.

NUREG/CR-4840, SAND88-3102, Methodology for External Event Screening Quantification - RMIEP Methodology, M. P. Bohn and J. A. Lambright, Sandia National Laboratories, Albuquerque, NM, July 1989.

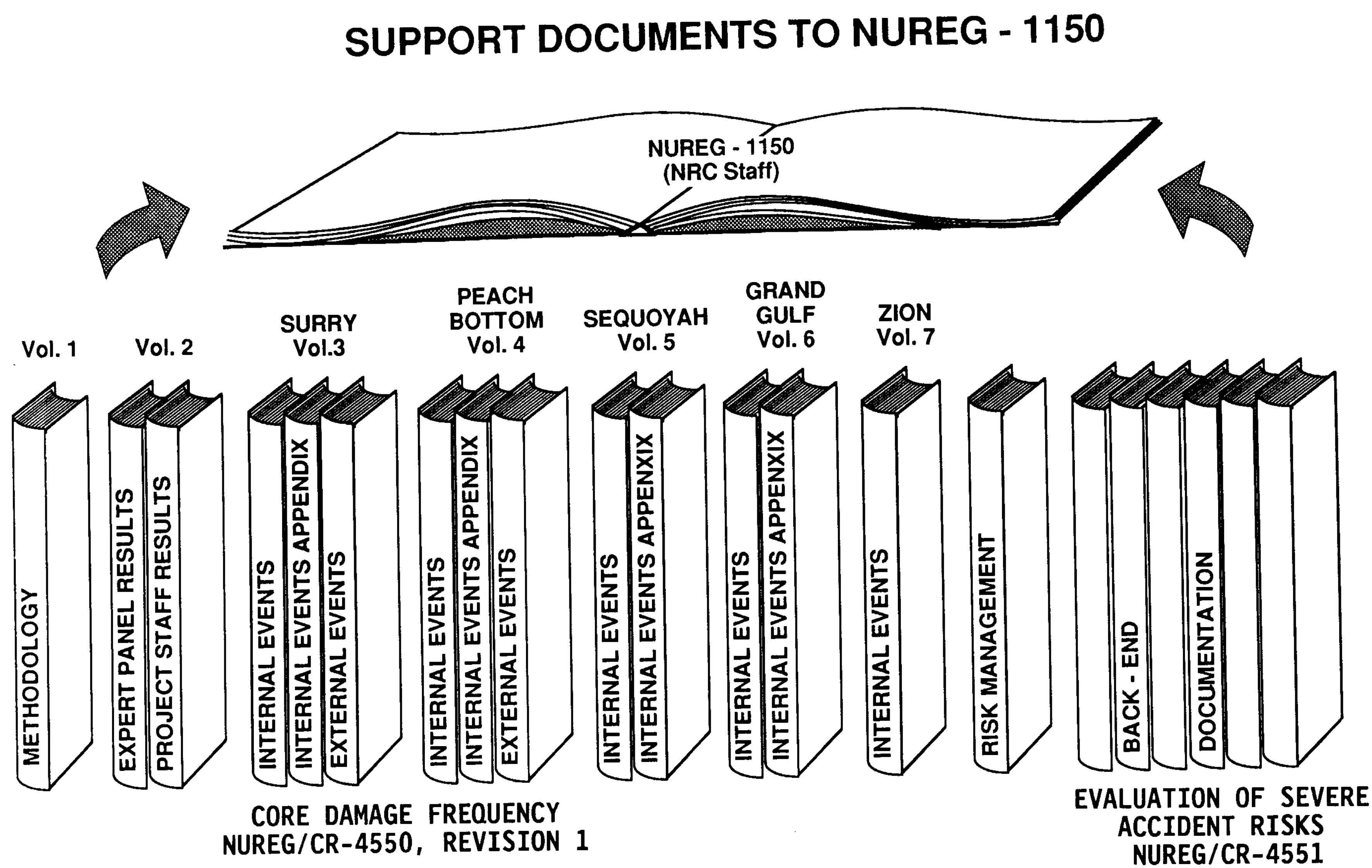

FIGURE 1. DOCUMENTATION FOR NUREG-1150.

Table 1.
NUREG-1150 Analysis Documentation

Original Documentation

NUREG/CR-4550
Analysis of Core Damage Frequency From Internal Events

Volume 1		Methodology
2		Summary (Not Published)
3		Surry Unit 1
4		Peach Bottom Unit 2
5		Sequoyah Unit 1
6		Grand Gulf Unit 1
7		Zion Unit 1

NUREG/CR-4551
Evaluation of Severe Accident Risks and the Potential for Risk Reduction

Volume 1	Surry Unit 1
2	Sequoyah Unit 1
3	Peach Bottom Unit 2
4	Grand Gulf Unit 1

NUREG/CR-4700
Containment Event Analysis for Potential Severe Accidents

Volume 1	Surry Unit 1
2	Sequoyah Unit 1
3	Peach Bottom Unit 2
4	Grand Gulf Unit 1

Revised Documentation

NUREG/CR-4550, Revision 1
Analysis of Core Damage Frequency

Volume 1		Methodology
2	Part 1	Expert Judgment Elicit. Expert Panel
	Part 2	Expert Judgment Elicit.--Project Staff
3	Part 1	Surry Unit 1 Internal Events
	Part 2	Surry Unit 1 Internal Events App.
	Part 3	Surry Unit 1 External Events
4	Part 1	Peach Bottom Unit 2 Internal Events
	Part 2	Peach Bottom Unit 2 Internal Events App.
	Part 3	Peach Bottom Unit 2 External Events
5	Part 1	Sequoyah Unit 1 Internal Events
	Part 2	Sequoyah Unit 1 Internal Events App.
6	Part 1	Grand Gulf Unit 1 Internal Events
	Part 2	Grand Gulf Unit 1 Internal Events App.
7		Zion Unit 1 Internal Events

NUREG/CR-4551, **Evaluation of Severe Accident Risks**

Volume 1		**Methodology**
2	Part 1	**Expert Judgment Elicit.--In-vessel**
	Part 2	**Expert Judgment Elicit.--Containment**
	Part 3	**Expert Judgment Elicit.--Structural**
	Part 4	**Expert Judgment Elicit.--Source-Term**
	Part 5	**Expert Judgment Elicit.--Supp. Calc.**
	Part 6	**Expert Judgment Elicit.--Proj. Staff**
	Part 7	**Expert Judgment Elicit.--Supp. Calc.**
	Part 8	**Expert Judgment Elicit.--MACCS Input**
3	Part 1	Surry Unit 1 Anal. and Results
	Part 2	Surry Unit 1 Appendices
4	Part 1	Peach Bottom Unit 2 Anal. and Results
	Part 2	Peach Bottom Unit 2 Appendices
5	Part 1	Sequoyah Unit 2 Anal. and Results
	Part 2	Sequoyah Unit 2 Appendices
6	Part 1	Grand Gulf Unit 1 Anal. and Results
	Part 2	Grand Gulf Unit 1 Appendices
7	Part 1	Zion Unit 1 Anal. and Results
	Part 2	Zion Unit 1 Appendices

NUREG/CR-4772, SAND86-1996, Accident Sequence Evaluation Program Human Reliability Analysis Procedure, A. D. Swain III, Sandia National Laboratories, Albuquerque, NM, February 1987.

NUREG/CR-5263, SAND88-3100, The Risk Management Implications of NUREG-1150 Methods and Results, A. L. Camp et al., Sandia National Laboratories, Albuquerque, NM, May 1989.

A Human Reliability Analysis for the ATWS Accident Sequence with MSIV Closure at the Peach Bottom Atomic Power Station, A-3272, W. J. Luckas, Jr. et al., Brookhaven National Laboratory, Upton, NY, 1986.

A brief flow chart for the documentation is given in Figure 2. Any related supporting documents to the back-end NUREG/CR-4551 analyses are delineated in NUREG/CR-4551. A complete list of the revised NUREG/CR-4550, Revision 1 volumes and parts is given below.

General

NUREG/CR-4550, Revision 1, Volume 1, SAND86-2084, Analysis of Core Damage Frequency: Methodology Guidelines for Internal Events.

NUREG/CR-4550, Revision 1, Volume 2, Part 1, SAND86-2084, Analysis of Core Damage Frequency: Expert Judgment Elicitation on Internal Events Issues - Expert Panel.

NUREG/CR-4550, Revision 1, Volume 2, Part 2, SAND86-2084, Analysis of Core Damage Frequency: Expert Judgment Elicitation on Internal Events Issues - Project Staff.

Parts 1 and 2 of Volume 2, NUREG/CR-4550 were published in one binder. This volume was published in April 1989 and distributed in May 1989 with an incorrect title, i.e., Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation, without the Revision 1 designation. The complete, correct title is: NUREG/CR-4550, Revision 1, Volume 2, SAND86-2084, Analysis of Core Damage Frequency: Expert Judgment Elicitation on Internal Events Issues.

Surry

NUREG/CR-4550, Revision 1, Volume 3, Part 1, SAND86-2084, Analysis of Core Damage Frequency: Surry Unit 1 Internal Events.

NUREG/CR-4550, Revision 1, Volume 3, Part 2, SAND86-2084, Analysis of Core Damage Frequency: Surry Unit 1 Internal Events Appendices.

NUREG/CR-4550, Revision 1, Volume 3, Part 3, SAND86-2084, Analysis of Core Damage Frequency: Surry Unit 1 External Events.

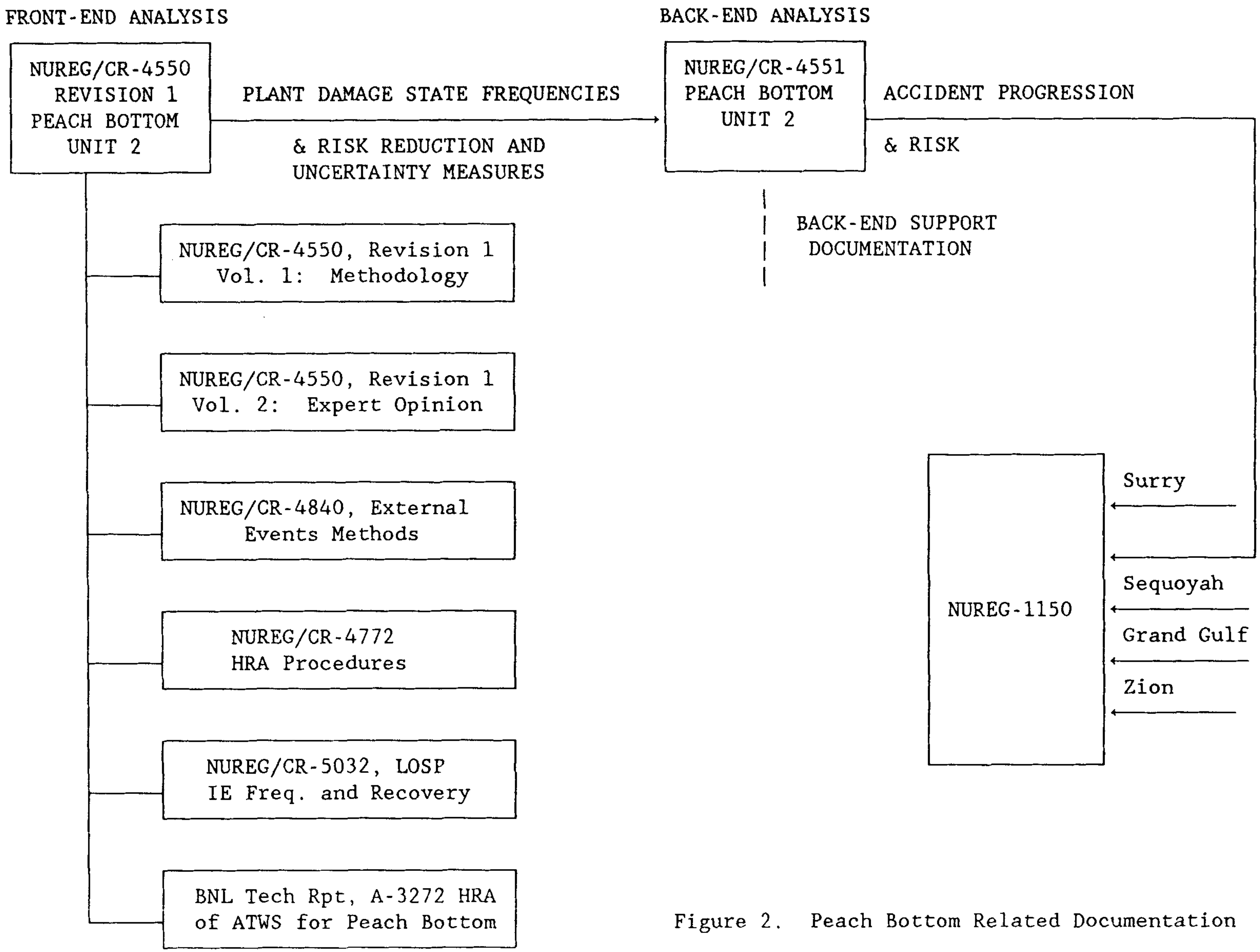

Figure 2. Peach Bottom Related Documentation

Peach Bottom

NUREG/CR-4697, EGG-2464, Containment Venting Analysis for the Peach Bottom Atomic Power Station, D. J. Hansen, et al., Idaho National Engineering Laboratory (EG&G Idaho, Inc.) February 1987.

NUREG/CR-4550, Revision 1, Volume 4, Part 1, SAND86-2084, Analysis of Core Damage Frequency: Peach Bottom Unit 2 Internal Events.

NUREG/CR-4550, Revision 1, Volume 4, Part 2, SAND86-2084, Analysis of Core Damage Frequency: Peach Bottom Unit 2 Internal Events Appendices.

NUREG/CR-4550, Revision 1, Volume 4, Part 3, SAND86-2084, Analysis of Core Damage Frequency: Peach Bottom Unit 2 External Events.

Sequoyah

NUREG/CR-4550, Revision 1, Volume 5, Part 1, SAND86-2084, Analysis of Core Damage Frequency: Sequoyah Unit 1 Internal Events.

NUREG/CR-4550, Revision 1, Volume 5, Part 2, SAND86-2084, Analysis of Core Damage Frequency: Sequoyah Unit 1 Internal Events Appendices.

Grand Gulf

NUREG/CR-4550, Revision 1, Volume 6, Part 1, SAND86-2084, Analysis of Core Damage Frequency: Grand Gulf Unit 1 Internal Events.

NUREG/CR-4550, Revision 1, Volume 6, Part 2, SAND86-2084, Analysis of Core Damage Frequency: Grand Gulf Unit 1 Internal Events Appendices.

Zion

NUREG/CR-4550, Revision 1, Volume 7, EGG-2554, Analysis of Core Damage Frequency: Zion Unit 1 Internal Events.

ACRONYMS AND INITIALISMS

ACP	ac power system
ACX	air cooling heat exchanger
ANS	American Nuclear Society
ADS	automatic depressurization system
AFW	auxiliary feedwater system
AOV	air-operated valve
ARI	alternate rod insertion
ARF	air return fan system
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CCF	common cause failure
CCU	containment atmosphere cleanup
CCW	component cooling water
CDF	core damage frequency
CDS	condensate system
CFC	containment emergency fan cooler
CGC	containment combustible gas control
CHP	charging pump
CHW	chilled water
CIS	containment isolation system
CLS	consequence limiting safeguards
CPC	charging pump cooling
CRD	control rod drive
CS	containment spray
CSR	containment spray recirculation
CSC	closed cycle cooling
CST	condensate storage tank
CSS	containment spray system
CVC	chemical and volume control
DCP	DC power system
DEP	depressurization
DG	diesel generator
DWS	drywell (wetwell) spray
ECCS	emergency core cooling system
ECW	emergency cooling water
EHS	emergency heat sink
EHV	emergency heating, ventilation
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
EPS	electric power system
ESF	engineered safety feature
ESW	essential service water
ESW	emergency service water
EVS	emergency ventilation system

FCD	functional control diagram
FHS	fuel handling system
FSAR	final safety analysis report
FW	feedwater
HEP	human error probabilities
HPCI	high pressure coolant injection
HPCS	high pressure core spray
HPR	high pressure recirculation
HPSI	high pressure safety injection
HPSW	high pressure service water
HRA	human reliability analysis
HTX	heat exchanger
IAS	instrument air system
ICS	ice condenser system
ICSR	inside containment spray recirculation
IE	initiating event
ILRT	integrated leak rate test
INEL	Idaho National Engineering Laboratory
IORV	inadvertent open relief valve
IREP	Interim Reliability Evaluation Program
ISO	isolation condenser
LER	licensee event report
LHS	latin hypercube sampling
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOSP	loss of offsite power
LLNL	Lawrence Livermore National Laboratory
LPCI	low pressure coolant injection
LPCS	Low pressure core spray
LPR	low pressure recirculation
LPSI	low pressure safety injection
LWR	light water reactor
MCC	motor control center
MCW	main circulating water
MFW	main feedwater
MOV	motor-operated valve
MSIV	main steam isolation valve
MSS	main steam system
NHV	normal heating, ventilation
NPRDS	national plant reliability and data system
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSW	normal service water
OEP	onsite electric power
OOS	out of service

ACRONYMS AND INITIALISMS (Cont.)

ORNL	Oak Ridge National Laboratory
OSR	outside containment spray recirculation
P&ID	piping and instrumentation diagram
PCS	power conversion system
PCV	primary containment venting
PDS	plant damage state
PECO	Philadelphia Electric Company
PLG	Pickard, Lowe, & Garrick
PORV	power-operated relief valve
PPS	primary pressure relief system
PRA	probabilistic risk analysis
PRUEP	Phenomenology and Risk Uncertainty Evaluation Program
PTS	pressurized thermal shock
QCG	quality control group
RBCW	reactor building cooling water
RCS	reactor coolant system
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RGW	radioactive gaseous water
RHR	residual heat removal
RMIEP	Risk Methods Integration and Evaluation Program
RLW	radioactive liquid waste
RMT	recirculation mode transfer
RPS	reactor protection system
RPSE	reactor protection system electrical
RPSM	reactor protection system mechanical
RPT	recirculation pump trip
RPV	reactor pressure vessel
RSS	Reactor Safety Study
RWCU	reactor water cleanup
RWST	refueling water storage tank
SAIC	Science Applications International Corporation
SARRP	Severe Accident Risk Reduction Program
SAS	service air system
SBO	station blackout
SCG	Senior Consultant Group
SDC	shutdown cooling
SGT	standby gas treatment
SIS	safety injection system
SLC	standby liquid control
SNL	Sandia National Laboratories
SPC	suppression pool cooling
SPM	suppression pool makeup
SRV	safety relief valve
SWS	service water system
TBCW	turbine building cooling water
TDP	turbine-driven pump

TEMAC	Top Event Matrix Analysis Code
TMI	Three Mile Island
UFSAR	updated final safety analysis report
VSS	vapor suppression system
/, $\overline{\ }$	In the text, two methods were used to show success of an event. Method one uses a slash preceding the event symbol, e.g., /LOSP or /C. Method two uses a bar over the event symbol, e.g., $\overline{\text{LOSP}}$ or $\overline{\text{C}}$.

ACKNOWLEDGEMENTS

The authors wish to acknowledge the efforts of all those involved in the Revision 1 analysis of the Peach Bottom plant. Major contributions were made by Ron Iman for his loss of offsite power analysis and TEMAC support, Michael Shortencarier for his development work on TEMAC, and Lanny Smith for his work on the ATWS thermal-hydraulic code simulations. Also, the efforts of Greg Krueger of Philadelphia Electric Company (PECO) and other PECO staff members are greatly appreciated. Greg was our interface with PECO throughout the analysis and documentation tasks. The continuous interaction between PECO and the analysis team were invaluable to the successful completion of the models and accident sequence analysis.

There were innumerable people involved in various support roles such as the reviewers, the quality control team, expert judgment elicitation participants, and the secretarial staff. In particular, Sarah J. Higgins organized numerous computer runs and Emily A. Preston prepared much of the documentation. A number of people at Science Applications International Corp. contributed to this work; Mary Drouin, Steve Miller, Nancy Cabber, Vickie Lucero, and Lorri Howe. Their efforts are much appreciated.

1. EXECUTIVE SUMMARY

This document presents the final results from one of several studies that will provide information to the Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research about Light Water Reactor (LWR) risk. The Office of Research will use the results of this work, along with other inputs, to prepare NUREG-1150. Risk from a selected group of five nuclear power plants is examined in NUREG-1150 by incorporating the results of wide-ranging research efforts that have taken place over the past several years. These results will provide the bases for updating the perception of risk from selected plants, developing methods for extrapolation to other plants, comparing NRC research to industry results, and resolving numerous severe accident issues. The level of detail and subjects covered are for the Probabilistic Risk Assessment (PRA) practitioner.

Peach Bottom was chosen as one of the five plants to be analyzed to accomplish these goals. The Peach Bottom Atomic Power Station is located in southeastern Pennsylvania in York County on the west shore of Conowingo Pond and includes two Boiling Water Reactor (BWR) units each of 1150 megawatts (electrical) capacity. The reactors are both housed in Mark I containments. Peach Bottom Unit 2, analyzed in this study, began commercial operation in July 1974 and is operated by Philadelphia Electric Company (PECO). The Peach Bottom plant was previously analyzed in the Reactor Safety Study (WASH-1400). Other plants that were chosen to be analyzed are Surry, Sequoyah, Grand Gulf, and Zion.

1.1 Objectives

The primary objective was to perform an analysis to support the NUREG-1150 project that is an efficient Level 1 Probabilistic Risk Assessment (PRA) that is as near to a state of the art as possible. Corresponding Level 2 and Level 3 analyses have also been performed and documented. External events were analyzed and are reported in Part 3 of this volume.

Direct objectives of the analysis are to identify potential, significant system failures, to support improved plant operations, to provide insights of value to utilities with plants of this type, and to support a detailed methodology that can be used by others including utilities. The perspectives gained from NUREG-1150 will be used to support the NRC severe accident policy and a variety of regulatory issues dealing with severe accidents.

This document presents the front-end part of the risk equation, i.e., the frequency of scenarios involving system failures which lead to severe core damage as a result of internal initiators.*

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* Core damage is defined as a significant core uncovery occurrence with reflooding of the core not imminently expected. The result is a prolonged uncovery of the core which leads to damaged fuel and an expected release of fission products from the fuel.

1.2 Approach

A standard Level 1 PRA approach formed the basis for this analysis. Event trees were constructed, the top events were modeled using large fault trees where required and quantified using the SETS and TEMAC computer codes.

There is a wealth of information available on Peach Bottom since it has been the subject of many studies. Using this information, an experienced PRA team analyzed only those aspects of the plant that they judged to be important. Thus, time was not spent analyzing areas that had been shown to be unimportant in the past. Also, if the analyst determined that a system could be represented adequately with a simplified model rather than a detailed fault tree, then the simplified approach was chosen. However, if the analyst determined that a system was important enough to warrant detailed modeling, then the appropriate modeling techniques were chosen.

As part of the basic PRA methodology, four areas merit comment. First, a human reliability analysis was performed on operator actions that surfaced in the PRA as potentially significant. Second, data was collected from several sources and verified for accuracy and applicability. Third, a recovery analysis was performed to assure proper credit was given for operator intervention during the accident. Finally, an extensive uncertainty analysis was performed. This required determining the uncertainty on the failure probabilities for basic events in the models. In some cases, no firm data existed, so expert judgment was formally elicited from people with extensive experience on each issue in question. This is the subject of Volume 2 of NUREG/CR-4550, Revision 1.

In addition to the typical Level 1 analysis, the results were reconstituted in a form suitable for input to the back-end accident progression event trees. Plant damage states* were defined in a joint effort between the front-end and back-end analysts. Statistical analyses identical to those for the accident sequences were performed on the plant damage states.

In order to maintain high quality, this work was reviewed by four different groups: an independent Senior Consultant Group (SCG), an independent Quality Control Group (QCG), Sandia staff and management, and the NRC. In addition, the staff at PECO were given an opportunity to review this work at various stages. PECO's comments were addressed in this analysis as were numerous comments received from the NRC, the public, and the nuclear industry.

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* A plant damage state is a grouping of accident sequences or parts of accident sequences that have similar characteristics such as vessel pressure, timing, containment response, and system failures which provides the necessary input for the accident progression event tree used in the Level 2 analysis.

1.3 Results

The Peach Bottom PRA identified two major accident types which contribute 89% of the core damage frequency (CDF). These accident types, station blackout [loss of offsite power (LOSP) transient with failure of the diesel generators] and Anticipated Transient Without Scram (ATWS), as well as other less important types of accidents, collectively cover a variety of plant damage states (see Figure 1-1). The mean core damage frequency at Peach Bottom was calculated to be 4.5E-6. The cumulative probability distribution and the corresponding probability density estimation for the total core damage frequency for Peach Bottom are given in Figure 1-2 where all of the accident sequences are combined statistically using a sample size of 1000. The corresponding statistics are:

Mean	4.5E-6
Standard Deviation	1.5E-5
Lower 5%	3.5E-7
Lower 25%	9.2E-7
Median	1.9E-6
Upper 25%	3.9E-6
Upper 5%	1.3E-5

Every accident sequence is the sum of one or more combinations of events that lead to core damage. These combinations of events are the detailed scenarios of the minimum sequence of failures (component and human) that result in core damage. They are defined as "cut sets." There were 1393 cut sets in the 18 dominant accident sequences in the final Peach Bottom front-end analysis. The top two cut sets contributed 36% of the total CDF. The top twenty cut sets contributed 68% of the total CDF. The top 350 cut sets account for 95% of the total CDF.

Among the most important results of the analysis are the results of the importance measure calculations. It is most illustrative to look at each of the importance measures for the total CDF. The risk reduction importance measure ranks the basic events by the reduction in CDF if that event probability were set to zero. The most significant risk reduction events for the Peach Bottom CDF are:

- Mechanical failure of the reactor protection system,
- Transient accident initiator with the power conversion system initially available,
- Transient accident initiator from loss of offsite power,
- Operator failure to restore the standby liquid control system after testing,
- Other operator failures to initiate systems or miscalibration of sensors, and
- Diesel generator failure-to-run.

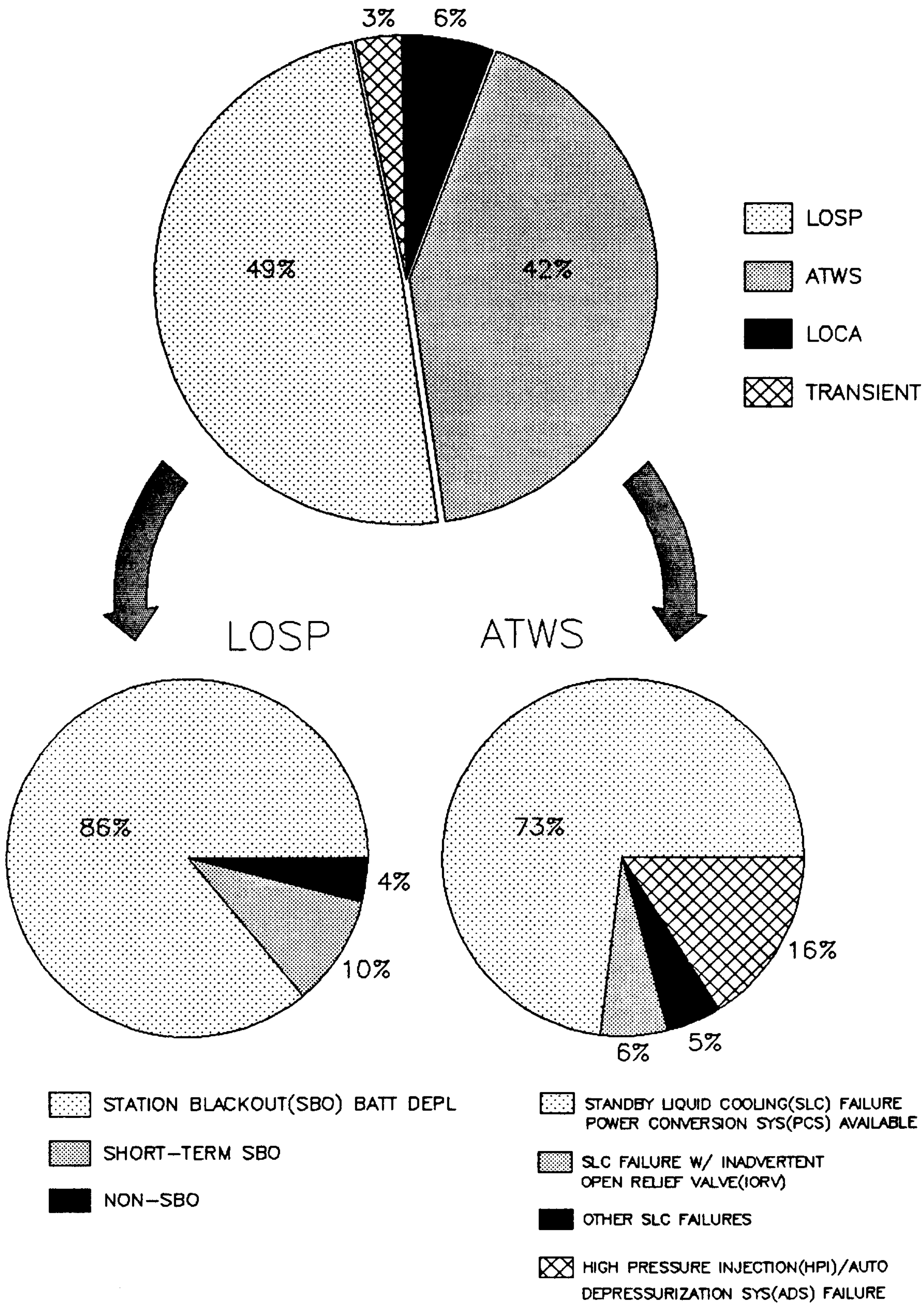
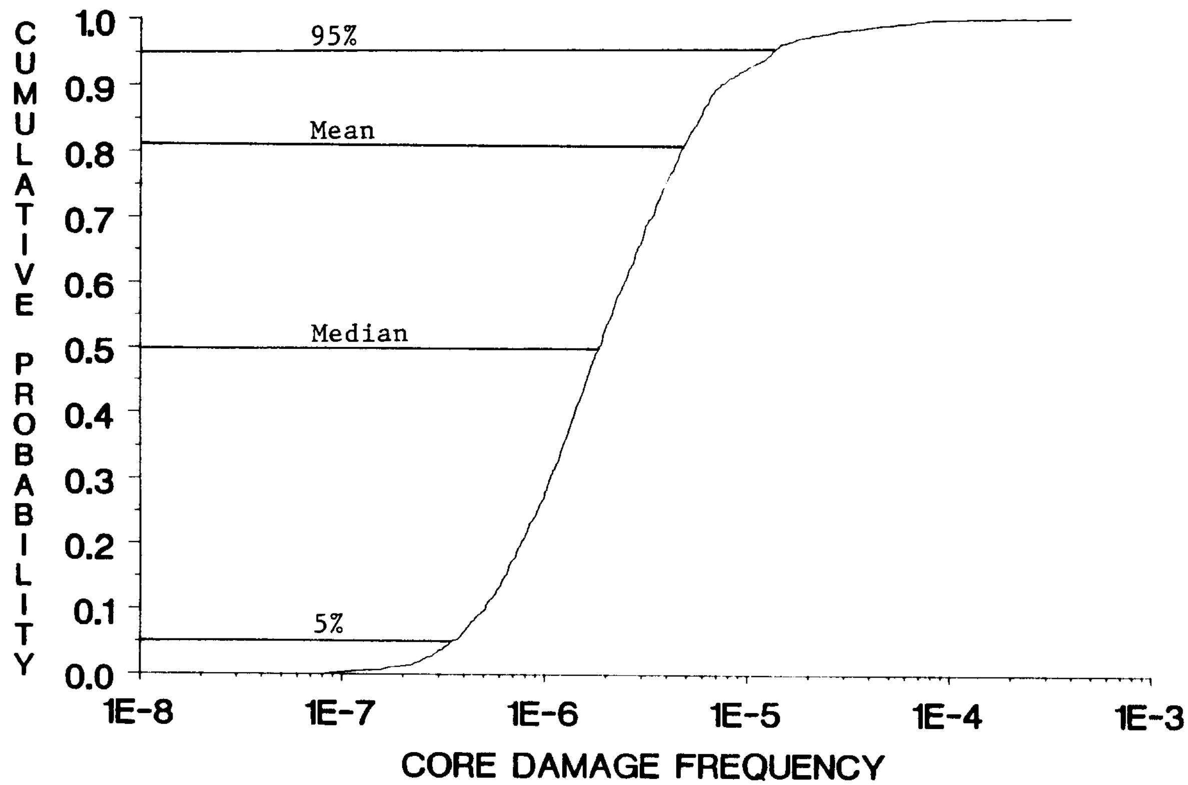
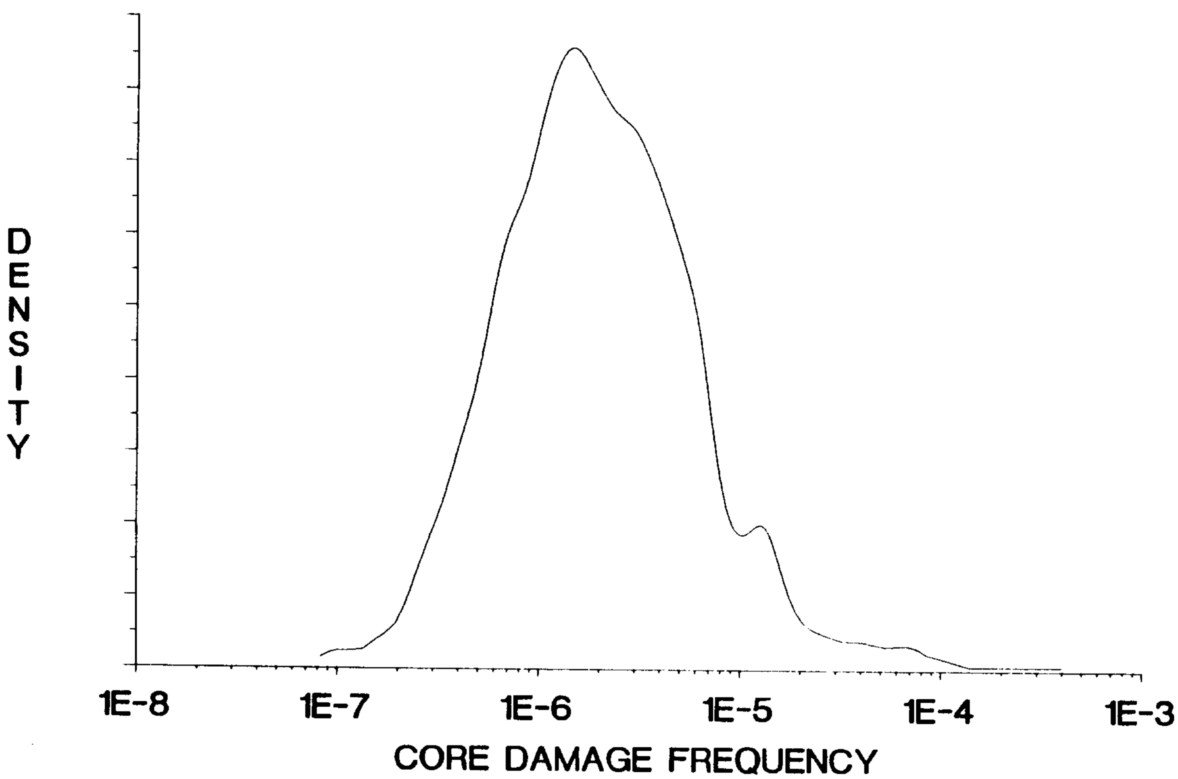


Figure 1-1. Peach Bottom Core Damage Frequency Types



UNCERTAINTY DISTRIBUTION FOR PEACH BOTTOM



DENSITY ESTIMATE FOR PEACH BOTTOM

Figure 1-2. Total Internal Event Core Damage Frequency for Peach Bottom

The inverse of risk reduction is risk increase, which estimates the CDF if an event probability is set to one. The importance of events ranked by this measure is that relaxed vigilance could cause significant CDF increases. Top risk increase events for Peach Bottom are:

- Mechanical failure of the reactor protection system,
- Operator miscalibration of the reactor vessel pressure permissive sensors used for low pressure injection,
- Common cause failure of the station batteries, and
- Two stuck-open safety-relief valves contributing to a loss of coolant injection following transient initiators.

Several events appear high in both risk measures, especially mechanical failure of the reactor protection system. Much more extensive lists of events relating to the risk measures are given in Section 5.4 and Appendix F of this report. The third importance measure is the relative importance of event uncertainties in the analysis. This will be discussed in Section 1.4.4.

1.4 Conclusions

One of the major purposes of the Peach Bottom analysis was to provide an updated perspective on our understanding of the risks from the plant relative to the results of the WASH-1400 analysis. It has been determined that changes to the plant design and its procedures, the evolution of Probabilistic Risk Assessment (PRA) methodology and an increasing understanding of severe accidents have all impacted the perspectives on the dominant risks for Peach Bottom.

This study concludes that station blackout (loss of all AC power) accidents and Anticipated Transients Without Scram (ATWS) scenarios are the dominant contributors to core damage at Peach Bottom. The possibility of successful containment venting and realistically allowing for successful core cooling after containment failure have considerably reduced the significance of the loss of long term heat removal accidents originally found to be important in the Reactor Safety Study (WASH-1400). Giving credit for more injection systems, using realistic system success criteria, and plant modifications have also collectively reduced the importance of loss of injection type sequences.

Given the considerable redundancy and diversity of coolant injection and heat removal features at Peach Bottom, it is not surprising that common features of the plant tend to drive the mean core damage frequency. These include common cause failures of equipment, failure of common support systems [AC power and Emergency Service Water (ESW)], and human error. In light of this conclusion, it must also be recognized that the calculated core damage frequency in this study is subject to the non-trivial uncertainties associated with the common cause and human error analyses.

The above insights can be considered applicable to other boiling water reactors of similar design to the extent that the redundancy arguments are true for other plants of interest. However, numerous subtleties in plant design and operational practices and procedures make it difficult to draw specific conclusions for other plants on the basis of this analysis without performing plant-specific reviews. Such reviews should consider plant-specific common cause failure potential and the location of equipment that might be subjected to possible phenomena such as steam entering the reactor building.

1.4.1 Plant Specific Conclusions

As stated above, the core damage profile is primarily made up of two general types of accidents as indicated below:

Accident Type	Mean Frequency	% Contribution to Mean Core Damage Frequency*
LOSP	2.2E-6	49
ATWS	1.9E-6	42
All Others	4.0E-7	9

*Does not account for the ~3% contribution of sequences <1E-8

These general accident types are made up of eighteen individual accident sequences or, alternatively, nine plant damage states.

1.4.2 Accident Sequence Conclusions

The accident sequence with the highest contribution to core damage frequency is a loss of offsite power transient with failure of the diesel generators (station blackout) and late failure of the high pressure systems. The high pressure systems are initially operating, but later in the sequence either battery depletion or harsh environments cause system failure. This is a late core damage sequence and contributes 36% of the total core damage frequency.

The second highest accident sequence contributor is a transient with the power conversion system initially available and mechanical failure of the reactor protection system (anticipated transient without scram). The standby liquid control system also fails, leading to core damage. This accident sequence contributes 31% of the total core damage frequency.

1.4.3 Plant Damage State Conclusions

From a plant damage state perspective, two plant damage states dominate the core damage frequency. Plant damage state 5 contributes 42% of the total. This plant damage state is a transient loss of offsite power and subsequent failure of all diesel generators (station blackout). The high pressure injection systems initially operate, but fail later due to battery depletion or harsh environments. The second highest contributor is an anticipated transient without scram with the standby liquid control system also failing. This plant damage state contributes 33% of the total core damage frequency.

1.4.4 Uncertainty Considerations

The process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model also can be used to assess the importance of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using an uncertainty importance measure is summarized below. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the following events were found to be important:

- Technical failure of the reactor protection system,
- Failure of the diesel generators to continue to run once started,
- Battery depletion time in station blackout accidents,
- Miscalibration of the low reactor pressure permissive instrumentation,
- Operator failure to restore the standby liquid control system after testing.

1.4.5 Comparison to Reactor Safety Study (WASH-1400)

In over ten years between WASH-1400 and this study, the Peach Bottom plant design, as well as the industry's understanding of reactor operation and safety, has changed substantially. Any comparison of dominant contributors to core damage frequency between these studies must be balanced by a knowledge of the differences in plant design, study methodology, and success criteria considerations.

It is difficult to directly compare the total core damage frequencies calculated in the two studies. WASH-1400 calculated a total core damage frequency of approximately 2.6E-5, which is a sum of individual sequence median values (note that the sum is not necessarily a median value).

This study has determined the median core damage frequency at Peach Bottom to be 1.9E-6 with a corresponding mean value of 4.5E-6. The modifications in plant configuration and procedures at Peach Bottom, consideration of realistic success criteria, as well as the evolution of analysis techniques since WASH-1400 have reduced the dominant results of the WASH-1400 study considerably. In fact, the two most dominant scenarios from the WASH-1400 study (transient with loss of long-term decay heat removal [TW] and ATWS [TC] have been decreased by factors of approximately 1000 and 25, respectively. However, a more complete consideration of failures of DC-powered systems during station blackout and a more comprehensive treatment of common cause failures and support system (e.g., power, cooling...) failures combine to yield a mean core damage frequency of 4.5E-6. Some of the significant comparisons leading to these insights are presented below.

- Transients with loss of long-term decay heat removal are dominant in WASH-1400, but not in this study. This is primarily because of the consideration of containment venting procedures now in place at Peach Bottom and an examination of the survivability of core cooling systems.

- ATWS sequence frequencies are reduced over an order of magnitude in this study as compared to WASH-1400 because a more detailed analysis was performed which more accurately treats the sequence thermal hydraulics and accounts for the provisions of the ATWS rule.

- Station blackout (loss of all AC) sequences are estimated to be a factor of five higher than in WASH-1400 because of a more complete consideration of potential failures of DC-powered systems during a blackout, a more complete common mode failure analysis (e.g., includes DC battery common mode failures), and a more complete analysis of support system effects on the AC power system (e.g., diesel cooling).

- All other transients and LOCAs combine to have a median CDF of 1.5E-6 in WASH-1400 and a median CDF of 7.5E-8 in this study. Thus, these sequences are a factor of 20 lower in this study.

- Based on the above, both studies conclude that transients, and not LOCAs, dominate the core damage frequency (and risk) at Peach Bottom. However, the types of transients are significantly different. WASH-1400 is dominated by ATWS and long-term heat removal failure sequences, while this study is dominated by station blackout scenarios (47%) and ATWS (42%).

Table 1-1 summarizes the comparable core damage frequencies for the most dominant sequences as well as for the total core damage frequency results of both studies. The sum of the median frequencies from WASH-1400 is 2.6E-5. Although the overall TEMAC median result is 1.9E-6, the sum of the individual PDS median frequencies, which is comparable to what was done in WASH-1400, is 9.1E-7. Thus, in comparable terms, the core damage frequency from the NUREG/CR-4550, Revision 1 analysis on Peach Bottom is about a factor of 30 less than the WASH-1400 value.

1.4.6 Other Insights

Some additional insights are noted by the team analysts as a result of performing the PRA update of Peach Bottom. The recent availabilities of the diesel generators at Peach Bottom generally are a factor of ten better than the industry average. This appears to be based on a deliberate attention to detail in the test and maintenance practices as well as an attempt to determine the root causes of failures so that effective actions can be taken.

The importance of the Control Rod Drive (CRD) and High Pressure Service Water systems as injection sources to the vessel (the latter as a last resort) came through clearly as the analysis evolved. The CRD system success probability might be further improved by examining whether the loss of air should be allowed to affect the operation of one of the CRD flow paths to the vessel. In addition, the use of CRD under depressurized conditions in the vessel could cause insufficient net positive suction head for the CRD pumps.

An air pressure limit for Safety Relief Valve (SRV) operation of approximately 100 psia could affect the capability to continue low pressure core cooling under accident conditions when the containment is at high pressure (i.e., SRVs will not stay open).

The conflicting requirements of first inhibiting the automatic depressurization system and then needing to rapidly depressurize in some ATWS sequences should be recognized.

The difficulties associated with venting the containment in a station blackout and the harsh reactor building environments caused by venting in ATWS scenarios have significant core damage and consequence effects.

Finally, the varied and more subtle failures of equipment because of unusual accident conditions are important factors. These failures include, for instance, turbine backpressure trip of the Reactor Core Isolation Cooling (RCIC) system when experiencing high containment pressure, the potential for High Pressure Coolant Injection (HPCI) and RCIC system failure on high suppression pool temperatures, the closing of the SRVs under very high containment pressures, the potential for loss of low pressure core spray and residual heat removal pumps under low pressure saturated conditions in the containment, and the possible effects of battery depletion when AC power is lost, among others. It is these subtle and perhaps "unexpected" failure modes which affect multiple equipment in the analyzed scenarios and ultimately contribute to the core damage potential at Peach Bottom.

Table 1-1
Comparison of NUREG/CR-4550, Revision 1 and WASH-1400 Sequences

General Accident Type	NUREG/CR-4550, Revision 1		Approximate WASH-1400	
	Mean (Median) Frequencies(1)	% of Total	Median Frequency	% of Total
Station Blackout	2.1E-6 (4.5E-7)	47%	1.0E-7	<1%
Anticipated Transient Without Scram (ATWS)	1.9E-6 (3.8E-7)	42%	1.0E-5	39%
Transient-Loss of Long-Term Heat Removal	<1E-8	--	1.4E-5	55%
Loss of Coolant Accidents (LOCAs)	2.6E-7 (4.4E-8)	6%	7.9E-7	3%
Other Transients	2.3E-7 (3.1E-8)	5%	6.9E-7	3%
Total	4.5E-6 (9.1E-7)(2)		2.6E-5	

(1) Sum of means (or medians) of individual plant damage states
(2) Statistically combined totals are 4.5E-6 (1.9E-6)

2. PROGRAM SCOPE

The Peach Bottom Probabilistic Risk Assessment (PRA) was conducted during two periods. During the first period, the objective was to complete a fast, efficient PRA in a short time. This was accomplished, and following a review and some revisions, the PRA was published as NUREG/CR-4550, Volume 4 in October 1986. This report received extensive distribution and considerable review. In response to the comments from reviewers and especially the NRC and Philadelphia Electric Company, an update of the report was initiated. During the interim period, several changes were made to the plant, and additional system and procedural details were examined. The result is the significantly revised analysis presented in this document, NUREG/CR-4550, Revision 1, Volume 4, Parts 1 and 2.

This report combines the tasks performed in the original analysis with the tasks accomplished during the revised analysis. While the original objective was to perform a fast, efficient PRA, it became necessary due to comments and criticism to examine additional details and to refine the models and techniques during the revised analysis. One target in the reanalysis was to reduce conservatism as much as possible. To give the reader a perspective of the scope of this work, a list of PRA tasks is given below describing what was done in this analysis. The level of detail is compared to a "state-of-the-art" PRA for each task and graded as (1) improved state of the art, (2) state of the art, (3) slightly abbreviated, (4) abbreviated, and (5) not analyzed.

- Initial Information Collection -- The information collected from past Peach Bottom studies and the Final Safety Analysis Report (FSAR) was put together in an initial set of event trees, fault trees, and questions for plant personnel. The pre-visit information gathering took a month. One week was spent at the plant gathering information first hand and regular contact with the plant was maintained throughout the course of the study. A confirmatory visit near the end of the first analysis and two subsequent visits during the revised analysis were conducted. Numerous changes were made to the event trees and fault trees. (Slightly abbreviated)

- Initiating Event Identification -- Initiating event information from plant-specific records and past studies were used. A search for support system initiators was conducted. During the revised analysis, these initiating events were reviewed. Interfacing system LOCAs (Initiating Event V) and reactor vessel rupture (Initiating Event R) were re-evaluated. The frequency and recovery of loss of offsite power were significantly improved. (State of the art)

- Event Tree Development (Non Anticipated Transient Without Scram, ATWS) -- Because the plant had been studied extensively in the past, functional event trees were not developed. Past studies and current NUREG-1150 containment analyses were used to identify the non-ATWS system event tree headings necessary to model all reactor functions. No significant shortcuts were used to develop the initial non-ATWS system event trees. Nevertheless, numerous refinements were made in the revised analysis. (Improved state of the art)

- Event Tree Development (ATWS) -- Detailed examinations of the plant, procedures, and updated thermal-hydraulic calculations were performed to identify the ATWS event tree headings and to develop the ATWS sequences. (Improved state of the art)

- System Modeling -- The level of modeling detail was at the discretion of the analyst. If a system could be shown to be relatively unimportant, or if a detailed model would have taken an excessive amount of time, simplifications were made. If the system was considered important, a detailed modeling effort was undertaken. The models are therefore a combination of detailed fault trees, simplified fault trees, and black box models. Fault trees for several systems were added in the revised analysis. The level of detail in many existing fault trees was also increased. Common cause failures were included in the fault trees rather than applying such failures by hand to the cut sets. Fault trees were expanded from pipe segment modeling to individual components. This was done to a large extent for the benefit of the external events analyses, which use the internal events analysis models. (Ranges from abbreviated to state of the art, depending on the system)

- Analysis of Dependent Failures -- A significant effort was made to identify, model, and quantify dependent failures. Intersystem dependencies were identified and modeled in the system analysis. Subtle interactions found in past PRAs were reviewed for their applicability to Peach Bottom. A review of licensee event reports (LERs) and other plant-specific reports for Peach Bottom was made to identify any unexpected interactions or common failures. (Slightly abbreviated)

- Human Reliability Analysis (HRA) -- Except for the ATWS scenarios, a screening procedure was developed to calculate human error probabilities. Although an HRA specialist was present during the plant visit, there was not as much time available to interview operators as desired. The screening procedure was somewhat conservative and values that yielded high results were flagged and reconsidered. During the recovery analysis conducted in the revised analysis, each human error event, either pre or post-accident, was carefully tabulated, described, and re-evaluated. Only errors of omission were considered in this analysis. The ATWS HRA was extremely detailed with three specialists spending considerable time analyzing ATWS operator responses. (Slightly abbreviated and state of the art)

- Data Base Development -- A data specialist was present during the initial plant visit. A week for data collection did not permit an extensive effort; however, a reasonable amount of plant-specific data was gathered. Where plant-specific data were lacking, generic data were used. (Slightly abbreviated)

- Accident Sequence Quantification -- While there were no shortcuts taken that should affect the results, a screening technique was used to avoid running every possible core damage accident sequence through the entire Boolean computer code. All the accident sequences with the potential for being greater than 1E-8 were completely analyzed. (State of the art)

- Plant Damage State Analysis -- The plant damage states (PDS) are defined by the back-end analyst, with the assistance of the front-end analyst to assure a clean interface between analyses. This requires continuous feedback while the accident progression event trees are being developed. There were 20 distinct PDSs that were grouped into nine larger PDSs for quantification. Finally, four super PDSs were formed covering very broad categories of accident types. (Improved state of the art)

- Physical Process of Reactor Meltdown Accidents -- Past thermal-hydraulic calculations and calculations performed by the NUREG-1150 containment analysts were used as required. New ATWS related calculations were run by the team analysts. (Slightly abbreviated)

- Radionuclide Release and Transport -- This was handled by the NUREG-1150 back-end analysts.

- Environmental Transport and Consequence Analysis -- This was handled by the NUREG-1150 back-end analysts.

- Seismic Risk Analysis -- This is considered in Part 3 of Volume 4. (State of the art)

- Fire Risk Analysis -- This is considered in Part 3 of Volume 4. (Slightly abbreviated)

- Flood Risk Analysis -- This is considered in Part 3 of Volume 4. (Slightly abbreviated)

- <u>Other External Hazards (e.g., Tornadoes)</u> -- This is considered in Part 3 of Volume 4. (Slightly abbreviated)

- <u>Treatment of Uncertainties</u> -- Statistical uncertainty in the failure data, uncertainty associated with the application of the failure data, and uncertainty caused by modeling assumptions and success criteria were all treated in the analysis. In the original analysis, modeling uncertainty was handled to a large extent by sensitivity studies. In the revised analysis, modeling uncertainty was incorporated directly into the data. Expert judgment elicitations were conducted on all issues that could significantly affect uncertainty. Furthermore, several model and informational issues from the original analysis were resolved by additional study. (Improved State of the art)

In addition to this comparison with a state-of-the-art PRA, it is informative to identify factors that PRAs do not normally treat. The following list of items not usually included in PRAs is taken with some modification from NUREG-1115 [1]:

- Partial Failures
- Design Adequacy
- Adequacy of Test and Maintenance Practices
- Effect of Aging on Component Reliability (also burn-in phenomena)
- Adequacy of Equipment Qualification
- Environmentally-Related Common Cause
- Similar Parts-Related Common Cause
- Sabotage

3. PROGRAM REVIEW

To assure quality, two groups were chartered with the responsibility of reviewing the work and providing timely feedback. Because the time available to complete the tasks in the original analysis was short, these reviews had to be intense, and Probabilistic Risk Assessment (PRA) team response time had to be almost instantaneous. In the revised analysis, more time was available, but the review meetings were still intense and informative. In addition to their review, public comments were received by the NRC and three other groups reviewed the work for their specific purposes.

3.1 Senior Consultant Group

The purpose of the Senior Consultant Group (SCG) was to provide a broad scope review of the methods and results of the reference plant PRAs. This high-level review was to further assure the validity and applicability of the products. However, the SCG was not expected to provide detailed quality control or assurance of the products. This group did not meet during the revised analysis.

The members of the SCG are listed below:

- Dennis C. Bley, PL&G
- Michael P. Bohn, SNL
- Gregory J. Kolb, SNL
- Joseph A. Murphy, NRC
- William E. Vesely, SAIC (formerly of BCL)

3.2 Quality Control Group

The goals of the Quality Control Group (QCG) were the following:

- to provide guidance regarding the methodologies to be utilized in the PRAs,

- to ensure the consistent application of the methodologies by all PRA teams, and

- to ensure the technical adequacy of the work

These goals were met via periodic review meetings with the PRA teams. At these meetings, the QCG discussed the methodologies and reviewed, in detail, all technical work performed.

The QCG was composed of the individuals listed below; also shown is each individual's technical specialty:

- Gregory J. Kolb, SNL (QCG team leader, systems analysis, original analysis only)

- Gareth W. Parry, NUS (uncertainty analysis, systems analysis, reliability data)

- John Wreathal, SAIC (human reliability analysis, revised analysis only)

- Barbara J. Bell, BCL (human reliability analysis)

- Arthur C. Payne, Jr., SNL (systems analysis, reliability data, back-end interface)

- Eddie A. Krantz, INEL (systems analysis, original analysis only)

- David M. Kunsman, SNL (systems analysis, back-end interface)

- Gary Boyd, SAROS (systems analysis, back-end interface)

3.3 Utility Interface

A constant interface was maintained with the utility throughout the duration of the original analysis. The Peach Bottom team leader was in constant contact with Peach Bottom engineering and plant personnel to ask questions and verify information. The Peach Bottom contacts also reviewed the results presented in the first draft of the study and provided comments that were considered in the revised analysis. The same close interface was carried through the revised analysis. The utility support was extremely helpful.

3.4 Uncertainty Review Panel

This panel was formed at the request of the NRC to consider the way in which uncertainty had been analyzed in the draft NUREG-1150 and the supporting documents. A three-day meeting was held on April 20-22, 1987, where a number of contributors to NUREG-1150 were invited to make presentations to the panel, as were others who were known to have views that were important to the assessment. The panel addressed all areas of the uncertainty methodology including the statistical methods used, the way the results were presented, and especially the use of expert judgment.

As a result of the panel's findings, significant changes were made to the analysis [47]. The most important improvement was in the elicitation of expert judgment, which became a major effort in the revised analysis for both the front-end and back-end analyses.

3.5 Peer Review Panel

After the publication of the draft NUREG-1150 and the supporting front-end and back-end documents, the NRC Commissioners recommended a peer review because of the potential importance of these documents to the NRC's regulatory process. Lawrence Livermore National Laboratory was selected to coordinate this effort. Although this review panel was initiated by the NRC, it functioned independently.

Fourteen members were selected including national and international experts in the fields of nuclear reactor safety, probabilistic risk assessment, and severe accident phenomenology. The individuals represented academics, research laboratories, electric utilities and consulting companies. The first phase of their review was to address the draft documentation. The second phase is to review the final NUREG-1150 and related documentation including this report. At least five formal meetings were held during the first phase, and testimony was given by numerous people including the Peach Bottom analysts. The findings are given in Reference 46. In general, the panel had a number of comments on NUREG-4550, and those comments relevant to the study have been addressed.

3.6 American Nuclear Society Committee

Many members of the American Nuclear Society (ANS) felt that the society should express its view regarding a document such as NUREG-1150 that has the potential to influence the perception of accident risks associated with nuclear power plants and have an impact on the regulatory process. Thus, the President of the ANS appointed a special committee to follow and comment upon the documentation and progress of the NUREG-1150 program.

Their findings and recommendations on the draft NUREG-1150 are found in Reference 48. These findings and recommendations were based on a review of the February 1987 draft NUREG-1150, and the supporting documents, a review of the public comments, briefings by the NRC staff and others, and visits to Sandia National Laboratories by the Chairman and Vice Chairman to observe the expert review panel process and to discuss the ongoing analysis leading to the revised document.

3.7 Public Comments

During the several months when public comments were solicited, a number (approximately 50) of individuals and organizations performed detailed reviews of the NUREG-1150 related documentation. Their comments were extensive. These comments were submitted to the NRC and sorted by subject. Those comments applicable to the front-end analysis and, in particular, the Peach Bottom analysis, were reviewed by the analysts and considered to the extent possible during the revised analysis.

4. TASK DESCRIPTIONS

This section contains information on the major tasks performed for this study. Section 4.1 provides a brief overview of the tasks. The remaining subsections within Section 4 address each individual task as it applies to the Peach Bottom analysis. Sections 5, Results, and 6, Summary and Conclusions, provide the information covered by the last task entitled "Interpretation of Results."

4.1 Task Flow Chart

The major tasks performed for this study are indicative of the general tasks performed in any Level 1 PRA. Figure 4.1-1 displays the major tasks carried out in this analysis and shows the primary information flow paths between each task. The entire process has been performed twice. The first time was during the initial analysis which began in July 1985 and resulted in the first draft of this report printed in October 1986. Following a comment and review period, the entire process was performed again in order to update the analysis and respond to comments received on the first draft. The following subsections reflect the combined effort for both the first draft phase and the reanalysis for each of the major tasks. Volume 1 of this document provides more detailed descriptions of the methodology used in carrying out each task [2]. The reader is referred to that volume and the subsections which follow in order to obtain a comprehensive description of how the Peach Bottom analysis was conducted.

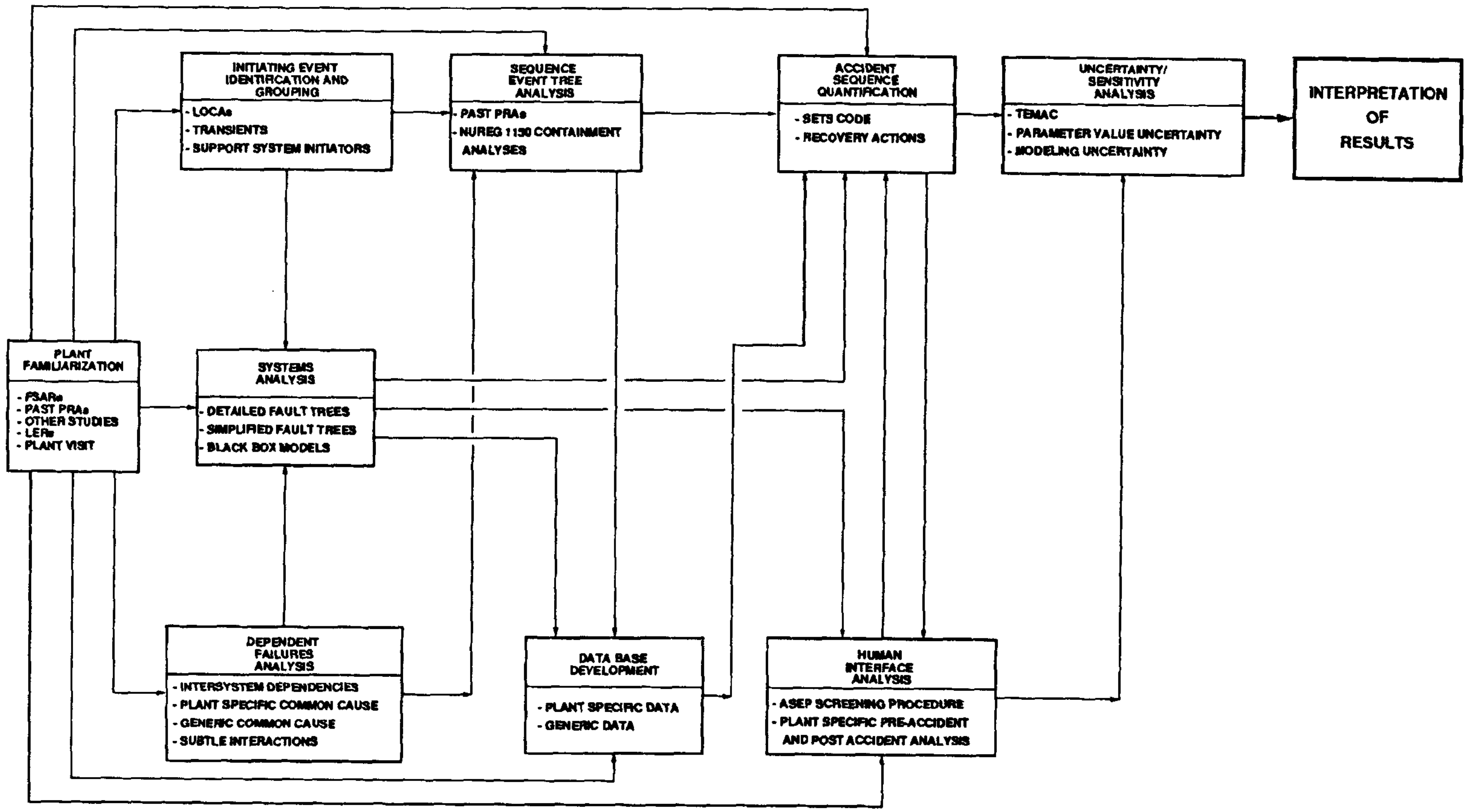


Figure 4.1-1. PRA Task Flow Chart

4.2 Plant Familiarization

4.2.1 Plant-Specific Nature of the Analysis

In order to assure that the analysis indeed reflected the Peach Bottom Unit 2 plant, a plant familiarization task was performed. During this effort, the analysts became familiar with the specific design, operational, and historical performance aspects of the unit. As a result, the analysis reflects the actual design, procedures, and operating experience at Peach Bottom during the analysis periods, to the extent possible. Therefore, the initiating event experience, the models, failure data, and human reliability analysis are based on Peach Bottom specific inputs.

The performance of this task included three major subtasks: (1) an initial plant visit, (2) a confirmatory plant visit near the end of the first draft period, and (3) a subsequent plant visit to begin the reanalysis effort. In addition, nearly continuous communication was maintained with the plant and the engineering staff to answer questions during both analysis phases. Prior to the initial plant visit, the Peach Bottom team reviewed the original Accident Sequence Evaluation Program (ASEP) analyses applicable to Peach Bottom [3], the fault tree and event tree sections of WASH-1400 [4], and Probabilistic Risk Assessment type studies related to Peach Bottom. Preliminary event trees, system fault trees, and simplified system schematics were constructed; preliminary success criteria and dependency matrices were developed to identify specific areas where information was needed for accurate models. Based on these initial activities, a package was prepared that identified the required plant specific information and data and gave a sampling of generic and specific questions the team would ask concerning system design and plant operation. This package was sent to Philadelphia Electric Company (PECO) so that their staff might better understand the team's needs. The following sections provide brief descriptions of each plant visit and the information obtained.

4.2.2 Initial Plant Visit

The purposes of the initial plant visit were to (1) gain specific knowledge of those Peach Bottom aspects which had been identified as important to safety/risk and (2) collect the necessary data. The visit occurred in July 1985. Two days were spent at PECO's main headquarters in Philadelphia, a third day at the Peach Bottom plant, and a fourth day at the Limerick simulator (Peach Bottom's operators are trained at this simulator). The Peach Bottom analysis team consisted of the overall program leader, the team leader, two system analysts, a data analyst, a containment analyst, and four human reliability analysts (three of whom concentrated on Anticipated Transient Without Scram (ATWS) scenarios). The team visited with PECO mechanical engineering staff members and various personnel in operations, training, and maintenance.

The preparatory package for the initial plant visit consisted generally of the following items:

- Request for Piping and Instrumentation Diagrams (P&IDs) and Functional Control Diagrams (FCDs) for all front line systems and their support systems,
- Request for Elementary Wiring Diagrams (one lines),
- Request for Layout Drawings (the reactor and control buildings),
- Request for Emergency Operating and Test/Maintenance Procedures,
- Request for Data Information (maintenance logs, LERs, etc.),
- Request for Post-Three Mile Island (TMI) and PRA modifications, and
- Lists of Questions (related to system design and plant operation).

The initial plant visit included the following events:

- Discussions with PECO engineering staff concerning
  - normal and emergency configurations and operation of the various systems of interest,
  - system interdependencies, and
  - design changes implemented at the plant;
- Discussions with PECO engineering and operational staff concerning
  - automatic and manual actions taken in response to various emergency conditions,
  - operational problem areas identified by plant personnel which might impact the analysis, and
  - detailed discussions regarding ATWS procedures;
- Discussions with PECO engineering and maintenance staff concerning
  - data: maintenance logs, LERS, etc., and
  - implementation regarding test/maintenance procedures;
- Discussions with PECO training staff concerning
  - training practices regarding various emergency conditions, and
  - detailed discussions regarding ATWS training.

4.2.3 Information Obtained

A considerable amount of information was obtained during and shortly following the initial plant visit. This information allowed the analysis to consider the specific features and operational aspects of Peach Bottom Unit 2. The information obtained consisted generally of the following:

- Information requested in the pre-visit package including:
  - the requested drawings
  - PECO's Emergency Operating Procedures [41] and examples of test and maintenance procedures
  - plant-specific failure data information on selected components considered likely to contribute the most to the overall results
  - recent or soon-to-be included plant modifications as a result of TMI action items, the recent ATWS rule, and utility-originated plant improvements
  - miscellaneous items regarding specific questions from the analysts.

- Peach Bottom monthly "hi-spot" reports for the period 1975-1985 [10] which summarize plant performance each month as well as provide information on every plant shutdown, and

- The Updated Final Safety Analysis Report for Peach Bottom Units 2 and 3 [11].

4.2.4 Confirmatory Plant Visit

The purpose of the confirmatory plant visit was to present the preliminary results of the first draft analysis and to confirm our knowledge regarding Peach Bottom. The plant visit occurred in December 1985. One day was spent at PECO's main headquarters and one day at the Peach Bottom plant. The Peach Bottom analysis team consisted of the overall program leader, the team leader, and three system analysts. The team visited with members of the PECO mechanical engineering staff and with various personnel in operations.

The final plant visit included the following activities:

- A presentation of the overall preliminary results,

- Discussions with engineering staff on major contributors and assumptions, and

- Discussions with operational staff on 'gray' areas concerning operator actions.

Additional information was supplied to the analysis team by PECO in response to issues raised during the final plant visit.

4.2.5. Subsequent Plant Visit for the Reanalysis Phase

In March 1988, a subsequent plant visit was made to the plant and PECO's engineering offices to learn of any changes or other factors which should be reflected in the reanalysis phase of the project. One day was spent at the engineering offices and one day at the plant site. Team members, including the team leader and two system analysts met with members of the PECO mechanical engineering staff and with operators at the plant. Updated drawings and new procedures were provided and discussed during the plant visit. While numerous miscellaneous changes or clarifications were identified, four primary changes in the plant and procedures were presented to the analysts which had a considerable impact on the reanalysis. These were:

- Modifications made to the Emergency Service Water (ESW) system hardware and operation since the first draft analysis,

- A revised station blackout procedure which accounted explicitly for stripping battery loads as well as actions to prevent HPCI/RCIC failure in the long term,

- A revised containment venting procedure which de-emphasized the use of local operations for venting and also required venting at 100 psig instead of 60 psig.

- Additional information on the containment's ability to withstand pressures closer to the 175 psia range instead of the earlier 130 psia.

Each of the above caused a significant impact in either the event tree or fault tree constructions or in the possible recovery actions and timing. This new information has been included in the reanalysis in order to properly reflect Peach Bottom's design and operational guidance as of early 1988.

4.3 Initiating Event Identification and Grouping

Following the initial plant familiarization stage of the analysis, the initiating events relevant to Peach Bottom were identified. Initiating events are those disruptions to the normal operation of the plant which cause a rapid shutdown of the plant, or a need to trip the plant, so as to challenge the safety systems in order to remove decay heat. The initiators included in this study are summarized in Table 4.3-1 along with their frequencies.

The selection of the initiators examined in this study is described in the following subsections. Discussions are included regarding information sources used, the initiating event selection process, the resulting list of initiators, and the underlying assumptions. The nomenclature used to identify each initiator is provided in Section 4.3.5. The final list of initiators forms the basis for the event tree task which defines the possible accident sequences that could occur for each initiator. It is these accident sequences that identify the possible scenarios leading to core damage (from internal initiators) for Peach Bottom Unit 2.

4.3.1 Scope of Events Considered

The scope of this work encompasses only the so-called internal initiators, i.e., those which directly affect the systems within the plant. External events such as fires, seismic events, and flooding are considered in Part 3 of NUREG/CR-4550 Revision 1.

Since a number of Probabilistic Risk Assessments (PRAs) on Boiling Water Reactor (BWR) plants have already been performed, this study made use of the combined list of initiators in those studies to derive its initiating event list. It should be noted that manual orderly shutdowns for refueling or administrative reasons were not considered. Table 4.3-2 summarizes the primary information sources used to identify the initiators examined in this study. The origi-nal WASH-1400 study, the Grand Gulf RSSMAP study, the IREP Browns Ferry study, and the Limerick and Shoreham PRAs were all reviewed for the lists of initiators in those studies based on actual events as reported in EPRI NP801 [13] and NP2230 [14]. In addition, suc-cess criteria implications from GE-NEDO 24708A and the initiators formerly covered by the Accident Sequence Evaluation Program (ASEP) were also used to assist in the identification of initiators for this analysis. This informa-tion was supplemented with actual plant trip data for both Peach Bottom units covering March 1976 to June 1985 as reported in PECO's monthly "hi-spot" reports. These actual plant shutdowns were reviewed to ensure that all initiating events that had occurred while at power at Peach Bottom were represented by the initiating event list. Finally, a review of the Peach Bottom design for special initiators was also undertaken. Plant design information from the Peach Bottom Updated Final Safety Analysis Report (UFSAR), coupled with information gained during the initial plant visit and subsequent telephone discussions, was used for the examination of special initiators. Special initiators are those events not typically included in general lists of initiating events. Such special

Table 4.3-1
Peach Bottom Initiating Events and Frequencies

INITIATOR NOMENCLATURE	DESCRIPTION	MEAN FREQUENCY (per year)
T1	Loss of offsite power (LOSP) transient	0.079
T2	Transient with the Power Conversion System (PCS) unavailable	0.05
T3A	Transient with the PCS initially available	2.5
T3B	Transient involving loss of feedwater (LOFW) but with the steam side of the PCS initially available	0.06
T3C	Transient due to an Inadvertent Open Relief Valve (IORV) in the primary system	0.19
TAC/x	Transient caused by loss of safety AC Bus "x"	5.0E-3
TDC/x	Transient caused by loss of safety DC BUS "x"	5.0E-3
A	Large LOCA	1.0E-4
S1	Intermediate LOCA	3.0E-4
S2	Small LOCA	3.0E-3
S3	Small-small LOCA	3.0E-2
V	Interfacing system LOCA (failure of a high/low pressure interface in the primary system)	<1E-8 (see Section 4.4)
R	Reactor Vessel Rupture	(see Section 4.4)

Table 4.3-2
Primary Information Sources Used to Identify Initiators

- ASEP prior work [3]
- WASH-1400 [4]
- Grand Gulf RSSMAP [5]
- IREP Browns Ferry [6]
- Limerick PRA [7]
- Shoreham PRA [8]
- GE-NEDO 24708A [9]
- PECO monthly "hi-spot" reports [10]
- Peach Bottom UFSAR [11]
- Minarick [12]

initiators which cause a plant trip and require decay heat removal are unique to the plant being analyzed. Examples would be loss of a particular DC bus or loss of service water. These are further discussed in Section 4.3.2.

PRAs typically divide initiating events into two major classes of events: loss of coolant accidents (LOCAs) and transients. While LOCAs of appreciable size have not occurred, as evidenced by operating experience, LOCAs are still examined as possible initiators since they would cause a plant trip, require emergency cooling if the Power Conversion System (PCS) were lost, and represent a possible threat to both the core and containment. During review of the above mentioned information sources, it was found that the Shoreham and Limerick plant analyses and General Electric's study of typical BWR 4 designs in NEDO 24708A supported the use of three LOCA sizes. These sizes are based on different mitigation success criteria as was done in the original WASH-1400 study of Peach Bottom.

The large LOCA, labeled A, is a steam or a liquid break in which the reactor vessel will rapidly depressurize. Low pressure system injection will be automatic, restoring water level in the reactor vessel. High pressure system injection flow rates are either inadequate to restore level (low pressure systems have much higher flow rates) or the high pressure turbine-driven systems cannot be run efficiently because of low steam pressure. Break sizes of approximately 0.1 square feet or larger are typical of this size LOCA.

The intermediate LOCA, labeled S1, is a steam or liquid break in which high pressure injection with the High Pressure Coolant Injection (HPCI) system is possible for a limited time period. This turbine-driven system can supply sufficient flow to the reactor until vessel pressure can no longer be maintained for successful HPCI operation. Low pressure injection must then be used to maintain water inventory in the core. Should HPCI fail initially, depressurization of the reactor vessel is required to allow for timely low pressure injection. Break sizes of approximately 0.004 to 0.1 square feet for liquid breaks and steam breaks of approximately 0.05 to 0.1 square feet are typical of this size LOCA.

The small LOCA, labeled S2, is small enough to allow for long-term successful mitigation by either HPCI or the Reactor Core Isolation Cooling (RCIC) system (a smaller capacity, turbine-driven system). Should both systems fail, depressurization is required for successful low pressure injection. This size LOCA can be approximated by a stuck-open Safety Relief Valve (SRV) for Peach Bottom. The break is any size smaller than that classified as an S1 LOCA above; e.g., less than 0.05 square feet for steam breaks and less than 0.004 square feet for liquid breaks.

In addition, a fourth LOCA category was defined to include the special recirculation pump seal leak. Such leaks have occurred in power plants, primarily because of the wearing-out of the pump seals during normal operation. Such leaks are well-instrumented and can be easily isolated.

Leaks up to a maximum of ~50-100 gpm could occur on a per pump basis although less than 5 gpm is more typical. Because the relative frequency of these leaks is considerably larger than for other LOCAs, and since these occurrences are easily detected and isolated, this type of LOCA was categorized as a separate small-small LOCA category, labeled S3.

A brief examination of possible LOCAs within mitigating systems was also performed. One LOCA source, in particular, received more attention than others since it could cause a plant trip and affect multiple safety systems. This was a LOCA in the Normal Service Water (NSW) piping where the piping interfaces with the Emergency Service Water (ESW) system piping to feed a number of emergency core cooling loads and the diesels (see the ESW system write-up in Section 4.6). A pipe break in this location could disturb normal service water flow so as to cause a plant trip along with possible loss of the NSW system. Subsequent ESW initiation would feed the break instead of cooling certain safety system loads. However, since (a) operation of the High Pressure Service Water (HPSW) is unaffected, as it has no dependency on ESW or NSW; (b) HPCI/RCIC are only affected indirectly by room cooling, therefore the systems can run 10 or more hours before failure of ESW or NSW would have any impact; (c) such a break could potentially be isolated; and (d) the probability of a LOCA having to occur in a specific location in a low pressure system is considered relatively low (<1E-6), we concluded that this initiator was not as important as other initiators of interest. Even with a coincident loss of offsite power, core damage would require the failure of HPCI and RCIC and the failure to recover AC power to systems such as the CRD system. Using arguments such as this, it was decided that LOCAs in the mitigating systems were probabilistically unimportant and, therefore, they were not included in this study. This finding is consistent with the scope of LOCAs analyzed in other PRAs.

Possible interfacing system LOCAs were also examined for inclusion in this study. Interfacing system LOCAs, or the so-called "V" sequence, are a breach of a high pressure to low pressure interface with the primary system. Such a breach could cause significant low pressure system leaks or even a pipe rupture and result in a loss of inventory from the primary system while at the same time failing a low pressure mitigating system. Possible bypass of the containment through the ruptured interface also represents a fission product escape path which could result in serious consequences. Based on actual experience as reported in References 12 and 49, focus for identifying sources for a possible "V" sequence included review of the high to low pressure interface in the Low Pressure Core Spray (LPCS) and Residual Heat Removal (RHR) systems. Precursors to the "V" sequence have occurred in BWRs during testing of both high and low pressure system valves which provide isolation from the primary system. Focus on the above low pressure systems is a result of the lower pressure design conditions of these systems which increases the chance of a significant loss of primary system inventory through a pipe break, relief valve, or pump seal rupture. Such a sequence has been examined as part of this study and is discussed in Section 4.4.

Transient initiators were selected primarily on the basis of the considerable prior work in BWR PRAs. In this earlier work, actual events have been grouped into major transient categories depending on the plant response to each transient. Where "like" responses are expected (i.e., the same systems are effectively failed or otherwise degraded resulting in similar overall plant effects and the same mitigating system success criteria apply), transients are grouped into major categories with each category identified as a transient initiator for analysis purposes. This categorization process significantly decreases the amount of analysis effort without affecting the results. Using the original WASH-1400 categories (T1, T2, T3) as a guide, the previously mentioned PRAs and the interim ASEP work were reviewed to determine whether expansion of these categories was necessary. In addition, actual operating history for Peach Bottom was reviewed as reported in PECO's monthly "hi-spot" reports which summarize, among other things, the causes for plant shutdowns. This information was coalesced into the list of transient initiators.

In general, it was found that transient events could remain grouped into the three main WASH-1400 transient categories. T1 events are those which involve a loss of offsite power to the plant. T2 events are those involving loss of the PCS and include, for example, Main Steam Isolation Valve (MSIV) closure events and loss of condenser vacuum. T3 events are those in which the PCS initially remains operational and allows for core heat to be removed as steam to the main condenser shortly after plant shutdown. Such events include turbine trips and IORV events. The T3 events were further subcategorized into three groups: IORV events, loss of feedwater events, and all other events of the T3 type.

While it was not within the scope of this study to perform a detailed analysis of a possible reactor vessel rupture as an initiating event, the possibility of such an occurrence has been considered. Instead, a review was conducted of previous work related to such a possibility to provide some insight as to the potential for such an event. Since, as a worst case, the initiator could preclude the ability to cool the core and hence define an accident sequence by itself, it is discussed as part of the Event Tree Section, 4.4, where accident sequences are defined in this report.

4.3.2 Support System and Special Initiators

Besides the traditional transient categories discussed above, a review was conducted to identify possible special initiators or support system failures acting as initiators. Two special initiators were identified and called TAC and TDC initiators. During the review of the Peach Bottom electrical design, it was noted that safety and non-safety loads are eventually shared off buses that ultimately derive their power from the 4160 VAC and 125/250 VDC safety buses. Loss of these buses could possibly cause a trip of the plant and simultaneous degradation of safety systems depending on the specific loads off each bus. While specific pathways to a plant trip were not explicitly identified for either the loss of a 4160 VAC or a 125/250 VDC safety bus, it was noted that an actual occurrence of the de-energization of a 4160 VAC safety bus on January 27, 1983 did indeed require a rapid shutdown of one of the units based on subsequent condenser water level anomalies. This fact and the sharing of safety and non-safety loads at Peach Bottom were used as sufficient argument to conservatively treat the loss of any of the above buses as a possible special initiator.

A search for other special initiators was also performed and included three major categories: loss of any service water system, loss of instrument air, and loss of heating and ventilation equipment. The NSW system, Turbine Building Cooling Water (TBCW) system, Reactor Building Cooling Water (RBCW) system, ESW system, and HPSW system were reviewed as possible sources for special initiators. Possible pipe breaks, the potential for causing a plant trip, and effects on safety systems such as loss of cooling or flooding were considered during the review. While detailed analyses were not possible because of the resources available for the study, no special initiators worthy of examination involving these systems were identified. This is based in part on the generally sharp separation between safety and non-safety cooling water systems (ESW, HPSW, and RBCW are standby safety systems; NSW and TBCW are normally running non-safety systems) and, thus, the unlikely possibility of both a plant trip and degrading safety systems at the same time (see earlier discussion on a LOCA for the NSW system). Possibilities of flooding seem small based on the low pressure operation of these systems and their locations with respect to most other safety systems.

Loss of instrument air/nitrogen can cause a plant trip through the dependency of the PCS, drywell coolers, and area ventilation systems on air supplies. Air or nitrogen is also supplied to the following accident mitigating systems: (1) the Automatic Depressurization System (ADS) valves, (2) the Emergency Ventilation System (EVS) dampers which provide room cooling for the diesels, switchgear, and DC systems, (3) the CRD full flow path, (4) some containment vent valves used for containment venting, and (5) the MSIVs. However, the MSIVs and ADS valves can remain open for significant periods of time since they are backed by accumulators and other air/nitrogen supplies (these have been tested to show they reliably hold air to the valves for ~one hour). The critical EVS dampers fail open. The CRD system can achieve near full flow conditions without air through an alternate passive path. Containment vent valves each have a separate air bottle which could be used to operate the valve locally. Furthermore, HPCI, RCIC, LPCS, LPCI, and HPSW are available to operate given a loss of instrument air. These points, along with the expected low probability of loss of air/nitrogen as an initiator (from pipe break or the required failure of multiple compressors - note: Peach Bottom has additional diesel compressors besides the main compressors), were used to eliminate loss of air/nitrogen as a special initiator on probabilistic grounds. This finding is further supported by the conclusions in a report on the effects of a loss of instrument air [15] and based on a discussion with one of the principal authors of that report.

Finally, heating and ventilation systems were reviewed but discarded as possible special initiators. This is again based on the degree of separation in the design of these systems at Peach Bottom, the low heat loads in critical equipment areas such as the AC bus rooms, and the generally slow effects of loss of heating and ventilation equipment which allow time for corrective action before a plant trip would occur. Also, PECO has performed analyses as part of the original FSAR questions to show that equipment in the control room, as an example, would not reach equipment qualification limits even with total loss of HVAC. In addition, Peach Bottom does not have a history of significant HVAC events.

4.3.3 Initiators Retained and Eliminated

Based on the above described process, the resulting list of initiators identified in Table 4.3-3 represents the initiators retained for analysis and hence the output of this task. These initiators form the categories of events which were examined to determine the possible accident sequences. Frequencies are also provided in the table for easy reference (see Section 4.9). Note that each initiator affects the plant differently or requires some change in the plant success criteria as evidenced by Table 4.3-4. More information on the success criteria associated with each initiator is contained in Section 4.4 and the development of the criteria followed the guidelines provided in NUREG/CR-4550, Volume 1.

Table 4.3-5 provides a summary of other possible initiators that were considered but eliminated from further analysis in the Peach Bottom study. Included are the primary reasons for each elimination during this screening step in the analysis.

4.3.4 Initiating Event Assumptions

The following represent the primary assumptions used in the identification and categorization of initiating events for this analysis:

- o All initiators are assumed to originate while the plant is at high power operation.

- o Manual shutdown in an orderly manner is not included.

- o The initiator list is reasonably complete. Disregarding external events, the wide range of sources used and the inclusion of actual operation history allows for a "reasonably complete" argument to be used. Any additional initiators would add further possibilities for core damage but should be of very low probability.

- o Losses of Divisions A, B, C, or D of the 4160 VAC or 125/250 VDC safety buses are conservatively assumed to lead to a loss of the PCS (including condensate) and are included as TAC/x and TDC/x initiators where "x" represents the divisional bus which is failed. Since explicit pathways for failing the PCS were not found for these bus losses (see Section 4.3.2), this analysis has taken a conservative stance by including these as possible initiators.

- The non-rigorous search for special initiators (due to resource constraints) adequately justifies the exclusion of such initiators except for TAC/x and TDC/x.

4.3.5 Initiating Event Nomenclature

This subsection addresses the nomenclature used to identify each type of initiator. Table 4.3-1 supplied earlier presents the initiators actually examined in the analysis. Other initiators were reviewed but excluded from the analysis effort. The nomenclature in the table defines the short-hand identification of each initiator that is used in the remainder of the report.

Table 4.3-3
Initiating Event Information Summary

INITIATOR	MEAN FREQUENCY	COMMENTS*	SOURCE OF INFORMATION*
T1 T2 T3A T3B T3C	0.079 0.05 2.5 0.06 0.19	o Major transient groups considered in original WASH-1400 PRA o Typical in BWR PRAs and in ASEP o Limerick PRA used: MSIV closure (like T2) Turbine trip (like T3A) LOSP (T1) LOFW (like T2 or T3B) Manual shutdown (like T3A) IORV (like T3C) o Shoreham PRA used above, and Loss of Condenser (like T2)	o WASH-1400 o ASEP Interim Reports o Limerick PRA Section 3.2 o Shoreham PRA Section 3.2 o Frequencies based on plant-specific experience
TAC/A TAC/B TAC/C TAC/D	5.0E-3 5.0E-3 5.0E-3 5.0E-3	o Safety 480 VAC buses (fed by 4160 V buses) share safety and PCS-related loads (Peach Bottom UFSAR) o Actual de-energization of 4160 VAC emergency bus caused need for scram (1-27-83) o Examined in ASEP, Shoreham PRA as an initiator o Loss of any 4160 VAC safety bus treated as a possible initiator for Peach Bottom	o Peach Bottom UFSAR Rev. 3, Section 8 o PECO "Hi-Spot" Reports o Shoreham PRA Section A.1.3.6.6 o ASEP Interim Reports

* The comments and sources of information relate to all the initiators in the group.

Table 4.3-3
Initiating Event Information Summary (Concluded)

INITIATOR	MEAN FREQUENCY	COMMENTS[b]	SOURCE OF INFORMATION[b]
TDC/A TDC/B TDC/C TDC/D	5.0E-3 5.0E-3 5.0E-3 5.0E-3	o 125 VDC buses share safety and PCS-related loads (Peach Bottom UFSAR) o Examined in ASEP, Shoreham PRA as an initiator o No direct cause for plant trip on a loss of a safety 125 VDC bus could be found. However, the sharing of safety and non-safety loads provided impetus to conservatively examine loss of any safety 125 VDC bus as a possible initiator.	o ASEP Interim Reports o Shoreham PRA Sec. A.1.3.6.1 o Peach Bottom UFSAR Sec. 8 & 7.10.3.5
A S1 S2	1.0E-4 3.0E-4 3.0E-3	o Original WASH-1400 PRA considered this LOCA breakdown o Limerick & Shoreham PRAs used these three sizes for LOCAs (Limerick system design is particularly close to Peach Bottom) o Further supported by success criteria implied by G.E. study which covers BWR-4 plants (Peach Bottom is a BWR-4)	o WASH-1400 o Shoreham PRA Table 1.2 o Limerick PRA Table 1.2 o GE-NEDO 24708A
S3	3.0E-2	o Typically included among S2 LOCAs but is unique in its frequency and that it can be easily isolated	
"V"	<1E-8 (see Note a)	o Shoreham & Limerick PRAs examined o Events have occurred at BWRs	o Shoreham PRA Appendix F o Limerick PRA Sec. 3.4.3.3 o Minarick and AEOD reports
"R" (vessel rupture)	(see Note a)	o Discussed qualitatively	o WASH-1400 and PTS work

NOTE: a) See discussions covered in Section 4.4.
b) The comments and sources of information relate to all the initiators in the group.

Table 4.3-4
Success Criteria Summary Information
(see Glossary for acronym definitions)

INITIATOR	REACTOR SUBCRITICAL	EMERGENCY CORE COOLING	EARLY CONTAINMENT OVERPRESSURE PROTECTION	LATE CONTAINMENT OVERPRESSURE PROTECTION
A	RPS or ARI & RPT or Manual Rods and RPT	1 of 4 LPCI or any 2 LPCS pumps	VSS	1 of 4 RHR & HtX (SPC or Spray modes) and associated HPSW or Containment Venting
S1	RPS or ARI & RPT or Manual Rods and RPT	HPCI (2 hours only) or DEP w/3 valves* and Any 2 LPCS pumps or DEP w/3 valves* and 1 of 4 LPCI or DEP w/3 valves* and 1 HPSW (inject mode)	VSS	1 of 4 RHR & HtX (SPC or Spray modes) and associated HPSW or Containment Venting

* Conservative for most breaks.

Table 4.3-4
Success Criteria Summary Information (Continued)

INITIATOR	REACTOR SUBCRITICAL	EMERGENCY CORE COOLING	EARLY CONTAINMENT OVERPRESSURE PROTECTION	LATE CONTAINMENT OVERPRESSURE PROTECTION
S2	RPS or ARI & RPT or Manual Rods and RPT or Timely SLC and RPT (for steam break)	HPCI or RCIC or 1 FW or DEP w/3 valves and Any 2 LPCS pumps or DEP w/3 valves and 1 of 4 LPCI or DEP w/3 valves and 1 Condensate or DEP w/3 valves and 1 HPSW (inject mode)	VSS	1 of 4 RHR & HtX (SPC or Spray Modes) and associated HPSW or Containment Venting or PCS
S3	If detected and isolated, treat like T3A. If not isolated, treat like S2 liquid LOCA.			

Table 4.3-4
Success Criteria Summary Information (Continued)

INITIATOR	REACTOR SUBCRITICAL	RCS OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	CONTAINMENT OVERPRESSURE PROTECTION
T1	RPS or ARI & RPT or Manual Rods and RPT or Timely SLC and RPT	SRVs open & close	HPCI or RCIC or CRD (~full flow) or 1 FW [see Note (a)] or DEP w/3 valves and Any 2 LPCS pumps or DEP w/3 valves and 1 of 4 LPCI or DEP w/3 valves and 1 Condensate [see Note (a)] or DEP w/3 valves and 1 HPSW (inject mode)	1 of 4 RHR & HtX (SDC, SPC, Spray Modes) and associated HPSW or PCS [see Note (a)] or Containment Venting

NOTE:
(a) Only available if offsite power is restored.

Table 4.3-4
Success Criteria Summary Information (Continued)

INITIATOR	REACTOR SUBCRITICAL	RCS OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	CONTAINMENT OVERPRESSURE PROTECTION
T2	RPS or ARI & RPT or Manual Rods and RPT or Timely SLC and RPT	SRVs open & close	HPCI or RCIC or CRD (~full flow) or 1 FW [see Note (a)] or DEP w/3 valves and Any 2 LPCS pumps or DEP w/3 valves and 1 of 4 LPCI or DEP w/3 valves and 1 Condensate or DEP w/3 valves and 1 HPSW (inject mode)	1 of 4 RHR & HtX (SDC, SPC, Spray Modes) and associated HPSW or PCS [see Note (b)] or Containment Venting

NOTES:

(a) Since feedwater is likely lost as part of the T2 initiator, feedwater must first be restored.

(b) T2 is a loss of the PCS so the PCS must first be restored.

Table 4.3-4
Success Criteria Summary Information (Continued)

INITIATOR	REACTOR SUBCRITICAL	RCS OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	CONTAINMENT OVERPRESSURE PROTECTION
T3 types	RPS <u>or</u> ARI & RPT <u>or</u> Manual Rods and RPT <u>or</u> Timely SLC and RPT	PCS <u>or</u> SRVs open & close	HPCI <u>or</u> RCIC <u>or</u> CRD (~full flow) <u>or</u> 1 FW <u>or</u> DEP w/3 valves and Any 2 LPCS pumps <u>or</u> DEP w/3 valves and 1 of 4 LPCI <u>or</u> DEP w/3 valves and Condensate <u>or</u> DEP w/3 valves and 1 HPSW (inject mode)	1 of 4 RHR & HtX (SDC, SPC, Spray modes) and associated HPSW <u>or</u> PCS <u>or</u> Containment Venting

Table 4.3-4
Success Criteria Summary Information (Concluded)

INITIATOR	REACTOR SUBCRITICAL	RCS OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	CONTAINMENT OVERPRESURE PROTECTION
TAC/X	Like T2 except Emergency Core Cooling & Residual Heat Removal have fewer AC pumps available to operate.			
TDC/X	Like T2 except Emergency Core Cooling & Residual Heat Removal have fewer AC pumps available to operate and HPCI or RCIC may be unavailable depending on which DC bus is affected.			

NOTE: Any transient with a stuck open relief valve will be treated as:

One valve stuck open ------- S2 steam LOCA

Two valves stuck open ------- S1 steam LOCA

Three valves stuck open ----- A steam LOCA

Table 4.3-5
Initiators Reviewed and Eliminated From Further Analysis

INITIATOR TYPE	PRIMARY REASONS FOR ELIMINATION
LOCAs in Secondary Side of Plant	o Isolation potential
LOCAs in Mitigating Systems	o Probability of occurrence o Isolation potential o Redundancy provided by other systems to prevent core damage
Reactor Vessel Rupture	o Qualitative discussion only
Loss of Service Water Systems	o Redundancy of systems o Functional and spatial separation of normally operating vs. standby systems o Probability of occurrence o Isolation potential
Loss of Instrument Air/ Nitrogen	o Ability of most key systems to adequately perform without air/nitrogen o Probability of occurrence
Loss of HVAC	o Redundancy in equipment o Relatively low heat loads in critical areas o Slow effects allow recovery before plant trip o Limited PECO analyses and historical performance

4.4 Event Tree Analysis

The next task involved the identification of the possible accident sequences for each initiator. This was done using the event tree approach which is commonly used in Probabilistic Risk Assessments (PRAs). The event trees are logic diagrams at the system level of detail which represent the combinations of system successes and failures forming possible sequences of events following each initiator. The philosophy behind the event tree analysis for Peach Bottom was to depict system successes and failures until the status of the core and containment are safe, vulnerable, or damaged and to display the status of other systems sufficiently to describe the plant damage states (see Section 4.11) applicable to each accident sequence.

The construction of the event trees was performed using the knowledge and experience base already represented by other Boiling Water Reactor (BWR) PRAs and with consideration of the generic event trees created as part of earlier ASEP efforts. Two major expansions of previous BWR event tree work were included, however, in this study.

(1) Formal analysis was conducted for more systems capable of core and containment cooling than considered before. Specifically, credit for the Control Rod Drive (CRD) system and the High Pressure Service Water (HPSW) system as injection sources to the reactor vessel was explicitly included in the success criteria and treated in the event trees and accompanying analyses. In addition, the Shutdown Cooling (SDC), Suppression Pool Cooling (SPC), and Containment Spray (CS) modes of the Residual Heat Removal (RHR) system, as well as the latest containment venting procedures (called containment venting in the tree, Y), were explicitly analyzed.

(2) The event tree analyses explicitly displayed and covered possible system success and failure paths beyond successful containment venting or containment failure. Therefore, the success or failure probabilities associated with continued core cooling were explicitly and formally analyzed rather than assumed.

The above expansion features of the event tree analyses provide, in general, more realistic analyses subject to less overall conservatism than previous analyses. However, as will become evident in the following subsections, conservative assumptions were still included in portions of the analyses so that the core damage potential would not be inadvertently underestimated. The above features of the analyses tend to provide lower core damage frequencies for some sequences than the reader may be accustomed to seeing in analyses for plants of similar design.

The following subsections address other aspects of the event tree analyses. Section 4.11 introduces the subject of plant damage states into which the dominant accident sequences were binned. Overall assumptions for the event tree analyses and a discussion of system success criteria are contained in Section 4.3.5. Each event tree used in the Peach Bottom-2 analysis is then displayed by each tree. The reader is referred to Section 4.4.16 for the nomenclature used in the event tree headings and resulting sequence identifiers.

4.4.1 General Event Tree Assumptions

There are a number of assumptions which generally apply to the event tree analyses performed for Peach Bottom-2 regardless of the specific initiator being examined. These assumptions are listed below with brief explanations as required.

(1) Low Pressure Core Spray (LPCS), Low Pressure Coolant Injection (LPCI), and RHR (all modes) pumps are assumed to fail following successful containment venting or containment failure by overpressure/temperature conditions.

The suppression pool is assumed to reach near atmospheric saturated conditions shortly after either successful venting or containment failure. Partial boiling of the pool water is assumed to decrease the net positive suction head (NPSH) for the LPCS/LPCI/RHR pumps such that these pumps cavitate, if running, causing subsequent failure.

(2) LPCS/LPCI/RHR (all modes) pumps, which use the suppression pool for suction, will successfully operate using pool water at a temperature approaching 350°F (corresponding to saturation conditions near point of containment failure by overpressure).

This assumption is based on (a) the corresponding pressure conditions of the containment which will assure adequate NPSH, (b) the pump seals and bearings being cooled by the Emergency Service Water system, (c) the findings of General Electric as reported in Section 5 of Reference 16, and (d) the fact that the RHR pumps normally pump water approaching such temperatures during the early phases of plant shutdown.

(3) Loss of the Vapor Suppression System (VSS) was considered but eliminated from the event tree as relatively improbable.

Loss of the VSS function could affect the ability of the Mark I containment to withstand steam release from the primary system through either a break or the opening of Safety Relief Valves (SRVs). The three most probable failure mechanisms appear to be downcomer pipe failure, stuck open wetwell/drywell vacuum breakers, or a broken SRV tail pipe. Based on References 4 and 17, best estimates for downcomer pipe or SRV pipe failures are <1E-5 and ~1E-7 respectively. Additionally, discussions with containment analysis personnel suggest that wetwell/drywell vacuum breaker demand is not expected in most scenarios of interest. Considering these probabilities in the context of other system failure probabilities led to the conclusion that VSS failure could be excluded from further analysis.

(4) High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) will fail at pool temperatures of ~210-260°F.

In all the accidents of interest, the HPCI system will eventually switch suction source from the condensate storage tank to the suppression pool automatically on high pool water level. Following procedures at Peach Bottom, the operator switches the RCIC system when he sees HPCI switch [18]. Switching back requires overriding certain circuits and therefore would not normally be performed. If, while the systems are running, the pool water should reach the 210-260°F range (nominally ~230°F), pump failure for both systems is assumed since these pumps are not externally cooled. This is supported, in part, by information supplied by Philadelphia Electric Company (PECO) [19].

(5) CRD in the enhanced mode (two pumps) is assumed to fail following reactor depressurization for SDC due to low NPSH.

The CRD system pumps water from the CST in the enhanced mode at approximately 200 gpm, which increases to near 300 gpm following reactor depressurization. The CST level is assumed to be too low at the time of reactor depressurization for SDC to prevent CRD pump cavitation due to insufficient NPSH.

In some event trees, the same event occurs more than once. A system may be successfully utilized in a sequence and later in the same sequence, following containment venting, may fail due to environmental conditions. In this analysis, credit is given for three injection systems (CRD (U4), Condensate (V1), High Pressure Service Water (V4)) to operate following the containment venting event (Y) in many of the event trees. If, in a particular event tree, the same injection system has been demanded before and after the containment venting event, then these events have different probabilities, although they have the same designation in the event tree. In this situation, the event demanded after containment venting refers to the survivability of the system, or its probability of successfully surviving containment venting. If the event is demanded only before containment venting, it refers to a hardware failure. If the event is demanded only after containment venting, it refers to hardware failure and survivability.

Core damage in many sequences is described as early or late. Early core damage refers to sequences in which loss of all coolant injection occurs soon after the initiating event and for which recovery is not performed. A late core damage designation is found in the T1 tree for sequences in which station blackout occurs and either HPCI or RCIC is functional. Injection may continue in these sequences for a substantial amount of time before injection fails and core damage occurs. A sequence designated as containment vulnerable indicates conditions (temperature and pressure) in containment constitute a risk of containment failure unless containment heat removal is effected.

4.4.2 Discussion of Success Criteria

The success criteria for the initiators of interest were presented earlier in Section 4.3.5. In the following subsections, the system success criteria for each initiator are presented again. The identification of initiators and the construction of the corresponding event trees is a very interactive process. Hence, many of the same information sources listed in Section 4.3 were used in the development of the success criteria and the event trees for each initiator [3-12].

Additional thermal-hydraulic analyses were performed for Anticipated Transients Without Scram (ATWS) scenarios as described in Section 4.4.15. For the most part, the other success criteria follow closely those used in the Limerick Probabilistic Safety Study [7] since Limerick and Peach Bottom have similar plant thermal ratings and similar emergency core cooling system designs and capacities. Any specific peculiarities in the criteria are noted for each initiator in subsequent subsections.

4.4.3 Large Loss of Coolant Accident (LOCA) Event Tree

This section contains information on the large LOCA event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.3.1 Success Criteria

A criterion specific to the large LOCA initiator is described below.

> For scenarios where core cooling is successful up to the time of containment venting or containment failure: one Condensate, one HPSW, or two CRD pump operation is assumed to be adequate to continue successful core cooling. This is based on the low decay heat loads reached by that time (many hours) and the fact that only small flow rates should be required to maintain sufficient vessel inventory and adequate core cooling.

4.4.3.2 Event Tree

Figure 4.4-1 displays the event tree for the large LOCA initiator. The following discussions define the event tree headings and describe the sequences presented. A bar over the event symbol or a slash preceding the event symbol both indicate success of the event.

The following event tree headings appear on the tree in the approximate chronological order that would be expected following a large LOCA.

- <u>A</u>: Initiating event, large LOCA.
- <u>C</u>: Success or failure of the Reactor Protection System (RPS). Success implies automatic scram by the control rods.
- <u>LOSP</u>: Success or failure to maintain offsite power.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

W1: Success or failure of RHR in the SPC mode. Success implies at least one RHR pump operating in the SPC mode with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

W3: Success or failure of RHR in the CS mode. Success implies at least one RHR pump operating in the CS mode with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

Y: Success or failure of containment venting. Success implies that the six-inch integrated leak test line or larger size line is open so as to prevent containment failure by over-pressure. As necessary, water makeup is also eventually supplied to the suppression pool.

V1: Success or failure of the Condensate System. Success implies at least one pump operating with sufficient makeup to the condenser hotwell for a continuing water supply.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

The following descriptions refer to the sequences found in Figure 4.4-1.

SEQUENCE 1 -- $A*\overline{C}*\overline{LOSP}*\overline{V2}*\overline{W1}$

Following the large LOCA (A), the RPS successfully inserts the rods into the core (/C). Offsite power remains available (/LOSP). High pressure cooling cannot be utilized because insufficient steam is available to run the turbines and LPCS is initiated to provide core coolant (/V2). The suppression pool temperature is increasing since residual heat from the reactor is being dumped to it. SPC is initiated to provide suppression pool cooling (/W1). With coolant makeup and containment overpressure protection provided, the core and containment are safe.

SEQUENCE 2 -- $A*\overline{C}*\overline{LOSP}*\overline{V2}*W1*\overline{W3}$

Same as Sequence 1 except containment overpressure protection is provided by the CSS mode of RHR (/W3) following the failure of SPC (W1).

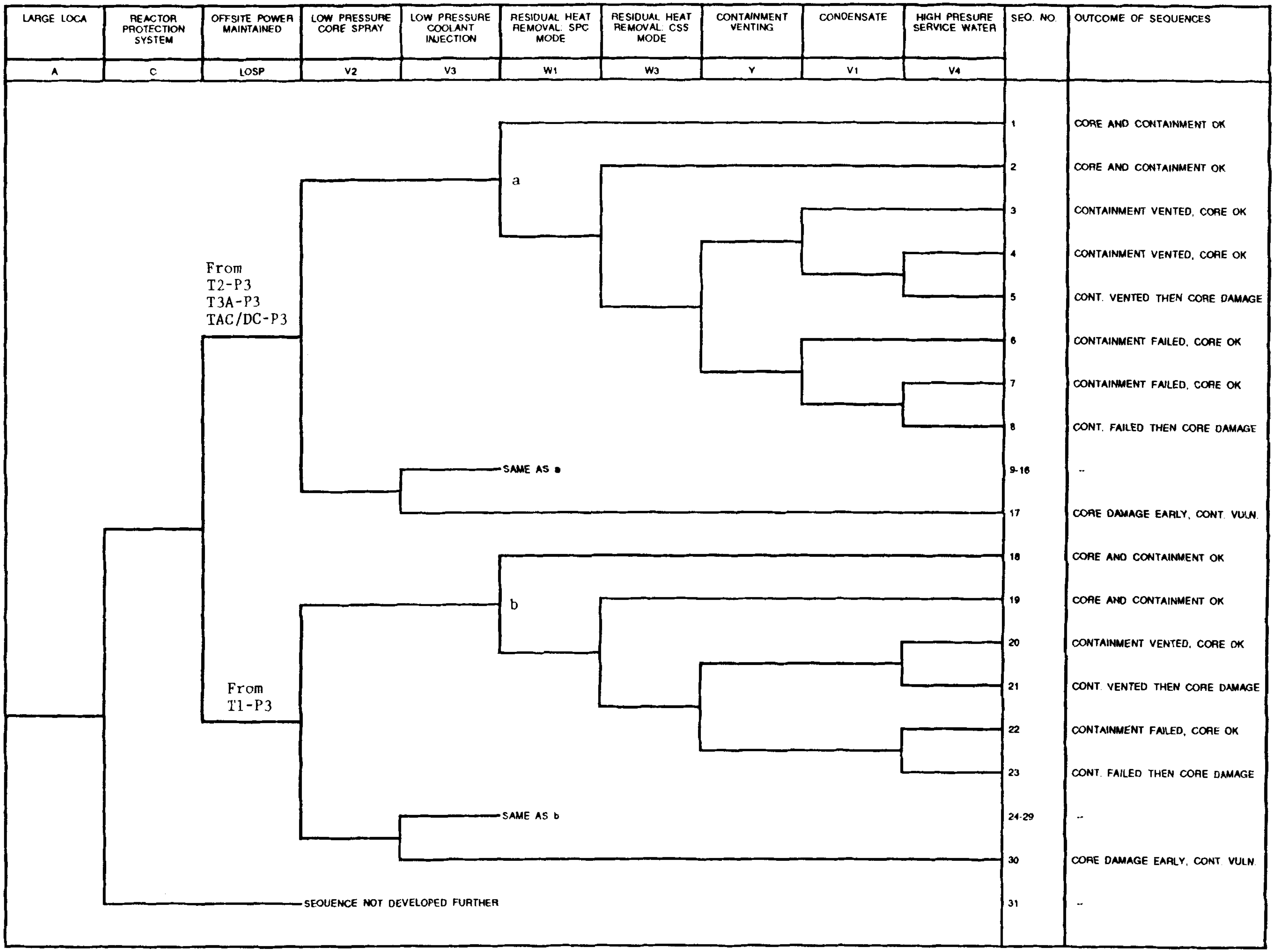


Figure 4.4-1. Large LOCA Event Tree

SEQUENCE 3 -- $A*\overline{C}*\overline{LOSP}*\overline{V2}*W1*W3*\overline{Y}*\overline{V1}$

Same as Sequence 1 except both SPC (W1) and CSS (W3) fail. The subsequent pressure rise in containment is alleviated by containment venting (/Y). LPCS failure is assumed following containment venting due to insufficient NPSH for the LPCS pumps. The operator then initiates Condensate (/V1) to continue to cool the core.

SEQUENCE 4 -- $A*\overline{C}*\overline{LOSP}*\overline{V2}*W1*W3*\overline{Y}*V1*\overline{V4}$

Same as Sequence 3 except HPSW provides core coolant (/V4) subsequent to Condensate failure (V1).

SEQUENCE 5 -- $A*\overline{C}*\overline{LOSP}*\overline{V2}*W1*W3*\overline{Y}*V1*V4$

Same as Sequence 4 except HPSW fails (V4) to cool the core. At this point all coolant makeup is lost, which leads to core damage in a vented containment.

SEQUENCES 6 TO 8

Same as Sequences 3 to 5 except containment venting fails (Y) leading to containment failure by overpressurization.

SEQUENCES 9 TO 16

Same as Sequences 1 to 8 except LPCS fails (V2) and LPCI provides initial low pressure coolant injection (/V3).

SEQUENCE 17 -- $A*\overline{C}*\overline{LOSP}*V2*V3$

Following the large LOCA (A), the RPS successfully inserts the rods into the core (/C). Offsite power remains available (/LOSP). LPCS and LPCI fail to provide low pressure core cooling, resulting in early core damage.

SEQUENCES 18 TO 19

Same as Sequences 1 and 2 except offsite power is not maintained (LOSP). Onsite power is established which enables LPCS to cool the core (/V2) and SPC (/W1) or CSS (/W3) to provide containment overpressure protection.

SEQUENCES 20 TO 21

Same as Sequences 4 and 5 except offsite power is lost (LOSP) and Condensate is therefore not available following successful containment venting.

SEQUENCES 22 TO 23

Same as Sequences 7 and 8 except offsite power is lost (LOSP) and Condensate is therefore not available following failure of containment venting.

SEQUENCES 24 TO 29

Same as Sequences 18 to 23 except LPCI provides initial low-pressure core cooling (/V3) following LPCS failure (V2).

SEQUENCE 30 -- A*$\bar{C}$*LOSP*V2*V3

Same as Sequence 17 except offsite power is also lost (LOSP).

SEQUENCE 31 -- A*C

Following the large LOCA (A), the RPS fails to properly insert the rods into the core (C). The sequence is not developed further due to its low probability.

4.4.4 Intermediate LOCA Event Tree

This section contains information on the intermediate LOCA event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.4.1 Success Criteria

A criterion specific to the intermediate LOCA initiator is described below.

> For scenarios where core cooling is successful up to the time of containment venting or containment failure: one Condensate, one HPSW, or two CRD pump operations is assumed to be adequate to continue successful core cooling. This is based on the low decay heat loads reached by that time (many hours) and the fact that only small flow rates should be required to maintain sufficient vessel inventory and adequate core cooling.

4.4.4.2 Event Tree

Figure 4.4-2 displays the event tree for the intermediate LOCA initiator. The following discussions define the event tree headings and describe the sequences presented.

The following event tree headings appear on the tree in the approximate chronological order that would be expected following an intermediate LOCA. For convenience, high and then low pressure injection systems are shown first, followed by containment-related systems, and finally by systems capable of long-term continued coolant injection.

S1: Initiating event, intermediate LOCA.

C: Success or failure of the RPS. Success implies automatic scram by the control rods.

LOSP: Success or failure to maintain offsite power.

U1: Success or failure of the HPCI system. Success implies operation of the HPCI system for ~1-2 hours until low primary system pressure causes isolation of HPCI either automatically or manually. U1' refers to the HPCI system without pump room ventilation.

X1: Success or failure of primary system depressurization. Success implies automatic or manual operation of the Automatic Depressurization System (ADS) or manual operation of other SRVs such that three valves or more are opened allowing low pressure injection. An intermediate LOCA may blow the vessel down sufficiently fast to preclude X1 operation.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

W1,W3: Success or failure of the RHR in the SPC mode or CS mode, respectively. Success implies at least one RHR pump operating in either the SPC or CS mode with the appropriate heat exchanger in the loop along with the HPSW in operation to the ultimate heat sink.

Y: Success or failure of containment venting. Success implies that the six-inch integrated leak test line or larger is open so as to prevent containment failure by overpressure. As necessary, water makeup is also eventually supplied to the suppression pool.

V1: Success or failure of the Condensate system. Success implies at least one pump operating with sufficient makeup to the condenser hotwell for a continuing water supply.

The following descriptions refer to the sequences found in Figure 4.4-2.

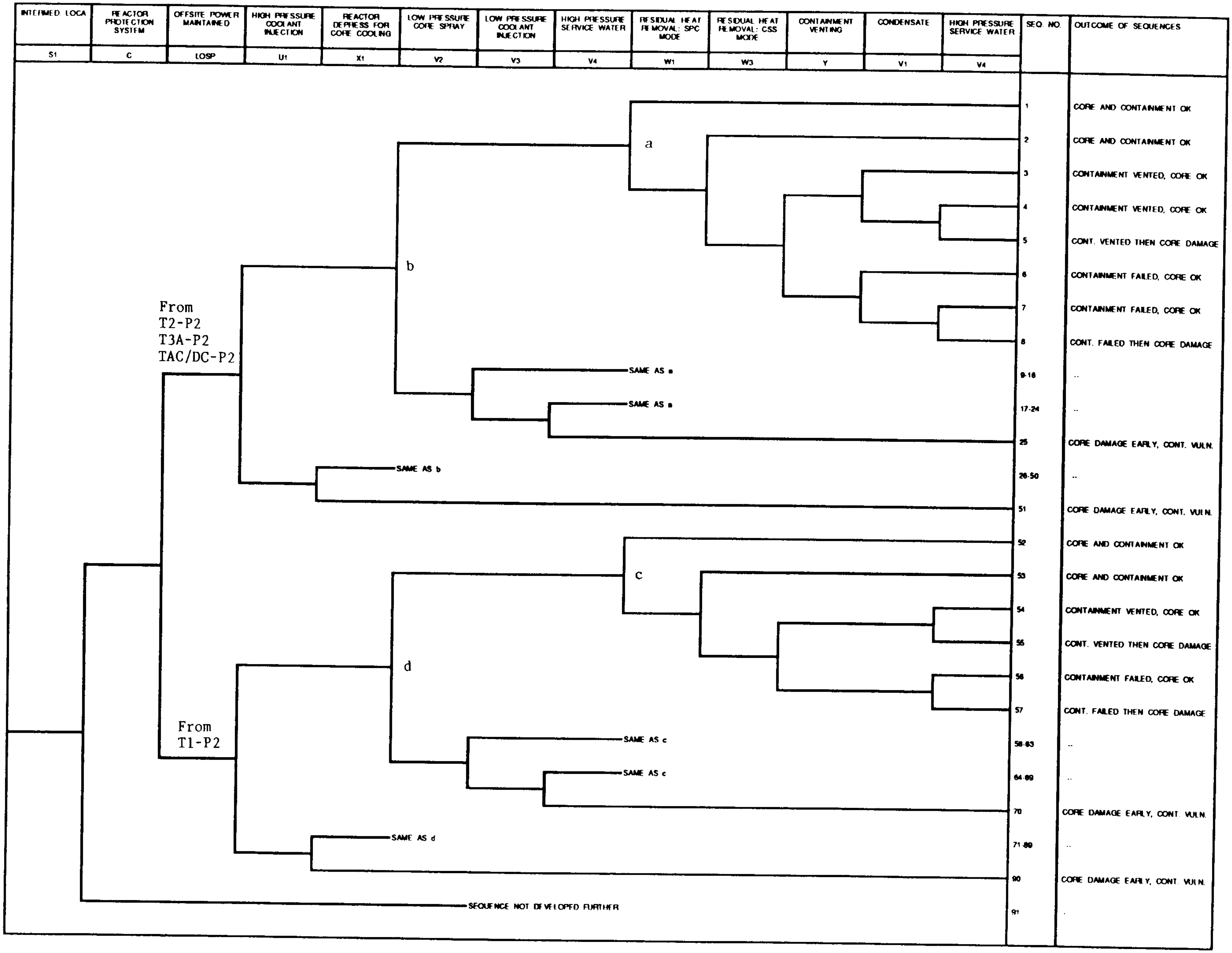


Figure 4.4-2. Intermediate LOCA Event Tree

SEQUENCE 1 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*\overline{W1}$

Following the intermediate LOCA (S1), the RPS successfully inserts the rods into the core (/C). Offsite power remains available (/LOSP) and HPCI (/U1') initially provides core coolant. The primary pressure decreases and steam is lost through the break, which eventually fails HPCI. LPCS is initiated to continue core cooling (/V2). Residual heat from the reactor is being transferred to the suppression pool. SPC is successfully initiated (/W1). With LPCS and SPC providing adequate coolant makeup and containment overpressure protection, the core and containment are safe.

SEQUENCE 2 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*\overline{W3}$

Same as Sequence 1 except CSS (/W3) provides containment overpressure protection following the failure of SPC (W1).

SEQUENCE 3 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*\overline{Y}*\overline{V1}$

Same as Sequence 1 except SPC (W1) and CSS (W3) fail to function, which causes the pressure to increase in containment. Containment venting is successful (/Y) which causes the LPCS pumps to fail due to low NPSH. Condensate is initiated (/V1) for coolant makeup resulting in no core damage in a vented containment.

SEQUENCE 4 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*\overline{Y}*V1*\overline{V4}$

Same as Sequence 3 except Condensate fails (V1) and HPSW is initiated to supply coolant makeup (/V4).

SEQUENCE 5 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*\overline{Y}*V1*V4$

Same as Sequence 4 except HPSW fails to provide coolant makeup (V4), resulting in core damage in a vented containment.

SEQUENCE 6 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*Y*\overline{V1}$

Same as Sequence 3 except containment venting fails (Y) following the loss of containment cooling resulting in a pressure rise in containment which leads to containment failure. This fails LPCS due to low NPSH. Condensate is initiated to provide coolant makeup(/V1). This results in no core damage in a failed containment.

SEQUENCE 7 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*Y*V1*\overline{V4}$

Same as Sequence 6 except HPSW provides coolant makeup (/V4) subsequent to Condensate failure (V1).

SEQUENCE 8 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*\overline{V2}*W1*W3*Y*V1*V4$

Same as Sequence 6 except both Condensate (V1) and HPSW (V4) fail to provide coolant makeup resulting in core damage in a failed containment.

SEQUENCES 9 TO 16

Same as Sequences 1 to 8 except early low-pressure coolant makeup is provided by LPCI (/V3) following failure of LPCS (V2).

SEQUENCES 17 TO 24

Same development as Sequences 9 to 16 except HPSW provides early low-pressure coolant makeup (/V4) following LPCI (V3) failure. HPSW demanded following containment venting refers to survivability.

SEQUENCE 25 -- $S1*\overline{C}*\overline{LOSP}*\overline{U1}'*V2*V3*V4$

Same as Sequence 1 except all efforts to establish early low-pressure core cooling with LPCS (V2), LPCI (V3) and HPSW (V4) fail, resulting in early core damage in a vulnerable containment.

SEQUENCES 26 TO 50

Same development as Sequences 1 to 25 except HPCI fails to initiate (U1') which requires depressurization of the primary system (/X1) to allow the low-pressure systems to provide coolant makeup.

SEQUENCE 51 -- $S1*\overline{C}*\overline{LOSP}*U1'*X1$

Same as Sequence 1 except HPCI fails to initiate (U1') and depressurization of the primary system is unsuccessful (X1), disabling the low-pressure core coolant systems, leading to early core damage in a vulnerable containment.

SEQUENCES 52 to 57

Same development as Sequences 1 to 8 except offsite power is lost (LOSP) early in the sequence and onsite emergency power is provided by the diesel generators. Since offsite power is not available, Condensate cannot be asked after the containment venting event, resulting in six sequences instead of eight.

SEQUENCES 58 TO 63

Same development as Sequences 52 to 57 except LPCI provides early coolant makeup (/V3) following LPCS failure (V2).

SEQUENCES 64 TO 69

Same development as Sequences 58 to 63 except HPSW provides early coolant makeup (/V4) following LPCI (V3) failure. HPSW demanded following containment venting refers to survivability.

SEQUENCE 70 -- S1*$\overline{C}$*LOSP*$\overline{U1}$'*V2*V3*V4

Following the intermediate LOCA (S1), the RPS successfully inserts the rods into the core (/C). Offsite power is lost (LOSP) and onsite power is established. HPCI provides coolant makeup (/U1') until the pressure in the primary reduces sufficiently to initiate the low-pressure coolant systems. LPCS (V2), LPCI (V3) and HPSW (V4) fail to operate, resulting in early core damage in a vulnerable containment.

SEQUENCES 71 TO 89

Same as Sequences 52 to 70 except HPCI fails to provide early coolant makeup (U1'), followed by successful depressurization (/X1) of the primary system to enable low-pressure systems to initiate.

SEQUENCE 90 -- S1*$\overline{C}$*LOSP*U1'*X1

Following the intermediate LOCA (S1), the RPS successfully inserts the rods into the core (/C). Offsite power is lost (LOSP) and onsite power is established. HPCI fails to provide coolant makeup (U1') followed by unsuccessful primary system depressurization (X1). This disables all low-pressure coolant systems, resulting in early core damage in a vulnerable containment.

SEQUENCE 91 -- S1*C

The RPS does not respond (C) to the intermediate LOCA and the sequence is not developed further due to a low probability.

4.4.5 Small LOCA Event Tree

This section contains information on the small LOCA event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.5.1 Success Criteria

Two criteria specific to the small LOCA initiator are described below.

(1) For scenarios in which core cooling has been provided for a period of a few hours or more, two CRD pump operation is considered adequate for continued success of core cooling should the other cooling systems then fail. This is based on the low decay heat

levels and relatively small flow rates required by that time to make up for the small break.

(2) For scenarios in which core cooling is successful up to the time of containment venting or containment failure, two CRD pumps or depressurization with operation of either one Condensate or one HPSW pump is considered to be adequate to continue successful core cooling.

4.4.5.2 Event Tree

Figure 4.4-3 displays the event tree for the small LOCA initiators. The following discussions define the event tree headings and describe the sequences presented.

The following event tree headings appear on the tree in the approximate chronological order that would be expected following a small LOCA. For convenience, the Residual Heat Removal (RHR) containment cooling choices are shown early in the tree to decrease the size of the event tree. Otherwise, the tendency is to show high and then low pressure injection systems, followed by containment venting, and finally long-term continued core cooling possibilities.

S2: Initiating event, small LOCA

C: Success or failure of the RPS. Success implies automatic scram by the control rods.

LOSP: Success or failure to maintain offsite power.

Q1: Success or failure of the Power Conversion System (PCS). Success implies operation of the balance of plant by removing heat through at least one Main Steam Isolation Valve (MSIV) with operation of the condenser and circulating water system as well as one feedwater train.

U1: Success or failure of the HPCI system. Success implies operation of the HPCI pump train so as to maintain sufficient coolant injection.

U2: Success or failure of the RCIC system. Success implies operation of the RCIC pump train so as to maintain sufficient coolant injection.

X: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS or manual operation of other SRVs such that three valves or more are opened allowing low pressure injection.

V1: Success or failure of the Condensate system. Success implies at least one pump operating with sufficient makeup to the condenser hotwell for a continuing water supply.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines. Conservative requirement since a small LOCA requires less makeup than two pumps provide.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

W1,W3: Success or failure of the RHR system in the SPC mode or CS mode, respectively. Success implies at least one RHR pump operating in either the SPC or CS mode with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

U4: Success or failure of the CRD system as an injection source. Success implies one pump operation.

Y: Success or failure of containment venting. Success implies that the six-inch integrated leak test line or larger size line is open so as to prevent containment failure by overpressure. As necessary, water makeup is also eventually supplied to the suppression pool.

R: Success or failure of the containment to withstand overpressurization. Success implies the containment ruptures before core damage.

X3: Success or failure of primary system depressurization. Success implies automatic or manual operation of ADS occurs subsequent to an initial depressurization to allow low pressure coolant injection.

The following descriptions refer to the sequences found in Figure 4.4-3.

SEQUENCE 1 -- $S2*\overline{C}*\overline{LOSP}*\overline{Q1}$

A small LOCA (S2) generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the PCS functions to remove heat from the core (/Q1), resulting in no core damage in a safe containment.

SEQUENCE 2 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*\overline{W1}$

Same as Sequence 1 except the PCS fails (Q1), HPCI is initiated to provide core coolant (/U1), and SPC provides containment overpressure protection (/W1).

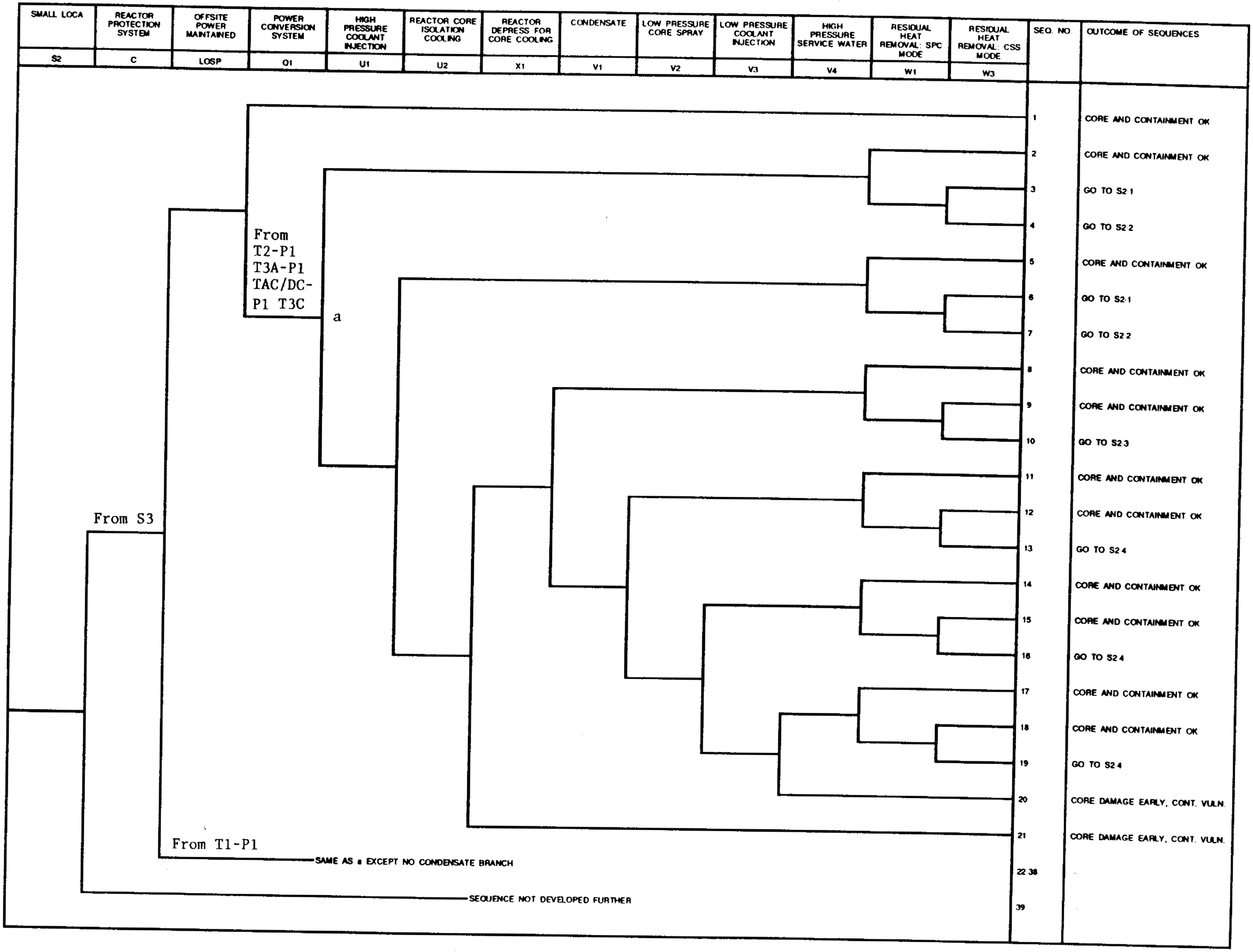


Figure 4.4-3. Small LOCA Event Tree (Page 1 of 3)

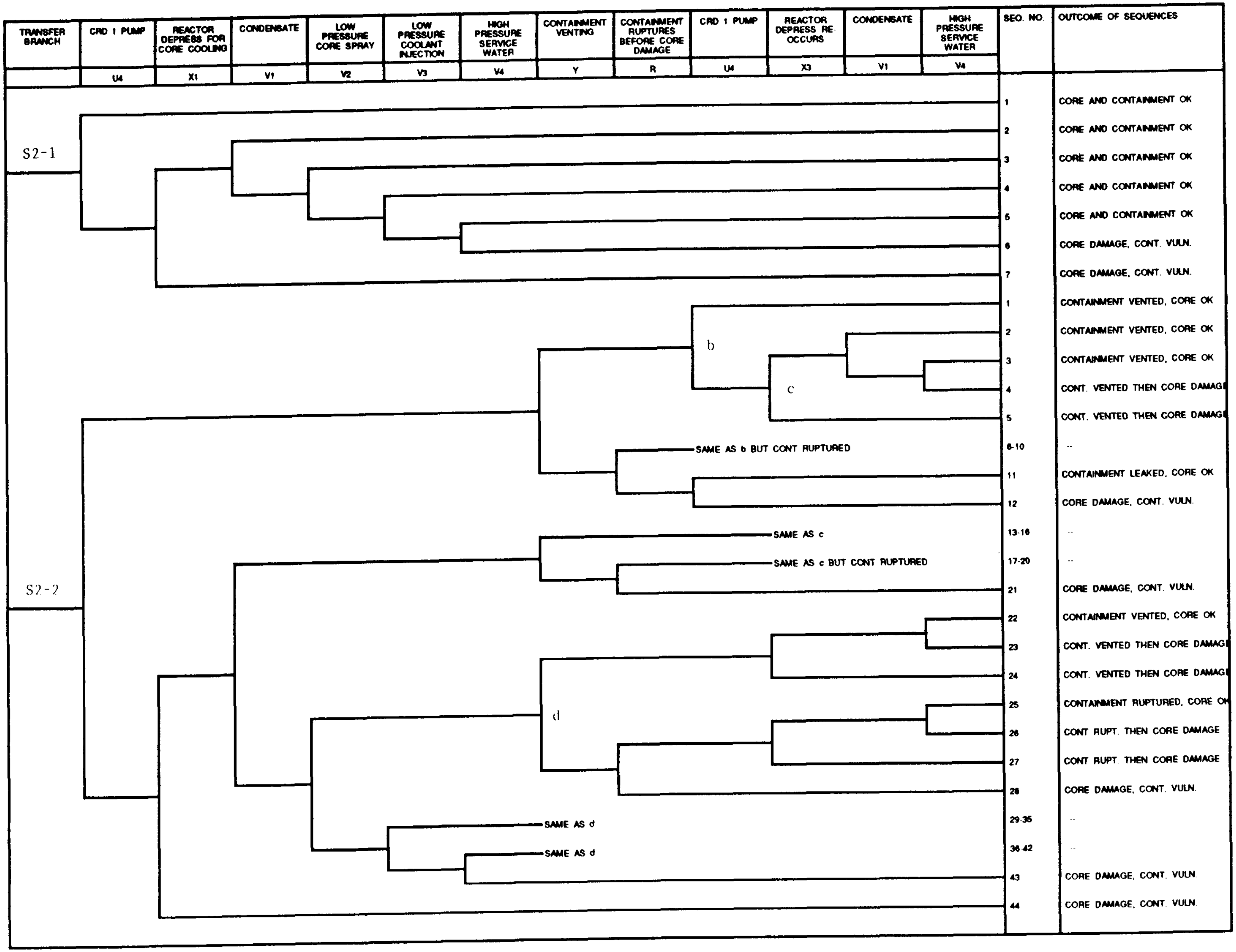


Figure 4.4-3. Small LOCA Event Tree (Page 2 of 3)

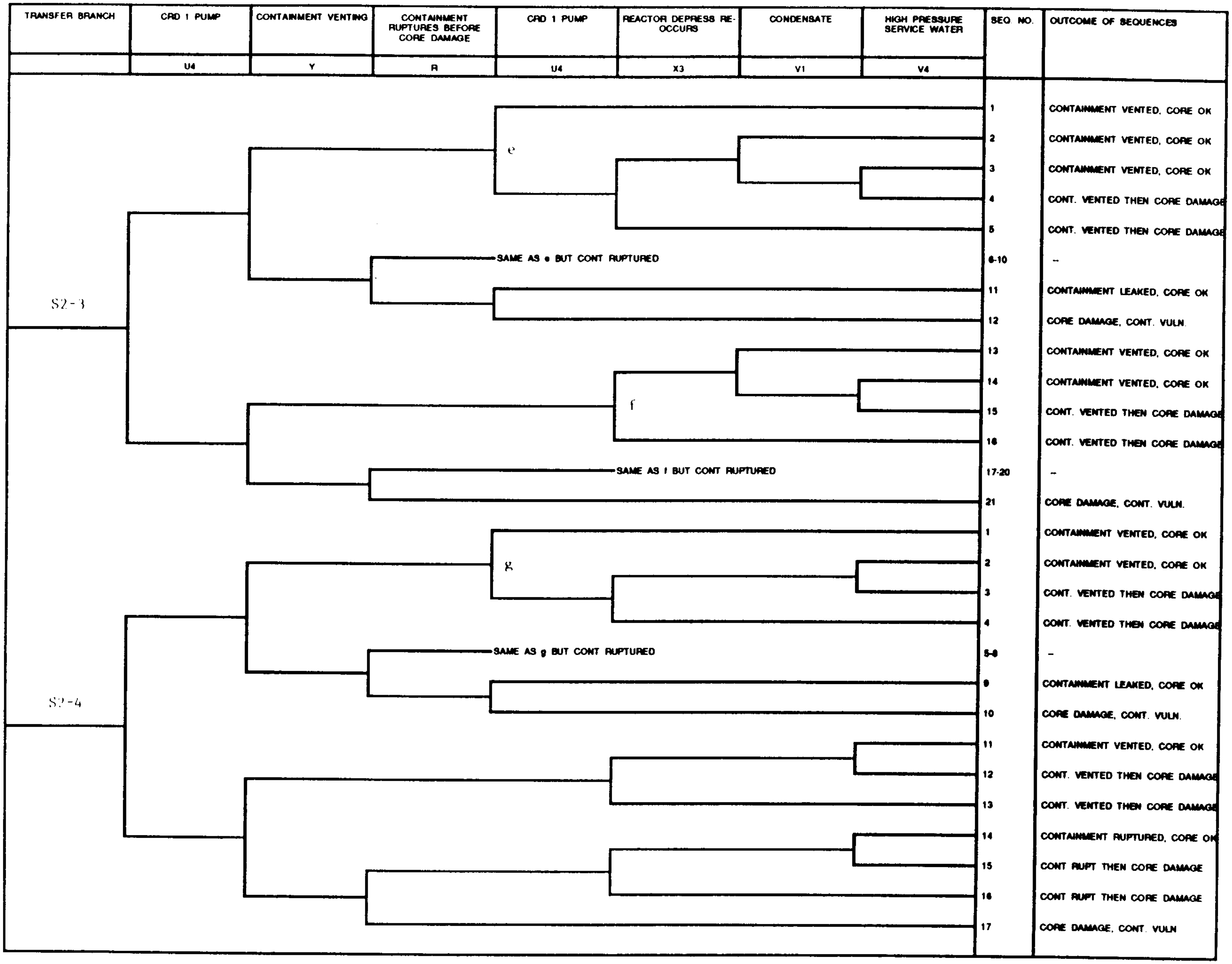


Figure 4.4-3. Small LOCA Event Tree (Page 3 of 3)

SEQUENCE 3-1 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*\overline{W3}*\overline{U4}$

Same as Sequence 2 except containment overpressure protection fails with SPC (W1) and CSS (/W3) is initiated. HPCI fails due to high suppression pool temperature reached before CSS is initiated and CRD is initiated to provide coolant makeup (/U4).

SEQUENCES 3-2 TO 3-5

Same as Sequence 3-1 except CRD fails (U4) and the primary system is depressurized (/X1) to allow the low-pressure coolant systems to cool the core. Either Condensate (/V1), LPCS (/V2), LPCI (/V3) or HPSW (/V4) functions to cool the core.

SEQUENCE 3-6 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*\overline{W3}*U4*\overline{X1}*V1*V2*V3*V4$

Same as Sequence 3-2 except all low-pressure core coolant systems fail (Condensate, LPCS, LPCI, HPSW) resulting in core damage in a vulnerable containment.

SEQUENCE 3-7 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*\overline{W3}*U4*X1$

Same as Sequence 3-1 except CRD fails to provide coolant makeup (U4) and subsequent primary system depressurization is unsuccessful (X1). Since all low-pressure cooling systems are disabled, core damage results in a vulnerable containment.

SEQUENCE 4-1 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 2 until both SPC (W1) and CSS (W3) fail to provide containment overpressure protection. HPCI eventually trips on high suppression pool temperatures (U1) and CRD is initiated (/U4). High containment pressure is reduced by containment venting (/Y). CRD survives the venting event and continues to provide coolant makeup, resulting in no core damage in a vented containment.

SEQUENCES 4-2 TO 4-3

Same as Sequence 4-1 except CRD does not survive containment venting (U4) and the primary system is depressurized (/X1) to allow Condensate (/V1) or HPSW (/V4) to continue core cooling.

SEQUENCE 4-4 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*V1*V4$

Same as Sequence 4-3 expect both Condensate (V1) and HPSW (V4) fail to provide core cooling, resulting in core damage in a vented containment.

SEQUENCE 4-5 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*\overline{U4}*\overline{Y}*U4'*X3$

Same as Sequence 4-2 except reactor depressurization is unsuccessful (X3), precluding the use of any low-pressure coolant systems, resulting in core damage in a vented containment.

SEQUENCES 4-6 TO 4-10

Same as Sequences 4-1 to 4-5 except containment venting is unsuccessful (Y) and overpressurization soon causes containment failure. All sequence outcomes are the same except the containment is not vented but failed.

SEQUENCE 4-11 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 4-1 except containment venting is unsuccessful (Y) and rupture of the containment does not occur (R), although a leak in the containment has developed. CRD survives and continues to provide core coolant resulting in no core damage in a leaking containment.

SEQUENCE 4-12 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 4-11 except CRD does not survive the containment over-pressurization and leak, resulting in core damage in a leaking containment.

SEQUENCES 4-13 TO 4-16

Same as Sequences 4-2 to 4-5 except CRD injection fails (U4) following HPCI failure, the primary system is depressurized (/X1), and Condensate continues core cooling (/V1) prior to venting.

SEQUENCES 4-17 TO 4-20

Same as Sequences 4-13 to 4-16 except containment venting fails (Y) and the containment ruptures (/R).

SEQUENCE 4-21 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*\overline{V1}*Y*R$

Same as Sequences 4-17 to 4-20 except the containment does not rupture (R) but only leaks following failure of containment venting. Increasing containment pressure eventually causes closure of the SRVs and a pressure rise in the vessel which precludes low pressure cooling, and core damage results in a leaking containment.

SEQUENCE 4-22 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*V1*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

A small LOCA (S2) occurs which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the PCS fails to remove heat from the core (Q1). HPCI is initiated for coolant makeup (/U1). Containment overpressure protection fails using SPC (W1) and CSS (W3), which eventually fails HPCI due to high suppression pool temperatures. CRD fails to supply sufficient makeup (U4) and the primary system is depressurized (/X1). Condensate fails (V1) followed by successful operation of LPCS (/V2) to cool the core. High containment pressure is alleviated by containment venting (/Y), which fails LPCS due to low NPSH. The reactor is again depressurized (/X3) and HPSW continues core cooling (/V4), resulting in no core damage in a vented containment.

SEQUENCE 4-23 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*V1*\overline{V2}*\overline{Y}*\overline{X3}*V4$

Same as Sequence 4-22 except HPSW fails to initiate (V4) following containment venting, at which point all coolant makeup is lost, resulting in core damage in a vented containment.

SEQUENCE 4-24 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*V1*\overline{V2}*\overline{Y}*X3$

Same as Sequence 4-22 except reactor depressurization following containment venting is unsuccessful (X3), precluding the use of HPSW, resulting in core damage in a vented containment.

SEQUENCES 4-25 TO 4-27

Same as Sequences 4-22 to 4-24 except containment venting is unsuccessful (Y) and the containment ruptures (/R).

SEQUENCE 4-28 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*V1*\overline{V2}*Y*R$

Same as Sequences 4-25 to 4-27 except the containment does not rupture (R) following containment venting which recloses the SRVs and precludes reactor depressurization and HPSW initiation, resulting in core damage in a leaking containment.

SEQUENCES 4-29 TO 4-35

Same as Sequences 4-22 to 4-28 except LPCS fails (V2) prior to containment venting and LPCI provides coolant makeup (/V3).

SEQUENCES 4-36 TO 4-42

Same as Sequences 4-29 to 4-35 except LPCI also fails (V3) and HPSW provides coolant makeup (/V4) prior to containment venting.

SEQUENCE 4-43 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*\overline{X1}*V1*V2*V3*V4$

Same as Sequences 4-36 to 4-42 except HPSW fails (V4), which leaves no system available for coolant makeup, resulting in core damage in a vulnerable containment.

SEQUENCE 4-44 -- $S2*\overline{C}*\overline{LOSP}*Q1*\overline{U1}*W1*W3*U4*X1$

Same as Sequence 4-22 until reactor depressurization is unsuccessful (X1) following CRD failure. All low-pressure coolant makeup is now lost, which leads to core damage in a vulnerable containment.

SEQUENCES 5 TO 7

Same as Sequences 2 to 4 except RCIC provides early high-pressure coolant makeup (/U2) following HPCI failure (U1).

SEQUENCES 8 TO 9

A small LOCA (S2) occurs which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the PCS fails to remove heat from the core (Q1). HPCI (U1) and RCIC (U2) fail to provide high-pressure coolant makeup. The reactor is depressurized (/X1) and Condensate successfully provides coolant makeup (/V1). Containment overpressure protection is provided by SPC (/W1) or CSS (/W3), resulting in no core damage in a safe containment.

SEQUENCE 10-1 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 8 until SPC (W1) and CSS (W3) fail to provide containment overpressure protection, resulting in the eventual loss of Condensate due to high primary system pressure, which occurs after SRVs shut on high containment pressure. CRD is initiated (/U4) to cool the core. High containment pressure is alleviated by venting (/Y). CRD continues to cool the core (/U4') resulting in no core damage in a vented containment.

SEQUENCES 10-2 TO 10-3

Same as Sequence 10-1 except CRD does not survive containment venting (U4'), the reactor is depressurized (/X3), and Condensate (/V1) or HPSW (/V4) provides coolant makeup.

SEQUENCE 10-4--$S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*V1*V4$

Same as Sequence 10-2 except Condensate (V1) and HPSW (V4) fail, at which point all coolant makeup is lost, resulting in core damage in a vented containment.

SEQUENCE 10-5 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*\overline{U4}*\overline{Y}*U4'*X3$

Same as Sequence 10-1 except CRD does not survive containment venting (U4') and reactor depressurization is unsuccessful (X3), leading to core damage in a vented containment.

SEQUENCES 10-6 TO 10-10

Same as Sequences 10-1 to 10-5 except the containment is not vented (Y) and eventually ruptures (/R).

SEQUENCE 10-11 -- $\overline{S2}*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 10-6 until the containment does not rupture but forms a leak, which does not affect CRD operation, resulting in no core damage in a leaking containment.

SEQUENCE 10-12 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 10-11 except CRD does not operate following the leak in containment (U4), resulting in core damage in a vulnerable containment.

SEQUENCES 10-13 TO 10-14

Same as Sequence 10-1 until CRD fails to initiate (U4) following the loss of Condensate. The containment is vented (/Y) to relieve the pressure and following reactor depressurization (X3), Condensate (/V1) or HPSW (/V4) provides core coolant, resulting in no core damage in a vented containment.

SEQUENCE 10-15 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*U4*\overline{Y}*\overline{X3}*V1*V4$

Same as Sequence 10-13 except both Condensate (V1) and HPSW (V4) fail, leaving no system available for coolant makeup, resulting in core damage in a vented containment.

SEQUENCE 10-16 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*U4*\overline{Y}*X3$

Same as Sequence 10-13 except reactor depressurization is unsuccessful (X3) following containment venting, which leaves Condensate and HPSW unavailable for coolant makeup, resulting in core damage in a vented containment.

SEQUENCES 10-17 TO 10-20

Same as Sequences 10-13 to 10-16 except containment venting is unsuccessful (Y), leaving the containment overpressurized, resulting in eventual rupture of the containment (/R).

SEQUENCE 10-21 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*\overline{V1}*W1*W3*U4*Y*R$

Same as Sequence 10-17 until the containment does not rupture (R), and core damage results in a vulnerable containment.

SEQUENCES 11 TO 12

Same as Sequences 8 to 9 except LPCS provides coolant makeup (/V2) following Condensate failure (V1).

SEQUENCES 13-1 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 11 until containment cooling with SPC (W1) and CSS (W3) fails. High containment pressure eventually closes the SRVs, which allows the primary system pressure to increase, resulting in the loss of LPCS (V2). CRD is successfully initiated in the one pump mode (/U4) to continue coolant makeup. Containment overpressure protection is accomplished by containment venting (/Y). CRD continues to provide coolant makeup (/U4'), resulting in no core damage in a vented containment.

SEQUENCE 13-2 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*\overline{V4}$

Same as Sequence 13-1 except CRD does not survive containment venting (U4'), the reactor is depressurized (/X3) to allow HPSW to continue coolant makeup (/V4).

SEQUENCES 13-3 TO 13-4

Same as Sequence 13-2 except either HPSW fails (V4) or reactor depressurization fails (X3), leaving no systems available for coolant makeup, resulting in core damage in a vented containment.

SEQUENCES 13-5 TO 13-8

Same as Sequences 13-1 to 13-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 13-9 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 13-5 until the containment does not rupture (R) but develops a leak. CRD continues to provide coolant makeup (/U4'), resulting in no core damage in a leaking containment.

SEQUENCE 13-10 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 13-9 except CRD does not continue to operate following the leak in containment (U4), resulting in core damage in a vulnerable containment.

SEQUENCE 13-11 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*U4*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 13-1 except CRD fails to initiate (U4) following the loss of LPCS. The containment is vented (/Y) and the primary system is depressurized (X3) to allow HPSW to provide coolant makeup (V4), resulting in no core damage in a vented containment.

SEQUENCES 13-12 TO 13-13

Same as Sequence 13-11 except either HPSW fails (V4) or reactor depressurization is unsuccessful (X3), leaving no core coolant system available, resulting in core damage in a vented containment.

SEQUENCES 13-14 TO 13-16

Same as Sequences 13-11 to 13-13 except containment venting fails (Y) and the containment ruptures (/R).

SEQUENCES 13-17 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W3*U4*Y*R$

Same as Sequence 13-14 until the containment does not rupture (R), causing closure of the SRVs and hence no low pressure cooling, resulting in core damage in a vulnerable containment.

SEQUENCES 14 TO 16

Same as Sequences 11 to 13 except LPCI provides early low-pressure coolant makeup (/V3) following LPCS failure (V2).

SEQUENCES 17 TO 19

Same as Sequences 14 to 16 except HPSW provides early low-pressure coolant makeup (/V4) following LPCI failure (V3).

SEQUENCE 20 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*\overline{X1}*V1*V2*V3*V4$

Same as Sequence 17 except HPSW fails to operate (V4). At this point all core coolant systems are lost, resulting in early core damage in a vulnerable containment.

SEQUENCE 21 -- $S2*\overline{C}*\overline{LOSP}*Q1*U1*U2*X1$

Following the small LOCA (S2) and successful reactor scram (/C), offsite power is maintained (/LOSP). The PCS fails to remove heat from the core (Q1). Both high-pressure injection systems, HPCI (U1) and RCIC (U2), fail to operate. Depressurization of the reactor is unsuccessful (X1), which leaves no system available for coolant makeup, resulting in early core damage in a vulnerable containment.

SEQUENCES 22 TO 38

Same as Sequences 2 to 21 except offsite power is not maintained (LOSP) early in the sequence. Onsite emergency power is utilized for core cooling systems, with the exception of the Condensate system, which requires off-site power to operate. All sequence outcomes are the same, except the success paths for Condensate events in the tree are eliminated.

SEQUENCE 39 -- S2*C

The RPS fails to scram the reactor (C) following the small LOCA (S2). This sequence has a low probability and is not developed further.

4.4.6 Small-Small (Recirculation Pump Seal) LOCA Event Tree

This section contains information on the small-small LOCA event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.6.1 Introduction

The recirculation pump seal LOCA (S3) was treated as either a small (S2) liquid LOCA or a transient with PCS initially available (T3A) depending on early actions of the operator (see Table 4.3-4 for corresponding success criteria). Experience suggests that the small-small LOCA category is dominated by recirculation pump seal failures. Such a leak would be easily identifiable for two reasons. First, the sources of such leaks are well-instrumented on recirculation pumps. Secondly, the Peach Bottom Emergency Procedure Guidelines (EPGs) call for the operator to first suspect a pump seal leak if drywell pressure begins to rise or unidentified leakage is detected. Procedures call for slowdown of the problem pump and then isolation of the pump. PCS operation would probably not be interrupted and power operation could possibly continue for a period of time.

4.4.6.2 Event Tree

The Small-Small LOCA event tree is depicted by Figure 4.4-4. The S3 LOCA analysis and the corresponding event tree assume that conditions proceed to the need for a reactor scram. Otherwise, if the operator should detect and isolate the leak <u>before</u> a reactor trip, the plant simply "rides" through the event resulting in no real challenge to the plant.

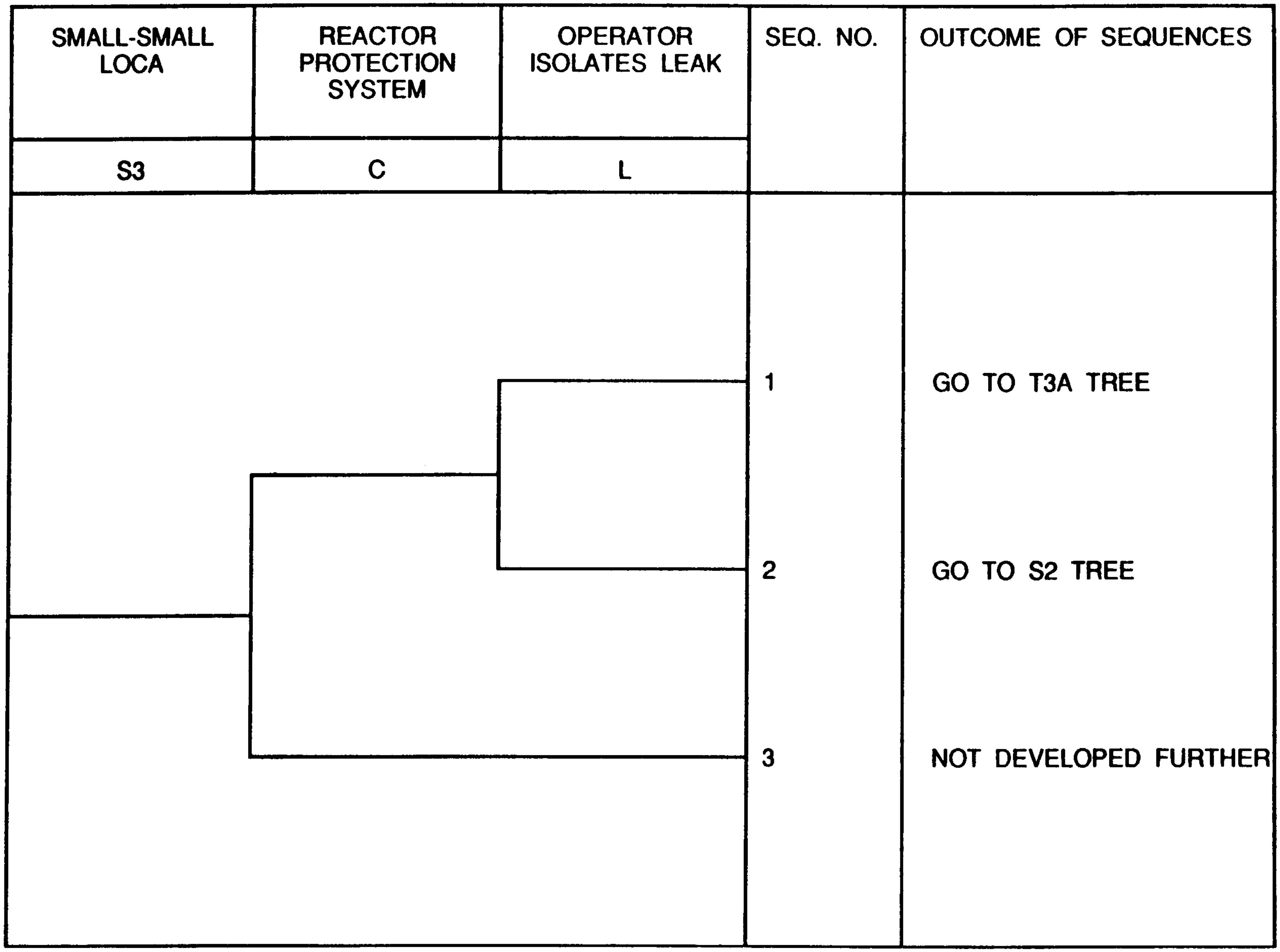


Figure 4.4-4. Small-Small LOCA Event Tree

The events in the tree include the following:

- S3: Initiating event, small-small LOCA (~50-to-100 gpm maximum).
- C: Success or failure of the Reactor Protection System (RPS). Success implies scram by the control rods.
- L: Success or failure of leak detection and isolation. Success implies the operator detects and isolates the leaky pump thus stopping the LOCA. With the reactor scrammed, the event becomes a transient with PCS most likely available.

  The course of events then follows the S2 LOCA or T3A transient tree as shown. See those tree descriptions for more information.

The following descriptions refer to the sequences found in Figure 4.4-4.

SEQUENCE 1 -- $S3*\overline{C}*\overline{L}$

A small-small LOCA occurs (S3) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). The operators isolate the leak (/L) and the sequence transfers to the T3A tree.

SEQUENCE 2 -- $S3*\overline{C}*L$

Same as Sequence 1 except the operator fails to detect the leak and the sequence transfers to the S2 tree.

SEQUENCE 3 -- S3*C

Following the small-small LOCA (S3), the RPS fails to scram the reactor and the sequence is not developed further since the probability of such a sequence (including additional failures which must occur to result in core damage) is sufficiently low.

4.4.7 Loss of Offsite Power Event Tree

This section contains information on the loss of offsite power event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.7.1 Success Criteria

Two criteria specific to the loss of offsite power initiator are described below.

(1) For scenarios in which core cooling has been provided for a period of approximately 6-8 hours or more, one CRD pump operation is considered adequate for continued success of core cooling. This

is based on the low decay heat levels reached by that time with no significant breach of the primary system. While the CRD failure model explicitly treats only the two pump criteria for success, single pump operation was treated as success during these long-term scenarios by eliminating (by hand) failures of the CRD system which would fail only one pump.

(2) For scenarios in which core cooling is successful up to the time of containment venting or containment failure, one CRD pump or depressurization with one HPSW pump operation is considered to be adequate to continue successful core cooling.

4.4.7.2 Event Tree

Figure 4.4-5 displays the event tree for the loss of offsite power initiator. The entire PCS, Feedwater, and Condensate systems are not shown in the tree since loss of offsite power also prevents operation of these systems. Should offsite power be restored, these systems could be used to mitigate the event. The following discussions define the event tree headings and describe the sequences presented.

The following event tree headings appear on the tree in the approximate chronological order that would be expected following a loss of offsite power. For convenience, the RHR containment cooling choices are shown early in the tree to decrease the size of the event tree. Otherwise, the tendency is to show high and then low pressure injection systems, followed by containment venting, and finally long-term continued core cooling possibilities. In addition, onsite AC power restoration is shown as a specific event so that station blackout sequences can be explicitly depicted.

T1: Initiating event, loss of offsite power.

C: Success or failure of the RPS. Success implies automatic scram by the control rods.

M: Success or failure of Reactor Coolant System (RCS) overpressure protection (if required) by automatic operation of the SRVs. Success implies prevention of RCS overpressure so as to avoid damage to the primary system.

P: Success or failure associated with reclosing of any SRVs which should open in response to reactor vessel pressure rises throughout the sequence. Success implies reclosure of all valves when vessel pressure drops below the closure setpoints. P1, P2 and P3 refer to the failure to reclose one, two and three SRVs, respectively.

B: Success or failure of the onsite AC power system (diesel generators and associated equipment and emergency buses) in response to the loss of offsite power. Success implies operation of at least one emergency AC power division so that AC-powered mitigating systems can be utilized. Failure implies loss of all AC, or station blackout.

U1: Success or failure of the HPCI system. Success implies operation of the HPCI pump train so as to maintain sufficient coolant injection. U1′ refers to the HPCI system without pump room ventilation.

U2: Success or failure of the RCIC system. Success implies operation of the RCIC pump train so as to maintain sufficient coolant injection. U2′ refers to the RCIC system without pump room ventilation.

X1: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS or manual operation of other SRVs such that three valves or more are opened allowing low pressure injection.

U3: Success or failure of the CRD system as an injection source. Success implies two pump operation.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

W1,W2,W3: Success or failure of the RHR system in the SPC, SDC, or CS mode, respectively. Success implies at least one RHR pump operating in any one of the three modes with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

X2: Success or failure of primary system depressurization. Success implies automatic or manual operation of any three of eleven ADS valves to allow the SDC mode of RHR to be initiated.

U4: Success or failure of the CRD system as an injection source. Success implies operation in the one pump mode.

Y: Success or failure of containment venting. Success implies that the six inch integrated leak test line or larger size line is open so as to prevent containment by overpressure. As necessary, water makeup is also eventually supplied to the suppression pool.

R: Success or failure of the containment to withstand overpressurization. Success implies the containment ruptures before core damage.

X3: Success or failure of primary system depressurization. Success implies automatic or manual operation of ADS occurs subsequent to initial depressurization to allow low pressure coolant injection.

The following descriptions refer to the sequences found in Figure 4.4-5.

SEQUENCE 1 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*\overline{W1}$

A loss-of-offsite power occurs (T1) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). The SRVs properly cycle to control reactor pressure (/M, /P) and onsite emergency AC power is established (/B). HPCI is initiated (/U1) for core cooling and SPC is initiated (/W1) for containment overpressure protection, resulting in a safe core and containment.

SEQUENCE 2 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*\overline{W2}$

Same as Sequence 1 but SPC fails to provide containment overpressure protection (W1) and SDC is initiated (/W2) following reactor depressurization (/X2).

SEQUENCES 3-1 TO 3-4

Same as Sequence 2 except SDC fails (W2) and CSS continues to provide containment overpressure protection (/W3). HPCI has failed due to high suppression pool temperatures and either CRD (/U4), LPCS (/V2), LPCI (/V3) or HPSW (/V4) continues core cooling.

SEQUENCE 3-5 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*\overline{W3}*U4*V2*V3*V4$

Same as Sequences 3-1 to 3-4 except CRD (U4), LPCS (V2), LPCI (V3) and HPSW (V4) fail, leaving no system available to cool the core, resulting in core damage in a vulnerable containment.

SEQUENCE 4-1 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 2 except SDC fails (W2), followed by CSS failure (W3), leaving the containment without overpressure protection. HPCI eventually fails due to high suppression pool temperatures and CRD is initiated (/U4). The containment is successfully vented (/Y) and CRD continues to provide core coolant (/U4'), resulting in no core damage in a vented containment.

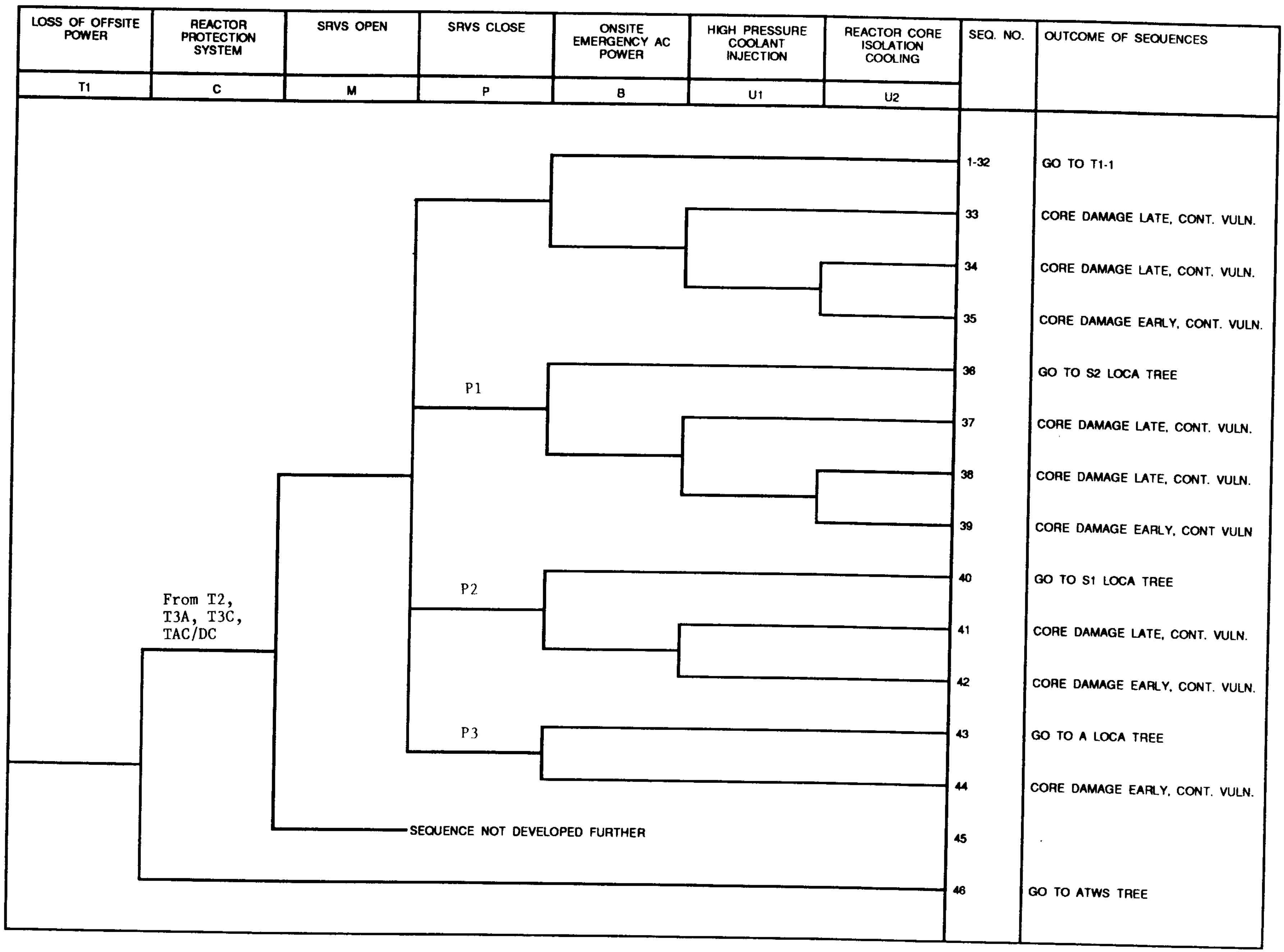


Figure 4.4-5. Loss of Offsite Power Event Tree (Page 1 of 5)

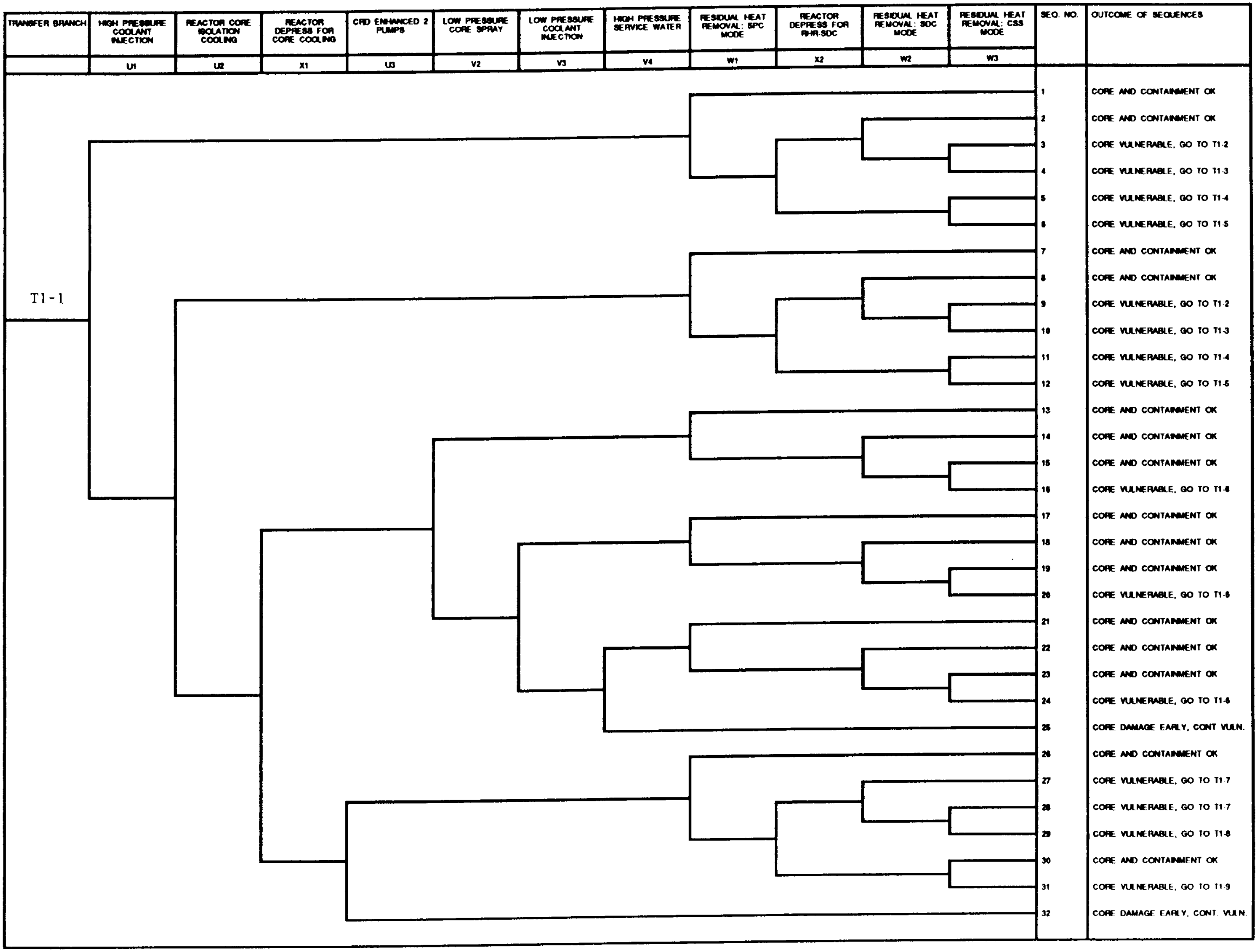


Figure 4.4-5. Loss of Offsite Power Event Tree (Page 2 of 5)

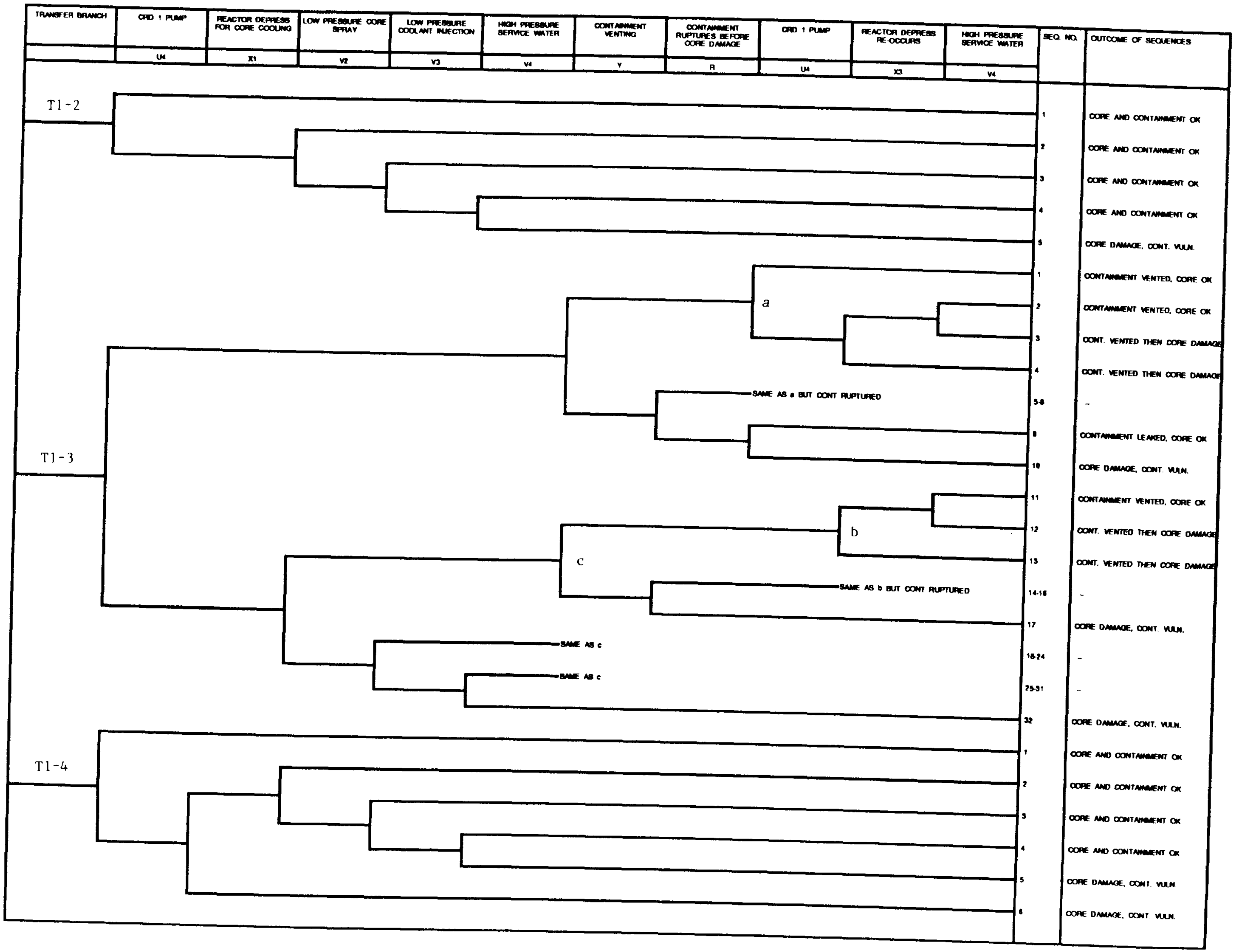


Figure 4.4-5. Loss of Offsite Power Event Tree (Page 3 of 5)

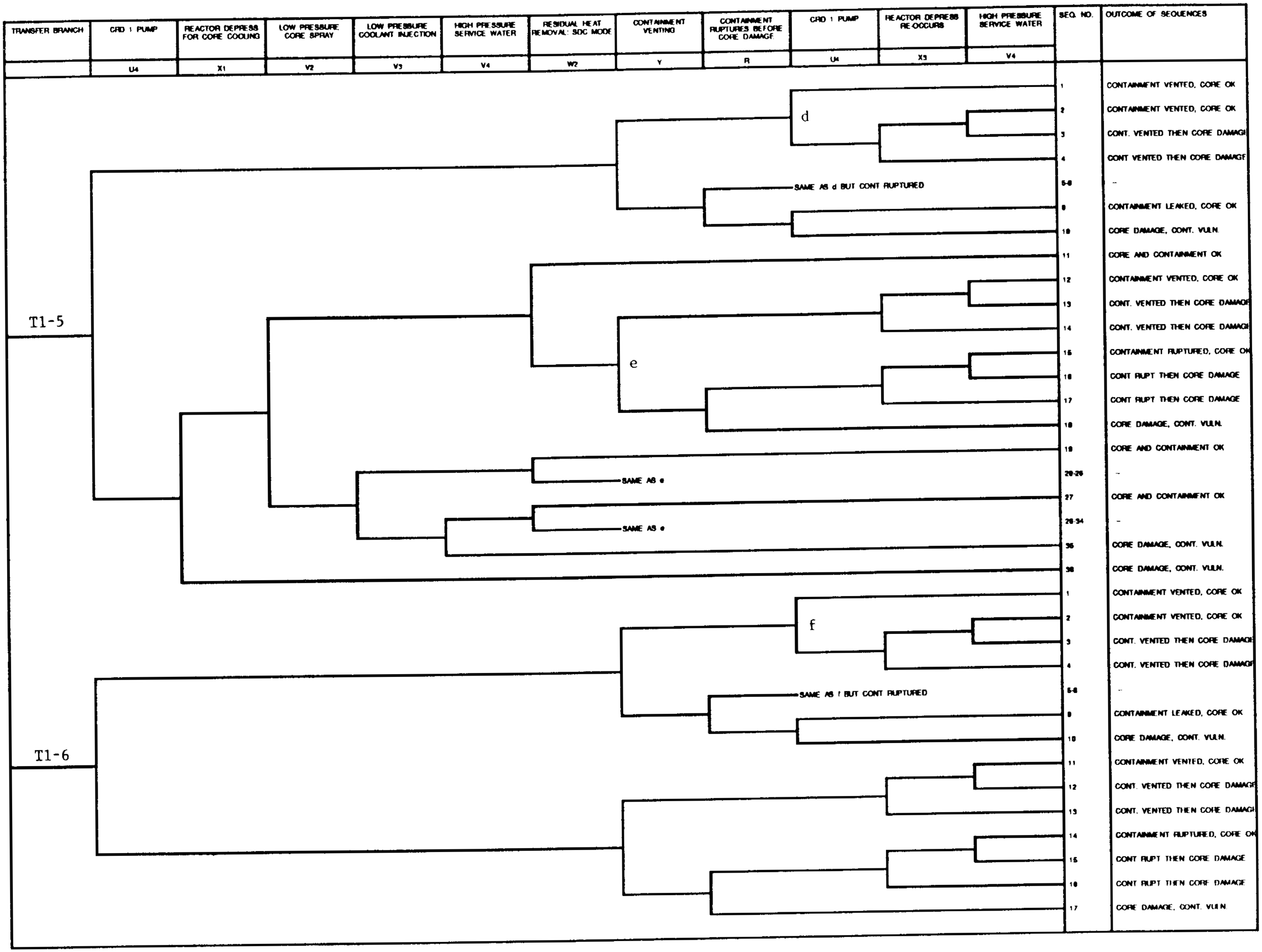


Figure 4.4-5. Loss of Offsite Power Event Tree (Page 4 of 5)

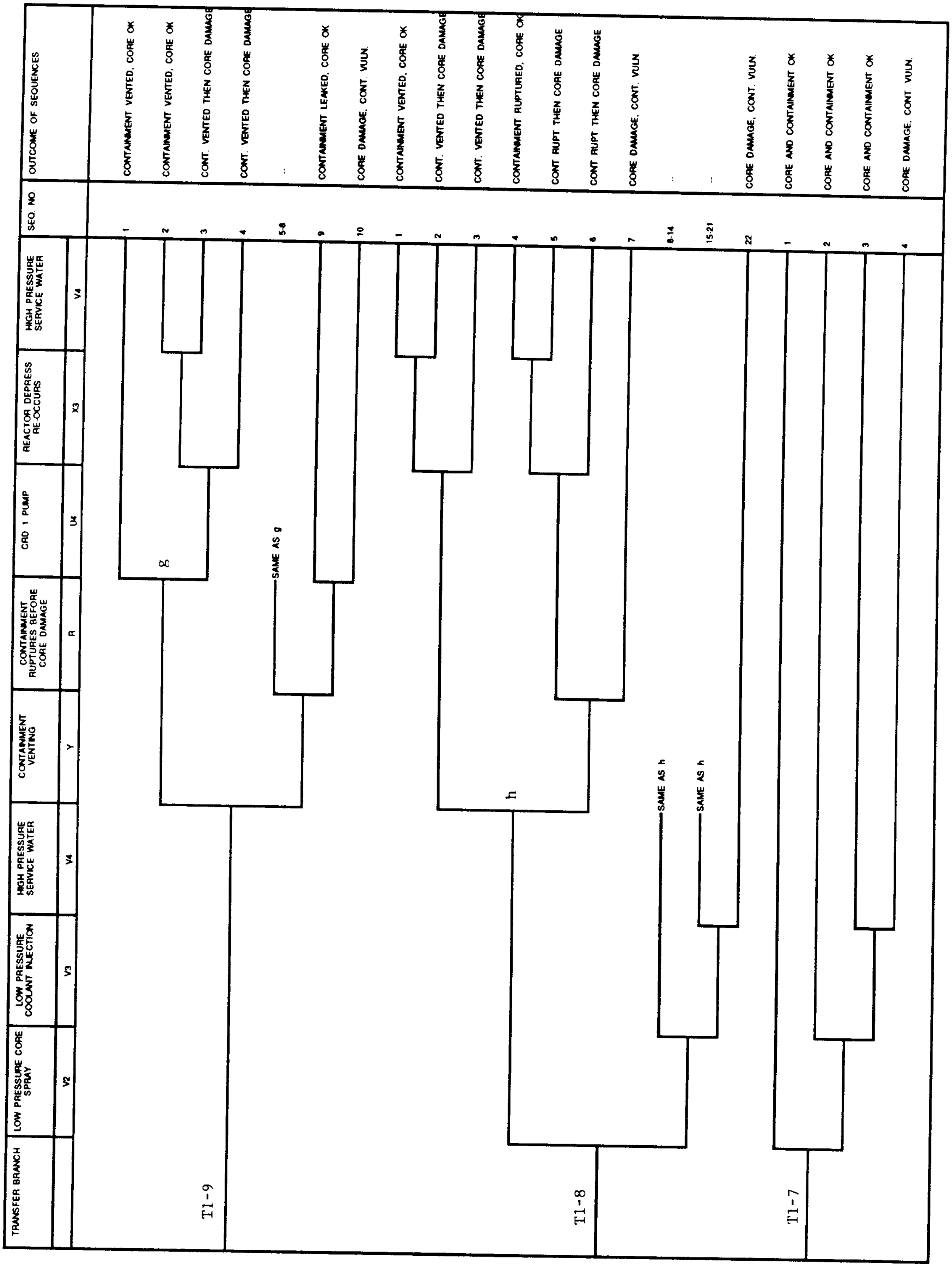


Figure 4.4-5. Loss of Offsite Power Event Tree (Page 5 of 5)

SEQUENCE 4-2 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*\overline{V4}$

Same as Sequence 4-1 except CRD fails during containment venting (U4'). Prior to containment venting, due to the loss of containment overpressure protection, high containment pressure forces the SRVs closed and the primary system pressure increases before injection is restored with CRD. The reactor is depressurized (/X3) and HPSW provides core coolant (/V4).

SEQUENCES 4-3 TO 4-4

Same as Sequence 4-2 except HPSW fails (V4), or reactor depressurization prior to HPSW operation is unsuccessful (X3), resulting in core damage in a vented containment.

SEQUENCES 4-5 TO 4-8

Same as Sequences 4-1 to 4-4 except containment venting fails (Y) and the containment ruptures before core damage (/R).

SEQUENCE 4-9 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 4-8 except the containment does not rupture (R) but develops a leak. CRD continues to operate (/U4'), resulting in no core damage in a leaking containment.

SEQUENCE 4-10 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 4-9 except CRD does not continue to operate (U4') following the containment leak and because high containment pressure, ADS cannot relieve primary pressure to allow HPSW to operate, resulting in core damage in a leaking containment.

SEQUENCE 4-11 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*U4*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 4-1 except CRD does not operate (U4) following HPCI failure. LPCS is initiated (/V2) to continue core cooling and the containment is eventually vented (/Y). The LPCS pumps then fail due to low NPSH and the reactor is depressurized to allow HPSW to cool the core (/V4), resulting in a safe core in a vented containment.

SEQUENCES 4-12 TO 4-13

Same as Sequence 4-11 except HPSW fails (V4), or depressurization prior to HPSW operation fails (X3), resulting in core damage in a vented containment.

SEQUENCES 4-14 TO 4-16

Same as Sequences 4-11 to 4-13 except containment venting is unsuccessful (Y) and the containment ruptures before core damage (/R).

SEQUENCE 4-17 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*U4*\overline{V2}*Y*R$

Same as Sequence 4-11 except containment venting fails (Y) and the containment does not rupture (R), thereby closing the SRVs due to high containment pressure and preventing low pressure cooling. This results in core damage in a leaking containment.

SEQUENCES 4-18 TO 4-24

Same as Sequences 4-11 to 4-17 except, following LPCS failure (V2), LPCI provides core coolant (/V3) prior to containment venting.

SEQUENCES 4-25 TO 4-31

Same as Sequences 4-18 to 4-24 except, following LPCI failure (V3), HPSW provides core coolant (/V4) prior to containment venting.

SEQUENCE 4-32 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*\overline{X2}*W2*W3*U4*V2*V3*V4$

Same as Sequence 4-11 except LPCS (V2), LPCI (V3), and HPSW (V4) fail and all core cooling is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 5-1 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*W1*X2*\overline{W3}*\overline{U4}$

Same as Sequence 2 except reactor depressurization for SDC is unsuccessful (X2) and CSS is initiated to provide containment overpressure protection (/W3). HPCI has failed due to high suppression pool temperatures before CSS is established and CRD is initiated to cool the core (/U4), resulting in a safe core and containment.

SEQUENCES 5-2 TO 5-4

Same as Sequence 5-1 except CRD fails to provide coolant injection (U4), the reactor is depressurized (/X1), and LPCS (/V2), LPCI (/V3) or HPSW (/V4) provide core cooling.

SEQUENCES 5-5 TO 5-6

Same as Sequence 5-2 except either reactor depressurization fails (X1) or LPCS (V2), LPCI (V3) and HPSW (V4) fail following depressurization, resulting in core damage in a vulnerable containment.

SEQUENCE 6-1 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 5 except CSS fails (W3), resulting in the loss of all containment overpressure protection. High suppression pool temperatures fail HPCI and CRD is initiated for core coolant (/U4). Increasing containment pressure is relieved by containment venting (/Y). CRD survives venting (/U4') and the core is safe in a vented containment.

SEQUENCE 6-2 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*\overline{V4}$

Same as Sequence 6-2 except CRD does not survive containment venting (U4'), the reactor is depressurized (/X1), and HPSW continues core cooling (/V4).

SEQUENCES 6-3 TO 6-4

Same as Sequence 6-2 except either reactor depressurization fails (X1), or HPSW fails (V4) following reactor depressurization, leading to core damage in a vented containment.

SEQUENCES 6-5 TO 6-8

Same as Sequences 6-1 to 6-4 except containment venting is unsuccessful (Y) and the containment ruptures (/R).

SEQUENCE 6-9 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 6-5 except the containment does not rupture (R), but develops a leak. This causes closure of the SRVs and the inability to use low pressure cooling. CRD continues coolant injection (/U4'), resulting in no core damage in a leaking containment.

SEQUENCE 6-10 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 6-9 except CRD fails (U4') following the containment leak, at which point all coolant makeup is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 6-11 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*\overline{W2}$

Same development as Sequence 6-1 until CRD fails to initiate (U4) following HPCI failure. The reactor is depressurized (/X1) to initiate LPCS for coolant injection (/V2). The reactor is sufficiently depressurized to initiate late SDC for containment overpressure protection (/W2), resulting in a safe core and containment.

SEQUENCE 6-12 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*W2*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 6-11 except SDC fails to provide containment overpressure protection (W2), followed by successful venting of the containment (/Y). Coolant injection is restored using HPSW (/V4) following reactor depressurization (/X3), resulting in a safe core in a vented containment.

SEQUENCES 6-13 TO 6-14

Same as Sequence 6-12 except either reactor depressurization fails (X3) or HPSW fails (V4) following reactor depressurization, resulting in core damage in a vented containment.

SEQUENCES 6-15 TO 6-17

Same as Sequences 6-12 to 6-14 except containment venting fails (Y) and the containment ruptures (/R).

SEQUENCE 6-18 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*W2*Y*R$

Same as Sequence 6-11 until containment overpressure protection with SDC fails (W2), followed by failure of containment venting (Y). The containment does not rupture (R), disallowing use of low pressure systems because of closure of the SRVs. Core damage results in a vulnerable containment.

SEQUENCES 6-19 TO 6-26

Same as Sequences 6-11 to 6-18 except LPCI provides coolant makeup (/V3) following failure of LPCS (V2).

SEQUENCES 6-27 TO 6-34

Same as Sequences 6-19 to 6-26 except HPSW provides coolant makeup (/V4) following failure of LPCI (V3).

SEQUENCE 6-35 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V2*V3*V4$

Same as Sequence 6-11 until LPCS fails (V2) following reactor depressurization, followed by failure of both LPCI (V3) and HPSW (V4), at which point all coolant makeup is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 6-36 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*\overline{U1}*W1*X2*W3*U4*X1$

Same as Sequence 6-11 until CRD fails to continue coolant makeup (U4) following HPCI failure. Reactor depressurization fails (X1), which disables all low-pressure core cooling systems, resulting in core damage in a vulnerable containment.

SEQUENCES 7 TO 12

Same as Sequences 1 to 6 except RCIC provides high pressure coolant makeup (/U2) following failure to initiate HPCI (U1).

SEQUENCES 13 TO 15

Same as Sequence 1 until failure to initiate HPCI (U1), followed by failure of RCIC (U2). The reactor is depressurized (/X1) and LPCS is initiated for coolant makeup (/V2). Containment overpressure protection is provided by SPC (/W1), SDC (/W2), or CSS (/W3), resulting in a safe core and containment.

SEQUENCES 16-1 TO 16-2

Same as Sequence 13 until SPC fails (W1), followed by failure of SDC (W2) and CSS (W3). Without containment overpressure protection, the pressure in containment rises until the SRVs close. Primary system pressure then rises, eventually failing LPCS (V2). CRD is initiated (/U4) for coolant makeup. High containment pressure is relieved by containment venting (/Y). CRD continues to cool the core, or the reactor is depressurized (/X1) and HPSW cools the core (/V4) if CRD does not survive the venting.

SEQUENCES 16-3 TO 16-4

Same as Sequence 16-1 except CRD does not survive containment venting and either reactor depressurization is unsuccessful (X1), or HPSW fails (V4) following reactor depressurization, resulting in core damage in a vented containment.

SEQUENCES 16-5 TO 16-8

Same as Sequences 16-1 to 16-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 16-9 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*\overline{U4}'$

Same as Sequence 16-5 except the containment does not rupture (R) but develops a leak. CRD survives (/U4') resulting in a safe core in a leaking containment.

SEQUENCE 16-10 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 16-9 except CRD does not survive the development of a leak in containment (U4'), all coolant systems are lost, and core damage results in a vulnerable containment.

SEQUENCE 16-11 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*U4*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 16-1 until CRD fails to initiate (U4) following loss of containment overpressure protection. Increasing containment pressure is relieved by containment venting (/Y) and HPSW is initiated to cool the core (/V4) following primary system depressurization (/X1). The core is safe in a vented containment.

SEQUENCES 16-12 TO 16-13

Same as Sequence 16-11 except either HPSW fails to cool the core (V4) or primary system depressurization fails (X1) prior to HPSW operation, resulting in core damage in a vented containment.

SEQUENCES 16-14 TO 16-16

Same as Sequences 16-11 to 16-13 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 16-17 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*U4*Y*R$

Same as Sequence 16-11 until containment venting fails (Y). The containment does not rupture (R) and continues to pressurize, resulting in core damage in a vulnerable containment since the SRVs are forced closed, preventing low pressure cooling.

SEQUENCES 17 TO 20

Same as Sequences 13 to 15 except LPCI provides early core coolant (/V3) following LPCS failure (V2).

SEQUENCES 21 TO 24

Same as Sequences 17 to 20 except HPSW provides early core coolant (/V4) following LPCI failure (V3).

SEQUENCE 25 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*\overline{X1}*V2*V3*V4$

Same as Sequences 21 to 24 until HPSW fails (V4), at which point all coolant makeup is lost, resulting in early core damage in a vulnerable containment.

SEQUENCE 26 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*\overline{W1}$

Same as Sequence 13 until reactor depressurization fails (X1) following failure to initiate high-pressure coolant systems. CRD is initiated in the two-pump mode to provide sufficient injection capacity (/U3). Containment overpressure protection is provided by SPC (/W1), resulting in a safe core and containment.

SEQUENCES 27-1 TO 27-3

Same as Sequence 26 until SPC fails to initiate (W1), the reactor is depressurized (/X2), and SDC provides containment overpressure protection (/W2). Reactor depressurization for SDC increases CRD flow rate which, when considering CST inventory is depleting, is assumed to fail the CRD pumps due to low NPSH. LPCS (/V2), LPCI (/V3) or HPSW (/V4) is initiated for core coolant, resulting in a safe core and containment.

SEQUENCE 27-4 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*\overline{W2}*V2*V3*V4$

Same as Sequence 27-1 until LPCS fails (V2) to initiate after CRD fails, followed by unsuccessful operation of LPCI (V3) and HPSW (V4), resulting in core damage in a vulnerable containment.

SEQUENCES 28-1 TO 28-4

Same as Sequences 27-1 to 27-4 except CSS provides containment overpressure protection (/W3) following SDC failure (W2).

SEQUENCE 29-1 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 28-1 until CSS fails to initiate (W3), at which point all containment cooling is lost. CRD failed due to reactor depressurization for SDC, so LPCS is initiated (/V2) to continue core cooling. Without containment overpressure protection, the pressure in containment is increasing and eventually closes the SRVs. Containment venting (/Y) is successful to relieve containment overpressurization, which fails LPCS due to low NPSH. Since the SRVs are closed, a pressure increase in the primary system begins until the reactor is again depressurized (/X3) and HPSW cools the core, resulting in a safe core in a vented containment.

SEQUENCES 29-2 TO 29-3

Same as Sequence 29-1 except either HPSW fails (V4) or reactor depressurization fails (X3) prior to HPSW operation, leaving no system available for coolant makeup, resulting in core damage in a vented containment.

SEQUENCES 29-4 TO 29-6

Same as Sequences 29-1 to 29-3 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 29-7 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*\overline{V2}*Y*R$

Same as Sequence 29-4 until the containment fails to rupture (R), which precludes HPSW operation because of forced closure of the SRVs. This results in core damage in a vulnerable containment.

SEQUENCES 29-8 TO 29-14

Same as Sequences 29-1 to 29-7 except LPCS fails to initiate (V2) following containment cooling failure and LPCI provides coolant makeup (/V3).

SEQUENCES 29-15 TO 29-21

Same as Sequences 29-8 to 29-14 except LPCI fails to initiate (V3) following containment cooling failure and HPSW provides coolant makeup (/V4).

SEQUENCE 29-22 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*V2*V3*V4$

Same as Sequence 29-11 until LPCS fails (V2) following containment cooling failure. LPCI (V3) and HPSW (V4) also fail to initiate, resulting in core damage in a vulnerable containment.

SEQUENCE 30 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*X2*\overline{W3}$

Same as Sequence 26 until SPC fails (W1), followed by failure of reactor depressurization for SDC (X2). CSS is initiated to provide containment overpressure protection (/W3). Since reactor depressurization was unsuccessful, CRD does not fail, resulting in a safe core and containment.

SEQUENCES 31-1 TO 31-2

Same as Sequence 30 until CSS fails (W3), at which point all containment overpressure protection is lost. Eventually containment venting is performed to relieve containment overpressure (/Y). CRD continues to cool the core in the one-pump mode (/U4), or CRD fails on containment venting and HPSW cools the core (/V4), resulting in a safe core in a vented containment.

SEQUENCES 31-3 TO 31-4

Same as Sequence 31-2 except HPSW fails (V4) or reactor depressurization fails prior to HPSW operation (X3), resulting in core damage in a vented containment.

SEQUENCES 31-5 TO 31-8

Same as Sequences 31-1 to 31-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 31-9 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*\overline{U4}$

Same as Sequence 31-5 except the containment does not rupture (R) but develops a leak. CRD continues to cool the core, resulting in a safe core in a leaked containment.

SEQUENCE 31-10 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*U4$

Same as Sequence 31-9 except CRD does not survive the containment leak (U4), resulting in core damage in a vulnerable containment.

SEQUENCE 32 -- $T1*\overline{C}*\overline{M}*\overline{P}*\overline{B}*U1*U2*X1*U3$

Same as Sequence 26 until CRD fails to initiate (U3) in the two-pump mode following failure to depressurize the reactor, which leaves no system available for coolant makeup. Early core damage results, with a vulnerable containment.

SEQUENCES 33 TO 34

A loss-of-offsite-power occurs (T1) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). The SRVs properly cycle to control reactor pressure (/M, /P) and onsite emergency power fails to be established (B). HPCI or RCIC is initiated (/U1', /U2') for coolant injection until it fails in the harsh environment or due to battery depletion, and core damage occurs late in a vulnerable containment.

SEQUENCE 35 -- $T1*\bar{C}*\bar{M}*\bar{P}*B*U1'*U2'$

Same as Sequence 34 except RCIC fails to operate (U2') and early core damage results with a vulnerable containment since no other coolant injection is possible without AC power.

SEQUENCE 36 -- $T1*\bar{C}*\bar{M}*P1*\bar{B}$

A loss-of-offsite-power occurs (T1) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). The SRVs open to relieve reactor pressure (/M) but one SRV fails to close (P1), creating a loss-of-coolant accident. Onsite emergency power is established (/B) and the sequence is transferred to the S2 LOCA tree.

SEQUENCES 37 TO 38

Same as Sequence 36 except onsite emergency power is not established (B) and HPCI (/U1') or RCIC (/U2') provides coolant injection until it fails in the harsh environment or due to battery depletion. This results in late core damage in a vulnerable containment.

SEQUENCE 39 -- $T1*\bar{C}*\bar{M}*P1*B*U1'*U2'$

Same as Sequence 37 except both HPCI (U1') and RCIC (U2') fail to provide coolant injection, resulting in early core damage in a vulnerable containment.

SEQUENCE 40 -- $T1*\bar{C}*\bar{M}*P2*\bar{B}$

Same as Sequence 36 except two SRVs fail to close (P2) and the sequence is transferred to the S1 LOCA tree.

SEQUENCES 41 TO 42

Same as Sequence 40 except onsite emergency power is not established (B) and late core damage in a vulnerable containment results if HPCI (/U1) provides temporary coolant injection. If HPCI fails to operate, early core damage results with a vulnerable containment. RCIC does not have enough capacity to provide sufficient coolant in an S1 LOCA situation.

SEQUENCE 43 -- $T1*C*\overline{M}*P3*\overline{B}$

Same as Sequence 40 except three or more SRVs fail to close (P3) and the sequence is transferred to the A LOCA tree.

SEQUENCE 44 -- $T1*\overline{C}*\overline{M}*P3*B$

Same as Sequence 43 except onsite emergency power is not maintained (B) and high pressure coolant systems cannot operate in a large LOCA situation, resulting in early core damage in a vulnerable containment.

SEQUENCE 45 -- $T1*\overline{C}*M$

A loss-of-offsite-power occurs (T1) which generates a scram condition and the RPS successfully inserts the rods into the core (/C). The SRVs do not open to reduce reactor pressure (M). The sequence is not developed further because of its low probability.

SEQUENCE 46 -- T1*C

A loss-of-offsite power occurs (T1) which generates a scram condition and the RPS fails to insert the rods into the core (C). The sequence is transferred to the ATWS tree.

4.4.8 <u>Transient Without PCS Initially Available Event Tree</u>

This section contains information on the transient without PCS initially available event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.8.1 Event Tree

The T2 transient event tree is shown in Figure 4.4-6. The following discussions define the event tree headings and the sequences.

The events in the tree include:

<u>T2</u>: Initiating event, transient without the PCS initially available.

<u>C</u>: Success or failure of the RPS. Success implies automatic scram by the control rods.

<u>LOSP</u>: Success or failure to maintain offsite power.

M: Success or failure of Reactor Coolant System (RCS) overpressure protection (if required) by automatic operation of the SRVs. Success implies prevention of RCS overpressure so as to avoid damage to the primary system.

P: Success or failure associated with reclosing of any SRVs which should open in response to reactor vessel pressure rises throughout the sequence. Success implies reclosure of all valves when vessel pressure drops below the closure setpoints. P1, P2 and P3 refer to the failure to reclose one, two, three or more SRVs, respectively.

U1: Success or failure of the HPCI system. Success implies operation of the HPCI pump train so as to maintain sufficient coolant injection.

U2: Success or failure of the RCIC system. Success implies operation of the RCIC pump train so as to provide coolant injection.

X1: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS or manual operation of other SRVs such that three valves or more are opened allowing low pressure injection.

V1: Success or failure of the Condensate system. Success implies at least one pump operating with sufficient makeup to the condenser hotwell for a continuing water supply.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

U3: Success or failure of the CRD system as an injection source. Success implies two pump operation.

W1,W2,W3: Success or failure of the RHR system in the SPC, SDC, or CS mode, respectively. Success implies at least one RHR pump operating in any one of the three modes with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

X2: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS to allow the SDC mode of RHR to be initiated.

U4: Success or failure of the CRD system as an injection source. Success implies operation in the one pump mode.

Y: Success or failure of containment venting. Success implies that the six-inch integrated leak test line or larger size line is open so as to prevent containment by overpressure. As necessary, water makeup is also eventually supplied to the suppression pool.

R: Success or failure of the containment to withstand overpressurization. Success implies the containment ruptures before core damage. Failure implies the containment does not rupture.

X3: Success or failure of primary system depressurization. Success implies automatic or manual operation of ADS occurs subsequent to initial depressurization to allow low pressure injection.

The following descriptions refer to the sequences found in Figure 4.4-6.

SEQUENCE 1 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*\overline{W1}$

A transient occurs without the PCS available (T2) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the SRVs properly cycle to control reactor pressure (/M, /P). HPCI is initiated for core coolant (/U1). Increasing suppression pool temperatures cause SPC to be initiated (/W1), and the core and containment are safe.

SEQUENCE 2 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*\overline{W2}$

Same as Sequence 1 except SPC fails to provide containment overpressure protection (W1), the reactor is depressurized (/X2), and SDC continues to cool the containment (/W2).

SEQUENCES 3-1 TO 3-5

Same as Sequence 2 until SDC fails (W2) and CSS is initiated to provide containment overpressure protection (/W3). By the time CSS is initiated, the environment within the containment has failed HPCI. Core coolant is provided by Condensate (/V1), CRD (/U4), LPCS (/V2), LPCI (/V3) or HPSW (/V4), resulting in a safe core and containment.

SEQUENCE 3-6 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*\overline{W3}*V1*U4*V2*V3*V4$

Same as Sequence 3-1 except all low-pressure cooling systems fail (Condensate, CRD (1 pump), LPCS, LPCI, HPSW) which results in core damage in a vulnerable containment.

SEQUENCE 4-1 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{V1}*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 2 until SDC fails to cool the containment (W2), followed by failure of CSS (W3), resulting in the loss of all containment overpressure protection. HPCI has failed due to the adverse containment environment, and Condensate is initiated for core coolant (/V1). Pressure buildup in containment eventually closes the ADS valves, resulting in a pressure rise in the primary. This higher primary pressure fails the Condensate system, and CRD is initiated to continue core cooling (/U4). Containment venting is performed to relieve high containment pressure (/Y). CRD survives containment venting (/U4') and the core is safe in a vented containment.

SEQUENCES 4-2 TO 4-3

Same as Sequence 4-1 except CRD does not survive containment venting. The reactor is depressurized again (/X3) and condensate (/V1) or HPSW (/V4) provide core coolant.

SEQUENCES 4-4 TO 4-5

Same as Sequence 4-3 except either reactor depressurization fails (X3), or HPSW fails (V4), which leaves no system available for core coolant, resulting in core damage in a vented containment.

SEQUENCES 4-6 TO 4-10

Same as Sequences 4-1 to 4-5 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 4-11 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{V1}*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 4-6 except the containment does not rupture (R) but develops a leak. CRD continues to provide core cooling (/U4').

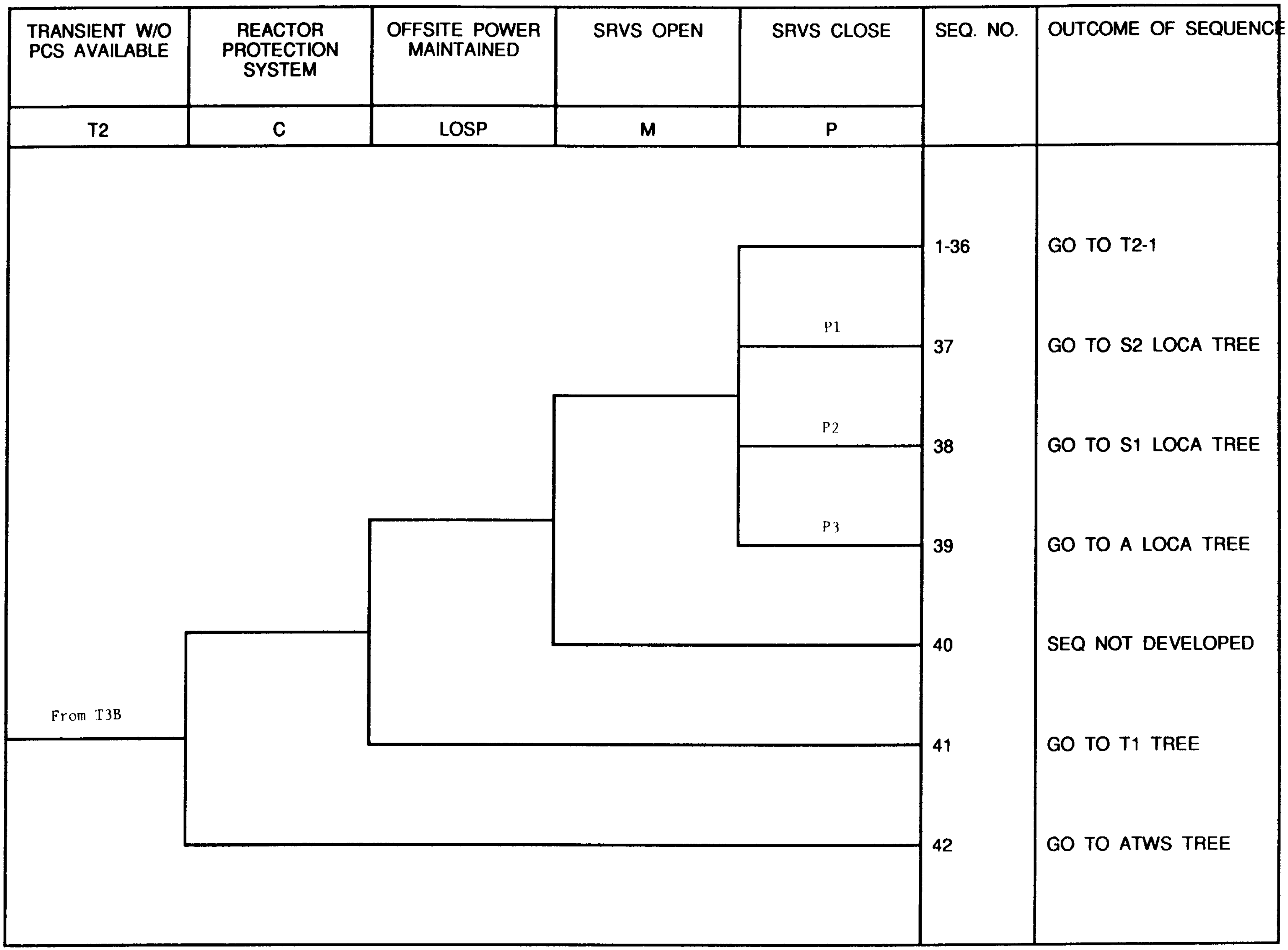


Figure 4.4-6. Transient Without PCS Initially Available Event Tree (Page 1 of 5)

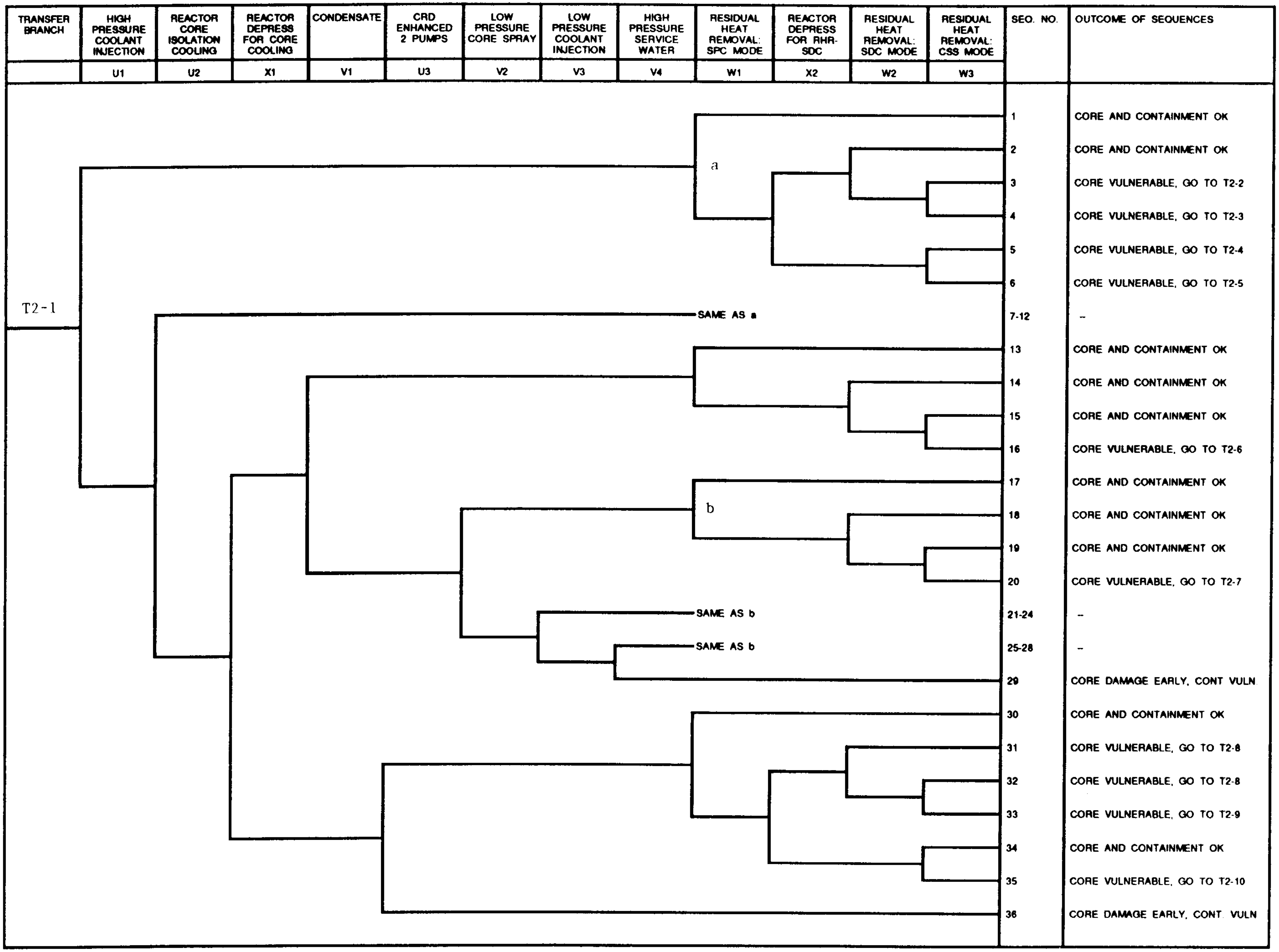


Figure 4.4-6. Transient Without PCS Initially Available Event Tree (Page 2 of 5)

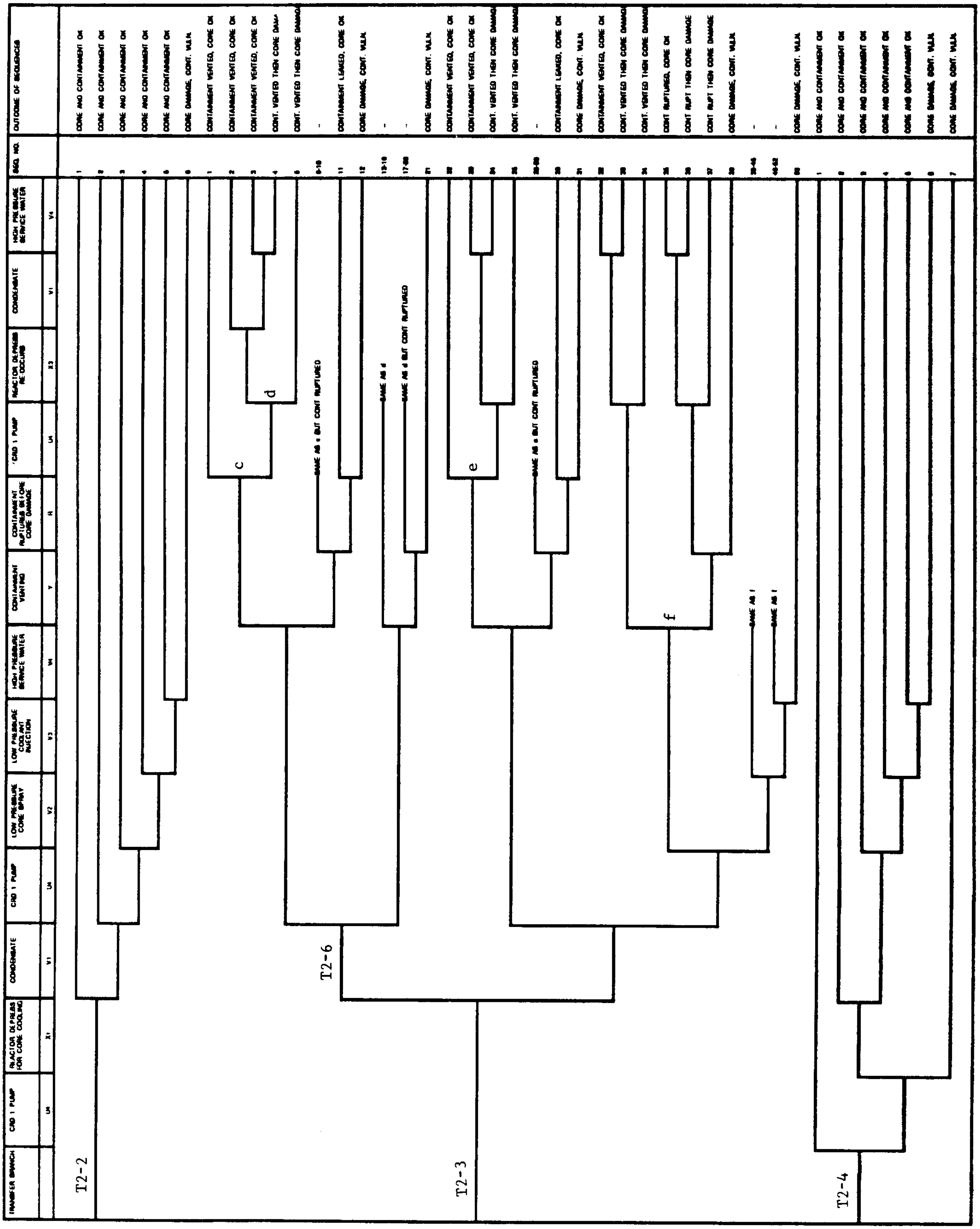


Figure 4.4-6. Transient Without PCS Initially Available Event Tree (Page 3 of 5)

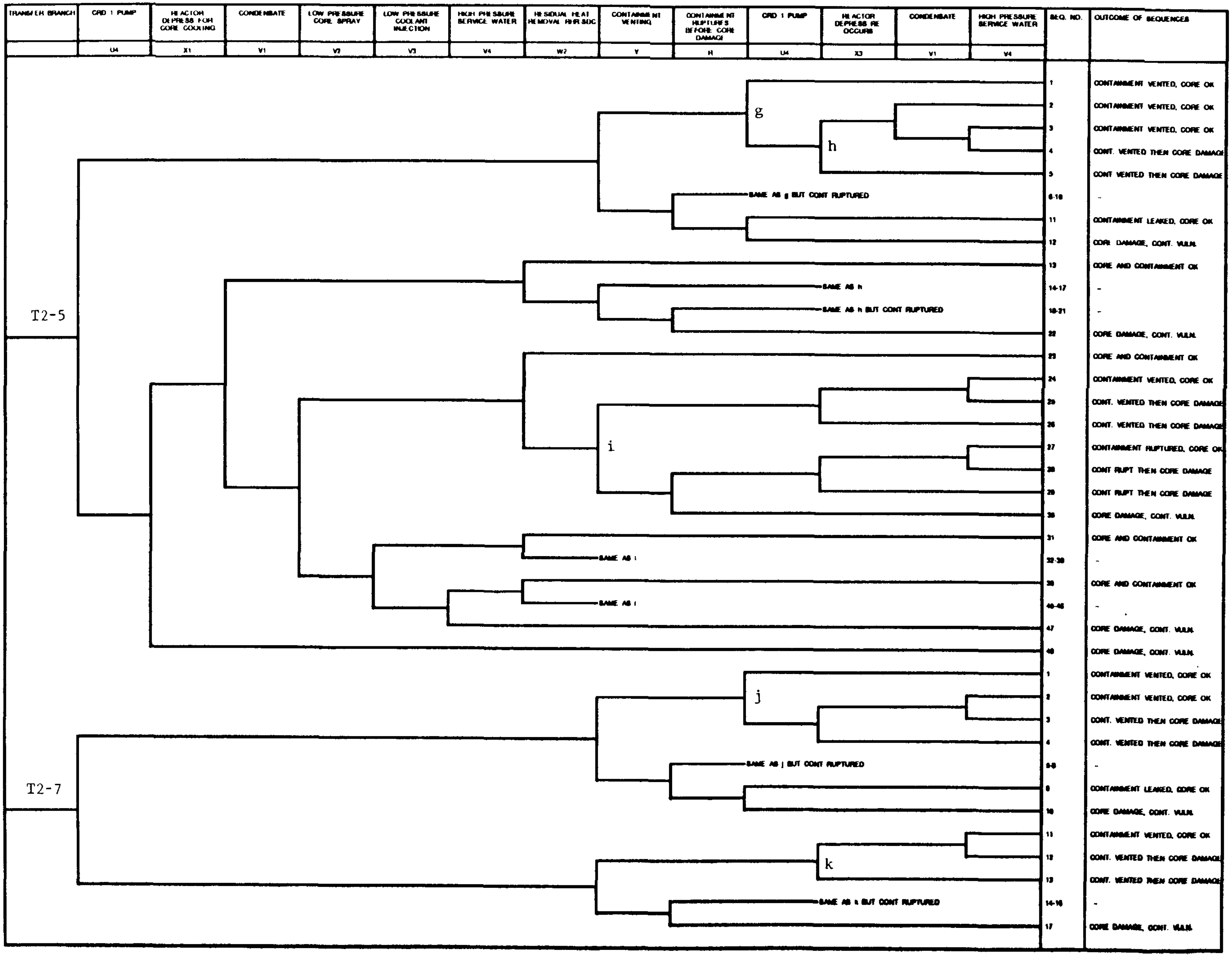


Figure 4.4-6. Transient Without PCS Initially Available Event Tree (Page 4 of 5)

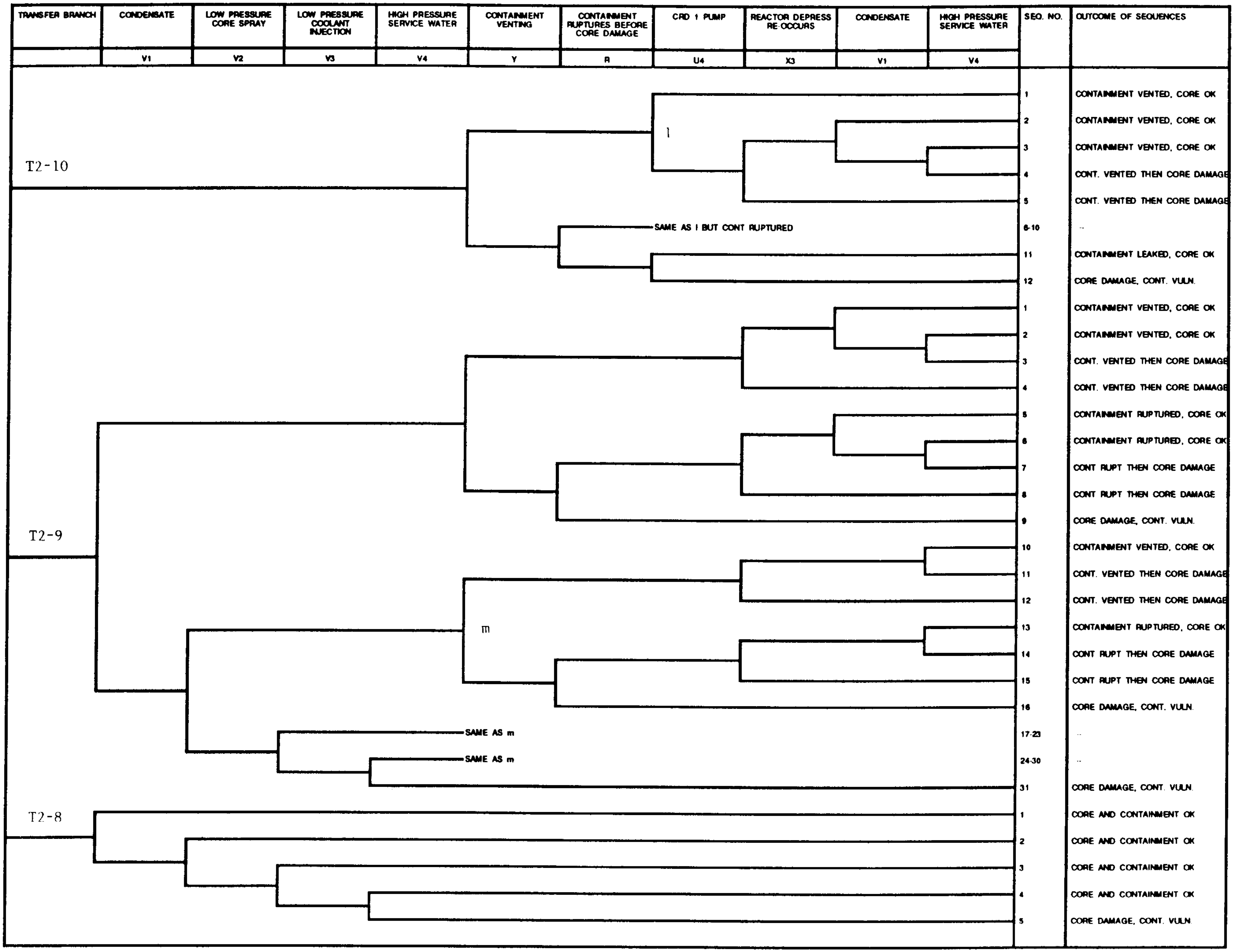


Figure 4.4-6. Transient Without PCS Initially Available Event Tree (Page 5 of 5)

SEQUENCE 4-12 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{V1}*\overline{U4}*Y*R*U4'$

Same as Sequence 4-11 except CRD fails (U4') following the leak in containment, leading to core damage in a vulnerable containment.

SEQUENCE 4-13 TO 4-16

Same as Sequences 4-2 to 4-5 except CRD fails to initiate (U4) following Condensate failure.

SEQUENCES 4-17 TO 4-20

Same as Sequences 4-13 to 4-16 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 4-21 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{V1}*U4*Y*R$

Same as Sequence 17 until the containment fails to rupture, which inhibits other low-pressure systems from operating, resulting in core damage in a vulnerable containment.

SEQUENCES 4-22 TO 4-23

Same as Sequence 4-1 until Condensate fails to initiate (V1) following containment overpressure protection failure. CRD provides core cooling (/U4) and eventually containment venting is necessary to relieve high containment pressure (/Y). CRD survives the venting event, or CRD fails and HPSW continues core cooling, resulting in a safe core in a vented containment.

SEQUENCES 4-24 TO 4-25

Same as Sequence 4-23 except the reactor fails to depressurize (X3) for HPSW, or HPSW fails to initiate (V4), resulting in core damage in a vented containment.

SEQUENCES 4-26 TO 4-29

Same as Sequences 4-22 to 4-25 except containment venting is unsuccessful (Y) and the containment eventually ruptures (/R).

SEQUENCE 4-30 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*V1*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 4-26 except the containment does not rupture (R) but develops a leak and CRD continues to provide core coolant (/U4').

SEQUENCE 4-31 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*V1*\overline{U4}*Y*R*U4'$

Same as Sequence 4-30 except CRD does not survive the containment leak (U4'), which leaves no system available for core coolant, resulting in core damage in a vulnerable containment.

SEQUENCE 4-32 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*V1*U4*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 4-22 until CRD does not initiate (U4) after Condensate failure and LPCS is initiated for core coolant (/V2). Containment venting is performed to relieve overpressure (/Y), which fails LPCS due to low NPSH. The reactor is depressurized again (/X3) and HPSW is initiated (/V4) to continue core cooling, resulting in a safe core in a vented containment.

SEQUENCES 4-33 TO 4-34

Same as Sequence 4-32 except HPSW fails (V4) or reactor depressurization prior to HPSW initiation fails (X3), resulting in core damage in a vented containment.

SEQUENCES 4-35 TO 4-37

Same as Sequences 4-32 to 4-34 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 4-38 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*V1*U4*\overline{V2}*Y*R$

Same as Sequence 4-37 until the containment fails to rupture (R), which forces the SRVs to close thus precluding the use of available core coolant systems, resulting in core damage in a vulnerable containment.

SEQUENCES 4-39 TO 4-45

Same as Sequences 4-32 to 4-38 except prior to containment venting, LPCI provides core coolant (/V3) following LPCS failure (V2).

SEQUENCES 4-46 TO 4-52

Same as Sequences 4-39 to 4-45 except prior to containment venting, HPSW provides core coolant (/V4) following LPCI failure (V3).

SEQUENCE 4-53 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*V1*U4*V2*V3*V4$

Same as Sequence 4-46 until HPSW fails (V4), which leaves no core coolant system available, resulting in core damage in a vulnerable containment.

SEQUENCES 5-1 TO 5-5

Same as Sequence 2 until depressurization for SDC fails (X2), followed by CSS initiation (/W3) for containment overpressure protection. HPCI fails prior to CSS initiation due to the adverse containment environment. CRD is initiated for core cooling (/U4), or, subsequent to CRD failure, the reactor is depressurized (/X1) and Condensate (/V1), LPCS (/V2), LPCI (/V3) or HPSW (/V4) continues core cooling, resulting in a safe core and containment.

SEQUENCES 5-6 TO 5-7

Same as Sequence 5-2 until reactor depressurization fails (X1) or all low pressure core coolant systems (Condensate, LPCS, LPCI, HPSW) fail to initiate, resulting in core damage in a vulnerable containment.

SEQUENCES 6-1 TO 6-3

Same as Sequence 1 until all containment overpressure protection is lost (SPC, reactor depressurization for SDC, and CSS). High suppression pool temperature fails HPCI (U1) and CRD is initiated for core coolant (/U4). High containment pressure is relieved by containment venting (/Y), and CRD (/U4), Condensate (/V1) or HPSW (/V4) continues core cooling, resulting in a safe core in a vented containment.

SEQUENCES 6-4 TO 6-5

Same as Sequence 6-2 except either reactor depressurization fails (X1) or Condensate (V1) and HPSW (V4) fail, which leaves no system available for core cooling, resulting in core damage in a vented containment.

SEQUENCES 6-6 TO 6-10

Same as Sequences 6-1 to 6-5 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 6-11 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 6-6 except the containment fails to rupture (R) but develops a leak. CRD survives venting (/U4'), resulting in a safe core in a leaking containment.

SEQUENCE 6-12 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 6-11 except CRD does not survive the containment leak (U4'), resulting in core damage in a vulnerable containment.

SEQUENCES 6-13 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V1}*\overline{W2}$

Same as Sequence 6-1 until CRD fails to initiate (U4) following loss of containment cooling. The reactor is depressurized (/X1) and Condensate is initiated for core coolant (/V1). Containment overpressure protection is established with SDC (W2), resulting in a safe core and containment.

SEQUENCES 6-14 TO 6-17

Same as Sequences 6-2 to 6-5 except CRD has failed (U4), the reactor is depressurized (/X1) and Condensate continues core cooling (/V1).

SEQUENCES 6-18 TO 6-21

Same as Sequences 6-14 to 6-17 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 6-22 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V1}*W2*Y*R$

Same as Sequence 6-13 until SDC fails (W2), followed by failure of containment venting (Y) and containment rupture (R), resulting in core damage in a vulnerable containment.

SEQUENCE 6-23 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V1*\overline{V2}*\overline{W2}$

Same as Sequence 6-13 except LPCS provides core cooling (/V2) following Condensate failure (/V1).

SEQUENCE 6-24 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V1*\overline{V2}*W2*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 6-23 except SDC fails to provide containment overpressure protection (W2) and containment venting is performed (/Y), followed by reactor depressurization (/X3) and HPSW initiation (/V4), resulting in a safe core in a vented containment.

SEQUENCES 6-25 to 6-26

Same as Sequence 6-24 except reactor depressurization prior to HPSW operation is unsuccessful (X3) or HPSW fails to initiate (V4), resulting in core damage in a vented containment.

SEQUENCES 6-27 TO 6-29

Same as Sequences 6-24 to 6-26 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 6-30 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V1*\overline{V2}*W2*Y*R$

Same as Sequence 6-27 until the containment fails to rupture (R), which leaves no system available for core cooling because of forced closure of the SRVs. This results in core damage in a vulnerable containment.

SEQUENCES 6-31 TO 6-38

Same as Sequences 6-23 to 6-30 except LPCI provides core coolant (/V3) following failure of LPCS to initiate (V2).

SEQUENCES 6-39 TO 6-46

Same as Sequences 6-31 to 6-38 except HPSW provides core coolant (/V4) following failure of LPCI to initiate (V3).

SEQUENCE 6-47 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V1*V2*V3*V4$

Same as Sequence 6-39 until HPSW fails (V4) and all core cooling is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 6-48 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*X1$

Same as Sequence 6-13 until depressurization following CRD failure is unsuccessful (X1), precluding the use of low pressure core coolant systems, resulting in core damage in a vulnerable containment.

SEQUENCES 7 TO 12

Same as Sequences 1 to 6 except RCIC provides early high pressure injection to the core (/U2) following failure of HPCI to initiate (U1).

SEQUENCES 13 TO 15

A transient occurs without the PCS available (T2) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the SRVs properly cycle to control reactor pressure (/M, /P). HPCI (U1) and RCIC (U2) fail to provide high pressure injection, the reactor is depressurized (/X1), and Condensate is initiated for core coolant (/V1). SPC (/W1), SDC (/W2) or CSS (/W3) provide containment overpressure protection, resulting in a safe core and containment.

SEQUENCES 16-1 TO 16-21

Same as Sequences 4-1 to 4-21 except, following failure of HPCI (U1) and RCIC (U2), Condensate provides early core coolant (/V1) prior to failure of containment overpressure protection.

SEQUENCES 17 TO 19

Same as Sequences 13 to 15 except LPCS provides early core coolant (/V2) following failure of Condensate (V1).

SEQUENCES 20-1 TO 20-2

Same as Sequence 17 until all containment overpressure protection fails (SPC, SDC, CSS), which causes increasing containment pressure, eventually closing the SRVs. The primary pressure subsequently increases which fails LPCS, and CRD is initiated to continue core cooling (/U4). Containment venting is performed to relieve high containment pressure (/Y), and CRD or HPSW continues to cool the core, resulting in a safe core in a vented containment.

SEQUENCES 20-3 TO 20-4

Same as Sequence 20-2 except HPSW fails to initiate (V4) or reactor depressurization prior to HPSW initiation fails (X3), resulting in core damage in a vented containment.

SEQUENCES 20-5 TO 20-8

Same as Sequences 20-1 to 20-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 20-9 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 20-5 except the containment fails to rupture and CRD survives (/U4'), resulting in a safe core in a leaking containment.

SEQUENCE 20-10 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*V1*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 20-9 except CRD does not continue core cooling (U4') following the development of a containment leak, resulting in core damage in a leaking containment.

SEQUENCES 20-11 TO 20-13

Same as Sequences 20-2 to 20-4 except CRD fails to initiate (U4) prior to the containment venting event.

SEQUENCES 20-14 TO 20-16

Same as Sequences 20-11 to 20-13 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 20-17

Same as Sequence 20-16 except the containment fails to rupture (R), resulting in core damage in a vulnerable containment.

SEQUENCES 21 TO 24

Same as Sequences 17 to 20 except LPCI provides early core coolant (/V3) following LPCS failure (V2).

SEQUENCES 25 TO 28

Same as Sequences 21 to 24 except HPSW provides early core coolant (/V4) following LPCI failure (V3).

SEQUENCE 29 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*V1*V2*V3*V4$

Same as Sequence 13 until all low pressure core coolant systems fail (Condensate, LPCS, LPCI, HPSW), which leaves no core coolant system available, resulting in early core damage in a vulnerable containment.

SEQUENCE 30 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*\overline{W1}$

Same as Sequence 13 until reactor depressurization fails (X1) and CRD is initiated in the enhanced mode (/U3) to provide sufficient cooling capacity. SPC is initiated for containment overpressure protection (/W1), resulting in a safe core and containment.

SEQUENCES 31-1 TO 31-4

Same as Sequence 30 until SPC fails (W1) and the reactor is depressurized (/X2) to initiate SDC (/W2). The decreased reactor pressure causes the CRD pump flow to increase, and, considering the CST level is decreasing, the CRD pumps are assumed to fail due to low NPSH. Condensate (/V1), LPCS (/V2), LPCI (/V3) or HPSW (/V4) provides core coolant, resulting in a safe core and containment.

SEQUENCE 31-5 -- $T2*\overline{C}*\overline{LOSP}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*V1*V2*V3*V4$

Same as Sequence 31-1 except all low pressure core coolant systems fail (Condensate, LPCS, LPCI, HPSW), resulting in core damage in a vulnerable containment.

SEQUENCES 32-1 TO 32-5

Same as Sequences 31-1 to 31-5 except SDC fails (W2) and CSS is initiated for containment overpressure protection (/W3).

SEQUENCES 33-1 TO 33-2

Same as Sequence 30 until all containment overpressure protection fails (SPC, SDC, CSS), although depressurization for SDC is successful. This depressurization increases the pump flow of CRD which, considering the CST level is continuously decreasing, is assumed to fail the CRD pumps due to low NPSH. Condensate is initiated to continue core cooling (/V1). High containment pressure is relieved by containment venting (/Y). The reactor is again depressurized (/X3) and Condensate (/V1) or HPSW (/V4) provides core coolant, resulting in a safe core in a vented containment.

SEQUENCES 33-3 TO 33-4

Same as Sequences 33-1 to 33-2 except HPSW fails (V4), or reactor depressurization prior to HPSW initiation fails (X3), resulting in core damage in a vented containment.

SEQUENCES 33-5 TO 33-8

Same as Sequences 33-1 to 33-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 33-9 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*\overline{V1}*Y*R$

Same as Sequence 33-5 until the containment fails to rupture (R), which leaves no coolant system operable, resulting in core damage in a vulnerable containment.

SEQUENCES 33-10 TO 33-16

Same as Sequences 33-1 to 33-9 except Condensate fails (V1) and LPCS provides core coolant (/V2) prior to the containment venting event, which results in two fewer sequences since no success path for Condensate exists subsequent to reactor depressurization (/X3).

SEQUENCES 33-17 TO 33-23

Same as Sequences 33-10 to 33-16 except following LPCS failure (V2), LPCI provides core coolant (/V3) prior to containment venting.

SEQUENCES 33-24 TO 33-30

Same as Sequences 33-17 to 33-23 except following LPCI failure (V3), HPSW provides core coolant (/V4) prior to containment venting.

SEQUENCES 33-31 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*V1*V2*V3*V4$

Same as Sequence 33-1 until Condensate fails (V1), followed by failure of LPCS (V2), LPCI(V3), and HPSW (V4), resulting in core damage in a vulnerable containment.

SEQUENCE 34 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*\overline{W3}$

Same as Sequence 30 until SPC fails (W1) to provide containment overpressure protection, followed by failure to depressurize the reactor (X2) for SDC. CSS is initiated (/W3) and CRD continues to function in the enhanced mode, resulting in a safe core and containment.

SEQUENCES 35-1 TO 35-3

Same as Sequence 34 until CSS fails (W3), after which all containment overpressure protection is lost, although CRD continues to provide core coolant. High containment pressure is relieved by containment venting (/Y), and CRD (/U4), Condensate (/V1), or HPSW (/V4) continues core cooling, resulting in a safe core in a vented containment.

SEQUENCES 35-4 TO 35-5

Same as Sequences 35-3 except HPSW fails (V4) or reactor depressurization prior to HPSW initiation fails (X3), which leaves all core coolant systems unavailable, resulting in core damage in a vented containment.

SEQUENCES 35-6 TO 35-10

Same as Sequences 35-1 to 35-5 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 35-11 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*\overline{U4}'$

Same as Sequence 35-6 except the containment does not rupture (R) and CRD continues in the 1 pump mode (/U4'), resulting in a safe core in a vulnerable containment.

SEQUENCE 35-12 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*U4'$

Same as Sequence 35-11 except CRD does not operate (U4') following the development of a containment leak, resulting in core damage in a vulnerable containment.

SEQUENCE 36 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*U3$

Same as Sequence 30 except CRD (2 pump mode) fails to initiate to provide core coolant (U3) following failure to depressurize the reactor (X1), which precludes the use of the low pressure core coolant systems, resulting in early core damage in a vulnerable containment.

SEQUENCE 37 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*P1$

A transient without the PCS available occurs (T2), which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the SRVs properly open to relieve the pressure (/M), but one SRV fails to close (P1) and the sequence is transferred to the S2 LOCA tree.

SEQUENCE 38 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*P2$

Same as Sequence 37 except two SRVs fail to close and the sequence is transferred to the S1 LOCA tree.

SEQUENCE 39 -- $T2*\overline{C}*\overline{LOSP}*\overline{M}*P3$

Same as Sequence 38 except three or more SRVs fail to close and the sequence is transferred to the A LOCA tree.

SEQUENCE 40 -- $T2*\overline{C}*\overline{LOSP}*M$

A transient occurs without PCS available (T2) which generates a reactor scram condition and the RPS successfully inserts the rods (/C). Offsite power is maintained (/LOSP). The SRVs fail to open to control reactor pressure (M) and the sequence is not developed further due to low probability.

SEQUENCE 41 -- $T2*\overline{C}*LOSP$

Same as Sequence 40 except offsite power is not maintained (LOSP) and the sequence is transferred to the T1 tree.

SEQUENCE 42 -- T2*C

Same as Sequence 40 except the RPS fails to scram the reactor, and the sequence is transferred to the ATWS tree.

4.4.9 Transient With PCS Initially Available Event Tree

This section contains information on the transient without the PCS initially available event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.9.1 Introduction

Transients in which the PCS remains initially available do not represent significant concerns for the plant unless the PCS is subsequently lost while the plant is being shut down. Should the PCS be lost, the sequence of events then proceeds similar to a transient in which the PCS was unavailable from the start. T3A represents all the transients of this type except Inadvertent Open Relief Valve (IORV) events and a loss of feedwater which can have somewhat different effects on plant conditions.

4.4.9.2 Event Tree

The T3A transient event tree is depicted by Figure 4.4-7. The following discussions define the event tree headings and the sequences.

The events in the tree include:

- <u>T3A</u>: Initiating event, transient with PCS initially available.
- <u>C</u>: Success or failure of Reactor Protection System (RPS). Success implies automatic scram by the control rods.
- <u>LOSP1</u>: Success or failure to maintain offsite power. The designation LOSP1 is used instead of LOSP for purposes of computational efficiency within the SETS code.
- <u>Q</u>: Continued success or subsequent failure of the PCS. Success implies continued operation of the PCS such that a safe cooldown of the plant is achieved using the PCS.
- <u>M</u>: Success or failure of Reactor Coolant System (RCS) overpressure protection (if required) by automatic operation of the SRVs. Success implies prevention of RCS overpressure so as to avoid damage to the primary system.
- <u>P</u>: Success or failure associated with reclosing of any SRVs which should open in response to reactor vessel pressure rises throughout the sequence. Success implies reclosure of all valves when vessel pressure drops below the closure setpoints. P1, P2 and P3 refer to the failure of one, two or three or more SRVs to reclose, respectively.

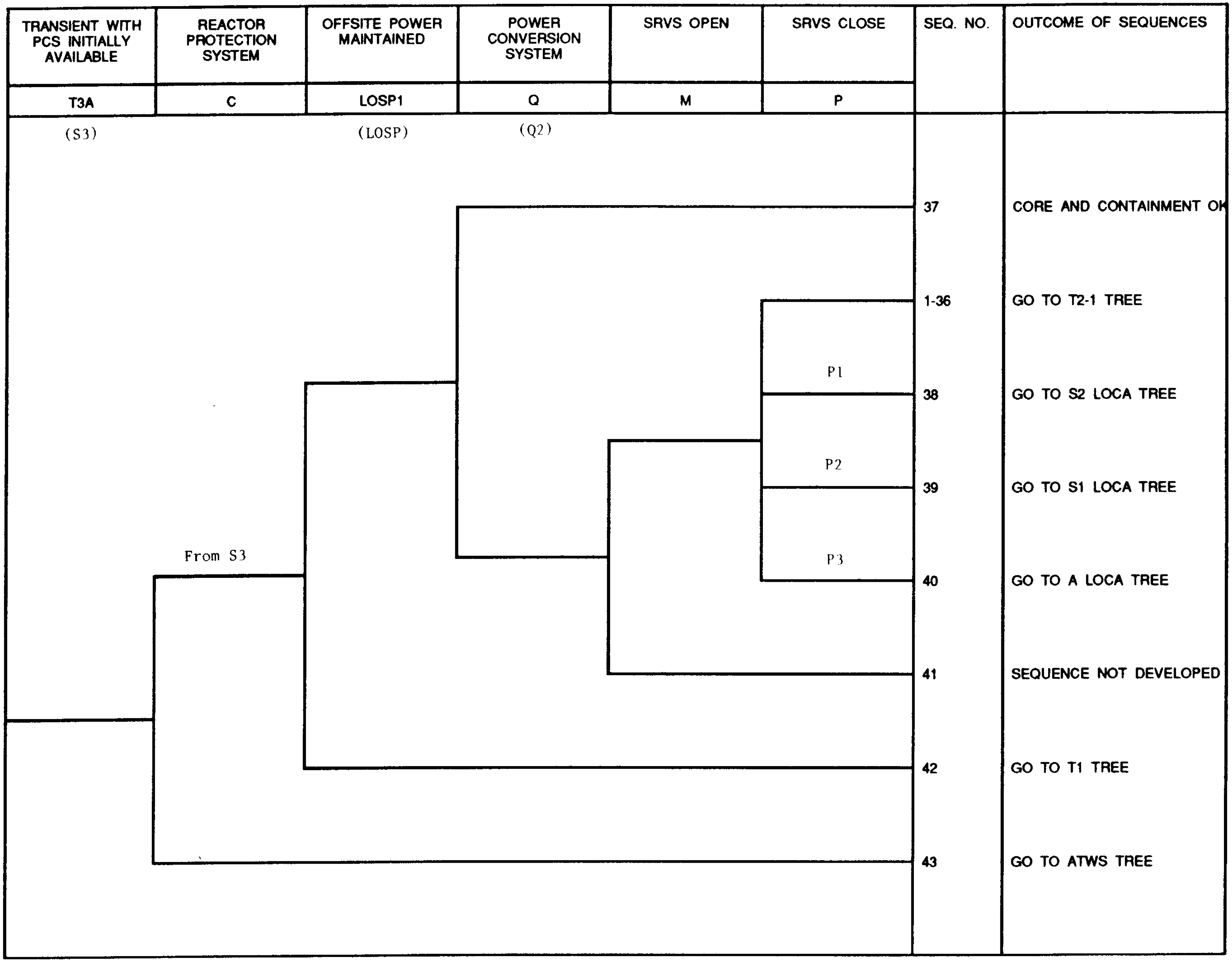


Figure 4.4-7. Transient With PCS Initially Available Event Tree

The following descriptions refer to the sequences found in Figure 4.4-7.

SEQUENCES 1 TO 36 -- $T3A*\overline{C}*\overline{LOSP}*Q*\overline{M}*\overline{P}$

A transient occurs with the PCS initially available (T3A) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP1). The PCS fails (Q) and the SRVs properly cycle to control reactor pressure (/M, /P). All sequences then transfer to the T2 tree at the T2-1 branch.

SEQUENCE 37 -- $T3A*\overline{C}*\overline{LOSP}*\overline{Q}$

Same as initial development of Sequences 1 to 36 except the PCS remains available (/Q), resulting in a safe core and containment.

SEQUENCE 38 -- $T3A*\overline{C}*\overline{LOSP}*Q*\overline{M}*P1$

Same as initial development of sequences 1 to 36 except one SRV fails to close (P1) and the sequence is transferred to the S2 LOCA tree.

SEQUENCE 39 -- $T3A*\overline{C}*\overline{LOSP}*Q*\overline{M}*P2$

Same as Sequence 38 except two SRVs fail to close (P2) and the sequence is transferred to the S1 LOCA tree.

SEQUENCE 40 -- $T2*\overline{C}*\overline{LOSP}*Q*\overline{M}*P3$

Same as Sequence 39 except three or more SRVs fail to close (P3) and the sequence is transferred to the A LOCA tree.

SEQUENCE 41 -- $T2*\overline{C}*\overline{LOSP}*Q*M$

Same as initial development of sequences 1 to 36 except the SRVs do not properly open to control reactor pressure (M) and the sequence is not developed further due to low probability.

SEQUENCE 42 -- $T2*\overline{C}*LOSP$

A transient occurs with the PCS initially available (T3A) and the RPS successfully scrams the reactor (/C). Offsite power is not maintained (LOSP) and the sequence is transferred to the T1 tree.

SEQUENCE 43 -- T2*C

A transient occurs with the PCS initially available (T3A), the RPS fails to successfully scram the reactor (C), and the sequence is transferred to the ATWS tree.

4.4.10 Loss of Feedwater Event Tree

This section contains information on the loss of feedwater event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.10.1 Introduction

A loss of feedwater event (T3B) is, in part, similar to a loss of PCS event except that only the feeder is definitely lost from the balance-of-plant. It is possible that the steam side of the PCS to the condenser may still be operable as well as the Condensate system. Coolant injection could be performed with systems such as HPCI, RCIC, or Condensate (as well as others) and heat removal might still be possible with the steam portion of the plant if condenser level and vacuum can be controlled. The success criteria would be as indicated for all T3-type transients already discussed.

To facilitate the analysis under the resource constraints of the study, the T3B event was conservatively analyzed as if the loss of feedwater event also included loss of the entire PCS as well as the Condensate system. Therefore, the T3B event was actually analyzed as a T2 transient which is described in Section 4.4.8.

While this "short-cut" is conservative, it was found at the conclusion of this study that this treatment of the T3B transient did not have a significant impact on the results.

4.4.10.2 Event Tree

The transfer tree for T3B is shown in Figure 4.4-8, since the event tree for T2 transients was conservatively used for the loss of feedwater initiator.

The following description refers to the sequence found in Figure 4.4-8.

SEQUENCE 1 -- T3B

A transient occurs in which feedwater is not available (T3B) and it is conservatively assumed that the entire PCS is lost and the sequence is transferred to the T2 tree.

4.4.11 Inadvertent Open Relief Valve Event Tree

This section contains information on the inadvertent open relief valve event tree. Success criteria considerations are presented along with the event tree and its description.

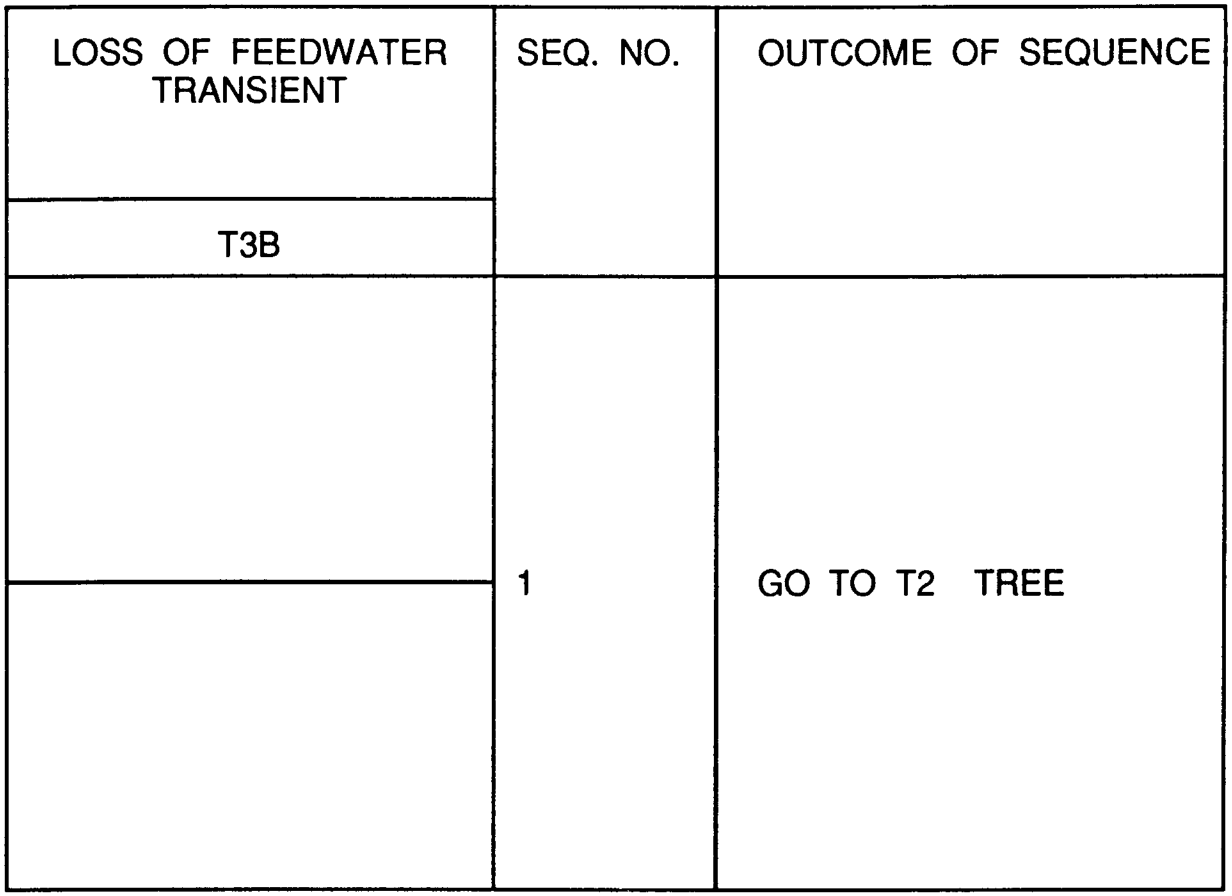


Figure 4.4-8. Loss of Feedwater Event Tree

4.4.11.1 Introduction

Should a primary system SRV inadvertently open during power operation, steam will be discharged to the suppression pool through the SRV tail pipe line. An open SRV will be easily detected by acoustical and temperature monitors on these lines. Procedures call for attempts to close the valve and, if unsuccessful, manually trip the plant and start shutdown procedures. Since the PCS is likely to be initially available, this event is categorized as another T3-type of transient (T3C).

It is separately analyzed since the open SRV will allow containment conditions to be at a somewhat higher stress level than other T3-type transients because of the initial steam release to the pool. It is, therefore, treated as a S2 steam LOCA and so is ultimately analyzed using the S2 success criteria (already described).

4.4.11.2 Event Tree

The T3C event tree is depicted by Figure 4.4-9. The following discussions define the event tree headings and the sequences.

The events in the tree include:

- T3C: Initiating event, inadvertent open relief valve transient.
- C1: Success or failure of reactor scram. Success implies manual trip of the reactor or automatic scram by the RPS.
- LOSP: Success or failure to maintain offsite power.
- Q1: Continued success or subsequent failure of the PCS. Success implies continued operation of the PCS such that cooldown of the plant is successfully achieved before containment conditions reach challenging levels from steam discharge from the stuck-open SRV.

The following descriptions refer to the sequences found in Figure 4.4-9.

SEQUENCE 1 -- $T3C*\overline{C1}*\overline{LOSP}*\overline{Q1}$

A relief valve inadvertently opens (T3C) which generates the need for a reactor scram which is performed manually or by the RPS (/C1). Offsite power is maintained (/LOSP) and the PCS functions properly to remove decay heat (/Q1) and the core and containment are safe.

SEQUENCE 2 -- $T3C*\overline{C1}*\overline{LOSP}*Q1$

Same as Sequence 1 except the PCS fails to remove decay heat (/Q1) and the sequence is transferred to the S2 LOCA tree.

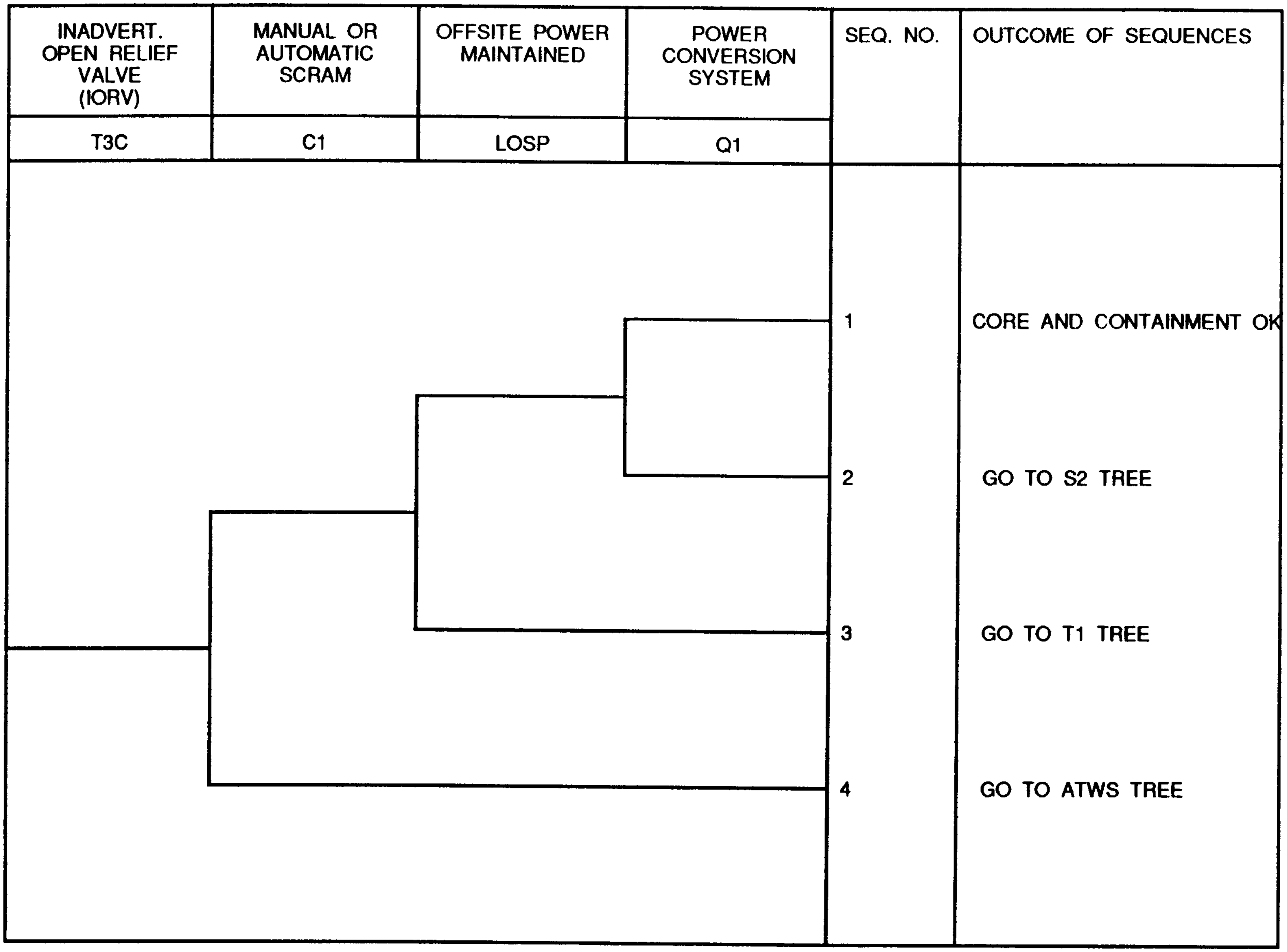


Figure 4.4-9. Inadvertent Open Relief Valve Event Tree

SEQUENCE 3 -- T3C*$\overline{C1}$*LOSP

Same as Sequence 2 except offsite power is not maintained and the sequence is transferred to the T1 tree.

SEQUENCE 4 -- T3C*C1

A relief valve inadvertently opens (T3C) and a manual or automatic scram is unsuccessful (C1) and the sequence is transferred to the ATWS tree.

4.4.12 Loss of an AC or DC Bus Event Tree

This section contains information on the loss of an AC or DC bus event tree. Success criteria considerations are presented along with the event tree and its description.

4.4.12.1 Introduction

A loss of an emergency AC or DC bus as an initiator was assumed to lead to a total loss of the PCS including the Condensate system.

4.4.12.2 Event Tree

The TAC/DC transient event tree is shown in Figure 4.4-10. The following discussions define the event tree headings and the sequences.

The events in the tree include:

TAC/DC: Initiating event, loss of an AC or DC bus.

C: Success or failure of the RPS. Success implies automatic scram by the control rods.

LOSP: Success or failure to maintain offsite power.

M: Success or failure of Reactor Coolant System (RCS) overpressure protection (if required) by automatic operation of the SRVs. Success implies prevention of RCS overpressure so as to avoid damage to the primary system.

P: Success or failure associated with reclosing of any SRVs which should open in response to reactor vessel pressure rises throughout the sequence. Success implies reclosure of all valves when vessel pressure drops below the closure setpoints. P1, P2 and P3 refer to the failure to reclose one, two or three or more SRVs, respectively.

U1: Success or failure of the HPCI system. Success implies operation of the HPCI pump train so as to maintain sufficient coolant injection.

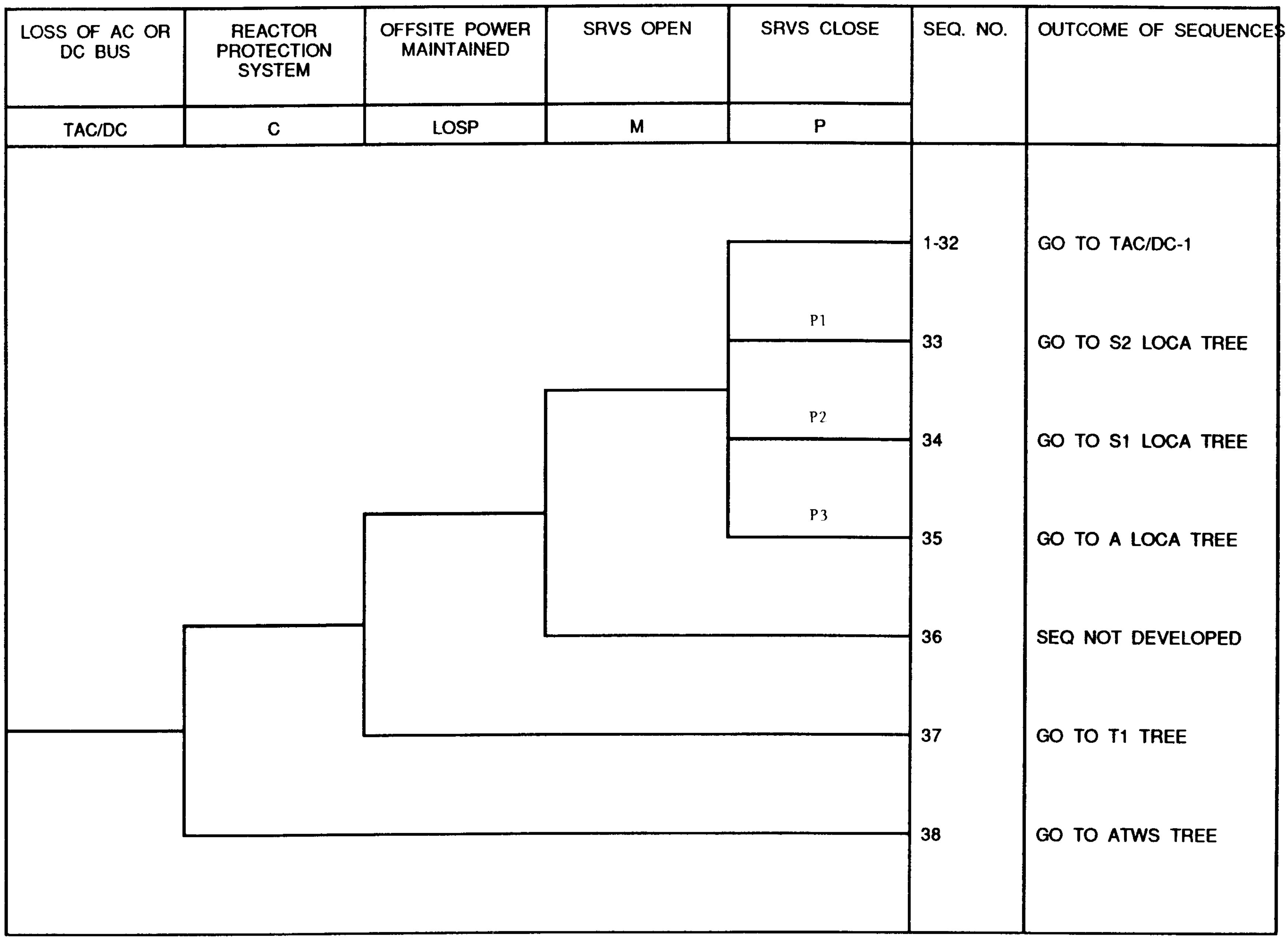


Figure 4.4-10. Loss of AC or DC Bus Event Tree (Page 1 of 5)

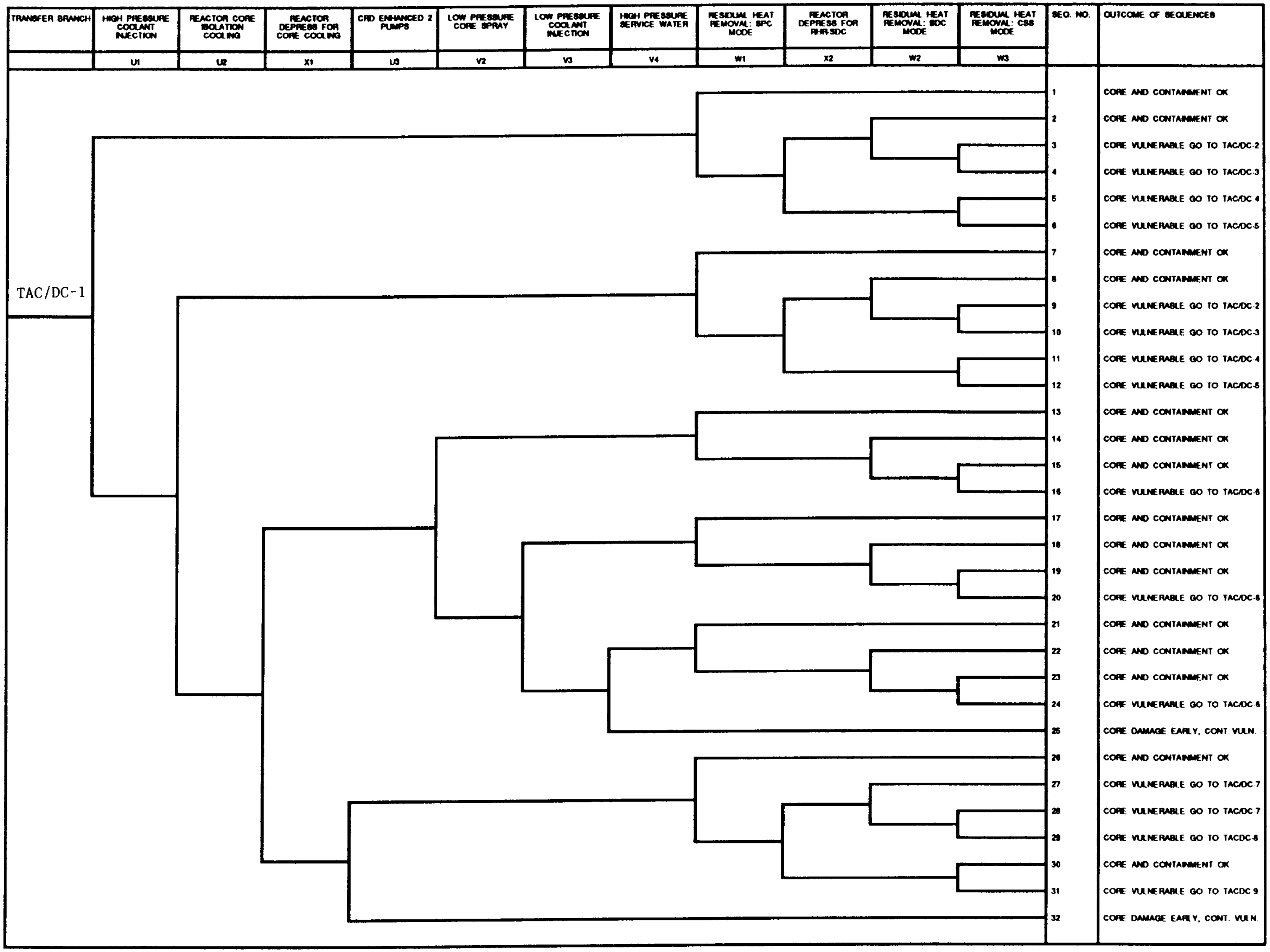


Figure 4.4-10. Loss of AC or DC Bus Event Tree (Page 2 of 5)

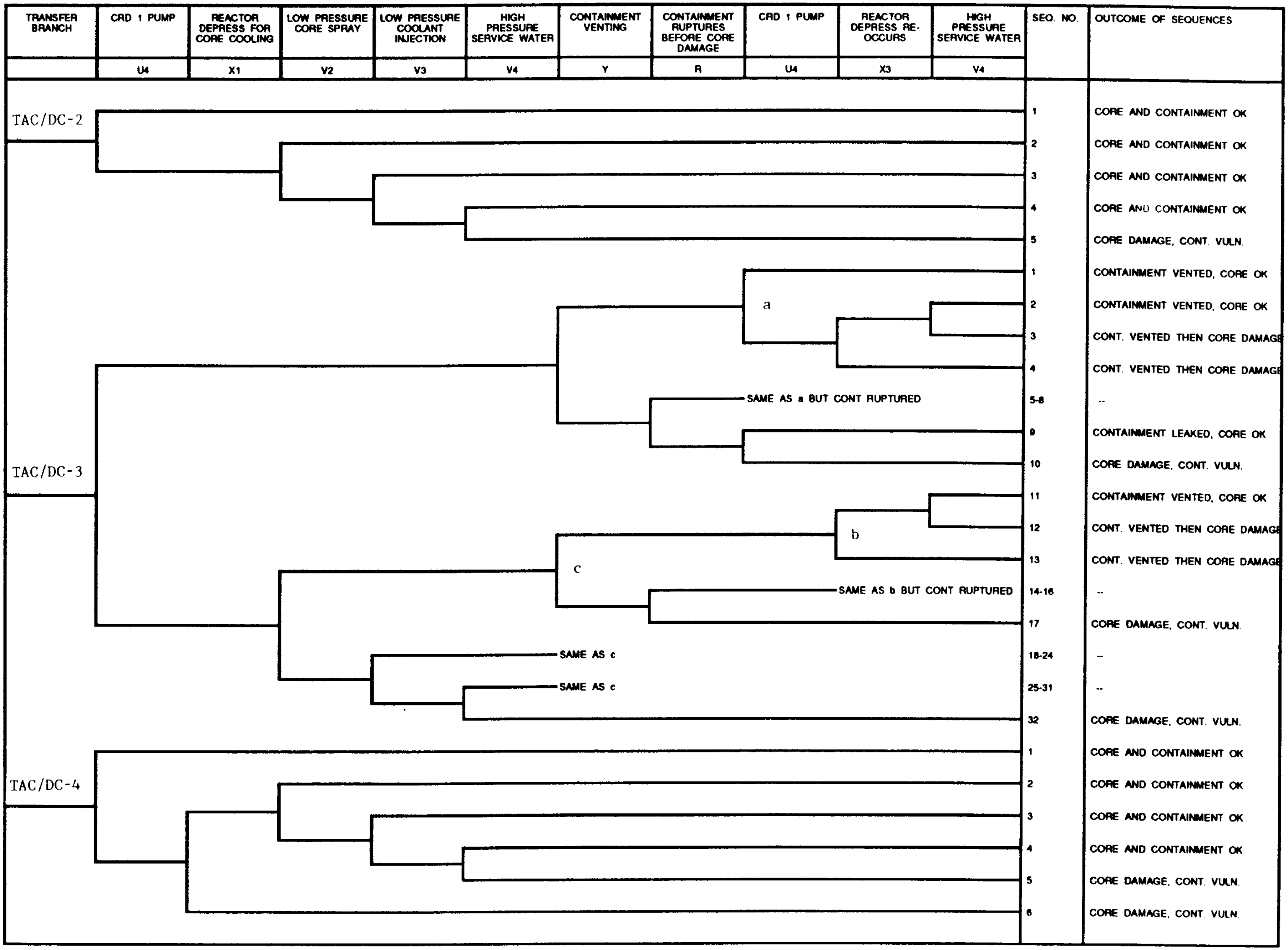


Figure 4.4-10. Loss of AC or DC Bus Event Tree (Page 3 of 5)

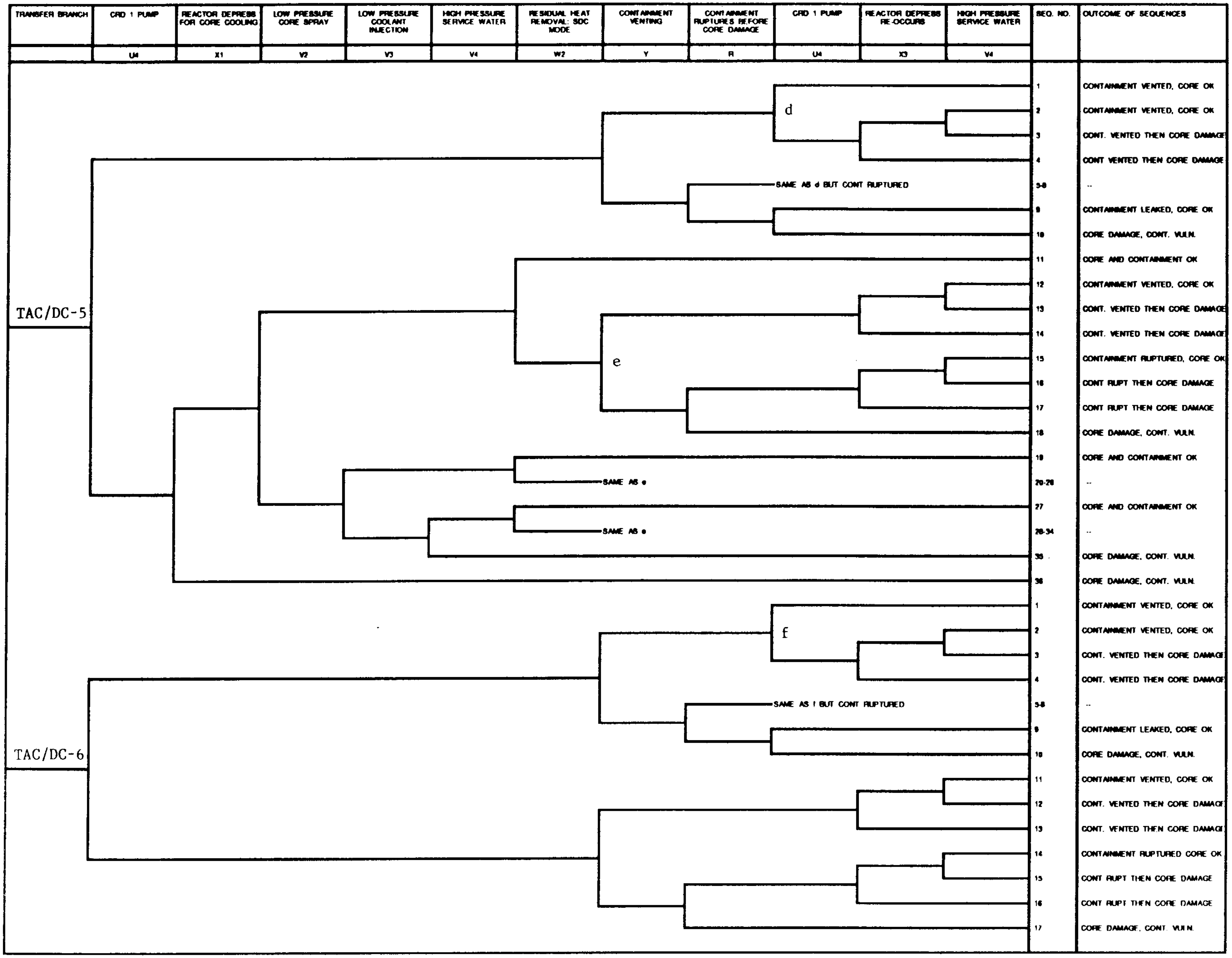


Figure 4.4-10. Loss of AC or DC Bus Event Tree (Page 4 of 5)

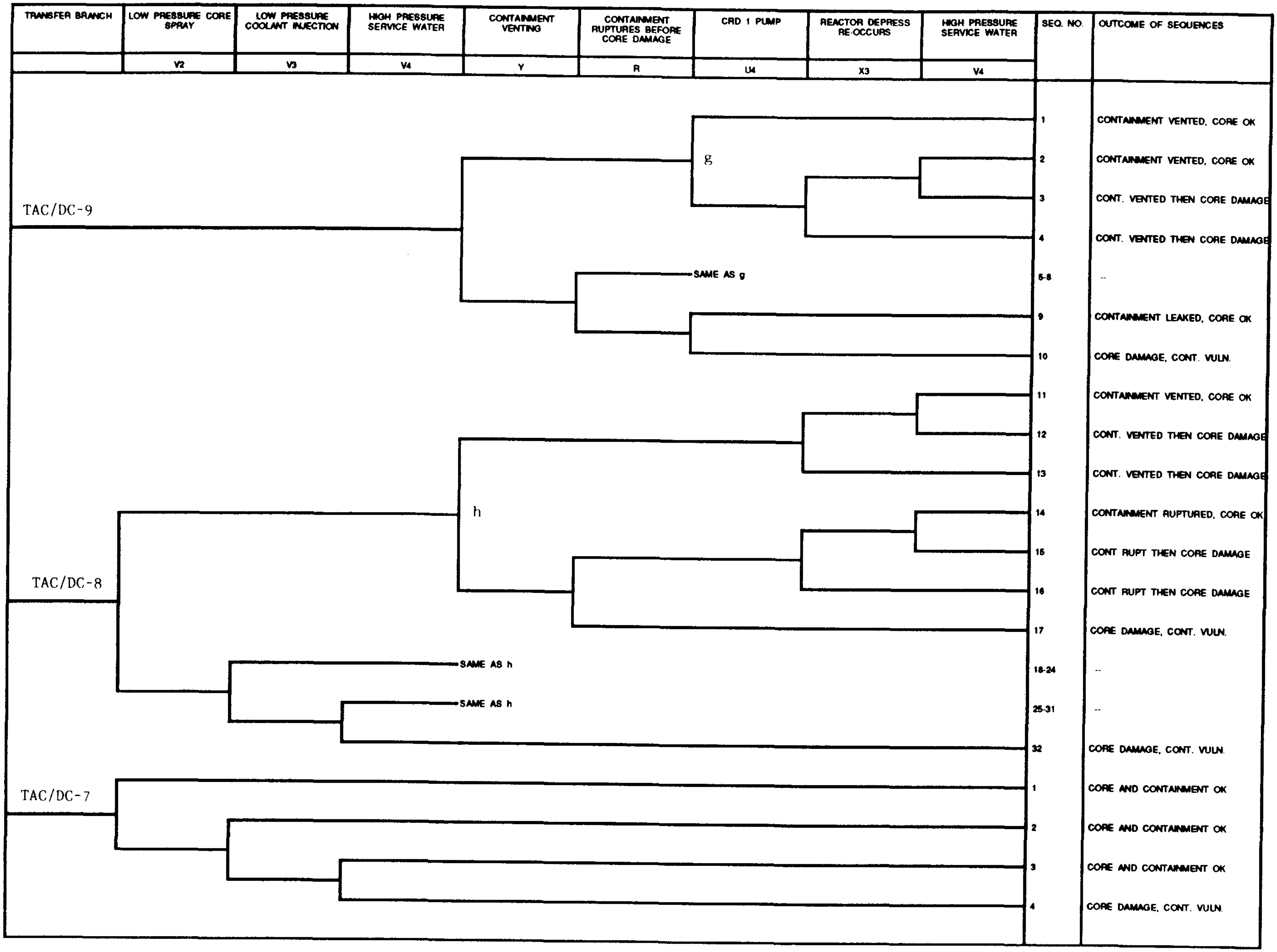


Figure 4.4-10. Loss of AC or DC Bus Event Tree (Page 5 of 5)

U2: Success or failure of the RCIC system. Success implies operation of the RCIC pump train so as to maintain sufficient coolant injection.

X: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS or manual operation of other SRVs such that three valves or more are opened allowing low pressure injection.

V1: Success or failure of the Condensate system. Success implies at least one pump operating with sufficient makeup to the condenser hotwell for a continuing water supply.

V2: Success or failure of the LPCS system. Success implies operation of any two of the four LPCS pumps through either or both LPCS injection lines.

V3: Success or failure of the LPCI mode of the RHR system. Success implies operation of one of four LPCI pumps through either LPCI injection line to the reactor vessel.

V4: Success or failure of the HPSW system in the inject mode to the reactor vessel through a LPCI injection line. Success implies manual operation of this injection source such that one HPSW pump successfully provides coolant to the reactor.

U3: Success or failure of the CRD system as an injection source. Success implies two pump operation.

W1,W2,W3: Success or failure of the RHR system in the SPC, SDC, or CS mode, respectively. Success implies at least one RHR pump operating in any one of the three modes with the appropriate heat exchanger in the loop along with the HPSW system in operation to the ultimate heat sink.

X2: Success or failure of primary system depressurization. Success implies automatic or manual operation of the ADS to allow the SDC mode of RHR to be initiated.

U4: Success or failure of the CRD system as an injection source. Success implies operation in the one pump mode.

Y: Success or failure of containment venting. Success implies that the six-inch integrated leak test line or larger size line is open so as to prevent containment by overpressure. As necessary, water makeup is also eventually supplied to the suppression pool.

R: Success or failure of the containment to withstand overpressurization. Success implies the containment ruptures before core damage.

X3: Success or failure of primary system depressurization. Success implies automatic or manual operation of ADS occurs subsequent to initial operation to allow low pressure injection.

The following descriptions refer to the sequences found in Figure 4.4-10.

SEQUENCE 1 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*\overline{W1}$

A loss of an AC or DC bus occurs (TAC/DC) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). The SRVs properly cycle to control reactor pressure (/M, /P) and onsite emergency power is established (/B). HPCI is initiated (/U1) for core cooling and SPC is initiated (/W1) for containment overpressure protection, resulting in a safe core and containment.

SEQUENCE 2 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*\overline{W2}$

Same as Sequence 1 but SPC fails to provide containment overpressure protection (W1) and SDC is initiated (/W2) following reactor depressurization (/X2).

SEQUENCES 3-1 TO 3-4

Same as Sequence 2 except SDC fails (W2) and CSS continues to protect the containment from overpressurization (/W3). HPCI fails due to the adverse containment environment and either CRD (/U4), LPCS (/V2), LPCI (/V3) or HPSW (/V4) continues core cooling.

SEQUENCE 3-5 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*\overline{W3}*U4*V2*V3*V4$

Same as Sequences 3-1 to 3-4 except CRD (U4), LPCS (V2), LPCI (V3) and HPSW (V4) fail, leaving no system available to cool the core, resulting in core damage in a vulnerable containment.

SEQUENCE 4-1 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*\overline{Y}*\overline{U4'}$

Same as Sequence 2 except SDC fails (W2), followed by CSS failure (W3), leaving the containment with no overpressure protection. HPCI eventually fails due to high suppression pool temperatures (U1) and CRD is initiated in the one pump mode (/U4). The containment is successfully vented (/Y) and CRD continues to provide core coolant (/U4'), resulting in no core damage in a vented containment.

SEQUENCE 4-2 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*\overline{V4}$

Same as Sequence 4-1 except CRD fails during containment venting (U4'). The reactor is depressurized (/X3) and HPSW provides core coolant (/V4).

SEQUENCES 4-3 TO 4-4

Same as Sequence 4-2 except HPSW fails (V4), or reactor depressurization prior to HPSW operation is unsuccessful (X3), resulting in core damage in a vented containment.

SEQUENCES 4-5 TO 4-8

Same as Sequences 4-1 to 4-4 except containment venting fails (Y) and the containment ruptures before core damage (/R).

SEQUENCE 4-9 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*Y*R*\overline{U4}'$

Same as Sequence 4-8 except the containment does not rupture (R) but develops a leak. CRD continues to operate (/U4'), resulting in no core damage in a leaking containment.

SEQUENCE 4-10 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 4-9 except CRD does not continue to operate (U4') following the containment leak which forces the SRVs closed and precludes low pressure cooling. This results in core damage in a vulnerable containment.

SEQUENCE 4-11 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*\overline{X2}*W2*W3*U4*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 4-1 except CRD does not operate (U4) following HPCI failure. LPCS is initiated (/V2) to continue core cooling and the containment is eventually vented (/Y). The LPCS pumps then fail due to low NPSH and the reactor is depressurized to allow HPSW to cool the core (/V4), resulting in a safe core in a vented containment.

SEQUENCES 4-12 TO 4-13

Same as Sequence 4-11 except HPSW fails (V4), or depressurization prior to HPSW operation fails (X3), resulting in core damage in a vented containment.

SEQUENCES 4-14 TO 4-16

Same as Sequences 4-11 to 4-13 except containment venting is unsuccessful (Y) and the containment ruptures (/R) before core damage.

SEQUENCE 4-17 -- $\mathrm{TAC/DC}*\overline{\mathrm{C}}*\overline{\mathrm{LOSP}}*\overline{\mathrm{M}}*\overline{\mathrm{P}}*\overline{\mathrm{U1}}*\mathrm{W1}*\overline{\mathrm{X2}}*\mathrm{W2}*\mathrm{W3}*\mathrm{U4}*\overline{\mathrm{V2}}*\mathrm{Y}*\mathrm{R}$

Same as Sequence 4-11 except containment venting fails (Y) and the containment does not rupture (R), resulting in core damage in a vulnerable containment.

SEQUENCES 4-18 TO 4-24

Same as Sequences 4-11 to 4-17 except, following LPCS failure (V2), LPCI provides core coolant (/V3) prior to containment venting.

SEQUENCES 4-25 TO 4-31

Same as Sequences 4-18 to 4-24 except, following LPCI failure (V3), HPSW provides core coolant (/V4) prior to containment venting.

SEQUENCE 4-32 -- $\mathrm{TAC/DC}*\overline{\mathrm{C}}*\overline{\mathrm{LOSP}}*\overline{\mathrm{M}}*\overline{\mathrm{P}}*\overline{\mathrm{U1}}*\mathrm{W1}*\overline{\mathrm{X2}}*\mathrm{W2}*\mathrm{W3}*\mathrm{U4}*\mathrm{V2}*\mathrm{V3}*\mathrm{V4}$

Same as Sequence 4-11 except LPCS (V2), LPCI (V3), and HPSW (V4) fail and all core cooling is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 5-1 -- $\mathrm{TAC/DC}*\overline{\mathrm{C}}*\overline{\mathrm{LOSP}}*\overline{\mathrm{M}}*\overline{\mathrm{P}}*\mathrm{U1}*\mathrm{W1}*\overline{\mathrm{X2}}*\overline{\mathrm{W3}}*\overline{\mathrm{U4}}$

Same as Sequence 2 except reactor depressurization for SDC is unsuccessful (X2) and CSS is initiated to provide containment overpressure protection (/W3). HPCI has failed due to high suppression pool temperatures and CRD (1 pump mode) is initiated to cool the core (/U4), resulting in a safe core and containment.

SEQUENCES 5-2 TO 5-4

Same as Sequence 5-1 except CRD fails to provide coolant injection (U4), the reactor is depressurized (/X1), and LPCS (/V2), LPCI (/V3) or HPSW (/V4) provide core cooling.

SEQUENCES 5-5 TO 5-6

Same as Sequence 5-2 except either reactor depressurization fails (X1) or LPCS (V2), LPCI (V3) and HPSW (V4) fail following depressurization, resulting in core damage in a vulnerable containment.

SEQUENCE 6-1 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*\overline{Y}*\overline{U4}'}$

Same as Sequence 5 except CSS fails (W3), resulting in the loss of all containment overpressure protection. High suppression pool temperatures fail HPCI and CRD (1 pump mode) is initiated for core coolant (/U4). Increasing containment pressure is relieved by containment venting (/Y). CRD survives venting (/U4') and the core is safe in a vented containment.

SEQUENCE 6-2 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*\overline{Y}*U4'*\overline{X3}*\overline{V4}}$

Same as Sequence 6-2 except CRD does not survive containment venting (U4'), the reactor is depressurized (/X1), and HPSW continues core cooling (/V4).

SEQUENCES 6-3 TO 6-4

Same as Sequence 6-2 except either reactor depressurization fails (X1), or HPSW fails (V4) following reactor depressurization, leading to core damage in a vented containment.

SEQUENCES 6-5 TO 6-8

Same as Sequences 6-1 to 6-4 except containment venting is unsuccessful (Y) and the containment ruptures (/R).

SEQUENCE 6-9 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*\overline{U4}'}$

Same as Sequence 6-5 except the containment does not rupture (R), but develops a leak. CRD continues coolant injection (/U4'), resulting in no core damage in a leaking containment.

SEQUENCE 6-10 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*\overline{U4}*Y*R*U4'}$

Same as Sequence 6-9 except CRD fails (U4') following the containment leak, at which point all coolant makeup is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 6-11 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*\overline{W2}}$

Same development as Sequence 6-1 until CRD fails to initiate (U4) following HPCI failure. The reactor is depressurized (/X1) to initiate LPCS for coolant injection (/V2). The reactor is sufficiently depressurized to initiate SDC for containment overpressure protection (/W2), resulting in a safe core and containment.

SEQUENCE 6-12 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*W2*\overline{Y}*\overline{X3}*\overline{V4}}$

Same as Sequence 6-11 except SDC fails to provide containment overpressure protection (W2), followed by successful venting of the containment (/Y). Coolant injection is restored using HPSW (/V4) following reactor depressurization (/X3), resulting in a safe core in a vented containment.

SEQUENCES 6-13 TO 6-14

Same as Sequence 6-12 except either reactor depressurization fails (X3) or HPSW fails (V4) following reactor depressurization, resulting in core damage in a vented containment.

SEQUENCES 6-15 TO 6-17

Same as Sequences 6-12 to 6-14 except containment venting fails (Y) and the containment ruptures (/R).

SEQUENCE 6-18 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*\overline{V2}*W2*Y*R}$

Same as Sequence 6-11 until containment overpressure protection with SDC fails (W2), followed by failure of containment venting (Y). The containment does not rupture (R), and core damage results in a vulnerable containment.

SEQUENCES 6-19 TO 6-26

Same as Sequences 6-11 to 6-18 except LPCI provides coolant makeup (/V3) following failure of LPCS (V2).

SEQUENCES 6-27 TO 6-34

Same as Sequences 6-19 to 6-26 except HPSW provides coolant makeup (/V4) following failure of LPCI (V2).

SEQUENCE 6-35 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*\overline{X1}*V2*V3*V4}$

Same as Sequence 6-11 until LPCS fails (V2) following reactor depressurization, followed by failure of both LPCI (V3) and HPSW (V4), at which point all coolant makeup is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 6-36 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*\overline{U1}*W1*X2*W3*U4*X1}$

Same as Sequence 6-11 until CRD fails to continue coolant makeup (U4) following HPCI failure. Reactor depressurization fails (X1), which disables all low-pressure core cooling systems, resulting in core damage in a vulnerable containment.

SEQUENCES 7 TO 12

Same as Sequences 1 to 6 except RCIC provides high pressure coolant makeup (/U2) following failure to initiate HPCI (U1).

SEQUENCE 13-15

Same as Sequence 1 until failure to initiate HPCI (U1), followed by failure of RCIC (U2). The reactor is depressurized (/X1) and LPCS is initiated for coolant makeup (/V2). Containment overpressure protection is provided by SPC (/W1), SDC (/W2), or CSS (/W3), resulting in a safe core and containment.

SEQUENCES 16-1 TO 16-2

Same as Sequence 13 until SPC fails (W1), followed by failure of SDC (W2) and CSS (W3). Without containment overpressure protection, the pressure in containment rises until the SRVs close. Primary system pressure then rises, eventually failing LPCS (V2). CRD is initiated (/U4) for coolant makeup. High containment pressure is relieved by containment venting (/Y). CRD continues to cool the core, or the reactor is depressurized (/X1) and HPSW cools the core (/V4) if CRD does not survive the venting.

SEQUENCES 16-3 TO 16-4

Same as Sequence 16-1 except CRD does not survive containment venting and either reactor depressurization is unsuccessful (X1), or HPSW fails (V4) following reactor depressurization, resulting in core damage in a vented containment.

SEQUENCES 16-5 TO 16-8

Same as Sequences 16-1 to 16-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 16-9 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*\overline{U4'}$

Same as Sequence 16-5 except the containment does not rupture (R) but develops a leak. CRD survives (/U4') resulting in a safe core in a leaking containment.

SEQUENCE 16-10 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*\overline{U4}*Y*R*U4'$

Same as Sequence 16-9 except CRD does not survive the development of a leak in containment (U4'), all coolant systems are lost, and core damage results in a vulnerable containment.

SEQUENCE 16-11 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*U4*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 16-1 until CRD fails to initiate (U4) following loss of containment overpressure protection. Increasing containment pressure is relieved by containment venting (/Y) and HPSW is initiated to cool the core (/V4) following primary system depressurization (/X1). The core is safe in a vented containment.

SEQUENCES 16-12 TO 16-13

Same as Sequence 16-11 except either HPSW fails to cool the core (V4) or primary system depressurization fails (X1) prior to HPSW operation, resulting in core damage in a vented containment.

SEQUENCES 16-14 TO 16-16

Same as Sequences 16-11 to 16-13 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 16-17 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*\overline{V2}*W1*W2*W3*U4*Y*R$

Same as Sequence 16-11 until containment venting fails (Y). The containment does not rupture (R) and continues to pressurize, resulting in core damage in a vulnerable containment since the SRVs are forced closed preventing low pressure cooling.

SEQUENCES 17 TO 20

Same as Sequences 13 to 15 except LPCI provides early core coolant (/V3) following LPCS failure (V2).

SEQUENCES 21 TO 24

Same as Sequences 17 to 20 except HPSW provides early core coolant (/V4) following LPCI failure (V3).

SEQUENCE 25 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*\overline{X1}*V2*V3*V4$

Same as Sequence 21 until HPSW fails (V4), at which point all coolant makeup is lost, resulting in early core damage in a vulnerable containment.

SEQUENCE 26 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*\overline{W1}$

Same as Sequence 13 until reactor depressurization fails (X1) following failure to initiate high-pressure coolant systems. CRD is initiated in the two-pump mode to provide sufficient injection capacity (/U3). Containment overpressure protection is provided by SPC (/W1), resulting in a safe core and containment.

SEQUENCES 27-1 TO 27-3

Same as Sequence 26 until SPC fails to provide containment overpressure protection (W1), the reactor is depressurized (/X2), and SDC is initiated (/W2). Reactor depressurization for SDC increases CRD flow rate which, when considering CST inventory is depleting, is assumed to fail the CRD pumps due to low NPSH. LPCS (/V2), LPCI (/V3) or HPSW (/V4) is initiated for core coolant, resulting in a safe core and containment.

SEQUENCE 27-4 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*\overline{W2}*V2*V3*V4$

Same as Sequence 27-1 until LPCS fails (V2) to initiate after CRD fails, followed by unsuccessful operation of LPCI (V3) and HPSW (V4), resulting in core damage in a vulnerable containment.

SEQUENCES 28-1 TO 28-4

Same as Sequences 27-1 to 27-4 except CSS provides containment overpressure protection (/W3) following SDC failure (W2).

SEQUENCE 29-1 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*\overline{V2}*\overline{Y}*\overline{X3}*\overline{V4}$

Same as Sequence 28-1 until CSS fails to initiate (W3), at which point all containment overpressure protection is lost. CRD failed due to reactor depressurization for SDC, so LPCS is initiated (/V2) to continue core cooling. Containment venting (/Y) is successful to relieve containment overpressurization, which fails LPCS due to low NPSH. The reactor is again depressurized (/X3) and HPSW cools the core, resulting in a safe core in a vented containment.

SEQUENCES 29-2 TO 29-3

Same as Sequence 29-1 except either HPSW fails (V4) or reactor depressurization fails (X3) prior to HPSW operation, leaving no system available for coolant makeup, resulting in core damage in a vented containment.

SEQUENCES 29-4 TO 29-6

Same as Sequences 29-1 to 29-3 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 29-7 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*\overline{V2}*Y*R$

Same as Sequence 29-4 until the containment fails to rupture (R), which precludes HPSW operation because of forced closure of the SRVs. This results in core damage in a vulnerable containment.

SEQUENCES 29-8 TO 29-14

Same as Sequences 29-1 to 29-7 except LPCS fails to initiate (V2) following containment cooling failure and LPCI provides coolant makeup (/V3).

SEQUENCES 29-15 TO 29-21

Same as Sequences 29-8 to 29-14 except LPCI fails to initiate (V3) following containment cooling failure and HPSW provides coolant makeup (/V4).

SEQUENCE 29-22 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*\overline{X2}*W2*W3*V2*V3*V4$

Same as Sequence 29-11 until LPCS fails (V2) following containment cooling failure. LPCI (V3) and HPSW (V4) also fail to initiate, resulting in core damage in a vulnerable containment.

SEQUENCE 30 -- $TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*\overline{W3}$

Same as Sequence 26 until SPC fails (W1), followed by failure of reactor depressurization for SDC (X2). CSS is initiated to provide containment overpressure protection (/W3). Since reactor depressurization was unsuccessful, CRD does not fail, resulting in a safe core and containment.

SEQUENCES 31-1 TO 31-2

Same as Sequence 30 until CSS fails (W3), at which point all containment overpressure protection is lost. Eventually containment venting is performed to relieve containment overpressure (/Y). CRD continues to cool the core in the one-pump mode (/U4), or CRD fails on containment venting and HPSW cools the core (/V4), resulting in a safe core in a vented containment.

SEQUENCES 31-3 TO 31-4

Same as Sequence 31-2 except HPSW fails (V4) or reactor depressurization fails prior to HPSW operation (X3), resulting in core damage in a vented containment.

SEQUENCES 31-5 TO 31-8

Same as Sequences 31-1 to 31-4 except containment venting fails (Y) and the containment eventually ruptures (/R).

SEQUENCE 31-9 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*\overline{U4}}$

Same as Sequence 31-5 except the containment does not rupture (R) but develops a leak. CRD continues to cool the core, resulting in a safe core in a leaked containment.

SEQUENCE 31-10 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*\overline{U3}*W1*X2*W3*Y*R*U4}$

Same as Sequence 31-9 except CRD does not survive the containment leak (U4), resulting in core damage in a vulnerable containment.

SEQUENCE 32 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*\overline{P}*U1*U2*X1*U3}$

Same as Sequence 26 until CRD fails to initiate (U3) in the two-pump mode following failure to depressurize the reactor, which leaves no system available for coolant makeup. Early core damage results in a vulnerable containment.

SEQUENCE 33 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*P1}$

A loss of an AC or DC bus occurs (TAC/DC) which generates a reactor scram condition and the RPS successfully inserts the rods into the core (/C). Offsite power is maintained (/LOSP) and the SRVs open to control the pressure (/M), but one SRV fails to close (P1) and the sequence is transferred to the S2 LOCA tree.

SEQUENCE 34 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*P2}$

Same as Sequence 33 except two SRVs fail to close (P2) and the sequence is transferred to the S1 LOCA tree.

SEQUENCE 35 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*\overline{M}*P3}$

Same as Sequence 33 except three or more SRVs fail to close (P3) and the sequence is transferred to the A LOCA tree.

SEQUENCE 36 -- $\mathrm{TAC/DC*\overline{C}*\overline{LOSP}*M}$

Same as Sequence 33 except the SRVs fail to open to control reactor pressure (M) and the sequence is not developed further due to low probability.

SEQUENCE 37 -- TAC/DC*$\overline{C}$*LOSP

A loss of an AC or DC bus occurs (TAC/DC) and the RPS successfully scrams the reactor (/C). Offsite power is not maintained (LOSP) and the sequence is transferred to the T1 tree.

SEQUENCE 38 -- TAC/DC*C

A loss of an AC or DC bus occurs (TAC/DC) and the RPS fails to scram the reactor (C) and the sequence is transferred to the ATWS tree.

4.4.13 "V" (Interfacing LOCA) Sequence

This type of a scenario typically involves the failure of a high-to-low pressure interface such that reactor pressure causes failure within a low-pressure system. This could possibly create an unmitigatable LOCA (worst case) with a fission product release path through the low-pressure system, thereby bypassing the suppression pool and containment.

References 12 and 49 suggest that, on the basis of precursor events, such a failure is most likely to occur while performing stroke valve testing of isolation valves during power operation. In Reference 49, the U.S. Nuclear Regulatory Commission performed an analysis in which it was estimated, based on precursor events, that the frequency of inadvertent pressurization of a low-pressure line is approximately 1E-2/year. Additional years experience since that analysis suggest this estimate should now be approximately 5E-3/year. In that analysis, it is judged that, given an inadvertent pressurization the probability of a significant open pathway, such as a pipe break, occurring so as to potentially cause core damage is 1E-2 to 1E-3. In Reference 50, the BWR Owner's Group provides a detailed analysis of a pipe rupture probability and estimates it at 3E-5. These results yield frequencies of a significant open pathway from the reactor vessel through a low-pressure system of ~E-5 to E-7. While such a pathway would fail the low-pressure system involved, four other factors must be considered in order to arrive at a core damage frequency from such an occurrence. First, Peach Bottom's emergency core cooling systems (ECCS) designs are highly compartmentalized in flood-proof rooms. This means that other ECCS would likely be available to makeup cooling to the core. Secondly, Condensate would likely still be available in such a sequence since the majority of the equipment is outside the reactor building and hence not subject to any adverse environment caused by the scenario. Third, High-Pressure Service Water (HPSW) may still be available to use for coolant makeup. Last, operation of the Safety Relief Valves (SRVs) to depressurize the reactor (thus slowing the leak rate) and reclosing of the necessary valves to stop the leak (the valves are typically located in their own rooms high above the pump rooms) are likely to occur since the operator would receive numerous alarms when the leak occurs. With all these mitigative features, the core damage frequency resulting from a "V" scenario is estimated to be at or below E-8.

Since it is not apparent that the precursor events reported in Reference 49 are applicable for Peach Bottom, depending on design and operational differences among plants, a separate analysis was performed for this study as reported below. Review of the piping interfaces with the primary system showed that the two LPCS injection lines and two LPCI injection lines were possible areas where the "V" sequence, as described, might occur. Testing procedures were reviewed. In each case, because of the equipment configuration and testing procedures, it was found that two hardware failures and two human errors would have to occur to initiate the "V" sequence during testing (refer to Figure 4.4-11 later for typical arrangements).

First, the testable check valve must leak or rupture and go undetected. Since the MOVs are stroke tested at least quarterly, and using 8E-7/hr (mean) and 2.7E-8/hr (mean) based on WASH-1400 data for leak and rupture failure rates [4] of the testable check valve, the probability that there has been a failure of the check valve between tests is ~9E-4 (mean value using $1/2\lambda t$ where λ are the rates above and t is equal to 3 months). Note that if the valve were to fail, detection is likely since Peach Bottom has disc position indication for such valves. The operator must then have failed to reclose the normally open MOV used to maintain the high-low pressure interface during the test.

Using ASEP's nominal Human Reliability Analysis (HRA) value of 0.02 for failure of a step-by-step task performed under moderate stress [25], and further reducing it by a factor of at least five (i.e., using the suggested lower bound value) to account for the clarity in the procedure and the nonstress situation, yields an operator failure probability to close the MOV of 4E-3 (mean). Following procedures, the operator is to open the bypass valve and then pressurize the line segment using the air test connection to near reactor pressure before opening the MOV being stroke-tested. Such a process would be virtually impossible if the previously mentioned MOV had not been closed to hold the pressure. Otherwise, pressure could not be maintained and the relief valve would lift before the pressure in the line could reach high pressure. Therefore, a nonrecovery probability is applied to failure to close the normally open MOV. This probability must be very small; estimated at 1E-4 to account for a possible plug in the line such that the operator could still pressurize the line segment. Then, an interlock exists between the normally open MOV and the MOV to be stroked such that both valves cannot be open at the same time. Failure of this interlock would have to occur and is estimated at 2.5E-2 based on possible limit switch failure (2.4E-2 per Indian Point study data [20]) or failure of the circuitry (1E-3 per ASEP generic data). Combining all these failures leads to a very small probability for the "V" sequence's occurring in this way (<<1E-8 per year).

Other lines were examined, such as the RHR shutdown cooling path and HPCI and RCIC lines. In such cases, these paths also appeared to offer low chances for the "V" scenario, considering similar interlock failure requirements or, in the case of HPCI and RCIC, the fact that an additional feedwater check valve would have to fail and that high-pressure piping exists for much of the system. In addition, these rooms are normally

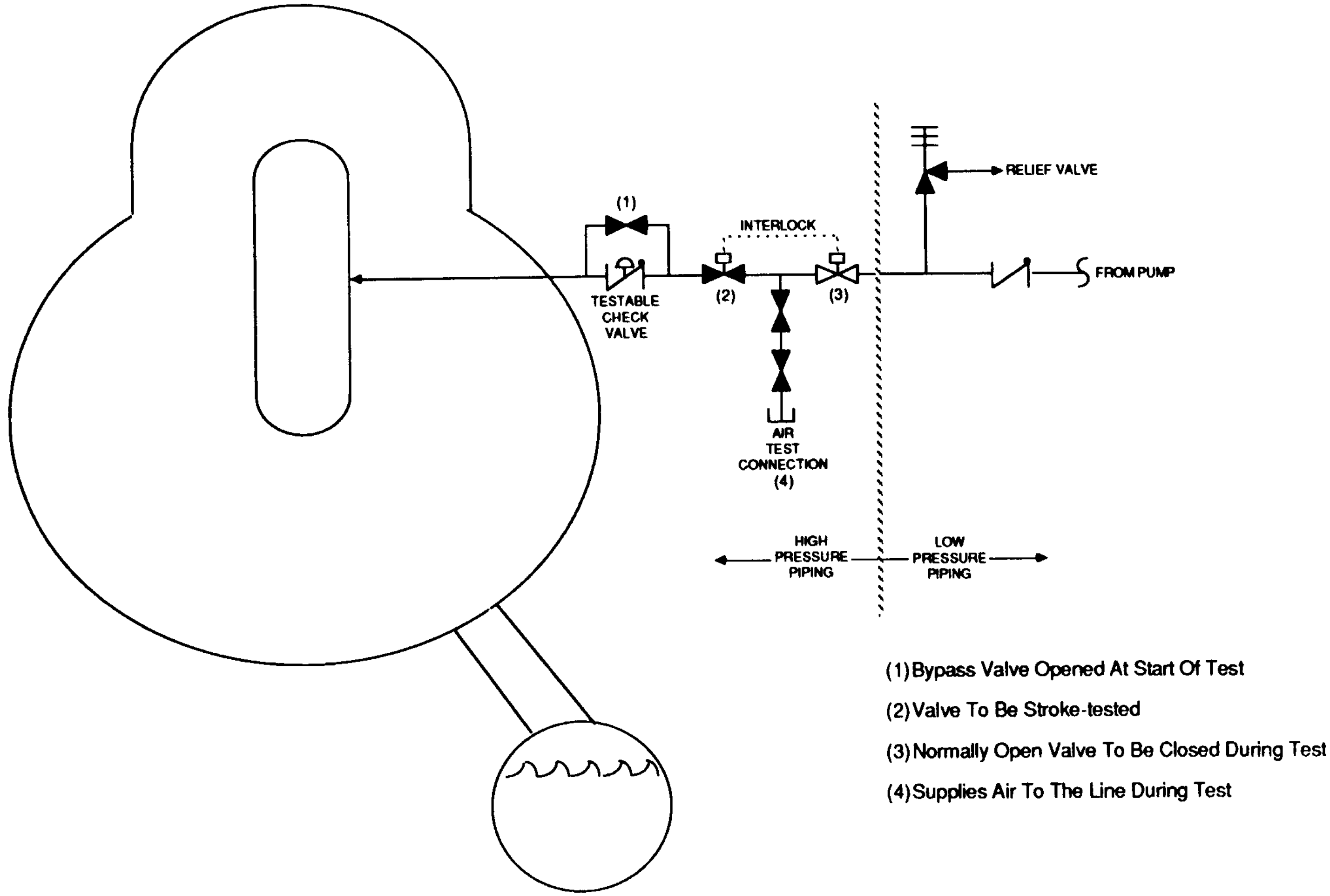


Figure 4.4-11. Typical Valve Arrangement for High-Low Pressure Interface

secured closed and leak tight so that only one room (and system) should be affected.

Also reviewed was the chance that two valves in series (typically a check valve and one MOV) leaked or ruptured between tests and went unnoticed (again refer to Figure 4.4-11). Allowing leak or rupture of the check valve and the MOV within a quarter year time period results in a probability of such an occurrence as approximately 8E-7 (mean) during any one quarter, or about 3E-6 per year. However, with pressure switches located in each line so as to detect such a dual failure, the probability of going undetected appears small. In addition, a catastrophic failure to create the LOCA would have to occur, and more than one room would have to be affected in order to prevent successful mitigation. These last two considerations would appear to suggest that at least another factor of 1E-2 should be applied before the "V" sequence actually leads to core damage.

On the basis of this review and the quantitative and qualitative arguments supplied above, it appears reasonable that the "V" scenario can be estimated at or below 1E-8 per year. This is the threshold value used in the Peach Bottom analysis for defining dominant accident sequences, and so the "V" sequence is not examined any further.

4.4.14 Discussion of Reactor Vessel Rupture (R) Event

The frequency of a rupture of the reactor vessel large enough to be beyond the capacity of the ECCS was estimated in the Reactor Safety Study (RSS) [4] to have a median value of 1.0E-7/yr. with an error spread of a factor of 10. This value is based on an Advisory Committee on Reactor Safeguards report which examined actual data on many types of non-nuclear pressure vessel failures and data from the United States Navy and commercial reactor experience. The important conclusions reached from this analysis are that the disruptive failure probability of reactor vessels designed to nuclear standards is less than 1.0E-6/yr., and the disrupture failure probability of such vessels beyond the capability of engineered safety feature is even lower. The RSS value of 1.0E-7/yr. represents the only estimate of a reactor vessel rupture beyond the capability of the ECCS used in previous PRAs.

Recent analyses of Pressurized Thermal Shock (PTS) in Pressurized Water Reactors (PWRs) are useful in determining the adequacy of the RSS estimate. The PTS analysis was conducted for three plants believed to be particularly susceptible to PTS and evaluated the frequency of flaw propagation through the vessel wall (i.e., vessel rupture) during overcooling transients. Overcooling transients are of particular concern for PTS because thermal stresses are superimposed upon hoop stresses present while the vessel is at or near operating pressure. The frequency of such overcooling transients was calculated using PRA techniques. The thermal-hydraulic conditions in the vessel downcomer region for the overcooling transients were calculated using thermal-hydraulic computer codes. The results from these calculations were used as boundary conditions for a probabilistic fracture-mechanics analysis of the reactor vessels.

The results of these PTS analysis indicate that the frequency of vessel rupture due to PTS is highly uncertain. For the H. B. Robinson plant [51], which is a PWR, the frequency of vessel rupture due to PTS was calculated to have a point estimate of 1.5E-8 and the following distribution:

95% Upper Bound	1.5E-5
Mean	8.4E-6
Median	2.3E-8
5% Lower Bound	1.9E-11

These values were calculated for a hypothetical reactor vessel as the results for the actual H. B. Robinson vessel were too low to permit an illustration of the probabilistic fracture-mechanics analysis method. The large uncertainty analysis is a result of the large uncertainty on the density of the flaws in the vessel.

Three general observations can be drawn from the PTS work concerning the potential for vessel rupture in a BWR. First, the potential of vessel rupture due to PTS in a BWR is generally expected as being substantially less than for a PWR. The fact that BWRs operate at a lower pressure reduces the hoop stress and the design of the vessel allows natural circulation, which reduces thermal stresses during overcooling transients. Second, the PTS calculations for scenarios involving small thermal transients provide some indication of the probability of vessel rupture due to random failure (i.e., flaw propagation occurring with hoop stresses only). A reactor trip situation at H. B. Robinson analyzed in Reference [51] provides such a minimal thermal transient. The frequency of vessel rupture for this situation was calculated as less than 1.0E-10/yr. Third, the frequency of vessel rupture due to PTS is highly uncertain, and the published results for H. B. Robinson are overly conservative since they were calculated for a hypothetical reactor vessel which would be more susceptible to PTS.

Based on these observations, the frequency of vessel rupture in a BWR used in the RSS is believed to be overly conservative. A frequency of less than 1.0E-8/yr. would appear to be more realistic. Therefore, vessel rupture was not considered further in this study.

4.4.15 Anticipated Transient Without Scram Event Tree

4.4.15.1 Event Tree

The ATWS event tree is shown in Figure 4.4-12. The following discussions define the event tree headings and the sequences.

Events in the tree include:

<u>T</u>: An initiating event occurs which requires the reactor to be tripped.

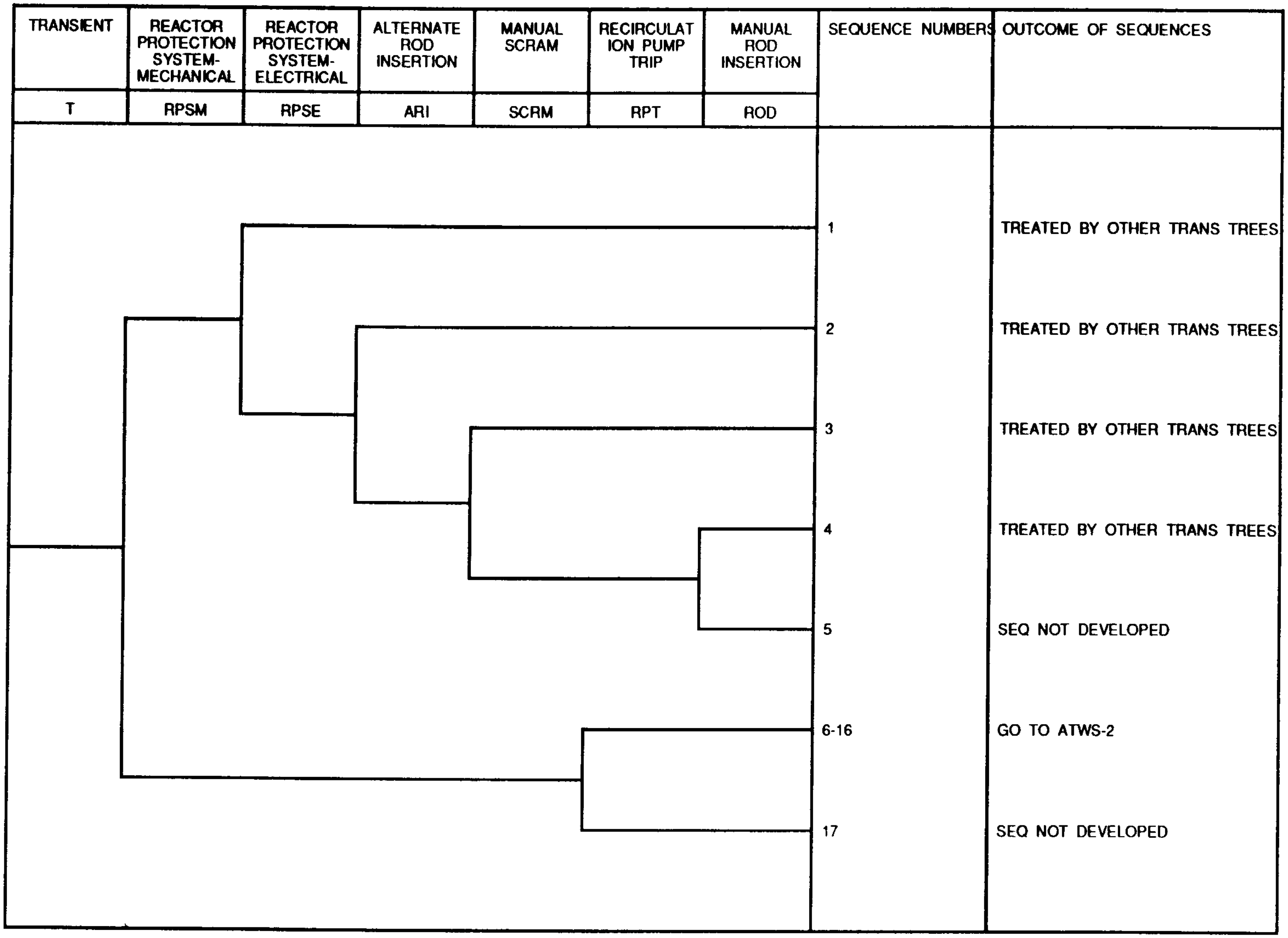


Figure 4.4-12. Anticipated Transient Without Scram Event Tree (Page 1 of 2)

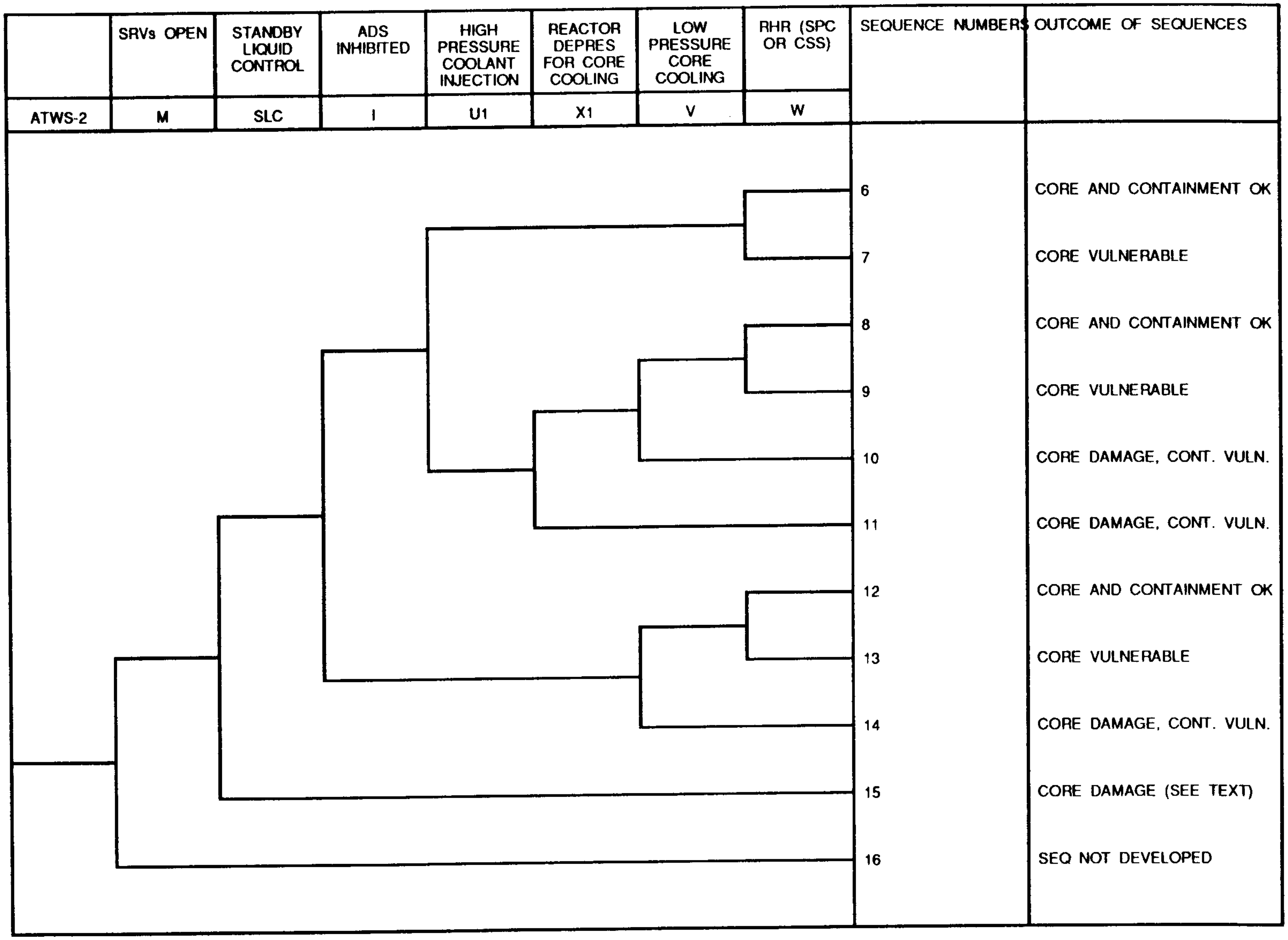


Figure 4.4-12. Anticipated Transient Without Scram Event Tree (Page 2 of 2)

RPSM: Success or failure of the Reactor Protection System-Mechanical (RPSM). Success implies the mechanical portion of the RPS functions properly and reactor scram is imminent upon receipt of the RPS electrical signal. Failure assumes that all rods are inoperable or otherwise left in the position that they occupied before the transient occurred and the operator cannot manually scram the reactor or manually insert the rods.

RPSE: Success or failure of the Reactor Protection System-Electrical (RPSE). Success implies the reactor scram signal operates and reactor subcriticality will be achieved if the rods insert. Failure implies the scram valves did not receive the RPS signal to scram and the control rods are not inserted into the reactor.

ARI: Success or failure of the Alternate Rod Insertion (ARI) system. Success implies the scram valves receive the actuation signal from the system separate from the previously failed RPSE system. Failure implies the actuation signal was not received by the scram valves and the rods are not inserted into the reactor.

SCRM: Success or failure of an attempt to manually scram the reactor. Success implies the operator has activated the reactor scram hydraulic system, the control rods are inserted into the reactor, and subcriticality is achieved. Failure implies the control rods are not inserted into the core.

RPT: Success or failure of a trip of the recirculation pumps. Success implies the recirculation pumps are automatically or manually tripped. RPT success reduces moderator effectiveness, thereby reducing both the power and pressure increase. If manual or automatic pump trip fails, the pumps will cavitate and fail when the operator drops the level to near the top of the active fuel.

ROD: Success or failure of manual rod insertion. Success implies the operator inserts the rods individually into the core and subcriticality is achieved. Failure implies operator cannot manually insert the control rods into the core.

M: Success or failure of overpressure protection by the SRVs. Success implies the SRVs open and the reactor vessel pressure drops or is otherwise stabilized. Failure implies that an insufficient number of SRVs operate to control pressure.

SLC: Success or failure of the Standby Liquid Control System. Success implies the operator initiates the SLC system and one or both pumps function to decrease the reactivity of the core. Failure implies insufficient boration of the core to achieve subcriticality in a timely manner (4 minutes used in this analysis).

I: Success or failure to inhibit the ADS system. Success implies the reactor remains at high pressure to allow HPCI to operate by preventing the ADS from activating to depressurize the reactor. Failure implies ADS is not inhibited and the reactor is subsequently depressurized because of low water level and high drywell pressure conditions.

U1: Success or failure of the High Pressure Coolant Injection (HPCI) system. Success implies HPCI automatically actuates or is manually actuated to provide coolant makeup. Failure implies HPCI does not initiate to provide coolant makeup or operates an insufficient amount of time.

X1: Success or failure of reactor depressurization. Success implies the operator lowers reactor pressure with SRVs to use low pressure cooling following high pressure cooling failure. Failure implies the reactor remains at high pressure.

V: Success or failure of low pressure systems to cool the core. Success implies the reactor water level is maintained so as to provide sufficient core cooling (defined as a reactor water level of two feet above the bottom of the active fuel) using the Condensate, LPCI, LPCS or other low pressure systems when the reactor pressure drops to approximately 400 psig. Failure implies low pressure cooling systems do not provide sufficient injection capacity to the reactor.

W: Success or failure of the RHR system in the SPC or CSS mode. Success implies that the RHR system is operated to provide sufficient containment overpressure protection so that containment integrity is not jeopardized. Failure implies that containment venting must be performed or containment failure occurs because of insufficient heat removal.

The following descriptions refer to the sequences found in Figure 4.4-12.

SEQUENCE 1 -- T*$\overline{\text{RPSM}}$*$\overline{\text{RPSE}}$

A transient occurs that requires the reactor to scram (T). The mechanical RPS functions successfully (/RPSM). The RPS electrical system sends the scram signal to the scram valves (/RPSE). All of the rods are assumed to go into the core and reactor shutdown is achieved. The event then becomes a normal transient and is transferred to the appropriate transient event tree depending on the initiating event.

SEQUENCE 2 -- T*$\overline{\text{RPSM}}$*RPSE*$\overline{\text{ARI}}$

A transient occurs that requires the reactor to scram (T). The mechanical RPS functions successfully (/RPSM) but the RPS electrical system fails (RPSE). A diverse scram signal is successfully sent to the alternate scram valves by the ARI and the reactor is scrammed (/ARI). The sequence is then the same as Sequence 1.

SEQUENCE 3 -- $T*\overline{RPSM}*RPSE*ARI*\overline{SCRM}$

Same as Sequence 2 except ARI fails to signal the scram valves to function. The operator then succeeds in scramming the reactor manually (/SCRM).

SEQUENCE 4 -- $T*\overline{RPSM}*RPSE*ARI*SCRM*\overline{ROD}$

Same as Sequence 3 except manual scram of the reactor fails (SCRM) and the operator successfully inserts the rods into the core by manually driving in the rods (/ROD).

SEQUENCE 5 -- $T*\overline{RPSM}*RPSE*ARI*SCRM*ROD$

Same as Sequence 4 except the operator fails to manually insert the control rods (ROD). This sequence is not developed further since the probability of this sequence is currently estimated to be below 1.0E-8.

SEQUENCE 6 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*\overline{U1}*\overline{W}$

A transient occurs that requires the reactor to scram (T). The mechanical RPS fails (RPSM) which eliminates any possibility to scram the reactor or manually insert the control rods. The recirculation pumps are tripped (/RPT) and the SRVs properly cycle to control reactor pressure (/M). SLC is initiated to inject borated water into the reactor to reduce reactivity (/SLC). The ADS valves are inhibited (/I) to maintain sufficient reactor pressure to initiate HPCI for coolant makeup (/U1). The RHR system is initiated in the SPC or CSS mode (/W) to provide containment overpressure protection, resulting in a safe core and containment.

SEQUENCE 7 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*\overline{U1}*W$

Same as Sequence 6 except the RHR system fails to provide containment overpressure protection (W). This results in a core vulnerable state. (Note: Since this sequence probability was estimated at or below 1.0E-8 at this point (see Section 4.10), resolution of the vulnerable state was not necessary).

SEQUENCE 8 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*U1*\overline{X1}*\overline{V}*\overline{W}$

Same as Sequence 6 until HPCI fails (U1). The reactor is depressurized (/X1) and a low pressure core cooling system is initiated for coolant makeup (/V). The RHR system in the SPC or CSS mode provides containment overpressure protection (/W), resulting in a safe core and containment.

SEQUENCE 9 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*U1*\overline{X1}*\overline{V}*W$

Same as Sequence 8 except the RHR system fails to provide containment overpressure protection (W), resulting in a core vulnerable state. (Note: Since this sequence probability was estimated at or below 1.0E-8 at this point (see Section 4.10), resolution of the vulnerable state was not necessary).

SEQUENCE 10 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*U1*\overline{X1}*V$

Same as Sequence 8 except low pressure core cooling fails (V), resulting in core damage in a vulnerable containment.

SEQUENCE 11 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*\overline{I}*U1*X1$

Same as Sequence 10 except reactor depressurization fails (X1) and core cooling capability is lost, resulting in core damage in a vulnerable containment.

SEQUENCE 12 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*I*\overline{V}*\overline{W}$

A transient occurs that requires the reactor to scram (T). The mechanical RPS fails (RPSM) which eliminates any possibility to scram the reactor or manually insert the control rods. The recirculation pumps are tripped (/RPT) and the SRVs properly cycle to control reactor pressure (/M). SLC is initiated to inject borated water into the reactor to reduce reactivity (/SLC). The ADS valves are not inhibited (I) and the reactor depressurizes which allows low pressure core cooling systems to operate (/V). The RHR system in the SPC or CSS mode is initiated for containment overpressure protection (/W), resulting in a safe core and containment.

SEQUENCE 13 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*I*\overline{V}*W$

Same as Sequence 12 except the RHR system fails to provide containment overpressure protection (W), resulting in a core vulnerable state. (Note: Since this sequence probability was estimated at or below 1.0E-8 at this point (see Section 4.10), resolution of the vulnerable state was not necessary).

SEQUENCE 14 -- $T*RPSM*\overline{RPT}*\overline{M}*\overline{SLC}*I*V$

Same as Sequence 12 except low pressure core cooling is unsuccessful (V), resulting in core damage in a vulnerable containment.

SEQUENCE 15 -- $T*RPSM*\overline{RPT}*\overline{M}*SLC$

A transient occurs that requires the reactor to scram (T). The mechanical RPS fails (RPSM) which eliminates any possibility to scram the reactor or manually insert the control rods. The recirculation pumps are tripped (/RPT) and the SRVs properly cycle to control reactor pressure (/M). SLC fails to initiate (SLC) which initiates a series of events that lead to core damage. Steam from the reactor vessel is continuously dumped into the suppression pool, which increases the pool temperature and pressure. HPCI might be used for core cooling until it fails on high suppression pool temperature, which is likely to occur in approximately 15 minutes. The reactor must then be depressurized to allow low pressure systems to cool the core. The containment is becoming overpressurized and venting is likely to be performed to prevent rupture of the containment. Low pressure core cooling systems (LPCS, LPCI) are assumed to fail during containment venting or subsequent containment failure due to insufficient NPSH for the pumps. Containment venting or containment failure will begin to fill the reactor building with steam and could potentially enter the turbine building by failing the blowout panels that lead to the turbine building. Core cooling must be initiated at this point with a large capacity system. Condensate could be initiated but is likely to fail because of limited capacity in the condenser or because of steam effects. HPSW is the final system with the capacity to provide sufficient cooling. However, the expert elicitation process indicates that the presence of steam in the reactor building will very likely fail HPSW valves with a probability of ≥0.7. Since all core cooling systems are very likely to be lost in this sequence, the development of the event tree was constructed to simply show that failure to scram and loss of SLC will lead to core damage. However, this assumption does not seem too conservative in light of the very high probabilities associated with the loss of both high and low pressure cooling systems of sufficient capacity to mitigate this accident sequence.

SEQUENCE 16 -- $T*RPSM*\overline{RPT}*M$

Same as Sequence 15 except the SRVs fail to control reactor pressure (M) and the sequence is not developed further due to an estimated probability below 1.0E-8.

SEQUENCE 17 -- T*RPSM*RPT

A transient occurs that requires the reactor to scram (T). The recirculation pumps fail to trip and the sequence is not developed further due to an estimated probability below 1.0E-8.

4.4.16 Event Tree Nomenclature

Table 4.4-1 contains a summary of the nomenclature used to identify the systems on the event trees.

Table 4.4-1
Event Tree Nomenclature

ARI	-	Failure of the Alternate Rod Insertion System
B	-	Failure of all AC power (station blackout)
C	-	Failure of the Reactor Protection System (RPS)
C1	-	Failure of RPS and manual scram
I	-	Failure to inhibit the ADS system
L	-	Failure of operator to isolate S3 "leak"
LOSP,LOSP1	-	Failure to maintain offsite power; Different Designations for this Event are for Different Frequencies
M	-	Failure of Safety Relief Valves (SRVs) to open
P	-	Failure of SRVs to close
P1,P2,P3	-	Failure of one, two or three SRVs to reclose
Q,Q1,Q2	-	Failure of the Power Conversion System (PCS); different designations for this event are for different frequencies
ROD	-	Failure to manually insert the control rods
RPSM	-	Failure of the mechanical RPS
RPSE	-	Failure of the electrical RPS
RPT	-	Failure to trip the recirculation pumps
SCRM	-	Failure to manually scram the reactor
SLC	-	Failure of the Standby Liquid Control System
U1	-	Failure of the High Pressure Coolant Injection (HPCI) system
U1'	-	Failure of HPCI without ventilation
U2	-	Failure of the Reactor Core Isolation Cooling (RCIC) system
U2'	-	Failure of RCIC without ventilation
U3	-	Failure of the Control Rod Drive (CRD) system (2 pump mode)
U4	-	Failure of the Control Rod Drive (CRD) system (1 pump mode)
U4'	-	Failure of CRD to survive containment venting
V1	-	Failure of the Condensate system
V1'	-	Failure of Condensate to survive containment venting
V2	-	Failure of the Low Pressure Core Spray (LPCS) system
V3	-	Failure of the Low Pressure Coolant Injection (LPCI) system
V4	-	Failure of the High Pressure Service Water (HPSW) system as an injection source to the reactor
V4'	-	Failure of HPSW (injection source) to survive containment venting
R	-	Rupture of the containment
W1	-	Failure of the Suppression Pool Cooling (SPC) mode of RHR
W2	-	Failure of the Shutdown Cooling (SDC) mode of the RHR
W3	-	Failure of the Containment Spray (CS) mode of the RHR
X1	-	Failure to depressurize the primary system via SRVs or the Automatic Depressurization System (ADS)
X2	-	Failure to depressurize the primary system to allow SDC to operate
X3	-	Failure to depressurize the primary system subsequent to an initial primary system depressurization
Y	-	Failure of Primary Containment Venting system (including makeup to the pool as required)

4.5 Plant Damage State Analysis

The plant damage states are the interface between the front-end analysis, or system analysis leading to core damage accident sequences, and the back-end analysis. In order to provide for this interface, the cut sets for the accident sequences contributing to core damage must be sorted into groups with common attributes relative to the back-end accident progression event trees. This could be accomplished by constructing a bridge tree between the sequence event tree and the containment event tree or by answering selected questions for each cut set that specify the state of the systems or phenomena when core damage occurs. The latter approach was chosen for Peach Bottom.

4.5.1 Plant Damage State Definitions

Sixteen questions were determined by the back-end analyst to properly describe the state of the systems as the plant accident progresses into a core damage situation. Each unique set of answers to the sixteen questions is defined as a plant damage state (PDS). Each unique plant damage state potentially results in a different challenge to the containment and ultimately a different source term release to the environment. Table 4.5-1 lists the sixteen questions posed for Peach Bottom. The total possible combination of answers, and hence plant damage states, is the product of the number of answers for each question. This is a very large and clearly unmanageable number. However, a number of combinations are not logical and many combinations are not significant for any given analysis. Thus, the expectation was that a reasonable number of plant damage states would evolve, which was the actual outcome of the analysis.

During the process of examining each cut set, certain information was useful in determining the answers and providing guidelines to simplifying the task. Questions 1 (initiating event) and 5 (stuck-open relief valve) can be answered by inspection of the accident sequence itself. Question 6 concerning success or failure of HPCI and RCIC may or may not be obvious from the accident sequence. If the initiator is a large or medium LOCA, steam to HPCI and RCIC will be lost early so that, effectively, both fail. The word "initially" used in these questions means during the period prior to the time of core damage.

Answers to several questions include a case where the system has not failed due to hardware, but due to loss of power. Thus, if power were restored, the system potentially could operate. The purpose of these questions, as well as some others, is to determine if water could be injected later during the accident progression. Injection could mitigate the core melt or it could cause detrimental effects. That is a back-end concern, but the answers to these front-end system questions establish the input state to the back-end analysis.

Similarly, several questions have answers indicating that the system is available. That is, the system may be operating, but the pressure is too high for injection, or perhaps the number of pumps is insufficient for

Table 4.5-1. Peach Bottom APET Questions for Plant Damage States

In order to define the plant damage states for Peach Bottom, the following information is needed for each cut set of each accident sequence such that each question is uniquely answered.

1. What is the Initiating Event (IE)?

    1) A-Large LOCA
    2) S1-Medium LOCA
    3) S2/3-Small/small-small LOCA
    4) T-Transient (all other transients)
    5) TC-Transient without scram (ATWS)
    6) IORV-Inadvertent open relief valve

2. Is there a Loss of Offsite Power (LOSP)?

    1) Seismic induced LOSP
    2) LOSP IE or random LOSP
    3) No LOSP

3. Is there a station blackout (Event B)?

    1) Yes - LOSP IE or random LOSP and loss of all Diesel Generators (DGs)
    2) No - At least one DG working

4. Is DC power available given a station blackout?

    1) No - All DC is failed
    2) Yes - At least one train of DC is working

5. Does a safety relief valve (SRV) stick open early?

    1) Yes - At least one SRV sticks open (P1,P2, or P3)
    2) No - No stuck open SRV

6. Are the High Pressure Injection system (HPI) and Reactor Core Isolation Cooling system (RCIC) initially working (Events U1 and U2)?

    1) No - Both HPCI and RCIC have initially failed.
    2) Yes - Either HPCI or RCIC is initially working.

    If these systems work initially, there is no core damage. There is no recovery after core damage since no steam will be available. Both systems work after LOSP and at high pressure so there are no recoverable or available questions.

7. Is the Control Rod Drive system (CRD) initially working (Events U3 and U4)?

Table 4.5-1. Peach Bottom APET Questions for Plant Damage States (Cont.)

1) fCRD - CRD is definitely failed.
2) rCRD - CRD is not on but has not failed either (i.e., depends on LOSP or T1 restored).
3) Yes - CRD is working.

(This assumes that if it can work then it's normally on; therefore, no availability question is asked).

8. What is the initial vessel pressure (Events X1 and X2)?

1) fADS - ADS has failed; therefore, the vessel can not go to low pressure.
2) High - Auto ADS has failed and the vessel can go to low pressure but the operator has not depressurized.
3) Low - Auto ADS or Manual depressurization has worked or any LOCA or transient and stuck open SRV has occurred except for ATWS.

9. What is the initial status of low pressure ECCS (Events V2 and V3)?

1) fLPC - Both LPCI and LPCS have failed and can not be recovered.
2) Recoverable - Both are not currently available but can be recovered (i.e., if LOSP and B or T1 and B restored).
3) Available - One pump is running but no injection due to high vessel pressure.
4) Yes - Either LPCS or LPCI is working

10. What is the initial status of Residual Heat Removal systems, RHR (SCS, SPC, CSS) i.e., W1, W2, and W3?

1) fRHR - All RHR modes are failed.
2) Recoverable - All RHR modes are currently unavailable but can be recovered after LOSP and B or T1 and B restored.
3) Yes - One RHR mode is available and working.

(no available question, since if on, it will work).

11. What is the initial status of Condensate (Event V1)?

1) fCOND - condensate is failed.
2) rCOND - condensate is recoverable (after LOSP or T1 restored).
3) aCOND - condensate is available but not injecting.
4) Yes - condensate is working (not possible given core damage).

Table 4.5-1. Peach Bottom APET Questions for Plant Damage States (Cont.)

12. What is the initial status of High Pressure Service Water system, HPSW (Event V4)?

    1) fHPSW - HPSW is failed.
    2) rHPSW - HPSW is recoverable. (after LOSP and B or T1 and B restored).
    3) aHPSW - HPSW is available. Manual lineup and actuation required.
    4) Yes - HPSW is working (not possible given core damage).

13. What is the initial status of the Containment Spray System (CCS) (Event W3)?

    1) fCSS - CSS is failed.
    2) rCSS - CSS is recoverable (after LOSP and B or T1 and B restored).
    3) aCSS - CSS is available, but manual actuation is required.
    4) Yes - CSS is working.

14. Is the containment vented before core damage (Event Y)?

    1) No - Containment is not vented.
    2) DW - Drywell vent (not likely at Peach Bottom).
    3) uDW - Drywell is vented in ATWS, but pressure still high.
    4) uWW - Wetwell is vented in ATWS, but pressure is still high.
    5) WW - Wetwell vent

15. What is the level of containment leakage?

    1) No leakage in excess of tech spec.
    2) Level 2 leakage occurs after accident (leak).
    3) Level 3 leakage occurs after accident (rupture).
    4) Level 2 leakage occurs before accident or isolation failure (leak).
    5) Level 3 leakage occurs before accident or isolation failure (rupture).

    (A leak vs. rupture depends on the sequence. In non-ATWS sequences, a leak would be about an 8 inch line or less. For ATWS sequences, a leak would be less than two 18 inch lines.)

16. What is the location of leakage?

    1) Containment intact
    2) Drywell
    3) Drywell Head
    4) Wetwell

success in preventing core damage, but could affect the back-end situation. Also, the system could be available if the operator should choose to use it.

The answer to Question 14 is 1 if anything fails that would prevent venting and X where venting is possible, but not asked in the system event trees.

Containment leakage is derived from the containment isolation system fault tree. Initially, if isolation failure occurs with probability one, it is in the drywell and designated as 22 for the answers to Questions 15 and 16. This is the case for loss of the 4160 volt AC bus B. If random failures of valves cause the leakage, the description is Y2 given LOSP and X2 otherwise. Subsequently, it was determined that containment isolation failure does not result in a significant leak at Peach Bottom. An isolation fault tree was constructed and two paths had the potential to be unisolated with a significant probability; the RBCW RCP seal cooling lines and, the drywell (DW) drain lines. From the back-end perspective, neither of these was important. The RBSW lines are not connected to the primary and leakage into the RBCW system is unlikely. The DW sump lines require a double random valve failure which has a probability low enough to be neglected.

A complete discussion of the plant damage states is given in the accident progression event trees section of NUREG/CR-4551, Volume 3.

4.5.2 Descriptions of the PDS Vector

The sixteen character vector describing the plant damage state (i.e., the answers to the 16 questions) can be subdivided into seven groups of questions that fit together logically.

Question 1 - What is the Initiating Event?

Questions 2, 3, and 4 - What Electric Power is available?

Question 5 - Do any Safety Relief Valves stick open?

Questions 6 and 7 - What is the status of the High Pressure Systems?

Question 8 - What is the status of RCS Depressurization?

Questions 9 to 13 - What is the status of the Low Pressure and decay heat removal Systems?

Questions 14 to 16 - Is the containment Vented or does Isolation fail?

As will be seen in Section 4.11, there are a limited number of answers to each of these groups of questions, and only a few combinations of these groups out of the large number possible actually show up as dominant in the analysis. This is explained further in Section 4.11 in the process of delineating and quantifying the plant damage states.

4.6 System Analysis

Section 4.6.1 provides an introduction to the system modeling performed in the Peach Bottom analysis. Sections 4.6.2 through 4.6.23 describe the modeling effort for each system. These subsections contain a system description, identification of interfaces and dependencies, discussion of operational constraints, a description of the models developed, specific assumptions used in modeling, and a discussion of any unique operational experience for each of the systems. Justification for those systems not modeled are presented in Section 4.6.24. The systems which were modeled in the Peach Bottom study are shown in Table 4.6-1. The nomenclature used to identify system failures is described in Section 4.6.25.

4.6.1 System Modeling Approach and Scope

System models were developed for each of the front line systems identified in the event tree headings and for all support systems required to operate the front line systems. Fault tree models were constructed for most of the systems using either detailed fault trees or simplified trees focusing on major failures. For those systems where fault tree models were not constructed, actual data could be used to represent the dominant failures of the systems (including interactions). For example, sufficient data exists to estimate the probability of loss of the power conversion system following a reactor trip without having to perform a fault tree analysis. These failure models were developed with top events corresponding to the success criteria used in the event tree analysis. Some systems have different success criteria in different circumstances and hence different top events. A few events in the event trees, such as the probability of a stuck-open valve, are single data values presented in the data section and hence are not discussed in this section.

Modeling of the systems was performed at the component level but with pipe segments, when deemed appropriate, indicated on the schematics. A pipe segment is a series collection of components within the system which could be modeled as one super-component or module independent from the rest of the system. The independent failure probability associated with a pipe segment could then be estimated as the sum of the individual failure probabilities of the components within the segment. Operator actions in response to plant conditions were included in the models where specific procedures for these actions were available. Operator errors of commission were not included in the fault tree analysis. Recovery actions for each accident sequence are handled at the cut set level of analysis and are covered in Section 4.8.

Details of the modeling process and assumptions were made throughout the system analysis process. The assumptions about the specific systems are provided in the system write-ups.

System schematics are provided for most of the systems analyzed. Figure 4.6.1-1 provides symbols and related abbreviations used in the schematics.

Table 4.6-1.
Systems Included in the Peach Bottom Study

SYSTEM	TYPE OF MODEL
Actuation and Control (ESF)	Fault Tree
Automatic and Manual Depressurization (ADS)	Fault Tree
Condensate (CDS)	Fault Tree
Containment Spray (CSS)	Fault Tree
Control Rod Drive (CRD)	Fault Tree
Electric Power (ACP,DCP)	Fault Tree
Emergency Service Water (ESW)	Fault Tree
Emergency Ventilation (EHV)	Fault Tree
High Pressure Coolant Injection (HCI)	Fault Tree
High Pressure Service Water (HSW)	Fault Tree
Instrument Air (IAS)	Fault Tree
Low Pressure Coolant Injection (LCI)	Fault Tree
Low Pressure Core Spray (LCS)	Fault Tree
Primary Containment Venting (PCV)	Fault Tree
Reactor Building Cooling Water (RBC)	Fault Tree
Reactor Core Isolation Cooling (RCI)	Fault Tree
Shutdown Cooling (SDC)	Fault Tree
Standby Liquid Control (SLC)	Fault Tree
Suppression Pool Cooling (RHR/SPC)	Fault Tree
Turbine Building Cooling (TBC)	Fault Tree
Reactor Protection (RPS)	Data Value
Power Conversion (PCS)	Data Value

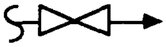
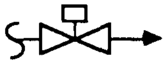
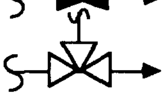
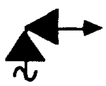
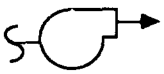
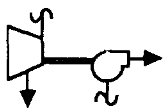
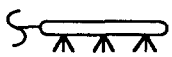
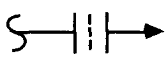
	Normally Open Manual Valve
	Normally Closed Manual Valve
	Normally Open Motor Operated Valve
	Normally Closed Motor Operated Valve
	Motor Driven Butterfly Valve
	Testable Check Valve
	Normally Open Air Operated Valve
	Normally Closed Air Operated Valve
	Normally Closed Explosive Valve
	Three Way Valve (Any shaded portion of valve implies valve is normally closed to flow in shaded direction)
	(Safety) Relief Valve (Normally Closed)
	Check Valve
	Motor Driven Check Valve
	Heat Exchanger Or Cooler
	Motor Driven Pump
	Turbine Driven Pump
	Positive Displacement Pump
	Heater
	Spray Header
	Orifice

Figure 4.6.1-1. Symbols and Abbreviations Used in Schematics.

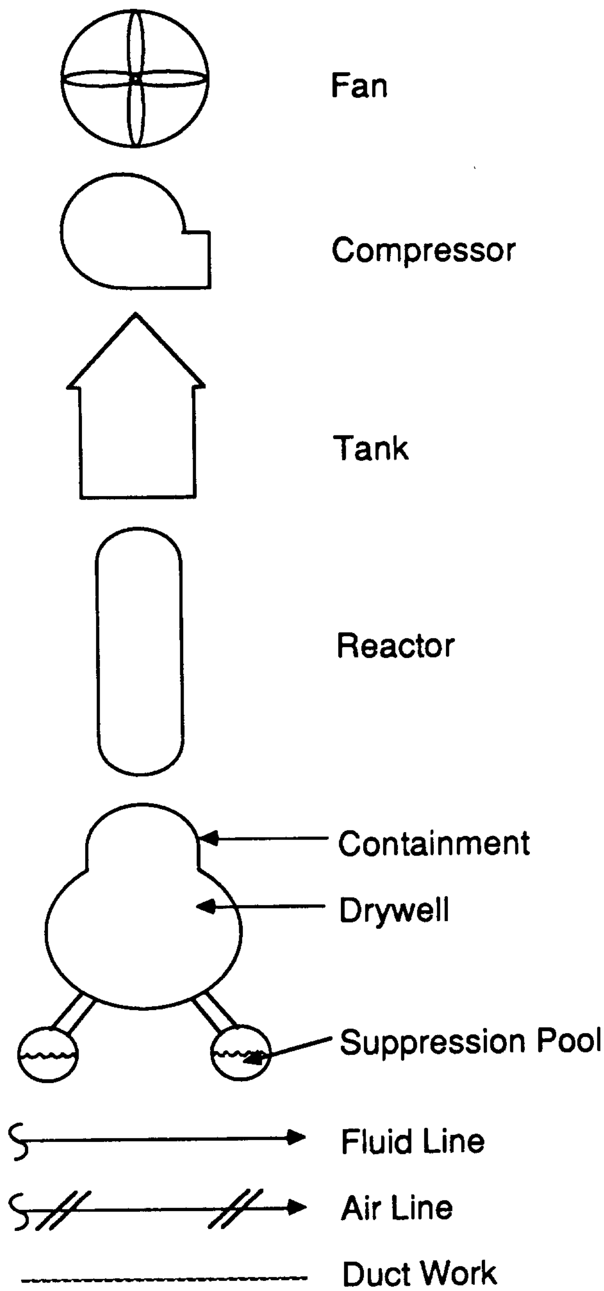


Figure 4.6.1-1. Symbols and Abbreviations Used in Schematics.
(Continued)

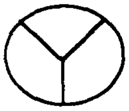
	Diesel Generator
	Charger
	Battery
	Inverter
	Transfer Switch
	Bus
LO	Locked Open
LC	Locked Closed
NC	Normally Closed
NO	Normally Open
FC	Fails Closed
FO	Fails Open

Figure 4.6.1-1. Symbols and Abbreviations Used in Schematics. (Concluded)

4.6.2 Identification of Systems

The systems modeled in the Peach Bottom analysis were: Actuation and Control (ESF), Automatic and Manual Depressurization (ADS), Condensate (CDS), Containment Spray (CS), Control Rod Drive (CRD) -- (Enhanced and One Pump), Electric Power (ACP,DCP), Emergency Service Water (ESW), Emergency Ventilation (EHV), High Pressure Coolant Injection (HPCI), High Pressure Service Water (HPSW), Instrument Air/Nitrogen (IAS), Low Pressure Coolant Injection (LPCI), Low Pressure Core Spray (LPCS), Primary Containment Venting (PCV), Reactor Building Cooling Water (RBCW), Reactor Core Isolation Cooling (RCIC), Shutdown Cooling (SDC), Standby Liquid Control (SLC), Suppression Pool Cooling (SPC), Turbine Building Cooling Water (TBCW) and as data values, the Reactor Protection System (RPS) and Power Conversion System (PCS).

4.6.3 Actuation and Control (Emergency Safeguard Features) System

4.6.3.1 ESF Description

The function of the ESF system is to initiate appropriate responses from various cooling systems so that the fuel is adequately cooled under abnormal or accident conditions.

Only that equipment required for the initiation and control of HPCI, RCIC, Automatic Depressurization System (ADS), LPCS and LPCI (the major Emergency Core Cooling System [ECCS] equipment) were modeled. Any additional unique instrumentation and isolation features were modeled as part of the associated systems. Actuation of other systems are addressed in the individual write-ups.

The ESF system is automatically initiated. Manual actuation is provided in the control room so that operator action is possible if there is a deficiency in the automatic actuation of the equipment or to provide control over long term accidents.

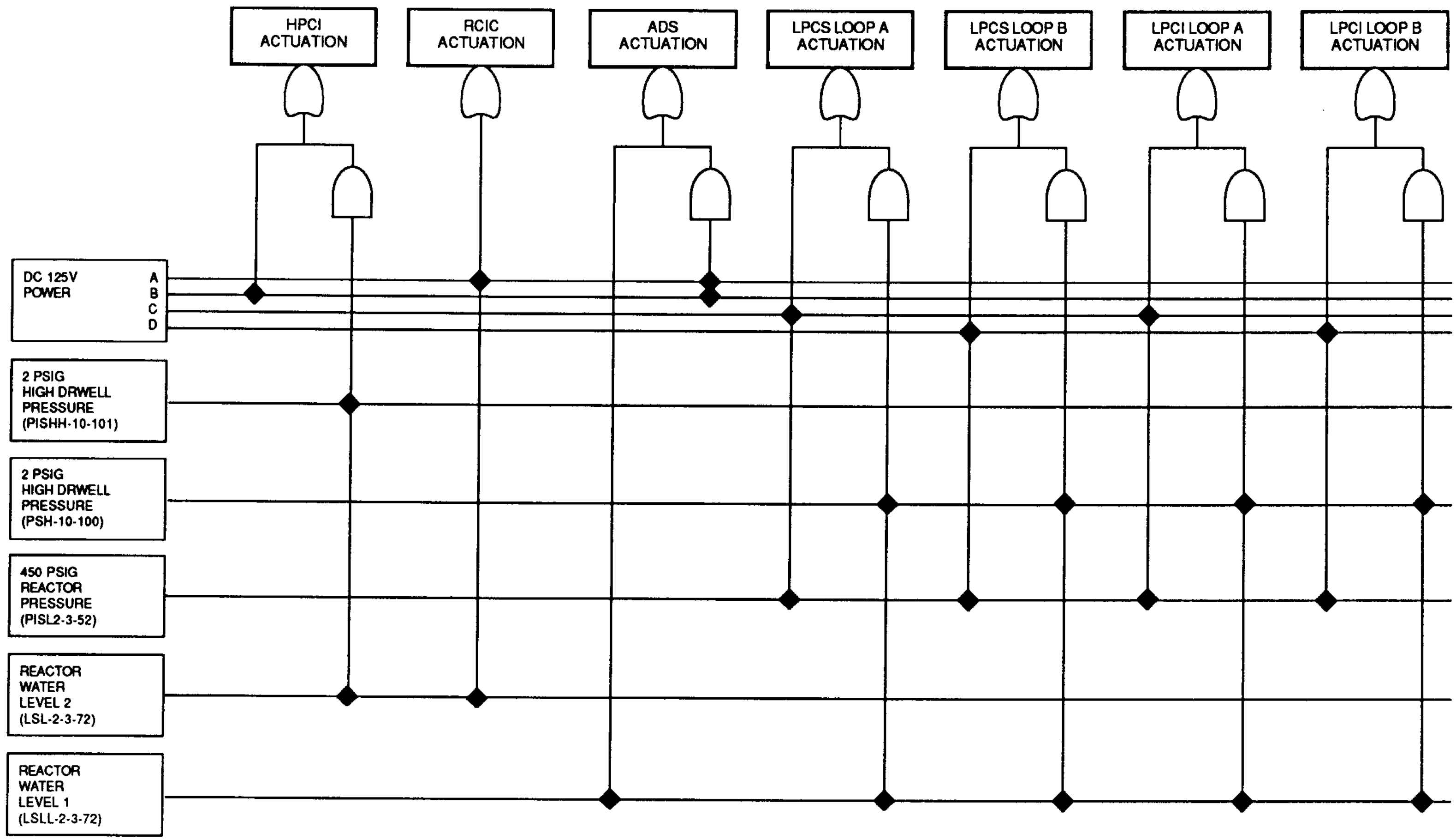
The success criteria for the ESF system is actuation of the cooling systems in time to limit fuel cladding temperature to acceptable levels. The specific success criteria of the actuation circuits depend on the success criteria for the front-line systems they support.

The response of the ESF systems is provided to the operator in the control room.

4.6.3.2 ESF Interfaces and Dependencies

A simplified dependency diagram of the HPCI, RCIC, ADS, LPCS and LPCI systems is provided by Figure 4.6.3-1. Shown are the major support needs for the systems as indicated by the solid diamonds.

Specific actuation and control descriptions can be found in the individual system sections.



Dependency Diagram Is Shown Using Failure Logic. Refer To The Fault Trees For Actual Failure Logic Details.

Figure 4.6.3-1. Actuation and Control Dependency Diagram.

4.6.3.3 ESF Test and Maintenance

Testing requirements for actuation and control are addressed in the individual system sections.

4.6.3.4 ESF Technical Specifications

All technical specifications for actuation and control are addressed in the individual system sections.

4.6.3.5 ESF Logic Model

The ESF system was modeled using a fault tree for generation of all signals required to actuate HPCI, RCIC, ADS, LPCS and LPCI. The fault tree model is presented in Appendix B.

Three human errors were incorporated into the ESF fault tree model. These errors are; operator miscalibration of all reactor level sensors, operator miscalibration of all high drywell pressure sensors, and operator miscalibration of all reactor pressure sensors.

4.6.3.6 ESF Assumptions

(1) Testing usually places components in the "trip" state. Therefore test unavailability or failure to restore after testing are not considered.

(2) Maintenance unavailability and failure to restore are considered part of the system data values, therefore no new values were added to the data list.

4.6.3.7 ESF Operating Experience

Any peculiarities in the operational history of the ESF system are addressed in the individual system sections.

4.6.4 Automatic and Manual Depressurization System

4.6.4.1 ADS Description

The ADS is designed to depressurize the primary system to a pressure at which the low pressure injection systems can inject coolant to the reactor vessel (event tree nomenclature--X1,X2,X3).

The Automatic Depressurization fault tree (event tree nomenclature--X1) is used for the automatic or, if required, manual operation of the ADS system to depressurize the primary system. This allows the low pressure injection systems to be used to cool the core. The Manual Depressurization fault tree (event tree nomenclature--X2) is used exclusively for manual operation of the ADS/SRV system to depressurize the primary system. This allows the SDC mode of the Residual Heat Removal (RHR) system to be used. A data value is used for the event tree question," Do the ADS/SRV valves reopen following containment failure or venting?" (event tree nomenclature--X3). This is strictly a survivability concern.

The ADS consists of five safety relief valves capable of being manually opened. Each valve discharges via a tailpipe line through a downcomer to the suppression pool. Relief valve capacity is approximately 820,000 lb/hr. A simplified schematic of the ADS is provided by Figure 4.6.4-1.

The ADS is automatically initiated. The operator may manually initiate the ADS or may depressurize the reactor vessel using the six relief valves that are not connected to ADS logic. The operator can inhibit ADS operation if a spurious ADS signal occurs or if the operator desires to do so (as in an Anticipated Transient Without Scram [ATWS] scenario).

The success criterion for the ADS is three of five valves opening to depressurize the reactor. For further information, refer to success criteria discussions in Section 4.4.

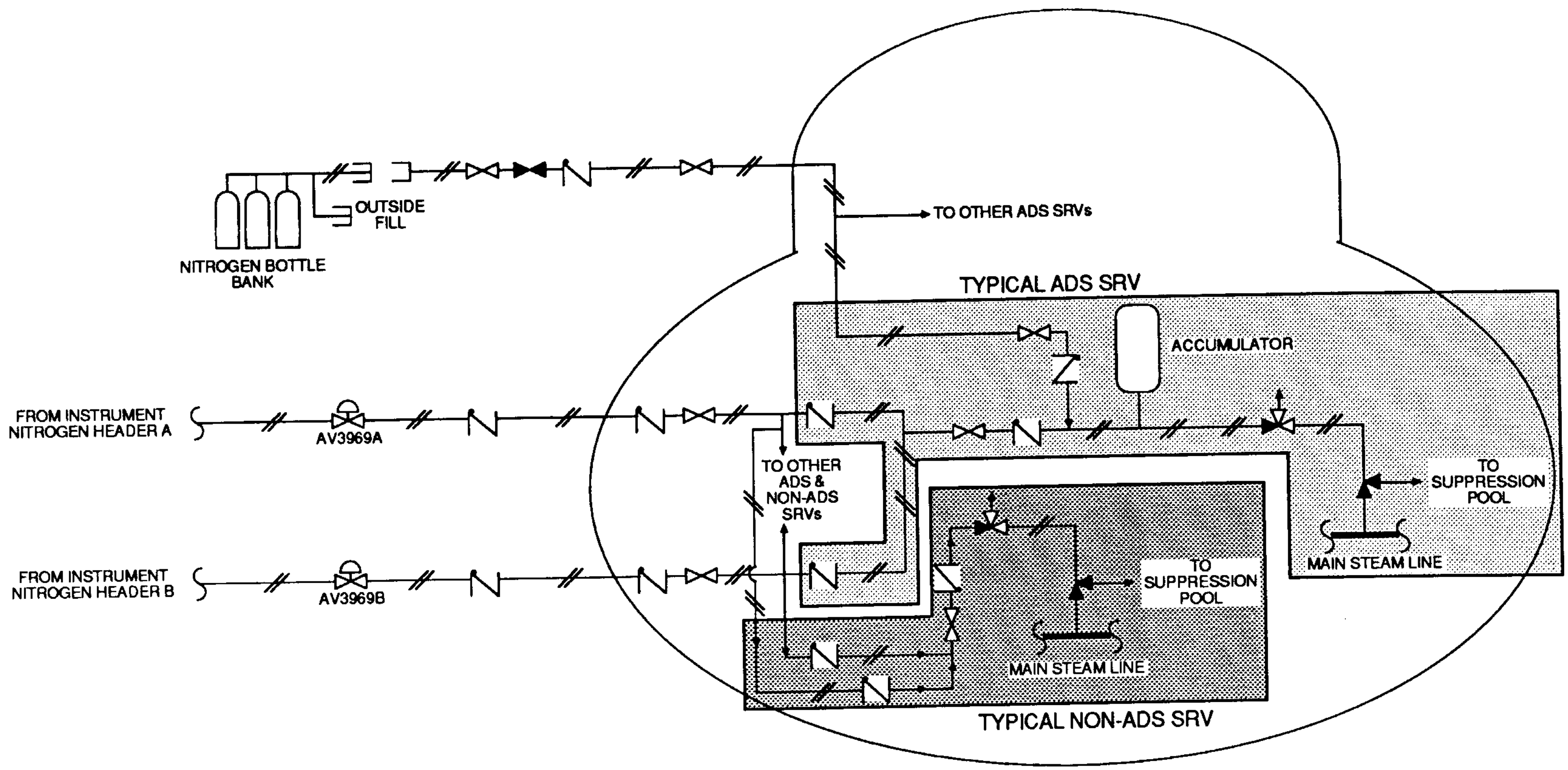
The ADS valves are located inside the containment. ADS performance is not normally affected by accident conditions since the equipment is qualified for accident conditions and the air/nitrogen supply pressure is judged to be sufficiently high to allow valve operation under most containment conditions. However, should containment pressure be excessively high (~85 psig or greater), the valves could not be kept open since the air/nitrogen supply pressure is limited to ~85 psig. This is based on discussions with Philadelphia Electric Company (PECO) personnel, who have indicated the supply is orificed to that limit.

4.6.4.2 ADS Interfaces and Dependencies

The ADS depends upon air/nitrogen and 125 VDC power sources. A simplified dependency diagram of the ADS is provided by Figure 4.6.4-2. Shown are the major support needs for the ADS as indicated by the solid diamonds. Air/nitrogen pressure is used to open the ADS valves. Accumulators for each ADS valve contain enough pressure for approximately five valve operations. In addition to the accumulators, there is a nitrogen bottle supply that can be manually valved in and an additional outside hook-up capability to a nitrogen truck or other source.

ADS logic consists of two divisions. Power dependencies for each division are the 125 VDC/A bus as a primary source and the 125 VDC/B bus as a backup source. ADS valve power is from either 125 VDC/A (the primary DC supply) or 125 VDC/D (backup DC supply). ADS logic is failed if 125 VDC/A and the relay that switches power fail. However, each relief valve has its own relay that switches power for solenoid operation.

Automatic ADS initiation occurs upon receipt of a low-low reactor water level signal (with an ~eight-minute time delay), a low-low level and high drywell pressure signal with a two minute delay. Any of these must be concurrent with one LPCI or two LPCS pumps running, for ADS to work automatically.



VALVES POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.4-1. Automatic and Manual Depressurization System Schematic.

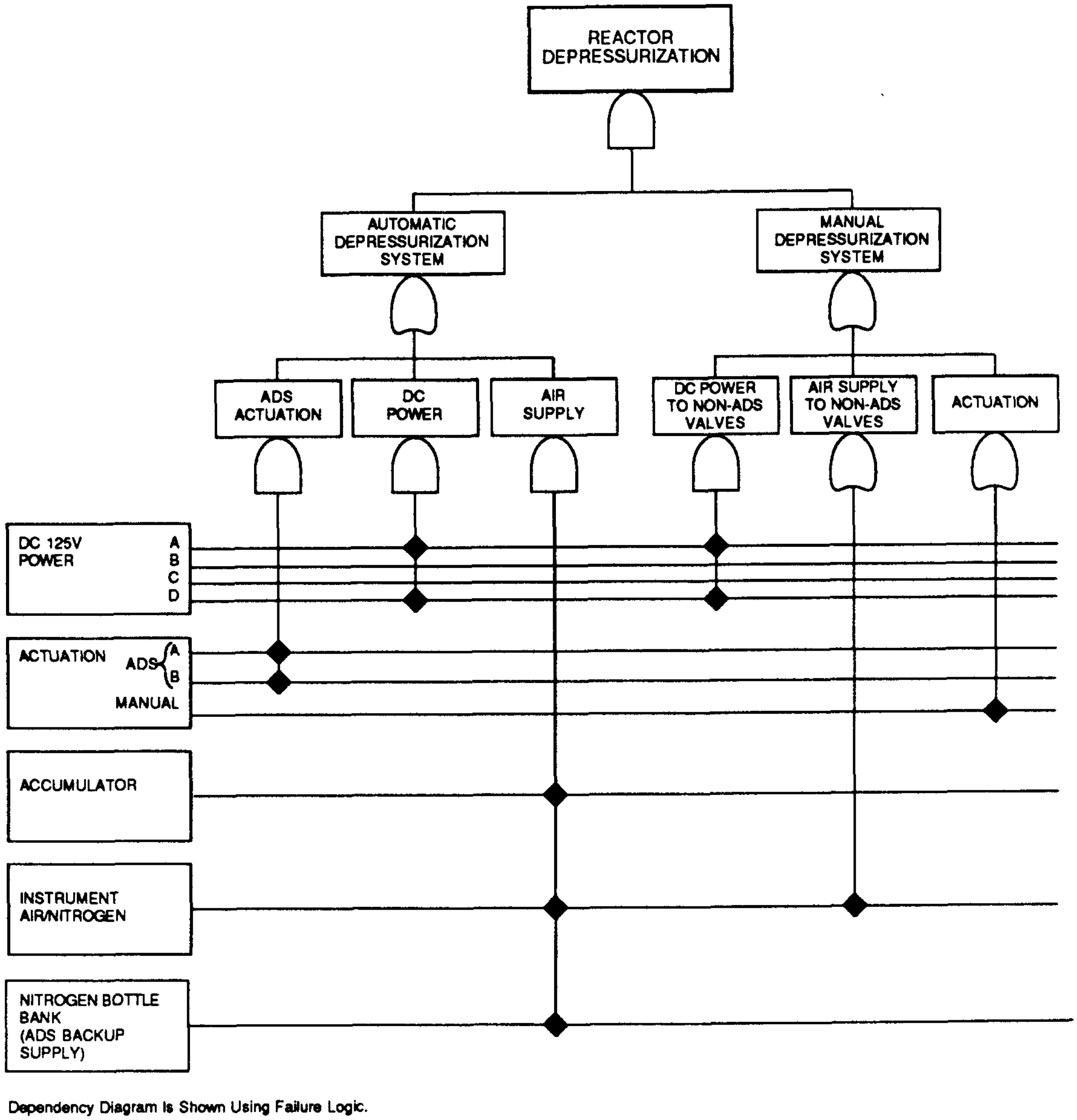


Figure 4.6.4-2. Automatic and Manual Depressurization System Dependency Diagram.

Low-low reactor water level sensors are shared with the LPCS and LPCI systems.

4.6.4.3 ADS Test and Maintenance

A simulated automatic actuation of the ADS is performed prior to startup after each refueling.

4.6.4.4 ADS Technical Specifications

If any one ADS valve is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the HPCI system is operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.4.5 ADS Logic Model

The ADS was modeled using two fault trees for the depressurization of the reactor either automatically or manually (see Appendix B and the discussion in 4.6.4.1).

Piping ruptures were considered to be negligible compared to other failures.

Four human errors were incorporated into the ADS fault tree model. These errors are (1) failure to valve in the backup nitrogen supply, (2) sensor miscalibration, (3) failure to manually depressurize, and (4) ADS inadvertently inhibited.

4.6.4.6 Assumptions in the ADS Model

(1) Although the random independent hardware failure of a significant number of either the ADS safety/relief valves or the non-ADS safety/relief valves is felt to be negligible compared to other system failures, an event for the hardware failures of these valves is included. Common mode failure of the valves is also included.

(2) Failure of the operator to manually initiate the ADS and/or to manually depressurize the reactor vessel in order to achieve low pressure core cooling, are felt to be strongly coupled and are assumed to be the same event.

(3) Failure of the accumulator is included in the undeveloped event representing ADS valve hardware failure.

4.6.4.7 ADS Operating Experience

Nothing was peculiar in the operational history of the ADS which would affect either system modeling or failure data.

4.6.5 Condensate System

4.6.5.1 CDS Description

The function of the CDS system is to take condensate from the main condenser and deliver it to the reactor at an elevated temperature and pressure (event tree nomenclature--V1).

The CDS system consists of the condenser hotwell, three condensate pumps, feedwater heaters and associated piping, valves, and controls. The condenser hotwell has a working capacity of approximately 100,000 gallons. The condensate pumps provide the required head to overcome the flow and static resistance of the condensate system, and provide excess over the suction pressure requirements of the feedwater pumps. The reactor vessel must be depressurized to approximately 600 psig in order to use condensate as an injection source without the use of the feedwater pumps. Injection to the reactor vessel is via the two feedwater lines. The CDS pumps have a 10,870 gpm rated flow head. A simplified schematic of the CDS system is provided by Figure 4.6.5-1.

The CDS system is normally running.

The success criteria for the CDS system is removal of decay heat (when the reactor has tripped). This can be sufficiently accomplished with only one pump train operational.

Virtually all of the CDS system is located in the turbine building.

4.6.5.2 CDS Interface and Dependencies

The CDS system requires offsite power, instrument air and TBCW for operation. A simplified dependency diagram is provided in Figure 4.6.5-2. Shown are the major support needs for the CDS system as indicated by the solid diamonds.

4.6.5.3 CDS Test and Maintenance

The CDS system has no special test and maintenance requirements.

4.6.5.4 Technical Specifications

The CDS system has no specific technical specifications.

4.6.5.5 CDS Logic Model

The CDS system was modeled using a fault tree for injection of water at an elevated temperature and pressure to the reactor vessel. The fault tree model representing the CDS system is presented in Appendix B. The fault tree has been simplified to cover only the major active components, interfaces and dependencies.

The CDS pumps, feedwater heaters and condenser hotwell were not explicitly modeled since the system is normally running and considerable

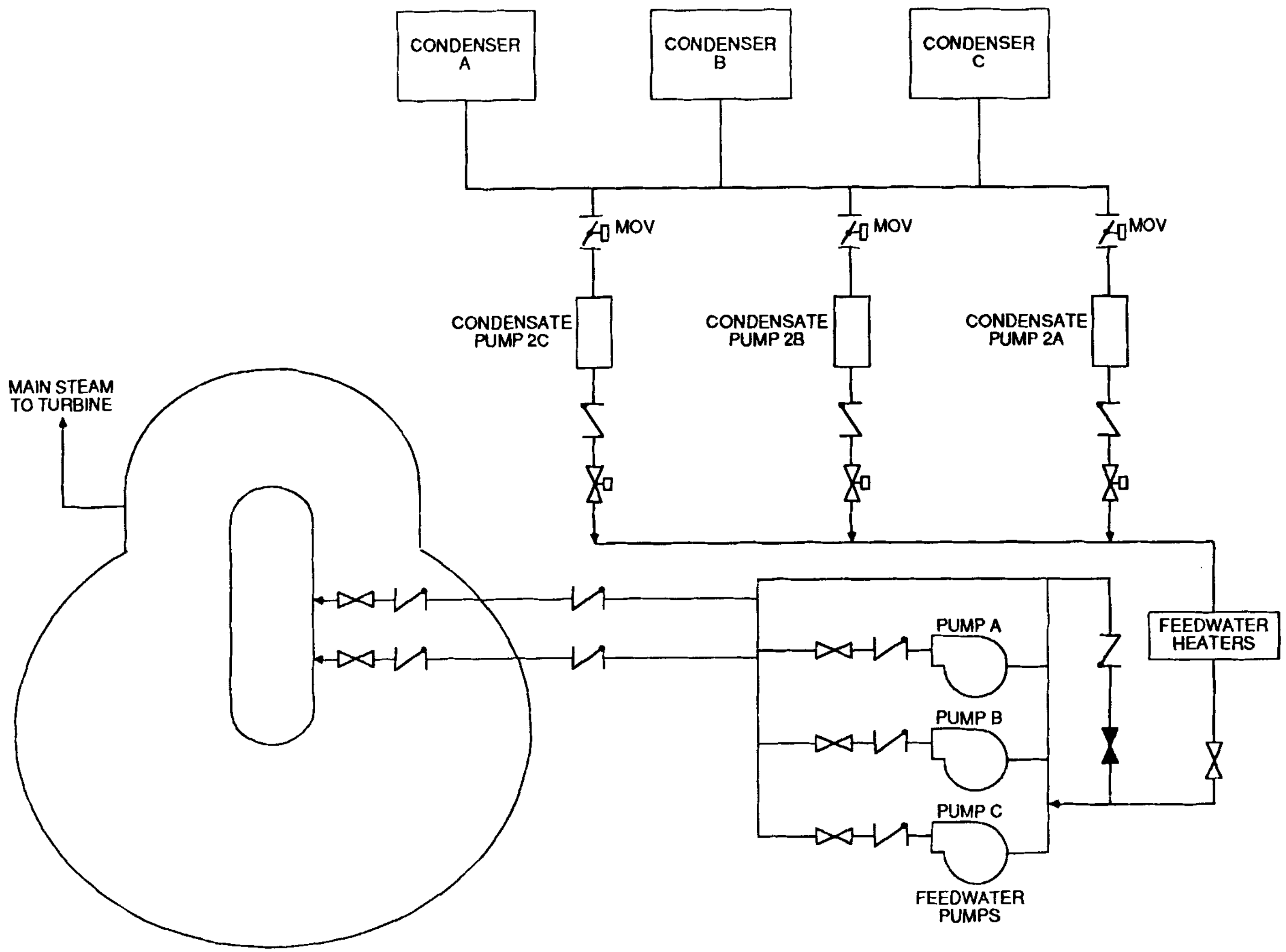
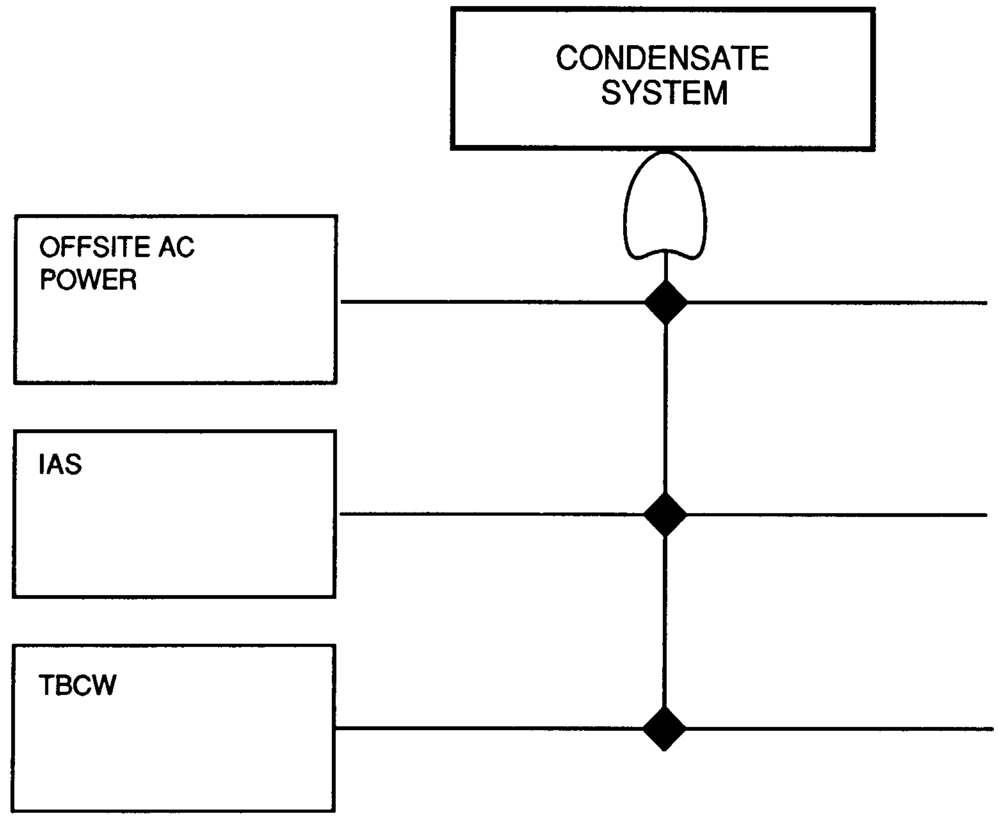


Figure 4.6.5-1. Condensate System Schematic.



Dependency Diagram Is Shown Using Failure Logic. Refer to the Fault Trees for Actual Failure Logic Details.

Figure 4.6.5-2. Condensate System Dependency Diagram.

redundancy exists (only need one of three trains working for success). All of the equipment hardware has been lumped into one event. The model focuses on the loss of the support systems as the most likely reasons that CDS would be lost.

4.6.5.6 Assumptions

Only major active components (lumped into one event) and major dependencies were modeled. These were assumed to dominate system failure.

4.6.5.7 CDS Operating Experience

There was nothing peculiar in the operational history of the CDS system which would affect system modeling.

4.6.6 Residual Heat Removal: Containment Spray System

4.6.6.1 CS Description

The function of the CS system is to suppress pressure in the drywell during accidents (event tree nomenclature--W3). The CS system is but one mode of the RHR system and, as such, shares components with other modes.

The RHR system is a two-loop system consisting of motor-operated valves and motor-driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm with a discharge head of 540 feet. Cooling water flow to the heat exchanger shell side is considered required for the CS mode. The CS suction source is the suppression pool. A simplified schematic of the CS (RHR) system is provided in Figure 4.6.6-1. Major components are shown as well as the pipe segment definitions (e.g., PS-25) used in the system fault tree with the CS portion highlighted.

The CS system is manually initiated and controlled and would be used if the LPCI mode (see 4.6.14) is not simultaneously required (i.e., LPCI is the preferred mode of RHR in accident situations).

The success criterion for the CS system is injection of flow from any one pump/heat exchanger train to the spray ring. For further information, refer to success criteria discussions in Section 4.4.

Most of the CS system is located in the reactor building. Local access to the CS system could be affected by either containment venting or failure. Room cooling failure is assessed to fail the CS pumps in ten hours (see Section 4.6.6.6).

4.6.6.2 CS Interfaces and Dependencies

Each CS pump is powered from a separate 4160 VAC bus with control and actuation power being supplied by a separate 125 VDC bus. All pumps require pump cooling. For further information on pump cooling, refer to Section 4.6.9.8. Each loop's normally closed spray valves receive motive

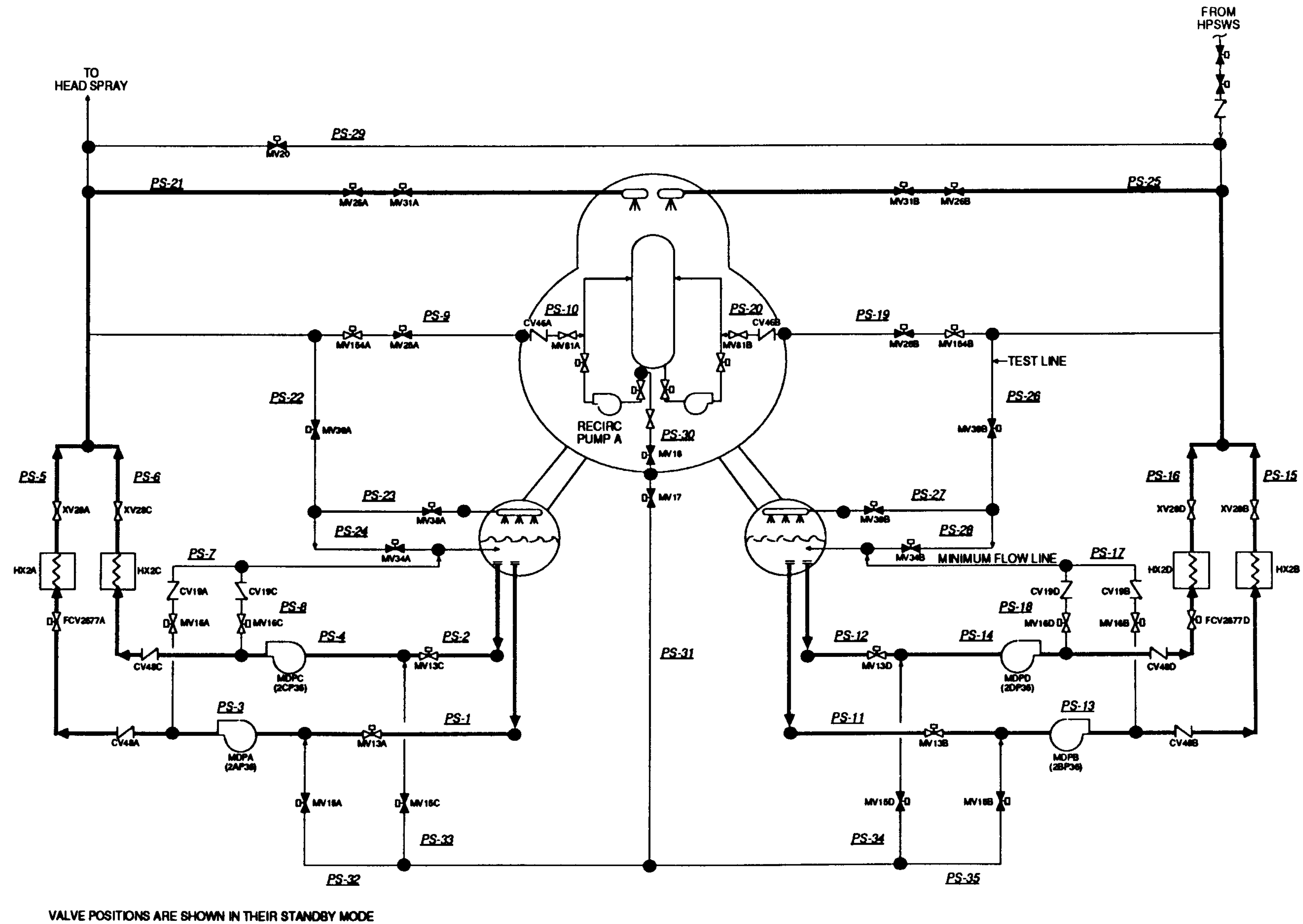


Figure 4.6.6-1. Containment Spray System Schematic.

power from one 480 VAC source. A simplified dependency diagram of the CS system is provided by Figure 4.6.6-2. Shown are the major support needs of the CS system as indicated by the solid diamonds.

Many components of the CS system are shared with the different modes of the RHR system. These commonalities are as follows: (1) the RHR pumps are common to the CS, SPC, LPCI, and SDC modes; (2) the suppression pool suction valve for each pump train is common to the CS, SPC, and LPCI modes; and (3) heat exchanger cooling is common to the CS, SDC, and SPC modes.

CS control circuitry is divided into two divisions. Division A is associated with control of components in Loop A, and Division B is associated with control of components in Loop B.

Reactor water level above the shroud (312 inches above vessel zero) and high drywell pressure (2 psig) permissive signals must be present before the CS system can be manually initiated. The water level signal can be overridden.

Although the CS has no isolation signals, there are permissives which will prevent the operation of certain components. CS pumps are demanded to stop or prevented from starting if the suppression pool suction valve or any of three SDC suction valves is not fully open.

4.6.6.3 CS Test and Maintenance

The CS surveillance requirements are the following: (1) pump operability---once/month, (2) MOV operability--once/month, (3) pump capacity test---once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.

4.6.6.4 CS Technical Specifications

Technical specifications exist based on sharing of the CS and LPCI modes. If any one LPCI pump or LPCI subsystem (i.e., loop A or B) is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the remaining LPCI components and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.6.5 CS Logic Model

The CS system was modeled using a fault tree for pressure suppression in the drywell. The major active components were modeled for the CS system. The fault tree model representing the CS system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to 1/3 of the main system piping was considered as a potential diversion path.

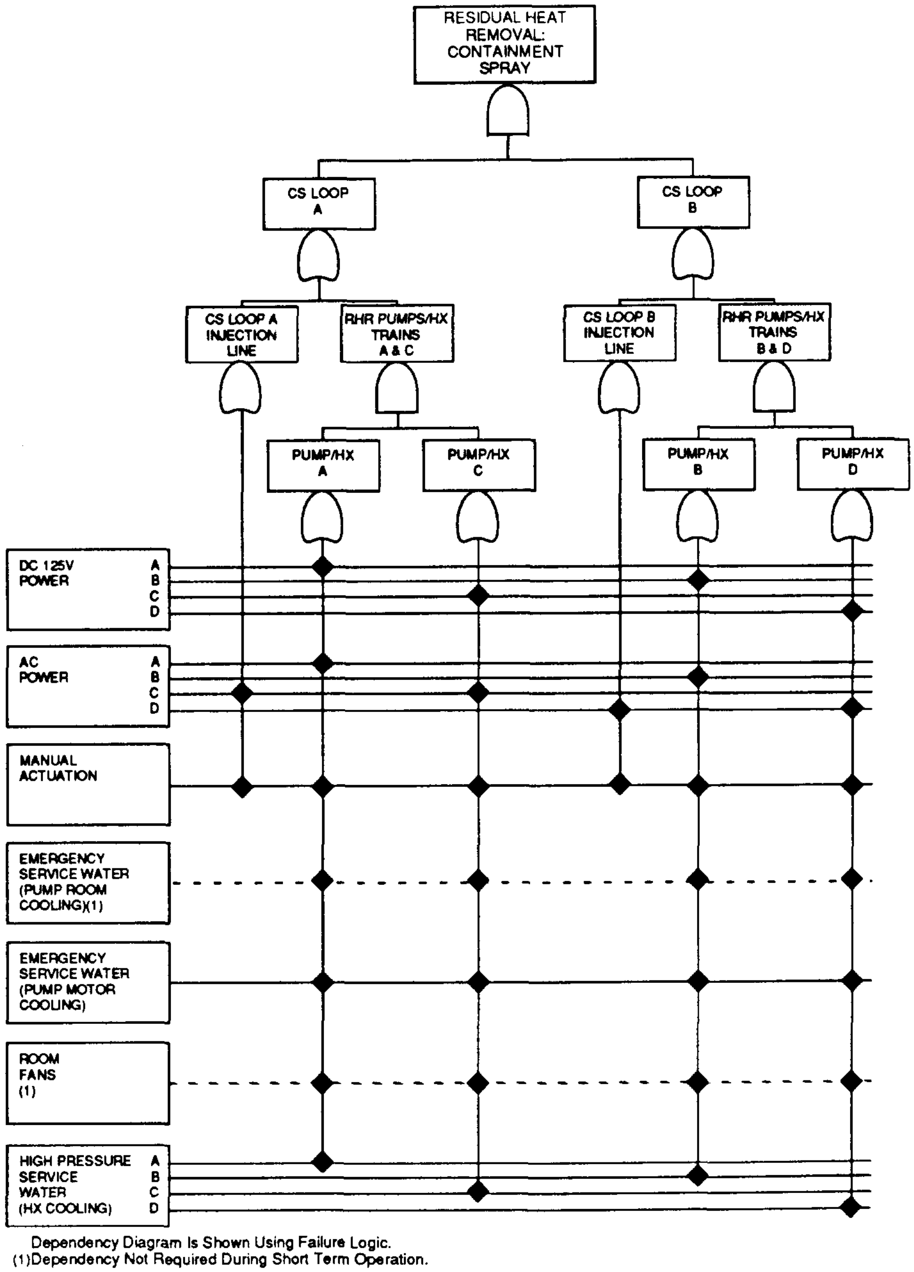


Figure 4.6.6-2. Containment Spray System Dependency Diagram.

Three human errors were incorporated into the CS fault tree model. These errors are failure of manual initiation, failure to override an erroneous shroud level permissive signal, and failure to properly restore key components following maintenance.

4.6.6.6 CS Assumptions

(1) Positions of all manual and motor-operated valves are indicated in the control room. Failure of these valves after testing and maintenance due to incorrect positioning is therefore felt to be negligible. The injection valves receive open signals on a real demand. Thus, unavailability from testing and failure to restore after testing is not important.

(2) During construction of the fault tree, it was necessary to determine which components could be taken Out Of Service (OOS) for maintenance. It was assumed that maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(3) Pump isolation because of spurious signals is assumed to be negligible compared to other system faults.

(4) The CS control circuitry was not modeled at a great level of detail. Only elements which were felt to be potentially important were included in the fault tree model. Except for the shroud water level permissive, high drywell pressure permissive, pump power permissive, and pump suction source relay, the hardware failures of relays and permissives are grouped into one term. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems.

(5) Based on a PECO response, it is assessed that the CS pumps will fail because of insufficient Net Positive Suction Head (NPSH) once the suppression pool has reached saturated conditions.

(6) Diversion of flow to the suppression pool is felt to be negligible compared to other system failures.

(7) A suction path must be available from either the suppression pool or the SDC path to start a CS pump.

(8) Failure of the suppression pool because of random failure or the plugging of all its strainers is assumed to be negligible compared to other system failures.

(9) The unavailability of the CS pumps due to testing does not defeat a real demand from operating the system. Therefore, it was not considered. Failure to restore the CS pumps after testing does not apply.

(10) Failure of room cooling (if not recovered) is assessed to fail CS in ten hours. This is based on utility calculations [52] which demonstrate that for approximately 50 hours or more without room cooling, operability is expected even with continuous pump operation. The ten hour CS failure value was chosen to be consistent with the general assumptions made for HPCI and RCIC (see Section 4.6.11). It is believed to be a conservative value.

4.6.6.7 CS Operating Experience

Nothing was peculiar in the operational history of the CS system which would affect either system modeling or failure data.

4.6.7 Control Rod Drive System--Enhanced and One Pump

4.6.7.1 CRD Description

The CRD system was modeled as a backup source of high pressure injection, event tree nomenclature--U3 (CRD Enhanced Mode--2 pumps required) and U4 (CRD-1 pump required).

The CRD pumps take suction from the condenser hotwell in the Condensate system or the Condensate Storage Tank (CST). A flow control station is installed downstream of the tap from the Condensate system and ties into the CRD pump suction line before the CRD suction filter. The flow control station will divert 250 gpm from the Condensate system. This will supply the CRD system with the remainder of the water being passed on to the CST. In the event that flow from the Condensate system is interrupted, the CST provides a backup source of water to ensure CRD system operability without operator action being required. A simplified schematic of the CRD system is provided by Figure 4.6.7-1.

The CRD pumps, together, can achieve a flow rate of approximately 210 gpm with the reactor fully pressurized and approximately 300 gpm with the reactor depressurized. Two discharge paths are provided for the CRD pumps. One discharge path is through an air-operated valve control station. When instrument air is lost, this path is closed. With both CRD pumps running and the reactor at nominal pressure, the second discharge path restricts flow, by means of an orifice, to approximately 180 gpm.

Normally one CRD pump is running, with the suction and discharge valves to the standby pump closed. Should the operator be required to realign the CRD system as a sole source of early high pressure injection, the standby CRD pump must be placed into operation to achieve sufficient flow to the reactor vessel.

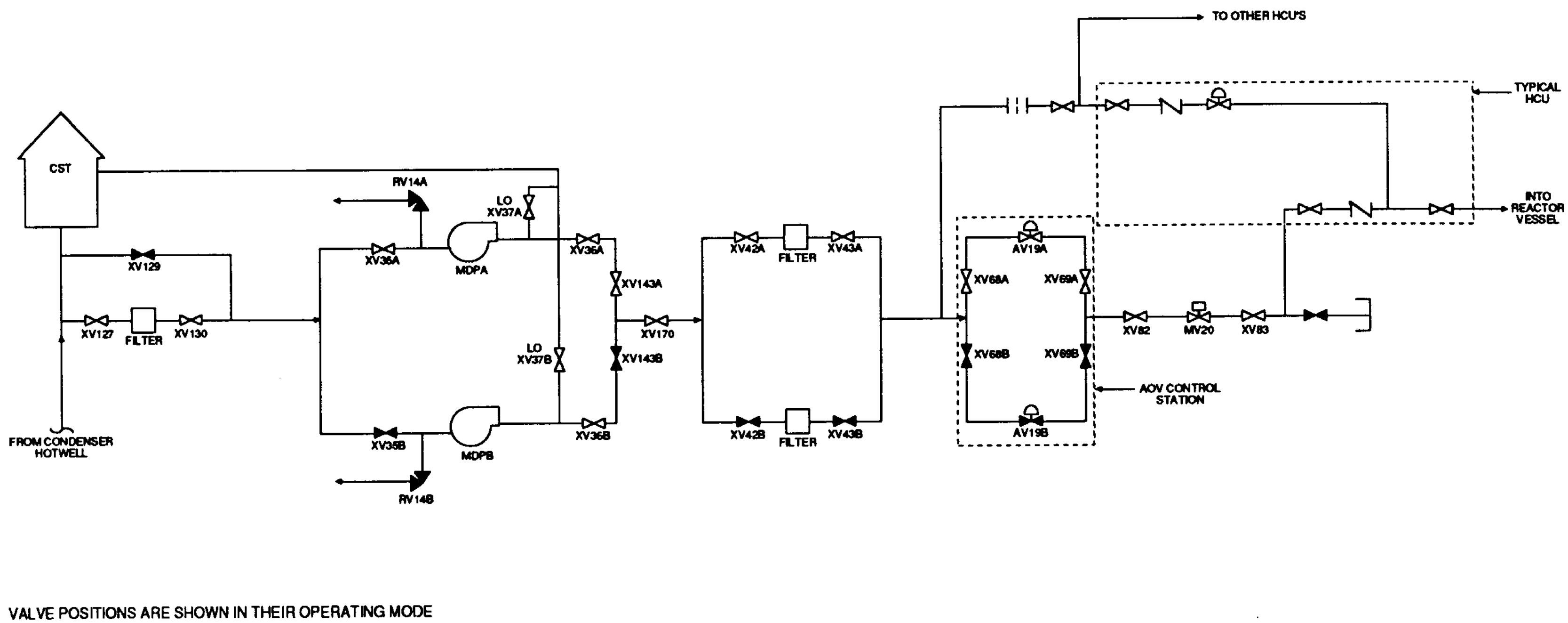


Figure 4.6.7-1. Control Rod Drive System Schematic.

In general, the CRD success criteria (as a sole injection source to the reactor) requires both pumps running and one of the two discharge paths available. If some other injection system has been operating successfully for ~6 or more hours following an initiator the CRD success criteria changes to one pump running and one of two discharge paths available. For further information, refer to success criteria discussions in Section 4.4.

Most of the CRD system (except for piping and a few valves) is located in the turbine building. Any physical impact of accident conditions on the ability of the CRD system to perform its function would be minimal. Since the system is located in a large open area, room cooling failure is not applicable to the CRD pumps.

4.6.7.2 CRD Interfaces and Dependencies

CRD Pump A is powered from 4160 VAC/A with control and actuation power supplied by 125 VDC/A. CRD Pump B is powered from 4160 VAC/D with control and actuation power supplied by 125 VDC/D. A simplified dependency diagram of the CRD system is provided by Figure 4.6.7-2. Shown are the major support needs for achieving full flow operation of the CRD system as indicated by the solid diamonds.

The CRD pumps receive no automatic initiation signals.

The CRD pumps are normally cooled by the TBCW system. If the TBCW is lost, cooling is performed by the RBCW system, which is automatically or manually transferred.

4.6.7.3 CRD Test and Maintenance

No specific CRD (in the high pressure injection mode) test and maintenance requirements are identified in the Peach Bottom technical specifications.

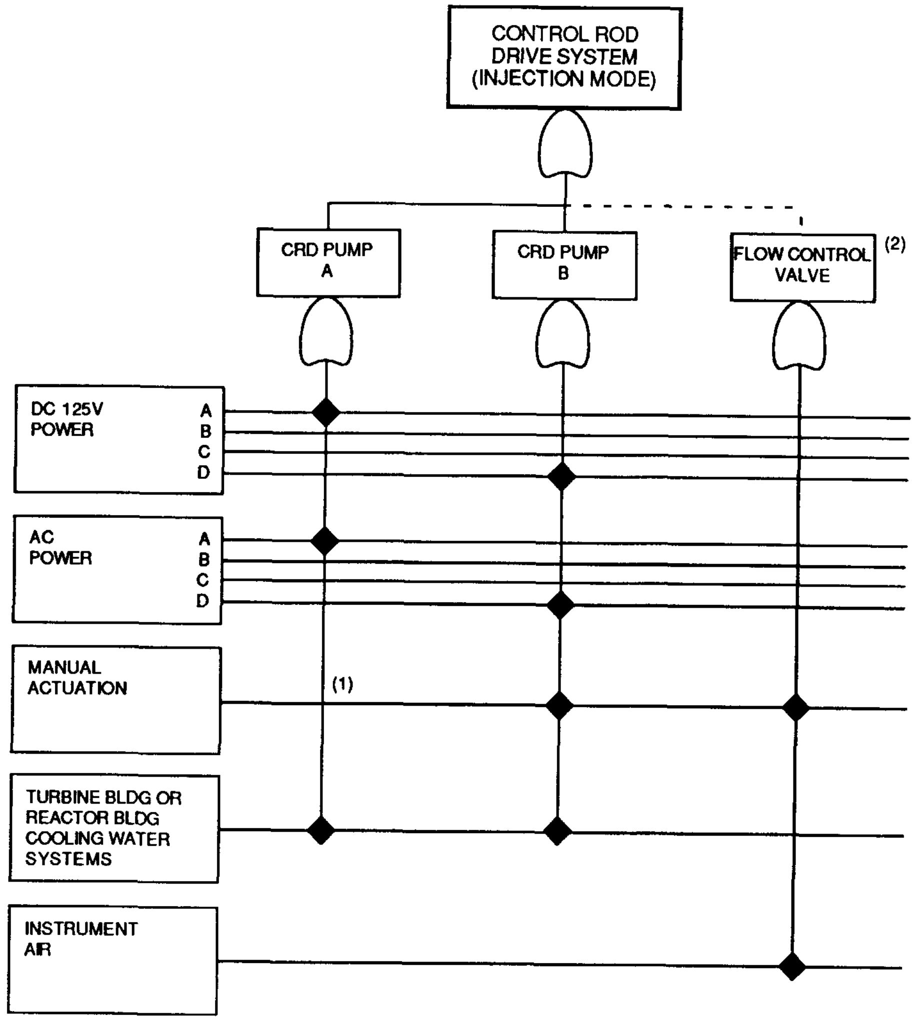
4.6.7.4 CRD Technical Specifications

No reference is made to the CRD high pressure injection mode in the Peach Bottom technical specifications.

4.6.7.5 CRD Logic Model

The CRD system was modeled using two fault trees for its high pressure injection mode. The enhanced mode fault tree has as its success criteria both pumps working. The success criteria for the one pump operation fault tree is one pump operational after ~6 or more hours.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.



Dependency Diagram Is Shown Using Failure Logic.
(1) CRD Pump A Normally Operating.
(2) Failure Of Flow Control Valve Does Not Fail Injection To Reactor Via CRD Because An Alternate Flow Path Is Available.

Figure 4.6.7-2. Control Rod Drive Dependency Diagram.

4.6.7.6 CRD Assumptions

(1) Pipe segments less than one third of the main system pipe diameter are not considered to be diversion paths.

(2) The orificed discharge path provides sufficient flow for successful high pressure injection as evidenced by the LTAS computer runs for Peach Bottom (See Appendix A).

(3) The test mode of the CRD system would place the system in a "run" configuration. Therefore, the unavailability of the system from testing is inapplicable. The same reasoning applies for a failure to restore the system after testing.

(4) The position (open or closed) of the train B valves do not affect a failure to restore the system after maintenance. However, maintenance staff could leave a breaker out of the circuit thereby defeating Pump B's ability to start. This has been addressed in the fault tree.

4.6.7.7 CRD Operating Experience

Nothing was peculiar in the operational history of the CRD system which would affect either system modeling or failure data.

4.6.8 Electric Power System

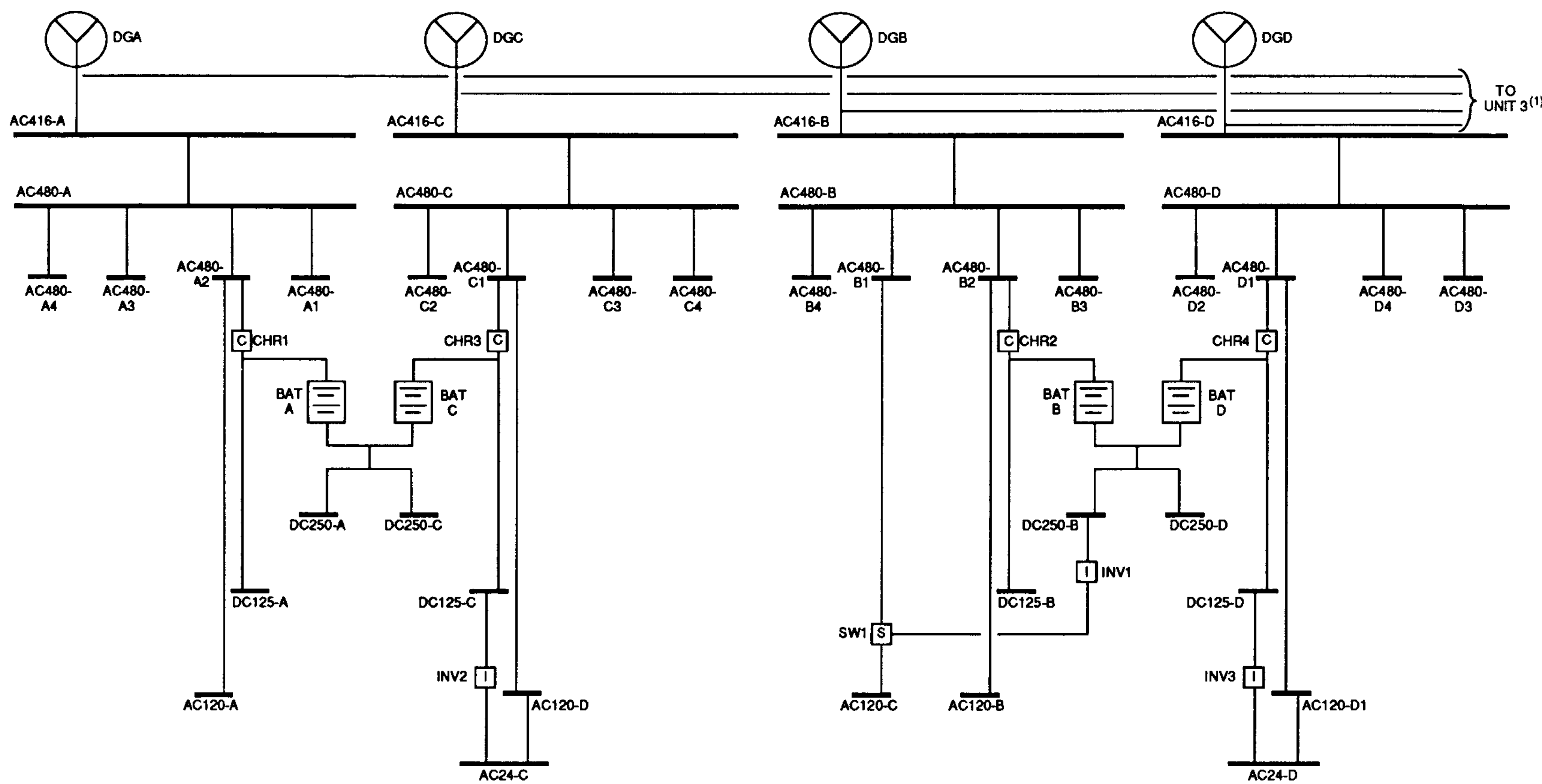
4.6.8.1 EPS Description

The EPS is designed to provide a diversity of dependable power sources which are physically isolated from each other.

The Peach Bottom station receives power from two separate offsite sources. If both offsite sources are lost, auxiliary power is supplied to both Unit 2 and Unit 3 from four onsite diesel generators shared between the two units. Loads important to plant safety are split and diversified. Station batteries provide control power for specific engineered safeguards and for other required functions when AC power is not available. A simplified schematic of the EPS is provided by Figure 4.6.8-1.

Each diesel generator unit consists of a diesel engine, a generator, and the associated auxiliaries mounted on a common base. The continuous rating of the diesel generators is 2600 kW. The engine is rated for a ten percent overload for any two of every twenty-four hours.

There are two independent 125/250 VDC systems or divisions per unit. Each division is comprised of two 125-V batteries, each with its own charger (i.e., each unit has four 125-V batteries). Each 125-V battery is a lead-calcium type with 58 cells. The chargers are full wave, silicon-controlled rectifiers. The two batteries for each unit are redundant. Loads are diversified between these systems so that each system serves loads which are identical and redundant. Power for larger



(1) GOES TO UNIT 3 BUSES (DG A, B, C, AND D ARE ARE SHARED BETWEEN UNITS 2 AND 3).

Figure 4.6.8-1. Electric Power System Schematic.

loads, such as DC motor-driven pumps and valves, is supplied at 250-V from two 125-V sources. Selected batteries from Unit 2 and from Unit 3 are needed to start Diesel Generators 1, 2, 3 and 4, respectively.

Each standby diesel generator automatically starts. The diesel generator may be stopped by the operator after determining that continued operation of the diesel is not required.

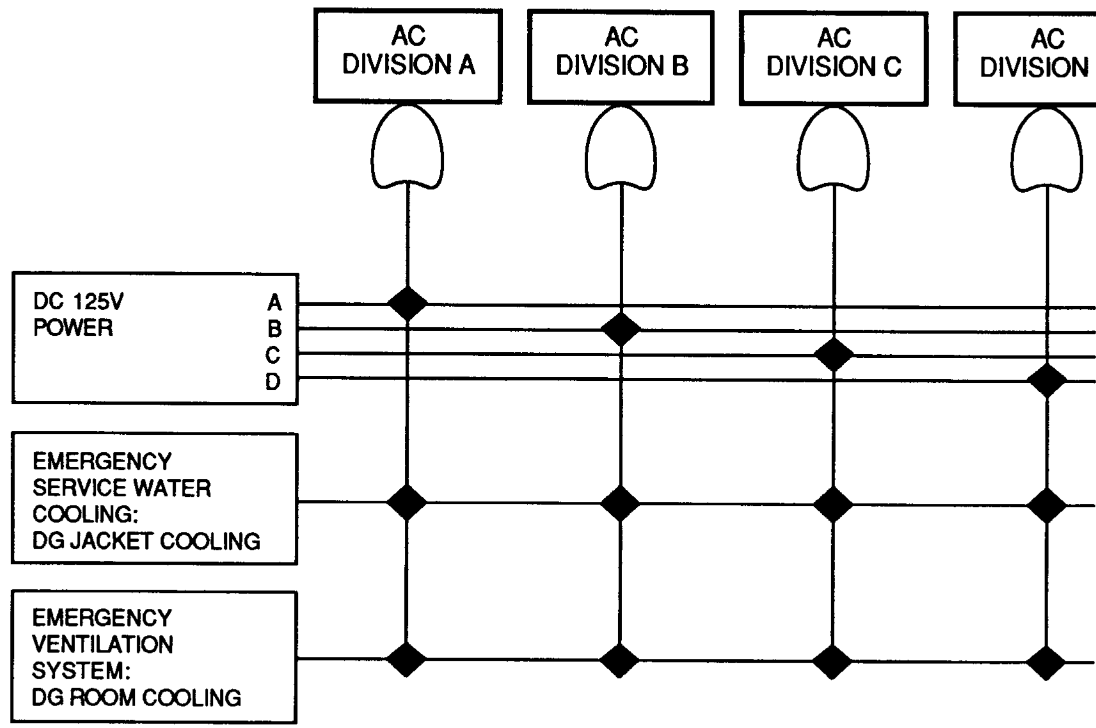
Most of the EPS is located in the diesel building and in compartmentalized rooms within the reactor building. Any physical impact of accident conditions on the ability of the EPS to perform its function would be minimal. It is assumed that room cooling is not required for the AC switchgear or DC battery rooms since the heat loads are small and no sizeable heat loads are near these rooms. Diesel generators are assumed to fail in less than 30 minutes without room cooling although it is recognized that diesel performance would degrade before actual failure of the diesel and provide a warning to the operators that a problem existed. Possible recovery actions (by opening doors) could therefore take place. Complete failure of the EPS would cause a station blackout. After a total loss of AC power, DC-driven components could operate until the station batteries are depleted (estimated at about 12 hours based on PECO input, see Section 4.12).

4.6.8.2 EPS Interfaces and Dependencies

Each standby diesel generator automatically starts on total loss of offsite power, low reactor water level, or high drywell pressure coincident with low reactor pressure. Two sources of offsite power are available to each 4-kV emergency bus. The failure of one offsite power source results in the automatic transfer to the other offsite source. When the diesel generators are demanded, essential loads are automatically sequenced onto the emergency bus. Nonessential 480 V loads are prevented from being automatically sequenced. Each diesel generator can be started locally, but can be electrically connected to its bus only from the main control room. A simplified dependency diagram of the EPS is provided by Figure 4.6.8-2. Shown are the major support needs of the EPS as indicated by the solid diamonds.

The diesel generator circuit breaker is tripped by protective devices under the following abnormal conditions: (1) engine overspeed, (2) jacket coolant high temperature, (3) jacket coolant low pressure, (4) lube oil high temperature, (5) lube oil low pressure, (6) crank case high pressure, (7) after-cooler coolant low pressure, (8) fuel oil low pressure, and (9) carbon dioxide fire extinguishing system discharge. Protective tripping of the diesels is announced in the main control room and locally at the unit. A two-out-of-three tripping logic prevents spurious trips of the diesels. These protective trips are overridden on a Loss of Coolant Accident (LOCA) signal.

Both the control and power battery systems operate ungrounded, with a ground detector alarm in the main control room.



Dependency Diagram Is Shown Using Failure Logic.

Figure 4.6.8-2. Electric Power System Dependency Diagram.

4.6.8.3 EPS Test and Maintenance

When it is determined that one diesel generator is inoperable, the other diesel generators are to be demonstrated operable immediately and daily thereafter. The diesel generators are tested by starting each generator every week. During these tests the starting air compressor, diesel fuel oil transfer pumps, and diesel starting time are checked. The diesel is started and brought up to full speed while isolated from its loads. Since the auto sequencing is turned off during the test, and so would not automatically operate, test unavailability was modeled. Once per operating cycle, the condition under which the diesel generator is required will be simulated. This test demonstrates that the diesel will start and accept the emergency load within a specified time sequence. Each diesel generator is given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

Unit batteries' specific gravity, voltage and temperature of the pilot cell, and overall battery voltage are measured weekly. Every three months, the voltage and specific gravity of each cell are checked while the battery is still floating on the bus. This test also includes temperature measurement of at least every fifth cell. Once per operating cycle, unit batteries are load discharge tested. Experience at Peach Bottom demonstrates that battery checks are staggered using different personnel to examine redundant battery trains.

4.6.8.4 EPS Technical Specifications

During any period when one diesel generator is inoperable, continued reactor operation is permissible for seven days if the remaining diesel generators are operable. If this requirement is not met, the reactor is to be placed in a cold shutdown condition within twenty-four hours. During any period when one 125-V battery system is inoperable, continued reactor operation is permissible during the succeeding three days.

The reactor cannot be taken critical unless all of the following conditions are satisfied: (1) both offsite sources and startup transformers are available and capable of automatically supplying power to the 4-kV emergency buses, (2) the 4 diesel generators are operable with a minimum of 104,000 gallons of diesel fuel on site, (3) the 4-kV emergency buses and the 480 V emergency load centers are energized, and (4) the 125-V batteries and their chargers are operable.

4.6.8.5 EPS Logic Models

The EPS was modeled using fault trees for its AC and DC power portions. Only the major buses and power sources were modeled in the fault trees. One human error, failure to restore the diesel systems after test or maintenance, was incorporated into the fault tree model. Human/EPS interactions were considered part of the recovery analysis. The fault tree model representing the EPS is presented in Appendix B.

4.6.8.6 EPS Assumptions

(1) A simplified lumped AC model is used. This is judged to be adequate since the failure of all AC buses is dominated by diesel generator failures.

(2) All valves powered from 480 V Motor Control Center (MCC) buses take their control power from the 120 V control bus associated with the same MCC bus.

(3) No safety load is connected to 120 VAC Buses 20Y33, 20Y34, 20Y35, 20Y50, and 00Y03 with the exception of accident monitoring sensors. The accident monitoring sensors are powered by 24 VAC buses.

(4) If an AC bus from Unit 3 is used by modeled equipment, the comparable bus from Unit 2 is used instead. Since the same diesel generator feeds the same emergency AC buses of both units, it is very likely that failure of one bus in Unit 3 is followed by failure of the similar bus in Unit 2.

(5) If a DC bus from Unit 3 supplies modeled equipment, the battery is assumed to be the sole source of power for that component.

(6) Short circuit faults and the potential effects of fault propagation are not modeled.

(7) Loss of ventilation can affect the diesel generators, but not the emergency switchgear or batteries as previously indicated.

(8) Unavailability of the diesels during tests is based on engineering judgment assuming that the diesels are unavailable approximately one hour during each test and that each diesel experiences an average of twenty tests per year.

4.6.8.7 EPS Operating Experience

The operational history of the Peach Bottom diesel generators justifies using plant specific failure data. In particular, operational data since 1980 indicate the diesels at Peach Bottom are achieving a much better reliability than the industry average.

4.6.9 Emergency Service Water System

4.6.9.1 ESW Description

The function of the ESW system is to provide a reliable supply of cooling water to selected equipment during a loss of offsite power.

The ESW system is common to both Units 2 and 3. The system has two full capacity pumps installed in parallel. The normal water supply to the suction of the ESW pumps is from Conowingo pond. The pump discharge consists of two headers with service loops to the diesel-engine coolers and selected equipment coolers. The modeled components supplied with cooling water are the LPCS pumps and pump room coolers, the RHR pumps and pump room coolers, the HPCI pump room cooler, and the RCIC pump room cooler. Valves in the supply headers provide loop isolation. A common discharge header directs effluent to Conowingo pond. A simplified schematic of the ESW system is provided by Figure 4.6.9-1. Major components are shown as well as the pipe segment definitions (e.g., PS-8) used in the system fault tree.

The ESW pumps are vertical, single-stage, turbine types with an 8000 gpm capacity. Their normal discharge head is 96 ft and their shutoff head is 132 ft.

The cooling for all modeled equipment, with the exception of the diesel generator coolers, is normally provided by the Normal Service Water (NSW) system which operates on offsite AC power only.

Should the preferred flow paths described above be unavailable or the bay level preclude normal flow path operation, the ESW system may also be operated in conjunction with the Emergency Heat Sink (EHS) in a closed or open loop fashion. In the closed loop mode, two ESW booster pumps take return water from various coolers, boost it in pressure, and deliver the water to the emergency cooling tower structure. The booster pumps are horizontal split types, with 8000 gpm flow at a head of 100 psig. One Emergency Cooling Water (ECW) pump then takes suction from the cooling tower structure. It delivers water through a motor-operated gate valve to the ESW heat loads. The ECW pump and motor are identical to those of the ESW pumps. The only difference between the ECW pump and the ESW pumps is pump column length. While the booster pumps would normally be used in this mode, they are not required since it has been demonstrated by recent tests that booster pump failure will not fail the cooling function of the ESW. In the open loop mode, the ECW pump delivers water from the cooling tower structure, through the ESW loads, and back to the bay. There is sufficient water supply in the cooling tower structure to last for days; hence the open loop mode is considered a success path.

Upon system automatic initiation, the operator checks discharge pressure for the two primary ESW pumps. If discharge pressure appears normal, the operator turns off one ESW pump at his discretion (i.e., he may not do this right away, but instead shut the pump off some time later). He also shuts down the ECW pump (the ECW pump also has an automatic trip in ~45 seconds if the discharge pressure is adequate). At some later time, if the operating ESW pump trips and the standby ESW pump fails to start, the operator must manually start the ECW pump. In the EHS closed loop mode, cooling tower fans must be manually started.

The success criterion for the ESW system is either of the ESW pumps or the ECW pump supplying cooling water to system heat loads.

Most of the ESW system is located in pump rooms external to the reactor and turbine buildings. Any physical impact of accident conditions on the

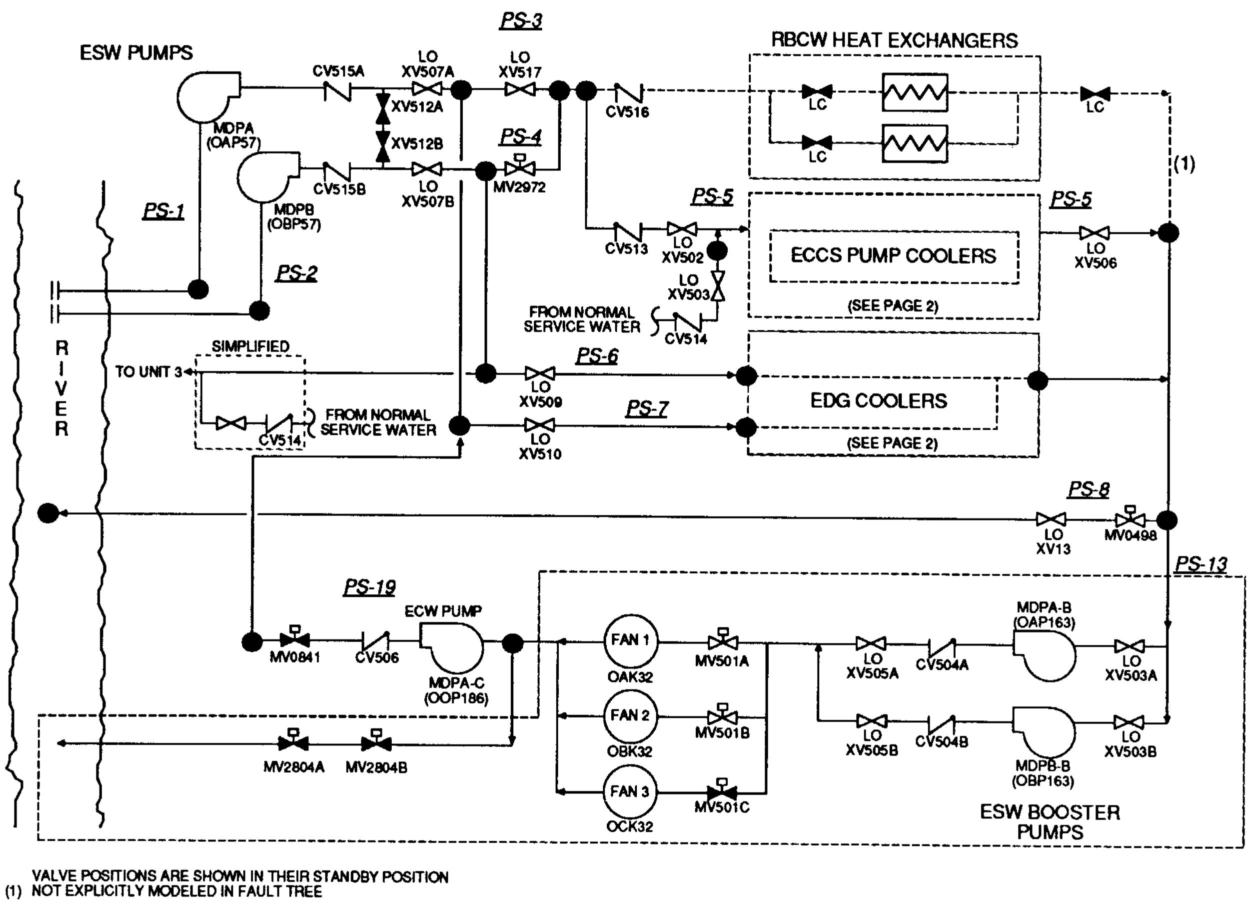
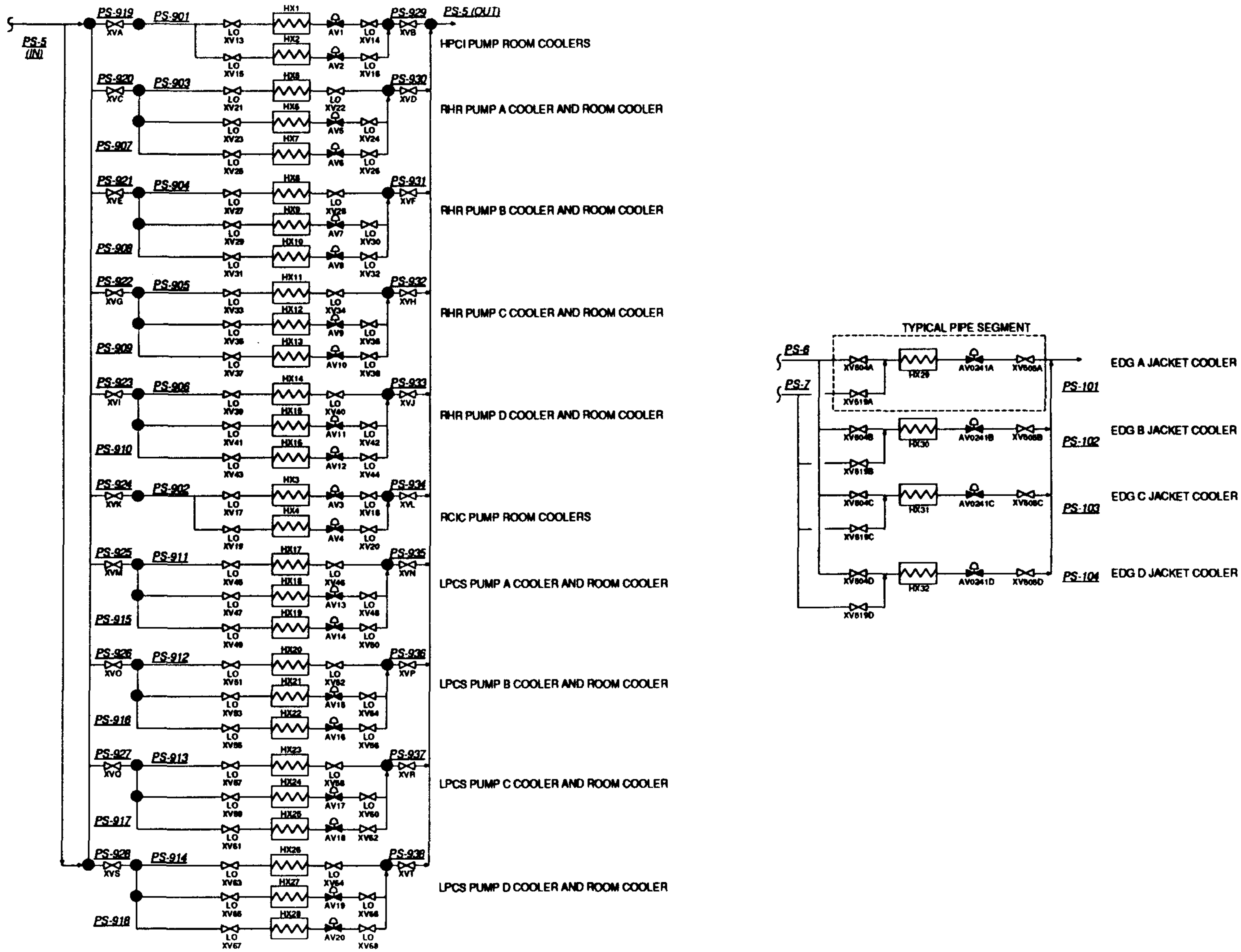


Figure 4.6.9-1. Emergency Service Water System Schematic.
(Page 1 of 2)



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.9-1. Emergency Service Water System Schematic.
(Page 2 of 2)

ability of the ESW system to perform its function would be minimal. Room cooling failure is assumed not to fail the ESW pumps, ESW booster pumps, and ECW pump.

Failure of the ESW system would quickly fail operating diesel generators and potentially fail the LPCS pumps and RHR pumps. The HPCI pumps and RCIC pumps would fail by a loss of their room cooling ten hours after a loss of the ESW system if other recovery actions were not taken.

4.6.9.2 ESW Interfaces and Dependencies

The ECW pump, ESW booster pumps, and ESW pumps are all self-cooled. ESW pump A and ESW booster pump A are powered from 4160 VAC/B with control and actuation power supplied by 125 VDC/B. ESW pump B and ESW booster pump B are powered from 4160 VAC/C with control and actuation power supplied by 125 VDC/C. The ECW pump is powered from 4160 VAC/D with control and actuation power supplied by 125 VDC/D. A simplified dependency diagram of the ESW system is provided by Figure 4.6.9-2. Shown are the major support needs for the ESW system as indicated by the solid diamonds.

Cooling tower fans are shared with the HPSW system. These fans are used in the EHS closed loop mode should the normal bay level be either too high or too low.

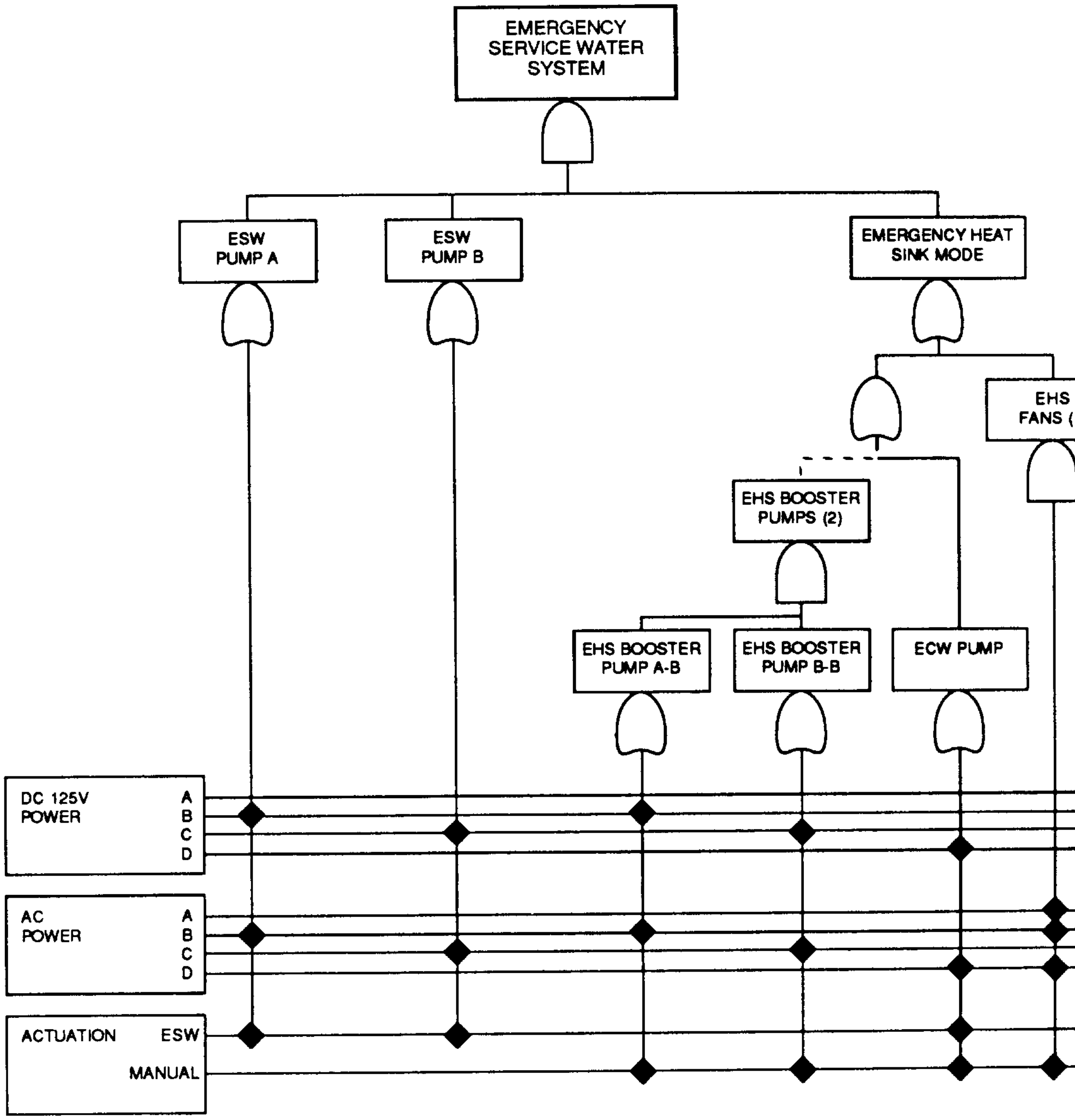
Both ESW pumps and the ECW pump start on a diesel start signal or a LOCA signal (low water level/high drywell pressure). If all three pumps start successfully, the operator will shut off one ESW pump and the ECW pump will automatically shut down as described above. If the running ESW pump fails, the other ESW pump will receive an auto start signal on low discharge pressure.

When both an ESW pump low discharge pressure signal and a diesel generator auto start signal occur, after a 30 second delay, the ECW pump discharge valve MV0841 opens.

For the closed loop mode, if an emergency cooling tower fan fails to start or trips on high vibration, its associated inlet valve automatically closes. High vibration alarms actuate in the control room.

4.6.9.3 ESW Test and Maintenance

The ESW system is tested once every three months as follows: (1) pump operability--the pump is manually started and flow capability checked and (2) valve operability--the automatic valves are stroked individually from their control switches. The associated pump room fans are tested for operability every three months. The ECW pump, ESW booster pumps and emergency cooling tower fans are tested once per operating cycle to verify operability. Because of diesel generator test requirements, the ESW system is realistically tested more often (~weekly).



Dependency Diagram Is Shown Using Failure Logic.
(1) Need In EHS Closed Loop Mode Only.
(2) Not Really Required To Operate; Shown For Information Only.

Figure 4.6.9-2. Emergency Service Water System Dependency Diagram.

4.6.9.4 ESW Technical Specifications

The ESW system shall be operable at all times when the reactor coolant temperature is greater than 212°F. If two ESW pumps become inoperable, the reactor may remain in operation for a period not to exceed one month. To consider the ECW pump operable as an equivalent ESW pump, at least one ESW booster pump and two emergency cooling tower fans must be operable. To consider the ESW pump operable, the associated pump room fans must be available for normal operation except that (1) one pump room supply and/or exhaust fan for each compartment may be out of service for one month or (2) temporary fans may be used in place of permanently installed fans to provide room temperatures of less than 120°F.

4.6.9.5 ESW Logic Models

The ESW system was modeled using fault trees for both its normal heat removal mode and its EHS open loop mode. The EHS closed loop mode was not modeled. The major active and some passive components were modeled for the ESW system. The fault tree model representing the ESW system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

Two human errors were incorporated into the ESW fault tree model. These errors are (1) operator failure to restart the ECW pump should the preferred path have a delayed failure and (2) operator failure to restore equipment properly after maintenance.

4.6.9.6 ESW Assumptions

(1) The ESW pumps do not require room cooling. These pumps, which are pumping cold water, are located in the service water pump structure which is a large building. By opening the door (which is not likely to be required) adequate cooling can be provided.

(2) The cross-tie valves between the two ESW pumps are not modeled. Each pump feeds into a common header; therefore, the cross-tie does not have significant impact on the dominant failure modes of the system. The only time the cross-tie is important is when manual valve 507A plugs and ESW Pump B (OBP57) fails or manual valve 507B plugs and ESW Pump A (OAP57) fails. These failures are judged to be negligible compared to the failure of both pumps.

(3) Diesel generator EDGA, EDGB, EDGC and EDGD jacket cooling failures, by means of one header failing because of valve plugging and the other because of ESW pump failure, are not modeled. This simplification was made since the likelihood of a manual valve's plugging and a pump's failing is insignificant compared to two pump failures.

(4) A system initiation signal starts both ESW pumps and the ECW pump. The operator shuts off one ESW pump and the ECW pump after checking discharge pressure. Failure of the operator to trip the two pumps is not considered a system failure mode.

(5) Cooling for the ECCS pump rooms is provided by fan cooling units. Operation of both the fan and coolant flow through the coil is needed for cooling the room.

(6) All of the air-operated valves in the ESW system fail open on loss of air.

(7) Both fan-coil units for each pump room receive the same operational signal and are supplied from the same power source.

(8) Test unavailability or failure to restore following test are not considered for the ESW system. Tests of the system typically involve simple start-up of the equipment such that little reconfiguration of the system has to be performed.

(9) No need for makeup is modeled for the EHS mode. This assumption is made because the amount of evaporation in the emergency cooling towers is expected to be low.

(10) Plugging of the strainers in the service water pump bay is considered insignificant. Since the NSW pumps are normally operating in the same bay, plugging of strainers would be easily detected prior to ESW operation. Plugging of strainers during ESW operation is considered very small since it would have to happen within minutes. After a few minutes, the EHS mode may be initiated.

(11) Closure of valve MV-0498 is virtually never expected. The valve has been placed in the open position with its wiring removed so that water flow will always be in the open loop mode.

4.6.9.7 ESW Operating Experience

Nothing was peculiar in the operational history of the ESW system which would affect either system modeling or failure data.

4.6.9.8 ESW Special Issues

There is one controversial issue regarding the need for ESW. That issue involves whether or not the LPCS/RHR pumps really require ESW cooling. PECO has stated that these pumps are designed to operate with working fluid temperatures approaching 160°F without pump cooling. This implies that in scenarios where the ESW system has been lost, these pumps could still operate; some RHR pumps would be placed in the suppression pool cooling mode and therefore keep the working fluid at less than 160°F. It is felt that there is significant validity to these arguments. However, because it is uncertain whether the suppression pool water can be maintained below 160°F in some sequences and whether PECO has properly accounted for pump heat addition to the system, the base case analysis assumes these pumps will fail upon loss of ESW cooling.

4.6.10 Emergency Ventilation System

4.6.10.1 EVS Description

The objective of the EVS is to maintain suitable temperatures in equipment rooms to preclude component failures.

The EVS cools the following: (1) standby diesel generator rooms, (2) pump structure service water pump rooms, and (3) pump rooms for the RHR, RCIC, HPCI and LPCS pumps. The pump rooms use small individual fan coolers in each room. A simplified schematic of the EVS is provided by Figure 4.6.10-1. Major components are shown as well as the pipe (duct) segment definitions (e.g., PS-4) used in the system fault tree.

The service water pumps, emergency switchgear, and battery rooms are assumed not to require room cooling. Pump room cooling loss for the RHR, RCIC, HPCI, and LPCS pumps is incorporated into the ESW and individual system models. Therefore, the EVS system model does not include ESW, RHR, RCIC, HPCI, and LPCS pump room cooling.

Each standby diesel generator room is provided with ventilation air supply fans and an exhaust relief damper. Diesel generator room cooling requires operation of one of two supply fans. Any physical impact of accident conditions on the ability of the EVS to perform its function would be minimal. It is estimated that failure of the EVS would fail operating diesel generators in less than 30 minutes. In actuality, the diesel may not fail, but a load drop is still likely.

4.6.10.2 EVS Interfaces and Dependencies

The standby diesel generator room fans are powered from their respective diesels. A simplified dependency diagram of the EVS is provided by Figure 4.6.10-2. Shown are the major support needs for the EVS as indicated by the solid diamonds.

Diesel Generator Room Fans 7, 9, 11, and 13 outside air supply dampers, AV25, AV28, AV31, and AV34, open on 65°F fan discharge temperature and fail open on a loss of instrument air. Diesel Generator Room Fans 7, 9, 11, and 13 room air supply dampers, AV26, AV29, AV32, and AV35, close on 65°F fan discharge temperature and fail closed on a loss of instrument air. Dampers AV27, AV30, AV33, and AV36 open on Fans 7, 9, 11, and 13, starting signals respectively and fail open on a loss of instrument air. Fans 7, 9, 11, 13 automatically start on a diesel generator actuation signal. Fans 8, 10, 12, and 14 automatically start on an automatic start signal of Fans 7, 9, 11, and 13 respectively. Diesel generator room supply fans trip on a carbon dioxide discharge signal except when a LOCA signal is already present.

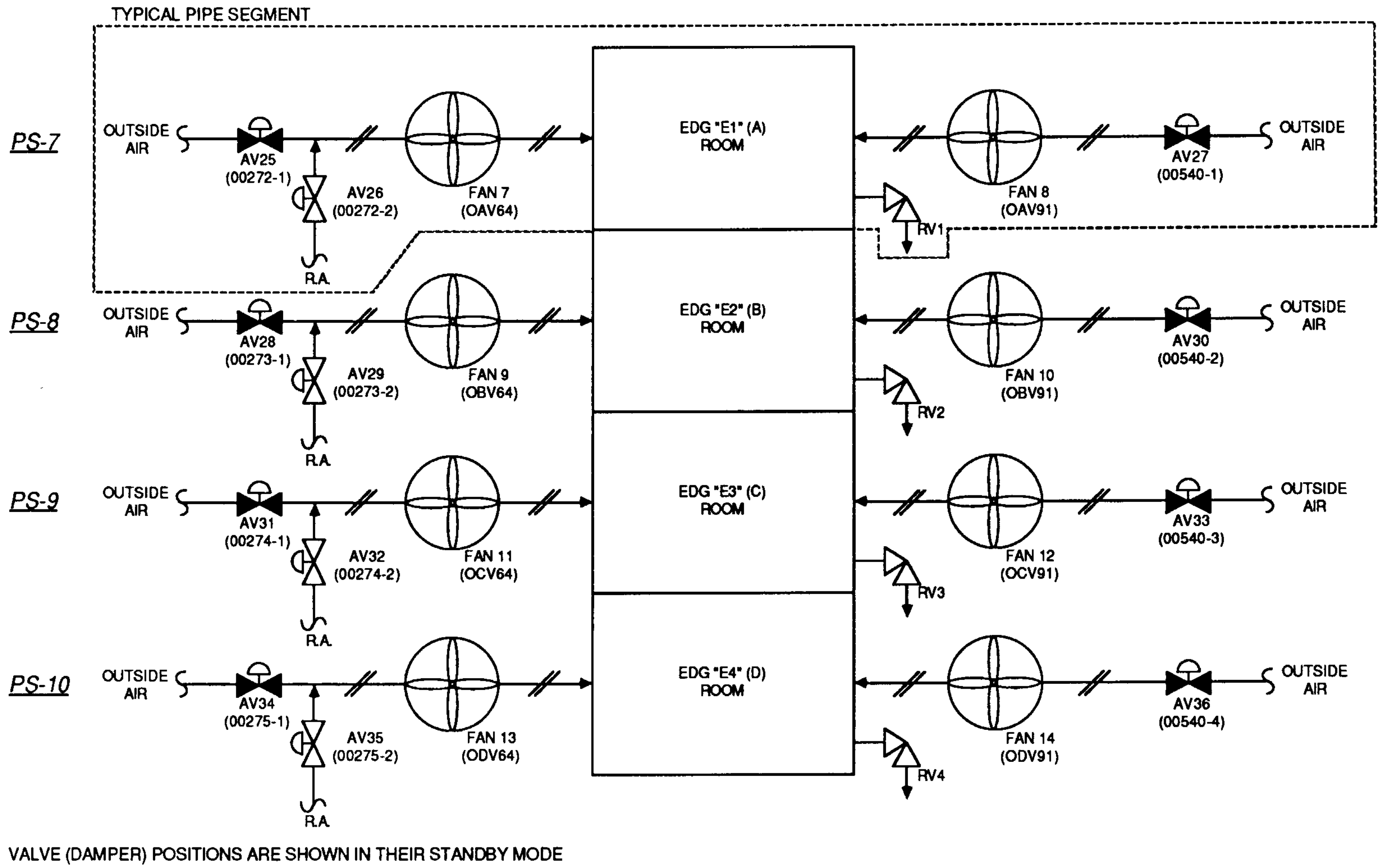
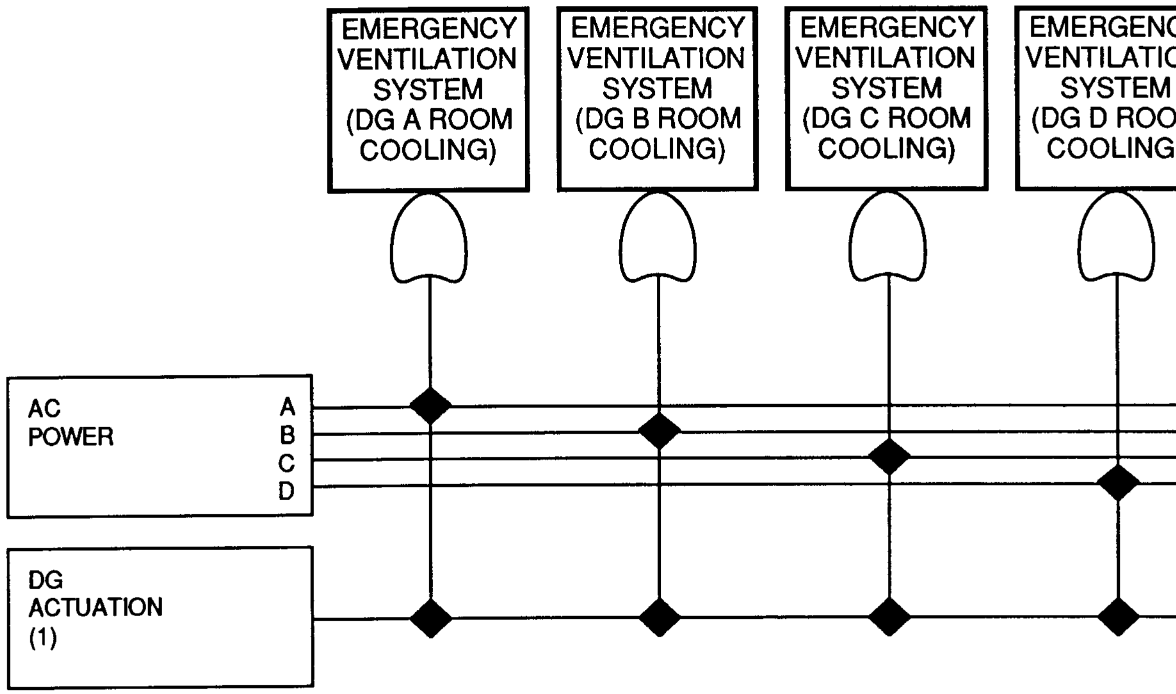


Figure 4.6.10-1. Emergency Ventilation System Schematic.



Dependency Diagram Is Shown Using Failure Logic.
(1) Each DG EVS Fan Is Auto Actuated By The Start Signal Of Its Corresponding DG.

Figure 4.6.10-2. Emergency Ventilation System Dependency Diagram.

4.6.10.3 EVS Test and Maintenance

No specific EVS test and maintenance requirements are identified in the Peach Bottom technical specifications.

4.6.10.4 EVS Technical Specifications

No reference is made to the EVS in the Peach Bottom technical specifications.

4.6.10.5 EVS Logic Models

The EVS was modeled using a fault tree. The major active and some passive components are shown as duct segments which were defined for the EVS. The fault tree model representing the EVS is presented in Appendix B.

Duct ruptures were considered to be negligible compared to other system failures.

One human error was incorporated into the EVS fault tree model. This error is failure to properly restore equipment following maintenance.

4.6.10.6 EVS Assumptions

(1) EVS failure is dominated by failure of fans and failure of closed dampers to open when demanded.

(2) Testing unavailabilities are negligible since tests include simple startup of the system.

4.6.10.7 EVS Operating Experience

Nothing was peculiar in the operational history of the EVS which would affect either system modeling or failure data.

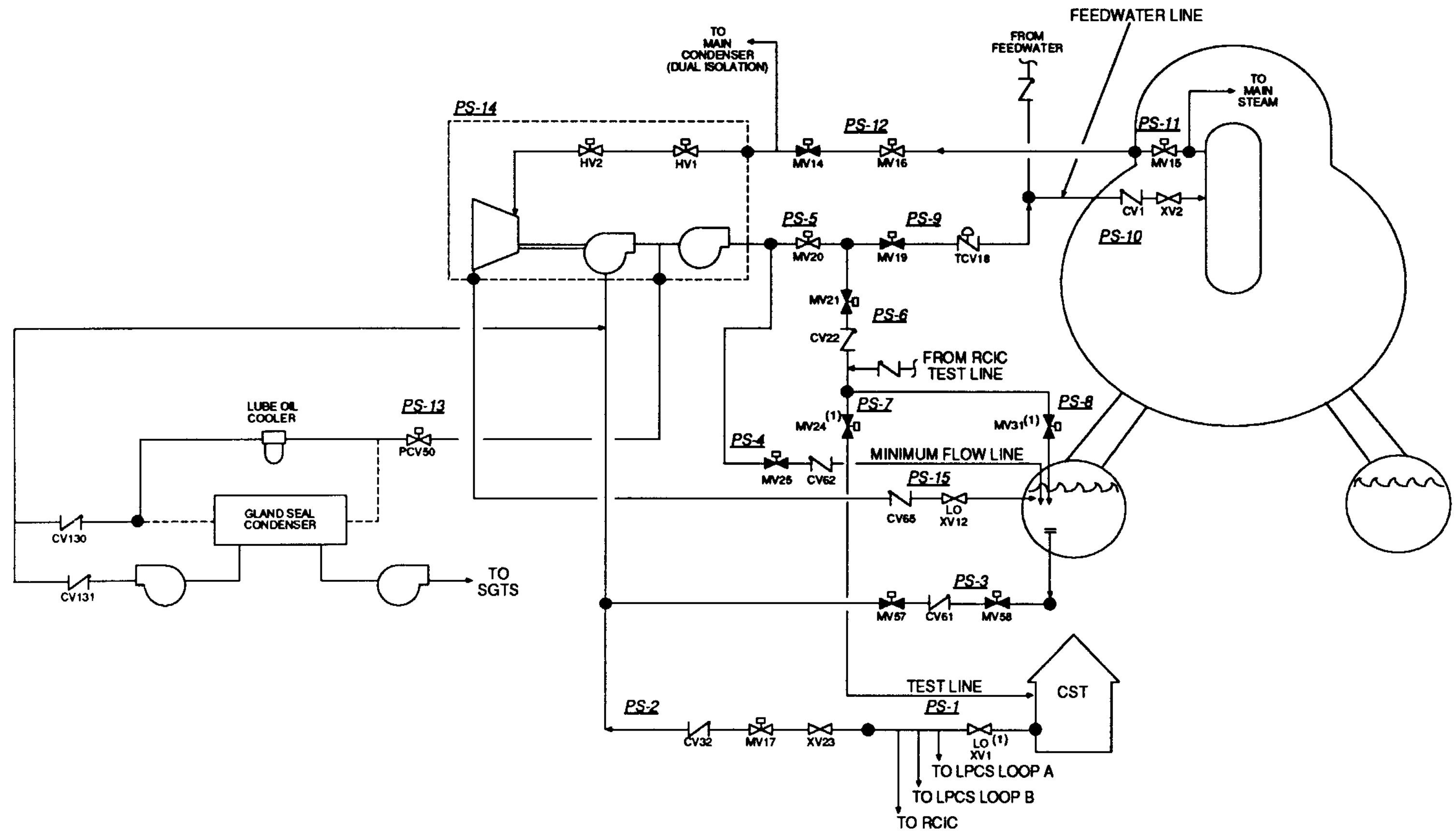
4.6.11 High Pressure Coolant Injection System

4.6.11.1 HPCI Description

The function of the HPCI system is to provide a makeup coolant source to the reactor vessel during accidents in which system pressure remains high (event tree nomenclature--U1).

The HPCI system consists of a single train with motor-operated valves and a turbine-driven pump. Suction is taken from either the CST or the suppression pool. Injection to the reactor vessel is via a feedwater line. The HPCI pump is rated at 5000 gpm flow with a discharge head of 1135 psig. A simplified schematic of the HPCI system is provided by Figure 4.6.11-1. Shown are major components that were modeled in the system fault tree.

The HPCI system is automatically initiated and controlled. Operator intervention is required as follows: (a) to prevent either vessel



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE
(1) VALVE ALSO SHOWN ON RCIC SCHEMATIC

Figure 4.6.11-1. High Pressure Coolant Injection System Schematic.

overfill if high level sensor failures occur, or continuous system trip/restart cycles, (b) to manually start the system given an auto--start failure, and (c) to setup the system for continuous operation under long-term station blackout conditions.

The success criteria for the HPCI system is injection at rated flow to the reactor vessel. For further information, refer to success criteria discussions in Section 4.4.

Most of the HPCI system is located in a separate room in the reactor building. Local access to the HPCI system could be affected by either containment venting or containment failure should steam be released to the reactor building area. Room cooling failure is estimated to fail the HPCI pump in ten hours (see Section 4.6.11.6).

4.6.11.2 HPCI Interfaces and Dependencies

The HPCI system major dependencies are DC power for short term and long term operation and room cooling for long term operation. Although there are AC-powered motor-operated valves, these valves are not required to change state during normal system operation since they are only used to isolate the system. A simplified dependency diagram of the HPCI system is provided by Figure 4.6.11-2. Shown are the major support needs for the HPCI system as indicated by the solid diamonds.

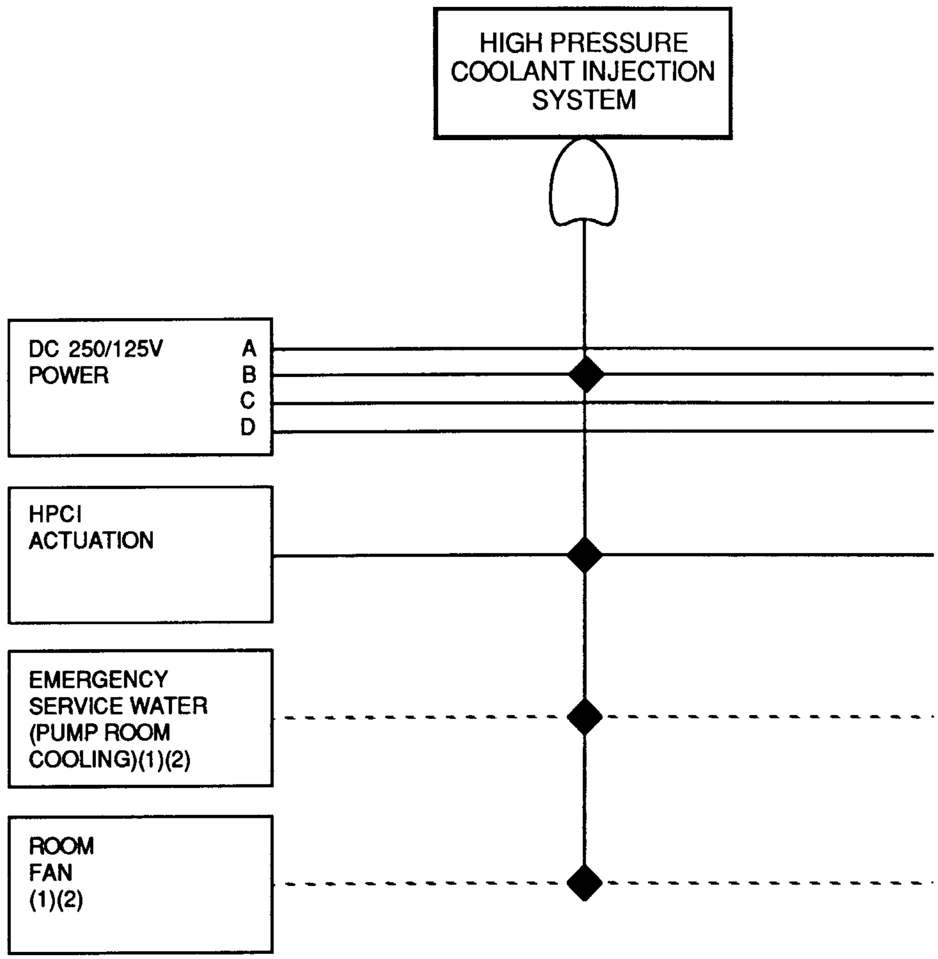
The HPCI system requires both 250 VDC/B and 125 VDC/B. 125 VDC/B is used for actuation and control power while an injection and a supply valve are powered from 250 VDC/B.

The HPCI and RCIC systems share a common CST suction valve. This is a normally open manual valve and is identified as XV-1 on the HPCI schematic. Failure of this valve will fail the CST as a suction source to both the HPCI and RCIC systems.

Upon system actuation, HPCI injection valves receive a signal to open and HPCI test valves receive a signal to close. The HPCI system is automatically initiated on the receipt of either a high drywell pressure (2 psig) or low reactor water level (490 inches above vessel zero) signal. The low reactor water level sensors are shared with the RCIC system.

The CST is the initial suction source for the HPCI system. Suction is automatically switched to the suppression pool upon either low CST level or high suppression pool level. Automatic switchover will not occur if there is an automatic isolation signal present. The CST suction valve does not close until both of the suppression pool suction valves are fully open.

The HPCI system is automatically isolated by high steam line space temperature, steam line high differential pressure (dP), or high turbine exhaust pressure (150 psig). Both the high temperature and high dP signals are used to detect a steam line break.



Dependency Diagram Is Shown Using Failure Logic.
(1) Dependency Not Required During Short Term Operation.
(2) Room Cooling Might Also Be Performed by Opening Doors.

Figure 4.6.11-2. High Pressure Coolant Injection System Dependency Diagram.

The HPCI turbine trips on high exhaust pressure, high reactor water level, low pump suction pressure, low steam pressure, or an auto isolation signal.

4.6.11.3 HPCI Test and Maintenance

The HPCI system surveillance requirements are the following: (1) pump operability--once/month, (2) motor-operated valve operability--once/month, (3) pump capacity test--once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.

4.6.11.4 HPCI Technical Specifications

If the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the ADS, RCIC, LPCI system, and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.11.5 HPCI Logic Model

The HPCI system was modeled using a fault tree for the injection of coolant to the reactor vessel. The major active components were modeled for the HPCI system. The fault tree model representing the HPCI system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only the piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

The gland seal condensate pump and the gland seal vacuum pump were not modeled since their operation is not essential to system operation.

Six human errors were incorporated into the HPCI fault tree model. These errors are (1) failure to trip the HPCI system and realign its suction source on low suction pressure, (2) failure to realign the suction source for the HPCI and RCIC systems in other circumstances, (3) failure to control HPCI flow (reactor level), (4) failure to manually backup automatic HPCI actuation, (5) miscalibration of CST level sensors, and (6) miscalibration of certain ESF sensors.

4.6.11.6 HPCI Assumptions

(1) The HPCI test return lines were not considered as potential diversion paths because the probability of two normally closed Motor Operated Valves (MOVs) failing to prevent flow was felt to be negligible compared to other system faults.

(2) Failure of the system to isolate given certain conditions was not considered since the system is effectively "non-operational." These conditions are: (a) high steam line space temperature, (b) high steam line dP, (c) low steam pressure, (d) high steam line exhaust pressure, and (e) manual isolation.

(3) Failure of the minimum flow line to open does not constitute system failure since the time between pump start and opening of the injection valve is small.

(4) The gland seal condensate pump and vacuum pump are not necessary for system operation. Therefore, their failures were not modeled.

(5) Spurious signals are felt to be negligible compared to other system failures because of their low probability of occurrence.

(6) The HPCI system is estimated to fail in a non-recoverable state if it fails to trip on low suction pressure or high reactor water level because of expected damage to the pump or turbine.

(7) HPCI pump bearing cooling fails if pump suction is from the suppression pool and the working fluid temperature reaches between 210 and 260°F. In the analysis, this was nominally assumed to occur at 250°F without any uncertainty in order to facilitate the analysis. Therefore, the uncertainty in the results does not reflect the temperature range over which failure might occur.

(8) The HPCI turbine auxiliary oil pump, stop valve, and governor valve failures were included in turbine failure data.

(9) System failure because of valves being left in the wrong position after test or maintenance is felt to be small compared to other system faults. The position of key manual and MOVs is indicated in the control room and the MOVs receive signals to realign on an actual demand. System operation must be assured of valve positions before startup of the plant following shutdown and concurrent maintenance activities. In addition, PECO maintains a control log of all "locked" valves in the plant to assure their correct position.

(10) Testing of TCV18 (PS-9) will not prevent flow from reaching the reactor vessel should a real demand occur.

(11) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. It was assumed that maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(12) An event for depletion of the CST was included for those cases where HPCI and/or RCIC operation was judged to be sufficiently long.

(13) Failure of the suppression pool by random failure or the plugging of its strainers is felt to be negligible compared to other system failures.

(14) If the HPCI or RCIC minimum flow line has been demanded open and subsequently fails to close on a system trip, there is the possibility that the CST will drain to the suppression pool because of their differences in elevation.

(15) Lube oil cooling is required for bearing cooling.

(16) The HPCI actuation circuitry was not modeled to a great degree of detail. Only elements which were felt to be potentially important were included in the fault tree model. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems. The power supply for the actuation circuitry was also included. Hardware failures of relays and certain permissives were grouped into one basic event.

(17) It is assumed that calibration of the low and low-low reactor vessel water level sensors is performed at the same time. Miscalibration of these sensors is assumed to be the same event.

(18) Failure to recover an initial loss of the normal suction source (the CST) will be treated as a recovery action. Operator error appears to dominate failures of suppression pool valves and their manual actuation circuitry. Failure of suppression pool valves from maintenance outages or support system failures appears elsewhere in the fault tree.

(19) Failure of the system to automatically realign to the suppression pool after a loss of the normal suction source (the CST) is treated explicitly with manual switchover being treated as a recovery action.

(20) The suction pressure trip is "ANDed" with a dummy event to account for the probability that low suction pressure exists.

(21) System unavailability due to testing is considered small compared with other system faults since it appears that the majority of testing requirements would not preclude proper system operation following a real demand. Hence this contribution to failure of the system is small compared with other system failure probabilities.

(22) Failure of room cooling (if not recovered) is estimated to fail HPCI in ten hours. This is based on utility calculations [52] which demonstrates that in 100 hours without room cooling, operability is expected assuming intermittent pump operation. Since in the accident sequences of interest continuous operation may be performed, this value was readjusted to 10 hours using engineering judgment.

4.6.11.7 HPCI Operating Experience

Nothing was peculiar in the operational history of the HPCI system which would affect system modeling. Plant operational data indicates a higher value for Turbine-Driven Pump (TDP) failure to run than the generic data base. The difference is that the generic value was calculated using plant operational hours instead of HPCI operational hours. The values compare closely when HPCI operational hours are used in the generic calculation. Therefore, the plant specific value for TDP failure to run is used.

4.6.12 High Pressure Service Water System

4.6.12.1 HPSW Description

The HPSW system is designed to supply cooling water from the ultimate heat sink to the RHR system heat exchangers under post-accident conditions and can provide an additional source of water to the reactor vessel (event tree nomenclature--V4) through a cross-tie to the RHR injection lines.

The HPSW system consists of four 4500 gpm pumps installed in parallel. The pumps are a vertical multi-stage turbine type with a discharge head of 700 ft. Each pump is sized to the design heat removal capacity of one RHR heat exchanger. Normal water supply to the suction of the pumps is from Conowingo Pond. In the EHS mode of system operation, suction comes from and discharge goes to the emergency cooling towers. The pump discharge is split into two headers with two pumps in each header. The headers are split by a normally closed, motor-operated gate valve. Each header delivers water to two RHR heat exchangers in parallel. The pump discharge head is sufficient to maintain the HPSW system at a higher pressure than the RHR system, thus precluding leakage of radioactivity and permitting operation in conjunction with the emergency cooling towers. As an injection source to the reactor vessel, the HPSW discharge to the RHR injection lines is from the pump B/D header. This connects to the RHR header. A simplified schematic of the HPSW system is provided by Figure 4.6.12-1. Major components are shown as well as the pipe segment definitions (e.g., PS-10) used in the system fault tree.

The operator is required to initiate the HPSW system. To initiate the system in the RHR cooling mode, the operator must start the appropriate HPSW pump and open the appropriate motor operated discharge valve depending on which RHR heat exchanger(s) is used. These discharge valves are arranged as one valve downstream of each of the four RHR heat exchangers. To inject water into the reactor vessel via the RHR system, the operator starts HPSW pumps B and/or D and opens MOV-176 and MOV-174.

The success criteria for the HPSW system in the RHR cooling mode is one of four pumps supplying flow to the appropriate one of four heat exchangers. This is based upon the RHR system success criteria. As a last effort injection source, either Pump B or D must supply flow through the cross-tie and corresponding RHR injection line under depressurized conditions in the reactor vessel. Pump A or C can be used with operation of a cross-tie valve. For further information, refer to the success criteria discussions in Section 4.4.

Most of the HPSW system is located in pump rooms external to the reactor and turbine buildings. Any physical impact of accident conditions on the ability of the HPSW system to perform its functions would be minimal except for the injection valves (MOV-174, 176) which are in the reactor building and could be affected by a harsh environment. Room cooling failure does not fail the HPSW pumps (see Section 4.6.12.6).

Failure of the HPSW system in the RHR cooling mode would fail the RHR cooling function. Failure of the HPSW system in the injection mode would fail one source of water for reactor makeup and containment spray.

4.6.12.2 HPSW Interfaces and Dependencies

The HPSW pumps have both a normal and a standby power supply. In the event of a loss of offsite power, each pump is powered by a different diesel generator. Corresponding DC power is required for all pumps for actuation purposes. The pumps are self-cooled and room cooling is not required. A simplified dependency diagram of the HPSW is provided by Figure 4.6.12-2. Shown are the major support needs for the HPSW system as indicated by the solid diamonds.

The HPSW system can inject water from the B/D header to the RHR system B header through a line containing two normally closed, motor-operated gate valves and a check valve.

Cooling tower fans are shared with the ESW system. These fans are used in the EHS mode of operation should the normal bay level be either too high or too low. The EHS mode requires power from three of the four divisions to operate the inline motor-operated valves.

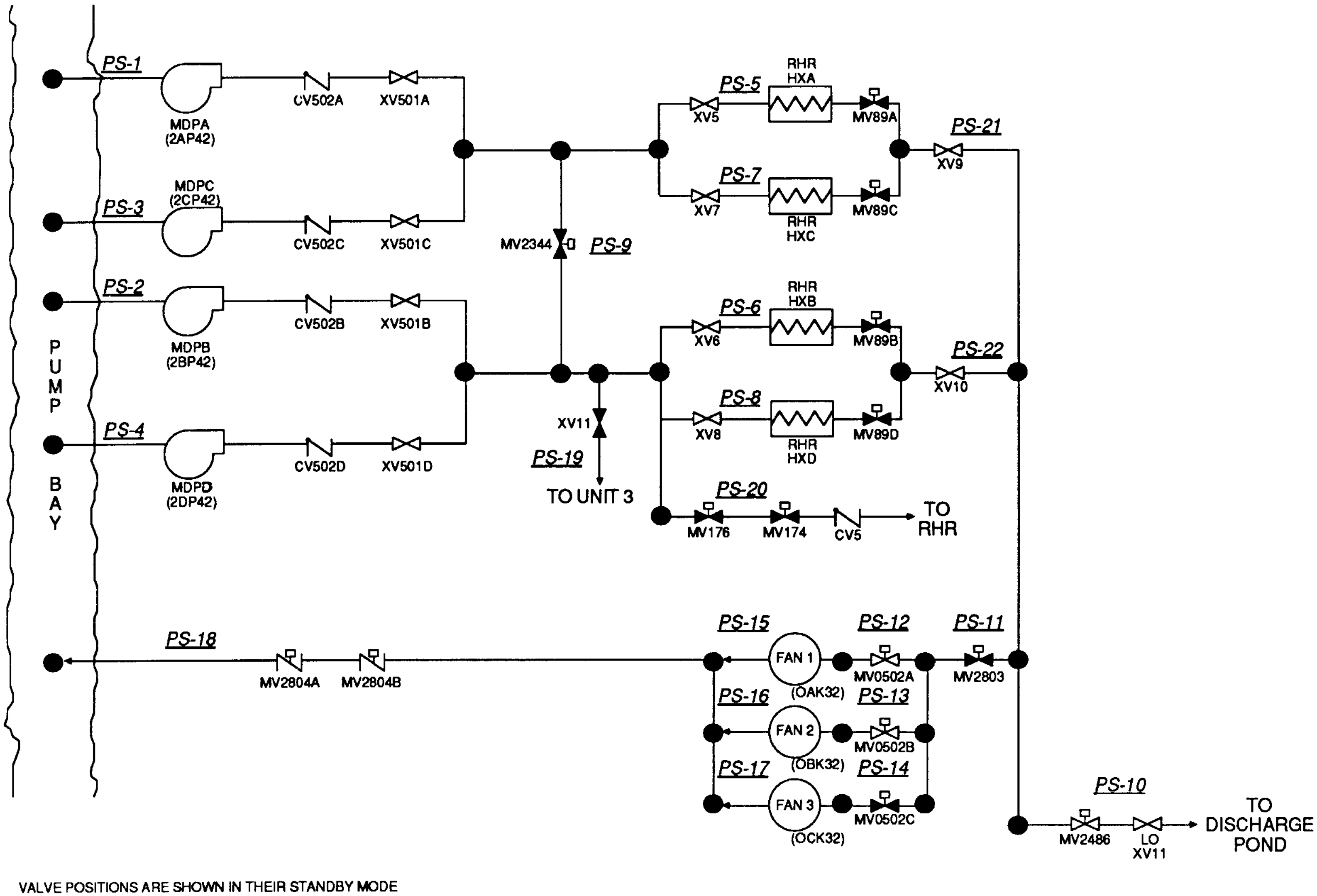
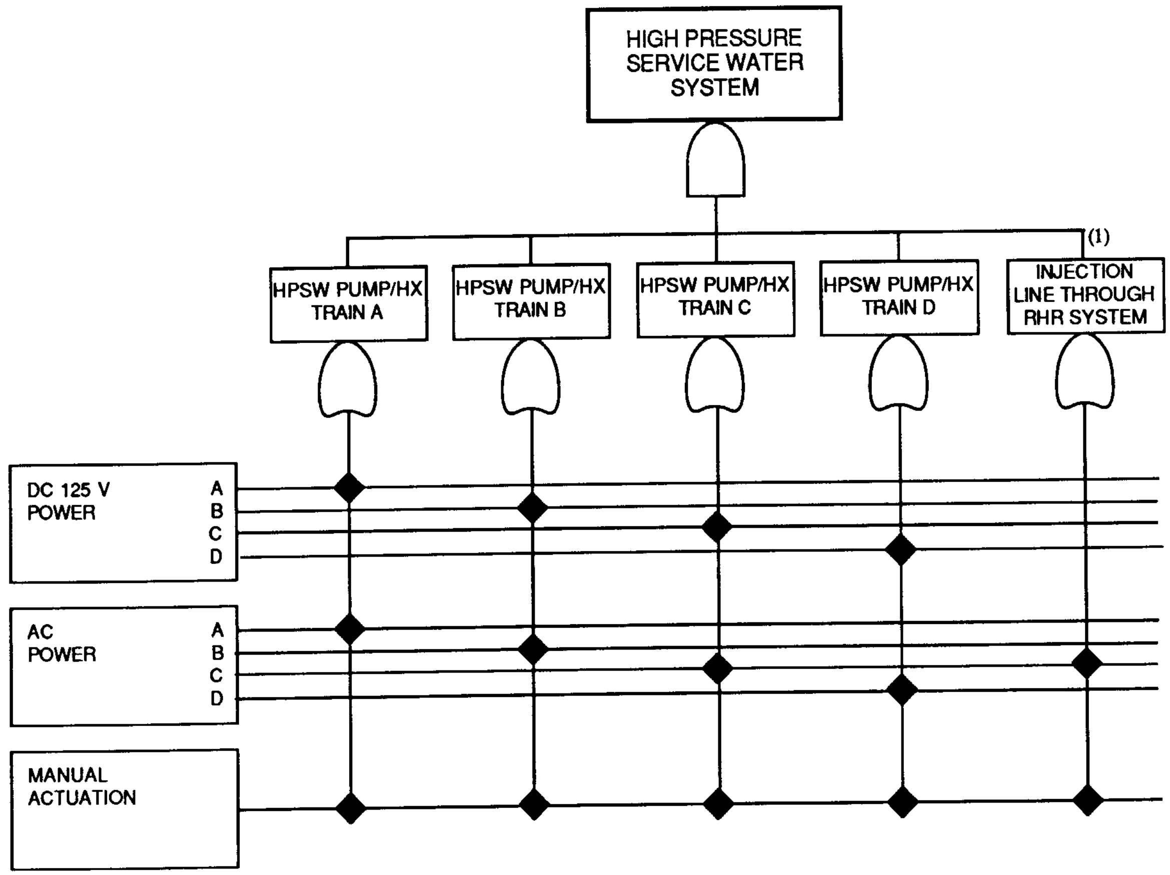


Figure 4.6.12-1. High Pressure Service Water System Schematic.



Dependency Diagram Is Shown Using Failure Logic.
(1) Used Only When The System Is Used As An Injection Source To The Reactor.

Figure 4.6.12-2. High Pressure Service Water Dependency Diagram.

The HPSW system is initiated manually, either locally or from the main control room.

4.6.12.3 HPSW Test and Maintenance

The HPSW surveillance requirements are the following: (1) pump operability--once/month, (2) motor-operated valve operability--once/month, and (3) pump capacity test--after pump maintenance and every three months.

4.6.12.4 HPSW Technical Specifications

The HPSW system shall be operable whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, as well as prior to reactor startup from a cold shutdown condition.

If any two HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible for thirty days. If three HPSW pumps are made or found to be inoperable, continued reactor operation is permissible for fifteen days. If three HPSW trains are made or found to be inoperable, the reactor can continue to operate for seven days. If these requirements cannot be met, the reactor is to be shut down.

4.6.12.5 HPSW Logic Models

The HPSW system was modeled using fault trees for both its heat removal mode (including the EHS configuration) and its vessel injection mode. The major active and some passive components were modeled for the HPSW system. The fault tree model representing the HPSW system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only the piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

Two human errors were modeled and include (1) failure of the operator to initiate the system, and (2) failure to restore equipment after maintenance.

4.6.12.6 HPSW Assumptions

(1) The HPSW pumps do not require room cooling. These pumps are located in a large building. By opening some doors (which is likely not to be necessary), adequate cooling can be provided for the pumps.

(2) The system is switched to the EHS mode when the sluice gates in the pump bay are closed and the water level drops. It is estimated that the EHS mode can also be switched on if MOV-2468 to the discharge pond fails closed.

(3) The design basis criteria follow. The emergency cooling towers require the fans for adequate heat removal. One induced-draft cooling tower is needed for heat removal from one RHR heat exchanger. One cooling tower is also needed for removal of heat from ESW loads. The cooling towers may be able to remove heat without induced-draft, but the success criteria would be different and would require further analysis. This has a negligible effect on system reliability since the emergency cooling towers are the secondary source of heat sink for the RHR heat exchangers.

(4) The emergency cooling tower reservoir is needed for successful operation of the HPSW system in the EHS mode. The HPSW system is switched to the EHS mode when the water level in the pump bay is already low. Without added water from the reservoir, the pumps will not have adequate NPSH either at the time of switchover or after when there will be further drainage from the pump bay.

(5) If the reservoir is providing water to the pump bay, failure of the pond discharge valve MV-2486 to close during the EHS mode of operation does not result in system failure. If this valve fails to close and the reservoir is supplying make up water, the reservoir will be depleted faster. Reservoir depletion will take three and a half days instead of seven days since approximately half the flow is diverted into the pond. This is considered easily recoverable.

(6) Test unavailability or failure to restore after test for the HPSW system is considered insignificant. The system is essentially aligned to its desired configuration for test.

4.6.12.7 HPSW Operating Experience

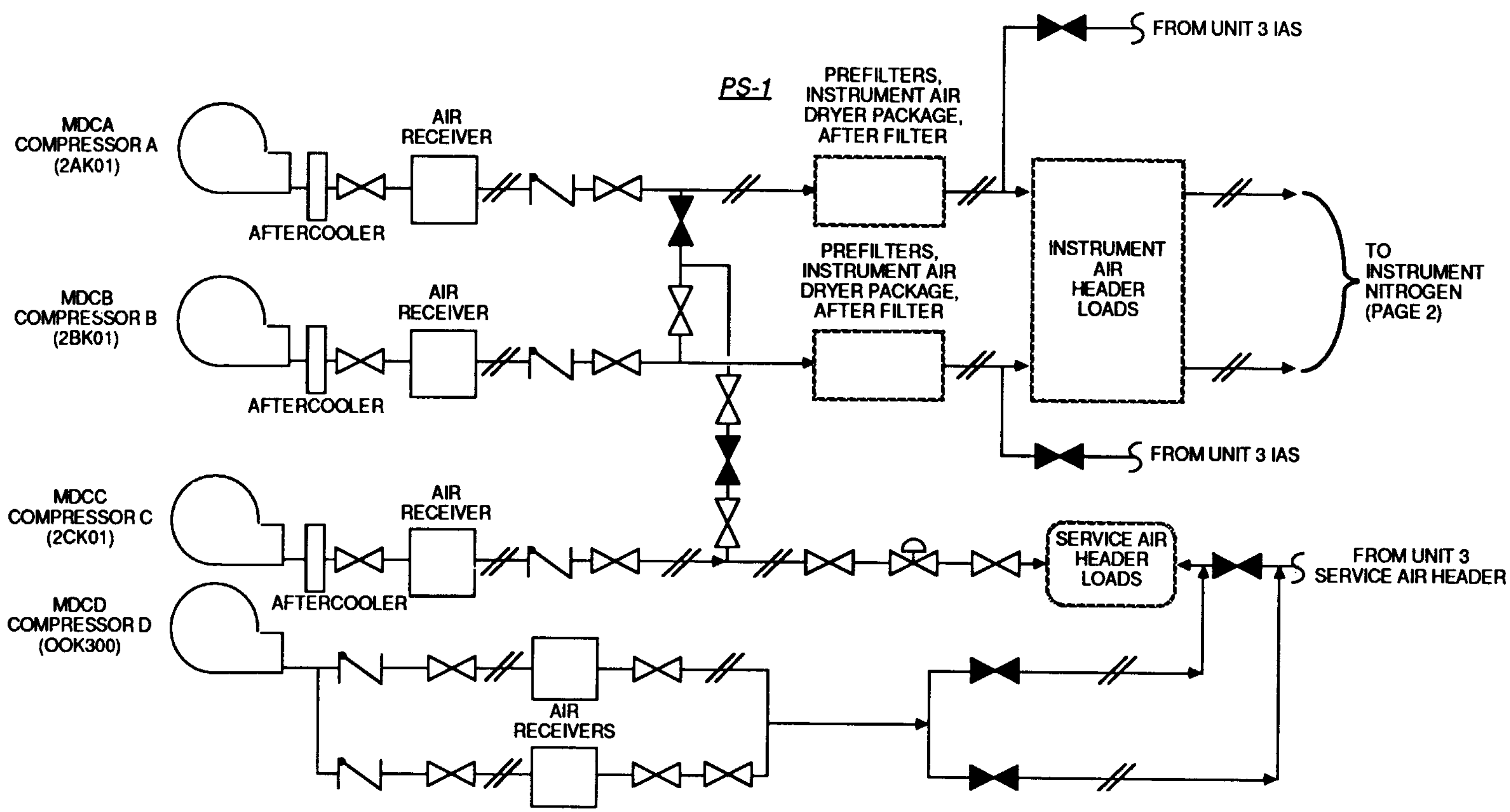
Nothing was peculiar in the operational history of the HPSW system which would affect either system modeling or failure data.

4.6.13 Instrument Air System

4.6.13.1 IAS Description

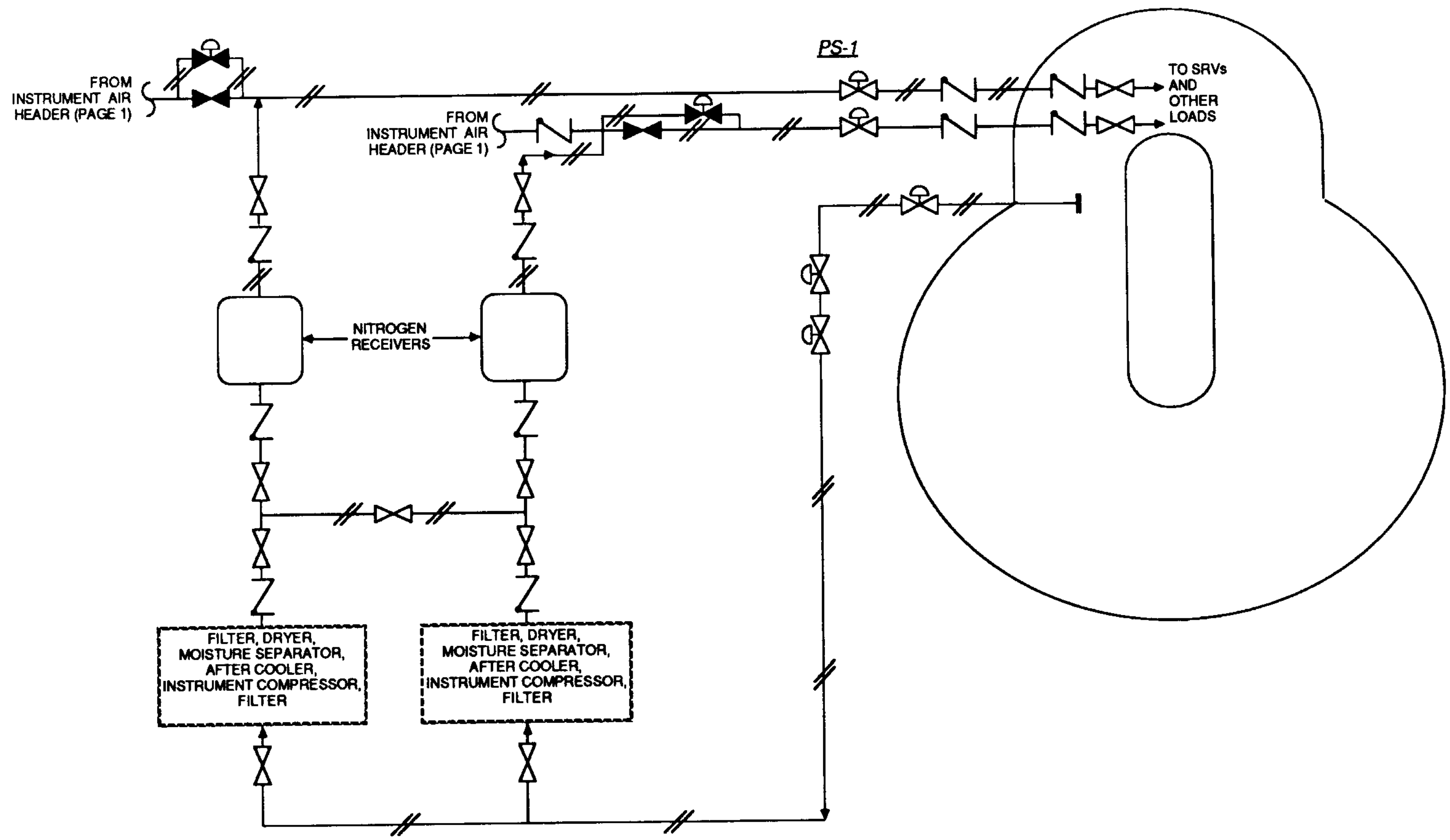
The IAS provides a pneumatic supply to support short-term and long-term operations of safety equipment.

The IAS and Service Air System (SAS) consist of three, in parallel, air compressors supplying a common discharge header via individual air receiver tanks, ductwork, valves, and instrumentation. A fourth air compressor is tied into the SAS header and is common to both units. Two compressors, one IAS and one SAS, normally supply all compressed air requirements. The other IAS compressor serves in a standby capacity. A simplified schematic of the IAS is provided by Figure 4.6.13-1. Shown is the tie-in with the Instrument Nitrogen System which is the preferred



VALVE (DAMPER) POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.13-1. Instrument Air/Nitrogen System Schematic.
(Page 1 of 2)



VALVE (DAMPER) POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.13-1. Instrument Air/Nitrogen System Schematic.
(Page 2 of 2)

supply to the Main Steam Isolation Valves (MSIVs) and ADS/SRVs. In addition to these compressors, the IAS is currently backed up by a portable diesel compressor and will be backed up by two diesel compressors (not shown) in the future, and can be served by the Unit 3 IAS/SAS.

Each of the three parallel compressors is a vertical, single-stage, double-acting, non-lubricated, reciprocating compressor rated at 377 scfm at 100 psig. Each has an aftercooler, moisture separator, and air receiver tank.

The standby SAS compressor consists of a non-lubricated compressor, aftercooler, moisture separator, and two receivers. This compressor is rated at 400 scfm at 100 psig.

The IAS supplies clean, dry, oil-free air to EVS and ESW system air valves, the CRD control system, and containment venting air valves and is a backup to the Instrument Nitrogen System.

When offsite power is lost, the air compressors trip. The operator is required to manually restart the air compressors when power is restored.

The success criterion for the IAS is that any one of the compressors supply air to system pneumatic loads.

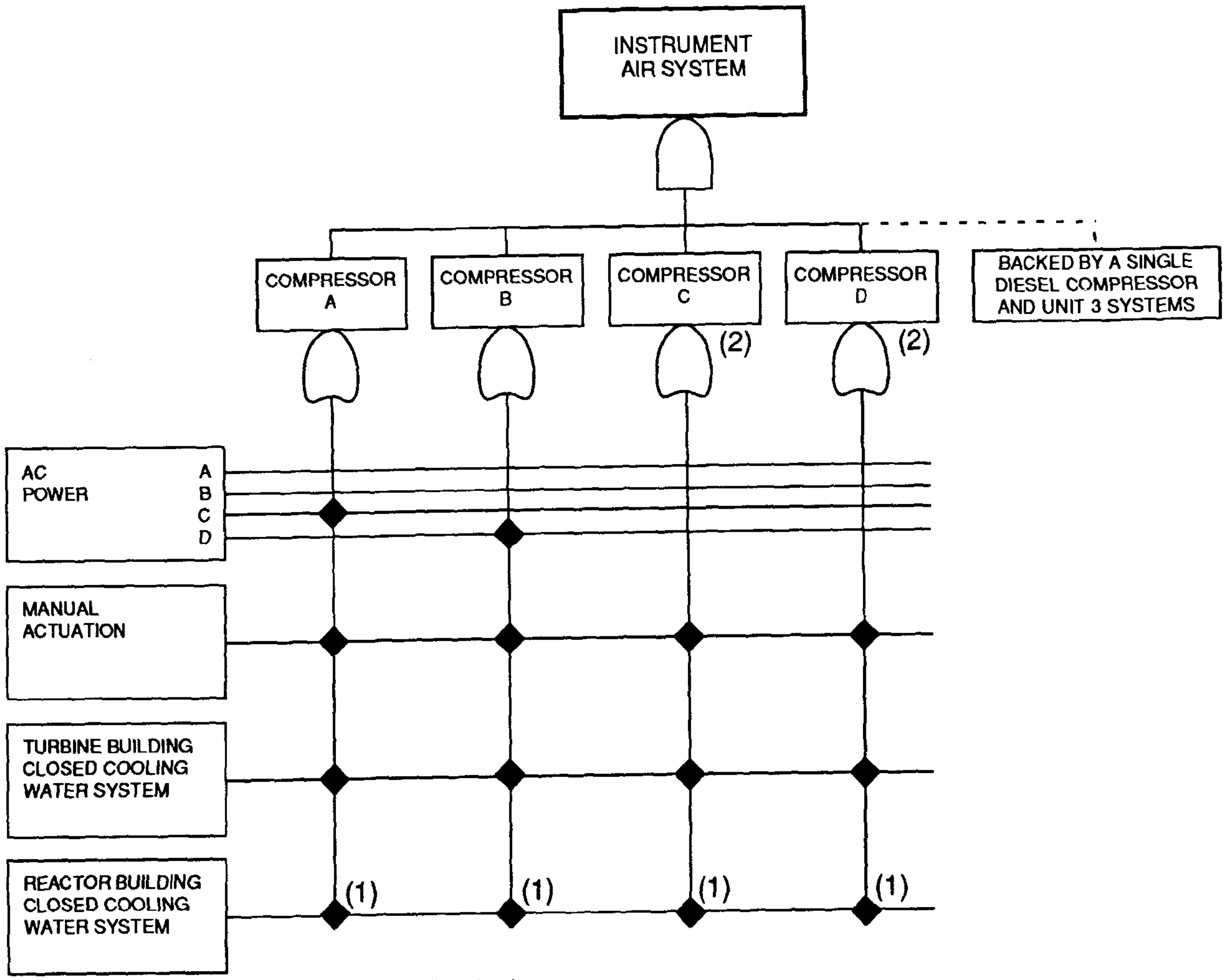
Any physical impact of accident conditions on the ability of the IAS to perform its functions would be minimal. Room cooling failure is deemed not to fail the IAS and SAS compressors. Even if this were to occur, the diesel compressors or unit 3 compressors could serve the necessary loads.

Failure of the IAS does not directly fail any safety systems because (1) accumulators are on the MSIVs and ADS valves, (2) instrument nitrogen is the preferred source to the MSIVs and ADS valves, and (3) other safety systems "fail-safe" on loss of air or have dedicated air bottles.

4.6.13.2 IAS Interfaces and Dependencies

Cooling requirements of system air compressors and aftercoolers are normally supplied by the TBCW system. In the event of offsite power failure, the RBCW system cools the air compressors and aftercoolers.

Motor-driven air compressor A is powered from 480 VAC/C with control and actuation power supplied by 120 VAC/C. Air Compressor B is powered from 480 VAC/D with control and actuation power supplied by 120 VAC/D. Air Compressors C and D are powered from non-safety Buses 20B13 and 20B31, respectively. Their control and actuation power comes from 120 VAC non-safety buses. Following a loss of offsite power, standby onsite power is provided to the air compressors to replenish compressed air storage as required. A simplified dependency diagram of the IAS is provided by Figure 4.6.13-2. In addition, two diesel compressors are normally on-line as backups.



Dependency Diagram Is Shown Using Failure Logic.
(1) RBCW Is A Backup If The TBCW Fails.
(2) Powered By Non-Safety Buses.

Figure 4.6.13-2. Instrument Air/Nitrogen System Dependency Diagram.

4.6.13.3 IAS Test and Maintenance

No IAS test and maintenance requirements are identified in the Peach Bottom technical specifications.

4.6.13.4 IAS Technical Specifications

IAS degradation does not limit plant operations.

4.6.13.5 IAS Logic Models

The IAS was modeled using a very simple fault tree covering only the failures of the compressors and loss of support system needs. The fault tree model representing the IAS is presented in Appendix B. This simplified modeling approach was used since the importance of this system to other systems modeled in the study is limited. Therefore, a detailed analysis was not warranted.

Ductwork ruptures were considered to be negligible compared to other system failures. Only ducts with a diameter of greater than or equal to 1/3 of the main system ducting was considered as a potential diversion path.

One human error was explicitly incorporated into the IAS fault tree model. This error is the operator's failing to restart the system following a loss of offsite power.

4.6.13.6 IAS Assumptions

(1) All IAS loads can be supplied from both IAS headers.

(2) The IAS trips on loss of offsite power and needs to be restarted manually.

(3) Failure of the TBCW system to provide cooling is dominated by TBCW pump failures and loss of offsite power.

(4) Failure of the RBCW system to provide cooling is dominated by RBCW pump failures and failure of switchover to the RBCW system for cooling.

(5) Due to the large number of compressors available even under partial losses of power, the IAS hardware was largely black-boxed with an assumed unavailability of 1E-4 using engineering judgment.

4.6.13.7 IAS Operating Experience

Nothing was peculiar in the operational history of the IAS which would affect either system modeling or failure data.

4.6.14 Low Pressure Coolant Injection System

4.6.14.1 LPCI Description

The function of the LPCI system is to provide a makeup coolant source to the reactor vessel during accidents in which system pressure is low (event tree nomenclature--V3). The ADS can be used in conjunction with the LPCI system to attain a low enough system pressure for injection to occur. The LPCI system is but one mode of the RHR system and, as such, shares components with other modes.

The RHR system is a two-loop system consisting of motor-operated valves and motor-driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm with a discharge head of 540 feet. Cooling water flow to the heat exchangers is not required for the LPCI mode. The LPCI suction source is the suppression pool. A simplified schematic of the LPCI (RHR) system is provided by Figure 4.6.14-1 with the LPCI portion highlighted. Major components are shown as well as the pipe segment definitions (e.g., PS-19) used in the system fault tree.

The LPCI system is automatically initiated and controlled. Operator intervention is required to manually start the system given an auto-start failure and to stop the system or control flow during an ATWS if required.

The success criterion for the LPCI system is injection of flow from any one pump to the reactor vessel. For further information, refer to success criteria discussions in Section 4.4.

Most of the LPCI system is located in the reactor building. Local access to the LPCI system could be affected by either containment venting or containment failure. Room cooling failure is deemed to fail the LPCI pumps in ten hours.

4.6.14.2 LPCI Interfaces and Dependencies

Each LPCI pump is powered from a separate 4160 VAC bus with control and actuation power being supplied by a separate 125 VDC bus. All pumps require pump cooling. For further information on pump cooling refer to Section 4.6.9.8.

Each loop's normally closed injection valve can receive motive power from one of two 480 VAC sources. The Loop A injection valve sources are either 480 VAC/A or 480 VAC/C, and the Loop B injection valve sources are either 480 VAC/B or 480 VAC/D. A simplified dependency diagram of the LPCI is provided by Figure 4.6.14-2. Shown are the major support needs for the LPCI system as indicated by the solid diamonds.

Many components of the LPCI system are shared with the different modes of the RHR system. These commonalities are as follows: (1) the RHR pumps are common to the LPCI, SPC, CS, and SDC modes; (2) the suppression pool suction valve for each pump train is common to the LPCI, SPC, and CS

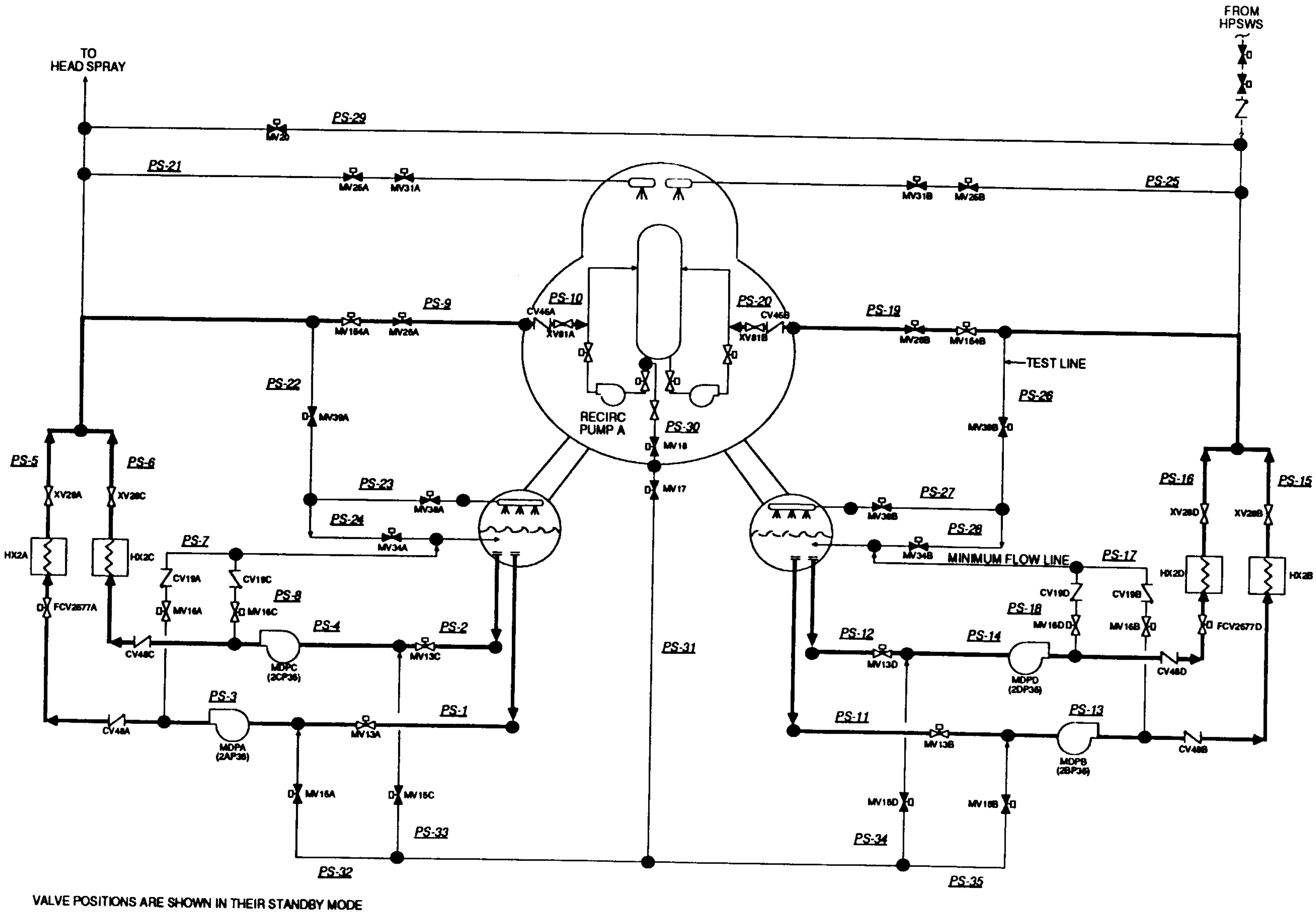


Figure 4.6.14-1. Low Pressure Coolant Injection System Schematic.

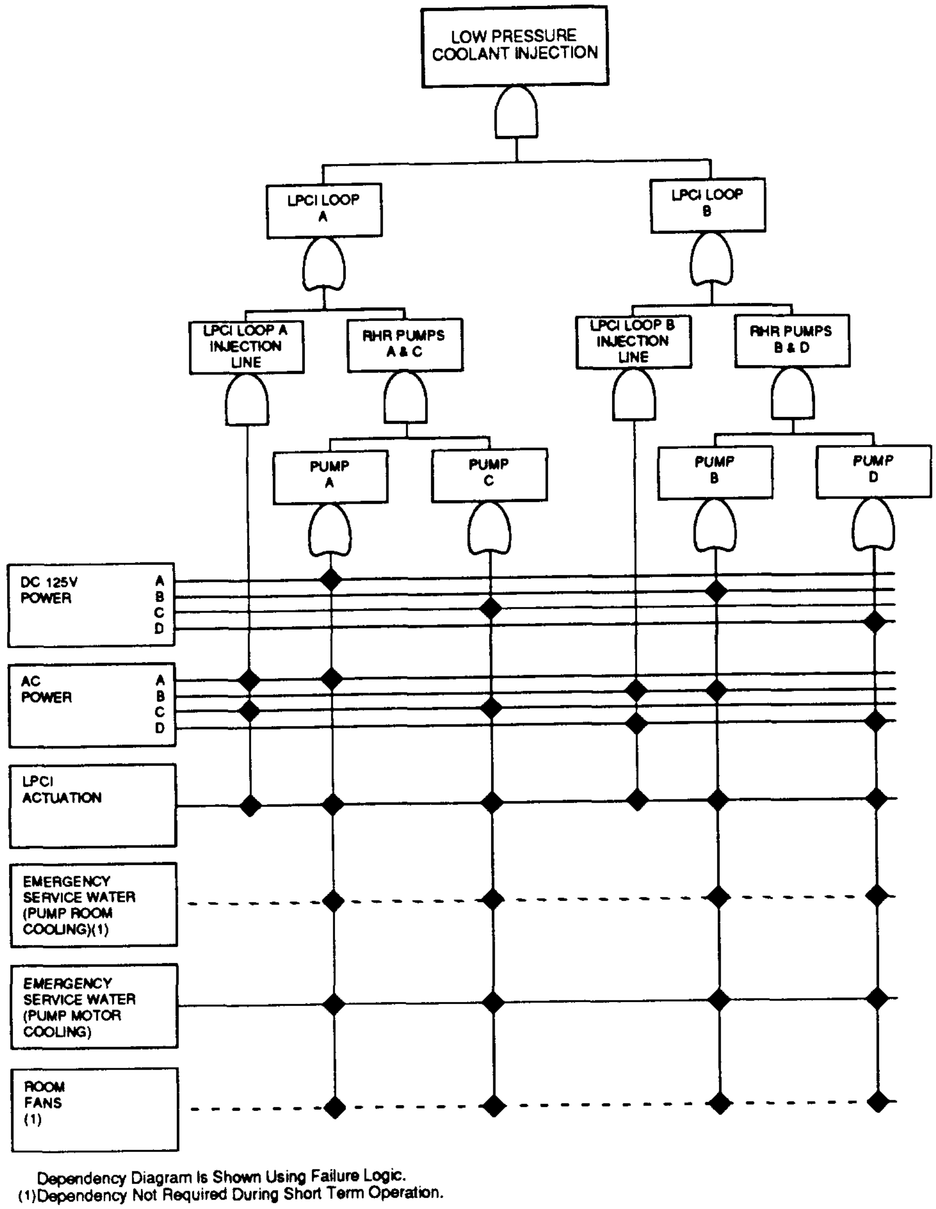


Figure 4.6.14-2. Low Pressure Coolant Injection System Dependency Diagram.

modes; and (3) Loops A and B injection valves are common to the LPCI, SDC, and HPSW injection modes.

Upon the receipt of a LPCI injection signal, start signals are sent to all pumps, Loops A and B injection valves are subsequently demanded to open when the reactor pressure is low enough, and the test return valves are demanded to close. The LPCI system is automatically initiated on the receipt of either a low-low reactor water level (378 inches above vessel zero) or high drywell pressure (2 psig) and low reactor pressure (450 psig). All actuation sensors are shared with the LPCS system.

LPCI actuation and control circuitry is divided into two divisions. Division A is associated with the actuation and control of components in Loop A, and Division B is associated with the actuation and control of components in Loop B. Each LPCI pump and loop injection valve receives an actuation signal from both divisions.

Although the LPCI system has no isolation signals, there are permissives which will prevent the operation of certain components. LPCI pumps are demanded to stop or prevented from starting if the suppression pool suction valve or any of three SDC suction valves are not fully open.

Loops A and B injection valves are prohibited from opening unless a low reactor pressure permissive (450 psig) is met and will reclose if reactor pressure becomes too high.

4.6.14.3 LPCI Test and Maintenance

The LPCI surveillance requirements are the following: (1) pump operability--once/month, (2) MOV operability--once/month, (3) pump capacity test--once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.

4.6.14.4 LPCI Technical Specifications

If any one LPCI pump is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the remaining LPCI components and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.14.5 LPCI Logic Model

The LPCI system was modeled using a fault tree for the injection of coolant to the reactor vessel. The major active components were modeled for the LPCI system. The fault tree model representing the LPCI system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

Three human errors were incorporated into the LPCI fault tree model. These errors are miscalibration of various sensors, failure to manually backup automatic actuation, and failure to properly restore key components following maintenance.

4.6.14.6 LPCI Assumptions

(1) Positions of all manual and motor-operated valves are indicated in the control room. Failure of these valves after testing and maintenance from incorrect positioning is therefore felt to be negligible. Test diverting flow causing LPCI system failure is also felt to be negligible since valves receive signals to close from both Divisions A and B actuation on a real demand. Thus, unavailability due to testing and failure to restore after testing is not important.

(2) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. It was judged that maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(3) Pump isolation because of spurious signals is assumed to be negligible compared to other system faults.

(4) The LPCI actuation circuitry was not modeled at a great level of detail. Only elements which were felt to be potentially important were included in the fault tree model. Hardware failure of relays and permissives are grouped into one term. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems.

(5) Based on a PECO response, it is estimated that the LPCI pumps will fail because of insufficient NPSH once the suppression pool has reached saturated conditions.

(6) A suction path must be available from either the suppression pool or the SDC path to start a LPCI pump.

(7) The unavailability of the LPCI pumps due to testing does not defeat a real demand from operating the system. Therefore, it was not considered. Failure to restore the LPCI pumps after testing does not apply.

(8) Failure of the suppression pool because of random failure or the plugging of all its strainers is assumed to be negligible compared to other system failures.

(9) It is assumed that calibration of the low and low-low reactor water level sensors is performed at the same time. Miscalibration of these sensors is considered to be the same event.

(10) Failure of room cooling (if not recovered) fails LPCI in ten hours. This is based on utility calculations [52] which demonstrate that for approximately 50 hours or more without room cooling, operability is expected even with continuous pump operation. The ten hour LPCI failure value was chosen to be consistent with the general assumptions made for HPCI and RCIC. It is believed to be a conservative value.

4.6.14.7 LPCI Operating Experience

Nothing was peculiar in the operational history of the LPCI system which would affect either system modeling or failure data.

4.6.15 Low Pressure Core Spray System

4.6.15.1 LPCS Description

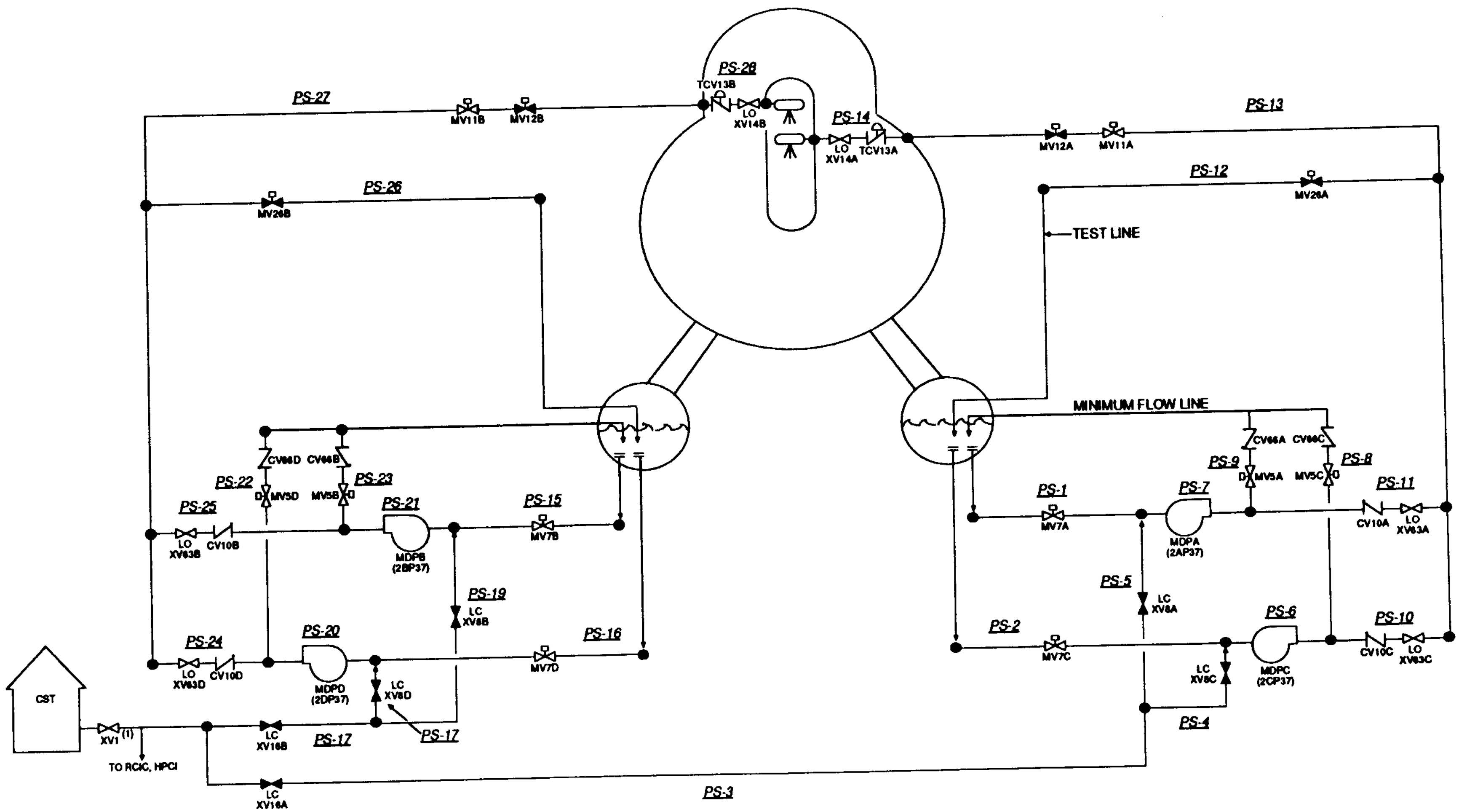
The function of the LPCS system is to provide makeup coolant to the reactor vessel during accidents in which system pressure is low (event tree nomenclature--V2.). The ADS can be used in conjunction with the LPCS system to attain a low enough system pressure for injection to occur.

The LPCS system is a two-loop system consisting of motor-operated valves and motor driven pumps. There are two fifty percent capacity pumps per loop, with each pump rated at 3125 gpm with a discharge head of 105 psig. The LPCS system normal suction source is the suppression pool. Pump suction can be manually realigned to the CST. A simplified schematic of the LPCS system is provided by Figure 4.6.15-1. Major components are shown as well as the pipe segment definitions (e.g., PS-27) used in the system fault tree.

The LPCS system is automatically initiated and controlled. Operator intervention is required to manually start the system given an auto-start failure and to stop the system or manually control flow during an ATWS if required.

The success criterion for the LPCS system is injection of flow from any two pumps to the reactor vessel. For further information, refer to success criteria discussions in Section 4.4.

Most of the LPCS system is located in the reactor building. Local access to the LPCS system could be affected by either containment venting or containment failure. Room cooling failure is assumed to fail the LPCS pumps in ten hours.



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY POSITION
(1) VALVE ALSO LOCATED ON HPCI SCHEMATIC, SEE HPCI SCHEMATIC FOR DEFINITION OF PIPE SEGMENT

Figure 4.6.15-1. Low Pressure Core Spray System Schematic.

4.6.15.2 LPCS Interfaces and Dependencies

Each LPCS pump is powered from a separate 4160 VAC bus with control and actuation power being supplied by a separate 125 VDC bus. All pumps require pump cooling. For further information on pump cooling, refer to Section 4.6.9.8.

Each loop's normally closed injection valve receives its motive power from a separate 480 VAC bus (480 VAC/C for Loop A, 480 VAC/D for Loop B). A simplified dependency diagram of the LPCS system is provided by Figure 4.6.15-2. Shown are the major support needs for the LPCS system as indicated by the solid diamonds at the appropriate places in the diagram.

Upon the receipt of a LPCS injection signal, start signals are sent to all LPCS pumps, both injection valves are demanded to open, and the test return valves are demanded to close. The LPCS system is automatically initiated on the receipt of either a low-low reactor water level (378 inches above vessel zero) or high drywell pressure (2 psig) and low reactor pressure (450 psig). All actuation sensors are shared with the LPCI system.

LPCS actuation and control circuitry is divided into two divisions. Division A is associated with the actuation and control of the components in Loop A, and Division B is associated with the actuation and control of the components in Loop B.

Each LPCS pump has a minimum flow line valve (normally open) which is demanded to open given a pump start.

Both injection valves are prohibited from opening unless a low reactor pressure permissive (450 psig) is met.

4.6.15.3 LPCS Test and Maintenance

The LPCS system surveillance requirements are the following: (1) pump operability--once/month, (2) MOV operability--once/month, (3) pump capacity test--once/three months, (4) simulated automatic actuation test---once/operating cycle, and (5) logic system functional test--once/six months.

4.6.15.4 LPCS Technical Specifications

If any one LPCS loop is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the remaining LPCS loop and the LPCI system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.15.5 LPCS Logic Model

The LPCS system was modeled using a fault tree for the injection of coolant to the reactor vessel. The major active components were modeled for the LPCS system. The fault tree model representing the LPCS system is presented in Appendix B.

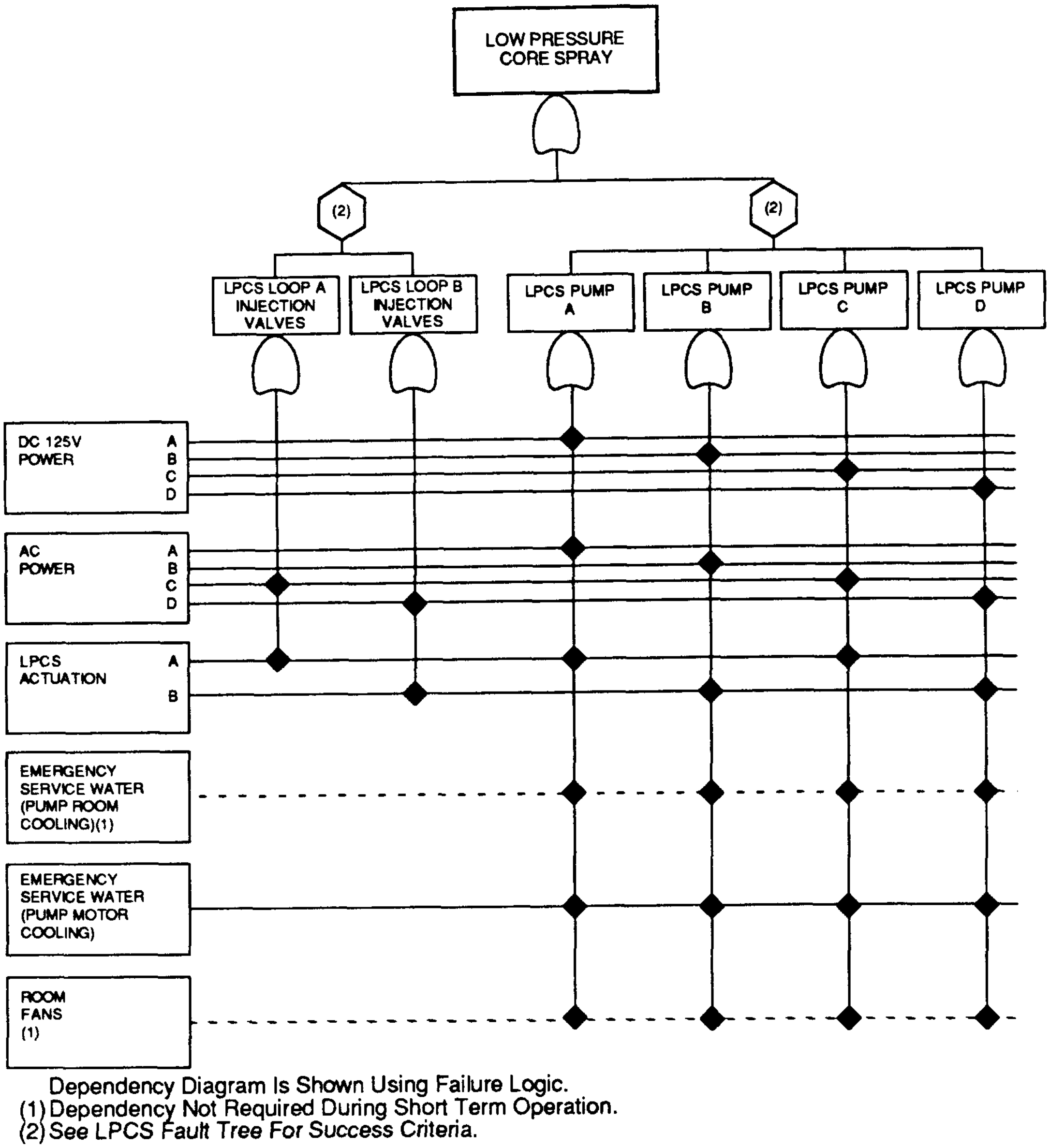


Figure 4.6.15-2. Low Pressure Core Spray System Dependency Diagram.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system was considered as a potential diversion path. Three human errors were incorporated into the LPCS fault tree model. These errors are miscalibration of various sensors, failure to manually backup automatic actuation, and failure to properly restore key components following maintenance.

4.6.15.6 LPCS Assumptions

(1) Positions of all manual and motor-operated valves are indicted in the control room. Failure of these valves after testing and maintenance due to incorrect positioning is therefore felt to be negligible. Test diverting flow causing LPCS system failure is also felt to be negligible since valves receive signals to close from both Divisions A and B actuation circuitry. The injection valves receive open signals on a real demand. Thus, unavailability due to testing and failure to restore after testing is not important.

(2) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. Maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for the component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(3) Pump isolation because of spurious signals is assumed to be negligible compared to other system faults.

(4) The LPCS actuation circuitry was not modeled at a great level of detail. Only elements which were felt to be potentially important were included in the fault tree model. Hardware failures of relays and permissives were grouped into one term. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems.

(5) Based on a PECO response, the LPCS pumps will fail because of insufficient NPSH once the suppression pool has reached saturated conditions.

(6) The CST is an alternate suction source which must be manually valved in and therefore is not explicitly included in the model but can be handled as a recovery action.

(7) The LPCS pumps do not trip on low pump suction pressure.

(8) The unavailability of the LPCS pumps from testing does not defeat a real demand from operating the system. Therefore, it was not considered. Failure to restore the LPCS pumps after testing does not apply.

(9) Failure of the suppression pool because of random failure or the plugging of all its strainers is assumed to be negligible compared to other system failures.

(10) It is assumed that calibration of the low and low-low reactor water level sensors is performed at the same time. Miscalibration of these sensors is considered to be the same event.

(11) Failure of room cooling (if not recovered) is assumed to fail LPCS in ten hours. This is based on utility calculations [52] which demonstrate that for approximately 50 hours or more without room cooling, operability is expected even with continuous pump operation. The ten hour LPCS failure value was chosen to be consistent with the general assumptions made for HPCI and RCIC. It is a conservative value.

4.6.15.7 LPCS Operation Experience

Nothing was peculiar in the operational history of the LPCS system which would affect either system modeling or failure data.

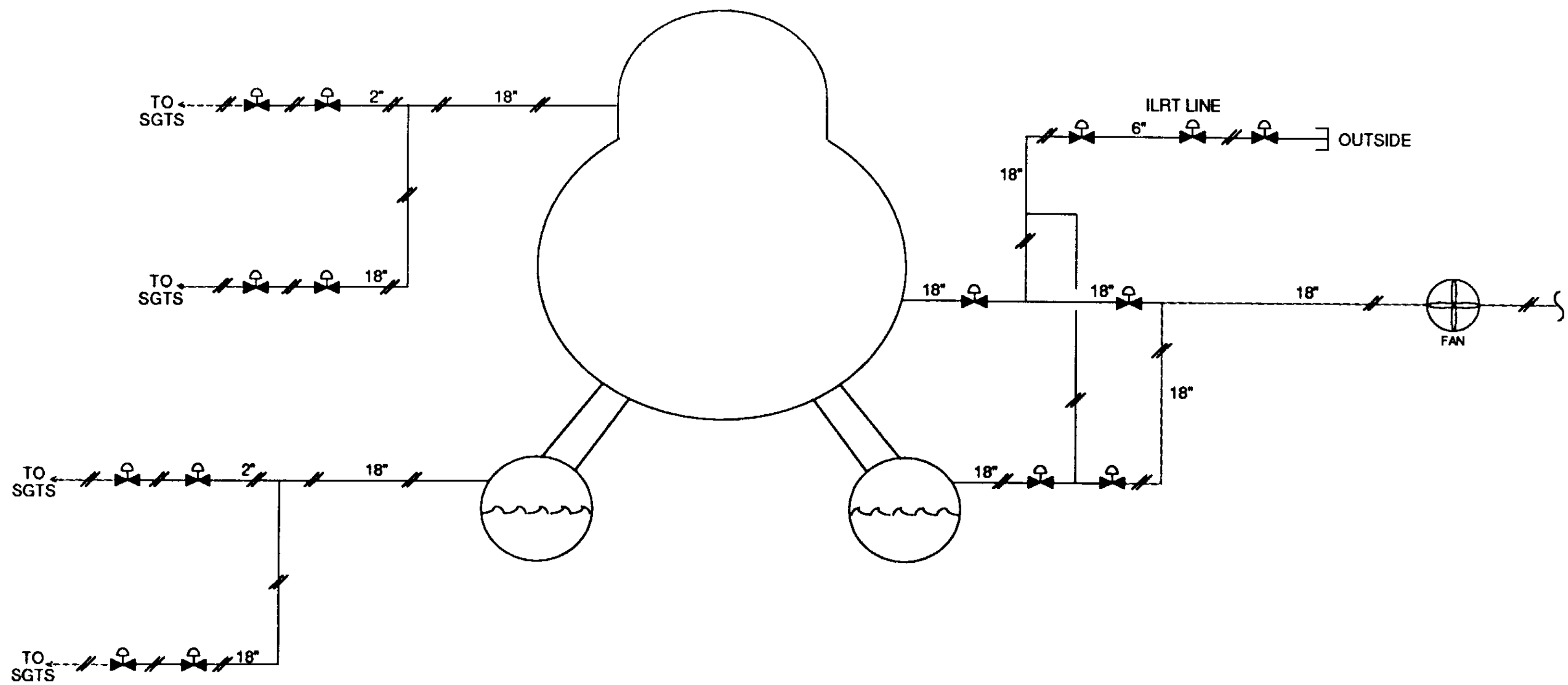
4.6.16 Primary Containment Venting System

4.6.16.1 PCV Description

When torus and containment sprays have failed to reduce primary containment pressure, the PCV is used to prevent a primary containment pressure limit from being exceeded (event tree nomenclature--Y).

The preferred primary containment vent paths include: (1) 2-in torus vent to the Standby Gas Treatment System (SGTS), (2) 6-in Integrated Leak Rate Test (ILRT) line from the torus, (3) 18-in torus vent path, (4) 18-in torus supply path, (5) 2-in drywell vent to the SGTS, (6) two 3-in drywell sump drain lines, (7) 6-in ILRT line from the drywell, (8) 18-in drywell vent path, and (9) 18-in drywell supply path. A simplified schematic of the PCV is provided by Figure 4.6.16-1.

For decay heat loads alone it is expected that the drywell pressure rise will be relatively slow. PCV success in this case is the 6-in vent path (or larger) being operational. However, if the rate of pressure rise is significantly faster as in the ATWS scenarios, success criteria dictate three or four 18-in vent paths as a minimum (assuming power levels ~15%). For further information, refer to success criteria discussions in Section 4.4.



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.16-1. Primary Containment Venting System Schematic.

Current venting procedure requires a vent path to be established if containment pressure rises to 100 psig (PECO is considering changing this to 60 psig). In the case of an ATWS, or if it can be inferred that the suppression pool is being bypassed, the operator is required to directly establish the 18-in vent paths.

4.6.16.2 PCV Interfaces and Dependencies

The PCV major dependencies are AC power and instrument air. A simplified dependency diagram of the PCV system is provided by Figure 4.6.16-2. Shown are the major support needs for the PCV system as indicated by the solid diamonds.

The drywell and torus vent paths to the SGTS are assumed to be successful whether or not the SGTS dampers are open. With the dampers closed, a rupture of the SGTS ducting in the reactor building is assumed to occur.

With a loss of instrument air, all air-operated valves fail closed. Backup air bottles are installed to facilitate opening air-operated valves locally.

With a loss of power, motor-operated valves fail in an "as is" position. These valves can still be opened with a handwheel or wrench on the stub protruding at the top of the motor operator.

4.6.16.3 PCV Test and Maintenance

The PCV system has no special test and maintenance requirements.

4.6.16.4 PCV Technical Specifications

The PCV system has no special technical specifications. However, the vent paths are used for inerting and de-inerting the containment as well as leak testing of the containment during refuelings.

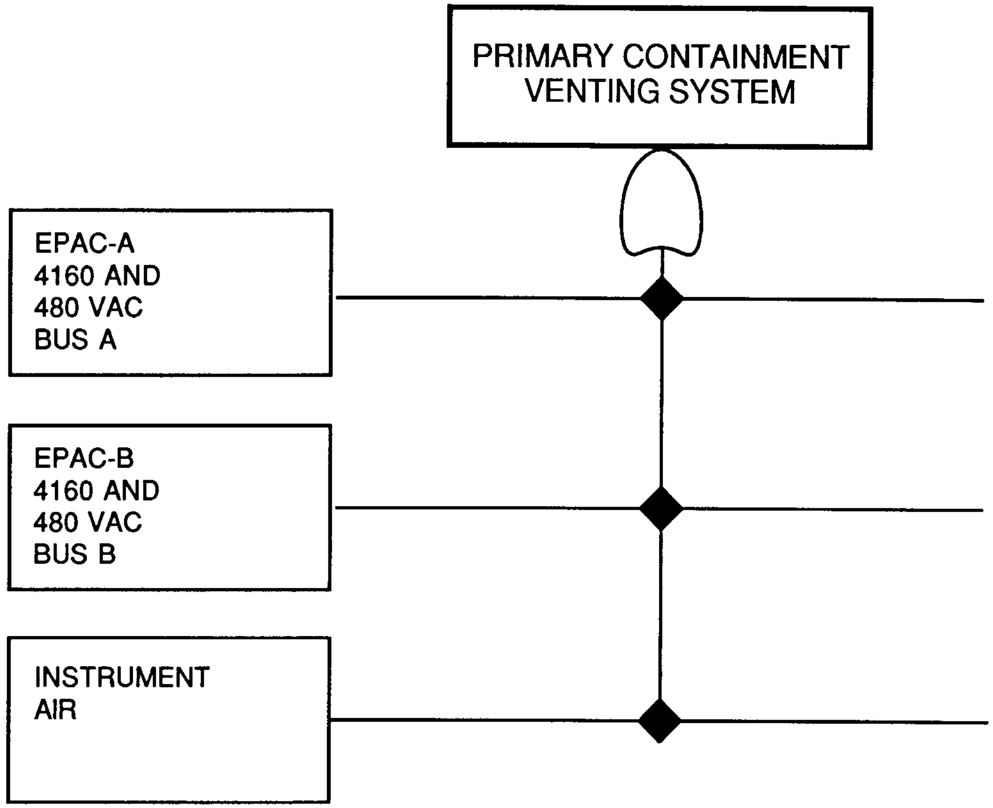
4.6.16.5 PCV Logic Models

The PCV system was modeled using a fault tree for reducing primary containment pressure. The fault tree has been simplified to cover only the major active components, interfaces and dependencies, and human errors. These have been lumped into one event. The PCV fault tree model is presented in Appendix B.

One human error was incorporated into the PCV fault tree model. That error was operator failure to vent.

4.6.16.6 PCV Assumptions

(1) Only major active components and major dependencies were modeled. These were assumed to dominate system failure.



Dependency Diagram Is Shown Using Failure Logic. Refer to the Fault Trees for Actual Failure Logic Details.

Figure 4.6.16-2. Primary Containment Venting System Dependency Diagram.

4.6.16.7 PCV Operational Experience

Nothing was peculiar in the operational history of the PCV system which would affect system modeling.

4.6.17 Reactor Building Cooling Water System

4.6.17.1 RBCW Description

The function of the RBCW system is to provide a means of cooling auxiliary plant equipment which is located primarily in the reactor building (e.g., recirculation pumps, sump coolers, radwaste, etc.). The RBCW system is a backup for cooling CRD pumps and IAS compressors and aftercoolers should the TBCW be lost.

The RBCW system is a closed loop system consisting of two full-capacity pumps, two full-capacity heat exchangers, one head tank, one chemical feed tank and associated piping, valves, and controls. The RBCW system is designed for an operating pressure of 140 psig. A simplified schematic of the RBCW system is provided by Figure 4.6.17-1.

The operator uses RBCW to cool certain critical loads if the TBCW system is lost. The RBCW system usually has one pump continuously operating. Control and instrumentation is designed for remote system startup from the main control room.

The success criteria for the RBCW system is one pump and one heat exchanger train operating, providing sufficient cooling to the loads. The cooling water pumps and heat exchangers are located in the reactor building auxiliary bay. The head tank is located on the reactor building refueling floor. The specific RBCW loads are distributed throughout different areas of the plant.

4.6.17.2 RBCW Interfaces and Dependencies

Cooling is maintained on critical equipment during failure of off-site power. Electrical power for operating the RBCW system pumps during such periods is supplied by the diesel generators.

In the event of off-site power failure,the ESW system can supply cooling water to the RBCW system. The RBCW system supply to the reactor cleanup system non-regenerative heat-exchanger is isolated, and the cooling water supply is maintained to the reactor recirculation pump motor oil and mechanical seal water coolers and the reactor building equipment drain sump cooler. In addition, cooling water is supplied to the drywell air cooling system and the drywell equipment drain sump cooler, which are nominally served by the chilled water system, and to the CRD pump oil coolers and air compressor jacket and after coolers, which are normally served by the TBCW system.

The RBCW system can also supply cooling water to the fuel pool cooling heat exchangers, via removable spool pieces, in the event of loss of normal cooling water.

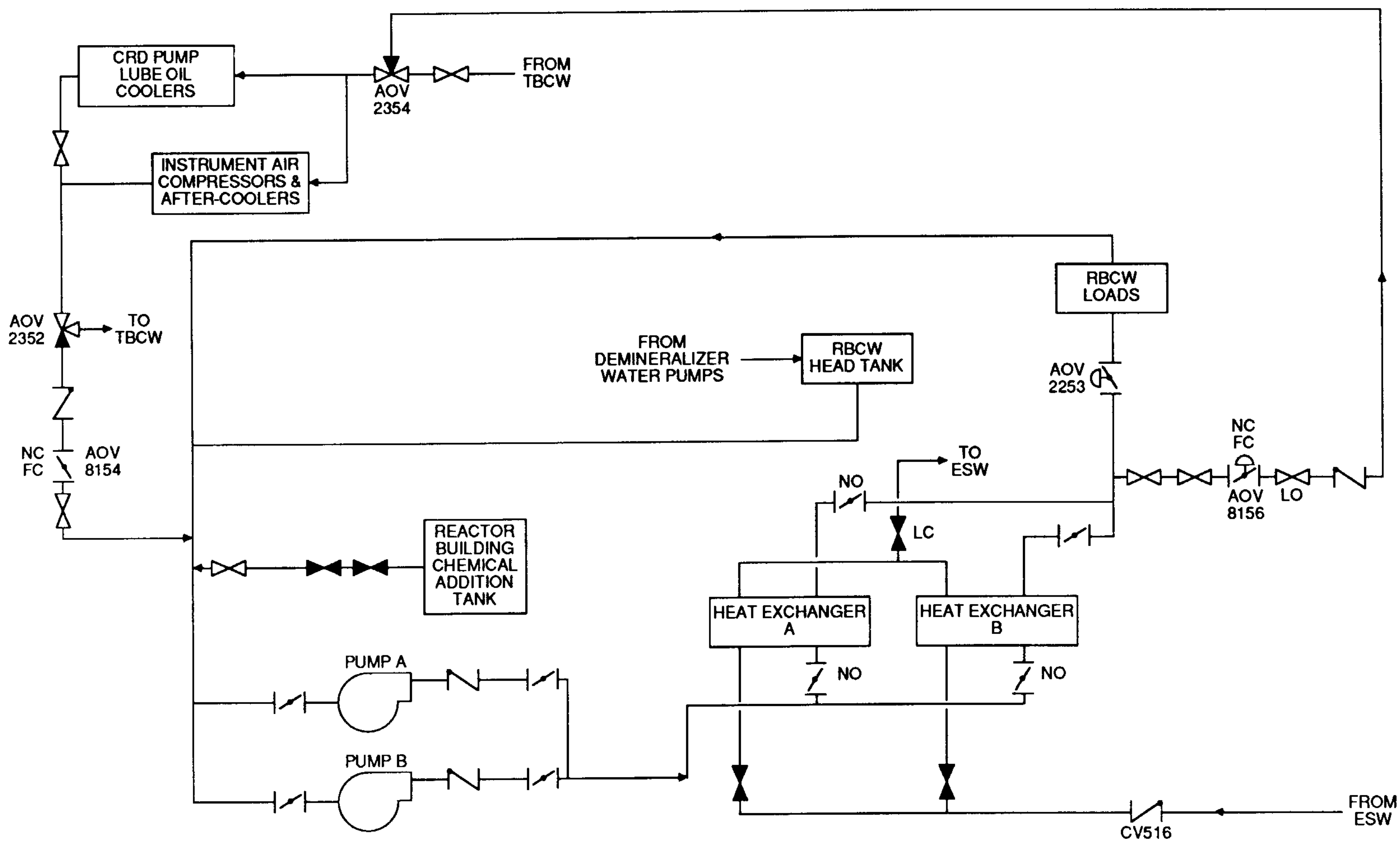


Figure 4.6.17-1. Reactor Building Cooling Water System Schematic.

A radiation monitor is provided at the cooling water return header to indicate, record, and alarm leakage of radioactivity.

A simplified dependency diagram of the RBCW system is provided by Figure 4.6.17-2. Shown are the major support needs as indicated by the solid diamonds.

4.6.17.3 RBCW Test and Maintenance

The RBCW system has no special test and maintenance requirements.

4.6.17.4 RBCW Technical Specifications

The RBCW system has no specific technical specifications.

4.6.17.5 RBCW Logic Model

The RBCW system was modeled using a fault tree for the loss of cooling water to auxiliary plant equipment. The fault tree has been simplified to cover only the major active components, interfaces and dependencies, and human errors.

The head tank and chemical addition tank were not modeled since they are passive devices and their failure probabilities are not expected to dominate system failure.

Seven human errors were incorporated into the RBCW fault tree. These errors are; failure to restore train 2354 valves after maintenance, failure to restore train 2352 valves after maintenance, operator failure to reclose the CRD-RBCW breakers given loss of off-site power occurs, failure to restore pump B train after maintenance, failure to restore manual valve 517 after maintenance, operator failure to open locked closed valves in the ESW system which cools RBCW, and pump B train failure to start due to operator error.

4.6.17.6 RBCW Assumptions

Only major active components and major dependencies were modeled. These were assumed to dominate system failure.

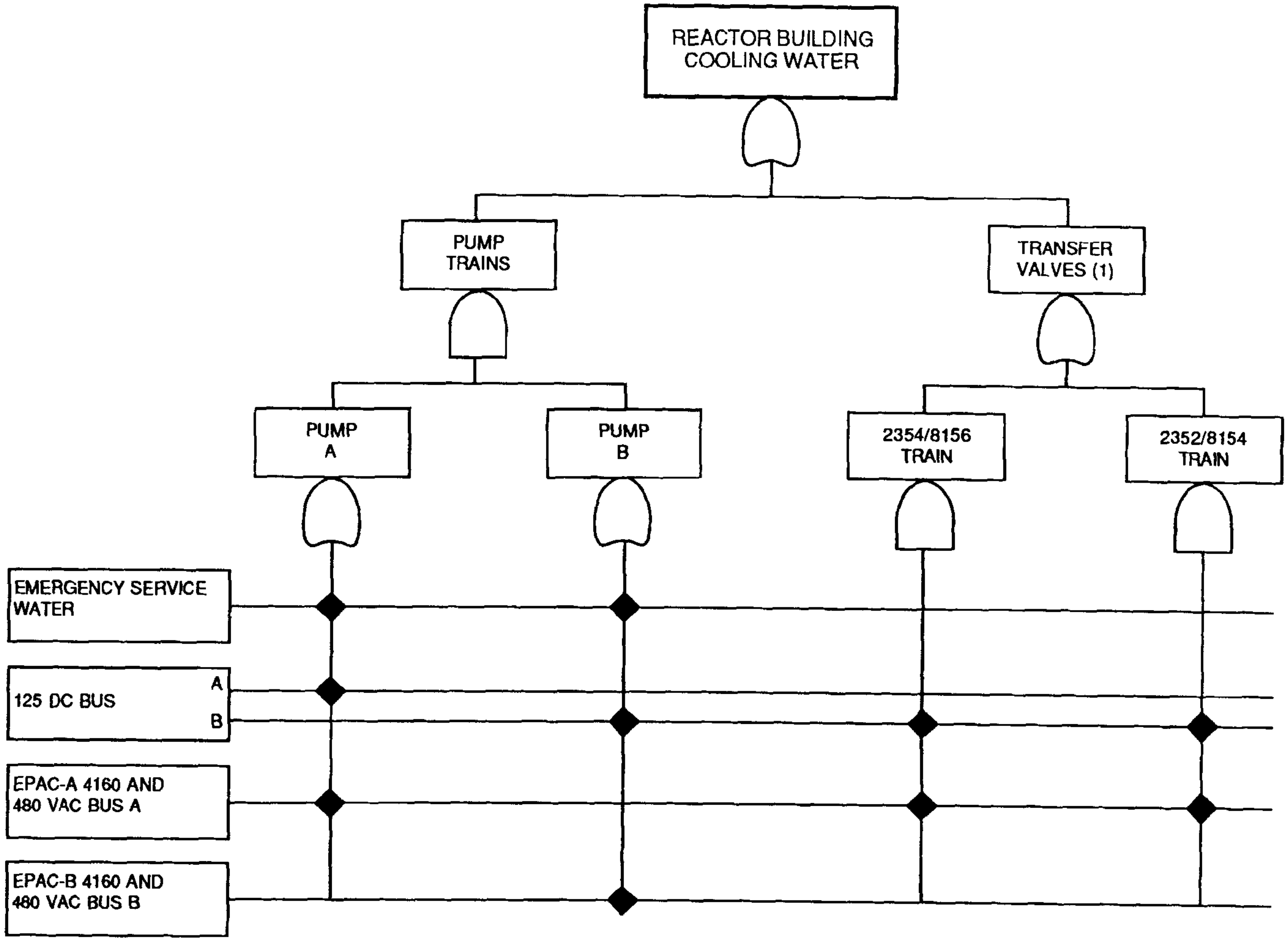
4.6.17.7 RBCW Operating Experience

There was nothing peculiar in the operational history of the RBCW system which would affect system modeling.

4.6.18 Reactor Core Isolation Cooling System

4.6.18.1 RCIC Description

The function of the RCIC system is to provide a makeup coolant source to the reactor vessel during accidents in which system pressure remains high (event tree nomenclature--U2).



Dependency Diagram Is Shown Using Failure Logic. Refer to the Fault Trees for Actual Failure Logic Details.
(1) For Cooling to the CRD Pumps and Instrument Air Equipment.

Figure 4.6.17-2. Reactor Building Cooling Water System Dependency Diagram.

The RCIC system consists of a single train with motor-operated valves and a turbine-driven pump. Suction is taken from either the CST or the suppression pool. Injection to the reactor vessel is via a feedwater line. The RCIC pump is rated at 600 gpm flow with a discharge head of 1135 psig. A simplified schematic of the RCIC system is provided by Figure 4.6.18-1. Major components are shown that were modeled in the system fault tree. The RCIC system is automatically initiated and controlled. Operator intervention is required as follows: (1) to prevent either vessel overfill or continuous system trip/restart cycles, (2) to manually start the system given an auto-start failure, and (3) to set up the system for continuous operation under long-term station blackout conditions.

The success criteria for the RCIC system is injection at rated flow to the reactor vessel. For further information, refer to success criteria discussions in Section 4.4.

Most of the RCIC system is located in a separate room in the reactor building. Local access to the RCIC system could be affected by either containment venting or containment failure should steam be released to the reactor building area. Room cooling failure is assumed to fail the RCIC pump in ten hours.

4.6.18.2 RCIC Interfaces and Dependencies

The RCIC system major dependencies are DC power for short term operation and room cooling for long term operation. Although there are AC powered motor-operated valves, these valves are not required to change state during normal system operation since they are only used to isolate the system. A simplified dependency diagram of the RCIC system is provided by Figure 4.6.18-2. Shown are the major support needs for the RCIC system as indicated by the solid diamonds.

The RCIC system requires both 250 VDC/A and 125 VDC/A. The 125 VDC/A is used for actuation and control power while an injection and a supply valve are powered from 250 VDC/A.

The RCIC and HPIC systems share a common CST suction valve. This is a normally open manual valve and is identified as XV-1 on the RCIC schematic. Failure of this valve will fail the CST as a suction source to both the RCIC and HPIC systems.

Upon system actuation, RCIC injection valves receive a signal to open and RCIC test valves receive a signal to close. The RCIC system is automatically initiated on the receipt of a low reactor water level signal (490 inches above vessel zero). The low reactor water level sensors are shared with the HPCI system.

The CST is the initial suction source for the RCIC system. Suction is automatically switched to the suppression pool on low CST level. Automatic switchover will not occur if there is an automatic isolation signal present. The CST suction valve does not close until both of the suppression pool suction valves are fully open.

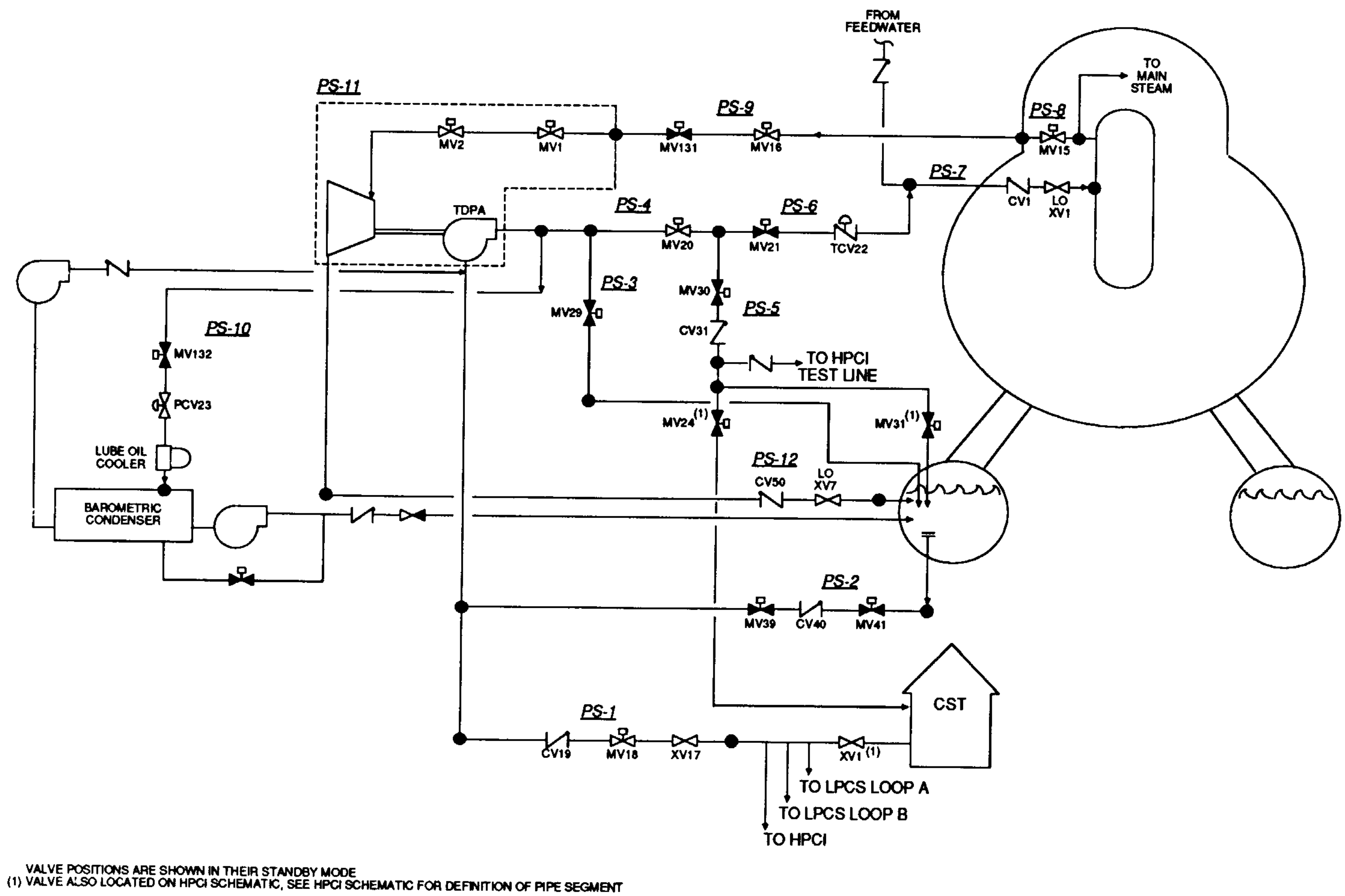
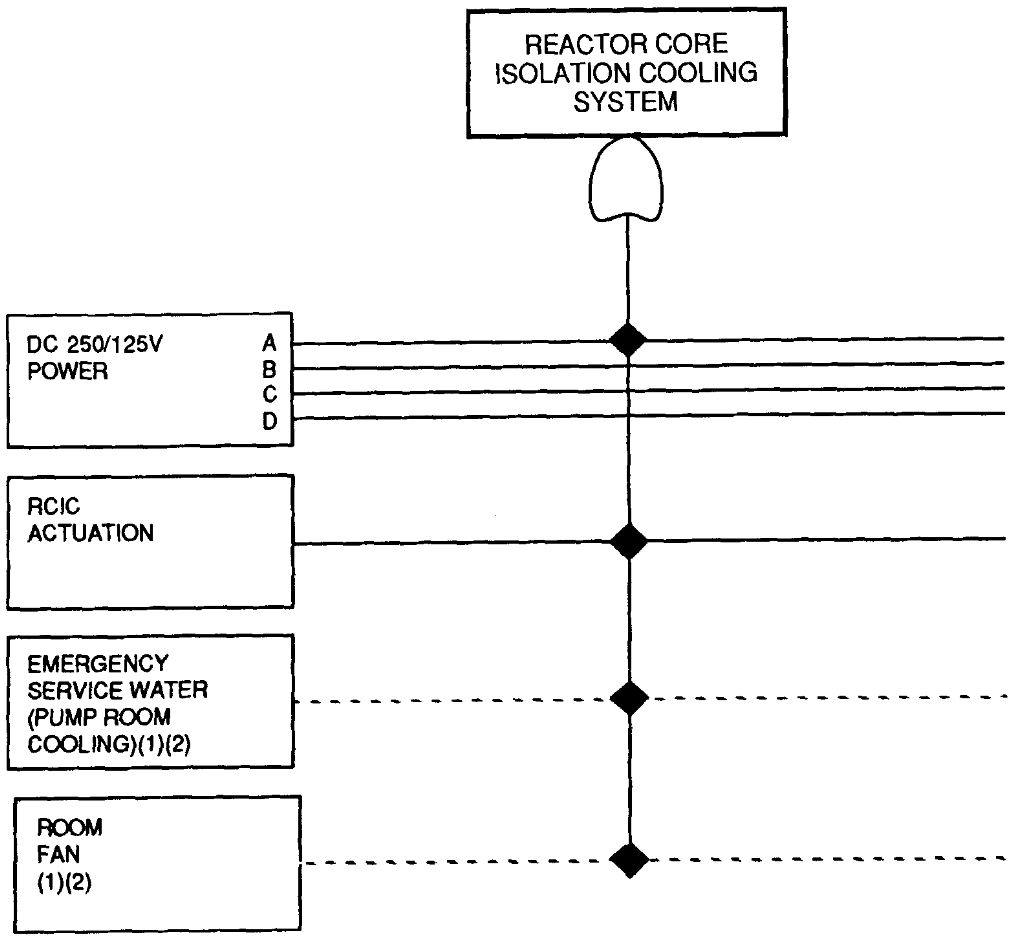


Figure 4.6.18-1. Reactor Core Isolation Cooling System Schematic.



Dependency Diagram Is Shown Using Failure Logic.
(1) Dependency Not Required During Short Term Operation.
(2) Room Cooling Can Also Be Performed By Opening Doors.

Figure 4.6.18-2. Reactor Core Isolation Cooling System Dependency Diagram.

The RCIC system is automatically isolated by high steam line space temperature, steam line high dP, or high turbine exhaust pressure (65 psia). Both the high temperature and high dP signals are used to detect a steam line break.

The RCIC turbine trips on high exhaust pressure, high reactor water level, low pump suction pressure, low steam pressure, or an auto isolation signal.

4.6.18.3 RCIC Test and Maintenance

The RCIC system surveillance requirements are the following: (1) pump operability--once/month, (2) motor-operated valve operability--once/month, (3) pump capacity test--once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.

4.6.18.4 RCIC Technical Specifications

If the RCIC system is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that ADS, HPIC, LPCI, and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.18.5 RCIC Logic Model

The RCIC system was modeled using a fault tree for the injection of coolant to the reactor vessel. The major active components were modeled for the RCIC system. The fault tree model representing the RCIC system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only the piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

The barometric condenser condensate pump and vacuum pump were not modeled since their operation is not essential to system operation.

Seven human errors were incorporated into the RCIC fault tree model. These errors are (1) failure to trip the RCIC system and realign its suction source on low suction pressure, (2) failure to realign the suction source for the RCIC and HPCI systems in other circumstances, (3) failure to control RCIC flow (reactor level), (4) failure to manually backup automatic RCIC actuation, (5) miscalibration of CST level sensors, (6) miscalibration of certain ESF sensors, and (7) failure to isolate the RCIC system given high exhaust pressure.

4.6.18.6 RCIC Assumptions

(1) The RCIC test return lines were not considered as potential diversion paths because the probability of two normally closed MOVs failing to prevent flow was felt to be negligible compared to other system faults.

(2) Failure of the system to isolate given certain conditions was not considered since the system is effectively "non-operational." These conditions are (a) high steam line space temperature, (b) high steam line dP, (c) low steam pressure, (d) high steam line exhaust pressure, and (e) manual isolation.

(3) Failure of the minimum flow line to open does not constitute system failure since the time between pump start and opening of the injection valve is small.

(4) The barometric condenser condensate pump and vacuum pump are not necessary for system operation. Therefore, their failures were not modeled.

(5) Spurious signals are felt to be negligible compared to other system failures because of their low probability of occurrence.

(6) The RCIC system is assumed to fail in a non-recoverable state if it fails to trip on low suction pressure or high reactor water level because of expected damage to the pump or turbine.

(7) RCIC pump bearing cooling fails if pump suction is from the suppression pool and the working fluid temperature reaches between 210 and 260°F. In the analyses, this was nominally assumed to occur at 250°F without any uncertainty in order to facilitate the analysis. Therefore, the uncertainty in the results does not reflect the temperature range over which failure might occur.

(8) The RCIC turbine shaft-driven oil pump, stop valve, and governor valve failures were included in turbine failure data.

(9) System failure because of valves being left in the wrong position after test or maintenance is felt to be small compared to other system faults. The position of key manual and motor-operated valves is indicated in the control room and the motor-operated valves receive signals to realign on an actual demand. System valves must be in their correct positions before startup of the plant following shutdown and concurrent maintenance activities. In addition, PECO maintains a control log of all "locked" valves in the plant to assure their correct position.

(10) Testing of TCV22 (PS-6) will not prevent flow from reaching the reactor vessel should a real demand occur.

(11) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. Maintenance would require components to be

effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(12) An event for depletion of the CST was included for those cases where RCIC and/or HPCI operation was judged to be sufficiently long.

(13) Failure of the suppression pool by random failure or the plugging of its strainers is felt to be negligible compared to other system failures.

(14) If the HPCI or RCIC minimum flow line has been demanded open and subsequently fails to close on a system trip, there is the possibility that the CST will drain to the suppression pool from their difference in elevation.

(15) Lube oil cooling is required for bearing cooling.

(16) The RCIC actuation circuitry was not modeled to a great degree of detail. Only elements which were felt to be potentially important were included in the fault tree model. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems. The power supply for the actuation circuitry was also included. Hardware failures of relays and certain permissives were grouped into one basic event.

(17) It is assumed that calibration of the low and low-low reactor vessel water level sensors is performed at the same time. Miscalibration of these sensors is assumed to be the same event.

(18) Failure to recover an initial loss of the normal suction source (the CST) will be treated as a recovery action. Operator error appears to dominate failures of suppression pool valves and their manual actuation circuitry. Failure of suppression pool valves from maintenance outages or support system failures appears elsewhere in the fault tree.

(19) Failure of the system to automatically realign to the suppression pool after a loss of the normal suction source (the CST) is treated explicitly with manual switchover being treated as a recovery action.

(20) The suction pressure trip is "ANDed" with a dummy event to account for the probability that low suction pressure exists.

(21) The operator is required to manually reset the RCIC turbine trip valve if either high steam flow or high steam line temperature occurs. Manual reset is not required for either high reactor water level or low suction pressure.

(22) System unavailability from testing is considered small compared with other system faults since it appears that the majority of testing requirements would not preclude proper system operation following a real demand. Hence this contribution to failure of the system is small compared with other system failure probabilities.

(23) Failure of room cooling (if not recovered) fails RCIC in ten hours. This is based on an utility calculations [52] which demonstrate that in 100 hours without room cooling, operability is expected assuming intermittent pump operation. Since in the accident sequences of interest continuous operation may be performed, this value was readjusted to ten hours using engineering judgement.

4.6.18.7 RCIC Operating Experience

Nothing was peculiar in the operational history of the RCIC system which would affect system modeling. Plant operational data indicates a higher value for TDP failure to run than the generic data base. The difference is that the generic value was calculated using plant operational hours instead of RCIC operational hours. The values compare closely when RCIC operational hours are used in the generic calculation. Therefore, the plant specific value for TDP failure to run is used.

4.6.19 Residual Heat Removal: Shutdown Cooling System

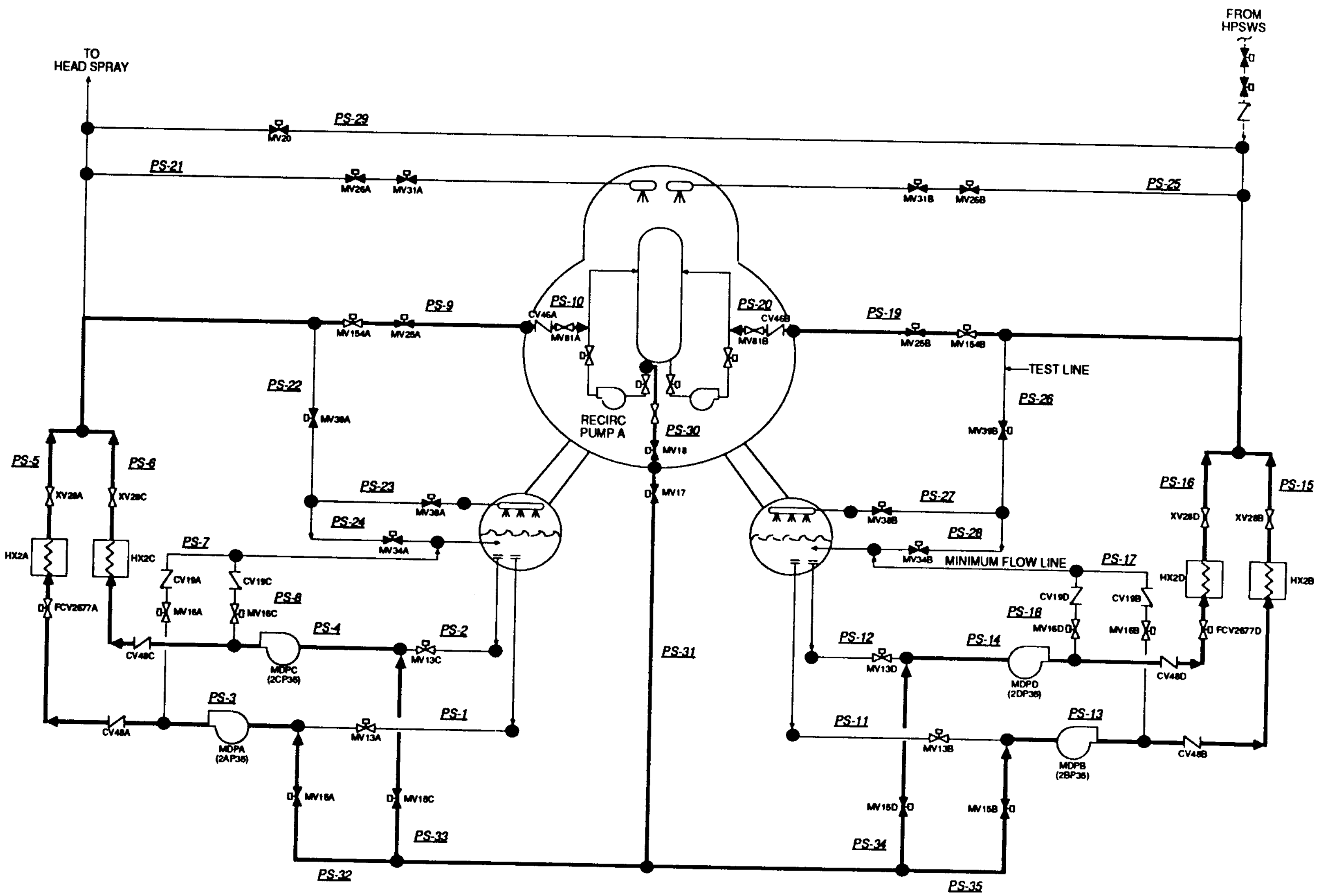
4.6.19.1 SDC Description

The function of the SDC system is to remove decay heat during accidents in which reactor vessel integrity is maintained (event tree nomenclature--W2). The SDC system is but one mode of the RHR system and, as such, shares components with other modes.

The RHR system is a two-loop system consisting of motor-operated valves and motor-driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm with a discharge head of 540 feet. Cooling water flow to the heat exchanger is required for the SDC mode. The SDC system suction source is one recirculation pump's suction line. A simplified schematic of the SDC (RHR) system is provided by Figure 4.6.19-1 with the SDC system highlighted. Major components are shown as well as the pipe segment definitions (e.g., PS-9) used in the system fault tree.

The SDC system is manually initiated and controlled.

The success criterion for the SDC system is injection of flow from any one pump/heat exchanger train to the reactor vessel. For further information, refer to success criteria discussions in Section 4.4.



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.19-1. Residual Heat Removal System - Shutdown Cooling Mode Schematic.

Most of the SDC system is located in the reactor building. Level access to the SDC system could be affected by either containment venting or containment failure. Room cooling failure is assumed to fail the SDC pumps in ten hours.

4.6.19.2 SDC Interfaces and Dependencies

Each SDC pump is powered from a separate 4160 VAC bus with control and actuation power being supplied by a separate 125 VDC bus. All pumps require pump cooling. For further information on pump cooling, refer to Section 4.6.9.8. A simplified dependency diagram of the SDC system is provided by Figure 4.6.19-2. Shown are the major support needs of the SDC system as indicated by the solid diamonds.

Each loop's normally closed injection valve receives motive power from one of two 480 VAC sources. The Loop A injection valve sources are either 480 VAC/A or 480 VAC/C, and the Loop B injection valve sources are either 480 VAC/B or 480 VAC/D.

Many components of the SDC system are shared with the different modes of the RHR system. These commonalities are as follows: (1) the RHR pumps are common to the SDC, SPC, CS, and LPCI modes; (2) Loops A and B injection valves are common to the SDC, LPCI, and HPSW injection modes; and (3) heat exchanger cooling is common to the CS, SDC, and SPC modes.

The two SDC suction valves (MV18 and MV17) are common to all four SDC pumps. MV18 requires 480 VAC/A and MV17 requires 250 VDC/B. Complete failure of the SDC system will occur if either of these valves fails to open.

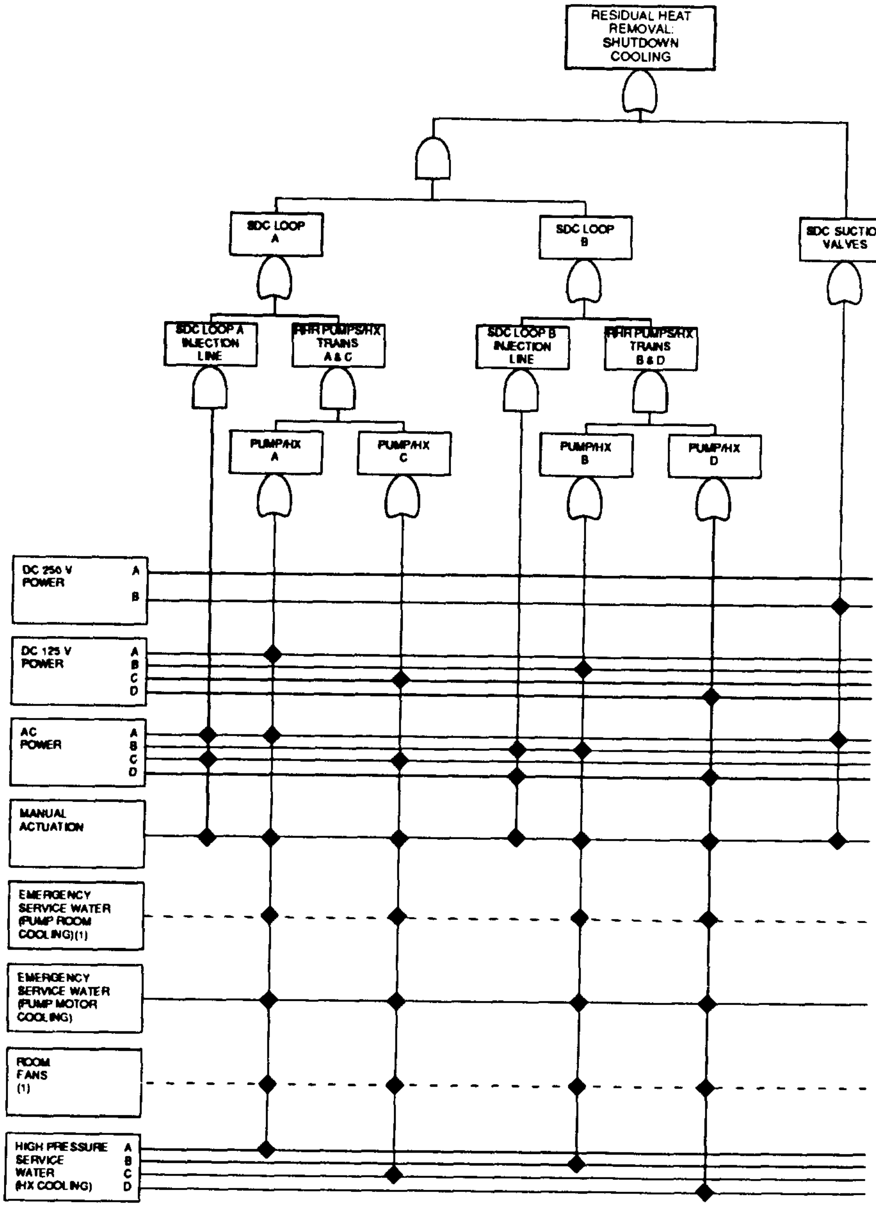
Each pump's suppression pool suction valve and SDC cooling suction valve are interlocked. One valve must be fully closed before the other valve can be opened.

SDC is initiated after emergency core injection is successful and reactor pressure is low. If an injection signal subsequently occurs, the RHR system will automatically be realigned to the LPCI mode. SDC cannot be initiated if any of the following conditions exist: (a) reactor pressure greater than 225 psig, (b) high drywell pressure, or (c) low reactor water level.

SDC pumps will stop or be prevented from starting if a suction path is not available.

4.6.19.3 SDC Test and Maintenance

The SDC surveillance requirements are the following: (1) pump operability-- once/month, (2) MOV operability--once/month, (3) pump capacity test---once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.



Dependency Diagram Is Shown Using Failure Logic.
(1) Dependency Not Required During Short Term Operation.

Figure 4.6.19-2. Residual Heat Removal System - Shutdown Cooling Mode Dependency Diagram.

4.6.19.4 SDC Technical Specifications

To the extent that the SDC and LPCI modes are shared, certain technical specifications are required because of the LPCI mode of the RHR system. If any one LPCI pump is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the remaining LPCI components and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.19.5 SDC Logic Model

The SDC system was modeled using a fault tree for removal of decay heat from the reactor vessel following transients. The major active components were modeled for the SDC system. The fault tree model representing the SDC system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system piping was considered as a diversion path.

Three human errors were incorporated into the SDC fault tree model. These errors are miscalibration of various sensors, failure of manual initiation, and failure to properly restore key components following maintenance.

4.6.19.6 SDC Assumptions

(1) Positions of all manual and motor-operated valves are indicated in the control room. Failure of these valves after testing and maintenance due to incorrect positioning is therefore felt to be negligible. The injection valves receive open signals on a real demand. Thus, unavailability due to testing and failure to restore after testing is not important.

(2) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. It was assumed that maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(3) Pump isolation because of spurious signals is assumed to be negligible compared to other system faults.

(4) The SDC control circuitry was not modeled at a great level of detail. Only elements which were felt to be potentially important were included in the fault tree model. Hardware failure of relays and permissive is grouped into one term. The permissive/isolation signal sensors and their support systems were explicitly modeled since they could be potentially important to system failure.

(5) Based on a PECO response, the SDC pumps will fail because of insufficient NPSH once the suppression pool has reached saturated conditions.

(6) SDC failure because of a test diverting flow is felt to be negligible because this mode is manually initiated and aligned.

(7) A suction path must be available from either the suppression pool or the SDC path to start a SDC pump.

(8) Failure of the suppression pool because of random failure or the plugging of all its strainers is assumed to be negligible compared to other system failures.

(9) The unavailability of the SDC pumps from testing does not defeat a real demand from operating the system. Therefore, it was not considered. Failure to restore the SDC pumps after testing does not apply.

(10) Pump room cooling is discussed in LPCI Section 4.6.14.6.

4.6.19.7 SDC Operating Experience

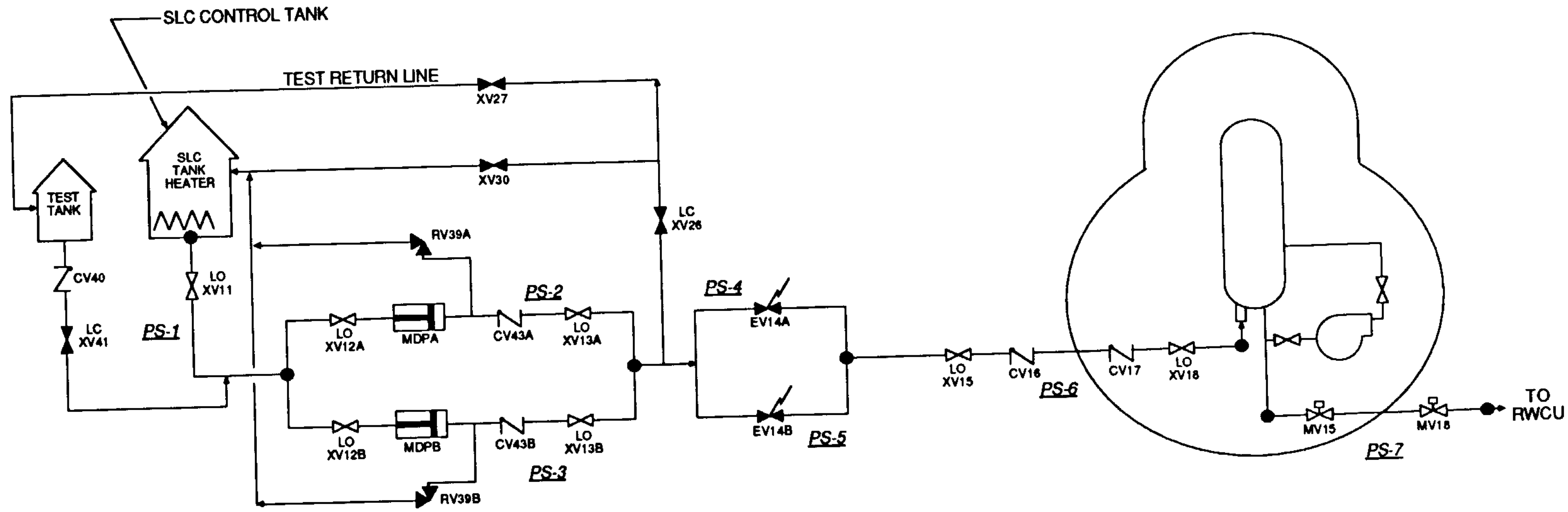
Nothing was peculiar in the operational history of the SDC system which would affect either system modeling or failure data.

4.6.20 Standby Liquid Control System

4.6.20.1 SLC Description

The SLC system provides a backup method, which is redundant but independent of the control rods, to establish and maintain the reactor subcritical (ATWS event tree nomenclature--SLC).

The suction for the SLC system comes from a control tank. The control tank has sodium pentaborate in solution with demineralized water. Two parallel positive displacement pumps are each sized to inject the sodium pentaborate solution into the reactor. Two parallel explosive valves are downstream of the pumps' common discharge. SLC discharge enters the reactor vessel near the bottom of the core shroud where it mixes with cooling water rising through the core. A simplified schematic of the SLC system is provided by Figure 4.6.20-1.



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.20-1. Standby Liquid Control System Schematic.

The operator manually activates the SLC system with a three-position keylock switch on the control room console. If the pump lights or the explosive valve light indicate that liquid may not be flowing, the operator can turn the keylock switch to the other side to operate the other pump.

The success criteria for the SLC system are one of two pumps running and one of two explosives valves open.

Most of the SLC system is located in the reactor building outside of the drywell. Local access to the SLC system could be affected by containment failure or containment venting.

4.6.20.2 SLC Interfaces and Dependencies

SLC Pump A is powered from 480 VAC/A with control and actuation power supplied by 125 VDC/A. SLC Pump B is powered from 480 VAC/B with control and actuation power supplied by 125 VDC/B. Both pumps are self-cooled and do not require room cooling. A simplified dependency diagram of the SLC system is provided by Figure 4.6.20-2. Shown are the major support needs for the SLC system as indicated by the solid diamonds.

The SLC system has a common test return line. This piping originates at the pumps combined discharge. If this line is not isolated following a test, pump discharge in the event of system actuation would preferentially flow to either the test or control tanks.

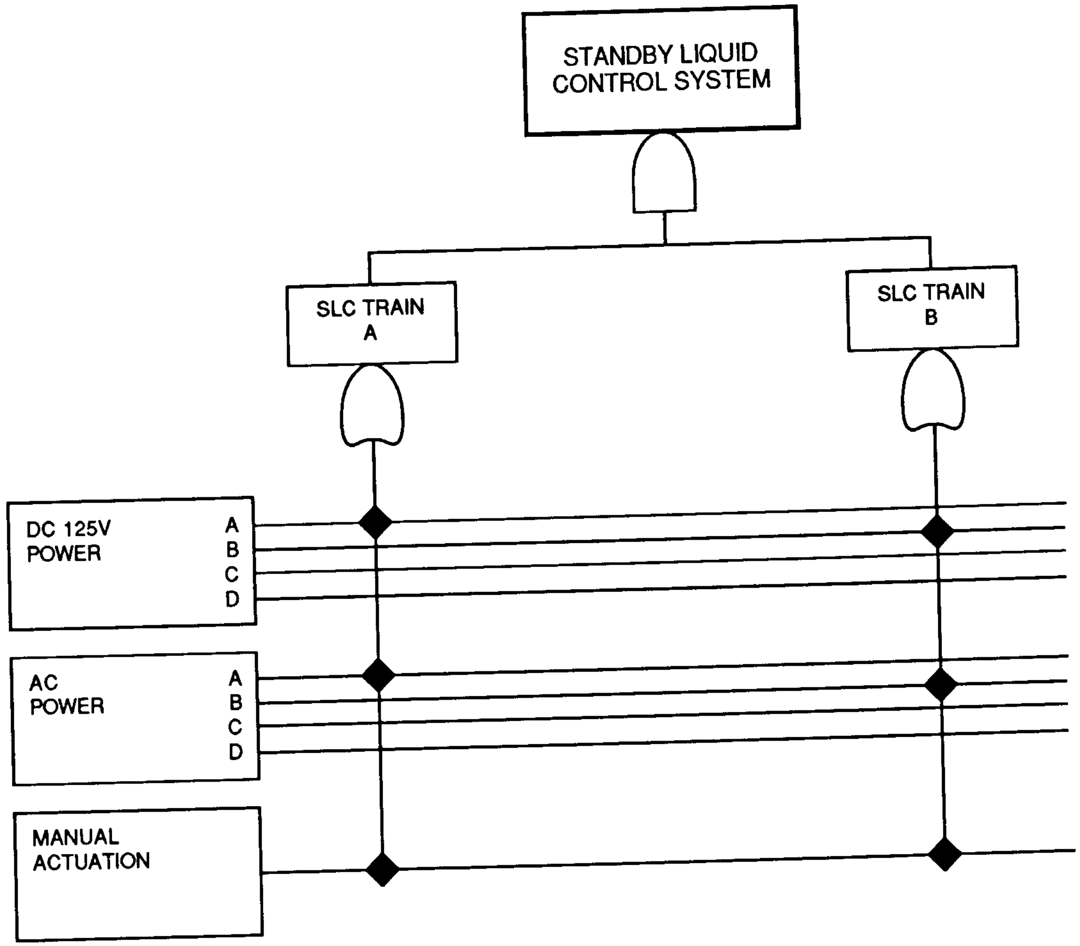
Switching from "Off" to either "Pump A" or "Pump B" on the three-position keylock switch starts the respective pump, opens both explosive valves, and closes the Reactor Water Cleanup (RWCU) system isolation valves (see PS-7, Figure 4.6.20-1). The RWCU isolation valves are closed to prevent loss or dilution of the boron.

The SLC pumps have control room informational lights. A green light indicates that power is available to the pump motor contractor but the contractor is open and the pump is not running. A red light indicates the contractor is closed and the pump is running.

The explosive valve shearing plunger is actuated by an explosive charge having dual ignition primers. Ignition circuit continuity is monitored by a trickle current. If either explosive valve circuit opens, a control room alarm actuates.

4.6.20.3 SLC Test and Maintenance

Once per month each pump loop is functionally tested by recirculating demineralized water to the test tank. The SLC system is tested once every operating cycle as follows: (1) relief valve settings are checked, (2) the system is manually initiated except for the explosive valves, and (3) one SLC pump takes suction from the test tank and discharges demineralized water into the reactor vessel. Both systems, including both explosive valves, are tested in the course of two operating cycles. When a component is found to be inoperable, its redundant component is to



Dependency Diagram Is Shown Using Failure Logic.

Figure 4.6.20-2. Standby Liquid Control System Dependency Diagram.

be demonstrated operable immediately and on a daily basis thereafter until the inoperable component is repaired.

4.6.20.4 SLC Technical Specifications

When fuel is in the reactor and prior to cold startup, the SLC system must be operable. With a redundant component inoperable, continued reactor operation is allowed for seven days.

5.6.20.5 SLC Logic Model

The SLC system was modeled using a fault tree for the injection of sodium pentaborate into the reactor vessel.

Besides major components, human errors were incorporated into the SLC system fault tree. These errors include operator failure to start the system and operator failure to properly restore the system following test and maintenance. Unavailability of the system during testing was also modeled.

4.6.20.6 SLC Assumptions

(1) Pipe segments less than one third of the main system pipe diameter are not considered to be diversion paths.

(2) Failure to heat the sodium pentaborate solution is not assumed to fail the system, based on information in the Peach Bottom UFSAR. [11]

4.6.20.7 SLC Operating Experience

Nothing was peculiar in the operational history of the SLC system which would affect either system modeling or failure data.

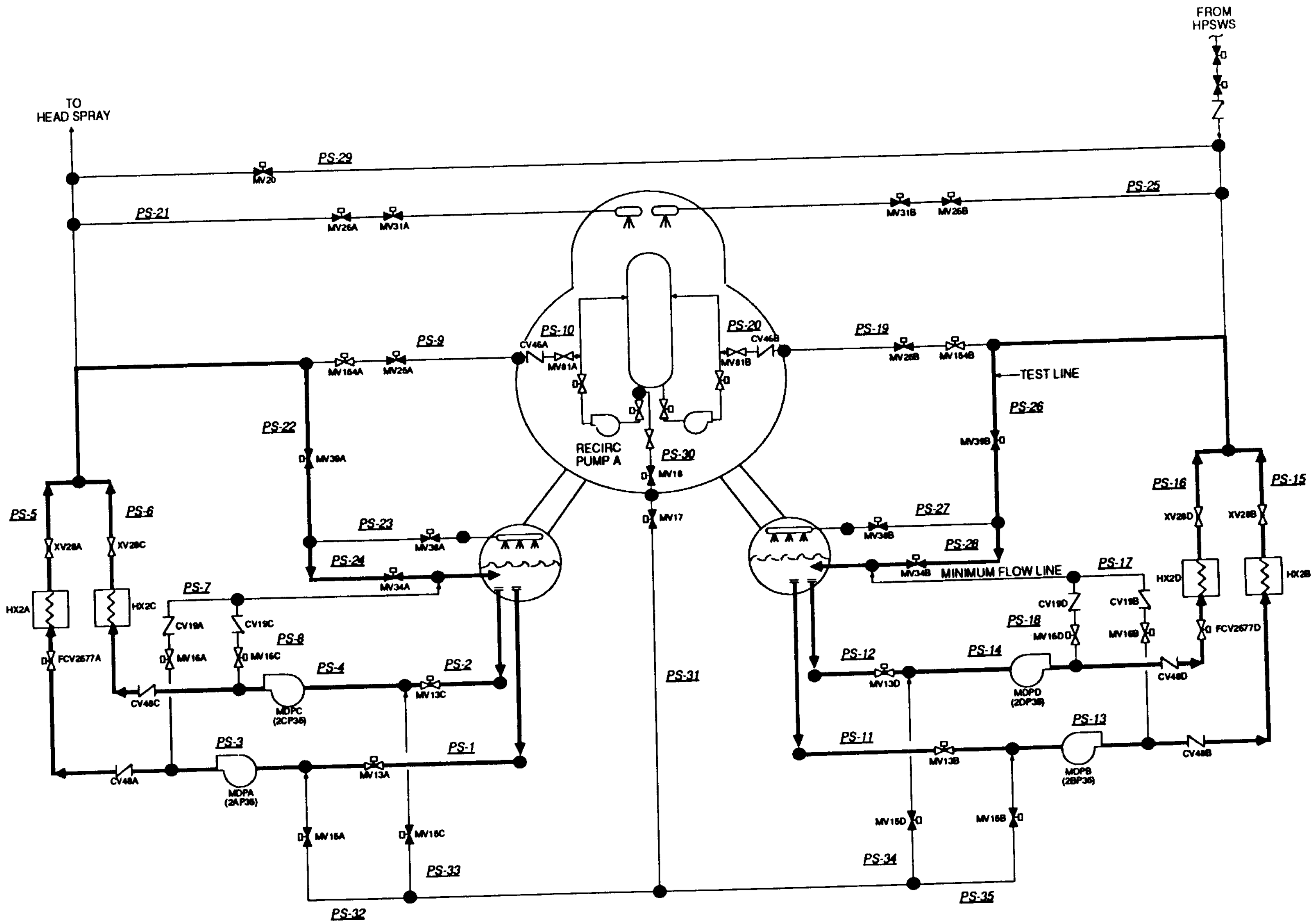
4.6.21 Residual Heat Removal: Suppression Pool Cooling System

4.6.21.1 SPC Description

The function of the SPC system is to remove decay heat from the suppression pool during accidents (event tree nomenclature--W1). The SPC system is but one mode of the RHR system and, as such, shares components with other modes.

The RHR system is a two-loop system consisting of motor-operated valves and motor-driven pumps. There are two pump/heat exchanger trains per loop, with each pump rated at 10,000 gpm with a discharge head of 540 feet. Cooling water flow to the heat exchanger is required for the SPC mode. The SPC suction source is the suppression pool. A simplified schematic of the SPC (RHR) system is provided by Figure 4.6.21-1 with the SPC mode highlighted. Major components are shown as well as the pipe segment definitions (e.g., PS-26) used in the system fault tree.

The SPC system is manually initiated and controlled.



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY MODE

Figure 4.6.21-1. Suppression Pool Cooling System Schematic.

The success criterion for the SPC system is injection of flow from any one pump/heat exchanger train to the suppression pool. For further information, refer to success criteria discussions in Section 4.4.

Most of the SPC system is located in the reactor building. Local access to the SPC system could be affected by either containment venting or containment failure. Room cooling failure fails the SPC pumps in ten hours.

4.6.21.2 SPC Interfaces and Dependencies

Each SPC pump is powered from a separate 4160 VAC bus with control and actuation power being supplied by a separate 125 VDC bus. All pumps require pump cooling. For further information on pump cooling, refer to Section 4.6.9.8. Each loop's normally closed suppression pool inlet valve receives motive power from one 480 VAC source. A simplified dependency diagram of the SPC system is provided by Figure 4.5.21-2. Shown are the major support needs of the SPC system as indicated by the solid diamonds.

Many components of the SPC system are shared with the different modes of the RHR system. These commonalities are as follows: (1) the RHR pumps are common to the SPC, LPCI, CS, and SDC modes; (2) the suppression pool suction valve for each pump train is common to the SPC, LPCI, and CS modes; and (3) heat exchanger cooling is common to the CS, SDC, and SPC modes.

SPC control circuitry is divided into two divisions. Division A is associated with control of components in Loop A, and Division B is associated with control of components in Loop B.

The SPC mode is manually initiated. If an injection signal is generated subsequent to the initiation of the SPC system, the SPC system will automatically realign to the LPCI mode. Besides a time delay, a permissive indicating that the reactor water level is above the shroud (312 inches above vessel zero) must be present prior to aligning to the SPC mode. However, this permissive may be overridden by a switch in the control room.

The SPC control circuitry is not common to the LPCI actuation and control circuitry but is shared with the CS mode. Reactor water level sensors are shared with the CS system.

Although the SPC system has no isolation signals, there are permissives which will prevent the operation of certain components. SPC pumps are demanded to stop or prevented from starting if the suppression pool suction valve or any of three SDC suction valves is not fully open.

4.6.21.3 SPC Test and Maintenance

The SPC surveillance requirements are the following: (1) pump operability--once/month, (2) MOV operability--once/month, (3) pump

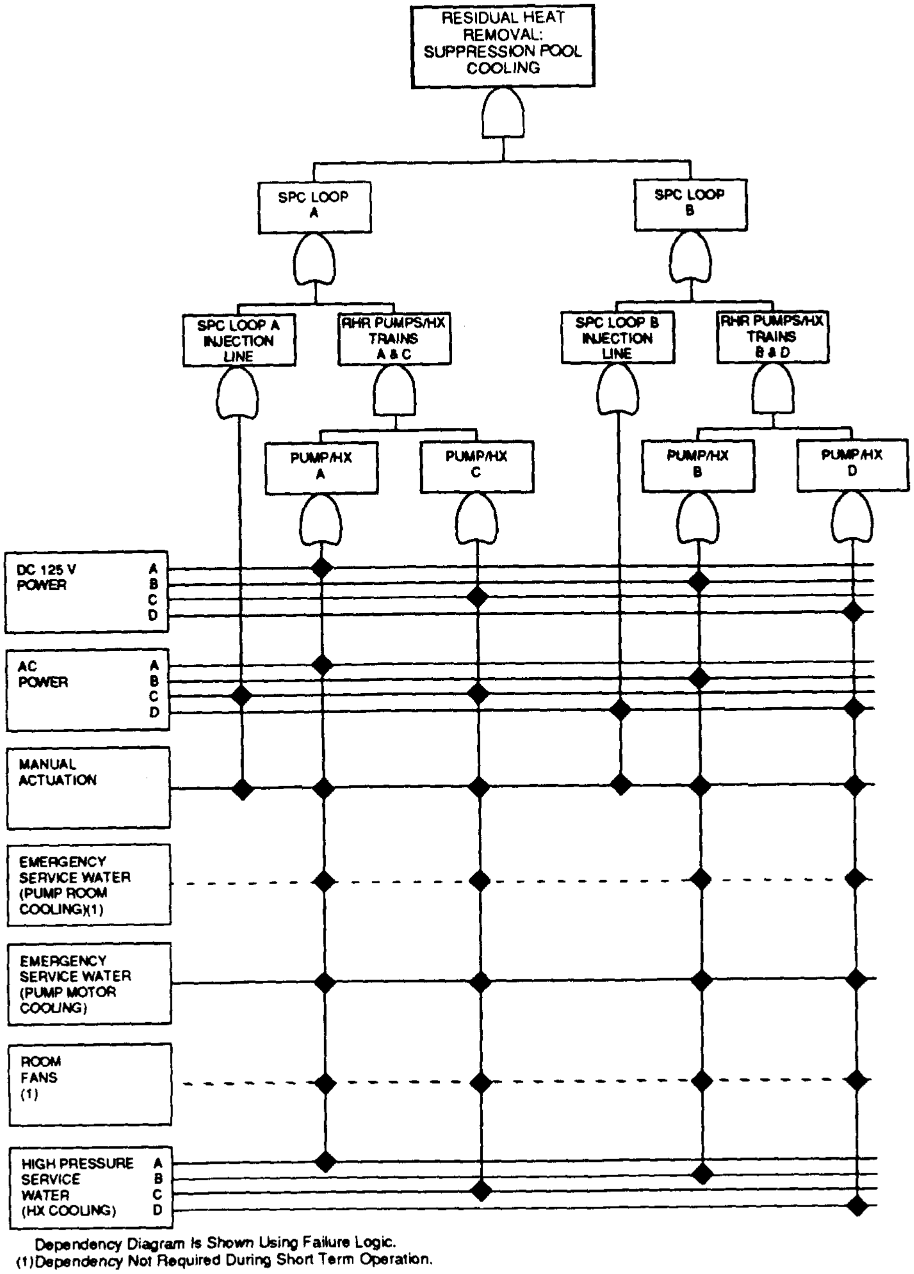


Figure 4.6.21-2. Suppression Pool Cooling System Dependency Diagram.

capacity test--once/three months, (4) simulated automatic actuation test--once/operating cycle, and (5) logic system functional test--once/six months.

4.6.21.4 SPC Technical Specifications

Technical specifications exist because of sharing of the SPC and LPCI modes of the RHR system. If any one LPCI pump is made or found to be inoperable for any reason, continued reactor operation is permissible for seven days provided that the remaining LPCI components and both loops of the LPCS system are operable. If this requirement cannot be met, the reactor is to be shut down.

4.6.21.5 SPC Logic Model

The SPC system was modeled using a fault tree for the removal of decay heat from the suppression pool. The major active components were modeled for the SPC system. The fault tree model representing the SPC system is presented in Appendix B.

Piping ruptures were considered to be negligible compared to other system failures. Only piping with a diameter of greater than or equal to one third of the main system piping was considered as a potential diversion path.

Three human errors were incorporated into the SPC fault tree model. These errors are failure of manual initiation, failure to override an erroneous shroud level permissive signal, and failure to properly restore key components following maintenance.

4.6.21.6 SPC Assumptions

(1) Positions of all manual and motor-operated valves are indicated in the control room. Failure of these valves after testing and maintenance due to incorrect positioning is therefore felt to be negligible. The injection valves receive open signals on a real demand. Thus, unavailability due to testing and failure to restore after testing is not important.

(2) During construction of the fault tree, it was necessary to determine which components could be taken OOS for maintenance. Maintenance would require components to be effectively removed from the system. Standard safety precautions of component isolation were used to decide which components could be taken OOS for maintenance while the plant was at power or normal operating pressure. The general guidelines used for component isolation were double blockage for high pressure piping or components and single blockage for low pressure piping or components.

(3) Pump isolation because of spurious signals is assumed to be negligible compared to other systems faults.

(4) The SPC control circuitry was not modeled at a great level of detail. Only elements which were felt to be potentially important were included in the fault tree model. Except for the shroud water level permissive, high drywell pressure permissive, pump power permissive, and pump suction source relay, the hardware failures of relays and permissives are grouped into one term. The initiating signal sensors and their support systems were explicitly modeled since they are shared between various ESF systems.

(5) Based on a PECO response, the SPC pumps will fail because of insufficient NPSH once the suppression pool has reached saturated conditions.

(6) Diversion of flow to the containment spray line is felt to be negligible compared to other system failures.

(7) A suction path must be available from either the suppression pool or the SDC path to start a SPC pump.

(8) Failure of the suppression pool because of random failure or the plugging of all its strainers is assumed to be negligible compared to other system failures.

(9) The unavailability of the SPC pumps from testing does not defeat a real demand from operating the system. Therefore, it was not considered. Failure to restore the SPC pumps after testing does not apply.

(10) Pump room cooling is discussed in LPCI Section 4.6.14.6.

4.6.21.7 SPC Operating Experience

Nothing was peculiar in the operational history of the SPC system which would affect either system modeling or failure data.

4.6.22 Turbine Building Cooling Water System

4.6.22.1 TBCW Description

The function of the TBCW system is to provide cooling water to auxiliary plant equipment associated with the power conversion system.

The TBCW system is a closed loop system consisting of two full-capacity pumps, two full-capacity heat exchangers, one head tank, one chemical fuel tank and associated piping, valves and controls. A simplified schematic of the TBCW system is provided by Figure 4.6.22-1.

The TBCW system is normally running. One pump is required to supply cooling to all TBCW loads.

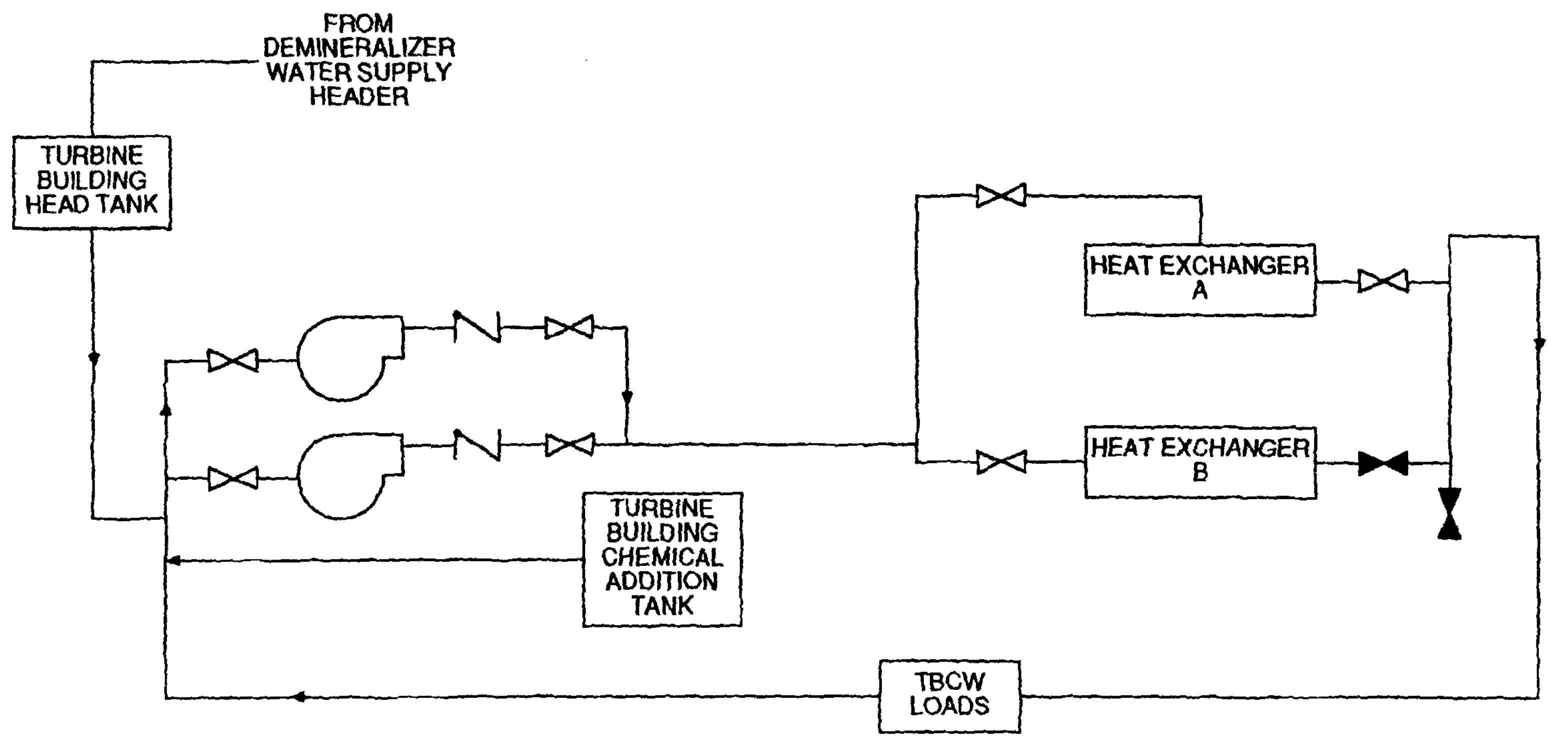


Figure 4.6.22-1. Turbine Building Cooling Water System Schematic.

The success criteria for TBCW is one of two pumps and either of the two heat exchangers operating. This will provide sufficient cooling to the TBCW loads.

The majority of the TBCW system including the cooling water pumps, heat exchangers and associated piping, valves and controls are located on the turbine building ground floor. The specific TBCW loads are distributed throughout different areas of the plant.

4.6.22.2 TBCW Interfaces and Dependencies

The TBCW system is not operated in the event of offsite power failure. Under loss of offsite power, the cooling water supply to the air compressor jackets and after coolers and the CRD pump lube oil coolers is maintained from the RBCW system. In order to operate, the TBCW system must have offsite AC power and NSW for the ultimate heat sink (see Figure 4.6.22-2).

4.6.22.3 TBCW Test and Maintenance

The TBCW system has no special test and maintenance requirements.

4.6.22.4 TBCW Technical Specifications

The TBCW system has no specific technical specifications.

4.6.22.5 TBCW Logic Model

The TBCW system was modeled using a fault tree for the loss of cooling water to auxiliary plant equipment. The fault tree has been simplified to cover only the major active components, interfaces, and dependencies, and human errors.

The head tank, heat exchangers and chemical addition tank were not modeled since they are passive devices and their failure probabilities are not expected to dominate system failure.

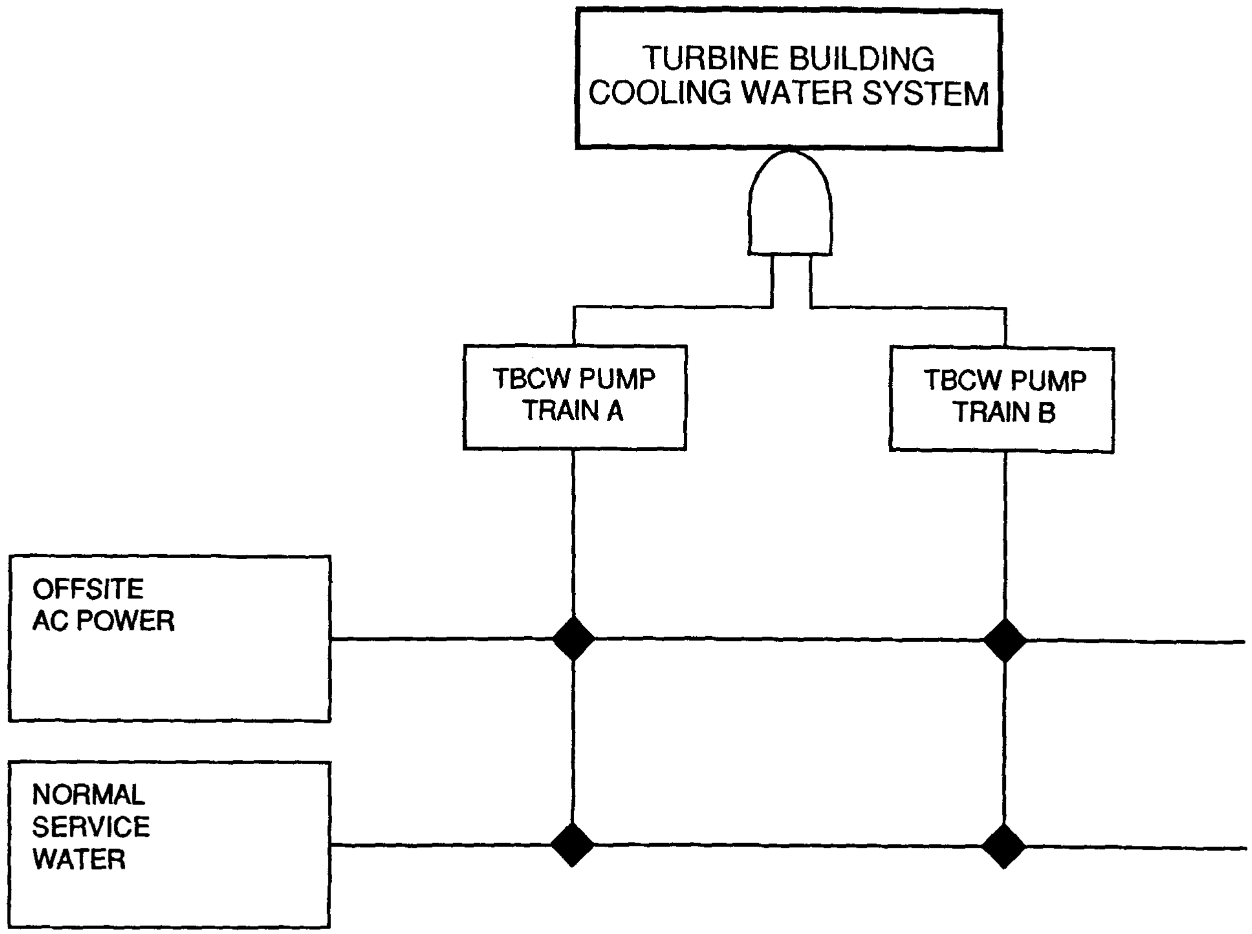
One human error was incorporated into the TBCW fault tree model. That error was failure to restore the pump B train after maintenance.

4.6.22.6 TBCW Assumptions

(1) Only major active components and major dependencies were modeled since it was assumed that these dominant system failure.

4.6.22.7 TBCW Operating Experience

There was nothing peculiar in the operational history of the TBCW system which would affect system modeling.



Dependency Diagram Is Shown Using Failure Logic. Refer to the Fault Trees for Actual Failure Logic Details.

Figure 4.6.22-2. Turbine Building Cooling Water System Dependency Diagram.

4.6.23 Reactor Protection System

4.6.23.1 RPS Description

The function of the RPS is to provide timely protection against the onset and consequences of conditions that threaten the integrities of the fuel barrier and the nuclear system process barrier (event tree nomenclature--C).

The RPS includes the motor-generated power supplies with associated control and indicating equipment, sensors, relays, bypass circuitry, and switches that cause rapid insertion of control rods (scram) to shut down the reactor.

4.6.23.2 RPS Interfaces and Dependencies

Power to each of the two reactor protection trip systems is supplied, via a separate bus, by high inertia AC motor-generator sets. Alternate power is available to either RPS bus from an electrical bus that can receive standby electrical power. The alternate power switch prevents simultaneously feeding both buses from the same source. DC power is supplied to the backup scram valve solenoids from the station batteries. Power is not needed to scram the reactor.

4.6.23.3 RPS Logic Models

The RPS was not modeled in any detail. RPS electrical failure and mechanical failure probabilities on demand were assigned values of 2E-5 and 1E-5 respectively (i.e., the system was simply treated as a data value).

4.6.23.4 RPS Operational Experience

Nothing was peculiar in the operational history of the RPS which would affect either system modeling or failure data.

4.6.24 Justification for Systems Not Modeled

All systems (front-line and their supports) that are important to providing core-cooling or containment cooling functions were modeled.

Other systems such as firewater, ECCS (keep-full systems), etc. which could provide cooling were not modeled since procedures don't exist to use them as reactor vessel injection sources or their flow rates are so small that it is uncertain if they would provide adequate cooling. These would be third or fourth order systems. In addition, the PCS/Feedwater system was not modeled but instead treated as a data value because of sufficient failure experience with this system.

4.6.25 System Analysis Nomenclature

A standard coding scheme was established to describe the basic events [2]. This consistency is necessary to assure that the dependencies and interfaces between the systems are properly accounted for when the individual system fault trees are merged with their support systems and the merged fault trees are linked together to perform the accident sequence quantification. In addition, the standard coding scheme provides the analyst or reviewer a traceability of the events from the cut sets resulting from the accident sequence quantification to the individual fault trees.

Each basic event is made up of a maximum of sixteen characters composed of four parts: a system identifier, an event or component type identifier, a failure mode code, and a unique event identifier. Each of these parts is separated by a dash for readability. The first three characters denote the system to which the basic event belongs or to which it is related. Table 4.6-2 contains a list of the system identifiers.

The second three letter code denotes the level of modeling corresponding to the basic event type. These event and component type identifiers are listed in Table 4.6-3. The third group consists of a two-letter code denoting the failure mode associated with the event (see Table 4.6-4). The final five chapters are for an alphanumeric event descriptor. These are used to identify individual components according to their numbering on the system schematics, or to use any other designator that will readily identify the event.

Eighteen special events were identified using a modification of the above coding scheme. These were specific Common Cause Failures (CCF) which were broken into a basic event multiplied by a Beta-factor. In this way, importance measures and uncertainty analysis of the Beta-factor itself could also be performed. The basic event was denoted using the same scheme as described above except for the five character unique event identifier which was replaced with a CCF term. The Beta-factor time was described by a BETA followed by a unique component type descriptor. The Beta-factor values are from generic common cause data [2]. All eighteen special events are incorporated into Table 4.9-1.

Table 4.6-2
System Identifiers

SYSTEM IDENTIFIER(XXX)	SYSTEM NAME
ACP	AC Power System
ARF	Air Return Fan System
ADS	Automatic Depressurization System
AFW	Auxiliary Feedwater System or Emergency Feedwater System
CPC	Charging Pump Cooling System
CHP	Charging Pump System
CVC	Chemical and Volume Control System
CHW	Chilled Water System
CSC	Closed Cycle Cooling System
CCW	Component Cooling Water System
CST	Condensate Storage Tank
CDS	Condensate System
CLS	Consequence Limiting Safeguards System
CCU	Containment Atmosphere Cleanup
CGC	Containment Combustible Gas Control
CFC	Containment Emergency Fan Cooler System
CIS	Containment Isolation System
CSR	Containment Spray Recirculation System
CSS	Containment Spray System
CRD	Control Rod Drive System
DCP	DC Power System
DWS	Drywell (Wetwell) Spray Mode of RHR System
EHV	Emergency Heating, Ventilation, and Air Conditioning System
ESF	Engineered Safety Feature Actuation System
ESW	Essential Service Water System
FHS	Fuel Handling System
HCI	High Pressure Coolant Injection System
HCS	High Pressure Core Spray System
HPR	High Pressure Recirculation System
HPI	High Pressure Safety Injection System
HSW	High Pressure Service Water System
ICS	Ice Condenser System
ISR	Inside Containment Spray Recirculation System
IAS	Instrument Air System
ISO	Isolation Condenser System
LCI	Low Pressure Coolant Injection System
LCS	Low Pressure Core Spray System
LPR	Low Pressure Recirculation System
LPI	Low Pressure Safety Injection System
MCW	Main Circulating Water System (Main Condenser Cooling Water)
MFW	Main Feedwater System
MSS	Main Steam System

Table 4.6-2
System Identifiers (Concluded)

SYSTEM IDENTIFIER(XXX)	SYSTEM NAME
NHV	Normal Heating, Ventilation, and Air Conditioning System
NSW	Normal Service Water
OEP	Onsite Electric Power System
OSR	Outside Containment Spray Recirculation System
PCS	Power Conversion System
PCV	Primary Containment Venting
PPS	Primary Pressure Relief System (PORV/SRV)
RGW	Radioactive Gaseous Water System
RLW	Radioactive Liquid Waste System
RBC	Reactor Building Cooling Water System
RCS	Reactor Coolant System
RCI	Reactor Core Isolation Cooling System
RPS	Reactor Protection System
RMT	Recirculation Mode Transfer System
RHR	Residual Heat Removal System
SIS	Safety Injection Actuation System
SWS	Service Water System
SDC	Shutdown Cooling Mode of RHR
SGT	Standby Gas Treatment System
SLC	Standby Liquid Control System
SPC	Suppression Pool Cooling System (or Suppression Pool Cooling Mode of the RHR System)
SPM	Suppression Pool Makeup System
TBC	Turbine Building Cooling Water System

Table 4.6-3
Event and Component Type Identifiers

COMPONENT	IDENTIFIER(YYY)
Air Cooling Heat Exchanger	ACX
Sensor/Transmitter Units:	
Flow	ASF
Level	ASL
Physical Position	ASD or ADS
Pressure	ASP
Radiation	ASR
Temperature	AST
Flux	ASX
Circuit Breaker	CRB
Calculational Unit	CAL
Electrical Cable	CBL
Signal Conditioner	CND
Control Rods:	
Hydraulically-Driven	CRH
Motor-Driven	CRM
Ducting	DCT
Motor-Driven Compressor	MDC
Motor-Driven Fan	FAN
Fuse	FUS
Diesel Generator	DGN
Hydrogen Recombiner Unit	HRU
Heat Exchanger	HTX
Inverter	INV
Electrical Isolation Device	ISO
Air Cleaning Unit	ACU
Load/Relay Unit	LOD
Logic Unit	LOG

Table 4.6-3
Event and Component Type Identifiers (Continued)

COMPONENT	IDENTIFIER(XXX)
Local Power Supply	LPS
Motor-Generator Unit	MGN
Motor-Operated Damper	MOD
Pumps:	
Engine-Driven	EDP
Motor-Driven	MDP
Turbine-Driven	TDP
Positive-Displacement	PDP
Manual Control Switch	XSW
Rectifier	REC
Transfer Switch	TSW
Transformer	TFM
Tank	TNK
Bistable Trip Unit	TXX
Air Heating Unit	AHU
Electrical Bus - DC	BDC
Electrical Bus - AC	BAC
Manual Damper	XDM
Pneumatic/Hydraulic Damper	PND
Battery	BAT
Valves:	
Check Valve	CKV
Hydraulic Valve	HDV
Safety/Relief Valve	SRV
Solenoid-Operated Valve	SOV
Motor-Operated Valve	MOV
Manual Valve	XVM
Air-Operated Valve	AOV
Testable Check Valve	TCV
Explosive Valve	EPV
Pressure Control Valve	PCV

Table 4.6-3
Event and Component Type Identifiers (Concluded)

COMPONENT	IDENTIFIER(YYY)
Filter	ALT
Instrumentation and Control Circuit	ICC
Strainer	STR
Heater Element	HTR
Pipe Segment	PSF
Pipe Train	PTF
Actuation Segment	ACS
Actuation Train	ACT
AC Electrical Train	TAC
DC Electrical Train	TDC
Operator Action	XHE
Common Cause Event	CCF
Miscellaneous Aggregation of Events	VFC
Phenomenological Events	PHN
System	SYS
Performance (Signal Operating)	PER
Power	PWR

Table 4.6-4
Failure Mode Codes*

FAILURE MODE	CODE (ZZ)
Valves, Contacts, Dampers	
Fail to Transfer	FT
Normally Open, Fail Open	OO
Normally Open, Fail Closed (Position)	OC
Normally Closed, Fail Open	CO
Normally Closed, Fail Closed	CC
Valves, Filters, Orifices, Nozzles	
Plugged	PG
Leak	CB
Pumps, Motors, Diesel, Turbines, Fans, Compressors	
Fail to Start	FS
Fail to Continue Running	FR
Sensors, Signal Conditioners, Bistable	
Fail High	HI
Fail Low	LO
No Output	NO
Segments, Trains, and Miscellaneous Agglomerations	
Loss of Flow, No Flow	LF or PF
Loss of Function	FC
Actuation Fails	FA
No Power, Loss of Power	LP
Failure (for miscellaneous fault agglomerations not based on segments or trains)	VF
Hardware	HW
Battery, Bus, Transformer	
No Power, Loss of Power	LP
Short	ST
Open	OP
Tank, Pipes, Seals, Tubes	
Leak	LK
Rupture	RP

* Grouping of failure modes by events or components are only suggestions. The failure mode listed may be used for any applicable event or component type.

Table 4.6-4
Failure Mode Codes* (Concluded)

FAILURE MODE	CODE (ZZ)
Human Errors	
Fail to Operate	FO
Miscalibrate	MC
Fail to Restore from Test or Maintenance	RE
Normal Operations (unavailable due to planned activity)	
Maintenance	MA
Test	TE
Test and Maintenance	TM

* Grouping of failure modes by events or components are only suggestions. The failure mode listed may be used for any applicable event or component type.

4.7 Dependent Failure Analysis

The system failure models and analyses explicitly accounted for the various system dependencies such as the need for power, room cooling, etc. These dependencies can be a source of possible system interactions as well as representing a common cause failure potential for the accident mitigating systems. In addition, specific tasks were performed as part of this study to address particular subtle interactions as well as common cause failures among components based on available failure data. The following subsections address each of these tasks performed as part of a more comprehensive dependent failure analysis.

4.7.1 Scope of Dependent Failure Analysis

Several attempts have been made to develop categories of dependent failures. The major purpose of this categorization is to allow the risk analyst to select a method for performing the dependent failure analysis. In the Peach Bottom Probabilistic Risk Assessment (PRA), essentially three categories of dependent failures were examined and explicitly included in the event and fault tree models: direct functional dependencies, common cause and subtle (peculiar or unexpected) interactions.

Direct functional dependencies are those dependencies that are required for a system to perform its function. Generally these dependencies include:

(1) Initiator Dependencies -- This includes the effects of events which cause a plant transient and causes or increases the probability of mitigating system failure. An example in the Peach Bottom PRA is a loss of an emergency AC bus initiating event. All such initiators are identified and discussed in Section 4.3.

(2) Support System Dependencies -- Failure of a single system such as Emergency Service Water (ESW) can fail multiple front-line systems which it supports. Inclusion of appropriate support systems as failure modes of front-line systems is used to ensure such dependencies are properly accounted for. Support system dependencies are further discussed in Section 4.7.2.

(3) Shared-Equipment Dependencies -- Components utilized by multiple systems when failed can potentially fail multiple systems. It is essential that the analyst uniquely identify such components in the system fault trees. Common component failures are included in the fault trees described in Section 4.6.

Common cause failures are those failures that result in failure of "like" components because of factors such as common maintenance or common manufacture. These dependencies are discussed further in Section 4.7.3.

Subtle interactions, or sometimes referred to as peculiar or unexpected interactions, are those physical interactions of the system with potential dependent failure mechanisms. They are called 'subtle interactions' because by their nature can be easily overlooked in a PRA unless the analyst explicitly looks for them. Two methods were employed to account for these types of interactions. Review of (1) the system design and interfaces and (2) the Licensee Event Reports (LERs) and other plant data were used to identify any peculiar or unexpected interactions. An example of this type of interaction in the Peach Bottom PRA is tripping of the Reactor Core Isolation Cooling (RCIC) turbine by a high turbine exhaust pressure signal following failure of containment heat removal. Such dependencies are included in the event tree construction described in Section 4.4. Additionally, many of these types of failures have been found in past analyses. Each of these interactions were reviewed for applicability to Peach Bottom. Section 4.7.4 presents descriptions of identified subtle interactions and the resolution of each for Peach Bottom.

4.7.2 Treatment of Direct Functional Dependencies

Operation of the so-called front-line core and containment cooling systems (e.g., HPCI, LPCS, RHR...) are directly or indirectly dependent on certain support systems. Examples of direct dependencies include AC/DC power to pumps and valves, service water cooling for pump bearings and seals, and instrument air to valves and dampers. Indirect dependencies include for example, room cooling via use of service water cooling and fans for room heat exchangers. By virtue of a delayed phenomena, front-line system failure or isolation is ultimately postulated because of room heat-up effects. In addition, some support systems are dependent on yet other supports (e.g., service water needs power).

Presented in each systems analysis section under Section 4.6, are descriptions of each system dependency which is modeled, accompanied by a dependency diagram which pictorially describes the relationship of each dependency to the system being analyzed. These dependencies are explicitly handled in the fault tree models for each system.

4.7.3 Common Cause Failure Analysis

The inclusion of residual dependent failures not already explicitly modeled but for which some data exists, were handled as non-descriptive common cause failures based on a review of plant specific failures and generic failure information. A review of Peach Bottom maintenance logs, "hi-spot" reports, and LERs was conducted to search for significant common cause events in the past five years of experience. No significant common cause failures were identified.

The fault tree for each system contains, where appropriate, common cause failure events. Such events (e.g., ESW-CCF-LF-AOVS) were modeled using the single event name in the fault tree but broken out into an independent failure term and a corresponding common cause factor for the dominant sequence cut sets. This was done so that the common cause factor uncertainty and importance measures could be calculated and examined separately. The choice of common cause events to be included was based on availability of estimates from an EPRI study [23] and other common cause failure analyses [37,38,39,40] for events involving 2 or more "like" component failures (e.g., common cause failure of four air-operated valves). Since the estimates are only readily available for common cause failures of "like" components within a system, common cause modeling cross system boundaries was not included in the Peach Bottom analysis.

The equipment failure common cause events explicitly modeled in the system fault trees are listed in Table 4.7-1. For those events appearing in the dominant accident sequences, the corresponding break out of these terms into an independent failure term and an overall common cause factor was used. Note that human-related common events such as miscalibration of "like" sensors is covered under the human interface analysis (see Section 4.8).

Too few Peach Bottom failure data were available to quantify plant-specific common cause factors. Therefore, EPRI report NP-3967 [23] and other analyses [37,38,39,40] were used to quantify all common cause probabilities with the exception of common cause battery failure. The calculated values were taken as mean values assuming an error factor of 3. In each case, the number of actual events as well as potential events were considered using the methodology in References 37 thru 40 to arrive at the data value for each event.

A battery failure common cause factor was determined utilizing the DC power study (NUREG-0666 [24]). That study suggests a worst case Beta factor of 0.4 for failure of a second battery given the first battery has failed. The first battery fails randomly at the probability assigned for a single battery failure. However, Peach Bottom's DC power system is better than the minimum system analyzed in the DC power study. Considering Peach Bottom's system, the report recommends a Beta factor of 0.4x0.02 or 8E-3 for the second battery. The estimate for additional coincident battery failures was arrived at assuming that the probability of common cause failure of each successive battery was half-way between unity and the common cause factor for the preceding battery (e.g., the common cause factor for the 3rd battery was (1.0 + 8E-3)/2 or approximately 0.5 resulting in an overall common cause factor of 8E-3 x 0.5 or 4E-3). This method is discussed in the dependent failure chapter of the NUREG/CR-4550 Methodology document [55]. Hence an overall failure rate for three batteries is determined by multiplying the random failure of the first battery times the factor, 4E-3. This approach was successively performed for the 4th battery, etc.

A summary of the common cause values used in this analysis is presented as part of the Data Section, 4.9.

4.7.4 Analysis of Subtle System Interactions

The first type of subtle interactions examined were 'peculiar' or 'unexpected' physical interactions or phenomenological dependencies. These are modeled by virtue of the event tree constructions. For example, HPCI success followed by containment cooling failure will ultimately lead to HPCI failure because of high suction water

Table 4.7-1
Peach Bottom Common Cause Events

EVENT NAME	DESCRIPTION
ACP-CCF-LP-DGS (ACP-DGN-LP-CCF*BETA-4DGNS)	Common cause failure of all four diesel generators
ADS-CCF-CC-ADSRV (ADS-AOV-CC-CCF*BETA-3SRVS)	Common cause failure of at least three ADS valves to open
ADS-CCF-CC-NADSV (ADS-AOV-CC-CCF*BETA-4SRVS)	Common cause failure of at least four non-ADS safety relief valves to open
ADS-CCF-LK-ACC (not separated into two events; value based on engineering judgment)	Common cause failure of ADS accumulators (leakage)
CSS-CCF-LF-MOVS (CSS-MOV-CC-CCF*BETA-2MOVS)	Common cause failure of the two containment spray injection valves to open
DCP-CCF-LP-BAT (DCP-BAT-LF-CCF*BETA-5BAT)	Common cause failure of at least five batteries to supply sufficient power to their loads
EHV-CCF-LF-AOVS (EHV-AOV-CC-CCF*BETA-6AOVS)	Common cause failure of at least six ventilation dampers (for diesel room cooling) to open
ESW-CCF-LF-AOVS (ESW-AOV-CC-CCF*BETA-3AOVS)	Common cause failure of at least three emergency service water valves (to supply diesel jacket cooling) to open
ESW-CCF-PF-MDPS (ESW-MDP-FS-CCF*BETA-2SWPS)	Common cause failure of the two primary emergency service water pumps
HSW-CCF-LF-MDPS (HSW-MDP-FS-CCF*BETA-4SWPS)	Common cause failure of all four high pressure service water pumps

Table 4.7-1
Peach Bottom Common Cause Events (Concluded)

EVENT NAME	DESCRIPTION
HSW-CCF-LF-MOVS (HSW-MOV-CC-CCF*BETA-4MOVS)	Common cause failure of all four high pressure service water valves (used for supply to RHR heat exchangers) to open
LCI-CCF-LF-MOVS (LCI-MOV-CC-CCF*BETA-2MOVS)	Common cause failure of the two LPCI injection valves to open
LCS-CCF-LF-MOVS (LCS-MOV-CC-CCF*BETA-2MOVS)	Common cause failure of the two LPCS injection valves to open
LCS-CCF-PF-MDPS (LCS-MDP-FS-CCF*BETA-3RHRMDPS)	Common cause failure of at least three LPCS pumps
RHR-CCF-PF-MDPS (RHR-MDP-FS-CCF*BETA-4RHRMDPS)	Common cause failure of all four RHR (also used for LPCI) pumps
SLC-CCF-PF-MDPS (SLC-MDP-FS-CCF*BETA-2SIPUMPS)	Common cause failure of both standby liquid pumps
SPC-CCF-LF-MOVS (SPC-MOV-CC-CCF*BETA-2MOVS)	Common cause failure of the two suppression pool cooling valves to open

temperature if the suppression pool is being used for suction. Hence other systems must then be used to prevent core damage. Such a dependency is explicitly covered by the event tree construction which requires success of such systems as Condensate, CRD, etc. following success of HPCI but failure of RHR (all modes). Further information on such dependencies is covered in each event tree writeup (See Section 4.4) where appropriate.

Past PRAs and actual events are available information sources for identifying particularly subtle failures which an analyst might normally overlook. As part of this effort, other knowledgeable experts in analyzing power plant safety were asked to identify subtle system interactions which they were aware of and which could cause mitigating system failures [21,22]. To the extent possible, recognizing resource and priority constraints, these interactions were to be reviewed for applicability to the Peach Bottom analysis. Any found to apply were appropriately accounted for in the analysis. The remainder of this section summarizes the Boiling Water Reactor (BWR)-related subtle interactions identified and their corresponding resolutions by the Peach Bottom analysts.

Air binding of cooling water systems

The failure or partial failure of cooling water systems has occurred because of air binding caused by leaks in a load being cooled. Plant air compressors usually are cooled by some cooling water system. Air inleakage into the cooling water system can cause failure of multiple systems because of air binding and loss of cooling.

The two most critical service water systems (Emergency Service Water, ESW, and High Pressure Service Water, HPSW) do not directly interface with air systems. Review of the Peach Bottom licensee event reports and maintenance records did not reveal problems in this area. Hence this does not seem to be significant at Peach Bottom and so is not explicitly modeled. (See Item #1 of Reference 21.)

Steam-line break isolation circuitry

Steam-driven systems usually have isolation circuitry to protect against steam-line breaks. This circuitry uses temperature readings as an indication of a line break and may include all locations containing the steam piping. Therefore, when assessing the need for room cooling, the cooling requirements of areas where temperature measurements are taken must be examined.

Failure modeling in the Peach Bottom system fault trees for High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) have accounted for this potential interaction. (See Item #2 of Reference 21.)

Passive component failures

This type of interaction involves component failure modes that might not otherwise be modeled (e.g., valve failure because of steam/disc separation, pipe breakage, blockage). These failures should be added to the models particularly where the impact of failure affects multiple trains of equipment. Additionally, these events can be potential initiators.

These were considered particularly wherever they might cause a disruption in normal plant operation and degrade mitigating systems. One source as a possible initiator (pipe break in the Normal Service Water (NSW) line near the Emergency Service Water (ESW) interface) is discussed in Section 4.3 but deemed insignificant. In other areas where passive failures (such as valve disk separation) were deemed as potential significant contributors, the failures were explicitly modeled in the system fault trees. (See Item #3 of Reference 21.)

Isolation of nonessential cooling water loads

This failure mode occurs when nonessential headers of important cooling systems are not isolated. Because such a failure can result in inadequate cooling of the essential loads, care should be taken when determining the impact of potential diversion paths from support cooling systems.

Diversion paths were considered for all systems, including cooling water systems. Possible significant ones are explicitly modeled in the fault trees. (See Item #4 of Reference 21.)

Cross-tied pumps' discharge check valve failures

This type of failure occurs when the discharge check valve in one train of a two-train, cross-tied system fails open. Various problems can result from this interaction, including functional failure of the system because of backflow, inability to actuate an idle pump because of the stuck-open valve, or system rupture from attempted actuation of an idle pump with a stuck-open valve.

Five years of plant data on major important systems reviewed for failure data did not mention problems of this type. Two areas in the ESW system were explicitly modeled (available test procedures were obtained) for this failure mode because of the possibility of occurrence and the fact that ESW failure could potentially affect so many other systems. (See Item #5 of Reference 21.)

Failures following station blackout

The treatment of the failure mode of reactor pump seals and battery depletion during a station blackout has varied among past PRAs and can be plant specific. Both failures can adversely affect the capability to cool the plant.

Seal loss of coolant accidents (LOCA) are not so significant for Boiling Water Reactors (BWRs) because of HPCI and RCIC capabilities. Battery depletion was considered and a nominal 12 hour time was used based on Philadelphia Electric Company input and internal expert opinion analysis. Uncertainty in the battery depletion time was explicitly factored in to the uncertainty analysis. (See Item #6 of Reference 21.)

Dependent events based on operating experience

There have been a number of recent activities to better scope out the problem of dependent and common cause events. Probably the best current collection of actual events that are in the nuclear data base are compiled in EPRI NP-3967 [23]. While there is considerable controversy on how to account for common cause events, the report clearly demonstrates the inaccuracy of models that do not specifically treat common cause events. While it has been a frequent criticism that quantification of these events leads to numbers but not indication of how to improve plants, a review of the events in EPRI NP-3967 will demonstrate that causes are known for a large percentage of these events.

A review of Peach Bottom maintenance logs and post-trip analysis reports since 1980 indicated that insufficient data exists to determine whether any actual common cause failures have occurred. However, potential common cause failures were included in the system models for the types of components listed in EPRI NP-3967. (See Item #7 of Reference 21.)

Main feedwater availability

The unavailability of main feedwater after a plant reactor trip is highly plant-specific. The consequences of this interaction will vary depending on whether the loss is total or partial and the potential for recovery.

With a recent change to a Level 1 trip for closure of Main Steam Isolation Valves (MSIVs), little experience exists at Peach Bottom. Many initiators will cause MSIV closure and hence loss of feedwater (turbine pumps). A conservative analysis was performed for Peach Bottom in which feedwater and condensate were assumed initially lost for most initiators. (See Item #8 of Reference 21.)

Turbine-driven pump failure by overfill

This interaction specifically involves failure of a turbine-driven pump because of steam generator or reactor vessel (for BWRs) overfilling. The loss of a turbine-driven pump can be immediate or delayed (i.e., water carryover through the steam lines to the turbine can lead to a sequence involving successful initial response followed by a later loss of the turbine-driven pump); therefore, its impact/consequence will vary depending on the timeframe of the loss.

HPCI and RCIC were modeled for this potential failure mode. In most cases, such an event would be prevented by high level trips of these systems. Feedwater, also turbine driven, was already conservatively assumed lost for most sequences. (See Item #12 of Reference 21.)

<u>DG load sequence problem</u>

The diesel generator load sequence system is a circuit designed to strip off non-essential loads from the diesel generators following loss of offsite power (LOSP). The design of such a circuit usually involves redundant means to strip all loads following a LOSP. However, such circuits may not always contain redundant means for subsequently reloading essential loads. In such a case failure of the load sequencing circuit could potentially result in common cause failure of multiple systems following a LOSP.

Peach Bottom uses individual time delay relays for the sequencing of most safety loads. Thus the potential for common cause failure of load sequencing was deemed quite low. The problem described here did not appear to be appropriate for consideration for Peach Bottom. (See Item #1 of Reference 22.)

<u>Sneak circuits</u>

The RCIC system at one Boiling Water Reactor was found to contain a sneak circuit which could result in an unintended isolation of the RCIC pump. This could occur during a loss of offsite power and subsequent energization of the RCIC steam leak detection circuit. Three subtle design aspects lead to the occurrence of this failure mode: (1) the RCIC system contains a steam leak detection isolation circuit, (2) the isolation circuitry is deenergized given a loss of offsite power (i.e., the circuitry is not fed by a non-interruptable battery-backed vital AC power supply), and (3) the isolation circuit contains a seal-in circuit.

The problem requires that some isolation-related control circuitry for HPCI/RCIC be AC powered. All such circuitry at Peach Bottom is DC powered and hence the problem does not exist at Peach Bottom. (See Item #2 of Reference 22.)

<u>Bus switching problems</u>

Two subtle aspects concerning bus switching have been identified at one power plant: (1) a safety-related DC power supply is also being used to perform a bus switching operation in the switchyard and safety-related loads are normally powered from the unit transformer rather than from offsite power, and (2) a safety-related AC bus does not have a diesel directly powering it; it must rely on diesel power from another bus via a breaker which only closes given a loss of offsite power.

Resources did not permit a detailed review of bus switching at Peach Bottom. The analysis methodology called for "simple" modeling of the onsite bus arrangement. Since there are not similar bus-to-bus cross feeds in normal use at Peach Bottom and since a diesel exists on all four division safety 4160V buses, the problem did not appear important for Peach Bottom. (See Item #3 of Reference 22.)

Normal operating configuration

This interaction involves the differences between the plant operations documentation (e.g., Piping and Instrumentation Diagrams, P&IDs) and the actual operating practices and configurations. For example: (1) the P&ID may show valves as normally closed which, during plant operation, are actually open; or (2) the P&ID indicates a room containing the high-pressure injection pumps with two room coolers, each receiving power and cooling water from different divisions when, in actuality, only one cooler is operating during normal plant operations plus the procedures relating to these coolers do not prohibit the operator to provide power and water to the cooler from two different divisions. Therefore, application of only the plant documentation could give erroneous results in the event analyses and quantification.

For the Peach Bottom study, the normal operating configurations and practices for all systems modeled were verified to the extent possible by plant visits and personnel interviews. All system fault tree models reflect the information obtained from these visits and interviews, thereby ensuring the most accurate representation of actual plant operating conditions, configuration, procedures, and practices. (See Item #4 of Reference 22.)

Room cooling

Several aspects concerning pump room cooling must be considered in a PRA systems analysis. First, a given plant's design may be such that, given loss of room cooling, the maximum room temperature remains below the temperature for which a pump and its control circuits are qualified. A system analyst may, therefore, conclude that the room cooling for the pump is not required. However, in some cases, a room temperature signal is used to trip the pump. The potential for reaching this temperature given loss of the room cooler should be examined.

Second, pump room coolers are often standby systems that actuate only upon actuation of the pump through a slave relay or by a thermostat. In either case, test procedures should be such that all of the actuation circuit is verified to function properly.

Finally, credit for opening pump room doors for cooling the room given failure of the room cooler should only be taken after considering administrative controls and technical specifications which may prohibit such action.

Peach Bottom predominantly uses slave relay type circuits and high room temperature trips of HPCI/RCIC because of the use of steam-line break detection thermocouples in the turbine rooms. There are typically numerous ways to detect loss of room cooling: steam line break detection circuitry, cooling trouble alarms, separate fire detection circuitry, etc. Failure of all indications seems small. Isolation and even failure of systems caused by high temperatures in rooms was considered for systems where appropriate (see individual systems analysis sections of this report). While it may be possible for plant staff to recover room cooling failures (such as opening doors to critical areas normally locked) credit was not given for such recovery due to the uncertainty as to whether or not such actions would successfully restore adequate cooling (some rooms represent closed-in, static areas where adequate flow is uncertain). (See Item #5, #6, #7 of Reference 22.)

Voltage droop

Not all LOSP events occur instantaneously. There have been events in which it took several minutes for the grid to degrade to the point at which offsite power was totally lost. During these several minutes, the grid voltage or frequency "dropped" out of tolerance causing the potential for breakers to open or fuses to blow on equipment normally powered from the grid. Particularly for the fuses, replacements need to be found before the equipment can be returned to service.

We did not rigorously pursue this issue. Effects of voltage droop and/or surges are subject to much uncertainty and speculation. In addition, nearly all of the systems analyzed in this study are normally in standby mode; therefore, their breakers should not be affected and fuses should remain intact. Balance-of-plant loads are normally powered by the unit generator and are not immediately affected by a grid voltage droop. There are also redundant means of separating the plant from the grid when the voltage and frequency are out of tolerance. Experience at Peach Bottom (no total losses of offsite power) makes this less important as well. Therefore, this interaction was not considered further. (See Item #8 of Reference 22.)

Terminal blocks in containment

A terminal block is located in an electrical junction box and is used to connect wire ends within a circuit. Many types of terminal blocks may not perform adequately in a steam environment. Instrument errors can occur in circuits that contain terminal blocks when exposed to a high temperature (>100°C) saturated steam environment. Such instrumentation failures can potentially prevent ECCS actuation following loss of coolant accidents.

Virtually all electrical portions of safety equipment are outside containment in BWRs. However, safety relief valve circuits do contain terminal blocks within containment. These and the possibility of terminal blocks for other systems being in the reactor building were considered in the analysis and treated as possible failure modes of the systems they serve. The redundancy of equipment and the fact that expected leakage currents are small compared with the normal current flow of the concerned circuits made this issue relatively unimportant. (See Item #9 of Reference 22.)

<u>Alternate core cooling systems</u>

There are methods of core cooling available, which although not preferred and not necessarily safety grade, could possibly be used in emergency situations. Some examples of such methods include:

- o use of service water to supply makeup to the reactor,
- o aligning a fire water pump to supply makeup to the reactor,
- o increasing control rod drive injection system flow,
- o aligning the boron injection pumps from a large water source.

In order to qualify as an alternate core cooling method during a transient (with scram) condition, several criteria are essential:

(1) Procedures must call out these systems and adequately describe their use (it is additionally useful if there is appropriate training on use of the systems and if procedures define the time order in which each system implementation should be attempted).

(2) The ability to deliver a flow rate of at least 200 gpm to the reactor must exist.

(3) The time required to establish flow from these systems must not be too long.

Appropriate systems, particularly the Control Rod Drive (CRD) and HPSW, are considered in the Peach Bottom analysis as alternate core cooling systems. (See Item #11 of Reference 22.)

<u>Level instrument error caused by high containment temperatures</u>

Level instruments could read high upon flashing of the reference legs when containment temperatures are high and the primary system is being depressurized.

Peach Bottom operators are very aware of this potential problem. The Emergency Procedure Guidelines (EPGs) call for maintaining primary pressure >80 psi above containment pressure so as to avoid this problem. As a further back-up, EPGs call for reflooding of reference legs if anomalies develop (there are ways to do this). Discussions with Oak Ridge National Laboratory personnel further substantiate that this is not a serious problem and will, at worst, only cause momentary anomalies if the vessel is <u>rapidly</u> depressurized (such as in a large LOCA). Everything considered, this did not seem to be significant at Peach Bottom. (Verbal concern raised at a quality assurance meeting.)

4.8 Human Reliability Analyses

This section contains a summary of the human interface analyses performed for the Peach Bottom study. Details of the study can be found in Appendix C.

4.8.1 Summary of Methodology and Scope

Only one type of human action error was analyzed in this study--errors of omission (e.g., failure to diagnose, miscalibration, failure to operate a system . . .). Errors of commission were considered outside the scope of this analysis. The human actions analyzed were divided into three categories: (1) pre-accident human actions such as component misalignment after test, (2) post-accident human actions such as failing to start a system for Loss of Coolant Accidents (LOCAs) and regular transients, and (3) post-accident human actions of Anticipated Transient Without Scram (ATWS) accident sequences. In the Peach Bottom analysis the post-accident human actions include any human action that occurs after the accident has started (i.e., from the time of the initiating event). With few exceptions, only those actions specified in the plant procedures were considered. System failures caused by hardware faults and maintenance outages were considered to be nonrecoverable. Additionally, only one human action event was allowed (per cut set) unless the actions could be judged to be independent.

The Human Error Probabilities (HEPs) evaluated for the pre-accident human errors and the post-accident human errors for LOCAs and regular transients are nominal values based on a simplification of the THERP method. This simplified method is documented as the "Accident Sequence Evaluation Program (ASEP) Human Reliability Analysis (HRA) Procedure" [25]. However, there were several human action errors that were not evaluated using the ASEP HRA procedure. These actions (or failure to perform) were estimated using the ASEP generic data base [2] and the specific analysis performed for offsite power recovery [26]. These include all human action errors regarding recovery of electrical faults and the Power Conversion System. These are explicitly noted in Section 4.8.2. Additionally, for the ATWS post-accident human actions a detailed HRA was performed by Brookhaven National Laboratory (BNL) specialists and described in detail in Section 4.8.5.

4.8.2 Human Actions Analyzed

The specific human actions analyzed in this study were identified in either the system failure models, by examining failures in the cut sets or the event trees. The system descriptions (see Section 4.6) summarize the human actions that were modeled as part of the system fault trees. These include all the pre-accident human errors such as misalignment after test and some of the post-accident human errors such as failure to back up auto start failure, system realignment failures, and manual start failure. In addition, other human action errors were analyzed. These included those actions the operators could successfully perform to mitigate the ongoing accident and prevent core damage or containment failure if taken in time and were identified in either the event trees or by examining the individual cut sets. In all cases, evaluation of the specific HEPs was such that individual events were given values of 1E-3 or higher unless justification could be provided for using a lower value. Similarly, 1E-4 was used as a cut-off for multiple, dependent events unless justification could also be provided to support a lower value. The analyses of the human actions are discussed in the following sections.

4.8.3 Analysis of Pre-Accident Errors

Pre-accident human errors were considered where appropriate, for all the systems analyzed in the Peach Bottom analysis. Pre-accident failures include all human action errors prior to the start of the accident: (1) failure to restore a component or system following either scheduled or unscheduled maintenance, (2) failure of a component or system because of miscalibration errors, (3) failure to restore a component or system following testing, or (4) other miscellaneous plant specific actions.

Each system was analyzed to identify components that might require maintenance while the plant is at power or may have been maintained while the plant was down; manual valves were assumed to be maintained infrequently and were not considered. For each component identified, the evaluation of the operator failing to perform the required task (i.e., restore) was performed in three steps. The first step (1) identified all activities (i.e., closing valves to isolate the component, pulling pump breakers, etc.) associated with performing each task (i.e., failure to restore pump after maintenance) and (2) determined dependence between the activities. Based on the activities, the next step involved identifying any potential for catching any errors made (e.g., written checks per shift on component status) for each task. The third step incorporated the results of the first two steps and evaluated the HEPs.

Systems that need to be realigned after testing were identified and a failure to restore the system to its proper alignment was modeled following the same three steps for failure to restore after maintenance.

Sensors were analyzed for potential miscalibration errors. The sensors were grouped as to their type and location; for example, all condensate storage tank low level sensors were put in one group and all high drywell pressure sensors were put in another group. A separate miscalibration error was assigned to each group. Failure to miscalibrate was also performed in three steps: (1) identification of the calibration activities, (2) identification of any potential to recover mistakes, and (3) computation of the HEP.

All failure to restore probabilities were calculated using the methodology presented in Reference 25. Table 4.8-1 lists the pre-accident failures used in this study. The detailed derivation of the probability of each pre-accident failure is presented in Appendix C.

4.8.4 Analysis of Post-Accident Errors (non-ATWS)

Post-accident human errors are those operator actions performed by the operator after the accident has started. With few exceptions, only those actions specifically addressed in the plant procedures are credited and evaluated. These include such actions as manually initiating a system, aligning and actuating a system for injection, recovering a failed system, etc. This section only discusses those actions involving LOCAs and regular transients (i.e., non-ATWS transients). If a single post accident HRA value was less than 10^{-3} or multiple HRA events were less than 10^{-4}, the HRA value was re-evaluated. This added further assurance that unrealistically low values were not used.

Post-accident human errors were identified in two steps: (1) system models and (2) sequence cut sets. When developing the system models, any post-accident operator action required for the system to successfully function when demanded was identified and added directly to the system model. This process only identified the action or task (e.g., manually align CRD for full flow) and did not identify the individual activities required in order to accomplish the task. These activities are identified as part of the task action (task evaluation) and discussed later.

The post-accident human errors considered in the Systems Analysis task were generally those actions performed by the operator for the system to properly function:

- o Manual operation of any components,
- o Manual initiation as backup to auto-initiation.

By identifying human action errors in the systems models, the potential for more than one human action event to appear in a cut set existed when linking the system models to form the accident sequence. This occurrence presents a problem when the actions are dependent. Only independent human actions can be multiplied together if the dependence among the actions is not considered. Since the failures (which dictate the conditions under which the operator is working) are identified in the sequence cut sets, it is impossible to evaluate the HEPs for post-accident human errors at the system model level. Therefore, these actions were assigned a screening value, generally 0.5, in the initial quantification step. Only the screening values were used unless the human failures were important (i.e., appeared in a dominant accident sequence). In this latter case, if a cut set appeared with one or more of these actions, the appropriate post-accident human error was assigned depending on whether the actions were dependent or independent, considering the sequence timing and specific failures that had occurred. The cut set was also evaluated for any additional recovery credit.

For example, the following cut set would be examined for any terms that are post-accident failures:

IE-T1*DCP-INV-LP-24C*ESW-CKV-CB-CV514

*ESW-XHE-FO-HCILV*ESF-XHE-FO-RCILV

*LOSPNR150MIN

The terms with '...-XHE-...' are post-accident failures. These terms are failures of the operator to manually control the operation of high pressure coolant injection and the reactor core isolation cooling systems. A Level 8 protective trip for these systems has failed because of the indicated 24 VDC failure. Since these two actions are highly dependent and essentially considered as one action, the two XHE terms were evaluated as one activity. That is, the operator is likely to either notice the Level 8 trip failure and control both systems, or he does not control either system. As a result, the human action error probability was evaluated as one event, its probability determined and then that probability was equally distributed to the two XHE terms such that the collective probability of ESF-XHE-FO-HCILV*ESF-XHE-FO-RCILV was equal to the correct value. In this example, an additional recovery term involving the restoration of power (LOSPNR150MIN) was added to the original cut set. In this case, since the initiating event (IE-T1) is a loss of offsite power, it was judged that activities associated with recovering AC power are independent of the Level 8 issue and hence the independent recovery action could be applied.

This basic approach was followed in evaluating each cut set for potential recovery. The majority of the HEPs for each recovery action were derived using the general HRA methodology outlined in Reference 25 which involved the following general steps:

(1) Identification of the sequence and subsequent accident conditions.

(2) Based on the cut set (and sequence), the timing of the events (i.e., occurrences, failures, alarms, indications, etc.) was established.

(3) Based on the cut set (and sequence), the symptoms and therefore the possible recovery actions (and required activities) were identified.

(4) The time available to the operator to diagnose and perform the action (and activities) was established.

(5) The probability of the operator failing to properly diagnose the accident was determined. This considered such things as operator training, simulator exercises, etc.

(6) The type of recovery action (whether 'dynamic' or 'step-by-step') was determined considering such things as the plant using symptom oriented procedures, operator training, etc.

(7) The stress-level of the operator was determined considering such things as time available, difficulty of the action, training, number and timing of equipment failures, etc.

(8) The probability of the operator failing to perform the recovery action was evaluated.

The exceptions to this procedure are: (1) the recovery of onsite power faults (e.g., the recovery of diesel generator hardware faults, (e.g., DGHWNR3HR), (2) the recovery of the PCS, (e.g., PCSNR13HR), and (3) the recovery of offsite power (e.g., LOSPNR12HR). The electrical fault and PCS recovery values came from the ASEP generic data base [2], and the recovery of offsite power was provided by Reference 26. Table 4.8-2 lists the post-accident events analyzed for Peach Bottom for LOCAs and transients. Appendix C contains the detailed analysis of these post-accident human actions.

4.8.5 Analysis of ATWS Post-Accident Errors

The post-accident human errors for ATWS sequences were identified and evaluated similar to that discussed for the LOCAs and transients including the re-evaluation on the 10^{-3} and 10^{-4} HRA values which check for unrealistically low values. Personnel from Brookhaven National Laboratory (BNL) performed a detailed HRA regarding the operational activities associated with postulated ATWS accident sequences at Peach Bottom, Unit 2.

Visits by BNL personnel and the Peach Bottom analysis team were made to the Peach Bottom Atomic Power Station and the Limerick training simulator (used by Peach Bottom operators for training) for the purpose of acquiring plant-specific information on (1) training, (2) procedures, (3) human engineering, and (4) experience and education levels of the operations crew. Interviews were conducted with training instructors and reactor operators.

A detailed task analysis was performed based on consideration of staffing, team interaction, and control room layout at Peach Bottom. ATWS scenarios developed by Oak Ridge National Laboratory, Idaho National Engineering Laboratory, and General Electric were reviewed [30,31,32,33, 34]. Thermal-hydraulic runs performed for various ATWS scenarios to determine the success criteria were included.

In the original analysis for Peach Bottom, the systems analysts provided the Brookhaven HRA analysts with an ATWS event tree (Case B in that analysis) which identified five major operator tasks that needed to be quantified. These were:

- Initiate Standby Liquid Control (SLC)
- Inhibit the Automatic Depressurization System (ADS)
- Control of Water Level Near the Top of the Active Fuel at High Pressure
- Manual Depressurization of the Reactor
- Control of Water Level Near the Top of the Active Fuel at Low Pressure

In addition, estimates were made for the following two events:

- Manual Scram
- Manual Rod Insertion

Preconditions for each of the above tasks differ as a result of the success or failure of previous tasks and safety systems. Each set of preconditions and relevant performance shaping factors were considered when the human error probabilities were assigned for the above events for each branch point on the ATWS event tree. These branch points were quantified using procedures which included a review of other PRAs and subjective judgment methods based on the structured assessment of performance shaping factors and the use of a time-reliability correlation. Because of the extensive nature of that analysis, the reader is referred to Reference 29 for details of that effort.

In the reanalysis phase of this project, the ATWS event tree was simplified considerably. This was done on the basis of improvements in the understanding of ATWS scenarios (with focus on only the most important phenomena and human actions) and as a result of comments received after the original analysis. As a result, the two previous assessments involving the "Control of Water Level" were no longer required, and the "Manual Rod Insertion" term did not have to be evaluated since this event only appeared in already non-dominant (<1E-8/year) accident sequences. The other events were used in the reanalysis with their original values as assessed in the first analysis, since the appropriate preconditions and performance shaping factors still applied. Table 4.8-3 summarizes the most critical human events in the reanalysis and the corresponding conditions and factors that most affected the ultimate value for each human error. The median value shown is out of the BNL analysis. The mean value was calculated using the uncertainty values provided in the BNL analysis.

4.8.6 Analysis of Innovative Long-Term Recovery Actions

There were no innovative long-term recovery actions applied in the analysis.

4.8.7 HRA Nomenclature

The three types of human actions in the Peach Bottom study are depicted in several forms as follows:

<u>Pre-Accident Human Actions</u> --

There were two types of actions modeled: (1) failure of the operator to restore and (2) miscalibration of equipment by the operator which is a common cause failure. These were designated in the analysis, respectively, as follows:

```
AAA-BBB-RE-CCCCC
|   |   |   |__________Unique Event Identifier
|   |   |______________Failure to Restore
|   |__________________Event or Component Identifier
|______________________System Identifier
```

```
AAA-XHE-MC-CCCCC
|   |   |   |__________Unique Event Identifier
|   |   |______________Miscalibration Failure
|   |__________________Common Cause Failure or Human Error
|______________________System Identifier
```

Post-Accident LOCA and Transient Human Actions --

These were modeled in two manners: (1) those actions (events) modeled explicitly in the fault trees and (2) those actions (events) added directly to the cut sets as a recovery action. These were designated in the analysis, respectively, as follows--

```
AAA-XHE-FO-CCCCC
|   |   |   |__________Unique Event Identifier
|   |   |______________Failure to Operate
|   |__________________Human Error
|______________________System Designator
```

```
AAAA-NR-ZZZZ
|     |  |_____________Time Period Identifier
|     |________________Action Not Recovered
|______________________Equipment and Failure Mode Being Recovered
```

Post-Accident ATWS Human Actions--

These were modeled in two manners: (1) those actions (events) modeled explicitly in the fault trees and (2) those actions (events) modeled explicitly in the event trees. These were designated in the analysis, respectively, as follows--

```
AAA-XHE-FO-CCCCC
|   |   |   |__________Unique Event Identifier
|   |   |______________Failure to Operate
|   |__________________Human Error
|______________________System Designator
```

```
FFF
|______________________Human Action Event Identifier
```

Two exceptions are noted to the above coding schemes. These include ADS-LOG-HW-INHIB and RAXV503NC which used a different nomenclature and are defined in Tables 4.8-1 and 4.8-2 respectively. The specific coding used for each human action modeled in the analysis is presented in those two tables with the exception of those events coded in the ATWS event tree (as top events). Section 4.4 depicts the coding used for the ATWS event tree headings.

Table 4.8-1

Summary of Pre-Accident Human Actions

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE[1]
<u>ACTUATION AND CONTROL SYSTEM</u>		
ESF-XHE-MC-CSTLV	Common cause miscalibration of CST low level sensors	6.65E-5
ESF-XHE-MC-HDPRS	Common cause miscalibration of high drywell pressure sensors	2.66E-4
ESF-XHE-MC-PRES	Operator miscalibrates reactor pressure sensors	5.32E-4
ESF-XHE-MC-VSLVL	Operator miscalibrates all reactor level sensors	1.33E-4
<u>AUTOMATIC DEPRESSURIZATION SYSTEM</u>		
ADS-LOG-HW-INHIB	ADS Inhibited	1.0E-5
<u>CONTROL ROD DRIVE SYSTEM</u>		
CRD-XHE-RE-PB	Motor-driven pump B not properly restored after maintenance	2.1E-3

(1) This evaluation can be found in Appendix C by referring to the event name as given in this table

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
<u>ELECTRIC POWER SYSTEM</u>		
ACP-DGN-RE-EDGA	Failure to restore Emergency Diesel Generator A after test or maintenance	7.98E-4
ACP-DGN-RE-EDG* (where * is B, C or D)	Failure to restore Emergency Diesel Generator * after test or maintenance	same as ACP-DGN-RE-EDGA
<u>EMERGENCY SERVICE WATER SYSTEM</u>		
ESW-CCF-MC-ECT	Common mode failure of all Emergency Cooling Towers due to miscalibration of vibration sensors	2.39E-8
ESW-MOV-RE-M2972	Failure to restore motor-operated valve 2972 after maintenance	7.98E-4
ESW-PTF-RE-DGA	Failure to restore diesel generator A cooling components after maintenance	3.13E-3
ESW-PTF-RE-DG* (where * is B, C or D)	Failure to restore diesel generator * cooling components after maintenance	same as ESW-PTF-RE-DGA

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
ESW-PTF-RE-ECW	Failure to restore Emergency Cooling Water (ECW) pump train after maintenance	2.19E-3
ESW-PTF-RE-HX1	Failure to restore Heat Exchanger 1 train after maintenance	2.19E-3
ESW-PTF-RE-HX* (where * is 10, 12, 13, 15, 16, 18, 19, 2, 21, 22, 24, 25, 27, 28, 3, 4, 6, 7, or 9)	Failure to restore Heat Exchanger * train after maintenance	same as ESW-PTF-RE-HX1
ESW-PTF-RE-MDPA	Failure to restore Emergency Service Water pump train A after maintenance	2.19E-3
ESW-PTF-RE-MDPB	Failure to restore Emergency Service Water pump train B after maintenance	2.19E-3
ESW-XVM-RE-XV517	Failure to restore manual valve 517 after maintenance	7.98E-4
ESW-XVM-RE-XVAB	Failure to restore manual valves A and B after maintenance	3.26E-3

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
ESW-XVM-RE-XV* (where * is CD, EF, GH, IJ, KL, MN, OP, QR, ST)	Failure to restore manual valves * after maintenance	same as ESW-XVM-RE-XVAB
<u>EMERGENCY VENTILATION SYSTEM</u>		
EHV-PTF-RE-0AV64	Failure to restore fan train 0AV64 after maintenance	2.39E-3
EHV-PTF-RE-* (where * is 0BV64, 0CV64 or 0DV64)	Failure to restore fan train * after maintenance	same as EHV-PTF-RE-0AV64
EHV-PTF-RE-0AV91	Failure to restore fan train 0AV91 after maintenance	1.60E-3
EHV-PTF-RE-* (where * is 0BV91, 0CV91 or 0DV91)	Failure to restore fan train * after maintenance	same as EHV-PTF-RE-0AV91
<u>HIGH PRESSURE SERVICE WATER</u>		
HSW-MOV-RE-2344	Failure to restore motor-operated valve 2344 after maintenance	7.98E-4

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
HSW-MOV-RE-2803	Failure to restore motor-operated valve 2803 after maintenance	7.98E-4
HSW-PTF-RE-ECTFA	Failure to restore fan A train after maintenance	3.19E-3
HSW-PTF-RE-* (where * is ECTFB or ECTFC)	Failure to restore fan * train after maintenance	same as HSW-PTF-RE-ECTFA
HSW-PTF-RE-HXA	Failure to restore heat exchanger A train after maintenance	3.19E-3
HSW-PTF-RE-* (where * is HXB, HXC or HXD)	Failure to restore heat exchanger * train after maintenance	same as HSW-PTF-RE-HXA
HSW-PTF-RE-MDPA	Failure to restore pump A train after maintenance	2.19E-3
HSW-PTF-RE-* (where * is MDPB, MDPC, or MDPD)	Failure to restore pump * train after maintenance	same as HSW-PTF-RE-MDPA

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
HSW-PTF-RE-PS10	Failure to restore pipe segment 10 after maintenance	1.6E-3
HSW-PTF-RE-PS18	Failure to restore pipe segment 18 after maintenance	2.39E-3
HSW-PTF-RE-PS20	Failure to restore pipe segment 20 after maintenance	2.39E-3
LOW PRESSURE COOLANT INJECTION SYSTEM		
LCI-MOV-RE-154A	Failure to restore motor-operated valve 154A after maintenance	3.19E-3
LCI-MOV-RE-154B	Failure to restore motor-operated valve 154B after maintenance	3.19E-3
LCI-PTF-RE-2AP35	Failure to restore pump train 2AP35 after maintenance	4.70E-3
LCI-PTF-RE-* (where * is 2BP35, 2CP35 or 2DP35)	Failure to restore pump train * after maintenance	same as LCI-PTF-RE-2AP35

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
LCI-PTF-RE-LOOPA	Failure to restore loop A valves after maintenance	4.70E-3
LCI-PTF-RE-LOOPB	Failure to restore loop B valves after maintenance	4.70E-3
<u>LOW PRESSURE CORE SPRAY SYSTEM</u>		
LCS-MOV-RE-MV11A	Failure to restore motor-operated valve 11A after maintenance	3.99E-3
LCS-MOV-RE-MV11B	Failure to restore motor-operated valve 11B after maintenance	3.99E-3
LCS-PTF-RE-2AP37	Failure to restore 2AP37 pump train after maintenance	4.7E-3
LCS-PTF-RE-* (where * is 2BP37, 2CP37 or 2DP37)	Failure to restore * pump train after maintenance	same as LCS-PTF-RE-2AP37

Table 4.8-1

Summary of Pre-Accident Human Actions
(Continued)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
REACTOR BUILDING COOLING WATER SYSTEM		
RBC-PTF-RE-2352	Failure to restore 2352 train of valves after maintenance	2.39E-3
RBC-PTF-RE-2354	Failure to restore 2354 train of valves after maintenance	2.39E-3
RBC-PTF-RE-PB	Failure to restore pump B train of after maintenance	3.19E-3
STANDBY LIQUID CONTROL SYSTEM		
SLC-XHE-RE-DIVER	Operator fails to restore system after test	3.19E-2
SLC-XHE-RE-EV14A	Explosive valve 14A not properly restored after maintenance	7.98E-3
SLC-XHE-RE-EV14B	Explosive valve 14B not properly restored after maintenance	7.98E-3
SLC-XHE-RE-MDPA	Motor-driven pump A train not properly restored after maintenance	3.13E-3

Table 4.8-1

Summary of Pre-Accident Human Actions
(Concluded)

PRE-ACCIDENT TERM	DESCRIPTION	MEAN VALUE
SLC-XHE-RE-MDPB	Motor-driven pump B train not properly restored after maintenance	3.13E-3
<u>TURBINE BUILDING COOLING WATER SYSTEM</u>		
TBC-PTF-RE-PUMPB	Failure to restore pump B train after maintenance	3.19E-3

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
ACTUATION AND CONTROL SYSTEM			
ESF-XHE-FO-ADSBT	.5	Operator fails to valve in Nitrogen bottle	Screening Value
ESF-XHE-FO-DATWS	2.0E-1	Operator fails to depressurize during an ATWS	ATWS HRA Methodology BNL [29]
ESF-XHE-FO-DEPRE	1.0E-2	Operator fails to manually depressurize reactor	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-DEPSD	.5	Operator fails to manually depressurize reactor	Screening Value
ESF-XHE-FO-HCICL	.5	Operator fails to control reactor level	Screening Value
ESF-XHE-FO-HCILV	3.3E-1[2]	Operator fails to control reactor level for HPCI	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-HCIL8	1.0E-1	Operator fails to control reactor levels for HPCI	HRA Methodology Developed from Tables C-37 through C-47

(2) This term always appears with ESF-XHE-FO-RCILV; together one gets $(.33)^2$ = the desired value of 0.1 where these terms are used.

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
ESF-XHE-FO-HCIRL	.5	Operator fails to realign HPCI suction source	Screening Value
ESF-XHE-FO-HPACT	.5	Operator fails to actuate HPCI	Screening Value
ESF-XHE-FO-HPSAC	6.0E-2	Operator fails to actuate HPCI	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-HPSAT	1.0E-1	Operator fails to back up high pressure system actuation	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-HPSRL	.5	Operator fails to realign HPCI suction source	Screening Value
ESF-XHE-FO-HSWIN	1.0E-1	Operator fails to realign HPSW for injection	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-LPSAT	.5	Operator fails to back up low pressure system actuation	Screening Value

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
ESF-XHE-FO-OVRID	.5	Operator fails to override shroud level permissive	Screening Value
ESF-XHE-FO-RCICL	.5	Operator fails to control reactor level for RCIC	Screening Value
ESF-XHE-FO-RCICO	.5	Operator fails to isolate RCIC	Screening Value
ESF-XHE-FO-RCILV[3]	3.3E-1	Operator fails to control reactor level for RCIC	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-RCIL8	1.0E-1	Operator fails to control reactor level for RCIC	HRA Methodology Developed from Tables C-37 through C-47
ESF-XHE-FO-RCIRL	.5	Operator fails to realign RCIC suction source	Screening Value
ESF-XHE-FO-RHRAT	1.0E-5	Operator fails to align RHR cooling mode	HRA Methodology Developed from Tables C-48 through C-58

(3) This term always appears with ESF-XHE-FO-HCILV; together one gets $(.33)^2$ = the desired value of 0.1 where these terms are used

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
CONTROL ROD DRIVE SYSTEM			
CRD-XHE-FO-BRKRS	.5	Reactor operator fails to reclose CRD/RBCW Breakers	Screening Value
CRD-XHE-FO-CRD	.5	Operator fails to initiate second pump and open valves	Screening Value
EMERGENCY SERVICE WATER SYSTEM			
ESW-XHE-FO-EHS	9.0E-1	Failure of operator to initiate emergency heat sink	HRA Methodology Developed from Tables C-59 through C-69
HIGH PRESSURE SERVICE WATER SYSTEM			
HSW-XHE-FO-PS9	.5	Operator fails to open bypass line valve 2344	Screening Value
PRIMARY CONTAINMENT VENTING SYSTEM			
PCV-XHE-FO-PCV	.5	Operator fails to vent	Screening Value
REACTOR BUILDING COOLING WATER SYSTEM			
RBC-XHE-FO-LCVAL	.5	Operator failure to open locked closed valves	Screening Value

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
RBC-XHE-FO-SWCH	.5	Operator fails to switch to RBCW system following LOSP	Screening Value
STANDBY LIQUID CONTROL SYSTEM			
SLC-XHE-FO-SLC	2.0E-2	Operator fails to initiate SLC	ATWS HRA Methodology BNL [29]
NON-RECOVERY			
DCHWNR68MIN	5.0E-1	DC hardware not recovered in 68 minutes	Methodology [2]
DCHWNR150MIN	4.0E-1	DC hardware not recovered in 150 minutes	Methodology [2]
DCHWNR9HR	3.0E-2	DC hardware not recovered in 9 hours	Methodology [2]
DCHWNR12HR	3.0E-2	DC hardware not recovered in 12 hours	Methodology [2]
DCHWNR13HR	3.0E-2	DC hardware not recovered in 13 hours	Methodology [2]
DCHWNR14HR	2.5E-2	DC hardware not recovered in 14 hours	Methodology [2]
DCHWNR17HR	2.5E-2	DC hardware not recovered in 17 hours	Methodology [2]

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
DCHWNR18HR	2.0E-2	DC hardware not recovered in 18 hours	Methodology [2]
DGACTNR3HR	3.0E-2	Diesel Generator (DG) actuation not recovered in 3 hours	Methodology [2]
DGACTNR5HR	3.0E-2	DG actuation not recovered in 5 hours	Methodology [2]
DGACTNR7HR	3.0E-2	DG actuation not recovered in 7 hours	Methodology [2]
DGACTNR9HR	2.0E-2	DG actuation not recovered in 9 hours	Methodology [2]
DGACTNR12HR	2.0E-2	DG actuation not recovered in 12 hours	Methodology [2]
DGACTNR16HR	1.0E-2	DG actuation not recovered in 16 hours	Methodology [2]
DGCCFNR3HR	7.0E-1	DG common cause failure not recovered in 3 hours	Methodology [2]
DGCCFNR5HR	6.0E-1	DG common cause failure not recovered in 5 hours	Methodology [2]
DGCCFNR7HR	5.0E-1	DG common cause failure not recovered in 7 hours	Methodology [2]
DGCCFNR9HR	4.5E-1	DG common cause failure not recovered in 9 hours	Methodology [2]

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
DGCCFNR12HR	4.0E-1	DG common cause failure not recovered in 12 hours	Methodology [2]
DGHWNR3HR	8.0E-1	DG hardware not recovered in 3 hours	Methodology [2]
DGHWNR5HR	7.0E-1	DG hardware not recovered in 5 hours	Methodology [2]
DGHWNR7HR	6.0E-1	DG hardware not recovered in 7 hours	Methodology [2]
DGHWNR9HR	5.8E-1	DG hardware not recovered in 9 hours	Methodology [2]
DGHWNR12HR	5.5E-1	DG hardware not recovered in 12 hours	Methodology [2]
DGHWNR16HR	5.0E-1	DG hardware not recovered in 16 hours	Methodology [2]
DGMANR3HR	7.0E-1	DG maintenance not recovered in 3 hours	Methodology [2]
DGMANR5HR	6.0E-1	DG maintenance not recovered in 5 hours	Methodology [2]
DGMANR7HR	5.0E-1	DG maintenance not recovered in 7 hours	Methodology [2]
DGMANR9HR	4.5E-1	DG maintenance not recovered in 9 hours	Methodology [2]

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Continued)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
DGMANR12HR	4.0E-1	DG maintenance not recovered in 12 hours	Methodology [2]
DGMANR16HR	2.0E-1	DG maintenance not recovered in 16 hours	Methodology [2]
PCSNR13HR	1.0E-2	Power Conversion System not recovered in 13 hours	Methodology [2]
LOSPNR15MIN	6.0E-1	Offsite power not restored in 15 minutes	Methodology [26] Appendix D
LOSPNR30MIN	3.1E-1	Offsite power not restored in 30 minutes	Methodology [26] Appendix D
LOSPNR45MIN	1.3E-1	Offsite power not restored in 45 minutes	Methodology [26] Appendix D
LOSPNR68MIN	1.1E-1	Offsite power not restored in 68 minutes	Methodology [26] Appendix D
LOSPNR150MIN	9.6E-2	Offsite power not restored in 150 minutes	Methodology [26] Appendix D

Table 4.8-2

Summary of Post-Accident LOCA and Transient Human Actions
(Concluded)

RECOVERY TERM	MEAN VALUE	DESCRIPTION	COMMENTS
LOSPNR5HR	4.8E-2	Offsite power not restored in 5 hours	Methodology [26] Appendix D
LOSPNR7HR	3.2E-2	Offsite power not restored in 7 hours	Methodology [26] Appendix D
LOSPNR9HR	2.3E-2	Offsite power not restored in 9 hours	Methodology [26] Appendix D
LOSPNR12HR	1.5E-2	Offsite power not restored in 12 hours	Methodology [26] Appendix D
LOSPNR13HR	1.3E-2	Offsite power not restored in 13 hours	Methodology [26] Appendix D
LOSPNR14HR	1.2E-2	Offsite power not restored in 14 hours	Methodology [26] Appendix D
LOSPNR17HR	8.1E-3	Offsite power not restored in 17 hours	Methodology [26] Appendix D
LOSPNR18HR	7.3E-3	Offsite power not restored in 18 hours	Methodology [26] Appendix D

Table 4.8-3

Most Important ATWS Human Errors from the BNL Analysis[1]

Event Description	Conditions/Factors	Median/Mean Human Error Probability
Manual Scram	--	<1E-4/<1E-4
Initiate SLC	• Mechanical failure of control rods • At least 4 minutes available • No reluctance	0.005/0.02
Inhibit ADS	• Mechanical failure of control rods • SLC successful	0.02/0.09
Manual Depressurization	• Mechanical failure of control rods • SLC successful • ADS originally inhibited	0.14/0.2

(1) See Reference 29

4.9 Data Base Development

This section describes the development of the data base. The first subsection identifies the sources used to establish the data base for requantification of the Peach Bottom sequences. The assumptions used in the data development, limitations and uncertainty distributions associated with the data, and the use of plant-specific and generic data are presented in subsequent subsections. Finally, the data is described on a system by system basis.

4.9.1 Sources of Information for the Data Base

A review of plant-specific data was conducted. Major system pump and valve histories as well as "hi-spot" reports [10] were reviewed. It was found in almost all cases that plant-specific data fell within the bounds of current Accident Sequence Evaluation Program (ASEP) generic data. This was determined with help from the QCG data specialist who used statistical tests to demonstrate the viability of using the generic data. The ASEP data was updated to incorporate the LaSalle information [45]. In a few cases, plant-specific data were used as noted in the data table. Other sources of data included WASH-1400, other Probabilistic Risk Assessments (PRAs), and miscellaneous reports as indicated in the data table. The initiating event plant frequencies are plant specific except for A, S1, S2, S3, and the bus initiators which are ASEP generic. Recovery data and other human error probabilities were derived from the Human Reliability Analysis (HRA) and generic ASEP recovery data as indicated in Section 4.8.

4.9.2 Assumptions and Limitations in the Data Base

The System Analysis section (4.6) describes assumptions associated with a particular system. There are generic assumptions applicable to several systems. These assumptions are described in this subsection.

Failure to restore the system was treated at the component level. The two main contributors to the failure to restore terms were; failure to restore the circuit breakers for pumps, and failure to restore the valves to operability after they had been isolated for maintenance. HRA and ASEP rules were used to obtain a nominal estimate of pre-accident and post-accident failure probabilities. The pre-accident and post-accident HRA values [25] make use of generic values but consider plant-specific procedures. Therefore, the HRA values are calculated with plant-specific considerations but are not plant-specific numbers. The pre-accident failure probabilities are based on Peach Bottom normal maintenance practices for isolating a portion of the system when the system is under maintenance and normal practices in restoring the system. In calculating the post-accident failure probabilities, the time available to perform recovery actions, indicators to operators for diagnosing a problem, and complexity of recovery actions were considered. Anticipated Transient Without Scram (ATWS) HRA values were derived from a detailed analysis covered in Reference 29.

In general, the beta common cause factor values were based on the data and methodology of Karl Fleming's report on reactor operating experience [23]. Higher order common cause factors for failure of more than two components are from Corey Atwood's common cause fault rate documents for valves, diesel generators, pumps and instrumentation [37,38,39,40]. There were some exceptions to the above technique. These include the battery common cause values which were based on NUREG-0666 [24], and the common cause value for air-operated valves which used a "generic" 0.1 beta value. Finally, multiple SRVs failing to close was based in part on the assumption of zero events in the available BWR reactor years. More on the treatment of common cause can be found in Section 4.7.

4.9.3 Plant-Specific Analysis and Use of Generic Data

When plant-specific data fell within the bounds of ASEP generic data, generic data were used. Plant-specific failure values that were based on zero or one failure were not used. It was felt that, in these instances, there were too few corresponding trials represented in the Peach Bottom experience base. The generic data represent a much larger experience base leading to a more certain estimate in the failure probabilities for most components. Therefore, generic data were once again used. Appendix D summarizes the plant-specific data values used in the Peach Bottom analysis.

4.9.4 Uncertainty Distributions

For most of the parameter estimates used in the study, lognormal uncertainty distributions were assumed. This is a common practice used in many of the PRAs conducted to date. Two general exceptions were made to this standard practice. First, the uncertainty distributions for human error events have a less extensive data base to draw upon than the component event data base. Confidence did exist in the mean estimates and the corresponding upper and lower bounds of the human error data. For this reason, a maximum entropy distribution was frequently used by fixing the mean and upper and lower bounds in the analysis. This type of distribution was used for many of the human-related events in the study. The other exception is the ATWS human-related error estimates. Since the ATWS analyses conducted by Brookhaven National Laboratory provided lognormal distributions for the human error uncertainties, these were used "as is" for most cases in the study. In a few cases, distributions were such that probabilities of greater than 1.0 were possible out at the 97th percentile or beyond. In these cases, log-uniform distributions were developed using parameters of the lognormal distributions (i.e., mean, variance) but with the upper bound limited to 1.0 in accordance with the axioms of probability.

The loss of offsite power initiating event and recovery times were modeled using Bayesian methods [26]. The modeling utilizes a composite probability model fitted to three sources of data as a method of predicting the time to recovery (including uncertainty) of loss of offsite power. The three sources are plant-centered losses, grid losses and severe weather losses. A Bayesian approach was also used to model the uncertainty in the frequency of the initiating events. Combining the composite model and the initiating event model yields a complete model that incorporates uncertainty into the loss of offsite power events.

4.9.5 Complete Data Base Description

This subsection contains the data used in the analysis to quantify the accident sequence frequencies. Table 4.9-1 presents the majority of the data used in the analysis. Additional data on recovery actions can be found in Section 4.8. The information in the table is presented in alphabetical order by major system heading. A miscellaneous heading has been developed for basic events that don't fit logically into a major system heading. Data for the initiating events and beta factor values are presented at the end of the table. Within each system category, the basic events are listed alphanumerically.

Table 4.9-1
Peach Bottom Event Data

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ACTUATION AND CONTROL:									
ESF-ACS-FC-MDPA	LPCI Pump A permissive fails to override Unit 3 stop signal	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	WASH-1400 [4]* The failure rate is 2.66E-6/hr with a mission time of 360 hrs and an error factor of 10
ESF-ACS-FC-MDPB	LPCI Pump B permissive fails to override Unit 3 stop signal	-	-	-	-	-	1.0E-3	same as ESF-ACS-FC-MDPA	--
ESF-ADS-FC-LI13A	Limit switch fails to indicate Motor-Operated Valve 13A fully open	3.75E-4/d	3	-	3.0E-4	3	3.75E-4	single component event	WASH-1400 [4]
ESF-ADS-FC-LI13B	Limit switch fails to indicate Motor-Operated Valve 13B fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--

* Instrumentation including sensor, transmitter, and process switch. Assume monthly test.

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ADS-FC-LI13C	Limit switch fails to indicate Motor-Operated Valve 13C fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ADS-FC-LI13D	Limit switch fails to indicate Motor-Operated Valve 13D fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASD-FC-SC15A	Limit switch fails to indicate Motor-Operated Valve 15A fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASD-FC-SC15B	Limit switch fails to indicate Motor-Operated Valve 15B fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASD-FC-SC15C	Limit switch fails to indicate Motor-Operated Valve 15C fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASD-FC-SC15D	Limit switch fails to indicate Motor-Operated Valve 15D fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASD-FC-SDC17	Limit switch fails to indicate Motor-Operated Valve 17 fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASD-FC-SDC18	Limit switch fails to indicate Motor-Operated Valve 18 fully open	-	-	-	-	-	3.75E-4	same as ESF-ADS-FC-LI13A	--
ESF-ASL-FC-LT72A	Reactor water level transmitter A fails	1.4E-6/hr	3	360	4.0E-4	3	5.0E-4	single component event	IEEE-500 [53]
ESF-ASL-FC-LT72B	Reactor water level transmitter B fails	-	-	-	-	-	5.0E-4	same as ESF-ASL-FC-LT72A	--
ESF-ASL-FC-LT72C	Reactor water level transmitter C fails	-	-	-	-	-	5.0E-4	same as ESF-ASL-FC-LT72A	--
ESF-ASL-FC-LT72D	Reactor water level transmitter D fails	-	-	-	-	-	5.0E-4	same as ESF-ASL-FC-LT72A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASL-FC-P101A	Hardware failure of LIS-2-3-101A no isolation signal produced	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	WASH-1400 [4], plant data
ESF-ASL-FC-P101B	Hardware failure of LIS-2-3-101B no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASL-FC-P101C	Hardware failure of LIS-2-3-101C no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASL-FC-P101D	Hardware failure of LIS-2-3-101D no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASL-HW-CSTL1	CST level Sensor 1 fails (LISL-171)	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	WASH-1400 [4], plant data
ESF-ASL-HW-CSTL2	CST level Sensor 2 fails (LISL-170)	-	-	-	-	-	1.0E-3	same as ESF-ASL-HW-CSTL1	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASL-HW-CSTL3	CST level Sensor 3 fails (LS-74)	-	-	-	-	-	1.0E-3	same as ESF-ASL-HW-CSTL1	--
ESF-ASL-HW-CSTL4	CST level Sensor 4 fails (LS-75)	-	-	-	-	-	1.0E-3	same as ESF-ASL-HW-CSTL1	--
ESF-ASL-LRXLEVEL	Reactor level below SDC permissive of 0"	-	-	-	-	-	1.0 or 0.0	Flag event, 1.0 (injection lost) 0.0 (injection working)	--
ESF-ASL-NO-RSXDA	Shroud water level permissive fails	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	WASH-1400 [4], plant data
ESF-ASL-NO-RSXDB	Shroud water level permissive fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-LH12A	Hardware failure of PIS-5-12A no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-LH12B	Hardware failure of PIS-5-12B no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASP-FC-LH12C	Hardware failure of PIS-5-12C no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-LH12D	Hardware failure of PIS-5-12D no isolation signal produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-LSPHC	HPCI pump suction pressure sensor fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-LSPRC	RCIC pump suction pressure sensor fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-P100A	Drywell pressure transmitter A fails	1.4E-6/hr	3	360	4.0E-4	3	5.0E-4	single component event	IEEE-500 [53]
ESF-ASP-FC-P100B	Drywell pressure transmitter B fails	-	-	-	-	-	5.0E-4	same as ESF-ASP-FC-P100A	--
ESF-ASP-FC-P100C	Drywell pressure transmitter C fails	-	-	-	-	-	5.0E-4	same as ESF-ASP-FC-P100A	--
ESF-ASP-FC-P100D	Drywell pressure transmitter D fails	-	-	-	-	-	5.0E-4	same as ESF-ASP-FC-P100A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASP-FC-P101A	HPCI high drywell pressure Sensor A fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASP-FC-P101B	HPCI high drywell pressure Sensor B fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASP-FC-P101C	HPCI high drywell pressure Sensor C fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASP-FC-P101D	HPCI high drywell pressure Sensor D fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-FC-P101A	--
ESF-ASP-FC-P128C	Hardware failure of PS-2-128C such that isolation signal is produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-P128D	Hardware failure of PS-2-128D such that isolation signal is produced	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASP-NOHDPEL	High drywell pressure signal not generated in early time frames	-	-	-	-	-	1.0 or 0.0	Flag event, 1.0 (do not expect high drywell pressure early)	--
ESF-ASP-NOHDPLT	High drywell pressure signal not generated in late time frame	-	-	-	-	-	1.0 or 0.0	Flag event, 1.0 (do not expect high drywell pressure late)	--
ESF-LOG-HW-RHRA	RHR control Logic A circuitry fails	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	Calvert Cliffs IREP [54], plant data
ESF-LOG-HW-RHRB	RHR control Logic B circuitry fails	-	-	-	-	-	1.61E-3	same as ESF-LOG-HW-RHRA	--
ESF-PER-LCI3ACT	Unit 3 LPCI pumps receive start signal (early time frame)	1.0E-3/d	10	-	-	10	1.0E-3	single event	engineering judgement
ESF-PER-LCI3ACT2	Other unit LPCI pumps receive start signal	5.0E-4/d	10	-	-	10	5.0E-4	single event	engineering judgement
ESF-PER-LIA3TEST	Other unit 3 LPCI pump A not under test	-	-	-	-	-	1.0	Flag event, 1.0 (pump not in test), always 1.0 for Peach Bottom sequences	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-ASP-FC-PL52A	LPCS, LPCI low reactor pressure Sensor A fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-PL52B	LPCS, LPCI low reactor pressure Sensor B fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-PL52C	LPCS, LPCI low reactor pressure Sensor C fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-FC-PL52D	LPCS, LPCI low reactor pressure Sensor D fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-HIDWPRES	Drywell pressure is above the SDC range	-	-	-	-	-	1.0 or 0.0	Flag event, 1.0 (expect high drywell pressure) 0.0 (sequence doesn't expect high drywell pressure)	--
ESF-ASP-HW-EX72A	RCIC high exhaust pressure Sensor A fails	-	-	-	-	-	1.0E-3	same as ESF-ASL-NO-RSXDA	--
ESF-ASP-HW-EX72B	RCIC high exhaust pressure Sensor B fails	-	-	-	-		1.0E-3	same as ESF-ASL-NO-RSXDA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-PER-LIB3TEST	Other unit 3 LPCI pump B not under test	-	-	-	-	-	1.0	same as ESF-PER-L1A3TEST	--
ESF-PER-LIC3TEST	Other unit 3 LPCI pump C not under test	-	-	-	-	-	1.0	same as ESF-PER-L1A3TEST	--
ESF-PER-LID3TEST	Other unit 3 LPCI pump D not under test	-	-	-	-	-	1.0	same as ESF-PER-L1A3TEST	--
ESF-PER-RXLNTMET	Reactor level below shroud level of 2/3 the core length	-	-	-	-	-	1.0 or 0.0	Flag event, 1.0 (injection lost) 0.0 (injection working)	--
ESF-PWR-FC-4160A	Bus 4160A power permissive sensor fails	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	WASH-1400 [4], plant data
ESF-PWR-FC-4160B	Bus 4160B power permissive sensor fails	-	-	-	-	-	1.0E-3	same as ESF-PWR-FC-4160A	--
ESF-PWR-FC-4160C	Bus 4160C power permissive sensor fails	-	-	-	-	-	1.0E-3	same as ESF-PWR-FC-4160A	--
ESF-PWR-FC-4160D	Bus 4160D power permissive sensor fails	-	-	-	-	-	1.0E-3	same as ESF-PWR-FC-4160A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-XHE-FO-ADSBT	Operator fails to valve in nitrogen bottle bank	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-DATWS	Operator fails to depressurize during an ATWS	-	-	-	-	-	2.0E-1	single event	HRA
ESF-XHE-FO-DEPRE	Operator fails to manually depressurize reactor	-	-	-	-	5	1.0E-2	single event	HRA
ESF-XHE-FO-DEPSD	Operator fails to manually depressurize the reactor	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-HCICL	Operator fails to control reactor level	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-HCILV	Operator fails to control reactor level for HPCI	-	-	-	-	-	3.3E-1	single event	HRA
ESF-XHE-FO-HCIL8	Operator fails to control reactor level for HPCI	-	-	-	-	-	1.0E-1	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-XHE-FO-HCIRL	Operator fails to realign HPCI suction source	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-HPACT	Operator fails to actuate HPCI	-	-	-	-	-	1.2E-1	single event	HRA
ESF-XHE-FO-HPSAC	Operator fails to actuate HPCI	-	-	-	-	-	6.0E-2	single event	HRA
ESF-XHE-FO-HPSAT	Operator fails to backup high pressure system actuation	-	-	-	-	10	1.0E-1	all sequences	HRA
ESF-XHE-FO-HPSRL	Operator fails to realign HPCI suction source	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-HSWIN	Operator fails to realign HPSW for injection	-	-	-	-	10	1.0E-1	single event	HRA
ESF-XHE-FO-LPSAT	Operator fails to backup low pressure system actuation	-	-	-	-	-	.5	single event	HRA/screening value

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-XHE-FO-OVRID	Operator fails to override shroud level permissive	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-RCICL	Operator fails to control reactor level for RCIC	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-RCICO	Operator fails to isolate RCIC	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-RCILV	Operator fails to to control reactor level for RCIC	-	-	-	-	-	3.3E-1	single event	HRA
ESF-XHE-FO-RCIL8	Operator fails to control reactor level for RCIC	-	-	-	-	-	1.0E-1	single event	HRA
ESF-XHE-FO-RCIRL	Operator fails to realign RCIC suction source	-	-	-	-	-	.5	single event	HRA/screening value
ESF-XHE-FO-RHRAT	Operator fails to align RHR cooling mode	-	-	-	-	10	1.0E-5	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESF-XHE-MC-CSTLV	Common cause mis-calibration of CST low level sensors	-	-	-	2.5E-5	10	6.65E-5	single event	HRA
ESF-XHE-MC-HDPRS	Common cause miscal-ibration of high drywell pressure sensors	-	-	-	1.0E-4	10	2.66E-4	single event	HRA
ESF-XHE-MC-PRES	Operator miscalibrates Rx pressure sensors	-	-	-	2.0E-4	10	5.32E-4	single event	HRA
ESF-XHE-MC-VSLVL	Operator miscalibrates all Rx level sensors	-	-	-	5.0E-5	10	1.33E-4	single event	HRA
AUTOMATIC DEPRESSURIZATION SYSTEM:									
ADS-ACT-HW-DIV1	Actuation circuit failure - Division 1	1.61E-3/d	5	-	-	5	1.61E-3	single component event	The failure rate is from the methods developed in the IREP Calvert Cliffs Report [54].
ADS-ACT-HW-DIV2	Actuation circuit failure - Division 2	-	-	-	-	-	1.61E-3	same as ADS-ACT-HW-DIV1	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ADS-PTF-VF-NADSV	Failure of 4 of 6 non-ADS valves to open	1.0E-6/d	10	-	-	10	1.0E-6	single event	engineering judgement
ADS-TSW-FT-DC125	Hardware failure of power transfer switches	1.0E-6/d	10	-	-	10	1.0E-6	single event	engineering judgement
CONDENSATE SYSTEM:									
CDS-SYS-FC-COND	Subsequent loss of the condensate system due to hardware/operator failure	-	-	-	-	-	1.0E-1	single event	engineering judgement, HRA
CONTAINMENT SPRAY SYSTEM (RHR):									
CSS-CCF-LF-MOVS	Common cause failure of CSS injection valves	3.0E-3/d x 0.049	10	-	-	-	1.47E-4	single event	ASEP generic, revision 1 [55]. Basic event converted to: CSS-MOV-CC-CCF *BETA-2MOVS
CSS-MOV-CC-MV26A	Motor-operated valve 26A fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
CSS-MOV-CC-MV26B	Motor-operated valves 26B fails to open	-	-	-	-	-	3.0E-3	same as CSS-MOV-CC-MV26A	--
CSS-MOV-CC-MV31A	Motor-operated valve 31A fails to open	-	-	-	-	-	3.0E-3	same as CSS-MOV-CC-MV26A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABLILITY: MEDIAN	UNAVAILABLILITY: EF	UNAVAILABLILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
CSS-MOV-CC-MV31B	Motor-operated valve 31B fails to open	-	-	-	-	-	3.0E-3	same as CSS-MOV-CC-MV26A	--
CSS-MOV-MA-MV26A	Motor-Operated Valve 26A out for maintenance	2.0E-4/d	10	-	7.5E-5	10	2.0E-4	single component event	Original ASEP generic document [2]
CSS-MOV-MA-MV26B	Motor-Operated Valve 26B out for maintenance	-	-	-	-	-	2.0E-4	same as CSS-MOV-MA-MV26A	--
CONTROL ROD DRIVE SYSTEM:									
CRD-MDP-FR-PA	Motor-Driven Pump A fails to run	3.0E-5/hr	10	24	2.7E-4	10	7.2E-4	single component event	ASEP generic, revision 1 [55]
CRD-MDP-FR-PB	Motor-Drive Pump B fails to run	-	-	-	-	-	7.2E-4	same as CRD-MDP-FR-PA	--
CRD-MDP-FS-PA	Motor-Driven Pump A fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
CRD-MDP-FS-PB	Motor-Drive Pump B fails to start	-	-	-	-	-	3.0E-3	same as CRD-MDP-FS-PA	--
CRD-PTF-MA-PB	Motor-Driven Pump B train out for maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
CRD-XHE-FO-BRKRS	Reactor operator fails to reclose CRD/RBCW breakers	-	-	-	-	-	.5	single event	HRA/screening value
CRD-XHE-FO-CRD	Operator fails to initiate second pump and open valves	-	-	-	-	-	.5	single event	HRA/screening value
CRD-XHE-RE-PB	Motor-Driven Pump B not properly restored after maintenance	-	-	-	4.0E-4	20	2.1E-3	single event	HRA
ELECTRIC POWER SYSTEM:									
ACP-BAC-LP-416A	4160V AC Bus A fails	1.25E-7/hr	5	40	3.1E-6	5	5.0E-6	single component event	Value typical of range (IREP [54] uses 1E-8/hr for all buses, Oconee NSAC [57] uses 4E-6/hr)
ACP-BAC-LP-416B	4160V AC Bus B fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--
ACP-BAC-LP-416C	4160V AC Bus C fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ACP-BAC-LP-416D	4160V AC Bus D fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--
ACP-CCF-LP-DGS	Common cause failure of diesel generators to start	3E-3/d x 0.013	3	-	-	-	3.9E-5	single event	Basic event converted to: ACP-DGN-LP-CCF *BETA-4DGNS, plant data used for ACP-DGN-LP-CCF, ASEP generic, revision 1 [55] used for BETA-4DGNS
ACP-DGN-FR-EDGA	Emergency diesel generator A fails to run	2.0E-3/hr	10	8 see Note (a)	4.5E-3	10	1.6E-2	single event	ASEP generic, revision 1 [55]
ACP-DGN-FR-EDGB	Emergency diesel generator B fails to run	-	-	-	-	-	1.6E-2	same as ACP-DGN-FR-EDGA	--
ACP-DGN-FR-EDGC	Emergency diesel generator C fails to run	-	-	-	-	-	1.6E-2	same as ACP-DGN-FR-EDGA	--
ACP-DGN-FR-EDGD	Emergency diesel generator D fails to run	-	-	-	-	-	1.6E-2	same as ACP-DGN-FR-EDGA	--

Note: (a) Number of hours based upon typical eight hour coping time requirement being incorporated in Station Blackout Rule.

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ACP-DGN-LP-EDGA	Emergency diesel generator A fails to start	3.0E-3/d	3	-	-	3	3.0E-3	single event	plant data
ACP-DGN-LP-EDGB	Emergency diesel generator B fails to start	-	-	-	-	-	3.0E-3	same as ACP-DGN-LP-EDGA	--
ACP-DGN-LP-EDGC	Emergency diesel generator C fails to start	-	-	-	-	-	3.0E-3	same as ACP-DGN-LP-EDGA	--
ACP-DGN-LP-EDGD	Emergency diesel generator D fails to start	-	-	-	-	-	3.0E-3	same as ACP-DGN-LP-EDGA	--
ACP-DGN-MA-EDGA	EDG A out for maintenance	6.0E-3/d	10	-	2.26E-3	10	6.0E-3	single component event	ASEP generic, revision 1 [55]
ACP-DGN-MA-EDGB	EDG B out for maintenance	-	-	-	-	-	6.0E-3	same as ACP-DGN-MA-EDGA	--
ACP-DGN-MA-EDGC	EDG C out for maintenance	-	-	-	-	-	6.0E-3	same as ACP-DGN-MA-EDGA	--
ACP-DGN-MA-EDGD	EDG D out for maintenance	-	-	-	-	-	6.0E-3	same as ACP-DGN-MA-EDGA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ACP-DGN-RE-EDGA	Failure to restore EDG A after test or maintenance	-	-	-	3.0E-4	10	7.98E-4	single event	HRA
ACP-DGN-RE-EDGB	Failure to restore EDG B after test or maintenance	-	-	-	-	-	7.98E-4	same as ACP-DGN-RE-EDGA	--
ACP-DGN-RE-EDGC	Failure to restore EDG C after test or maintenance	-	-	-	-	-	7.98E-4	same as ACP-DGN-RE-EDGA	--
ACP-DGN-RE-EDGD	Failure to restore EDG D after test or maintenance	-	-	-	-	-	7.98E-4	same as ACP-DGN-RE-EDGA	--
ACP-DGN-TE-EDGA	EDG A unavailable due to test	2.3E-3/d	3	-	-	3	2.3E-3	single event	engineering judgement
ACP-DGN-TE-EDGB	EDG B unavailable due to test	-	-	-	-	-	2.3E-3	same as ACP-DGN-TE-EDGA	--
ACP-DGN-TE-EDGC	EDG C unavailable due to test	-	-	-	-	-	2.3E-3	same as ACP-DGN-TE-EDGA	--
ACP-DGN-TE-EDGD	EDG D unavailable due to test	-	-	-	-	-	2.3E-3	same as ACP-DGN-TE-EDGA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
DCP-BAT-LP-A2	Unit 2 Battery A fails	1E-6/hr	3	2160 hr	8.6E-4	3	1.08E-3	single component event	ASEP generic, revision 1 [55]
DCP-BAT-LP-B2	Unit 2 Battery B fails	-	-	-	-	-	1.08E-3	same as DCP-BAT-LP-A2	--
DCP-BAT-LP-C2	Unit 2 Battery C fails	-	-	-	-	-	1.08E-3	same as DCP-BAT-LP-A2	--
DCP-BAT-LP-C3	Unit 3 Battery C fails	-	-	-	-	-	1.08E-3	same as DCP-BAT-LP-A2	--
DCP-BAT-LP-D2	Unit 2 Battery D fails	-	-	-	-	-	1.08E-3	same as DCP-BAT-LP-A2	--
DCP-BAT-LP-D3	Unit 3 Battery D fails	-	-	-	-	-	1.08E-3	same as DCP-BAT-LP-A2	--
DCP-BDC-LP-125A	125VDC Bus A fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--
DCP-BDC-LP-125B	125VDC Bus B fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--
DCP-BDC-LP-125C	125VDC Bus C fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--
DCP-BDC-LP-125D	125VDC Bus D fails	-	-	-	-	-	5.0E-6	same as ACP-BAC-LP-416A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
DCP-CCF-LP-BAT	Common cause failure of batteries	1.08E-3/d x 0.0025	10	-	-	-	2.25E-6	single event	ASEP generic, revision 1 [55]; Beta factor calculated from NUREG-0666 [24]/ Basic event converted to: DCP-BAT-LF-CCF *BETA-5BAT
DCP-INV-LP-24C	24V Inverter C fails	1.0E-4/hr	3	40	3.2E-3	3	4.0E-3	single component event	IREP [56]
DCP-INV-LP-24D	24V Inverter D fails	-	-	-	-	-	4.0E-3	same as DCP-INV-LP-24C	--
DCP-REC-LP-1	Unit 2 Charger A fails	1.0E-6/hr	3	8 see Note (b)	4.8E-6	3	8.0E-6	single component event	ASEP generic, revision 1 [55]
DCP-REC-LP-2	Unit 2 Charger B fails	-	-	-	-	-	8.0E-6	same as DCP-REC-LP-1	--
DCP-REC-LP-3	Unit 2 Charger C fails	-	-	-	-	-	8.0E-6	same as DCP-REC-LP-1	--

Note:
(b) Corresponds to the diesel generator run time used in the analysis.

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
DCP-REC-LP-4	Unit 2 Charger D fails	-	-	-	-	-	8.0E-6	same as DCP-REC-LP-1	--
DGACTA	EDG A actuation fails	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	Failure rate from methods developed in Calvert Cliffs IREP report [27]
DGACTB	EDG B actuation fails	-	-	-	-	-	1.61E-3	same as DGACTA	--
DGACTC	EDG C actuation fails	-	-	-	-	-	1.61E-3	same as DGACTA	--
DGACTD	EDG D actuation fails	-	-	-	-	-	1.61E-3	same as DGACTA	--
EMERGENCY SERVICE WATER SYSTEM:									
ESW-ACX-FC-HX1	Cooling coil unit on HX1 fails	1.25E-6/hr	3	40	4.0E-5	3	5.0E-5	single event	IEEE-500 [53]
ESW-ACX-FC-HXN* where N* = 2,3,4,6,7,9,10, 12,13,15,16,18, 19,21,22,24,25, 27,28	Cooling coil unit on HXN* fails	-	-	-	-	-	5.0E-5	same as ESW-ACX-FC-HX1	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-AOV-CC-0241A	Air-Operated Valve 0241A fails to open	1.0E-3/d	3	-	8.0E-4	3	1.0E-3	single component event	ASEP generic, revision 1 [55]
ESW-AOV-CC-0241B	AOV 0241B fails to open	-	-	-	-	-	1.0E-3	same as ESW-AOV-CC-0241A	--
ESW-AOV-CC-0241C	AOV 0241C fails to open	-	-	-	-	-	1.0E-3	same as ESW-AOV-CC-0241A	--
ESW-AOV-CC-0241D	AOV 0241D fails to open	-	-	-	-	-	1.0E-3	same as ESW-AOV-CC-0241A	--
ESW-AOV-CC-AV1	Air operated valve 1 fails to open	1.0E-3/d	3	-	8.0E-4	3	1.0E-3	single component event	Original ASEP generic document [2]
ESW-AOV-CC-AVM* where M* = 2,3,4,5,6,7,8,9, 10,11,12,13,14, 15,16,17,18,19,20	Air operated valve M* fails to open	-	-	-	-	-	1.0E-3	same as ESW-AOV-CC-AV1	--
ESW-AOV-MA-0241A	Valve 0241A out for maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	Original ASEP generic document [2]
ESW-AOV-MA-0241B	Valve 0241B out for maintenance	-	-	-	-	-	2.0E-4	same as ESW-AOV-MA-0241A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-AOV-MA-0241C	Valve 0241C out for maintenance	-	-	-	-	-	2.0E-4	same as ESW-AOV-MA-0241A	--
ESW-AOV-MA-0241D	Valve 0241D out for maintenance	-	-	-	-	-	2.0E-4	same as ESW-AOV-MA-0241A	--
ESW-AOV-MA-AV1	Valve 1 out for maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	Original ASEP generic document [2]
ESW-AOV-MA-AVM* where M* = 2,3,4,5,6,7,8,9, 10,11,12,13,14, 15,16,17,18,19,20	Valve M* out for maintenance	-	-	-	-	-	2.0E-4	same as ESW-AOV-MA-AV1	--
ESW-CCF-LF-AOVS	Common cause loss of flow to air operated valves	1.0E-3/d x 0.055	3	-	-	-	5.5E-5	single event	Original ASEP generic document [2]. Basic event converted to: ESW-AOV-CC-CCF *BETA-3AOVS
ESW-CCF-MC-ECT	Common cause failure of all emergency cooling towers	-	-	-	9.0E-9	10	2.39E-8	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-CKV-HW-CV516	Check valve 516 fails to open	-	-	-	-	-	1.0E-4	same as ESW-CKV-HW-C515A	--
ESW-FAN-FR-HX1	Heat exchanger 1 fan unit fails to run	1.25E-5/hr	3	40	4.0E-4	3	5.0E-4	single event	WASH-1400 [4]
ESW-FAN-FR-HXP* where P* = 2,3,4,6,7,9,10, 12,13,15,16,18, 19,21,22,24,25, 27,28	Fan unit for HXP* fails to run	-	-	-	-	-	5.0E-4	same as ESW-FAN-FR-HX1	--
ESW-FAN-FS-HX1	Fan unit for HX1 fails to start	3.75E-4/d	3	-	3.0E-4	3	3.75E-4	single component event	WASH-1400 [4]
ESW-FAN-FS-HXP* where P* = 2,3,4,6,7,9,10, 12,13,15,16,18, 19,21,22,24,25, 27,28	Fan unit for HXP* fails to start	-	-	-	-	-	3.75E-4	same as ESW-FAN-FS-HX1	--
ESW-FAN-MA-HX1	Fan unit for HX1 out for maintenance	1.86E-3	10	-	7.0E-4	10	1.86E-3	single component event	WASH-1400 [4]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-FAN-MA-HXP* where P* = 2,3,4,6,7,9,10, 12,13,15,16,18, 19,21,22,24,25, 27,28	Fan unit for HXP* out for maintenance	-	-	-	-	-	1.86E-3	same as ESW-FAN-MA-HX1	--
ESW-MDP-FR-ECW	ECW pump fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
ESW-MDP-FR-MDPA	ESW Pump A fails to run	-	-	-	-	-	1.2E-3	same as ESW-MDP-FR-ECW	--
ESW-MDP-FR-MDPB	ESW Pump B fails to run	-	-	-	-	-	1.2E-3	same as ESW-MDP-FR-ECW	--
ESW-MDP-FS-ECW	ECW pump fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
ESW-MDP-FS-MDPA	ESW Pump A fails to start	-	-	-	-	-	3.0E-3	same as ESW-MDP-FS-ECW	--
ESW-MDP-FS-MDPB	ESW Pump B fails to start	-	-	-	-	-	3.0E-3	same as ESW-MDP-FS-ECW	--
ESW-MDP-MA-ECW	ECW pump out for maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-MDP-MA-MDPA	ESW Pump A out for maintenance	-	-	-	-	-	2.0E-3	same as ESW-MDP-MA-ECW	--
ESW-MDP-MA-MDPB	ESW Pump B out for maintenance	-	-	-	-	-	2.0E-3	same as ESW-MDP-MA-ECW	--
ESW-MOV-CC-M0841	Motor-operated valve 0841 fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
ESW-MOV-MA-M0841	Valve 0841 out for maintenance	8E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	Original ASEP generic document [2]
ESW-MOV-MA-M2972	Motor-operated valve 2972 out for maintenance	-	-	-	-	-	2.0E-4	same as ESW-MOV-MA-M0841	--
ESW-MOV-PG-M2972	Motor-operated valve 2972 plugs	1E-7/hr	3	720 hr	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
ESW-MOV-RE-M2972	Failure to restore MOV 2972 after maintenance	-	-	-	3.0E-4/d	10	7.98E-4	single component event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-CCF-PF-MDPS	Common cause pump failure	3.0E-3/d x .026	10	-	-	-	7.8E-5	single event	ASEP generic, revision 1 [55] Basic event converted to: ESW-MDP-FS-CCF *BETA-2SWPS
ESW-CKV-CB-C515A	Check valve 515A fails	3.0E-3/d	3	-	-	3	3.0E-3	single component event	ASEP generic, revision 1 [55]; Event represents back leakage with testing done every 3 months
ESW-CKV-CB-C515B	Check valve 515B fails	-	-	-	-	-	3.0E-3	same as ESW-CKV-CB-C515A	--
ESW-CKV-HW-C515A	Check Valve 515A fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
ESW-CKV-HW-C515B	Check valve 515B fails to open	-	-	-	-	-	1.0E-4	same as ESW-CKV-HW-C515A	--
ESW-CKV-HW-CV506	Check valve 506 fails to open	-	-	-	-	-	1.0E-4	same as ESW-CKV-HW-C515A	--
ESW-CKV-HW-CV513	Check valve 513 fails to open	-	-	-	-	-	1.0E-4	same as ESW-CKV-HW-C515A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-PTF-RE-DGA	Failure to restore DGN A cooling components after maintenance	-	-	-	1.0E-3	12	3.13E-3	single event	HRA
ESW-PTF-RE-DGB	Failure to restore DGN B cooling components after maintenance	-	-	-	-	-	3.13E-3	same as ESW-PTF-RE-DGA	--
ESW-PTF-RE-DGC	Failure to restore DGN C cooling components after maintenance	-	-	-	-	-	3.13E-3	same as ESW-PTF-RE-DGA	--
ESW-PTF-RE-DGD	Failure to restore DGN D cooling components after maintenance	-	-	-	-	-	3.13E-3	same as ESW-PTF-RE-DGA	--
ESW-PTF-RE-ECW	Failure to restore ECW pump train after maintenance	-	-	-	7.0E-4	12	2.19E-3	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-PTF-RE-HX1	Failure to restore HX1 train after maintenance	-	-	-	7.0E-4	12	2.19E-3	single event	HRA
ESW-PTF-RE-HXQ* where Q* = 2,3,4,6,7,9,10, 12,13,15,16,18, 19,21,22,24,25, 27,28	Failure to restore HXQ* train after maintenance	-	-	-	-	-	2.19E-3	same as ESW-PTF-RE-HX1	--
ESW-PTF-RE-MDPA	Failure to restore ESW pump A trains after maintenance	-	-	-	7.0E-4	12	2.19E-3	single event	HRA
ESW-PTF-RE-MDPB	Failure to restore ESW pump B trains after maintenance	-	-	-	-	-	2.19E-3	same as ESW-PTF-RE-MDPA	--
ESW-TNK-LL-PS13	Insufficient water in cooling tower basin	-	-	-	-	-	1.0E-5	single event	engineering judgement
ESW-XHE-FO-EHS	Failure of operator to initiate emergency heat sink	-	-	-	-	-	9.0E-1	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-XVM-MA-XV517	Manual valve 517 unavailable due to maintenance	8E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	Original ASEP generic document [2]
ESW-XVM-PG-D504A	Manual Valve 504A plugs	1E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
ESW-XVM-PG-D504B	Manual valve 504B plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D504C	Manual valve 504C plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D504D	Manual valve 504D plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D505A	Manual valve 505A plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D505B	Manual valve 505B plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D505C	Manual valve 505B plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D505D	Manual valve 505D plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D505A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-XVM-PG-D519A	Manual Valve 519A plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D519B	Manual valve 519B plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D519C	Manual valve 519C plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-D519D	Manual valve 519D plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-X507A	Manual Valve 507A plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-X507B	Manual valve 507B plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-D504A	--
ESW-XVM-PG-XV13	Manual valve 13 plugs	1E-7/d	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-XVM-PG-XVY* where Y* = 15,16,17,18,19, 20,21,22,23,24, 25,26,27,28,29, 30,31,32,33,34, 35,36,37,38,39, 40,41,42,43,44, 45,46,47,48,49, 50,502,506,509, 51,510,517,52, 53,54,55,56,57, 58,59,60,61,62, 63,64,65,66,67, 68,A,B,C,D,E,F, G,H,I,J,K,L,M,N, O,P,Q,R,S,T	Manual valve XVY* plugs	-	-	-	-	-	4.0E-5	same as ESW-XVM-PG-XV13	--
ESW-XVM-RE-XV517	Failure to restore XV517 after maintenance	-	-	-	3.0E-4	10	7.98E-4	single component event	HRA
ESW-XVM-RE-XVAB	Failure to restore manual valves after maintenance	-	-	-	9.0E-4	14	3.26E-3	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
ESW-XVM-RE-XVZ* where Z* = CD,EF,GH,IJ,KL, MN,OP,QR,ST	Failure to restore manual valves after maintenance	-	-	-	-	-	3.26E-3	same as ESW-XVM-RE-XVAB	--
EMERGENCY VENTILATION SYSTEM:									
EHV-AOV-CC-AV25	Air-operated valve 25 fails to open	1.0E-3/d	3	-	8.0E-4	3	1.0E-3	single component event	Original ASEP generic document [2]
EHV-AOV-CC-AV27	AOV 27 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-AOV-CC-AV28	AOV 28 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-AOV-CC-AV30	AOV 30 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-AOV-CC-AV31	AOV 31 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-AOV-CC-AV33	AOV 33 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-AOV-CC-AV34	AOV 34 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-AOV-CC-AV36	AOV 36 fails to open	-	-	-	-	-	1.0E-3	same as EHV-AOV-CC-AV25	--
EHV-CCF-LF-AOVS	Common cause failure of DG room AOV'S to open	1.0E-3/d x 0.036	3	-	-	-	3.6E-5	single event	Original ASEP generic document [2]; Basic event converted to: EHV-AOV-CC-CCF *BETA-6AOVS
EHV-FAN-FR-OAV64	EDG Room E1 supply fan OAV64 (Fan 7) fails to run	1.25E-5/hr	3	40	4.0E-4	3	5.0E-4	single event	WASH-1400 [4]
EHV-FAN-FR-OAV91	EDG Room E1 supply fan OAV91 (Fan 8) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--
EHV-FAN-FR-OBV64	EDG Room E2 supply fan OBV64 (Fan 9) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--
EHV-FAN-FR-OBV91	EDG Room E2 supply fan OBV91 (Fan 10) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--
EHV-FAN-FR-OCV64	EDG Room E3 supply fan OCV64 (Fan 11) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-FAN-FR-OCV91	EDG Room E3 supply fan OCV91 (Fan 12) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--
EHV-FAN-FR-ODV64	EDG Room E4 supply fan ODV64 (Fan 13) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV64	--
EHV-FAN-FR-ODV91	EDG Room E4 supply fan ODV91 (Fan 14) fails to run	-	-	-	-	-	5.0E-4	same as EHV-FAN-FR-OAV91	--
EHV-FAN-FS-OAV64	EDG Room E1 supply fan OAV64 (Fan 7) fails to start	3.75E-4/d	3	-	3.0E-4	3	3.75E-4	single event	Original ASEP generic document [2]
EHV-FAN-FS-OAV91	EDG Room E1 supply fan OAV91 (Fan 8) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-OBV64	EDG Room E2 supply fan OBV64 (Fan 9) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-OBV91	EDG Room E2 supply fan OBV91 (Fan 10) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-FAN-FS-OCV64	EDG room E3 supply fan OCV64 (Fan 11) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-OCV91	EDG Room E3 supply Fan OCV91 (Fan 12) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-ODV64	EDG Room E4 supply Fan ODV64 (Fan 13) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-ODV91	EDG Room E4 supply Fan ODV91 (Fan 14) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-PTF-MA-OAV64	Fan train OAV64 out for maintenance	2.0E-3/d	3	-	1.6E-3	3	2.0E-3	single event	Original ASEP generic document [2]
EHV-PTF-MA-OAV91	Fan train OAV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-OBV64	Fan train OBV64 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-OBV91	Fan train OBV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-FAN-FS-OCV64	EDG room E3 supply fan OCV64 (Fan 11) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-OCV91	EDG Room E3 supply Fan OCV91 (Fan 12) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-ODV64	EDG Room E4 supply Fan ODV64 (Fan 13) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-FAN-FS-ODV91	EDG Room E4 supply Fan ODV91 (Fan 14) fails to start	-	-	-	-	-	3.75E-4	same as EHV-FAN-FS-OAV64	--
EHV-PTF-MA-OAV64	Fan train OAV64 out for maintenance	2.0E-3/d	3	-	1.6E-3	3	2.0E-3	single event	Original ASEP generic document [2]
EHV-PTF-MA-OAV91	Fan train OAV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-OBV64	Fan train OBV64 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-OBV91	Fan train OBV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-PTF-MA-OCV64	Fan Train OCV64 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-OCV91	Fan train OCV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-ODV64	Fan Train ODV64 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-MA-ODV91	Fan train ODV91 out for maintenance	-	-	-	-	-	2.0E-3	same as EHV-PTF-MA-OAV64	--
EHV-PTF-RE-OAV64	Failure to restore fan train after maintenance (OAV64)	-	-	-	9.0E-4	10	2.39E-3	single event	HRA
EHV-PTF-RE-OAV91	Failure to restore fan train after maintenance (OAV91)	-	-	-	6.0E-4	10	1.6E-3	single event	HRA
EHV-PTF-RE-OBV64	Failure to restore fan train after maintenance (OBV64)	-	-	-	-	-	2.39E-3	same as EHV-PTF-RE-OAV64	--
EHV-PTF-RE-OBV91	Failure to restore fan train after maintenance (OBV91)	-	-	-	-	-	1.6E-3	same as EHV-PTF-RE-OAV91	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILABILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
EHV-PTF-RE-OCV64	Failure to restore fan train after maintenance (OCV64)	-	-	-	-	-	2.39E-3	same as EHV-PTF-RE-OAV64	--
EHV-PTF-RE-OCV91	Failure to restore fan train after maintenance (OCV91)	-	-	-	-	-	1.6E-3	same as EHV-PTF-RE-OAV91	--
EHV-PTF-RE-ODV64	Failure to restore fan train after maintenance (ODV64)	-	-	-	-	-	2.39E-3	same as EHV-PTF-RE-OAV64	--
EHV-PTF-RE-ODV91	Failure to restore fan train after maintenance (ODV91)	-	-	-	-	-	1.6E-3	same as EHV-PTF-RE-OAV91	--
EHV-SRV-CC-RV1	Relief damper 1 fails to open	3.0E-4/d	10	-	1.0E-4	10	3.0E-4	single component event	Design of damper closely approximates relief valve, IREP [56].
EHV-SRV-CC-RV2	Relief damper 2 fails to open	-	-	-	-	-	3.0E-4	same as EHV-SRV-CC-RV1	--
EHV-SRV-CC-RV3	Relief damper 3 fails to open	-	-	-	-	-	3.0E-4	same as EHV-SRV-CC-RV1	--
EHV-SRV-CC-RV4	Relief damper 4 fails to open	-	-	-	-	-	3.0E-4	same as EHV-SRV-CC-RV1	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILABILITY: MEDIAN	UNAVAILABILITY: EF	UNAVAILABILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HIGH PRESSURE COOLANT INJECTION SYSTEM:									
HCI-ACT-HW-HPCI	Actuation circuitry fails	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	Failure rate from methods developed in Calvert Cliffs IREP report [27]
HCI-ACT-HW-LOCST	Failure of low CST level circuitry for HPCI	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	Failure rate from methods developed in Calvert Cliffs IREP report [27]
HCI-CKV-HW-CV32	Check Valve 32 fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
HCI-CKV-HW-CV61	Check Valve 61 fails to open	-	-	-	-	-	1.0E-4	same as HCI-CKV-HW-CV32	--
HCI-CKV-HW-CV65	Check Valve 65 fails to open	-	-	-	-	-	1.0E-4	same as HCI-CKV-HW-CV32	--
HCI-ICC-HW-FC108	Flow controller fails	1.25E-4/d	3	-	1.0E-4	3	1.25E-4	single component event	WASH-1400 [4]
HCI-MOV-CC-MV14	Motor-Operated Valve 14 fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
HCI-MOV-CC-MV19	Motor-Operated Valve 19 fails to open	-	-	-	-	-	3.0E-3	same as HCI-MOV-CC-MV14	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HCI-MOV-CC-MV57	Motor-Operated Valve 57 fails to open	-	-	-	-	-	3.0E-3	same as HCI-MOV-CC-MV14	--
HCI-MOV-CC-MV58	Motor-Operated Valve 58 fails to open	-	-	-	-	-	3.0E-3	same as HCI-MOV-CC-MV14	--
HCI-MOV-HW-MV15	Hardware failures in PS-11 (HPCI steam supply line)	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	original ASEP generic document [55]
HCI-MOV-HW-MV20	Hardware failure in PS-5 (MV20 injection valve)	-	-	-	-	-	4.0E-5	same as HCI-MOV-HW-MV15	--
HCI-MOV-MA-MV14	Motor-Operated Valve 14 out for maintenance	2.0E-4/d	10	-	7.52E-4	10	2.0E-4	single component event	original ASEP generic document [2]
HCI-MOV-MA-MV17	Motor-Operated Valve 17 out for maintenance	-	-	-	-	-	2.0E-4	same as HCI-MOV-MA-MV14	--
HCI-MOV-MA-MV20	Motor-Operated Valve 20 out for maintenance	-	-	-	-	-	2.0E-4	same as HCI-MOV-MA-MV14	--
HCI-MOV-MA-MV57	Motor-Operated Valve 57 out for maintenance	-	-	-	-	-	2.0E-4	same as HCI-MOV-MA-MV14	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HCI-MOV-MA-PCV50	Pressure Control Valve 50 out for maintenance	-	-	-	-	-	2.0E-4	same as HCI-MOV-MA-MV14	--
HCI-MOV-PG-MV16	Motor-Operated Valve 16 plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
HCI-MOV-PG-MV17	Motor-Operated Valve 17 plugs	-	-	-	-	-	4.0E-5	same as HCI-MOV-PG-MV16	--
HCI-PSF-HW-COL13	Pressure Control Valve 50 plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
HCI-PTF-VF-NOSUC	HPCI suction line failures (low suction sensors)	1.61E-3/d	5	-	-	5	1.61E-3	single component event	failure rate is from methods developed in Calvert Cliffs IREP [54]
HCI-TCV-HW-TCV18	Test Check Valve 18 fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
HCI-TDP-FO-20S37	Turbine-driven pump fails to run for 1 hour	5.0E-3/hr	10	1	-	-	5.0E-3	S1 and some transient sequences	plant data
HCI-TDP-FR-20S37	Turbine-driven pump fails to run	5.0E-3/hr	10	10	-	10	5.0E-2	S2 and most transient sequences--10 hrs	plant data

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HCI-TDP-FS-20S37	Turbine-driven pump fails to start	3.0E-2/d	10	-	1.13E-2	10	3.0E-2	single component event	ASEP generic, revision 1 [55]
HCI-TDP-MA-20S37	Turbine-driven pump out for maintenance	1.0E-2/d	10	-	3.8E-3	10	1.0E-2	single component event	ASEP generic, revision 1 [55]
HCI-XVM-HW-CST01	Manual Valve 1 plugs	-	-	-	-	-	4.0E-5	same as HCI-PSF-HW-COL13	--
HCI-XVM-PG-XV12	Manual Valve 12 plugs	-	-	-	-	-	4.0E-5	same as HCI-PSF-HW-COL13	--
HCI-XVM-PG-XV23	Manual Valve 23 plugs	-	-	-	-	-	4.0E-5	same as HCI-PSF-HW-COL13	--
HIGH PRESSURE SERVICE WATER:									
HSW-CCF-LF-MDPS	Common cause failure of HSW pumps to start	3.0E-3/d x 0.0096	10	-	-	-	2.88E-5	single event	ASEP generic, revision 1 [55] Basic event converted to HSW-MDP-FS-CCF *BETA-4SWPS
HSW-CCF-LF-MOVS	Common cause failure of RHR HX valves to open	3.0E-3/d x 0.032	10	-	-	-	9.6E-5	single event	ASEP generic, revision 1 [55] Basic event converted to HSW-MDP-FS-CCF *BETA-4MOVS

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-CKV-HW-C502A	Check Valve 502A fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, REVISION 1 [55]
HSW-CKV-HW-C502B	Check Valve 502B fails to open	-	-	-	-	-	1.0E-4	same as HSW-CKV-HW-C502A	--
HSW-CKV-HW-C502C	Check Valve 502C fails to open	-	-	-	-	-	1.0E-4	same as HSW-CKV-HW-C502A	--
HSW-CKV-HW-C502D	Check valve 502D fails to open	-	-	-	-	-	1.0E-4	same as HSW-CKV-HW-C502A	--
HSW-CKV-HW-CV5	Check Valve 5 fails to open	-	-	-	-	-	1.0E-4	same as HSW-CKV-HW-C502A	--
HSW-FAN-FR-ECTFA	Fan A fails to run	6.65E-6/hr	10	40	1.0E-4	10	2.66E-4	single component event	IEEE-500 [53]
HSW-FAN-FR-ECTFB	Fan B fails to run	-	-	-	-	-	2.66E-4	same as HSW-FAN-FR-ECTFA	--
HSW-FAN-FR-ECTFC	Fan C fails to run	-	-	-	-	-	2.66E-4	same as HSW-FAN-FR-ECTFA	--
HSW-FAN-FS-ECTFA	Emergency Cooling Tower Fan A fails to start	3.5E-3/d	3	-	2.8E-3	3	3.5E-3	single component event	IEEE-500 [53]
HSW-FAN-FS-ECTFB	Emergency Cooling Tower Fan B fails to start	-	-	-	-	-	3.5E-3	same as HSW-FAN-FS-ECTFA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-FAN-FS-ECTFC	Emergency Cooling Tower Fan C fails to start	-	-	-	-	-	3.5E-3	same as HSW-FAN-FS-ECTFA	--
HSW-FAN-MA-ECTFA	Fan A out for maintenance	1.86E-3/d	10	-	7.0E-4	10	1.86E-3	single component event	WASH-1400 [4]
HSW-FAN-MA-ECTFB	Fan B out for maintenance	-	-	-	-	-	1.86E-3	same as HSW-FAN-MA-ECTFA	--
HSW-FAN-MA-ECTFC	Fan C out for maintenance	-	-	-	-	-	1.86E-3	same as HSW-FAN-MA-ECTFA	--
HSW-HTX-PG-HXA	RHR Heat Exchanger fails by blockage	5.7E-6/hr	10	40	-	10	2.28E-4	single component event	ASEP generic, revision 1 [55]
HSW-HTX-PG-HXB	RHR Heat Exchanger B fails by blockage	-	-	-	-	-	2.28E-4	same as HSW-HTX-PG-HXA	--
HSW-HTX-PG-HXC	RHR Heat exchanger C fails by blockage	-	-	-	-	-	2.28E-4	same as HSW-HTX-PG-HXA	--
HSW-HTX-PG-HXD	RHR heat exchanger D fails by blockage	-	-	-	-	-	2.28E-4	same as HSW-HTX-PG-HXA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-HTX-RP-HXA	RHR Heat Exchanger A fails by rupture	3.0E-6/hr	10	40	-	10	1.2E-4	single component event	ASEP generic, revision 1 [55]
HSW-HTX-RP-HXB	RHR Heat Exchanger B fails by rupture	-	-	-	-	-	1.2E-4	same as HSW-HTX-RP-HXA	--
HSW-HTX-RP-HXC	RHR heat exchanger C fails by rupture	-	-	-	-	-	1.2E-4	same as HSW-HTX-RP-HXA	--
HSW-HTX-RP-HXD	RHR heat exchanger D fails by rupture	-	-	-	-	-	1.2E-4	same as HSW-HTX-RP-HXA	--
HSW-MDP-FR-MDPA	HPSW Pump A fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
HSW-MDP-FR-MDPB	HPSW Pump B fails to run	-	-	-	-	-	1.2E-3	same as HSW-MDP-FR-MDPA	--
HSW-MDP-FR-MDPC	HPSW Pump C fails to run	-	-	-	-	-	1.2E-3	same as HSW-MDP-FR-MDPA	--
HSW-MDP-FR-MDPD	HPSW Pump D fails to run	-	-	-	-	-	1.2E-3	same as HSW-MDP-FR-MDPA	--
HSW-MDP-FS-MDPA	HPSW Pump A fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-MDP-FS-MDPB	HPSW Pump B fails to start	-	-	-	-	-	3.0E-3	same as HSW-MDP-FS-MDPA	--
HSW-MDP-FS-MDPC	HPSW Pump C fails to start	-	-	-	-	-	3.0E-3	same as HSW-MDP-FS-MDPA	--
HSW-MDP-FS-MDPD	HPSW Pump D fails to start	-	-	-	-	-	3.0E-3	same as HSW-MDP-FS-MDPA	--
HSW-MDP-MA-MDPA	HPSW Pump A out for maintenance	2.0E-3/d	10	-	7.5E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
HSW-MDP-MA-MDPB	HPSW Pump B out for maintenance	-	-	-	-	-	2.0E-3	same as HSW-MDP-MA-MDPA	--
HSW-MDP-MA-MDPC	HPSW Pump C out for maintenance	-	-	-	-	-	2.0E-3	same as HSW-MDP-MA-MDPA	--
HSW-MDP-MA-MDPD	HPSW Pump D out for maintenance	-	-	-	-	-	2.0E-3	same as HSW-MDP-MA-MDPA	--
HSW-MOV-CC-2344	Motor-Operated Valve 2344 fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
HSW-MOV-CC-2804A	Motor-Operated Valve 2804A fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-MOV-CC-2804B	Motor-Operated Valve 2804B fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-M2803	MOV 2803 fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-M502C	Motor-operated Valve 502C fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-MV174	Motor-Operated Valve 174 fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-MV176	Motor-Operated Valve 176 fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-MV89A	Motor-Operated Valve 89A fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-MV89B	Motor-operated valve 89B fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-CC-MV89C	Motor-operated valve 89C fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-MOV-CC-MV89D	Motor-operated valve 89D fails to open	-	-	-	-	-	3.0E-3	same as HSW-MOV-CC-2344	--
HSW-MOV-MA-2344	Valve 2344 out for maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
HSW-MOV-MA-2804A	MOV 2804A out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-2804B	MOV 2804B out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-M2486	MOV 2486 out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-M2803	MOV 2803 out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-M502A	MOV 502A out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-M502B	MOV 502B out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-M502C	MOV 502C out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-MOV-MA-MV174	MOV 174 out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-MV176	Motor-Operated Valve 176 out for maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
HSW-MOV-MA-MV89A	MOV 89A out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-MV89B	MOV 89B out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-MV89C	MOV 89C out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-MA-MV89D	MOV 89D out for maintenance	-	-	-	-	-	2.0E-4	same as HSW-MOV-MA-2344	--
HSW-MOV-PG-M2486	Motor-Operated Valve 2486 plugs	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
HSW-MOV-PG-M502A	Motor-operated Valve 502A plugs	-	-	-	-	-	4.0E-5	same as HSW-MOV-PG-M2486	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILIBILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-MOV-PG-M502B	Motor-operated valve 502B plugs	-	-	-	-	-	4.0E-5	same as HSW-MOV-PG-M2486	--
HSW-MOV-RE-2344	Failure to restore valve 2344 after maintenance	-	-	-	3.0E-4	10	7.98E-4	single component event	HRA
HSW-MOV-RE-M2803	Failure to restore valve 2803 after maintenance	-	-	-	-	-	7.98E-4	same as HSW-MOV-RE-2344	--
HSW-PTF-RE-ECTFA	Failure to restore fan A train after maintenance	-	-	-	1.2E-3	10	3.19E-3	single event	HRA
HSW-PTF-RE-ECTFB	Failure to restore Fan B train after maintenance	-	-	-	-	-	3.19E-3	same as HSW-PTF-RE-ECTFA	--
HSW-PTF-RE-ECTFC	Failure to restore Fan C train after maintenance	-	-	-	-	-	3.19E-3	same as HSW-PTF-RE-ECTFA	--
HSW-PTF-RE-HXA	Failure to restore HXA train after maintenance	-	-	-	1.2E-3	10	3.19E-3	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILIBILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-PTF-RE-HXB	Failure to restore HXB train after maintenance	-	-	-	-	-	3.19E-3	same as HSW-PTF-RE-HXA	--
HSW-PTF-RE-HXC	Failure to restore HXC train after maintenance	-	-	-	-	-	3.19E-3	same as HSW-PTF-RE-HXA	--
HSW-PTF-RE-HXD	Failure to restore HXD train after maintenance	-	-	-	-	-	3.19E-3	same as HSW-PTF-RE-HXA	--
HSW-PTF-RE-MDPA	Failure to restore Pump A train after maintenance	-	-	-	7.0E-4	12	2.19E-3	single event	HRA
HSW-PTF-RE-MDPB	Failure to restore Pump B train after maintenance	-	-	-	-	-	2.19E-3	same as HSW-PTF-RE-MDPA	--
HSW-PTF-RE-MDPC	Failure to restore Pump C train after maintenance	-	-	-	-	-	2.19E-3	same as HSW-PTF-RE-MDPA	--
HSW-PTF-RE-MDPD	Failure to restore Pump D train after maintenance	-	-	-	-	-	2.19E-3	same as HSW-PTF-RE-MDPA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-PTF-RE-PS10	Failure to restore PS-10 after maintenance	-	-	-	6.0E-4	10	1.60E-3	single event	HRA
HSW-PTF-RE-PS18	Failure to restore PS-18 after maintenance	-	-	-	9.0E-4	10	2.39E-3	single event	HRA
HSW-PTF-RE-PS20	Failure to restore PS-20 after maintenance	-	-	-	-	-	2.39E-3	same as HSW-PTF-RE-PS18	--
HSW-TNK-LF-RESVR	Reservoir fails	-	-	-	-	10	1.0E-5	single event	engineering judgment
HSW-VFC-LF-PPBAY	Flow from pump bay fails	-	-	-	-	10	1.0E-5	single event	engineering judgment
HSW-XHE-FO-PS9	Operator fails to open bypass line valve 2344	-	-	-	-	-	.5	single event	HRA/screening value
HSW-XVM-OO-516A	Manual Valve 516A fails to remain closed	1.25E-4/d	3	-	1.0E-4	3	1.25E-4	single component event	WASH-1400 [4]
HSW-XMV-PG-X501A	Manual Valve 501A plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
HSW-XVM-PG-X501B	Manual valve 501B plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-X501C	Manual valve 501C plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-X501D	Manual valve 501D plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-X515B	Manual Valve 10 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV11	Manual Valve 11 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV5	Manual Valve 5 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV6	Manual valve 6 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV7	Manual valve 7 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV8	Manual Valve 8 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--
HSW-XVM-PG-XV9	Manual Valve 9 plugs	-	-	-	-	-	4.0E-5	same as HSW-XVM-PG-X501A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
INSTRUMENT AIR SYSTEM:									
IAS-PTF-HW-IAS	Insufficient Air Pressure from system	1.0E-4/d	10	-	3.8E-5	10	1.0E-4	single event	engineering judgement
LOW PRESSURE COOLANT INJECTION SYSTEM:									
LCI-ACT-HW-DIV1	Actuation circuitry Division 1 fails	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	failure rate from methods developed in Calvert Cliffs IREP [27]
LCI-ACT-HW-DIV2	Actuation circuitry Division 2 fails	-	-	-	-	-	1.61E-3	same as LCI-ACT-HW-DIV1	--
LCI-CCF-LF-MOVS	LCS Pump common cause failure	3.0E-3/d x 0.049	10	-	-	-	1.47E-4	single event	ASEP generic, revision 1 [55] Basic event converted to LCI-MOV-CC-CCF *BETA-2MOVS
LCI-CKV-HW-CV19A	Check Valve 19A fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
LCI-CKV-HW-CV19B	Check Valve 19B fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV19C	Check valve 19C fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-CKV-HW-CV19D	Check valve 19D fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV46A	Check Valve 46A fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV46B	Check Valve 46B fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV48A	Check Valve 48A fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV48B	Check Valve 48B fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV48C	Check Valve 48C fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-CKV-HW-CV48D	Check Valve 48D fails to open	-	-	-	-	-	1.0E-4	same as LCI-CKV-HW-CV19A	--
LCI-MDP-FR-2AP35	Motor-Driven Pump 2AP35 fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
LCI-MDP-FR-2BP35	Motor-Driven Pump 2BP35 fails to run	-	-	-	-	-	1.2E-3	same as LCI-MDP-FR-2AP35	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-MDP-FR-2CP35	Motor-Driven Pump 2CP35 fails to run	-	-	-	-	-	1.2E-3	same as LCI-MDP-FR-2AP35	--
LCI-MDP-FR-2DP35	Motor-Driven Pump 2DP35 fails to run	-	-	-	-	-	1.2E-3	same as LCI-MDP-FR-2AP35	--
LCI-MDP-FS-2AP35	Motor-Driven Pump 2AP35 fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
LCI-MDP-FS-2BP35	Motor-Driven Pump 2BP35 fails to start	-	-	-	-	-	3.0E-3	same as LCI-MDP-FS-2AP35	--
LCI-MDP-FS-2CP35	Motor-Driven Pump 2CP35 fails to start	-	-	-	-	-	3.0E-3	same as LCI-MDP-FS-2AP35	--
LCI-MDP-FS-2DP35	Motor-Driven Pump 2DP35 fails to start	-	-	-	-	-	3.0E-3	same as LCI-MDP-FS-2AP35	--
LCI-MDP-MA-2AP35	Motor-Driven Pump 2AP35 out for maintenance	2.0E-3/d	10	-	7.5E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
LCI-MDP-MA-2BP35	Motor-Driven Pump 2BP35 out for maintenance	-	-	-	-	-	2.0E-3	same as LCI-MDP-MA-2AP35	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		Failure Rate			Unavailibility				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-MDP-MA-2CP35	Motor-Driven Pump 2CP35 out for maintenance	-	-	-	-	-	2.0E-3	same as LCI-MDP-MA-2AP35	--
LCI-MDP-MA-2DP35	Motor-Driven Pump 2DP35 out for maintenance	-	-	-	-	-	2.0E-3	same as LCI-MDP-MA-2AP35	--
LCI-MOV-CC-MV25A	Motor-Operated Valve 25A fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
LCI-MOV-CC-MV25B	Motor-Operated Valve 25B fails to open	-	-	-	-	-	3.0E-3	same as LCI-MOV-CC-MV25A	--
LCI-MOV-HW-MV13A	Motor-Operated Valve 13A plugs	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
LCI-MOV-HW-MV13B	Motor-Operated Valve 13B plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-HW-MV13A	--
LCI-MOV-HW-MV13C	Motor-Operated Valve 13C plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-HW-MV13A	--
LCI-MOV-HW-MV13D	Motor-Operated Valve 13D plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-HW-MV13A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-MOV-MA-154A	Motor-Operated Valve 154A out for maintenance	2.0E-4/d	10	-	7.54E-4	10	2.0E-4	single component event	original ASEP generic document [2]
LCI-MOV-MA-154B	Motor-Operated Valve 154B out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-MA-2677A	Motor-Operated Valve 2677A out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-MA-2677D	Motor-Operated Valve 2677D out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-MA-MV16A	Motor-Operated Valve 16A out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-MA-MV16B	Motor-Operated Valve 16B out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-MA-MV16C	Motor-Operated Valve 16C out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-MOV-MA-MV16D	Motor-Operated Valve 16D out for maintenance	-	-	-	-	-	2.0E-4	same as LCI-MOV-MA-154A	--
LCI-MOV-PG-154A	Motor-Operated Valve 154A plugs	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
LCI-MOV-PG-154B	Motor-Operated Valve 154B plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-2677A	MOV 2677A plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-2677D	Valve 2677D plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-MV16A	Motor-Operated Valve 16A plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-MV16B	Motor-Operated valve 16B plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-MV16C	Motor-Operated Valve 16B plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--
LCI-MOV-PG-MV16D	Motor-Operated valve 16D plugs	-	-	-	-	-	4.0E-5	same as LCI-MOV-PG-154A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE RATE (PER DEMAND OR HOUR)	FAILURE RATE EF	MISSION TIME ~(HRS)	UNAVAILIBILITY MEDIAN	UNAVAILIBILITY EF	UNAVAILIBILITY MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-MOV-RE-154A	Failure to restore MOV 154A after maintenance	-	-	-	1.2E-3	10	3.19E-3	single event	HRA
LCI-MOV-RE-154B	Failure to restore MOV 154B after maintenance	-	-	-	-	-	3.19E-3	same as LCI-MOV-RE-154A	--
LCI-PTF-RE-2AP35	Failure to restore pump A train after maintenance	-	-	-	1.5E-3	12	4.70E-3	single event	HRA
LCI-PTF-RE-2BP35	Failure to restore pump B train after maintenance	-	-	-	-	-	4.70E-3	same as LCI-PTF-RE-2AP35	--
LCI-PTF-RE-2CP35	Failure to restore pump C train after maintenance	-	-	-	-	-	4.70E-3	same as LCI-PTF-RE-2AP35	--
LCI-PTF-RE-2DP35	Failure to restore pump D train after maintenance	-	-	-	-	-	4.70E-3	same as LCI-PTF-RE-2AP35	--
LCI-PTF-RE-LOOPA	Failure to restore loop A valves after maintenance	-	-	-	1.2E-3	10	3.19E-3	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCI-PTF-RE-LOOPB	Failure to restore loop B valves after maintenance	-	-	-	-	-	3.19E-3	same as LCI-PTF-RE-LOOPA	--
LCI-TSW-FT-ATOC	Transfer ABT fails	1.25E-3/d	3	-	1.0E-3	3	1.25E-3	single component event	WASH-1400 [4]
LCI-TSW-FT-BTOD	Transfer ABT fails	-	-	-	-	-	1.25E-3	same as LCI-TSW-FT-ATOC	--
LCI-XVM-PG-XV81A	Manual Valve 81A plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
LCI-XVM-PG-XV81B	Manual Valve 81B plugs	-	-	-	-	-	4.0E-5	same as LCI-XVM-PG-XV81A	--
LOW PRESSURE CORE SPRAY SYSTEM:									
LCS-ACT-HW-LOOPA	Loop A actuation fails	1.61-3/d	5	-	1.0E-3	5	1.61E-3	single component event	failure rate from methods developed in Calvert Cliffs IREP [27]
LCS-ACT-HW-LOOPB	Loop B actuation fails	-	-	-	-	-	1.61E-3	same as LCS-ACT-HW-LOOPA	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-CCF-LF-MOVS	Common cause failure of injection valves to open	3.0E-3/d x 0.049	10	-	-	-	1.47E-4	single event	ASEP generic, revision 1 [55] Basic event converted to LCS-MOV-CC-CCF *BETA-2MOVS
LCS-CCF-PF-MDPS	Common cause failure of three pumps to start	3.0E-3/d x 0.11	10	-	-	-	3.3E-4	single event	ASEP generic, revision 1 [55] Basic event converted to LCS-MDP-FS-CCF *BETA-3RHRMDPS
LCS-CKV-HW-CV10A	Check Valve 10A fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
LCS-CKV-HW-CV10B	Check Valve 10B fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-CKV-HW-CV10C	Check Valve 10C fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-CKV-HW-CV10D	Check Valve 10D fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-CKV-HW-CV66A	Check valve 66A fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-CKV-HW-CV66B	Check Valve 66B fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-CKV-HW-CV66C	Check Valve 66C fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-CKV-HW-CV66D	Check Valve 66D fails to open	-	-	-	-	-	1.0E-4	same as LCS-CKV-HW-CV10A	--
LCS-MDP-FR-2AP37	Pump 2AP37 fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.20E-3	single component event	ASEP generic, revision 1 [55]
LCS-MDP-FR-2BP37	Pump 2BP37 fails to run	-	-	-	-	-	1.20E-3	same as LCS-MDP-FR-2AP37	--
LCS-MDP-FR-2CP37	Pump 2CP37 fails to run	-	-	-	-	-	1.20E-3	same as LCS-MDP-FR-2AP37	--
LCS-MDP-FR-2DP37	Pump 2DP37 fails to run	-	-	-	-	-	1.20E-3	same as LCS-MDP-FR-2AP37	--
LCS-MDP-FS-2AP37	Pump 2AP37 fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
LCS-MDP-FS-2BP37	Pump 2BP37 fails to start	-	-	-	-	-	3.0E-3	same as LCS-MDP-FS-2AP37	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-MDP-FS-2CP37	Pump 2CP37 fails to start	-	-	-	-	-	3.0E-3	same as LCS-MDP-FS-2AP37	--
LCS-MDP-FS-2DP37	Pump 2DP37 fails to start	-	-	-	-	-	3.0E-3	same as LCS-MDP-FS-2AP37	--
LCS-MDP-MA-2AP37	Pump 2AP37 out for maintenance	2.0E-3/d	10	-	7.5E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
LCS-MDP-MA-2BP37	Pump 2BP37 out for maintenance	-	-	-	-	-	2.0E-3	same as LCS-MDP-MA-2AP37	--
LCS-MDP-MA-2CP37	Pump 2CP37 out for maintenance	-	-	-	-	-	2.0E-3	same as LCS-MDP-MA-2AP37	--
LCS-MDP-MA-2DP37	Pump 2DP37 out for maintenance	-	-	-	-	-	2.0E-3	same as LCS-MDP-MA-2AP37	--
LCS-MOV-CC-MV12A	Motor-Operated valve 12A fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
LCS-MOV-CC-MV12B	Motor-Operated valve 12B fails to open	-	-	-	-	-	3.0E-3	same as LCS-MOV-CC-MV12A	--
LCS-MOV-CO-MV26A	Motor-Operated Valve 26A fails to remain closed	5.0E-7/hr	10	360	6.77E-5	10	1.8E-4	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-MOV-CO-MV26B	Motor-Operated Valve 26B fails to remain closed	-	-	-	-	-	1.8E-4	same as LCS-MOV-CO-MV26A	--
LCS-MOV-HW-MV7A	Motor-Operated Valve 7A fails to remain open	1.25E-7/hr	3	360	3.6E-5	3	4.5E-5	single component event	WASH-1400 [4]
LCS-MOV-HW-MV7B	Motor-Operated Valve 7B fails to remain open	-	-	-	-	-	4.5E-5	same as LCS-MOV-HW-MV7A	--
LCS-MOV-HW-MV7C	Motor-Operated Valve 7C fails to remain open	-	-	-	-	-	4.5E-5	same as LCS-MOV-HW-MV7A	--
LCS-MOV-HW-MV7D	Motor-Operated Valve 7D fails to remain open	-	-	-	-	-	4.5E-5	same as LCS-MOV-HW-MV7A	--
LCS-MOV-MA-MV11A	Motor-Operated Valve 11A out for maintenance	2.0E-4/d	10	-	7.5E-4	10	2.0E-4	single component event	original ASEP generic document [2]
LCS-MOV-MA-MV11B	Motor-Operated Valve 11B out for maintenance	-	-	-	-	-	2.0E-4	same as LCS-MOV-MA-MV11A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-MOV-MA-MV5A	Motor-Operated Valve 5A out for maintenance	-	-	-	-	-	2.0E-4	same as LCS-MOV-MA-MV11A	--
LCS-MOV-MA-MV5B	Motor-Operated Valve 5B out for maintenance	-	-	-	-	-	2.0E-4	same as LCS-MOV-MA-MV11A	--
LCS-MOV-MA-MV5C	Motor-Operated Valve 5C out for maintenance	-	-	-	-	-	2.0E-4	same as LCS-MOV-MA-MV11A	--
LCS-MOV-MA-MV5D	Motor-Operated Valve 5D out for maintenance	-	-	-	-	-	2.0E-4	same as LCS-MOV-MA-MV11A	--
LCS-MOV-PG-MV11A	Motor-Operated Valve 11A plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
LCS-MOV-PG-MV11B	Motor-Operated valve 11B plugs	-	-	-	-	-	4.0E-5	same as LCS-MOV-PG-MV11A	--
LCS-MOV-PG-MV5A	Motor-Operated Valve 5A plugs	-	-	-	-	-	4.0E-5	same as LCS-MOV-PG-MV11A	--
LCS-MOV-PG-MV5B	Motor-Operated Valve 5B plugs	-	-	-	-	-	4.0E-5	same as LCS-MOV-PG-MV11A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-MOV-PG-MV5C	Motor-Operated Valve 5C plugs	-	-	-	-	-	4.0E-5	same as LCS-MOV-PG-MV11A	--
LCS-MOV-PG-MV5D	Motor-Operated Valve 5D plugs	-	-	-	-	-	4.0E-5	same as LCS-MOV-PG-MV11A	--
LCS-MOV-RE-MV11A	Failure to restore MOV 11A after maintenance	-	-	-	1.5E-3	10	3.99E-3	single event	HRA
LCS-MOV-RE-MV11B	Failure to restore MOV 11B after maintenance	-	-	-	-	-	3.99E-3	same as LCS-MOV-RE-MV11A	--
LCS-PTF-RE-2AP37	Failure to restore pump 2AP37 train after maintenance	-	-	-	1.5E-3	12	4.7E-3	single event	HRA
LCS-PTF-RE-2BP37	Failure to restore Pump 2BP37 train after maintenance	-	-	-	-	-	4.7E-3	same as LCS-PTF-RE-2AP37	--
LCS-PTF-RE-2CP37	Failure to restore pump 2CP37 train after maintenance	-	-	-	-	-	4.7E-3	same as LCS-PTF-RE-2AP37	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
LCS-PTF-RE-2DP37	Failure to restore Pump 2DP37 train after maintenance	-	-	-	-	-	4.7E-3	same as LCS-PTF-RE-2AP37	--
LCS-TCV-HW-TV13A	Test check valve 13A fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
LCS-TCV-HW-TV13B	Check valve 13B fails to open	-	-	-	-	-	1.0E-4	same as LCS-TCV-HW-TV13A	--
LCS-XVM-PG-XV14A	Manual Valve 14A plugs	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
LCS-XVM-PG-XV14B	Manual valve 14B plugs	-	-	-	-	-	4.0E-5	same as LCS-XVM-PG-XV14A	--
LCS-XVM-PG-XV63A	Manual valve 63A plugs	-	-	-	-	-	4.0E-5	same as LCS-XVM-PG-XV14A	--
LCS-XVM-PG-XV63B	Manual valve 63B plugs	-	-	-	-	-	4.0E-5	same as LCS-XVM-PG-XV14A	--
LCS-XVM-PG-XV63C	Manual Valve 63C plugs	-	-	-	-	-	4.0E-5	same as LCS-XVM-PG-XV14A	--
LCS-XVM-PG-XV63D	Manual Valve 63D plugs	-	-	-	-	-	4.0E-5	same as LCS-XVM-PG-XV14A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
MISCELLANEOUS:									
CST-PSF-CSTLOST	CST inventory is lost	-	-	-	-	-	1.0E-5	single event	engineering judgement
CST-PSF-DEPLETED	CST depleted	-	-	-	-	-	1.0 or 0.0	house event, flag, for most sequences 0.0	--
LOSP	Subsequent loss of offsite power following a plant trip	2.0E-4/d	3	-	1.2E-4	3	2.0E-4	--	ASEP generic, revision 1 [55]
LOSP1	Artifical value for LOSP to allow SETS and TEMAC runs	-	-	-	-	3	5.0E-4	--	--
NSW-SYS-FO-NSW	Normal service water fails to operate given PCS not failed	1.0E-3/d	-	-	-	10	1.0E-3	single event	ASEP generic, revision 1 [55], engineering judgement
NSW-SYS-FO-NSW-1	Normal service water fails to operate given PCS failed or isolated	1.0E-1/d	10	-	-	10	1.0E-1	single event	ASEP generic, revision 1 [55], engineering judgement

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
P1	Failure of one safety relief valve to close (stuck open)	-	-	-	-	-	9.6E-2	single event	ASEP generic, revision 1 [55] & plant specific
P2	Failure of two safety relief valves to close	-	-	-	-	10	2.0E-3	single event	based on NRC LER Data Summary [58] and 2 events in BWR reactor experience
P3	Failure of three safety relief valves to close	-	-	-	-	10	2.0E-4	single event	based on NRC LER Data Summary [58] and no events in BWR reactor experience
Q	Failure of the Power Conversion System for a T3A initiating event	-	-	-	-	3	1.0E-2	single event	WASH-1400 [4]
Q1	Failure of the Power Conversion System for a T3C and S2 initiating event	-	-	-	-	-	5.0E-1	single event	engineering judgement considering plant specific design

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RPSM or RPSM1	Mechanical failure of the Reactor Protection System	-	-	-	-	10	1.0E-5	single event	based on mech. contribution from NUREG-0460 [35]; RPSM1 is a dummy necessary for SETS runs
RPSE	Electrical failure of the Reactor Protection System	-	-	-	-	-	2.0E-5	single event	based on elec. contribution from NUREG-0460 [35]
ARI	Alternate Rod Insertion system failure	-	-	-	-	-	5E-3	single event	Estimate using IDCOR input
SCRM	Manual scram failure	-	-	-	-	-	2E-3	single event	Estimate based on ASEP generic, revision 1 [55]; value for HRA and instrumentation circuit actuation
RPT	Recirculation pump trip failure	-	-	-	-	-	<1E-4	single event	engineering judgement

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SF-ADS	Automatic ADS failed but manual ADS available	-	-	-	-	-	2.0E-1	single event	HRA - split fraction added to T-SLC sequence combinations
PRIMARY CONTAINMENT VENTING:									
PCV-SYS-HW-SYSTM	Local Equipment faults	1.0E-4/d	10	-	3.76E-5	5	1.0E-4	single event	ASEP generic, revision 1 [55] for AOVS & engineering judgement
PCV-XHE-FO-PCV	Operator fails to vent	-	-	-	-	-	.5	single event	HRA/screening value
REACTOR BUILDING COOLING WATER SYSTEM:									
RBC-AOV-FT-A2352	Hardware failure of AOV 2352	1.0E-3/d	3	-	8.0E-4	3	1.0E-3	single component event	ASEP generic, revision 1 [55]
RBC-AOV-FT-A2354	Hardware failure of AOV 2354	-	-	-	-	-	1.0E-3	same as RBC-AOV-FT-A2352	--
RBC-AOV-FT-A8154	Hardware failure of AOV 8154	-	-	-	-	-	1.0E-3	same as RBC-AOV-FT-A2352	--
RBC-AOV-FT-A8156	Hardware failure of AOV 8156	-	-	-	-	-	1.0E-3	same as RBC-AOV-FT-A2352	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RBC-AOV-MA-A2352	AOV 2352 unavailable due to maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
RBC-AOV-MA-A2354	AOV 2354 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-MA-A8154	AOV 8154 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-MA-A8156	AOV 8156 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-OO-2253	Failure of non-emergency load isolation valve to close	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	IREP [56]
RBC-MDP-FR-PA	RBCW Pump A fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
RBC-MDP-FR-PB	RBCW Pump B fails to run	-	-	-	-	-	1.2E-3	same as RBC-MDP-FR-PA	--
RBC-MDP-FS-PA	Pump A fails to start due to hardware failure	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RBC-AOV-MA-A2352	AOV 2352 unavailable due to maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
RBC-AOV-MA-A2354	AOV 2354 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-MA-A8154	AOV 8154 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-MA-A8156	AOV 8156 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-AOV-MA-A2352	--
RBC-AOV-OO-2253	Failure of non-emergency load isolation valve to close	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	IREP [56]
RBC-MDP-FR-PA	RBCW Pump A fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
RBC-MDP-FR-PB	RBCW Pump B fails to run	-	-	-	-	-	1.2E-3	same as RBC-MDP-FR-PA	--
RBC-MDP-FS-PA	Pump A fails to start due to hardware failure	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RBC-MDP-FS-PB	Pump B fails to start due to hardware failure	-	-	-	-	-	3.0E-3	same as RBC-MDP-FS-PA	--
RBC-PTF-MA-PB	RBCW Pump B train out for maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
RBC-PTF-RE-2352	Failure to restore 2352 train of valves after maintenance	-	-	-	9.0E-4	10	2.39E-3	single event	HRA
RBC-PTF-RE-2354	Failure to restore 2354 train of valves after maintainence	-	-	-	-	-	2.39E-3	same as RBC-PTF-RE-2352	--
RBC-PTF-RE-PB	Failure to restore Pump B train after maintenance	-	-	-	1.2E-3	10	3.19E-3	single component event	HRA
RBC-SOV-FT-S2352	Hardware failure of SOV 2352	1.0E-3/d	3	-	8.0E-4	3	1.0E-3	single component event	original ASEP generic document [2]
RBC-SOV-FT-S2354	Hardware failure of SOV 2354	-	-	-	-	-	1.0E-3	same as RBC-SOV-FT-S2352	--
RBC-SOV-MA-S2352	SOV 2352 unavailable due to maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RBC-SOV-MA-S2354	SOV 2354 unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as RBC-SOV-MA-S2352	--
RBC-XHE-FO-LCVAL	Operator failure to open locked closed valves	-	-	-	-	-	.5	single event	HRA/screening value
RBC-XHE-FO-SWCH	Operator fails to switch to RBCWS following LOSP	-	-	-	-	-	.5	single event, 30 minutes	generic experience, HRA/screening value
REACTOR CORE ISOLATION COOLING SYSTEM:									
RCI-ACT-HW-LOCST	Failure of low CST circuitry for RCIC	1.0E-3/d	10	-	3.8E-4	10	1.0E-3	single component event	plant data
RCI-ACT-HW-RCIC	Actuation fails	1.61E-3/d	5	-	1.0E-3	5	1.61E-3	single component event	WASH-1400 [4], plant data
RCI-AOV-MA-PCV23	Pressure Control Valve 23 out for mainenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
RCI-AOV-VF-PCV23	Pressure Control Valve 23 fails	3.0E-4/d	3	-	1.13E-4	3	3.0E-4	single component event	ASEP generic, revision 1 [55] for MOVs and engineering judgement

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RCI-CKV-HW-CV19	Check Valve 19 fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
RCI-CKV-HW-CV40	Check Valve 40 fails to open	-	-	-	-	-	1.0E-4	same as RCI-CKV-HW-CV19	--
RCI-CKV-HW-CV50	Check Valve 50 fails to open	-	-	-	-	-	1.0E-4	same as RCI-CKV-HW-CV19	--
RCI-ICC-HW-FIC91	Flow controller fails	1.25E-4/d	3	-	1.0E-4	3	1.25E-4	single component event	WASH-1400 [4]
RCI-MOV-CC-MV131	Motor-Operated Valve 131 fails to open	3.0E-3/d	10	-	1.13E-3	3	3.0E-3	single component event	ASEP generic, revision 1 [55]
RCI-MOV-CC-MV132	Motor-Operated Valve 132 fails to open	-	-	-	-	-	3.0E-3	same as RCI-MOV-CC-MV131	--
RCI-MOV-CC-MV21	Motor-Operated Valve 21 fails to open	-	-	-	-	-	3.0E-3	same as RCI-MOV-CC-MV131	--
RCI-MOV-CC-MV39	Motor-Operated Valve 39 fails to open	-	-	-	-	-	3.0E-3	same as RCI-MOV-CC-MV131	--
RCI-MOV-CC-MV41	Motor-Operated Valve 41 fails to open	-	-	-	-	-	3.0E-3	same as RCI-MOV-CC-MV131	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RCI-MOV-HW-MV20	Motor-Operated Valve 20 fails to remain open	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
RCI-MOV-MA-MV131	Motor-Operated Valve 131 out for maintenance	2.0E-4/d	10	-	7.5E-5	10	2.0E-4	single component event	original ASEP generic document [2]
RCI-MOV-MA-MV132	Motor-Operated Valve 132 out for maintenance	-	-	-	-	-	2.0E-4	same as RCI-MOV-MA-MV131	--
RCI-MOV-MA-MV18	Motor-Operated Valve 18 out for maintenance	-	-	-	-	-	2.0E-4	same as RCI-MOV-MA-MV131	--
RCI-MOV-MA-MV20	Motor-Operated Valve 20 out for maintenance	-	-	-	-	-	2.0E-4	same as RCI-MOV-MA-MV131	--
RCI-MOV-MA-MV39	Motor-Operated Valve 39 out for maintenance	-	-	-	-	-	2.0E-4	same as RCI-MOV-MA-MV131	--
RCI-MOV-PG-MV15	Motor-Operated Valve 15 plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
RCI-MOV-PG-MV16	MOV 16 plugs	-	-	-	-	-	4.0E-5	same as RCI-MOV-PG-MV15	--
RCI-MOV-PG-MV18	MOV 18 plugs	-	-	-	-	-	4.0E-5	same as RCI-MOV-PG-MV15	--

Table 4.9-1
Peach Bottom Event Data (Continued)

		FAILURE RATE			UNAVAILIBILITY				
BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	RATE (PER DEMAND OR HOUR)	EF	MISSION TIME ~(HRS)	MEDIAN	EF	MEAN	DESCRIPTION	SOURCE/ COMMENTS
RCI-TCV-HW-TCV22	Test Check Valve 22 fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
RCI-TDP-FO-20S38	Failure of turbine driven pump to operate	5.0E-3/hr	10	1	-	10	5.0E-3	single component event	plant data
RCI-TDP-FR-20S38	Turbine-driven pump 20S38 fails to run	5.0E-3/hr	10	10	-	10	5.0E-2	single component event	plant data
RCI-TDP-FS-20S38	Turbine-driven pump 20S38 fails to start	3.0E-2/d	10	-	1.13E-2	10	3.0E-2	single component event	ASEP generic, revision 1 [55]
RCI-TDP-MA-20S38	Turbine-driven pump out for maintenance	1.0E-2/d	10	-	3.8E-3	10	1.0E-2	single component event	ASEP generic, revision 1 [55]
RCI-XVM-PG-XV17	Manual Valve 17 plugs	4.0E-5/d	3	-	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
RCI-XVM-PG-XV9	Manual Valve 9 plugs	-	-	-	-	-	4.0E-5	same as RCI-XVM-PG-XV17	--
SHUTDOWN COOLING SYSTEM (RHR):									
SDC-MOV-CC-MV15A	Motor-Operated Valve 15A fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SDC-MOV-CC-MV15B	Motor-Operated Valve 15B fails to open	-	-	-	-	-	3.0E-3	same as SDC-MOV-CC-MV15A	--
SDC-MOV-CC-MV15C	Motor-Operated Valve 15C fails to open	-	-	-	-	-	3.0E-3	same as SDC-MOV-CC-MV15A	--
SDC-MOV-CC-MV15D	Motor-Operated Valve 15D fails to open	-	-	-	-	-	3.0E-3	same as SDC-MOV-CC-MV15A	--
SDC-MOV-CC-MV17	Motor-Operated Valve 17 fails to open	-	-	-	-	-	3.0E-3	same as SDC-MOV-CC-MV15A	--
SDC-MOV-CC-MV18	Motor-Operated Valve 18 fails to open	-	-	-	-	-	3.0E-3	same as SDC-MOV-CC-MV15A	--
SDC-MOV-MA-MV15A	Motor-Operated Valve 15A out for maintenance	2.0E-4/d	10	-	7.5E-5	10	2.0E-4	single component event	original ASEP generic document [2]
SDC-MOV-MA-MV15B	Motor-Operated Valve 15B out for maintenance	-	-	-	-	-	2.0E-4	same as SDC-MOV-MA-MV15A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SDC-MOV-MA-MV15C	Motor-Operated Valve 15C out for maintenance	-	-	-	-	-	2.0E-4	same as SDC-MOV-MA-MV15A	--
SDC-MOV-MA-MV15D	Motor-Operated Valve 15D out for maintenance	-	-	-	-	-	2.0E-4	same as SDC-MOV-MA-MV15A	--
SDC-MOV-00-MV13A	Motor-Operated Valve 13A fails to close	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
SDC-MOV-00-MV13B	Motor-Operated Valve 13B fails to close	-	-	-	-	-	3.0E-3	same as SDC-MOV-00-MV13A	--
SDC-MOV-00-MV13C	Motor-Operated Valve 13C fails to close	-	-	-	-	-	3.0E-3	same as SDC-MOV-00-MV13A	--
SDC-MOV-00-MV13D	Motor-Operated Valve 13D fails to close	-	-	-	-	-	3.0E-3	same as SDC-MOV-00-MV13A	--
SDC-XVM-PG-XV1	Manual Valve 1 plugs	1.0E-7/hr	3	720	3.2E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
STANDBY LIQUID CONTROL SYSTEM:									
SLC-CCF-LF-MOVS	Common cause failure of motor-operated valves	3.0E-3 x 0.049	10	-	-	-	1.47E-4	single event	ASEP generic, revision 1 [55]; Basic event converted to: SLC-MOVCC-CCF *BETA-2MOVS
SLC-CCF-PF-MDPS	Common cause pump failure to start	3.0E-3/d x 0.21	10	-	-	-	6.3E-4	single event	ASEP generic, revision 1 [55] & engineering judgement; Basic event converted to: SLC-MDP-FS-CCF *BETA-2SIPUMPS
SLC-CKV-HW-CV16	Check valve 16 fails to open	1.0E-4/d	3	-	8.0E-5	3	1.0E-4	single component event	ASEP generic, revision 1 [55]
SLC-CKV-HW-CV17	Check valve 17 fails to open	-	-	-	-	-	1.0E-4	same as SLC-CKV-HW-CV16	--
SLC-CKV-HW-CV43A	Check valve 43A fails to open	-	-	-	-	-	1.0E-4	same as SLC-CKV-HW-CV16	--
SLC-CKV-HW-CV43B	Check valve 43B fails to open	-	-	-	-	-	1.0E-4	same as SLC-CKV-HW-CV16	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SLC-EPV-HW-EV14A	Explosive valve 14A fails to operate	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	original ASEP generic document [2]
SLC-EPV-HW-EV14B	Explosive valve 14B fails to operate	-	-	-	-	-	3.0E-3	same sa SLC-EPV-HW-EV14A	--
SLC-EPV-MA-EV14A	Explosive valve 14A unavailable due to maintenance	2.0E-4/d	10	-	7.52E-5	10	2.0E-4	single component event	original ASEP generic document [2]
SLC-EPV-MA-EV14B	Explosive valve 14B unavailable due to maintenance	-	-	-	-	-	2.0E-4	same as SLC-EPV-MA-EV14A	--
SLC-MDP-FR-MDPA	Pump A fails to run	3.0E-5/hr	10	.5	5.64E-6	10	1.5E-5	single component event	ASEP generic, revision 1 [55]
SLC-MDP-FR-MDPB	Pump B fails to run	-	-	-	-	-	1.5E-5	same as SLC-MDP-FR-MDPA	--
SLC-MDP-FS-MDPA	Pump A fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
SLC-MDP-FS-MDPB	Pump B fails to start	-	-	-	-	-	3.0E-3	same as SLC-MDP-FS-MDPA	--
SLC-MDP-MA-MDPA	Motor-driven Pump A unavailable due to maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
SLC-MOV-OO-MV15	Motor-Operated valve 15 fails to close	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SLC-MOV-OO-MV18	Motor-Operated valve 18 fails to close	-	-	-	-	-	3.0E-3	same as SLC-MOV-OO-MV15	--
SLC-PSF-MA-MDPB	Motor-Driven Pump B unavailable due to maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single event	ASEP generic, revision 1 [55]
SLC-SRV-CC-RV39A	Relief valve 39A fails open	3.0E-4/d	10	-	1.0E-4	10	3.0E-4	single component event	ASEP generic, revision 1 [55]
SLC-SRV-CC-RV39B	Relief valve 39B fails open	-	-	-	-	-	3.0E-4	same as SLC-SRV-CC-RV39A	--
SLC-SYS-TE-SLC	SLC system unavailable during test	3.4E-3/d	-	-	-	-	3.4E-3	single event	HRA
SLC-XHE-FO-SLC	Operator fails to initiate SLC	-	-	-	-	-	2.0E-2	single event	HRA
SLC-XHE-RE-DIVER	Operator fails to restore system after test	-	-	-	1.2E-2	10	3.19E-2	single event	HRA
SLC-XHE-RE-EV14A	Explosive valve 14A not properly restored after maintenance	-	-	-	3.0E-3	10	7.98E-3	single event	HRA

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SLC-XHE-RE-EV14B	Explosive valve 14B not properly restored after maintenance	-	-	-	-	-	7.98E-3	same as SLC-XHE-RE-EV14A	--
SLC-XHE-RE-MDPA	Motor-Driven Pump A Train not properly restored after maintenance	-	-	-	1.0E-3	12	3.13E-3	single event	HRA
SLC-XHE-RE-MDPB	Motor-Driven Pump B Train not properly restored after maintenance	-	-	-	-	-	3.13E-3	same as SLC-XHE-RE-MDPA	--
SLC-XVM-PG-XV11	Pump Suction Manual Value 11 plugged	1.0E-7/hr	3	720	3.25E-5	3	4.0E-5	single component event	ASEP generic, revision 1 [55]
SLC-XVM-PG-XV12A	Manual valve 12A plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--
SLC-XVM-PG-XV12B	Manual valve 12B plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--
SLC-XVM-PG-XV13A	Manual valve 13A plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
SLC-XVM-PG-XV13B	Manual valve 13B plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--
SLC-XVM-PG-XV15	Manual valve 15 plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--
SLC-XVM-PG-XV18	Manual valve 18 plugs	-	-	-	-	-	4.0E-5	same as SLC-XVM-PG-XV11	--
SUPPRESSION POOL COOLING SYSTEM (RHR):									
RHR-CCF-PF-MDPS	RHR common cause pump failure	3.0E-3/d x 0.1	10	-	-	-	3.0E-4	single component event	ASEP generic, revision 1 [55]; Basic event converted to: RHR-MDP-FS-CCF *BETA-4RHRMDPS
RHR-MOV-CC-MV34A	Motor-Operated Valve 34A fails to open	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]
RHR-MOV-CC-MV34B	Motor-Operated Valve 34B fails to open	-	-	-	-	-	3.0E-3	same as RHR-MOV-CC-MV34A	--
RHR-MOV-CC-MV39A	Motor-Operated Valve 39A fails to open	-	-	-	-	-	3.0E-3	same as RHR-MOV-CC-MV34A	--

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
RHR-MOV-CC-MV39B	Motor-Operated Valve 39B fails to open	-	-	-	-	-	3.0E-3	same as RHR-MOV-CC-MV34A	--
RHR-MOV-MA-MV39A	Motor-Operated Valve 39A out for maintenance	2.0E-4/d	10	-	7.5E-5	10	2.0E-4	single component event	original ASEP generic document [2]
RHR-MOV-MA-MV39B	Motor-Operated Valve 39B out for maintenance	-	-	-	-	-	2.0E-4	same as RHR-MOV-MA-MV39A	--
SPC-CCF-LF-MOVS	Common cause failure of SPC injection valves	3.0E-3/d x 0.049	10	-	-	10	1.47E-4	single event	ASEP generic, revision 1 [55]; Basic event converted to: SPC-CCF-LF-MOVS *BETA-2MOVS
TURBINE BUILDING COOLING WATER SYSTEM:									
TBC-MDP-FR-PUMPA	Pump A fails to run	3.0E-5/hr	10	40	4.5E-4	10	1.2E-3	single component event	ASEP generic, revision 1 [55]
TBC-MDP-FR-PUMPB	Pump B fails to run	-	-	-	-	-	1.2E-3	same as TBC-MDP-FR-PUMPA	--
TBC-PTF-FS-PUMPB	Pump B train fails to start	3.0E-3/d	10	-	1.13E-3	10	3.0E-3	single component event	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
TBC-PTF-MA-PUMPB	Pump B train unavailable due to maintenance	2.0E-3/d	10	-	7.52E-4	10	2.0E-3	single component event	ASEP generic, revision 1 [55]
TBC-PTF-RE-PUMPB	Failure to restore Pump B train after maintenance	-	-	-	1.2E-3	10	3.19E-3	single event	HRA
INITIATING EVENTS:									
Large LOCA (A)	--	N/A	N/A	N/A	-	10	1.0E-4	--	WASH-1400 [4], other PRAs
Intermediate LOCA (S1)	--	N/A	N/A	N/A	-	10	3.0E-4	--	WASH-1400 [4], other PRAs
Small LOCA (S2)	--	N/A	N/A	N/A	-	10	3.0E-3	--	WASH-1400 [4], original ASEP generic document [2]
Small-Small LOCA (S3)	--	N/A	N/A	N/A	-	10	3.0E-2	--	original ASEP generic document [2], NRC summary of actual experience
LOSP (T1)	--	N/A	N/A	N/A	-	3	7.9E-2	--	plant specific See Appendix D

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
Transient without PCS inititally available (T2)	--	N/A	N/A	N/A	-	3	5.0E-2	--	plant specific See Appendix D
Transient with PCS initially available (T3A)	--	N/A	N/A	N/A	-	-	2.5	--	plant specific See Appendix D
Loss of feedwater transient (T3B)	--	N/A	N/A	N/A	--	3	6.0E-2	--	plant specific See Appendix D
Inadvertent opening of a relief valve (IORV) (T3C)	--	N/A	N/A	N/A	--	3	1.9E-1	--	plant specific See Appendix D
Loss of an emergency AC/DC bus (TAC/DC)	--	N/A	N/A	N/A	--	3	5.0E-3	--	ASEP generic, revision 1 [55]
BETA FACTORS VALUES:									
BETA-2MOVS	Common cause factor for two motor-operated valves	-	-	-	-	3	4.9E-2	--	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Continued)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
BETA-2SIPUMPS	Common cause factor for two safety injection pumps	-	-	-	-	3	2.1E-1	--	ASEP generic, revision 1 [55]
BETA-2SWPS	Common cause factor for two service water pumps	-	-	-	-	3	2.6E-2	--	ASEP generic, revision 1 [55]
BETA-3AOVS	Common cause factor for three air-operated valves	-	-	-	-	3	5.5E-2	--	ASEP generic/ engineering judgement
BETA-3RHRMDPS	Common cause factor for 3 RHR pumps	-	-	-	-	3	1.1E-1	--	ASEP generic, revision 1 [55]
BETA-3SRVS	Common cause factor for 3 safety relief valves (fail to open)	-	-	-	-	3	1.5E-1	--	ASEP generic, revision 1 [55]
BETA-4DGNS	Common cause factor for 4 diesel generators	-	-	-	-	3	1.3E-2	--	ASEP generic, revision 1 [55]
BETA-4MOVS	Common cause factor for 4 motor-operated valves	-	-	-	-	3	3.2E-2	--	ASEP generic, revision 1 [55]

Table 4.9-1
Peach Bottom Event Data (Concluded)

BASIC EVENT	COMPONENT FAILURE MODE DESCRIPTION	FAILURE RATE: RATE (PER DEMAND OR HOUR)	FAILURE RATE: EF	MISSION TIME ~(HRS)	UNAVAILIBILITY: MEDIAN	UNAVAILIBILITY: EF	UNAVAILIBILITY: MEAN	DESCRIPTION	SOURCE/ COMMENTS
BETA-4RHRMDPS	Common cause factor for 4 RHR pumps	-	-	-	-	3	1.0E-1	--	ASEP generic, revision 1 [55]
BETA-4SRVS	Common cause factor for 4 safety relief valves (fail to open)	-	-	-	-	3	1.2E-1	--	ASEP generic, revision 1 [55]
BETA-4SWPS	Common cause factor for 4 service water pumps	-	-	-	-	3	9.6E-3	--	ASEP generic, revision 1 [55]
BETA-5BAT	Common cause factor for 5 batteries	-	-	-	-	3	2.5E-3	--	plant specific/ NUREG-0666 [24]
BETA-6AOVS	Common cause factor for 6 air-operated valves	-	-	-	-	3	3.6E-2	--	ASEP generic, revision 1 [55] & engineering judgement

4.10 Accident Sequence Quantification

4.10.1 General Approach

The accident sequences developed in the event tree analysis were analyzed to determine the core damage sequences with the highest contributions to the total core damage frequency. The sequences were quantified by combining the Boolean equations derived from the system failure models using the event tree logic associated with the sequences, and reducing the resultant equation to form minimal cut sets. System successes were explicitly included in the sequence logic. The sequence minimal cut sets were quantified using the data (mean values) established for the project. While in general these were the steps followed, the actual process was actually more complicated and included a number of screening steps. The following paragraphs describe the explicit sequence quantification process and identify plant-specific quantification issues.

The quantification of accident sequences was performed using a step-by-step, screening approach, building upon small quantification efforts until whole sequences (where necessary) were quantified.

First, as part of each system failure model quality assurance check, system minimal cut sets were obtained without and then with support system (e.g. power, cooling, etc.) failures included. After being reviewed for accuracy, these fault tree models were linked together using the SETS code [42] to form portions of entire accident sequences given in the event trees discussed in Section 4.4. As the linking process was performed, success states of certain systems were explicitly accounted for when forming these partial sequence Boolean expressions. In addition, certain failures were precluded when necessary to obtain the correct minimal cut sets. For example, high pressure coolant injection (HPCI) failure following an intermediate LOCA (S1) should not include long term loss of room cooling failures since HPCI will fail on loss of steam pressure in less than one hour. The mean data values were applied to the basic events in these Boolean expressions except for human errors which were assigned screening values (0.5 or greater). At this point in the quantification process, initiator frequencies and recovery actions were not yet included.

During this process, adjustments were made to the cut sets for three primary reasons. First, double test and maintenance terms not allowed by technical specifications were eliminated by hand. Second, a variety of adjustments had to be made because of the complexity of the Emergency Service Water (ESW) system. In this latter case, the ESW system fault tree was constructed so as to simplify the model. This simplification process resulted in the ESW system fault tree yielding conservative answers by providing failure cut sets which in fact do not fail the system. An example is the immediate failure to start of the two primary ESW pumps and failure of the operator to start the backup emergency cooling water (ECW) pump. The ECW pump starts automatically with the start of the ESW pumps. The ECW pump automatically trips after about 45 seconds if the discharge pressure from the two main pumps is adequate, but the ESW pump must be manually started if the discharge pressure falls below normal after the ECW pump automatic trip. Therefore, operator failure to start the ECW pump after immediate failure to start of the two main pumps is meaningless and was removed from the sequence cut sets. The failure-to-run type cut sets which would be allowable were "captured" using other terms in the ESW system fault tree. Similar adjustments to other ESW failure terms had to be made. Third, again because of simplifications in the ESW model, subsequent terms representing required failures of the Normal Service Water (NSW) system had to be added to some sequence cut sets. This was done to accurately reflect loss of service water cooling to some loads. For example, a cut set would describe the failure of ESW to provide cooling to some vital load when, in fact, NSW had not failed to provide cooling to this load. Hence a NSW failure term had to be added to the sequence cut sets to obtain a proper evaluation.

Examination of the resulting partial sequence expressions showed that in some cases, the probability of system successes and failures were sufficiently low and could be eliminated. That is, it could be shown that even if additional system failures that are required to cause core damage were assumed to fail at a probability of 1.0 and the initiator frequency (frequencies can be greater than 1.0 per year) was included, a sequence core damage frequency estimate of less than 1E-8 would result for the full sequence expression. These partial sequences were therefore eliminated from further analysis. The remaining partial sequence expressions had the potential of being greater than 1E-8 in core damage frequency when analyzed further.

The above remaining expressions were then combined with the initiator, other independent multipliers per the sequence being quantified, and some recovery actions (by hand). Other multipliers included for example, P1, the probability of a stuck-open relief valve. This is not analyzed by a fault tree but with a data value that is independent of the other system faults. This portion of the analysis was performed by applying the appropriate terms and data to the sequence expressions, thereby creating quantified but still only partial accident sequence expressions. These expressions included the initiator, system failures, and partial recovery. In some cases, more realistic human error values were used to assist in the elimination process. Again, these expressions were screened and those with a core damage frequency less than 1E-8 were eliminated from further analysis.

The results of the above process identified sequence expressions with the potential of leading to core damage frequency estimates of 1E-8 or greater. In some cases, the expressions already represented a core damage sequence. Other expressions were of the AW-, SW-, or TW-type sequence in which core cooling was thus far successful but containment cooling was failed and containment venting ("Y" event) success or failure could lead to a core damage state depending on the success or failure of continued core cooling (a so-called core vulnerable sequence).

At this point, the Peach Bottom analysts used information supplied by the containment response analysts using the inputs for the expert elicitation issue on equipment operability in harsh environments to determine the resulting estimates for continued core cooling following success or failure of venting. This was done to account for the potentially significant interaction between the containment status and the survivability of long term core cooling in Boiling Water Reactors (BWRs). Phenomenological failures such as potential for pipe failures following containment failure were considered. The loss of the Low Pressure Core Spray and Low Pressure Coolant Injection systems under pool saturated conditions was treated as per the general event tree assumptions in Section 4.4. Possible steam environments in the reactor building, such as when containment venting success into the SGTS ductwork causes overpressure failure of the ductwork, were also considered as to their effects on long term cooling systems (e.g., effects on operability of motor operated valves, motor control centers etc.). Simple Boolean expressions to cover the above considerations were constructed by the containment analysts, and estimates of the core damage potential were made by combining the partial sequence frequencies from the Peach Bottom analyses with these simple Boolean expressions and performing simple hand calculations. The results of this combined effort identified those sequences worthy of complete analysis whenever core damage potentials appeared greater than 1E-8.

Following the above process, sequences which appeared to have frequency estimates greater than 1E-8 were completely analyzed. This was done by setting the human errors to their correct values (not screening values) and applying complete recovery to each accident sequence cut set. The result of this process yielded the final dominant sequences for the Peach Bottom analysis. After obtaining sequence frequency point estimates, uncertainty estimates were added to the data. The quantified uncertainty analysis was then performed using the TEMAC code [28].

4.10.2 Identification of Sequences Analyzed

By following the screening quantification approach presented above, the dominant core damage sequences were identified and completely quantified. The quantification accounted for applicable recovery actions in each sequence. A dominant sequence was defined as a unique initiator coupled with a set of system successes and failures, and recovery actions, that resulted in an estimated core damage frequency of greater than 1E-8 (per year).

Table 4.10-1 summarizes how all of the sequences that lead to core damage were analyzed and how most were eliminated during the quantification process. Depicted are an accident sequence identifier (1, 2, 3...) for reference purposes; the event tree sequence (e.g., A-5 is sequence number 5 in the A (large LOCA) event tree); the entire sequence Boolean expression; the estimated frequency for the sequence based on how much of the screening process were performed; what expression was quantified (usually a combination of computer run and hand multiplication); and comments about the quantification. During this screening process, only a sufficient level of effort was performed in each case to eliminate the sequence. For example, typically only partial recovery was performed and often, human errors were kept at screening values. Therefore, the values shown in Table 4.10-1 are generally conservative estimates of the real sequence frequencies. Those sequences eliminated from further analysis by this process are designated by a "yes" in the appropriate column of Table 4.10-1. Sequences not eliminated were analyzed completely with all human errors set at correct values and with full recovery applied.

4.10.3 Application of Operator Recovery Actions

The specific operator and recovery actions used in the Peach Bottom analysis have been previously discussed in Section 4.8. As mentioned earlier, some recovery actions were included at the cut set level for the partial sequence expressions during one of the screening steps in the quantification process. In each case, applicable non-recovery terms were applied to the computerized Boolean expressions and the partial sequence expressions were re-quantified based on the appropriate non-recovery probability depending on sequence timing. As per the general methodology guidelines [2], no component hardware or test/maintenance unavailabilities were considered recoverable except for the few electrical cases identified by the recovery actions discussed in Section 4.8. Recovery actions were selected on the basis of that action which would (1) if successful, mitigate the potential core damage scenario depending on the specific cut set involved and (2) be expected to be performed considering the sequence timing, the specific failures involved, and Peach Bottom's procedures.

While in most cases only one human action event was allowed per accident sequence cut set, there were some exceptions. If the human action events could be shown to be independent, then more than one event (i.e., recovery actions) was allowed. In the Peach Bottom analysis, there are two important situations where credit was given for more than one recovery action:

(1) Failure to recover electric power <u>and</u> failure to recover an injection system,

(2) Failure to recover the Power Conversion System (PCS)/Condensate and failure to operate or recover other cooling methods.

When applying the recovery actions, care was taken to assure that the actions could be performed when considering venting success or failure (if appropriate). This was done to assure that the recovery actions were still valid even under possible severe containment or reactor building conditions. The specific recovery actions used to eliminate some sequences in the screening process are covered in Table 4.10-1.

The specific and complete recovery actions applied to the potentially dominant sequences are covered in Table 4.10-2 for those sequences shown as not eliminated in Table 4.10-1. In Table 4.10-2, the same accident sequence identifier, event tree sequence, and Boolean expression are shown as in Table 4.10-1. The recovery actions applied to each potentially dominant sequence are also summarized.

Table 4.10-3 summarizes how each of the sequences in Table 4.10-2 were affected by the complete quantification and recovery process. Shown are the point estimate frequencies before and after full recovery and a final resolution as to what sequences remained dominant (depicted by a "no" in the appropriate column of Table 4.10-3). Section 5 summarizes the final dominant sequences and presents the mean and uncertainty values following the complete uncertainty analysis.

The Peach Bottom analysis arrived at 18 dominant accident sequences. The dominant sequence frequencies were calculated completely by coupling the initiator, system cut sets, and non-recovery terms using the TEMAC code [28] and proper data values. In the case of quantifying the dominant sequences using TEMAC, two event name transformations were performed. First, all important common cause events were transformed into two events; an event name representing the first component failure and a second event name representing the common cause failure factor (see Section 4.7 for additional details). This was done to calculate importance measures for each common cause factor individually. Second, each important cut set involving battery depletion failures in station blackout scenarios were broken out into five cut sets. One represented the chance of battery depletion occurring in three hours and the subsequent need to recover AC power in approximately five hours so as to prevent core damage. Another represented the probability of battery depletion occurring in five hours, another in seven hours, another in nine hours, and the last for battery depletion occurring in greater than approximately ten hours. In the latter case, other equipment failures due to loss of room cooling dominate long-term failures, particularly of HPCI and RCIC. This transformation was done to better analyze the uncertainty effects in the battery depletion time by discretizing the battery depletion curve into the five time periods mentioned above and providing a weighting factor representing the chance that the batteries deplete their power in each time period. More on this is covered in Section 4.12 of this report. The resulting 18 dominant sequences depict the most significant sequences contributing to the core damage frequency at Peach Bottom.

Of the over one thousand possible unique accident sequences that lead to core damage as depicted on the event trees, 18 dominant sequences were identified during the various phases of the screening process. Based on the details of the quantification during the screening process, it is estimated by the analysts that the total core damage contribution of the eliminated sequences (using a truncation value for individual cut sets of the order of E-10) is approximately 5E-7/year. This means that collectively the screened or eliminated sequences (each of which is below 1E-8/year) constitute nearly 5% of the total core damage frequency. As a result, the 18 dominant sequences reported in the Results, Section 5, are estimated to represent approximately 95% of the total core damage frequency due to internal initiators.

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
1	A-5	A*/C*/LOSP*/V2*W1,3*/Y*V1,4	<6E-9	A*/V2*W1,3*/Y*V1,4 (with and without LOSP failure)	Yes	o Realistic failure to start RHR o Partial recovery included o Little steam effects on V1, V4 with /Y
2	A-8	A*/C*/LOSP*/V2*W1,3*Y*V1,4	<6E-9	A*/V2*W1,3*Y (with and without LOSP failure)	Yes	o Realistic failure to start RHR o Partial recovery included o Y approx. = 0.1 based on human error and cut sets that would lead to Y (failure)
3	A-13	A*/C*/LOSP*V2*/V3*W1,3*/Y*V1,4	<6E-9	A*V2*/V3*W1,3*/Y*V1,4 (with and without LOSP failure)	Yes	o Realistic failure to start RHR o Partial recovery included o Little steam effects on V1, V4 with /Y
4	A-16	A*/C*/LOSP*V2*/V3*W1,3*Y*V1,4	<6E-9	A*V2*/V3*W1,W3*Y (with and without LOSP failure)	Yes	o Realistic failure to start RHR o Partial recovery included o Y approx. = 0.1 based on human error and cut sets that would lead to Y (failure)

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
5	A-17	A*/C*/LOSP*V2,3	>5E-8	A*V2*V3 (with and without LOSP failure)	No	---
6	A-21,23, 27,29,30	Like Sequences 1-5 above with LOSP failure	each <E-9	Covered by A-5,8,13, 16,17, respectively	Yes (all)	o Subsequent LOSP failure =2E-4
7	A-31	A*C	<1E-9	A*C	Yes	o A(1E-4)*C(1E-5)
8	S1-5	S1*/C*/LOSP*/U1*/V2*W1,3* /Y*V1,4	<1E-8	Covered by A-5 with A replaced by S1	Yes	o Like A-5
9	S1-8	S1*/C*/LOSP*/U1*/V2*W1,3* Y*V1,4	<1E-8	Covered by A-8 with A replaced by S1	Yes	o Like A-8
10	S1-13	S1*/C*/LOSP*/U1*V2*/V3*W1,3* /Y*V1,4	<1E-8	Covered by A-13 with A replaced by S1	Yes	o Like A-5
11	S1-16	S1*/C*/LOSP*/U1*V2*/V3*W1,3* Y*V1,4	<1E-8	Covered by A-16 with A replaced by S1	Yes	o Like A-8
12	S1-21	S1*/C*/LOSP*/U1*V2*V3*/V4* W1,3*/Y*V1,4	<1E-8	S1*V2,3*/V4*W1,3*/Y*V1,V4 (with and without LOSP failure)	Yes	o Like A-5
13	S1-24	S1*/C*/LOSP*/U1*V2*V3*/V4* W1,3*Y*V1,4	<1E-8	S1*V2,3*/V4*W1,3*Y (with and without LOSP failure)	Yes	o Like A-8

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
14	S1-25	S1*/C*/LOSP*/U1*V2*V3*V4	>1E-7	S1*/U1*V2,3,4 (with and without LOSP failure)	No	---
15	S1-30,33,38 41,46,49	Like Sequences 8-13 above with U1*/X1(a)	<1E-8	Covered by S1-5,8,13, 16,21,24, respectively	Yes (all)	o U1 failure implies sequences $\leq$ previous S1 sequences
16	S1-50	S1*/C*/LOSP*U1*/X1*V2,3,4	<7E-9	S1*U1*/X1*V2,3,4 (with and without LOSP failure)	Yes	o No recovery o Adjusted screen value to 0.1 for failure to operate HPSW
17	S1-51	S1*/C*/LOSP*U1*X1	<1E-8	S1*U1*X1 (with and without LOSP failure)	Yes	o All opertor error values at high screen values
18	S1-55,57, 61,63,67,69	Like Sequences 8-13 with LOSP failure	each <E-9	Covered by S1-5,8,13, 16,21,24 respectively	Yes (all)	o Subsequent LOSP failure =2E-4
19	S1-70	Like Sequence 14 with LOSP failure	<E-9	Covered by S1-25	Yes	o Subsequent LOSP failure =2E-4
20	S1-74,76, 80,82,86,88 89,90	Like Sequences 15-17 with LOSP failure	each <E-9	Covered by S1-30,33,38 41,46,49,50,51 respectively	Yes (all)	o Subsequent LOSP failure =2E-4
21	S1-91	S1*C	<3E-9	S1*C	Yes	o S1(3E-4)*C(1E-5)

(a) 8-13 refers to the identifier numbers shown in the first column under the heading "Accident Sequence". (Similar for references to other sequences).

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
22	S2-3-6	S2*/C*/LOSP*Q1*/U1*W1*/W3*U4*/X1*V1,2,3,4	<3E-9	S2*/U1*W1*/W3*U4*/X1 (with and without LOSP failure)	Yes	o Not all system failures included in calculation o Allowed for recovery of offsite power/PCS
23	S2-3-7	S2*/C*/LOSP*Q1*/U1*W1*/W3*U4*X1	<3E-10	S2*/U1*W1*/W3*U4*/X1 (with and without LOSP failure)	Yes	o Allowed for recovery of offsite power/PCS
24	S2-4-4	S2*/C*/LOSP*Q1*/U1*W1,3*/U4*/Y*U4*/X3*V1,4	<E-10	Entire Boolean expression (with and without LOSP failure)	Yes	o Allowed for recovery of offsite power/PCS o U4 failure after /Y ≤E-2
25	S2-4-5	S2*/C*/LOSP*Q1*/U1*W1,3*/U4*/Y*U4*X3	<E-10	Entire Boolean expression (with and without LOSP failure)	Yes	o Like S2-4-4
	S2-4-9,10,12	Similar to Sequences 24 and 25	<E-9	S2*/U1*W1,3*/U4*Y (with and without LOSP failure)	Yes (all)	o Allowed for recovery of offsite power/PCS o Adjusted screen value to 1E-2 for failure to vent containment

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
26	S2-4-15,16, 19,20,21,23, 24,26,27,28, 30,31,33,34, 35,37,38,40, 41,42,43	See S2 Event Tree	each <3E-9	S2*/U1*W1,3*U4*/X1 (with and without LOSP failure)	Yes (all)	o Virtually all cut sets have subsequent LOSP failure = 2E-4 o Not all system failures included in calculation o LOSP recovery not yet included
27	S2-4-44	S2*/C*/LOSP*Q1*/U1*W1,3* U4*X1	<E-10	S2*/U1*W1,3*U4*X1 (with and without LOSP failure)	Yes	o Virtually all cut sets have subsequent LOSP failure = 2E-4 o LOSP recovery not yet included
28	S2-6-6	Like Sequence 22 with U1*/U2	<3E-9	S2*U1*/U2*W1*/W3*U4*/X1 (with and without LOSP failure)	Yes	o Like S2-3-6
29	S2-6-7	Like Sequence 23 with U1*/U2	<3E-10	S2*U1*/U2*W1*/W3*U4*X1 (with and without LOSP failure)	Yes	o Like S2-3-7
30	S2-7-4,5, 9,10,12,15, 16,19,20,21, 23,24,26,27, 28,30,31,33, 34,35,37,38, 40,41,42,43,44	Like Sequences 24-27 with U1*/U2	each <3E-9	Just like corresponding S2-4-4,5,9,10,12, etc. above except with /U1 replaced by U1*/U2	Yes (all)	o Like corresponding S2-4-4,5,9,10,12, etc. sequences

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
31	S2-10-4,5,9, 10,12,15,16, 19,20,21	See S2 Event Tree	each <1E-8	S2*U1*U2*/X1*/V1*W1,3	Yes (all)	o Allowed for recovery of PCS o Not all system failures included in calculation
32	S2-13-3,4,7, 8,10,12,13,15, 16,17	See S2 Event Tree	each <E-9	S2*U1*U2*/X1*V1*/V2*W1,3 (with and without LOSP failure)	Yes (all)	o Like S2-10-... above
33	S2-16-3,4,7, 8,10,12,13,15, 16,17	See S2 Event Tree	each <E-10	S2*U1*U2*/X1*V1,2*/V3* W1,3 (with and with LOSP failure)	Yes (all)	o All cut sets have subsequent LOSP failure = 2E-4
34	S2-19-3,4,7, 8,10,12,13, 15,16,17	See S2 Event Tree	each <E-8	S2*U1*U2*/X1*V1,2,3*/V4* W1,3 (with and without LOSP failure)	Yes (all)	o Allowed for recovery of offsite power/PCS o Not all system failures included in calculation
35	S2-20	S2*/C*/LOSP*Q1*U1,2*/X1* V1,2,3,4	<E-9	Entire Boolean expression (with and without LOSP failure)	Yes	o Allowed for recovery of PCS o Subsequent LOSP failure = 2E-4
36	S2-21	S2*/C*LOSP*Q1*U1,2*X1	<E-8	Entire Boolean expression (with and without LOSP failure)	Yes	o Like S2-20 o Operator errors still at high screening values

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
37	(b)S2-23-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
38	S2-24-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
39	S2-26-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
40	S2-27-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
41	S2-30-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences

(b) S2-23-... refers to all the core damage accident sequences to which sequence number 23 on the S2 tree transfers. (Similar for other sequences).

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
42	S2-33-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
43	S2-36-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
44	S2-37-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
45	S2-38-...	Similar to Sequences 22-36 with LOSP failure and no /V1	each <E-8	Covered by all other S2 sequences above (LOSP failure already included in above sequences)	Yes (all)	o See above sequences
46	S2-39	S2*C	<E-8	S2*C (requires other system failures)	Yes	o S2(3E-3)*C(1E-5)* other failures (<.3) such as HPCI or SLC...
	S3-1-...	(appear later in table)				

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
47	S3-2	See Event Tree	Each <E-9	S3*L	Yes (all)	o S3(3E-2)*L(1E-2) =<S2(3E-3) o Hence if all S2 sequences are each non-dominant (<E-8), S3*L type sequences are each <E-9
48	S3-3	S3*C	<E-8	S3*C* (requires other system failures)	Yes	o S3(3E-2)*C(1E-5)* other failures (<.03) such as HPCI and ADS, or SLC...
49	T1-3-5	T1*/C*/M*/P*/B*/U1*W1*/X2* W2*/W3*U4*V2,3,4	<E-10	T1*/U1*W1*/X2*W2*/W3*/U4* V2,3,4	Yes	o No cut sets >E-10 before recovery
50	T1-4-3,4	See T1 Event Tree	Each <E-8	T1*/U1*W1*/X2*W2,3*U4* /Y*U4	Yes (all)	o Allowed for recovery of offsite power o U4 failure after /Y ≤E-2 o Little steam effects on X3,V4 with /Y
51	T1-4-7,8,10	See T1 Event Tree	Each <E-8	T1*/U1*W1*/X2*W2,3*/U4*Y	Yes (all)	o Allowed for recovery of offsite power o Adjusted screen value to 1E-2 for failure to vent containment o Realistic failure to start RHR

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
52	T1-4-12,13, 19,20,26,27	See T1 Event Tree	Each <5E-9	T1*/U1*W1*/X2*W2,3*U4*/Y	Yes (all)	o Allowed for recovery of offsite power o Realistic failure to start RHR o Not all system failures included in calculation
53	T1-4-15,16, 17,22,23,24, 29,30,31	See T1 Event Tree	Each <5E-9	T1*/U1*W1*/X2*W2,3*U4*Y	Yes	o Like T1-4-12,13, etc. above
54	T1-4-32	T1*/C*/M*/P*/B*/U1*W1*/X2* W2,3*U4*V2,3,4	<E-10	T1*/U1*W1*/X2*W2,3*U4* V2,3,4	Yes	o Not cut sets >E-10 before recovery
55	T1-5-5,6	See T1 Event Tree	Each <E-8	T1*/U1*W1*X2*/W3*U4	Yes (all)	o Allowed for recovery of offsite power o Lower screen value for depressurization failure (0.1) o Other system failures must also occur
56	T1-6-3,4	See T1 Event Tree	Each <E-9	T1*/U1*W1*X2*W3*/U4*/Y*U4	Yes (all)	o Allowed for recovery of offsite power o Realistic failure to start RHR o U4 failure after /Y ≤1E-2 o Lower screen value for depressurization failure (0.1)

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
57	T1-6-7,8,10	See T1 Event Tree	Each <E-8	T1*/U1*W1*X2*W3*/U4*Y	Yes (all)	o Allowed for recovery of offsite power o Adjusted screen value to 1E-2 for failure to vent containment o Realistic failure to start RHR o Other system failures must also occur
58	T1-6-13,14, 16,17,18,21, 22,24,25,26, 29,30,32,33, 34,35	See T1 Event Tree	Each <E-8	T1*/U1*W1*X2*W3*U4*/X1	Yes (all)	o Allowed for recovery of offsite power o Realistic failure to start RHR o Other system failures must also occur o Consideration for realistic values for operator failure to start U4 in 10 hours.
59	T1-6-36	T1*/C*/M*/P*/B*/U1*W1*X2* W3*U4*X1	<E-9	T1*/U1*W1*X2*W3*U4*X1	Yes	o Realistic failure to start RHR o No other recovery applied to calculation

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
60	T1-9-...	Similar to Sequences 49-59 with U1*/U2	Each <E-8	Did not run sequences by computer. They are just like corresponding T1 sequences above except U1*/U2 replaces /U1.	Yes (all)	o Additional failure (U1) ensures sequence estimates at or below sequence estimates with /U1
61	T1-10-...	Similar to Sequences 49-59 with U1*/U2	Each <E-8	Did not run sequences by computer. They are just like corresponding T1 sequences above except U1*/U2 replaces /U1.	Yes (all)	o Additional failure (U1) ensures sequence estimates at or below sequence estimates with /U1
62	T1-11-...	Similar to Sequences 49-59 with U1*/U2	Each <E-8	Did not run sequences by computer. They are just like corresponding T1 sequences above except U1*/U2 replaces /U1.	Yes (all)	o Additional failure (U1) ensures sequence estimates at or below sequence estimates with /U1
63	T1-12-...	Similar to Sequences 49-59 with U1*/U2	Each <E-8	Did not run sequences by computer. They are just like corresponding T1 sequences above except U1*/U2 replaces /U1.	Yes (all)	o Additional failure (U1) ensures sequence estimates at or below sequence estimates with /U1
64	T1-16-3,4	See T1 Event Tree	Each <E-8	T1*U1,2*/X1*/V2*W1,2,3*/U4*/Y	Yes (all)	o Realistic failure to start RHR o Allowed for recovery of offsite power o U4 failure after /Y ≤E-2

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
65	T1-16-7,8,10	See T1 Event Tree	Each <E-9	T1*U1,2*/X1*/V2*W1,2,3* /U4*Y	Yes (all)	o Realistic failure to start RHR o Allowed for recovery of offsite power
66	T1-16-12,13	See T1 Event Tree	Each <E-9	T1*U1,2*/X1*/V2*W1,2,3* U4*/Y	Yes (all)	o Allowed for recovery of offsite power
67	T1-16-15,16,17	See T1 Event Tree	Each <E-9	T1*U1,2*/X1*/V2*W1,2,3* U4*Y	Yes (all)	o Realistic failure to start RHR o Allowed for recovery of offsite power
68	T1-20-...	See T1 Event Tree	Each <E-8	T1*U1,2*/X1*V2*/V3*W1,2,3	Yes (all)	o Realistic failure to start RHR o Allowed for recovery of offsite power o Other system failures must occur
69	T1-24-3,4	See T1 Event Tree	Each <E-8	T1*U1,2*/X1*V2,3*/V4* W1,2,3*/U4*/Y	Yes (all)	o Allowed for recovery of offsite power o U4 failure after /Y ≤E-2
70	T1-24-7,8,10	See T1 Event Tree	Each <E-8	T1*U1,2*/X1*V2,3*/V4* W1,2,3*/U4*Y	Yes (all)	o Allowed for recovery of offsite power o Ajusted screen value to 1E-2 for failure to vent containment

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
71	T1-24-12,13	See T1 Event Tree	Each <E-9	T1*U1,2*/X1*V2,3*/V4* W1,2,3*U4*/Y	Yes (all)	o Allowed for recovery of offsite power
72	T1-24-15,16,17	See T1 Event Tree	Each <E-9	T1*U1,2*/X1*V2,3*/V4* W1,2,3*U4*Y	Yes (all)	o Allowed for recovery of offsite power
73	T1-25	T1*/C*/M*/P*/B*U1,2*/X1*V2,3,4	<E-8	Entire Boolean expression	Yes	o Allowed for recovery of offsite power o Realistic operator failure to start HPSW
74	T1-27-4	T1*/C*/M*/P*/B*U1,2*X1*/U3* W1*/X2*/W2*V2,3,4	<E-10	Entire Boolean	Yes expression	--
75	T1-28-4	T1*/C*/M*/P*/B*U1,2*X1*/U3*W1* /X2*W2*/W3*V2,3,4	<E-10	Entire Boolean expression	Yes	--
76	T1-29-2,3	See T1 Event Tree	Each <E-10	T1*U1,2*X1*/U3*W1*/X2* W2,3*/V2*/Y	Yes (all)	--
77	T1-29-5,6,7	See T1 Event Tree	Each <E-10	T1*U1,2*X1*/U3*W1*/X2* W2,3*/V2*Y	Yes (all)	--
78	T1-29-9,10	See T1 Event Tree	Each <E-10	T1*U1,2*X1*/U3*W1*/X2* W2,3*V2*/V3*/Y	Yes (all)	--
79	T1-29-12,13,14	See T1 Event Tree	Each <E-10	T1*U1,2*X1*/U3*W1*/X2* W2,3*V2*/V3*Y	Yes (all)	--

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
80	T1-29-16,17	See T1 Event Tree	Each <E-8	T1*U1,2*X1*/U3*W1*/X2* W2,3*V2,3*/V4*/Y	Yes (all)	o Allowed for recovery of offsite power o Near realistic value for failure to depressurize (1E-3)
81	T1-29-19,20,21	See T1 Event Tree	Each <E-8	T1*U1,2*X1*/U3*W1*/X2* W2,3*V2,3*/V4*Y	Yes (all)	o Same as T1-29-16,17
82	T1-29-22	T1*/C*/M*/P*/B*U1,2*X1*/U3* W1*/X2*W2,3*V2,3,4	<5E-9	Entire Boolean expression	Yes	o Same as T1-29-16,17
83	T1-31,3,4	See T1 Event Tree	Each <5E-9	T1*U1,2*X1*/U3*W1*X2* W3*/Y	Yes (all)	o Same as T1-29-16,17
84	T1-31-7,8,10	See T1 Event Tree	Each <5E-9	T1*U1,2*X1*/U3*W1*X2* W3*Y	Yes (all)	o Same as T1-29-16,17
85	T1-32	T1*/C*/M*/P*/B*U1,2*X1*U3	<E-8	Entire Boolean expression	Yes	o Allowed for recovery of offsite power o Realistic operator error estimates
86	T1-33	T1*/C*/M*/P*B*/U1	>E-6	Entire Boolean expression	No	--
87	T1-34	T1*/C*/M*/P*B*U1*/U2	>E-7	Entire Boolean expression	No	--

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
88	T1-35	T1*/C*/M*/P*B*U1,2	>E-6	Entire Boolean expression	No	--
89	T1-36/S2-...	See Event Tree	Each <E-8	Made use of S2 solved expressions except replace S2 with T1*P1 (LOSP must also be failed)	Yes (all)	o Allowed for recovery of offsite power o Examined S2 cut sets and adjusted for different initiator
90	T1-37	T1*/C*/M*P1*B*/U1	>E-6	Used T1-33 expression but with P1 included	No	--
91	T1-38	T1*/C*/M*P1*B*U1*/U2	>E-7	Used T1-34 expression but with P1 included	No	--
92	T1-39	T1*/C*/M*P1*B*U1,2	>E-7	Used T1-35 expression but with P1 included	No	--
93	T1-40/S1-...	See Event Tree	Each <E-8 except for Sequence 94	Made use of S1 solved expressions except replace S1 with T1*P2 (LOSP must also be failed)	Yes (all) except for Sequence 94	o Allowed for recovery of offsite power o Examined S1 cut sets and adjusted for different initiator
94	T1-40/S1-70	T1*/C*/M*P2*/B*/U1*V2,3,4	>E-8	As above	No	--
95	T1-41	T1*/C*/M*P2*B*/U1	>E-6	Entire Boolean expression	No	--

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
96	T1-42	T1*/C*/M*P2*B*U1	>E-7	Entire Boolean expression	No	--
97	T1-43/A-...	See Event Tree	Each <E-8	Made use of A solved expressions except replace A with T1*P3 (LOSP must also be failed)	Yes (all)	o Allowed for recovery of offsite power o Examine A cut sets and adjusted for different initiator
98	T1-44	T1*/C*/M*P3*B	>E-7	Entire Boolean expression	No	--
99	T1-45	T1*/C*M	<E-8	Did not computerize	Yes	o M failure believed to be negligible
100	T1-46	T1*C	>E-8	Treated in ATWS sequences (later in table)	No	--
101	T2-all sequences	See Event Tree	Each <E-8 except for Sequences 102-104	Used all past T1,S2,S1,A expressions as necessary adjusting for different initiator	Yes (all) except for Sequences 102-104	o Allowed for recovery of PCS where applicable o Subsequent LOSP failure = 2E-4 o Realistic operator errors in some cases
102	T2-38/S1-25	T2*/C*/LOSP*/M*P2*/U1*V2,3,4	>E-8	As above	No	--
103	T2-39/A-17	T2*/C*/LOSP*/M*P3*V2,3	>E-8	As above	No	--

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
104	T2-42	T2*C	>E-8	Treated in ATWS sequences (later in table)	No	--
105	T3A-all sequences	See Event Tree	See Comments	See Comments	See Comments	o Results are similar to T2 (see next table)
106	T3B-all sequences	See Event Tree	See Comments	See Comments	See Comments	o Results are similar to T2 (see next table)
107	T3C-all sequences	See Event Tree	See Comments	See Comments	See Comments	o Results are similar to T2 except no S1 or A transfers o All sequences estimated at <E-8
108	S3-1-...	See Event Tree	See Comments	See Comments	See Comments	o Just like T3A sequences o All sequences estimated at <E-8
109	TAC-DC-all sequences	See Event Tree	Each <E-8	Used all past T1,S2,S1,A expressions as necessary adjusting for different initiator	Yes (all)	o Allowed for recovery of bus and PCS in longer term sequences where appropriate o Subsequent LOSP failure = 2E-4

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
110	ATWS-5 for T1,T2,T3A,B,C	T*/RPSM*RPSE*ARI*SCRM*ROD	Each <E-8	T*/RPSM*RPSE*ARI*SCRM	Yes (all)	o Largest T = T2 = 2.5/year o RPSE = 2E-5 o ARI $\leq 10^{-2}$ o SCRM $\leq 10^{-2}$ o Other failures must occur before core damage
111	ATWS-17 for T1,T2, T3A,B,C	T*RPSM*RPT	Each <E-8	Entire expression	Yes (all)	o Largest T = T2 = 2.5/year o RPSM = 1E-5 o RPT = negligible considering electrical faults required and high probability of pump cavitation if they don't trip
112	ATWS-7,9,10, 13,14 for T1,T2,T3,A,B,C initiators	See Event Tree	Each <E-8	Entire expression	Yes (all)	o Failure of RHR (event W) and/or Low Pressure Cooling (event V) $\sim 10^{-4}$ or less with successful SLC and subsequent shutdown
113	ATWS-11 for T1,T2,T3A,B,C initiators	T*RPSM*/RPT*/M*/SLC*/I*U1*X1	>E-8	Entire expression	No	--

Table 4.10-1
Accident Sequences Quantified Before Full Recovery Applied (Concluded)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE(S)	SEQUENCE BOOLEAN EQUATION	ESTIMATED FREQUENCY	EXPRESSION QUANTIFIED	SEQUENCE(S) ELIMINATED	COMMENTS
114	ATWS-15 for T1,T2,T3A,B,C initiators	T*RPSM*/RPT*/M*SLC	>E-8	Entire expression	No	--
115	ATWS-16 for T1,T2,T3A,B,C initiators	T*RPSM*/RPT*M	<E-8	Did not computerize	Yes	o M failure believed to be negligible

Table 4.10-2
Potentially Dominant Accident Sequences Prior to Full Recovery

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	APPLICABLE RECOVERY ACTIONS	
5	A-17	A*/C*/LOSP*V2,3	PCSNR13HR:	Non-recovery of PCS/condensate in 13 hours following room cooling failure of LPCS/LPCI
14	S1-25	S1*/C*/LOSP*/U1*V2*V3*V4	No recovery on cut sets >E-10	
86	T1-33	T1*/C*/M*/P*B*/U1	LOSPNRXX:	Offsite power not restored in time XX depending on cut set
			DGCCFNRXX:	At least one diesel not restored in time XX after common cause failure
			DGHWNRXX:	Failed diesel not restored in time XX after a hard-ware failure
			DGMANRXX:	Diesel not restored to service in time XX when down for maintenance
			DGACTNRXX:	Actuation failure of diesel not recovered in time XX
			DCHWNRXX:	DC system not restored in time XX after a hardware failure
			RAXV503NC:	Operator doesn't close XV503 after backleakage occurs to NSW from the ESW system
87	T1-34	T1*/C*/M*/P*B*U1*/U2	LOSPNRXX:	Offsite power not restored in time XX depending on cut set
			DGCCFNRXX:	At least one diesel not restored in time XX after common cause failure
			DGHWNRXX:	Failed diesel not restored in time XX after a hard-ware failure
			DGMANRXX:	Diesel not restored to service in time XX when down for maintenance

Table 4.10-2
Potentially Dominant Accident Sequences Prior to Full Recovery (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	APPLICABLE RECOVERY ACTIONS	
			DCHWNRXX:	DC system not restored in time XX after a hardware failure
			RAXV503NC:	Operator doesn't close XV503 after backleakage occurs to NSW from the ESW system
88	T1-35	T1*/C*/M*/P*B*U1,2	LOSPNRXX:	Offsite power not restored in time XX depending on cut set
			DCHWNRXX:	DC system not restored in time XX after a hardware failure
90	T1-37	T1*/C*/M*P1*B*/U1	LOSPNRXX:	Offsite power not restored in time XX depending on cut set
			DGCCFNRXX:	At least one diesel not restored in time XX after common cause failure
			DGHWNRXX:	Failed diesel not restored in time XX after a hard-ware failure
			DGMANRXX:	Diesel not restored to service in time XX when down for maintenance
			DGACTNRXX:	Actuation failure of diesel not recovered in time XX
			DCHWNRXX:	DC system not restored in time XX after a hardware failure
91	T1-38	T1*/C*/M*P1*B*U1*/U2	LOSPNRXX:	Offsite power not restored in time XX depending on cut set
			DGCCFNRXX:	At least one diesel not restored in time XX after common cause failure

Table 4.10-2
Potentially Dominant Accident Sequences Prior to Full Recovery (Continued)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	APPLICABLE RECOVERY ACTIONS
			DGHWNRXX: Failed diesel not restored in time XX after a hardware failure
			DGMANRXX: Diesel not restored to service in time XX when down for maintenance
			DCHWNRXX: DC system not restored in time XX after a hardware failure
92	T1-39	T1*/C*/M*P1*B*U1,2	LOSPNRXX: Offsite power not restored in time XX depending on cut set
			DCHWNRXX: DC system not restored in time XX after a hardware failure
94	T1-40/S1-70	T1*/C*/M*P2*/B*/U1*V2,3,4	No recovery on cut sets >E-10
95	T1-41	T1*/C*/M*P2*B*/U1	LOSPNRXX: Offsite power not restored in time XX depending on cut set
96	T1-42	T1*/C*/M*P2*B*U1	LOSPNRXX: Offsite power not restored in time XX depending on cut set
98	T1-44	T1*/C*/M*P3*B	LOSPNRXX: Offsite power not restored in time XX depending on cut set
102	T2-38/S1-25	T2*/C*/LOSP*/M*P2*/U1*V2,3,4	No recovery on cut sets >E-10
103	T2-39/A-17	T2*/C*/LOSP*/M*P3*V2,3	PCSNR13HR: Non-recovery of PCS/condensate in 13 hours following room cooling failure of LPCS/LPCI

Table 4.10-2
Potentially Dominant Accident Sequences Prior to Full Recovery (Concluded)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	APPLICABLE RECOVERY ACTIONS
105	T3A-39/S1-25	T3A*/C*/LOSP*Q*/M*P2*/U1*V2,3,4	No recovery on cut sets >E-10
106	T3B/T2-38/S1-25	T3B*/C*/LOSP*/M*P2*/U1*V2,3,4	No recovery on cut sets >E-10
	T3B/T2-39/A-17	T3B*/C*/LOSP*/M*P3*V2,3	PCSNR13HR: Non-recovery of PCS/condensate in 13 hours following room cooling failure of LPCS/LPCI
113	ATWS-11	T1*RPSM*/RPT*/M*/SLC*/I*U1*X1	No Recovery
	ATWS-11	T2*RPSM*/RPT*/M*/SLC*/I*U1*X1	No Recovery
	ATWS-11	T3A*RPSM*/RPT*/M*/SLC*/I*U1*X1	No Recovery
	ATWS-11	T3B*RPSM*/RPT*/M*/SLC*/I*U1*X1	No Recovery
	ATWS-11	T3C*RPSM*/RPT*/M*/SLC*/I*U1*X1	No Recovery
114	ATWS-15	T1*RPSM*/RPT*/M*SLC	No Recovery
	ATWS-15	T2*RPSM*/RPT*/M*SLC	No Recovery
	ATWS-15	T3A*RPSM*/RPT*/M*SLC	No Recovery
	ATWS-15	T3B*RPSM*/RPT*/M*SLC	No Recovery
	ATWS-15	T3C*RPSM*/RPT*/M*SLC	No Recovery

Table 4.10-3
Potentially Dominant Accident Sequences Before and After Full Recovery

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	SEQUENCE FREQUENCY BEFORE RECOVERY*	SEQUENCE FREQUENCY AFTER RECOVERY*	SEQUENCE ELIMINATED
5	A-17	A*/C*/LOSP*V2,3	5.3E-8	5.3E-8	No
14	S1-25	S1*/C*/LOSP*/U1*V2*V3*V4	1.6E-7	1.6E-7	No
86	T1-33	T1*/C*/M*/P*B*/U1	1.6E-4	9.3E-7	No
87	T1-34	T1*/C*/M*/P*B*U1*/U2	1.1E-5	6.6E-8	No
88	T1-35	T1*/C*/M*/P*B*U1,2	1.8E-7	1.8E-7	No
90	T1-37	T1*/C*/M*P1*B*/U1	1.5E-5	8.1E-8	No
91	T1-38	T1*/C*/M*P1*B*U1*/U2	7.5E-8	7.4E-10	Yes
92	T1-39	T1*/C*/M*P1*B*U1,2	1.7E-8	1.7E-8	No
94	T1-40/S1-70	T1*/C*/M*P2*/B*/U1*V2,3,4	9.0E-8	9.0E-8	No
95	T1-41	T1*/C*/M*P2*B*/U1	8.4E-8	5.7E-9	Yes
96	T1-42	T1*/C*/M*P2*B*U1	8.7E-10	5.2E-10	Yes
98	T1-44	T1*/C*/M*P3*B	6.3E-9	1.6E-9	Yes
102	T2-38/S1-25	T2*/C*/LOSP*/M*P2*/U1*V2,3,4	5.3E-8	5.3E-8	No
103	T2-39/A-17	T2*/C*/LOSP*/M*P3*V2,3	5.3E-9	5.3E-9	Yes

* Point estimates.

Table 4.10-3
Potentially Dominant Accident Sequences Before and After Full Recovery (Concluded)

ACCIDENT SEQUENCE	EVENT TREE SEQUENCE	SEQUENCE BOOLEAN EQUATION	SEQUENCE FREQUENCY BEFORE RECOVERY*	SEQUENCE FREQUENCY AFTER RECOVERY*	SEQUENCE ELIMINATED
105	T3A-39/S1-25	T3A*/C*/LOSP*Q*/M*P2*/U1*V2,3,4	2.7E-8	2.7E-8	No
106	T3B/T2-38/S1-25	T3B*/C*/LOSP*/M*P2*/U1*V2,3,4	6.4E-8	6.4E-8	No
	T3B/T2-39/A-17	T3B*/C*/LOSP*/M*P3*V2,3	6.5E-9	6.5E-9	Yes
113	ATWS-11	T1*RPSM*/RPT*/M*/SLC*/I*U1*X1	Ran as 1 computerized run with all T's combined. Separated cut sets by hand according to specific initiator. No recovery actions.	8.1E-9	Yes
	ATWS-11	T2*RPSM*/RPT*/M*/SLC*/I*U1*X1		5.1E-9	Yes
	ATWS-11	T3A*RPSM*/RPT*/M*/SLC*/I*U1*X1		2.6E-7	No
	ATWS-11	T3B*RPSM*/RPT*/M*/SLC*/I*U1*X1		6.1E-9	Yes
	ATWS-11	T3C*RPSM*/RPT*/M*/SLC*/I*U1*X1		1.9E-8	No
114	ATWS-15	T1*RPSM*/RPT*/M*SLC	Ran as 1 computerized run with all T's combined. Separated cut sets by hand according to specific initiator. No recovery actions.	4.4E-8	No
	ATWS-15	T2*RPSM*/RPT*/M*SLC		2.8E-8	No
	ATWS-15	T3A*RPSM*/RPT*/M*SLC		1.4E-6	No
	ATWS-15	T3B*RPSM*/RPT*/M*SLC		3.4E-8	No
	ATWS-15	T3C*RPSM*/RPT*/M*SLC		1.1E-7	No

* Point estimates.

4.11 Plant Damage State Quantification

4.11.1 General Approach

Given the definition of the plant damage states in terms of the sixteen character vector (see Section 4.5), the task is to answer each question for every cut set of every accident sequence retained in the accident sequence quantification. Actually, many questions can be answered at the accident sequence level, e.g., all of the cut sets in the sequence have the same answer to the question asked. A few questions are answered differently; here, each cut set in the sequence must address the question. These few questions that require cut set examination may result in the division of a given accident sequence into two or more damage states. This does not change the overall core damage frequency, but only partitions the cut sets of the applicable accident sequence into two or more possible PDSs.

4.11.2 Identification of Plant Damage States Analyzed

The results of the PDS identification are given in Table 4.11-1. Each of the 18 dominant accident sequences is divided by cut set into the PDSs shown. Each PDS is defined by a 16 character vector depicting the applicable answers to each of the 16 questions which establish the PDS. Since the sixteen character vector is not in itself very expressive, a brief description is given in Table 4.11-2. The 16 questions are presented in Table 4.5-1. For the mechanics of manipulating the cut sets into PDSs, all that is needed is the PDS number and the accident sequence/cut set number. One simplification that was made by the back-end analysts was to combine large and medium LOCAs into one group and answer Question 1 with a 1 for both cases.

4.11.3 Quantification of Plant Damage States

When the cut sets from the 18 dominant accident sequences are sorted by PDS, 20 distinct plant damage states result. These are given in Table 4.11-3. These 20 were subsequently collapsed to 9 PDSs by the back-end analysts as shown in Table 4.11-4.

Events representing battery depletion uncertainty were then applied to three long-term station blackout accident sequences (Numbers 1, 6, and 7). This is discussed in Section 4.12. The end result is an expansion of the cut sets for those three accident sequences by a factor of 5. Theoretically, this expansion of cut sets should not change the core damage frequency. However, since this substitution also provided a more accurate evaluation of recovery, depending on when battery depletion might occur, and 1 of the 3 sequences was a very high contributor to core damage frequency, the core damage frequency did increase by approximately 12%. Table 4.11-5 summarizes the point estimate core damage frequencies before and after this change.

Table 4.11-1. Plant Damage States by Accident Sequence Before Simplification

Accident Sequence		Cut Sets(1)	PDS 16-Character Vector
1.	T1-BNU11	1-130	4-21S-2-22-S-22222-122
2.	T3A-C-SLC	1-9*ADS	5-322-2-23-2-33333-XX2
		1-9*/ADS*/VENT	5-322-2-23-3-43333-1X2
		1-9*/ADS*VENT	5-322-2-23-3-43333-4X2
3.	T3A-CU11X	1-14	5-322-2-23-2-33333-XX2
4.	S1-V2V3V4NU11	1-3	1-322-2-13-3-13113-XX2
5.	T1-BU11U21	1	4-211-2-12-1-22222-122
6.	T1-P1BNU11	1-57	4-21S-1-22-3-22222-122
7.	T1-BU11NU21	1-79	4-21S-2-22-S-22222-122
8.	T3C-C-SLC	1-6*ADS	5-322-1-23-2-33333-XX2
		1-6*/ADS*VENT	5-322-1-23-3-43333-1X2
		1-6*/ADS*/VENT	5-322-1-23-3-43333-4X2
9.	T1-P2V234NU11B	2,3	4-222-1-13-3-11131-XY2
		1,4,5	4-222-1-13-3-13113-XY2
10.	T2-P2V234NU11	1	4-322-1-13-3-13113-XX2
11.	T3B-P2V234NU11	1	4-322-1-13-3-13113-XX2
		2	4-322-1-13-3-11131-XX2
12.	A-V2V3	1-3	1-322-2-13-3-13113-XX2
13.	T1-C-SLC	1-4*ADS	5-222-2-23-2-33233-XY2
		1-4*/ADS*VENT	5-222-2-23-3-43233-1Y2
		1-4*/ADS*/VENT	5-222-2-23-3-43233-4X2
14.	T3B-C-SLC	1-4*ADS	5-322-2-23-2-33333-XX2
		1-4*/ADS*VENT	5-322-2-23-3-43333-1X2
		1-4*/ADS*/VENT	5-222-2-23-3-43233-4X2
15.	T2-C-SLC	1-4*ADS	5-322-2-23-2-33333-XX2
		1-4*/ADS*VENT	5-322-2-23-3-43333-1X2
		1-4*/ADS*/VENT	5-322-2-23-3-43333-4X2
16.	T3A-P2V234NU11	1	4-322-1-13-3-13113-XX2
17.	T3C-CU11X	1-5	5-322-1-23-2-33333-XX2
18.	T1-P1BU11U21	1	4-211-1-12-3-22222-122

(1) See Appendix E for more details.

Table 4.11-2. Plant Damage State (PDS) Vector Groups

Question	1	2,3,4	5	6,7	8	9,10,11,12,13	14,15,16
Description	Initiating Event	Electric Power	Stuck Open SRVs	High Press. Systems	ADS-RCS Depress.	Low Press & DHR Systems	Venting & Containment Isolation
Answers	1 A, S1 4 T 5 ATWS	212 SBO 211 SBO&DC Failure 21S SBO&Special Battery Depletion Consider-ations 222 LOSP Only 322 No LOSP	1 Yes 2 No	23 HPCI or RCIC Success,CRD Success 22 HPCI or RCIC Success,CRD Recoverable 13 HPCI&RCIC Fail, But CRD OK 12 HPCI&RCIC Fail, CRD Recoverable 11 HPCI,RCIC& CRD Fail	1 ADS Fail 2 ADS Available 3 ADS Success S ADS Special Battery Depletion Considerations 43333 Low Press. Systems Working or Available 43233 Low Press. Systems Working or Available, Condensate Recoverable	33333 Low Press. Systems Available 33233 Low Press. Systems Available Condensate Recoverable 22222 Low Press. Systems Recoverable 22122 Low Press. Systems Recoverable Except Condensate Fails 11131 Low Press. Sys. Fail Except HPSW 33113 Low Press. Systems Available Except Condensate & HPSW Fail 13113 Low Press. Systems Fail Except RHR & CSS	122 No Venting & Isolation Fails 1X2 No Venting, Random Isolation Failures 4X2 Wetwell Vented in ATWS, Random Isolation Failures 1Y2 No Venting, Mostly Random Failures -Some Sequence Dependence XX2 Random Failures XY2 Mostly Random Failures -Some Sequence Dependence

Table 4.11-3. Interim Peach Bottom Plant Damage States

PDS#	PDS Vector	Contributing Accident Sequence Cut Sets[1]
1.	1-322-2-13-3-13113-XX2	4(1-3)+12(1-3)
2.	4-322-1-13-3-13113-XX2	11(1)+10(1)+16(1)
3.	4-322-1-13-3-11131-XX2	11(2)
4.	4-222-1-13-3-11131-XY2	9(2,3)
5.	4-222-1-13-3-13113-XY2	9(1,4,5)
6.	4-211-1-12-3-22222-122	18(1)
7.	4-211-2-12-1-22222-122	5(1)
8.	4-21S-1-22-3-22222-122	6(1-57)
9.	4-21S-2-22-S-22222-122	1(1-130)+7(1-79)
10.	5-322-1-23-2-33333-XX2	17(1-5)
11.	5-322-1-23-2-33333-XX2	8(1-6)*ADS)
12.	5-322-1-23-3-43333-1X2	8(1-6*/ADS*VENT)
13.	5-322-1-23-3-43333-4X2	8(1-6*/ADS*/VENT)
14.	5-322-2-23-2-33333-XX2	2(1-9*ADS)+14(1-4*ADS)+ 15(1-4*ADS)
15.	5-322-2-23-3-43333-1X2	2(1-9*/ADS*VENT)+14(1-4*/ADS*VENT) +15(1-4*/ADS*VENT)
16.	5-322-2-23-3-43333-4X2	2(1-9*/ADS*/VENT)+14(1-4*/ADS*/VENT)+ 15(1-4*/ADS*/VENT)
17.	5-322-2-23-2-33333-XX2	3(1-14)
18.	5-222-2-23-2-33233-XY2	13(1-4*ADS)
19.	5-222-2-23-3-43233-1Y2	13(1-4*/ADS*VENT)
20.	5-222-2-23-3-43233-4X2	13(1-4*/ADS*/VENT)

(1) See Appendix E for more details. This column gives the cut sets for the accident sequences that go into that PDS, e.g., 7(2) means cut set #2 from accident sequence #7. Also, 13(1-4*ADS) means cut sets 1 through 4 of accident sequence 13 are all multiplied by the split fraction ADS.

(2) X in question 14 means the vent set point is not reached by the time of core damage, therefore, random or operator failure is possible later in the sequence (handled in the APET). The 1 for question 14 in PDSs 6-9 implies station blackout, so that without AC venting can not occur until AC is recovered (also handled in the APET). The 1 for question 14 as it applies to PDSs 12, 15, and 19 and the 4 applied to PDSs 13, 16, and 20 implies the venting set point is reached before core damage and random or operator failure may or may not occur.

Table 4.11-4. Final Peach Bottom Plant Damage States

PDS#	PDS Vector	Interim PDSs Included	Accident Sequence Cut Sets Involved
1.	1-322-2-13-3-13113-111	1	4(1-3)+12(1-3)
2.	4-622-1-13-3-13113-111	2,5	9(1,4,5)+11(1)+10(1)+16(1)
3.	4-622-1-13-3-11131-111	3,4	9(2,3)+11(2)
4.	4-211-6-12-1-22222-111	6,7	5(1)+18(1)
5.	4-212-6-22-3-22222-111	8,9	1(1-130)+6(1-57)+7(1-79)
6.	5-322-6-23-2-33333-111	10,17	3(1-14)+17(1-5)
7.	5-322-1-23-6-33333-611	11,12,13	8(1-6)
8.	5-322-2-23-6-33333-611	14,15,16	2(1-9)+14(1-4)+15(1-4)
9.	5-222-2-23-6-33233-611	18,19,20	13(1-4)

Notes:

1) Venting may be required before core damage for PDSs 7, 8, and 9. Venting is not possible until AC power is restored for PDSs 4 and 5. For all other PDSs venting may fail due to operator or random failure, but is not required until after core damage occurs, so it is handled in the APET.

2) Containment isolation failures were either unlikely or not possible in the defined PDSs.

3) The digit 6 was used for several questions in the sixteen character PDS vector, since it had not been used previously, to depict several conditions depending on the questions as explained below:

Question 2 - If LOSP has occurred, all systems respond the same. The APET will handle any differences using TEMAC 4 to split the cut sets.

Question 5 - Differences caused by a stuck open SRV are handled in the APET using split fractions.

Questions 8 and 14 - The difference is manual ADS, which can be handled in the APET using split fractions. The low pressure response is also handled by the APET, depending on primary system pressure results. The venting response depends on whether or not there is a quasi-stable state with low pressure injection working.

Table 4.11-5. Core Damage Frequency by Plant Damage States

PDS#	PDS Vector	Number of Cut Sets and Frequency Before Battery Depletion Added		Number of Cut Sets and Frequency After Battery Depletion Added	
1.	1-322-2-13-3-13113-111	6	2.13E-7	No Change, Except as Noted Below	
2.	4-622-1-13-3-13113-111	6	2.27E-7		
3.	4-322-1-16-3-11131-111	3	5.83E-9		
4.	4-211-6-12-1-22222-111	2	1.95E-7		
5.	4-212-1-22-3-22222-111	266	6.95E-7	1330	1.07E-6
6.	5-322-6-23-2-33333-111	19	2.82E-7		
7.	5-322-1-23-6-33333-611	6	1.07E-7		
8.	5-322-2-23-6-33333-611	17	1.47E-6		
9.	5-222-2-23-6-33233-611	4	4.43E-8		
	Total Point Estimates	329	3.24E-6	1393	3.62E-6

Note: In accounting for battery depletion in more detail, the total number of cut sets was expanded from 329 to 1393 and the total core damage frequency increased from 3.24E-6 to 3.62E-6.

These plant damage state groupings of accident sequence cut sets are input to the uncertainty analysis (Section 4.12) and to the back-end analysis. A brief description of each plant damage state is provided in the next section.

4.11.4 Description of Plant Damage States

Each of the plant damage states is described below in words using the groups of questions previously delineated in Section 4.5.

PDS-1 1-322-2-13-3-13113-111

This PDS is composed of two accident sequences, A-V2V3 and S1-V2V3V4NU11. A-V2V3 is a large LOCA initiator followed by immediate failure of the LPCS and LPCI systems (other high or low pressure systems can not mitigate this sequence in time or fail as a result of the initiator). The result is early core damage. S1-V2V3V4NU11 is a medium LOCA initiator followed by initial success of HPCI. HPCI fails soon thereafter due to low vessel pressure and LPCS, LPCI, and HPSW (insufficient time or operator error) all fail (other systems fail or can not mitigate the LOCA). This again results in early core damage. CRD is working in both sequences and all containment heat removal is working. LPCI has failed due to miscalibration of the level sensors and the injection valves can not be opened; this fails HPSW also. Venting will work if needed, but will not be demanded before CD.

PDS-2 4-622-1-13-3-13113-111

This PDS is composed of four sequences: T3A-P2V234NU11, T3B-P2V234NU11, T2-P2V234NU11, and T1-P2V234NU11B. This PDS is similar to PDS-1. Different transient initiators with subsequent failure of SRVs result in the equivalent of an intermediate LOCA (P2). The sequences then follow the same pattern as in PDS-1. Containment heat removal is working, but steam is directed through the SRVs to the suppression pool, not to the drywell as in a LOCA. HPCI works early, but fails on low vessel pressure, and all other high or low pressure systems are failed. The low pressure injection valves fail which, in turn, fail LPCI, LPCS, and HPSW. This results in early core damage. Venting will not occur before CD.

PDS-3 4-622-1-13-3-11131-111

This PDS is composed of two sequences: T3B-P2V234NU11, and T1-P2V234NU11B. This PDS is similar to PDS-1. These transient initiators with subsequent failure of SRVs result in the equivalent of an intermediate LOCA (P2). The sequences then follow the same pattern as in PDS-1. Containment heat removal is not working, but steam is directed through the SRVs to the suppression pool, not to the drywell as in a LOCA. CRD is not working in some cut sets. This PDS is also similar to PDS-2, except that containment heat removal is not working and HPSW has failed by operator error or can not be used in time (makes it similar to PDS-1).

PDS-4 4-211-6-12-1-22222-111

This PDS is composed of two sequences: T1-P1BU11U21, and T1-BU11U21. The first sequence is a station blackout, followed by one stuck open SRV. High pressure injection fails and early core damage results. Vessel pressure remains low; DC power has also failed. For the second sequence, there is no stuck open SRV, so the vessel is at high pressure. Venting is not possible unless AC is restored. AC systems are available with recovery of AC power.

PDS-5 4-212-6-22-3-22222-111

This PDS is composed of three sequences: T1-P1BNU11, T1-BNU11, and T1-BU11NU21. These sequences involve a station blackout with or without one stuck open SRV and initially successful operation of HPCI or RCIC. Battery depletion may or may not occur before core damage. The vessel remains at low pressure if a SRV is stuck open, otherwise, it repressurizes on loss of DC. AC systems are available on recovery of AC power. Venting not possible until AC is restored.

PDS-6 5-322-6-23-2-33333-111

This PDS is composed of two sequences: T3C-CU11X, and T3A-CU11X. This is an IORV or a loss of AC bus with failure to scram, SLC works, HPCI works initially, and the vessel is not manually depressurized. HPCI fails on high suppression pool temperature. The containment is not vented before core damage, but venting is operable. TEMAC 4 is used to calculate a split fraction for the SRV open or not.

PDS-7 5-322-1-23-6-33333-611

This PDS is composed of one sequence: T3C-C-SLC. This is an IORV with failure to scram and SLC also fails. HPCI fails on high suppression pool temperature, the reactor is: a) not manually depressurized, or b) is manually depressurized to use low pressure systems. If a) then early CD results and venting will not occur before CD. If b) then the containment will pressurize until venting, containment failure, or SRV reclosure on high containment pressure. In all b) cases, the low pressure injection systems will fail due to low NPSH or harsh environments and CD will result. Venting will be tried before CD. The CRD system is working in all cases.

PDS-8 5-322-2-23-6-33333-611

This PDS is composed of three sequences: T3A-C-SLC, T3B-C-SLC, and T2-C-SLC. This is a loss of AC bus or PCS with failure to scram, and SLC also fails. HPCI fails on high suppression pool temperature, and the reactor is a) not manually depressurized or b) is manually depressurized to use the low pressure systems. If a) then early CD results and venting will not occur before CD. If b) then the containment will pressurize until either venting, containment failure, or SRV reclosure on high containment pressure. In all b) cases, the low pressure injection systems will fail due to low NPSH or harsh environments and CD will result. Venting will be tried before CD. The CRD system is working in all cases. This PDS is similar to PDS-7, except that a SRV is not stuck open.

PDS-9 5-222-2-23-6-33233-611

This PDS is composed of one sequence: T1-C-SLC. This is a LOSP with failure to scram and SLC fails. HPCI fails on high suppression pool temperature and the reactor is: a) not manually depressurized, or b) is manually depressurized to use the low pressure systems. If a) then early CD results and venting will not occur before CD. If b) then the containment will pressurize until venting, containment failure, or SRV reclosure on high containment pressure. In all b) cases, the low pressure injection systems will fail due to low NPSH or harsh environments and CD will result. Venting will be tried before CD. The CRD system is working in all cases. This PDS is similar to PDS-8 except for LOSP.

4.12 Uncertainty Analysis

There are various sources of uncertainty in the numerical results of this study. This section discusses the sources and treatment of uncertainty for the Peach Bottom study. Uncertainty in the analysis comes from every step of the process. Uncertainty can be both qualitative and quantitative in nature, and arise from the data base used to determine parameter values, modeling assumptions, and completeness of the analysis. Uncertainty in the models and model parameters is propagated through the quantification process so that the core damage and risk estimates incorporate the uncertainties of the analysis.

4.12.1 Sources and Treatment of Uncertainties

Two basic types of uncertainty were addressed in the Peach Bottom study: parameter value uncertainty and modeling uncertainty. The parameters of interest are those of the probability models for the basic events of the logic models and include failure rates, component unavailabilities, initiating event frequencies, and human error probabilities. The essential difference between the parameter value uncertainty and modeling uncertainty is the following: parameters can take on any of a continuous range of values and the fact that there is uncertainty as to which value is correct does not change the structure of the logic model. In general, a few discrete modeling hypotheses are proposed and the different hypotheses may well lead to different logic models.

Sources of parameter uncertainty include lack of data on component failure modes, interpretation of data and component performance records, and the use of industry-wide data for plant specific analyses. Modeling uncertainty reflects limitations of knowledge regarding phenomenological impacts on component performance, physical propagation of accident progression through the plant systems, and human response to abnormal conditions.

Parameter value uncertainties have been handled by defining a probability distribution on the value of each parameter such that the nth percentile of the distribution represents the value below which the analyst has a degree of belief of n/100 that the true value lies. This subjective approach to the representation of uncertainty makes the propagation of parameter value uncertainty through the evaluation of the bottom line results mathematically straightforward using constrained Monte Carlo (e.g., Latin Hypercube) or other sampling techniques. The uncertainty ranges characterized by the distributions vary in origin. If the estimates are based on plant specific data, the range should be characteristic of the statistical uncertainty. If the estimates are generic (or non-plant specific) the range should be characteristic of those factors which may affect the failure properties of the component in the different uses and environments from which the data for the estimates have been gathered. In this instance, the range should include plant-to-plant variation.

Modeling uncertainties are treated similarly by defining discrete or continuous probability distributions over the different modeling hypotheses.

Previous studies have incorporated modeling uncertainties into their analyses by performing sensitivity analyses to identify which modeling hypotheses are most significant. The method for the Peach Bottom analysis was to use expert judgment to elicit from a panel of experts a model which weighs the various hypotheses for each modeling uncertainty, and then to propagate the model uncertainty through the accident sequence so as to include the various hypotheses in the final overall core damage and risk estimates. The expert elicitation process used in the NUREG-1150 plant analyses is described in NUREG/CR-4550 [44], Revision 1, Volume 2.

It should be noted that the separation between parameter and modeling uncertainty is not always clear. Parameter uncertainty is uncertainty in the value of a parameter due to variability in the data. Such factors as number of components, demands, failures, and the time between component maintenance all impact the estimate of parameter value. However, to incorporate these factors into a quantification of parameter uncertainty, models must be selected, thus introducing modeling uncertainty into the measure of parameter uncertainty. Here, modeling uncertainty involves issues such as the choice of parameter distribution (e.g., lognormal, maximum entropy, or Bayesian update of a noninformed prior), the definition of component boundaries and classification of data into component failures and succeses.

4.12.2 Development of Parameter Distributions.

Probabilistic distributions for parameter values were developed from several sources of information including plant specific data, industry-wide data summaries and analyses, past PRAs, formal expert opinion elicitation and informal expert opinion elicitation.

If sufficient plant specific data was available for a particular component failure mode, then the estimate and uncertainty model for that parameter value were based on statistical analysis of the data. Often, sufficient plant data did not exist for certain parameters, so generic estimates and uncertainty models based on industry data were used for many parameter values. The primary body of generic parameter estimates used for the Peach Bottom analysis in the ASEP Generic Data Base in the methodology document for the supporting analysis of NUREG-1150, NUREG/CR-4550, Revision 1, Volume 1 [2]. This set of generic parameter uncertainty models is based on extensive review of data analyses, such as LER summaries, NPRDS reports, common cause data analyses by Fleming [42] and Atwood [38,39,40], and generic parameter estimations developed for the NRC sponsored Risk Methods Integration and Evaluation Program (RMIEP) [45].

Informal or project staff expert opinion elicitations documented in NUREG/CR-4550, Revision 1, Volume 2, Part 2, were used to assess parameter value uncertainties which could not be obtained from plant specific or generic data.

Loss of offsite power recovery parameters and initiating event frequencies were modeled by industry data with a composite statistical model which combined probability models for plant centered, grid, and weather related losses together. The model was adjusted to be site specific for Peach Bottom, and is documented in NUREG/CR-5032 [26]. The results for Peach Bottom are shown in Appendix D.

Human error probabilities and uncertainties were developed by applying the rules for Human Reliability Assessment (HRA) from NUREG/CR-1278 [25]. These rules recommend using lognormal distributions to model HRA parameter uncertainty; however, some adjustment to this recommendation had to be made. The rules generated probabilistically incorrect distributions for several parameters. Using the mean and error factor recommended for certain HRA results, lognormal distributions were developed, which had probability quantiles greater than 1.0. For these parameters, the distribution was changed to a maximum entropy distribution, with the maximum value defined as either 1.0 or the mean multiplied by the range factor, whichever was less. The minimum value was defined by dividing the mean by the range factor.

4.12.3 Elicitation of Expert Judgment

Modeling uncertainty was treated using the elicitation of expert judgment. This process and its results are discussed in Volume 2 of NUREG/CR-4550 [44]. The elicitation of expert judgment was done in two phases. The first phase was a formal process where a panel of nationally recognized PRA experts were convened to assess the ten most significant modeling issues. The second phase was a less formal process where the project staff were elicited. Issues not covered by the expert panel, but still deemed as significant were put before the analysts working on the various plant analyses. The informal elicitations followed the same methods and rules as the expert panel process.

The formal expert panel elicitations are documented in Volume 2, Part 1 of NUREG/CR-4550, Revision 1. Among the ten issues reviewed by the panel, none effected the Peach Bottom accident sequence analysis. However, there are issues which are relevant to the Peach Bottom containment event tree analysis. The informal elicitation process did involve several issues of interest to the Peach Bottom front-end analysis. These were:

- Common Cause Beta Factor Uncertainty Ranges
- Common Cause Factors for AOVs
- Station Battery Depletion Time
- Conowingo Hydrogenerator Recovery of AC Power

These issues and their resolutions are documented in Volume 2 Part 2 of NUREG/CR-4550, Revision 1, but are briefly summarized below.

The uncertainty ranges for the common cause Beta factors used in the plant analyses were scrutinized by reviewing the common cause data in the Fleming common cause analysis, EPRI-NP-3967 [42], for misclassification of data. The conclusion of the elicitation was that the existing common cause uncertainty models accounted for any reasonable misclassification of the data.

The Fleming report did not have an analysis for AOVs, so the uncertainty model for common cause for AOVs was assessed as part of the informal elicitations. Based on the results in EPRI-NP-3967 for several types of valves and valves as a total family of components, a common cause Beta factor model of a lognormal distribution with a mean of 0.1 and an error factor of 3 was developed.

Station battery depletion time was assessed for each plant individually. A cumulative probability distribution was developed to model the failure probability of the station batteries versus time for station blackout sequences. Because the batteries could fail over a range of times, the uncertainty of battery failure time was incorporated into the accident sequence model by discretizing the battery failure distribution into four areas, with each area centered at equal increments of time, or time parameters. The curve was discretized only out to 10 hours. After 10 hours, core damage will result due to other failures regardless of the state of the batteries. Figure 4.12-1 shows the discretization of the battery failure curve. The total probability of each block was calculated and assigned to the mean time of the block. The four time parameters were incorporated into the accident sequence models by being linked together with fault tree "OR" logic, and replacing a single "Battery-Fails" event in the fault trees with the set of four mutually exclusive linked time parameters. The probability associated with each time parameter was used in the point estimate calculations, but for the uncertainty analysis the time parameters were used as switches, always taking on the value of either 0.0 or 1.0. The number of times each time parameter was sampled at 1.0 was proportional to its probability. Furthermore, the sampling of the time parameters was correlated so that, for each sample of the accident sequence model, only one of the time parameters would be valued at 1.0, with the others at 0.0. This correlation was imposed on the sampling because, although battery failure may occur over a range of time, it can only occur once during an accident.

The implementation of the battery depletion issue discussed above is the substitution of five terms into nine separate cases. The following expansion is used in each case:

INJ-FAILS+BAT-DEP-3HR+BAT-DEP-5HR+BAT-DEP-7HR+BAT-DEP-9HR.

The resulting specific substitutions are given in Table 4.12-1. When these substitutions are made to the applicable cut sets in the four affected accident sequences 1188 additional cut sets are formed which account for battery depletion.

Conowingo dam is a hydroelectric generator close to Peach Bottom. It is capable of supplying power to Peach Bottom as a source of power restoration for loss of power incidents. Because there are procedures at Peach Bottom to start up Conowingo and connect it to the plant, Conowingo was considered as a potential recovery option for loss of power incidents. Conowingo cannot supply power for approximately 40 minutes after procedures are initiated. Sequences with failure to restore power within 60 minutes results in core damage. The informal elicitation

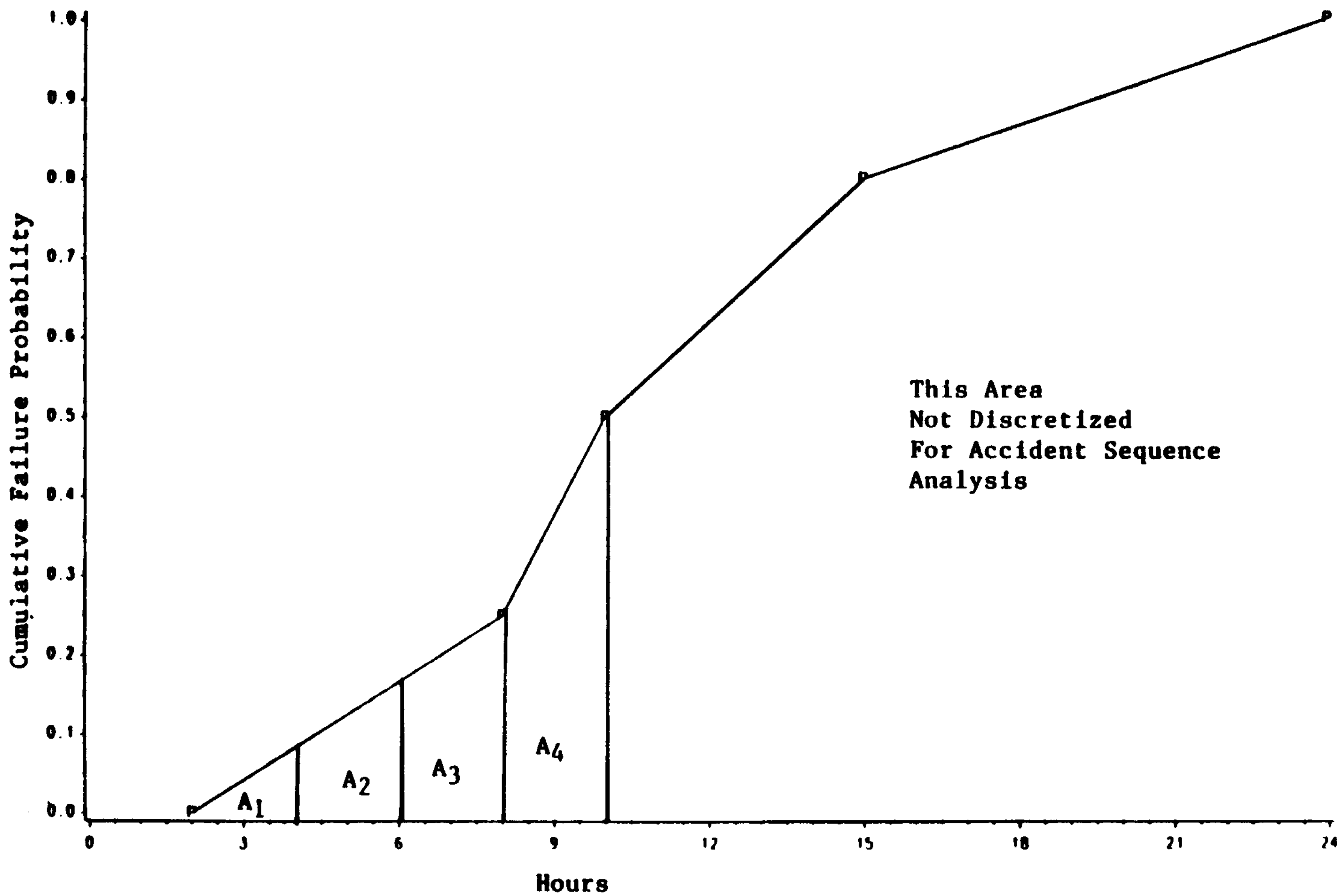


Figure 4.12-1. Battery Depletion Time - Peach Bottom

Table 4.12-1. Battery Depletion Cut Set Substitutions

1. <u>LOSPNR13HR</u> =

   LOSPNR13HR*INJ-FAILS +
   LOSPNR5HR*BAT-DEP-3HR +
   LOSPNR7HR*BAT-DEP-5HR +
   LOSPNR9HR*BAT-DEP-7HR +
   LOSPNR12HR*BAT-DEP-9HR .

2. <u>LOSPNR13HR*DGCCFNR12HR</u> =

   LOSPNR13HR*DGCCFNR12HR*INJ-FAILS +
   LOSPNR5HR*DGCCFNR3HR*BAT-DEP-3HR +
   LOSPNR7HR*DGCCFNR5HR*BAT-DEP-5HR +
   LOSPNR9HR*DGCCFNR7HR*BAT-DEP-7HR +
   LOSPNR12HR*DGCCFNR9HR*BAT-DEP-9HR .

3. <u>LOSPNR13HR*DGHWNR12HR</u> =

   LOSPNR13HR*DGHWNR12HR*INJ-FAILS +
   LOSPNR5HR*DGHWNR3HR*BAT-DEP-3HR +
   LOSPNR7HR*DGHWNR5HR*BAT-DEP-5HR +
   LOSPNR9HR*DGHWNR7HR*BAT-DEP-7HR +
   LOSPNR12HR*DGHWNR9HR*BAT-DEP-9HR .

4. <u>LOSPNR13HR*DGMANR12HR</u> =

   LOSPNR13HR*DGMANR12HR*INJ-FAILS +
   LOSPNR5HR*DGMANR3HR*BAT-DEP-3HR +
   LOSPNR7HR*DGMANR5HR*BAT-DEP-5HR +
   LOSPNR9HR*DGMANR7HR*BAT-DEP-7HR +
   LOSPNR12HR*DGMANR9HR*BAT-DEP-9HR .

5. <u>LOSPNR18HR*DGHWNR12HR</u> =

   LOSPNR18HR*DGHWNR12HR*INJ-FAILS +
   LOSPNR9HR*DGHWNR3HR*BAT-DEP-3HR +
   LOSPNR12HR*DGHWNR5HR*BAT-DEP-5HR +
   LOSPNR14HR*DGHWNR7HR*BAT-DEP-7HR +
   LOSPNR17HR*DGHWNR9HR*BAT-DEP-9HR .

6. <u>LOSPNR18HR*DGMANR12HR</u> =

   LOSPNR18HR*DGMANR12HR*INJ-FAILS +
   LOSPNR9HR*DGMANR3HR*BAT-DEP-3HR +
   LOSPNR12HR*DGMANR5HR*BAT-DEP-5HR +
   LOSPNR14HR*DGMANR7HR*BAT-DEP-7HR +
   LOSPNR17HR*DGMANR9HR*BAT-DEP-9HR .

Table 4.12-1. Battery Depletion Cut Set Substitutions (Cont.)

7. LOSPNR18HR*DCHWNR18HR =

LOSPNR18HR*DCHWNR18HR*INJ-FAILS +
LOSPNR9HR*DCHWNR9HR*BAT-DEP-3HR +
LOSPNR12HR*DCHWNR12HR*BAT-DEP-5HR +
LOSPNR14HR*DCHWNR14HR*BAT-DEP-7HR +
LOSPNR17HR*DCHWNR17HR*BAT-DEP-9HR .

8. LOSPNR18HR =

LOSPNR18HR*INJ-FAILS +
LOSPNR9HR*BAT-DEP-3HR +
LOSPNR12HR*BAT-DEP-5HR +
LOSPNR14HR*BAT-DEP-7HR +
LOSPNR17HR*BAT-DEP-9HR .

9. LOSPNR18HR*DGACTNR12HR =

LOSPNR18HR*DGACTNR12HR*INJ-FAILS +
LOSPNR9HR*DGACTNR3HR*BAT-DEP-3HR +
LOSPNR12HR*DGACTNR5HR*BAT-DEP-5HR +
LOSPNR14HR*DGACTNR7HR*BAT-DEP-7HR +
LOSPNR17HR*DGACTNR9HR*BAT-DEP-9HR .

yielded a probability of failure to restore power via Conowingo within 40 to 60 minutes of 0.60. This probability was applied only to the recovery of plant centered power loss model. It was assumed that grid and weather related power losses would disable Conowingo as well as Peach Bottom

4.12.4 Quantification of Accident Sequence Uncertainty

The uncertainty of the parameter values was propagated through the accident sequence models using two computer codes. A Latin Hypercube Sampling (LHS) algorithm was used to generate the samples for all of the parameter values. The code is documented in NUREG/CR-3624 [43]. The Top Event Matrix Analysis Code (TEMAC) was used to quantify the uncertainty of the accident sequence equation using the parameter value samples generated by the LHS code. TEMAC is documented in NUREG/CR-4598 [28].

LHS is a constrained Monte Carlo technique which forces all parts of the distribution to be sampled. The LHS code is also flexible in that it can sample a variety of random variable distributions. Furthermore, parameter distributions for similar events were correlated. For example, if two similar components (e.g., MOV-FTO-XX and MOV-FTO-YY) are modeled from the same probability distribution, then the sampling of these two distributions is perfectly correlated, meaning the same value is used for both events in a given sample member. For basic events which are modeled with very similar but slightly different distributions (e.g., MOV XX fails to remain closed for 100 hours and MOV YY fails to remain closed for 200 hours), the LHS code permits an induced correlation between the samples. However, LHS does not allow the correlation coefficient for this case to be equal to 1.0. LHS did permit sampling with a coefficient of 0.99 in these cases.

TEMAC uses the LHS parameter samples and the accident sequence equations (cut sets) as input to quantify the core damage estimates. TEMAC generates a sample of the accident sequence frequency, a point estimate of the frequency, and various importance measures and ranking for the base events. The TEMAC users manual, NUREG/CR-4598, describes the code's calculations and output in detail. A brief description of the calculations generated for Peach Bottom is given below. These calculations for Peach Bottom are displayed in Section 5.0 of this report.

<u>Descriptive Statistics for the Top Event</u>

The following descriptive statistics are considered in TEMAC for the top event frequencies and each accident sequence, and plant damage state:

- Size of the LHS sample
- The nominal estimate of the top event (quantified with all base events and initiating events set equal to a user-specified nominal value)
- Mean of the sample
- Standard deviation of the sample
- 0.5, 0.25, 0.50, 0.75, and 0.95 quantiles of the sample

The entire sample of the top event generated by TEMAC is plotted to show the cumulative probability distribution and probability density functions of the frequency. These are given in Section 5.1.

Risk Reduction by Basic and Initiating Events

Risk reduction is a measure of the change in top event frequency due to a proportional change in the base event probability. This measure yields a ranking of the base events by importance, or contribution, to top event frequency. The risk reduction figure of merit is the potential reduction in the top event frequency if a base event probability is quantified as 0.0, or perfectly reliable. This measure is useful in identifying which components, human actions, maintenance practices, and initiating events should be the focus of efforts to improve reliability and reduce risk. Uncertainty intervals for risk reduction are also calculated. These are the 0.05 and 0.95 quantiles of the risk reduction calculations generated by performing n such calculations over the LHS matrix of base and initiating events samples (n being the size of the Latin hypercube sample). The risk reduction uncertainty intervals show the uncertainty in a base event's contribution to risk due to the uncertainty of the top event frequency. Initiating events are ranked separately from base events.

Risk Increase by Base Event

Risk increase (sometimes called risk achievement) is the increase in risk that results should a particular base event's probability be set equal to 1.0. This measure is meaningful only for probabilities and is not used for initiating events. This measure is useful to assess which elements of the risk model are the most crucial for maintaining risk at current levels. An increase in component reliability or human error probability for risk increase dominant events will maximize the risk for a particular accident sequence model. Uncertainty intervals for risk increase are calculated as with risk reduction.

Uncertainty Importance

The uncertainty importance measure focuses on the contribution to the variance of the frequency of the top event attributable to each of the base and initiating events that jointly constitute the top event. In particular, if F is a composite of these events, where F represents the frequency of the top event, it is reasonable to expect a reduction in the Var(F) if the value of an event, X_j, is known with certainty. If X_j is known with certainty, then the variance of F is conditional on the specific value of X_j and is denoted by $Var(F|X_j)$. Moreover, the conditional reduction in the variance of F attributable to ascertaining the true value of the event X_j is expressed as

$$Var(F) - Var(F|X_j).$$

The unconditional variance of F, Var(F), can be expressed in terms of the expected value of the conditional variance, $E_{X_j}[Var(F|X_j)]$, and the variance of the conditional expectation, $Var_{X_j}[E(F|X_j)]$, as follows.

$$\mathrm{Var}(F) = E_{X_j}[\mathrm{Var}(F|X_j)] + \mathrm{Var}_{X_j}[E(F|X_j)]$$

or

$$\mathrm{Var}(F) - E_{X_j}[\mathrm{Var}(F|X_j)] = \mathrm{Var}_{X_j}[E(F|X_j)]$$

The square root of the left-hand side of the above equation is the measure referred to as uncertainty importance for event X_j.

The uncertainty importance measure requires calculating the variance of a conditional expectation of a random variable, $\mathrm{Var}_{X_j}[E(F|X_j)]$. If the random variable has a long-tailed distribution, such as occurs when lognormal distributions are used with large error factors, then its variance is extremely difficult to estimate. This estimation problem is directly attributable to the scale of the numbers involved. The scaling problem can be overcome by performing uncertainty importance calculations based on a logarithmic scale for the top event frequencies. The log scale produces a reliable ordering of the events and expresses the results in terms of log-based risk.

However, the log-based uncertainty importance calculations do not readily translate back to a linear scale; thus, the uncertainty importance calculations in TEMAC are given only in terms of log-based risk. TEMAC does, however, provide the analyst with information that aids in the interpretation of the results of the log-based uncertainty importance calculation. This is accomplished by computing the ratio, $R_{.05}$, of the .05 quantile of the distribution of the top event frequency when X_j is held constant at its mean value, to the .05 quantile of the top event frequency when X_j is not held constant. A similar ratio, $R_{.95}$, is calculated by TEMAC for the .95 quantiles.

If $R_{.05}$ and $R_{.95}$ are both greater than 1.0, then the distribution of the frequency of the top event with X_j held constant at its mean value has shifted to the right, or shows an overall higher level of risk. On the other hand, if $R_{.05}$ and $R_{.95}$ are both less than 1.0, then the distribution of the frequency of the top event with X_j held constant at its mean value has shifted to the left, or shows an overall lower level of risk. If $R_{.05} > 1.0$ and $R_{.95} < 1.0$, then the overall uncertainty in the distribution of the top event frequency has decreased. Likewise, if $R_{.05} < 1.0$ and $R_{.95} > 1.0$, then the overall uncertainty in the distribution of the top event frequency has increased.

A basic difference between uncertainty importance and risk reduction and increase should be noted. Risk increase and risk reduction importance measures illustrate the impact on risk due to changes in an individual base event's frequency. The uncertainty importance measure illustrates the change in the uncertainty of the risk estimate due to a reduction in the uncertainty of random variable estimates used to model base event parameters. As such it can relate to more than a single base event in the case where events are correlated to a common distribution. Risk reduction and increase are useful in evaluating how specific plant components and procedures may impact risk. Uncertainty importance is useful in pointing out weaknesses in particular parameter estimates which can be used to model several base event estimates.

Presentation of Cut Set Results

TEMAC prints out a ranked listing of the cut sets of the top event equation. The cut sets are ranked by their frequency. For each cut set, TEMAC shows the number of the cut set (this is simply determined by the order in which the cut sets are read into TEMAC, there is no implication between rank and number), the order of the cut set (number of events in the cut set), the frequency of the cut set, the cumulative normalized cut set frequency, and a listing of the cut set. The cumulative normalized cut set frequency for a particular cut set shows what fraction of the top event frequency is modeled by that cut set and all other higher ranked cut sets. This measure is convenient for review and screening of a top event equation. It tells the analyst which cut sets of the equation can be eliminated from further consideration and still retain some minimum threshold of the top event frequency (e.g., 99%). TEMAC also writes the top event equation with the cut sets order by frequency to an output file. This ranked structuring of the input equation is useful if it is desired to screen out low frequency cut sets from further TEMAC analyses.

5. RESULTS

The final results of the Revised Peach Bottom Probabilistic Risk Assessment (PRA) for NUREG-1150 are presented in this section. The overall results are discussed first, followed by a brief discussion of the top cut sets, and then a breakdown by accident sequence and plant damage state. The top or dominant cut sets are described for their corresponding accident sequences and not repeated for each plant damage state. The importance measures are presented next, relative to the overall results. Cut sets and importance measures are given in Appendix F for every accident sequence and plant damage state. The results presented in this section emphasize the combined results. The last subsection is a comparison of the results with WASH-1400.

5.1 Characterization of Core Damage Frequency and Uncertainty

The models and data producing the results provided here constitute the most representative analysis of the Peach Bottom Unit 2 plant that the analysts could achieve. All modeling issues were reexamined since the original analysis, and their uncertainties were integrated into a single uncertainty analysis. Expert judgment was elicited on important issues to help determine the appropriate uncertainty ranges.

The total core damage frequency (CDF) and uncertainty presented in this subsection result from a combined analysis of all cut sets from all the accident sequences. The mean core damage frequency for the Peach Bottom Unit 2 plant is 4.5E-6 per reactor year for the internal initiating events. The cumulative distribution function is given in Figure 5-1. The probability density function (PDF) is approximated in Figure 5-2. Both figures were generated from the TEMAC sample of 1000.

The total frequency includes all the accident sequences with core damage frequencies greater than 1E-8. The corresponding point estimate is 3.62E-6 per reactor year. There were approximately 50 accident sequences with point estimates in the range of 1E-9 to 1E-8 that were eliminated or not fully quantified. If each of these 50 sequences were given a 3E-9 average point estimate frequency, this results in another 1.5E-7 not accounted for in the overall frequency. Thus, it appears that approximately 3% of the core damage frequency may have been omitted. The purpose of using point estimates here is that the accident sequences eliminated were in terms of point estimates. This is a reasonable approximation for the savings in resources achieved by not expanding the analysis.

The descriptive statistics for the total Peach Bottom Unit 2 core damage frequency are:

Sample Size	1000
Mean	4.50E-6
STD DEV	1.54E-5
LOWER 5%	3.46E-7
LOWER 25%	9.21E-7
MEDIAN	1.85E-6

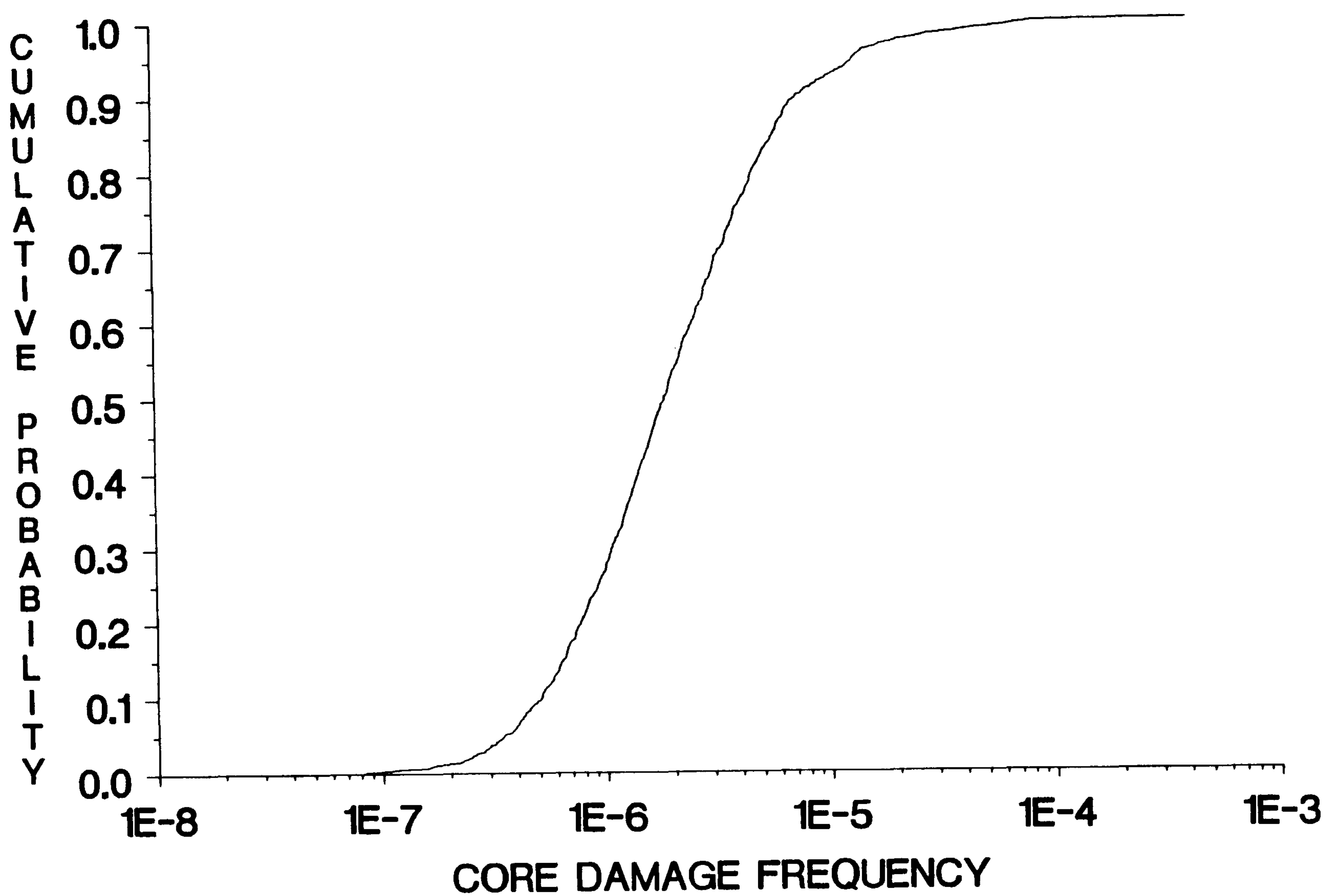


FIGURE 5-1. UNCERTAINTY DISTRIBUTION FOR PEACH BOTTOM CORE DAMAGE FREQUENCY

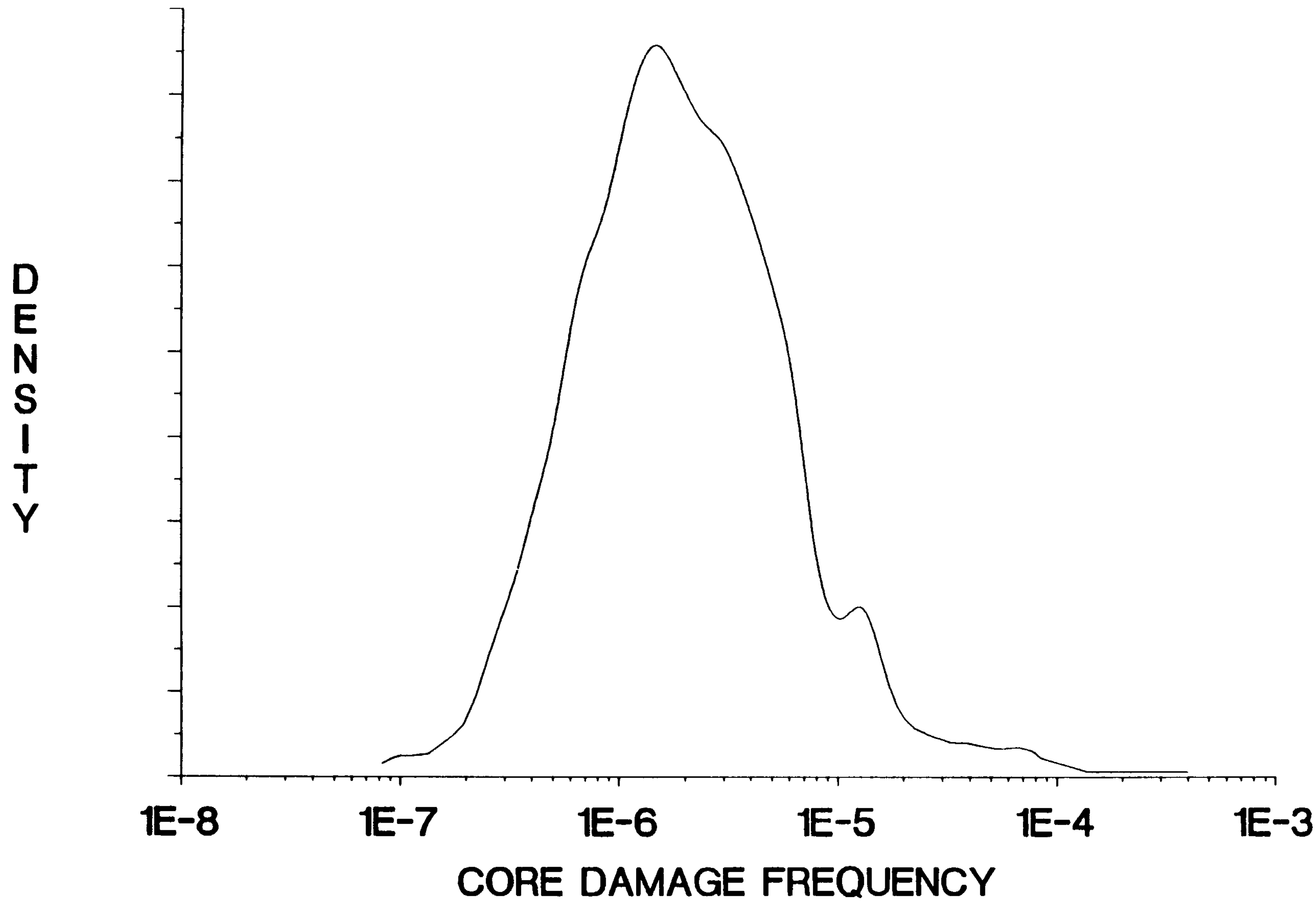


FIGURE 5-2. DENSITY ESTIMATION FOR PEACH BOTTOM CORE DAMAGE FREQUENCY

UPPER 25% 3.88E-6
UPPER 5% 1.33E-5

The corresponding statistics are shown in Appendix F for each accident sequence and plant damage state. There were 18 accident sequences with core damage frequencies greater than 1E-8. Two accident sequences contributed 68% of the total core damage frequency. The first is T1-BNU11, which is a loss of offsite power (LOSP) transient (T1) with subsequent station blackout (B) and eventually battery depletion causing loss of all injection. High pressure coolant injection is successful (NU11) early in the accident sequence. The second sequence is T3A-C-SLC, which is a transient with the power conversion system initially available (T3A), but failure to scram (C) and failure of the standby liquid control system (SLC) cause core damage. All 18 accident sequences are discussed in more detail in Section 5.2.

In order to perform the back-end analysis, the accident sequences must be placed in plant damage states corresponding to the necessary input for the accident progression event tree. This resulted in nine plant damage states, which are discussed in Section 5.3.

Each accident sequence has one or more cut sets. A cut set is one specific sequence of events within the more general definition given by the accident sequence that leads to core damage. There can be a number of cut sets, each causing the same end result, but by different component failures or events occurring given the accident initiator. It is very informative to consider the top or dominant cut sets for the total core damage frequency ranked in order of contribution. There are 1393 cut sets considered in the Peach Bottom analysis. A tabulation of the contribution to core damage frequency versus the accumulated number of cut sets starting with the highest contributor is given below.

% of Total Core Damage Frequency	Cumulative Number of Cut Sets	
50%	6	
60%	11	
70%	24	
75%	38	All Cut Sets >1E-8
80%	62	
85%	103	
90%	174	
95%	350	
99%	873	
100%	1393	

Clearly, a few cut sets dominate. The top 20 cut sets are given in Table 5-1. The point estimate frequency, % of the total point estimate frequency, and corresponding accident sequence and plant damage state are given for each of the cut sets. Point estimates must be used here, since means are not calculated for the cut sets. Mean frequencies are calculated for the total core damage frequency and each accident sequence and plant damage state. While these cut sets are described in more

Table 5-1. Top Peach Bottom Cut Sets Contributing to Core Damage Frequency

	Pt. Est. Freq.	% of Total Pt. Est. Frequency	Cut Set*	Corresponding Accident Sequence and PDS		
1.	8.0E-7	22.0	IE-T3A*RPSM*SLC-XHE-RE-DIVER*NR	2.	T3A-C-SLC	PDS-8
2.	5.0E-7	13.8	IE-T3A*RPSM*SLC-XHE-FO-SLC*NR	2.	T3A-C-SLC	PDS-8
3.	1.8E-7	4.9	IE-T1*DCP-BAT-LF-CCF*BETA-5BAT*NR	5.	T1-BU11U21	PDS-4
4.	1.6E-7	4.4	IE-S1*ESF-XHE-MC-PRES*NR	4.	S1-V2V3V4NU11	PDS-1
5.	1.5E-7	4.2	IE-T3A*ESF-XHE-FO-DATWS*HCI-TDP-FS-20S37 *RPSM*NR	3.	T3A-CU11X	PDS-6
6.	8.5E-8	2.4	IE-T3A*RPSM*SLC-SYS-TE-SLC*NR	2.	T3A-C-SLC	PDS-8
7.	8.4E-8	2.3	IE-T1*P2*ESF-XHE-MC-PRES*NR	9.	T1-P2V234NU11B	PDS-2
8.	6.4E-8	1.8	IE-T3B*P2*ESF-XHE-MC-PRES*NR	11.	T3B-P2V234NU11	PDS-2
9.	6.1E-8	1.7	IE-T3C*RPSM*SLC-XHE-RE-DIVER*NR	8.	T3C-C-SLC	PDS-7
10.	5.3E-8	1.5	IE-T2*P2*ESF-XHE-MC-PRES*NR	10.	T2-P2V234NU11	PDS-2
11.	5.3E-8	1.5	IE-A*ESF-XHE-MC-PRES*NR	12.	A-V2V3	PDS-1
12.	5.0E-8	1.4	IE-T3A*RPSM*ESF-XHE-FO-DATWS* *HCI-TDP-MA-20S37*NR	3.	T3A-CU11X	PDS-6
13.	3.8E-8	1.1	IE-T3C*RPSM*SLC-XHE-FO-SLC*NR	8.	T3C-C-SLC	PDS-7
14.	3.7E-8	1.0	IE-T1*ESF-XHE-FO-EHS*ACP-DGN-FR-EDGB *ACP-DGN-FR-EDGC*1NJ-FAILS *DGHWNR12HR*LOSPNR18HR	1.	T1-BNU11	PDS-5
15.	2.8E-8	0.8	IE-T1*ESW-AOV-CC-CCF*BETA-3AOVS *INJ-FAILS*LOSPNR13HR	1.	T1-BNU11	PDS-5
16.	2.8E-8	0.8	IE-T1*ESW=XHE-FO-EHS*ACP-DGN-FR-EDGB *ACP-DGN-FR-EDGC*BAT-DEP-3HR *DGHWNR3HR*LOSPNR9HR	1.	T1-BNU11	PDS-5
17.	2.7E-8	0.7	IE-T3A*P2*Q*ESF-XHE-MC-PRES*NR	16.	T3A-P2V234NU11	PDS-2
18.	2.5E-8	0.7	IE-T1*RPSM*SLC-XHE-RE-DIVER*NR	13.	T1-C-SLC	PDS-9
19.	2.5E-8	0.7	IE-T3A*RPSM*ESF-XHE-FO-DATWS *HCI-TDP-FO-20S37*NR	3.	T3A-CU11X	PDS-6
20.	2.1E-8	0.6	IE-T1*ESW-XHE-FO-EHS*ACP-DGN-FR-EDGB *ACP-DGN-FR-EDGC*BAT-DEP-9HR *DGHWNR9HR-LOSPNR17HR	1.	T1-BNU11	PDS-5

* See Tables 4.8-1, 4.8-2, and 4.9-1 for definition of events

detail under the accident sequence descriptions in Section 5.2, it is interesting to note that 12 of the 18 accident sequences are represented in the top 20 cut sets.

Furthermore, all but one plant damage state is represented. The one exception is PDS-3, which has a mean core damage frequency less than 1E-8.

While there is considerable diversity in the events, accident sequences, initiating events, and plant damage states involved in the top 20 cut sets, two observations are made. First, since mechanical failure of the reactor protection system (RPSM) is in several of these cut sets, RPSM is a dominant event. This is discussed further in Section 5.4. Second, the top two cut sets contribute 22% and 14% individually to the total core damage frequency.

The cut set with the highest frequency is a transient with the power conversion system (PCS) initially available (IE-T3A) including the events of RPSM failure and standby liquid control (SLC) system unavailability due to operator failure to restore the system after testing. There is no recovery (NR) allowed due to the rapid progression of events leading to core damage. This cut set results from the T3A-C-SLC accident sequence. The C acronym designates an ATWS, which is caused by the failure of the RPS. The frequency of this cut set is 8.0E-7, and it contributes 22.0% of the total core damage frequency as shown in Table 5-1. The cut set is:

IE-T3A*RPSM*SLC-XHE-RE-DIVER*NR

where

f(IE-T3A) = 2.5E-0
P(RPSM) = 1.0E-5
P(SLC-XHE-RE-DIVER) = 3.2E-2
P(NR) = 1.0E-0

This cut set frequency is 8.0E-7, which may vary slightly in this report due to differences in round-off of the numbers. In general, the number associated with the initiating event is a frequency which can be greater than 1. The events representing the component failures or human errors are conditional probabilities. Thus, the product is referred to as a frequency. This concept is repeated for every cut set, but it is useful to describe a few examples. The reader may then reconstruct the frequency of other cut sets as desired.

The second highest frequency cut set is very similar to the one previously discussed. The only difference is that operator failure to restore the system after testing (probability of 3.2E-2) is replaced with operator failure to initiate the SLC in a timely fashion (probability of 2.0E-2). This results in a cut set core damage frequency of 5.0E-7, which contributes 13.8% of the total CDF. The third highest frequency cut set is a transient caused by loss of offsite power (IE-T1) followed by a common cause failure of multiple emergency batteries and no recovery

(NR) due to difficulties of restoring power and cooling systems without adequate DC power in time to prevent core damage. This is a station blackout (B) cut set. The high pressure injection systems HPCI (U11) and RCIC (U21) fail due to loss of DC power, resulting in failure to start and control those systems and to monitor the condition of the reactor coolant system. Without AC or DC power, all coolant injection is failed, and core damage occurs. This cut set comes from the T1-BU11U21 accident sequence. The frequency of this cut set is 1.8E-7, and it contributes 4.9% of the total core damage frequency. The cut set is:

IE-T1*DCP-BAT-LF-CCF*BETA-5BAT*NR

where

f(IE-T1) = 7.9E-2
P(DCP-BAT-LF-CCF) = 9.0E-4
P(BETA-5BAT) = 2.5E-3
P(NR) = 1.0E-0

The event DCP-BAT-LF-CCF is the common cause failure (CCF) of the DC power (DCP) system batteries (BAT) due to local faults (LF) where BETA-5BAT is the corresponding common cause factor representing the failure relationship between five batteries. The product of the four mean values given above is 1.8E-7.

The fourth highest frequency cut set is an intermediate size LOCA (IE-S1) with early success of the high pressure coolant injection (HPCI) system (NU11) and subsequent failure of all low pressure systems (V2V3V4) caused by operator miscalibration of the reactor pressure permissive sensors (ESF-XHE-MC-PRES). No recovery (NR) is considered possible due to difficulties associated with diagnosing the specific fault and the relatively short time for recovery. The frequency of this cut set is 1.6E-7 and it contributes 4.4% of the total core damage frequency. The cut is:

IE-S1*ESF-XHE-MC-PRES*NR

where

f(IE-S1) = 3.0E-4
P(ESF-XHE-MC-PRES) = 5.3E-4
P(NR) = 1.0E-0

Code names for events are very important to PRA analyses. Simple names such as E-35 for the thirty-fifth event to be named could be used. Generally, descriptive names are used so that the analyst can recognize the events in terms of the real hardware or operator actions. Table 5-2 provides a quick reference to the events discussed in Section 5. In addition, a short list of system code names is given below.

ACP AC Power System
DCP DC Power System

Table 5-2. Description of Important Events for the Peach Bottom Core Damage Frequency Results

Term	Description	Mean Value
ACP-DGN-FR-EDGZ	Emergency Diesel Generator Z Fails to Run	1.6E-2
BAT-DEP-ZHR	Batteries Deplete in Z Hours	8.3E-2
BAT-DEP-9HR	Batteries Deplete in 9 Hours	2.5E-1
BETA-3AOVS	Common Cause Factor for Three Air-Operated Valves	5.5E-2
BETA-6AOVS	Common Cause Factor for Six Air Operated Valves	3.6E-2
BETA-5BAT	Common Cause Factor for Five Batteries	2.5E-3
DCP-BAT-LF-CCF	Common Cause Failure of Batteries	9.0E-4
DGCCFNR3HR	Failure to Recover Common Cause Failure of Diesel Generators in 3 Hours	7.0E-1
DGHWNR9H2	Failure to Recover Diesel Generator Hardware in 9 Hours	5.8E-1
DGHWNR12HR	Failure to Recover Diesel Generator Hardware in 12 Hours	5.5E-1
EHV-AOV-CC-CCF	Common Cause Failure of DG Room AOVs to Open	1.0E-3
EHV-SRV-CC-RVZ	Relief Damper Z Fails to Open	3.0E-4
ESF-ASP-FC-PL52Z	LPCS, LPCI Low Reactor Pressure Sensor Z Fails	1.0E-3
ESF-XHE-FO-DATWS	Operator Fails to Depressurize During an ATWS	2.0E-1
ESF-XHE-MC-PRES	Operator Miscalibrates Reactor Pressure Sensors	5.3E-4
ESW-AOV-CC-CCF	Common Cause Loss of Flow to Air Operated Valves	1.0E-3
ESW-AOV-CC-0241Z	Air Operated Valve 0241Z Fails to Open	1.0E-3
ESW-CKV-CB-CV515Z	Check Valve 515Z Fails	3.0E-3
ESW-CKV-HW-CV513	Check Valve 513 Fails to Open	1.0E-4
ESW-MDP-FR-MDPZ	Motor Driven Pump Z Fails to Run	1.2E-3
ESW-MDP-FS-CCF	Emergency Service Water Motor Driven Pumps Common Cause Failure to Start	3.0E-3
ESW-MDP-FS-ECW	Emergency Cooling Water Motor Driven Pump Fails to Start	3.0E-3
ESW-XHE-FO-EHS	Failure of Operator to Initiate Emergency Heat Sink	9.0E-1
ESW-XVM-PG-XV502	Emergency Service Water Manual Valve 502 Plugs	4.0E-5
HCI-TDP-FO-20S37	Turbine Driven Pump Fails to Run for One Hour	5.0E-3

Table 5-2. Description of Important Events for the Peach Bottom Core Damage Frequency Results (Cont.)

Term	Description	Mean Value
HCI-TDP-FS-20S37	Turbine Driven Pump Fails to Start	3.0E-2
HCI-TDP-MA-20S37	Turbine Driven Pump Out for Maintenance	1.0E-2
INJ-FAILS	Failure of Injection Systems	5.0E-1
LOSPNRZZZZZ	Failure to Recover Offsite Power in the Time Given	See Sect. 4.10.3
NR	No Recovery Applied	1.0E-0
PZ	Z Stuck Open Safety Relief Valves	See Sect. 4.9
Q	Failure of the Power Conversion System	1.0E-2
RPSM	Mechanical Failure of Reactor Protection System	1.0E-5
SLC-CKV-HW-CVZZ	Check Valve ZZ Fails to Open	1.0E-4
SLC-MDP-FS-CCF	Common Cause Pump Failure to Start of Two SI Pumps	3.0E-3
SLC-SYS-TE-SLC	System Unavailable During Test	3.4E-3
SLC-XHE-FO-SLC	Operator Fails to Initiate SLC	2.0E-2
SLC-XHE-RE-DIVER	Operator Fails to Restore System After Test	3.2E-2
<u>Initiating Events</u>		
IE-A	Large LOCA	1.0E-4
IE-SI	Intermediate LOCA	3.0E-4
IE-T1	Loss of Offsite Power	7.9E-2
IE-T2	Transient, PCS Initially Unavailable	5.0E-2
IE-T3A	Transient, PCS Initially Available	2.5E-0
IE-T3B	Transient, Loss of Feedwater	6.0E-2
IE-T3C	Transient, Inadvertent Opening of a Relief Valve (IORV)	1.9E-1

Note: The Zs represent identifiers for a specific component or train. For more information, refer to Table 4.9-1 and to the system schematic in Section 4.6. For events not shown, also refer to Table 4.9-1.

EHV Emergency Ventilation System
ESF Emergency Safeguard Actuation System
ESW Emergency Service Water System
HCI High Pressure Coolant Injection System
RPS Reactor Protection System
SLC Standby Liquid Control System

5.2 Accident Sequence Results

The 18 Peach Bottom accident sequences leading to core damage are tabulated in Table 5-3 with a short description and the corresponding statistics. The top two accident sequences contribute 36% and 31% to the total core damage frequency. The next six accident sequences combined contribute another 23%. The last ten accident sequences contribute the other 10% of core damage frequency. Each of the accident sequences are discussed below including its top cut sets and key events. The importance measures are discussed for the overall core damage frequency in Section 5.4. Importance measures are given for the accident sequences in Appendix F.

5.2.1 Accident Sequence 1 T1-BNU11 650 Cut Sets

1.64E-6 Mean CDF 36.4% of the Total CDF

This accident sequence is initiated by a loss of offsite power (T1). The SRVs properly control the reactor pressure, but failure of all diesel generators occurs, (B) which results in a station blackout. HPCI is initially successful (NU11) but fails later due to either the harsh environment (e.g., loss of room cooling effects) or subsequent battery depletion, resulting in late core damage in a vulnerable containment. HPCI failure is nominally expected to occur (including RCIC failure) in about ten hours, with core damage resulting in about 13 hours as a result of coolant boiloff. More time is allowed for those cut sets in which the diesel generators have initially started but then failed to run.

The top cut set has a frequency of 3.7E-8, but 49 cut sets comprise 50% of the accident sequence CDF. The significance of this is that there are many combinations of events with comparable frequencies that cause station blackout. Key event contributors are: operator failure to initiate the emergency heat sink causing emergency service water failure and subsequent diesel generator failure due to loss of cooling, failure of the diesel generators to continue to run after successfully starting, failure of the injection systems due to the harsh environment, battery depletion resulting in loss of system control, and failure to recover power.

5.2.2 Accident Sequence 2 T3A-C-SLC 9 Cut Sets

1.40E-6 Mean CDF 31.3% of the Total CDF

The initiating event for this sequence is a transient with the power conversion system initially available (T3A). The reactor protection

Table 5-3. Peach Bottom Accident Sequence Core Damage Frequencies

	Accident Sequence	Simplified Description	5%	Median	Mean	95%	% of Total
1.	T1-BNU11	Station Blackout, Batt Dep	3.1E-8	3.4E-7	1.6E-6	4.1E-6	36.4
2.	T3A-C-SLC	ATWS, SLC Fails	1.7E-8	2.7E-7	1.4E-6	5.4E-6	31.3
3.	T3A-CU11X	ATWS, Inj Fails, ADS Fails	2.7E-9	5.3E-8	2.8E-7	1.0E-6	6.2
4.	S1-V2V3V4NU11	Med LOCA, LPI Fails	9.7E-10	2.3E-8	2.1E-7	6.4E-7	4.7
5.	T1-BU11U21	Station Blackout, Inj Fails	3.3E-9	4.6E-8	1.9E-7	6.5E-7	4.2
6.	T1-P1BNU11	Station Blackout, 1 Open SRV Batt Dep	1.1E-9	2.1E-8	1.3E-7	3.5E-7	2.9
7.	T1-BU11NU21	Station Blackout, Batt Dep	1.2E-9	1.7E-8	1.3E-7	3.0E-7	2.7
8.	T3C-C-SLC	ATWS, SLC Fails, IORV	1.2E-9	2.3E-8	1.1E-7	3.8E-7	2.4
9.	T1-P2V234NU11B	LOSP, 2 Open SRVs, LPI Fails	5.6E-10	1.4E-8	8.7E-8	3.5E-7	1.9
10.	T2-P2V234NU11	PCS Fails, 2 Open SRVs, LPI Fails	2.1E-10	5.9E-9	5.7E-8	1.8E-7	1.3
11.	T3B-P2V234NU11	FW Trans, 2 Open SRVs, LPI Fails	2.3E-10	7.5E-9	5.6E-8	2.2E-7	1.2
12.	A-V2V3	Large LOCA, LPI Fails	3.3E-10	7.8E-9	4.6E-8	1.9E-7	1.0
13.	T1-C-SLC	LOSP, ATWS, SLC Fails	4.3E-10	1.0E-8	4.4E-8	1.6E-7	1.0
14.	T3B-C-SLC	ATWS, SLC Fails	4.2E-10	6.7E-9	3.3E-8	1.4E-7	0.7
15.	T2-C-SLC	ATWS, SLC Fails	3.7E-10	5.6E-9	2.7E-8	1.1E-7	0.6
16.	T3A-P2V234NU11	PCS Trans, 2 Open SRVs, LPI Fails	6.4E-11	2.4E-9	2.5E-8	9.6E-8	0.6
17.	T3C-CU11X	ATWS, IORV, Inj Fails, ADS Fails	1.7E-10	3.5E-9	2.2E-8	7.2E-8	0.5
18.	T1-P1BU11U21	Station Blackout, Open SRV HPI Fails	1.3E-10	3.3E-9	1.7E-8	6.8E-8	0.4
		Total Core Damage Frequency	3.5E-7	1.9E-6	4.5E-6	1.3E-5	100.0

system fails (C) to scram the reactor, resulting in an ATWS. The SRVs open and the standby liquid control system (SLC) fails, leading to core damage. The high pressure systems (HPCI/RCIC) fail in less than one-half hour due to high suppression pool temperature (the pool will be used for suction following high pool level indication).

Feedwater flow is stopped because of a likely closure of the MSIVs and depletion of the condenser hotwell, followed by either failure to depressure to go to low pressure cooling or failure of low pressure cooling. This could occur because of containment venting, containment failure, or reclosure of the SRVs upon high containment pressure. These events are likely to fail the low pressure systems because of low NPSH, steam in the reactor building, or the inability to keep the reactor vessel depressurized.

The top two cut sets for the total CDF come from this accident sequence. These were discussed in Section 5.1. In essence, each of these cut sets involves mechanical failure of the reactor protection system (RPSM) and operator failure to initiate the SLC system or properly restore the system after testing. All other cut sets include RPSM and some hardware failure in the SLC. The key events for this accident sequence are RPSM and operator failure to initiate the SLC system or restore it properly after testing.

5.2.3 Accident Sequence 3 T3A-CU11X 14 Cut Sets

2.79E-7 Mean CDF 6.2% of the Total CDF

This accident sequence is a transient with the power conversion system initially available (T3A), leading to an ATWS following reactor protection system failure (C), and likely loss of feedwater. The SLC system operates, but HPCI (U11) and reactor depressurization fail (X), leading to core damage since low pressure cooling can not be initiated. Core damage occurs in less than fifteen minutes.

Every cut set includes the event of failure of the operator to depressurize the primary system (ESF-XHE-FO-DATWS), given that in a sequence with an ATWS and mechanical failure of the RPS (RPSM), the operator has performed the event of inhibiting the ADS. There is no feasible recovery (NR) because of the short time available to prevent core damage. In addition, each cut set has some component failure of the HPCI system. The top three cut sets cover 86% of the CDF and involve the turbine driven pump failing to start and run or being out for maintenance. The key events are RPSM and failure to depressurize the primary system. The top three cut sets rank 5th, 12th, and 19th in total core damage frequency.

5.2.4 Accident Sequence 4 S1-V2V3V4NU11 3 Cut Sets

2.12E-7 Mean CDF 4.7% of the Total CDF

This sequence is a medium size LOCA (S1) with initial HPCI success (NU11) but subsequent HPCI failure in about one-half to one hour due to loss of vessel steam pressure to operate the steam-driven HPCI pumps. Subsequent failure of the LPCS (V2), LPCI (V3), and HPSW (V4) low pressure injection systems leads to core damage in 1 to 2 hours following the initiator in a containment vulnerable situation.

There are three cut sets in this Peach Bottom accident sequence, one of which one is significant. This cut set is IE-S1*ESF-XHE-MC-PRES*NR, in which the operator miscalibrates the reactor pressure sensors which permit opening of the low pressure cooling injection valves once vessel pressure is sufficiently low. This causes failure of all applicable low pressure systems and no feasible recovery (NR) in the time available. The initiating event, IE-S1, and the primary event, ESF-XHE-MC-PRES, are the dominant events for all measures. The dominant cut set in this accident sequence ranks fourth in total core damage frequency.

5.2.5 Accident Sequence 5 T1-BU11U21 1 Cut Set

1.90E-7 Mean CDF 4.2% of the Total CDF

This sequence is initiated by a loss of offsite power (T1) with successful scram and proper pressure control by the SRVs. This is followed by failure of all diesel generators (B) due to common cause failure of multiple batteries, resulting in a station blackout. Battery failure leads to loss of diesel generator start and loading capability and failure of both HPCI (U11) and RCIC (U21). Additionally, low pressure cooling capability is lost because low pressure pumps require AC power, resulting in no injection for coolant makeup. This leads to early core damage (~1 hour) and a vulnerable containment. The one cut set is IE-T1*DCP-BAT-LF-CCF*BETA-5BAT*NR. All four events are key events for this accident sequence. This cut set ranks third in total core damage frequency.

5.2.6 Accident Sequence 6 T1-P1BNU11 285 Cut Sets

1.31E-7 Mean CDF 2.9% of the Total CDF

This sequence is initiated by loss of offsite power (T1) followed by one stuck open safety relief valve (P1). Subsequent failure of all diesel generators (B) results in a station blackout. Although HPCI is initially successful (NU11), it later fails because of either harsh environments (see Section 5.2.1) or as a result of subsequent battery depletion, resulting in core damage in 10 to 13 hours. This time may be longer, depending on whether the diesel generators failed to start or failed to run after successfully starting. Note that while the stuck-open valve would cause boiloff slightly faster than for accident sequence 1, the difference in timing is not significant (1 to 2 hours) and was not specifically accounted for in the recovery probabilities (i.e., differences in value are small).

As in the case of accident sequence 1, many cut sets include combinations of diesel generator failure and battery depletion or subsequent injection failure. Key events are the stuck open SRV (P1), failure of the operator to initiate the emergency heat sink (ESW-XHE-FO-ESH), failure of the diesel generators to run, subsequent injection failure, battery depletion, and failure to recover offsite power.

5.2.7 Accident Sequence 7 Tl-BU11NU21 395 Cut Sets

1.25E-7 Mean CDF 2.7% of the Total CDF

This sequence is initiated by a loss of offsite power (Tl), followed by loss of all diesels (B) which results in a station blackout. HPCI then fails, followed by either battery depletion or RCIC injection failure due to the harsh environment. RCIC (NU21) is successful initially. Core damage occurs late (similar to accident sequence 1) in a vulnerable containment. Similar to sequences 1 and 6, combinations of diesel generator failure and battery depletion or subsequent injection failure dominate the results. Key events are failure of the operator to initiate the emergency heat sink (ESW-XHE-FO-EHS), failure of the HPCI turbine driven pump to start or run, failure of the diesel generators to run, subsequent injection failure, battery depletion, and failure to recover offsite power.

5.2.8 Accident Sequence 8 T3C-C-SLC 6 Cut Sets

1.14E-7 Mean CDF 2.4% of the Total CDF

This sequence is a transient with inadvertent opening of a relief valve (T3C), which creats a LOCA. An ATWS results from failure to scram (C), followed by standby liquid control (SLC) system failure to shutdown the reactor, leading to core damage. This scenario is very similar to accident sequence 2 except for the additional failure of a stuck-open valve.

All of the 6 cut sets involve mechanical failure of the reactor protection system (RPSM) and non-recovery (NR). Operator or hardware failures in the SLC system also contribute. Key events are RPSM and operator failure to restore the system after testing (SLC-XHE-RE-DIVER). The top cut set ranks ninth overall.

5.2.9 Accident Sequence 9 Tl-P2V234NU11B 5 Cut Sets

8.73E-8 Mean CDF 1.9% of the Total CEF

This is a loss of offsite power transient (Tl) not leading to station blackout (NB). High pressure injection initially operats (NU11), but two SRVs fail to close (P2). This causes the equivalent of an S1 LOCA, so the plant response is similar to that of sequence 4. When the low pressure systems are demanded, they fail (V234), resulting in core damage.

There are five cut sets in this sequence. However, the top cut set, IE-T1*P2*ESF-XHE-MC-PRES*NR, dominates the sequence CDF and ranks 7th in the cut sets comprising the total core damage frequency. Key events are the two stuck open SRVs (P2), and operator miscalibration of the reactor pressure sensors (ESF-XHE-MC-PRES).

5.2.10 Accident Sequence 10 T2-P2V234NU11 1 Cut Set

5.72E-8 Mean CDF   1.3% of the Total CDF

This sequence is a transient with the PCS initially unavailable (T2), followed by two stuck open SRVs (P2) and early success of the HPCI system (NU11). Later in the sequence, low pressure systems fail (V234) upon demand, leading to core damage. The plant response and timing is similar to that of accident sequences 4 and 9. The one cut set is IE-T2*P2*ESF-XHE-MC-PRES*NR. The key events are those in this cut set. This cut set ranks tenth in the total core damage frequency.

5.2.11 Accident Sequence 11 T3B-P2V234NU11 2 Cut Sets

6.41E-8 Mean CDF   1.2% of the Total CDF

This sequence is a transient with the PCS initially available (T3B), followed by two stuck open SRVs (P2) and successful early high pressure injection with the HPCI system (NU11). Subsequent HPCI failure and failure of available low pressure systems (V234) occurs, similar to sequences 4, 9, and 10. The cut set IE-T3B*P2*ESF-XHE-MC-PRES*NR dominates the accident sequence and its events are the key events. This cut set ranks eighth in the total core damage frequency.

5.2.12 Accident Sequence 12 A-V2V3 3 Cut Sets

4.63E-8 Mean CDF   1.0% of the Total CDF

This is a large LOCA (A) followed by failure of the applicable low pressure systems (V2 and V3) leading to core damage. One cut set, IE-A*ESF-XHE-MC-PRES*NR, dominates the results. This cut set represents operator miscalibration of the reactor pressure sensors with no feasible recovery in the short time period available. Core damage results in approximately fifteen minutes with no injection. Key events are those events in the dominant cut set. This cut set ranks twelfth in the total core damage frequency.

5.2.13 Accident Sequence 13 T1-C-SLC 4 Cut Sets

4.37E-8 Mean CDF   1.0% of the Total CDF

This sequence is a transient caused by loss of offsite power (T1), leading to an ATWS following failure to scram (C). Failure of the SLC system to shutdown the reactor leads to core damage. The plant response and timing is similar to that of sequence 2. The top two cut sets dominate the sequence. The top cut set ranks eighteenth overall. The cut sets involve RPSM, non-recovery (NR), and operator failure to initiate the SLC (SLC-XHE-FO-SLC), and failure to restore the system after testing (SLC-XHE-RE-DIVER). These are also the key events for this sequence.

5.2.14 Accident Sequence 14 T3B-C-SLC 4 Cut Sets

3.29E-8 Mean CDF 0.7% of the Total CDF

This sequence is a transient caused by loss of feedwater (T3B), leading to an ATWS following failure to scram (C). The SLC system then fails to shutdown the reactor. The discussion of sequence 13 applies here also. None of the cut sets rank in the top 20 cut sets overall.

5.2.15 Accident Sequence 15 T2-C-SLC 4 Cut Sets

2.69E-8 Mean CDF 0.6% of the Total CDF

This sequence is a transient with the PCS initially unavailable (T2), leading to an ATWS following failure to scram (C). The SLC system then fails to shutdown the reactor. The discussion of sequence 13 applies here also. None of the cut sets rank in the top 20 cut sets overall.

5.2.16 Accident Sequence 16 T3A-P2V234NU11 1 Cut Set

2.45E-8 Mean CDF 0.6% of the Total CDF

This is a transient sequence with the PCS initially available (T3A) but later unavailable (Q), followed by two stuck open relief valves (P2). Initial success of the HPCI system (NU11) is followed by failure of HPCI and all applicable low pressure systems (V234), which leads to core damage. The plant response is similar to that for sequences 4, 9, 10 and 11. The one cut set ranks 17th overall. This cut set is IE-T3A*P2*Q*ESF-XHE-MC-PRES*NR. All of the events in this cut set are key events for the accident sequence.

5.2.17 Accident Sequence 17 T3C-CU11X 5 Cut Sets

2.20E-8 Mean CDF 0.5% of the Total CDF

This sequence is a transient caused by an inadvertent opening of a relief valve (T3C), leading to an ATWS following failure to scram (C). Failure of the HPCI system and failure to depressurize the reactor (X) occur, similar to sequence 3. None of the cut sets are ranked in the top 20 overall. The five cut sets can be characterized by IE-T3C*RPSM*ESF-XHE-FO-DATWS* [HPCI system failure] *NR. ESF-XHE-FO-DATWS is operator failure to depressurize during an ATWS. HPCI system failures are failure of the turbine driven pump or failure of either a motor operated valve (MOV) in the steam line to the turbine or a discharge MOV. RPSM and ESF-XHE-FO-DATWS are the key events.

5.2.18 Accident Sequence 18 T1-P1BU11U21 1 Cut Set

1.70E-8 Mean CDF 0.4% of the Total CDF

This is a loss of offsite power transient (T1) with one stuck open SRV (P1) and failures of both HPCI (U11) and RCIC (U21). Station blackout results (B) due to common cause failure of the five batteries (DCP-BAT-LF-CCF*BETA-5BAT). The sequence description follows that given for sequence 5 except the event of a stuck-open valve occurs. The one cut set in this accident sequence is not ranked in the top 20 cut sets. The key events are those given above.

5.3 Plant Damage State Results

The accident sequences must be grouped for input in to the accident progression event tree for the back-end analysis. The complete definitions of the plant damage states are given in sections 4.5 and 4.11. The resulting mapping from accident sequences to plant damage states is given in Table 5-4. Thus, the primary contributors to each plant damage state can be inferred from the corresponding accident sequences and are not repeated here. The core damage frequency statistics for each plant damage state are given in Table 5-5 along with an abbreviated description of the PDS. Plant damage states 5 and 8 contribute 42.0% and 32.5%, respectively. PDS-3 is relatively insignificant at 0.1% of the total CDF. Each plant damage state is discussed below in order of its contribution to the total CDF.

5.3.1 Plant Damage State 5 1330 Cut Sets

1.90E-6 Mean CDF 42.0% of the Total CDF

This PDS is composed of three sequences: T1-P1BNU11, T1-BNU11, and T1-BU11NU21. These sequences involve a station blackout with or without one stuck open SRV and initially successful operation of HPCI or RCIC, hence core damage is 10 or more hours following the initiating event. Battery depletion may or may not occur before core damage. The vessel remains at low pressure if an SRV is stuck open, otherwise, it repressurizes on loss of DC. AC systems are available on recovery of AC power. Venting is not possible until AC is restored.

The cut sets in PDS-5 are characterized by diesel generator failure given LOSP and failure to recover offsite power. This may be due to hardware failures and subsequent failure to repair the diesel generators or cooling failures which, in turn, cause diesel generator failure. In all cases, failure of injection is the end result because of battery depletion or harsh environment conditions. PDS-5 cut sets involve all the combinations of these failures leading to a large number of cut sets with a more uniform distribution of contribution per individual cut set.

Key events are the operator failure to initiate the emergency heat sink when required, diesel generator failure to run, HPCI and RCIC subsequent failure due to harsh environments, battery depletion, failure to recover diesel generator failures, and failure to recover offsite power. Four cut sets in this PDS rank in the top 20 cut sets overall; they rank 14th, 15th, 16th, and 20th.

5.3.2 Plant Damage State 8 17 Cut Sets

1.46E-6 Mean CDF 32.5% of the Total CDF

Table 5-4. Peach Bottom Accident Sequences Included in Each Plant Damage State (PDS)

PDS	Accident Sequences and Cut Sets*	Number of Cut Sets
1	4(S1-V2V3V4NU11)(CS1-3) + 12(A-V2V3)(CS1-3)	6
2	9(T1-P2V234NU11B)(CS1,4,5) + 11(T3B-P2V234NU11)(CS1) + 10(T2-P2V234NU11)(CS1) + 16(T3A-P2V234NU11)(CS1)	6
3	9(T1-P2V234NU11B)(CS2,3) + 11(T3B-P2V234NU11)(CS2)	3
4	5(T1-BU11U21)(CS1) + 18(T1-P1BU11U21)(CS1)	2
5	1(T1-BNU11)(CS1-650) + 6(T1-P1BNU11)(CS1-285) + 7(T1-BU11NU21)(CS1-395)	1330
6	3(T3A-CU11X)(CS1-14) + 17(T3C-CU11X)(CS1-5)	19
7	8(T3C-C-SLC)(CS1-6)	6
8	2(T3A-C-SLC)(CS1-9) + 14(T3B-C-SLC)(CS1-4) + 15(T2-C-SLC)(CS1-4)	17
9	13(T1-C-SLC)(CS1-4)	4
	Total Cut Sets	1393

*Accident Sequence Number (accident sequence code name)(cut sets included)

Table 5-5. Peach Bottom Plant Damage State Core Damage Frequencies

	Plant Damage State Code	Simplified Description	5%	Median	Mean	95%	% of Total
1.	1-322-2-13-3-13113-111	LOCA-HPIFAILS-LPIFAILS	2.5E-9	4.4E-8	2.6E-7	7.8E-7	5.7
2.	4-622-1-13-3-13113-111	TRANS-SORV-LPIFAILS	1.1E-9	3.0E-8	2.2E-7	8.1E-7	4.9
3.	4-622-1-13-3-11131-111	TRANS-SORV-LPIFAILS	5.9E-11	1.2E-9	6.1E-9	2.7E-8	0.1
4.	4-211-6-12-1-22222-111	TRANS-SBO-NODC-HPIFAILS-NOADS	3.5E-9	5.0E-8	2.1E-7	7.1E-7	4.6
5.	4-212-6-22-3-22222-111	TRANS-SBO-BATDEP	3.5E-8	4.0E-7	1.9E-6	4.8E-6	42.0
6.	5-322-6-23-2-33333-111	ATWS-HPIFAILS-LPIAVAIL	3.2E-9	5.9E-8	3.0E-7	1.1E-6	6.7
7.	5-322-1-23-6-33333-611	ATWS-IORV-SLCFAILS	1.2E-9	2.3E-8	1.1E-7	3.8E-7	2.5
8.	5-322-2-23-6-33333-611	ATWS-SLC FAILS	1.8E-8	2.9E-7	1.5E-6	5.6E-6	32.5
9.	5-222-2-23-6-33233-611	ATWS-LOSP-LPIAVAIL	4.3E-10	1.0E-8	4.4E-8	1.6E-7	1.0
		Total Core Damage Frequency	3.5E-7	1.9E-6	4.5E-6	1.3E-5	100.0

This PDS is composed of three sequences: T3A-C-SLC, T3B-C-SLC, and T2-C-SLC. Each includes a transient with subsequent failure to scram and SLC failure. HPCI fails early on high suppression pool temperature, when it transfers suction to the pool on high pool level. Then the reactor is a) not manually depressurized or b) is manually depressurized to use the low pressure systems. If b), then the containment will pressurize until either venting or containment failure occurs. Also, SRV reclosure on high containment pressure could occur. In all b) cases, the low pressure injection systems will fail due to low NPSH, harsh environments, or the inability to keep the reactor vessel depressurized. Hence, core damage results. Venting will likely be tried before core damage. The CRD system is working in all cases but supplies inadequate cooling. This PDS is similar to PDS-7, except that a SRV is stuck open in that case. All the cut sets in this plant damage state are characterized by a transient initiating event followed by RPSM, a SLC operator or hardware failure and no feasible recovery (NR). Key events are RPSM and operator failure to restore the SLC after testing. Three cut sets in this PDS rank in the top 20 cut sets over all; they rank 1st, 2nd, and 6th.

5.3.3 Plant Damage State 6 19 Cut Sets

3:00E-7 Mean CDF 6.7% of the Total CDF

This PDS is composed of two sequences: T3C-CU11X, and T3A-CU11X. These are transients with failure to scram, SLC success, HPCI failure, and manual vessel depressurization failure which precludes low pressure cooling. The containment is not vented before core damage, but venting is operable. The cut sets in this PDS all include an initiating event followed by RPSM and operator failure to depressurize after an ATWS (ESF-XHE-FO-DATWS), HPCI system failures, and no feasible recovery (NR). The key events are RPSM and ESF-XHE-FO-DATWS. Three cut sets from this PDS are in the top 20 cut sets overall; they rank 5th, 12th, and 19th.

5.3.4 Plant Damage State 1 6 Cut Sets

2.58E-7 Mean CDF 5.7% of the Total CDF

This PDS is composed of two accident sequences, A-V2V3 and S1-V2V3V4NU11. A-V2V3 is a large LOCA initiator followed by immediate failure of the LPCS and LPCI systems (other high or low pressure systems can not mitigate this sequence in time or fail as a result of the initiator). The result is early core damage. S1-V2V3V4NU11 is a medium LOCA initiator followed by initial success of HPCI. HPCI fails soon thereafter (~1/2 to 1 hour) due to low vessel pressure and LPCS, LPCI, and HPSW (insufficient time or operator error) all fail (other systems fail or can not mitigate the LOCA). This again results in early core damage. CRD is working in both sequences and all containment heat removal is potentially operable. LPCS and LPCI have failed due to miscalibration of the pressure permissive sensors and so the injection valves can not be opened; this fails HPSW also. Venting will work if needed, but will not be demanded before core damage. All the cut sets in this PDS can be characterized by an initiating event followed by failure or miscalibration of the reactor pressure sensors and no feasible recovery. The key event is ESF-XHE-MC-PRES. Two cut sets are in the top 20 cut sets overall; they rank 4th and 11th.

5.3.5 Plant Damage State 2 6 Cut Sets

2.19E-7 Mean CDF 4.9% for the Total CDF

This PDS is composed of four sequences: T3A-P2V234NU11, T3B-P2V234NU11, T2-P2V234NU11, and T1-P2V234NU11B. This PDS is similar to PDS-1. Different initiators with subsequent failure of SRVs result in the equivalent of an intermediate LOCA (P2). The sequences then follow the same pattern as in PDS-1. Containment overpressure protection is working, but steam is directed through the SRVs to the suppression pool, not to the drywell as in a LOCA. HPCI works early, but fails on low vessel pressure in about 1/2 to 1 hour, and all other high or low pressure systems are inoperable. The low pressure injection valves can not be opened which, in turn, fail LPCI, LPCS, and HPSW. This results in early core damage. Venting will not be demanded before core damage. The cut sets in this PDS can be characterized by an initiating event followed by two stuck open SRVs (P2) and operator miscalibration of the reactor pressure sensors (ESF-XHE-MC-PRES) with no feasible recovery (NR). These are also the key events. Four cut sets from this accident sequence are in the top 20 and cut sets overall; they rank 7th, 8th, 10th, and 17th.

5.3.6 Plant Damage State 4 2 Cut Sets

2.07E-7 Mean CDF 4.6% of the Total CDF

This PDS is composed of two sequences: T1-P1BU11U21, and T1-BU11U21. The first sequence is a station blackout and includes one stuck-open SRV. High pressure injection fails because of DC power common cause failure, so early core damage results. Vessel pressure remains low, however, because of the stuck-open valve. For the second sequence, there is no stuck-open SRV, so the vessel is at high pressure during core melt. Venting is not possible unless AC is restored. AC systems are available with recovery of AC power. There are two cut sets in this plant damage state characterized by IE-T1 followed by common cause failure of the five batteries (DCP-BAT-LF-CCF*BETA-5BAT) and no feasible recovery (NR). One cut set also has one stuck-open SRV. The key events are those mentioned above. One cut set ranks third in the top 20 cut sets overall.

5.3.7 Plant Damage State 7 6 Cut Sets

1.14E-7 Mean CDF 2.5% of the Total CDF

This PDS is composed of one sequence: T3C-C-SLC. This is an IORV with failure to scram and SLC failure. HPCI fails early in the sequence because of high suppression pool temperature when HPCI suction switches to the pool on high pool level. Then, the reactor is: a) not manually depressurized, or b) is manually depressurized to use low pressure systems. If a) then early CD results and venting will not occur before CD. If b) then the containment will pressurize until venting or containment failure occurs. SRV reclosure on high containment pressure may also occur. In all b) cases, the low pressure injection systems will fail due to low NPSH, a harsh environment or SRV reclosure. Core damage will result but under low pressure conditions because of the IORV initiator. Venting will likely be tried before CD. The CRD system is working in all cases. There are two dominant cut sets in this PDS that can be characterized by the initiator (T3C) followed by RPSM, operator failure to initiate SLC (SLC-XHE-FO-SLC) or operator failure to restore the SLC system after testing and no feasible recovery (NR). These same events are the key events. Two cut sets from this PDS are in the top 20 cut sets overall; they rank 9th and 13th.

5.3.8 Plant Damage State 9 4 Cut Sets

4.37E-8 Mean CDF 1.0% of the Total CDF

This PDS is composed of one sequence: T1-C-SLC. This is a LOSP with failure to scram and SLC failure. HPCI fails on high suppression pool temperature as above, and the reactor is: a) not manually depressurized, or b) is manually depressurized to use the low pressure systems. The possible sequence of events is as above since this PDS is similar to PDS-8 except for LOSP. The four cut sets in this PDS can be characterized by the T1 initiating event, followed by RPSM, operator failure to initiate the SLC system (SLC-XHE-FO-SLC) or operator failure to restore the SLC system after testing (SLC-XHE-RE-DIVER) and no feasible recovery (NR). Key events are those mentioned above. One cut set from this PDS ranks in the top 20 cut sets overall at 18th.

5.3.9 Plant Damage State 3 3 Cut Sets

6.05E-9 Mean CDF 0.1% of the Total CDF

This PDS is composed of two sequences: T3B-P2V234NU11 and T1-P2V234NU11B. This PDS is similar to PDS-1. These transient initiators with subsequent failure of the SRVs result in the equivalent of an intermediate LOCA (P2). The sequences then follow the same pattern as in PDS-1. However, containment overpressure protection is not working, but steam is directed through the SRVs to the suppression pool, not to the drywell as in a LOCA. CRD is also not working in some cut sets. This PDS is also similar to PDS-2, except that containment overpressure protection is not working and HPSW has failed by operator error or can not be used in time (makes it similar to PDS-1). The three cut sets in this PDS are characterized by a transient initiator and two stuck open SRVs (P2) followed by operator failure to realign the HPSW system for injection (ESF-XHE-FO-HSWIN) and valves failing closed in the emergency service water system. ESF-XHE-FO-HSWIN and P2 are key events. No cut sets from this PDS are in the top 20 cut sets overall.

5.3.10 Plant Damage State Split Fractions

When the accident sequences were categorized by plant damage state, there were initially 20 unique plant damage states. In Section 4.11, these PDSs were called interim PDSs since they were later combined to form nine final plant damage states to save analysis resources. In doing this, the back-end analysts need to know the proportion of each of the final nine PDSs that come from the twenty interim PDSs. The proportions were calculated and called split fractions as shown in Table 5-6.

PDS-1 did not require a split fraction. The split fractions for PDS-2 and 3 were calculated from the ratio of point estimates for the cut sets included since there were two accident sequences that were divided between the two PDSs and only point estimates were calculated for individual cut sets. The split fractions for PDS-4 through 6 were calculated from ratios of the accident sequence mean value. PDS-7 through 9 were determined from the products of the appropriate ADS failure (0.200) and failure to vent (0.002). For example, the probability of /ADS*/VENT = 0.800*0.998 = 0.798.

5.3.11 Super Plant Damage States

The accident sequences or plant damage states can be categorized by type relative to the initiators. One breakdown is given in Figure 5-7 considering in order of precedence ATWS, LOSP, transients, and LOCAs. These four types are called super plant damage states. This is what is used in NUREG-1150. The precedence is that a LOSP, which becomes an ATWS, is grouped with ATWS, and transient induced LOCAs, i.e., stuck open SRVs, are grouped with transients and not with the LOCAs. TEMAC runs were made on the four super plant damage states shown in Table 5-7 to obtain the statistics presented in the table.

Another informative subdivision is station blackout. Short-term station blackout results from two accident sequences, T1-BU11U21 and T1-P1BU11U21, and accounts for 4.6% of the total CDF. These two sequences constitute the entire PDS-4. Long-term station blackout with subsequent battery depletion or other long-term failure of injection results from three accident sequences, T1-BNU11, T1-P1BNU11, and T1-BU11NU21, and accounts for 42.0% of the total CDF. These three sequences constitute the entire PDS-5. Thus, PDS-4 and PDS-5 together represent all of the station blackout scenarios and constitute 46.6% of the total CDF.

5.4 Importance Measures

The importance measures examined in this study are risk reduction, risk increase, and uncertainty. Each of these measures were evaluated for the total CDF, each accident sequence, and each plant damage state. These results can be found in Appendix F. Only the importance measures for the total CDF are discussed here in the results section since the most insights are to be gained by identifying those events affecting the overall CDF. Definitions for the three importance measures are given below:

Table 5-6. Peach Bottom Plant Damage State Split Fractions

Final PDS	Interim PDS	Variable	Split Fraction
1	1		None Required
2	2	/LOSP	0.630
	5	LOSP	0.370
3	3	/LOSP	0.052
	4	LOSP	0.948
4	6	SRV	0.082
	7	/SRV	0.098
5	8	SRV	0.069
	9	/SRV	0.931
6	10	SRV	0.073
	17	/SRV	0.927
7	11	ADS	0.200
	12	/ADS*VENT	0.002
	13	/ADS*/VENT	0.798
8	14	ADS	0.200
	15	/ADS*VENT	0.002
	16	/ADS*/VENT	0.798
9	18	ADS	0.200
	19	/ADS*VENT	0.002
	20	/ADS*/VENT	0.798

Table 5-7. Peach Bottom Super Plant Damage States

Super Plant Damage State	Contributing Accident Sequences		5%	Median	Mean	95%	% of Total
ATWS (PDS-6, 7,8 and 9)	2	T3A-C-SLC	3.1E-8	4.4E-7	1.9E-6	6.6E-6	42.2
	3	T3A-CU11X					
	8	T3C-C-SLC					
	13	T1-C-SLC					
	14	T3B-C-SLC					
	15	T2-C-SLC					
	17	T3C-C-CU11X					
LOSP (PDS-4, 5 and parts of 2 and 3)	1	T1-BNU11	8.3E-8	6.2E-7	2.2E-6	6.0E-6	48.9
	5	T1-BU11U21					
	6	T1-P1BNU11					
	7	T1-BU11NU21					
	9	T1-P2V234NU11B					
	18	T1-P1BU11U21					
Transient (Parts of PDS-2 and 3)	10	T2-P2V234NU11	6.1E-10	1.9E-8	1.4E-7	4.7E-7	3.1
	11	T3B-P2V234NU11					
	16	T3A-P2V234NU11					
LOCA (PDS-1)	4	S1-V2V3V4NU11	2.5E-9	4.4E-8	2.6E-7	7.8E-7	5.8
	12	A-V2V3					
Total Core Damage Frequency			3.5E-7	1.9E-6	4.5E-6	1.3E-5	100.0

Risk Reduction - A measure of how much the results are reduced given a specific event is assumed to be totally reliable (probability of failure = 0). A large value indicates that a significant reduction in the core damage frequency is possible by improving the reliability associated with that event.

Risk Reduction - Opposite of risk reduction (probability of failure = 1). A large effect indicates the importance of maintaining the reliability of the specific event and not letting it get worse.

Uncertainty - A measure of how much the uncertainty in the results is affected by the uncertainty associated with a specific event. The larger the measure, the more the uncertainty in the results is driven by the uncertainty in the value of the specific event.

The top 20 events for each of the measures are given in Tables 5-8, 5-9, and 5-10. Additional detail is given in Appendix F for the total core damage frequency and for each accident sequence and plant damage state. Definitions of the events found in these three tables are in Table 5-2.

Most of the events in Table 5-8 covering the risk reduction measure were discussed often for the accident sequence and plant damage states previously presented. NR is at the top of the list because most of the significant cut sets did not have a feasible recovery action. In a way, it is an artificial event since it was used by the analyst to tag the cut set as having been examined for potential recovery, but none was identified. The second ranked risk reduction event is RPSM, which is the mechanical failure of the reactor protection system. This event is key to the ATWS sequences. The next four events (excluding the two initiators) are operator errors or failures. The eighth and ninth ranked events for risk reduction are failure of diesel generators to run.

The risk reduction frequency is the magnitude the core damage frequency could be reduced if that event were set to zero. This provides perspective as to a possible priority for the potential for improvement.

The top 20 risk increase events are given in Table 5-9. No initiating events are included since the frequency of an initiating event is not limited to 1.0, whereas the probability of a failure event is limited to 1.0 by definition. RPSM leads the list, which means that the core damage frequency is very sensitive to RPSM performance. Any degradation in the RPSM would have a big impact on core damage frequency. Operator miscalibration of the reactor pressure sensors (ESF-XHE-MC-PRES) is ranked second. Two stuck-open SRVs, common cause failures of batteries and emergency service water AOVs are important using this measure. It is interesting to note that several events occur on both lists indicating that the core damage frequency is particularly sensitive to these contributors:

Table 5-8. Peach Bottom Risk Reduction Events

Ranking	Event	Risk Reduction Frequency
1	NR	2.5E-6
2	RPSM	1.9E-6
3	IE-T3A	1.7E-6
4	IE-T1	1.4E-6
5	SLC-XHE-RE-DIVER	9.2E-7
6	ESW-XHE-FO-EHS	6.2E-7
7	SLC-XHE-FO-SLC	5.8E-7
8	ESF-XHE-MC-PRES	4.4E-7
9	ACP-DGN-FR-EDGC	3.7E-7
10	ACP-DGN-FR-EDGB	3.7E-7
11	INJ-FAILS	3.5E-7
12	ESF-XHE-FO-DATWS	2.8E-7
13	BAT-DEP-3HR	2.6E-7
14	P2	2.3E-7
15	BAT-DEP-9HR	2.1E-7
16	LOSPNR9HR	2.0E-7
17	LOSPNR18HR	2.0E-7
18	DCP-BAT-LF-CCF	2.0E-7
19	BETA-5BAT	2.0E-7
20	HCI-TDP-FS-20S37	1.8E-7

Table 5-9. Peach Bottom Risk Increase Events

Ranking	Event	Risk Increase Frequency
1	RPSM	1.9E-1
2	ESF-XHE-MC-PRES	8.3E-4
3	DCP-BAT-LF-CCF	2.2E-4
4	P2	1.2E-4
5	ESW-AOV-CC-CCF	9.7E-5
6	BETA-5BAT	7.8E-5
7	EHV-AOV-CC-CCF	6.3E-5
8	ESF-CKV-HW-CV513	4.3E-5
9	ESW-CKV-CB-C515B	4.1E-5
10	ESF-CKV-CB-C515A	4.1E-5
11	ESW-XVM-PG-XV502	4.0E-5
12	ESF-AOV-CC-0241B	3.9E-5
13	ESW-AOV-CC-0241C	3.9E-5
14	SLC-SYS-TE-SLC	2.9E-5
15	SLC-XHE-FO-SLC	2.8E-5
16	SLC-XHE-RE-DIVER	2.8E-5
17	LOSPNR18HR	2.8E-5
18	EHV-SRV-CC-RV3	2.7E-5
19	EHV-SRV-CC-RV2	2.7E-5
20	SLC-CKV-HW-CV17	2.7E-5

Table 5-10. Peach Bottom Uncertainty Importance

Ranking	Event	Uncertainty Importance Factor*
1	RPSM	29.7
2	ACP-DGN-FR-EDGB	11.5
3	ACP-DGN-FR-EDGD	11.5
4	ACP-DGN-FR-EDGC	11.5
5	IE-T1	7.7
6	ESF-XHE-MC-PRES	6.4
7	IE-T3A	5.8
8	SLC-XHE-RE-DIVER	5.4
9	SLC-XHE-FO-SLC	3.0
10	BAT-DEP-3HR	2.9
11	DCP-BAT-LF-CCF	2.9
12	ESF-MDP-FS-ECW	1.6
13	ESW-MDP-FS-MDPA	1.6
14	SLC-MDP-FS-CCF	1.6
15	ESW-MDP-FS-CCF	1.6
16	ESW-MDP-FS-MDPB	1.6
17	P2	1.5
18	INJ-FAILS	1.5
19	LOSPNR13HR	1.4
20	BETA-6AOVS	1.3

*Percent reduction in the uncertainty of the log of risk.

Event	Risk Reduction Ranking	Risk Increase Ranking
RPSM	2	1
SLC-XHE-RF-DIVER	3	16
SLC-XHE-FO-SLC	7	15
ESF-XHE-MC-PRES	8	2
P2	14	4
LOSPNR18HR	17	17
DCP-BAT-LF-CCF	18	3
BETA-5BAT	19	6

The last measure is the uncertainty importance given in Table 5-10. Again, RPSM is at the top of the list followed by diesel generator failure to run, initiating events T1 and T3A, and the three operator errors found on both the previous tables; ESF-XHE-MC-PRES, SLC-XHE-RE-DIVER, and SLC-XHE-FO-SLC. The uncertainty importance factor is not readily explainable, but if thought of as a relative measure, the 29.7 value for RPSM versus the 1.5 value for P2 gives an insight into how significant the contribution of RPSM is to the overall core damage frequency uncertainty.

5.5 Comparison of Results with The Reactor Safety Study (WASH-1400)

A comparison of the results of this study with those of WASH-1400 [4] is useful with full recognition of study differences in order to produce meaningful insights. In the over ten years between WASH-1400 and this study, the Peach Bottom plant design, as well as the industry's understanding of reactor operation and safety, has changed substantially. Any comparison of dominant contributors to core damage frequency between these studies must be balanced by a knowledge of the differences in plant design, study methodology, and success criteria considerations.

It is difficult to directly compare the total core damage frequencies calculated in the two studies. WASH-1400 calculated a total core damage frequency of approximately 2.6E-5, which is a sum of individual sequence median values (note that the sum is not necessarily a median value). This study has determined the median core damage frequency at Peach Bottom to be 1.9E-6 with a corresponding mean value of 4.5E-6. The modifications in plant configuration and procedures at Peach Bottom, consideration of realistic success criteria, as well as the evolution of analysis techniques since WASH-1400 have reduced the dominant results of the WASH-1400 study considerably. In fact, the two most dominant scenarios from the WASH-1400 study (transient with loss of long-term decay heat removal [TW] and ATWS [TC] have been decreased by approximately three orders of magnitude and over one order of magnitude, respectively. However, a more complete consideration of failures of DC-powered systems during station blackout and a more comprehensive treatment of common cause failures and support system (e.g., power, cooling...) failures combine to yield a mean core damage frequency of 4.5E-6. Some of the significant comparisons leading to these insights are presented below.

- Transients with loss of long-term decay heat removal are dominant in WASH-1400, but not in this study. This is primarily because of the consideration of containment venting procedures now in place at Peach Bottom as well as examining the survivability of core cooling systems even if the containment should fail.

- ATWS sequence frequencies are reduced over an order of magnitude in this study as compared to WASH-1400 because a more detailed analysis was performed which accounts for the provisions of the ATWS rule that have been put in place since WASH-1400. The corresponding procedures and plant modifications have reduced the core damage contribution from these sequences.

- Station blackout (loss of all AC) sequences are estimated to be a factor of five higher than in WASH-1400 because of a more complete consideration of potential failures of DC-powered systems during a blackout, a more complete common mode failure analysis (e.g., includes DC battery common mode failures), and a more complete analysis of support system effects on the AC power system (e.g., diesel cooling).

- All other transients and LOCAs combine to have a median CDF of 1.5E-6 in WASH-1400 and a median CDF of 7.5E-8 in this study. Thus, these sequences are a factor of 20 lower in this study.

- Based on the above, both studies conclude that transients, and not LOCAs, dominate the core damage frequency (and risk) at Peach Bottom. However, the types of transients are significantly different. WASH-1400 is dominated by ATWS and long-term heat removal failure sequences while this study is dominated by station blackout scenarios (47%) and ATWS (42%).

Table 5.11 summarizes the comparable core damage frequencies for the most dominant sequences as well as for the total core damage frequency results of both studies. The sum of the median frequencies from WASH-1400 is 2.6E-5. Although the overall TEMAC median result is 1.9E-6, the sum of the individual PDS median frequencies, which is comparable to what was done in WASH-1400, is 9.1E-7. Thus, in comparable terms, the core damage frequency from the NUREG/CR-4550, Revision 1 analysis on Peach Bottom is about a factor of 30 less than WASH-1400.

Table 5.11. Comparison of NUREG/CR-4550, Revision 1 & WASH-1400 Sequences (Most Dominant Only)

General Accident Type	4550, Revision 1 Plant Damage States	4550, Revision 1 Frequency Mean (Median) [See Note (a)]	4550, Revision 1 % of Total	Similar WASH-1400 Sequences	Approximate WASH-1400 Frequency Median Values	Approximate WASH-1400 % of Total
Station Blackout	4,5	2.1E-6 (4.5E-7)	47%	Part of TQUV & TPQUV	1.0E-7	<1%
ATWS	6,7,8,9	1.9E-6 (3.8E-7)	42%	TC	1.0E-5	39%
Transient-Loss of Long-Term Heat Removal	None	<1E-8		TW	1.4E-5	55%
LOCA	1	2.6E-7 (4.4E-8)	6%	A, S1, S2	7.9E-7	3%
Other Transient	2,3	2.3E-7 (3.1E-8)	5%	Part of TQUV & TPQUV	6.9E-7	3%
TOTAL		4.5E-6 (9.1E-7) Note (b)			2.6E-5	

NOTE

(a) Sum of means (or medians) of individual plant damage states

(b) Statistically combined totals are 4.5E-6 (1.9E-6)

6. CONCLUSIONS

The following subsections present overall conclusions and other insights based on the results of the Peach Bottom NUREG/CR-4550, Revision 1 analysis.

6.1 General Conclusions

One of the major purposes of the Peach Bottom analysis was to provide an updated perspective on our understanding of the risks from the plant, relative to the results of the WASH-1400 analysis [4]. It has been determined that changes to the plant design and its procedures, the evolution of Probabilistic Risk Assessment (PRA) methodology, and our increasing understanding of severe accidents have all impacted our perspectives on the dominant risks for Peach Bottom. While both WASH-1400 and this study agree that transients (and not loss of coolant accidents) dominate the Peach Bottom core damage frequency, our understanding of the most important types of transient scenarios has changed.

Unlike WASH-1400, this study concludes that station blackout (loss of all AC power) accidents and Anticipated Transients Without Scram (ATWS) scenarios are the dominant contributors to core damage. All other types of accidents are relatively insignificant. The possibility of successful containment venting and realistically allowing for successful core cooling after containment failure have considerably reduced the significance of the loss of long-term heat removal accidents which were originally found as important in WASH-1400. Giving credit for more injection systems, using best estimate system success criteria, and plant modifications have also collectively reduced the importance of loss of injection type sequences.

Given the considerable redundancy and diversity of coolant injection and heat removal features at Peach Bottom, it is not surprising that common features of the plant tend to drive the core damage frequency. These include common cause failures of equipment, failure of common support systems [AC power and Emergency Service Water (ESW)], and human error. In light of this conclusion, it must also be recognized that the calculated core damage frequency in this study is subject to the non-trivial uncertainties associated with the state-of-the-art in common cause and human error analyses. This calculated frequency is 4.5E-6 (mean value) as compared to 2.6E-5 (sum of individual sequence median values) in WASH-1400.

The above insights can be considered applicable to other boiling water reactors of similar design to the extent that the redundancy arguments are true for other plants of interest. However, numerous subtleties in plant design and operational practices and procedures make it difficult to draw specific conclusions for other plants on the basis of this analysis without performing plant-specific reviews, particularly as related to common cause potential and the location of equipment relative to possible phenomena such as steam entering the reactor building.

More specific conclusions and insights applicable to Peach Bottom are presented in the following subsections.

6.2 Plant Specific Conclusions

As stated above, the core damage profile is primarily made up of two general types of accidents as indicated below:

Accident Type	Mean Frequency	% Contribution to Mean Core Damage Frequency*
Station Blackout	2.1E-6	47%
ATWS	1.9E-6	42%
All Others	4.9E-7	11%

*Does not account for the <5% contribution of sequences <1E-8.

Making up the general accident types are eighteen individual, dominant sequences and nine plant damage states. These states were defined to properly bin "like" sequences with similar plant effects and to facilitate the subsequent containment analyses and development of risk profiles presented in other reports.

The accident sequence with the highest estimated mean frequency is a loss of all AC power leading to core damage in the long term. Core damage results from late failure of the turbine-driven High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems because of battery depletion or harsh environment effects, and the non-recovery of AC power in time to prevent core damage. This sequence represents approximately 36 percent of the total core damage frequency. The next most dominant sequence involves an ATWS event with failure of Standby Liquid Control (SLC) and ultimately core damage. This represents approximately 31 percent of the total core damage frequency. The remaining one-third is represented by sixteen accident sequences which individually contribute about 6 percent or less to the results.

From the plant damage state perspective, two plant damage states make up approximately 75 percent of the total frequency of core damage. These involve loss of all AC power with late failure of injection and ATWS scenarios with failure of SLC and the vessel at either high or low pressure during core melt. The remaining 25 percent are made up of seven other plant damage states which individually contribute about 7 percent or less to the results.

6.3 Uncertainty Considerations

The above conclusions are incomplete without considering the results of the uncertainty calculations. The total mean core damage frequency (4.5E-6) has an 95 percent upper bound value of 1.3E-5 and 5 percent lower bound of 3.5E-7 because of statistical uncertainty in the failure data as well as uncertainty in modeling issues.

Based on the uncertainty importance calculated in the study, the uncertainty in the results is driven by uncertainties in the scram failure probability, and the failure to run probabilities associated with the diesel generators. In addition, the battery depletion time in station blackout accidents and a variety of human error and equipment common cause failures also contribute significantly to the uncertainty in the results.

6.4 Other Insights

Based on the other two importance measures evaluated for the study, the following insights are noted. Failures which, if reduced significantly, would have the greatest effect in lowering the core damage potential include common mechanical failure of the control rods, human failures associated with SLC, human failures associated with miscalculation of the low-pressure permissive circuitry, operator failure associated with the use of the emergency heat sink mode of ESW, diesel failure-to-run probabilities, and two initiators (LOSP and transients with PCS initially available).

There are features whose availabilities should not be allowed to increase significantly or they could increase the core damage frequency considerably. These include common mechanical failure of the control rods, the probability of two or more stuck-open safety relief valves, battery common cause and independent hardware faults, and miscalibration of the low reactor pressure permissive circuitry for low-pressure cooling.

Besides the above, some additional insights are noted by the team analysts as a result of performing the PRA update of Peach Bottom. The recent availabilities of the diesel generators at Peach Bottom generally are a factor of ten better than the industry average. This appears to be based on a deliberate attention to detail in the test and maintenance practices as well as an attempt to determine the root causes of failures so that effective actions can be taken.

The importance of the Control Rod Drive (CRD) and High-Pressure Service Water (HPSW) systems as injection sources to the vessel (the latter as a last resort) came through clearly as the analysis evolved. The CRD system success probability might be further improved by examining whether the loss of air should be allowed to affect the operation of one of the CRD flow paths to the vessel. In addition, use of CRD under depressurized conditions in the vessel could cause insufficient net positive suction head for the CRD pumps.

An air pressure limit for SRV operation of approximately 100 psia could affect the capability to continue low-pressure core cooling under accident conditions when the containment is at high pressure (i.e., SRVs will not stay open). The purpose for this limit should perhaps be reviewed.

The conflicting requirements of first inhibiting the automatic depressurization system and later requiring a rapid depressurization in some ATWS sequences should also be recognized in operator training.

The difficulties associated with venting the containment in a station blackout and the harsh reactor building environments caused by venting in ATWS scenarios could have significant core damage and consequence effects which may need to be addressed.

Finally, the varied and more subtle failures of equipment because of unusual accident conditions should be made apparent to operational staff with sufficient warnings in the procedures of the possibilities of such occurrences. These include, for instance, turbine backpressure trip of RCIC when experiencing high containment pressure, the potential for HPCI and RCIC failure on high suppression pool temperatures, the closing of the SRVs under very high containment pressures, the potential for loss of low pressure core spray and residual heat removal pumps under low pressure saturated conditions in the containment, the possible effects of battery depletion in loss of all AC-type sequences, among others. It is these subtle and perhaps "unexpected" failure modes which affect multiple equipment in the analyzed scenarios and ultimately contribute to the core damage potential at Peach Bottom.

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11. ABSTRACT (200 words or less)

This document contains the appendices for the accident sequence analysis of internally initiated events for the Peach Bottom, Unit 2 Nuclear Power Plant. This is one of the five plant analyses conducted as part of the NUREG-1150 effort for the Nuclear Regulatory Commission. The work performed and described here is an extensive reanalysis of that published in October 1986 as NUREG/CR-4550, Volume 4. It addresses comments from numerous reviewers and significant changes to the plant systems and procedures made since the first report. The uncertainty analysis and presentation of results are also much improved, and considerable effort was expended on an improved analysis of loss of offsite power. The content and detail of this report is directed toward PRA practitioners who need to know how the work was done and the details for use in further studies.

The mean core damage frequency is 4.5E-6 with 5% and 95% uncertainty bounds of 3.5E-7 and 1.3E-5, respectively. Station blackout type accidents (loss of all AC power) contributed about 46% of the core damage frequency with Anticipated Transient Without Scram (ATWS) accidents contributing another 42%. The numerical results are driven by loss of offsite power, transients with the power conversion system initially available, operator errors, and mechanical failure to scram. External events were also analyzed using the internal event fault tree and event tree models as a basis, and are reported separately in Part 3 of NUREG/CR-4550, Volume 4, Revision 1.

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