New Hampshire
Yankee

NYN-91034

March 1, 1991

Document Control Desk
United States Nuclear Regulatory Commission
Washington, D.C. 20555

References: (a) Facility Operating License No. NPF-86, Docket No. 50-443

(b) USNRC Generic Letter 88-20, dated November 23, 1988, "Individual Plant Examination for Severe Accident Vulnerabilities"

(c) Supplement 1 to USNRC Generic Letter 88-20, dated August 29, 1989, "Individual Plant Examination for Severe Accident Vulnerabilities"

(d) NUREG-1335, dated August 1989, "Individual Plant Examination: Submittal Guidance"

(e) NHY Letter NYN-89136, dated November 1, 1989, "Response to Generic Letter 88-20", T.C. Feigenbaum to USNRC

Subject: Supplementary Response to Generic Letter 88-20

Gentlemen:

In the initial response to Generic Letter 88-20 [Reference (e)], New Hampshire Yankee (NHY) stated that the intent of Generic Letter 88-20 had been met for Seabrook Station by the completion of the Seabrook Station Probabilistic Safety Assessment (SSPSA) and the establishment of an ongoing risk management process. The SSPSA, a full-scope, level three probabilistic safety assessment applicable specifically to Seabrook Station, was initially submitted to the NRC for review in 1984. The risk management process was described by Reference (e) as having two principal objectives. The first was to maintain a current SSPSA by performing periodic updates. To this end, the most recent SSPSA update was completed in December 1990. This update addressed the Station configuration as of July 1990. The second risk management objective was to apply probabilistic safety assessment techniques to proposed Station design changes and improvements in methods of operation. NHY procedures have implemented this objective by requiring that significant Station design and methodology changes are evaluated for their impact on the SSPSA. In preparing the original SSPSA and in performing numerous subsequent risk studies, NHY and Yankee Atomic Electric Company (YAEC) personnel have been closely involved with technical experts in many aspects of accident behavior. Ongoing risk management activities are performed in-house by NHY and YAEC engineers. The risk management process addresses the request of Generic Letter 88-20 to base the IPE on the current plant configuration and to involve utility personnel in the IPE studies.

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In Reference (e), NHY stated that its Individual Plant Examination (IPE) Report would be submitted by March 1, 1991; and that the IPE Report would summarize the updated SSPSA and its associated studies. Enclosed is a copy of the IPE Report for Seabrook Station. This report arranges the findings and results of the updated SSPSA in a format consistent with that of the IPE submittal guidance (NUREG-1335). It should be noted that, because the SSPSA and its updates are full-scope evaluations, external hazard events are included within the scope, results and conclusions of the IPE Report. Based upon the extensive evaluations summarized in this report, NHY concludes that no significant core damage vulnerabilities exist at Seabrook Station. That is, there are no design features, equipment vulnerabilities or operator actions that would result in a significant likelihood of a severe accident over the life of Seabrook Station.

New Hampshire Yankee would be pleased to discuss the contents of the IPE Report with NRC Staff personnel should this be desired to facilitate NRC review. If you have any questions regarding the IPE Report, please contact Mr. Kenneth L. Kiper at (603) 474-9521, extension 4049.

Very truly yours,

Bruce L. Drawbridge
Executive Director
Nuclear Production

TCF:GK/act

cc: Mr. Thomas T. Martin
Regional Administrator
United States Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Gordon E. Edison, Sr. Project Manager
Project Directorate 1-3
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Noel Dudley
NRC Senior Resident Inspector
P.O. Box 1149
Seabrook, NH 03874

cc: (without enclosure)

Mr. Dave Modeen
Nuclear Management and Resources Council
1776 Eye Street N.W., Suite 300
Washington, D.C. 20006-2496

STATE OF NEW HAMPSHIRE

Rockingham, ss. March 1, 1991

Then personally appeared before me, the above-named Bruce L. Drawbridge, being duly sworn, did state that he is Executive Director of Nuclear Production of the New Hampshire Yankee Division of Public Service Company of New Hampshire, that he is duly authorized to execute and file the foregoing information in the name and on the behalf of New Hampshire Yankee Division of the Public Service Company and that the statements therein are true to the best of his knowledge and belief.

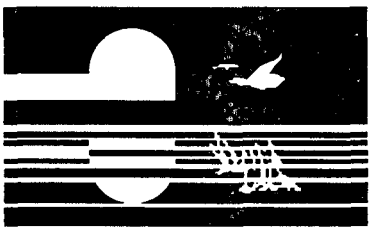
Beverly E. Silloway

Beverly E. Silloway, Notary Public
My Commission Expires: February 28, 1995

Individual Plant Examination

Report for Seabrook Station

March 1991



New Hampshire
Yankee

INDIVIDUAL PLANT EXAMINATION

REPORT FOR

SEABROOK STATION

Response to Generic Letter 88-20

New Hampshire Yankee Engineering Report No. 91-01

Prepared By:

Kenneth L. Kiper — 2-15-91 (Date)
New Hampshire Yankee

Patrick J. O'Regan — 2-14-91 (Date)
Yankee Atomic Electric Company

Darvin M. Kapitz — 2/14/91 (Date)
Yankee Atomic Electric Company

John F. Bretti — 2/14/91 (Date)
Yankee Atomic Electric Company

Reviewed By:

Larry W. Rau — 2/15/91 (Date)
New Hampshire Yankee

James R. Chapman — 2/14/91 (Date)
Yankee Atomic Electric Company

Approved By:

Joseph M. Vargas — 2/19/91 (Date)
New Hampshire Yankee

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INDIVIDUAL PLANT EXAMINATION REPORT FOR SEABROOK STATION

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1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

This report was written in response to Generic Letter 88-20 (Reference 1) which requested "each existing plant (to) perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission." In our initial response to the Generic Letter (Reference 2), we stated that "NHY has already fulfilled the intent of Generic Letter 88-20" based on the existing Seabrook Station Probabilistic Safety Assessment (SSPSA, Reference 3) and subsequent studies that have been extensively reviewed by the NRC. This report summarizes the results and conclusions of the SSPSA as it has been updated through July 1990. The guidance of NUREG-1335 (Reference 4) was used in preparing this report.

The SSPSA is a full- scope, level three probabilistic safety assessment applicable specifically to Seabrook Station. The SSPSA was completed in 1983 by a team of contractors lead by PLG, Inc., with significant utility involvement in the form of information input and detailed technical reviews. It was submitted to the NRC for information in January 1984 (Reference 5). Two reviews of the SSPSA were performed for the NRC: one of the plant model (by Lawrence Livermore National Lab), and another of the containment model (by Brookhaven National Lab). The conclusions and comments from these reviews are summarized in Appendix A. The SSPSA provides the base-line risk model for subsequent updates as well as for this IPE report.

A number of applied-risk studies have been performed for Seabrook Station since the original SSPSA was completed in 1983. These studies were performed to address specific issues as well as to update and enhance the risk modeling by incorporating new insights regarding severe accident behavior. A summary of these studies is given in Appendix B for background information and to indicate the extent of the ongoing risk management program at NHY.

As part of the risk management program, three major updates of the original SSPSA have been made in order to account for important plant configuration changes as well as updates to analysis and methodology. These updates are titled the "Seabrook Station Probabilistic Safety Study" (SSPSS) to distinguish the study that will be updated from the original base line study - the SSPSA. These updates, SSPSS-1986 (Reference 6),

SSPSS - 1989 (Reference 7), and SSPSS - 1990 (Reference 8), are also summarized in Appendix B. The method of maintaining the SSPSS updated is described in Appendix C.

Because the SSPSA and its updates (SSPSS) are full-scope evaluations, external hazard events are an integral part of the model. As a result, they are included in results and conclusions of this report. Their significance relative to internal events highlights the importance of considering the full spectrum of risks in evaluating potential plant enhancements. A summary of external events analysis is contained in Appendix D. This analysis will be updated in the future consistent with the ongoing risk management program as well as the consideration of the anticipated Generic Letter on IPE External Events.

Thus, this report is a summary of the results of the current SSPSS (1990 Update) and a description of the process of achieving the results.

1.2 Plant Familiarization

As part of performing the original SSPSA, extensive reviews of plant documentation and numerous plant walkdowns were performed by contractors (principally PLG, Inc.) with assistance from utility personnel - engineers, operators, training instructors, and plant management from NHY and YAEC. Walkdowns of Unit 1, which was nearing completion, and Unit 2, which was in the early stages of construction, allowed detailed inspections of all areas of the plant. Since completion of the original SSPSA, subsequent studies have been performed using the same contractor team with significant utility personnel involvement. In addition, the updates to the SSPSA have been the direct responsibility of the utility. Thus, through all stages of risk studies, utility personnel and the original contractor team have developed a thorough familiarization with Seabrook Station.

As a part of the process of maintaining the SSPSS up-to-date with the plant, the Reliability and Safety Engineering Department (RSED) at Seabrook Station reviews each permanent plant change prior to its installation. In addition to permitting the inclusion of risk insights in the design, the review also allows an ongoing familiarization with the plant as it changes.

1.3 Overall Methodology

The basic methodology of the current SSPSS is the same as that used in the original study. This methodology, which can be described as "the event tree linking

approach," is based on the objectives to model important dependencies explicitly in the event trees and to describe risk as a listing and analysis of scenarios. Figure 1-1 shows the risk model at the highest level, consisting of sequences of plant, containment, and site response. At each level in the risk model, results in terms of sequences are produced which allows the risk profile of the plant to be viewed from a number of perspectives. (Note: The site model is not included in this report but is an integral part of the SSPSS.)

These three models are developed and linked via logic rules (between plant and containment) and bins (release categories between containment and site). Figure 1-2 illustrates the plant model, which consists of sequences made up of:

- Initiating events,
- Support system availability given the initiating event, and
- Frontline system and operator response to the initiator, given the support systems status.

Analyses of systems, operators, hazards, and data are performed to build up these models.

These blocks were built and then combined using the PLG computer code RISKMAN (Reference 9) to effectively create a single, large tree from initiating event to release categories within the computer. This software allows linking of all scenarios with significant frequency without the need for support states or impact vectors to accomplish the linking. While support states and plant damage states were used in the original SSPSA to support the linking, the current software allows linking of sequences by use of logic rules.

As Figure 1-2 illustrates, initiating events are analyzed and quantified from data (e.g., general transients), hazards (e.g., flood initiators), and systems analyses (e.g., loss of support systems). The support system availability analysis is based primarily on systems analysis (also including seismic equipment fragility). The plant response is based on systems analysis and operator action analysis. The systems analyses were performed using fault trees and/or reliability block diagrams in order to quantify the contributions of hardware failures, common cause failure, human errors, and unavailability due to test and maintenance.

At the lowest level is data analysis, which includes generic data for initiating event frequencies, component failure rates and maintenance unavailabilities, and common cause failure fractions (beta factors).

This systematic, structured approach to constructing the risk model allows for a high level of completeness and permits unraveling the results to understand the key risk-controlling factors that drive the results. This, in turn, supports the development of engineering insights needed to use the PSA as a risk management tool.

1.4 Summary of Major Findings

This section summarizes the major results and findings of the current SSPSS-1990. Detailed results are located in Section 3.4; Appendix B summarizes results from previous updates. In addition, conclusions and results are found in Section 13 of the SSPSA (1983) and Section 9 of the SSPSS (1990). While the order of sequences and the relative contributions of initiators have changed since the original study, most of the significant insights at the core damage level have not changed. Important new insights regarding contributors to public risk have been gained by studies done subsequent to the SSPSA.

1.4.1 Core Damage Frequency Results

The key results of the current SSPSS-1990 are a quantitative estimate of risk and, more importantly, an understanding of the basis for this risk. As a quantitative measure of risk, the annual frequency of a core damage accident at Seabrook is estimated to be 1.1E-4 per year (mean value). This measure of risk has decreased by about a factor of two from the original SSPSA due primarily to updates in analysis and data.

The basis for the core damage risk can be understood by examining important contributors. First, almost 70% of the core damage frequency is due to two functional accident sequences:

1. Station blackout (loss of off-site and on-site AC power), resulting in a Reactor Coolant Pump (RCP) seal LOCA with no primary system makeup (35%); and

2. Loss of component cooling also resulting in a RCP seal LOCA with no makeup (34%).

These sequences point out the most significant limiting characteristic of the plant to severe accidents - RCP seal LOCA. This is the result of the design configuration and not from unreliable equipment or operator actions. This issue is also strongly affected by uncertainties in the size and timing of the LOCA and the effect of depressurization.

Sequences can be broken down into initiating events and system and operator responses. Figure 1-3 illustrates the risk contributors by initiating event category. Transient initiators, including general transients (e.g., turbine trip, loss of feedwater), loss of off-site power (LOSP), and loss of support trains (e.g., loss of a train of Primary Component Cooling (PCC)), account for 83% of the core damage frequency. These initiators result in a "normal" plant trip but have subsequent loss of RCS integrity and failure of makeup or loss of all decay heat removal. This includes the RCP seal LOCA sequences identified above, as well as sequences with failure of secondary cooling (Emergency Feedwater) and primary cooling (feed and bleed). LOCA initiators, e.g., losses of primary inventory large enough to result in a plant trip and SI actuation, account for about 8% of the core damage frequency. Most of these contributors are due to loss of RHR following a small LOCA. Finally, transient initiators with subsequent failure of reactor trip, anticipated transients without scram (ATWS), account for about 9%.

Systems important to core damage risk include primarily support systems - AC power, Primary Component Cooling Water (PCC), Service Water (SW) - consistent with the important sequences. The important operator actions are primarily recovery actions due to the automated design of safety features. These recovery actions include electric power recovery, signal recovery, service water recovery (manual cooling tower actuation), and EFW recovery (manual start-up feed pump start, turbine-driven pump restart).

Accident sequences can also be classified as internal event or external event-initiated. From Figure 1-4, it can be seen that, based on mean values, external events make a significant contribution to the calculated core damage frequency, dominated by fire, seismic, and flood initiators. This relative contribution which is due in part to the relatively high levels of uncertainty for externals, has increased from the original SSPSA due to emphasis on updating the analysis and data for internal events. Additional analysis and update of external events is planned as part of the ongoing risk management program.

Thus, based on the extensive evaluations summarized in this report, no significant core damage vulnerabilities exist at Seabrook Station. That is, there are no design features, equipment unreliabilities, or operator actions that would result in a significant

likelihood (greater than one percent) of a severe accident over the life of the Seabrook Station. The most significant limiting characteristic of the plant, as identified above, is the RCP seal LOCA. Several potential enhancements are being evaluated, including high temperature seal O-rings and an independent seal cooling system. The final decision regarding these enhancements will not be made until the update to external events, described above, has been completed.

1.4.2 Containment Performance Results

Containment performance results are shown in Figure 1-5 for four general containment responses following a core melt accident. These are described as follows, in order of increasing severity:

- Intact Containment - 20.2%

  Given a core melt, with the containment isolated, with containment spray and heat removal functioning, and with the containment surviving the initial blowdown, containment will remain intact in the long term. The release to the environment is restricted to the containment leakage limited by Technical Specifications, with essentially no public health effects.

- Late Containment Failure - 65.4%

  Given a core melt, with the containment isolated and with the containment surviving the initial blowdown, but with no containment heat removal, the containment will eventually overpressurize and fail structurally. The time to containment failure is very long (>24 hours) because of the large volume in containment and the strength of the containment. This allows adequate time for an immediate emergency response, i.e., evacuation, so that the only potential public health effects are latent effects. This long time would also allow opportunities for recovery of failed equipment (e.g., diesel generator, spray pump); however, no credit has been taken for post-core melt recovery. This issue will be addressed as part of accident management.

- Early, Small Containment Failure/Bypass - 14.2%

  Given a core melt, with the containment isolated except for a single three-inch diameter opening, or with initial blowdown causing failure of penetrations of less than three-inch diameter, a larger initial leakage results. This size

opening is too small to relieve the pressure buildup in containment, so eventually the containment also fails due to overpressurization, as in the previous failure mode. This results in a very small potential for early health effects, but generally, the health effects are similar to late containment failure.

- Early, Large Containment Failure/Bypass - 0.2%

  Given a core melt, with a large containment penetration not isolated, or with the containment bypassed (i.e., inter-system LOCA), or with gross structural failure due to the initial blowdown, a large opening exists with the potential for early as well as latent health effects. This category of containment performance is used to define the term "unusually poor" containment performance used in the Generic Letter and NUREG-1335. This category has a small conditional probability given a core melt (0.002) as well as a small absolute value (2.1E-7 per year).

Figure 1-6 shows the specific containment failure modes that contribute to "unusually poor" containment performance. Containment isolation failure includes failure of the eight-inch containment on-line purge valves to close due to loss of protection signals and failure of the operators to manually initiate equipment. This operator action was quantified with conservative screening values. A more detailed evaluation of this action, planned for a future update, is expected to reduce this contribution. The other significant contributors, induced steam generator tube rupture and direct containment heating, are phenomenological issues associated with high pressure in the RCS at time of core melt. A very small contribution comes from a number of causes, including LOCA outside containment (RHR isolation valve failure, RHR pipe rupture outside containment); steam and/or hydrogen explosion at vessel blowdown, and external hazards impacting containment (aircraft, turbine missile).

The frequency of early, large containment failure in the SSPSA (1983) was about 2.4E-6 per year, with a conditional probability of 1.0%. This was dominated by LOCAs outside containment - initiated by catastrophic failure of RHR isolation valves with subsequent RHR pipe rupture due to overpressure. The current results reflect detailed analysis of LOCAs outside containment (Reference 11) as well as new containment performance issues - direct containment heating and induced steam generator tube rupture (Reference 10). Thus, the current results are not only significantly lower, but also more complete.

Thus, based on the evaluations summarized in this IPE Report, no significant containment performance vulnerabilities exist at Seabrook Station. Potential changes in emergency procedures have been identified to reduce the likelihood of direct containment heating and induced SGTR - i.e., assure that the RCS is at low pressure at time of core melt. These changes will be evaluated as part of an integrated accident management plan.

1.4.3 Vulnerability Findings

The basic finding of the extensive evaluations summarized in this report is that there are no fundamental weaknesses or vulnerabilities with regard to severe accidents at Seabrook Station. The term vulnerabilities, as used in this report, refers to "those components, systems, operator actions, and/or plant design configurations that contribute significantly to an unacceptably high severe accident risk." Based on this definition, as expanded in Section 3.4.2.3 of the report, no vulnerabilities exist because of the low risk from severe accidents.

Several risk enhancement opportunities have been identified to address specific limiting plant features, including RCP seal LOCA and loss of SF_6 switchyard. In addition, the need for an updated analysis of fire and seismic hazards has been identified based on the relative importance of external events. These are being evaluated as part of the ongoing risk management program.

Figure 1-1

Risk Model Overview

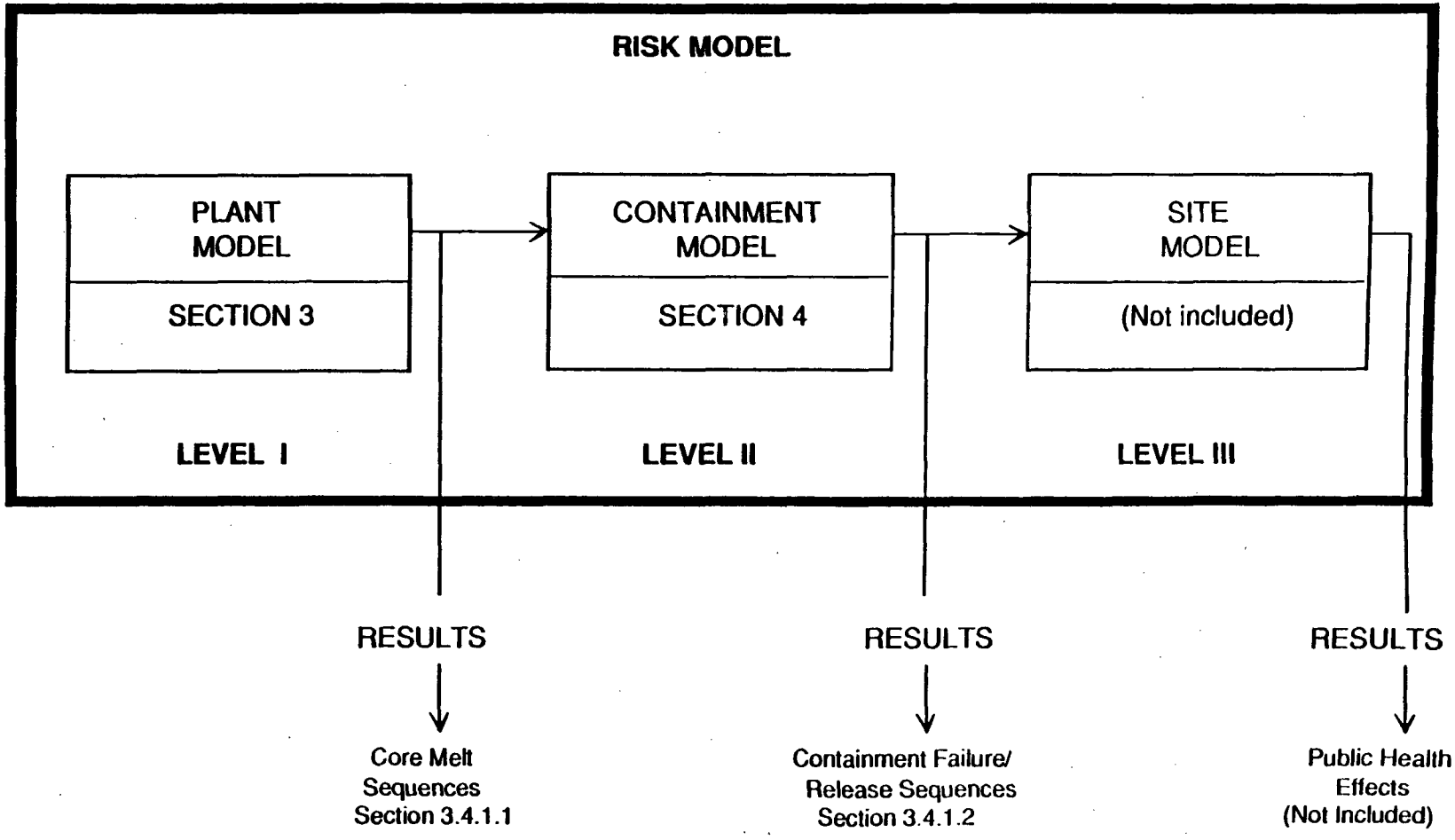


Figure 1-2 Plant Model Overview (with IPE Report Section References)

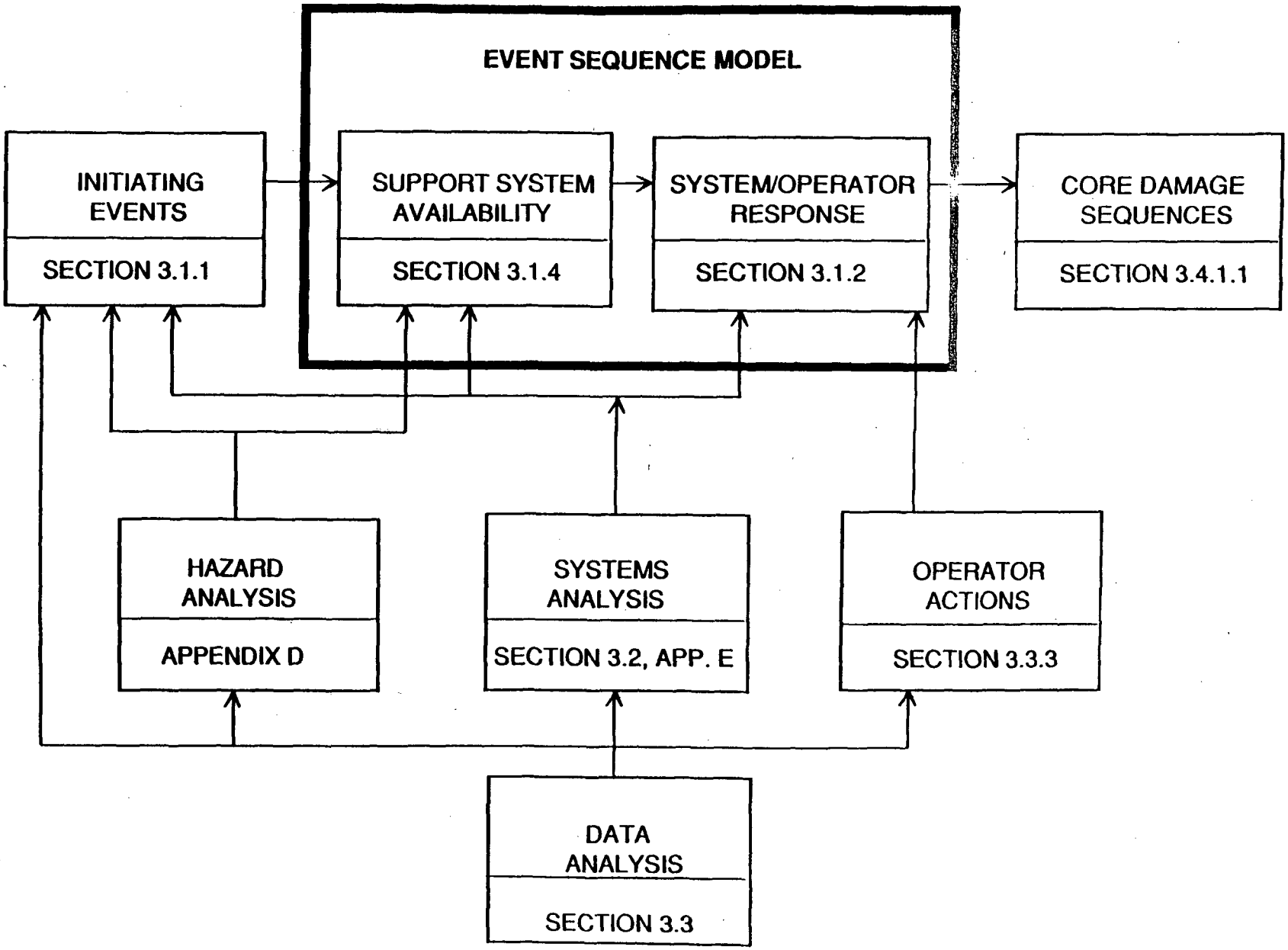


FIGURE 1-3
CONTRIBUTIONS TO CORE DAMAGE FREQUENCY
Accidents Grouped by Initiating Event

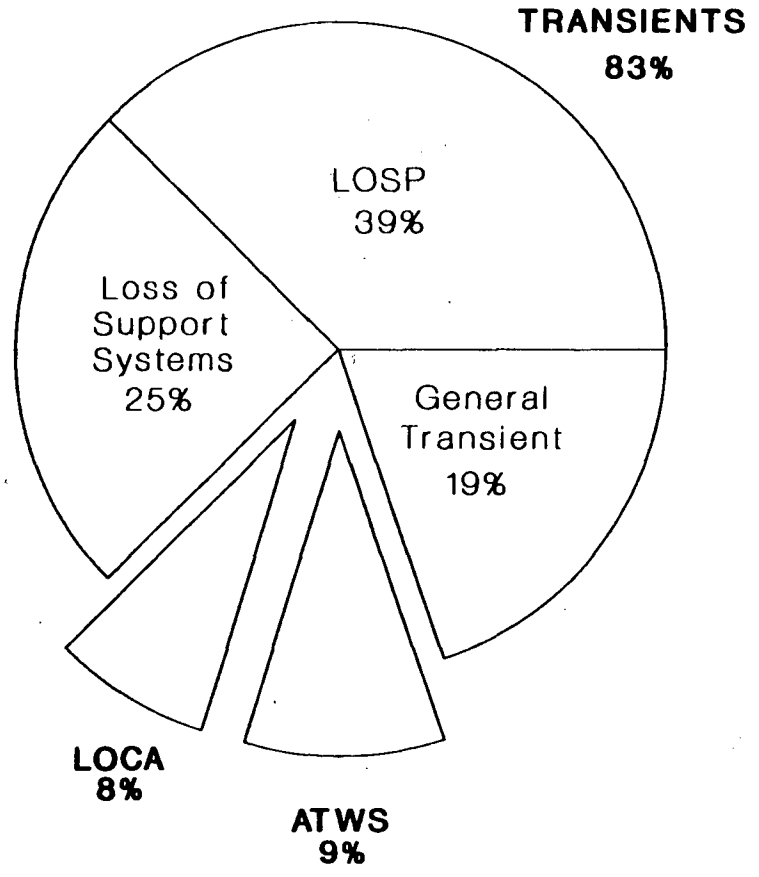


FIGURE 1-4
CONTRIBUTIONS TO CORE DAMAGE FREQUENCY
Accidents Grouped by Internal and External Initiating Events

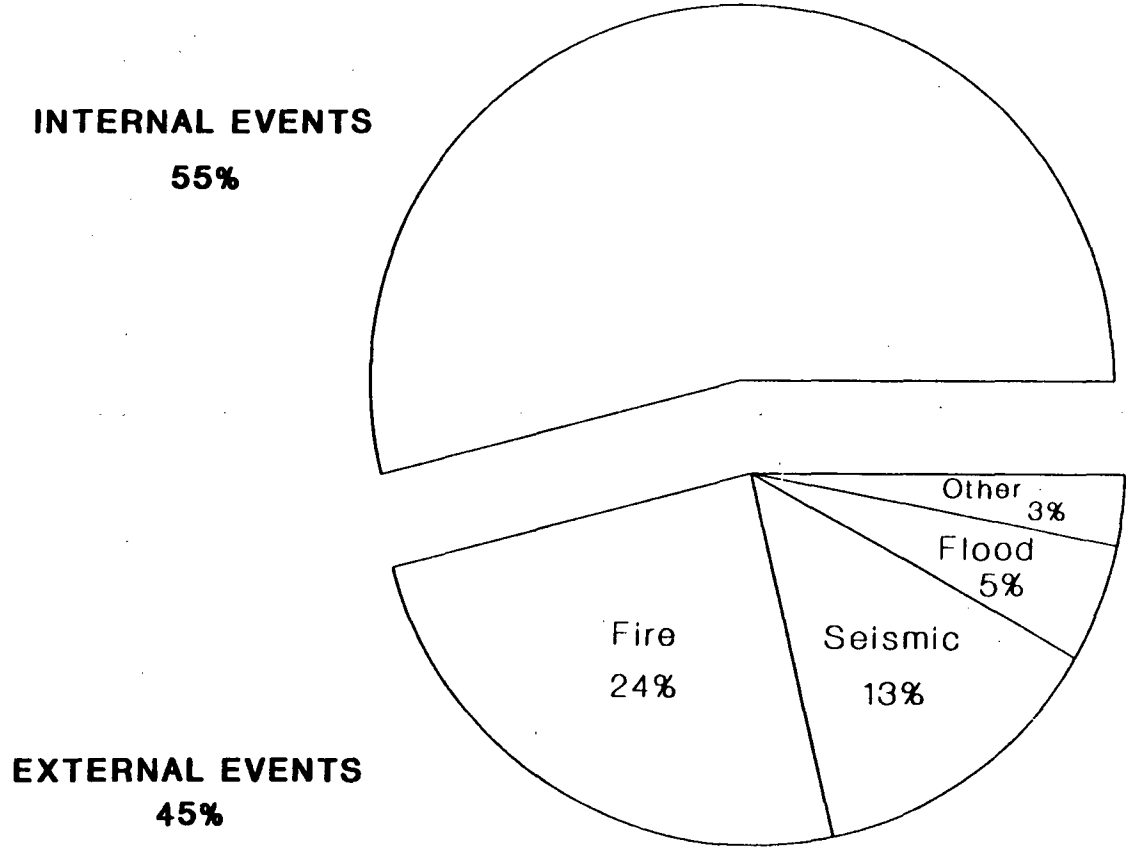
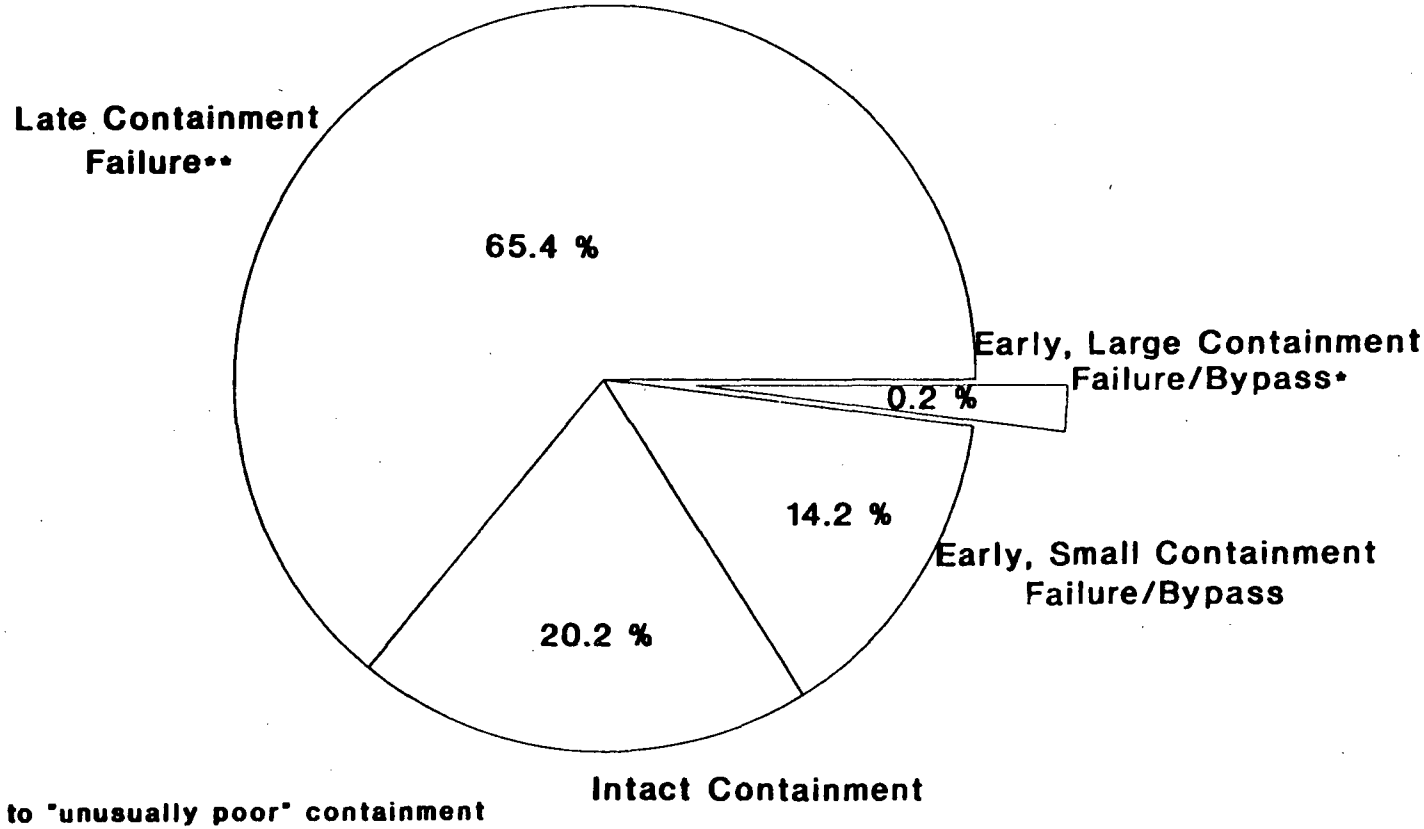


FIGURE 1-5
CONTAINMENT PERFORMANCE RESULTS
(Conditional Failure Probability Given Core Damage)

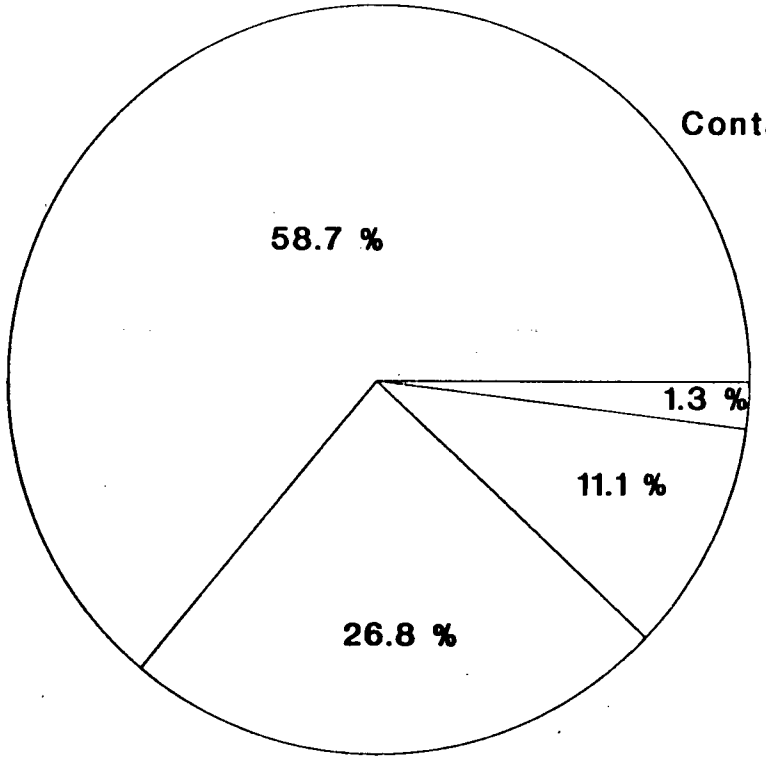


* Equivalent to "unusually poor" containment performance, as defined in GL 88-20

** The containment failure probability of late containment failure is believed to be overestimated relative to containment intact. No credit has been taken for post-core melt recovery actions.

FIGURE 1-6

CONTAINMENT FAILURE MODE CONTRIBUTIONS TO EARLY, LARGE CONTAINMENT FAILURE/BYPASS
("unusually poor" containment performance)



Containment Isolation Failure

Other

Direct Containment Heating

Induced Steam Generator Tube Rupture

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

The Individual Plant Examination (IPE) for Seabrook Station was performed on the basis of the original SSPSA (Reference 3) and ongoing risk management activities. This report summarizes the examination process that was performed in 1982 and 1983 for the SSPSA, the continuing process of updating the risk model, and the results of the latest update (SSPSS-1990, Reference 8).

The method of examination used in the SSPSA and subsequent updates is the standard PRA method, consistent with the first method identified in Section 4 of the Generic Letter 88-20 (Reference 1). This method requires the following elements:

- Provide at least a Level I analysis. The risk models in the SSPSA and update models are generally to Level III, although results in this report are limited to Level II.

- Use current methods and information. State-of-the-art methods and current plant information were used in the original SSPSA and in subsequent updates.

- Consider the most current severe accident phenomenological issues. Several studies done since the SSPSA were to specifically address these evolving issues. These are described in Section 4 and Appendix B.

- Certify that the PRA is based on the most current design. This is accomplished via the model updates, as discussed in the next section and in Appendix C.

2.2 Conformance With Generic Letter and Supporting Material

The generic letter identified that for plants using an existing PRA, the following three items were needed:

- "Certify that the PRA meets the intent of the generic letter, in particular with respect to utility involvement."

- "Certify that it reflects the current plant design and operation."

- "Submit the results on a shorter schedule.

This report is being submitted on a shorter schedule. The other two items are satisfied in the SSPSA and its updates as described below.

2.2.1 Intent of the Generic Letter

The Generic Letter 88-20 identified four purposes for each utility in performing the IPE. New Hampshire Yankee (NHY) has satisfied these as follows:

1. Develop Appreciation of Severe Accident Behavior - In preparing the original SSPSA and in performing numerous subsequent risk studies, NHY and Yankee Atomic Electric Company (YAEC) personnel have been closely involved with technical experts in many aspects of accident behavior. Appendix B lists the major studies and indicates increasing utility involvement to the extent that the SSPSS updates have been done virtually all in-house.

2. Understand Most Likely Severe Accident Sequences - The SSPSA and the SSPSS updates have consistently shown the same set of dominant sequences, i.e., RCP seal LOCA due to station blackout or loss of other basic support systems. In an effort to better understand these sequences and to reflect the most current data and analyses, several studies were done involving electric power recovery (Reference 12), RCP seal leakage (Reference 13), and off-site power recovery (Reference 14). These studies allowed examination of the important parameters input to each analysis (e.g., battery lifetime). In addition, a series of studies of the dominant sequences to early containment failure (References 10, 11, and 15) has allowed a detailed understanding of risk from this perspective.

3. Gain a More Quantitative Understanding of the Probabilities of Core Damage and Releases - The use of the RISKMAN suite of computer codes (Reference 9) has allowed an unraveling of the risk at either core damage or release categories. In addition to a ranked list of sequences, this software quantifies the core damage or release contribution for each initiating event and the risk importance of each split fraction, i.e., system or operator failure. This permits a quantitative understanding of the contributors to risk in terms of initiating events, specific accident sequences, group of sequences, and equipment and operator action importance measures.

4. Reduce, if Necessary, the Overall Probabilities of Core Damage and Releases - The original SSPSA did not identify any significant vulnerabilities, i.e., any plant design or operational features that would result in significant risk when compared to the proposed safety goals. As a result of subsequent studies to better understand the early release risk, as well as ongoing risk management activities to update the PSA, the quantitative estimate of risk has been significantly reduced. These studies included a detailed analysis of the LOCA outside containment sequences (Reference 11), an updated equipment fragility analysis (Reference 16), and updated generic data in SSPSS-1989 and SSPSS-1990 updates. As a result, the mean estimate of core damage has decreased from 2.3E-4/yr in the SSPSA (1983) to 1.1E-4/yr in the SSPSS-1990. Also, the conditional probability of early containment failure or bypass has decreased from 1.0% to 0.2%. These results illustrate the importance of evaluating conservative analyses before considering plant design change.

Thus, NHY has satisfied both the letter and the spirit of the generic letter by its original study, the SSPSA, and by the subsequent studies that have had increasing utility involvement. The current risk management activities are performed and directed in NHY by the Reliability and Safety Engineering Department.

2.2.2 Current Plant Design/Operations

The original SSPSA (Reference 3) was performed in 1982 and 1983, and involved the then-current plant design documents, plant procedures (some of which were in draft), and plant configuration. Since then, for each subsequent risk study, the applicable plant documents were reviewed and changes incorporated as necessary. Specifically, the emergency operating procedures have been reviewed extensively and modeled in the studies. The process of updating for the SSPSS has involved a review of plant design documents and/or design change requests. The current update, SSPSS-1990, has been updated for design changes through July 1990. This process has been proceduralized as part of the risk management program.

2.3 General Methodology

This section contains a summary of the general methodology used to develop the Seabrook Station risk models. Descriptions of specific aspects of the methodology are

included throughout this report. A detailed description of this methodology is contained in the SSPSA (Section 4 and Appendix A). A description of the RISKMAN software methodology used in the SSPSS-1990 is contained in Reference 9.

2.3.1 Seabrook Risk Model

A PSA is basically a listing and analysis of accident scenarios, and a full scope PSA can contain literally billions of scenarios depending on how finely the scenarios are described. To provide some logic to the qualitative progression of an accident scenario, the overall risk model can be thought of as three linked models: the plant model, the containment model, and the site model, as shown in Figure 1-1. A single accident scenario progressing to off-site consequences spans all three of these models. For most accident scenarios, the input to the containment and site models depends only on the state of the plant or containment and not on the history of the arrival to that state. This property is a result of the detailed treatment of the plant damage states that form the interface between the plant and containment models.

1. The Plant Model

A wide set of different accident sequences must be considered in the plant model. This requires detailed modeling of the plant, systems, and components, and their interdependencies. Physical and human interactions with the plant can affect the frequency of occurrence of an accident scenario as well as its outcome and are also included.

Event frequencies and their associated uncertainties are quantified using historical experience in both nuclear and non-nuclear industries as applicable. The plant model contains all the systems reliability aspects, including the engineered safety features of the containment. The containment model (explained below) deals only with the phenomenological issues of containment response once core damage occurs.

2. The Containment Model

The containment model represents the subsequent progress of a scenario once core damage or melt is experienced. The outcome of the scenario is principally determined by the physical processes of the scenario (for example, the pressure and temperature response, the cooling of core debris, etc.) as well as the passive response of the containment itself.

The containment event tree models the scenario in approximately chronological order and gives special consideration to effects that specific plant features have on the accident simulation. The results of the model are a continuation of the scenario structure, expressed by release categories, quantification of their frequencies, and a source term for estimating accident consequences.

3. The Site Model

The site model represents the progression of scenarios from the release categories output from the containment model to the actual off-site impacts. This model is outside the scope of the IPE report and, thus, will not be discussed further.

2.3.2 Logical Structure of a PSA Model

The logical structure of a PSA model is shown in Figure 1-2. The first step in the model is to identify "initiating events." These are identified using several independent approaches including a fault tree analysis of the plant energy balance, a "master logic diagram," which is another form of a fault tree, failure modes and effects analysis of plant systems, and cross-checks against reactor operating experience and events identified in other PSAs.

Once the initiating events are identified, scenarios or accident sequences that could result are identified using a "plant event tree." The top events of the tree represent the functioning of the various systems, so that each path through the tree represents an accident sequence. At the end of each sequence, the plant either is in a stable, recovered condition, or has suffered some degree of core damage. A set of plant states is defined, and each path through the tree is assigned to one of these states.

2.3.3 Dependent Event Methodology

Dependent events include common cause initiating events, intersystem dependencies, and intercomponent dependencies. In view of the different types of dependent failures, the variety of physical and human interactions that cause them, and the multifaceted needs of PSA, there is no single approach or method of dependent failure analysis. Rather, it is necessary to apply a number of different techniques and to analyze a great deal of information and data to truly capture the essence of dependent failures in

PRA studies and system reliability analyses. There are two general approaches to dependent failure analysis, explicit and parametric, explained as follows.

Explicit methods involve the identification of specific causes of multiple failures and, in some cases, the use of models that are appropriate for a specific physical or human interaction. The explicit modeling category includes the direct incorporation of dependent failure causes into the event tree and fault tree logic models as well as the seismicity and fragility models used to analyze the risks of seismic-induced accident sequences.

Parametric methods are used to estimate the reliability characteristics of systems subject to common cause failure. In these methods, parameters are used to model the effects of the failure dependence without having to enumerate the specific causes directly in the model. In as much as the parameters are estimated from experience data in much the same manner as component failure rates are estimated, the parametric methods implicitly account for all causes of multiple failures present in the systems from which the data is collected. This approach is consistent with the way in which independent failures are normally modeled in that the root causes of failure are implicit in the assignment of component failure rates, but not explicitly modeled.

In the SSPSA and in the SSPSS Updates, a comprehensive approach for the treatment of dependent failures has been followed. Table 2-1 lists the coverage of dependent failures throughout this study. As can be seen from Table 2-1, the major thrust of dependent failure analysis is focused in the event sequence and systems analysis tasks. In the former task, the master logic diagram method, the heat balance fault tree method and a specialized failure modes and effects analysis procedure are applied to identify common cause initiating events. Also, functional and shared equipment dependencies among systems (types 2A and 2B) are modeled explicitly in the event tree logic.

The systems analysis task involves the analysis of all types of dependent failures, principally because this is the task in which the plant is conceptually disassembled and reconstructed to facilitate risk quantification and to acquire an intimate knowledge of plant design, operation, and maintenance. The methods employed in the task include explicit modeling and an advanced version of the beta factor method which provides a means of incorporating all relevant experience with common cause failures. All remaining tasks draw heavily from the systems analysis task in their analysis of dependent failures.

In the data analysis task, evidence relative to initiating events and common cause failures is used to quantify the frequency of initiating events, to explicitly model common cause failures, and to develop beta factors in support of systems and event sequence analyses.

The event sequence and systems analyses tasks provide a thorough coverage of interactions between physically connected and functionally related systems. These basic tasks also address interactions between nonconnected and nonfunctionally related systems; however, the information normally processed in these tasks provides a limited ability to incorporate all possible interactions in this category. To address this gap, a special task was performed in the SSPSA to address spatial interactions between and among all systems. This task includes the use of a separate plant model that explicitly models all localized interactions and the performance of in-depth physical inspection of the plant layout. This task provided a comprehensive coverage of all physical interactions, including those in categories 1A, 2C, and 3C, and enabled a more comprehensive treatment of external events.

Among the dependent failures that are explicitly modeled are the so-called external events which comprise a major segment of the possible causes of physical interactions leading to multiple failures.

The above tasks largely address the physical interactions that give rise to dependent failures. The remaining interactions are related to human actions. These interactions, especially those that influence multiple systems and the unfolding of accident sequences, were the subject of a second special task performed on this project to effect a full treatment of dependent failures. In this task, insights from running key accident sequences on the Seabrook Station training simulator with an operations crew were used to help model operator actions such as cognitive errors, errors of omission and commission, and recovery actions in the event sequence logic. Operating experience data was collected and analyzed to supplement the available evidence on human error rates and reliability.

2.3.4 Quantification

The process of integrating and quantifying the Seabrook Station risk model involved use of the RISKMAN suite of computer codes. This software package uses IRRAS for Boolean reduction of systems analysis fault trees. In addition, common cause failures and multiple system alignments can be added before reduction. Quantification can be either point-estimate or Monte Carlo uncertainties. The results of the systems analysis are written in terms of split fractions for use by the event trees. The event trees are quantified

using a user-specified cutoff frequency which allows the construction of very large trees and the bypassing of insignificant sequences. The results also report the "unaccounted for" frequency to allow the user to evaluate the effect of the cutoff. For final runs, a 1.0E-10 cutoff was typically used in the plant model quantification, and a 5.0E-13 cutoff was used in the containment model quantification.

The results of the event tree quantification can be reported in terms of a list of dominant sequences, the initiating event contribution to core damage or release, and the top event or split fraction importance rankings. Each of these reports allows a different view of the Seabrook Station risk and aids in identifying any potential vulnerabilities.

In addition, the logic rules or split fraction values can be changed and the model quickly rerun to permit sensitivity studies.

2.4 Information Assembly

The SSPSS includes, directly or by reference, the plant information used in the examination. Most of the plant layout information is contained in the FSAR (Reference 17). Additional details on containment design can be found in the SSPSS-1990, Section 7.0 (and other risk studies referenced therein).

When the original SSPSA was performed in 1982 and 1983, insights from the Zion and Indian Point studies were factored in directly by the consultant (PLG) who was also directly involved in the earlier studies. Insights from these studies included the importance of dependencies and common mode failures: support system failures, common mode component failures, and external hazards. The SSPSS updates have used results from other studies in the areas of generic data and specific analyses, e.g., the RCP seal LOCA assumptions from NUREG-1150 (Reference 18).

The results of the Zion NUREG-1150 analysis (Reference 64) were reviewed based on its similar design to Seabrook Station. The RCP seal LOCA is the dominant contributor to core damage frequency at both plants. This is due to a similar design for seal cooling, i.e., thermal barrier cooling and seal injection cooling, both of which depend on primary component cooling water. However, specific sequences differ due to differences in plant specific design and modeling. Most important is the difference in configuration of component cooling and service water systems. At Zion, six SW pumps and five PCC pumps can supply either, or both, units. At Seabrook, there are two physically separated trains with three SW pumps per train; including a Cooling Tower pump which is diverse in size

and location. In addition, PCC consists of two physically separated trains with two pumps per train. Thus, common cause failure due to pipe breaks, important to the Zion results, do not apply to Seabrook. Other design differences (e.g., Zion swing diesel, manual switchover to low pressure recirculation) and the use of substantial amounts of plant-specific data are responsible for other differences in results - for example the relative importance of recirculation and unimportance of station blackout at Zion versus Seabrook.

As described in the previous section, the "process used to confirm that the IPE represents the as-built, as-operated plant," is based on the ongoing risk management activities which have been formalized in an engineering procedure. Per this procedure, the Reliability and Safety Engineering Department reviews each design change document prior to implementation. This allows an early identification of important plant changes as well as an opportunity to provide input to the change based on risk insights.

A number of walkdowns have been performed during the initial SSPSA and for subsequent studies covering every aspect of the plant. During each walkdown, the utility personnel from Engineering and/or Operations were involved. These walkdowns included:

- Systems walkdowns - system familiarity.
- Spatial interactions walkdowns - including considerations of fire, flood, and seismic affects.
- Containment walkdowns.
- Containment bypass walkdowns.

In addition to these walkdowns, the PSA team performed extensive investigations on the Seabrook Station training simulator to examine operator action behavior during specific accident sequences.

TABLE 2-1

Coverage of Dependent Failure Types in SSPSS-1990 Project Tasks

Dependent Failure Type	Subtypes	SSPSS Plant Analysis Tasks: Event Sequence Analysis	Systems Analysis	Data Analysis	Spatial Interactions	External Events Analysis	Human Actions Analysis
1. Common Cause Initiating Event	1A Physical Interaction	XX*	X		X	X	
	1B Human Interaction	XX	X	X		X	X
2. Intersystem Dependency	2A Functional Dependency	XX	X				
	2B Shared Equipment	XX	X				
	2C Physical Interaction		X		X	XX	
	2D Human Interaction		X			X	XX
3. Intercomponent	3A Functional Dependency		XX				
	3B Shared Equipment		XX				
	3C Physical Interaction		XX	X	X	X	
	3D Human Interaction		XX	X		X	X

* X = contributing analyses; XX = principal analyses.

3.0 FRONT-END ANALYSIS

3.1 Accident Sequence Delineation

The determination of accident sequences results from the event sequence model, as illustrated in Figure 3.1-1. This figure shows the connection between the various event trees that are discussed in the following sections. As can be seen from this figure, each initiating event is processed first through the Seismic Response tree (bypassed for nonseismic initiating events) and then through the support systems tree. At this point the sequence consists of a specific support system state, i.e., success or failure of one or both trains of AC/DC power, Primary Component Cooling (PCC), Service Water (SW), etc. The initiating event (with specific support system information) is then processed through its appropriate frontline trees.

The frontline trees model the response of standby safety systems and the operator to the initiating event with specific support system status, e.g., secondary cooling (EFW), safety injection.

As shown in Figure 3.1-1, the frontline trees consist of an early response tree (e.g., General Transient) and a late response tree (LL2 for large LOCA sequences, LT12 for all other sequences).

Out of the early response tree, the sequence is either a stable plant configuration, a core damage sequence, or a degraded plant configuration with the potential for success.

The appropriate long-term response tree then addresses issues such as long-term heat removal (i.e., the potential failure of recirculation cooling for those sequences where successful core cooling was obtained in the early response tree) and, for core damage sequences, the status of the RCS and containment (e.g., high/low pressure melt, wet or dry containment, and containment isolation). This information is required for the linking to the Containment Event Tree (the Level II interface).

At this point, the Recovery event tree is then processed for those potential core damage sequences where recovery actions are possible. This tree credits only specific sequences to recover failed equipment (restart diesel generators and/or recover off-site power if either is recoverable) or provide for alternate success paths (provide makeup to the Refueling Water Storage Tank in order to maintain long-term cooling for small LOCA sequences).

At this stage, the core damage sequences are processed through the Containment Event Tree (CET, see Section 4). This tree deals with phenomenological issues related to containment failure mode and timing of failure. The result of the CET analysis is a list of sequences contributing to specific Release Categories with their corresponding frequency. For the purposes of this report, the Release Categories have been grouped into four broad classes of containment performance:

- Early, large containment failure or bypass.
- Early, small containment failure or bypass.
- Late containment failure.
- Containment intact.

Each of the above steps is explained in more detail in the sections that follow.

3.1.1 Initiating Events

Seventy-two initiating events were selected for quantification in the current SSPSS-1990. These initiating events, their annual frequency, and a brief description are provided in Table 3.1-1. These initiating events are again displayed in Table 3.1-2 in order to present the coverages of major classes of initiating events - Loss of Primary Coolant, General Transients (GTs), Common Cause Events (Support Systems and Externals), and ATWS's. Note that of the 72 initiating events, 49 are actually distinct initiators. (See footnote in Table 3.1-2 for explanation.) Table 3.1-2 also summarizes the success criteria for each initiator. Table 3.1-3 provides an additional listing of initiating events along with their corresponding impact upon event tree top events. These dependencies are carried through the quantification process by appropriate event tree modeling and logic rule assignments.

Three different methods were used to logically identify all possible candidate initiating events as follows:

- Master Logic Diagram
- Heat Balance Fault Tree
- Failure Modes and Effects Analysis

The Master Logic Diagram method began with a top event "potential off-site release," and developed the logical conditions needed to accomplish this with increasing detail. This method identified most of the initiating event categories that were finally selected for quantification. The Heat Balance Fault Tree method was developed in the

SSPSA project and resulted in a more detailed structure for defining initiating event categories and enhanced completeness. Failure Modes and Effects Analysis (FMEA) was used to systematically identify support system failure modes that result in common cause initiating events. These multiple and diverse methods of searching for initiators, as well as reviews of other lists of initiators, yielded high confidence of the completeness of the accident sequence definition.

The original analysis resulted in 58 initiating events. Since that time, and as part of updating the risk results, a number of initiating events have been added, deleted, and revised. For instance, the "LOSP" initiating event has been updated to incorporate more recent New England power grid loss data. The LSF6 initiator (loss of SF6 bus duct connection from the off-site grid to on-site power) was added to better reflect the plant-specific nature of Seabrook's switchyard reliability.

As Seabrook Station has had limited operating experience, the transient initiating event frequencies are generally based upon generic data, as discussed in Section 3.3.1. For several initiating events where industry data may not be applicable (e.g., L1CCA, loss of Train A Primary Component Cooling) a Seabrook-specific systems analysis was conducted.

External events such as fires (FCRCC, FET1, etc.) utilized industry data for event frequency per location and Seabrook-specific analysis for plant response. The external events generally contain the hazard frequency (e.g., the frequency of fire in the PAB) and consequential failures (e.g., fire-induced failures of PCC). These initiators are modeled through the event trees to account for random occurring failures that may occur to create an important sequence.

3.1.2 Frontline Event Trees

The frontline event trees consist of the early response (GT, LOCA, steam line break, steam generator tube rupture, and ATWS) and late response (long-term) trees, as shown in Figure 3.1-1. These trees are described below along with the general success criteria for each tree (also, see Table 3.1-2 for a summary of success criteria for frontline systems).

3.1.2.1 General Transient (GT) Tree

The GT event tree, shown in Figure 3.1-2, is used to analyze the early plant response to those non-LOCA initiating events which result in a plant trip with no safety injection signal. Sequences entering the GT tree consist of initiating events and various

combinations of available support systems. Initiating events in this category include loss of main feedwater, loss of off-site power, loss of condenser vacuum, loss of RCS flow, and reactor overpower conditions. Transient external events, such as fires, floods and seismic induced transients, and support system faults are also included. Only initiating events which result in a successful reactor trip are analyzed with this tree. Those transient initiating events in which reactor trip is unsuccessful are mapped to the ATWS tree (see Section 3.1.2.5).

The top events in the GT tree are listed in Figure 3.1-2. The first top events question secondary cooling; turbine trip (TT), emergency feedwater and steam relief (EF), emergency feedwater recovery (FR), and operator action to control feedwater flow during overcooling sequences (OM). A total loss of PCC guarantees an RCP seal LOCA (NL). Top event RW questions RWST availability for feed and bleed cooling or containment spray operation. Feed and bleed cooling requires operation of one of the four high pressure injection pumps (H2) and opening of both PORVs (OR). Top Event OR also includes the necessary operator actions to initiate feed and bleed cooling. Operator actions to control RCS pressure during overcooling sequences are modeled in Top Event OP. Failure of Top Event OP is assumed to create a pressurized thermal shock concern and possible threat to reactor vessel integrity (RV). If successful, secondary cooling is established, operator actions to achieve long-term plant stabilization and cooldown are questioned in Top Event OQ. Top Event OQ also models operator actions to depressurize the RCS, in the event of a seal LOCA, to slow the break flow and extend the time for recovery. The operability of the RWST isolation valves (RA and RB) is modeled for those sequences requiring RHR miniflow or Containment Building spray operation. If feed and bleed cooling is successful, the RHR pumps are required to operate in the minimum recirculation mode (miniflow) for up to six hours (Top Events L1 and L2). Containment Building sprays (Top Events CA and CB) are questioned for core melt sequences.

The GT tree models the plant response for approximately five to ten hours immediately following the initiating event. Success sequences for GTs are:

- Secondary cooling (at least one of two EFW pumps to two steam generators).
- Feed and bleed cooling (at least one of four HPI pumps and two open PORVs) and sump recirculation.

Those sequences which would result in normal RHR cooldown are mapped to a success state and not analyzed further. Those which result in feed and bleed cooling are transferred to long-term response tree LT12 (see Section 3.1.2.6) to question recirculation cooling. Those sequences which result in a core melt condition are also transferred to LT12 to question containment status: spray, heat removal, and isolation.

3.1.2.2 Loss of Coolant Event Trees

The LOCA event trees model all primary system breaks of sufficient size to exceed the makeup capabilities of the normal charging system. Since there is a large range of possible break sizes, with significantly differing plant responses, the LOCA plant model utilizes three distinct event trees: small, medium, and large LOCAs. The LOCA trees model the early response, i.e., the ECCS injection phase. The long term response is modelled in event tree LT12 for small and medium LOCA's, and event tree LL2 for large LOCA's. The three LOCA event trees are described as follows:

- Small LOCA - SLOCA (See Figure 3.1-3)

  The SLOCA tree models breaks from 0.5" to 2.0" in diameter. The plant response for a small LOCA is similar to that of a GT, except that RCS makeup is required. Reactor trip and turbine trip signals are generated on low pressurizer pressure. The model assumes that a successful reactor trip has occurred prior to entering the tree. If reactor trip failed in the preceding support tree, the sequence is transferred to the ATWS tree as an ATWS - SLOCA.

  The SLOCA tree first questions secondary system response, both for loss of secondary cooling and also for potential overcooling events. These top events are turbine trip (TT), emergency feed and secondary cooling (EF) and operator actions to control EFW (OM). OM is only asked if TT fails and EF succeeds, and if failed, leads to an overcooling event.

  High pressure makeup to the RCS depends on the availability of water from the RWST (RW) and one of four trains of high pressure injection operable (H2). If high pressure injection is available and an overcooling event has occurred, then operator action to control high pressure injection is questioned (OP). If operators do not control HPI for an overcooling event, then the potential for a PTS challenge to reactor vessel integrity is asked (RV).

If secondary cooling is not available, then operator initiation of feed and bleed cooling is required (OR). Feed and bleed cooling requires one train of HPI and both PORVs to open (included in OR). For sequences with successful secondary cooling, long term stabilization of the plant is modelled in top event OQ. This models two possible operator actions: (1) with HPI available, controlling EFW flow and establishing RCS pressure and temperature at RHR entry conditions, or (2) with failure of HPI and at least one RCP available, rapidly depressurizing the RCS using the secondary system and initiating low pressure injection.

The isolation valves between the RWST and the common RHR/CBS pump suction for each train (RA,RB) must be open to provide a suction supply to these pumps. The RHR pumps must run successfully in mini-flow mode for up to 6 hours (L1,L2) while the RCS pressure is above the pump shut-off head.

For small LOCAS with secondary cooling (EF) and high pressure injection (H2) available, normal RHR cooldown (LR) is questioned. Finally, Containment Building sprays (CA, CB) are questioned for those sequences ending in core melt conditions.

The success sequences for the SLOCA model are:

- Secondary cooling (one of two EFW pumps to two steam generators), high pressure makeup (one of four HPI trains), and normal RHR cooldown.

- Secondary cooling, high pressure makeup, and, in the long term response tree (LT12), sump recirculation cooling.

- Secondary cooling and RCS depressurization, and, in the LT12 tree, low pressure injection and sump recirculation cooling.

- Feed and bleed cooling (one of four HPIs pump and two open PORVs) and, in the LT12 tree, sump recirculation cooling.

The SLOCA sequences either terminate in a success state or pass to the long term response tree LT12 (see Section 3.1.2.6).

- Medium LOCA - MLOCA (See Figure 3.1-4)

The medium LOCA event tree models RCS breaks from 2.0" to 6.0" in diameter. Breaks in the smaller end of the break spectra resemble the small LOCA plant response, while breaks at the larger end of the spectra will depressurize rapidly and respond similar to a large LOCA. In general, the smaller break sizes are more demanding on system performance, and therefore the success criteria of top events reflects these smaller breaks. Reactor pressure is assumed to remain above the shut-off head of the low pressure injection pumps for one to two hours, therefore requiring high pressure injection and RHR mini-flow recirculation during this time.

The MLOCA model first questions the availability of the RWST (RW). Successful safety injection is achieved by two of four HPI trains (H1). With emergency feedwater available (EF) and operator action to cool down and depressurize the RCS (OD), the RCS can be brought to low pressure conditions allowing RHR injection. The isolation valves between the RWST and RHR/CBS suction must be open (RA,RB) or these pumps will fail due to loss of suction. The model requires that the RHR trains run successfully in mini-flow for up to two hours (L1,L2) while the RCS pressure remains high. For core melt sequences, the operation of both trains of containment building spray (CA,CB) are questioned for their impact on containment performance and radionuclide release, as well as their effect on the rate of RWST injection (and, thus, the time to switchover to sump recirculation).

The success sequences for the MLOCA model are:

- High pressure injection (2 of 4 HP pumps) and sump recirculation cooling (in LT12).

- Secondary cooling (1 of 2 EFW pumps to 2 steam generators), RCS depressurization and, in LT12, low pressure injection (1 of 2 RHR trains), and sump recirculation cooling.

- Large LOCA - LLOCA (See Figure 3.1-5)

The large LOCA event tree models RCS breaks from 6.0 inches to double-ended design basis breaks of 29.0 inches. For these events the RCS depressurization and void formation will initially render the core subcritical,

with the high boron concentration of injected water maintaining subcriticality. This tree questions availability of RWST water (RW), position of RWST isolation valves (RA,RB) and operation of the low pressure injection system consisting of accumulators and RHR trains (LA,LB). Automatic initiation of containment building sprays (CA,CB) to reduce containment pressure and possible radionuclide release is modelled. Upon depletion of the RWST due to the combined action of the RHR and CBS pumps (approximately twenty minutes), transfer is made to the late tree (LL2) to model long term plant response.

Successful sequences require injection from the three accumulators of the intact loops and injection from one of the two RHR trains in low pressure injection mode and, in LL2, sump recirculation cooling.

3.1.2.3 Steam Line Break Event Trees - SLBI, SLBO

The Steam Line Break event trees model a major break of a single steam line. The steam line break events are divided into steam line breaks inside containment - SLBI (see Figure 3.1-6), and steam line breaks outside of containment - SLBO (see Figure 3.1-7). The RCS response and top events modelled are very similar for both events. They differ in the fact that for SLBI events there will be an automatic initiation of containment building sprays which result in a much more rapid decrease in RWST inventory. The SLBO model allows for successful cooling by either secondary cooling ending in a success state, or long-term recirculation cooling which transfers to long term tree LT12. The SLBI tree model assumes that Containment Building spray operates, resulting in RWST depletion prior to SI termination. This requires switchover to recirculation, which transfers to long term tree LT12.

The start-up feed pump is modelled as not available for secondary cooling (i.e., no recovery actions to manually start pump), and that the steam dump valves are not available because of MSIV closure. A reactor trip is assumed prior to entering both event trees. Since most top events for both trees are identical, only one set of top events will be described, with the minor differences noted.

The first event questioned is the closure of any three of the four MSIV's to prevent uncontrolled blowdown of the intact steam generators (MS). Uncontrolled blowdown of more than one steam generator results in an potential overcooling event and possible PTS condition. One of two EFW pumps must deliver feed flow to at least two steam generators to provide decay heat removal (EF). If top event MS fails, the operator must control EFW

flow to prevent overcooling (OM). The SLBO event tree models a RCP seal LOCA (NL) if a total loss of PCC has occurred. The RWST must be available (RW) to supply borated water to the ECCS and Containment Building spray system. One of four trains of HPI (H2) must operate to supply borated water to the RCS.

Failure of MSIV closure and EFW control, along with successful HPI, require that the operator throttle HPI (OP) to prevent an overpressure condition and resulting PTS challenge to the reactor vessel. If the operator fails to control HPI, the integrity of the vessel is questioned (RV). If EFW fails and HPI is successful, operator action is necessary to open both PORVS and initiate feed and bleed cooling (OR).

Plant stabilization and cooldown (OQ) consists of either of two operations. If EFW and HPI are successful, cooldown is accomplished by controlling EFW flow and securing HPI if appropriate. If HPI fails, it is conservatively assumed that the operator must depressurize the RCS using the isolated steam generators and initiate low pressure injection in order to maintain adequate boron concentrations.

The isolation valves between the RWST and the RHR/CBS suction lines (RA,RB) must remain open to prevent pump failure due to lack of suction. RHR trains must operate successfully in mini-flow mode for about two hours (L1,L2) while the RCS remains above RHR entry conditions. The operation of the CBS (CA,CB) trains is modelled to reduce containment pressure for the SLBI event and for core melt sequences for both events.

For both initiators, successful mitigation requires the following:

1. Steam line isolation to limit the loss of steam generator inventory and the degree of RCS cooldown,

2. Reactor trip to limit decay heat,

3. Boron injection to ensure reactor core subcriticality in the event that main steam line isolation fails,

4. Long term heat removal.

SLBO sequences either terminate in a success state signifying closed loop RHR cooling, or transfer to long term response tree LT12 for recirculation cooling or core melt sequences. All SLBI sequences transfer to LT12 for recirculation cooling and containment spray operation.

3.1.2.4 Steam Generator Tube Rupture Tree (See Figure 3.1-8)

The Steam Generator Tube Rupture (SGTR) tree models the complete double ended break of a steam generator tube. This event is similar to the small LOCA, except that the inventory loss bypasses the containment, therefore, water will not be available in the containment sump for recirculation cooling and makeup, and the leak must eventually be terminated to maintain RCS inventory.

The SGTR Tree first questions secondary system response, both for loss of secondary cooling, and also for potential overcooling events. Closure of the turbine stop valves, control valves or the MSIVs will result in a successful Turbine Trip (TT) and prevent an overcooling of the RCS. Top event EF models the standby feedwater systems; EFW and startup feed pump; and secondary cooling using the condenser or atmospheric relief valves. If TT fails and EF is successful, operation action is required to control feedwater and prevent overcooling (OM).

If a total loss of PCC occurs, an RCP seal LOCA is assumed (NL). Total loss of PCC also guarantees loss of high pressure injection due to loss of cooling to high pressure ECCS pumps.

Availability of the RWST (RW) is required to provide borated water for RCS makeup, containment sump water for recirculation cooling, and containment sprays, if necessary. High pressure makeup (H2) can be provided by one of four trains of high presure injection (H2). If high pressure injection is available and an overcooling event has occurred, then operator action to control high pressure is questioned (OP). If operators do not control HPI for an overcooling event, then PTS challenge to reactor vessel integrity is asked (RV).

Successful operator termination of the leak (O4) is achieved by depressurizing the RCS to the steam generator atmospheric relief valve setpoint, or if a relief valve in the faulted steam generator remains open, depressurizing the RCS to atmospheric pressure. The preferred method of depressurization is using the remaining intact steam generators. If secondary cooling is not available, feed and bleed cooling must be initiated for RCS depressurization. The failure of O4 increases the likelihood of failure to isolate the faulted steam generator from the environment (SL).

If the leak is terminated and the steam generator isolated from the environment, the cooldown is similar to a normal plant cooldown, but with the added requirement of depressurizing the faulted steam generator simultaneously with the RCS. This is modelled in top event OQ.

Failure of either top EF or H2 requires early or midterm operator action to maintain RCS inventory and achieve a stable condition. These operator actions are modelled in top event 05. Failure of EF and success of H2 requires continued feed and bleed cooling to reach closed loop RHR entry conditions. If EF succeeds, H2 fails, and SL succeeds, stable conditions can be maintained, but a source of makeup and cooldown to RHR conditions are eventually required. Success of EF and failure of both H2 and SL require rapid cooldown of the RCS using the steam generators and initiation of low pressure injection.

The isolation valves between the RWST and the CBS and RHR trains are questioned (RA, RB) to determine the availability of these systems. Failure of either of these valves will result in failure of the associated RHR pump in mini-flow operation and loss of CBS spray pump supply. RHR pumps must operate in the mini-flow mode for several hours (L1,L2) until closed loop RHR or containment sump recirculation cooling is established. For extended feed and bleed sequences, eventual switchover to closed loop RHR cooling is required (LR). The operation of the two trains of containment spray are questioned in top events CA and CB. They will initiate automatically on high pressure from extended feed and bleed. Their operation is important also for core melt sequences, where their operation will affect containment performance by reducing containment pressure for core melt sequences.

The success sequences for the SGTR event are:

- Makeup provided by HPI, operator depressurizes using EFW, leak terminated, and closed loop RHR cooling initiated.

- EFW failure, RCS depressurized with feed and bleed (1 of 4 HPI pumps and 2 open PORVs), leak terminated, closed loop RHR cooling.

- HPI failure, rapid depressurization of RCS using secondary system, low pressure injection, and closed loop RHR cooling.

The SGTR sequences either end in a success state or pass to long-term response tree LT12. Those transferring to LT12 are either being cooled successfully in feed and bleed mode, or have resulted in a core melt condition. If the atmospheric or safety valve of the faulted steam generator remains open, this information is passed to LT12 to indicate that a containment bypass condition exists.

3.1.2.5 ATWS Tree (See Figure 3.1-9)

The ATWS tree is used to model those initiating events which require a reactor trip and experience a failure to automatically or manually trip the reactor within one minute of the initiating event. Initiating events not analyzed for ATWS events include RT, where the reactor is tripped by definition; and larger LOCA's, where RCS voiding will achieve reactor shutdown. Non-seismic ATWS initiating events have been consolidated into six initiating events, based on their similar system response to the criticality event. The initiating event frequency is the sum of the initiating events in the given group. The initiating event groups and their characteristics are as follows:

- Loss of Off-Site Power Events - Since these events will automatically have a loss of power to the control rod drives, the only mechanism for trip failure is the mechanical binding of the rods.

- Small LOCA and SGTR - Operator action not needed to initiate emergency boration due to SI signal.

- Turbine Trip - Turbine trip is a guaranteed success and MFW is available (not isolated).

- Partial and Total Loss of Main Feedwater - The start-up feed pump is available.

- Core Power Excursion and Loss of Primary Flow - Main feedwater is available, not isolated.

- MSIV Closure, Excessive FW, and Steam Relief Valve Opening - Main feedwater isolated and no credit for steam dump to condenser.

Seismic ATWS events are modelled as separate initiating events based on the acceleration level of the seismic event.

The ATWS tree has been updated since the original study. The current model addresses issues brought up during subsequent reviews and advancements in the state of knowledge. Revisions have resulted in the SSPSS-1990 model being more consistent with the Westinghouse Owner's Group (Reference 19) and NUREG-1150 (Reference 18) analyses.

The model first questions whether the power level at the time of the initiating event is less than 40% of rated power (PL). Although ATWS events below 70% power will

not exceed the ASME Level C service criteria, the events were conservatively divided at 40% power because the ATWS Mitigation System (AMSAC) will be activated above 40% power. Below 40% power AMSAC and turbine trip are not questioned.

The next top events model the secondary system response for RCS heat removal following the initiator. If main feedwater is available (MF), the RCS will not exceed Level C for any initiating events. If the AMSAC functions successfully (AM) signals will be generated to initiate EFW and a turbine trip. Successful turbine trip (TT) will both prolong the time of effective heat removal and also result in the RCS temperature increasing, allowing for a negative temperature coefficient to reduce reactor power. Turbine trip failure is assumed to result in peak RCS pressure exceeding the ASME Level C service criteria and core melt. If there is a loss of main feedwater, initiation of EFW or the start-up feed pump is required (EF). No credit is given for main feedwater or condensate to supply long term secondary cooling. Failure of EF in those sequences is assumed to lead to core melt.

With failure to trip, the control rods may be driven into the core either by the automatic control system or by manual action (MR). This will be successful in reducing peak RCS pressure if the insertion begins within one minute of the initiator, and continues for at least one minute.

For initiating events above 40% power, the automatic response of the pressurizer PORVs and safety valves is questioned (PS). The pressure relief required is a function of estimated Moderator Temperature Coefficient (MTC) for the given period in the cycle, secondary heat removal capacity, and control rod insertion. Based on the preceding system responses, integrity of the reactor vessel is questioned (RV). If RCS pressure exceeds 3,200 psig, the vessel is assumed to fail and an excessive LOCA results.

If the peak RCS pressure has not exceeded 3200 psig, top events to bring the reactor to shutdown condition are examined (OH). Operators may bring the reactor subcritical by emergency boration, opening the trip breakers and bypass breakers, or tripping motor generator sets. One of these must be achieved within ten minutes for success. Successful injection of borated water requires one of two centrifugal charging pumps for all sequences except small LOCA's, for which success may be obtained by either one centrifugal charging pump or one of two SI pumps. One of two pressurizer PORVs must also open for successful emergency boration (PR). If a pressurizer PORV or safety valve fails to reseat, a small LOCA results (P2). Top event OQ models operator action to normalize plant conditions following the ATWS. This top event includes RCS depressurization using the secondary system if top event PR fails.

Sequences in which a small LOCA has occurred, either from an RCS break or relief valve failing open, require RWST isolation valves to open (RA,RB) and RHR pumps to operate in mini-flow mode (L1,L2) while RCS pressure remains high. Normal RHR shutdown cooling is also questioned (LR) for these sequences. For melt sequences the operation of the CBS trains to reduce containment pressure are questioned (CA,CB).

All ATWS sequences either terminate in a success state or proceed to long term response tree LT12.

3.1.2.6 Long Term Response Trees

The Long Term Response trees (LT12, LL2) are entered from all early response trees except the V-sequence (containment bypass) trees (see Section 3.1.3.2). Entry conditions can be either successful cooling or core damage. Transfer to LT12 is usually at the time of RWST low-low level, except for core melt sequences in which coolant injection has failed. Successful cooling is by recirculation cooling, either by feed and bleed, or discharging through the RCS break for larger LOCA's. Initiating events which achieve successful closed-loop RHR cooling are mapped directly to a success state at the first top event (SU) in long term response tree LT12.

Two long term response trees are used in the plant model analysis: (1) LT12 for all initiators except large LOCA's, and (2) LL2 for large LOCA's. The large LOCA long term response is unique in its early entry, guaranteed low RCS pressure, and requirement for eventual switchover to hot leg recirculation.

- Long Term Response Tree LT12 (See Figure 3.1-10)

  The long term tree LT12 is entered from all initiating events except the large LOCA. Entry conditions can be either a state of adequate cooling, usually by feed and bleed, or core melt sequences.

  If adequate cooling is being maintained at the time of entry, functions required to perform recirculation cooling are asked. These include operation of containment sump isolation valves (ZA,ZB) and RHR pumps and heat exchangers (L5,L6). If RCS pressure is above RHR pump shut-off head (RP), HPI suction must be aligned with RHR discharge (O3) for high pressure Recirculation Cooling (RC).

  If core damage occurs, water may or may not have been injected and collected in the containment sump. If water has been injected and collected in the

containment sump (WS), sump isolation valves, CBS sprays and associated heat exchangers must start and run for seven days (XC,XD) or continue to run (XA and VA, XB and VB), and containment isolation is questioned. If water has not been injected and the sumps are dry, only containment isolation is questioned. The model first questions whether the large containment air purge line isolation valves are isolated (C2) and then whether the small penetrations are isolated (CI).

- Large LOCA Long Term Response - LL2 (See Figure 3.1-11)

  An additional long term response event tree was developed to address the unique conditions of a large LOCA. Characteristics of a large LOCA include very early depletion of the RWST (approximately 20 minutes) and guaranteed low pressure RCS conditions. Therefore, RCS pressure and high pressure recirculation are not questioned. Entry conditions into the tree can be either successful cooling using low pressure injection or core melt conditions.

  If the core is being adequately cooled at the point of entry, the model questions the availability of the containment sump Water Supply (WS) and isolation valves opening (ZA,ZB), operation of RHR pumps (LC,LD) and associated heat exchangers (HA,HB), and pump room cooling (CV). Since pump room cooling will not impact equipment in the short time period modelled in the large LOCA early response tree, it is modelled explicitly as a top event in the long term LOCA tree. If an RHR pump is operational, but both RHR heat exchangers are inoperable, successful heat removal can be accomplished by one train of CBS pump and heat exchanger, (XA and VA, XB and VB) operating in recirculation mode while the RHR pump provides RCS makeup. Approximately 18 hours after a large LOCA, the operator is required to realign low pressure recirculation from discharging to the cold leg to discharging to the hot leg to prevent boron precipitation (HE,HS).

  If core melt has occurred at the time of entry, the model questions the availability of water in the reactor cavity (WS), CBS pumps and associated heat exchangers, and containment isolation (CI,C2).

All sequences from the Long Term Response trees pass to the containment tree (see Section 4.0). Sequences from certain initiating events transfer to the Recovery tree (see Section 3.1.3.1) prior to entering the containment tree.

3.1.3 Special Trees

3.1.3.1 Recovery Tree (See Figure 3.1-12)

The Recovery Tree models operator actions to recover vital systems following certain initiating events. This is the last stage event tree of the plant model, and is entered from long term response tree LT12. Initiating events which are not modelled for recovery terminate following LT12 and do not enter the Recovery Tree. The Recovery Tree models the following recovery actions.

- Recovery of Electric Power - ER

  This models the recovery of electric power given a loss of off-site power and failure of one or more diesel generators. The likelihood of recovery depends on the number of diesels which can be recovered, whether off-site power can be restored, and whether EFW is available at the time at which power is lost.

- Makeup to the RWST RM

  This models operator action to provide makeup to the RWST in order to maintain long term cooling for small LOCA sequences.

Initiating events for which recovery is modelled include transients, with the exception of ATWS and external events for electric power recovery, and small LOCA for RWST makeup.

Sequences in which the vital system was recovered are mapped to a success state. Those sequences in which recovery was unsuccessful are mapped to the appropriate plant damage state, but with a resulting lower frequency reflecting the possibility of recovery for a fraction of the sequences.

3.1.3.2 Interfacing Systems LOCA (V - Sequence) (See Figure 3.1-13)

An evaluation of potential paths for loss of RCS inventory outside containment was performed for the original SSPSA, Section D.13 and was enhanced in Reference 45 (RAI 25). From these evaluations, two general paths for LOCA outside containment were judged to be potentially significant - a steam line breach with a steam generator tube rupture, evaluated in Section 3.1.2.4, and failure of RHR injection line or suction line isolation valves. This second general path into the RHR System, the classic V-sequence first analyzed in WASH-1400, is discussed below.

An evaluation was made of potential interfacing systems LOCA, i.e., lines that connect to the RCS. From this evaluation, the only LOCAs of concern were RCS leakage greater than 150 gpm either past two series check valves in the RHR injection line, or two series motor operated valves in the RHR suction line.

Extensive upgrades have been made to the original SSPSA interfacing systems LOCA model, which assumed catastrophic RHR piping failure, guaranteed melt, and early release. The current model allows for various leak rates, a spectrum of various RHR break sizes, and possible recovery actions.

The model first questions the size of the leak through the RHR isolation valves (LE) and the operation of the RHR relief valves (VO) to estimate the resulting impact on the RHR system. The possible conditions are as follows:

- 150 gpm < Leak flow < 1800 gpm, RHR relief valves open

  Leak flows of this rate are within the capacity of the RHR relief system. RCS leakage is directed to the containment sump via the pressurizer relief tank and is available for recirculation cooling. This is mitigated by the ECCS systems as a typical LOCA.

- Leak flow > 150 gpm, RHR relief valves fail to open

  This scenario leads to an overpressurization of the RHR system, resulting in a break in the RHR piping. This is modelled as a core melt similar to the original SSPSA model.

- Leak flow > 1800 gpm, RHR relief valves open

  These conditions lead to a possible, but not guaranteed, failure of RHR piping, heat exchanger, or pump seals. These sequences continue by examining break size, location, equipment survivability, and possible operator recovery actions.

If the leakage flow passed the RHR isolation valves is above 1800 gpm, with successful RHR relief valve operation, the first question asked is whether a break in the RHR piping or heat exchanger occurs (PI). If this type of break occurs, a guaranteed core melt is assumed. If the RHR piping and heat exchanger survive the pressure increase, questions pertaining to RHR pump seals are examined (SI). If the RHR pump seals fail, the break size of the pump seals is questioned. The potential break sizes are grouped as follows:

- Break <.09 square inches (LX)

  A break in this range is within the capability of the RHR/CBS vault sump pumps. Successful operation of the sump pumps prevents failure of the RHR, SI, and CBS pumps due to submergence.

- Break from .09 to 1.05 square inches (LY)

  A break size of 1.05 inches results in a leak rate of 150 gpm at a pressure slightly below the RHR relief valve setting of 450 psig. Makeup to the RWST is estimated to be limited to approximately 150 gpm. Therefore, for breaks in this range, makeup to the RCS can potentially be maintained by injection from RWST.

- Break from 1.05 square inches to 2.06 square inches (LZ)

  A break of 2.06 inches corresponds to the maximum expected break size for the failure of both RHR pump seals. Breaks in this range guarantee failure of all pumps in both vaults.

If the leak occurs on the injection side and the operator diagnoses the event (O1), the leak can be terminated by closing the motor operated valve on the RHR pump discharge (O2). The survivability of CBS, SI, and RHR pumps (CS,SS,RS) is examined as a function of the leak size and operator action to terminate the leak. These pumps are assumed to fail if the leak is above .09 square inches and the operator fails to terminate the leak, or if the leak is above 1.05 square inches.

The final question asked in the interfacing systems LOCA event tree is operator action to initiate makeup to the RWST (O3).

The interfacing system LOCA event tree does not transfer to any long term tree. Sequences are mapped directly to the following end states:

- Success states DLOC or DILOC in which makeup is successfully maintained by charging pumps. DLOC refers to cases where the leak has been terminated, while DILOC refers to cases where it has not been terminated.

- Small and medium LOCA states. These LOCA inside containment and states would be used to update these LOCA initiating event frequencies if they were of significant frequency.

- Core melt sequences where the leak has been terminated, therefore initially containing releases within the containment.

- Core melt sequences where the leak has not been terminated and containment bypass results.

3.1.3.3 Seismic Tree (See Figure 3.1-14)

The Seismic Tree, the first stage event tree of the plant model, was added to clarify the modelling of system response to seismic initiating events. All initiating events enter the seismic tree. If the initiating event is a nonseismic event, the first top event, QS, is set to 0.0, and the sequence is passed directly through the tree without questioning the following seismic top events. All resultant sequences, for both seismic and non-seismic events, pass to the Support System Tree (see Section 3.1.4). Adding the Seismic Tree provides the following:

1. The seismic and non-seismic causes of a system failure can now be differentiated with seismic and non-seismic top events.

2. Two explicit seismic causes to station blackout are now modelled as top events to better model potential recovery. Seismic ac power failure is separated into a top event describing switchgear failure due to relay chatter (QK) and a top event describing mechanical failure of the diesel generator (QD). Failure due to relay chatter has the potential for recovery, while mechanical failure of the diesel generator is assumed to be unrecoverable.

If the initiating event is a seismic event, top event QS set to 1.0, and the following top events are questioned:

QY - Loss of off-site power due to failure of the station switchyard or grid. This fails top event OG in the support tree and requires initiation of emergency on-site power. No credit for recovery of off-site power is modelled.

QK - Failure of ac power due to relay chatter in the 4.16 kV switchgear. This fails both trains of emergency power, resulting in a station blackout condition. The top event split fractions include the potential for recovery actions for seismic events less than or equal to 0.7g.

QD - Seismic failure of the diesel generators. Similar effect as failure of QK, except recovery assumed impossible.

QR - Structural failure of the RWST. This results in failure of all sequences requiring RCS safety injection and in failure of CBS pumps to operate.

All sequences of the Seismic Tree, both for seismic or non-seismic events, transfer to the second stage Support Tree.

3.1.4 Support Systems Event Tree (See Figure 3.1-15)

The Support Systems event tree analyzes the performance of various support systems required following a given initiating event. Support system states may impact both frontline systems and other support systems. Each sequence describes a set of support system conditions which impact the quantification of the subsequent event trees in the sequence analysis. All initiating events pass through the support tree. For all initiating events, the support tree is preceded by the seismic tree, which impacts support system quantification if the initiating event is a seismic event.

The Support tree analyzes the following systems:

- Electric Power (OG,DA,DB,GA,GB)

  Top events question off-site power (OG), two trains of dc power (DA,DB) and two trains of emergency ac power (GA,GB). If ac power is lost, possible recovery actions are addressed in the Recovery tree, following the Long Term Response event tree.

- Signals (SA,SB,EA,EB,OS)

  Both trains of Solid State Protection System (SA,SB) and both trains of the Engineered Safety Features Actuation System (EA,EB) are explicitly modelled

as top events. In addition, a top event (OS) is included to model possible signal recovery actions for some initiating events where timely operator initiation of equipment is possible.

- Reactor Trip (RT,MT)

  Both automatic and manual reactor trip are modelled as top events. For loss of signals, manual reactor trip is required within one minute for success. Failure of these events results in an ATWS initiating event.

- Primary Component Cooling and Service Water (PA,PB,WA,WB)

  Both trains of Primary Component Cooling (PA,PB) and both trains of Service Water (WA,WB) are modelled. The possibility of recovery of Service Water using the cooling tower is included in the model.

- Ventilation (EH)

  Ventilation to the pumps in the ECCS vaults (RHR,SI, and CBS) and the charging pump cubicles is also modelled.

The sequences of initiators and resulting support systems available are mapped to early response trees (e.g. GT, ATWS, SGTR) based on initiating event.

3.1.5 Sequence Grouping and Back-End Interface

As discussed in Section 3.3.7, the quantification process for Seabrook currently links all core damage sequences through the Containment Event Tree (CET) by the use of logic rules. This effectively links sequences together from initiating events to release category and eliminates the necessity of grouping core damage sequences into plant damage states prior to conducting the CET analysis.

This updated methodology greatly facilitates the quantification process and makes the plant damage state binning used in the original SSPSA unnecessary. However, the logic contained in the plant damage state definitions is still used to write the logic rules for the containment event tree. Specific information on back-end analysis and its quantification process is provided in Section 4.0.

TABLE 3.1-1

Initiating Events

Number	Initiating Event	Frequency PER YR	Description
LOCAS			
1	ELOCA	2.66E-07	Excessive LOCA
2	LLOCA	2.03E-04	Large LOCA
3	MLOCA	4.65E-04	Medium LOCA
4	SLOCA	1.79E-02	Small LOCA
5	SGTR	2.84E-02	Steam Generator Tube Rupture
6	VI	4.50E-06	Interfacing Systems LOCA - RHR Injection Valves Failure
7	VS	3.26E-06	Interfacing Systems LOCA - RHR Suction Valves Failure
General Transients			
8	LCV	1.18E-01	Loss of Condenser Vacuum
9	LOPF	1.76E-01	Loss of Primary Flow
10	TT	1.07E+00	Turbine Trip
11	MSIV	8.66E-02	Closure of One MSIV
12	RT	1.35E+00	Reactor Trip
13	MSRV	4.19E-03	Main Steam Relief Valve Opening
14	SI	2.99E-02	Inadvertent Safety Injection
15	TLMFW	1.62E-01	Total Loss of Main Feedwater
16	PLMFW	1.13E+00	Partial Loss of Main Feedwater
17	CPEXC	2.68E-02	Core Power Excursion
18	EXEW	1.68E-01	Excessive Feedwater Flow
19	AMSIV	1.93E-02	Closure of all MSIVs
20	SLBI	4.65E-04	Steam Line Break Inside Containment
21	SLBO	6.04E-03	Steam Line Break Outside Containment
Support System Failures			
22	L1SWA	3.56E-03	Loss of Train A Service Water
23	L1SWB	3.56E-03	Loss of Train B Service Water

Initiating Events

Number	Initiating Event	Frequency PER YR	Description
24	L1CCA	2.93E-03	Loss of Train A Primary Component Cooling
25	L1CCB	2.93E-03	Loss of Train B Primary Component Cooling
26	LDCA	3.20E-03	Loss of Train A Essential DC Power
27	LDCB	3.20E-03	Loss of Train B Essential DC Power
28	LSF6	8.96E-04	Loss of Off-Site Power due to Faults on SF6 System
29	LOSP	4.84E-02	Loss of Off-Site Power
Seismic Events			
30	E1T	3.13E-03	Seismic 0.1G Transient Event
31	E2T	4.26E-04	Seismic 0.2G Transient Event
32	E3T	1.12E-04	Seismic 0.3G Transient Event
33	E4T	4.44E-05	Seismic 0.4G Transient Event
34	E5T	1.98E-05	Seismic 0.5G Transient Event
35	E7T	1.92E-05	Seismic 0.7G Transient Event
36	E10T	3.65E-06	Seismic 1.0G Transient Event
37	E14T	7.70E-07	Seismic 1.4G Transient Event
38	E20T	1.44E-07	Seismic 2.0G Transient Event
39	E5L	1.19E-07	Seismic 0.5G Large LOCA
40	E7L	1.01E-06	Seismic 0.7G Large LOCA
41	E10L	8.76E-07	Seismic 1.0G Large LOCA
42	E14L	4.47E-07	Seismic 1.4G Large LOCA
43	E20L	1.23E-07	Seismic 2.0G Large LOCA
44	E4AT	2.22E-07	ATWS - Seismic 0.4G Event
45	E5AT	3.56E-07	ATWS - Seismic 0.5G Event
46	E7AT	1.57E-06	ATWS - Seismic 0.7G Event
47	E10AT	8.76E-07	ATWS - Seismic 1.0G Event
48	E14AT	3.61E-07	ATWS - Seismic 1.4G Event
49	E20AT	1.00E-07	ATWS - Seismic 2.0G Event
Fires			
50	FSRCC	1.76E-06	Fire in Cable Spreading Room - PCC Loss

Initiating Events

Number	Initiating Event	Frequency PER YR	Description
51	FSRAC	2.54E-07	Fire in Cable Spreading Room - AC Power Loss
52	FCRCC	7.18E-06	Fire in Control Room - PCC Loss
53	FCRSW	1.68E-06	Fire in Control Room - SWS Loss
54	FCRAC	1.68E-06	Fire in Control Room - AC Power Loss
55	FTBLP	1.23E-03	Fire in Turbine Building - LOSP
56	FPCC	3.12E-06	Fire in PCC Area
57	FET1	2.52E-04	Fire in Electric Tunnel 1
58	FET3	1.26E-04	Fire in Electric Tunnel 3
Floods			
59	FL1SG	5.40E-06	Flood in Turbine Building - LOSP and Loss of one Vital Switchgear Room
60	FL2SG	1.90E-07	Flood in Turbine Building - LOSP and Loss of Both Vital Switchgear Rooms
61	FLSW	1.10E-06	External Flooding - Loss of Service Water
62	FLLP	6.90E-04	Flood in Turbine Building - LOSP
Other Externals			
63	APC	8.50E-09	Airplane Crash - Containment Building Impact
64	TMLL	1.40E-08	Turbine Missile - Containment Impact Causing Large LOCA
65	TCTL	1.90E-04	Truck Crash Into Transmission (SF6) Lines, Loss of Off-Site Power
66	APAB	1.40E-07	Aircraft Crash - PAB Impact

Initiating Events

Number	Initiating Event	Frequency PER YR	Description
ATWS			
67	ALOMF	1.29E+00	ATWS - Loss of Main Feedwater (PLMFW,TLMFW)
68	AGT	3.96E-01	ATWS - General Transient (AMSIV,MSIV,LCV, EXEW,MSRV)
69	ATT	1.07E+00	ATWS - Turbine Trip
70	AMFW	2.03E-01	ATWS Event - MFW Available (CPEXC,LOPF)
71	ALOSP	5.14E-02	ATWS - LOSP (FL1SG,FSRAC,FCRAC,FL2SG,LOSP, FLLP,TCTL,LSF6,FTBLP)
72	ASLOC	4.63E-02	ATWS - Small LOCA (SLOCA + SGTR)

TABLE 3.1-2

Initiating Event Groups

Group	Number	Initiating Event	Success Criteria for Frontline Systems
Loss of Coolant Inventory	7		
		ELOCA	No successful core cooling possible - leakage exceeds the ECCS capability to maintain core cooling. One train of CBS (pump and heat exchanger) for successful containment cooling.
		LLOCA	Accumulators (three of three) and one train of RHR (including the associated RHR or CBS heat exchanger) operating successfully and successful transfer to cold leg recirculation, then hot leg recirculation (approximately 20 hours after the initiating event). Also, reactor trip is not required.
		MLOCA	If the RCS pressure is above the RHR pump shut-off head, successful core cooling is achieved if one train of RHR and one train of High Pressure Injection (HPI) (same train) are successful. If the RCS pressure is below the RHR pump shut-off head, the success criteria is one train of RHR operating successfully.
		SLOCA, SGTR	HPI and secondary heat removal successful or feed and bleed cooling successful along with success of normal RHR cooldown. If normal RHR cooldown is unsuccessful, core cooling can still be established with success of long-term sump recirculation cooling or continued HPI injection with RWST makeup.
		VI, VS	Success occurs for the RHR pump seal failure with piping intact if core makeup is provided by the charging pumps. Sequences which result in a LOCA inside containment are included as part of the initiating event frequencies for small and medium LOCAs.

TABLE 3.1-2
(Continued)

Initiating Event Groups

Group	Number	Initiating Event	Success Criteria for Frontline Systems
General Transients	14	LCV, SLBO, LOPF, TT, MSIV, SLBI, RT, MSRV, SI, TLMFW, PLMFW, CPEXC, EXFW, AMSIV	Secondary decay heat removal available via EFW system or primary decay heat removal via feed and bleed cooling. If core cooling is maintained using feed and bleed, success also requires successful transfer to and operation of sump recirculation cooling.
Common Cause Initiating Events:			
- Support System Faults	8	L1SWA, L1SWB, LSF6, LOSP, LDCA, LDCB, L1CCA, L1CCB	Same as General Transient, described above.
- External/Spatial			
Seismic	1 (9)[a]	E1T, E2T, E3T, E4T, E5T, E7T, E1OT, E14T, E20T	Same as General Transient, described above.
	1 (5)[a]	E5L, E7L, E10L, E14L, E20L	Same as large LOCA (LLOCA), described above.
	1 (6)[a]	E4AT, E5AT, E7AT, E10AT, E14AT, E20AT	Same as ATWS described below.

TABLE 3.1-2
(Continued)

Initiating Event Groups

Group	Number	Initiating Event	Success Criteria for Frontline Systems
Fires	9	FSRCC, FPCC, FET1, FET3, FCRSW, FCRCC, FCRAC, FSRAC, FTBLP	Same as General Transients.
Floods	4	FLLP, FLSW, FL1SC, FL2SG	Same as General Transients.
Others	2	TMLL, APC	Same as large LOCA (LLOCA).
	2	TCTL, APAB	Same as General Transients.
Anticipated Transients Without Scram (ATWS)	(6)[b]	ALOMF, AMFW, ACT, ATT, ASLOC, ALOSP	Secondary decay heat removal must be maintained via the EFW system or the feed and bleed mode of cooling. For sequences where main feedwater is not available (or isolated) and the power level is greater than 40% full power, turbine trip must be successful and sufficient RCS pressure relief must be available to maintain RCS pressure below 3200 psig. Subcriticality must be achieved via manual control rod insertion or chemical shutdown.
Total =	49 (72)		

(a) Note that the 20 seismic initiating events are just three distinct initiators: seismic-initiated general transient, large LOCA, and ATWS. The 20 initiating events are used to account for the different plant effects due to different discrete acceleration values.

(b) Note that the six ATWS initiating events are actually transients (or groups of transients) with subsequent reactor trip failure. These initiators are defined separately to aid in the quantification process.

TABLE 3.1-3

Impact of Initiating Events on Plant Model Top Events

	Initiating Event Category	Impacts of Initiating Event on Event Tree Top Events
1.	Excessive LOCA (Reactor Vessel Failure)	Generates SI, CBS actuation, and containment isolation. Guaranteed failure of Top Events LA and LB. Guaranteed core melt.
2.	Large LOCA	Generates SI, CBS actuation, and containment isolation. Low pressure operation shortens time to switchover to sump recirculation. Top Event RT success not required.
3.	Medium LOCA	Generates SI, CBS actuation, and containment isolation. Low pressure operation shortens time to switchover to sump recirculation. Top Event RT success not required.
4.	Small LOCA	Generates SI and containment isolation. RHR miniflow operation (Top Events L1 and L2) assumed to last for six hours.
5.	Steam Generator Tube Rupture	Impacts Top Events C2 and CI. Generates SI and requires operator action to mitigate.
6.-7.	Interfacing Systems LOCA	Generates SI and containment isolation.
8.	Loss of Condenser Vacuum	Guaranteed failure of Top Event MF in ATWS event tree and failure of condenser steam dump portion of Top Event EF.
9.	Loss of Primary Flow	No impact.
10.	Turbine Trip	Guaranteed success of Top Event TT.
11.	Closure of One Main Steam Isolation Valve (MSIV)	No impact.
12.	Reactor Trip	Guaranteed success of Top Event RT.

Impact of Initiating Events on Plant Model Top Events

	Initiating Event Category	Impacts of Initiating Event on Event Tree Top Events
13.	Inadvertent Opening of Main Steam Relief Valves	Causes a core power excursion. Modelled as a SLBO event.
14.	Inadvertent Safety Injection	Prevents auto start of startup feed pump in Top Event EF.
15.	Total Loss of Main Feedwater	Startup feed pump auto starts (Top Event EF).
16.	Partial Loss of Main Feedwater	Automatic start of the startup feed pump included in Top Event EF.
17.	Core Power Excursion	No impact.
18.	Excessive Feedwater Flow	Results in main feedwater isolation.
19.	Closure of All Main Steam Isolation Valves	Impacts secondary steam relief portion of Top Event EF. Guarantees success of Top Events TT and MS.
20.	Steam Line Break Inside Containment	Causes an overcooling event. Requires main steam line isolation Top Event (MS) to prevent potential challenge to reactor vessel. Containment Spray Actuation Signal (CSAS) generated.
21.	Steam Line Break Outside Containment	Causes an overcooling event. Requires main steam line isolation (Top Event MS) to prevent potential challenge to reactor vessel.
22.	Loss of Train A Service Water	Guaranteed failure of Top Event WA.
23.	Loss of Train B Service Water	Guaranteed failure of Top Event WB.
24.	Loss of Train A Primary Component Cooling (PCC)	Guaranteed failure of Top Event PA.
25.	Loss of Train B Primary Component Cooling (PCC)	Guaranteed failure of Top Event PB.

Impact of Initiating Events on Plant Model Top Events

	Initiating Event Category	Impacts of Initiating Event on Event Tree Top Events
26.	Loss of Train A DC Bus	Guaranteed failure of Top Event DA. Guaranteed failure of Top Event GA (if Top Event OG fails).
27.	Loss of Train B DC Bus	Guaranteed failure of Top Event DB. Guaranteed failure of Top Event GB (if Top Event OG fails).
28.	Loss of Off-Site Power Due to Fault on the SF_6 System	Guaranteed failure to Top Event OG. Normally operating equipment must restart (e.g., SW pumps, PCC pumps). Fails condenser steam dump in Top Event EF. Off-Site power not recoverable in Top Event ER.
29.	Loss of Off-Site Power	Guaranteed failure of Top Event OG. Normally operating equipment must restart (e.g., SW pumps, PCC pumps). Fails steam dump to condenser in Top Event EF.
30.-49.	Seismic Events	
	• General Transients	Guaranteed failure of Top Event QS.
	• Large LOCAS	Guaranteed failure of Top Event QS. Large LOCA occurs.
	• ATWSs	Guaranteed failure of Top Events QS and RT.
50.-58.	Fires	
	• Fire in Cable Spreading Room - Loss of PCC	Guaranteed failure of Top Events PA and PB. Guaranteed core melt.
	• Fire in Cable Spreading Room Causing Loss of AC Power	Guaranteed failure of off-site power (OG) and both trains of emergency ac power (GA,GB). Guaranteed core melt.

Impact of Initiating Events on Plant Model Top Events

	Initiating Event Category	Impacts of Initiating Event on Event Tree Top Events
	• Fire in PCC Area	Guaranteed failure of Top Events PA and PB. Guaranteed core melt.
	• Fire in Electric Tunnel 1	Guaranteed failure of Top Event WA.
	• Fire in Electric Tunnel 3	Top Event OG fails. Guaranteed failure of Top Event WA.
	• Fire in Turbine Building Causing Loss of Off-Site Power.	Guaranteed failure of Top Event OG. Off-site power not recoverable in Top Event ER.
	• Fire in Control Room Causing Loss of AC Power	Guaranteed failure of off-site power (OG) and both trains of emergency ac power (GA and GB). Guaranteed core melt.
	• Fire in Control Room Causing Loss of Service Water	Guaranteed failure of Top Events WA and WB. No recovery of service water. Guaranteed core melt.
	• Fire in Control Room Causing Loss of Primary Component Cooling (PCC)	Guaranteed failure of Top Events PA and PB. Guaranteed core melt.
59.-62.	Floods	
	• Flood in Turbine Building Causing Loss of Off-Site Power	Guaranteed failure of Top Event OG. Off-site power not recoverable in Top Event ER.
	• External Flood Causing Loss of All Service Water	Guaranteed failure of Top Events WA and WB. No recovery of service water. Guaranteed core melt.
	• Flood in Turbine Building Causing LOSP and Loss of One Vital Switchgear Room	Guaranteed failure of Top Event OG. Guaranteed failure of Train A of emergency ac power (GA). No recovery of ac power.

Impact of Initiating Events on Plant Model Top Events

	Initiating Event Category	Impacts of Initiating Event on Event Tree Top Events
	• Flood in Turbine Building Causing LOSP and Loss of Both Vital Switchgear Rooms	Guaranteed failure of Top Event OG and both trains of emergency ac power (GA,GB). No recovery of ac power. Guaranteed core melt.
63.-66.	Others	
	• Turbine Missile Causing Large LOCA	Guaranteed failure of Top Events CA, CB, and C2.
	• Aircraft Crash into Containment	Guaranteed failure of Top Events CA, CB and C2.
	• Truck Crash into Transmission (SF_6) Lines	Guaranteed failure of Top Event OG. Off-site power not recoverable in Top Event ER.
	• Aircraft into the PAB	Guaranteed failure of Top Events PA and PB.
67.-72.	Anticipated Transient Without Scram (ATWS)	EFW (Top Event EF) must feed all four steam generators for success.

FIGURE 3.1-1

Accident Sequence Model

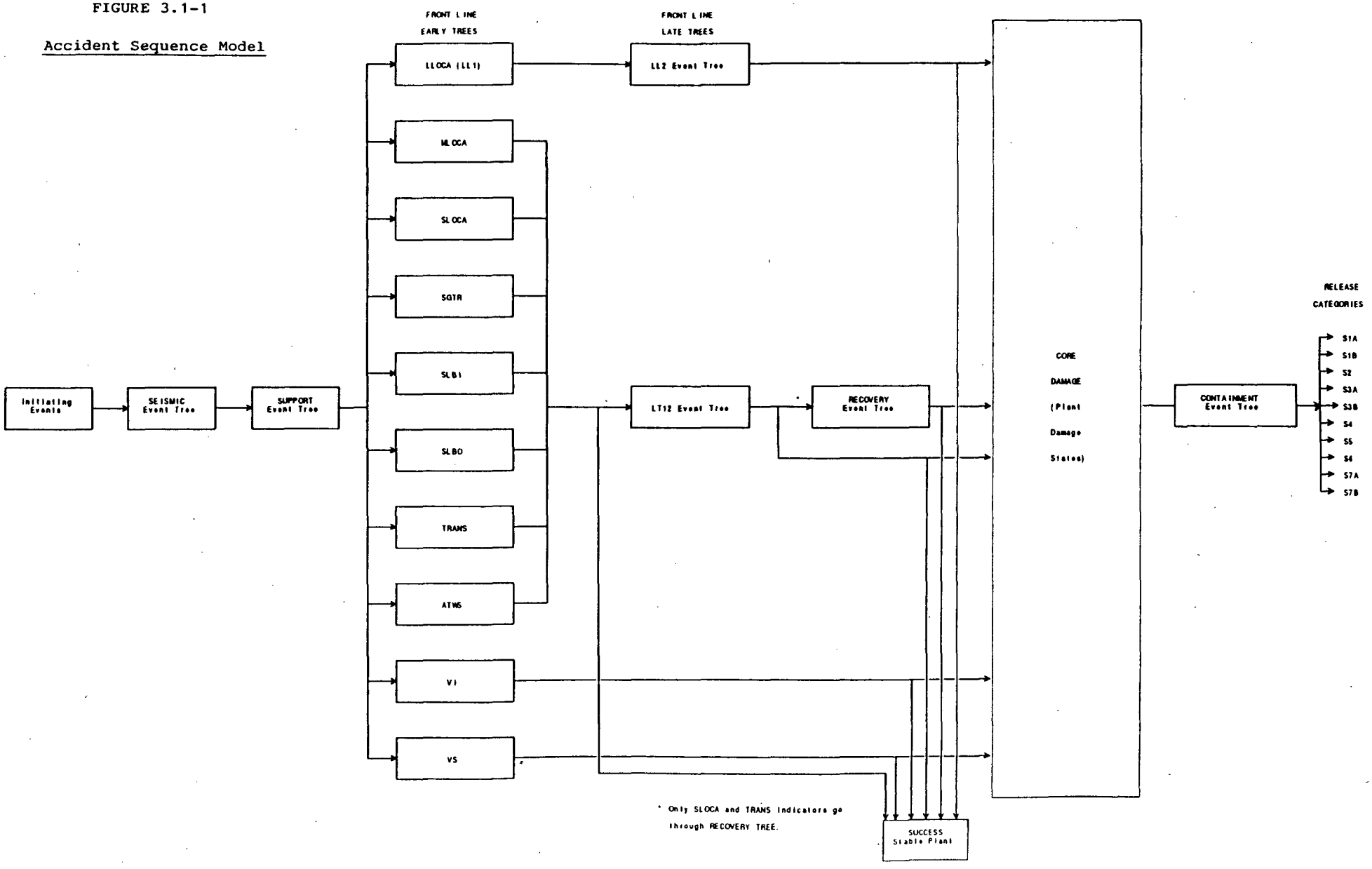


Figure 3.1-2 General Transient Event Tree
(Sheet 1 of 2)

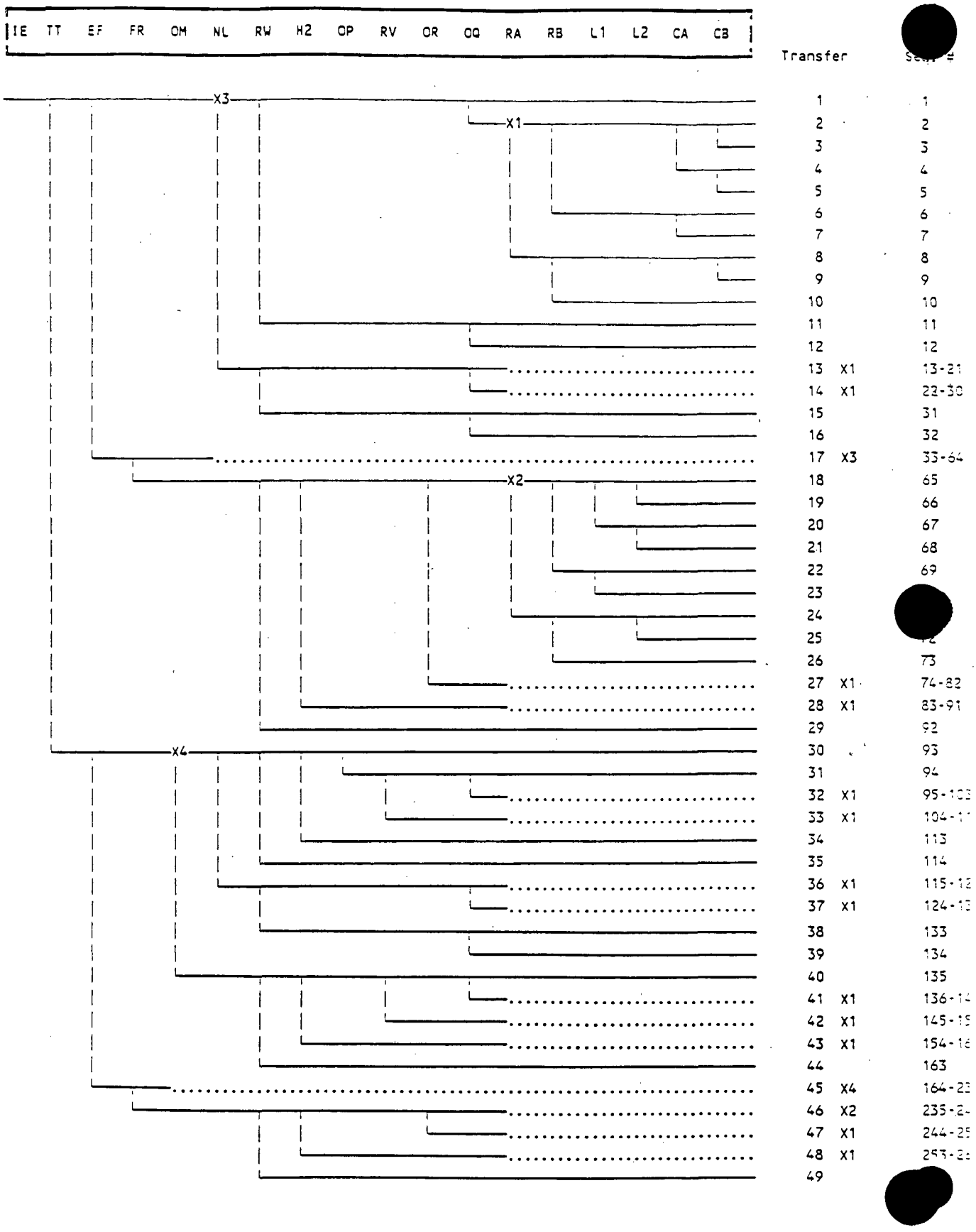


Figure 3.1-2 General Transient Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
TT	TURBINE TRIP
EF	EMERGENCY FEEDWATER
FR	EMERGENCY FEEDWATER RECOVERY (TDP AND/OR SUFP)
OM	OPERATOR CONTROLS EFW
NL	NO RCP SEAL LOCA
RW	REFUELING WATER STORAGE TANK
H2	HIGH PRESSURE INJECTION
OP	OPERATOR CONTROLS HPI
RV	NO REACTOR VESSEL RUPTURE
OR	OPERATOR INITIATES FEED AND BLEED (2 PORVS)
OQ	OPERATOR STABILIZES PLANT
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	LPI TRAIN A IN MINIFLOW
L2	LPI TRAIN B IN MINIFLOW
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-3 Small LOCA Event Tree
(Sheet 1 of 2)

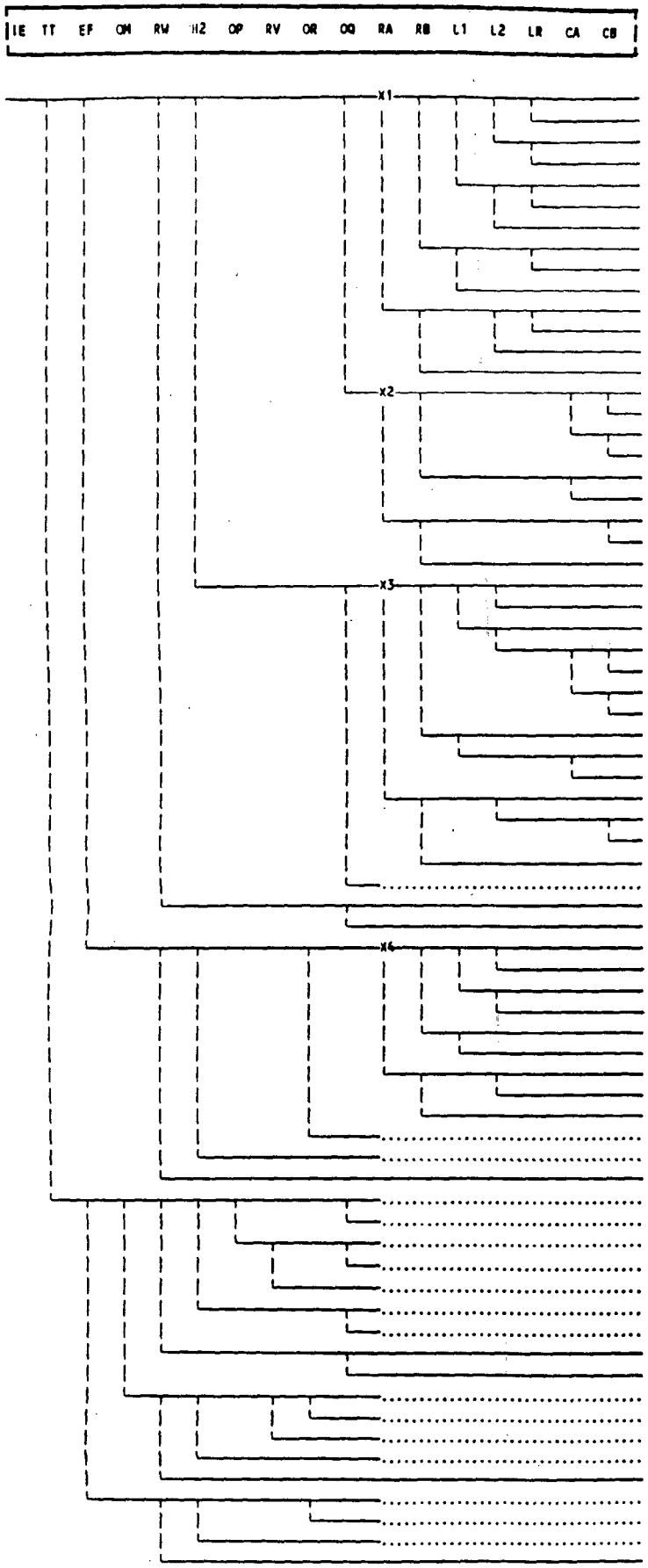
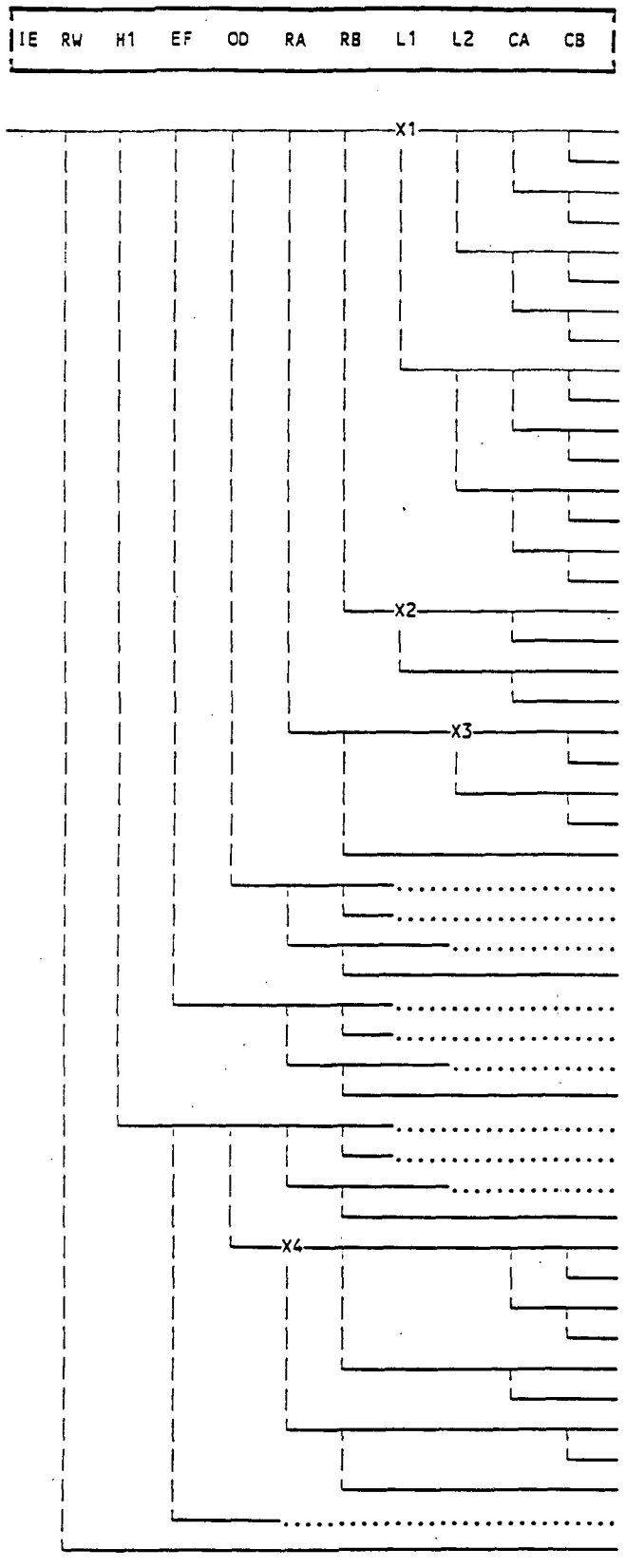


Figure 3.1-3 Small LOCA Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
TT	TURBINE TRIP
EF	EMERGENCY FEEDWATER
OM	MANUAL CONTROL OF EFW
RW	REFUELING WATER STORAGE TANK
H2	HIGH PRESSURE INJECTION
OP	OPERATOR CONTROLS HPI
RV	NO REACTOR VESSEL RUPTURE
OR	OPERATOR INITIATES FEED AND BLEED (2 PORVS)
OQ	OPERATOR STABILIZES PLANT
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	LPI TRAIN A IN MINIFLOW
L2	LPI TRAIN B IN MINIFLOW
LR	LONG TERM NORMAL RHR COOLING (HOT LEG)
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-4 Medium LOCA Event Tree
(Sheet 1 of 2)



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46		109
47	X4	110-118
48		119

Figure 3.1-4 Medium LOCA Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
RW	REFUELING WATER STORAGE TABK
H1	HIGH PRESSURE INJECTION
EF	EMERGENCY FEEDWATER
OO	OPERATOR DEPRESSURIZES
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	LPI TRAIN A IN MINI-FLOW
L2	LPI TRAIN B IN MINI-FLOW
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-5 Large LOCA Event Tree
(Sheet 1 of 2)

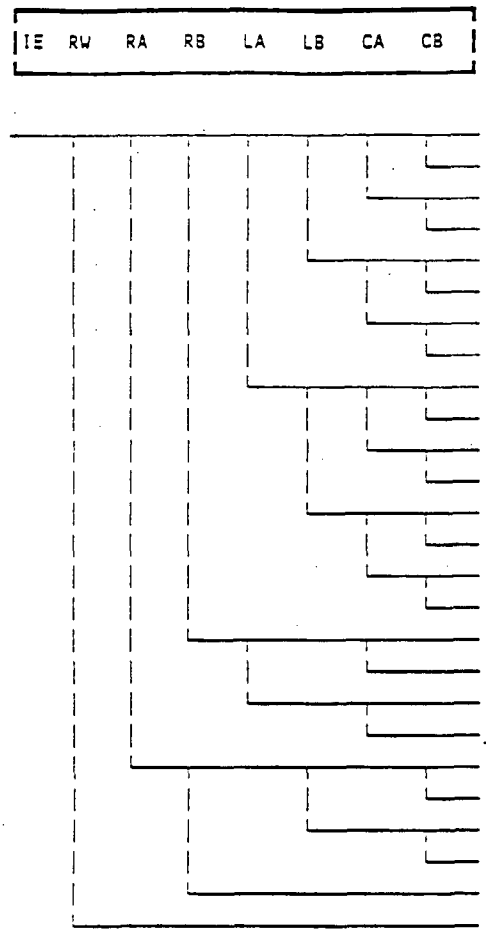

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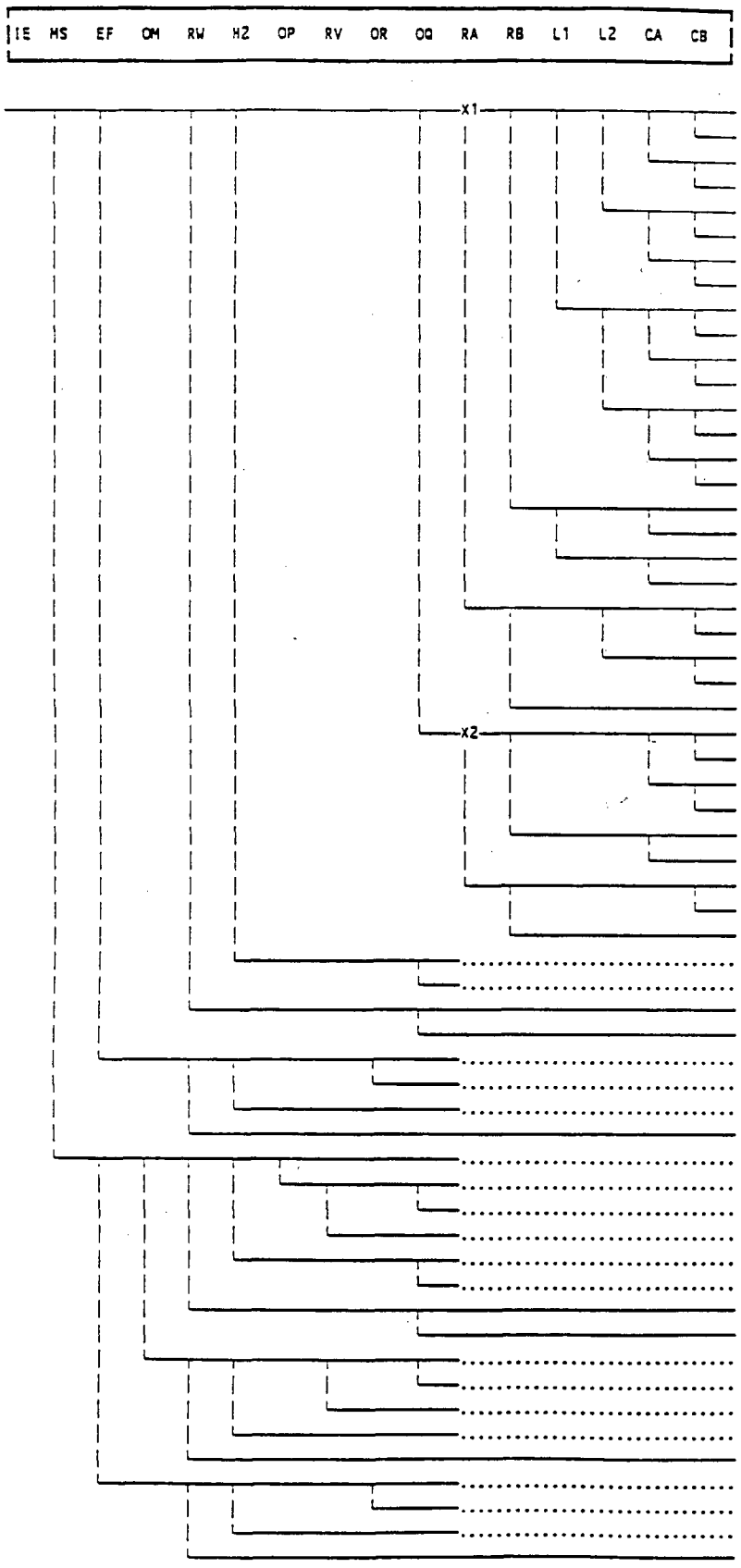
Figure 3.1-5 Large LOCA Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
RW	RWST AVAILABLE
RA	RWST TRAIN A ISOLATION VALVE OPEN
RB	RWST TRAIN B ISOLATION VALVE OPEN
LA	LPI TRAIN A
LB	LPI TRAIN B
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-6 Steamline Break Inside Containment (SLBI) Event Tree
(Sheet 1 of 2)



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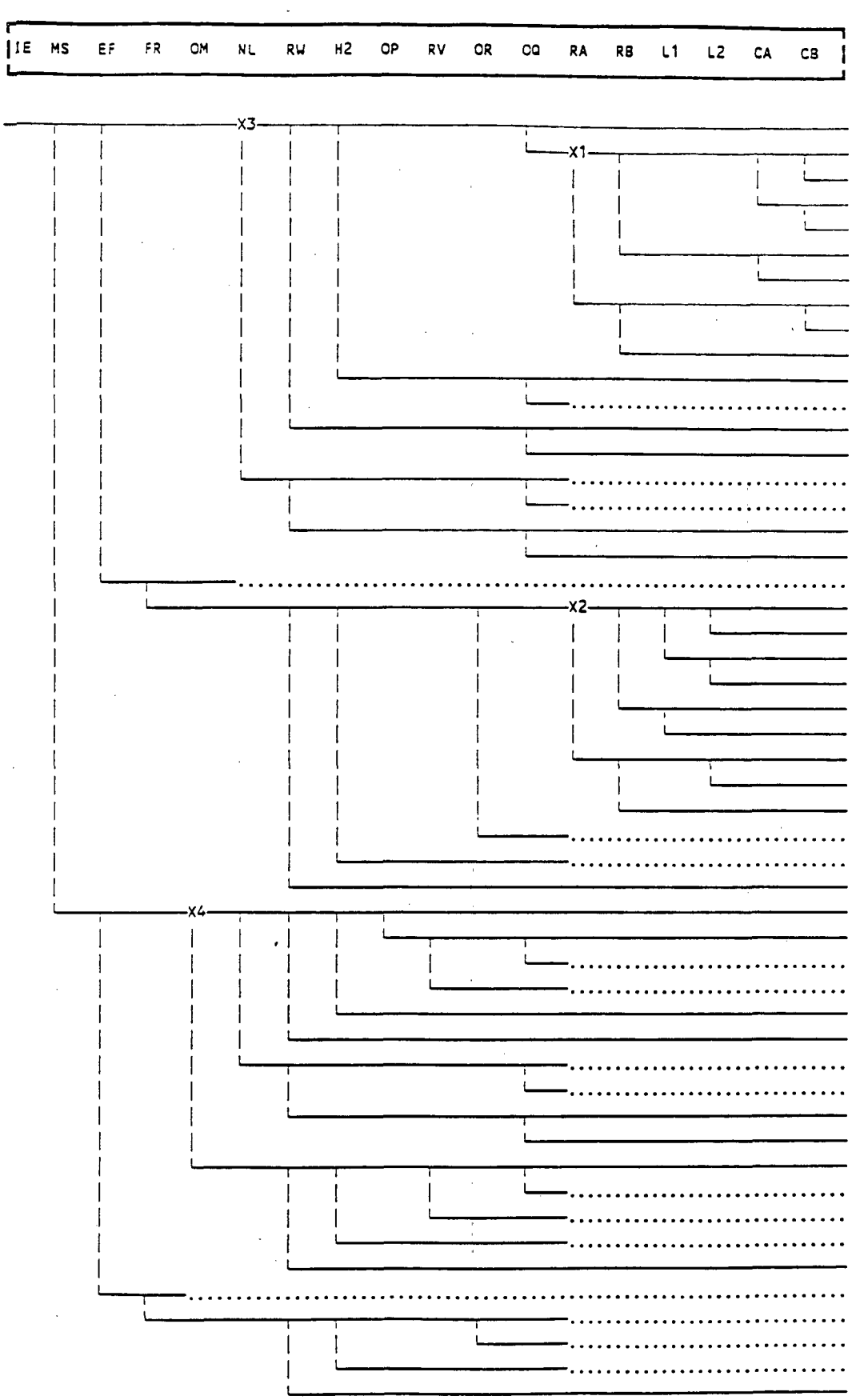
52	X2	244-252
53	X2	253-261
54	X2	262-270
55		271
56	X1	272-296
57	X2	297-305
58	X2	306-314
59		315

Figure 3.1-6 Steamline Break Inside Containment (SLBI) Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
MS	MAIN STEAM ISOLATION
EF	EMERGENCY FEEDWATER
OM	OPERATOR CONTROLS EFW
RW	REFUELING WATER STORAGE TANK
H2	HIGH PRESSURE INJECTION
OP	OPERATOR ACTION TO CONTROL HPI
RV	NO REACTOR VESSEL RUPTURE
OR	OPERATOR INITIATES FEED AND BLEED (2 PORVS)
OQ	OPERATOR STABILIZES PLANT
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	LPI TRAIN A IN MINIFLOW
L2	LPI TRAIN B IN MINIFLOW
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-7 Steamline Break Outside Containment (SLBO) Event Tree
(Sheet 1 of 2)



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43	X1	156-164
44	X1	165-173
45	X1	174-182
46		183
47	X4	184-254
48	X2	255
49	X1	[redacted]
50	X1	[redacted]
51		282

Figure 3.1-7 Steamline Break Outside Containment (SLBO) Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
MS	MAIN STEAM ISOLATION
EF	EMERGENCY FEEDWATER
FR	EFW RECOVERY (TURBINE-DRIVEN EFW PUMP AND/OR SUFP)
OM	OPERATOR CONTROLS EFW
NL	NO RCP SEAL LOCA
RW	REFUELING WATER STORAGE TANK
HZ	HIGH PRESSURE INFECTION
OP	OPERATOR CONTROLS HPI
RV	NO REACTOR VESSEL RUPTURE
OR	OPERATOR INITIATES FEED AND BLEED (2 PORVS)
OQ	OPERATOR STABILIZES PLANT
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	LPI TRAIN A IN MINIFLOW
L2	LPI TRAIN B IN MINIFLOW
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-6 Steam Generator Tube Rupture (SGTR) Event Tree
(Sheet 1 of 2)

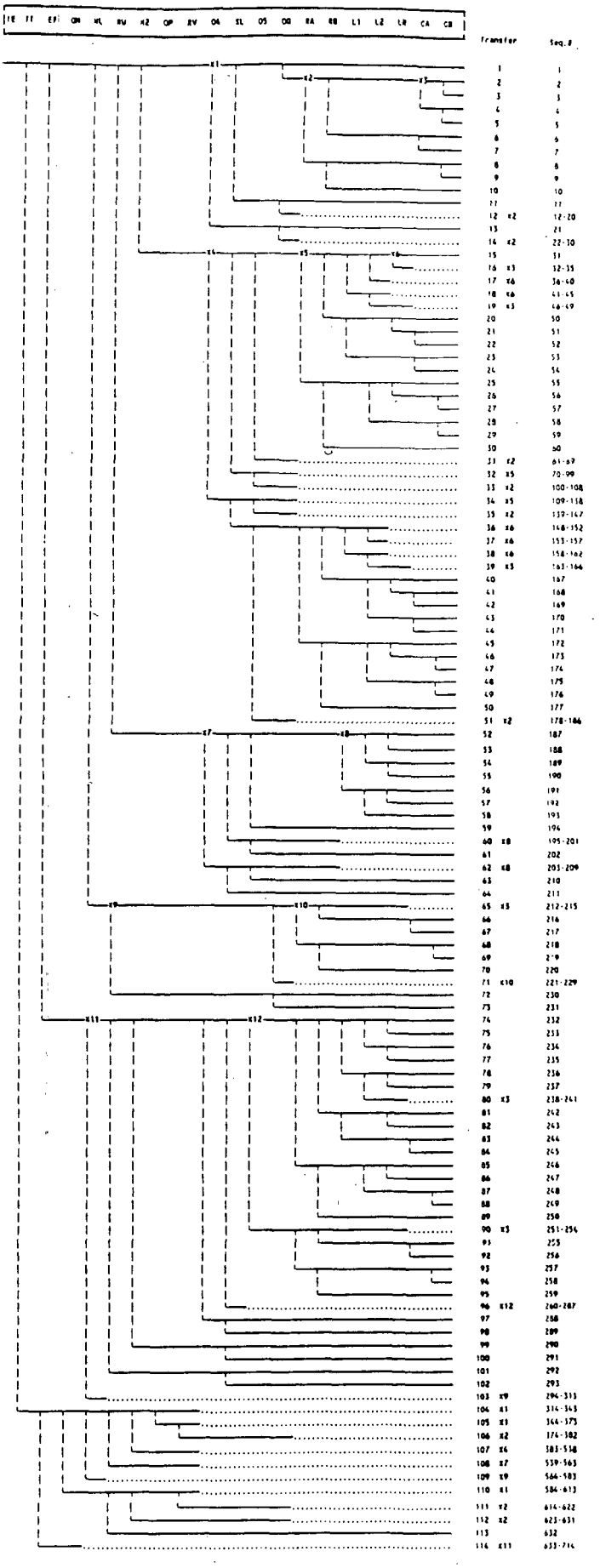


Figure 3.1-8 Steam Generator Tube Rupture (SGTR) Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
TT	TURBINE TRIP
EF	EMERGENCY FEEDWATER
OM	OPERATOR CONTROLS OM (OVERCOOLING ONLY)
NL	NO RCP SEAL LOCA
RW	RWST AVAILABLE
H2	HPI AVAILABLE
OP	OPERATOR CONTROLS HPI (OVERCOOLING ONLY)
RV	NO REACTOR VESSEL RUPTURE
O4	OPERATOR CONTROLS BREAK FLOW
SL	STEAM LEAK
O5	OPERATOR DEPRESSURIZES AND RCS MAKEUP
OQ	OPERATOR STABILIZES PLANT
RA	RWST ISOLATION VALVE TRAIN A
RB	RWST ISOLATION VALVE TRAIN B
L1	RHR MINIFLOW TRAIN A
L2	RHR MINIFLOW TRAIN B
LR	LONG TERM SHUTDOWN RHR COOLING (HOT LEG)
CA	CBS TRAIN A
CB	CBS TRAIN B

Figure 3.1-9 ATWS Event Tree
(Sheet 1 of 2)

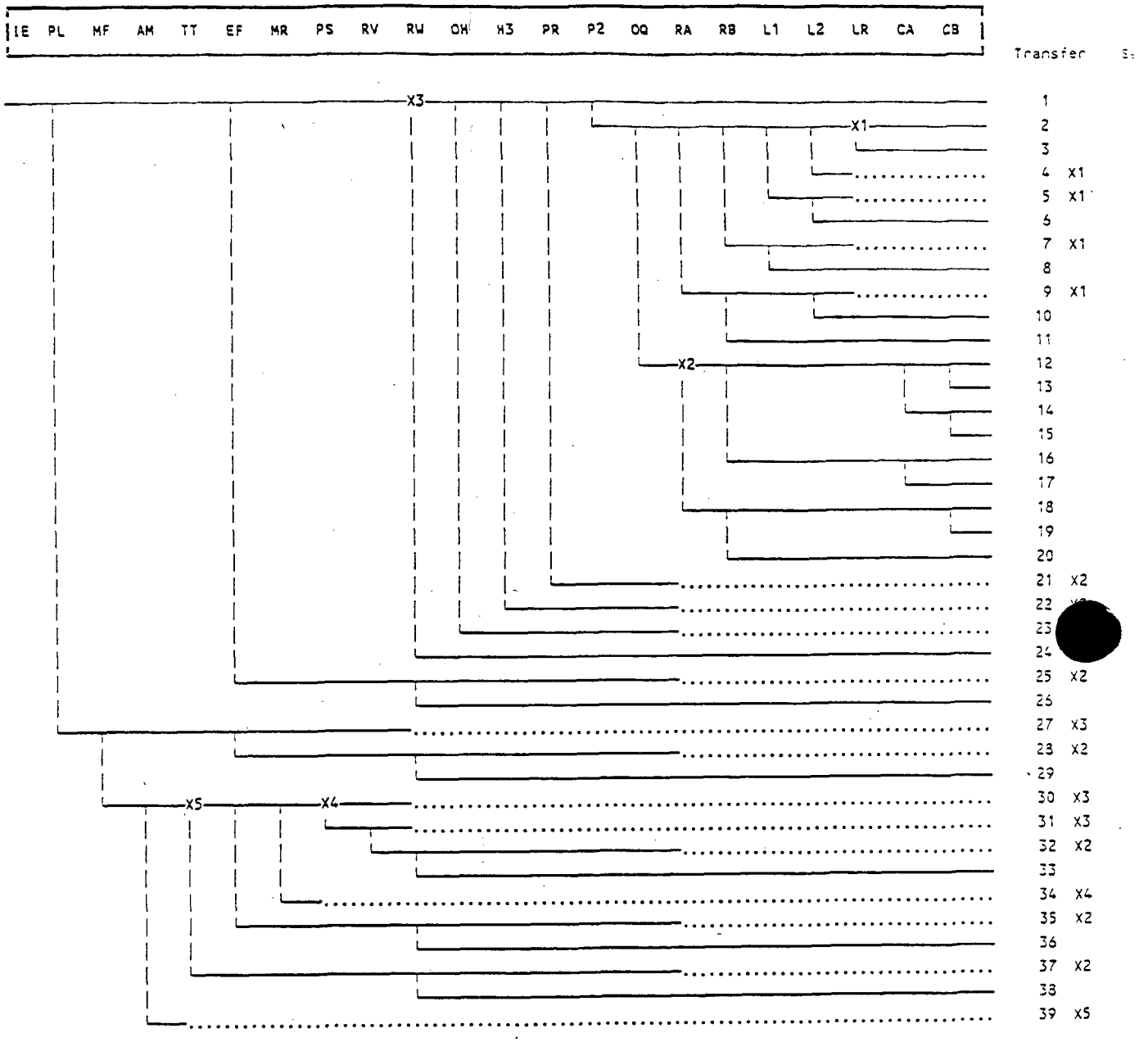


Figure 3.1-9 ATWS Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
PL	INITIAL POWER LEVEL BELOW 40 PERCENT
MF	MAIN FEEDWATER AVAILABLE
AM	ATWS MITIGATION SYSTEM
TT	TURBINE TRIP AVAILABLE
EF	EMERGENCY FEEDWATER AVAIALBLE
MR	OPERATOR PERFORMS MANUAL ROD INSERTION
PS	PRIMARY PRESSURE RELIEF FOR SPIKE
RV	RCS PRESSURE BOUNDARY INTACT
RW	RWST AVAILABLE
OH	OPERATOR PERFORMS EMERGENCY BORATION
H3	HIGH HEAD CCPS AVAILABLE FOR BORATION
PR	PORVS AVAILABLE FOR EMERGENCY BORATION
P2	SAFETY VALVES AND PORVS RESEAT
OQ	OPERATOR DEPRESSURIZES DURING INDUCED SMALL LOCA
RA	RWST ISOLATION VALVE TO TRAIN A RHR/CBS
RB	RWST ISOLATION VALVE TO TRAIN B RHR/CBS
L1	TRAIN A RHR (PUMP AND HEAT EXCHANGER)
L2	TRAIN B RHR (PUMP AND HEAT EXCHANGER)
LR	NORMAL RHR COOLDOWN FUNCTION
CA	TRAIN A CBS PUMP
CB	TRAIN B CBS PUMP

Figure 3.1-10 Long-Term Response LT12 Event Tree
(Sheet 1 of 2)

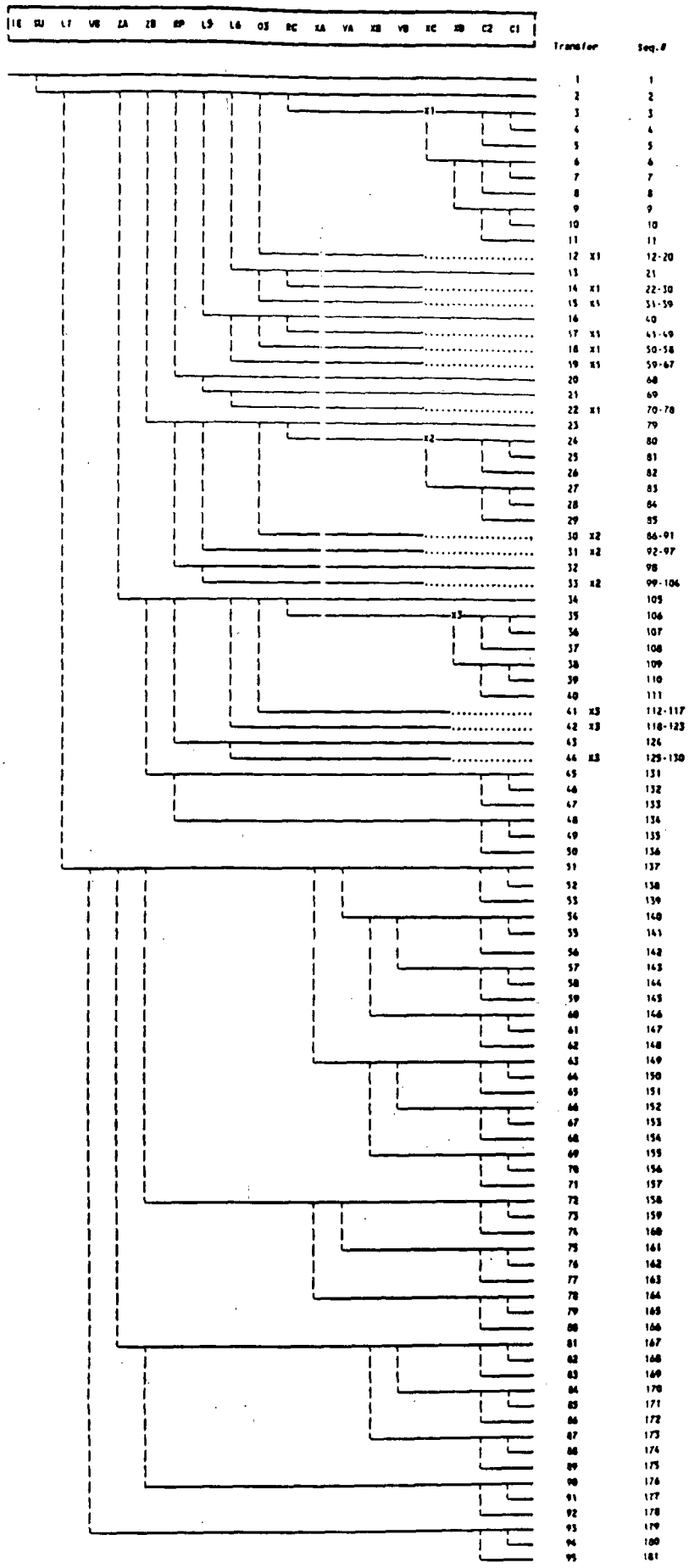


Figure 3.1-10 Long-Term Response LT12 Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
SU	ENTRY STATE SWITCH (0.0 IF SUCCESS;1.0 IF NOT)
LT	LATE TREE SWITCH - 0.0(LT1) OR 1.0(LT2)
WS	WATER IN CONTAINMENT - 0.0(YES) OR 1.0(NO)
ZA	RECIRCULATION SWITCHOVER TRAIN A
ZB	RECIRCULATION SWITCHOVER TRAIN B
RP	RCS PRESSURE
L5	RHR TRAIN A
L6	RHR TRAIN B
O3	OPERATOR ALIGNS HPI RECIRCULATION
RC	HIGH PRESSURE RECIRCULATION COOLING
XA	CBS PUMP RECIRCULATION TRAIN A
VA	CBS HEAT EXCHANGER OUTLET VALVE TRAIN A
XB	CBS PUMP RECIRCULATION TRAIN B
VB	CBS HEAT EXCHANGER OUTLET VALVE TRAIN B
XC	CBS PUMP RECIRCULATION TRAIN A (WITH HX VALVE)
XD	CBS PUMP RECIRCULATION TRAIN B (WITH HX VAVLE)
C2	CONTAINMENT ISOLATION (LARGE OPENING; >3" DIA.)
CI	CONTAINMENT ISOLATION (SMALL OPENING; <3" DIA.)

Figure 3.1-11 Large LOCA Long-Term Response Event Tree
(Sheet 1 of 2)

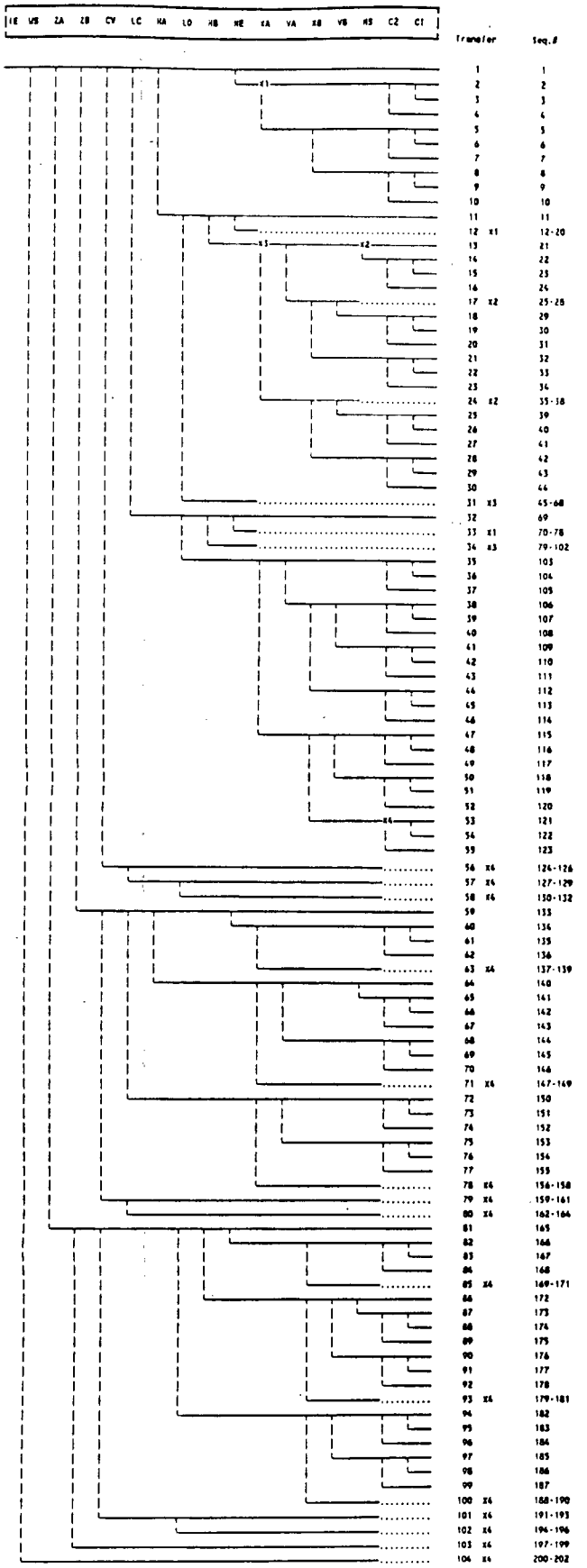
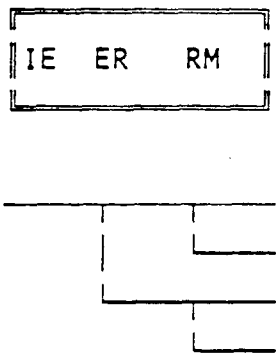


Figure 3.1-11 Large LOCA Long-Term Response Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
WS	WATER SUPPLY
ZA	CONTAINMENT SUMP VALVE A
ZB	CONTAINMENT SUMP VALVE B
CV	CONTAINMENT ENCLOSURE BUILDING VENTILATION
LC	RHR PUMP TRAIN A
HA	RHR HEAT EXCHANGER A
LD	RHR PUMP TRAIN B
HB	RHR HEAT EXCHANGER B
HE	HOT LEG RECIRCULATION LINEUP
XA	CBS PUMP A - NO COOLING REQUIRED
VA	CBS HEAT EXCHANGER A
XB	CBS PUMP B - NO COOLING REQUIRED
VB	CBS HEAT EXCHANGER B
HS	HOT LEG RECIRCULATION LINEUP
C2	CONTAINMENT ISOLATION (LARGE OPENING; >3" DIA.)
CI	CONTAINMENT ISOLATION (SMALL OPENING; <3" DIA.)

Figure 3.1-12 Recovery Event Tree



Top Event Designator.....	Top Event Description.....
ER	STATION BLACKOUT RECOVERY
RM	OPERATOR PROVIDES MAKEUP TO RWST

Figure 3.1-13(a) Interfacing Systems LOCA (Suction Line) Event Tree
(Sheet 1 of 2)

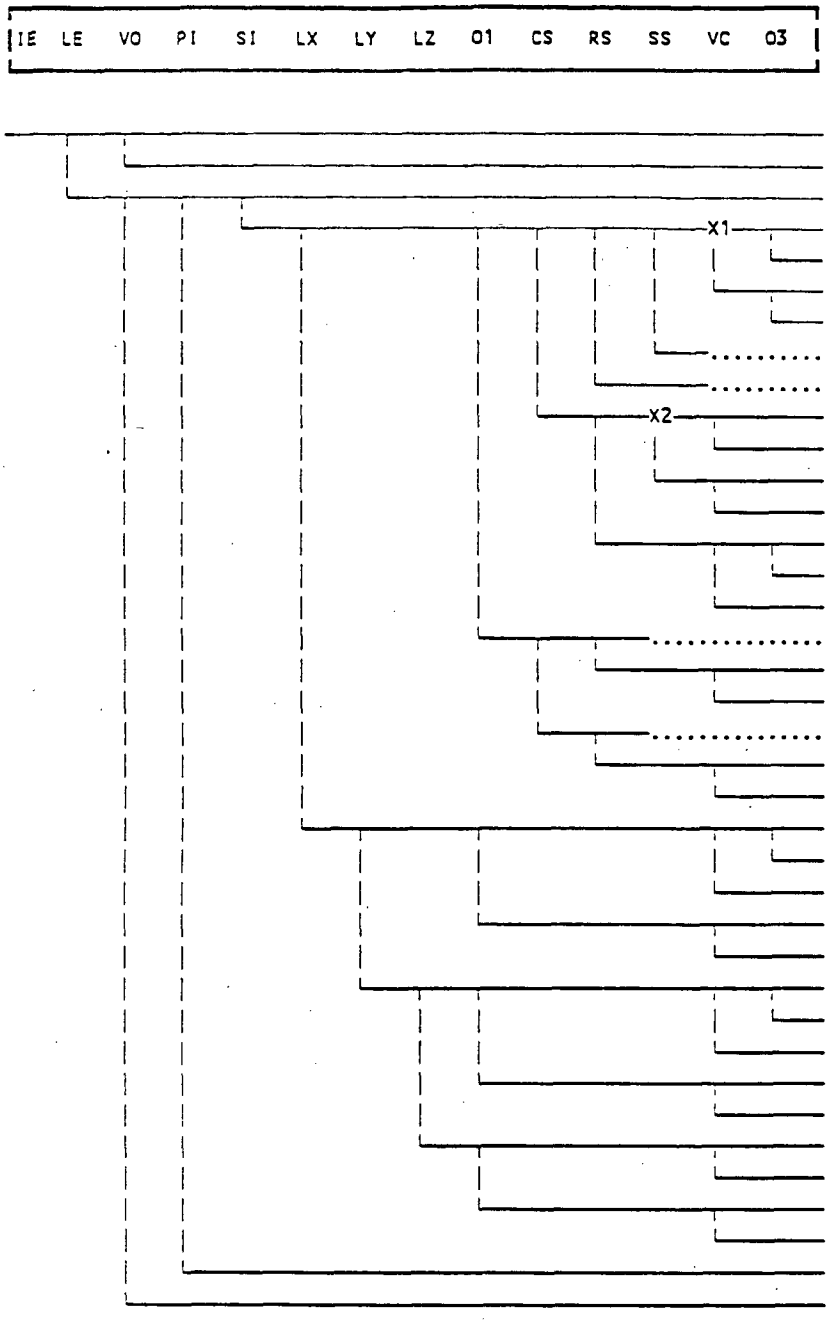


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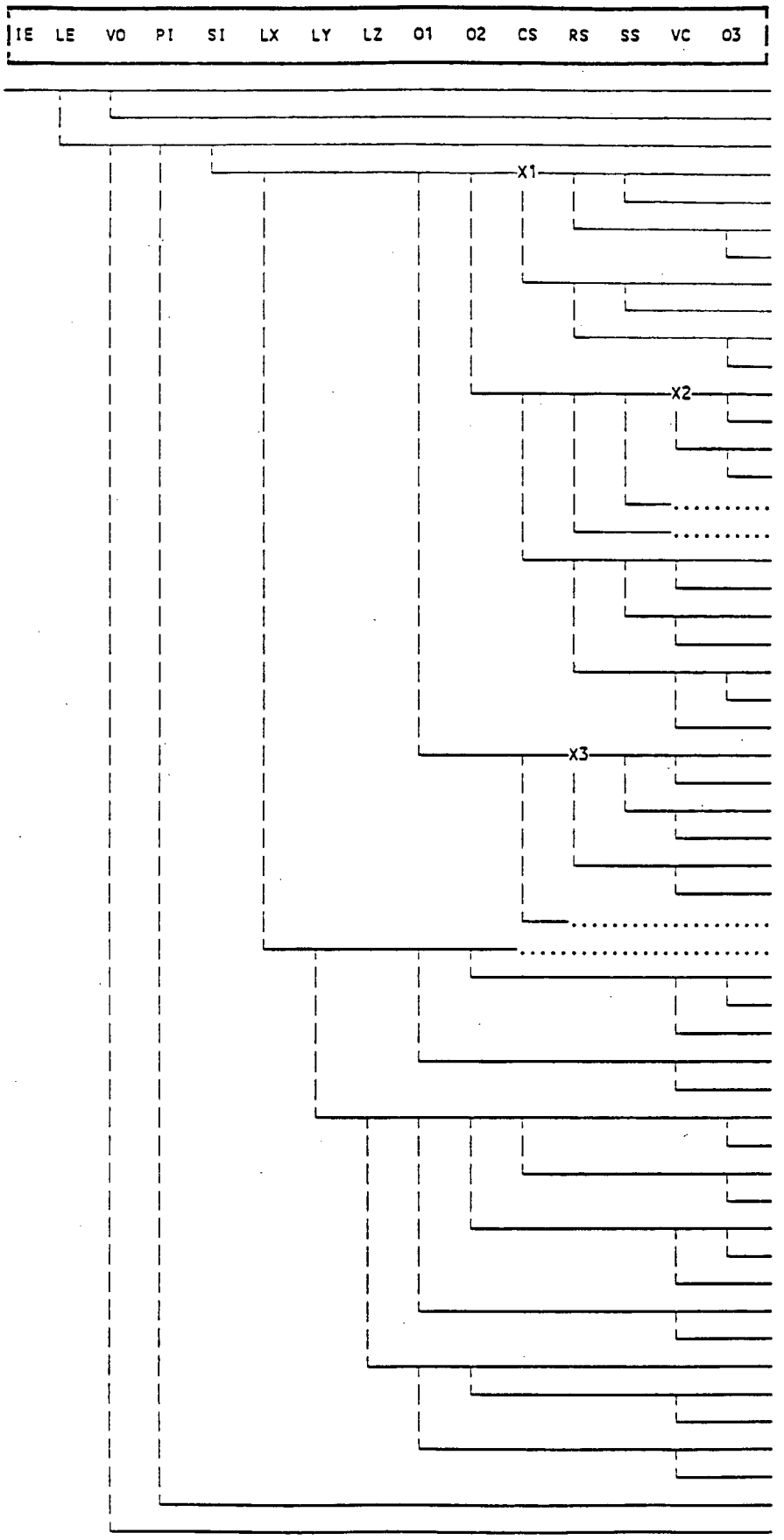
Figure 3.1-13(a) Interfacing Systems LOCA (Suction Line) Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
LE	LEAKAGE < RELIEF VALVE CAPACITY
VO	RELIEF VALVES OPEN
PI	RHR AND HX REMAIN INTACT
SI	RHR PUMP SEALS REMAIN INTACT
LX	PUMP SEAL LEAK BETWEEN 0 AND .09 IN. SQ.
LY	PUMP SEAL LEAK BETWEEN .09 AND 1.05 IN. SQ.
LZ	PUMP SEAL LEAK BETWEEN 1.05 AND 2.6 IN. SQ.
O1	OPERATOR DIAGNOSES EVENT
CS	CBS PUMPS SURVIVE VAULT ENVIRONMENT
RS	RHR PUMPS SURVIVE VAULT ENVIRONMENT
SS	SI PUMPS SURVIVE VAULT ENVIRONMENT
VC	RELIEF VALVES CLOSE
O3	OPERATOR ESTABLISHES RWST MAKEUP

Figure 3.1-13(b) Interfacing Systems LOCA (Injection Line) Event Tree
(Sheet 1 of 2)

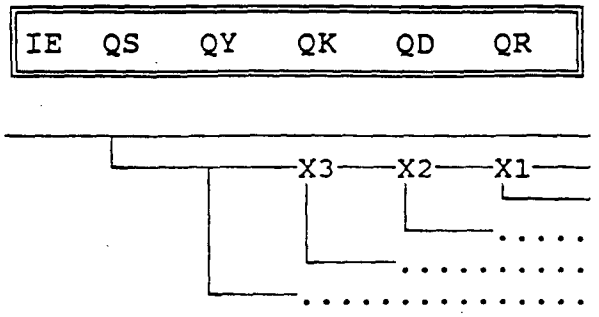


Transfer		Seq.#
1		1
2		2
3		3
4		4
5		5
6		6
7		7
8		8
9		9
10		10
11		11
12		12
13		13
14		14
15		15
16	X2	16-19
17	X2	20-23
18		24
19		25
20		26
21		27
22		28
23		29
24		30
25		31
26		32
27		33
28		34
29		35
30		36
31	X3	37-42
32	X1	43-50
33		51
34		52
35		53
36		54
37		55
38		56
39		57
40		58
41		59
42		60
43		61
44		62
45		63
46		64
47		65
48		66
49		67
50		68
51		69
52		70
53		71

Figure 3.1-13(b) Interfacing Systems LOCA (Injection Line) Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
LE	LEAKAGE < RELIEF VALVE CAPACITY
VO	RELIEF VALVES OPEN
PI	RHR AND HX REMAIN INTACT
SI	RHR PUMP SEALS REMAIN INTACT
LX	PUMP SEAL LEAK BETWEEN 0 AND .09 SQ. IN.
LY	PUMP SEAL LEAK BETWEEN .09 AND 1.05 SQ. IN.
LZ	PUMP SEAL LEAK BETWEEN 1.05 AND 2.6 SQ. IN.
O1	OPERATOR DIAGNOSES EVENT
O2	OPERATOR TERMINATES VALVE LEAKAGE
CS	CBS PUMPS SURVIVE VAULT ENVIRONMENT
RS	RHR PUMPS SURVIVE VAULT ENVIRONMENT
SS	SI PUMPS SURVIVE VAULT ENVIRONMENT
VC	RELIEF VALVES CLOSE
O3	OPERATOR ESTABLISHES RWST MAKEUP

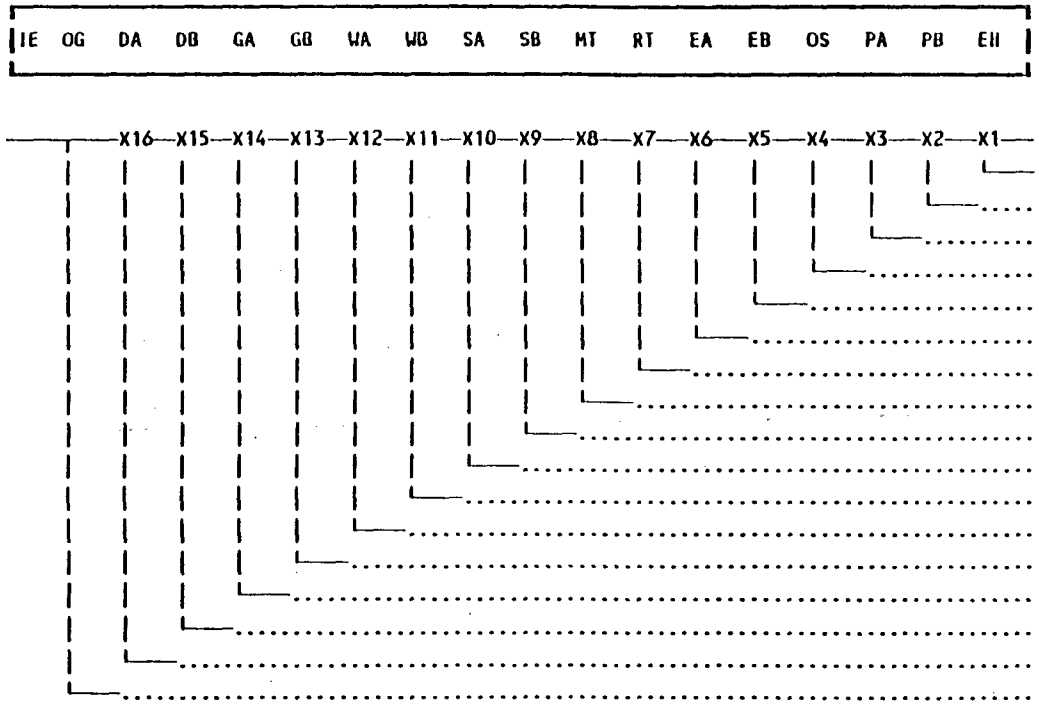
Figure 3.1-14 Seismic Event Tree



Transfer		Seq.#
1		1
2		2
3		3
4	X1	4-5
5	X2	6-9
6	X3	10-17

Top Event Designator.....	Top Event Description.....
QS	SEISMIC EVENT SWITCH
QY	SWITCHYARD
QK	4.16 KV SWITCHGEAR (RELAY CHATTER)
QD	DIESEL GENERATORS
QR	REFUELING WATER STORAGE TANK (RWST)

Figure 3.1-15 Support Systems Event Tree
(Sheet 1 of 2)



Transfer		Seq.#
1		1
2		2
3	X1	3-4
4	X2	5-8
5	X3	9-16
6	X4	17-32
7	X5	33-64
8	X6	65-128
9	X7	129-256
10	X8	257-512
11	X9	513-1024
12	X10	1025-2048
13	X11	2049-4096
14	X12	4097-8192
15	X13	8193-16384
16	X14	16385-32768
17	X15	32769-65536
18	X16	65537-131072

Figure 3.1-15 Support Systems Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
OG	OFFSITE GRID
DA	DC BUS A TRAIN
DB	DC BUS B TRAIN
GA	DIESEL GENERATOR A TRAIN
GB	DIESEL GENERATOR B TRAIN
WA	SERVICE WATER A TRAIN
WB	SERVICE WATER B TRAIN
SA	SOLID STATE PROTECTION A TRAIN
SB	SOLID STATE PROTECTION B TRAIN
MT	MANUAL REACTOR TRIP (INITIATE SIGNAL ONLY)
RT	REACTOR TRIP (BREAKERS AND CRDMS)
EA	ESFAS A TRAIN
EB	ESFAS B TRAIN
OS	OPERATOR RECOVERS SIGNAL FAILURE
PA	PRIMARY COMPONENT COOLING A TRAIN
PB	PRIMART COMPONENT COOLING B TRAIN
EH	CONTAINMENT ENCLOSURE AIR HANDLING

3.2 System Analysis

3.2.1 System Description

Table 3.2-1 presents a summary of those systems that have been quantified in the SSPSS-1990, with a brief description of their functions. Appendix E to this report contains a more detailed summary description of each individual system. This appendix provides a system description which includes its function, configurations, dependencies, and operational characteristics. A simplified P&ID sketch (or line diagram) for each system is also provided. Finally, a summary of the quantification is given, with a breakdown of the contributions.

The details of the systems analyses for the 1990 model are contained in documentation notebooks. These results have evolved from the system analyses documented in Appendix D of the original SSPSA.

3.2.2 System Analysis

In addition to containing descriptions of each system, Appendix E contains a summary of the system analysis that was performed for each system. The appendix describes the individual system models, their relationship to event tree top events, applicable success criteria, and the various boundary (analysis) conditions that were addressed during the quantification process.

The original SSPSA system analysis involved the use of reliability block diagrams and resultant system equations. Fault trees were used to verify the equations but were not retained as part of the documentation. These analyses were performed by analysts familiar with plant response and the potential for system importance. Specific boundary conditions, the effect of testing and maintenance configurations, and the impact of common mode failures were often incorporated by inspection and conservative engineering judgment.

As part of maintaining a living PSA, periodic updates to system modeling and software have allowed a more complete treatment of common cause and different system alignments. A number of the system analyses have been upgraded utilizing the fault tree approach that is now available through the RISKMAN suite of codes (Reference 9).

Currently, the system analyst develops the system logic model based upon the system's function and the event tree top event success criteria. The logic model relates a system state, such as success or failure, to combinations of more basic events such as component failure rates. The following tasks are representative of the process followed during the more recent system analysis updates at New Hampshire Yankee:

1. Reliability Block Diagram Development

   A piping and instrumentation diagram or schematic diagram, such as an elementary electrical drawing, is used as a basis for constructing the block diagram. The block diagram portrays the "success paths" of the system. These paths are combinations of component success states that enable successful functioning of the system. The success paths, which have the same logical information contained in a listing of the minimal cut sets, provide the basis for calculating system unavailability.

2. Fault Tree Model Development

   Fault tree models of each system top event are constructed to provide the logic structure of deriving the algebraic unavailability equations that are used to quantify the top event split fractions. The development of the fault tree is based on the block diagrams and converts the success logic of the block diagrams to failure logic. Fault trees serve three purposes: (1) to provide a cross-check of the model logic, (2) to provide an analysis format that can be easily reviewed, and (3) to allow the generation of minimal cut sets to be used by RISKMAN to develop algebraic equations. Basic events associated with common cause failures are added to the fault trees prior to Boolean reduction in accordance with NUREG/CR-4780 (Reference 20).

3. Common Cause Modeling

   To incorporate common cause events into the system analysis, the analyst must understand the factors that determine the dependence or independence among the components in the system. Such factors include how groups of components are used, the extent of their diversity (if any), the physical proximity or separation of redundant components, and the susceptibilities of system components to varied environmental stresses. Similarity in design, manufacture, and type among components of different trains implies the existence of strong dependencies. On the other hand, common cause effects would not be expected for dissimilar equipment. To account for these factors,

the analyst must identify those components in the system that will be included or eliminated from the common cause analysis and categorize common cause groups of components for systems of interest.

4. System Unavailability and Boundary Conditions

Having developed the logic model, the next step is to convert the logic model into an algebraic model in parameters that can be quantified. The initial conditions for the normal alignment assume that no equipment is unavailable due to test or maintenance at the time of the initiating event and that all support systems are available. However, when the system is under maintenance conditions or test alignments, the equipment may be functionally unavailable due to system configuration changes, such as valve position changes. Therefore, in addition to the component failure modes of the system identified in the logic model development task, the analyst must also identify all of the important causes for the unavailability of components in the system. These may include:

- Functional unavailability due to lack of required support.

- Independent and dependent hardware failures. These random failures include undetected failure while in standby, failures on demand, and failures during operation.

- Test and maintenance. System unavailability may change when test or maintenance is in progress. Since Technical Specifications do not allow systems with redundant trains to be disabled during test and maintenance, additional failures must occur for the system to fail.

- Human errors. System misalignments or miscalibrations may occur due to errors of omission and commission.

In summary, the first step in analyzing system failure for each top event split fraction is to identify all important unavailability causes for the system components in the fault tree. The fault tree provides the logic structure for evaluating system failure; i.e., it identifies the logic combinations of component failure modes that are necessary and sufficient to prevent the system from meeting its success criteria.

Using the fault trees as the basic logic framework, the systems analysts then convert fault trees to fault tree input files using RISKMAN. RISKMAN generates a set of basic cut sets from the fault tree input. These basic cut sets are reduced using the initial conditions and boundary conditions to produce a set of minimal cut sets for each of the system alignments and boundary conditions. The minimal cut sets are then converted into equations that can be used to quantify each of the system split fractions.

A listing of generic data used for component failure, component unavailability, and initiating events is provided in Section 3.3.1.

3.2.3 System Dependencies

Table 3.2-2 presents a summary of the system-to-system dependencies as analyzed in the SSPSS. More information on these dependencies is provided in the systems descriptions in Appendix E. This table highlights support system dependencies - support-to-support and support-to-frontline system dependencies. Other dependencies, such as frontline-to-frontline system, operator interdependencies, and initiating event-to-top event dependencies, are addressed in Section 3.1. Because Seabrook is a newer vintage plant, trainwise separation (cooling, power and control supply) and spatial considerations have been an element of the design process and, as such, many of the issues regarding dependencies raised on earlier plant designs are not important to Seabrook. Also, because of the trainwise separation, cross-connects between trains are not available. This has the effect of simplifying the analysis but also provides fewer options for recovery.

TABLE 3.2-1

System Summary

System	System Description
AC Power	Provides ac motive and control power necessary for normal operation and the response to abnormal events.
DC Power	Provides dc control power for diesel generator starting, switchgear and component manipulation as well as plant status monitoring.
Primary Component Cooling Water	Provides cooling water to prevent overheating of components necessary for normal operation to maintain core heat removal, and RCP seal integrity.
Service Water	Provides cooling water to transfer heat from the primary (PCC) and secondary (SCC) loads and the diesel generators to the ultimate heat sink (Atlantic Ocean or the atmosphere).
Solid State Protection System	Processes the output from sensors which monitor various plant parameters. Upon reaching unsatisfactory conditions, signals are sent to the RTS and ESFAS to initiate protective actions.
Engineered Safety Features Actuation System	Receives signals from the output logic channels of the SSPS and initiates any of a variety of equipment actuations - ECCS pump starts, valve strokes, etc.
Reactor Trip System	Initiates a reactor shutdown (trip) upon receipt of a trip signal from the SSPS.
Emergency Air Handling	Provides ventilation and component cooling to permit continuous operation of ECCS and CBS pumps.
Instrument Air	Provides air for pneumatic instruments and controls.
Emergency Core Cooling	Removes the stored and fission product decay heat from the reactor core following an accident. Its functions include high pressure injection, high pressure recirculation, low pressure injection, low pressure recirculation and RHR shutdown cooling.
Reactor Coolant Pressure Relief	Provides primary system pressure relief for overpressure transients and for cooling in the 'feed and bleed' mode.

TABLE 3.2-1
(Continued)

System Summary

System	System Description
Emergency Feedwater	Supplies water to the steam generators in order to remove heat from the reactor coolant system during events when the main feedwater is not available.
Main Steam	With regards to accident mitigation, provides for adequate secondary side heat removal, prevents excessive heat removal and provides overpressure protection of the main steam piping.
Containment Building Spray	Maintains the containment building pressure and temperature within design limits in the event of a main steam line break or LOCA. Also serves as an active means of containment building heat removal and fission product scrubbing.
Containment Isolation	Guards against the atmospheric release of fission products in the event of an accident by isolating those lines penetrating the containment which are not required for the operation of the engineered safety features systems.

TABLE 3.2-2

System Dependencies

Support Systems	Train	Support Systems																	
		DC		DG		IP		SSPS		RT		ESFAS		SW		PCC		EAH	
		A	B	A	B	A	B	A	B	A	B	A	B	A	B	A	B	A	B
Off-Site Power[1]	--	o	o	o	o	o	o			o	o			o	o	o	o	o	o
DC Power[2]	A	-		X		X		o		o		o		o		o		o	
	B		-		X		X		o		o		o		o		o		o
Diesel Generator[3]	A	o		-										X		X		X	
	B		o		-										X		X		X
Instrument Power[4]	A					-		o		o		X		o		o	o		
	B						-		o		o		X		o	o	o		
SSPS[5]	A			o				-		X		X		o					
	B				o				-		X		X		o				
Reactor Trip[6]	A									-									
	B										-								
ESFAS[5]	A			o								-							
	B				o								-						
Service Water[7]	A			X										-		X			
	B				X										-		X		
PCC[7]	A															-		X	
	B																-		X
EAH	A																	-	
	B																		-

Legend:

o = Dependency Exists
X = Direct Dependency - Guaranteed Failure
- = Same System and Train
DC = 125V DC Bus
DG = Diesel Generator and Associated Bus
IP = 120 V AC Instrument Power
SSPS = Solid State Protection System
RT = Reactor Trip
ESFAS = Engineered Safety Features Actuation System
SW = Service Water
PCC = Primary Component Cooling Water
EAH = Enclosure Building Ventilation

System Dependencies

Support Systems	Train	Front-Line Systems																							
		RHR		CVCS		SI		CBS		EFW			RC		MS		TT	CI		CBA		IA			
		A	B	A	B	A	B	A	B	A	B	S	A	B	A	B	-	-		A	B	-			
Off-Site Power[8]	--									o	o	X			o	o	o	o		o	o	o			
DC Power[9]	A	X		X		X		X		o		X	X		o		o	o		X		X			
	B		X		X		X		X	o	X			X		o	o	o			X	X			
Diesel Generator[10]	A	X		X		X		X		o		X					o	o		X		X			
	B		X		X		X		X	o	X						o	o			X	X			
Instrument Power[11]	A														o		o								
	B															o	o								
SSPS[12]	A	X		X		X		X		o		o			o		o	o		X		o			
	B		X		X		X		X	o	X	o				o	o	o			X	o			
Reactor Trip[13]	A																X								
	B																X								
ESFAS[12]	A	X		X		X		X		o		o			o		o	o		X		o			
	B		X		X		X		X	o	X	o				o	o	o			X	o			
Service Water[14]	A																					o			
	B																					o			
PCC[14]	A	o		X		X		o																	
	B		o		X		X		o																
EAH[15]	-	o	o		o	o	o	o	o												-				

Legend:

o = Dependency exists.
X = Direct dependency - guaranteed failure.
RHR = Residual Heat Removal System.
CVCS = Charging portion of Chemical Volume Control System.
SI = Safety Injection System.
CBS = Containment Building Spray System.
EFW = Emergency Feedwater. EFW includes the start-up feed pump (S), the turbine-driven pump (A), and electric-driven pump(B). The start-up pump is assumed unavailable with SI signal, loss of off-site power, or any feedwater isolation event.
RC = Primary safety and relief valves. Relief valves fail closed on loss of DC.
MS = Main steam isolation valves, atmospheric relief valves, and steam dump valves to condenser.
TT = Turbine trip.
CI = Containment Isolation (Phase A or B).
CBA = Control Building HVAC.
IA = Instrument Air. Assumed unavailable with SI signal or loss of off-site power.

System Dependencies

(1) Loss of off-site power results in SW, PCC, and EAH (normally operating systems) having to restart from emergency power. RT is generated (loss of power to control rod drive motor generator sets) independent of SSPS logic or RT breakers. Diesel generators are started on loss of bus voltage. Batteries must provide DC power for DG starts. Instrument power is provided temporarily from DC until DG provides AC source.

(2) With off-site power available, the DC bus is normally supplied by the battery charger with the battery as a backup. DC power is assumed to be available when off-site power is available due to the high reliability of the DC Power System for the 24-hour mission time. Failure of a DC bus is included as an initiating event. With loss of off-site power, loss of DC results in DG failure and failure of IP. SW, PCC, and EH failures are due to DC-induced failure of DG. SSPS, RT, and ESFAS dependencies are due to DC-induced failure of IP.

(3) DC batteries will eventually fail without restoration of AC power. SW, PCC, and EAH are directly dependent on AC power.

(4) Loss of 120 V AC instrument power results in loss of power to SSPS and ESFAS. Except for containment pressure, input parameters to SSPS are de-energized to actuate. The RT breaker undervoltage device will open the RT breaker. Also, the tower actuation ("TA") signal will fail. Isolation of PCC nonessential loads due to low head tank level occurs.

(5) DG receives start signal from SSPS and ESFAS. However, undervoltage at bus provides DG start without SSPS or ESFAS. RT and ESFAS are dependent on SSPS logic and signals. SCC isolation failure (for SI signal and LOSP, or TA signal) is assumed to fail SW.

(6) Reactor trip is essentially a frontline system and has no impact on support systems.

(7) Service Water includes cooling towers and associated pumps. For loss of off-site power, no credit is taken for cooling towers or standby SW pumps. DG's require SW for cooling. EAH is dependent on PCC.

System Dependencies

(8) On loss of off-site power, EFW turbine (A) starts (steam admission valves open), Electric Pump (B) starts (bus undervoltage), and start-up feed pump (S) fails without operator action on loss of Bus 4. The Steam Dump Valves (SDVs) fail closed with loss of air, turbine trip can be initiated by generator trip or reactor trip, many containment isolation valves fail safe (closed), CBA must be restarted, and instrument air is generally assumed unavailable due to loss of secondary cooling. For systems where loss of instrument air causes success, e.g., containment isolation valves fail closed, credit is given for operator action to align fire water cooling to the air compressors.

(9) DC power is assumed available with off-site power available. With loss of off-site power, DC failures guarantee DG and IP failures and, thus, major components lose control power and operating power. EFW turbine pump (A) starts (steam admission valves open). The start-up feed pump (S) is unavailable with loss of off-site power (Bus 4), but can be manually powered by Train A emergency power (Bus E5). The Atmospheric Relief Valves (ARVs) fail closed. The SDVs fail closed with loss of off-site power or IP. MSIVs fail as is (in the short term) upon loss of both DC trains. Turbine trip should occur from reactor trip or generator trip. Pressurizer PORVs fail closed, some CI valves fail safe (closed), and CBA air conditioning fails.

(10) On loss of DG or its associated bus with LOSP, major components lose operating power and power is lost to some CIS valves. Loss of off-site power starts EFW turbine (A) and start-up feed pump (S) is unavailable. The start-up feed pump can be manually powered from Train A power. TT is generated from generator trip or RT.

(11) The SDVs fail closed. TT should be generated from RT.

(12) No automatic start of major components with SSPS or ESFAS failure. With off-site power available, failure of both trains of SSPS or ESFAS will fail EFW turbine (A). EFW motor-driven pump (B) fails with SSPS Train B or ESFAS Train B failure if off-site power is available. The start-up feed pump (S) is blocked with an SI signal.

MSIV closure is successful with either SSPS and ESFAS train. Failure of both SSPS trains or both ESFAS trains will fail MSIV closure. Both SSPS or ESFAS train failure will fail the containment isolation function. Failure of both SSPS

System Dependencies

trains or both ESFAS trains (for high SG level) will fail normal turbine trip. SSPS and ESFAS isolate Secondary Component Cooling (SCC) from service water. Instrument air is dependent on SCC for compressor cooling. The CBA emergency cleanup filter fans fail to get autostart without SSPS or ESFAS.

(13) Turbine trip failure requires failure of both reactor trip breakers to open and failure of the ATWS Mitigating Systems Actuation Circuitry (AMSAC). Turbine trip may also be generated by generator trip.

(14) Failure of PCC results in failure of CVCS and SI pumps. RHR and CBS pumps also fail in the sump recirculation mode, and RHR will fail in mini-flow recirculation. Service water isolates SCC cooling to instrument air compressors requiring operator action to align fire water cooling.

(15) EAH failure (both trains) results in long-term RHR, CVCS, SI, and CBS pumps.

Other Considerations

- SI Accumulators are included with LPI function of the RHR.
- RWST supplies RHR, CBS, SI, and CVCS pumps in the injection mode.
- Condensate storage tank supplies motor-driven and turbine-driven EFW pumps and start-up feed pump.
- The following ventilation systems are included in the appropriate system analysis:
  - Switchgear area part of CBA and is included in electric power analysis.
  - EFW pumphouse included in EFW system analysis.
  - Service water and cooling tower switchgear ventilation systems are included in SW system analysis. Cooling tower pump room ventilation is also included.
  - PCC area included in PCC system analysis.
  - DG ventilation included in electric power analysis.

3.3 Sequence Quantification

3.3.1 List of Generic Data

This section presents the generic data utilized in the Seabrook Station PSS for component failure rates, component maintenance unavailabilities, and initiating event frequency data.

The data methodology used in the current SSPSS is, in general, based on the Bayesian treatment of data and the concept of "probability of frequency," as developed by PLG, Inc. The general methodology is documented in PLG-0500, Volume 1, "Methodology" (Reference 21) and is given in more detail for maintenance data and common cause data in PLG-0500, Volume 3 (Reference 22) and Volume 4 (Reference 23), respectively. This methodology is consistent with that used in the SSPSA (documented in Section 6.0 of the SSPSA).

Component Failure Rate Data

The component failure rate distributions are summarized in Table 3.3-1. These distributions are based on generic estimates (e.g., WASH-1400, Reference 24) and relevant data from other operating plants (where available), combined in a "stage one" Bayesian update (using methodology described in Reference 21). The specific bases for these distributions are found in PLG-0500, Volume 2 (Reference 25), except as described in footnotes to Table 3.3-1. Seabrook-specific data has not yet been included in these distributions based on limited operation experience, as described in Section 3.3.2.

The component failure rate distributions presented in Table 3.3-1 are generally consistent with the values used in the SSPSA.

Component Maintenance Data

The component maintenance data distributions for maintenance frequency and duration are summarized in Table 3.3-2. These distributions are based on relevant maintenance data from operating plants (e.g., Zion, Indian Point), combined in a "stage one" Bayesian update (using methodology described in Reference 21). The specific bases for these distributions are found in PLG-0500, Volume 3 (Reference 22). Seabrook-specific data has not yet been included in the distributions based on limited operating experience.

The maintenance distributions presented in Table 3.3-2 have been updated from the corresponding distributions in the SSPSA. First, the number of maintenance distributions used in the current model has been increased from the SSPSA maintenance model in order to more accurately account for differences in maintenance frequency and duration among various components. For example, the number of maintenance duration distributions has increased from four in the SSPSA to 17 in the current model.

In addition, the data base has been updated with additional maintenance data from other operating plants. This has resulted in generally increased estimates of maintenance frequencies and decreased estimates of maintenance durations. Thus, for example, for a standby pump the mean maintenance frequency has increased from 8.4E-5 per hour in the SSPSA to 1.2E-4 per hour in the current model. Similarly, for a pump with a seven-day allowed outage time per Technical Specifications, the mean maintenance duration has decreased from 40.4 hours to 28.7 hours.

Initiating Event Frequency

The frequencies for "internal" initiating events - i.e., transients and LOCAs - are given in Table 3.3-3. These distributions are based on generic data from operating plants combined in a "stage one" Bayesian update. The specific bases for these distributions are given in PLG-0500, Volume 6 (Reference 26). Seabrook-specific data has not yet been included in these distributions based on limited operating experience.

The distributions for transient initiators have been updated from the SSPSA with additional generic data through 1987. This updated data significantly reduced transient frequency estimates; for example:

Transient	Current (Mean)	SSPSA (Mean)	Ratio
Reactor Trip	1.35/yr	3.13/yr	0.4
Turbine	1.07	1.95	0.5
Loss of Condenser Vacuum	0.12	0.42	0.3

The distributions for LOCAs in Table 3.3-3 are equivalent to the corresponding distributions in the SSPSA.

The complete set of initiating events (internal and external events) used in the SSPSS-1990 is more fully discussed in Section 3.1.1.

3.3.2 Plant-Specific Data and Analysis

Because Seabrook Station has limited operational experience (commercial operational date of July 1990), plant-specific initiating events and component failure data have not been incorporated into the risk models. However, as discussed in Sections 3.1 and 3.2, plant-unique configurations are accounted for in developing various aspects of the Seabrook risk model. For example, the following initiating events were quantified using a model of Seabrook-specific configurations with generic failure rate data:

L1SWA(B) - Reactor trip due to loss of one train of Service Water System.

L1CCA(B) - Reactor trip due to loss of one train of Component Coolant Water System.

LDCA(B) - Reactor trip due to loss of one train of essential dc power.

LSF6 - Loss of off-site power due to SF6 bus duct failures in switchyard or transmission lines.

A data acquisition plan has been developed, as part of the Risk Management Program, to capture operational data related to component failure rates and maintenance unavailabilities. When sufficient data has been collected, the existing generic data distributions will be updated using the "stage two" Bayesian update option within the RISKMAN suite of codes.

A preliminary evaluation of the diesel generator data indicates better performance than the generic data used. For example, one failure has been recorded out of 149 starts for both diesels. The generic failure rate for "diesel fail to start" is:

- Generic Data (Without the Seabrook Data) - 2.14E-2 (mean).
- Seabrook Data (One Failure Out of 149 Starts) - 6.7E-3 (point estimate).
- Bayesian Update (Including the Seabrook Data) - 1.07E-2 (mean).

Thus, the Bayesian update is a factor of two less than the generic data.

3.3.3 Human Failure Data

The type of human interactions included in the SSPSS include the following (based on the classification scheme in EPRI-6560L, Reference 33):

- Type A - Pre-Initiating Event Interactions

  These cover routine operator or technician actions that can inadvertently disable safety equipment during test or maintenance. These actions are modeled in the systems analyses as contributions to train or system unavailability. Table 3.3-4 lists these actions, along with the event tree top event that includes the effects of the action. These actions were, in general, quantified using the handbook methods (NUREG/CR-1278, Reference 27), as documented in the SSPSA.

- Type B - Initiating Event Related Interactions

  These actions have been implicitly accounted for in the quantification of initiating events (see Section 3.1.1).

- Type C - Post-Initiating Event-Related Interactions

  These actions are dynamic operator responses to various initiating events and include the following general groups:

  - Actions taken during an event sequence which supplement the automatic response of plant systems for event mitigation (e.g., MT = manual reactor trip);

  - Actions required of operators for plant control (e.g., OM = operator controls EFW); and

  - Actions taken during an event sequence which lead to recovery of failed systems (e.g., ER = electric power recovery).

Table 3.3-5 lists these operations, including a description of the action, the time available, the procedure that directs the action, and a reference for the analysis and quantification of the action. Most of these actions were analyzed in the SSPSA, as discussed below. Other actions identified since then have been

quantified using a conservative screening value (generally 0.1 or 0.01), as noted in Table 3.3-5. An update to the quantification of important operator actions is planned as part of the living PSA.

The evaluation of dynamic operator actions (Type C) was performed in detail as part of the SSPSA (Section 10). The process of this evaluation is illustrated in Figure 3.3-1 and is explained below.

The first step was to make use of event sequence diagrams to identify operator tasks of interest, the context in which they arise, and the relevant accident sequences.

The definition of possible operator action sequences of interest allows one to focus on the second step - the types of plant information required and the factors which affect the operators' performance. The systems-human interaction information consists of systems knowledge, plant procedures, and simulator information. The systems knowledge and plant procedures are used as a basis to develop the operator action trees. The Seabrook training simulator experience was used to supplement and integrate the other systems-human interaction information (e.g., the number and types of alarms received during a transient scenario) and was used in conjunction with the other systems-human interaction information to estimate human error rates where existing human performance data do not exist and/or are not applicable.

Plant systems knowledge is explicitly defined in the systems analyses of the SSPSA (Appendix D). Plant procedures used in the human action analysis include the Westinghouse Emergency Response Guidelines (ERG), generic Westinghouse operating procedures and, wherever possible, the prospective Seabrook operating and emergency procedures.

Approximately 20 different transient scenarios were conducted at the Seabrook simulator under various auxiliary system states. The sequences were performed with and without operator action in order to better appreciate performance shaping factors such as the time available, number and importance of alarms, presentation of information, and the stress level. The simulator experience provided an opportunity to place into perspective operator interaction with the plant response while keeping in mind that the basis for operator action times and response is the detailed reactor kinetic and thermal-hydraulic plant safety analyses.

A team of PLG engineers, a cognitive psychologist, and a Control Room team of four prospective Seabrook operators (two reactor operators, one senior reactor operator, and a Shift Technical Advisor qualified as a senior reactor operator) participated in the transient scenario exercises. During the course of these transient scenarios, the simulator action was "frozen" in time in order to discuss possible operator course of actions due to parameter indications which might lead to misdiagnosis and the effect of that misdiagnosis on future operator performance. This information was used as input for the operators' plant status confusion matrix. The transients were performed with and without the prospective Seabrook Station procedures.

Another input into defining the factors that affect operator performance was an operator-plant status confusion matrix. This was developed in order to aid in the determination of whether the operators could misdiagnose one plant event as another. The list of initiating events in the SSPSA was used as an initial screen of events since it represents a reasonably complete list of plant transient or events. The development of this matrix helped identify possible operator errors, the results of these errors, and the possibility of recovery. The operator-plant status confusion matrix lists plant events where event symptoms may be confused.

The importance of the results of the confusion matrix is less important today with improved training programs (including simulator use) and procedures. The Control Room operators are provided and trained in the use of emergency restoration guidelines which guide the Control Room operators in the maintenance of safety functions (i.e., reactor shutdown, core cooled, and sufficient coolant inventory available). These guidelines do not require immediate correct diagnosis of the cause of the transient in order to select the correct procedure and therefore correct course of action. The results of the confusion matrix, therefore, are more important where the Control Room operators are required to take additional actions to prevent the release of radiation to the environment prior to any core damage.

The third major step shown in Figure 3.3-1 is the development of operator action trees. The operator action trees are developed from the system-human interaction information and the operator-plant status confusion matrix. They identify potential failure states should the operators fail to take timely proper action or take no action in the course of an accident sequence. A generalized sequence is that once the event occurs, the operators check their indications, perform a diagnosis, and then take action. The operator action tree is not a model of how the

operators think, for that is an interactive process (i.e., between checking of parameters, diagnosis, review of procedures, discussion). The operator action tree, however, is used to estimate the probability of arriving at an end state in an operator action sequence.

The final step is the quantification of the operator response defined by the previous step. Human performance quantification is performed using estimates from NUREG/CR-1278, NUREG/CR-2815 (Reference 65), and other PRAs. These are combined using expert opinion for applicability to the Seabrook-specific actions. An attempt was made to anchor these estimates by quantifying one action, stabilizing high pressure injection, based on historical events. This experience data was used in a Bayesian update of the expert estimates to create a consensus distribution.

The results of the human action modeling in terms of a rank of important actions is discussed in Section 3.4.2.

3.3.4 Common Cause Failure Data

Common cause failures have been treated in the SSPSS either explicitly, by identifying the causes of dependent failures, and incorporating them into the system or event sequence models, or implicitly, by using parameters to account for their contribution to system unavailability. These two methods are discussed below:

- Explicit Common Cause Failure Modeling

  Explicit methods involve the identification of specific causes of multiple related failures. These common causes are directly incorporated into the event tree and fault tree logic models. Hardware dependencies, involving common support systems, are described in Section 3.2.3 for system dependencies and in Section 3.1.1 for initiating events. Common cause failures resulting from "external" initiating events are especially important because they cross functional boundaries. Finally, human interactions are evaluated to identify the possibility for common cause failures.

- Parametric Common Cause Failure Modeling

  Parametric methods are used to model the effects of failure dependence without having to enumerate the specific causes directly in the model. Because the parameters are estimated from experience data in much the same

manner as component failure rates are estimated, the parametric methods implicitly account for all causes of multiple failures present in the systems from which the data is collected. This approach is consistent with the way in which independent failures are normally modeled in that the root causes of failure are implicit in the assignment of component failure rates, but not explicitly modeled.

The specific parametric method used in the SSPSS is the multiple greek letter (MGL) method (Reference 21), which is an expansion of the beta factor method developed and used in the SSPSA (see SSPSA, Section 4.3.5). In the MGL method, parameters are defined as follows:

ß = Conditional probability that the cause of the component failure will be shared by one or more additional components, given that a specific component has failed.

γ = Conditional probability that the cause of a component failure will be shared by one or more additional components, given that two specific components have failed.

Additional parameters can be defined by increasingly larger sets of components, but parameters are generally limited to three (four or more components failing) due to the lack of data.

The MGL method is used in the systems analysis to model common cause between or among identical components in similar configurations. Table 3.3-6 lists the common cause components groupings modeled for each system. The common cause grouping is added to the fault tree using RISKMAN (Reference 9) and is quantified using generic distributions (from PLG-0500, Reference 23) and Seabrook-specific data distributions (from a plant specialization of generic common cause data). Table 3.3-7 lists the generic and specific beta factors used in the SSPSS-1990. Generic gamma and delta factors used are from PLG-0500.

The modeling of common cause in the systems analysis has expanded from the modeling in the SSPSA to include more components. In addition, the common cause data distributions have changed based on updated data.

3.3.5 Quantification of Unavailability of Systems and Functions

The PC-based software package RISKMAN (Reference 9) was used to quantify the unavailabilities of the systems analyzed in the SSPSS-1990. As discussed in Section 3.2.2, failure expressions for the system analyses were derived using one of two methods: the block diagram method or the fault tree method. The first method involves using logic block diagrams to develop logic expressions (including common cause contribution and test maintenance configuration) by inspection. This method was used extensively in the original SSPSA. It was adequate for most systems because of the "clear" train separation that exists at Seabrook Station. The second method involves entering the normal configuration logic by means of a fault tree, which also accounts for common cause and different system alignments. The software then uses the constructed fault tree to generate an equation (cut set) file. The cut sets for the fault trees are generated using the IRRAS fault tree code (Reference 9). This method has been used on higher risk-important systems in order to more accurately account for common cause and maintenance contributions.

Appendix E provides a summary of each system analyzed in the SSPSS-1990. Section E.1, Table E.1-1, lists the unavailabilities calculated for the systems analyzed. This table also identifies the support states and boundary conditions (i.e., initiating events) which are associated with the system. Support system availability affects the availability of the system to perform its function, while the initiating event often determines the functional requirements of the system.

For the most part, the system analyses model only automatic responses. Proceduralized operator actions are modeled separately as top events in the event trees. However, there are a few systems which model operator actions in the system analysis. These exceptions are noted in Table E.1-1.

The system unavailabilities calculated for two train systems and for multi-system functions (e.g., high head injection) are converted into single train split fractions for use in the event tree quantification. This allows the conditional split fractions (e.g., B train failed given A train successful) to be calculated from the two-train and single-train system unavailabilities. The method of calculating split fractions from system unavailabilities is described in Section E.1.

3.3.6 Generation of Support System States and Quantification of Their Probabilities

The effect of support system unavailability on frontline systems is determined in the event tree sequence quantification by the success or failure of the relevant support systems. This quantification process uses rules with support system event tree directly linked to frontline trees. As such, the concept of "support system states" is not applicable to the Seabrook quantification process. Further information on the support and frontline tree quantifications are provided in the following section and in Sections 3.1.2 and 3.1.4.

3.3.7 Quantification of Sequence Frequencies

The quantification of sequences is performed by linking initiating events with event trees as illustrated in Figure 3.1-1. Initiating events and event trees are described in Section 3.1. Table 3.3-8 also summarizes the event trees that are linked for each initiating event group. As shown, all initiating events pass through the SEISMIC and SUPPORT event trees, and then to the appropriate frontline and long-term trees. The frequency of each initiating event, a sequence cutoff frequency, and the logic for linking to event trees for each initiator are input to the RISKMAN code for quantification of sequences. For example, the large LOCA quantification input includes the LLOCA frequency and a list of event trees in order of linking (SEISMIC, SUPPORT, LL1, and LL2). The linking of these four event trees is similar to creating one very large event tree for the large LOCA initiator. For Seabrook, a cutoff frequency of 1.0E-10 was used for core melt sequence quantification of all initiating events.

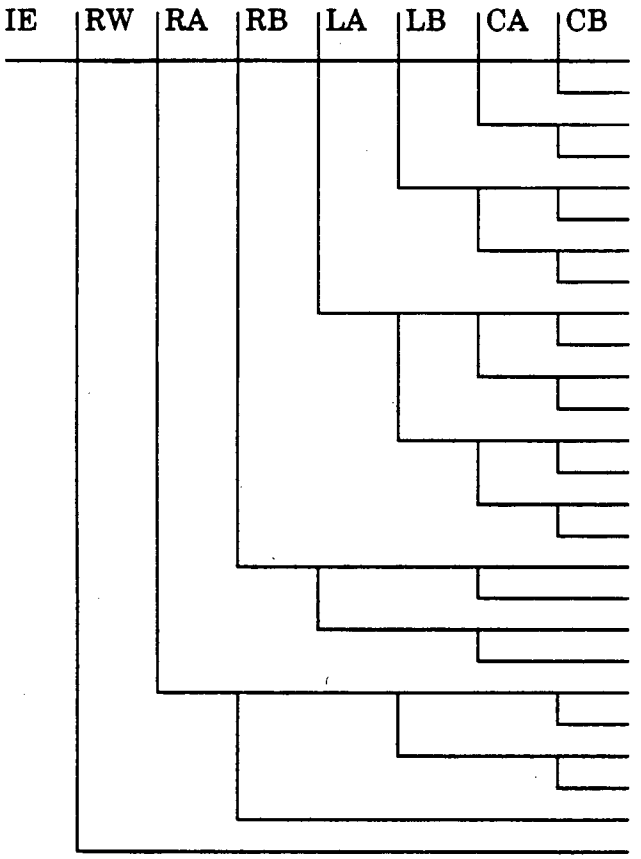
In each event tree, the success or failure of each top event (branch point) depends on the tree structure, top event rules, and the quantitative value assigned to the top event failure. Each failure value for a top event is referred to as a split fraction. A top event may have several split fractions due to top event dependencies on initiating events and success or failure of systems asked previously in the event tree models. The choice of split fractions for each top event during sequence quantification is based on logic rules. An example is provided below to illustrate the process.

Top event dependencies on initiating events are summarized in Table 3.1-3. Dependencies between systems are described in Section 3.2.3. Split fraction rules are one way that these dependencies are accounted for during sequence quantification. Another

way that dependencies are included is through the tree structure itself. In this case, an earlier top event failure guarantees failure of top events later in the same tree. Therefore, there is no branch point for these later top events (a pass through). These are discussed further in the example below.

The large LOCA event tree (LL1) is used as an example to illustrate the use of rules and the quantification of sequences. The LL1 event tree, its top event descriptions, and the split fraction rules used to quantify the LL1 tree are provided below. Note that the split fractions have a three-letter code where the first two are the same as the event tree top event and last one identifies the specific split fraction. When the third letter is an "F", this usually represents a guaranteed failure of the top event. In addition, in the logic rules the symbols "+", "*", and "-" represent "or", "and", "not" logic, respectively. "INIT=" is used to represent initiating events in the rules. "S", "F", and "B" are used to represent success, failure, and bypass of top events.

Large LOCA Frontline Event Tree (LL1)



Top Event Descriptions
IE - LLOCA Initiating Event
RW-RWST Available
RA - RWST Train A Valve Open
RB - RWST Train B Valve Open
LA - LPI Train A
LB - LPI Train B
CA - CBS Train A
CB - CBS Train B

Tree Structure Dependencies
RWST Train A Valve (RA) is a common supply to LA and CA.

RWST Train B Valve (RB) is a common supply to LB and CB.

RWST (RW) is common supply to RA, RB, LA, LB, CA, and CB.

The following rules are used to assign split fraction to top events during quantification of the LL1 tree:

Split Fraction	Split Fraction Logic (Rule)
RWF	QR=F
RW1	1
RA1	1
RB1	1
LAF	INIT=ELOCA + POWERA + ESFASA
LA2	1
LBF	INIT=ELOCA + POWERB + ESFASB
LBA	LA=F
LB1	LA=S
CAF	INIT=APC + POWERA + ESFASA + INIT=TMLL
CA2	1
CBF	INIT=APC + POWERB + ESFASB + INIT=TMLL
CBA	CA=F
CB1	CA=S

The meaning of the above split fractions and rules is described below:

Top event RW has two split fractions; the rule QR=F directs the quantification to use the value RWF (guaranteed failure of the RWST) if top event QR in the seismic event tree has failed along the sequence being quantified. Top event QR is only asked in the SEISMIC event tree for seismic initiating events and QR represents the failure split fraction for the RWST due to seismic fragility. If the initiating event is not seismic, a top event in the SEISMIC tree and its rules do not allow QR to be asked (no branches - pass through) and, therefore, it cannot fail. In this case, the quantification code passes to the next RW rule which is RW1. The "1" logic says to always use split fraction RW1 if previous rules did not apply. The split fraction RW1 represents unavailability of the RWST given no seismic failure.

The above example for RW illustrates how rules are used to evaluate seismic failures (switch on the seismic failure top events) in the SEISMIC tree for seismic initiating events and how dependencies in the latter trees are quantified based on seismic success or failure.

Top event RW in the LL1 tree also illustrates another method of modeling dependencies in event trees when top events later in the tree depend on RW. In this case, there is no branching when RW fails because the RWST is the only water source for low pressure injection (LA and LB) and containment spray (CA and CB). The pass through represents guaranteed failure of these top events. This approach to modeling dependencies must be taken into account in the long-term tree (LL2) where the top event, L1, depends on success of LA in the LL1 event tree. Therefore, the rule for guaranteed failure of L1 must consider the tree structure in LL1 and would look like the following:

L1F -LA=S

which says if LA is not success, L1 is guaranteed failure. Note that a pass through is not success. Another way to write this rule, which is also correct, would be as follows:

L1F LA=F + LA=B

which says if LA fails or if LA is a bypass (pass through), L1 is guaranteed to fail.

The rules for RA and RB simply say to always use RA1 and RB1. This is because the supply paths are normally open and failure modes are passive. That is, there are no boundary conditions associated with availability of support systems.

The rules for LA and LB include a guaranteed failure (LAF and LBF) if one of the following occur:

- Excessive LOCA initiating event (INIT=ELOCA) because by definition, it is beyond the capacity of the ECCS.

- Unavailability of AC power Train A (POWERA) to the pump and valves associated with LA (similar for LB). POWERA and POWERB are defined as macros in the SUPPORT event tree which introduces another method of using rules. For example, POWERA is defined as follows in the SUPPORT tree:

  POWERA:= OG=F*(GA=F + WA=F)

  POWERA can be used as a rule instead of OG=F*(GA=F + WA=F) after it is defined as a macro. This is true in the SUPPORT tree as well as all trees that follow the SUPPORT tree. When POWERA is used, it says if off-site ac power

fails (OG=F) and Diesel Generator Train A fails (GA=F) or Service Water Train A fails (WA=F), then LA is guaranteed to fail (LAF).

- Unavailability of signals (ESFASA) to start the pump and open valves associated with LA operation (similar for LB). ESFASA and ESFASB define macros in the SUPPORT event tree that define failure of the engineered safety feature actuation system trains.

If the first rule is not satisfied, that is ac power and signals are available and no excessive LOCA, the split fraction LA2 rule will be satisfied because of the "1". For LB, the split fraction is conditional on whether LA fails (LA=F) or succeeds (LA=S). When a two train system is split into separate top events, there are dependencies between these tops that requires the second top to be quantified conditional on the success or failure of the first top event. This explained further in Section E.1.

The containment spray top event rules (CA and CB) are similar to LA and LB except that an airplane crash into containment (INIT=APC) and turbine missile into containment (TMLL) large LOCA initiating events cause failure of containment spray.

The initiating event frequencies used in the quantification are given in Section 3.1.1 along with the event trees. The event tree top event failure fractions (split fractions) are based on human failure analysis results in Section 3.3.3 and the systems analysis results in Section 3.2 and Appendix E. A list of top event split fractions used to quantify the event trees is provided in Table 3.4-4 and is referred to as the master frequency file. The rules used to quantify the event trees are provided in Appendix F.

The binning of sequences to SUCCESS or plant damage states is based on binning rules defined for the last tree (RECOVERY, LT12, LL2, VS, and VI) that is linked to the initiating event. These rules are provided in Appendix F. When the containment event tree (CET) is linked on the end to each initiating event to obtain Level II results, the plant damage state bins are no longer required. Actually, the plant damage states binning rules are converted to macros and are used in the last plant model event tree split fraction rules such that the CET top event rules (split fraction rules) can use them. Binning rules are defined for collecting CET sequences in the appropriate release category. The CET split fraction and binning rules are provided in Appendix F.

3.3.8 Internal Flooding Analysis

The analysis of internal flooding was conducted as part of the spatial interaction study of the SSPSA. This analysis has recently been updated to better account for the as-built plant and to address industry experience with flooding. The analysis of internal flooding consists of the following steps:

1. Identification of critical locations where a single flood can simultaneously cause an initiating event (such as a LOCA, reactor trip, LOSP, etc.) and impact systems necessary to mitigate that event.

2. Calculation of the probability distribution of the frequency of floods in these areas.

3. Calculation of flood severity and its mitigation possibilities.

4. Establishment of the important scenarios and their frequencies.

A flood may lead to a LOCA if a valve inadvertently opens between the RCS and low pressure piping or containment atmosphere. This would require a hot short by wetting of bare conductors. It was determined that the only feasible LOCA from flooding was a small LOCA from an inadvertent opening of a PORV and its failure to reclose. This event was dismissed because of the small likelihood of a significant flood in the Control Room, which is the only location where base conductors related to the PORV control circuit can be found. Also, loss of off-site power can result from flooding of equipment in the northwest corner of the ground floor of the Turbine Building. All other initiating events are considered as general transients. It is assumed that all floods considered will lead to a reactor trip either directly from the flood or from the operator's judgment to scram the reactor.

The flooding analysis in Section 9.5 of the SSPSA is similar to the fire analysis and the primary basis for component locations was the Seabrook fire protection evaluation and fire protection safe shutdown study. To supplement this information, a three-day walkdown by four analysts (consisting of utility and consultant personnel) was performed to identify additional components not covered in the fire studies. Critical locations were identified by combining a plant systems-location matrix and a plant level fault tree to identify location minimal cut sets leading to core damage and release. Generation of these location

cut sets assumed that all components in an affected area were disabled by the flood; that is, equipment fragilities were not considered. Conservative frequencies were applied to develop an initial ranking of floods leading to core damage. These locations were then examined in greater detail to identify scenarios which, in fact, would be potential contributors to core damage.

The resulting flood locations from the screening analysis were the following:

EFW Pumphouse

Since floods in this area would impact only EFW equipment, its analysis is included into the EFW unavailability. The potential exists for flooding of electrical tunnels, but doors or plugs would need to be open to result in any significant spill. Also, these cables are expected to remain functional when submerged.

Control Building

The Control Building consists of three levels. Flooding of the first level is discussed later as part of the Turbine Building floods. The second level consists of the Cable Spreading Room and two HVAC Rooms. The Cable Spreading Room contains no large sources of water. Also, there are no splices of cables; therefore, the cables will not be affected by flooding or spray. The third level contains the Control Room. The only sources of water are the Fire Protection System in the stairwell and the portable water piping. Equipment would withstand up to 4" in the Control Room. Because of the existing drain paths, flooding of this area is not considered a significant hazard.

Primary Auxiliary Building (PAB)

The PAB can be divided into three elevations for flooding analysis. Elevations below 7'-0" contain no vital equipment. Elevation 7'-0" contains charging pumps, but no equipment required for safe shutdown. Elevation 25'-0" contains the four PCC pumps. The minimum time to reach this elevation would be one hour. Due to the time available for operator intervention, failure of the PCC pumps due to flooding is deemed almost impossible. Due to the metal partition between the two trains of pumps and shields above each pump, failure of all pumps due to a single spray source seems unlikely.

Turbine Building

The only significant scenarios quantified for the plant model involve flooding originating in the Turbine Building. Floods reaching one foot or more in depth on the floor of the turbine can result in loss of off-site power due to flooding of the relay cabinets in the northwest corner of the building. It is assumed that recovery of off-site power is not possible for this event. Only breaks in the service water and circulating water piping can produce floods of this magnitude.

The Train A (Emergency Bus E5) Switchgear Room is connected to the ground floor of the Turbine Building through a closed, but not watertight, door. Given that a flood in the Turbine Building reaches a depth of one foot, there is the potential that water will leak into the Train A Switchgear Room. It is assumed that a few inches of water in the Switchgear Room is sufficient to cause loss of Emergency bus E5. This results in a loss of off-site power and loss of one train of emergency power.

The two emergency Switchgear Rooms are also connected by a normally closed, but not watertight, door. If the flood water leaked from the Train A to Train B Switchgear Room, the result would be a loss of off-site power and total loss of emergency power, or station blackout. This is assumed to result in a guaranteed core melt, i.e., no recovery possible. The following three flood initiating events are included in Table 3.1-1:

FLLP - Flood in Turbine Building, loss of off-site power - 6.9E-4/yr

FL1SG - Flood in Turbine Building, loss of off-site power and loss of one vital Switchgear Room - 5.4E-6/yr

FL2SG - Flood in Turbine Building, loss of off-site power and loss of both vital Switchgear Rooms - 1.9E-7/yr

A confirmatory walkdown and analysis was performed to examine the as-built plant configuration. This analysis focused on penetrations between RHR equipment vaults, drain paths, and indications to the operator of significant flooding. In addition, industry experience identified in SOER 85-5 (Reference 66), NRC Information Notice 83-44 (References 62 and 63), and other sources was reviewed for its significance to Seabrook.

This analysis identified plant features that were not evaluated in the original analysis. For example, penetrations between the RHR equipment vaults above about 8 feet over the floor are sealed for fire separation but not flood. However, a flood that would impact the equipment vaults to this height would take time and is judged to not be significant. Also, the Switchgear Rooms contain drains that were not considered in the original analysis. These drains would likely remove water at the rate it leaked under the door unless the drains were clogged. This adds a conservative factor to this analysis. However, none of these features was important to the flooding risk. Also, no new scenarios were identified.

TABLE 3.3-1

Component Failure Rate Data[a],[b]

	Name of Distribution		Mean	5th	Med	95th
Major Rotating Equipment						
1.	ZIPMOS	NORMALLY OPERATED MOTOR DRIVEN PUMPS--FAIL. TO START ON DEMAND	2.35E-03	2.51E-04	1.44E-03	6.42E-03
2.	ZIPMOR	NORMALLY OPERATED MOTOR DRIVEN PUMP - FAILURE DURING OPERATION	3.36E-05	2.75E-06	1.64E-05	9.00E-05
3.	ZIPMSS	STANDBY MOTOR DRIVEN PUMPS--FAILURE TO START ON DEMAND	3.29E-03	2.22E-04	1.64E-03	1.01E-02
4.	ZIPMSR	STANDBY MOTOR DRIVEN PUMP - FAILURE DURING OPERATION	3.42E-05	2.83E-06	1.80E-05	8.19E-05
5.	ZIPTSS	TURBINE DRIVEN AUX.FEEDWATER PUMP - FAILURE TO START ON DEMAND	3.31E-02	5.75E-03	2.50E-02	7.10E-02
6.	ZIPTSR	TURBINE DRIVEN AUX. FEEDWATER PUMP - FAILURE DURING OPERATION	1.03E-03	6.10E-05	4.62E-04	2.91E-03
7.	ZIFN2S	VENTILATION FAN - FAILURE TO START ON DEMAND	4.84E-04	4.95E-05	2.83E-04	1.24E-03[c]
8.	ZIFN2R	VENTILATION FAN - FAILURE DURING OPERATION	7.89E-06	1.82E-06	6.04E-06	1.49E-05[c]
9.	ZIFN1S	COOLING TOWER FAN - FAILURE TO START ON DEMAND	2.93E-03	3.31E-04	1.73E-03	7.19E-03[c]
10.	ZIFN1R	COOLING TOWER FAN - FAILURE DURING OPERATION	7.89E-06	1.82E-06	6.04E-06	1.49E-05[c]
11.	ZICHLS	CONTROL ROOM VENTILATION CHILLER - FAILURE TO START ON DEMAND	8.07E-03	8.25E-04	4.72E-03	2.06E-02
12.	ZICHLR	CONTROL ROOM VENTILATION CHILLER - FAILURE DURING OPERATION	9.45E-05	2.21E-05	7.08E-05	1.99E-04
13.	ZICMPS	AIR COMPRESSOR FAILURE TO START ON DEMAND	3.29E-03	2.22E-04	1.64E-03	1.01E-02
14.	ZICMPR	AIR COMPRESSOR FAILURE DURING OPERATION	9.81E-05	1.28E-05	5.67E-05	2.76E-04
15.	ZIDGSS	DIESEL GENERATOR - FAILURE TO START ON DEMAND	2.14E-02	2.84E-03	1.34E-02	5.29E-02

Component Failure Rate Data

		Name of Distribution	Mean	5th	Med	95th
16.	ZIDGS1	DIESEL GENERATOR - FAILURE DURING FIRST HR OF OPERATION	1.69E-02	1.27E-03	9.89E-03	4.71E-02
17.	ZIDGS2	DIESEL GENERATOR - FAILURE TO RUN AFTER FIRST HOUR	2.50E-03	2.43E-04	1.60E-03	5.80E-03
Valves and Dampers						
1.	ZIVMOD	MOTOR OPERATED VALVE - FAILURE TO OPEN/CLOSE ON DEMAND	4.30E-03	7.27E-04	2.83E-03	1.10E-02
2.	ZIVMOT	M.O.V. / TRANSFER CLOSED OR TRANSFER OPEN DURING OPERATION	9.27E-08	1.03E-08	5.02E-08	2.37E-07
3.	ZIVMOE	MOV FAILURE TO CLOSE ON DEMAND WHILE SHOWING CLOSED	1.07E-04	1.51E-05	6.60E-05	2.45E-04
4.	ZIVMCX	VALVE (MOTOR-OPERATED OR CHECK) - DISC RUPTURE	1.55E-08	1.03E-10	4.09E-09	4.18E-08
5.	ZIVSOD	SOLENOID VALVE (DIRECT ACTING) FAILURE TO OPERATE ON DEMAND	2.43E-03	7.64E-05	9.79E-04	6.94E-03
6.	ZIVSOT	SOLENOID VALVE / TRANSFER OPEN OR SHUT DURING OPERATION	1.27E-06	5.21E-08	4.91E-07	3.59E-06
7.	ZIVAOD	AIR OPERATED VALVE / FAILURE TO OPERATE ON DEMAND	1.52E-03	2.83E-04	1.14E-03	3.16E-03
8.	ZIVAOF	AIR OPERATED VALVE FAILURE TO TRANSFER TO FAILED POSITION	2.66E-04	7.57E-06	1.04E-04	7.62E-04
9.	ZIVAOT	AIR OPERATED VALVE/TRANSFER OPEN/SHUT DURING OPERATION	2.67E-07	1.78E-08	1.20E-07	6.71E-07
10.	ZIVE1D	ELECTRO-HYDRAULIC VALVE (EXCEPT TSV,TCV) FAILURE TO OPERATE	1.52E-03	2.83E-04	1.14E-03	3.16E-03
11.	ZIVE1T	ELECTRO-HYDRAULIC VALVE (EXCEPT TSV,TCV) TRANSFER OPEN/CLOSED	2.67E-07	1.78E-08	1.20E-07	6.71E-07
12.	ZIVTCD	BUTTERFLY TEMPERATURE CONTROL VALVE - FLT TO OPERATE ON DEMAND	1.52E-03	2.83E-04	1.14E-03	3.16E-03[(c)]

Component Failure Rate Data

		Name of Distribution	Mean	5th	Med	95th
13.	ZIVTCF	BUTTERFLY TEMPERATURE CONTROL VALVE - FAIL TO TRFR TO FAILED PSN	2.66E-04	7.57E-06	1.04E-04	7.62E-04[c]
14.	ZIVTCT	BUTTERFLY TEMPERATURE CONTROL VALVE-TRFR OPEN/ SHUT DURING OPS.	4.20E-08	1.69E-09	1.41E-08	1.31E-07[c]
15.	ZIVCSD	CHECK VALVE (STOP) - FAILURE TO OPERATE ON DEMAND	9.13E-04	7.01E-05	4.21E-04	2.35E-03[c]
16.	ZIVCSL	CHECK VALVE (STOP) - GROSS LEAKAGE DURING OPERATION	5.36E-07	8.21E-08	3.46E-07	1.37E-06
17.	ZIVCSP	CHECK VALVE (STOP) - TRANSFER CLOSED/PLUGGED	1.04E-08	2.43E-09	7.80E-09	2.19E-08
18.	ZIVCOD	CHECK VALVE (OTHER THAN STOP) - FAILURE TO OPERATE ON DEMAND	2.69E-04	5.56E-05	1.50E-04	5.43E-04
19.	ZIVCOL	CHECK VALVE (OTHER THAN STOP) - GROSS LEAKAGE DURING OPERATION	5.36E-07	8.21E-08	3.46E-07	1.37E-06
20.	ZIVCOP	CHECK VALVE (OTHER THAN STOP) - TRFR CLOSED/PLUGGED	1.04E-08	2.43E-09	7.80E-09	2.19E-08
21.	ZIVHOT	MANUAL VALVE - TRANSFER OPEN/SHUT DURING OPERATION	4.20E-08	1.69E-09	1.41E-08	1.31E-07
22.	ZIVR2O	RELIEF VALVE (EXCEPT PORV, SAFETY) - FAILURE TO OPEN ON DEMAND	2.42E-05	7.55E-07	9.72E-06	6.92E-05
23.	ZIVR2T	RELIEF VALVE (OTHER THAN PORV OR SAFETY) - PREMATURE OPEN	6.06E-06	1.08E-06	3.94E-06	1.73E-05
24.	ZIVR1O	PRIMARY SAFETY VALVE - FAILURE TO OPEN ON DEMAND	3.28E-04	1.34E-05	1.41E-04	1.08E-03
25.	ZIVR1S	PRIMARY SAFETY VALVE - FAILURE TO RESEAT ON DEMAND	2.87E-03	8.84E-05	1.15E-03	8.21E-03
26.	ZIVR1W	PRIMARY SAFETY VALVE - FAILURE TO RESEAT AFTER WATER REL	1.01E-01	2.88E-03	1.20E-01	2.50E-01
27.	ZIVR3O	PORV FAILURE TO OPEN ON DEMAND	4.27E-03	9.95E-04	3.20E-03	8.98E-03
28.	ZIVR3C	PORV - FAILURE TO RESEAT ON DEMAND	2.50E-02	5.85E-03	1.87E-02	5.25E-02

Component Failure Rate Data

		Name of Distribution	Mean	5th	Med	95th
29.	ZIVE2D	TURBINE STOP/CONTROL VALVE FAILURE TO OPERATE ON DEMAND	1.25E-04	2.92E-05	9.37E-05	2.63E-04
30.	ZIVE21	TURBINE STOP/CONTROL VALVE TRANSFER CLOSED DURING OPERATION	2.88E-05	8.23E-07	1.13E-05	8.25E-05
31.	ZIVE22	TURBINE STOP/CONTROL VALVE TRANSFER OPEN DURING OPERATION	1.24E-05	3.54E-07	4.85E-06	3.55E-05
32.	ZIDAOD	PNEUMATIC DAMPER - FAILURE TO OPERATE ON DEMAND	1.52E-03	2.83E-04	1.14E-03	3.16E-03
33.	ZIDAOT	PNEUMATIC DAMPER - TRANSFER OPEN OR SHUT DURING OPERATION	2.67E-07	1.78E-08	1.20E-07	6.71E-07
34.	ZIDAOF	PNEUMATIC DAMPER - FAILURE TO TRANSFER TO FAILED POSITION	2.66E-04	7.57E-06	1.04E-04	7.62E-04[d]
35.	ZIDFRI	FIRE DAMPER - INADVERTENT ACTUATION	4.20E-08	1.69E-09	1.41E-08	1.31E-07[d]
36.	ZIDBDD	BACKDRAFT DAMPER - FAILURE TO OPEN ON DEMAND	2.69E-04	5.56E-05	1.50E-04	5.43E-04[d]
37.	ZIDBDT	BACKDRAFT DAMPER - TRANSFER CLOSED	1.04E-08	2.43E-09	7.80E-09	2.19E-08[d]
38.	ZIDHOT	MANUAL DAMPER - TRANSFER OPEN/SHUT DURING OPERATION	4.20E-08	1.69E-09	1.41E-08	1.31E-07[e]
39.	ZIDMOD	MOTOR OPERATED DAMPER - FAILURE TO OPEN/CLOSE ON DEMAND	4.30E-03	7.27E-04	2.83E-03	1.10E-02[e]
40.	ZIDMOT	MOTOR OPERATED DAMPER - TRANSFER CLOSED OR OPEN	9.27E-08	1.03E-08	5.02E-08	2.37E-07[e]
Pressure Vessels						
1.	ZIHXRB	HEAT EXCHANGER - RUPTURE/EXCESSIVE LEAKAGE DURING OPERATION	1.95E-06	3.16E-07	1.33E-06	5.20E-06
2.	ZITK1B	STORAGE TANK - RUPTURE DURING OPERATION	2.66E-08	7.59E-10	1.04E-08	7.63E-08
3.	ZIPP2B	PIPE, LESS THAN THREE INCH, PER PIPE SECTION	8.60E-09	1.98E-11	1.80E-09	2.02E-08
4.	ZIPP1B	PIPE, GREATER THAN THREE INCH, PER PIPE SECTION	8.60E-10	1.98E-12	1.80E-10	2.02E-09

Component Failure Rate Data

		Name of Distribution	Mean	5th	Med	95th
Nozzles, Strainers, Sumps, and Filters						
1.	ZISPNP	CONTAINMENT BUILDING SPRAY NOZZLES (TRAIN) PLUGGED	7.06E-08	2.70E-09	3.02E-08	2.00E-07
2.	ZISC1P	SERVICE WATER STRAINER - FAILURE DURING OPERATION	6.22E-06	8.08E-07	3.90E-06	1.58E-05(c)
3.	ZIFL1P	VENTILATION FILTER	1.07E-06	3.04E-08	4.16E-07	3.05E-06
4.	ZIFL2P	VENTILATION LOUVRE - PLUGGED	1.07E-07	3.04E-09	4.16E-08	3.05E-07(d)
5.	ZIRSCP	CONTAINMENT SUMP - PLUG DURING OPERATION	8.76E-06	3.07E-07	3.19E-06	3.19E-05*
6.	ZIFA1P	FILTER, AIR - PLUG DURING OPERATION	4.98E-06	1.75E-07	1.81E-06	1.81E-05(e)*
7.	ZIFA2P	FILTER, OIL REMOVAL - PLUG DURING OPERATION	3.01E-05	1.05E-06	1.10E-05	1.09E-04(e)*
8.	ZIFA3P	FILTER, COMPRESSED AIR - PLUG DURING OPERATION	1.50E-05	5.25E-07	5.45E-06	5.44E-05(e)*
Electrical Equipment						
1.	ZIXR1R	TRANSFORMER (GST,UAT,RAT) - FAILURE DURING OPERATION	1.56E-06	2.83E-07	1.10E-06	3.16E-06
2.	ZIXR2R	TRANSFORMER (STN.SERVICE,4.16KV TO 480V) - FAILURE DURING OPS.	6.87E-07	1.34E-07	4.47E-07	1.41E-06
3.	ZIXR3R	TRANSFORMER(INSTRUMENT) (480V TO 120V) - FAILURE DURING OPS.	1.55E-06	7.44E-08	6.57E-07	4.18E-06
4.	ZICB1O	CIRCUIT BREAKER (480 VAC AND ABOVE) - FAILURE TO OPEN ON DEMAND	6.49E-04	5.95E-05	3.67E-04	1.41E-03
5.	ZICB1C	CIRCUIT BREAKER (480 VAC AND ABOVE) - FAIL TO CLOSE ON DEMAND	1.61E-03	2.80E-04	1.22E-03	3.23E-03
6.	ZICB1T	CIRCUIT BREAKER (480 VAC AND ABOVE) - TRANSFER OPEN DURING OPS.	8.28E-07	5.08E-08	3.99E-07	2.36E-06
7.	ZICB2O	CIRCUIT BKR (AC OR DC,LT. 480V) - FAILURE TO OPEN ON DEMAND	8.39E-04	2.39E-05	3.28E-04	2.40E-03

TABLE 3.3-1
(Continued)

Component Failure Rate Data

		Name of Distribution	Mean	5th	Med	95th
8.	ZICB2C	CIRCUIT BKR (AC OR DC,LT. 480V) - FAILURE TO CLOSE ON DEMAND	2.27E-04	6.48E-06	8.89E-05	6.52E-04
9.	ZICB2T	CIRCUIT BREAKER (AC OR DC, LT.480V) - TRANSFER OPEN DURING OPS.	2.68E-07	2.50E-08	1.41E-07	9.11E-07
10.	ZISWBD	BISTABLE FAILURE TO OPERATE ON DEMAND	3.89E-07	5.98E-08	2.58E-07	9.16E-07
11.	ZISWBI	BISTABLE SPURIOUS OPERATION	2.21E-06	2.56E-09	4.01E-07	4.61E-06
12.	ZIBS1R	BUS - FAILURE DURING OPERATION	4.98E-07	7.73E-08	3.36E-07	1.17E-06
13.	ZIBATR	125V DC BATTERY - FAILURE OF OUTPUT DURING OPERATION	7.53E-07	5.88E-08	3.81E-07	1.73E-06
14.	ZIBATD	125V DC BATTERY - FAILURE OF OUTPUT ON DEMAND	4.84E-04	7.51E-05	3.26E-04	1.15E-03
15.	ZIBCHR	BATTERY CHARGER - FAILURE DURING OPERATION	1.86E-05	8.54E-07	7.58E-06	5.09E-05
16.	ZIMGSR	MOTOR GENERATOR - FAILURE DURING OPERATION	3.59E-05	1.10E-06	1.23E-05	1.19E-04[c]
17.	ZIPS1R	POWER SUPPLY	1.71E-05	1.18E-06	7.25E-06	4.39E-05
18.	ZIFU1R	FUSE - FAIL OPEN DURING OPERATION	9.20E-07	2.83E-08	3.16E-07	2.83E-06
19.	ZIRL1D	RELAY - FAILURE TO OPERATE ON DEMAND	2.41E-04	1.41E-05	1.35E-04	6.40E-04
20.	ZIRL1R	RELAY - FAILURE DURING OPERATION	4.20E-07	2.83E-08	1.90E-07	1.41E-06
21.	ZIINVR	INVERTER - FAILURE DURING OPERATION	1.83E-05	1.73E-06	1.14E-05	4.16E-05
22.	ZICCOS	CONTROL CABLE - OPEN OR SHORTED DURING OPERATION	4.63E-06	8.40E-07	3.23E-06	1.05E-05[c]
23.	ZIPSLR	POWER SUPPLY +5V DC,+24V DC ESFAS - FAILURE DURING OPERATION	5.33E-05	1.52E-06	2.08E-05	1.53E-04[c]
24.	ZIPSHR	POWER SUPPLY +120V DC ESFAS - FAILURE DURING OPERATION	1.33E-04	3.80E-06	5.21E-05	3.81E-04[c]
25.	ZICB3D	REACTOR TRIP BREAKER - FAILURE ON DEMAND	1.94E-03	4.53E-04	1.45E-03	4.07E-03

Component Failure Rate Data

	Name of Distribution		Mean	5th	Med	95th
26.	ZISTCD	REACTOR TRIP BREAKER SHUNT TRIP COIL FAIL TO OPEN	1.19E-04	2.88E-05	8.94E-05	2.53E-04(e)*
27.	ZIUVCD	REACTOR TRIP BREAKER UNDERVOLTAGE DEVICE FAIL TO OPEN	2.61E-03	6.34E-04	1.96E-03	5.28E-03(e)*
28.	ZIPBSD	PUSHBUTTON SWITCH - FAIL ON DEMAND	1.97E-05	7.75E-07	7.56E-06	7.09E-05(e)*
29.	ZIHTRR	HEAT TRACING LINES - FAIL DURING OPERATION	8.50E-06	2.08E-07	2.61E-06	3.15E-05(e)*
Off-Site Sources of Power						
1.	OG1	LOSS OF OFF-SITE GRID DUE TO UNIT TRIP	(f)			
Electronic Equipment						
1.	ZISEQD	ECCAS/LOP SEQUENCER - FAILURE ON DEMAND	2.40E-06	6.83E-08	9.37E-07	6.87E-06
2.	ZISMDR	SIGNAL MODIFIER - FAILURE DURING OPERATION	2.94E-06	4.66E-07	2.04E-06	6.42E-06
3.	ZICL1R	TRIP LOGIC MODULE - FAILURE DURING OPERATION	2.93E-06	8.35E-08	1.15E-06	8.39E-06
4.	ZILC1D	TRIP LOGIC MODULE - FAILURE TO TRIP ON DEMAND	8.52E-05	2.43E-06	3.33E-05	2.44E-04
5.	ZIOASR	OPERATIONAL AMPLIFIER SIGNAL MATRIX - NO OUTPUT DURING OPS.	6.23E-06	1.78E-07	2.44E-06	1.79E-05(d)
Instrumentation						
1.	ZITRFR	FLOW TRANSMITTER - FAIL DURING OPERATION	6.25E-06	6.03E-07	4.18E-06	1.41E-05
2.	ZITRLR	LEVEL TRANSMITTER - FAILURE DURING OPERATION	1.57E-05	3.51E-06	1.12E-05	3.34E-05
3.	ZITRPR	PRESSURE TRANSMITTER - FAILURE DURING OPERATION	7.60E-06	8.11E-07	4.69E-06	1.79E-05
4.	ZISWPD	PRESSURE SWITCH - FAIL TO OPERATE ON DEMAND	2.69E-04	1.41E-05	1.25E-04	7.69E-04
5.	ZITM1X	TEMPERATURE MONITOR LOOPS - NO OUTPUT	2.65E-06	2.89E-08	5.45E-07	9.86E-06(e)*

Component Failure Rate Data

	Name of Distribution	Mean	5th	Med	95th
Containment Building					
1. ZECISS	SMALL PREEXISTING LEAK IN CONTAINMENT	3.73E-03	9.36E-05	1.13E-03	1.24E-02(g)*
2. ZECISL	LARGE PREEXISTING LEAK IN CONTAINMENT	9.32E-05	1.63E-06	2.43E-05	3.48E-04(h)*
Scram Rods					
1. ZICRAD	SINGLE CONTROL ROD - FAIL TO INSERT ON DEMAND	3.20E-05	2.00E-06	1.02E-05	9.12E-05

NOTES:

(a) Failure modes "fail during operation" and "transfer open or closed" are in units of FAILURES PER HOUR.

(b) Component failure rate variables designated "ZI ____" in this table correspond to variables designated "ZT ____" in PLG-0500 (Reference 25).

(c) These variables names have been changed (from earlier data base versions) to be consistent with PLG-0500:

Variable Names:	New	Old
	ZIFN1S	ZIFN2S
	ZIFN1R	ZIFN2R
	ZIFN2S	ZIFN1S
	ZIFN2R	ZIFN1R
	ZIVTCD	ZIVBFD
	ZIVTCF	ZIVBFF
	ZIVTCT	ZIVBFT
	ZIVCSD	ZIVSCD
	ZISC1P	ZISCIP
	ZIMGSR	ZIMG1R
	ZICCOS	ZICCO

Variable Names:	New	Old
	ZIPSLR	ZIPS2R
	ZIPSHR	ZIPS3R

Component Failure Rate Data

(d) These variables are not defined in PLG-0500 (Reference 25) but are listed in the SSPSA (Section 6.2) and are defined below:

ZIDAOF - Due to unavailability of data for "pneumatic damper failure to transfer to failed position," the distribution for AOVs for the same failure mode (ZIVAOF) is used.

ZIDFRI - Due to unavailability of data for "fire damper inadvertent actuation," the distribution for manual valves for the same failure mode (ZIVHOT) is used.

ZIDBDD - Due to unavailability of data for "backdraft damper failure to open on demand," the distribution for check valves for the same failure mode (ZIVCOD) is used.

ZIDBDT - Due to unavailability of data for "backdraft damper transfer closed," the distribution for check valves for the same failure mode (ZIVCOP) is used.

ZIFL2P - Ventilation Lourve, plugged (basis not available).

ZIOASR - Operational amplifier, no output (basis not available).

(e) These variables are found in PLG-0500 (Reference 25) but not in the SSPSA:

ZIDHOT	ZIFA1P	ZIUVCD
ZIDMOD	ZIFA2P	ZIPBSD
ZIDMOT	ZIFA3P	ZIHTRR
ZIRSCP	ZISTCD	ZITM1X

(f) OG1 - This distribution is based on a more recent plant-specific review of loss of off-site power data, documented in PLG-0726 (Reference 14). The distribution in PLG-0500 is <u>not</u> used.

(g) ZECISS - This distribution is based on a data review of NPE documented in Section 3.17 of SSPSS-1990. This <u>is</u> consistent with the generic distribution in PLG-0500.

(h) ZECISL - This distribution is based on a data review and analysis documented in Section 3.17 of SSPSS-1990. This is <u>not</u> consistent with the generic distribution in PLG-0500.

* These distributions were created for the SSPSS-1990 using RISKMAN4 (DATAMAN) software with basic input data as described in PLG-0500 or in the footnotes above:

ZIRSCP	ZISTCD	ZIPBSD
ZIFA1P	ZIUVCD	ZIHTRR
ZIFA2P	ZECISS	ZITM1X
XIFA3P	ZECISL	

TABLE 3.3-2

Maintenance Frequency and Duration Data

	Name of Distribution	Mean	5th	Med	95th
1.	ZMELEF MAINTENANCE FREQUENCY - BATTERIES, BATTERY CHARGERS AND INVERTERS	2.49E-05	3.87E-06	1.41E-05	4.14E-05
2.	ZMBUSF MAINTENANCE FREQUENCY - BUSES	2.66E-06	1.29E-07	9.86E-07	7.04E-06
3.	ZMXFRF MAINTENANCE FREQUENCY - TRANSFORMERS	4.40E-06	1.21E-07	1.26E-06	1.25E-05
4.	ZMCMPF MAINTENANCE FREQUENCY - COMPRESSORS	2.93E-04	1.22E-05	1.06E-04	7.85E-04
5.	ZMCHLF MAINTENANCE FREQUENCY - CHILLERS	1.38E-04	7.90E-06	5.29E-05	3.76E-04
6.	ZMFN1F MAINTENANCE FREQUENCY - LARGE FANS	1.47E-04	3.85E-06	4.03E-05	4.05E-04
7.	ZMFN2F MAINTENANCE FREQUENCY - SMALL FANS	2.09E-04	8.85E-06	7.13E-05	5.74E-04
8.	ZMDGSF MAINTENANCE FREQUENCY - DIESEL GENERATORS	1.03E-03	1.65E-04	5.99E-04	2.13E-03
9.	ZMGTSF MAINTENANCE FREQUENCY - GAS TURBINES	1.92E-04	1.33E-05	9.31E-05	4.04E-04
10.	ZMHXRF MAINTENANCE FREQUENCY - HEAT EXCHANGERS	4.15E-05	2.38E-06	1.62E-05	1.12E-04
11.	ZMSC1F MAINTENANCE FREQUENCY - STRAINERS	9.27E-05	5.33E-06	3.69E-05	2.27E-04
12.	ZMPSWF MAINTENANCE FREQUENCY - OPERATING SERVICE WATER PUMPS	3.35E-04	2.64E-05	1.39E-04	8.46E-04
13.	ZMPOPF MAINTENANCE FREQUENCY - OTHER OPERATING PUMPS	1.58E-04	1.29E-05	7.35E-05	3.87E-04
14.	ZMPMSF MAINTENANCE FREQUENCY - OTHER STANDBY MOTOR (OR DIESEL)-DRIVEN	1.17E-04	7.96E-06	4.52E-05	3.27E-04
15.	ZMPTSF MAINTENANCE FREQUENCY - OTHER STANDBY TURBINE-DRIVEN PUMPS	4.19E-04	5.99E-05	2.41E-04	8.89E-04
16.	ZMPPDF MAINTENANCE FREQUENCY - POS. DISPL. PUMPS	6.37E-04	5.73E-05	3.41E-04	1.35E-03
17.	ZMVLVF MAINTENANCE FREQUENCY - VALVES	2.74E-05	3.94E-06	1.41E-05	5.72E-05
18.	ZMPNSD MAINTENANCE DURATION - PUMPS - NO TECH SPECS	2.66E+02	1.99E+00	4.72E+01	8.15E+02

Maintenance Frequency and Duration Data

	Name of Distribution	Mean	5th	Med	95th
19.	ZMPSSD MAINTENANCE DURATION - PUMPS - SHORT TECH SPECS	7.47E+00	1.24E+00	5.43E+00	1.82E+01
20.	ZMPMSD MAINTENANCE DURATION - PUMPS - 72 HOUR TECH SPECS	1.11E+01	1.16E+00	6.20E+00	3.08E+01
21.	ZMPLSD MAINTENANCE DURATION - PUMPS - 168 HOUR TECH SPECS	2.87E+01	2.58E+00	1.57E+01	7.27E+01
22.	ZMVNSD MAINTENANCE DURATION - VALVES - NO TECH SPECS	1.32E+02	7.23E-01	1.69E+01	4.10E+02
23.	ZMVSSD MAINTENANCE DURATION - VALVES - SHORT TECH SPECS	4.05E+00	6.83E-01	2.70E+00	9.52E+00
24.	ZMVLSD MAINTENANCE DURATION - VALVES - LONG TECH SPECS	1.89E+01	1.54E+00	1.01E+01	5.13E+01
25.	ZMOSSD MAINTENANCE DURATION - OTHER EQUIPMENT - 24 HR TECH SPECS	6.26E+00	5.46E-01	3.42E+00	2.02E+01
26.	ZMOMSDMAINTENANCE DURATION - OTHER EQUIPMENT - 48 & 72 HR TECH SPECS	1.31E+01	7.84E-01	6.01E+00	4.04E+01
27.	ZMOLSD MAINTENANCE DURATION - OTHER EQUIPMENT - LONG TECH SPECS	3.72E+01	8.20E+00	2.75E+01	7.41E+01
28.	ZMONSD MAINTENANCE DURATION - OTHER EQUIPMENT - NO TECH SPECS	3.85E+01	1.37E+00	1.37E+01	1.17E+02
29.	ZMHXNDMAINTENANCE DURATION - HEAT EXCHANGERS - NO TECH SPECS	5.83E+02	6.34E+01	3.68E+02	1.53E+03
30.	ZMCHNDMAINTENANCE DURATION - CHILLERS - NO TECH SPECS	4.70E+02	1.68E+01	2.02E+02	1.53E+03

TABLE 3.3-3

Initiating Event Frequency[a]

		Name of Distribution	Mean	5th	Med	95th
1.	ZEEXL	EXCESSIVE LOCA	2.66E-07	7.10E-09	8.75E-08	8.07E-07
2.	ZELLOC	LARGE LOCA	2.03E-04	6.73E-06	8.11E-05	5.75E-04
3.	ZEMLOC	MEDIUM LOCA	4.65E-04	1.86E-05	2.00E-04	1.11E-03
4.	ZESLOC	SMALL LOCA, NONISOLABLE	5.83E-03	1.14E-04	1.80E-03	1.65E-02
5.	ZESLI	SMALL LOCA, ISOLABLE	2.30E-02	4.12E-04	8.73E-03	4.84E-02[b]
6.	ZESLBI	STEAMLINE BREAK INSIDE CONTAINMENT	4.65E-04	1.86E-05	2.00E-04	1.11E-03
7.	ZESLBO	STEAM LINE BREAK OUTSIDE CONTAINMENT	6.04E-03	1.84E-04	2.18E-03	1.74E-02
8.	IMSRV	INADVERTENT OPENING OF MAIN STEAM RELIEF VALVES	4.19E-03	7.64E-05	1.12E-03	1.14E-02
9.	SGTR	STEAM GENERATOR TUBE RUPTURE	2.84E-02	2.06E-04	5.91E-03	8.66E-02
10.	OMSIV	CLOSURE OF ONE MSIV	8.66E-02	6.31E-03	4.46E-02	2.22E-01
11.	AMSIV	INADVERTENT CLOSURE OF ALL MSIV'S	1.93E-02	5.97E-04	1.13E-02	5.64E-02
12.	EXFW	EXCESSIVE FEEDWATER FLOW	1.68E-01	8.40E-03	7.10E-02	5.27E-01
13.	CPEX	CORE POWER EXCURSION	2.68E-02	9.05E-04	1.50E-02	5.16E-02
14.	RT	REACTOR TRIP	1.35E+00	4.09E-01	1.07E+00	2.79E+00
15.	PLMFW	PARTIAL LOSS OF MAIN FEEDWATER	1.13E+00	2.02E-01	8.04E-01	2.57E+00
16.	TLMFW	TOTAL LOSS OF MAIN FEEDWATER	1.62E-01	1.80E-02	9.72E-02	4.06E-01
17.	LOCV	LOSS OF CONDENSER VACUUM	1.18E-01	1.90E-02	8.36E-02	2.64E-01
18.	LPF	LOSS OF PRIMARY FLOW	1.76E-01	8.41E-03	7.54E-02	5.12E-01
19.	TT	TURBINE TRIP	1.07E+00	3.80E-01	9.21E-01	1.85E+00
20.	ISI	INADVERTENT SAFETY INJECTION SIGNAL	2.99E-02	2.98E-04	7.81E-03	8.77E-02

Initiating Event Frequency

NOTES:

(a) Initiating event frequencies are in units for EVENTS PER CALENDAR YEAR.

(b) Small LOCA initiating event frequency (ZESL) is calculated as follows:

ZESL = ZESLOC + 0.5* ZESLI

based on assuming half (0.5) of the isolable LOCAs are isolated before safety injection would occur.

TABLE 3.3-4

Pre-Initiating Event Human Actions

Human Action Basic Event	Related System Analysis	Human Action	Mean Value	Reference
HE1	SSPS	Human error of miscalibration of a single instrument loop	2.75E-03	SSPSA, Appendix D.6
HE2	SSPS	Human error of miscalibration of a second instrument loop, given first miscalibration	1.45E-01	SSPSA, Appendix D.6
HE3	SSPS	Human error - failure to detect miscalibration	5.25E-03	SSPSA, Appendix D.6
HE1	ACP	Human error - miscalibration of undervoltage relay	4.7E-03	SSPSA, Appendix D.2
HE2	ACP	Human error - miscalibration of second undervoltage relay given first miscalibration	1.73E-01	SSPSA, Appendix D.2
ZHE01A	DCP	Human error to cross-connect dc buses during battery maintenance	2.2E-03	SSPSS-1990, Section 3.3
ZHE02A	EFW	Operators fail to realign EFW flow when required during testing.	2.2E-02	SSPSS-1990, Section 3.14
ZHE01B	EFW	Operators fail to realign EFW flow path after testing.	4.7E-03	SSPSS-1990, Section 3.14
ZHEC02	EFW	Operators fail to realign startup feed pump after test.	4.7E-03	SSPSS-1990, Section 3.14
ZHEOEF	EFW	Second operator fails to find an EFW alignment error.	5.1E-02	SSPSS-1990, Section 3.14

TABLE 3.3-5

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
OD1	OD	Rapidly depressurize steam generators to cooldown and depressurize the RCS, using either the steam dump or atmospheric dump valves.	30 Min.	2.6E-02	MLOCA	FR-C.1, Step 11	SSPSA, Section 10.3.2
OD2	OD	Rapidly depressurize steam generators to cooldown and depressurize the RCS, using either the steam dump or atmospheric dump valves.	60 Min.	1.3E-02	GT,SLOCA	FR-C.1, Step 11	SSPSA, Section 10.3.2
041	04	Operators depressurize the RCS using pressurizer spray, EFW, and S/Gs.	30 Min.	5.0E-02	SGTR	E-3, Step 14	SSPSA, Section 10.3.9
042	04	Depressurize and cooldown by feed and bleed given EFW failure.	30 Min.	7.0E-02	SGTR	FR-H.1, Step 10	SSPSA, Section 10.3.9
051	05	Operators continue feed and bleed cooling to about 350°F and 400 psig.	30 Min.	5.0E-02	SGTR	FR-H.1, Step 10	SSPSA, Section 10.3.9
052	05	Operators depressurize the RCS given failure of high pressure injection.	30 Min.	9.0E-02	SGTR	FR-C.1, Step 11	SSPSA, Section 10.3.9
053	05	Same as OD2 but used in SGTR event tree.	60 Min.	1.3E-02	SGTR	FR-C.1, Step 11	SSPSA, Section 10.3.2 OD2
OM1	OM	Control EFW flow to prevent overcooling (PTS concern) given turbine trip failure and main steam line isolation failure.	20 Min.	6.2E-02	GT, SLOCA, SGTR, SLBI	FR-P.1 Step 1, E-2, Step 4	SSPSA, Section 10.3.3

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
OP1	OP	Control HPI flow to prevent overpressurization (PTS concern) given turbine trip failure.	30 Min.	2.3E-02	GT, SLOCA, SGTR, SLBI	FR-P.1 Step 6, ES-1.1	SSPSA, Section 10.3.4
OR(3)	OR	Operator action - feed and bleed given failure of EFW.	2 Hrs.	1.7E-02	GT, SLOCA, SGTR, SLBI, SLBO	FR-H.1, Step 10	SSPSA, Section 10.3.6
OQ1	OQ	Long-term plant stabilization.	>3 Hrs.	0.0(4)	GT, SLOCA, SLBI, SLBO	E-0	SSPSA, Section 10.3.4
OQ2	OQ	Operator action - plant stabilization - depressurize S/Gs to cooldown and depressurize the RCS, using either the steam dump or atmospheric dump valves.	60 Min.	1.3E-02	GT, SLOCA, SLBO, SLBI	FR-C.1 Step 11	SSPSA, Section 10.3.2
OQ3	OQ	Plant stabilization given failure to control primary-to-secondary leak.	>3 Hrs.	1.0E-04	SGTR	E-3	(10)
O31	O3	Operators manually align ECCS flow for low pressure/high pressure sump recirculation.	6-8 Hrs.	8.0E-04	Late Trees	ES-1.3	SSPSA, Section 10.3.7
O31	HE, HS	Operator action - low pressure hot leg recirculation. Included with hardware failure of HE1, HE2, HS1, and HS2.	18-20 Hrs.	8.0E-04	LLOCA Late Tree	ES-1.4	SSPSA, Section 10.3.7

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
LR(3)	LR	Operators align RHR for long-term recirculation given all support systems available.	6-8 Hrs.	8.0E-04	SLOCA	ES-1.2 Step 30	SSPSA, Section 10.3.7
OH(3)	OH	Operator initiates emergency boration or trips the reactor following an ATWS event. Included with hardware in OH1.	10 Min.	5.3E-03	ATWS	FR-S.1, Steps 4 and 5	WCAP-11992
MT1	MT	Operators manually generate a reactor trip signal given failure of the SSPS.	1 Min.	1.0E-02	SUPPORT	E-0, Step 1	WCAP-11992
RMU	RM	Operator provides makeup to RSWT - SLOCA.	4-6 Hrs.	1.0E-01	SLOCA	ECA-1.1	(5)
MR(3)	MR	Manual rod insertion (auto and manual) given ATWS. Included with hardware in MR1.	1 Min.	2.0E-02	ATWS	FR-S.1, Step 1	WCAP-11992
OS1	OS	Recover ESFAS - long response time available.	60 Min.	1.0E-02	SUPPORT	E-0	(6)
OS2	OS	Recover ESFAS - LOCAs, seismic events.	20 Min.	1.0E-01	SUPPORT	E-0	(5)
O11	01	Diagnose V-Sequence.	0.5-1.0 Hr.	6.5E-03	VI, VS	ECA-1.2	PLG-0432
O22	O2	Operators isolate the RHR System LOCA and throttle flow into the Primary System.	0.5-1.0 Hr.	9.1E-03	VI	ECA-1.2	PLG-0432

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
O3C	O3	Operators provide makeup to the RWST during an interfacing system LOCA.	0.5-2.0 Hrs.	4.9E-03	VI, VS	ECA-1.1	PLG-0432
ER1(7)	ER	Recovery of 1 of 2 D/Gs or Off-Site Power (OSP) given EFW available.	7-14 Hrs.	1.1E-02	RECOVERY	ECA-0.0	PLG-0507, Revision 2
ER2(7)	ER	Recovery of 1 of 2 D/Gs or OSP given EFW not available.	2 Hrs.	6.4E-02	RECOVERY	ECA-0.0	PLG-0507, Revision 2
ER3(7)	ER	Recovery of one D/G or OSP - given EFW available.	7-14 Hrs.	1.2E-02	RECOVERY	E-0, Step 3	PLG-0507, Revision 2
ER4(7)	ER	Recovery of one D/G or OSP - no EFW.	2 Hrs.	6.7E-02	RECOVERY	E-0, Step 3	PLG-0507, Revision 2
ER5(7)	ER	Recovery of 1 of 2 D/Gs, no OSP - EFW available.	7-14 Hrs.	7.1E-01	RECOVERY	ECA-0.0	PLG-0507, Revision 2
ER6(7)	ER	Recovery of 1 of 2 D/Gs, no OSP - no EFW.	2 Hrs.	8.9E-01	RECOVERY	ECA-0.0	PLG-0507, Revision 2
ER7(7)	ER	Recovery of one D/G, no OSP - EFW available.	7-14 Hrs.	5.6E-01	RECOVERY	E-0, Step 3	PLG-0507, Revision 2
ER8(7)	ER	Recovery of one D/G, no OSP - no EFW.	2 Hrs.	6.9E-01	RECOVERY	E-0, Step 3	PLG-0507, Revision 2
ER9(7)	ER	Recovery of off-site power - EFW available.	7-14 Hrs.	4.8E-02	RECOVERY	E-0, Step 3	PLG-0507, Revision 2
ERA(7)	ER	Recovery of off-site power - no EFW.	2 Hrs.	1.9E-01	RECOVERY	E-0, Step 3	PLG-0507, Revision 2

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
OP TA(8)	WA, WB	Recover service water by actuation of cooling tower.	20 Min.	7.4E-03	GT, SLBO	E-0, Step 10 Also, OS1216.01	SSPSS, Section 3.5
CI(OP) (8)	CI	Manually close RCP seal return MOV outside containment. Included with hardware in CIC, CID, and CIH.	30 Min.	1.0E-03	LTGT	ECA-0.0, Step 8A	SSPSS, Section 3.17
QK(OP)	QK	Operator recovers relay chattering for seismic events up to and including 0.7g. Included with hardware in QK1-QK7.	30 Min.	1.0E-01	QUAKE	(9)	SSPSS, Section 4.2
S40(OP)	ER	Operators shed nonessential dc loads within 40 minutes to conserve battery life.	40 Min.	6.0E-01	RECOVERY	ECA-0.0, Step 14a	PLG-507
OLS(OP)	ER	Operators shed nonessential dc loads within 70 minutes to conserve battery life.	70 Min.	3.2E-02	RECOVERY	ECA-0.0, Step 14a	PLG-507
N/A	WA, WB	Operators isolate SW flow to D/Gs during SI signal with off-site power available (small LOCAs only).	4-6 Hrs.	0.0(4)	SUPPORT	E-1, Step 11	FSAR 9.2
N/A	PA, PB	Operators manually adjust flow to PCC components during post-LOCA recirculation.	1-6 Hrs.	0.0(4)	SUPPORT	ES-1.2 Steps 3f and 12d	SSPSS, Section 3.6
ZEFOP1 (8)	FR	Operators diagnose the loss of the turbine-driven EFW pump.	30 Min.	2.7E-03	GT, SLBO	FR-H.1	SSPSS, Section 3.14

TABLE 3.3-5
(Continued)

Post-Initiating Event Human Actions

Split Frac.	Top Event	Operator Action	Time Avail.(1)	Mean Value	Event Trees	Operator Guidelines(2)	Reference
ZEFOP2 (8)	FR	Auxiliary operators respond (in sufficient time) given successful diagnosis.	1 Hrs.	1.1E-03	GT, SLBO	FR-H.1	SSPSS, Section 3.14
ZEFOP3 (8)	FR	AO restores TDP given successful diagnosis and response.	2 Hrs.	1.9E-03	GT, SLBO	FR-H.1	SSPSS, Section 3.14
OPDIAS (8)	FR	Control Room operators diagnose the demand for EFW and start the SUFP given TDP and MDP failure.	2 Hrs.	1.0E-03	GT, SLBO	FR-H.1	SSPSS, Section 3.14
OPACTS (8)	FR	Operators align SUFP flow given successful diagnosis of demand.	2 Hrs.	2.0E-03	GT, SLBO	FR-H.1	SSPSS, Section 3.14
N/A	EF, FR	Operators manually open and control the atmospheric relief valves on loss of instrument air.	>4 Hrs.	0.0(4)	GT, SLBO, SLBI, SLOCA, MLOCA, ATWS	E-0, FR-H.1	SSPSS, Section 3.14
HE4	FCRAC FSRAC FCRSW	Operators regain control of the plant from the remote safe shutdown panels given a fire in the Control Room/ Cable Spreading Room affecting power or service water.	>4 Hrs.	5.3E-02	Initiating Event	OS1200.02	SSPSA, Section 9.4.6
HE2	FCRCC FSRCC	Operators regain control of the plant from the remote safe shutdown panels given a fire in the Control Room/Cable Spreading Room affecting Primary Component Cooling.	1 Hr.	2.3E-01	Initiating Event	OS1200.02	SSPSA, Section 9.4.6

Post-Initiating Event Human Actions

NOTES:

(1) Time available to begin action - see reference for basis for value.
(2) Operator guidelines - procedure references:
- FR - Functional Restoration Procedures. Includes procedures for Subcriticality (S), Core Cooling (C), Heat Sink (H), Integrity (P), Containment (Z), Inventory (I), Emergency Recirculation (F), and RDMS (R).
- ECA - Emergency Contingency Actions.
- E - Refers to Emergency Response Procedures. Includes for Reactor Trip or Safety Injection (E-0), Loss of Reactor or Secondary Coolant (E-1), and Steam Generator Tube Rupture (E-3).

(3) This operator action is combined with hardware failure in the split fraction.
(4) Because of the long time available to perform these actions and their relative simplicity, these actions were assumed to be guaranteed successful.
(5) This action was added to the SSPSS-1990 model using a 0.1 screening value. It will be examined in greater detail if it is found to be important to risk.
(6) This action was added to the SSPSS-1990 model using a 0.01 screening value. It will be examined in greater detail if it is found to be important to risk.
(7) Split fraction includes operator action implicitly through analysis of recovery time.
(8) Basic event used in systems analysis fault tree.
(9) While there are no explicit procedures to recovery from relay chatter, the EOPs provide explicit guidance to check operating systems. In addition, relay chatter does not necessarily lead to failure. This will be examined in greater detail in the future.
(10) This split fraction is conservatively quantified using a screening value based on the long-time available and multiple parallel success paths - continued feed and bleed cooling or secondary cooling with primary makeup or depressurizing and closed loop RHR cooling.

Components Modeled Using Common Cause

System	Components Modeled With Common Cause	Number of Components
AC Power System	• Diesel Generators (D/Gs)	2
	• D/G Fuel Oil Pumps	2
	• D/G Feeder Breakers	2
	• Incoming Switchgear Breakers	2
	• D/G Ventilation Fans	4
	• D/G Switchgear Ventilation Fans	4
Service Water System	• Service Water (SW) Pumps	4
	• SW Switchgear Ventilation Fans	2
	• Motor-Operated Valves (MOVs)	2,4*
	• Cooling Tower (CT) Pumps	2
	• CT Roof Exhaust Fans	2
	• CT Switchgear Ventilation Fans	2
	• CT Fans	3
Primary Component Cooling System	• PCC Pumps	4
	• PAH Ventilation Fans	2
Solid State Protection System	• Input Relays	6**
	• Parameter Channels	3**
	• Logic Channels	2
Engineered Safety Features Actuation System	• Master Relays	26***
	• Slave Relays	72***
Reactor Trip System	• Reactor Trip Breakers	2
	• Undervoltage Coils	2
	• Shunt Trip Coils	2
Emergency Air Handling System	• Pump Room Return Fans	2
	• Other Fans	4
Emergency Core Cooling System	• Charging Pumps	2
	• Safety Injection Pumps	2
	• RHR Pumps	2
	• MOVs	2*
Reactor Coolant System Pressure Relief	• Power-Operated Relief Valves (PORVs)	2
Emergency Feedwater System	• EFW Pumps (Pumps, Not Drivers)	2
Main Steam System	• Atmospheric Relief Valves (ARVs)	4
	• Main Steam Isolation Valves (MSIVs)	

Components Modeled Using Common Cause

System	Components Modeled With Common Cause	Number of Components
Containment Building Spray System	• CBS Pumps	2
	• MOVs	2,4*
Containment Isolation System	• MOVs	2*
	• AOVs	2

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* MOV common cause groupings have been made based upon valve function (e.g., open/close) and upon valve location.

** In actuality, there are eight input relays and four parameter channels. However, the model conservatively assumes that each parameter uses 2-out-of-3 logic. Most utilize 2-out-of-4 logic.

*** Number of relays in common cause model is dependent on initiating event.

TABLE 3.3-7

Generic and Plant-Specific Beta Factors for Common Cause Factor

Type	Applicable Components	Failure Mode	Mean Value
Generic Type A	• Operating Pumps	Failure to Run	1.00E-3
	• Operating Fans	Failure to Run	
	• Diesel Generators	Failure to Start	
	• Logic Trip Modules	Failure to Actuate	
	• Motor Generator Sets	Failure to Run	
Generic Type B	• Standby Pumps	Failure to Run	1.00E-2
	• Standby Fans	Failure to Run	
	• Turbine-Driven Pumps	Failure to Run	
	• Positive Displacement Pumps	Failure to Run	
	• Operating Pumps	Failure to Start or Restart	
	• Operating Fans	Failure to Start or Restart	
	• Diesel Generators	Failure to Run	
	• Mechanical Relief Valves	Failure to Open	
	• Mechanical Relief Valves	Failure to Reseat	
	• Mechanical Relief Valves	Premature Opening	
	• Check Valves	Failure to Open or Reopen	
	• Check Valves	Failure to Reseat	
	• Motor Generator Sets	Failure to Restart	
Generic Type C	• Standby Pumps	Failure to Start	7.00E-2
	• Standby Fans	Failure to Start	
	• Turbine-Driven Pumps	Failure to Start	
	• Positive Displacement Pumps	Failure to Start	
	• Motor-Operated Valves	Failure to Open	
	• Motor-Operated Valves	Failure to Close	
	• Electrohydraulic Valves	Failure to Open	
	• Electrohydraulic Valves	Failure to Close	
	• Mechanical Relays	Failure to Actuate	
	• Bistables	Failure to Actuate	
	• Switches	Failure to Actuate	
	• Target Rock Relief Valves (two-stage)	Failure to Open On Overpressure	
	• Target Rock Relief Valves (two-stage)	Failure to Open on Signal	
	• Solenoid Valves	Failure to Operate on Demand	
	• Circuit Breakers	Failure to Open	
	• Circuit Breakers	Failure to Close	
	• Air/Motor-Operated Dampers	Failure to Open	
	• Air/Motor-Operated Dampers	Failure to Close	
Plant Specific	• Primary Component Cooling Pumps	Failure to Start	7.02E-3
	• Primary Component Cooling Pumps	Failure to Run	2.40E-2
	• Service Water Pumps	Failure to Start	7.02E-3
	• Service Water Pumps	Failure to Run	6.22E-2

TABLE 3.3-8

Event Trees Linked for Quantification

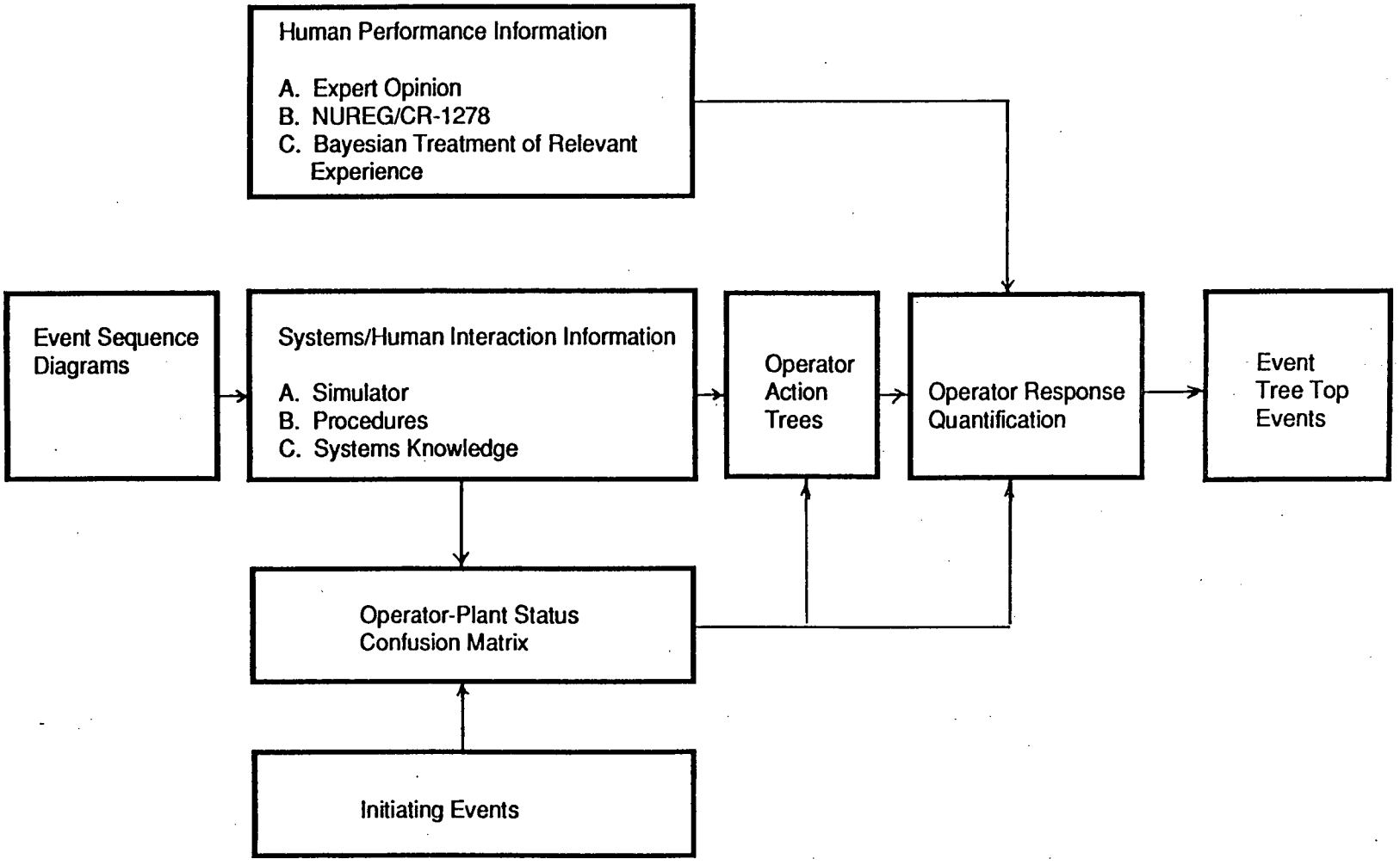
Initiating Event	Seismic[a]	Supp	GT	SLBI	SLBO	SLOCA	SGTR	MLOCA	LL1	ATWS	VS	VI	LT12	LL2	REC
Excessive LOCA	1	2							3					4	
Large LOCA	1	2							3					4	
Medium LOCA	1	2						3					4		
Small LOCA	1	2				3							4		5
SGTR	1	2					3						4		
V Sequence, VS	1	2									3				
V Sequence, VI	1	2										3			
Steam Line Break Inside Containment	1	2		3									4		
Steam Line Break Outside Containment	1	2			3								4		
Transient[b]	1	2	3										4		5(c)
ATWS	1	2								3			4		
Seismic Transient[a]	1	2	3										4		
Seismic LLOCA[a]	1	2							3					4	
Seismic ATWS[a]	1	2								3			4		

NOTES:

(a) All initiating events enter the seismic event tree, but the first top event allows all initiating events except seismic initiators to pass through without questioning seismic failures.
(b) The general category of transients contains internal events and loss of support systems including those due fires and floods and other hazards.
(c) Recovery of sequences in the recovery event tree only applies to certain transient initiators and sequences.

Figure 3.3-1

Human Actions Analysis Methodology



3.4 Results and Screening Process

3.4.1 Application of Generic Letter Screening Criteria

The Generic Letter provided specific screening criteria to report potentially important sequences and system failures that might lead to core damage or "unusually poor" containment performance. The following two sections explain how this screening criteria was applied for the systemic sequences out of the SSPSS-1990.

3.4.1.1 Core Damage Sequence Screening

Table 3.4-1 provides a listing of the 100 highest ranking sequences to core damage. This is consistent with the guidance from NUREG-1335 that the number of unique sequences to be reported "should not exceed the 100 most significant sequences." Additional, secondary screening criteria are addressed as follows:

- "Any systemic sequences that contribute 1.0E-7 or more per reactor year to core damage." - Table 3.4-1 lists all sequences with frequency equal to or greater than about 1.7E-7. To include all sequences greater than 1.0E-7 would have required about 40 additional sequences.

- "All systemic sequences within the upper 95% of the total core damage frequency." - Table 3.4-1 contains approximately 78% of the total core damage frequency. More than 300 sequences would be needed to reach 95% of the total core damage frequency.

Thus, Table 3.4-1 is restricted to 100 sequences based on the limitation cited above. Almost any number of additional sequences is available out of the event tree quantification. This listing of sequences contains the sequence rank; the frequency (events per reactor year); the initiating event; the sequence of support system availability, operator response and equipment response; and the percentage contribution to the total core damage frequency. The list includes external events (starred) consistent with the scope of the SSPSS.

These sequences can be understood by reference to Table 3.4-3 for a listing of initiating events, and to Table 3.4-4(a) for a listing of split fractions. In addition, it is helpful to note that the sequences of split fractions are separated by slashes (/) between each segment of the plant model, as follows:

/Seismic Failures/Support Systems/Early Response/Long-Term Response/.

Some sequences (station blackout and small LOCAs) also include an additional segment, a recovery split fraction, separated by a slash from long-term response. Also, the split fractions that end in "F" indicate guaranteed failures, i.e., systems or operator actions that fail due to "upstream" dependencies. Finally, the sequence listing provides only the failed split fractions. To see the complete sequence, including success terms, refer to the event trees in Section 3.1.

The top 20 sequences are discussed below. The additional sequences are generally different "flavors" of the same types of sequences.

The first sequence listed in Table 3.4-1 consists of a fire in the Control Room (FCRCC) which results in a loss of the PCC System (PAF and PBF). This sequence includes the evacuation of the Control Room and the failure to control the plant from the remote safe shutdown panels. As a result of the loss of PCC, a seal LOCA (NL) develops and high pressure injection fails. Core damage is assumed to follow, with no credit for recovery. Section 3.4.2 further explains the seal LOCA scenario. Other fire sequences in the top 20 which cause loss of support systems (PCC, ac power, service water) include Nos. 7, 12, 15, 16, and 19.

Sequence No. 2 is a station blackout sequence - loss of the 345 kV connection to the off-site grid due to a fire in the Turbine Building (FTBLP), failure of on-site ac power (diesel generator, GAl and GBA), and failure to recover the diesels' power in time to prevent core damage (ER5). This sequence also results in a seal LOCA scenario, which is discussed further in the following section. Sequence Nos. 3, 4, 5, and 17 are similar station blackout sequences except that loss of off-site power is due to other causes such as SF6 bus duct faults (LSF6), grid disturbances (LOSP), and flooding (FLLP), which affect the recoverability of power.

Sequence No. 6 is a reactor trip (RT) with subsequent failure of PCC (PA1 and PBA). Similar transients with loss of PCC are included in Sequence Nos. 8, 9, and 13. Sequence Nos. 10 and 11 are similar transients with loss of PCC, but initiated by loss of one train of PCC (LlCCB or LlCCA) with subsequent failure of the opposite train of PCC (PA4 or PB4). The plant response is similar to the first sequence - RCP seal LOCA with no makeup.

Sequence No. 14 is a 1.0g seismic event that causes failure of the off-site grid connection (QYA) and chatter of the 4,160 V breaker relays (QKA), which is assumed to cause nonrecoverable loss of on-site ac power. Similar seismic sequences occur below sequence No. 20.

Sequence No. 18 is initiated by a loss of one vital dc bus (LDCB) which results in a plant trip from loss of main feedwater. This also fails one train of ESF actuation (EBF). With the operator failure to recover the one train of signals (OS1), only the turbine-driven pump (EFD) is automatically actuated. With failure of secondary cooling (EFD) and unavailability of the feed and bleed option (due to conservative success criteria which requires both PORVs to be available), core damage is assumed to occur. In addition, if the operators fail to recover the signal failure(s), it is also assumed that they will fail to perform feed and bleed cooling, which is a more complex task.

Sequence No. 20 is an ATWS (anticipated transient without scram) loss of main feedwater initiator and subsequent failure of reactor trip breaker to open (RT1). At a power level above 40% (PL1), a Turbine Trip (TT) is required within 60 seconds; however, TT fails due to failure of the reactor trip breakers and the ATWS Mitigation System (AM1). Failure to prevent overcooling results in a pressure spike that is assumed to fail the reactor vessel. This analysis is consistent with the latest WOG analysis (Reference 19).

3.4.1.2 Containment Performance Screening

Table 3.4-2 provides a listing of the 100 highest ranking sequences to core melt with "unusually poor" containment performance. The definition of "unusually poor" containment performance is given below in terms of release categories. This list of sequences is consistent with the screening guidance from NUREG-1335 to not exceed 100 sequences. The additional, secondary screening criteria are addressed, as follows:

- "Systemic sequences that contribute to a containment bypass frequency in excess of 1.0E-8 per year." - Table 3.4-2 lists all sequences with large, early containment failure/bypass having frequencies greater than about 3.0E-10.

- "All systemic sequences within the upper 95% of the total containment failure frequency." - Table 3.4-2 lists all relevant sequences within the upper 80% of the total.

Thus, Table 3.4-2 is also restricted to 100 sequences based on the limitation cited above. Almost any number of additional sequences is available out of the event tree quantification. This listing of sequences is presented similar to Table 3.4-1 with the addition of the containment event tree (after the last slash on each sequence). The containment event tree split fractions are defined in Table 3.4-4(b). The sequence listing also indicates the release category to which the sequence is mapped.

The definition of containment release categories is contained in Table 3.4-5. As shown in this table, the term "unusually poor" containment performance has been defined as early, large containment failure or bypass, and includes Release Categories S1A, S1B, S6, and S7A. These release categories are described below, along with a reference to the sequences in Table 3.4-2.

Release category S1A involves gross containment structural failure due to overpressurization at the time of reactor vessel melt-through. The overpressurization may result from direct containment heating, hydrogen explosion, or steam explosion. Sequences 20, 25, etc. are high pressure core melt sequences with some chance of containment failure due to direct containment heating (top event CN). Release category S1B involves gross containment structural failure due to exterior forces - i.e., airplane crash or turbine missile crash into containment. Because of the low frequency, these sequences do not show up in Table 3.4-2. Release category S6 includes containment isolation failure, i.e., failure of the eight-inch containment purge valves to close on demand. A large number of sequences (Nos. 1, 2, 4, 5, 6, 7, etc.) include failures of signals, failure of operator (OS) to recover signal failures, failure of EFW (EF), and failure of purge valve to close, if open, due to loss of signal (C2M). Finally, release category S7A includes two containment bypass modes. First, induced steam generator tube rupture involves rupture of the SG tubes due to high RCS temperatures and pressures without steam generator makeup and with direct release through the secondary steam relief valves. Sequence Nos. 3, 8, 10, 11, etc., are transient high-pressure core melt sequences with failure of EFW. Second, the V-sequence pipe rupture is the "classic" LOCA through the RHR suction or injection lines. Pipe rupture allows containment release through the enclosure building to the environment. Sequence Nos. 15 and 27 are pipe rupture sequences from the RHR injection check valves (VI) or suction motor-operated valves (VS).

The other release categories not included in the definition of "unusually poor" containment performance are distinctly different in timing and magnitude of release. Release category S2 includes a small containment isolation valve failure ($\leq$ 3-inch diameter) with long-term overpressure failure. Release category S7B involves two distinct failure modes: (1) steam generator tube rupture (SGTR) with subsequent steam release (e.g., stuck open steam safety valve), and (2) the V-sequence, LOCA through the RHR suction or injection valves, but with the more likely failure of the RHR pump seals rather than piping. The second release mode is through a larger opening than SGTR, but the release is drastically reduced because it is through a thirty-foot pool of water. Release categories S3A, S3B, and S4 involve long term (>24-hour) failure of the containment.

This failure is a structural failure of containment shell due to overpressurization or basemat melt-through. Finally, release category S5 involves intact containment with only Technical Specification-allowed leakage. Section 4.0 contains the details of containment event tree and the source terms assigned to each release category.

Thus, the list of sequences in Table 3.4-2 and the subsequent discussion about containment performance are based on this definition of "unusually poor" containment performance as early, large containment failure/bypass.

3.4.2 Vulnerability Screening

The following two sections discuss the results of the vulnerability screening for core damage and then for containment performance. The final section addresses the specific finding resulting from this screening.

3.4.2.1 Core Damage Results

The core damage vulnerabilities are examined by presenting the core damage results from several different perspectives: functional sequences, initiating event contribution, dominant systems, and dominant operator actions.

1. Functional Sequences

   Table 3.4-6 presents the dominant functional sequences (i.e., sequences in terms of functional failures rather than system failures) with respect to core damage frequency. These sequences account for about 98% of the core damage total. By looking at the first two functional sequences, the importance of the RCP seal LOCA becomes apparent. More than 69% of the core damage total is due to seal LOCAs resulting from station blackout (35%) or transients with PCC failure (34%). These two functional sequences are discussed in detail below.

   The third sequence type is a general transient with subsequent loss of secondary-side and primary-side cooling due to hardware or operator failures. Secondary-side cooling involves feeding the steam generators via EFW or startup feed pump and bleeding via the atmospheric relief valves. Primary-side cooling involves feeding the vessel via charging pumps or SI pumps and bleeding via the pressurizer PORVs.

The fourth type is an "anticipated transient without scram" (ATWS) with failure of the reactor trip breakers to open or failure of the operator to manually scram or emergency borate within ten minutes. For the "limiting" ATWS, i.e., Loss of Main Feedwater (LOMF), the accident analysis requires a turbine trip within 60 seconds, EFW actuation within 60 seconds, and reactor trip within 10 minutes to assure acceptable results. Thus, for a LOMF with failures of reactor trip breakers and the ATWS Mitigation System, the resulting ATWS without turbine trip is assumed to challenge the reactor vessel integrity, leading to core damage. For less challenging initiators (e.g., turbine trip), it is assumed that the operator has additional time to manually start EFW.

The fifth sequence type is initiated with a LOCA, a primary system leakage greater than the normal makeup can supply (>0.5 in diameter). The dominant functional failure given LOCA is failure of low pressure injection or low pressure recirculation (i.e., RHR).

These five functional sequences account for approximately 98% of the core damage frequency. There are, obviously, many variations in these functional sequences to account for this frequency.

Seal LOCA Sequence

As described above, seal LOCA sequences account for nearly 70% of the core damage frequency. These important functional sequences are described in some detail, as follows:

Based on the best available information, a loss of coolant is expected from the RCP seal package as a result of an extended loss of seal cooling (>30 minutes). Seal cooling is normally provided by charging pump seal injection and backed up by the Thermal Barrier Cooling System. Because of this redundancy, loss of seal cooling due to failure of charging pump and independent failure of the Thermal Barrier Cooling System is highly unlikely. However, both seal cooling methods depend on the Primary Component Cooling (PCC) Water System. Without PCC, the charging pump will overheat and fail in a short time (five to ten minutes), and the Thermal Barrier Cooling System will fail to remove heat from the RCS coolant leaking up the pump shaft. The result

of overheating the seals is a LOCA ranging in size from 20 gpm per pump (if the seals stay intact) to 480 gpm per pump (if the seals disintegrate). The various leak rates between these ranges were assigned probability of occurrence estimates based on expert judgment documented in NUREG-1150 (Reference 18). This information was integrated with the possibility of operator action to depressurize the primary system in order to determine the time to core uncovery. This time is used to estimate the likelihood of recovery prior to core damage.

The most likely seal LOCA sequences from Table 3.4-6 are station blackout and transients with failure of PCC. Figures 3.4-1 and 3.4-2 present event sequence diagrams for these generalized sequences. In the first sequence, a plant upset condition is initiated by disturbances to the plant connection to the off-site grid. This could be caused by the grid itself, outside influences (e.g., seismic events, high winds), or within the plant boundaries (e.g., SF6 faults, fires or floods in the Turbine Building). Following a loss of the 345 kV grid, the emergency AC Power System is needed and can fail due to a variety of causes, internal (e.g., crankshaft failure) or external (Service Water System failure) to the diesel generators. The failure modes of the diesels and off-site power will affect their recoverability. A station blackout would result in unavailability of charging pumps and thermal barrier cooling, leading to a RCP seal LOCA. The operability of EFW (turbine-driven pump) and operator actions to depressurize the RCS and shed DC loads from the vital batteries will extend the time available for recovery but are not sufficient to prevent core damage without recovery of power. If electric power is not restored in the 2 to more than 24 hours available, core damage occurs. The variation in time from 2 hours to more than 24 hours is based on the specific mitigating action taken as well as the uncertainty in seal leakage rate. The differences in time available are accounted for in the evaluation of likelihood of electric power recovery.

Figure 3.4-2 presents the other general seal LOCA sequence. This is initiated by a general plant transient (e.g., reactor trip, turbine trip, loss of feedwater) which is expected to occur several times per year. Subsequent to the trip, PCC fails due to hardware faults within the system or failure of support systems (SW, electric power). Loss of all seal cooling results in a seal LOCA. The time to core uncovery ranges between two hours and more than 24 hours, due to the uncertainty in seal LOCA size and timing and the possible mitigating actions. No credit is taken for recovering PCC so the variation in time does not affect this functional sequence.

2. Major Contributors by Initiating Events

After looking at sequences, additional risk insights can be gained by examining the initiating events, the events that cause a plant upset and begin the accident scenarios. One way of reducing risk is to decrease the frequency of initiating events.

Table 3.4-3 provides a list of initiating events, ranked by their contribution to core damage frequency. Losses of off-site power (LOSP, LSF6), fire-induced transients (FTBLP, FCRCC), and general transients (RT, TT, PLMFW) lead the list. Table 3.4-7 shows a breakdown of initiating events by basic accident types - transient (83%), LOCA (8%), and ATWS (9%), where the values in parenthesis are the percentage of the total core damage frequency attributed to each initiator.

Transient initiators include general transients (e.g., reactor trip, turbine trip, and loss of feedwater), loss of off-site power, and loss of support systems (e.g., loss of one train of PCC). As discussed previously, the most likely core damage sequence for transient initiators involves the RCP seal LOCA. Other transients, such as failure of EFW and feed and bleed cooling, eventually result in a LOCA from the PORV lifting. However, in each case, the LOCA is a result subsequent to the initiator and not the accident initiator itself.

The LOCA category in Table 3.4-7 includes scenarios initiated by breaks in the primary system piping - from small LOCAs up to vessel rupture, steam generator tube rupture (SGTR), and LOCA outside containment (interfacing system LOCAs). The frequencies of these events are estimated from industry experience (for small LOCAs and SGTRs) or from expert opinion where no such initiators have occurred.

ATWS initiators are a subcategory of transients; however, the failure to scram makes these sequences significantly different. ATWS sequences include a severe overpressure transient that is modeled to have some potential to fail the RCS pressure boundary, conservatively modeled as vessel rupture.

Table 3.4-7 also shows the breakdown between internal initiating events (i.e., events internal to the plant systems) and external events (i.e., hazard events - fires, floods, seismic) for each initiator type. External hazards are most

significant to transients and to losses of off-site power and support systems in particular.

External initiators make up almost half of the core damage frequency, most of which are transient events. The sizable contribution of external events illustrates the need to consider hazard events in any attempt to significantly reduce the total frequency of core damage.

Finally, Tables 3.4-8a and 3.4-8b provide a more detailed list of initiating event contributors to core damage, separated into internal and external events, respectively. Fire, seismic, and flooding make up most of the external event risk.

3. Dominant Systems

Table 3.4-9 lists dominant systems based on their importance rank. The importance ranking is based on the percentage of core damage sequences in which a system (or train) has failed due to internal failures, i.e., component failures and maintenance unavailability. This excludes the contribution to system failure from support system failures or from external events (fire, floods). Thus, this ranking is based on the direct effect of the system failure and not dependent failures. Based on this ranking, the dominant systems are support systems -- diesel generators, Primary Component Cooling (PCC), and Service Water (SW). In addition, EFW is an important system, consistent with the importance of transients, as discussed for functional sequences.

These systems listed in Table 3.4-9 are discussed below. See Appendix E for more details.

1) Diesel Generators (DG) - This system is important because of the frequency of a loss of off-site power (expected about once per 20 years) and because of the generic lower reliability of diesels (as compared to other systems). Given loss of off-site power, failure of this system causes loss of all major safety systems except DC power and the turbine-driven EFW pump. Based on the RCP seal LOCA model, loss of coolant is also a result of this system failure. Some credit is given in top event ER to recover the diesels for minor problems that can be fixed in a few hours.

2) Primary Component Cooling (PCC) System - This system provides the cooling function to all safeguards pumps (including seal injection) and

heat exchangers. Loss of the PCC System will fail the thermal barrier system which provides cooling to the RCP seals. Thus, failure of PCC results in loss of all RCP seal cooling with subsequent RCP seal LOCA. No recovery of PCC failures is credited in the plant model.

3) Service Water (SW) System - The SW System supplies cooling water to the PCC System and to the diesel generators when running. SW failures also lead to RCP seal LOCA due to loss of thermal barrier cooling and RCP seal injection. The cooling tower is credited for SW recovery for all loss of SW sequences except LOSP where the diesels were assumed to fail destructively due to loss of SW before the cooling tower can be manually initiated.

4) Emergency Feedwater (EFW) System - The EFW consists of the turbine-driven and motor-driven EFW pumps. The EFW model also includes steam relief through the Atmospheric Relief Valves (ARVs) or Steam Dump Valves (SDVs), as appropriate, and the manual or automatic start of the start-up feed pump depending on the initiating event. The use of alternate means of feedwater, e.g., condensate pumps, was not included in the model.

Successful EFW operation allows plant stabilization and cooldown for scenarios where the RCS remains intact. For accident scenarios where a loss of AC power has occurred, the successful operation of EFW System extends the time available for restoration of electric power.

EFW does not show up as a more dominant system because the system is relatively reliable, and because feed-and-bleed cooling is fully redundant to the normal secondary cooling function.

5) Residual Heat Removal (RHR) System - This system is required to start and run in response to a LOCA initiator to provide low pressure injection and recirculation, or to support high pressure recirculation. This system is important because of the frequency of small LOCA (expected about once in the life of the plant) and the reliability of RHR (two pumps) as compared to HPI (charging pumps and SI pumps - four pumps). No recovery of RHR is modeled; however, operator action to provide makeup to the RWST upon loss of containment sump recirculation extends the time for injection until the recirculation function is restored or the plant reaches cooldown (top event RM).

6) & 7) ESFAS/SSPS - Given failure of the Engineered Safety Features Actuation System (ESFAS) or Solid State Protection System (SSPS), all automatic start signals are lost to ECCS equipment. In addition, for plant transients (except LOSP), an ATWS event will occur on failure of SSPS and manual reactor trip (top event MT).

Loss of signals with a plant transient would cause operator confusion and complicate subsequent recovery actions but does not, by itself, cause failure of equipment. Some credit is given in top event OS for manual actions on loss of signals.

8) Emergency Air Handling (EAH) - EAH provides room cooling to safeguards pumps -- RHR, SI, CBS, and charging pumps. The ventilation system is relatively unimportant because it does not effect support systems and because support system failures, which fail EAH, also fail the safeguards pumps cooled by EAH. In addition, the present model, which assumes failure of pumps within six hours without ventilation, is believed to be very conservative based on environmental qualification reports.

9) Off-Site Grid - This system consists of the major transformers (UAT and RAT), the switchyard, and the transmission line connections to the 345 kV grids. This system ranking includes only failures of the off-site grid subsequent to the plant trip. If the initiating event contributions to loss of off-site power were included, the total system importance is much higher. The plant model includes recovery of off-site power as a function of time available.

10) High Pressure Injection (HPI) - This function consists of two charging pumps and two high head SI pumps. Because of this level of redundancy, this function is very reliable. While HPI would be important in similar sequences as RHR (i.e., small LOCAs), it is less important because of its higher reliability compared to RHR.

In general, systems are important to core damage risk, not because they are unreliable, but because of their function. System importance is a combination of the following:

- The system's function in preventing core damage;

- Whether it has functional redundancy in another system; and
- The frequency at which the system is demanded.

Within this list of important systems, the specific ranking is a balance between system reliability and frequency that the system is demanded. Thus, PCC (frequently demanded, high reliability) ranks just below diesel generators (less frequently demanded, lower reliability).

4. Dominant Operator Actions

In addition to hardware failures, operator actions analysis is useful in understanding the risk insights. The SSPSS quantified the impact of a number of normal, emergency, and recovery actions. (See Section 3.3). Routine actions, such as test and maintenance activities, are included in the system's analysis to the extent that they contribute to system unavailability. Important operator actions in response to a plant upset condition are discussed below.

Table 3.4-10 lists the dominant operator action failures ranked by their contribution to core damage sequences (i.e., risk importance). By far the dominant action is electrical power recovery (ER), consistent with the importance of station blackout sequences. This action includes recovery of off-site power or diesels and is a function of the time available to core uncovery. In addition to ER, signal recovery (No. 2) and EFW recovery (No. 3) are important. These recoveries include the likelihood of equipment operating or being repaired but are, in general, dominated by human errors. Operator actions in response to ATWS (No. 4) and SGTR (No. 5) are important manual actions involving diagnosing the initiator, as well as responding accordingly. Feed-and-bleed failure (No. 6) is the action to establish primary heat removal when secondary heat removal (EFW) has failed. Operator actions to provide makeup to the RWST (No. 7) upon loss of containment sump recirculation extends the time for injection until the recirculation function is restored or the plant reaches cooldown. Operator action to switchover to high pressure recirculation (No. 8) is of relatively low importance because of low demand frequencies (i.e., only small LOCA with no depressurization). Depressurization for station blackout sequences (No. 9) allows additional time for recovery but is not sufficient to result in a successful sequence. The final operator action, controlling EFW (No. 10) is modeled in the general transient tree to prevent overfilling the steam generators with potential for overcooling.

Table 3.4-10 also provides the Risk Achievement Worth for each important operator action top event in order to examine the sensitivity of the quantification. The one operator action that stands out for this risk measure compared to risk importance is Top Event 03 - switchover to high pressure recirculation. While this event contributes to less than 1% of the core damage frequency total, because it is quantified as a highly reliable action (mean value = 8.0E-4), it has a relatively high Risk Assessment Worth. This action, which would occur six to eight hours after initiation of a small LOCA, is expected to be highly reliable because of the time available and the alarms and procedural guidance.

Based on this ranking, short-term operator actions are, in general, less important than hardware failures due to the automated design of the safety features. Most of the important actions are in response to hardware failures -- repair/restoration actions (electric power recovery) and manual start (signal recovery, EFW recovery).

3.4.2.2 Containment Performance Results

As discussed in Section 3.4.1.2, containment performance has been defined in terms of containment failure modes and release categories. Table 3.4-5 defines the release categories and also presents the quantitative results of the containment performance evaluation. As explained previously, the focus of this evaluation, the "unusually poor" containment performance, is the early, large containment failure/bypass class. This class has a mean annual frequency of about 2.3E-7, or about 0.2% of core damage total. The release categories failure modes in this class are discussed below, in order of their contribution to large, early containment failure/bypass.

- Large Containment Isolation Valve Failure - 54.2% - Release Category S6

  This containment bypass is through the 8-inch containment on-line purge (COP) line, which communicates directly with the environment (through filters). This line, which has two air-operated isolation valves, is used during operation to purge containment for habitability and pressure considerations. The model assumes the line is used 10% of the time. If the valves are open, they are designed to close given a containment isolation ("T") signal or loss of instrument air. The dominant scenarios to this failure mode (see Table 3.4-2)

are failures of signals due to hardware or calibration problems (top events SA, SB, EA, and EB) and subsequent failure of operator to recover this signal failure and operate equipment manually (top event OS). These failures are modeled to result in core damage and, if the purge valves are open, containment isolation failure occurs due to loss of signals.

The values used for OS (see Table 3.3-5) are conservative screening values. Also, a more realistic model would identify two different actions and time frames - core damage mitigation in the first few hours and containment isolation for a number of additional hours. Thus, it is believed that this contribution is over-estimated based on a conservative operator model. This will be addressed in the future if needed.

- Induced Steam Generator Tube Rupture (ISGTR) - 26.8% - Release Category S7A

ISGTR is a containment bypass which occurs following a high pressure core melt with dry steam generators. If the steam generator tubes fail due to high temperature with the RCS intact, the release is out of the steam lines to the environment. The present results take no credit for operators depressurizing the RCS during high pressure core melts or feeding the dry steam generators with fire water. These procedural and hardware enhancements are addressed in Section 6.2.

The types of sequences important to ISGTR are transients with failure of EFW and failure of feed and bleed cooling; for example, a transient initiated by failure of one train of DC power, failure of the turbine driven EFW pump to start, and failure of the operator to manually start the other EFW pumps. Feed and bleed cooling fails due to unavailability of one PORV due to loss of DC bus (with modeling assumption that two PORVs are needed). Another important sequence is station blackout with failure of the turbine-driven EFW pump.

- Direct Containment Heating (DCH) - 11.1% - Release Category S1A

DCH is a potential failure mode applicable to high pressure core melt sequences. This failure mode is due to the postulated event of rapid heat transfer to the containment resulting from high pressure ejection of finely

dispersed core debris into and out of the reactor cavity. This pressure load could combine with other pressure loads, such as RCS depressurization and hydrogen burns, to challenge the containment structural integrity.

The dominant sequences are the dominant transient sequences contributing to core damage frequency - station blackout and transients with PCC failure. The present results take no credit for operators depressurizing the RCS.

- Large, Pre-Existing Leaks - 4.5% - Release Category S6

  This is a large (greater than three inch diameter) penetration open in containment with no indication or automatic isolation. The probability of a large pre-existing leak is based on data review and is added to all top event C2 split fractions.

- V-Sequence, Pipe Failure - 2.1% - Release Category S7A

  This is a containment bypass through failed open RHR suction or injection valves with release to the enclosure building out the rupture of the RHR piping. This is the "classic" V-sequence, but based on analysis in PLG-0432 (Reference 11) this is also relatively unlikely because of the strength of the RHR piping. The more likely sequence is failure of the RHR pump seals which results in the release point being covered with 30 feet of water, and thus, a much reduced source term (release category S7B).

- Steam/Hydrogen Explosion - 0.8% - Release Category S1A

  The pressure loads due to steam explosion and/or hydrogen explosion at the time of vessel rupture have a small likelihood of challenging containment structural integrity.

- External Impact on Containment (Aircraft Crash, Turbine Missile Impact) - 0.5% - Release Category S1B

  These failure modes are very low frequency based on the strength of the primary and secondary containments and the low likelihood of the hazard.

The containment performance results can also be examined in terms of initiating event contribution (Table 3.4-11) and top event important (Table 3.4-12). These results are consistent with the results by failure mode discussed above. Thus, general transients are

the dominant initiating event type contributing to early, large containment failure/bypass. These initiators contribute primarily to the release category S6, large containment isolation valve failure. The results of this release category are dominated by the conservative assumption regarding operator action OS, as discussed above. Table 3.4-12 shows quantitatively the importance of top events OS, C2, EA/EB, and SA/SB - all related to the sequences in release category S6. Induced steam generator tube rupture shows up in Table 3.4-12 in top event IS as well as in top events EF (dry steam generators) and HL (no hot leg rupture, thus high RCS pressure is maintained). Top event CN, containment structural failure, is due primarily to direct containment heating, release category S1A. Finally, top events ER and GA/GB are important contributors to high pressure core melt sequences, which is a necessary condition for ISGTR and DCH.

3.4.2.3 Vulnerability Findings

The basic finding of the extensive evaluations summarized in this report is that there are no fundamental weaknesses or vulnerabilities with regard to severe accidents at Seabrook Station. The basis for this statement, including the definition of vulnerabilities, is presented below. The term vulnerabilities, as used in this report, refers to those components, systems, operator actions, and/or plant design configurations that contribute significantly to an unacceptably high severe accident risk. To further clarify this definition, the term, "contribute significantly" is used to mean - responsible for a majority (i.e., more than 50 percent) of the total frequency for a given risk measure. Thus, if this plant feature could be made perfect, the risk would be significantly decreased, by at least a factor of two. This magnitude reduction is needed to be significant because of the level of uncertainty, which is generally at least a factor of three between the median and the extremes (5th or 95th percentile). Contributing significantly is not sufficient to define a vulnerability, since it is possible to have a dominant contributor to a very low risk. The term "unacceptably high severe accident risk" must also be included in the definition. For use in this report, this term has been defined, for the two risk measures of interest, as follows:

- core damage - mean frequency substantially exceeding 1.0E-4 per year, and
- containment performance - mean frequency of large, early release substantially exceeding 1.0E-6 per year,

where "substantially exceeding" means a factor of two or more.

These values have been chosen based in part on their appearance in various draft or proposed safety goals. In addition, the value of 1.0E-4 per year represents a high confidence (95%) of low probability (0.01) of a significant financial risk (i.e., a core damage accident) over the life of the plant. (This assumes: a 40-year plant life times 2.5E-4 per year, the estimated 95th percentile, equals 0.01). The value of 1.0E-6 per year represents a low conditional probability (0.01) of a substantial, early release, given a core melt. Thus, while these values do not define "unacceptable risk" in absolute terms, there is some objective basis for these definitions.

Using this definition of vulnerabilities and the results from the previous sections, the finding of no vulnerabilities can be explained. For core damage risk, the mean frequency (1.1E-4/yr) does not substantially exceed 1.0E-4/yr. This statement can be made rigorously because of the high level of completeness in the risk analysis. Careful attention was given in the SSPSA to identify and address every potential plant initiator, including all conceivable external hazard events. For containment performance, the mean frequency of large, early releases (2.3E-7 per year) is substantially below 1.0E-6/yr. This statement is also rigorous based on the extensive investigation of potential early containment failure/bypass mechanisms. Thus, no vulnerabilities exist based on the low level of risk.

For both risk measures, risk enhancement opportunities exist, i.e., limiting plant features which, if improved, would significantly reduce the already low risk. For core damage risk, the reactor coolant pump seal LOCA accounts for almost 70% of the frequency. As discussed in Section 3.4.2.1, this issue is due to (1) the basic design of the Westinghouse RCP seal package, (2) the plant design configuration that supplies seal cooling from systems that are dependent on Primary Component Cooling and power, and (3) the uncertainty in the size and timing of the loss of coolant. An additional plant feature, the SF_6 switchyard and transmission lines (included as initiator LSF6), accounts for about six percent of the total core damage frequency. While this is not significant, this may understate the importance of this system because of its long time to repair (> one week) versus other plant equipment (typically one day). For containment performance, after the conservative treatment of signal failure, the dominant plant feature is high RCS pressure at time of core melt, which makes possible direct containment heating and induced steam generator tube rupture. Potential plant enhancements to address these risk enhancement opportunities are addressed in Section 6.0.

3.4.3 Decay Heat Removal Evaluation

This section provides an evaluation of the decay heat removal capability at Seabrook Station to support resolution of Unresolved Safety Issue A-45, "Shutdown Decay Heat Removal Requirements." The objective of USI A-45 is to determine the adequacy of the decay heat removal function at operating plants and to identify any cost-beneficial improvements. The USI is limited to considerations of the decay heat removal function needed immediately after a trip from power operation. The results presented below are generally based on a 24-hour mission time after trip. Additional considerations of decay heat removal for shutdown cooling are addressed in the Shutdown Study (Reference 28) summarized in Appendix B.

The decay heat removal function is accomplished by the following systems:

- For Loss-Of-Coolant Accidents (LOCAs), decay heat is transferred from the core to the containment using high or low pressure safety injection systems (i.e., charging pumps, SI pumps, and RHR pumps). During recirculation, decay heat is removed from the core and containment via the RHR System or CBS System. Each system has two pumps which take suction from the recirculation sumps and cool the water via heat exchangers, two for each system. The heat exchangers transfer heat to the PCC System and then to the Service Water System to either the ocean (through the intake/discharge tunnels) or to the atmosphere (through the cooling tower).

- For transients and small break LOCAs, the Emergency Feedwater (EFW) or Main Feedwater (MFW) Systems provide decay heat removal through the steam generators to the atmosphere (through atmospheric relief valves) or to the condenser (through steam dump valves) and finally to the ocean. In the event that EFW and MFW fail, feed and bleed cooling (consisting of the high pressure charging pumps and SI pumps and the PORVs), together with the recirculation function (discussed above) are utilized for successful decay heat removal.

The following special features of these systems enhance the reliability of the decay heat removal function, as follows:

1. The EFW System at Seabrook has two 100% capacity redundant trains. Both trains (one motor driven, one turbine driven) feed all four Steam Generators (SGs).

2. The Feedwater System also contains a diverse motor-driven start-up feed pump located in a different building from the EFW pumps. This pump is normally powered from a nonsafety grade supply but can be easily aligned to the safety grade emergency bus as called for in plant procedures. In addition, this pump can be aligned to take suction from the safety grade Condensate Storage Tank (CST), as appropriate.

3. The emergency operating procedures (Functional Restoration Procedure FR-H.1) include explicit instructions to restore EFW, to align the startup feed pump, or to establish feed flow from the condensate system. No credit has been taken in the present model for the condensate system.

4. Feed and bleed cooling is proceduralized (Function Restoration Procedure FR-H.1). Also, recent analyses show that the success criteria in the existing SSPSS is quite conservative. For instance, the existing model requires the operator to open two out of two PORVs for successful feed and bleed cooling. However, recent analyses (Reference 29) show a variety of PORV (one or two of two) and SI pumps (one of four, two high head and two medium head SI pumps) combinations can maintain feed and bleed decay heat removal with and without operator action. This analysis has not yet been incorporated into the Seabrook models.

5. For large LOCA sequences with failure of the RHR heat exchangers during recirculation, decay heat removal can be accomplished via the Containment Building Spray (CBS) heat exchangers.

6. For long-term operation, plant procedures call for monitoring of CST level and, as necessary, to provide makeup via the normal demineralized water source or, as necessary, via the firewater water system which has its own dedicated water supply and diesel-driven fire pump.

7. For losses of the RCS recirculation function, plant procedures direct the operator to provide makeup to the RWST. This allows plant stabilization while the recirculation function is restored or the plant is cooled down.

8. The steam generators have a large inventory of secondary coolant which translates into more than 60 minutes of decay heat removal via boiloff. This provides significant time for operator action to manually start or align

equipment or to perform simple recoveries. This study models operator action to manually start and align the startup feed pump and to recover the turbine-driven EFW pump in the event of the overspeed trip.

Due to the configuration of EFW, MFW, the feed and bleed cooling, and ECCS recirculation, as well as the special features listed above, decay heat removal does not represent a significant risk potential at Seabrook. As can be seen in Tables 3.4-9 (System Importance) and 3.4-10 (Operator Action Importance), the contributions to core damage for systems (EFW, RHR) and human errors related to the Decay Heat Removal (DHR) function are small. The DHR-related systems rank fourth and fifth, EFW (15%) and RHR (4%), respectively. For important human errors, the recovery of EFW (FR - 4%), feed and bleed (OR - 1.6%), makeup to the RWST (RM - 1.0%), and switchover to recirculation (O3 - 0.6%) appear, but both on an absolute and relative basis, do not contribute significantly to core damage results.

Appendix 5 of Generic Letter 88-20 (DHR Insights) and NUREG/CR-5230 (Reference 30) were reviewed and the Seabrook-specific application of these insights is discussed as follows:

Redundancy - Secondary heat removal accomplished by two diverse self-cooled safety grade trains (EFW), a frequently used and spatially diverse, self-cooled, start-up feed pump with separate safety grade power supply. In addition, feed and bleed cooling is available via four safety grade and spatially independent pumps.

Separation - As stated above, the EFW and startup feed pumps are spatially independent and each of the four feed and bleed pumps are spatially independent. Also, their support systems are spatially independent as a minimum by train.

Physical Protection - Seabrook Station is a recent vintage plant and, as such, issues such as physical protection (i.e., separation, ruggedness, interaction, etc.) have been addressed in the design phase. Results of the PSA highlight the robustness of the Seabrook design.

In summary, this examination shows that the likelihood and nature of DHR failure-related events are well understood and represent a negligible contribution to risk at Seabrook.

3.4.4 USI and GSI Evaluation

This report addresses two unresolved safety issues:

- A17, "System Iterations," was resolved in Generic Letter 89-18 by referring the "internal flooding" issue to IPE. This issue has been addressed in Section 3.3.8 and it was concluded that no significant vulnerabilities exist in that area.

- A45, "Decay Heat Removal," is addressed in the previous section. The conclusion from that evaluation is that there are no significant vulnerabilities in this area.

No other issues have been identified that are resolved via this IPE Report. However, it is expected that the risk analysis for Seabrook Station will be used in the future to directly resolve or to help prioritize generic issues. Specifically, Generic Issue B-23, Reactor Coolant Pump Seal Integrity is being evaluated as part of NHY's risk management program. In addition, Supplement 4 to Generic Letter 88-20, which is scheduled to be issued in early 1991, may identify other issues that can be resolved in the IPE process.

TABLE 3.4-1

Dominant Core Damage Sequences

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
*	1.	FCRCC	6.7936E-06	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	6.1
*	2.	FTBLP	6.0884E-06	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5	5.4
	3.	LSF6	4.4351E-06	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5	4.0
	4.	LOSP	3.6623E-06	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER1	3.3
*	5.	FLLP	3.4154E-06	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5	3.1
	6.	RT	3.3592E-06	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	3.0
*	7.	FPCC	2.9521E-06	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	2.6
	8.	PLMFW	2.8122E-06	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	2.5
	9.	TT	2.6621E-06	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	2.5
	10.	L1CCB	1.7747E-06	//PA4*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	1.6
	11.	L1CCA	1.7747E-06	//PAF*PB4*EHF/NLF/SUF*LTF*XAF*XBF/	1.6
*	12.	FSRCC	1.6653E-06	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	1.5
	13.	LOSP	1.6279E-06	//OGF*PA2*PBB*EHF/NLF/SUF*LTF*XAF*XBF/	1.5
*	14.	E10T	1.6252E-06	/QSF*QYA*QKA/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*CAF *CBF/SUF*LTF*WSF*CI4	1.5
*	15.	FCRSW	1.5936E-06	//WAF*WBF*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	1.4
*	16.	FCRAC	1.5786E-06	//OGF*DAF*DBF*GAF*GBF*WAF*WBF*SAF*SBF*EAF*EBF*OSF*PAF *PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CIP	1.4
	17.	LOSP	1.4762E-06	//OGF*WA3*WBC*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER9	1.3

* External Initiating Event

TABLE 3.4-1
(Continued)

Dominant Core Damage Sequences

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
	18.	LDCB	1.4365E-06	//DBF*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/	1.3
*	19.	FET1	1.3303E-06	//WAF*WBD*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	1.2
	20.	ALOMF	1.1118E-06	//RT1/PL1*MFF*AM1*TTF/SUF*LTF	1.0
*	21.	FLSW	1.0434E-06	//WAF*WBF*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	0.9
	22.	SLOCA	1.0286E-06	///L13*L2C/SUF*LTF/RMU	0.9
	23.	SLOCA	1.0051E-06	//EH1/LRF/SUF*L5F*L6F*XCF*XDF/RMF	0.9
	24.	LDCB	1.0038E-06	//DBF*EBF/EFD*FR4*ORF*CBF/SUF*LTF/	0.9
	25.	TCTL	9.4048E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5	0.8
	26.	LLOCA	8.3217E-07	///LA2*LBA/LCF*LDF	0.7
*	27.	FTBLP	7.7668E-07	//OGF*WA3*WBC*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ERF	0.7
*	28.	E7T	7.7558E-07	/QSF*QY7*QD7/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.7
	29.	RT	7.3204E-07	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/	0.7
*	30.	E7T	7.2936E-07	/QSF*QY7*QK7/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.7
	31.	RT	6.9629E-07	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/	0.6
*	32.	E2T	6.9198E-07	/QSF*QY2/OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.6
*	33.	E10T	6.7692E-07	/QSF*QYA*QKA*QRA/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*RWF/SUF*LTF*WSF*CI4	0.6
*	34.	FET3	6.6514E-07	//WAF*WBD*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF	0.6

* External Initiating Event

TABLE 3.4-1
(Continued)

Dominant Core Damage Sequences

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
* 35.	E7AT	6.0405E-07	/QSF*QY7/OGF*RTF/PL1*MFF*MRF*PRF/SUF*LTF	0.5
36.	LOSP	6.0375E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FR5*H2F*CAF*CBF/SUF*LTF*WSF/ER2	0.5
37.	TT	5.8013E-07	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/	0.5
38.	ALOMF	5.7728E-07	//RT1/PL1*MFF*OH1/SUF*LTF	0.5
39.	LSF6	5.6578E-07	//OGF*WA3*WBC*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ERF	0.5
40.	LDCA	5.6371E-07	//DAF*EAF/EFA*FR2*ORF*CAF/SUF*LTF*XAF/	0.5
* 41.	FTBLP	5.5974E-07	//OGF*GB1*WAC*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.5
* 42.	FTBLP	5.5974E-07	//OGF*GA1*WAF*WBG*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.5
43.	TT	5.5179E-07	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/	0.5
44.	SGTR	5.2185E-07	///EFA*O42/SUF*LTF*WSF	0.5
45.	ATT	5.2028E-07	//RT1/PL1*EFM/SUF*LTF	0.5
46.	RT	5.0644E-07	//WA5*WBJ*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	0.5
47.	ATT	4.9150E-07	//RT1/PL1*OH1/SUF*LTF	0.4
48.	LOSP	4.8483E-07	//OGF*GB1*WAC*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER3	0.4
49.	LOSP	4.8483E-07	//OGF*GA1*WAF*WBG*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER3	0.4
50.	LOSP	4.8012E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*NLF*CAF*CBF/SUF*LTF*WSF/ER2	0.4
51.	LOPF	4.3788E-07	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	0.4
* 52.	FLLP	4.3570E-07	//OGF*WA3*WBC*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ERF	0.4
53.	PLMFW	4.2396E-07	//WA5*WBJ*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	0.4

* External Initiating Event

TABLE 3.4-1
(Continued)

Dominant Core Damage Sequences

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
	54.	EXFW	4.1798E-07	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	0.4
	55.	LSF6	4.0774E-07	//OGF*GA1*WAF*WBG*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.4
	56.	LSF6	4.0774E-07	//OGF*GB1*WAC*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.4
*	57.	E10L	4.0683E-07	/QSF*QYA*QKA/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/LAF*LBF*CAF*CBF/WSF*CI4	0.4
	58.	TLMFW	4.0316E-07	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	0.4
	59.	TT	4.0134E-07	//WA5*WBJ*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	0.4
*	60.	E10T	3.9871E-07	/QSF*QYA*QKA*QDA/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.4
*	61.	E3T	3.9013E-07	/QSF*QY3/OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.3
*	62.	FTBLP	3.8427E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6	0.3
*	63.	FL1SG	3.6495E-07	//OGF*DAF*GAF*GB2*WAF*WBF*SAF*EAF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF	0.3
	64.	AGT	3.4128E-07	//RT1/PL1*MFF*AM1*TTF/SUF*LTF	0.3
	65.	SGTR	3.3732E-07	///EFA*O51/SUF*LTF	0.3
	66.	LCV	3.2904E-07	///EFB*FR1*OR4/SUF*LTF/	0.3
	67.	SLBI	3.2605E-07	////SUF*O31	0.3
*	68.	E7AT	3.1455E-07	/QSF*QY7/OGF*RTF/PRF/SUF*LTF	0.3
*	69.	FLLP	3.1400E-07	//OGF*GA1*WAF*WBG*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.3
*	70.	FLLP	3.1400E-07	//OGF*GB1*WAC*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER7	0.3
	71.	RT	2.9824E-07	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*CIM/	0.3

* External Initiating Event

TABLE 3.4-1
(Continued)

Dominant Core Damage Sequences

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
72.	LCV	2.9355E-07	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	0.3
73.	ALOMF	2.9350E-07	//RT1/OH1/SUF*LTF	0.3
74.	LSF6	2.7992E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6	0.3
75.	SLBI	2.6010E-07	///L14*L2C/SUF*LTF	0.2
76.	ATT	2.5626E-07	//RT1/EFM/SUF*LTF	0.2
* 77.	E5T	2.5247E-07	/QSF*QY5*QK5/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.2
78.	SLBO	2.5060E-07	//EB3*OS2/EFD*FRF*ORF*CBF/SUF*LTF*ZBF	0.2
79.	MLOCA	2.4959E-07	///L12*L2A/SUF*LTF	0.2
80.	ELOCA	2.4480E-07	///LAF*LBF/LCF*LDF	0.2
* 81.	E10AT	2.4284E-07	/QSF*QYA*QKA/OGF*GAF*GBF*WAF*WBF*RTF*PAF*PBF*EHF/PL1*MFF*MRF*H3F/SUF*LTF*ZAF*ZBF*CI4	0.2
82.	ATT	2.4208E-07	//RT1/OH1/SUF*LTF	0.2
* 83.	FSRAC	2.3867E-07	//OGF*DAF*DBF*GAF*GBF*WAF*WBF*SAF*SBF*EAF*EBF*OSF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CIP	0.2
84.	TT	2.3635E-07	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*CIM/	0.2
* 85.	E14T	2.1844E-07	/QSF*QYB*QKB*QRB/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*RWF/SUF*LTF*WSF*CI4	0.2
* 86.	FLLP	2.1556E-07	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6	0.2
87.	MSIV	2.1546E-07	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF/	0.2

* External Initiating Event

TABLE 3.4-1
(Continued)

Dominant Core Damage Sequences

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent CDF
* 88.	E4T	2.1285E-07	/QSF*QY4/OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CI4	0.2
89.	AGT	1.9421E-07	//RT1/PL1*MFF*EFN/SUF*LTF	0.2
90.	SLOCA	1.9251E-07	//PB1/L14*L2F/SUF*LTF/RMF	0.2
91.	SLOCA	1.9250E-07	//PA1/L1F*L24/SUF*LTF*XAF/RMF	0.2
92.	LOSP	1.9122E-07	//OGF*DB1*GBF*WBF*SBF*EBF*PBF/EFD*FR5*ORF*CBF/SUF*LTF*ZBF/ERA	0.2
* 93.	E14T	1.8459E-07	/QSF*QYB*QKB*QDB*QRB/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/NLF*RWF/SUF*LTF*WSF*CI4	0.2
* 94.	FL2SG	1.7853E-07	//OGF*DAF*DBF*GAF*GBF*WAF*WBF*SAF*SBF*EAF*EBF*OSF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*CIP	0.2
95.	AGT	1.7620E-07	//RT1/PL1*MFF*OH1/SUF*LTF	0.2
96.	MSIV	1.7413E-07	///EFA*FR1*OR4/SUF*LTF/	0.2
* 97.	E10L	1.6945E-07	/QSF*QYA*QKA*QRA/OGF*GAF*GBF*WAF*WBF*PAF*PBF*EHF/RWF/WSF*CI4	0.2
98.	L1CCA	1.6637E-07	//WB5*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	0.1
99.	L1CCB	1.6636E-07	//WA5*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/	0.1
* 100.	E3T	1.6624E-07	/QSF*QY3/OGF*GB1*WBF*PBF/EFD*FRF*H2F*CBF/SUF*LTF*ZBF	0.1
Others		2.46E-05		22.0
TOTAL		1.12E-04		100.0%

* External Initiating Event

TABLE 3.4-2

Dominant Containment Performance Sequences - Large, Early Containment Failure/Bypass

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
1.	RT	3.3175E-08	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	14.8	S6
2.	TT	2.6291E-08	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	11.7	S6
3.	LDCB	6.0332E-09	//DBF*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	2.7	S7A
4.	RT	5.4869E-09	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	2.4	S6
5.	TT	4.3483E-09	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	1.9	S6
6.	LOPF	4.3244E-09	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	1.9	S6
7.	SLBO	4.2444E-09	//EAC*EBE*OS2/MSF*EFF*FRF*ORF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	1.9	S6
8.	LDCB	4.2160E-09	//DBF*EBF/EFD*FR4*ORF*CBF/SUF*LTF//SCF*R1F*VDF*DPF*HL1*IS1	1.9	S7A
9.	EXFW	4.1279E-09	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	1.8	S6
10.	RT	3.0746E-09	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	1.4	S7A
11.	RT	2.9244E-09	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	1.3	S7A
12.	LCV	2.8993E-09	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	1.3	S6
13.	LOSP	2.5358E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FR5*H2F*CAF*CBF/SUF*LTF*WSF/ER2/SCF*R1F*VDF*DPF*HL1*IS1	1.1	S7A
14.	TT	2.4365E-09	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	1.1	S7A

* External Initiating Event

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
	15.	VI	2.4031E-09	///LE1*PI1/SCF	1.1	S7A
	16.	LDCA	2.3676E-09	//DAF*EAF/EFA*FR2*ORF*CAF/SUF*LTF*XAF//SCF*R1F*VDF*DPF*HL1*IS1	1.1	S7A
	17.	TT	2.3175E-09	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	1.0	S7A
	18	ATT	2.1852E-09	//RT1/PL1*EFM/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	1.0	S7A
	19	MSIV	2.1278E-09	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.9	S6
*	20	FCRCC	2.0354E-09	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.9	S1A
	21	SGTR	1.9489E-09	//EAB*EBC*OS1/EFF*H2F/SUF*LTF*WSF*C2M/SCF	0.8	S6
	22	RT	1.8623E-09	//SA6*EAF*EBK*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.8	S6
	23	RT	1.8618E-09	//SB6*EAE*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.8	S6
	24	SLBI	1.8198E-09	//SAE*SBJ*EAF*EBF*OS2/MSF*EFF*ORF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	0.8	S6
*	25	FTBLP	1.7900E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5/SCF*R1F*VDF*DPF*HL1*CC2*CHF*CYF*CND*SMF	0.8	S1A
	26	MLOCA	1.7857E-09	//SAA*SBA*EAF*EBF*OS2/H1F*EFF/SUF*LTF*ZAF*ZBF*C2M/SCF	0.8	S6
	27	VS	1.7409E-09	///LE1*PI1/SCF	0.8	S7A
*	28	FTBLP	1.6139E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6/SCF*R1F*VDF*DPF*HL1*IS1	0.7	S7A

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
* 29	E1T	1.5210E-09	/QSF/EAE*EBJ*OS2/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C25/SCF	0.7	S6
30	TT	1.4759E-09	//SA6*EAF*EBK*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.7	S6
31	TT	1.4755E-09	//SB6*EAE*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.7	S6
32	SGTR	1.4167E-09	///EFA*O51/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	0.6	S7A
33	LCV	1.3820E-09	///EFB*FR1*OR4/SUF*LTF//SCF*R1F*VDF*DPF*HL1*IS1	0.6	S7A
34	LSF6	1.3039E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5/SCF*R1F*VDF*DPF*HL1*CC2*CHF*CYF*CND*SMF	0.6	S1A
35	SLOCA	1.2397E-09	//EAB*EBC*OS1/EFF*H2F*CAF*CBF/SUF*LTF*WSF*C2M/RMF/SCF	0.5	S6
36	LSF6	1.1757E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6/SCF*R1F*VDF*DPF*HL1*IS1	0.5	S7A
37	LOSP	1.0767E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER1/SCF*R1F*VDF*DPF*HL1*CC2*CHF*CYF*CND*SMF	0.5	S1A
38	ATT	1.0763E-09	//RT1/EFM/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	0.5	S7A
39	SLBO	1.0525E-09	//EB3*OS2/EFD*FRF*ORF*CBF/SUF*LTF*ZBF/SCF*R1F*VDF*DPF*HL1*IS1	0.5	S7A
40	RT	1.0065E-09	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.5	S1A
* 41	FLLP	1.0041E-09	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER5/SCF*R1F*VDF*DPF*HL1*CC2*CHF*CYF*CND*SMF	0.4	S1A

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
*	42	FLLP	9.0537E-10	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/EFD*FRF*H2F*CAF*CBF/SUF*LTF*WSF/ER6/SCF*R1F*VDF*DPF*HL1*IS1	0.4	S7A
*	43	FPCC	8.8448E-10	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.4	S1A
	44	PLMFW	8.4256E-10	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.4	S1A
	45	AGT	8.1567E-10	//RT1/PL1*MFF*EFN/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	0.4	S7A
	46	LOSP	8.0313E-10	//OGF*DB1*GBF*WBF*SBF*EBF*PBF/EFD*FR5*ORF*CBF/SUF*LTF*ZBF/ERA/SCF*R1F*VDF*DPF*HL1*IS1	0.4	S7A
	47	LOSP	8.0257E-10	//OGF*EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.4	S6
	48	TT	7.9760E-10	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.4	S1A
	49	LLOCA	7.9458E-10	//SAA*SBA*EAF*EBF*OS2/LAF*LBF*CAF*CBF/WSF*C2M/SCF	0.4	S6
*	50	FCRCC	7.4803E-10	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF*C2A/SCF	0.3	S6
	51	SI	7.3555E-10	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.3	S6
	52	MSIV	7.3133E-10	///EFA*FR1*OR4/SUF*LTF//SCF*R1F*VDF*DPF*HL1*IS1	0.3	S7A
	53	LOPF	7.1523E-10	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.3	S6
*	54	E3T	6.9819E-10	/QSF*QY3/OGF*GB1*WBF*PBF/EFD*FRF*H2F*CBF/SUF*LTF*ZBF/SCF*R1F*VDF*DPF*HL1*IS1	0.3	S7A

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
	55	LOSP	6.9611E-10	//OGF*WA3*WBC*PAF*PBF*EHF/EFD*FR5*H2F*CAF*CBF/SUF*LTF*WSF/ERA/SCF*R1F*VDF*DPF*HL1*IS1	0.3	S7A
	56	EXFW	6.8272E-10	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.3	S6
	57	SLBO	6.6142E-10	//SA6*EAF*EBG*OS2/MSF*EFF*FRF*ORF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	0.3	S6
	58	SLBO	6.6123E-10	//SB4*EAC*EBF*OS2/MSF*EFF*FRF*ORF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	0.3	S6
	59	CPEXC	6.5849E-10	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.3	S6
	60	LOSP	6.1171E-10	//OGF*GB1*WBF*PBF/EFD*FR5*OR4*CBF/SUF*LTF*ZBF/ER4/SCF*R1F*VDF*DPF*HL1*IS1	0.3	S7A
*	61	FTBLP	6.0208E-10	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*C2D/ER5/SCF	0.3	S6
	62	LDCB	5.8834E-10	//DBF*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF*C2E//SCF	0.3	S6
	63	LDCB	5.3877E-10	//DA2*DBF*EBF*OSF/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	64	L1CCA	5.3172E-10	//PAF*PB4*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A
	65	L1CCB	5.3172E-10	//PA4*PBF*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A
	66	SLOCA	5.2738E-10	///EFA*OR4/SUF*LTF/RMF/SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
*	67	FSRCC	4.9894E-10	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
	68	LOSP	4.8773E-10	//OGF*PA2*PBB*EHF/NLF/SUF*LTF*XAF*XBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A
	69	LCV	4.7953E-10	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.2	S6
*	70	FCRSW	4.7747E-10	//WAF*WBF*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A
	71	AMSIV	4.7421E-10	//EAE*EBJ*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.2	S6
	72	LSF6	4.3859E-10	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*C2D/ER5/SCF	0.2	S6
	73	LOSP	4.3399E-10	//OGF*WA3*WBC*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF/ER9/SCF*R1F*VDF*DPF*HL1*CC2*CHF*CYF*CND*SMF	0.2	S1A
	74	LDCB	4.2579E-10	//DBF*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNG*SMF	0.2	S1A
	75	AMFW	4.1457E-10	//RT1/PL1*EFM/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	76	LOPF	4.0078E-10	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
*	77	FET1	3.9857E-10	//WAF*WBD*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.2	S1A
	78	AGT	3.9832E-10	//RT1/EFM/SUF*LTF/SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	79	LOSP	3.9229E-10	//OGF*GB1*WBF*PBF/EFD*FR5*L14*L2F/SUF*LTF*ZBF/ER4/SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	80	LOSP	3.8302E-10	//OGF/EFB*FR1*OR4/SUF*LTF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

	CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
	81	EXFW	3.8256E-10	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	82	LOPF	3.8120E-10	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
*	83	E4T	3.8091E-10	/QSF*QY4/OGF*GB1*WBF*PBF/EFD*FRF*H2F*CBF/SUF*LTF*ZBF/SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	84	RT	3.6987E-10	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF*C2A//SCF	0.2	S6
	85	EXFW	3.6388E-10	//EB5*OS1/EFD*FRF*ORF*CBF/SUF*LTF*ZBF//SCF*R1F*VDF*DPF*HL1*IS1	0.2	S7A
	86	LOSP	3.6216E-10	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*C2D/ER1/SCF	0.2	S6
	87	SLBI	3.5582E-10	//EAD*EBH*OS2/MSF*EFF*ORF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	0.2	S6
	88	ALOMF	3.5398E-10	//SA6*SBL*MT1*RTF*EAF*EBF*OS2/PL1*MFF*AMF*TTF*CAF*CBF/SUF*LTF*WSF*C2M/SCF	0.2	S6
	89	MSIV	3.5193E-10	//SA6*SBL*EAF*EBF*OS1/EFF*FRF*H2F*CAF*CBF/SUF*LTF*WSF*C2M//SCF	0.2	S6
	90	MLOCA	3.4353E-10	//EAA*EBA*OS2/H1F*EFF/SUF*LTF*ZAF*ZBF*C2M/SCF	0.2	S6
*	91	FLLP	3.3775E-10	//OGF*GA1*GBA*WAF*WBF*PAF*PBF*EHF/NLF*CAF*CBF/SUF*LTF*WSF*C2D/ER5/SCF	0.2	S6
	92	ALOMF	3.3186E-10	//RT1/PL1*MFF*AM1*TTF/SUF*LTF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNG*SMF	0.1	S1A
*	93	FPCC	3.2505E-10	//PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF*C2A/SCF	0.1	S6

* External Initiating Event

TABLE 3.4-2
(Continued)

Dominant Containment Performance Sequences -
Large, Early Containment Failure/Bypass

CD Rank	Initiating Event	Sequence Frequency	Sequence Listing - Failed Split Fractions	Percent Contribution	Release Category
* 94	FLSW	3.1263E-10	//WAF*WBF*PAF*PBF*EHF/NLF/SUF*LTF*XAF*XBF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.1	S1A
95	LOSP	3.1153E-10	//OGF*GA1*WAF*WBG*PAF*PBF*EHF/EFD*FR5*H2F*CAF*CBF/SUF*LTF*WSF/ER4/SCF*R1F*VDF*DPF*HL1*IS1	0.1	S7A
96	LOSP	3.1153E-10	//OGF*GB1*WAC*WBF*PAF*PBF*EHF/EFD*FR5*H2F*CAF*CBF/SUF*LTF*WSF/ER4/SCF*R1F*VDF*DPF*HL1*IS1	0.1	S7A
97	PLMFW	3.0964E-10	//PA1*PBA*EHF/NLF/SUF*LTF*XAF*XBF*C2A//SCF	0.1	S6
98	SLOCA	3.0703E-10	///L13*L2C/SUF*LTF/RMU/SCF*R1F*VDF*DPF*HL1*CHF*CYF*NG*SMF	0.1	S1A
99	SLOCA	3.0113E-10	//EH1/LRF/SUF/*LSF*LGF*XCF*XDF/RMF/SCF*R1F*VDF*DPF*HL1*CHF*CYF*CNK*SMF	0.1	S1A
100	RT	2.9982E-10	//SB6*EBF*OS1/EFD*FRF*ORF*CBF/SUF*LTF*2BF*C2E//SCF	0.1	S6
Others	4.36E-8			19.4	
Total	2.25E-7			100.0%	

* External Initiating Event

TABLE 3.4-3

Initiating Event Contribution

Initiating Event	Initiating Event Frequency	Description	Core Damage Frequency	Percent CDF
LOSP	4.84E-02	LOSS OF OFF-SITE POWER	1.09E-05	9.7
FTBLP	1.23E-03	FIRE IN TURBINE BUILDING - LOSP	9.27E-06	8.3
FCRCC	7.18E-06	FIRE IN CONTROL ROOM - PCC LOSS	7.17E-06	6.4
LSF6	8.96E-04	LOSS OF OFF-SITE POWER DUE TO FAULT ON SF6 SYSTEM	6.75E-06	6.0
RT	1.35E+00	REACTOR TRIP	6.38E-06	5.7
FLLP	6.90E-04	FLOOD IN TURBINE BUILDING - LOSP	5.20E-06	4.6
TT	1.07E+00	TURBINE TRIP	5.06E-06	4.5
PLMFW	1.13E+00	PARTIAL LOSS OF MAIN FEEDWATER	3.73E-06	3.3
SLOCA	1.79E-02	SMALL LOCA	3.55E-06	3.2
ALOMF	1.29E+00	ATWS - LOSS OF MAIN FEEDWATER (PLMFW + TLMFW)	3.20E-06	2.9
E10T	3.65E-06	SEISMIC 1.0G TRANSIENT EVENT	3.19E-06	2.9
FPCC	3.12E-06	FIRE IN PCC AREA	3.11E-06	2.8
LDCB	3.20E-03	LOSS OF TRAIN B DC POWER	2.83E-06	2.5
ATT	1.07E+00	ATWS - TURBINE TRIP	2.21E-06	2.0
L1CCA	2.93E-03	LOSS OF TRAIN A PRIMARY COMPONENT COOLING	2.13E-06	1.9
L1CCB	2.93E-03	LOSS OF TRAIN B PRIMARY COMPONENT COOLING	2.12E-06	1.9
E7T	1.92E-05	SEISMIC 0.7G TRANSIENT EVENT	2.12E-06	1.9
FSRCC	1.76E-06	FIRE IN CABLE SPREADING ROOM - PCC LOSS	1.75E-06	1.6
FCRAC	1.68E-06	FIRE IN CONTROL ROOM - AC POWER LOSS	1.68E-06	1.5

TABLE 3.4-3
(Continued)

Initiating Event Contribution

Initiating Event	Initiating Event Frequency	Description	Core Damage Frequency	Percent CDF
FCRSW	1.68E-06	FIRE IN CONTROL ROOM - SWS LOSS	1.67E-06	1.5
FET1	2.52E-04	FIRE IN ELECTRIC TUNNEL 1	1.57E-06	1.4
E7AT	1.57E-06	ATWS - SEISMIC 0.7G EVENT	1.57E-06	1.4
AGT	3.96E-01	ATWS - GENERAL TRANSIENT (AMSIV+MSIV+LCV+EXFW+MSRV)	1.44E-06	1.3
TCTL	1.90E-04	TRUCK CRASH INTO TRANSMISSION (SF6) LINES	1.43E-06	1.3
SLBI	4.65E-04	STEAMLINE BREAK INSIDE CONTAINMENT	1.40E-06	1.3
SGTR	2.84E-02	STEAM GENERATOR TUBE RUPTURE	1.37E-06	1.2
LLOCA	2.03E-04	LARGE LOCA	1.35E-06	1.2
FLSW	1.10E-06	EXTERNAL FLOODING - LOSS OF SERVICE WATER	1.09E-06	1.0
E2T	4.26E-04	SEISMIC 0.2G TRANSIENT EVENT	1.07E-06	1.0
MLOCA	4.65E-04	MEDIUM LOCA	1.00E-06	0.9
LCV	1.18E-01	LOSS OF CONDENSER VACUUM	9.36E-07	0.8
E3T	1.12E-04	SEISMIC 0.3G TRANSIENT EVENT	9.10E-07	0.8
E10AT	8.76E-07	ATWS - SEISMIC 1.0G EVENT	8.76E-07	0.8
LOPF	1.76E-01	LOSS OF PRIMARY FLOW	8.40E-07	0.8
EXFW	1.68E-01	EXCESSIVE FEEDWATER FLOW	8.02E-07	0.7
E10L	8.76E-07	SEISMIC 1.0G LLOCA	8.00E-07	0.7
FET3	1.26E-04	FIRE IN ELECTRIC TUNNEL 3	7.90E-07	0.7
E14T	7.70E-07	SEISMIC 1.4G TRANSIENT EVENT	7.62E-07	0.7

TABLE 3.4-3
(Continued)

Initiating Event Contribution

Initiating Event	Initiating Event Frequency	Description	Core Damage Frequency	Percent CDF
E4T	4.44E-05	SEISMIC 0.4G TRANSIENT EVENT	7.25E-07	0.6
E5T	1.98E-05	SEISMIC 0.5G TRANSIENT EVENT	7.01E-07	0.6
LDCA	3.20E-03	LOSS OF TRAIN A DC POWER	6.42E-07	0.6
MSIV	8.66E-02	CLOSURE OF ONE MSIV	6.13E-07	0.5
TLMFW	1.62E-01	TOTAL LOSS OF MAIN FEEDWATER	5.42E-07	0.5
AMFW	2.03E-01	ATWS EVENT - MFW AVAILABLE (CPEXC + LOPF)	5.16E-07	0.5
FL1SG	5.40E-06	FLOOD IN TURBINE BUILDING - LOSP AND LOSS OF ONE VITAL SWITCHGEAR ROOM	4.77E-07	0.4
E14L	4.47E-07	SEISMIC 1.4G LLOCA	4.45E-07	0.4
E1T	3.13E-03	SEISMIC 0.1G TRANSIENT EVENT	4.41E-07	0.4
SLBO	6.04E-03	STEAMLINE BREAK OUTSIDE CONTAINMENT	4.40E-07	0.4
E14AT	3.61E-07	ATWS - SEISMIC 1.4G EVENT	3.61E-07	0.3
E5AT	3.56E-07	ATWS - SEISMIC 0.5G EVENT	3.56E-07	0.3
ELOCA	2.66E-07	EXCESSIVE LOCA	2.66E-07	0.2
ASLOC	4.63E-02	ATWS - SMALL LOCA (SLOCA + SGTR)	2.64E-07	0.2
FSRAC	2.54E-07	FIRE IN CABLE SPREADING ROOM - AC POWER LOSS	2.54E-07	0.2
MSRV	4.19E-03	MAIN STEAM RELIEF VALVE OPENING	2.40E-07	0.2
E4AT	2.22E-07	ATWS - SEISMIC 0.4G	2.22E-07	0.2
E7L	1.01E-06	SEISMIC 0.7G LARGE LOCA	2.11E-07	0.2

TABLE 3.4-3
(Continued)

Initiating Event Contribution

Initiating Event	Initiating Event Frequency	Description	Core Damage Frequency	Percent CDF
SI	2.99E-02	INADVERTENT SAFETY INJECTION	2.10E-07	0.2
L1SWB	3.56E-03	LOSS OF TRAIN B SERVICE WATER	2.07E-07	0.2
ALOSP	5.14E-02	ATWS - LOSP (FL1SG+FSRAC+FCRAC+FL2SG+LOSP+FLLP+TCTL+LSF6+FTBLP)	2.07E-07	0.2
L1SWA	3.56E-03	LOSS OF TRAIN A SERVICE WATER	2.02E-07	0.2
CPEXC	2.68E-02	CORE POWER EXCURSION	1.91E-07	0.2
FL2SG	1.90E-07	FLOOD IN TURBINE BUILDING - LOSP AND LOSS OF BOTH VITAL SWITCHGEAR ROOMS	1.90E-07	0.2
AMSIV	1.93E-02	CLOSURE OF ALL MSIVS	1.56E-07	0.1
E20T	1.44E-07	SEISMIC 2.0G TRANSIENT EVENT	1.44E-07	0.1
APAB	1.40E-07	AIRCRAFT CRASH - PAB IMPACT	1.40E-07	0.1
E20L	1.23E-07	SEISMIC 2.0G LARGE LOCA	1.23E-07	0.1
E20AT	1.00E-07	ATWS - SEISMIC 2.0G EVENT	1.00E-07	0.1
VS	3.26E-06	INTERFACING SYSTEMS LOCA - RHR SUCTION	3.60E-08	<0.1
VI	4.50E-06	INTERFACING SYSTEMS LOCA - RHR INJECTION	1.13E-08	<0.1
E5L	1.19E-07	SEISMIC 0.5G LLOCA	9.35E-09	<0.1
APC	8.50E-09	AIRPLANE CRASH - CONTAINMENT BUILDING IMPACT	6.66E-10	<0.1
TMLL	1.40E-08	TURBINE MISSILE - LLOCA	5.22E-10	<0.1

TABLE 3.4-4(a)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
AM1	1.0000E-02	ATWS MITIGATION SYSTEM (AMSAC) - PLMFW, TLMFW
AMF	1.0000E+00	ATWS MITIGATION SYSTEM (AMSAC) - G.F.
C21	1.0960E-04	Large CIS lines - SEISMIC - BOTH SIGNALS/BUSES AVAILABLE
C22	9.8370E-05	Large CIS lines - SEISMIC - LOSS OF ALL POWER
C23	4.0240E-04	Large CIS lines - SEISMIC - ALL POWER NOT LOST, ONE TRAIN SIGNALS LOST
C24	1.5120E-04	Large CIS lines - SEISMIC - LOSS OF ONE SIGNAL, ALL POWER LOST
C25	2.0010E-01	Large CIS lines - SEISMIC - ALL POWER NOT LOST, BOTH SIGNALS LOST
C2A	1.0960E-04	Large CIS lines - ALL SUPPORT AVAILABLE
C2B	1.0960E-04	Large CIS lines - LOSS OF TRAIN B POWER, ALL SIGNALS AVAIL
C2C	1.0960E-04	Large CIS lines - LOSS OF TRAIN A POWER, ALL SIGNALS AVAIL
C2D	9.8370E-05	Large CIS lines - LOSS OF BOTH TRAINS OF POWER, ALL SIGNALS AVAIL
C2E	4.0240E-04	Large CIS lines - LOSS OF TRAIN B SIGNAL, BOTH BUSES AVAIL
C2F	4.0240E-04	Large CIS lines - LOSS OF TRAIN B SIGNAL AND POWER
C2G	4.0240E-04	Large CIS lines - LOSS OF TRAIN B SIGNAL AND TRAIN A POWER
C2H	9.8370E-05	Large CIS lines - LOSS OF TRAIN B SIGNAL AND ALL POWER
C2I	4.0240E-04	Large CIS lines - LOSS OF TRAIN A SIGNAL, ALL POWER AVAIL
C2J	4.0240E-04	Large CIS lines - LOSS OF TRAIN A SIGNAL AND TRAIN B POWER
C2K	4.0240E-04	Large CIS lines - LOSS OF TRAIN A SIGNAL AND POWER
C2L	9.8370E-05	Large CIS lines - LOSS OF TRAIN A SIGNAL AND ALL POWER
C2M	1.0010E-01	Large CIS lines - LOSS OF ALL SIGNALS, ALL POWER AVAIL
C2N	1.0010E-01	Large CIS lines - LOSS OF ALL SIGNALS AND B TRAIN POWER
C2O	1.0010E-01	Large CIS lines - LOSS OF ALL SIGNALS AND A TRAIN POWER
C2P	9.8370E-05	Large CIS lines - LOSS OF ALL SIGNALS AND ALL POWER
C2T	1.0000E+00	Large CIS lines - APC AND TMLL - GUARANTEED FAILURE
CA1	9.7590E-03	CBS injection - train A - all support available.
CA2	9.7590E-03	CBS injection - train A single train.
CAF	1.0000E+00	CBS injection - train A - guaranteed failure.
CB1	9.2180E-03	CBS injection - train B - all support available.
CB2	9.7590E-03	CBS injection - train B single train.
CBA	6.4700E-02	CBS injection - train B after train A fails.
CBF	1.0000E+00	CBS injection - train B - guaranteed failure.
CI1	4.4950E-03	Small CIS lines - SEISMIC - ALL SUPPORT AVAIL
CI2	8.4490E-03	Small CIS lines - SEISMIC - LOSS OF B TRAIN POWER, SIGNALS AVAIL
CI3	8.4490E-03	Small CIS lines - SEISMIC - LOSS OF A TRAIN POWER, SIGNALS AVAIL
CI4	1.0000E+00	Small CIS lines - SEISMIC - G.F. - LOSS OF ALL POWER AND SIGNALS
CI5	1.6830E-02	Small CIS lines - SEISMIC - LOSS OF B TRAIN SIGNAL, ALL POWER AVAIL
CI6	1.6830E-02	Small CIS lines - SEISMIC - LOSS OF B TRAIN SIGNAL AND POWER
CI7	1.6830E-02	Small CIS lines - SEISMIC - LOSS OF A TRAIN SIGNAL, POWER AVAIL
CI8	1.6830E-02	Small CIS lines - SEISMIC - LOSS OF A TRAIN SIGNAL AND POWER
CIA	4.4950E-03	Small CIS lines - ALL SUPPORT AVAILABLE

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
CIB	8.4490E-03	Small CIS lines - LOSS OF TRAIN B POWER, ALL SIGNALS AVAIL
CIC	4.1680E-03	Small CIS lines - LOSS OF TRAIN A POWER, ALL SIGNALS AVAIL
CID	5.1640E-03	Small CIS lines - LOSS OF BOTH TRAINS OF POWER, ALL SIGNALS AVAIL
CIE	1.7100E-02	Small CIS lines - LOSS OF TRAIN B SIGNAL, BOTH BUSES AVAIL
CIF	1.7100E-02	Small CIS lines - LOSS OF TRAIN B SIGNAL AND POWER
CIG	1.0000E+00	Small CIS lines - LOSS OF TRAIN B SIGNAL AND TRAIN A POWER
CIH	1.2130E-02	Small CIS lines - LOSS OF TRAIN B SIGNAL AND ALL POWER
CII	4.4500E-03	Small CIS lines - LOSS OF TRAIN A SIGNAL, ALL POWER AVAIL
CIJ	1.0000E+00	Small CIS lines - LOSS OF TRAIN A SIGNAL AND TRAIN B POWER
CIK	1.6830E-02	Small CIS lines - LOSS OF TRAIN A SIGNAL AND POWER
CIL	1.0000E+00	Small CIS lines - LOSS OF TRAIN A SIGNAL AND ALL POWER
CIM	1.0000E+00	Small CIS lines - LOSS OF ALL SIGNALS, ALL POWER AVAIL
CIN	1.0000E+00	Small CIS lines - LOSS OF ALL SIGNALS AND B TRAIN POWER
CIO	1.0000E+00	Small CIS lines - LOSS OF ALL SIGNALS AND A TRAIN POWER
CIP	1.0000E+00	Small CIS lines - LOSS OF ALL SIGNALS AND ALL POWER
CIT	1.0000E+00	Small CIS lines - APC AND TMLL - GUARANTEED FAILURE
CSA	1.1000E-01	CBS pumps survive vault environment (seal leak < 0.09 sq. in.)
CSB	1.0000E-01	CBS pumps survive vault environment (operator terminates interfacing LOCA and 0.09 < seal leak < 1.05 sq.")
CSC	4.4000E-01	CBS pumps survive vault environment (operator terminates interfacing LOCA and seal leak > 0.09 sq.")
CSD	7.5000E-01	CBS pumps survive vault environment (operator terminates interfacing LOCA and 1.05 < seal leak < 2.6 sq.")
CSF	1.0000E+00	V-SEQUENCE, GUARANTEED FAILURE
CV1	0.0000E+00	EAH operating (yes) - long term trees.
CVF	1.0000E+00	EAH operating (no) - long term trees.
DA1	8.9220E-04	DC Train A - Loss of AC Power
DA2	8.9220E-04	DC Train A - Single Train - Loss of AC Power
DAF	1.0000E+00	DC Train A - Guaranteed Failure
DB1	8.9160E-04	DC Train B - DA Success - Loss of AC Power
DB2	8.9220E-04	DC Train B - Single Train - Loss of AC Power
DBA	1.5340E-03	DC Train B - DA Failure - Loss of AC Power
DBF	1.0000E+00	DC Train B - Guaranteed Failure
EA1	1.0790E-02	ESFAS Train A - LLOCA/MLOCA - all support available.
EA2	8.5500E-03	ESFAS Train A - SLOCA - SGTR - all support available.
EA3	9.2210E-03	ESFAS Train A - SLBOC - all support available.
EA4	1.1460E-02	ESFAS Train A - SLBIC - all support available.
EA5	1.1600E-03	ESFAS Train A - GT - all support available.

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
EAA	1.0790E-02	ESFAS Train A - LLOCA/MLOCA - one support train avail.
EAB	8.5500E-03	ESFAS Train A - SLOCA - SGTR - one support train avail
EAC	9.2210E-03	ESFAS Train A - SLBOC - one support train available.
EAD	1.1460E-02	ESFAS Train A - SLBIC - one support train available.
EAE	1.1600E-03	ESFAS Train A - GT - one support train available.
EAF	1.0000E+00	ESFAS Train A - Guaranteed failure.
EB1	1.0830E-02	ESFAS Train B - LLOCA/MLOCA - all support available.
EB2	8.5530E-03	ESFAS Train B - SLOCA/SGTR - all support available.
EB3	9.2350E-03	ESFAS Train B - SLBOC - all support available.
EB4	1.1510E-02	ESFAS Train B - SLBIC - all support available.
EB5	1.1370E-03	ESFAS Train B - GT - all support available.
EBA	7.1040E-03	ESFAS Train B after train A failure - LLOCA/MLOCA.
EBB	1.0790E-02	ESFAS Train B - LLOCA/MLOCA - one support train avail.
EBC	8.1560E-03	ESFAS Train B after train A failure - SLOCA/SGTR
EBD	8.5500E-03	ESFAS Train B - SLOCA/SGTR - one support train avail
EBE	7.6730E-03	ESFAS Train B after train A failure - SLBOC.
EBF	1.0000E+00	ESFAS Train B - Guaranteed failure.
EBG	9.2210E-03	ESFAS Train B - SLBOC - one support train available.
EBH	6.7980E-03	ESFAS Train B after train A failure - SLBIC.
EBI	1.1460E-02	ESFAS Train B - SLBIC - one support train available.
EBJ	2.1300E-02	ESFAS Train B after train A failure - GT.
EBK	1.1600E-03	ESFAS Train B - GT - one support train available.
EFA	2.7300E-04	ASSA (MDP*TDP + ARV*SDV)
EFB	3.7860E-04	ASSA and LOSP (MDP*TDP + ARV)
EFC	4.7580E-02	(TDP + ARV*SDV)
EFD	4.7690E-02	NO AC POWER (TDP + ARV)
EFE	5.3380E-03	MDP + ARV*SDV
EFF	1.0000E+00	GUARANTEED FAILURE OF EFW
EFG	5.4440E-03	MDP
EFH	1.5940E-06	LOMF, ASSA (MDP*TDP*SFP + ARV*SDV)
EFI	1.0720E-04	MDP*TDP*SFP + ARV
EFJ	2.7210E-04	TDP*SFP + ARV*SDV
EFK	3.7770E-04	TDP*SFP + ARV
EFL	5.7190E-03	LOMF, NO SIGNALS (SFP + ARV*SDV)
EFM	5.6420E-03	ATWS - FEEDING ALL 4 SGs - MDP*TDP + ARV*SDV
EFN	5.7480E-03	ATWS - FEEDING ALL 4 SGs - MDP*TDP + ARV
EFO	5.3240E-02	ATWS - FEEDING ALL 4 SGs - TDP + ARV*SDV
EFP	5.3350E-02	ATWS - FEEDING ALL 4 SGs - TDP + ARV

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
EFQ	1.1000E-02	ATWS - FEEDING ALL 4 SGs - MDP + ARV*SDV
EFR	1.1110E-02	ATWS - FEEDING ALL 4 SGs - MDP + ARV
EFS	3.2300E-05	ATWS - LOMF, ASSA (MDP*TDP*SFP + ARV*SDV)
EFT	5.7190E-03	ATWS - LOMF, NO SIGNALS (SFP + ARV*SDV)
EFU	1.0690E-04	ASSA (MDP*TDP*EFRTDP*EFRSFP + ARV) - RECOVERY
EFV	2.5650E-04	ASSA (MDP*TDP*EFRTDP + ARV) - RECOVERY
EFW	1.5190E-04	MDP*EFRSFP + ARV - SUFP RECOVERY
EFX	3.3360E-04	TDP*EFRTDP*EFRSFP + ARV - RECOVERY
EFY	2.6410E-02	TDP*EFRTDP + ARV
EFZ	1.3400E-06	ASSA (MDP*TDP*EFRTDP*EFRSFP + ARV*SDV) - RECOVERY
EH1	6.0410E-05	EAH - GT or T signal.
EH2	1.3220E-03	EAH - Single T signal.
EH3	1.0150E-02	EAH - Single PCC Train
EH4	1.1380E-02	EAH - Single T signal, Single PCC train.
EH5	6.9920E-05	EAH - LOSP
EH6	1.3310E-03	EAH - LOSP, Single T Signal
EH7	1.5670E-03	EAH - LOSP, Single EBus or Single PCC Train
EH8	2.0970E-03	EAH - LOSP, Single T Signal, Single EBus(or PCC Train)
EHF	1.0000E+00	EAH - Guaranteed Failure (Both T, No AC power).
ER1	1.0910E-02	RECOVERY OF 1 OF 2 D/Gs OR OFF-SITE POWER - EFW
ER2	6.4010E-02	RECOVERY OF 1 OF 2 D/Gs OR OFF-SITE POWER - NO EFW
ER3	1.2230E-02	RECOVERY OF ONE D/G OR OFF-SITE POWER - EFW
ER4	6.6590E-02	RECOVERY OF ONE D/G OR OFF-SITE POWER - NO EFW
ER5	7.1370E-01	RECOVERY OF 1 OF 2 D/Gs, NO OFF-SITE POWER - EFW
ER6	8.8780E-01	RECOVERY OF 1 OF 2 D/Gs, NO OFF-SITE POWER - NO EFW
ER7	5.5560E-01	RECOVERY OF ONE D/G, NO OFF-SITE POWER - EFW
ER8	6.9320E-01	RECOVERY OF ONE D/G, NO OFF-SITE POWER - NO EFW
ER9	4.8300E-02	RECOVERY OF OFF-SITE POWER - EFW
ERA	1.9300E-01	RECOVERY OF OFF-SITE POWER - NO EFW
ERF	1.0000E+00	STATION BLACKOUT RECOVERY - GUARANTEED FAILURE
ERS	0.0000E+00	RECOVERY OF OFF-SITE POWER, D/Gs - NOT NEEDED
FR0	1.0000E+00	EFW RECOVERY - DEFAULT S.F. ASSIGNMENT - G.F.
FR1	2.8240E-01	MDP & TDP AVAILABLE - TDP OR SUFP EFW PUMP RECOVERY
FR2	6.7750E-01	MDP & TDP AVAILABLE - TDP EFW PUMP RECOVERY
FR3	2.7900E-02	ONLY MDP AVAILABLE - SUFP RECOVERY
FR4	6.9950E-03	ONLY TDP AVAILABLE - TDP OR SUFP RECOVERY
FR5	5.5380E-01	ONLY TDP AVAILABLE - TDP RECOVERY
FRA	4.9080E-03	MDP & TDP AVAILABLE - TDP OR SUFP RECOVERY W/ SDVs
FRF	1.0000E+00	EFW (TDP AND/OR SUFP) RECOVERY - GUARANTEED FAILURE

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
FRS	0.0000E+00	EFW RECOVERY - GUARANTEED SUCCESS IF EF=S
GA1	7.3430E-02	Diesel Generator Train A - Given LOSP
GA2	7.3430E-02	Diesel Generator Train A - Single Train - Given LOSP
GAF	1.0000E+00	Diesel Generator Train A - Guaranteed Failure
GAS	0.0000E+00	DIESEL GENERATOR NOT ASKED, OG SUCCESS
GB1	7.1190E-02	Diesel Generator Train B - Given Train A Success, LOSP
GB2	7.3430E-02	Diesel Generator Train B - Single Train, LOSP
GBA	1.0170E-01	Diesel Generator Train B - Given Train A Failure, LOSP
GBF	1.0000E+00	Diesel Generator Train B - Guaranteed Failure
GBS	0.0000E+00	DIESEL GENERATOR NOT ASKED, OG SUCCESS
H11	3.4520E-05	High pressure injection - MLOCA - all support.
H12	3.0930E-02	HPI - MLOCA - loss of one AC power train (signal).
H13	1.6560E-02	HPI - MLOCA - loss of one PCC train.
H1F	1.0000E+00	HPI - MLOCA - guaranteed failure.
H21	1.1680E-06	HPI - SLOCA, etc. - all support available.
H22	1.9500E-04	HPI - SLOCA - loss of one AC power train.
H23	7.3340E-05	HPI - SLOCA - loss of one PCC train.
H2F	1.0000E+00	HPI - SLOCA - guaranteed failure.
H31	1.1270E-03	HPI - ATWS - all support available.
H32	2.2260E-02	HPI - ATWS - loss of one AC power train.
H33	8.0780E-03	HPI - ATWS - loss of one PCC train.
H3A	1.1680E-06	HPI - ATWS feed and bleed - all support available.
H3B	1.9500E-04	HPI - ATWS feed and bleed loss of one AC train.
H3C	7.3340E-05	HPI - ATWS feed and bleed loss of one PCC train.
H3F	1.0000E+00	HPI - ATWS - guaranteed failure.
HA1	4.3630E-03	RHR HX cooling - LLOCA - Train A all support avail.
HA2	4.3630E-03	RHR HX cooling - LLOCA - Train A - single train.
HAF	1.0000E+00	RHR HX cooling - LLOCA - Train A - guaranteed fail.
HB1	4.0340E-03	RHR HX cooling - LLOCA - Train B all support avail.
HB2	4.3630E-03	RHR HX cooling - LLOCA - Train B - single train.
HBA	7.9420E-02	RHR HX cooling - LLOCA - Train B after train A fail
HBF	1.0000E+00	RHR HX cooling - LLOCA - Train B - guaranteed fail.
HE1	8.0070E-04	Oper. action - LLOCA recirculation.
HE2	8.0110E-04	Oper. action - LLOCA recirc. - single train.
HEF	1.0000E+00	Oper. action - recirc. - guaranteed failure.
HS1	8.0070E-04	Oper. action - LLOCA hot leg recirculation.
HS2	8.0110E-04	Oper. action - LLOCA hot leg recirc. - single train
HSF	1.0000E+00	Oper. action - hot leg recirc. - guaranteed failure

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
L11	1.7850E-02	RHR miniflow train A - MLOCA - all support avail.
L12	1.7850E-02	RHR miniflow train A - MLOCA - loss of one AC, etc.
L13	1.8100E-02	RHR miniflow train A - SLOCA - all support avail.
L14	1.8100E-02	RHR miniflow train A - SLOCA - loss of one AC, etc.
L1F	1.0000E+00	RHR miniflow train A - guaranteed failure.
L21	1.7570E-02	RHR miniflow train B - MLOCA - all support avail.
L22	1.7850E-02	RHR miniflow train B - MLOCA - loss of one AC, etc.
L23	1.7810E-02	RHR miniflow train B - SLOCA - all support avail.
L24	1.8100E-02	RHR miniflow train B - SLOCA - loss of one AC, etc.
L2A	3.3520E-02	RHR miniflow train B after train A - MLOCA.
L2C	3.3610E-02	RHR miniflow train B after train A - MLOCA.
L2F	1.0000E+00	RHR miniflow train B - guaranteed failure.
L51	1.0110E-03	LPR - Train A (including HX) - all support avail.
L52	1.0110E-03	LPR (with HX) - Train A - single train.
L53	1.1210E-03	HPR - Train A (including HX) - all support avail.
L54	1.1210E-03	HPR (with HX) - Train A - single train.
L5F	1.0000E+00	LPR (with HX) - Train A - guaranteed failure.
L61	1.0040E-03	LPR - Train B (including HX) - all support avail.
L62	1.0110E-03	LPR (with HX) - Train B - single train.
L63	1.1140E-03	HPR - Train B (including HX) - all support avail.
L64	1.1210E-03	HPR (with HX) - Train B - single train.
L6A	7.8940E-03	LPR - Train B after train A fails.
L6B	7.3320E-03	HPR - Train B after train A fails.
L6F	1.0000E+00	LPR (with HX) - Train B - guaranteed failure.
LA1	1.5360E-02	LPI - Train A - LLOCA - all support available.
LA2	1.5360E-02	LPI - Train A - LLOCA - single train.
LAF	1.0000E+00	LPI - Train A - LLOCA - guaranteed failure.
LB1	1.1080E-02	LPI - Train B - LLOCA - all support available.
LB2	1.5360E-02	LPI - Train B - LLOCA - single train.
LBA	2.9000E-01	LPI - Train B after train A fails.
LBF	1.0000E+00	LPI - Train B - LLOCA - guaranteed failure.
LC1	1.1940E-03	Low pressure recirc. (LPR) - Train A - all support.
LC2	1.1940E-03	LPR - Train A - single train (one AC, PCC).
LCF	1.0000E+00	LPR - Train A - guaranteed failure.
LD1	1.1830E-03	LPR - Train B - all support available.
LD2	1.1940E-03	LPR - Train B - single train (one AC, PCC).
LDA	1.0540E-02	LPR - Train B after train A fails.
LDF	1.0000E+00	LPR - Train B - guaranteed failure.
LE1	9.0000E-02	V-SEQUENCE, LR IN RMEPS

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
LR1	1.3170E-03	Operators align RHR for long-term recirculation given all support systems available - SLOCA
LR2	1.1990E-02	Operators align RHR for long-term recirculation given only one support system available - SLOCA
LRF	1.0000E+00	Long term cooling - SLOCA - guaranteed failure.
LTF	1.0000E+00	LATE TREE SWITCH (LT2 TREE)
LTS	0.0000E+00	LATE TREE SWITCH (LT1 TREE)
LX1	9.1900E-01	0.0 < PUMP SEAL LEAK < 0.09 SQUARE INCHES
LY1	5.6920E-01	0.09 < PUMP SEAL LEAK < 1.05 SQUARE INCHES
LZ1	2.0000E-02	1.05 < PUMP SEAL LEAK < 2.6 SQUARE INCHES
MF1	0.0000E+00	MFW REMAINS AVAIL. FOR 1ST 4-5 MIN. - LOPF, TT, CPEXC
MFF	1.0000E+00	MFW REMAINS AVAILABLE - G.F.
MR1	2.0000E-02	MANUAL ROD INSERTION (AUTO AND MANUAL) - ATWS
MR2	3.0000E-02	MANUAL ROD INSERTION (AUTO ONLY) - ATWS
MRF	1.0000E+00	MANUAL ROD INSERTION - ATWS (G.F.)
MS1	1.3230E-04	MSIV isolation - SLB or turbine trip failure
MSF	1.0000E+00	MSIV isolation - guaranteed failure
MT1	1.0000E-02	Operator manually generates reactor trip signal given SSPS failure
MTF	1.0000E+00	MANUAL REACTOR TRIP - GUARANTEED FAILURE
MTS	0.0000E+00	MANUAL REACTOR TRIP - NOT NEEDED
NL1	0.0000E+00	No reactor coolant pump seal failure.
NLF	1.0000E+00	Guaranteed reactor coolant pump seal failure.
O11	6.5000E-03	V-SEQUENCE, O1 IN RMEPS
O21	1.0000E+00	VS-SEQUENCE, O2 IN RMEPS (VS)
O22	9.1000E-03	VI-SEQUENCE, O2A IN RMEPS (VI)
O31	8.0000E-04	Operators align ECCS flow for low/high pressure sump recirc
O32	0.0000E+00	Oper. action - not asked
O3C	4.9000E-03	Operators provide makeup to RWST during V-Sequence
O3F	1.0000E+00	Oper. action - guaranteed failure.
O41	5.0000E-02	Operators depressurize the RCS using pzr spray, EFW, and S/Gs - SGTR
O42	7.0000E-02	Operators depressurize and cool down by feed and bleed - SGTR
O4F	1.0000E+00	FAILURE OF EVENT O4 IN SGTR EVENT TREE
O51	5.0000E-02	Operators continue feed and bleed cooling to 350 deg. F and 400 psig - SGTR
O52	9.0000E-02	Operators depressurize the RCS given failure of HPI - SGTR
O53	1.3000E-02	Operators rapidly depressurize S/Gs to cool down and depressurize the RCS - SGTR
OD1	2.6000E-02	Operators rapidly depressurize S/Gs to cool down and depressurize the RCS - MLOCA

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
OD2	1.3000E-02	Operators rapidly depressurize S/Gs to cool down and depressurize the RCS - General Transients, Small LOCA
ODF	1.0000E+00	Operator action - depressurize - guaranteed failure.
OG1	5.7200E-04	LOSS OF OFF-SITE GRID GIVEN NO LOSP INITIATING EVENT
OGF	1.0000E+00	GUARANTEED FAILURE OF OFF-SITE GRID
OH1	5.3100E-03	Operator initiates emergency boration or trips the reactor following ATWS event
OH3	0.0000E+00	Oper. action - manual shutdown - guaranteed success.
OHF	1.0000E+00	Oper. action - manual shutdown - guaranteed failure.
OM1	6.2000E-02	Oper. action - control EFW flow - overcooling.
OM2	0.0000E+00	Oper. action - control EFW flow - not asked.
OMF	1.0000E+00	Oper. action - control EFW flow - guaranteed failure.
OP1	2.3000E-02	Oper. action - control HPI flow - overcooling.
OP2	0.0000E+00	Oper. action - control HPI flow - not asked.
OPF	1.0000E+00	Oper. action - control HPI flow - guaranteed failure.
OQ1	0.0000E+00	Oper. action - plant stabilization.
OQ2	1.3000E-02	Oper. action - plant stabilization - dep. SG's.
OQ3	1.0000E-04	Oper. action - plant stabilization - SGTR
OQF	1.0000E+00	Oper. action - guaranteed failure.
OR1	1.7000E-02	Oper. action - feed and bleed - SGTR break flow.
OR2	0.0000E+00	Oper. action - feed and bleed - not asked.
OR4	2.7460E-02	Oper. action - feed and bleed - OR1 + PR1.
ORF	1.0000E+00	Oper. action - feed and bleed - guaranteed failure.
OS1	1.0000E-02	OPER. ACTION - RECOVER ESFAS - LONG RESPONSE TIME AVAI
OS2	1.0000E-01	OPER. ACTION - RECOVER ESFAS - LOCAs, FIRES, SEISMICS
OSF	1.0000E+00	OPER. ACTION - RECOVER ESFAS - G.F.
OSS	0.0000E+00	OPER. ACTION - RECOVER ESFAS - BYPASS
OT1	0.0000E+00	Oper. action - manual trip turbine ATWS - NA.
OTF	1.0000E+00	Oper. action - manual trip turbine ATWS.(guar failure)
P21	5.8690E-02	Safety and relief valves reseat - ATWS
P2F	1.0000E+00	SAFETY AND RELIEF VALVES RESEAT - G.F.
PA1	6.3730E-04	PCC Train A - no P signal - off-site power available
PA2	1.6270E-03	PCC Train A - LOSP.
PA3	6.3730E-04	PCC Train A - P signal - off-site power available.
PA4	6.3730E-04	PCC Train A - single train - no P signal, no LOSP.
PA5	1.6270E-03	PCC Train A - single train - LOSP.
PA6	6.3730E-04	PCC Train A - single train - P signal required.
PAF	1.0000E+00	PCC Train A - guaranteed failure.
PB1	6.3510E-04	PCC Train B - no P signal - off-site power available
PB2	1.5850E-03	PCC Train B - LOSP.

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
PB3	6.3510E-04	PCC Train B - P signal - off-site power available.
PB4	6.3730E-04	PCC Train B - single train - no P signal, no LOSP.
PB5	1.6270E-03	PCC Train B - single train - LOSP.
PB6	6.3730E-04	PCC Train B - single train - P signal required.
PBA	4.1160E-03	PCC Train B after train A fails - no P, no LOSP.
PBB	2.7300E-02	PCC Train B after train A fails - LOSP.
PBC	4.1160E-03	PCC Train B after train A fails - P signal.
PBF	1.0000E+00	PCC Train B - guaranteed failure.
PI1	6.0000E-03	V-SEQUENCE, PI IN RMEPS
PL1	6.7000E-01	Plant power level (> 40%) - ATWS
PR1	1.0460E-02	PORV in feed and bleed.
PR2	3.3140E-04	PORV lift - ATWS - chemical shutdown 1/2 PORV
PR3	5.2300E-03	PORV lift - ATWS - chemical shtdwn 1/1 PORV - Sngl Trn
PR4	2.7460E-02	PR1 + OR : OR - Operator initiates feed and bleed.
PR5	1.7330E-02	PR2 + OR : OR - Operator initiates feed and bleed.
PR6	2.2230E-02	PR3 + OR : OR - Operator initiates feed and bleed.
PRF	1.0000E+00	Feed and bleed guaranteed failure.
PS1	1.2840E-03	Primary pressure relief - Severe ATWS - 1/2 PORV
PS2	6.1830E-03	Prim press relief - Severe ATWS,sngl trn - 1/1 PORV
PS3	9.8250E-04	Primary pressure relief - ATWS
PS4	0.0000E+00	Primary pressure relief ATWS - not required.
PSA	1.4800E-03	RCS PRESSURE RELIEF - MRI and 50% EFW Flow
PSB	4.4500E-02	RCS PRESSURE RELIEF - No MRI and 50% EFW Flow
PSF	1.0000E+00	Primary pressure relief ATWS - guaranteed failure.
QD1	0.0000E+00	DIESEL GENERATOR AT .1G
QD2	0.0000E+00	DIESEL GENERATOR AT .2G
QD3	0.0000E+00	DIESEL GENERATOR AT .3G
QD4	1.0000E-03	DIESEL GENERATOR AT .4G
QD5	8.0000E-03	DIESEL GENERATOR AT .5G
QD7	5.3000E-02	DIESEL GENERATOR AT .7G
QDA	1.9700E-01	DIESEL GENERATOR AT 1.0G
QDB	4.5800E-01	DIESEL GENERATOR AT 1.4G
QDC	7.2000E-01	DIESEL GENERATOR AT 2.0G
QK1	0.0000E+00	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .1G
QK2	0.0000E+00	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .2G
QK3	7.0000E-04	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .3G
QK4	5.5000E-03	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .4G
QK5	1.7100E-02	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .5G
QK7	5.0000E-02	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT .7G

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
QKA	8.4300E-01	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT 1.0G
QKB	9.8000E-01	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT 1.4G
QKC	9.9900E-01	4.16 KV SWITCHGEAR (RELAY CHATTERING) AT 2.0G
QR1	0.0000E+00	RWST AT .1G
QR2	0.0000E+00	RWST AT .2G
QR3	0.0000E+00	RWST AT .3G
QR4	7.0000E-03	RWST AT .4G
QR5	2.4000E-02	RWST AT .5G
QR7	1.0200E-01	RWST AT .7G
QRA	2.9400E-01	RWST AT 1.0G
QRB	5.7200E-01	RWST AT 1.4G
QRC	8.0400E-01	RWST AT 2.0G
QSF	1.0000E+00	SEISMIC EVENT
QSS	0.0000E+00	NON-SEISMIC EVENT
QY1	7.0000E-03	OFF-SITE POWER AT SEISMIC LEVEL .1G
QY2	2.3300E-01	OFF-SITE POWER AT SEISMIC LEVEL .2G
QY3	5.0000E-01	OFF-SITE POWER AT SEISMIC LEVEL .3G
QY4	6.9700E-01	OFF-SITE POWER AT SEISMIC LEVEL .4G
QY5	8.2500E-01	OFF-SITE POWER AT SEISMIC LEVEL .5G
QY7	9.5700E-01	OFF-SITE POWER AT SEISMIC LEVEL .7G
QYA	9.9800E-01	OFF-SITE POWER AT SEISMIC LEVEL 1.0G
QYB	1.0000E+00	OFF-SITE POWER AT SEISMIC LEVEL 1.4G
QYC	1.0000E+00	OFF-SITE POWER AT SEISMIC LEVEL 2.0G
RA1	1.0260E-04	RWST outlet valve - train A - LLOCA.
RA2	1.0270E-04	RWST outlet valve - train A - MLOCA.
RA3	1.0300E-04	RWST outlet valve - train A - SLOCA, etc.
RB1	1.0260E-04	RWST outlet valve - train B - LLOCA.
RB2	1.0270E-04	RWST outlet valve - train B - MLOCA.
RB3	1.0300E-04	RWST outlet valve - train B - SLOCA, etc.
RC1	1.8360E-08	HPR - high pressure pumps - all support available.
RC2	8.2010E-07	HPR (pumps) - loss of train A LPR.
RC3	1.8610E-08	HPR (pumps) - loss of train B LPR.
RC4	1.2430E-06	HPR (pumps) - loss of one train of AC power.
RC5	1.2410E-06	HPR (pumps) - loss of one train of PCC.
RC6	0.0000E+00	HPR (pumps) - guaranteed success
RCF	1.0000E+00	HPR (pumps) - guaranteed failure.
RMF	1.0000E+00	OPERATOR PROVIDES MAKEUP TO RWST - G.F.
RMS	0.0000E+00	OPERATOR PROVIDES MAKEUP TO RWST - N/A
RMU	1.0000E-01	OPERATOR PROVIDES MAKEUP TO RWST - SLOCA

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
RP1	0.0000E+00	Reactor pressure > 180 psig.
RPF	1.0000E+00	Reactor pressure < 180 psig.
RSA	5.6000E-01	RHR pumps survive vault environment (seal leak < 0.09 square inches)
RSB	5.5000E-01	RHR pumps survive vault environment (operator terminates interfacing LOCA and 0.09 < seal leak < 1.05 sq.")
RSC	8.5000E-01	RHR pumps survive vault environment (seal leak > 1.05 sq.")
RSF	1.0000E+00	V-SEQUENCE, GUARANTEED FAILURE
RT1	1.3710E-04	Reactor Trip - Both SSPS Trains Avail. - No Op. Action
RT2	1.7710E-03	Reactor Trip - Single SSPS Train Avail. - No Op.Action
RT3	7.4800E-06	Reactor Trip - SSPS Not Required (LOSP, etc.)
RTF	1.0000E+00	Reactor Trip - Guaranteed Failure
RTS	0.0000E+00	Reactor Trip - Guaranteed Success
RV1	0.0000E+00	No Reactor pressure vessel failure.
RV2	1.0000E-02	Reactor pressure vessel failure.
RV3	1.0000E+00	RCS BOUNDARY FAILURE GIVEN OVERPRESSURE - ATWS
RW1	2.7620E-08	RWST - LLOCA
RW2	5.5240E-08	RWST - MLOCA
RW3	1.6570E-07	RWST - SLOCA
RWF	1.0000E+00	RWST - GUARANTEED FAILURE (SEISMIC)
SA1	1.6060E-03	SSPS Train A - LLOCA/MLOCA - all support available.
SA2	1.2070E-03	SSPS Train A - SLOCA - all support available.
SA3	1.2090E-03	SSPS Train A - SGTR - all support available.
SA4	1.2090E-03	SSPS Train A - SLBO - all support available.
SA5	1.6060E-03	SSPS Train A - SLBIC - all support available.
SA6	1.2090E-03	SSPS Train A - GT - all support available.
SA8	0.0000E+00	SSPS Train A - Guaranteed success.
SAA	1.6060E-03	SSPS Train A - LLOCA/MLOCA - one support train avail.
SAB	1.2070E-03	SSPS Train A - SLOCA One support train available.
SAC	1.2090E-03	SSPS Train A - SGTR - one support train available.
SAD	1.2090E-03	SSPS Train A - SLBOC - one support train available.
SAE	1.6060E-03	SSPS Train A - SLBIC - one support train available.
SAF	1.0000E+00	SSPS Train A - Guaranteed failure.
SAG	1.2090E-03	SSPS Train A - GT - one support train available.
SB1	1.2070E-03	SSPS Train B - LLOCA/MLOCA - all support available.
SB2	1.2060E-03	SSPS Train B - SLOCA - all support available.
SB3	1.2060E-03	SSPS Train B - SGTR - all support available.
SB4	1.2060E-03	SSPS Train B - SLBOC - all support available.
SB5	1.2070E-03	SSPS Train B - SLBIC - all support available.
SB6	1.2060E-03	SSPS Train B - GT - all support available.

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
SBA	2.4990E-01	SSPS Train B after train A failed - LLOCA/MLOCA.
SBB	1.6060E-03	SSPS Train B - LLOCA/MLOCA - one support train avail.
SBC	1.6750E-03	SSPS Train B after train A failed - SLOCA.
SBD	1.2070E-03	SSPS Train B - SLOCA - one support train available.
SBE	3.4060E-03	SSPS Train B after train A failed - SGTR.
SBF	1.0000E+00	SSPS Train B - Guaranteed failure.
SBG	1.2090E-03	SSPS Train B - SGTR - one support train available.
SBH	3.4060E-03	SSPS Train B after train A failed - SLBOC.
SBI	1.2090E-03	SSPS Train B - SLBOC - one support train available.
SBJ	2.4990E-01	SSPS Train B after train A failed - SLBIC.
SBK	1.6060E-03	SSPS Train B - SLBIC - one support train available.
SBL	3.4060E-03	SSPS Train B after train A failed - GT.
SBM	1.2090E-03	SSPS Train B - GT - one support train available.
SI1	9.9000E-01	V-Sequence - RHR Pump Seals Remain Intact
SL1	1.0800E-04	No secondary side leak to atmosphere - SGTR.
SL2	5.7700E-03	No secondary side leak to atmosphere - SGTR.
SL3	5.4600E-03	No secondary side leak to atmosphere - SGTR
SL4	1.1000E-02	No secondary side leak to atmosphere - SGTR.
SL5	4.2000E-04	No secondary side leak to atmosphere - SGTR.
SL6	9.3900E-03	No secondary side leak to atmosphere - SGTR (O4 fail).
SL7	2.0100E-01	No secondary side leak to atmosphere - SGTR (O4 fail).
SSA	1.1000E-01	VS-SEQUENCE, SSA IN RMEPS (VS)
SSB	1.0000E-01	VI-SEQUENCE, SSA IN RMEPS (VI)
SSC	3.3000E-01	VI-SEQUENCE, SSB IN RMEPS (VI)
SSF	1.0000E+00	V-SEQUENCE, GUARANTEED FAILURE
SUF	1.0000E+00	ENTRY TO LT1 OR LT2
SUS	0.0000E+00	SUCCESSFUL CORE COOLING
TT1	4.4560E-06	Turbine trip (non - TT events)
TT2	0.0000E+00	Turbine trip (TT events) - guaranteed success
TT3	5.8950E-10	Turbine trip * MS1
TTF	1.0000E+00	Turbine trip - guaranteed failure
VA1	4.2860E-03	CBS recirc cooling - train A - all support avail.
VA2	4.2860E-03	CBS recirc cooling - train A single train.
VA3	0.0000E+00	CBS no recirc cooling required - Train A.
VAF	1.0000E+00	CBS recirc cooling - train A - guaranteed failure.
VB1	3.9840E-03	CBS recirc cooling - train B - all support avail.
VB2	4.2860E-03	CBS recirc cooling - train B single train.
VB3	0.0000E+00	CBS no recirc cooling required - Train B.

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
VBA	7.4360E-02	CBS recirc cooling - train B after train A fails.
VBF	1.0000E+00	CBS recirc cooling - train B - guaranteed failure.
VC1	1.0000E-01	V-Sequence - Relief Valves Close
VO1	4.8000E-05	V-Sequence - Relief Valves Open
WA1	5.5650E-03	SW Train A - after SI, no LOSP.
WA2	1.2860E-03	SW Train A - no SI - no LOSP.
WA3	1.3370E-02	SW Train A - after LOSP.
WA4	4.3440E-03	SW Train A - after SI, no LOSP. - W/ OP ACTION
WA5	6.0130E-05	SW Train A - no SI - no LOSP. - W/ OP ACTION
WA6	4.6760E-02	SW Train A - CT Operation given I.E. = L1SWA
WAA	5.5650E-03	SW Train A - single train after SI.
WAB	1.2860E-03	SW Train A - single train no SI.
WAC	1.3370E-02	SW Train A - single train LOSP.
WAD	4.3440E-03	SW Train A - single train after SI. - W/ OP ACTION
WAE	6.0130E-05	SW Train A - single train no SI. - W/ OP ACTION
WAF	1.0000E+00	SW Train A - guaranteed failure.
WB1	5.2240E-03	SW Train B after SI - no LOSP.
WB2	1.2470E-03	SW Train B - no SI - no LOSP.
WB3	1.2750E-02	SW Train B - after LOSP.
WB4	4.0300E-03	SW Train B after SI - no LOSP. - W/ OP ACTION
WB5	5.9740E-05	SW Train B - no SI - no LOSP. - W/ OP ACTION
WB6	4.6760E-02	SW Train B - CT Operation given I.E. = L1SWB
WBA	6.6500E-02	SW Train B after train A fails - SI.
WBB	3.1830E-02	SW Train B after train A fails - no SI.
WBC	5.9090E-02	SW Train B after train A fails - LOSP.
WBD	5.5650E-03	SW Train B - single train after SI.
WBE	1.2860E-03	SW Train B - single train no SI.
WBF	1.0000E+00	SW Train B - guaranteed failure.
WBG	1.3370E-02	SW Train B - single train LOSP.
WBH	7.6310E-02	SW Train B after train A fails - SI. - W/ OP ACTION
WBI	4.3440E-03	SW Train B - single train after SI. - W/ OP ACTION
WBJ	6.5760E-03	SW Train B after train A fails - no SI. - W/ OP ACTION
WBK	6.0130E-05	SW Train B - single train no SI. - W/ OP ACTION
WS1	0.0000E+00	Water in containment - Yes.
WSF	1.0000E+00	Water in containment - No.
XA1	6.3520E-03	CBS recirc w/o cooling - train A - all support avail
XA2	6.3520E-03	CBS recirc w/o cooling - train A single train.
XA3	1.9870E-02	CBS FOR LT2Y (START & RUN)

TABLE 3.4-4(a)
(Continued)

Plant Event Tree Split Fractions

Split Fraction	Unavailability	Description
XAF	1.0000E+00	CBS recirc w/o cooling - train A - guaranteed failure
XB1	6.2260E-03	CBS recirc w/o cooling - train B - all support avail
XB2	6.3520E-03	CBS recirc w/o cooling - train B single train.
XB3	1.9870E-02	CBS FOR LT2Y (START & RUN)
XBA	2.6070E-02	CBS recirc w/o cooling - train B after train A fails.
XBB	5.9440E-02	CBS FOR LT2Y (START & RUN)
XBF	1.0000E+00	CBS recirculation - train B - guaranteed failure.
XC1	1.9870E-02	CBS recirc - train A - LT1, LT2 all support avail.
XC2	1.9870E-02	CBS recirc - train A single train.
XCF	1.0000E+00	CBS recirc - train A - guaranteed failure.
XD1	0.0000E+00	CBS recirc - train B - LT1, LT2 all support avail.
XD2	1.9870E-02	CBS recirc - train B single train.
XDA	5.9440E-02	CBS recirc - train B after train A fails.
XDF	1.0000E+00	CBS recirc - train B - guaranteed failure.
ZA1	4.8780E-03	Contnmnt sump isol. - Train A LLOCA - all support.
ZA2	4.8780E-03	Contnmnt sump isol. - Train A LLOCA - one train.
ZA3	4.3170E-03	Contnmnt sump isol. - Train A SLOCA - all support.
ZA4	4.3170E-03	Contnmnt sump isol. - Train A SLOCA - one train.
ZAF	1.0000E+00	Contnmnt sump isol. - Train A - guaranteed failure.
ZB1	4.5490E-03	Contnmnt sump isol. - Train B LLOCA - all support.
ZB2	4.8780E-03	Contnmnt sump isol. - Train B LLOCA - one train.
ZB3	3.9880E-03	Contnmnt sump isol. - Train B SLOCA - all support.
ZB4	4.3170E-03	Contnmnt sump isol. - Train B SLOCA - one train.
ZBA	7.2060E-02	Contnmnt sump isol. - Train B after A fails LLOCA.
ZBB	8.0150E-02	Contnmnt sump isol. - Train B after A fails SLOCA.
ZBF	1.0000E+00	Contnmnt sump isol. - Train B - guaranteed failure.

TABLE 3.4-4(b)

Containment Event Tree Split Fractions

Split Fraction	Unavailability	Description
CC1	1.0000E-03	DEBRIS COOLED - CAVITY FLOODED, DEBRIS DISPERSED
CC2	1.0000E+00	CAVITY DRY DEBRIS GUAR. NOT TO COOL 1D/5D/3D/7D
CCA	1.0000E-02	DEBRIS COOLED - CAVITY FLOODED, DEBRIS NOT DISPERSED
CD2	5.0000E-01	3,4,7 & 8 STATES DEBRIS DISPERSED HIGH PRESS.
CDF	1.0000E+00	1,2 5 & 6 DEBRIS DISPERSED GF LOW PRESS.
CHF	1.0000E+00	DIRECT CONTAINMENT HEATING - PDS 3,4,7,8 AND CD=S DISPERSED
CHS	0.0000E+00	DIRECT CONTAINMENT HEATING - LOW PRESS G.S.
CN1	1.0000E-06	2A/6A CONTAINMENT INTACT / DEBRIS IN CAVITY
CN2	1.0000E-05	2C/6C 2D/6D 2A/6A CONTAINMENT INTACT
CN3	5.5000E-05	3D/7D CONTAINMENT INTACT
CN4	1.0000E-06	4A/8A CONTAINMENT INTACT
CN5	1.0000E-06	4C/8C CONTAINMENT INTACT
CN6	1.0000E-05	4D/8D CONTAINMENT INTACT
CN7	2.5000E-05	1D/5D CONTAINMENT INTACT
CNA	6.3500E-05	2A/6A CONTAINMENT INTACT CH FAILED
CNB	1.0000E-04	2C/6C 2D/6D CONTAINMENT INTACT CH FAILED
CNC	2.6000E-04	3D/7D CONTAINMENT INTACT CH FAILED
CND	1.0000E-03	3D/7D CONTAINMENT INTACT CD SUCCESS, CH FAILED
CNE	3.7000E-04	4A/8A CONTAINMENT INTACT CH FAILED
CNG	1.0000E-03	4A/8A CONTAINMENT INTACT CD SUCCESS, CH FAILED
CNH	2.7000E-04	4C/8C CONTAINMENT INTACT CH FAILED
CNI	1.0000E-03	4C/8C CONTAINMENT INTACT CD SUCCESS, CH FAILED
CNJ	2.5000E-04	4D/8D CONTAINMENT INTACT CH FAILED
CNK	1.0000E-03	4D/8D CONTAINMENT INTACT CD SUCCESS, CH FAILED
CNL	1.2000E-04	1D/5D CONTAINMENT INTACT CH FAILED
CY1	1.8000E-02	1D/5D NO PREVIOUS BURN
CY2	1.2000E-01	2A/6A NO PREVIOUS BURN
CY3	1.1000E-01	2C/6C 2D/6D NO PREVIOUS BURN
CY4	3.5000E-01	4A/8A NO PREVIOUS BURN
CY5	2.5000E-01	4C/8C NO PREVIOUS BURN
CY6	9.0000E-02	4D/8D NO PREVIOUS BURN
CY7	1.0000E-01	3D/7D NO PREVIOUS BURN
CYA	1.0000E-02	PREVIOUS BURN (VH=F) ALL PDS STATES
CYF	1.0000E+00	DEBRIS DISPERSED GUARANTEED H2 BURN
DP1	1.2200E-02	OPERATOR DEPRESSURIZES RCS - HIGH PRESS. 2 TRAINS OF DC
DP2	1.0000E-01	OPERATOR DEPRESSURIZES RCS - HIGH PRESS. 1 TRAIN OF DC
DPF	1.0000E+00	OPERATOR DEPRESSURIZES RCS - NO DC FOR PORV'S
DPS	0.0000E+00	OPERATOR DEPRESSURIZES RCS - LOW PRESS. SEQUENCES
HL1	6.0000E-01	HOT LEG FAILURE - HIGH PRESS. DP FAILURE
HLF	1.0000E+00	HOT LEG FAILURE - LOW PRESS. PDS - RCS ALREADY DEPRESS.

TABLE 3.4-4(b)

Containment Event Tree Split Fractions

Split Fraction	Unavailability	Description
IS1	7.0000E-03	INDUCED S/G TUBE RUPTURE - HL=F & OP=F AND HIGH PRESS
ISS	0.0000E+00	NO INDUCED S/G TUBE RUPTURE - HL=S OR OP=S OR LOW PRESS.
LH1	3.2000E-01	2A/6A 4A/8A LATE H2 BURN
LH2	1.0000E-02	PDS 1-8 C&D LATE H2 BURN
LHA	1.0000E-02	2A/6A 4A/8A LATE H2 BURN CH OR VH FAILED
LHB	7.0000E-01	2A/6A 4A/8A LATE H2 BURN CC FAILED
LHC	1.0000E-03	PDS 1-8 C&D LATE H2 BURN CH OR VH FAILED
LHD	1.0000E-02	PDS 1-8 C&D LATE H2 BURN CC FAILED
LM1	3.0000E-01	PDS 2,4,6&8 A,C&D BASEMAT INTACT
LM2	1.0000E+00	3D/7D BASEMAT INTACT (GF-DEBRIS NOT COOLED)
LMA	1.0000E+00	PDS 2,4,6&8 A,C&D BASEMAT INTACT CD FAILED
LS1	1.0000E-06	2A/6A SHELL INTACT
LS2	1.0000E+00	1,2,5,6 C,D SHELL INTACT DEFAULT & (CC S, LH F)
LS3	8.9000E-01	3D/7D SHELL INTACT (CC F,LH S) CC GF FOR 3D/7D
LS4	1.0000E-06	4A/8A SHELL INTACT
LS5	1.0000E+00	4C/8C 4D/8D SHELL INTACT DEFAULT & (CC S, LH F)
LSA	4.5000E-05	2A/6A SHELL INTACT CC SUCCESS, LH FAILED
LSB	5.0000E-06	2A/6A SHELL INTACT CC FAILED, LH SUCCESS
LSC	2.2000E-04	2A/6A SHELL INTACT CC FAILED, LH FAILED
LSD	9.9000E-01	2C/6C 2D/6D SHELL INTACT (CC F, LH S) & (CC F, LH F)
LSE	9.8000E-01	3D/7D SHELL INTACT (CC F, LH F)
LSG	1.3000E-04	4A/8A SHELL INTACT CC SUCCESS, LH FAILED
LSH	5.0000E-06	4A/8A SHELL INTACT CC FAILED, LH SUCCESS
LSI	6.5000E-04	4A/8A SHELL INTACT CC FAILED, LH FAILED
LSJ	9.9000E-01	4C/8C 4D/8D SHELL INTACT (CC F, LH S) OR (CC F, LH F)
LSK	6.7000E-01	1D/5D SHELL UNTACT (CC F, LH S)
LSL	7.5000E-01	1D/5D SHELL INTACT (CC F, LH F)
R1F	1.0000E+00	DIRECT MAP TO S1B
R1S	0.0000E+00	NO MAPPING TO S1B
R2F	1.0000E+00	DIRECT MAP TO S2
R2S	0.0000E+00	NO MAPPING TO S2
R3F	1.0000E+00	DIRECT MAP TO S6
R3S	0.0000E+00	NO MAPPING TO S6
R4F	1.0000E+00	DIRECT MAP TO S7A
R4S	0.0000E+00	NO MAPPING TO S7A
R5F	1.0000E+00	DIRECT MAP TO S7B
R5S	0.0000E+00	NO MAPPING TO S7B
REF	1.0000E+00	OPERATOR RECOVERS CONTAINMENT COOLING
SCF	1.0000E+00	CORE DAMAGE STATE - PROCESS THROUGH CONTAINMENT TREE
SCS	0.0000E+00	PLANT SUCCESS SEQUENCE - NO CORE MELT

TABLE 3.4-4(b)

Containment Event Tree Split Fractions

Split Fraction	Unavailability	Description
SM1	1.0000E-02	2A/6A SMALL LEAK
SM2	1.0000E-02	2C/6C SMALL LEAK
SM3	2.0000E-02	3D/7D SMALL LEAK
SM4	2.0000E-02	4A/8A SMALL LEAK
SM5	2.0000E-02	4C/8C 4D/8D SMALL LEAK
SMA	1.0000E-02	2A/6A SMALL LEAK LS FAILED
SMB	2.4000E-01	2C/6C 2D/6D SMALL LEAK LS FAILED
SMC	1.2000E-01	3D/7D SMALL LEAK LS FAILED
SMD	2.0000E-02	4A/8A SMALL LEAK LS FAILED
SME	2.4000E-01	4C/8C 4D/8D SMALL LEAK, LS FAILED
SMF	1.0000E+00	SMALL LEAK - G. F. DUE TO DCH
VDF	1.0000E+00	DEBRIS NOT COOLED IN VESSEL
VH0	0.0000E+00	NO H2 BURN IF DEBRIS COOLED IN VESSEL
VHA	3.5000E-01	2A/6A NO H2 BURN DEBRIS NOT COOLED IN VESSEL (VD)
VHB	2.4000E-01	2C/6C 2D/6D NO H2 DEBRIS NOT COOLED IN VESSEL
VHC	2.0000E-02	3D/7D NO H2 BURN DEBRIS NOT COOLED IN VESSEL
VHD	4.0000E-03	4A/8A NO H2 BURN DEBRIS NOT COOLED IN VESSEL
VHE	3.3000E-03	4C/8C NO H2 BURN DEBRIS NOT COOLED IN VESSEL
VHG	3.0000E-04	4D/8D NO H2 BURN DEBRIS NOT COOLED IN VESSEL
VHH	3.7000E-02	1D/5D NO H2 BURN DEBRIS NOT COOLED IN VESSEL
VI1	1.0000E-04	2A/6A 2C/6C 2D/6D CONT. INTACT NO H2 BURN
VI3	1.0000E-06	3D/7D 4A/8A 4C/8C 4D/8D CONT INTACT NO H2 BURN
VIA	2.7500E-04	2A/6A CONT INTACT H2 BURN
VIB	2.3000E-04	2C/6C 2D/6D CONT INTACT H2 BURN
VIC	1.1000E-05	3D/7D CONT INTACT H2 BURN
VID	8.0000E-05	4A/8A CONT INTACT H2 BURN
VIE	9.0000E-05	4C/8C CONT INTACT H2 BURN
VIG	1.1000E-05	4D/8D CONT INTACT H2 BURN

TABLE 3.4-5

Release Category Definition

Classification	Release Category	Containment Failure Mode	Frequency
*Early Large Containment Failure/Bypass			TOTAL = 2.25E-07 (0.2%)
	S1A	• Direct containment heating.	11.1%
		• Steam/hydrogen explosion.	0.8%
	S1B	• Aircraft crash into containment.	0.3%
		• Turbine missile impact into containment.	0.2%
	S6	• Large containment isolation failure (i.e., purge valves).	54.2%
		• Large, pre-existing leaks	4.5%
	S7A	• Induced steam generator tube rupture.	26.8%
		• V-sequence - pipe failure (dry).	2.1%
Early Small Containment Failure/Bypass			TOTAL = 1.60E-05(14.2%)
	S7B	• SGTR with steam leak.	1.0%
		• V-sequence - RHR pump seal failure (wet).	0.2%
	S2	• Early small leak (<3" dia.) late overpressurization.	96.2%
		• Small, pre-existing leaks	2.6%
Late Containment Failure			TOTAL = 7.34E-05(65.4%)
	S3A	• Late overpressurization with dry containment.	39.0%
	S3B	• Late overpressurization with wet containment.	56.3%
	S4	• Late basemat melt through.	4.8%
Containment Intact	S5	• Intact	2.27E-05 (20.2%)
			TOTAL 1.12E-04 (100%)

* Classification used to define "unusually poor" containment performance.

TABLE 3.4-6

List of Dominant Functional Sequences

	Dominant Functional Sequences				
	Initiating Event[a]	Support Function Failure[b]	Mainline Function Failure[c]	Sequence Type	Core Damage Contribution
1	LOSP	On-Site AC	RCS Integrity (GF) * HPI (GF)	Station Blackout/Seal LOCA	35.1%
2	GT	Comp. Cooling	RCS Integrity (GF) * HPI (GF)	Transient/ Seal LOCA	34.3%
3	GT	--	SC * PC	Transient/ Feed and Bleed	12.1%
4	GT	--	Criticality* RCS Integrity	ATWS	9.0%
5	LOCA	--	RCS Integrity (GF) * LP	LOCA	7.1%
				TOTAL	**97.6%**

GF = Guaranteed failure (due to initiating event or support function failure).

(a) Initiating Events (includes External Hazards):

LOSP - Loss of off-site power from all causes
GT - General transient (i.e., reactor trip, turbine trip, loss of feedwater, etc.)
LOCA - Loss-of-coolant accident

(b) Support Functions:

On-Site AC - Diesel generators and support systems
Comp. Cooling - Primary Component Coolant Water System (PCC) and support systems (i.e., AC power, Service Water)

(c) Mainline Functions:

RCS Integrity - No primary system leakage greater than normal makeup
HPI - High pressure injection (charging, SI pumps)
LP - Low pressure injection and recirculation (RHR)
SC - Secondary cooling (Emergency Feedwater, atmospheric relief valves)
PC - Primary cooling - feed and bleed cooling (HPI and PORVs)
Criticality - Control rods inserted into core

TABLE 3.4-7

Initiating Event Contributions to Core Damage

Initiating Event Type	Percentage of Total Core Damage		
	Total	Internal Events	External Events
Transients	83.0%	42.1%	40.9%
- General Transient	- 19.0%	- 19.0%	- 0.0%
- LOSP	- 39.5%	- 15.8%	- 23.7%
- Loss of Support Systems	- 24.5%	- 7.3%	- 17.2%
LOCA	8.0%	6.6%	1.4%
ATWS	9.0%	5.9%	3.1%
TOTAL	**100.0%**	**54.6%**	**45.4%**

TABLE 3.4-8(a)

Internal Initiating Events - Contribution to Core Damage Total

Internal Initiating Event	Percentage of Total Core Damage Frequency	Total
1. General Transients		17.4%
• Reactor Trip	5.7%	
• Loss of Feedwater	3.8%	
• Turbine Trip	4.5%	
• Others	3.4%	
2. Loss of Off-Site Power (LOSP)		15.8%
• LOSP Based on Industry Data	9.8%	
• LOSP Due to Fault on SF_6 System	6.0%	
3. Support System Failure		7.3%
• Single Train PCC Failure	3.8%	
• Single Train SW Failure	0.4%	
• Single Train DC Power Failure	3.1%	
4. Loss-of-Coolant Accident (LOCA)		6.6%
• Small LOCA (Break <2" Diameter)	3.1%	
• Medium/Large LOCA (Break >2" Diameter)	2.3%	
• Steam Generator Tube Rupture	1.2%	
• V-Sequence (Interfacing Systems LOCA)	0.03%	
5. Anticipated Transient Without Scram (ATWS)		5.9%
6. Others		1.6%
• Steam Line Break	1.6%	
Internal Initiating Events - TOTAL	**54.6%**	**54.6%**

TABLE 3.4-8(b)

External Initiating Events - Contribution to Core Damage Total

External Initiating Event	Percentage of Total Core Damage Frequency	Total
1. Fire		24.4%
• Fire in Control Room Causing Loss of Support Systems (AC power, PCC, SW)	9.4%	
• Fire in Turbine Building - Loss of Off-Site Power	8.3%	
• Fire in PCC Pump Area	2.8%	
• Fire in Electrical Tunnels	2.1%	
• Fire in Cable Spreading Room - Loss of Support Systems	1.8%	
2. Seismic Event		13.4%
• Seismic-Initiated Station Blackout	8.9%	
• Seismic-Initiated ATWS	3.1%	
• Seismic-Initiated Large LOCA	1.4%	
3. Flood		6.2%
• Internal Flood in Turbine Building - LOSP	5.2%	
• External Flood Causing SW Failure	1.0%	
4. Transportation		1.3%
• Truck Crash into SF_6 Lines - LOSP	1.3%	
5. Others		0.1%
• Aircraft Crash - Loss of PCC	0.1%	
• All Others	<<0.1%	
External Initiating Events - TOTAL	**45.4%**	**45.4%**

TABLE 3.4-9

System Importance Ranking*

System Failure (Top Event)	Ranking	Percent of Core Damage Frequency
Diesel Generators (GA, GB)	1	27.5%
Primary Component Cooling Water (PA, PB)	2	17.5%
Service Water (WA, WB)	3	15.7%
Emergency Feedwater (EF)	4	14.8%
Residual Heat Removal (LA, LB; L1, L2; L5, L6)	5	3.8%
Emergency Safeguards Features Actuation System (EA, EB)	6	3.2%
Solid State Protection System (SA, SB)	7	2.6%
Emergency Air Handling (EH)	8	1.4%
Off-Site Power (OG)	9	1.2%
High Pressure Injection (HP)	10	0.4%

---

* System importance ranking is based on the percentage of the total core damage sequence frequency in which a given system (or train of the system) has failed. This includes only the failures that are those internal to the system -- i.e., component failure and maintenance unavailability. System failure due to support system failures or due to external events (e.g., fires, floods) is excluded.

TABLE 3.4-10

Operator Action Importance Ranking Measures

Operator Action Failure (Top Event)	Risk* Importance	Risk** Achievement
Electric Power Recovery (ER)	26.4%	5.7
Signal Recovery (OS)	6.0%	6.5
EFW Recovery (FR)	4.0%	2.4
Manual Reactor Shutdown (OH)	2.3%	4.6
SGTR - Control Break Flow (O4), Depressurize (O5)	2.2%	1.1
Feed and Bleed (OR)	1.6%	1.6
Makeup to the RWST (RM)	1.0%	1.1
Switchover to High Pressure Recirculation (O3)	0.6%	6.1
Depressurize - SBO (OD)	0.3%	1.0
Control EFW (OM)	<0.1%	1.0

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* Risk Importance is the percentage of the total core damage sequence frequency in which a given operator action has failed.

** Risk Achievement Worth is the increase in core damage frequency total with the operator action assumed to be failed.

TABLE 3.4-11

Initiating Event Contribution to Core Damage Sequence With Early, Large Containment Failure/Bypass

Initiating Event Type	Percentage Contribution to Early, Large Containment Failure/Bypass
Transients	86.4%
- General transients	- 62.1%
- LOSP	- 11.9%
- Loss of Support Systems	- 12.4%
LOCA	9.1%
ATWS	3.5%
TOTAL (2.25E-07 per year)	100.0%

TABLE 3.4-12

Top Event Importance Ranking for Containment Performance

Top Event	Description	Rank	Percent Contribution to Early, Large Containment Failure/Bypass
OS	Operator failure to recover signals	1	67.6%
C2	Containment isolation failure - purge valves	2	60.3%
EA, EB	ESFAS failure	3	51.2%
HL	No RCS hot leg rupture	4	37.5%
EF	Emergency Feedwater System failure	5	28.6%
IS	Induced steam generator tube rupture	6	25.9%
SA, SB	SSPS failure	7	18.0%
CN	Containment structural failure	8	11.6%
ER	Electric power recovery	9	10.2%
GA, GB	Diesel generator failure	10	9.6%

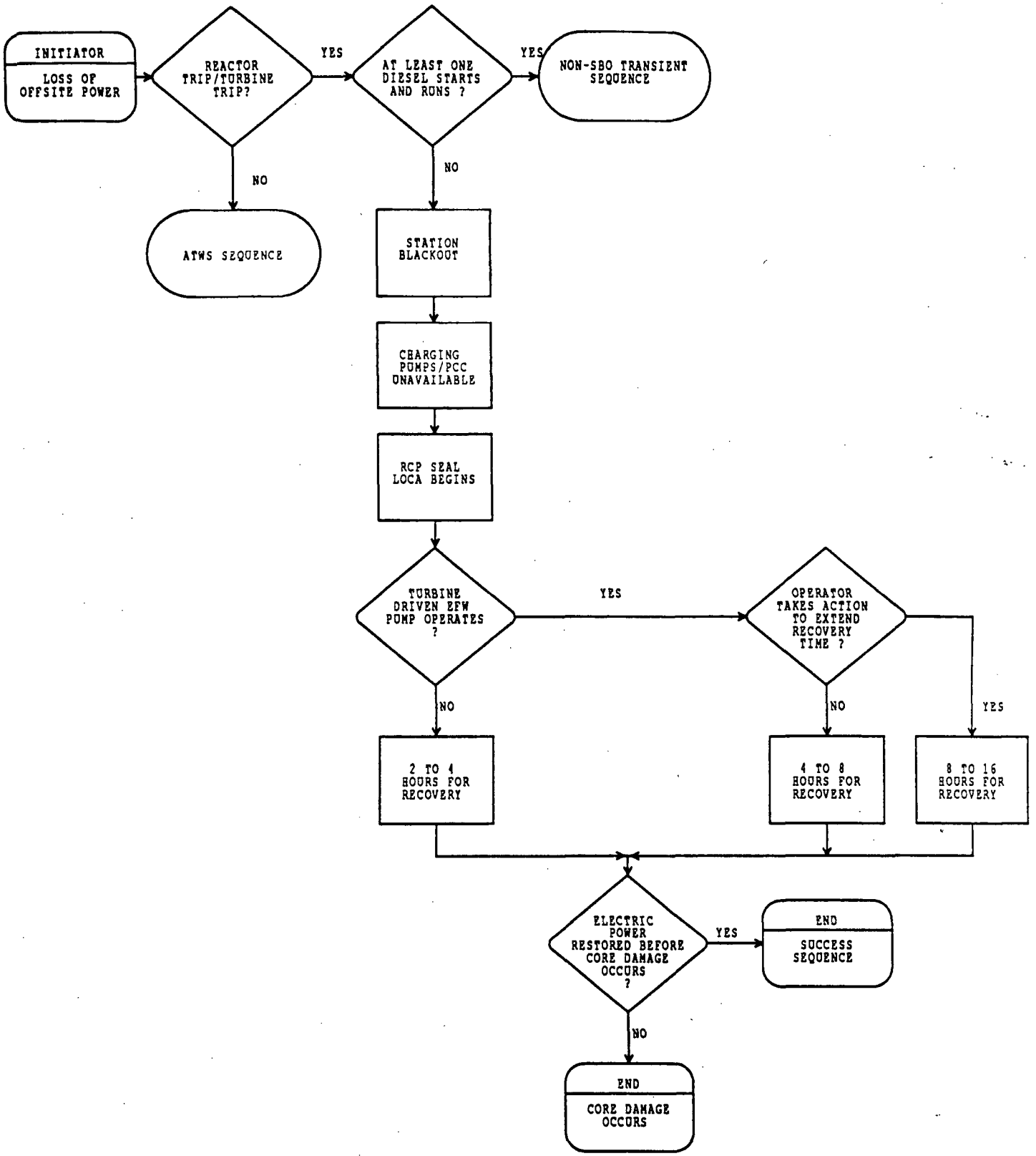


Figure 3.4-1

Station Blackout (SBO) Event Sequence Diagram

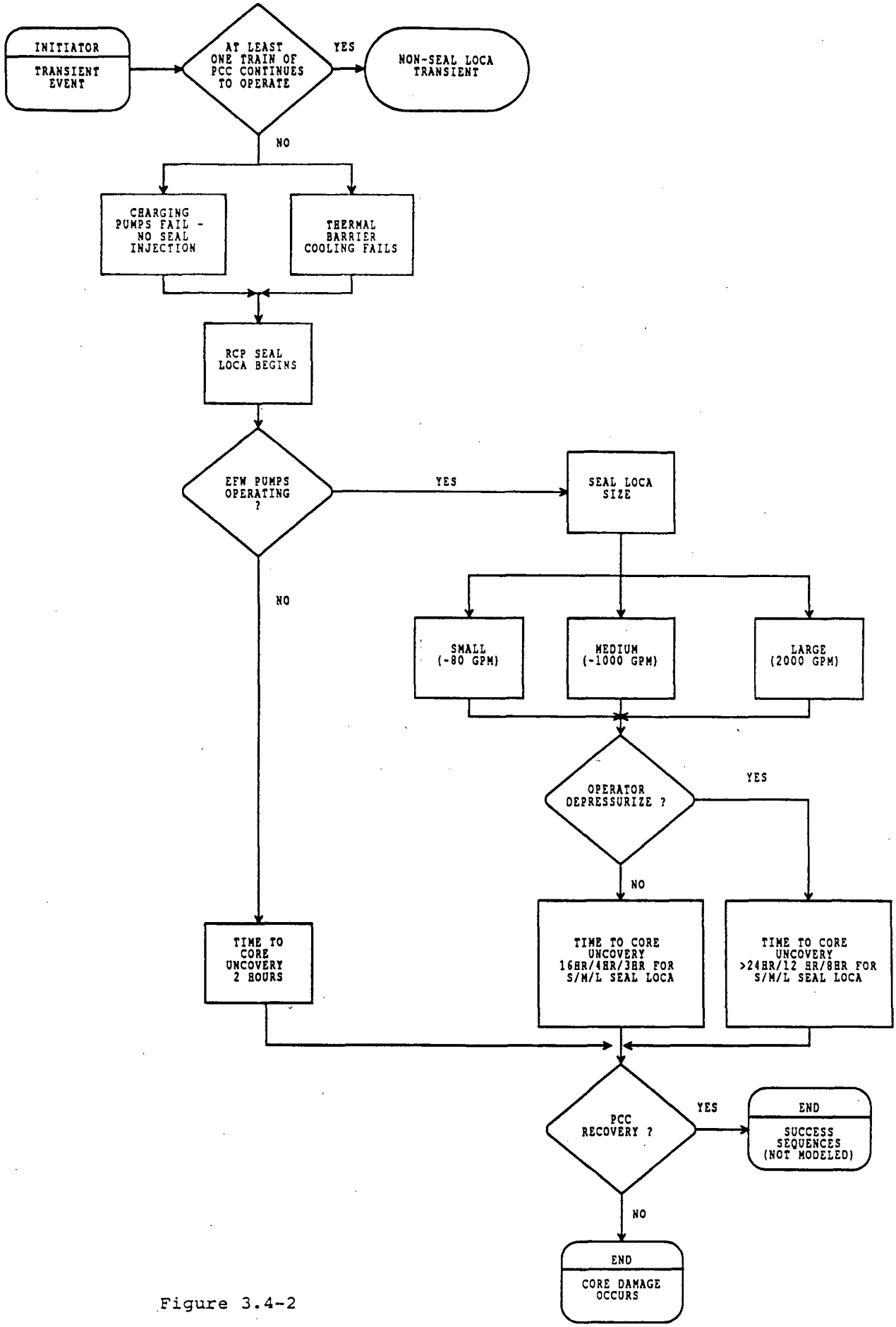


Figure 3.4-2

Seal LOCA Event Sequence Diagram

4.0 BACK-END ANALYSIS

The back-end or Level II analysis addresses the physical progression of accident sequences from the time of core damage to the point of release of radionuclides from the containment. This section summarizes the current Level II model and results as documented in the SSPSS-1990 (Reference 8). The current model is based extensively on the core and containment response analysis from the original SSPSA, Section 11.0, and Appendix H (Reference 3). The scope of the SSPSA was to:

- Define the Plant Damage States (PDS), as an interface between Level I and Level II models.
- Develop the structure for a Containment Event Tree (CET) and quantify it for each plant damage state.
- Determine containment failure modes and time-dependent failure probability distributions.
- Define radionuclide release categories as an interface between Level II and Level III analysis.
- Quantify the C matrix, where the matrix elements C_{ij} express the conditional probability that Plant Damage State i will result in Release Category j.
- Determine the uncertainty in the containment response quantification for dominant sequences.

The end product of these tasks was a characterization of the impact of each plant state on the mode, timing, and magnitude of radionuclides released from the plant. The conclusions of the analysis in the SSPSA were that:

- The Seabrook containment is by far the strongest containment analyzed in any PRA to date (1983).
- Due to the high containment failure pressure, the time to containment failure is very long.

- Accident sequences without containment heat removal have the shortest time to containment failure for a wet cavity condition (i.e., Refueling Water Storage Tank (RWST) injected).

- Uncertainties associated with the analysis methods then available for the radionuclide source terms have been found to be extremely large, yielding very conservative point estimates.

Subsequent studies (References 10, 11, and 15) were performed to better understand the early containment failure/bypass mechanisms and to more realistically estimate the resultant source terms. (See Appendix B for a summary of each study.) These studies served to extend and refine the present state of knowledge in areas including LOCA outside containment, induced steam generator tube rupture, and direct containment heating. These additional studies, however, did not change the basic Level II conclusions of the SSPSA.

In addition to enhanced analyses, the computer software used to quantify the containment event tree has also been enhanced. The present computer model links sequences from initiator through the plant model and then through the containment event tree without the use of bins (so-called Plant Damage States). The logic used to create the Plant Damage States is now used in the software as logic rules.

The enhanced analyses and software have been integrated with the original analysis in the SSPSA to produce the model described in this report and documented in the SSPSS-1990.

4.1 Plant Data and Plant Description

The Seabrook Unit 1 containment can be described as a large, dry PWR containment. It is a Seismic Category 1 reinforced concrete structure in the form of a right vertical cylinder with a hemispherical dome and flat foundation mat founded on bedrock. The inside face is lined with a welded carbon steel plate, providing a high degree of leak tightness. A protective 4' thick concrete mat, which forms the floor of the containment, protects the liner over the foundation mat. The containment structure provides biological shielding for normal and accident conditions. The approximate dimensions of the containment are:

Inside Diameter	140'
Inside Height	219'
Vertical Wall Thickness	4'-6"
Dome Thickness	3'-6"
Foundation Mat Thickness	10'

Containment penetrations are provided in the lower portion of the structure and consist of a personnel lock and an equipment hatch/personnel lock, a fuel transfer tube and piping, electrical, instrumentation, and ventilation penetrations.

The containment is designed to withstand all credible conditions of loading, including normal loads, construction loads, test loads, severe environmental loads, and extreme environmental and abnormal loads. The maximum design pressure is 52 psig. The maximum liner temperature associated with the design pressure response is 217°F.

The containment enclosure, surrounding the containment, is designed in a similar configuration as a vertical right cylindrical Seismic Category 1, reinforced concrete structure with dome and ring base. The approximate dimensions of the structure are: inside diameter -158', vertical wall thickness - varies from 1'-3" to 3', and dome thickness - 1'-3".

The containment enclosure is designed to entrap, filter, and then discharge any leakage from the containment structure. To accomplish this, the space between the containment enclosure and the containment structure, as well as the penetration and safeguards pump areas, are maintained at a negative pressure following a LOCA by fans which take suction from the containment enclosure and exhaust to atmosphere through charcoal filters. Leakage through all joints and penetrations has been minimized.

A Containment Building Spray (CBS) System is utilized for post-accident containment heat removal. The CBS is designed to spray water containing boron and sodium hydroxide into the containment atmosphere after a LOCA to cool it and remove iodine. The pumps initially take suction from the RWST and discharge into the containment atmosphere through the spray headers located in the containment dome. After a prescribed amount of water is removed from the tank, the pump suction is transferred to the containment sump, and cooling is continued by recirculating sump water through the spray heat exchangers and back through the spray headers. The spray is actuated by a containment spray actuation signal which is generated at a designated containment pressure. The system is completely redundant and can withstand any single active failure.

The Containment Isolation System establishes and/or maintains isolation of the containment from the outside environment in order to prevent the release of fission products. Automatic trip isolation signals actuate the appropriate valves to a closed position whenever safety injection occurs or high containment pressure is experienced. Double barrier protection is provided for all lines that penetrate the containment boundary.

In performing the core and containment response analyses, comparisons were made between the Seabrook Station design and other reference designs (i.e, Indian Point and Zion). The design comparisons are documented in Table 11.2-1 of the SSPSA and Table 4-5 of PLG-0432. Based on these comparisons, the following important features and distinctions of the Seabrook design were identified:

1. The NSSS and Containment Building are of the same general configuration, type, and size as that for the reference plants.

2. The intermediate floors in the Containment Building are largely grated and good mixing paths for the containment atmosphere appear to exist for all regions of the containment. Thus, the potential for localized hydrogen combustion, discussed in Generic Letter 88-20, Supplement 3, is not significant.

3. The reactor cavity and instrument tunnel configuration is somewhat different in detail from the reference plant configurations, but these differences are not significant with respect to debris behavior following release from the vessel.

4. The Seabrook design does not include safety-related fan coolers. Also, the nonsafety-related fan coolers are not of sufficient capacity for containment heat removal under accident conditions. The containment heat removal function is integrated into the Containment Spray System design by including a separate spray heat exchanger that is not typically present in other designs.

5. Another difference is related to the height of the curb on the containment floor surrounding the reactor cavity. The Seabrook design includes a 30" high curb, while in the comparison plants, the curb height is only 6" high. This means that in the Seabrook design, most of the RWST contents must be injected before water will spillover into the cavity. In the comparison plant designs, only a small fraction of the RWST must be injected before flooding of the cavity occurs. However, for both Seabrook and the comparison plants, the reactor cavity is full of water (wet) with RWST injection and is dry without RWST injection.

6. A comparison of the base mat concrete composition indicates another important design difference. The Seabrook Station uses a concrete aggregate of a basaltic composition, while in the comparison plants, the concrete aggregate is a limestone composition. During core concrete interaction, the limestone concrete generates noncondensible carbon dioxide and carbon monoxide, while the basaltic aggregate does not.

7. The Seabrook Station design includes a Containment Enclosure Building (or secondary containment), while the other designs do not. This additional containment structure has the potential to provide additional attenuation of radionuclide releases. However, its value is limited in containment pressure challenges because of its low capacity in comparison to the primary containment.

8. The Auxiliary Building of the Seabrook Station was compared to that of the Zion plant. The Auxiliary Building features are important for the radionuclide release path characteristics in the V-sequence (LOCA outside containment through RHR System).

   This release is expected to occur in the portion of the Auxiliary Building containing the RHR System (i.e., RHR equipment vaults). In the Seabrook design, the RHR cubicles have no openings in the lower 30'. Therefore, a deep pool of water would cover the most likely release site for radionuclides. In the reference designs, there is no potential for this beneficial flooding of the location where radionuclides are expected to be released.

9. The design pressure of the RHR piping outside containment is sufficiently high (600 psi) that it is unlikely to fail at RCS pressures. In the postulated LOCA outside containment scenario, the most likely failure location in the RHR System is the RHR pump seals, located near the bottom of the equipment vault. This feature, in conjunction with the equipment vault configuration discussed above, results in a release through a deep pool of water.

10. The final and most significant feature is related to the inherent containment failure characteristics. The Seabrook containment design has a high pressure capacity relative to the comparison plants. This high containment failure pressure results in a very long time to containment failure if the containment does not fail from the initial blowdown pressure spike.

4.2 Plant Models and Methods for Physical Processes

In the assessment of degraded core accidents, a wide spectrum of potential accident scenarios must be considered, which requires an assessment of many different physical phenomena. As discussed in Section 4.1, the Seabrook design is very similar to the Indian Point and Zion plants. Therefore, many of the physical processes considered in the analyses of these plants (References 31 and 32) are applicable to the Seabrook design. As a result, the Seabrook back-end methodology utilized is very similar to that used in the Zion and Indian Point studies. Where appropriate, when additional experimental and analytical results have become available since these studies were completed (References 31 and 32), an update of the physical processes was provided. Areas of particular importance are debris bed cooling, debris dispersal, and in the physical processes associated with core debris in a dry reactor cavity. A detailed description of the physical processes considered can be found in Appendix H.2.1 of Reference 3.

A description of the computer code used to perform severe accident analysis is provided in Appendix H.2.2 of Reference 3. The computer codes include the MARCH (Reference 34), COCOCLASS9 (Reference 35), MODMESH, and CORCON-MOD1 (Reference 36). MARCH was used to model the core and primary system transient behavior and to obtain mass and energy releases from the primary system until vessel failure. COCOCLASS9 is a modified version of the Westinghouse COCO computer code. This code utilizes the mass and energy releases computed by MARCH to model the Containment Building pressurization and hydrogen burn phenomena. Energetic interaction of molten debris and reactor cavity water at the time of vessel failure is modelled with the MODMESH computer program. A modified CORCON-MOD1 code was used to replace the MARCH subroutine INTER. CORCON models the core-concrete interaction after dryout in the reactor cavity occurs.

The input parameters used in the containment analyses are contained in Tables 2.2.2-1 and 2.2.2-2 of Appendix H.2.2 of Reference 3. These tables provide the initial conditions and the containment design data used in the analysis.

4.3 Bins and Damage States

In the SSPSA, accident scenarios from Level I analysis were grouped into Plant Damage States. Each Plant Damage State collects or bins all those sequences for which the core melt progression, the release of fission products from the fuel, the containment

environment, the source-term mitigation, and the containment states are similar. For each Plant Damage State, the CET was quantified separately. This was accomplished by determining the dominant sequences for each Plant Damage State.

The SSPSA defined 39 possible Plant Damage States as shown in Table 4-1 of Reference 3. In Reference 11, "Risk Management and Emergency Planning Study" (RMEPS), three additional damage states were added (1 FV, 1 FPV, 7 FPV) for V-sequence scenarios. The two FPV states represent conditions where the RHR pump seals are submerged, and therefore, the releases are filtered. The 1 FV states represents a release path resulting from a RHR System pipe failure.

In the current SSPSS, the ability to directly link the Level I sequences to the CET is possible. Plant Damage States are now used in the form of logic rules that determine which split fraction is used for each top event in the CET. In addition, these logic rules have been expanded to include availability of various safety equipment. An example of this is the Emergency Feedwater System, which is important in determining the consequences of an Induced Steam Generator Tube Rupture (ISGTR). The availability of this system can be retrieved directly from the accident sequence definitions and used in the CET quantification.

The final binning of results occurs between the containment model and the site consequence model. This is accomplished by defining Release Categories. The Release Categories group sequences considering three basic aspects which impact source terms. These aspects are: the containment failure mode, the availability of the spray system for radionuclide scrubbing, and whether the cavity is wet or dry for the core concrete reaction. Definitions of the Release Categories used in the SSPSS are given in Section 4.7.

4.4 Containment Failure Characterization

The Seabrook containment structure was subject to an extensive structural analysis as part of the original SSPSA in order to understand the internal pressure at which the containment is realistically expected to fail, the location of the failure, and the size of the leak. This analysis yielded a median failure pressure for the primary containment of 187 psia for dry containment sequences and 210 psia for wet containment sequences. The secondary containment (Enclosure Building) was analyzed to fail at the same time the primary containment fails during a severe accident. The following structural failure modes were considered in the SSPSA:

- Membrane Failure
- Base Slab Failure
- Shear Failure at Wall/Base mat Intersection
- Penetrations
- Liner Buckling
- Fuel Transfer Tube

The structural analysis identified failures that were subsequently characterized as one of three types: Type A, Type B, or Type C. Type A failures are small failures that do not arrest the pressure increase in the containment and eventually progress to Type B or Type C failures. The Type A failures do not increase in size as the containment pressure increases until ultimate failure. These small failures have an upper bound size that results in an increase of containment leakage to the environment from 0.1% volume/day to 40% volume/day. Type B failures are larger failures that are self-limiting. The failure is of sufficient size that it is capable of stopping the pressure increase, but the containment does not blowdown catastrophically. A number of Type B locations were identified and principally involve piping penetrations. Type C failures are gross structural failures that result in a rapid complete blowdown of the containment atmosphere to the environment. The conditional probability that the containment fails in a benign manner (Type B) is 0.88 at 187 psia and 0.76 at 210 psia. Thus, given an overpressurization of containment, the most likely ultimate failure mode is penetration failures rather than gross structural failure.

The lowest failure pressures expected for the primary containment involve penetration failures. A median pressure of approximately 181 psia is expected to result in small leaks (Type A) around penetrations. The lowest major structural failure is expected in the cylindrical wall of the containment resulting from hoop failure at a median pressure of 216 psig. Table 4-2 lists the major failure modes and their median pressures. The results of the structured analysis were combined to generate the conditional containment failure curve (Figure 4-1).

Additional structural analysis was performed as part of the Brookhaven National Laboratory (BNL) review of the EPZ Study (see Appendix A, Section A.2.2, for description of the review). The BNL review recommended a median failure criteria defined as 1% strain. BNL determined a median failure pressure of 158 psia (for dry containment conditions, higher for wet containment). The BNL analysis was suspended at this pressure with the containment structure calculated to be in a general yield condition when continued calculations became extremely numerically intensive.

4.5 Containment Event Tree

The Seabrook Containment Event Tree (CET) is shown in Figure 4-2. The present CET has evolved from the original CET presented in the SSPSA which consisted of 12 top events and 154 sequences. Additional issues identified since the SSPSA have resulted in a new CET that has a total of 19 top events and 989 sequences. The CET structure groups the accident progression into three phases: pre-Reactor Pressure Vessel (RPV) failure, RPV failure through debris quench or dryout, and long-term behavior.

New issues incorporated into the CET include Direct Containment Heating (DCH) and Induced Steam Generator Tube Rupture (ISTGR), identified in Reference 10. The CET also contains top events to address possible alternate success paths, for which no credit has been taken, including:

- In-vessel debris retention after the start of core damage (Top Event VD).

- Recovery of long-term containment heat removal after vessel failure (Top Event RE).

The logic for the fully developed CET is not employed in sequences where the containment has a large initial isolation failure or is bypassed based on the initiating event or failure of CI in the plant model. These accident sequences are passed via a branch (top event RI) directly to the appropriate Release Category.

The top events used in the CET are discussed below:

1. Top Event IE - Initiating Event

   This top event is the entry point (the initiating event) for the Containment Event Tree. This is linked to the plant model for core damage accident sequences. This allows the status of containment isolation, containment water inventory, containment heat removal, timing of reactor pressure vessel failure, and pressure at the time of vessel failure to be referenced via logic rules in the CET. The split fractions defined for each top event are conditional on the plant status and potentially on the result of previous top events.

2. Top Event SC - Success

The software employed in the most recent update requires that all sequences map to an end-state. This top event maps sequences without core melt through to a success end-state.

3. Top Event R1 - Direct Map

This top event is used to provide a direct mapping to a CET end-state for those sequences where the containment is not initially isolated or is bypassed. Sequences with Containment Isolation System failures are mapped directly to Release Category S2 (for small leakage) and S6 (for large penetration openings). The V-sequence (LOCA outside containment) is mapped directly to Release Category S7A (for pipe breaks) and S7B (for pump seal failure).

4. Top Event VD - Debris Cooled In-Vessel

This top event represents those sequences where the accident is terminated with the debris still in the vessel. No credit is taken for this possibility in the SSPSA because prior to evidence from the TMI vessel examination, it was believed that molten debris contact with the lower head instrument tube welds would fail the welds. Possible mechanisms which might be credited include:

- Recovery of in-vessel core cooling after the start of core damage.
- Heat transfer from the debris through the lower head to a water-filled cavity.

5. Top Event DP - Depressurization

This top event represents an operator action to depressurize the RCS using the Pressurizer Power Operated Relief Valves (PORV). The Seabrook Station PORVs are DC powered with an AC powered normally opened block valve upstream. Each PORV is powered from a different train of electric power, and a single PORV is sufficient to accomplish the depressurization. The act of depressurizing reduces the potential for DCH and essentially eliminates the potential for ISGTR. No credit is taken for this action in the current model.

6. Top Event HL - Hot Leg Failure

This top event represents the potential failure of the RCS hot leg (the pressurizer surge line) by thermal creep rupture prior to RPV failure. Seabrook's hot legs are fabricated from stainless steel. Success in the preceding top event (DP=S) precludes the possibility of hot leg failure since the differential pressure is eliminated. Reference 10 evaluated the time/temperature required to fail the hot leg. The study evaluated the potential for surge line failure and concluded that it was extremely unlikely to occur prior to hot leg failure since the temperature of the gas entering the pressurizer is colder than the hot leg fluid.

7. Top Event IS - Induced Steam Generator Tube Rupture (ISGTR)

This top event represents the possibility that the steam generator tubes experience such high temperatures that they fail from thermal creep rupture prior to the hot leg or vessel failing. This top event is dependent on the preceding three top events (DP, HL, VD) failing. In addition, the secondary side must be dry (i.e., secondary cooling failure) or the tubes will never reach the required temperature. Reference 10 concluded that ISGTR events are unlikely even without depressurization (Top Event DP). That study also evaluated several hardware changes to improve the likelihood that secondary cooling is available (see Section 6.0 for evaluations of these changes).

8. Top Event VH - No Early H2 Burn

This top event represents the possibility of an early hydrogen burn. Hydrogen generated by either quenching of the core (VD=S) or from continued metal-water reaction in the damaged core. An early hydrogen burn impacts a number of subsequent top events and increases the probability of early containment failure by increasing both containment pressure and temperature. However, the burn also consumes hydrogen which decreases the probability of a later burn when the containment pressure might be higher.

9. Top Event VI - Containment Intact (Early)

This top event represents the possibility that an early containment failure may result from either the initial primary system blowdown forces or an early hydrogen burn. A detailed evaluation of the accident loads and containment strength was performed to evaluate this event.

10. Top Event CD - Debris Dispersed

This top event represents the possibility that the debris is dispersed from the cavity at vessel failure. This event is true if more than 50% of the core material is relocated to the lower containment floor. The debris can only be dispersed if the vessel is at high pressure. Therefore, this top event is dependent on the top events where the potential for operator depressurization or hot leg failure is evaluated.

11. Top Event CC - Debris Cooled in Cavity

This top event represents the possibility that debris in the cavity is cooled. The debris is unlikely to be cooled if the RWST is not injected. The Seabrook lower compartment configuration includes a 30" high lip around the openings into the cavity. Approximately two-thirds of the RWST contents must be injected before overflowing into the cavity. Also, debris coolability is influenced by the debris depth and particle size. Therefore, this event is dependent on the amount of material ejected from the cavity by the RPV blowdown which is evaluated in the preceding top event.

12. Top Event CH - No Direct Containment Heating (DCH)

This top event represents the possibility of a DCH event occurring during the vessel blowdown. This top event serves as a switch, based on previous top events, to track when a DCH event can occur. The probability of containment failure due to the pressure rise from DCH is included in Top Event CN.

The impact of a DCH event at Seabrook is mitigated by the cavity opening arrangement which does not easily permit the debris to be dispersed above the lower compartment level. The PORV study evaluated the pressure increase resulting from a DCH event and found the peak pressure to within the capacity of the containment. This event is precluded if the RPV pressure at failure is low, the operator depressurizes using the PORV(DP=S), or the hot leg fails (HL=S).

13. Top Event CY - No H2 Burn at Vessel Failure

This top event represents the possibility of a hydrogen burn occurring as the vessel blowdown occurs or as the debris is quenched in the cavity. Hydrogen previously contained in the oxygen depleted vessel is now available in the containment where oxygen is present. The burn may occur at the vessel failure location or a global burn may occur slightly later as the hydrogen generated in the cavity is mixed with the containment atmosphere. An evaluation of the impact of DCH with a simultaneous hydrogen burn showed that the containment structure was unlikely to fail even if a hydrogen burn was forced to occur simultaneously. A previous burn before vessel breach reduces the probability of this burn since the hydrogen concentrations will be lower.

14. Top Event CN - Containment Intact

This top event represents the possibility that the containment will fail simultaneously or slightly after the vessel blowdown. The additional possible pressure loads considered include those arising from the vessel blowdown, debris quench including potential steam explosion, DCH, and hydrogen burns. Not all of the possible loads may occur in a given sequence, based on previous CET top events and plant/containment conditions. The source terms associated with this type of failure could be significant since many of the depletion processes have not been active long enough to have large impact.

15. Top Event RE - Containment Recovery

This top event represents the possibility that the operators could recover containment heat removal capabilities after the vessel has failed. No credit is taken for this possibility at the present time. The recovery action considered most likely is recovery of electric power and therefore, ESF pumping capability. Recovery actions could have unintended negative consequences such as:

- CBS activation could reduce the steam concentration to such an extent as to make the hydrogen mixture combustible.

- Flooding containment, without spraying, could produce additional steam from quenching of the debris in the cavity and possibly the lower compartment. The additional steam will produce a greater pressure load on the containment.

- Injection of water without heat removal could shorten time to failure.

These impacts would be reflected in the choice of split fraction for the remaining top events.

16. Top Event LH - No Late H2 Burn

This top event represents the possibility that the containment will experience a late hydrogen burn. The combustible gas concentration will be impacted by the extent of debris quenching and core-concrete attack. The combustible gas mixture will be primarily hydrogen since little carbon monoxide will be produced from the core attack of the basaltic concrete used in the Seabrook containment structure. The mixture may be inerted by the large amount of steam present in the containment if the CBS System is not operating. Previous burns diminish the probability of a late burn.

17. Top Event LS - Shell Intact (Long-Term)

This top event represents the possibility that the containment structure may fail in the long term. Failure may be a result of a slow pressurization from an uncooled debris bed, lack of containment heat removal (even if sprays operate), or a rapid pressure spike from a hydrogen burn. A detailed analysis of the loads present and the containment structural strength, including the impact of a very hot containment atmosphere, was performed to evaluate this top event.

18. Top Event LM - Basemat Intact (Long-Term)

This top event represents the possibility that the containment may fail from basemat melt-through. This event is dependent on the outcome of Top Event CC, i.e., whether the debris is cooled in cavity since a quenched debris bed will not attack the basemat. An unquenched debris bed does not necessarily penetrate the basemat since the debris may be sufficiently diluted by inclusion of the ablated concrete into the debris pool that the penetration stops.

19. Top Event SM - Small Leak

This top event addresses the possibility that a containment failure may be either a small leak or a gross failure given that a failure has occurred in a prior top event. The top event is evaluated in terms of containment strength/leakage areas and temperature/pressure loads imposed by the accident. Slow pressure increases tend to produce small failures or an increase in leakage rate while the sharp pressure rise associated with combustible gas ignition, vessel blowdown, and direct containment heating are more prone to gross containment failure. The early failures are mapped to different end-states to reflect the difference between a puff or continuous release, while the late failures are combined since the consequence model can not distinguish them once the nearby population is evacuated.

4.6 Accident Progression and CET Quantification

The starting point for the containment analysis is the Plant Damage State (PDS) definitions. The PDS condenses a large number of accident sequences from the plant model which progress to core melt into a manageable (i.e., small) and a well defined set of states. Each PDS contains sequences with similar accident signatures, time to loss of secondary heat removal (if relevant), time to start of core melt, primary system pressure at vessel failure, etc. PDSs with low frequency are conservatively assigned to higher consequence potential dominant PDSs to reduce the number of PDSs requiring evaluation.

The end points for the Containment Event Tree are Release Categories. From the SSPSA and follow on studies, a total of nine different Release Categories have been defined. The CET logic is largely bypassed for a number of PDSs which involve containment isolation failures or bypass sequences and are mapped directly to Release Categories. Each Release Category has associated with it a conservative and best-estimate source term. The logic rules for binning CET sequences into Release Categories is given in Appendix F.2.2.

Each top event probability is represented in the CET by a split fraction. The split fraction value is determined by the PDS and the success or failure of earlier top events. Each split fraction value is based on a review of the relevant physical phenomenon for a

given PDS and may be set to a guaranteed success, if the phenomenon is physically impossible. It should be noted that the split fractions identify the probability that the conditions for a given phenomenon will occur. The phenomenon will always occur if those conditions are present and not occur if they are absent. Therefore, the split fractions do not represent the probability that for a given set of conditions a phenomenon will occur.

The SSPSA and PORV Study (Reference 10) evaluated the split fraction numerical values which are used in the present CET. A complete description of the method employed is given in those two studies. The phenomenon of severe accidents is subject to much uncertainty, hence, much controversy. In order to address these issues, these studies included an examination of the uncertainties in the physical processes.

Appendix F.1.2 lists the split fraction logic rules used in the Containment Event Tree. Each sequence in the CET is evaluated using these split fraction values (see Table 3.4-4(b)) to determine its conditional probability. The split fractions were assigned initial conservative values based on prior studies and the CET was quantified to identify the important split fractions. Those split fractions with a high importance (i.e., large impact on risk) were evaluated in detail. Split fractions that were evaluated in detail include the hydrogen burn, DCH, ISTGR, and containment failure top events. The analysis for hydrogen burns evaluated the containment conditions at the time of the possible burn based on the thermal hydraulic evaluation of the representative accident sequences. Other conditions determined from the thermal-hydraulic analyses include time of core uncovery, RPV failure time, and containment temperature and pressure. The evaluation of the various split fractions for the containment failure modes combines the range of pressure load profiles and the containment strength profiles.

4.7 Radionuclide Release Characterization

As described in previous sections, the Seabrook source terms have been categorized using nine Release Categories. Conservative and realistic source terms have been defined for each Release Category. Each Release Category described the following information:

- Start time for the release of radioactive material to the environment.
- Duration of the release.

- Warning time.
- Energy contained in the release plume.
- Release elevation.
- Nuclide release fractions.

Table 4-3 lists the Release Category information for the Seabrook source terms.

Release Category S1A characterizes an early gross failure with a large early release. It represents containment failures resulting from an early overpressurization caused by DCH or steam explosions. It is represented by a two-puff release.

Release Category S1B characterizes an early failure with a large early release. It represents containment failures resulting from an external source such as an aircraft or turbine missile. The conservative source term is represented by a single-puff release, while the realistic source term employs three puffs.

Release Category S2 characterizes an early leakage failure with late overpressurization. It represents inadequate isolation of small (< 3") penetrations or a small pre-existing leak. It is represented by a multipuff release.

Release Category S3A characterizes a late failure. It represents a late overpressurization event due to loss of containment heat removal, with a dry reactor cavity, i.e., no RWST injected. This Release Category also includes the failure mode base mat melt through identified in Table 3.4-5 as Release Category S4.

Release Category S3B also characterizes a late containment failure but includes a vaporization release. It represents a late overpressurization due to the lack of debris cooling.

Release Category S5 characterizes an intact containment with only low leakage. It represents the state where containment systems function long term to provide containment scrubbing and heat removal. Only A-type PDSs can enter this state.

Release Category S6 characterizes an early large failure with an early large release. It represents inadequate isolation of large (> 3") penetrations or a large pre-existing leak. It is represented by a multipuff release.

Release Category S7A characterizes a large early release from a bypassed containment. It represents a bypass via a large RHR pipe break in the RHR vaults which is unscrubbed. This also includes a bypass via a large, induced steam generator tube rupture.

Release Category S7B characterizes a small early release from a bypassed containment. It represents a failure of an RHR pump seal which is submerged and has a scrubbed release. This also includes a steam generator tube rupture with steamline bypass.

CORE MELT TIME SINCE REACTOR SHUTDOWN	CONDITIONS AT TIME OF REACTOR VESSEL MELT-THROUGH		CONTAINMENT INTACT AT TIME OF CORE MELT START							
			YES				NO			
			CONTAINMENT FUNCTIONS AVAILABLE				ACTIVITY RELEASE			
	PRESSURE IN REACTOR VESSEL (≳ 300 PSIA)	RWST INJECTION INITIATED	HEAT AND FISSION PRODUCT REMOVAL (A)	HEAT REMOVAL ONLY (B)	FISSION PRODUCT REMOVAL ONLY (C)	NONE (D)	FILTERED (E)	UNFILTERED		
								OPENING > 3 INCH DIAMETER (F)	OPENING < 3 INCH DIAMETER (FP)	AIRCRAFT CRASH (FA)
EARLY < 6 HOURS	LOW	NO (1)				1D		1F	1FP	1FA
		YES (2)	2A		2C	2D	2E	2F	2FP	2FA
	HIGH	NO (3)				3D		3F	3FP	
		YES (4)	4A		4C	4D	4E	4F	4FP	
LATE > 6 HOURS	LOW	NO (5)								
		YES (6)	6A		6C	6D	6E	6F	6FP	6FA
	HIGH	NO (7)				7D		7F	7FP	
		YES (8)	8A		8C	8D	8E	8F	8FP	
CORE MELT WITH NONISOLATED STEAM GENERATOR TUBE RUPTURE (9)			9A		9C	9D				

PRECLUDED BY SEABROOK STATION DESIGN FEATURES

NOT USED BECAUSE OF UNCERTAINTIES IN FAN COOLER CAPABILITY

PHYSICALLY POSSIBLE BUT NOT USED IN SSPSA RISK MODEL

TABLE 4-1

PLANT DAMAGE STATE MATRIX

TABLE 4-2

Seabrook Primary Containment Structural Failure Modes

Failure Mode	Median Pressure (psig)
Wall Hoop Failure	216
Dome Hoop or Meridiantal Failure	223
Wall Meridiantal Failure	281
Base Slab Shear Failure	323
Base Slab Flextural Failure	400
Wall Shear Failure at Base	408

TABLE 4-3

Conservative (C) and Realistic (R) Source Terms for Each Release Category

Release Category	Puff	Start (Hour)	Duration (Hour)	Kr-Xe	Org-I	I2-Br	CS-Rb	Te	Ba-Sr	Ru	La
S1A-C	1	3.0	1.0	3.0-7	2.1-9	3.7-8	3.3-8	5.7-9	3.7-9	1.1-9	1.1-10
	2	4.0	0.01	9.5-1	6.6-3	7.2-2	2.3-2	2.4-2	3.3-3	4.1-1	9.8-5
S1A-R	1	18.1	1.0	2.4-7	1.7-9	2.2-7	3.2-7	3.3-8	2.1-8	6.2-9	6.2-10
	2	19.1	0.2	9.4-1	6.6-3	8.0-1	7.4-1	3.9-1	9.3-2	4.6-1	2.8-3
S1B-C	1	0.074	0.5	9.0-1	7.0-3	7.0-1	5.0-1	3.0-1	6.0-2	2.0-2	4.0-3
S1B-R	1	0.074	2.0	2.0-1			2.2-2	4.0-3	3.0-3	8.0-4	8.0-5
	2	2.0	4.0	3.0-1			2.8-2	5.0-3	3.0-3	1.0-3	1.0-4
	3	6.0	6.0	5.0-1			2.0-3	4.0-3	2.0-4	2.0-4	4.0-5
S2-C	1	2.2	2.0	3.0-2	2.1-4	4.3-3	2.3-2	4.2-3	2.8-3	8.4-4	8.4-5
	2	4.2	4.0	7.0-2	5.0-4	1.3-3	4.8-2	3.9-2	5.5-3	3.4-3	5.2-4
	3	16.0	18.0	2.3-2	1.6-3	2.3-3	1.3-1	1.5-1	1.4-2	1.1-2	1.9-3
	4	24.0	52.0	8.8-1	5.0-3	7.1-3	1.1-1	1.3-1	1.2-2	9.8-3	1.7-3
S2-R	1	2.2	12.0	1.5-1			4.0-3	7.0-4	5.0-4	2.0-4	2.0-5
	2	14.2	8.0	2.0-1			7.0-3	8.0-4	8.0-4	6.0-4	1.0-4
	3	22.2	4.0	4.7-2			1.4-4	1.4-4	1.4-5	7.2-6	1.4-6
S3A-C	1	28.0	1.0	1.0			1.5-2	1.9-2	1.6-3	1.5-3	2.5-4
S3A-R	1	89.0	0.0	1.0			1.0-3	2.0-3	1.0-5	1.0-5	1.0-5
S3B-C	1	22	1.0	1.0			2.6-2	4.9-3	3.3-3	9.7-4	9.7-5
S3B-R	1	53.0	1.0	7.0-1			8.5-4	1.6-4	1.1-4	3.1-5	3.1-6
S5-C	1	4.3	24.0	1.4-2		5.0-7	5.0-7	1.0-7	6.0-8	2.0-8	2.0-9
S5-R	1	4.3	24.0	1.4-2		5.0-7	5.0-7	1.0-7	6.0-8	2.0-8	2.0-9

TABLE 4-3
(Continued)

Conservative (C) and Realistic (R) Source Terms for Each Release Category

Release Category	Puff	Start (Hour)	Duration (Hour)	Kr-Xe	Org-I	I2-Br	CS-Rb	Te	Ba-Sr	Ru	La
SS6-C	1	1.75	1.0	1.5-1	1.1-3	1.0-1	1.1-1	2.0-2	1.4-2	4.1-3	4.1-4
	2	2.75	4.0	4.2-1	2.9-3	7.0-2	1.9-1	6.3-2	2.2-2	9.0-3	1.0-3
	3	15.75	18.5	3.2-1	2.2-3	1.0-2	1.3-1	3.2-1	1.1-2	2.0-2	3.8-3
S6-R	1	4.0	2.0	2.0-1			4.0-3	9.0-5	3.0-4	2.0-5	2.0-5
	2	6.0	4.0	3.0-1			5.0-3	1.0-4	3.0-4	3.0-5	3.0-5
	3	10.0	10.0	5.0-1			1.0-3	9.0-5	2.0-5	1.0-5	1.0-5
S7A-C	1	2.5	0.5	9.0-1	7.0-3	7.0-1	5.0-1	3.0-1	6.0-2	2.0-2	4.0-3
S7A-R	1	2.5	0.5	9.0-1	7.0-3	7.0-1	5.0-1	3.0-1	6.0-2	2.0-2	4.0-3
S7B-C	1	8.5	7.0	9.0-1	7.0-4	7.0-2	5.0-2	3.0-2	6.0-3	2.0-3	4.0-4
S7B-R	1	8.5	7.0	9.0-1	7.0-6	7.0-4	5.0-4	3.0-4	6.0-5	2.0-5	4.0-6

Note: Exponential notation is indicated in abbreviated form; i.e., 3.0-7 = 3.0E-7.

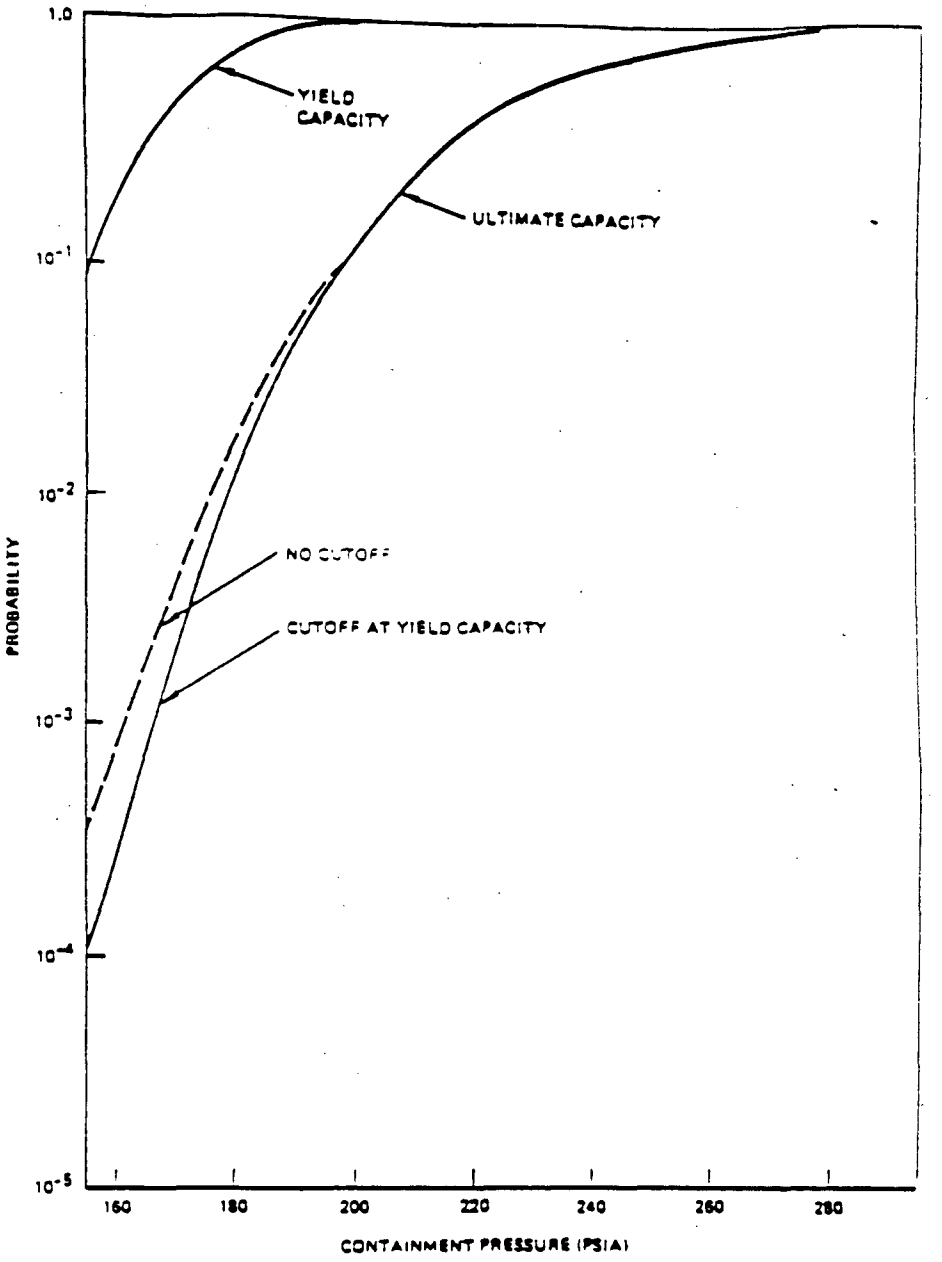
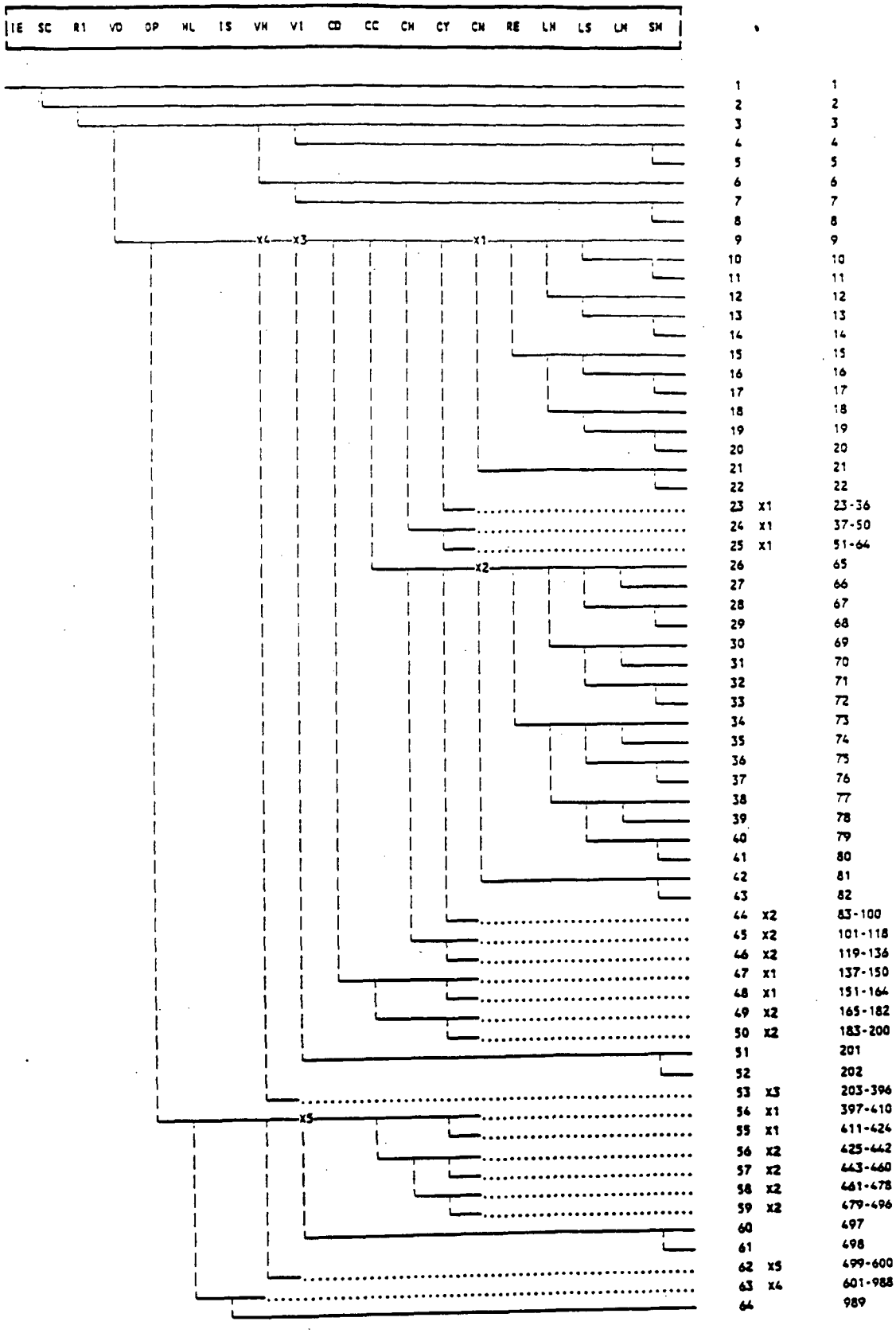


FIGURE 4-1

Containment Conditional Failure Probability
Versus
Internal Pressure

WPP44/142

Figure 4-2 Containment Event Tree
(Sheet 1 of 2)



1		1
2		2
3		3
4		4
5		5
6		6
7		7
8		8
9		9
10		10
11		11
12		12
13		13
14		14
15		15
16		16
17		17
18		18
19		19
20		20
21		21
22		22
23	x1	23-36
24	x1	37-50
25	x1	51-64
26		65
27		66
28		67
29		68
30		69
31		70
32		71
33		72
34		73
35		74
36		75
37		76
38		77
39		78
40		79
41		80
42		81
43		82
44	x2	83-100
45	x2	101-118
46	x2	119-136
47	x1	137-150
48	x1	151-164
49	x2	165-182
50	x2	183-200
51		201
52		202
53	x3	203-396
54	x1	397-410
55	x1	411-424
56	x2	425-442
57	x2	443-460
58	x2	461-478
59	x2	479-496
60		497
61		498
62	x5	499-600
63	x4	601-988
64		989

Figure 4-2 Containment Event Tree
(Sheet 2 of 2)

Top Event Designator.....	Top Event Description.....
IE	
SC	SUCCESS MAPPING - NO CORE MELT
R1	DIRECT MAP
VD	DEBRIS COOLED IN VESSEL
DP	OPERATOR DEPRESSURIZES RCS
HL	HOT LEG FAILS FIRST
IS	INDUCED SGTR
VH	NO H2 BURN / DEBRIS IN VESSEL
VI	CONT. INYACT / DEBRIS IN VESSEL
CD	DEBRIS DISPERSED IN CAVITY
CC	DEBRIS COOLED IN CAVITY
CH	NO DCH / DEBRIS IN CAVITY
CY	NO H2 BURN / DEBRIS IN CAVITY
CN	CONT. INTACT / DEBRIS IN CAVITY
RE	OPERATOR RECOVERS CONTAINMENT COOLING
LH	NO LATE H2 BURN
LS	SHELL INTACT / LONG TERM
LM	BASEMAT INTACT / LONG TERM
SM	SMALL LEAK

5.0 UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

The IPE Program is run by the Seabrook "PSA Team" consisting of engineers from the Reliability and Safety Engineering Department within New Hampshire Yankee (NHY) and engineers in the Safety Assessment Group within Yankee Atomic Electric Company (YAEC). The close involvement with YAEC allows the development of experts in specific risk assessment areas and the cross-fertilization of ideas from groups working on different types of plants. The original SSPSA (Reference 3) and a number of the subsequent studies were performed by PLG, Inc. with support from various subcontractors. In each of these studies, the Seabrook PSA Team was involved providing information, reviewing analyses, and assuring the studies reflected the plant configuration. As the PSA Team has evolved since the SSPSA, increasingly more work has been done in-house, with consultants used only in areas of special expertise.

The following discussion of independent reviews focuses on the reviews done in 1982 and 1983 for the SSPSA. These independent reviews are important to this IPE Report because the SSPSA forms the basis for the present risk model.

In follow-on studies to the SSPSA, a variety of technical reviews have been used. The most common is a peer review by a qualified individual within NHY or YAEC. The level of independence achieved in the SSPSA is not practical in smaller scope studies. The Risk Management and Emergency Planning Study (Reference 11) was an exception in that it had a detailed review by an independent group of leading technical experts.

5.2 Composition of Independent Review Teams

The original SSPSA was performed with a multilayered technical review to assure the technical accuracy of the analyses, documentation, and results. Table 5-1 identifies the seven layers of review that extended from the analyst through the contractor technical management to the utility personnel. In particular, the review of the SSPSA by NHY and YAEC personnel was more than six person-years of effort. This review included two separate independent review processes: the Quality Assurance (QA) Review, lead by the PLG QA Manager, audited by a review team from YAEC; and the Technical Review Board, made up of experts, including two PSNH personnel.

The QA review was performed by PLG personnel not connected with the SSPSA to assure compliance with PLG QA Manual (Reference 37). The QA review was enhanced through an audit by YAEC to verify compliance with the QA Manual and PLG internal audits to assure that QA procedures were followed and documented. These procedures included a formal program to document and verify all computer software that was developed and used on the Seabrook project.

The Technical Review Board was made up of the individuals shown on Table 5-2. This board was set up specifically to provide an independent technical review of the documents, analyses, and results obtained during the project. Independence for this board meant that no reviewers on the board was allowed to contribute to a document or deliverable other than reviewing it. Draft sections of the report were sent out to specific members (i.e., not all members reviewed all the report) according to their expertise. Two meetings of the complete board were held, one to review early results and one to review the final report. The first meeting focused on recommendations to enhance the analyses before results were finalized. The second meeting focused on the documentation of the final results. Review comments made in writing and during the meeting were resolved by the Project Manager.

5.3 Areas of Review and Major Comments

As described in Section 5.2, the SSPSA was performed with two levels of independent review, as well as a number of interactive reviews.

The QA review focused on adequacy of document control, computer code control and verification, analyst training, and the independent technical review process. In particular, the YAEC QA audit early in the project raised the comments listed in Table 5-3. These comments were addressed by the PLG Manager of QA, as indicated in Table 5-3.

The Technical Review Board was responsible for reviewing every aspect of the risk analysis and every section of the report. Because of the diversity of experience, some of the reviewers commented on methods and techniques of the analyses, while others commented on the results or the accuracy of model to the real plant (References 46 - 49). A sample of the extensive comments is given in Table 5-4.

The resolution of the specific comments are addressed in Table 5-4. The Technical Review Board also had positive comments, e.g., "The study is conducted according to the highest quality standards and using methodologies that often go beyond the state-of-the-art" (Reference 46).

5.4 Resolution of Comments

Each comment raised by the independent review boards discussed in Section 5.3 was resolved by the Project Manager and his team to the satisfaction of the boards. The resolution of the specific comments are given in Table 5-3 for the QA review and Table 5-4 for the Technical Review Board comments.

TABLE 5-1

Levels of Review for the SSPSA

Level	Review Objective	Person Responsible
1	Check all calculations, computer input and output, proofread documents prepared by Publications Department for technical accuracy.	Analyst/Author
2	Double check all calculations, review documentation for technical accuracy, ensure consistency of documentation within technical area (e.g., systems), and ensure that the right tools are used.	Task Leader
3	Spot check calculations, ensure that acceptable PRA methods and procedures are utilized, perform independent review of all deliverables, supports calculations, and documents, as necessary, focusing on reasonableness of results and conclusions and whether project documentation adequately reflects what was done; recommend corrective action when appropriate.	Technical Review Board
4	Review all deliverables, ensure project objectives are met, ensure consistency among technical areas, responsible for resolution of all review comments and assignment of work needed to resolve review issues.	Project Manager
5	Review results and conclusions of key deliverables for technical credibility and efficiency of methods employed.	Project Director
6	Review all deliverables for appropriateness of assumptions regarding interpretation of plant documentation, safety analyses, and modelling of plant and site-unique characteristics.	Client (PSNH and YAEC)
7	Perform QA audits, conduct QA training, and maintain QA records.	PLG QA Manager

TABLE 5-2

Technical Review Board (TRB) for the SSPSA

TRB Member	Affiliation
Frank R. Hubbard, Chairman	Pickard, Lowe, and Garrick, Inc.
George Apostolakis*	University of California, Los Angeles
William K. Brunot	Private Consultant
Vijay K. Dhir*	University of California, Los Angeles
William T. Hussey, QA Manager	Pickard, Lowe, and Garrick, Inc.
Mohammad Modarres*	University of Maryland
Donald A. Norman	University of California, San Diego
Norman C. Rasmussen	Massachusetts Institute of Technology
James E. Shapley	Pickard, Lowe, and Garrick, Inc.
Walter B. Sturgeon	Public Service Company of New Hampshire
Juliette Zivic	Public Service Company of New Hampshire

*Also associated with Pickard, Lowe, and Garrick, Inc.

TABLE 5-3

Quality Assurance Review of the SSPSA Process

QA Review Comments	Resolution
• Need to log and track correspondence between PLG and client.	• Correspondence log has been established.
• QA Manual should reflect responsibilities of QA Manager distinct from Project Manager.	• New QA Manual will reflect the activities and responsibilities of the QA Manager.
• An audit was needed per PLG QA Procedure early in the project.	• Audits of Document Control and Computer programs have been initiated.
• Assure that all certified computer codes are documented, verified, reviewed, and approved.	• An audit of computer code documentation has been initiated.
• Provide training to project participants in QA activities.	• A QA Training Program is being prepared.
• Document process of selecting subcontractors and maintaining interfaces.	• QA Manual is being revised to address this.
• Identify most current documents used under the Seabrook Project activity.	• The Seabrook PSA is being done as a base line documented in FSAR Amendment 47 and drawings and documents issued before December 3, 1982.

TABLE 5-4

Selected SSPSA Technical Review Board Comments/Resolutions

TRB Comments	Resolution
• V-sequence valve failure modes - how can a command open both MOVs?	• Only by defeating an interlock that prevents opening at greater than 600 psia. This is a low probability event.
• Reactor trip - very conservative to assume no more than one stuck rod for success criteria.	• Agree, but this conservatism does not influence the results.
• Seismic dependency assumption - very conservative.	• Agree, but more realistic treatment of this dependence is beyond the state-of-the-art.
• Value for manual trip is too conservative if operators are trained to always hit manual scram on scram signal.	• Agree, but difficult to justify lower value for short time (2 minutes).
• Calculation of risk from two units (i.e., Seabrook Units 1 and 2 was questioned.	• Comment clarified, results are correct.
• Release Categories should be described based on the outcome of the containment tree (i.e., the release type) rather than the cause.	• Cause information is used only to the extent needed to characterize the source term.
• Use of depressurization and low pressure pumps as a success path redundant to high pressure safety injection is questionable. This option should not be used unless safety injection failure becomes important.	• modelled actions are consistent with existing emergency procedures and, thus, are left in the model.
• An operator error to fail to control EFW (i.e., overfilling the generators) should be included.	• Such an action was included in the model.
• Initiating event frequencies for large and medium LOCAs and transients seem to be too high.	• Large and medium LOCAs are consistent with WASH-1400. Transient frequencies are based on EPRI-NP-2230.
• Use of operator response viewed on the simulator should be done with care since the operators have not yet been trained on the Seabrook control board or emergency procedures.	• Comment noted, no impact on PSA.

TABLE 5-4
(Continued)

SSPSA Technical Review Board Comments/Resolutions

TRB Comments	Resolution
• Too many conservatisms appear in the report to make it useful for future risk management.	• Treatment of dominant risk contributors is realistic enough for report publication. Results are continuously being updated to reflect new evidence which often supports a lower core melt frequency but occasionally suggests a higher value.

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

Because of the safety features built into Seabrook Station and based on the extensive evaluation summarized in this report, no fundamental weakness or vulnerability has been uncovered with regard to severe accidents. A summary of these important safety features is given in Section 6.1. The process of evaluating important risk contributors for potential plant improvements is described in Section 6.2.

6.1 Unique Safety Features

A number of important safety features contribute to the low level of risk at Seabrook Station. While most of these safety features are shared by plants of a similar vintage, features that are more or less unique to Seabrook include the following:

- Secondary Cooling Function

  Three 100% capacity trains, diverse power, and location (turbine-driven pump, electric-driven pump from essential Bus E5, both in EFW Pumphouse; start-up feed pump with back-up power from essential Bus E6 in the turbine hall).

  - System dependent only on essential ac and dc power (pumps self-cooled, not dependent on ventilation; valves are normally open MOVs or fail open AOVs).

  - Large steam generator inventories allow extended time (>1 hour) to recover forced cooling.

- Component Cooling/Service Water Systems

  - Component cooling has two 100% pumps per train; service water has three 100% pumps per train (including one cooling tower pump per train).

  - Service water provides two diverse ultimate heat sinks, ocean via the intake/discharge tunnels, and atmosphere via the safety grade cooling towers.

- ECCS

  - Four spatially independent high pressure pumps: two charging pumps (in separate cubicles in the PAB) and two safety injection pumps (in separate RHR equipment vaults) - each 100% pumps for small LOCA and feed and bleed cooling.

  - Relatively few components that must change state upon initiation (e.g., injection lines contain check valves in series rather than normally closed motor-operated valves).

  - Switchover to recirculation is automatic for low pressure pumps.

- Off-Site Power

  - Reliable off-site grid.

  - Three incoming lines from separate directions.

  - SF_6 switchyard eliminates concern about salt spray on insulation.

- Containment

  - Large, massive containment structure - ultimate failure pressure of more than three times design.

  - Base mat concrete composition minimizes the production of noncondensible gases, thereby delaying containment pressure rise.

  - Capability to use either RHR or CBS heat exchangers for containment heat removal during recirculation phase.

  - Containment isolation is very reliable with fail-closed air-operated valves. Seal return is the only path dependent on AC power, and it is identified in emergency operating procedures.

- Other

  - ATWS Mitigation System (AMSAC) provides alternate means of turbine trip and EFW actuation.

- Thermal Barrier Cooling System provides diverse, back-up means of RCP seal cooling.

- Pressurizer PORVs require only DC power to operate; no dependency on air or other support equipment.

- RHR design pressure and pump location in equipment vaults substantially reduce risk from interfacing systems LOCA.

- Seismic design has substantial margin. No structures and only a few components with estimated median capacity below 2g.

6.2 Potential Plant Improvements

A number of potential plant improvements have been identified and are being analyzed for their cost-benefit in reducing the frequency of core damage (financial risk) and significant off-site release (public risk). These are described in the sections that follow. In general, the following process is used in considering potential plant improvements:

1. Confirmation that the dominant sequences are realistic. We must be satisfied that the risk assessment does not contain some undo conservative assumptions. This may require evaluation of success criteria, recovery potential, etc. This is consistent with the interactive nature of risk assessments which attempt to apply the most realistic modelling to the most important issues.

2. Next, the potential for enhanced procedures and/or training is evaluated. This can provide benefit in the quantification or added assurance that operator models are conservative.

3. For some sequences that are dominated by hardware failures and maintenance malfunctions of systems based on generic data, system reliability improvements are considered. These are focused on reliability-centered maintenance programs which are using the output from SSPSS to help set priorities. In the long-term, equipment failure data and maintenance unavailabilities will be monitored to allow plant-specific updates.

4. Finally, functional improvements are evaluated that can address a number of sequences. Where substantial risk improvement can be shown, a cost estimate is made to provide sufficient information to estimate the cost-benefit ratio.

6.2.1 Core Damage Risk Improvements

From the analysis of results in Section 3.4.2.1, several conclusions were reached that impact potential improvements. First, the dominant functional sequences, station blackout, and transient with PCC failure have the RCP seal LOCA in common. Thus, a plant improvement that would provide independent seal cooling to the RCP would offset a maximum of 70% of the sequence frequencies. Second, the evaluation of initiating event contribution to core damage indicates the importance of external events (about 45%). Thus, any plant enhancements need to consider the impact of external events. Finally, the list of important systems is lead by diesel generators, PCC, and SW which is consistent with the functional sequences.

In order to accurately judge the effect of plant enhancements, as well as the need for more detailed analyses, an evaluation of the dominant sequences was performed. Table 6-1 lists the top 24 core damage sequences (with frequency greater than or equal to 1.0E-6/year) with an indication of potential plant and analysis enhancements that are being considered. As can be seen, the specific scenario can impact the design of the functional improvement. For example, the first sequence, fire in the Control Room leading to loss of all PCC and eventual RCP seal LOCA, can be addressed by an independent Seal Cooling System. However, this system may need to be automatic because the sequence includes the failure of the operators to cool down using the remote safe shutdown panel. For this sequence, as with others identified, the first step will be to update the fire analysis including reanalyzing the human action included therein.

Table 6-2 lists potential plant enhancements with their effect at reducing core damage frequency. The most effective enhancements are the independent, automatic seal injection pump or larger charging pump. The smaller seal injection pump is nearly as effective at CDF reduction because of the low importance of feed and bleed cooling. This table shows quantitatively the importance of considering all accident sequences, including external hazards. Since the importance of external hazards will directly affect the benefit of any enhancement, the next step in a logical risk management program is a review and update of the hazards analyses. This is planned for 1991.

In addition to these future enhancements, significant work was done from the SSPSA to the SSPSS 1990 to better understand and quantify the dominant core damage sequences. The electric power recovery model was enhanced to account for the time dependency of actions and availability of equipment (EFW, batteries) using a Monte Carlo simulation (Reference 12). This was updated to account for more current information on grid restoration times and battery lifetimes (Reference 14). Also, the seal LOCA issue was addressed in more detail in order to better account for the most current judgment on leak rates (Reference 13).

6.2.2 Containment Performance Risk Improvements

From the analysis of results in Section 3.4.3.2, the conclusion was presented that there are no vulnerabilities to "unusually poor" containment performance. Nonetheless, potential improvements listed in Table 6-3 have been considered to address the various contributors to early, large containment failure or bypass. For the highest ranking enhancement, restricting purge valve operation, the first step would be to more carefully consider the operator model for signal recovery. The present, conservative model credits only operator actions to recover signals to prevent core melt. Additional time is available for manual containment isolation. All of these failure modes are low frequency and have relatively large uncertainty. Also, several of these failure modes (e.g., direct containment heating and induced SGTR) are dominated by phenomenological uncertainties. Thus, any changes, even procedural changes, need to be considered in light of the frequency and uncertainty. Procedural changes, in particular, will be evaluated in an integrated fashion along with long-term containment recovery actions as part of a future accident management program.

A number of plant and model enhancements have been evaluated as part of work done since the SSPSA to better understand the early public health risk. The issues that have been addressed are discussed below:

1. Interfacing LOCA

   In the original SSPSA, the interfacing system's LOCA sequence dominated early large release frequency at approximately 2E-6 per year. Initial efforts to evaluate this sequence identified the potential for the low pressure piping to survive RCS pressure, the potential for operators to isolate certain sequences, and the potential for pump seal leaks to be covered by water in the equipment vault. This was evaluated and documented in PLG-0432 (Reference 11) and is incorporated into the present SSPSS model and results.

The interfacing LOCA update identified the potential to improve emergency operating procedures. In the interim, a training module was developed to address diagnosing a spectrum of interfacing LOCA scenarios. This was reviewed by the NRC and BNL at the Seabrook simulator (Reference 67). Recently, the Emergency Operating Procedures (Revision 12, September 12, 1990) have been revised to address interfacing LOCA concerns.

The total core damage release frequency from interfacing LOCA sequences is presently 4E-8 per year. A Leakage Monitoring System was investigated to identify valve leakage. The system would only reduce the frequency for certain failure modes and the risk is judged too low to justify equipment modifications.

2. Seismic Events

Seismic hazard provided a small contribution (approximately 6E-7 per year) in the original SSPSA. More detailed evaluations of equipment fragilities, modelling assumptions, and containment isolation valve dependence on relay chatter essentially eliminated the seismic contribution to early large release. The present total frequency is less than 1E-8 per year.

3. Pre-Existing Leaks

Pre-existing containment leaks were not explicitly modelled in the original SSPSA as they were judged to be a small contributor to containment isolation failure. After the NRC questioned this judgment (Reference 67), industry events were evaluated with regard to their applicability to Seabrook Station, and pre-existing leaks were explicitly modelled as a contribution to containment isolation failure. The total frequency of early large release from containment isolation failure is 1.2E-7/yr, and the contribution from large pre-existing leaks is 1.0E-8/yr.

A conceptual Containment Leakage Monitoring System was investigated, but the present judgment is that risk is too low to justify equipment modifications.

4. Induced SGTR (Bypass)

These sequences were not modelled in the original SSPSA as they were believed to be very unlikely. The NRC and their contractors have identified this phenomena as having large uncertainties and, because of potential containment bypass, should be evaluated (Reference 67). This issue was evaluated in PLG-0550 (Reference 10) and is conservatively incorporated in the present model and results.

Sequences that can potentially induce failure of the steam generator tubes must be high pressure core melt sequences with the steam generator dry. PLG-0550 evaluated two potential improvements:

- Depressurization capability using the primary PORVs and revising EOPs to allow depressurization for station blackout sequences.

- A hard-piped crosstie between fire water and feedwater.

The total frequency of induced SGTR sequences without credit for either of the above improvements is 6.0E-8/yr. Because of the low risk, no improvements are presently planned. However, the Westinghouse Owners Group (WOG) is expected to consider depressurization for station blackout sequences in the future. At the same time, WOG is expected to address concerns identified in PLG-0550 regarding the jogging or starting of reactor coolant pumps during high pressure sequences with dry steam generators. The hard piped crosstie to fire water will be appropriately considered as a potential improvement to core damage frequency.

5. Direct Containment Heating Loads

These sequences were not explicitly modelled in the original SSPSA as they were judged to be very unlikely. A conditional probability of 10^{-4} was used for early containment structure failure. The NRC and their contractors have identified the potential loads from high pressure melts (i.e., direct containment heating phenomena) as having large uncertainties requiring additional analysis (Reference 67). PLG-0550 evaluated loads being considered by the NRC contractors and the probability of containment failure using containment capacities based on the NRC and BNL reviews. The present models and results explicitly model the probability of early containment failure due to direct containment heating loads.

The total frequency is 2.5E-8/yr. This frequency takes no credit for operators depressurizing the Primary System with PORVs for high pressure sequences. As discussed above, this action may be considered by the WOG in the future for station blackout sequences. Seabrook-specific analyses in PLG-0550 indicate potential benefits of depressurization, but the risk is already very low.

6. Other Early Large Releases

The original SSPSA included turbine missiles and airplane crash as potential initiators that fail containment. The frequency of these are very low, such that potential improvements cannot be justified.

7. Shutdown Events

Because of the very low risks being demonstrated for Seabrook Station, the NRC questioned the relative contribution from shutdown events with the equipment hatch off (Reference 67). A detailed study of such events was conducted and is documented in Reference 28. This study concluded that additional, more restrictive procedures were required to reduce the frequency of an early large release during shutdown. The total frequency is 5.5E-7 per year.

TABLE 6-1

Potential Plant and Analysis Enhancements to Reduce the Frequency of the Dominant Core Damage Sequence

CD Rank	Sequence (a)	Functional Sequence (b)	Similar Sequences (c)	Potential Plant/Analysis Enhancements: More Detailed Analysis (d)	System Reliability Improvement (e)	Functional Improvement (f)	Enhanced Operator Action
1	Fire in Control Room, loss of PCC.	2	12, 15	Update fire analysis.	--	(1), (3-auto)	--
2	Fire in Turbine Building, loss of off-site power and failure of diesels and failure to recover diesels.	1	--	Update fire analysis.	DG	(1), (3-auto), (4)	--
3	Loss of off-site power due to SF_6 failures and failure of diesels and failure to recover diesels.	1	25	--	SF_6, DG	(1), (2), (3), (4), (5)	--
4	Loss of off-site power (grid-related) and failure of diesels and failure to recover power.	1	--	--	DG	(1), (2), (3), (4), (5)	--
5	Flood in Turbine Building loss of off-site power and failure of diesels and failure to recover diesels.	1	--	--	DG	(1), (3-auto), (4)	--
6	Reactor trip and failure of PCC.	2	8, 9, 10, 11, 13, 51, 54, 58, 72, 87	--	PCC	(1), (2), (3)	Connect fire water to PCC loads.
7	Fire in the PAB, loss of PCC.	2		Update fire analysis.	--	(1), (3-auto)	
8	Partial loss of main feedwater and failure of PCC.	2	6	--	PCC	(1), (2), (3)	Connect fire water to PCC loads.
9	Turbine trip and failure of PCC.	2	6	--	PCC	(1), (2), (3)	Connect fire water to PCC loads.

TABLE 6-1
(Continued)

Potential Plant and Analysis Enhancements to Reduce the Frequency of the Dominant Core Damage Sequence

CD Rank	Sequence (a)	Functional Sequence (b)	Similar Sequences (c)	Potential Plant/Analysis Enhancements: More Detailed Analysis (d)	System Reliability Improvement (e)	Functional Improvement (f)	Enhanced Operator Action
10	Reactor trip due to loss of PCC Train B and failure of PCC Train A.	2	6	--	PCC	(1), (2), (3)	
11	Reactor trip due to loss of PCC Train A and failure of PCC Train B.	2	6	--	PCC	(1), (2), (3)	Connect fire water to PCC loads.
12	Fire in Cable Spreading Room, loss of PCC.	2	1	Update fire analysis.	--	(1), (3-auto)	--
13	Loss of off-site power (grid-related) and failure of PCC.	2	6	--	PCC	(1), (2), (3),	Connect fire water to PCC loads.
14	Large seismic event (1.0g) and loss of off-site power due to seismic event and seismic-induced relay chatter, loss of on-site ac power.	1	--	Update seismic analysis (relay - chatter).	--	--	Operator response to relay chattering.
15	Fire in Control Room, loss of service water.	2	1	Update fire analysis.	--	(1), (3-auto)	--
16	Fire in Control Room, loss of all ac power.	1	--	Update fire analysis.	--	(1), (3-auto), (4)	--
17	Loss of off-site power and failure of service water.	1	--	--	SW	(1), (3-auto)	Operator training caution regarding diesel generators dependence on service water.

TABLE 6-1
(Continued)

Potential Plant and Analysis Enhancements to Reduce the Frequency of the Dominant Core Damage Sequence

CD Rank	Sequence (a)	Functional Sequence (b)	Similar Sequences (c)	Potential Plant/Analysis Enhancements: More Detailed Analysis (d)	System Reliability Improvement (e)	Functional Improvement (f)	Enhanced Operator Action
18	Loss of DC Bus B, failure of turbine-driven EFW pump, and failure of operator action to manually initiate Train A signals.	3	--	Revise conservative quantification of operator action.	EFW	--	--
19	Fire in electrical Tunnel 1, loss of Train A service water and failure of Train B service water.	2	--	Update fire analysis.	SW	(1), (3-auto)	--
20	Loss of main feedwater and failure of reactor trip breaker to open and reactor power at high level (>10%) and failure of ATWS Mitigation System (resulting in failure of turbine trip and EFW).	4	--	--	RT Breakers	(6)	--
21	External flood, loss of service water.	1	--	--	--	(1), (3-auto)	--
22	Small LOCA and failure of RHR in miniflow recirculation and failure to makeup to RWST.	5	--	Revise conservative quantification of operator action.	RHR	--	--
23	Small LOCA and failure of ventilation to ECCS.	5	--	Analyze assumption of need for ventilation.	--	--	--
24	Loss of DC Bus B and failure of EFW and failure to recover EFW or start-up feed pump.	3	--	Update success criteria for feed and bleed (one PORV) rather than two PORVs.	EFW	(7)	--

TABLE 6-1
(Continued)

Potential Plant and Analysis Enhancements to Reduce the Frequency of the Dominant Core Damage Sequence

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(a) Core damage sequence list is from Table 3.4-1.

(b) "Functional Sequences" are from Table 3.4-6, and include the following type:

1 - Station blackout/seal LOCA.
2 - Transient with loss of component cooling/seal LOCA.
3 - Transient with failure of EFW and feed and bleed cooling.
4 - ATWS.
5 - Small LOCA with failure at RHR.

(c) "Similar Sequences" are sequences from the top 100 core damage sequences from Table 3.4-1.

(d) "More Detailed Analysis" - Potential analysis updates includes:

- Fire Analysis - Better plant documentation and methods for quantifying operator actions are available now.
- Seismic Analysis - Conservative modeling of relay chatter issue.
- Conservative operator actions, success criteria.

(e) "System Reliability Improvements" - For most systems, this is the reliability-centered maintenance program which will evaluate risk-important systems on a priority basis. Also, the SF_6 is being evaluated for reliability upgrades related to plant availability.

(f) "Functional Improvements" - See Table 6-2 for list of potential improvements.

TABLE 6-2

Potential Plant Enhancements to Reduce Core Damage Frequency (CDF)

	Enhancement	Benefit	Percent CDF Reduction
1.	Independent, automatic seal injection pump.	In the event of loss of ac power or loss of PCC, would allow RCP seal cooling preventing seal LOCA; would allow successful cooldown as long as secondary cooling is available.	59%
2.	Independent, manual seal injection pump.	In the event of loss of AC power or loss of PCC, would allow RCP sealing cooling with operator action, preventing seal LOCA. (Not credited for hazard initiators that may impact operator action, e.g., Control Room fire, seismic events).	28%
3.	Independent, manual charging pump.	In the event of loss of ac power or loss of PCC, would allow RCP seal cooling, with operator action, preventing seal LOCA; would also provide decay heat removal through feed and bleed cooling.	28% (61% if automatic)
4.	Alternate emergency ac power source (e.g., swing diesel).	In the event of loss of all ac power, would allow restoration of ac power within a few hours from a redundant power source to power one emergency bus.	24%
5.	Alternate off-site power source that bypasses switchyard.	In the event of an extended loss of off-site power due to SF_6 bus failures, would allow restoration of off-site power within a few hours.	8%

TABLE 6-2
(Continued)

Potential Plant Enhancements to Reduce Core Damage Frequency (CDF)

	Enhancement	Benefit	Percent CDF Reduction
6.	Alternate scram button to remove power from MG sets to control rod drives.	In the event of an ATWS due to breaker failure, would allow reactor shutdown with operator action.	5%
7.	DC power enhancement: - independent ac source for battery chargers. - credit operator action to crosstie batteries within each train. - additional batteries.	In the event of loss of ac power, would extend battery lifetime to allow additional time for recovery.	4%

TABLE 6-3

Potential Plant Enhancements to Reduce Off-Site Release

	Enhancement	Benefit	Percent Reduction in Large Early Failure Probabilities
1.	Administrative control to reduce time the purge valves are open.	In the event of a core melt with failure of containment isolation signals, would reduce the likelihood of a large containment penetration open.	47%
2.	Procedure to direct depressurization of RCS.	In the event of a core melt, would reduce the potential for direct containment heating and for induced steam generator tube rupture.	34%
3.	Alternate, independent emergency feedwater pump (e.g., diesel firewater pump hard piped to discharge of startup feed pump).	In the event of a high pressure core melt, would reduce the potential for induced steam generator tube rupture.	24%
4.	Containment leakage monitoring.	In the event of a core melt, would reduce the potential for pre-existing containment leakage.	4%
5.	RHR isolation valve leakage monitoring system.	In the event of failure of the upstream RHR isolation valve to isolate RCS, would allow time for stable shutdown and depressurization as long as second isolation valve did not fail on demand; would preclude challenge of RHR piping and LOCA outside containment.	<2%

7.0 SUMMARY AND CONCLUSION

This report summarizes the systematic, plant-specific examination for vulnerabilities to severe accidents at Seabrook Station. The results and conclusions presented are based on the Seabrook Station Probabilistic Safety Study - 1990 Update (SSPSS 1990). This current risk assessment has evolved from the original full scope, Level 3, Seabrook Station Probabilistic Safety Assessment (SSPSA) which is the base line risk assessment.

The results of the current study indicate a very low risk from severe accidents. Using the definition of vulnerabilities from Section 3.4.2.3 (i.e., those components, systems, operator actions, and/or plant design configurations that contribute significantly to an unacceptably high severe accident risk), no severe accident vulnerabilities were uncovered based on extensive risk studies. The major contributor to the core damage severe accident risk is the direct dependency of RCP seal cooling on Primary Component Cooling and AC electric power. In addition, external events were identified as a significant contributor to the core damage risk. As a result, plant enhancements under consideration to address this issue must be made considering the impact of fires, seismic events, etc. Consistent with the iterative nature of risk assessments, the important external events will be re-evaluated prior to any final decisions about plant enhancements.

The major contributors to "unusually poor" containment performance, i.e., large early release, are (1) a conservative model of signal failure and (2) phenomenological issues, direct containment heating and induced steam generator tube rupture, which require high primary system pressure to be of concern. The conservative modeling will be addressed by more careful consideration of operator actions. The other issues can be addressed by procedural changes. Because of their low frequency, these changes will be considered in coordination with an integrated accident management plan.

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APPENDIX A

NRC REVIEWS OF SEABROOK RISK STUDIES

APPENDIX A

NRC Reviews of Seabrook Risk Studies

The U.S. Nuclear Regulatory Commission (NRC) and its contractors have reviewed the Seabrook Station Probabilistic Safety Assessment and its follow-on studies. The following three major reviews are summarized in this Appendix:

- LLNL review of the Level I analysis, including external events of original SSPSA,
- BNL review of the Level II analysis of SSPSA, and
- BNL review of the risk management and emergency planning studies.

A.1 Level I Review - Lawrence Livermore National Laboratory (LLNL)

In the fall of 1984, the Lawrence Livermore National Laboratory reviewed the SSPSA for the NRC. A draft technical report entitled, "A Review of the Seabrook Station Probabilistic Safety Assessment," dated December 12, 1984 was issued by the NRC with a letter dated April 4, 1985 (Reference 50). The staff summary stated that, "our review of the Seabrook PSA did not identify any safety issues which merit immediate attention ... Overall, the review did not identify a discrepancy or error which is estimated, at this point, to significantly change the quantitative results of the PSA." It was concluded that the dominant sequences were reasonable, and often conservative, and that a new evaluation would not find the probability of core melt significantly larger.

The review was performed by a project team composed of personnel from the NRC staff, LLNL staff, subcontractors, and consultants. It included a site visit and meeting with Seabrook Station personnel. The review covered all major areas of the Level I plant analysis and evaluation in the SSPSA. This included initiating events, event trees, success criteria (for functions and systems), fault trees, human factors, component and operating experience data, and the treatment of uncertainty.

NRC Reviews of Seabrook Risk Studies

The scope of the review also included an examination of several issues of particular interest to the NRC, including, (1) reactor coolant pump seal LOCAs, (2) depletion of station batteries during station blackout, (3) pressurized thermal shock, (4) steam generator tube rupture with stuck open secondary steam relief valves, and (5) stuck-open safety/relief valve.

A review of the draft technical report performed by NHY identified a significant number of discrepancies and points of disagreement. These comments were documented in a letter to the NRC, dated May 17, 1986 (Reference 51). The draft technical report (which was issued in December 1984), does not reflect the NHY comments.

A summary of the significant LLNL comments and NHY resolutions is contained in Table A-1.

NRC Reviews of Seabrook Risk Studies

A.2 Level II Review - Brookhaven National Laboratory (BNL)

In 1985, BNL conducted a review for the NRC of the Seabrook SSPSA containment and source term analysis (NUREG/CR-4552, Reference 39). The objective of the review was to provide a perspective on severe accident propagation, containment response and failure modes together with radiological source term characteristics. Principal design characteristics were compared with those of Zion, Indian Point, and Millstone 3 designs.

The review concluded that the probability of prompt containment failure was negligible, failure during the first few hours is unlikely, and the timing of overpressure failure is very long compared to WASH-1400. Their assessment of containment failure indicated that failure would most likely be through a relatively benign mode. Also, most core melt accidents would be effectively mitigated by containment spray operation. The review also concluded that the point estimate release fractions were comparable with WASH-1400, and the energy of release was somewhat higher.

A.3 Risk Management and Emergency Planning Studies Review - Brookhaven National Laboratory

Subsequent to the review of the SSPSA summarized above, BNL conducted a technical evaluation (Reference 67) of the Seabrook Station emergency planning studies, PLG-0432 and PLG-0465 (References 11 and 15). These studies reviewed the bases of NUREG-0396 for the current 10-mile evacuation emergency planning zone, and taking account for Seabrook-specific plant features and improvements, argued that a reduced EPZ was justified for Seabrook. PLG-0465 depends heavily on earlier Seabrook studies - the Seabrook Station Risk Management and Emergency Planning Study (PLG-0432, Reference 11) and the SSPSA. This BNL evaluation focused on the results of both PLG-0432 and PLG-0465.

At the request of the NRC, the review concentrated on the following areas:

- Interfacing system LOCA's
- Containment function
  - Isolation failure

  - Pre-existing leaks
  - Structural capacity

- Containment loads

- Seabrook-specific WASH-1400 Source Terms

- Site consequences model

During the review process, two additional areas originally outside the scope of the review were added as potentially important to risk at Seabrook. These areas were:

- Accidents at shutdown

- Induced Steam Generator Tube Rupture (SGTR)

These areas were not originally included in the review because in the past they were not thought to be dominant risk contributors. However, since the early health risk quantified in PLG-0432 and PLG-0465 are very low, events previously considered unimportant were felt to have the potential to influence the Seabrook risk estimates.

The approach taken by BNL was to perform sensitivity studies in selected areas of PLG-0465 to assess the impact on results. The conditional risk indices from PLG-0465 were used to assess the impact of varying the probability of sequences and containment performance on risk estimates. The review focused on assessing ways in which the Seabrook containment could fail or be bypassed early during a severe core melt accident.

The review found no areas of significant disagreement with PLG-0465, but did conclude that certain areas required additional analysis to validate the conclusions of PLG-0465. The major areas where it was felt that further analysis was required were direct containment heating, induced steam generator tube rupture, and shutdown events. Subsequent to this review, these and other areas of the review have been revisited and the results incorporated into the current Seabrook model.

Table A-2 summarizes the major findings of this review and Seabrook resolution of these issues where appropriate.

TABLE A-1

Significant LLNL Review Comments/Resolutions

LLNL Review Comments	NHY Resolutions
1. ATWS	
• Failure to scram should be divided into failures due to electrical failures and those due to mechanical failures.	• The current plant model differentiates between ATWS events resulting from electrical failures and those resulting from mechanical failures. The cause of the ATWS effects the success of possible operator actions and the required equipment actuation.
• Rather than using the favorable MTC which was analyzed to exist for 95% of the cycle, the relative fractions of cycle time the unfavorable MTC would occur should be factored into the quantification.	• The current model factors the relative cycle fractions that an unfavorable MTC exists.
• Whenever the ASME Level C criteria is exceeded, a core melt should be assumed, as in the ATWS rule, rather than an SLOCA.	• The current model conservatively assumes a core melt if ASME Level C stress limits are exceeded.
• A Diverse Turbine Trip System should be included in the model since this will be required for ATWS mitigation.	• The current model includes the Seabrook AMSAC System as a top event in the ATWS event tree.
2. V-SEQUENCE	
• The V-sequence analysis ignores possible lower consequence-higher probability scenarios.	• The current V-sequence analysis considers breaks of various sizes, from those which can be mitigated by the RHR relief valves, to large breaks of the RHR piping.
• Questioned whether the injection leg V-sequence would be more likely than the assumed suction line V-sequence.	• The current model considers both suction line and injection line failures as separate initiating events, with different possible scenarios for each.

TABLE A-1
(Continued)

Significant LLNL Review Comments/Resolutions

LLNL Review Comments	NHY Resolutions
3. ESFAS-SSPS	
• The ESFAS success criteria is overly conservative for most analysis.	• A top event for ESFAS recovery is included in the support tree. This allows for possible ESFAS recovery for the more slowly developing initiating events.
• The common cause contribution to SSPS appears invalid and incomplete.	• The common cause contribution of SSPS is dominated by calibration errors of the instrument transmitters. This common cause contribution is modelled in the current model
4. PCC-SW	
• Loss of a single train of PCC or SW will result in a plant shutdown and, therefore, should be included as an initiating.	• The loss of a single train of PCC and a single train of SW are included as initiating events in the current model.
5. EFW	
• No human actions for recovery of EFW were considered.	• EFW recovery is now explicitly modelled in the appropriate front line event trees and includes human actions to start the turbine driven EFW pump.
6. LOSP	
• The frequency of loss of off-site power was considered to be optimistic.	• An update to the Seabrook loss of off-site power was performed using industry data up to June, 1987. A two-step Bayesian update was performed, with industry data as the prior, and Seabrook site-specific data as the posterior distribution. This analysis is documented in PLG-0726 (Reference 14).

Significant LLNL Review Comments/Resolutions

LLNL Review Comments	NHY Resolutions
7. RCP Seal LOCA	
• The leak rate of RCP seal during a seal LOCA condition was questioned.	• An updated analysis of the RCP seal LOCA was performed and documented in PLG-0724, January, 1990 (Reference 13). This analysis examined a spectrum of expected leak rates and operator actions, and also a sensitivity analysis.
8. Seismic Fragilities	
• The seismic fragilities were felt to be overly conservative compared to previous PRAs. It was recommended that they be re-examined.	• The Seabrook seismic fragilities for important components were updated and documented in "Seismic Fragilities of Structures and Components at the Seabrook Generating Station Units 1 and 2," NTS Engineering, June 1986 (Reference 16). These resulting fragilities were more realistic and are incorporated in the current model.

TABLE A-2

Significant BNL Review Comments/Resolutions

BNL Review Comments	NHY Resolutions
1. Interfacing System LOCA's The frequency of interfacing system LOCA's is believed to be higher than estimated in PLG-0465 (Reference 15). However, even with the higher frequency estimates, interfacing system LOCA's were not a dominant contributor to Seabrook risk.	No significant effect on conclusions.
2. Shutdown Events Events occurring during shutdown may be significant and warrant further study.	A complete probabilistic safety assessment of potential accidents, including potential source terms and consequences has been performed for Seabrook since the BNL review. This study is documented in "Seabrook Station Probabilistic Safety Study - Shutdown (Modes 4, 5, and 6)," May 1988, (Reference 28). See Appendix B, Section B.7 for a summary of the Shutdown Study.
3. Induced Steam Generator Tube Rupture Sensitivity studies indicate that induced steam generator tube rupture is potentially risk important for core melt pressures.	The current Seabrook containment model in the SSPSS-1990 includes the potential for induced steam generator tube rupture at high pressure conditions with dry steam generators.

Significant BNL Review Comments/Resolutions

	BNL Review Comments	NHY Resolutions
4.	**Containment Isolation Failure and Pre-Existing Leaks**	
	Purge and vent valves in a fully closed configuration should provide reliable isolation under severe accident conditions up to the pressure corresponding to 1% hoop strain.	Agree.
	The approach to both large and small pre-existing leaks was reasonable.	The contribution from large and small pre-existing leaks is included in the current containment isolation model.
5.	**Containment Loads**	
	Seabrook-specific loads were not developed, but BNL-developed Zion-specific loads were used to obtain an estimate of Seabrook containment response. If the Zion median load were applied to the Seabrook containment the probability of early failure was judged to be very low. There are large uncertainties in containment loads, especially for high pressure due to direct containment heating, and therefore, the PLG-0465 contention that the contribution to early containment failure due to DCH is negligible cannot be confirmed.	The current Seabrook containment model explicitly models direct containment heating for high pressure core melts, with the potential for early-large containment failures. The quantification was based on a subsequent study (Reference 10) performed by NHY and its contractors using Sandia developed DCH pressure loads. This study resulted in a conditional probability of .001 for early containment failure following a DCH event. The potential for Containment failure due to DCH has been included in the SSPSS-1990 model.
6.	**Source Terms**	
	The fission product source terms used in PLG-0465 were reviewed in terms of their consistency with WASH-1400 and found appropriate.	Agree.

TABLE A-2
(Continued)

Significant BNL Review Comments/Resolutions

	BNL Review Comments	NHY Resolutions
8.	**Consequence Model** A comparison was made between the Seabrook developed CRACIT predictions of dose versus distance with results from the Sandia National Laboratory developed MACCS code. The results compared favorably and the dose versus distance results of PLG-0465 appear reasonable.	Agree.

APPENDIX B

SEABROOK RISK STUDIES

APPENDIX B

Seabrook Risk Studies

A number of risk studies have been performed for Seabrook Station since the original SSPSA was completed in 1983. These studies were performed to address specific issues and also resulted in updates and improvements to risk modeling. In the following sections a brief description of each major study is provided to document the changes that culminated in the SSPSS-1990.

B.1 Seabrook Station Probabilistic Safety Assessment - SSPSA (PLG-0300, Reference 3)

In December 1983, a full-scope, Level 3 PSA was completed for Seabrook Station. The purpose of the SSPSA was to provide a base line risk assessment and an integrated plant and site model for use as a risk management tool. The study was provided to the NRC and to the public for information in January 1984, (Reference 5).

The key findings of the SSPSA were:

- The mean severe core damage frequency was found to be 2.3E-4 events per reactor-year.

- Both the societal and individual risk provisions of the NRC safety goals were met by wide margins; therefore, the risk to public health and safety was estimated to be extremely small.

- Different risk factors were found to have different key contributors. Interfacing systems Loss-of-Coolant Accident (LOCA) events and, to a lesser extent, seismic-induced transient events with failure of containment isolation were the principal contributors to early health risk. The contributors to core melt frequency and latent health risk were comprised of a large group of initiators, including loss of off-site power, transient events, fires, and seismic events. A common event in many dominant sequences and in more than two-thirds of the total severe core damage frequency was the reactor coolant pump seal LOCA.

- The dominant contributors to severe core damage frequency were support system faults, external events, and internal hazards that affected both the core cooling and containment heat removal systems. As a result, a major fraction of the severe core damage frequency, about 73%, was associated with sequences in which long-term containment overpressurization was indicated.

- Only about 1% of the core melt frequency was associated with early containment failure or bypass. This percentage is more than 30 times less than that assumed in the Reactor Safety Study (Reference 24) for PWR plants. Its low value is the result of the high strength of the Seabrook containment as determined by more detailed analysis.

- In contrast with previous PSA containment analysis, the time of containment overpressurization due to failure to remove decay heat was found to be very long (several days instead of several hours).

B.2 Risk-Based Evaluation of Technical Specifications for Seabrook Station (PLG-0451, Reference 55)

PLG-0451 was published in August of 1985. Its primary objectives were to evaluate proposed changes to the Technical Specifications at Seabrook, to optimize them with respect to risk, and to quantify the risk impact of variations in the allowable outage times and surveillance-testing frequencies.

In performing this study, the key systems from the PSA were re-evaluated using new information about system success criteria and common cause failures. The quantitative impact of these re-evaluations on the results of the PSA was a small increase in the overall core damage frequency and a reduction in the likelihood of sequences involving containment isolation failure following seismically induced, station blackout events.

Seabrook Risk Studies

B.3 Seabrook Station Risk Management Emergency Planning Study (RMEPS) (PLG-0432, Reference 11)

The objective of the RMEPS, published in December 1985, was to update the SSPSA plant, containment, and consequence models to account for the latest available information on the plant systems, procedures, and accident source terms. One of the more significant changes was a major reassessment of the risk of interfacing systems LOCA scenarios to account for more realistic failures and operator responses. This affected both the frequency of this event and the consequences in terms of source term release. An updated risk model was then used to evaluate alternative emergency planning strategies to the standard ten-mile EPZ for maintaining public health risk at acceptably low levels.

The key conclusions of the RMEPS study are summarized, as follows:

- The updated risk assessment provided in the RMEPS showed that the acute health risk to the population surrounding Seabrook is very low in absolute terms and in relation to any known standards of acceptability of safety goals.

- Because the acute health risk levels are already very low assuming no evacuation, the potential for risk reduction by evacuation or sheltering to various distances from the site is also very low in absolute terms.

- Of the small amount of risk reduction achievable through prompt evacuation, a very large portion is achieved with close-in evacuation. More than 70% of the risk benefits from evacuation are realized with a one-mile evacuation distance. More than 95% of the risk benefits from evacuation are realized with a two-mile evacuation distance.

- There is no measurable difference in risk reduction between evacuation to ten-miles, and the combination of evacuation to two miles and sheltering for a distance of two to ten miles.

- Using the same rational basis as used in NUREG-0396 (Reference 31) to select a ten-mile EPZ for all U.S. sites, the results of RMEPS support an EPZ of less than one mile.

- Because of large margins between calculated risk levels and levels of acceptability, the above conclusions are generally insensitive to key uncertainties in the risk estimates.

B.4 Seabrook Station Emergency Planning Sensitivity Study (EPSS) (PLG-0465, Reference 15)

RMEPS, discussed in the previous section, used an updated version of the SSPSA model to evaluate emergency planning options for Seabrook. The RMEPS evaluation was based, in part, on new insights about the nature and magnitude of radioactive release source terms relative to the source-term technology that was used to develop the generic requirement for a ten-mile EPZ (Reference 56). The purpose of the EPSS was to determine the radius of the EPZ that can be justified for Seabrook Station without considering any advances in the source-term methodology since the completion of the Reactor Safety Study (Reference 24) in 1975. In the EPSS, the analyses performed in RMEPS were reassessed to delete credit for advances in source-term technology since WASH-1400.

The principal conclusion of EPSS and RMEPS was that an EPZ at Seabrook Station of one-mile radius or less was more fully justified for its risk management effectiveness that the current ten-mile EPZ was justified by the results of NUREG-0396. In RMEPS, this conclusion was based on the then most current information about all the risk factors of importance in emergency planning at Seabrook Station, including the most up-to-date PRA technology about source-term analyses. In the EPSS, a one-mile EPZ was still shown to be justified without accounting for any new insights about source terms since WASH-1400.

Seabrook Risk Studies

Because of significant differences between the EPSS and RMEPS regarding source terms, the absolute numerical results are correspondingly different. The principal difference is that the margins between the results and the acceptance criteria are smaller in the EPSS. In addition, the source term analysis in the RMEPS is still considered to be the best estimate while the analysis in the EPSS is a conservative sensitivity.

B.5 Risk Management Actions to Ensure Containment Effectiveness at Seabrook (PLG-0550, Reference 10)

Brookhaven National Laboratory (BNL) and the NRC staff reviewed New Hampshire Yankee's one-mile EPZ request (see Appendix A for a summary of that review). Although BNL did not challenge the principal conclusions of the New Hampshire Yankee submittals, it did request that additional work be performed to reduce the uncertainties associated with the frequencies of early containment failure or bypass scenarios. Four specific issues were identified in this review for further evaluation. They were:

- Possible early containment failure resulting from direct heating of the containment atmosphere by core debris (direct containment heating - DCH).
- Possible bypass of the containment by thermally induced failures of steam generator tubes (induced steam generator tube rupture - ISGTR).
- Risks associated with potential accidents initiated with the reactor shutdown.
- The potential for early radioactive releases from pre-existing containment leaks.

The purpose of PLG-0550 (referred to as the "PORV" Study) was to examine the first two issues, the potential risks from DCH and ISGTR scenarios, and to identify potential improvements in the plant hardware or procedures that could enhance the capability of the plant to respond to such events. (The other two issues are addressed in other studies described below - shutdown events in Section B.7 and pre-existing leaks in Section B.8).

The significant conclusions of the PORV Study were:

- A conservative assessment of the mean frequency of early containment failure or bypass because of ISGTR is 6E-10 per year.

- A conservative assessment of the mean frequency of containment failure because of DCH is 8.8E-8 per year. The result for DCH is based, in part, on the BNL assessment of the pressure capacity of the Seabrook containment and DCH pressure loads based on the current NRC contractor estimates.

- New procedures and plant modifications have been identified which can further reduce the risk from DCH and ISGTR. These include the expanded use of pressurizer PORVs to depressurize the RCS and the use of firewater pumps to feed the steam generators.

- This detailed examination of the DCH and ISGTR issues at Seabrook Station upholds the principal conclusions of PLG-0432 and PLG-0465. In view of the results and the fact that these issues were not addressed in NUREG-0396 (Reference 56), it would not be necessary to evacuate beyond one mile to achieve the level of protection of health and safety that had been perceived in NUREG-0396 for a ten-mile evacuation zone.

B.6 Seabrook Station Probabilistic Safety Study - 1986 Update, (SSPSS-1986, Reference 6)

This study was the first in-house effort to update the entire PSA to reflect the plant configuration as of mid-1986. A number of changes had been made from the SSPSA to this Study due to changes in the plant design from 1983 until 1986 and also due to model changes and enhancements in documentation. Significant changes are listed below:

- Plant Changes

  The following plant changes were reflected in the risk model.

- Technical Specifications - The allowed outage times have been changed for a number of systems, including Service Water System and Primary Component Cooling Water System (the standby pumps are now in the Technical Specifications), ECCS (AOT extended from 72 hours to seven days), Emergency Feedwater System (startup feed pump is now included along with two EFW pumps), containment on-line purge valves (allowed open time changed from 1,000 hr/year to unlimited duration but open only within guidelines).

- IST Pump Test Frequency - For all safety pumps except EFW pumps, the test frequency has been extended from monthly to quarterly.

- Startup Feed Pump - The startup feed pump is now self-cooled, rather than cooled by SCC; tested monthly with other EFW pumps.

- Turbine-Driven EFW Pump - New AOVs were added to the steam admission lines to the turbine driver.

- Atmospheric Relief Valves - ARVs are now powered by instrument air with gas accumulator backup rather than electrohydraulic.

- Boron Injection Tank and Associated Recirculation Pump and Bypass Line - These components have been removed.

- Enclosure Building Air Handling System - New one-out-of-two standby fans were added in the RHR vault return flow path.

- Reactor Trip Breakers - Shunt trip coil is now actuated by the automatic trip signal as well as the UV device.

- RCP Thermal Barrier Cooling System - The design has been finalized, includes several manual valves not in the SSPSA model.

- Model Changes

  The following model changes were made in this study:

  - Event Tree Qualification - The documentation and traceability of the event tree split fractions back to systems and operator action were enhanced by the use of unique split fraction identifiers. Also, the method for binning event tree quantification was better documented.

  - Seismic Analysis - The seismic fragilities of important components to the seismic risk were reanalyzed based on actual seismic qualification reports (Reference 16).

  - Systems Analysis - Quantification was done using RISKMAN-3 software. This enhances the traceability of the systems analysis back to the data as well as improves transcription errors.

  - Systems Analysis - Common cause treatment was expanded in this study to include more than two components failing together in common cause.

These changes impacted the system and plant models, and resulted in a mean core damage frequency of 2.7E-4/year. Other parts of the risk model - data, human action, containment, and consequence analyses - were unchanged from the original SSPSA model.

B.7 Seabrook Station Probabilistic Safety Study of Shutdown Events (Reference 28)

A study of risk during shutdown was initiated by New Hampshire Yankee in January 1987 to provide an explicit assessment of risk specifically for Seabrook Station use in future decision-making. The shutdown study was completed in May 1988.

This issue was identified during the BNL review of the reduced EPZ request (see Appendix A). This issue was raised because other analyzed contributors to early release scenarios were found to be so small that the relative importance of shutdown events could be significant. Also, because of their omission from previous PRAs and nearly all severe accident research programs, much less is known about the nature of the initiation and progression of shutdown events in comparison to power operation events.

This study concluded that the frequency of core damage is small, but not negligible, in comparison to power operation. This result was based on several improvements identified during the study. The risk at shutdown is strongly influenced by different plant configurations and equipment unavailabilities unique to shutdown. Modeling of operator action is important because of the absence of automatic control during shutdown. After assuming more restrictive controls on containment penetrations (i.e., hatch, purge valves, etc.), the frequency of early release was estimated to be comparable to that from power operation. However, the consequences were assessed to be less than power operation due to reduced source terms.

B.8 PRA of 40% Power Operation at the Seabrook Station (PLG-0631, Reference 59)

The objective of this study was to examine the risk of operation at 40% power and other possible compensating measures that would have allowed limited operation before the final resolution of emergency planning issues in Massachusetts. While that objective is no longer of interest, this study also rebaselined the total risk of operating Seabrook at 100% power, in order to adequately evaluate compensating measures and potential plant improvements. This rebaselining included conclusions reached in previous studies regarding direct containment heating, induced steam generator tube rupture, pre-existing containment leaks, and accidents occurring during shutdown.

B.9 Seabrook Station Probabilistic Safety Study - SSPSS-1989 Update (Reference 7)

This in-house study revises the 1986 update with plant changes through July of 1989. This update also includes enhanced system modeling, advanced PC-based software, and the containment failure/source-term enhancements.

The results of this study are summarized in the Management Summary Report - 1989 (Reference 60). These results indicated a core damage frequency of 1.1E-4/yr. This frequency is reduced approximately a factor of two from the original SSPSA results due to the changes listed below. However, the importance of the RCP seal LOCA remains the same - contributing 70% of the core damage frequency total. The estimate of early containment failure is decreased by a factor of 5 to 0.2% of the core damage frequency. This change is due to the incorporation of containment failure/source team enhancements brought together in the "40% Power Study," above.

This update included the following significant changes from the 1986 update:

- Systems
  - No significant design changes that impacted the risk model were found.
- Data
  - Initiating event frequencies were updated with data through 1987.
  - Common cause and maintenance distributions were updated based on additional industry data.
- Software
  - RISKMAN Release 2 software was used for system and plant models.
- Electric Power Recovery Model
  - Updated model with more current recovery data (Reference 12).

- Recovery - Recovery actions were integrated into the event tree model via a recovery tree placed at the end of the plant model.

B.10 Seabrook Station Probabilistic Safety Study - 1990 Update, (SSPSS-1990, Reference 8)

This study updates and replaces the 1989 update, with plant changes through July 1990.

The results of this study are summarized in the IPE Report. The significant changes include the following:

- Plant - ATWS Mitigation System was implemented which provides a diverse turbine trip and EFW actuation signal. This hardware update and an update of the ATWS analysis based on WCAP-11993 (Reference 19) were included in this update.

- Electric Power Recovery Model - This model is updated based on more current, PSNH-specific data for recovery of 345 kV grid, update of off-site power data, battery lifetime analysis update, and an update of the RCP seal LOCA analysis that is an input to this model.

- Computer Model - RISKMAN Release 2 software was used to create a fully integrated plant - containment model from initiating event to release category.

- Recovery - New recovery actions were added (OS, Signal Failure Recovery and RM, RWST makeup).

- Present recovery actions moved in plant model (EFW recovery added after event EF in frontline trees, SW recovery added to top events WA, WB in the support tree.

- Containment Event Tree - Updated to explicitly model ISGTR and DCH.

APPENDIX C

RISK MANAGEMENT PROGRAM

APPENDIX C

Risk Management Program

The Risk Management Program at New Hampshire Yankee utilizes the tools and insights developed via the SSPSA and subsequent studies. The purpose of this program is to provide quantitative and objective inputs into decision making processes to assure that the level of plant safety is maintained in a cost-effective manner.

The foundation for the Risk Management Program is the Seabrook Station Probabilistic Safety Study (SSPSS) and the concept called the "living PSA". In order to be usable in an ongoing program, the tools must be:

- Functional - Capable of performing detailed evaluations of model or data changes in a rapid manner.

- Updated - Reflect the current configuration of the plant in order to accurately evaluate changes.

- Documented - Fully documented so that models, methods, and assumptions can be traced from the raw data to the final risk assembly.

To accomplish these objectives, the concept of a living PSA has been developed. This consists not only of the SSPSS but also the process for documenting and updating the study in the future. The form of the documentation has evolved from the original SSPSA (8-volume report) to the present form to support this "living PSA" concept. The SSPSS-1990 consists of the following:

- Documentation Notebooks - Table C-1 lists these notebooks.

- Computer Codes - An improved PC-version of PLG RISKMAN suite of codes (Reference 9) is used to allow integrated computer modeling of data, systems, and event trees.

- Computer Files - Input/Output files with hard copy in notebooks.

- A Summary Report - The IPE Report serves as the summary report of the results for the 1990 update.

Thus, the living PSA is no longer a document to occupy space on a bookshelf but has become fully functional, supporting the updating of the models and documented in more detail than was possible in a formal report. The formal documentation has been reduced to the essential background information and results that will be useful to the general, technical audience.

Updating the living PSA consists of two general tasks. The first is updating the PSA models as the plant configuration changes, i.e., hardware and operational changes. The hardware changes are detected in the design review process, during which the changes are evaluated for their effect on the SSPSS as well as considerations of design adequacies using the SSPSS. Other significant, operational changes are identified and evaluated during periodic systems or operational evolution reviews. In general, it has been found that most plant hardware and procedure changes tend to be minor changes to the original plant with regard to their impact on plant risk. Also, the PSA is a macroscopic model of the plant. So long as the underlying plant logic does not change (e.g., success criteria), small changes to the original plant are usually not significant to the PSA models and results. Thus, the update effort is aimed at maintaining the models current with the overall plant configuration.

Secondly, the living PSA is updated by acquiring plant-specific data on component failure rates and maintenance unavailabilities. The present models are based on generic industry data and, thus, the results represent something of an industry average (although in plant-specific logic models). Plant-specific data will allow the SSPSS to better reflect the specific operating and maintenance characteristics of Seabrook Station. In addition, comparison with average data may allow identification of equipment whose performance is much better or worse than expected. The short-term data plan will be limited to the critical components identified in the systems analyses in the SSPSS. In the long-term, a large number of components will be monitored in conjunction with NPRDS to allow more accurate quantification of failure rates by component group (e.g., large MOVs, normally operating pumps, etc.).

Risk Management Program

With the living PSA as the basis for the ongoing Risk Management Program, a number of applications are possible, including the following:

- Risk Communication:
  - Current state of plant risk.
  - Training for operations/engineering.
- Design and Operational Changes: Evaluate the impact on risk, propose modifications to improve risk impact.
- Safety Enhancements: Identify cost-effective modifications to improve/assure the level of risk.
- Prioritization: Aid in prioritizing future design changes, maintenance activities (RCM), inspection/surveillance activities; input into a general cost/benefit program for major design changes.
- Regulatory Requirements: Provide input into backfit requirements.
- Operator Training: Provide risk-basis for the training for licensed operators.

TABLE C-1

Documentation Notebooks

1.0 INTRODUCTION

2.0 PLANT MODEL

- 2.1 Plant Model Overview
- 2.2 Initiating Events
- 2.3 Event Trees
  - 2.3.1 Seismic Response Tree
  - 2.3.2 Support Systems
  - 2.3.3 General Transient
  - 2.3.4 Long-Term Response
  - 2.3.5 LOCA
  - 2.3.6 Steam Line Break
  - 2.3.7 Steam Generator Tube Rupture
  - 2.3.8 ATWS
  - 2.3.9 V Sequence
  - 2.3.10 Recovery Actions

3.0 SYSTEMS ANALYSIS

- 3.1 Systems Overview
- 3.2 AC Power
- 3.3 DC Power
- 3.4 Off-Site Power
- 3.5 Service Water
- 3.6 Primary Componet Cooling Water
- 3.7 Solid State Protection System
- 3.8 Emergency Safeguards Features Actuation System
- 3.9 Reactor Trip System
- 3.10 Emergency Air Handling
- 3.11 Instrument Air
- 3.12 Emergency Core Cooling System
- 3.13 RCS Pressure Relief
- 3.14 Emergency Feedwater
- 3.15 Main Steam
- 3.16 Containment Building Spray
- 3.17 Containment Isolation

Documentation Notebooks

4.0 HAZARDS

- 4.1 Fire
- 4.2 Seismic
- 4.3 Other Hazards

5.0 HUMAN ACTIONS

6.0 DATA ANALYSIS

7.0 CONTAINMENT ANALYSIS

8.0 SITE/CONSEQUENCE ANALYSIS

9.0 RESULTS

APPENDIX D

EXTERNAL EVENTS SUMMARY

APPENDIX D

External Events Summary

The Seabrook Station PSS addresses various external events, including seismic events, internal and external floods, fires, turbine missiles, and aircraft and truck impacts. These events have been integrated into the plant model from the original SSPSA and have shown to be important to risk. For the SSPSS-1990, external events account for a larger percentage of the core damage risk due to reductions in the internal events contribution. Future work is planned in external events, specifically, fire and seismic analyses. However, the current results are believed to reasonably reflect the current state of risk.

Of the external events analyzed, fires and earthquakes contribute the most to the total core damage frequency. Table D-1 presents the contribution to total core damage frequency for various external event groupings. Fire hazards, seismic hazards, and "other" hazards are addressed in the following paragraphs.

D.1 Fire Hazards

Section 4.1 of the Seabrook Station PSS addresses internal fire hazards. The results, in terms of dominant sequences that were quantified, are summarized in Table D-2.

The fire analysis is unchanged from the original SSPSA (Section 9.4) except that the frequencies of fires in the major areas - Control Room, Cable Spreading Room, Auxiliary Building, and Turbine Building - have been updated with more current date (see Reference 40, PLG-0602).

Data and risk models in the SSPSA fire analysis were specialized to take into account important characteristics unique to fires. The methodology used for the evaluation of risk from fires is adapted from Reference 41.

The occurrence of fires and their effects on plant safety are complex issues. Therefore, the following conservative assumptions were made in order to perform the analysis.

APPENDIX D
(Continued)

External Events Summary

- The analysis is based on the location of important cables and equipment given in Reference 42. Additional information is provided in References 43 and 44. The separation between the two safety trains was investigated as part of the fire protection of the safe shutdown capability (Reference 43).

- The possibility of hot shorts in the control cables and their impact on the plant are considered explicitly only for a limited number of components and fire zones, where these effects were judged to have a significant impact on the vital systems availability.

- The frequencies of fires are derived from evidence collected from all U.S. nuclear power generating stations, as updated in Reference 40.

- The analysis of the accident sequences here is not as detailed as it could be. A more detailed analysis would explicitly include the timing of events, the possibility of errors of commission, etc. The purpose was only to find those scenarios which might dominate a plant damage state using conservative models.

- Except for the Cable Spreading Room and the Control Room, the impact of fires on instrumentation is not analyzed explicitly. It should be noted that whenever a fire is postulated in an area where it can affect instrumentation, the question of completeness of the analysis becomes very important. It is very difficult to know what information reaches the operators and how they respond. However, the impact of such events on the fire risk is judged to be covered by the conservatisms in the model and the uncertainties of the dominant scenarios.

- Several scenarios involve service water valves or air handling units. The impact of the same fire on the availability of the cooling towers could not be established explicitly from References 42 and 43. It was assumed that the cooling towers remain unaffected.

External Events Summary

As part of a spatial interactions evaluation and walkdown, candidate fire scenarios were identified for each location in the plant. These scenarios were conservatively quantified in order to identify the most important ones, which were then, subjected to a detailed evaluation. The evaluation included estimating a severity and a geometry factor, detection, and suppression capabilities, and operator response.

D.2 Seismic Hazards

Seismic hazards are addressed in Section 4.2 of the Seabrook Station PSS. The seismic risk analysis consists of using a seismicity analysis to determine the frequency of ground motion acceleration of various levels at the site and a fragility analysis to determine the ground acceleration at which plant structures and components are predicted to fail. This evaluation is based on the SSPSA Section 9.2 with the updated fragility analysis described below.

The seismicity study was a Seabrook site-specific evaluation performed by Dames & Moore (see SSPSA, Appendix F, Section F.1). Six sets of curves were developed to reflect uncertainty from different seismologic hypothesis. The fragility analysis evaluated the seismic capacity of all major safety-related structures and components. This evaluation was conducted by Structural Mechanics Associates and is documented in the SSPSA, Appendix F, Section F.2. This evaluation was updated in Reference 16 to account for seismic qualification reports for specific components that were shown to be important in the SSPSA.

As a result of these evaluations, no structures and only a small set of components have a median acceleration capacity less than 2.0g. These components are listed in Table D-3 and are discussed below. The key components that are important to core melt or release mitigation are indicated by an asterisk.

1. Off-Site Power Supply*

   Loss of off-site power would cause an immediate loss of load and reactor trip requiring automatic start and load of the standby diesels. It is assumed for this analysis that seismic loss of off-site power is unrecoverable. The lower bound threshold for the median fragility curve for off-site power is set at about 0.1g at which level the assumption of no recovery is conservative.

2. 4,160 V Switchgear, Instrument Buses, MCCs, 480 V Transformers and Buses*

Chattering of the 4,160 V switchgear could potentially cause loss of off-site power and failure of the diesels to be able to load onto the bus. This failure mode has the lowest fragility of any essential electrical equipment listed. The current conservative treatment of chattering assumes failure for accelerations above 0.7g; at or below 0.7g, it is assumed that the likelihood of not recovering loss of power due to chatter is 0.1. (Seismic chattering will be examined in more detail in the future).

3. Spray Additive Tank (SAT)

Seismic failure of the SAT would not initiate an accident sequence. In addition, the SAT was judged not to be significant to reduction in the fission product aerosol inventory in containment following a core melt. This is due to the effectiveness of containment spray. Also, the SAT plays no role in core melt mitigation.

4. RWST*

Loss of RWST does not initiate an accident sequence but does result in failure of ECCS injection, which is needed to mitigate a seismic large LOCA.

5. Control Room Evaporator Units

It is assumed that on loss of Control Room evaporator units, there will be no immediate impact on Control Room habitability, and operators will take the appropriate action long term. This failure is assumed to not cause an initiating event.

6. Reactor Internals*

Seismic failure of the reactor internals is assumed to cause control rods to jam and not position for reactor shutdown. This failure would initiate an ATWS.

External Events Summary

7. Diesel Generators*

Failure of diesels due to a seismic event, which also causes loss of off-site power, results in a station blackout which, it is assumed, cannot be recovered. The diesels are assumed to be coupled in that, if one fails seismically, the second is guaranteed failed.

8. Steam Generators*

Seismic failure of steam generators is assumed to cause leaks in the Primary Coolant System by failure of anchor bolts allowing the generators to tilt, initiating a large LOCA sequence.

9. Cooling Tower Fans

The cooling tower fans and pumps are redundant to the service water pumps, which have a much higher equipment fragility. Thus, cooling tower fans' seismic failures are not of high risk importance.

10. Reactor Coolant Pumps*

As with steam generators, seismic failure of RCPs is assumed to cause a large LOCA.

11. Reactor Building Crane

During plant operation, the crane is parked in a position where it does not endanger equipment that could initiate or mitigate an accident sequence. That reason and its relatively high seismic capacity are the basis for eliminating the crane from further consideration.

12. MSIVs

Seismic failure of the MSIVs is assumed to cause the valves to close and, thus, force the safety valves to open to relieve the steam. This is not modelled

further because of the high fragility and because of the other components that would have to fail before this became a core melt accident.

These key seismic components are used in the plant model as sequence initiator and subsequent system component failures. These initiators included:

- Transients - Seismic events at 0.1g or greater, assume causing reactor trip.
- ATWS - Seismic events at 0.4g or greater, resulting in failure of reactor internals, causing control rods to jam outside the core.
- Large LOCA - Seismic events at 0.5g or greater, resulting in failure of steam generators or reactor coolant pumps, severing the RCS piping.

These seismic initiators are linked to seismic-induced system failures (i.e., off-site power, diesel generators, 4,160 V buses - relay chatter, and/or RWST) in the seismic tree and to nonseismic hardware failures in the plant event trees to create seismic sequences.

D.3 Other External Hazards

Other external hazards are discussed in Section 4.3 of the Seabrook Station PSS. They include the following:

- Internal Flooding
- External Flooding
- Hazardous Chemicals/Transport
- Wind and Tornados
- Turbine Missiles
- Aircraft Crash

Table D-4 summarizes the results for these hazards.

These hazards were each analyzed based on a Seabrook site-specific evaluation. A conservative evaluation was performed using FSAR information when available, with a screening criteria of 1E-7 for core damage sequences and a 1E-9 for core damage/early

release sequences. The present evaluation of these hazards is essentially the same as in the SSPSA (Section 9) except:

- The frequencies were adjusted to account for plant availability (i.e., the conditional likelihood of the plant being at power when the hazard occurs).
- Tornado analysis was revised slightly based on a change in the calculation of tornado frequency.
- Turbine missile analysis was updated to delete Unit 2 missiles on Unit 1.
- A consistent screening criteria, discussed above, was applied.

TABLE D-1

External Initiating Events - Contribution to Core Damage Total

	External Initiating Event	Percentage of Total Core Damage Frequency*	
1.	Fire	-	24.4%
	• Fire in Control Room Causing Loss of Support Systems (ac power, PCC, SW)	9.4%	
	• Fire in Turbine Building - Loss of Off-Site Power	8.3%	
	• Fire in PCC Pump Area	2.8%	
	• Fire in Electrical Tunnels	2.1%	
	• Fire in Cable Spreading Room - Loss Support Systems	1.8%	
2.	Seismic Event	-	13.4%
	• Seismic-Initiated Station Blackout	8.9%	
	• Seismic-Initiated ATWS	3.1%	
	• Seismic-Initiated Large LOCA	1.4%	
3.	Flood	-	6.2%
	• Internal Flood in Turbine Building	5.2%	
	• External Flood Causing SW Failure	1.0%	
4.	LOSP	-	1.3%
	• Truck Crash Into SF_6 Lines	1.3%	
5.	Others	-	0.1%
	• Aircraft Crash - Loss of PCC	0.1%	
	• All Others	<<0.1%	
	External Initiating Events - TOTAL	45.4%	45.4%

* The results presented are based on mean values. The distribution for external events generally have higher levels of uncertainty than internal-initiated sequences.

TABLE D-2

Fire Scenario Event Frequencies

Fire Initiating Event	Scenario Description	Mean Frequency (Per Calendar Year)
FSRCC	Fire in the Cable Spreading Room causing loss of PCC.	1.76E-6
FSRAC	Fire in the Cable Spreading Room causing loss of all ac power (station blackout).	2.54E-7
FCRCC	Fire in the Control Room causing loss of PCC.	7.18E-6
FCRSW	Fire in the Control Room causing loss of SW.	1.68E-6
FCRAC	Fire in the Control Room causing loss of all ac power (station blackout).	1.68E-6
FET1	Fire in the Electrical Tunnel No. 1 causing loss of one train of ac power.	2.52E-4
FET3	Fire in the Electrical Tunnel No. 3 causing loss of one train of ac power.	1.26E-4
FPCC	Fire in the PAB causing loss of PCC.	3.12E-6
FTBLP	Fire in the Turbine Building causing loss of off-site power.	1.23E-3

TABLE D-3

Seabrook Components for Seismic Analysis (With Medium Acceleration Capacity <2.0g)

Equipment	Median Acceleration Capacity a	β_R	β_U
* Off-Site Power Supply	0.30	0.25	0.50
* 4160 V Switchgear (Chatter/Structural)	0.70/>2	0.30	0.21
Spray Additive Tank	0.75	0.40	0.32
120 V AC Instrument Buses (Chatter/ Structural)	1.48/1.37	0.23	0.50/0.28
480 V Motor Control Centers (Chatter/ Structural)	1.41/1.78	0.30	0.37/0.20
480 V Transformers, Buses (Chatter/ Structural)	1.78/1.96	0.28/0.30	0.30/0.48
* RWST	1.31	0.41	0.31
Control Room Evaporator Units (Diesel Generator Building)	1.18	0.16	0.50
* Reactor Internals	1.50	0.38	0.44
* Diesel Generators	1.51	0.36	0.35
* Steam Generators	1.71	0.36	0.39
Service Water Cooling Tower Fans	1.71	0.41	0.39
* Reactor Coolant Pumps	1.74	0.35	0.32
Reactor Building Crane	1.75	0.25	0.55
MSIVs	1.86	0.41	0.41

* Key Seismic Components Included in Model

TABLE D-4

Summary of Results for Other Hazards

Category	Initiator	Mean Frequency (Per Year)	Impact
Internal Flood	FLLP	6.9E-4	Nonrecoverable loss of off-site power.
	FL1SG	5.4E-6	Nonrecoverable loss of off-site power and failure of Bus E5 (Train A).
	FL2SG	1.9E-7	Nonrecoverable station blackout.
External Flood	FLSW	1.1E-6	Failure of all SW (pumphouse and tower).
Chemicals/ Transport	TCTL	1.9E-4	Nonrecoverable loss of off-site power.
Winds/Tornados	(None)	--	
Turbine Missile	TMLL	1.4E-8	Large LOCA, failure of CBS and CIS.
Aircraft Crash	APC	8.5E-9	Large LOCA, failure of CBS and CIS.
	APAB	1.4E-7	Failure of all PCC.

APPENDIX E

SYSTEMS ANALYSIS SUMMARY

APPENDIX E

Systems Analysis Summary

This section contains summaries of the systems analysis from the SSPSS-1990, Section 3.0. Each summary contains: (1) a brief description of the system function, configuration, dependencies, and operation; (2) a brief description of the system model, i.e., top events, success criteria and analysis conditons; and (3) the quantitative results. The systems analyzed in the following:

AC Power - Section E.2
DC Power - Section E.3
Off-Site Power - Section E.4
Service Water - Section E.5
PCC - Section E.6
SSPS - Section E.7
ESFAS - Section E.8
RTS - Section E.9
EAH - Section E.10
Instrument Air - Section E.11
ECCS - Section E.12
RCS Pressure Relief - Section E.13
EFW - Section E.14
Main Steam - Section E.15
CBS - Section E.16
CIS - Section E.17

The results of these systems analyses are used to calculate plant model split fractions, as explained in Section E.1.

E.1 Split Fraction Quantification

A summary of system unavailability results from Sections E.2 through E.17 is presented in Table E-1. In order to use the system's results in the plant model, the event tree split fractions must be calculated from systems and operator action results. Split

fractions, which serve as the interface between the system's models and the plant model, are the conditional likelihood of failure of the top event, given success or failure of the previous top events. The top event may include:

- A system [e.g., split fraction RT1 = RT1 (RTS System result)],
- A combination of systems [e.g., split fraction EFA = EF1 (EFW System result) + ARVSRV (Steam Relief System result)],
- An operator action [e.g., split fraction OM1 = OMS (operator action result)],
- A combination of system and operator action [e.g., split fraction OR4 = PR1 (Primary Relief System result) + OR1 (operator action, initiate feed and bleed)], or
- A train of a system [e.g., split fractions WA1, WBA are single train values for Service Water System].

The calculation of split fractions from systems results is performed using the equations listed in Table E-2. Table 3.4-4(a) presents the quantitative results for each split fraction in the Master Frequency File.

The calculation of split fractions from systems results is straightforward except where the results for two-train systems have to be separated into Train A and Train B split fractions. Because of the dependencies between the two trains (due to maintenance, common cause failures, etc.), at least two conditional split fractions must be calculated for the second top event given success or failure of the first top event. In the following examples, WA and WB are top events representing two trains of Service Water System. In this case, two fault tree models are generated and quantified, one for a single train (SW4) and the other for both trains (SW1). From these two values, all split fractions for Top Events WA and WB can be calculated as follows:

WA1 = SW4 - Train WA failure.

WBA = $\frac{SW1}{SW4}$ - Train WB failure, conditional on WA failure.

WB1 = $\frac{(SW4 - SW1)}{(1-SW4)}$ - Train WB failure, conditional on WA success.

This can be seen from the following derivation:

The single train failure can be defined in terms of system results and split fractions, as follows:

$\overline{WA}$ ≡ Single train failure (including common cause failures) ≡ SW4
≡ WA1

Then, the failure of both trains is equal to:

$\overline{WA} * \overline{WB}$ ≡ Failure of both trains ≡ SW1
≡ WA1 * WBA

Where WBA ≡ $\overline{WB}$ given $\overline{WA}$ (i.e., WB failed, given WA failure)

Based on symmetry between trains,

$\overline{WA} * \overline{WB}$ ≡ WA * WB

This equation can be written in terms of split fractions, as follows:

WA1 * (1-WBA) = (1-WA1) * WB1

Where WB1 ≡ $\overline{WB}$ given WA (i.e., WB failed, given WA success)

Systems Analysis Summary

Writing these equations in terms of system unavailabilities (SW1, SW4) and split fractions (WA1, WB1, WBA):

$$SW1 = WA1 * WBA$$
$$SW4 = WA1$$
$$WA1 * (1 - WBA) = (1 - WA1) * WB1$$

With three equations, the three split fractions (WA1, WB1, WBA) are determined in terms of system unavailabilities:

$$\rightarrow WA1 = SW4 \qquad \text{(Equation E.1)}$$

$$\rightarrow WBA = \frac{SW1}{SW4} \qquad \text{(Equation E.2)}$$

$$SW4 * \left(1 - \frac{SW1}{SW4}\right) = (1 - SW4) * WB1$$

$$\rightarrow \quad WB1 = \frac{SW4}{(1 - SW4)} * \left(\frac{SW4 - SW1}{SW4}\right) = \frac{(SW4 - SW1)}{(1 - SW4)} \qquad \text{(Equation E.3)}$$

The split fractions are quantified in RISKMAN using equations similar to Equations E.1, E.2, and E.3 for all two-train systems modelled as individual trains. Table E-2 contains the actual equations used to translate system results to split fraction values.

Table E-1 System Unavailabilities (Sheet 1 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
Emergency AC Power	DGAB	7.4650E-03	Loss of Both Diesel Generators
	DG1	7.3430E-02	Loss of One Diesel Generator
DC Power System	DC1	2.7310E-10	Loss of Both DC Trains - Offsite Power Available
	DC2	1.3690E-06	Loss of Both DC Trains - LOSP (Station blackout for 1st 6 hrs)
	DC3	1.2130E-05	Loss of One DC Train - Offsite Power Available
	DC4	8.9220E-04	Loss of One DC Train - LOSP (Station blackout for 1st 6 hrs)
Service Water System	SW1	3.3150E-04	Loss of Both SW Trains Given SI Signal, No LOSP *
	SW2	3.9540E-07	Loss of Both SW Trains Given No SI Signal, No LOSP *
	SW3	7.9000E-04	Loss of Both SW Trains Given Loss of Offsite Power
	SW4	4.3440E-03	Loss of One SW Train Given SI Signal, No LOSP *
	SW5	6.0130E-05	Loss of One SW Train Given No SI Signal, No LOSP *
	SW6	1.3370E-02	Loss of One SW Train Given Loss of Offsite Power
	SW7	3.7010E-04	Loss of Both SW Trains Given SI Signal, No LOSP
	SW8	4.0930E-05	Loss of Both SW Trains Given No SI Signal, No LOSP
	SW9	5.5650E-03	Loss of One SW Train Given SI Signal, No LOSP
	SWA	1.2860E-03	Loss of One SW Train Given No SI Signal, No LOSP
Primary Component	PCC1	2.6230E-06	Loss of Both PCC Trains Given No LOSP & No P Signal
Cooling Water System	PCC2	4.4420E-05	Loss of Both PCC Trains Given Offsite Power Unavailable
	PCC3	2.6230E-06	Loss of Both PCC Trains Given P Signal Present
	PCC4	6.3730E-04	Loss of One PCC Train Given No LOSP & No P Signal
	PCC5	1.6270E-03	Loss of One PCC Train Given Offsite Power Unavailable (LOSP)
	PCC6	6.3730E-04	Loss of One PCC Train Given P Signal Present
Solid State	SC1	4.0140E-04	Loss of Both SSPS Trains Given LLOCA, MLOCA, or SLBI
Protection System	SCA	1.6060E-03	Loss of One SSPS Train Given LLOCA, MLOCA, or SLBI
(SSPS)	SC2	2.0220E-06	Loss of Both SSPS Trains Given Small LOCA

* Some human recovery actions have been included in the system analysis

Table E-1 System Unavailabilities (Sheet 2 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
SSPS (cont)	SCB	1.2070E-03	Loss of One SSPS Train Given Small LOCA
	SC3	4.1180E-06	Loss of Both SSPS Trains Given GT, SGTR, or SLBO
	SCC	1.2090E-03	Loss of One SSPS Train Given GT, SGTR, or SLBO
Engineered Safety	EC1	7.6650E-05	Failure of Both ESFAS Trains Given LLOCA or MLOCA
Features Actuation	ECA	1.0790E-02	Failure of One ESFAS Train Given LLOCA or MLOCA
System (ESFAS)	EC2	7.7900E-05	Loss of Both ESFAS Trains Given SLBI
	ECB	1.1460E-02	Loss of One ESFAS Train Given SLBI
	EC3	6.9730E-05	Loss of Both ESFAS Trains Given Small LOCA
	ECC	8.5500E-03	Loss of One ESFAS Train Given Small LOCA
	EC4	6.9730E-05	Loss of Both ESFAS Trains Given SGTR
	ECD	8.5500E-03	Loss of One ESFAS Train Given SGTR
	EC5	7.0750E-05	Loss of Both ESFAS Trains Given SLBO
	ECG	9.2210E-03	Loss of One ESFAS Train Given SLBO
	EC6	2.4710E-05	Loss of Both ESFAS Trains Given General Transient
	ECH	1.1600E-03	Loss of One ESFAS Train Given General Transient
Reactor Trip System	RT1	1.3710E-04	Failure of RTS Given Both SSPS Trains Available
	RT2	1.7710E-03	Failure of RTS Given One SSPS Train Available
	RT3	7.4800E-06	Failure of Control Rod Drives
Emergency Air	EH1	6.0410E-05	Loss of EAH Given All Support Available
Handling System	EH2	1.3220E-03	Loss of EAH Given Loss of One T Signal
	EH3	1.0150E-02	Loss of EAH Given Loss of One PCC Train
	EH4	1.1380E-02	Loss of EAH Given Loss of One PCC Train and One T Signal
	EH5	6.9920E-05	Loss of EAH Given LOSP and All Support Available
	EH6	1.3310E-03	Loss of EAH Given LOSP and Loss of One T Signal
	EH7	1.5670E-03	Loss of EAH Given LOSP and Loss of One Emergency Bus
	EH8	2.0970E-03	Loss of EAH Given LOSP, Loss of 1 Emer. Bus and 1 T Signal

* Some human recovery actions have been included in the sy[illegible]analysis

Table E-1 System Unavailabilities (Sheet 3 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
Emergency Core	LI1	4.4540E-03	LPI(1)LL LP Injection BC 1 (ASSA), BC3(1 PCC Train)
Cooling System (ECCS)	LI2	1.5360E-02	LPI(2)LL - BC 2 (1 Emergency Bus or 1 Signal)
	LP1	1.2590E-05	LPR(1)LL LP Recirc BC 1
	LP2	1.1940E-03	LPR(2)LL - BC 2 (1 EBus or 1 Sig), BC3(1 PCC Train)
	R51	7.3900E-07	HLR(1)LL Hot Leg Recirc. BC 1 (ASSA), BC3 (1 PCC)
	R52	1.1140E-06	HLR(2)LL - BC 2 (1 Emergency Bus or 1 Signal)
	HB1	3.4650E-04	RHRHX(1)LL RHR Heat Exchange Cooling - BC 1 (ASSA)
	HB2	4.3630E-03	RHRHX(2)LL - BC 2 (1 EBus or 1 Signal), BC3 (1 PCC)
	H11	3.4520E-05	HPI(1)ML - MLOCA - BC 1 (ASSA)
	H12	3.0930E-02	HPI(2)ML - BC 2 (1 Emergency Bus or 1 Signal)
	H13	1.6560E-02	HPI(3)ML - BC 3 (1 PCC Train)
	MM1	5.9830E-04	RHRM(1)ML Miniflow Recirc BC 1 (ASSA)
	MM2	1.7850E-02	RHRM(2)ML - BC 2 (1 EBus or 1 Signal), BC 3 (1 PCC)
	H31	1.1270E-03	HPI(1)ATWS - BC 1 (All Support Sytems Available)
	H32	2.2260E-02	HPI(2)ATWS - BC 2 (1 Emergency Bus or 1 Signal)
	H33	8.0780E-03	HPI(3)ATWS - BC 3 (1 PCC Train)
	H21	1.1680E-06	HPI(1)SL - SLOCA & TRANS - BC 1, (ASSA)
	H22	1.9500E-04	HPI(2)SL - BC 2 (1 Emergency AC Bus or 1 Sig Avail)
	H23	7.3340E-05	HPI(3)SL - BC 3 (1 Train of PCC Available)
	T1A	1.6490E-08	HPR(1A)SL HP Recirc BC 1A (ASSA, SI Pump B Unavail)
	T1B	1.6490E-08	HPR(1B)SL - BC 1B (ASSA, SI Pump A Unavailable)
	T1C	4.0460E-08	HPR(1C)SL - BC 1C (ASSA, 1 CVCS Pump Unavailable)
	T2A	1.2440E-06	HPR(2A)SL - BC 2A (Train B EBus or Signal Unavail)
	T2B	1.2420E-06	HPR(2B)SL - BC 2B (Train A EBus or Signal Unavail)
	T3A	1.2420E-06	HPR(3A)SL - BC 3A (PCC Train B Unavailable)
	T3B	1.2400E-06	HPR(3B)SL - BC 3B (PCC Train A Unavailable)
	TA1	3.2180E-06	HPR(4AA)SL-BC 4AA-Smp A or RHR Pmp A Unav;SI B Unav
	TA2	1.9630E-08	HPR(4AB)SL-BC 4AB-Smp A or RHR Pmp A Unav;SI A Unav
	TA3	4.2630E-08	HPR(4AC)SL-BC 4AC-Smp A or RHR Pmp A Unav;1CVCSP Un

* Some human recovery actions have been included in the system analysis

Table E-1 System Unavailabilities (Sheet 4 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
ECCS (cont)	TB1	1.6500E-08	HPR(4BA)SL-BC 4BA -Sump B or RHR B Unavl;SI B Unavl
	TB2	1.6490E-08	HPR(4BB)SL-BC 4BB -Sump B or RHR B Unavl;SI A Unavl
	TB3	4.1430E-08	HPR(4BC)SL-BC 4BC -Sump B or RHR B Unavl;1 CVCSP Un
	SM1	6.0840E-04	RHRM(1)SL Miniflow Recirc BC 1 (ASSA)
	SM2	1.8100E-02	RHRM(2)SL - BC 2 (1 EBus or 1 Signal), BC3 (1 PCC)
	LR1	5.1670E-04	LR(1)SL Long Term Cooling - BC 1 (ASSA)
	LR3	1.1190E-02	LR(3)SL Long Term Cooling - BC 3 (1 PCC Train)
	SP1	8.2190E-06	RHRHPR(1)SL RHR pumps for HPR - BC 1 (ASSA)
	SP2	1.1210E-03	RHRHPR(2)SL - BC 2 (1 EBus or 1 Signal), BC3(1 PCC)
	SR1	7.9810E-06	LPR(1)SL RHR pumps for LPR - BC 1 (ASSA)
	SR2	1.0110E-03	LPR(2)SL - BC 2 (1 EBus or 1 Signal), BC3 (1 PCC)
	RWLL	2.7620E-08	RWST (1 HOUR) - LLOCA
	RWML	5.5240E-08	RWST (2 HOURS) - MLOCA
	RWSL	1.6570E-07	RWST (6 HOURS) - SLOCA
	RALL	1.0260E-04	RWST Outlet Valve V2 - LLOCA
	RBLL	1.0260E-04	RWST Outlet Valve V5 - LLOCA
	RAML	1.0270E-04	RWST Outlet Valve V2 - MLOCA
	RBML	1.0270E-04	RWST Outlet Valve V5 - MLOCA
	RASL	1.0300E-04	RWST Outlet Valve V2 - SLOCA
	RBSL	1.0300E-04	RWST Outlet Valve V5 - SLOCA
	V11	3.5150E-04	CRS(1)LL Containment Recirc. Sumps - BC 1, BC 3
	V12	4.8780E-03	CRS(2)LL - BC 2 (1 Emergency Bus or 1 Signal)
	V21	3.4600E-04	CRS(1)SL Containment Recirc Sumps - BC 1, BC 3
	V22	4.3170E-03	CRS(2)SL - BC 2 (1 Emergency Bus or 1 Signal)
Reactor Coolant System Pressure Relief (RCPR)	PS1	1.2840E-03	RC Pressure Relief - Severe ATWS
	PS2	6.1830E-03	RC Pressure Relief - Severe ATWS, Single Train AC/DC
	PS3	9.8250E-04	RC Pressure Relief - ATWS
	P21	5.8690E-02	Safety and Relief Valves Reseat - ATWS

* Some human recovery actions have been included in the sys[illegible]nalysis

Table E-1 System Unavailabilities (Sheet 5 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
RCPR (cont)	PR1	1.0460E-02	PORV in Feed and Bleed
	PR2	3.3140E-04	PORV in Chemical Shutdown - ATWS
	PR3	5.2300E-03	PORV in Chemical Shutdown - ATWS, Single Train AC/DC
Emergency Feedwater	EF1	2.7300E-04	Normal Configuration - No Startup Feed Pump
System (EFW)	EF2	4.7580E-02	Turbine Driven Pump Only
	EF3	5.3380E-03	Motor Driven Pump Only
	EF4	5.7190E-03	Startup Feed Pump - Auto Start
	EF5	5.6420E-03	EFW1 - Feeding All 4 SGs - ATWS (TDP and MDP)
	EF6	5.3240E-02	EFW2 - Feeding All 4 SGs - ATWS (TDP only)
	EF7	1.1000E-02	EFW3 - Feeding All 4 SGs - ATWS (MDP only)
	EFRTDP	5.5280E-01	TDP Recovery Fraction *
	EFRSFP	8.6670E-03	SFP Recovery (Manual Actuation) *
Main Steam System	ARVSDV	3.2430E-08	Atmos Relief Valves & Cond Steam Dump Valves
	ARVV	1.0560E-04	Atmos Relief Valves Only
	MS1	1.3230E-04	MSIV Isolation - SLB or Turbine Trip Failure
	IV1	1.5230E-03	MSIV and Bypass Isolated - SL Tree - SGTR
	SS1	5.3610E-03	Steaming SG Isolated - SL Tree - SGTR
	SO1	9.5780E-03	Safety Valves Open/Close,Steam Relief - SL Tree - SGTR
	SO2	2.9140E-01	Safety Valves Open/Close,Water Relief - SL Tree - SGTR
	SV1	4.8310E-08	Safety Valve Action for ATWS
	TT1	4.4560E-06	Turbine Trip for Non - TT Failure Initiating Events
	TT1	0.0000E+00	Turbine Trip for Non - TT Failure Initiating Events
Containment Building	CBSCA1	6.3140E-04	CBS INJECTION - All Support Systems Available (ASSA)
Spray (CBS) System	CBSCA2	9.7590E-03	CBS INJECTION - Single Support Train Available (SSTA)
	CBSXA1	1.6560E-04	CBS PUMP RECIRC W/O HX COOLING - ASSA
	CBSXA2	6.3520E-03	CBS PUMP RECIRC W/O HX COOLING - SSTA

* Some human recovery actions have been included in the system analysis

Table E-1 System Unavailabilities (Sheet 6 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
CBS (cont)	CBSVA1	3.1870E-04	CBS HEAT EXCHANGER COOLING DURING RECIRC - ASSA
	CBSVA2	4.2860E-03	CBS HEAT EXCHANGER COOLING DURING RECIRC - SSTA
	CBSXC1	1.1810E-03	CBS RECIRC: START & RUN W/ HX COOLING - ASSA
	CBSXC2	1.9870E-02	CBS RECIRC: START & RUN W/ HX COOLING - SSTA
Containment Isolation	CIA	4.4950E-03	All Support Available
System (CIS)	CIB	8.4490E-03	Loss of Train B Power and All Signals Available
Small Lines (< 3")	CIC	4.1680E-03	Loss of Train A Power and All Signals Available *
	CID	5.1640E-03	Loss of Both Trains of Power and All Signals Available *
	CIE	1.7100E-02	Loss of Train B Signal and Both Buses Available
	CIF	1.7100E-02	Loss of Train B Signal and Power
	CIG	1.0000E+00	Loss of Train B Signal and Train A Power
	CIH	1.2130E-02	Loss of Train B Signal and All Power *
	CII	4.4500E-03	Loss of Train A Signal and All Power Available
	CIJ	1.0000E+00	Loss of Train A Signal and Train B Power
	CIK	1.6830E-02	Loss of Train A Signal and Power
	CIL	1.0000E+00	Loss of Train A Signal and All Power
	CIM	1.0000E+00	Loss of All Signals With All Power Available
	CIN	1.0000E+00	Loss of All Signals and Train B Power
	CIO	1.0000E+00	Loss of All Signals and Train A Power
	CIP	1.0000E+00	Loss of All Signals and All Power
	CIT	1.0000E+00	Loss of CIS Given APC or TMLL - Guaranteed Failure
	CI1	4.4950E-03	Seismic - All Support Available
	CI2	8.4490E-03	Seismic - Loss of Train B Power and All Signals Available
	CI3	8.4490E-03	Seismic - Loss of Train A Power and All Signals Available
	CI4	1.0000E+00	Seismic - Loss of All Power and Signals - Guaranteed Failure
	CI5	1.6830E-02	Seismic - Loss of Train B Signal and All Power Available
	CI6	1.6830E-02	Seismic - Loss of Train B Signal and Power
	CI7	1.6830E-02	Seismic - Loss of Train A Signal and All Power Available

* Some human recovery actions have been included in the system analysis

Table E-1 System Unavailabilities (Sheet 7 of 7)

SYSTEM/FUNCTION	SPLIT FRACTION	UNAVAILABILITY	DESCRIPTION
Small CIS (cont)	CI8	1.6830E-02	Seismic - Loss of Train A Signal and Power
Containment Isolation	C2A	1.0960E-04	All Support Available
System (CIS)	C2B	1.0960E-04	Loss of Train B Power and All Signals Available
Large Lines (> 3")	C2C	1.0960E-04	Loss of Train A Power and All Signals Available
	C2D	9.8370E-05	Loss of Both Trains of Power and All Signals Available
	C2E	4.0240E-04	Loss of Train B Signal and All Power Available
	C2F	4.0240E-04	Loss of Train B Signal and Power
	C2G	4.0240E-04	Loss of Train B Signal and Train A Power
	C2H	9.8370E-05	Loss of Train B Signal and All Power
	C2I	4.0240E-04	Loss of Train A Signal and All Power Available
	C2J	4.0240E-04	Loss of Train A Signal and Train B Power
	C2K	4.0240E-04	Loss of Train A Signal and Power
	C2L	9.8370E-05	Loss of Train A Signal and All Power
	C2M	1.0010E-01	Loss of All Signals and All Power Available
	C2N	1.0010E-01	Loss of All Signals and Train B Power
	C2O	1.0010E-01	Loss of All Signals and Train A Power
	C2P	9.8370E-05	Loss of All Signals and All Power
	C2T	1.0000E+00	Failure of CIS Given APC or TMLL - Guaranteed Failure
	C21	1.0960E-04	Seismic - All Support Available
	C22	9.8370E-05	Seismic - Loss of All Power
	C23	4.0240E-04	Seismic - All Power Not Lost and One Train of Signals Lost
	C24	1.5120E-04	Seismic - Loss of All Power and One Signal Train
	C25	2.0010E-01	Seismic - All Signals Lost and All Power Not Lost

* Some human recovery actions have been included in the system analysis

Event Tree Split Fraction Equations

* SYSTEMS ANALYSES-TO-EVENT TREE SPLIT FRACTION EQUATIONS

*** OFF-SITE POWER (OSP) SYSTEM

OG1 = OG1
OGF = 1.0

*** EMERGENCY AC POWER (ACP) SYSTEM

GA1 = DG1
GA2 = DG1
GAS = 0.0
GAF = 1.0

GB1 = (DG1-DGAB) / (1 - DG1)
GBA = DGAB / DG1
GB2 = DG1
GBS = 0.0
GBF = 1.0

*** DC POWER (DCP) SYSTEM

DA1 = DC4
DA2 = DC4
DAF = 1.0

DB1 = (DC4-DC2) / (1 - DC4)
DBA = DC2 / DC4
DB2 = DC4
DBF = 1.0

*** SERVICE WATER (SW) SYSTEM

WA1 = WAA = SW9
WA2 = WAB = SWA
WA3 = WAC = SW6
WA4 = WAD = SW4
WA5 = WAE = SW5
WA6 = SW5 / SWA
WAF = 1.0

Event Tree Split Fraction Equations

WB1 = (SW9-SW7) / (1 - SW9)
WBA = SW7 / SW9
WBD = SW9
WB2 = (SWA-SW8) / (1 - SWA)
WBB = SW8 / SWA

WBE = SWA
WB3 = (SW6-SW3) / (1 - SW6)
WBC = SW3 / SW6
WBG = SW6
WB4 = (SW4-SW1) / (1-SW4)
WBH = SW1 / SW4
WBI = SW4
WB5 = (SW5-SW2) / (1-SW5)
WBJ = SW2 / SW5
WBK = SW5
WB6 = SW5 / SWA
WBF = 1.0

*** PRIMARY COMPONENT COOLING (PCC) WATER SYSTEM

PA1 = PA4 = PCC4
PA2 = PA5 = PCC5
PA3 = PA6 = PCC6
PAF = 1.0

PB1 = (PCC4-PCC1) / (1 - PCC4)
PBA = PCC1 / PCC4
PB2 = (PCC5-PCC2) / (1 - PCC5)
PBB = PCC2 / PCC5
PB3 = (PCC6-PCC3) / (1 - PCC6)
PBC = PCC3 / PCC6
PB4 = PCC4
PB5 = PCC5
PB6 = PCC6
PBF = 1.0

*** SOLID STATE PROTECTION SYSTEM (SSPS)

SA1 = SAA = SCA
SA2 = SAB = SCB
SA3 = SAC = SCC
SA4 = SAD = SCC
SA5 = SAE = SCA
SA6 = SAG = SCC
SA8 = 0.0
SAF = 1.0

Event Tree Split Fraction Equations

SB1 = (SCA-SC1) / (1 - SCA)
SBA = SC1 / SCA
SBB = SCA
SB2 = (SCB-SC2) / (1 - SCB)
SBC = SC2 / SCB

SBD = SCB
SB3 = (SCC-SC3) / (1 - SCC)
SBE = SC3 / SCC
SBG = SCC
SB4 = (SCC-SC3) / (1 - SCC)
SBH = SC3 / SCC
SBI = SCC
SB5 = (SCA-SC1) / (1 - SCA)
SBJ = SC1 / SCA
SBK = SCA
SB6 = (SCC-SC3) / (1 - SCC)
SBL = SC3 / SCC
SBM = SCC
SBF = 1.0

*** ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)

EA1 = EAA = ECA
EA2 = EAB = ECC
EA3 = EAC = ECG
EA4 = EAD = ECB
EA5 = EAE = ECH
EAF = 1.0

EB1 = (ECA-EC1) / (1 - ECA)
EBA = EC1 / ECA
EBB = ECA
EB2 = (ECC-EC3) / (1 - ECC)
EBC = EC3 / ECC
EBD = ECC
EB3 = (ECG-EC5) / (1 - ECG)
EBE = EC5 / ECG
EBG = ECG
EB4 = (ECB-EC2) / (1 - ECB)
EBH = EC2 / ECB
EBI = ECB
EB5 = (ECH-EC6) / (1 - ECH)
EBJ = EC6 / ECH
EBK = ECH
EBF = 1.0

Event Tree Split Fraction Equations

*** REACTOR TRIP SYSTEM (RTS)

RT1 = RT1
RT2 = RT2
RT3 = RT3
RTS = 0.0

RTF = 1.0

*** ATWS MITIGATING SYSTEMS ACTUATION CIRCUITRY (AMSAC)

AM1 = 0.01
AMF = 1.0

*** EMERGENCY AIR HANDLING (EAH) SYSTEM

EH1 = EH1
EH2 = EH2
EH3 = EH3
EH4 = EH4
EH5 = EH5
EH6 = EH6
EH7 = EH7
EH8 = EH8
EHF = 1.0
CV1 = 0.0
CVF = 1.0

*** EMERGENCY CORE COOLING SYSTEM (ECCS)

RW1 = RWLL
RW2 = RWML
RW3 = RWSL
RWF = 1.0

RA1 = RALL
RA2 = RAML
RA3 = RASL

RB1 = RBLL
RB2 = RBML
RB3 = RBSL

H11 = H11
H12 = H12
H13 = H13
H1F = 1.0

Event Tree Split Fraction Equations

H21 = H21
H22 = H22
H23 = H23
H2F = 1.0

H31 = H31
H32 = H32
H33 = H33
H3A = H21
H3B = H22
H3C = H23
H3F = 1.0

LA1 = LI2
LA2 = LI2
LAF = 1.0

LB1 = (LI2-LI1) / (1 - LI2)
LBA = LI1 / LI2
LB2 = LI2
LBF = 1.0

L11 = MM2
L12 = MM2
L13 = SM2
L14 = SM2
L1F = 1.0

L21 = (MM2-MM1) / (1 - MM2)
L2A = MM1 / MM2
L22 = MM2
L23 = (SM2-SM1) / (1 - SM2)
L2C = SM1 / SM2
L24 = SM2
L2F = 1.0

LR1 = LR1 + O31
LR2 = LR3 + O31
LRF = 1.0

ZA1 = V12
ZA2 = V12
ZA3 = V22
ZA4 = V22
ZAF = 1.0

ZB1 = (V12-V11) / (1 - V12)
ZBA = V11 / V12
ZB2 = V12
ZB3 = (V22-V21) / (1 - V22)
ZBB = V21 / V22
ZB4 = V22
ZBF = 1.0

Event Tree Split Fraction Equations

LC1 = LP2
LC2 = LP2
LCF = 1.0

LD1 = (LP2-LP1) / (1 - LP2)
LDA = LP1 / LP2
LD2 = LP2
LDF = 1.0

HA1 = HB2
HA2 = HB2
HAF = 1.0

HB1 = (HB2-HB1) / (1 - HB2)
HBA = HB1 / HB2
HB2 = HB2
HBF = 1.0

L51 = SR2
L52 = SR2
L53 = SP2
L54 = SP2
L5F = 1.0

L61 = (SR2-SR1) / (1 - SR2)
L6A = SR1 / SR2
L62 = SR2
L63 = (SP2-SP1) / (1 - SP2)
L6B = SP1 / SP2
L64 = SP2
L6F = 1.0

RC1 = .25*T1A + .25*T1B + .25*T1C
RC2 = .25*TA1 + .25*TA2 + .25*TA3
RC3 = .25*TB1 + .25*TB2 + .25*TB3
RC4 = .5*T2A + .5*T2B
RC5 = .5*T3A + .5*T3B
RC6 = 0.0
RCF = 1.0

RP1 = 0.0
RPF = 1.0

WS1 = 0.0
WSF = 1.0

*** REACTOR VESSEL

RV1 = 0.0
RV2 = 0.01
RV3 = 1.0

Event Tree Split Fraction Equations

*** RCP SEAL LOCA

NL1 1 NL1 = 0.0
NLF 1 NLF = 1.0

*** RCS PRESSURE RELIEF

PSA = 1.48E-03
PSB = 4.45E-02
PS1 = PS1
PS2 = PS2
PS3 = PS3
PS4 = 0.0
PSF = 1.0
P21 = P21
P2F = 1.0
PR1 = PR1
PR2 = PR2
PR3 = PR3
PR4 = PR1 + OR1
PR5 = PR2 + OR1
PR6 = PR3 + OR1
PRF = 1.0

*** REACTOR POWER LEVEL (FRACTION > 40%)

PL1 = 0.67

*** MAIN FEEDWATER (MFW) SYSTEM

MF1 = 0.0
MFF = 1.0

*** EMERGENCY FEEDWATER (EFW) SYSTEM

EFA = EF1 + ARVSDV
EFB = EF1 + ARVV
EFC = EF2 + ARVSDV
EFD = EF2 + ARVV
EFE = EF3 + ARVSDV
EFG = EF3 + ARVV
EFH = EF1*EF4 + ARVSDV
EFI = EF1*EF4 + ARVV
EFJ = EF2*EF4 + ARVSDV
EFK = EF2*EF4 + ARVV
EFL = EF4 + ARVSDV

Event Tree Split Fraction Equations

EFM = EF5 + ARVSDV
EFN = EF5 + ARVV
EFO = EF6 + ARVSDV
EFP = EF6 + ARVV
EFQ = EF7 + ARVSDV
EFR = EF7 + ARVV
EFS = EF5*EF4 + ARVSDV
EFT = EF4 + ARVSDV
EFU = EF1*EFRTDP*EFRSFP + ARVV
EFV = EF1*EFRTDP + ARVV
EFW = EF3*EFRSFP + ARVV
EFX = EF2*EFRTDP*EFRSFP + ARVV
EFY = EF2*EFRTDP + ARVV
EFZ = EF1*EFRTDP*EFRSFP + ARVSDV
EFF = 1.0

*** MAIN STEAM SYSTEM (MSS)

MS1 = MS1
MSF = 1.0
TT1 = TT1
TT2 = 0.0
TT3 = TT1*MS1
TTF = 1.0
SL1 = 0.000108
SL2 = 0.00577
SL3 = 0.00546
SL4 = 0.011
SL5 = 0.00042
SL6 = 0.00939
SL7 = 0.201

*** CONTAINMENT BUILDING SPRAY (CBS) SYSTEM

CA1 = CA2 = CBSCA2
CAF = 1.0

CB1 = (CBSCA2-CBSCA1) / (1 - CBSCA2)
CBA = CBSCA1 / CBSCA2
CB2 = CBSCA2
CBF = 1.0

XA1 = XA2 = CBSXA2
XA3 = CBSXC2
XAF = 1.0

XB1 = (CBSXA2-CBSXA1) / (1 - CBSXA2)
XBA = CBSXA1 / CBSXA2
XB2 = CBSXA2

Event Tree Split Fraction Equations

XB3 = CBSXC2
XBB = CBSXC1 / CBSXC2
XBF = 1.0

VA1 = VA2 = CBSVA2
VA3 = 0.0
VAF = 1.0

VB1 = (CBSVA2-CBSVA1) / (1 - CBSVA2)
VBA = CBSVA1 / CBSVA2
VB2 = CBSVA2
VB3 = 0.0
VBF = 1.0

XC1 = XC2 = CBSXC2
XCF = 1.0

XD1 = (CBSXC2-CBSXC2) / (1 - CBSXC2)
XDA = CBSXC1 / CBSXC2
XD2 = CBSXC2
XDF = 1.0

*** CONTAINMENT ISOLATION SYSTEM (CIS)

CIA = CIA
CIB = CIB
CIC = CIC
CID = CID
CIE = CIE
CIF = CIF
CIG = CIG
CIH = CIH
CII = CII
CIJ = CIJ
CIK = CIK
CIL = CIL
CIM = CIM
CIN = CIN
CIO = CIO
CIP = CIP
CIT = 1.0
CI1 = CI1
CI2 = CI2
CI3 = CI3
CI4 = 1.0
CI5 = CI5
CI6 = CI6
CI7 = CI7
CI8 = CI8

Event Tree Split Fraction Equations

C2A = C2A
C2B = C2B
C2C = C2C
C2D = C2D
C2E = C2E
C2F = C2F
C2G = C2G
C2H = C2H
C2I = C2I
C2J = C2J
C2K = C2K
C2L = C2L
C2M = C2M
C2N = C2N
C2O = C2O
C2P = C2P
C2T = 1.0
C21 = C21
C22 = C22
C23 = C23
C24 = C24
C25 = C25

*** OPERATOR ACTIONS

OD1 = 0.0260
OD2 = 0.0130
O41 = 0.0500
O42 = 0.0700
O4F = 1.0
O51 = 0.0500
O52 = 0.0900
O53 = 0.0130
ODF = 1.00
OM1 = 0.0620
OM2 = 0.00
OMF = 1.00
OP1 = 0.0230
OP2 = 0.00
OPF = 1.00
OR1 = 0.0170
OR2 = 0.00
OR4 = OR1 + PR1
ORF = 1.00
ON1 = 0.0
ON2 = 0.0130
ONF = 1.00
O31 = 0.0008
O32 = 0.00
O3F = 1.00
HE1 = O31 + R51

Event Tree Split Fraction Equations

HE2 = O31 + R52
HEF = 1.00
HS1 = O31 + R51
HS2 = O31 + R52
HSF = 1.00
OT1 = 0.00
OTF = 1.00
OH1 = 0.00531
OH3 = 0.00
OHF = 1.00
MT1 = 1.0E-02
MTS = 0.0
MTF = 1.0
RMU = 0.1
RMS = 0.0
RMF = 1.0
MR1 = 0.02
MR2 = 0.03
MRF = 1.0
OS1 = 0.01
OS2 = 0.1
OSS = 0.0
OSF = 1.0

*** INTERFACING SYSTEM LOCA (V-SEQUENCE)

CSA = 1.100E-01
CSB = 1.000E-01
CSC = 4.400E-01
CSD = 7.500E-01
CSF = 1.000E+00
LE1 = 9.000E-02
LX1 = 9.190E-01
LY1 = 5.692E-01
LZ1 = 2.000E-02
O11 = 6.500E-03
O21 = 1.000E+00
O22 = 9.100E-03
O3C = 4.900E-03
RSA = 5.600E-01
RSB = 5.500E-01
RSC = 8.500E-01
RSF = 1.000E+00
SI1 = 9.900E-01
SSA = 1.100E-01
SSB = 1.000E-01
SSC = 3.300E-01
SSF = 1.000E+00
VC1 = 1.000E-01
VO1 = 4.800E-05
PI1 = 6.000E-03

Event Tree Split Fraction Equations

*** MISC. SPLIT FRACTIONS

LTF = 1.0
LTS = 0.0
SUF = 1.0
SUS = 0.0

*** SEISMIC HAZARD

QSF = 1.000
QSS = 0.000
QY1 = 0.007
QY2 = 0.233
QY3 = 0.500
QY4 = 0.697
QY5 = 0.825
QY7 = 0.957
QYA = 0.998
QYB = 1.000
QYC = 1.000
QK1 = 0.000
QK2 = 0.000
QK3 = 0.0007
QK4 = 0.0055
QK5 = 0.0171
QK7 = 0.0500
QKA = 0.843
QKB = 0.980
QKC = 0.999
QD1 = 0.000
QD2 = 0.000
QD3 = 0.000
QD4 = 0.001
QD5 = 0.008
QD7 = 0.053
QDA = 0.197
QDB = 0.458
QDC = 0.720
QR1 = 0.000
QR2 = 0.000
QR3 = 0.000
QR4 = 0.007
QR5 = 0.024
QR7 = 0.102
QRA = 0.294
QRB = 0.572
QRC = 0.804

Event Tree Split Fraction Equations

*** ELECTRIC POWER RECOVERY

ERS = 0.0
ER1 = 1.091E-2
ER2 = 6.401E-2
ER3 = 1.223E-2
ER4 = 6.659E-2
ER5 = 7.137E-1
ER6 = 8.878E-1
ER7 = 5.556E-1
ER8 = 6.932E-1
ER9 = 4.830E-2
ERA = 1.930E-1
ERF = 1.0

*** EMERGENCY FEEDWATER (EFW) RECOVERY

FRS = 0.0
FR0 = 1.0
FR1 = 2.824E-1
FR2 = 6.775E-1
FR3 = 2.790E-2
FR4 = 6.995E-3
FR5 = 5.538E-1
FRA = 4.908E-3
FRF = 1.0

E.2 AC Power System

E.2.1 System Description

Function

The function of the AC Power System is to provide ac motive power and control power necessary for normal operation, as well as mitigating any abnormal events that could affect the reactor core, the reactor heat removal systems, or systems that affect the release of radioactivity to the environment. The AC Power System also provides the normal power supply to the 120V ac instrumentation needed for monitoring key plant parameters and for input to safeguards actuation logic and reactor trip logic.

Configuration

The Vital AC Power System is a Class 1E system which is divided into two load groups (Train A and Train B). Each load group consists of a Class 1E 4.16 kV bus, an emergency diesel generator, 480 V load centers, 120 V ac preferred power supplies, and 120 V ac instrument power supplies (see Figures E.2-1 and E.2-2). The system also includes the Switchgear Ventilation System and Diesel Generator Building Ventilation System as well as other diesel generator support systems.

Dependencies

The 345 kV transmission network provides off-site power to the station during startup and following a unit trip. In the event of a loss of off-site power, emergency ac power is supplied by the Class 1E Distribution System from two diesel generators (one per train). The diesel generators are supported by the Service Water System, the vital 125 V DC Power System (for operation and control), and the diesel generator support systems included with the system boundary (Fuel Oil Transfer System, dc bus supply breakers, and the Ventilation System).

E.2 AC Power System (Continued)

Operation

During normal plant power operation, the plant loads are powered off the main turbine generator through the Unit Auxiliary Transformers (UATs). If the unit trips and off-site power is available, power is routed through the Generator Step-Up Transformer (GSU) to the UATs to supply the 13.8 kV and 4.16 kV buses. If the GSUs or UATs are unavailable, the Reserve Auxiliary Transformers (RATs) are energized directly from the 345 kV switchyard. If off-site power is unavailable or if the UATs and RATs fail, the Class 1E System is powered by the emergency diesel generator.

E.2.2 System Model

The system model includes normal ac power from the switchyard to the essential 4.16 kV buses and emergency power from the diesel generators (including support systems).

System unavailability is quantified using the fault tree method (IRRAS) and RISKMAN software to generate an equation file.

Top Events

The results of the AC Power System quantification are used as input to the support systems event tree (SUPPORT). Top Events GA and GB model the AC Power System (Trains A and B, respectively) in its functional relationship to the other support systems (e.g., Service Water System, PCC System, etc.). Recovery of the AC Power System is included in Top Event ER in the recovery event tree (RECOVERY). Top Event ER models restoration of off-site power and/or recovery of the diesel generator(s) and is described in Section 2.3.9 of the SSPSS-1990. Top Events GA, GB, and ER are only asked when the normal power supply through the UATs/RATs is unavailable (i.e., loss of off-site power). Thus, Top Events GA and GB are primarily diesel generator unavailabilities.

E.2 AC Power System (Continued)

Success Criteria

The success criteria for the AC Power System is that at least one of two ac power buses (E5 or E6) remains operable. For the "Off-Site Power Available" case, the mission time is 24 hours, at which time the plant has been cooled down, the decay heat has greatly decreased, and the time for repair if failures do occur is very long. For the "Loss of Off-Site Power" case, the mission time for the diesel generator and its support systems is six hours. The electric power recovery model (Section 2.3.10 of the SSPSS-1990) evaluates diesel generator failure as well as diesel and off-site power recovery over a mission time of 24 hours.

Analysis Conditions

The AC Power System is in service during all modes of unit operation. The following assumptions have been made about system operation:

1. The AC Power System is assumed to be energized as specified for power operation in the Technical Specifications.

2. All 480 V and 120 V ac buses are assumed to be energized by their normal power source. Crossties between buses of different load groups are assumed to be de-energized.

3. Since failure of the 480 V buses is dominated by failure of the associated 4.16 kV or 13.8 kV bus, the 480 V buses are combined at the higher voltage level in a bounding model.

4. Failure of relays that actuate breakers are included in the breaker failure data.

5. Unavailability of switchgear ventilation is assumed to cause failure of the associated train of 4.160 kV Class 1E switchgear after approximately two hours of operation with no ventilation.

6. Failure of service water to the diesel generator water jacket heat exchangers or failure of D/G building ventilation is assumed to cause failure of the associated diesel generator due to overheating.

E.2 AC Power System (Continued)

7. Failure of the fuel oil transfer pump causes failure of the diesel engine due to lack of fuel.

8. Unavailability of the emergency power sequencer is equivalent to power failure at the 4.16 kV Class 1E level since it prevents automatic starting of the equipment supplied from the bus.

E.2.3 Results

The dominant contributors to system unavailability for ac power given LOSP (two train and single train) are provided below (the definitions of the cut set basic events are given at the end of this section):

DGAB	7.465E-03	TOTAL - LOSS OF TWO DIESEL GENERATORS		
DGAB	8.361E-04	1	*	DGABR5
DGAB	6.664E-04	2	*	DGA_S*DGA_R1
DGAB	4.906E-04	2	*	DGA_S*DGA_R5
DGAB	7.969E-04	1	*	DGA_S*DGA_S
DGAB	4.045E-04	2	*	DGA_R1*DGA_R5
DGAB	6.488E-04	1	*	DGA_R1*DGA_R1
DGAB	2.265E-04	1	*	DGSABS
DGAB	1.672E-04	1	*	DGABR1
DGAB	3.182E-04	1	*	DGA_R5*DGA_R5
DGAB	1.265E-04	2	*	DGA_S*DGFOAS
DGAB	1.125E-04	1	*	BRKR2C
DGAB	1.759E-03	Maintenance Unavailability		
DG1	7.343E-02	TOTAL - LOSS OF ONE DIESEL GENERATOR		
DG1	2.057E-02	1	*	DGA_S
DG1	1.650E-02	1	*	DGA_R1
DG1	1.203E-02	1	*	DGA_R5
DG1	3.015E-03	1	*	DGFOAS
DG1	1.461E-03	1	*	DGBKAC
DG1	9.128E-04	2	*	DG1VFS
DG1	9.128E-04	2	*	DG1SFS
DG1	7.555E-04	1	*	DGFIRA
DG1	1.515E-02	Maintenance Unavailability		

Cut Set Basic Events	Definition
DGABR5	Common cause failure of both D/Gs to run after first hour.
DGA_S	Single D/G fails to start on demand.
DGA_R1	Single D/G fails to run for the first hour.
DGA_R5	Single D/G fails to run after first hour.
DGSABS	Common cause failure of both D/G fuel oil pumps to start on demand.

E.2 AC Power System (Continued)

Cut Set Basic Events	Definition
DGABR1	Common cause failure of both D/Gs to run.
DGFOAS	Single D/G fuel oil pump fails to start.
BRKR2C	Both D/G feeder breakers fail to close (common cause)
DGBKAC	Single D/G feeder breaker fails to close.
DG1VFS	Single D/G ventilation fan fails to start on demand.
DG1SFS	Single D/G switchgear ventilation fan fails to start on demand.
DGFIRA	D/G or switchgear fire.

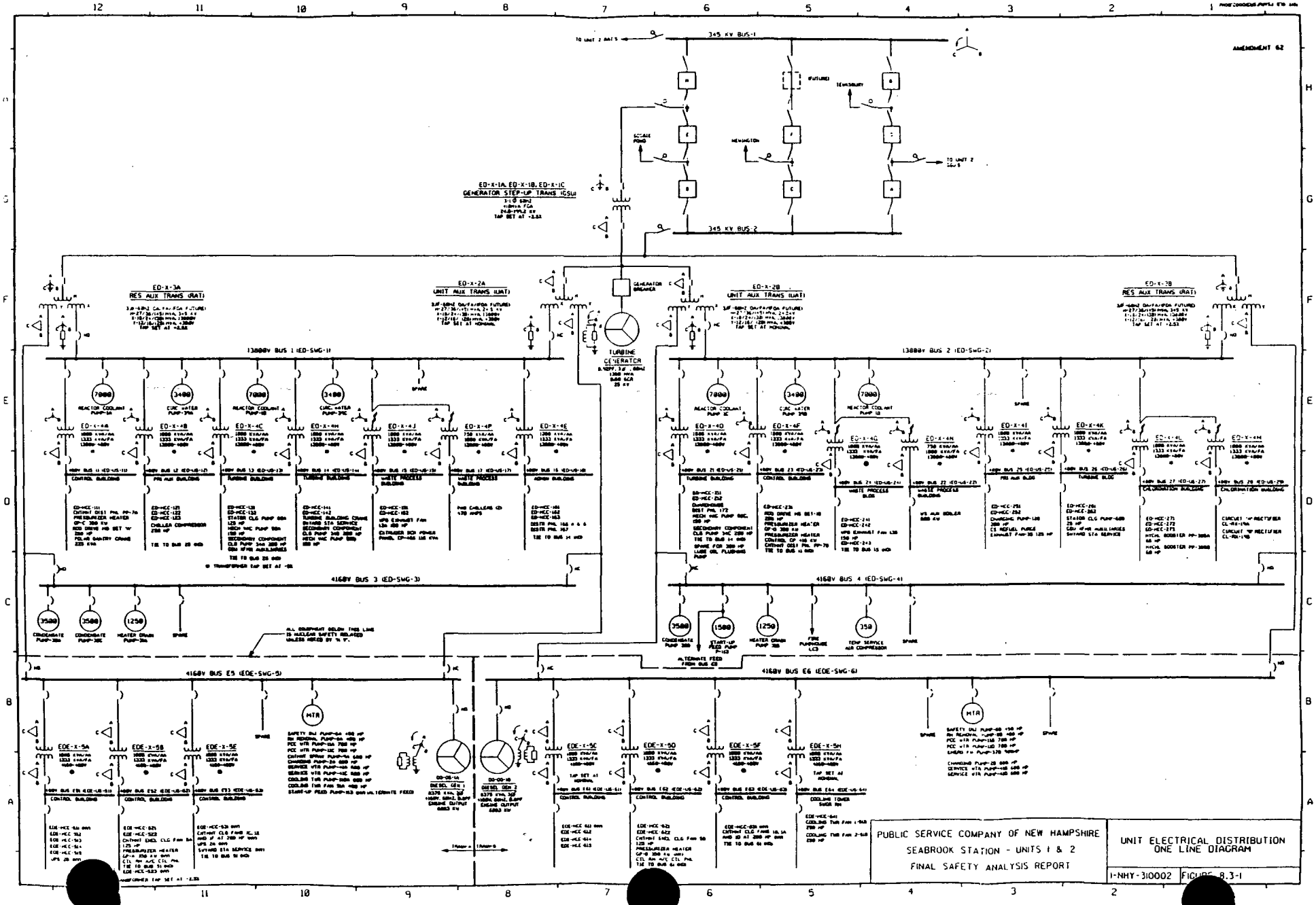

Figure E.2-1 Unit Electrical Di[redacted]on One Line Diagra

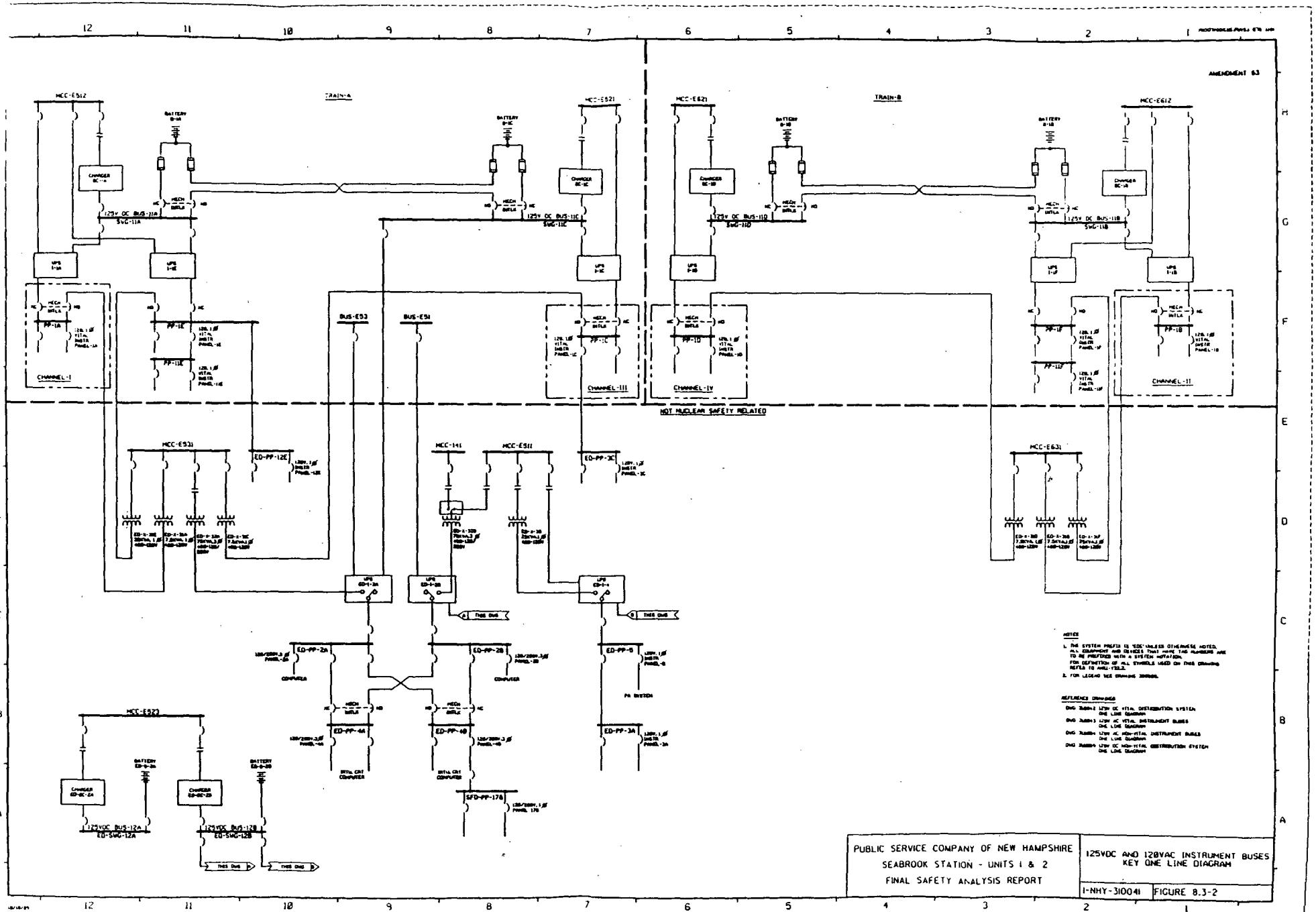

Figure E.2-2 125V DC and 120V AC Instrument Buses Key One Line Diagram

E.3 DC Power System

E.3.1 System Description

Function

The primary function of the vital DC Power System is to supply and distribute 125 V dc control power for switchgear and controller operation. This system also provides backup power to the uninterruptible power supplies (UPS) for 120 V ac instrument buses and provides operation and control power for the emergency diesel generators.

The nonessential dc power systems provide power for turbine and generator auxiliaries for emergency lighting, power to the station computers, and power to the 345 kV substation.

Configuration

The DC Power System consists of a Class 1E (essential) DC Distribution System, a non-Class 1E DC Distribution System, and switchgear/battery room HVAC.

The essential DC Distribution System (see Figure E.3-1 for Train A and Figure E.3-2 for Train B) is divided into four buses (two per train) and is composed of 59-cell lead-calcium batteries, battery racks, battery chargers, dc distribution panels, dc power cables, and ground detection equipment.

The non-Class 1E dc distribution consists of battery chargers, batteries, switchboards, distribution panels, and cables. A second nonvital DC Power System supplies power for the 345 kV switchyard Protective Relay and Control Systems and consists of two independent 12V lead-calcium batteries and associated battery chargers.

The Switchgear and Battery Room HVAC consists of supply fans, exhaust fans, battery room exhaust fans, a hot water heater unit, filters, and various duct work, dampers, controls, and accessories.

E.3 DC Power System (Continued)

Dependencies

The essential AC and DC Systems depend on each other through the battery chargers, inverters, and dc control power for the diesel generators and switchgear. Loss of essential 4.16 kV ac power will cause loss of power to the associated battery charger. The switchgear and battery room HVAC provides heating, ventilation, and air conditioning to the components in the DC System.

Operation

The DC Power System operates as an essentially passive system. With loss of power to the battery chargers, the batteries automatically supply power to the associated dc bus. Manual crossties are provided between a battery and the opposite bus within each train (e.g., between Batteries B-1A and Bus 11C).

Potential for Event Initiation

The loss of one dc power train (i.e., loss of Bus 11A for Train A or loss of Bus 11B for Train B) results in reactor trip due to loss of main feedwater.

E.3.2 System Model

The system model includes the vital DC Power System only. A related system model is used to quantify the loss of one dc bus initiating event.

System unavailability is quantified using the fault tree method (IRRAS) and RISKMAN software to generate an equation file.

Top Events

The unavailabilities calculated in the DC Power System analysis are used as top events DA and DB in the support systems event tree for Train A dc power and Train B dc power, respectively. DC power unavailability is questioned in the support tree only when the normal power through the UATs/RATs is unavailable (i.e., loss of off-site power). Battery lifetime during a station blackout is modeled in the electric power recovery model (see Section 2.3.10 of SSPS-1990).

E.3 DC Power System (Continued)

Success Criteria

The success criteria for the DC Power System is that at least one of the two essential dc buses (either Bus 11A or 11B) remain operable for the mission duration. The dc Buses 11A and 11B are considered essential buses since they alone provide dc control power. The other buses (11C and 11D) provide power to instrument channels (whose signals fail safe on loss of dc). For the "ac power available case", a standard mission time of 24 hours is used. For the "ac power unavailable case", a 24-hour mission time is also used. However, for this case, the battery chargers are assumed to be unavailable during the first six hours (due to loss of ac power) and the batteries unavailable after six hours (due to battery discharge depletion).

Analysis Conditions

The DC Power System is assumed to be energized as specified for power operation in the Technical Specifications. Failures that cause a loss of power not resulting in an initiating event will be quickly detected by plant personnel due to failure of auxiliary equipment. These failures will be repaired or the unit shut down in accordance with the operability criteria in the Technical Specifications.

All dc buses are assumed to be energized by their primary power source. The crossties between the buses are assumed to be de-energized (open).

Loss of ventilation has a negligible effect on operability of the DC Power System. The DC Power System breakers are all manual; thus, no spurious operation of bus protection or fault lockout relays due to overheating is feasible. Also, ventilation in the battery room is to prevent hydrogen gas buildup, which is only a long-term concern.

No operator actions are included in the system model. No credit is taken for manual cross-tying the other dc bus/battery in each train.

The nonessential dc power system is not modeled because it has no direct safety function.

E.3 DC Power System (Continued)

E.3.3 Results

Quantitative results for the DC Power System are shown below (the definitions of cut set basic events are given at the end of this section):

```
DC1     2.731E-10 TOTAL Both DC Trains - Off-Site Power Available
DC1     2.666E-10  1  *  BUS11A*BUS11B
DC1     2.069E-12  1  *  B1A_D*BCA_18*BUS11B
DC1     2.069E-12  1  *  B1B_D*BCB_18*BUS11A
DC1     6.896E-13  1  *  B1B_D*BCB_6*BUS11A
DC1     6.896E-13  1  *  B1A_D*BCA_6*BUS11B

DC2     1.369E-06 TOTAL Both DC Trains - LOSP (SBO for 1st 6 hours)
DC2     4.729E-07  1  *  BCA_18*BCB_18
DC2     4.567E-07  1  *  B1A_D*B1B_D
DC2     1.641E-07  1  *  B1B_D*BCA_18
DC2     1.641E-07  1  *  B1A_D*BCB_18
DC2     2.426E-08  1  *  B1B_D*BAMNT
DC2     2.426E-08  1  *  B1A_D*BBMNT

DC3     1.213E-05 TOTAL One DC Train - Off-Site Power Available
DC3     1.188E-05  1  *  BUS11A
DC3     1.641E-07  1  *  B1A_D*BCA_18
DC3     5.471E-08  1  *  B1A_D*BCA_6
DC3     2.426E-08  1  *  B1A_D*BCAMNT

DC4     8.922E-04 TOTAL One DC Train - LOSP (SBO for 1st 6 hours)
DC4     4.875E-04  1  *  B1A_D
DC4     3.387E-04  1  *  BCA_18
DC4     4.959E-05  1  *  BAMNT
DC4     1.188E-05  1  *  BUS11A
DC4     4.483E-06  1  *  B1A_6
```

Quantitative results for the Loss of One DC Bus Initiating Event are shown below:

```
LDCA    3.205E-03 TOTAL LOSS OF ONE BUS INITIATING EVENT
LDCA    3.205E-03 NORMAL (System in the Normal Alignment)
LDCA    1.000E+00 Fraction of time in: NORMAL
LDCA    3.205E-03 Conditional system failure frequency given: NORMAL
LDCA    3.071E-03  1  *  BUS1YR
LDCA    7.366E-05  1  *  BCMNT*MBATD
LDCA    5.470E-05  1  *  BATD*BC1YR
LDCA    4.558E-06  1  *  BATMNT*BC30D*OP_ER
LDCA    8.196E-07  1  *  BATD*BCK1YR
LDCA    3.889E-07  1  *  BATMNT*BC30D*CCBKR
```

E.3 DC Power System (Continued)

Cut Set Basic Events	Definitions
System:	
B1A_D, B1B_D	Battery fail to discharge on demand.
B1A_6, B1B_6	Battery fails to operate for first six hours.
BUS11A, BUS11B	125 V dc bus fails during operation.
BCA_6, BCB_6	Battery charger fails to operate for first six hours.
BCA_18, BCB_18	Battery charger fails to operate after first six hours.
BCAMNT, BCBMNT Maintenance on battery charger.	
BAMNT, BBMNT	Maintenance on battery.
Initiating Event:	
BUS1YR	Bus failure over one year.
BCMNT	Battery charger maintenance during one year.
MBATD	Battery fails on demand during battery charger maintenance.
BATD	Battery fails on demand.
BC1YR	Battery charger fails.
BATMNT	Battery in maintenance during one year.
BC30D	Battery charger fails during battery maintenance.
OP_ER	Operator error to cross-connect buses during battery maintenance.
BCK1YR	Battery charger breaker fails - one year.
CCBKR	Failure of cross-connect breaker during battery maintenance.

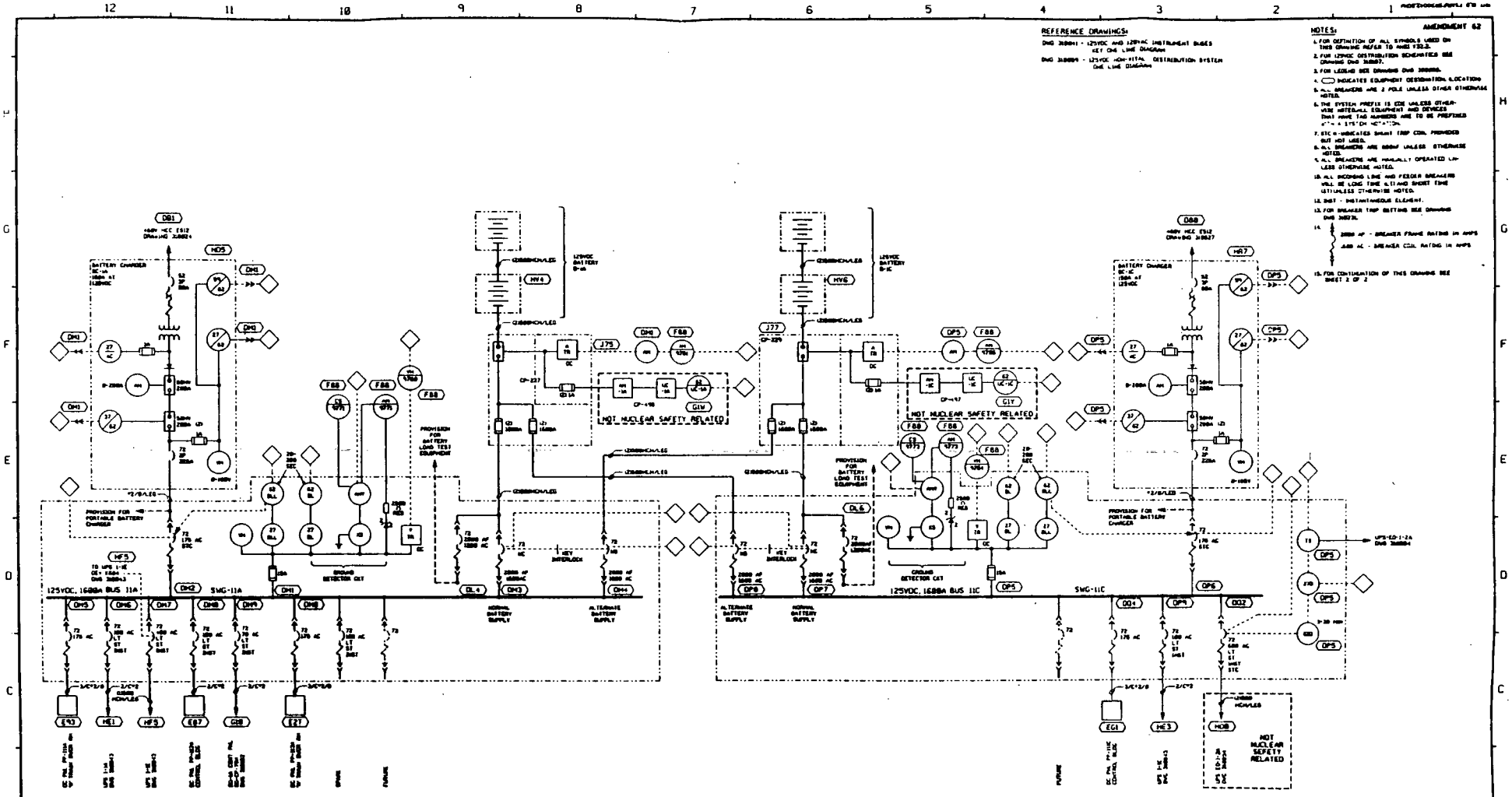

Figure E.3-1 125V DC Vital Distribution System (Train A)

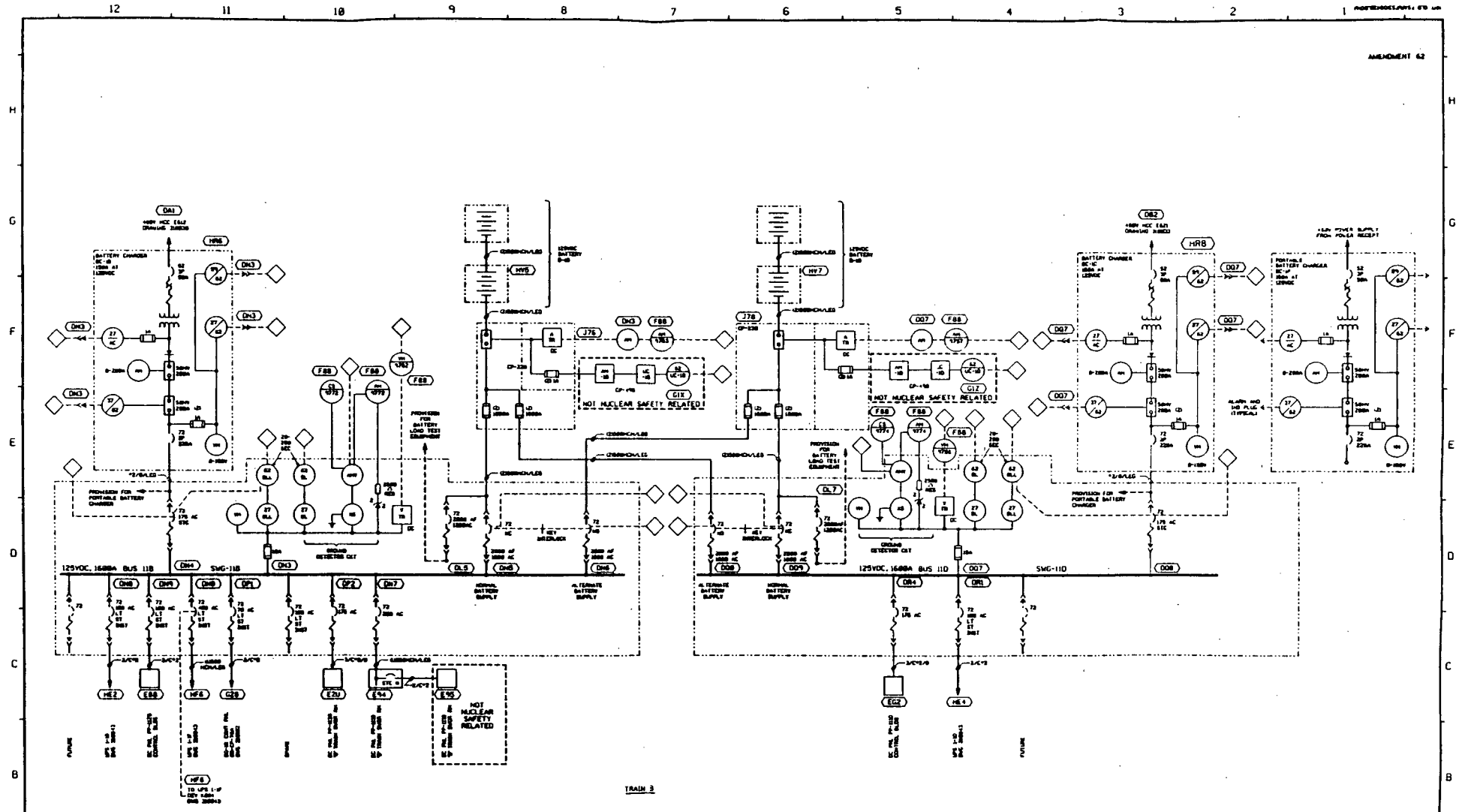

Figure E.3-2 125V DC Vital Distribution System (Train B)

E.4 Off-Site Power System

E.4.1 System Description

Function

The Off-Site Power System provides the normal source of power to essential and nonessential electric power plant loads when the main generator breaker is open. The Nonessential AC Power System provides power at 13.8 kV, 4.16 kV, and 480 V to plant loads that are necessary for power operation but not for safe shutdown.

Configuration

The Class 1E System is divided into two load group trains, designated as Train A and Train B. Each load group supplies all equipment necessary to safely shut down the reactor under any of the design basis accident conditions analyzed in the Seabrook Final Safety Analysis Report (FSAR). Each load group train consists of a Class 1E 4.16 kV bus, an emergency diesel generator, 480 V load centers, 120 V ac preferred power supplies, 120 V ac instrument and control power supplies, and two 125 V dc batteries.

The unit has two UATs and two RATs. Both UATs are supplied directly from the unit generator when the unit is up and feeding the grid. The UATs are supplied from the grid through the switchyard and backfed through the GSU when the unit is not up.

The 345 kV switchyard configuration is breaker-and-a-half, with two primary 345 kV buses and three cross-connect buses (see Figure E.4-1). The switchyard consists of metal enclosed SF_6 gas-insulated components connected by an integral bus system.

E.4 Off-Site Power System (Continued)

Dependencies

The Off-Site Power System has a number of support systems, including nonessential dc power, the SF_6 gas system, protective relaying, transformer cooling, etc. These support systems are included within the system due to the method of modeling. Because data is collected for total loss of off-site power, the Off-Site Power System is modeled as one supercomponent, including support systems.

If off-site power is lost, essential ac power is provided by the diesel generators. Also, without off-site power, the RCPs are inoperable, the start-up feed pump cannot automatically start, and the main condenser is not available for steam dump operation.

Operation

During normal plant power operation, the plant loads are powered off the main turbine generator through the UATs. If the plant's turbine generator trips and power is available from the grid, power is backfed through the GSU to the UATs to supply the 13.8 kV and 4.16 kV buses. If power becomes unavailable via the GSU and UATs, the buses are automatically transferred to the RATs, which are normally energized via the 345 kV switchyard. If off-site power is unavailable, or if the UATs and RATs are failed, the Class 1E System is powered by the emergency diesel generator.

The operators and plant personnel may interface with the system directly for remote and local circuit breaker operations. Manual operations can provide additional flexibility in recovery operations following severe system transients.

E.4.2 System Model

The system model includes the off-site grid, transmission lines, and switchyard.

E.4 Off-Site Power System (Continued)

Top Event

Loss of off-site power is included in the plant model as a top event in the support systems event tree (top event OG) and as two separate initiating events: loss of off-site power (LOSP) and loss of off-site power due to failures in the SF_6 switchyard and transmission lines (LSF6). LSF6 differs from LOSP in that the failure is essentially nonrecoverable in 24 hours. Top event OG quantifies the conditional frequency of loss of off-site power given a unit trip.

The Top Event OG and initiating event LOSP are each quantified directly from applicable data from generic experience. The initiating event LSF6 is quantified using a logic model of the switchyard.

Recovery of off-site power is modeled as top event ER in the recovery event tree (see Section 2.3.10 of the SSPSS-1990)

E.4.3 Results

Event	Description	Mean Value	Units
LOSP	Loss of Off-Site Power Initiating Event	6.91E-2	Per site year.
		4.84E-2	Per site year while operating.*
OG1	Conditional Loss of Off-Site Power Due to Unit Trip	5.72E-4	-----
LSF6	Loss of Off-Site Power Initiating Event Due to SF_6 System Failure (Unrecoverable)	1.28E-3	Per site year.
		8.96E-4	Per site year while operating.*

E.4 Off-Site Power System (Continued)

---

* The use of units of "per site year while operating" assumes a plant availability factor of 0.70. The units of "per site year" account for initiating events in a calendar year while the plant is in any mode; units of "per site year while operating" include the conditional likelihood that the plant is at power (Modes 1 and 2).

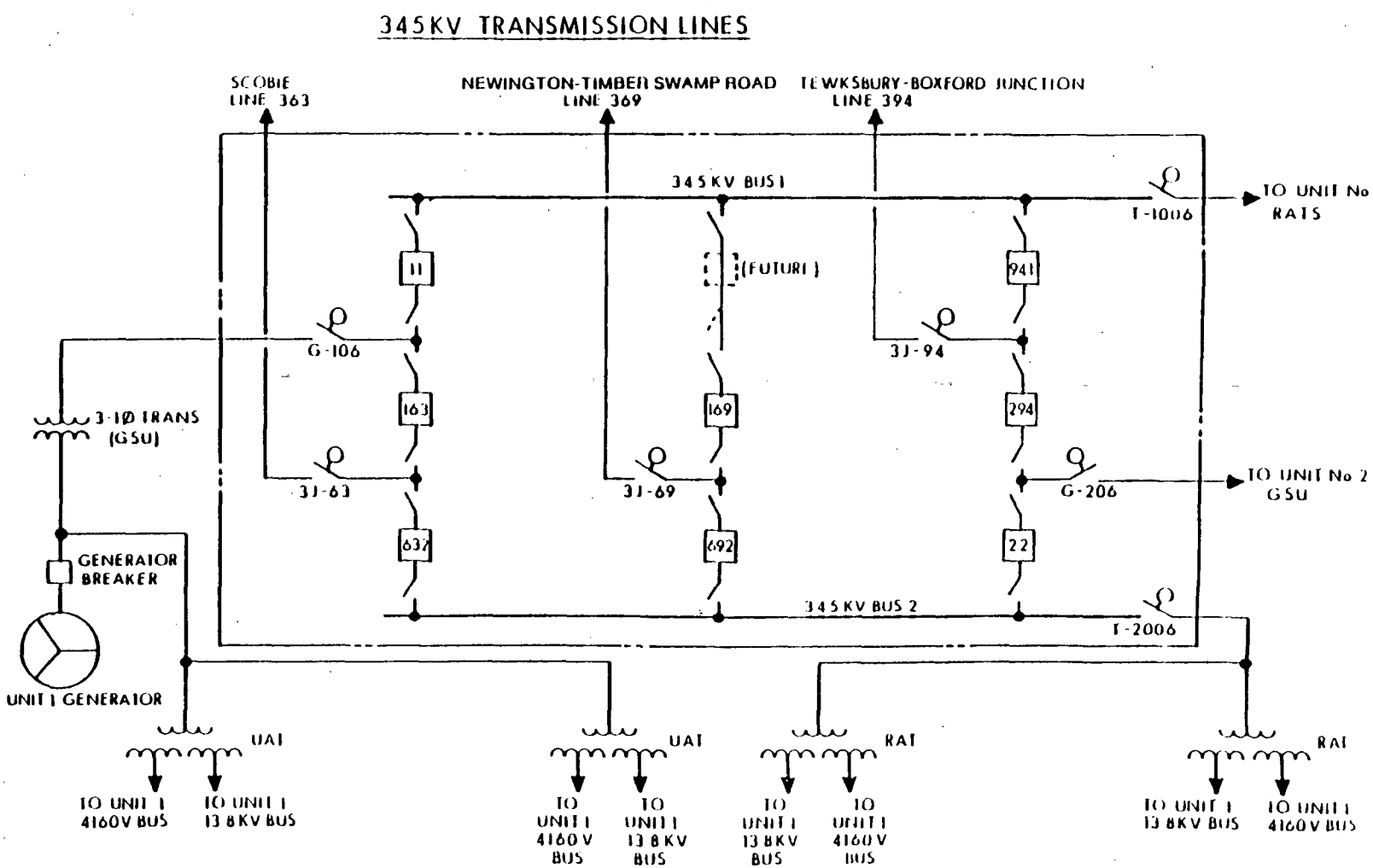

FIGURE E.4-1 SWITCHING DIAGRAM FOR SEABROOK 1 AND 2

E.5 Service Water System

E.5.1 System Description

Function

The Service Water System (SWS) provides cooling water to transfer the heat from primary (safety-related) and secondary (nonsafety-related) loads to the ultimate heat sink, either the Atlantic Ocean or the atmosphere. During a loss of off-site power, the SWS also provides cooling to the diesel generator jacket water coolers.

Configuration

The SWS consists of a seawater service water system (normal), a cooling tower system (alternate), and their associated ventilation systems. The seawater service water system includes two independent and redundant trains which take suction from a common bay in the service water pumphouse. Each train contains two parallel service water pumps, one normally operating and the other in standby. The Cooling Tower System also includes two independent trains, with one cooling tower pump per train. Fans are provided to remove heat from the cooling tower (see Figure E.5-1).

Dependencies

Support for the normal SWS is provided by the Service Water Pumphouse Heating and Ventilation System and by the Electric Power System. Support for the Cooling Tower System is provided by its associated Heating and Ventilation System and by the Electric Power System.

E.5 Service Water System (Continued)

Operation

The SWS is operable during all modes of operation with one pump per train in standby mode. If the operating service water pump trips, the standby pump automatically starts. If the discharge pressure in a service water train falls below its low-low pressure setpoint, a train-associated tower actuation (TA) signal is generated which starts the associated cooling tower pump and stops that train's service water pumps. Given a TA signal, an S signal, or a loss of off-site power, the secondary heat loads are isolated to conserve cooling water to safeguards equipment.

Potential for Event Initiation

Loss of service water is a potential initiating event because the system is required to supply cooling water to the plant PCC system and SCC system heat exchangers at all times during operation. Loss of either train of the SWS would affect the plant power generation through PCC cooling to the RCPs.

E.5.2 System Model

The SWS analysis includes several system models:

1. Availability of "normal" service water, i.e., using the service water pumphouse,
2. Availability of cooling towers, assumed to start only on manual actuation, and
3. Initiating event - loss of one train of service water

System unavailability is quantified using the fault tree method (IRRAS) and RISKMAN software.

Top Event Definition

The SWS System is analyzed for Top Event WA (loss of SWS Train A) and Top Event WB (loss of SWS Train B) in the support systems event tree under three boundary conditions:

Case 1 - SI signal with off-site power available

E.5 Service Water System (Continued)

Case 2 - No SI signal and off-site power available (i.e., general transient)
Case 3 - Loss of off-site power

For all three cases, the SWS must continue to supply service water to the PCC heat loads after an initiating event occurs. Case 2 is applied to initiating events which require isolation of the nonsafety-related heat loads (i.e., secondary component cooling). Case 3 is applied to initiating events which also require isolation of the secondary heat loads. In addition, for Case 3, the SWS pumps must restart and operate throughout the mission time. The mission time for all three cases is 24 hours.

Success Criteria

System success criteria is one of two trains continuing to operate for 24 hours after event initiation.

The model also assumes loss of pumphouse switchgear ventilation and cooling tower ventilation systems result in failure of SW and CT pumps, respectively. Loss of pumphouse ventilation is assumed to have no effect for the 24-hour mission time.

The model assumes that isolation of the secondary heat loads is required for a loss of off-site power concurrent with an S signal or for a TA signal. For small LOCA, steam generator tube rupture, and steam line break outside containment initiating events with off-site power available, it is assumed that isolation of secondary heat loads is not required. Thus, for these three initiators, Service Water is quantified for Case 2 (no SI signal with off-site power available).

Analysis Conditions

- Operator actions to initiate cooling tower operation are modeled. No credit has been taken for the automatic generation of a TA signal.

- Failure of the operators to close the spray bypass MOVs SW-V139 and SW-V140 is assumed to have no effect on system performance for the mission time. Closure of these valves controls cooling tower water temperature by redirecting all cooling tower return flow to the spray headers (instead of the tower basin).

E.5 Service Water System (Continued)

- The SWS is analyzed for various combinations of support states, including loss of off-site power, S signal, TA signal, and single AC power train availability.

- No credit is given for manually initiating the cooling tower for LOSP-initiated sequences because of the time dependence between diesel cooling and recovery from SW failure.

E.5.3 Results

The SWS System quantification results are shown below:

```
SW1 = SWN1*SWCT1+SWSCC1+E1_A*E1_B
SW1      3.315E-04 TOTAL SI Signal, No LOSP (Both Trains Avail.)
SW1      2.987E-04  MOVSC2
SW1      3.371E-05  MOVSC1 * MOVSC1
SW1      2.491E-07  SGVFS2 * OP_TA
SW1      1.619E-08  OP_TA * Z
SW1      1.386E-08  OP_TA * SGVFR2
SW1      1.153E-08  SGVFS2 * (4 * MOVC2)
SW1      1.153E-08  SGVFS2 * (4 * TMOVO2)
SW1      1.350E-05 Maintenance Unavailability

SW2 = SWN1*(SWCT1+SWSCC1)+E1_A*E1_B
SW2      3.954E-07 TOTAL No SI Signal, No. LOSP (Both Trains Avail.)
SW2      2.491E-07  SGVFS2 * OP_TA
SW2      1.619E-08  OP_TA * Z
SW2      1.386E-08  OP_TA * SGVFR2
SW2      1.153E-08  SGVFS2 * (4 * MOVC2)
SW2      1.153E-08  SGVFS2 * (4 * TMOVO2)
SW2      1.005E-08  SGVFS2 * MOVSC2
SW2      7.543E-09  SGVFS2 * CTPPS2
SW2      7.481E-09  OP_TA * SWPPR4
SW2      5.847E-09  SGVFS2 * (2 * CTFNS2)
SW2      1.360E-08 Maintenance Unavailability

SW3 = SWN3 + SWSCC1 + E1_A*E1_B
SW3      7.900E-04 TOTAL Loss of Off-Site Power (Both Trains Avail.)
SW3      2.987E-04  1  *  MOVOG2
SW3      2.987E-04  1  *  MOVSC2
SW3      3.366E-05  1  *  SGVFS2
SW3      1.925E-05  2  *  PRSOG1*MOVOG1
SW3      1.691E-05  1  *  PRSOG2
SW3      3.371E-05  1  *  MOVOG1*MOVOG1
SW3      3.371E-05  1  *  MOVSC1*MOVSC1
SW3      1.106E-05  1  *  SWPPR2
SW3      1.076E-05  1  *  PRSOG1*PRSOG1
SW3      9.798E-06  2  *  SWPPR3
SW3      6.031E-06  2  *  SWPPR1*MOVOG1
```

E.5 Service Water System (Continued)

SW3 3.515E-06 2 * PRSOG1*SWPPR1
SW3 3.060E-05 MAINTENANCE Unavailability

SW4 = SWN4 * SWCT4 + SWSCC4 + E1_A
SW4 4.344E-03 TOTAL - SI Signal, No LOSP, One Train - Op Actions
SW4 4.080E-03 MOVSC1
SW4 2.987E-04 MOVSC2
SW4 4.137E-06 E1_A
SW4 3.720E-06 SGVFS1 * (2 * MOVC1)
SW4 3.720E-06 SGVFS1 * (2 * TMOVO1)
SW4 3.374E-06 SGVFS1 * OP_TA
SW4 2.370E-06 S1 * (2 * MOVC1)
SW4 2.370E-06 S1 * (2 * TMOVO1)
SW4 2.178E-06 Y1 * (2 * MOVC1)
SW4 2.178E-06 Y1 * (2 * TMOVO1)
SW4 2.149E-06 S1 * OP_TA
SW4 9.400E-06 MAINTENANCE Contribution

SW5 = SWN4 * (SWCT4 + SWSCC4) + E1_A
SW5 6.013E-05 TOTAL - No SI Signal, No LOSP, One Train - Op Actions
SW5 4.137E-06 E1_A
SW5 3.720E-06 SGVFS1* (2 * MOVC1)
SW5 3.720E-06 SGVFS1 * (2 * TMOVO1)
SW5 3.374E-06 SGVFS1 * OP_TA
SW5 2.370E-06 S1 * (2 * MOVC1)
SW5 2.370E-06 S1 * (2 * TMOVO1)
SW5 2.178E-06 Y1 * (2 * MOVC1)
SW5 2.178E-06 Y1 * (2 * TMOVO1)
SW5 2.149E-06 S1 * OP_TA
SW5 1.975E-06 Y1 * OP_TA
SW5 1.860E-06 SGVFS1 * MOVSC1
SW5 1.379E-06 SGVFS1 * CTPPS1
SW5 1.250E-06 SGVFS1 * CTFNS1
SW5 1.185E-06 S1 * MOVSC1
SW5 1.089E-06 Y1 * MOVSC1
SW5 9.400E-06 MAINTENANCE Contribution

SW6 = SWN6 + SWSCC4 + E1_A
SW6 1.337E-02 TOTAL - Loss of Off-Site Power, One Train - No Op
SW6 4.080E-03 1 * MOVOG1
SW6 4.080E-03 1 * MOVSC1
SW6 2.358E-03 1 * PRSOG1
SW6 7.439E-04 1 * SWPPR1
SW6 4.559E-04 1 * SGVFS1
SW6 2.987E-04 1 * MOVOG2
SW6 2.987E-04 1 * MOVSC2
SW6 2.904E-04 1 * S1
SW6 2.669E-04 1 * Y1
SW6 4.600E-04 MAINTENANCE Contribution

SW7 = SWN1 + SWSCC1 + E1_A*E1_B
SW7 3.701E-04 TOTAL - SI, No LOSP, ASSA - No Operator Actions
SW7 2.987E-04 1 * MOVSC2
SW7 3.371E-05 1 * MOVSC1 * MOVSC1
SW7 3.366E-05 1 * SGVFS2

E.5 Service Water System (Continued)

```
SW7     2.188E-06  1  *  Z
SW7     1.873E-06  1  *  SGVFR2
SW7     1.011E-06  1  *  SWPPR4
SW7     5.078E-07  1  *  SGVFS1*SGVFS1
SW7     2.471E-07  1  *  T
SW7     1.150E-05 MAINTENANCE Contribution

SW8 = SWN1 + E1_A*E1_B
SW8     4.093E-05 TOTAL - No SI, No LOSP, ASSA - No Operator Actions
SW8     3.366E-05  1  *  SGVFS2
SW8     2.188E-06  1  *  Z
SW8     1.873E-06  1  *  SGVFR2
SW8     1.011E-06  1  *  SWPPR4
SW8     2.471E-07  1  *  T
SW8     5.078E-07  1  *  SGVFS1*SGVFS1
SW8     1.792E-07  1  *  S1*S2
SW8     1.670E-07  2  *  SGVFS1*SGVFR1
SW8     1.630E-07  1  *  Y1*Y2
SW8     1.757E-06 MAINTENANCE Contribution

SW9 = SWN4 + SWSCC4 + E1_A
SW9     5.565E-03 TOTAL - SI, No LOSP, One Train - No Op Actions
SW9     4.080E-03  1  *  MOVSC1
SW9     4.559E-04  1  *  SGVFS1
SW9     2.987E-04  1  *  MOVSC2
SW9     2.904E-04  1  *  S1
SW9     2.669E-04  1  *  Y1
SW9     1.857E-04  1  *  SGVFR1
SW9     3.366E-05  1  *  SGVFS2
SW9     1.106E-05  1  *  SWPPR2
SW9     1.340E-04 MAINTENANCE Contribution

SWA =  SWN4 + E1_A
SWA     1.286E-03 TOTAL - No SI, No LOSP, One Train - No Op Actions
SWA     4.559E-04  1  *  SGVFS1
SWA     2.904E-04  1  *  S1
SWA     2.669E-04  1  *  Y1
SWA     1.857E-04  1  *  SGVFR1
SWA     3.366E-05  1  *  SGVFS2
SWA     1.106E-05  1  *  SWPPR2
SWA     9.798E-06  2  *  SWPPR3
SWA     4.137E-06  1  *  E1_A
SWA     3.015E-06  1  *  SWPPR1*MOVO1
SWA     6.700E-05 MAINTENANCE Contribution /
```

The results for the Loss of One SWS Train Initiating Event are shown below:

```
L1SW    3.560E-03 TOTAL
L1SW    5.805E-05  1  *  PR2 $
L1SW    5.316E-04  1  *  E1_A $
L1SW    1.335E-03  1  *  PS1 * PR1 $
L1SW    2.107E-05  1  *  PR1 * PR1X $
L1SW    1.623E-03 MAINTENANCE CONTRIBUTION
```

E.5 Service Water System (Continued)

Cut Set Basic Events	Definitions
System:	
MOVSC2	Common cause failure of both SCC isolation valves to close on demand.
MOVSC1	Single SCC isolation valve fails to close on demand.
SGVFS2	Common cause failure of both SW switchgear ventilation fans to start on demand.
SGVFS1	Single SW switchgear ventilation fan fails to start on demand.
OP_TA	Operator fails to manually initiate cooling tower operation.
Z	SW intake MOV transfers closed.
SGVFR2	Common cause failure of both SW switchgear ventilation fans to run.
SGVFR1	Single SW switchgear ventilation fan fails to run.
MOVC2	Common cause failure of two MOVs to close on demand.
MOVC1	Single MOV fails to close on demand.
TMOV02	Common cause failure of two MOVs to open on demand.
TMOV01	Single MOV fails to open on demand.
CTPPS2	Common cause failure of both cooling tower pumps to start on demand.
CTPPS1	Single cooling tower pump fails to start on demand.
SWPPR4	Common cause failure of all four SW pumps to run.
CTFNS2	Common cause failure of two cooling tower fans to start on demand.
PRSOG2	Both normally operating SW pumps fail to restart given LOSP.
PRSOG1	Single SW pump fails to restart given LOSP.
MOVOG2	Common cause failure of two SW pump discharge MOVs to reopen on LOSP.
MOVOG1	Single SW pump discharge MOV fails to reopen on LOSP.
SWPPR3	Common cause failure of three SW pumps to run.
SWPPR2	Common cause failure of two SW pumps to run.
SWPPR1	Single SW pump fails to run.
E1_A	Common train discharge line valves transfer closed.
S1	Pumphouse switchgear dampers/filters fail.
Y1	Pumphouse switchgear ventilation fails.
CTFNS1	Single cooling tower fan fails to start on demand.
T	Tornado check damper transfer closed.
Initiating Event:	
PR2	Two SW pumps fail to run (common cause).
PR1, PR1X	Single SW pump fails to run.
E1_A	Common train discharge line valves transfer closed.
PS1	Single SW pump fails to start on demand.

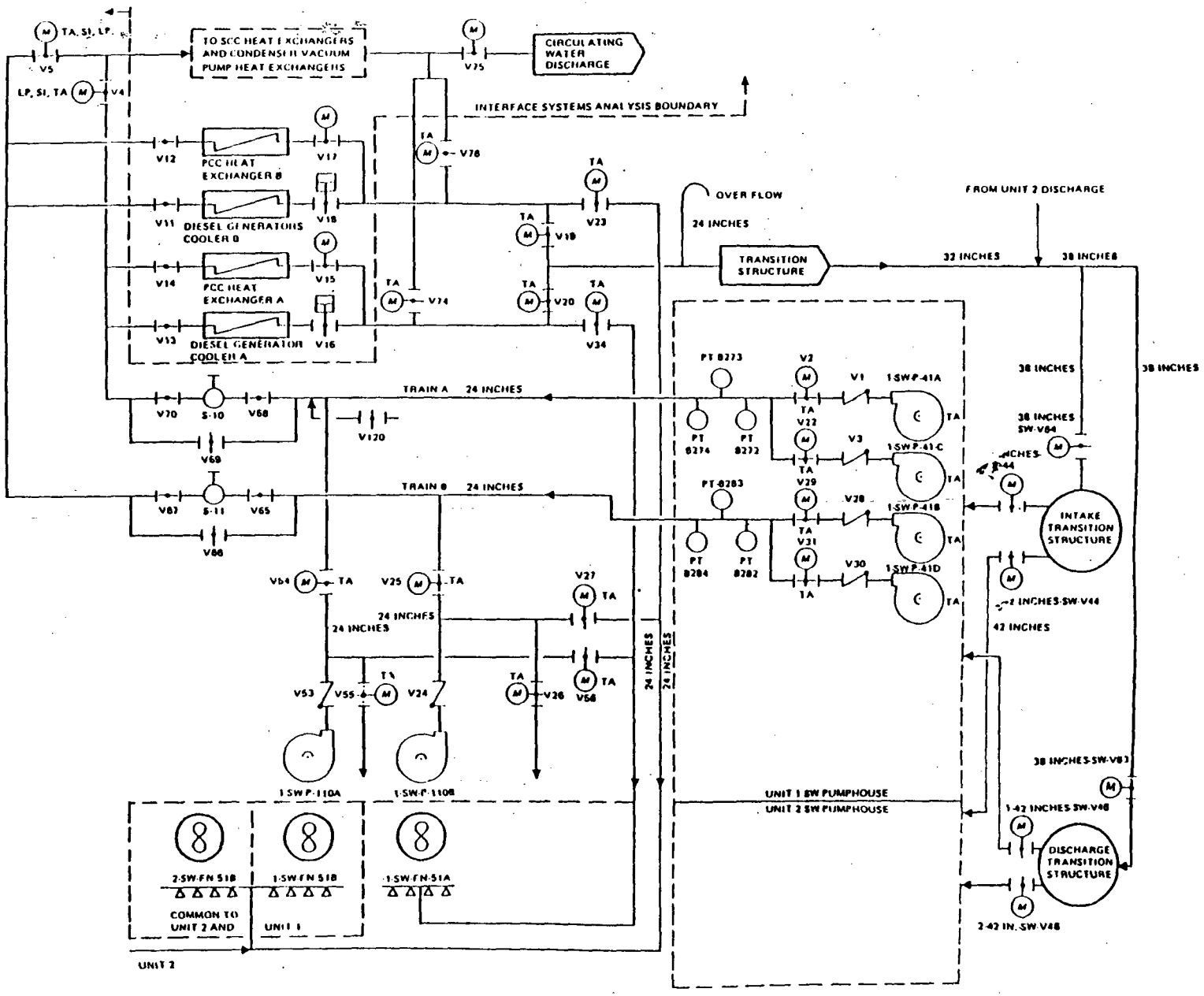

FIGURE E.5-1 SERVICE WATER SYSTEM (INCLUDES ULTIMATE HEAT SINK)

E.6 Primary Component Cooling Water (PCC) System

E.6.1 System Description

Function

The PCC System supplies cooling water to prevent overheating of components which are needed for plant operation and to maintain core heat removal and RCP seal integrity.

Configuration

The PCC System consists of two separate closed-loop cooling systems. Each loop, or train, contains two full-capacity centrifugal PCC pumps, one vertical shell and straight tube heat exchanger, and one head tank. One pump operates in each loop, while the second pump serves as a standby. (See Figure E.6-1 for Loop A; the other loop is similar.)

The RCP Thermal Barrier Cooling System (RCPTB) includes two heat exchangers, two full- capacity recirculation pumps, a head/relief tank, and motor-operated valves.

Dependencies

The PCC System depends on the Service Water System to provide cooling to the PCC heat exchangers. A subsystem of the PAH Ventilation System provides redundant ventilation in the PCC pump area should the normal PAH Ventilation System fail to provide adequate ventilation (e.g., during a loss of off-site power).

The PCC, PAH Ventilation, and RCP Thermal Barrier Cooling Systems are dependent upon the essential Electric Power System for ac motor power for fans and pumps; control power (ac and/or dc) for the automatic operation of motors, dampers, valves, and actuation signals; and for monitoring and indication of system parameters. The pneumatic dampers and air- operated valves require compressed air for normal functioning; they fail safe on loss of instrument air.

E.6 Primary Component Cooling Water System (Continued)

The PCC System is also dependent on SSPS/ESFAS to provide isolation signals to nonessential loads.

Operation

During normal operation, both loops of the PCC System are operating with one pump per loop in operation and the other in standby. The pumps and the heat exchanger valves can be controlled from the main control board and from the remote safe shutdown panel. Given a P signal, the nonessential loads inside containment supplied by PCC are isolated. Given a T signal, the nonessential loads outside containment are isolated.

Potential for Event Initiation

Loss of either train of PCC during normal plant operation requires a reactor trip within ten minutes following a loss of PCC to the RCP motor coolers.

E.6.2 System Model

The PCC System model includes two analyses:

1. Availability of PCC, and
2. Initiating event involving loss of one train of PCC.

System unavailability is quantified using the fault tree method (IRRAS) and RISKMAN software.

Top Event Definition

The PCC System is analyzed for Top Event PA (loss of PCC Loop A) and Top Event PB (loss of PCC Loop B) in the support systems event tree under three general boundary conditions. In the first case, the unit requires a continuous supply of PCC after an initiating event occurs (with off-site power available). The second case corresponds to an unavailability of off-site power. For this case, the unit requires the PCC pumps to restart and operate for 24 hours after the

E.6 Primary Component Cooling Water System (Continued)

emergency power sequencer functions. The third case is applied to initiating events which lead to the generation of a P signal, which requires nonessential cooling loads in the containment to be isolated.

The PAH Ventilation Subsystem is combined with the PCC System for quantification of loss of off-site power cases, and is included in Top Events PA and PB.

The RCP Thermal Barrier Cooling System quantification is not included in Top Events PA and PB, nor is it used in the event tree model. Either seal injection from the charging pumps or seal cooling from the RCP Thermal Barrier Cooling System is sufficient to prevent thermal degradation of the RCP seals and subsequent leakage. However, since both of these methods require PCC, RCP seal failure (Top Event NL) is conditioned on availability of PCC alone. Thus, the RCP Thermal Barrier Cooling System is not included in any top event.

Success Criteria

Success of the PCC System is defined as success of one of two trains, with success of a train corresponding to success of one of two PCC pumps per loop to start automatically (for LOSP) and continue to operate for 24 hours.

For loss of off-site power, success of the PAH ventilation subsystem is defined as success of either supply damper, fan, and exhaust damper train to start automatically when activated by its temperature-sensing device.

Analysis Conditions

The PCC System analysis assumes the plant is operating at normal full power operation prior to the initiating event, with one pump in each loop operating and the other in standby. Failure of the PAH ventilation subsystem to operate for 24 hours is assumed to cause failure of the PCC System for the loss of off-site power case.

No credit is taken for operator actions to recover failed equipment over the 24-hour period of this analysis.

E.6 Primary Component Cooling Water System (Continued)

The flow to PCC components may require some manual adjustment during the post-LOCA recirculation phase. These actions are assumed to be performed correctly and are not included in this analysis.

E.6.3 Results (See below for definition of cut set basic events.)

PCC1 2.623E-06 TOTAL Off-Site Power & No P Signal
PCC1 1.725E-06 1 * BLKCA*BLKCB
PCC1 8.256E-07 1 * PR4
PCC1 1.448E-08 4 * PR1*PR3
PCC1 7.722E-08 Maintenance Unavailability

PCC2 4.442E-05 TOTAL Off-Site Power Unavailable
PCC2 3.411E-05 1 * VF2S
PCC2 2.846E-06 1 * BLCKG
PCC2 1.908E-06 1 * VF2R
PCC2 1.725E-06 1 * BLKCA*BLKCB
PCC2 1.440E-06 1 * PS4
PCC2 8.696E-07 Maintenance Unavailability

PCC3 2.623E-06 TOTAL P Signal Present
PCC3 1.725E-06 1 * BLKCA*BLKCB
PCC3 8.256E-07 1 * PR4
PCC3 1.448E-08 4 * PR1*PR3
PCC3 7.722E-08 Maintenance Unavailability

PCC4 6.373E-04 TOTAL Off-Site Power & No P Signal (One Train)
PCC4 6.180E-04 1 * BLKCA
PCC4 4.300E-06 1 * PR2
PCC4 2.917E-06 2 * PR3
PCC4 1.937E-06 1 * PR1*PR1
PCC4 1.859E-06 1 * PS1*PR1
PCC4 1.799E-05 MAINT1 Pump Maintenance on Train A

PCC5 1.627E-03 TOTAL Off-Site Power Unavailable (One Train)
PCC5 6.180E-04 1 * BLKCA
PCC5 4.459E-04 1 * VF1S
PCC5 2.742E-04 1 * H2
PCC5 1.886E-04 1 * VF1R
PCC5 3.411E-05 1 * VF2S
PCC5 5.686E-05 Maintenance Unavailability

PCC6 6.373E-04 TOTAL P Signal Present (One Train)
PCC6 6.180E-04 1 * BLKCA
PCC6 4.300E-06 1 * PR2
PCC6 2.917E-06 2 * PR3
PCC6 1.937E-06 1 * PR1*PR1
PCC6 1.859E-06 1 * PS1*PR1
PCC6 1.799E-05 Maintenance Unavailability

E.6 Primary Component Cooling Water System (Continued)

Cut Set Basic Events	Definitions
PS1	PCC pump fails to start on demand.
PS4	All four PCC pumps fail to start - common cause.
PR1	PCC pump fails to run for 24 hours.
PR2	Two PCC pumps fail to run for 24 hours - common cause.
PR3	Three PCC pumps fail to run for 24 hours - common cause.
PR4	All four PCC pumps fail to run for 24 hours - common cause.
BLKCA, BLKCB	Failure of PCC-to-RCP water cooling valves.
BLCKG	PAH intake louver/exhaust dampers fail.
VF1S	PAH ventilation fan fails to start.
VF1R	PAH ventilation fan fails to run for 24 hours.
VF2S	Both PAH ventilation fans fail to start - common cause.
VF2R	Both PAH ventilation fans fail to run or 24 hours - common cause.

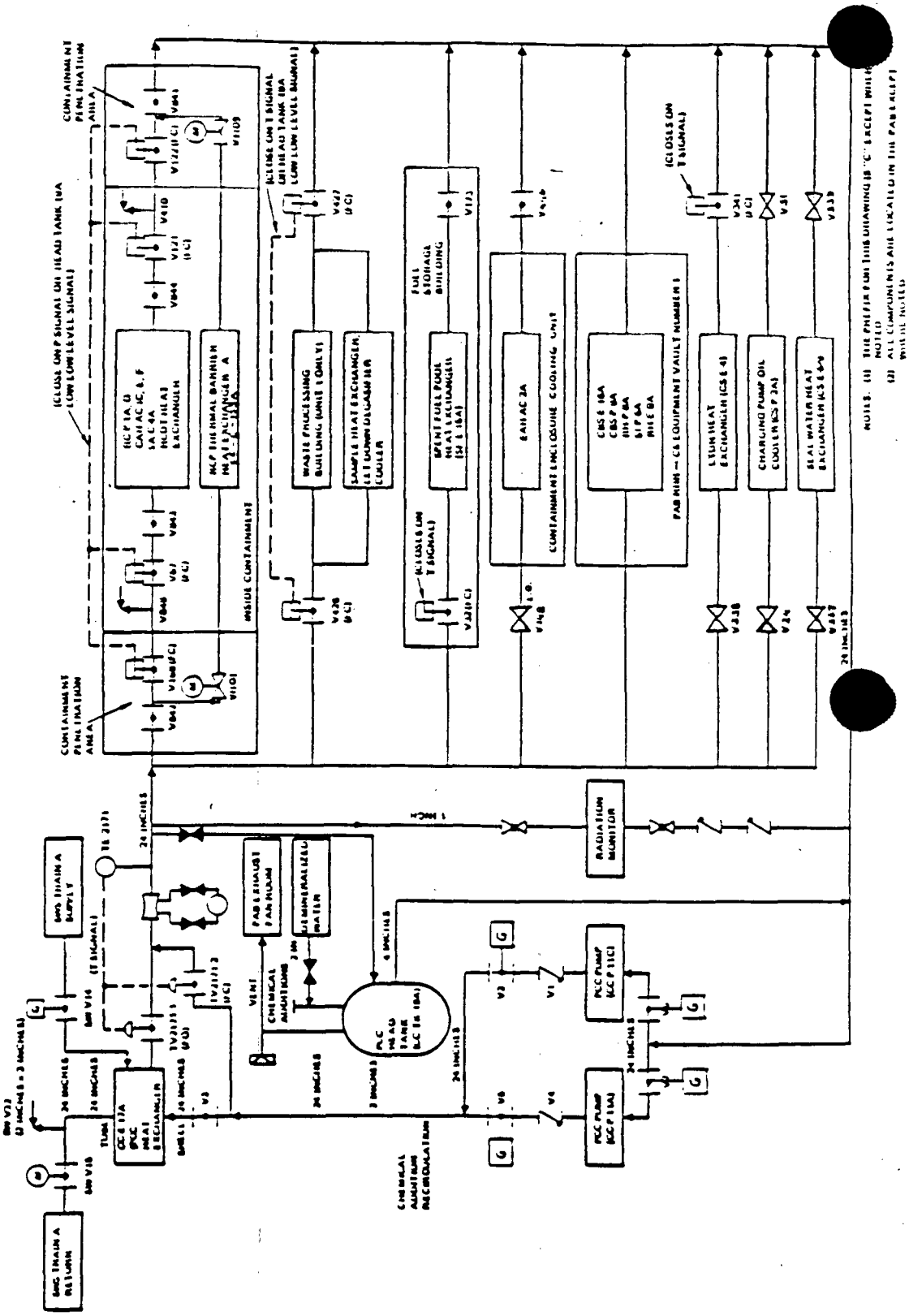

FIGURE E.6-1 PRIMARY COMPONENT COOLING WATER SYSTEM SIMPLIFIED SCHEMATIC DIAGRAM (LOOP A)

E.7 Solid State Protection System

E.7.1 System Description

Function

The Solid State Protection System (SSPS) processes the output from sensors which monitor various plant parameters to determine if safe operating limits are being maintained and if primary system or containment boundaries are in jeopardy. When the SSPS has determined that unsatisfactory conditions exist, signals are sent to the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) to initiate protective actions (see Figure E.7-1).

Configuration

The SSPS is comprised of redundant nuclear and non-nuclear detector channels serving as input to two identical and independent logic trains, A and B. The SSPS consists of the following major components:

- Detectors, amplifiers, and cables
- Bistables
- SSPS logic matrices (input and output)
- Manual actuation circuits

(See Figure E.7-2.)

Dependencies

The SSPS is dependent on the 120 V ac vital instrument power for the input sensors, logic channels, and output relays. The input sensors fail safe on loss of power. The output relays are disabled on loss of power.

Because the ESFAS train associated with a particular SSPS train receives 120 V ac vital instrument power from the primary source of power for the SSPS train, failure of a single 120 V ac instrument panel (PP-1A or PP-1B) is assumed to fail a single ESFAS train.

E.7 Solid State Protection System (Continued)

Operation

During normal operation, the SSPS continuously monitors plant parameters and analyzes this information through its logic channels. When conditions exist, the SSPS sends actuation signals to the RTS and/or ESFAS. Manual initiation of required signals is also provided.

E.7.2 System Model

System unavailability is quantified using the fault tree method (IRRAS) and RISKMAN software.

Top Event Definition

The quantification of the SSPS appears in the support systems event tree as Top Event SA (SSPS Train A) and as Top Event SB (SSPS Train B). Top Events SA and SB only model automatic generation of the required signal(s). Top Event MT, which follows Top Events SA and SB in the support tree, models operator actions to generate a manual reactor trip signal. Top Event OS, which follows Top Events EA and EB in the support tree, models operator actions to manually generate an SI signal or locally start pumps from the MCB if required.

Success Criteria

System success is defined as at least one SSPS train sending an activating signal to the RTS and/or ESFAS, as required, to initiate protective actions when an unsafe condition exists.

Analysis Conditions

The SSPS is analyzed under the following assumptions:

1. The reactor is operating at normal full power prior to the occurrence of any initiating event.

2. Only one parameter is available to generate the required signal.

E.7 Solid State Protection System (Continued)

3. All signals are conservatively modeled as using 2-out-of-3 logic (most utilize a 2-out-of-4 logic).

4. Manual initiation of SSPS signals is not analyzed in the system analysis but is included as a separate top event in the support tree.

E.7.3 Results (See below for definition of cut set basic events.)

```
SC1     4.014E-04 TOTAL BOTH SSPS TRAINS AVAILABLE - LL/ML/SLBI
SC1     3.993E-04  1  *  HE1
SC1     3.288E-07  1  *  IR6_6
SC1     5.914E-07  1  *  PS1*PS2
SC1     7.835E-08  1  *  LC2_2
SC1     3.810E-08  3  *  PC2_3
SC1     2.637E-07 Maintenance Unavailability
SC1     1.030E-06 Test Unavailability

SCA     1.606E-03 TOTAL ONE SSPS TRAIN AVAILABLE - LL/ML/SLBI
SCA     3.993E-04  1  *  HE1
SCA     3.310E-04  1  *  PS1
SCA     8.159E-05  1  *  LC
SCA     9.339E-06  3  *  IR2_6
SCA     8.355E-07  10  *  IR3_6
SCA     1.575E-04 Maintenance Unavailability
SCA     6.250E-04 Test Unavailability

SC2     2.022E-06 TOTAL BOTH SSPS TRAINS AVAILABLE - SMALL LOCA
SC2     3.288E-07  1  *  IR6_6
SC2     5.914E-07  1  *  PS1*PS2
SC2     7.835E-08  1  *  LC2_2
SC2     1.379E-07 Maintenance Unavailability
SC2     5.305E-07 Test Unavailability

SCB     1.207E-03 TOTAL ONE SSPS TRAIN AVAILABLE - SMALL LOCA
SCB     3.310E-04  1  *  PS1
SCB     8.159E-05  1  *  LC
SCB     9.339E-06  3  *  IR2_6
SCB     8.355E-07  10  *  IR3_6
SCB     5.346E-07  3  *  IR*IR
SCB     1.575E-04 Maintenance Unavailability
SCB     6.250E-04 Test Unavailability

SC3     4.118E-06 TOTAL BOTH SSPS TRAINS AVAILABLE - GT/SGTR/SLBO
SC3     2.096E-06  1  *  HE3
SC3     3.288E-07  1  *  IR6_6
SC3     5.914E-07  1  *  PS1*PS2
SC3     7.835E-08  1  *  LC2_2
SC3     3.810E-08  3  *  PC2_3
SC3     1.386E-07 Maintenance Unavailability
SC3     5.331E-07 Test Unavailability
```

E.7 Solid State Protection System (Continued)

```
SCC     1.209E-03 TOTAL ONE SSPS TRAIN AVAILABLE - GT/SGTR/SLBO
SCC     3.310E-04  1  *  PS1
SCC     8.159E-05  1  *  LC
SCC     9.339E-06  3  *  IR2_6
SCC     2.096E-06  1  *  HE3
SCC     1.575E-04 Maintenance Unavailability
SCC     6.250E-04 Test Unavailability
```

Cut Set Basic Events	Definitions
LC	Logic channel fails.
LC2_2	Both logic channels fail - common cause.
PS1, PS2	DC power supply fails.
IR	Input relay fails.
IR2_6	Two input relays fail - common cause.
IR3_6	Three input relays fail - common cause.
IR6_6	More than three (and up to six) input relays fail - common cause.
PC	Parameter channel fails.
PC2_3	Two parameter channels fail - common cause.
PC3_3	Three parameter channels fail - common cause.
HE1, HE2, HE3	Human error of miscalibration.

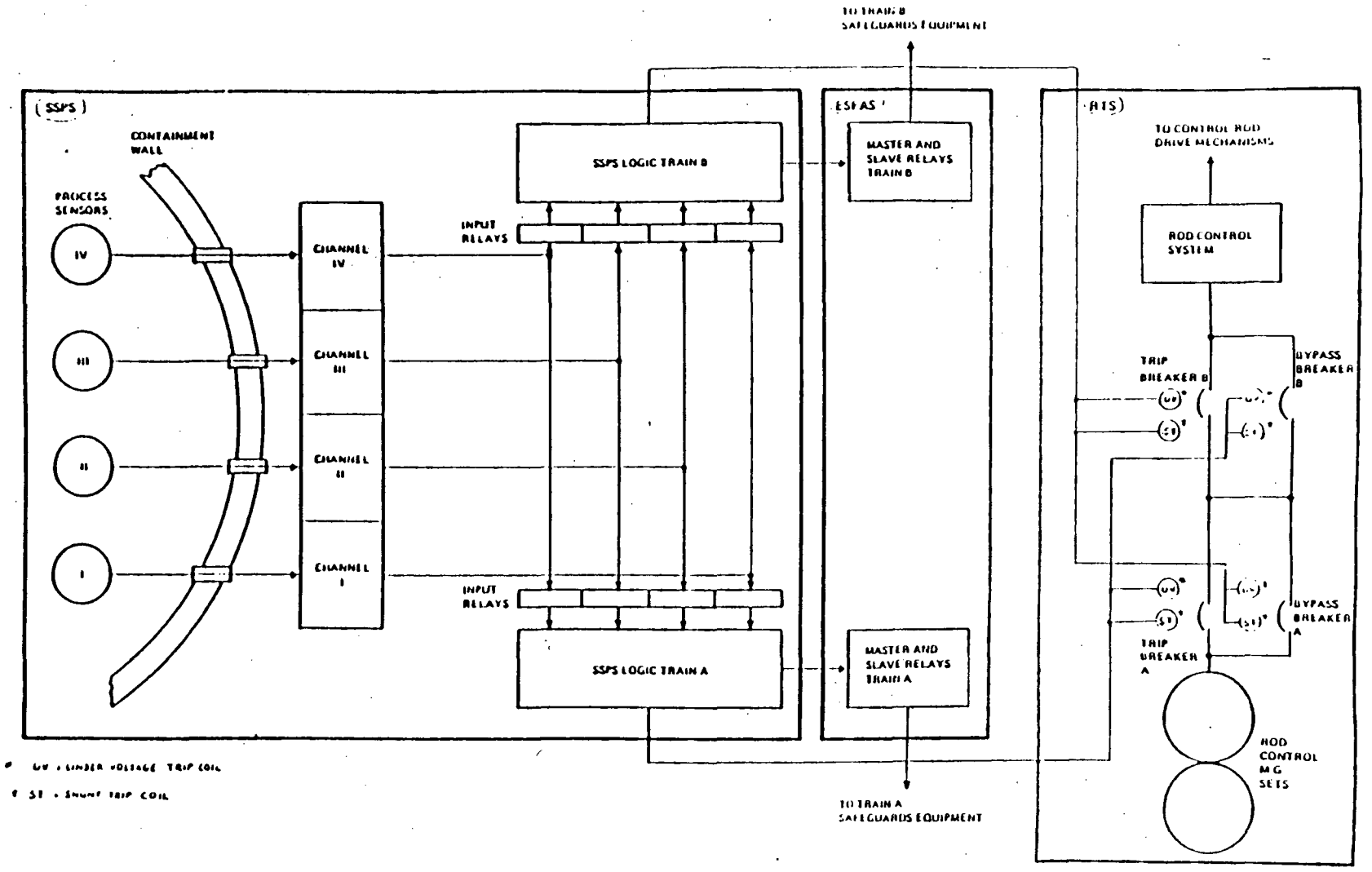

FIGURE E.7-1 SSPS, ESFAS, AND RTS

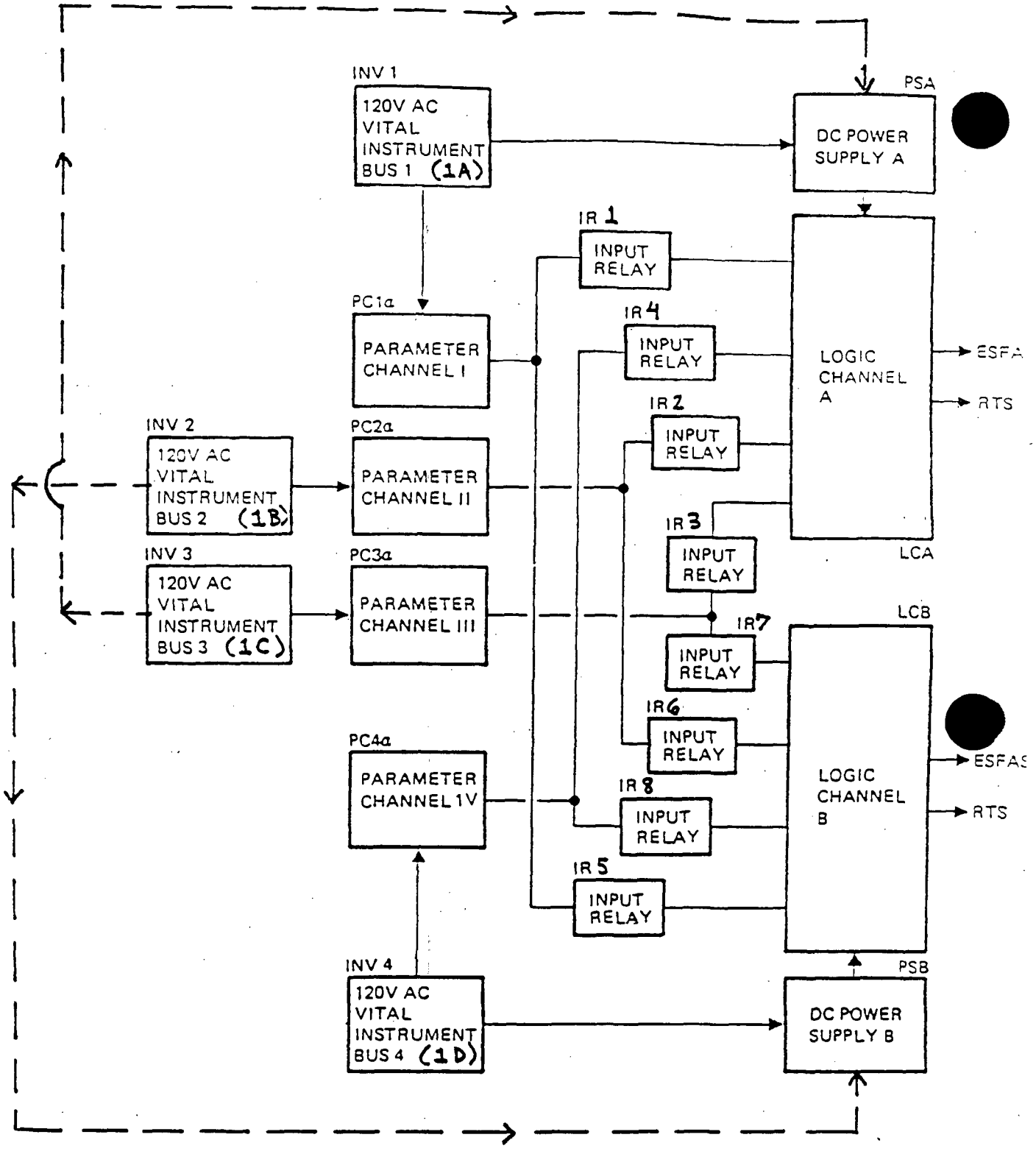

FIGURE E.7-2 SOLID STATE LOGIC SYSTEM BLOCK DIAGRAM MODEL

E.8 Engineered Safety Features Actuation System (ESFAS)

E.8.1 System Description

Function

ESFAS provides signals to the engineered safety features systems to provide the proper sequencing of safety systems functions.

Configuration

The ESFAS consists of two redundant trains of master and slave relays (see Figure E.8-1). The master relays receive input signals from the SSPS. The system is also composed of a series of manual actuation circuits. Support is provided by the 120V AC instrument power. The following major systems have components which are actuated by this system:

- Safety Injection
- Residual Heat Removal
- Chemical and Volume Control
- Emergency Feedwater
- Containment Building Spray
- Main Steam
- Main Feedwater
- Service Water
- Primary Component Cooling
- Emergency Diesel Generators
- Containment Ventilation
- Containment Isolation

Dependencies

Support for the ESFAS is provided by the 120V AC Vital Instrument Power System. Loss of a single 120 V ac instrument power bus fails the associated ESFAS train. In addition, the ESFAS depends upon the SSPS for the generation of input signals.

E.8 Engineered Safety Features Actuation System (ESFAS) (Continued)

Operation

The ESFAS receives signals from the output logic channels of the SSPS. Depending on the signal received, the ESFAS will automatically initiate any or all of a variety of equipment actuations. The system can be manually actuated for the following protective actions:

- Main Steam Line Isolation
- Safety Injection Actuation
- Containment Spray and Containment Isolation
- Emergency Feedwater (EFW) System Actuation

E.8.2 System Model

The ESFAS model analyzes and quantifies the unavailability of the master and slave relays given signals from the SSPS logic trains to actuate safety equipment.

Quantification is performed using the fault tree method (IRRAS) and RISKMAN software.

Top Event

The ESFAS is included in the support systems event tree (SUPPORT) as Top Events EA and EB for Trains A and B, respectively, and also as Top Event OS. Top Events EA and EB only model automatic actuation of the required equipment. Top Event OS models operator recovery of ESFAS and SSPS failure(s).

Success Criteria

System success for the ESFAS is defined as processing all signals received from the SSPS to actuate the required equipment for accident mitigation, dependent on the initiating event.

Analysis Conditions

The ESFAS is analyzed under the following assumptions:

E.8 Engineered Safety Features Actuation System (ESFAS) (Continued)

- The reactor is operating at normal full power prior to the occurrence of any initiating event.

- The ESFAS is operated and maintained in accordance with the plant Technical Specifications.

E.8.3 Results

The dominant contributors to system unavailability for all support states are provided as follows. (The definitions of the cut set basic events are given at the end of this section.)

```
EC1     7.665E-05 TOTAL LLOCA/MLOCA - Both Trains of SSPS Available
EC1     2.853E-05  1  *  CCSR1
EC1     8.876E-06  1  *  CCMR1
EC1     4.337E-06  1  *  SR615A*SR615B
EC1     2.776E-06  1  *  SR601A*SR601B
EC1     2.776E-06  1  *  SR602A*SR602B
EC1     2.776E-06  1  *  SR625A*SR625B
EC1     3.430E-06 Maintenance Unavailability

ECA     1.079E-02 TOTAL LLOCA/MLOCA - One Train
ECA     1.120E-03  1  *  SR615A
ECA     8.957E-04  1  *  SR625A
ECA     8.957E-04  1  *  SR601A
ECA     8.957E-04  1  *  SR602A
ECA     6.718E-04  1  *  SR605A
ECA     6.718E-04  1  *  SR622A
ECA     6.718E-04  1  *  SR623A
ECA     1.577E-04 Maintenance Unavailability

EC2     7.790E-05 TOTAL Steam Line Break inside Containment - Both T
EC2     2.789E-05  1  *  CCSR2
EC2     8.850E-06  1  *  CCMR2
EC2     4.337E-06  1  *  SR615A*SR615B
EC2     2.776E-06  1  *  SR601A*SR601B
EC2     2.776E-06  1  *  SR602A*SR602B
EC2     2.776E-06  1  *  SR625A*SR625B
EC2     3.644E-06 Maintenance Unavailability

ECB     1.146E-02 TOTAL Steam Line Break inside Containment - One Tr
ECB     1.120E-03  1  *  SR615A
ECB     8.957E-04  1  *  SR601A
ECB     8.957E-04  1  *  SR602A
ECB     8.957E-04  1  *  SR625A
ECB     6.718E-04  1  *  SR605A
ECB     6.718E-04  1  *  SR622A
```

E.8 Engineered Safety Features Actuation System (ESFAS) (Continued)

ECB 6.718E-04 1 * SR623A
ECB 1.578E-04 Maintenance Unavailability

EC3 6.973E-05 TOTAL Small LOCA - Both Trains of SSPS Available
EC3 2.931E-05 1 * CCSR3
EC3 8.984E-06 1 * CCMR3
EC3 4.337E-06 1 * SR615A*SR615B
EC3 2.776E-06 1 * SR601A*SR601B
EC3 2.776E-06 1 * SR602A*SR602B
EC3 1.561E-06 1 * SR605A*SR605B
EC3 1.561E-06 1 * SR622A*SR622B
EC3 1.561E-06 1 * SR623A*SR623B
EC3 2.710E-06 Maintenance Unavailability

ECC 8.550E-03 TOTAL Small LOCA - One Train
ECC 1.120E-03 1 * SR615A
ECC 8.957E-04 1 * SR601A
ECC 8.957E-04 1 * SR602A
ECC 6.718E-04 1 * SR605A
ECC 6.718E-04 1 * SR622A
ECC 6.718E-04 1 * SR623A
ECC 1.573E-04 Maintenance Unavailability

EC4 6.973E-05 TOTAL Steam Generator Tube Ruptures - Both Trains
EC4 2.931E-05 1 * CCSR4
EC4 8.984E-06 1 * CCMR4
EC4 4.337E-06 1 * SR615A*SR615B
EC4 2.776E-06 1 * SR601A*SR601B
EC4 2.776E-06 1 * SR602A*SR602B
EC4 1.561E-06 1 * SR605A*SR605B
EC4 1.561E-06 1 * SR622A*SR622B
EC4 1.561E-06 1 * SR623A*SR623B
EC4 2.710E-06 Maintenance Unavailability

ECD 8.550E-03 TOTAL Steam Generator Tube Ruptures - One Train
ECD 1.120E-03 1 * SR615A
ECD 8.957E-04 1 * SR601A
ECD 8.957E-04 1 * SR602A
ECD 6.718E-04 1 * SR605A
ECD 6.718E-04 1 * SR622A
ECD 6.718E-04 1 * SR623A
ECD 4.479E-04 1 * SR607A
ECD 1.573E-04 Maintenance Unavailability

EC5 7.075E-05 TOTAL Steam Line Break outside Containment - Both
EC5 2.845E-05 1 * CCSR5
EC5 8.941E-06 1 * CCMR5
EC5 4.337E-06 1 * SR615A*SR615B
EC5 2.776E-06 1 * SR601A*SR601B
EC5 2.776E-06 1 * SR602A*SR602B
EC5 1.561E-06 1 * SR605A*SR605B
EC5 1.561E-06 1 * SR622A*SR622B
EC5 1.561E-06 1 * SR623A*SR623B
EC5 2.924E-06 MAINTA Maintenance on Train A

E.8 Engineered Safety Features Actuation System (ESFAS) (Continued)

ECG	9.221E-03			TOTAL Steam Line Break outside Containment - One Train
ECG	1.120E-03	1	*	SR615A
ECG	8.957E-04	1	*	SR601A
ECG	8.957E-04	1	*	SR602A
ECG	6.718E-04	1	*	SR605A
ECG	6.718E-04	1	*	SR622A
ECG	6.718E-04	1	*	SR623A
ECG	4.479E-04	1	*	SR607A
ECG	4.479E-04	1	*	SR627A
ECG	1.575E-04			Maintenance Unavailability
EC6	2.471E-05			TOTAL Transient - Both Trains of SSPS Available
EC6	1.138E-05	1	*	CCMR6
EC6	1.138E-05	1	*	CCSR6
EC6	1.735E-07	1	*	MR515A*MR515B
EC6	1.735E-07	1	*	MR515A*SR640B
EC6	1.735E-07	1	*	MR515B*SR640A
EC6	1.735E-07	1	*	MR516A*MR516B
EC6	1.735E-07	1	*	MR516A*SR641B
EC6	1.735E-07	1	*	MR516B*SR641A
EC6	1.735E-07	1	*	SR640A*SR640B
EC6	1.735E-07	1	*	SR641A*SR641B
EC6	3.282E-07			Maintenance Unavailability
ECH	1.160E-03			TOTAL Transient - One Train
ECH	2.239E-04	1	*	MR515A
ECH	2.239E-04	1	*	MR516A
ECH	2.239E-04	1	*	SR640A
ECH	2.239E-04	1	*	SR641A
ECH	1.081E-04	1	*	INVA
ECH	1.562E-04			Maintenance Unavailability

Cut Set Basic Events	Definitions
SR***A, SR***B	Slave relays fail.
MR***A, MR***B	Mater relays fail.
CCMR1,2,3,4,5,6	Common cause failure between master relays.
CCSR1,2,3,4,5,6	Common cause failure between slave relays.
INVA, INVB	120 V ac vital instrument power inverter fails.

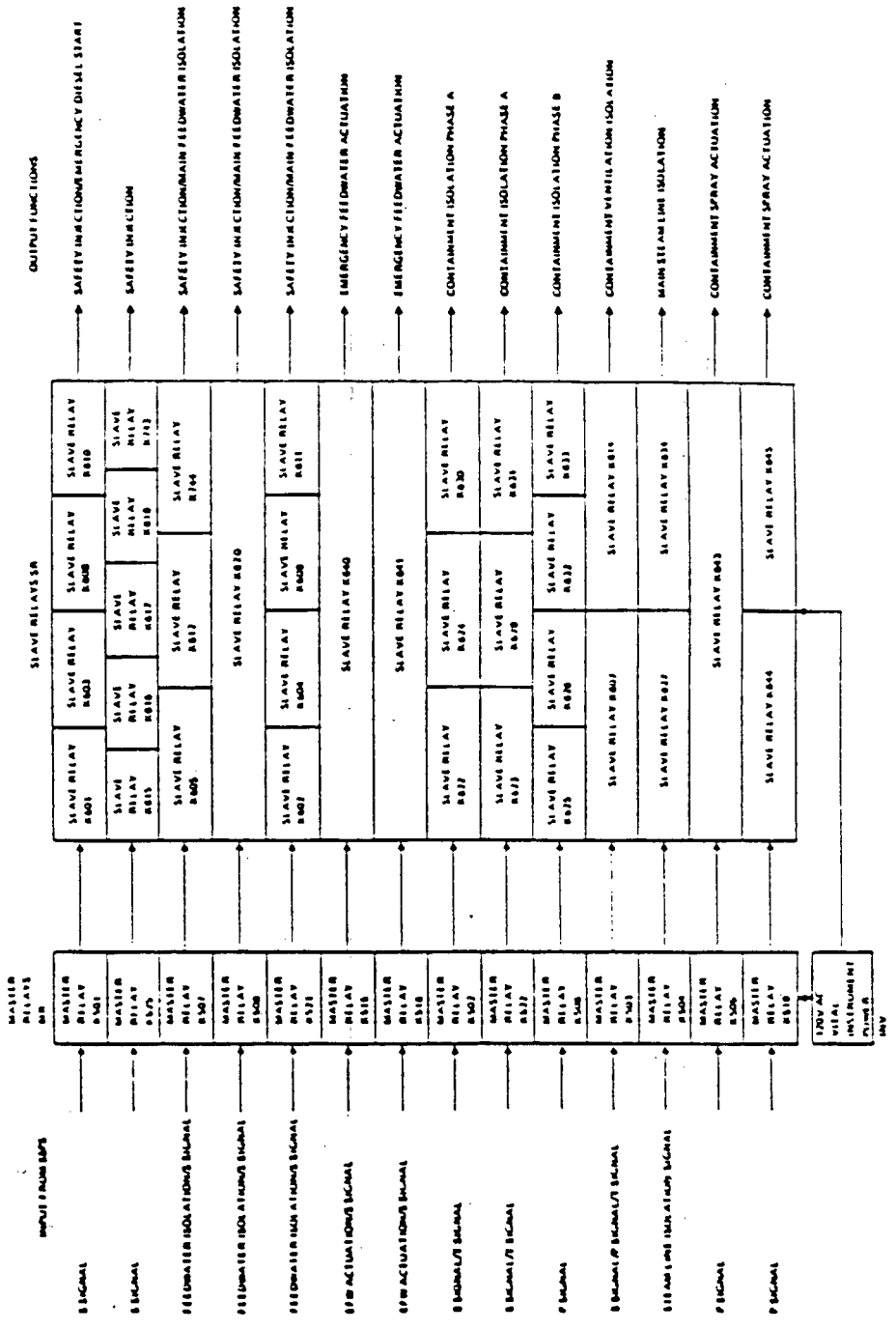

FIGURE E.8-1 ESFAS BLOCK DIAGRAM MODEL
(One train shown - other train identical)

E.9 Reactor Trip System

E.9.1 System Description

Function

The RTS initiates a trip (reactor shutdown) upon receiving a trip signal from the Solid State Protection System (SSPS), thereby ensuring integrity of the fuel, core, and reactor coolant system.

Configuration

The RTS consists of the following major components (see Figure E.9-1):

- Reactor trip switchgear (two reactor trip breakers and two bypass breakers).
- Control rods.
- Undervoltage and shunt trip coils (two undervoltage coils and two shunt trip coils).
- Manual actuation circuits.
- Two control rod motor generator sets.

Dependencies

With off-site power available, the RTS requires an output signal from the SSPS. The reactor trip breaker receives dc power for the undervoltage coil from the SSPS, and the SSPS receives power from the 120 V ac instrument bus. The reactor trip breaker requires dc power for the operation of the shunt trip coil. Loss of 120 V ac power would initiate a reactor trip.

E.9 Reactor Trip System (Continued)

Operation

During normal reactor operation, the reactor trip breakers are closed (and bypass breakers open) allowing power to the control rod drive mechanisms. When a reactor trip is required, the SSPS generates a signal which de-energizes the undervoltage coils and energizes the shunt trip coils. The trip plungers are released and spring pressure forces the breakers open, interrupting power to the control rod drive mechanisms and causing the control rod cluster assemblies to fall into the reactor core.

A manual reactor trip signal can be generated by two switches. Either switch actuates Train A breaker, Train B breaker, and the bypass breakers.

E.9.2 System Model

The model for the Reactor Trip System includes the undervoltage and shunt trip coils, the reactor trip breakers, and the control rod drives.

Quantification is performed using the fault tree method (IRRAS) and RISKMAN software.

Top Event

The RTS is included in the support system's event tree as top event RT. All sequences require that a reactor trip signal be generated automatically from the SSPS (Top Events SA or SB) or manually (Top Event MT) if a signal is not already present (from the initiating event). The systems analysis for the RTS does not include any manual actions. Top Event MR in the ATWS event tree, however, models automatic or manual rod insertion in the event Top Event RT fails.

Success Criteria

The success criteria for the RTS is defined as no more than one control rod assembly failing to insert into the core upon demand. Thus, failure of the RTS is defined as failure of two or more control rods to insert upon actuation.

E.9 Reactor Trip System (Continued)

Analysis Conditions

The RTS is analyzed under the following assumptions:

- The reactor is operating at normal power prior to the initiating event.
- Manual actuation of a reactor trip is not considered in this portion of the analysis.
- The RTS is operated and maintained in accordance with the plant Technical Specifications.

E.9.3 Results

The dominant contributors to system unavailability for all support states are given as follows. (The definitions of the cut set basic events are provided at the end of this section.)

RT1	1.371E-04	TOTAL Both Trains of Signals (Automatic or Manual)
RT1	1.242E-04	1 * CB2
RT1	4.140E-06	1 * CB1 * CB1
RT1	7.475E-06	1 * CR
RT1	2.487E-09	1 * UV2 * ST2
RT1	1.229E-06	Unavailability due to testing
RT2	1.771E-03	TOTAL One Train of Signals Available (Automatic or Manual)
RT2	1.638E-03	1 * CB1
RT2	1.242E-04	1 * CB2
RT2	7.475E-06	1 * CR
RT2	1.229E-06	Unavailability due to testing
RT3	7.480E-06	TOTAL Loss of Off-Site Power
RT3	7.475E-06	1 * CR
RT3	5.191E-09	Unavailability due to testing

Cut set Basic Events	Definitions
CB1	Reactor trip breaker fails to open.
CB2	Both reactor trip breakers fail to open - common cause.
UV2	Both undervoltage (UV) coils fail - common cause.
ST2	Both shunt trip coils fail - common cause.
CR	Control rod drives fail.

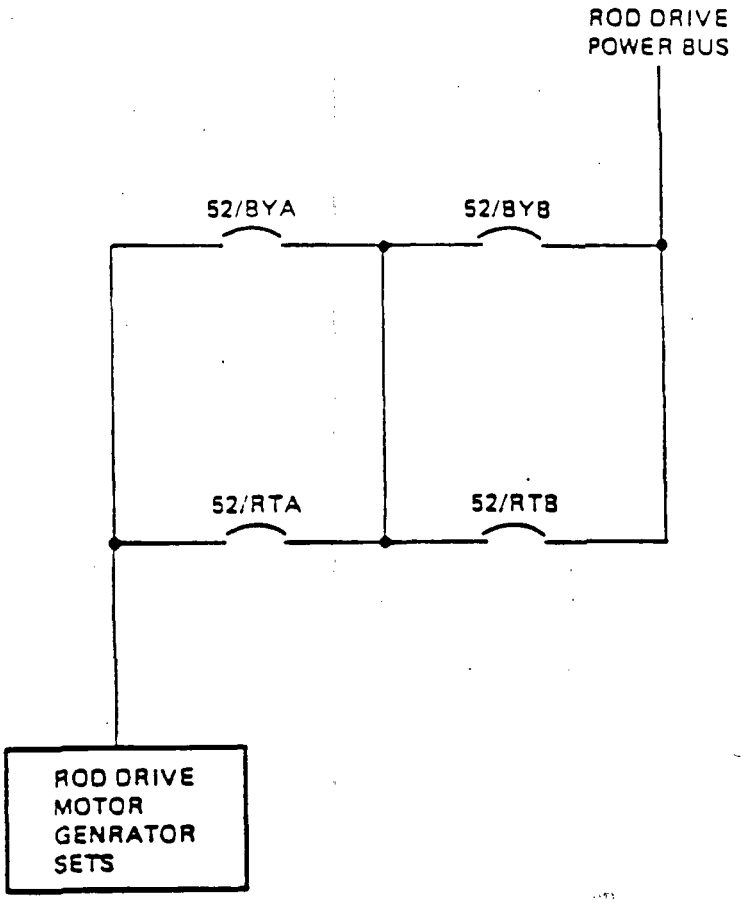

FIGURE E.9-1 ARRANGEMENT OF REACTOR TRIP BREAKERS

E.10 Emergency Air Handling (EAH) System

E.10.1 System Description

Function

The Emergency Air Handling (EAH) System provides ventilation and component cooling to permit continuous operation of equipment in the following areas:

- Charging pump areas
- Safety injection pump areas
- Residual heat removal and containment spray equipment areas
- Containment structure annulus enclosure area

Configuration

The EAH System is comprised of the Containment Enclosure Cooling System, the Containment Enclosure Emergency Air Cleaning System (CEEACS), and the Main Steam and Feedwater Pipe Chase Ventilation System. The CEEACS does not perform a component cooling function and, therefore, is not considered in the analysis of the EAH System. The Main Steam and Feedwater Pipe Chase Ventilation System is not required during emergency conditions and is not analyzed further in this study.

The Containment Enclosure Cooling System consists of two trains, each having supply and return fans and a cooler unit (see Figure E.10-1).

Dependencies

The Containment Enclosure Cooling System is dependent on the PCC System, normal and emergency electric power, containment isolation (T signal), and instrument air.

Operation

One train (supply fan, return fans, and cooler units) of the Containment Enclosure Cooling System is required to operate continuously to maintain the areas served within the

E.10 Emergency Air Handling (EAH) System (Continued)

design limits with the redundant train serving as a standby. Under emergency conditions (T signal), the PAB Air Handling System is isolated from the Containment Enclosure Cooling System.

E.10.2 System Model

The EAH System model includes the Containment Enclosure Cooling System only. The Main Steam and Feedwater Pipe Chase Ventilation System and CEEACS are not modeled since they do not provide component cooling functions to safeguards equipment.

Quantification is performed using the fault tree method (IRRAS) and RISKMAN software.

Top Event

The results for the EAH System quantification are used directly as Top Event EH in the support systems event tree.

Success Criteria

Success of the EAH System has been defined in this analysis as one of two trains providing ventilation to the emergency equipment areas for 24 hours following an initiating event. Success also includes isolation of the containment enclosure area cooling from the PAB Air Handling System.

Analysis Conditions

The EAH System (Containment Enclosure Cooling System) is analyzed under the following assumptions:

- The plant is assumed to be at normal full power operation prior to the initiating event, with one train running and the other in standby.

E.10 Emergency Air Handling (EAH) System (Continued)

- Except for manual startup of the standby unit on loss of PCC to the operating unit, only automatic operation of the ventilation trains are considered in the analysis.

- Failure of the EAH System to operate for 24 hours is conservatively assumed to cause long-term failure of the charging, SI, RHR, and CBS pumps.

- Fire dampers are included in the analysis since their failure to remain open can block ventilation to components. Balancing dampers, on the other hand, are not included because they are mechanically held in position and are not required to change position.

E.10.3 Results

The dominant contributors to system unavailability for all support states are provided below (definitions of the cut set basic events are provided at the end of this section):

```
EH1     6.099E-05 TOTAL All Support Systems Available
EH1     3.531E-05  1  *  SFNS2
EH1     2.296E-05  1  *  C
EH1     5.344E-07  1  *  SFNS  *  SFNS
EH1     1.194E-07  1  *  B180  *  SFNS
EH1     1.500E-06 Maintenance Unavailability

EH2     1.321E-03 TOTAL Loss of One T Signal
EH2     5.376E-04  1  *  D
EH2     4.625E-04  1  *  SFNS
EH2     2.626E-04  1  *  B180
EH2     3.531E-05  1  *  SFNS2
EH2     1.497E-05 Maintenance Unavailability

EH3     1.018E-02 TOTAL Loss of One PCC Train
EH3     5.515E-04  1  *  B5
EH3     4.929E-04  1  *  FNS
EH3     3.531E-05  1  *  SFNS2
EH3     2.296E-05  1  *  C
EH3     9.078E-03 Maintenance Unavailability

EH4     1.142E-02 TOTAL Loss of One PCC Train, One T Signal
EH4     5.515E-04  1  *  B5
EH4     5.376E-04  1  *  D
EH4     4.929E-04  1  *  FNS
```

E.10 Emergency Air Handling (EAH) System (Continued)

EH4	4.625E-04	1 * SFNS
EH4	2.626E-04	1 * B180
EH4	9.078E-03	Maintenance Unavailability
EH5	7.071E-05	TOTAL LOSP, All Support Systems Available
EH5	3.531E-05	1 * SFNS2
EH5	2.296E-05	1 * C
EH5	2.847E-06	2 * FNS2
EH5	1.204E-06	2 * FNS * FNS
EH5	7.315E-07	4 * FNS3
EH5	5.900E-06	Maintenance Unavailability
EH6	1.331E-03	TOTAL LOSP, Loss of One T Signal
EH6	5.376E-04	1 * D
EH6	4.625E-04	1 * SFNS
EH6	2.626E-04	1 * B180
EH6	3.531E-05	1 * SFNS2
EH6	1.937E-05	Maintenance Unavailability
EH7	1.582E-03	TOTAL LOSP, Loss of One Emergency Bus
EH7	4.929E-04	1 * FNS
EH7	4.625E-04	1 * SFNS
EH7	2.626E-04	1 * B180
EH7	2.626E-04	1 * B31
EH7	3.531E-05	1 * SFNS2
EH7	2.296E-05	1 * C
EH7	1.976E-05	Maintenance Unavailability
EH8	2.103E-03	TOTAL LOSP, Loss of One Emer. Bus and One T Signal
EH8	5.376E-04	1 * D
EH8	4.929E-04	1 * FNS
EH8	4.625E-04	1 * SFNS
EH8	2.626E-04	1 * B180
EH8	2.626E-04	1 * B31
EH8	3.531E-05	1 * SFNS2
EH8	2.643E-05	Maintenance Unavailability

Cut set Basic Events	Definitions
SFNS2	Failure of both CCP Room return fans to start - common cause.
SFNS	CCP Room return fan fails to start - common cause.
C, D	Pneumatic damper fails.
A180, B180	Damper fails.
B5	Damper, filter, valve, heat exchanger fails.
FNS	Fan fails to start.
FNS2	Failure of two fans to start - common cause.
FNS3	Failure of three fans to start - common cause.
B31	Back draft damper fails to open on demand.

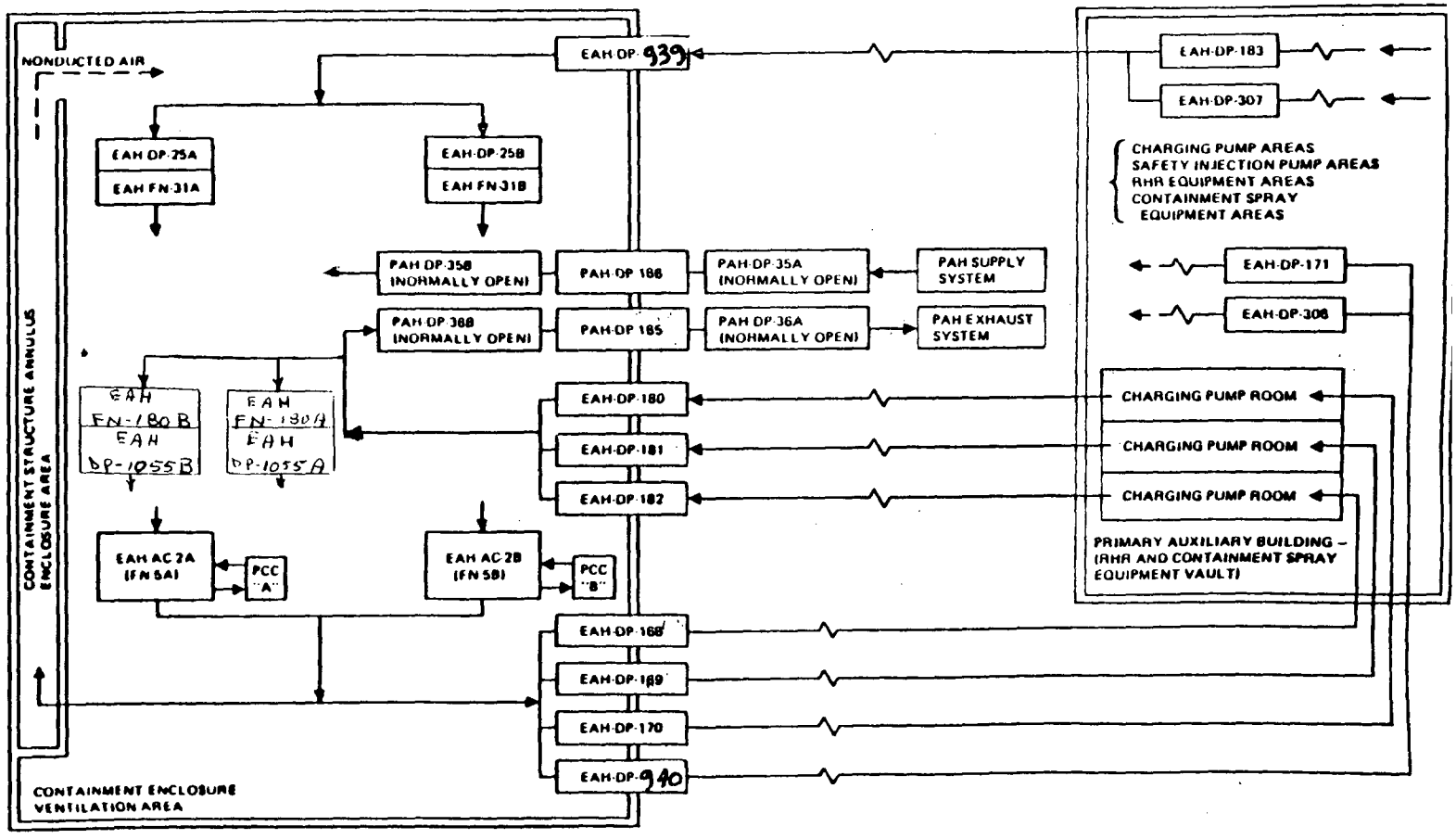

FIGURE E.10-1 CONTAINMENT ENCLOSURE COOLING SYSTEM SIMPLIFIED SCHEMATIC

E.11 Instrument Air System

E.11.1 System Description

Function

The function of the Instrument Air (IA) System is to provide air for pneumatic instruments and controls. It is not a safety system, but, if operating, provides a means of controlling the air-operated valves whose functioning, during or after a transient, could provide a means of gaining additional core and Reactor Coolant System cooling capability.

Configuration

The IA Subsystem (see Figure E.11-1) consists of:

- Three intake filters.
- Three 100% capacity compressors - rated at 350 cfm capacity at 100 psig.
- Three aftercoolers/moisture separators.
- Two air receivers.
- Two instrument air dryers.
- Instruments/controls.
- Piping and valves (including two instrument air headers).

Pneumatic devices in safety class systems are designed to fail in the safest position upon loss of air. However, in a few cases, it is desirable to maintain pneumatic control for modulating valves. in these instances, high pressure gas bottles are provided for backup to the IA System. These include the EFW steam supply valves, ARVs, and the PCC temperature control valves.

Dependencies

Water to cool the compressor cooling jackets and aftercoolers is supplied by the Secondary Component Cooling (SCC) Water System. Cooling for the SCC heat exchangers is provided by the Service Water System (SWS). On loss of off-site power or an S signal or a TA signal, SCC is isolated from SWS cooling. Thus, for these conditions, without manual intervention, the air compressors will overheat and fail. An alternative source of water for

E.11 Instrument Air System (Continued)

the compressor cooling jackets can be provided by the Fire Protection System through a normally closed manual gate valve (SCC-V318).

The IA System is designated in the FSAR as nonsafety-related, but two of the three plant air compressors are connected to the emergency diesel generator Buses A and B, making them available following a loss of off-site power if cooling is available.

System Operation

Normally, the plant air subsystem operates continuously, since compressed air is required during all operating modes. However, the system is designed such that individual parts of the plant can be isolated and shut down as required.

E.11.2 System Model

The IA System does not show up as a single top event, but is considered in several top events. The availability of IA is modeled in each affected system in a conservative manner, as follows:

- Top Event EAH - Containment Enclosure Air Handling System

  Several dampers in the EAH System are normally open, but fail closed. It is conservatively assumed the IA is always available for EAH so that no credit is taken for dampers failing closed on loss of IA.

- Top Event EF - Emergency Feedwater System and Secondary Cooling

  ARV - Secondary cooling includes Atmospheric Relief Valves (ARVs) and Condenser Steam Dump Valves (SDVs). The ARVs are air-operated to open, fail-closed valves, powered by instrument air with a gas accumulator backup. If the gas is used up before the nine hour mission time for secondary

E.11 Instrument Air System (Continued)

cooling, either the operator will manually open the ARVs or the secondary steam safeties will open extending the time for operator action. This operator action is judged to be very reliable and the ARV failure will be dominated by valve hardware failures. Thus, due to the gas accumulator and operator actions, instrument air is assumed to be unnecessary for successful operation of ARVs.

SDV - The SDVs are air-operated to open and are assumed unavailable if IA is not available.

TDP - The valves which admit steam to power the turbine-driven EFW pump are air-operated, fail open valves. On loss of instrument air, MS-V393 and V394 will open, and after a time delay, MS-V395 will open, admitting steam to the turbine.

Feedwater Isolation - The feedwater isolation and bypass valve air-operated fail closed. The loss of instrument air leads to an initiating event - loss of main feedwater - which has been included implicitly in the data analysis of initiating events.

Feedwater isolation also affects the Start-Up Feed Pump (SFP) flow path. The SFP normally injects through the main feed line which isolates on loss of instrument air. The SFP is credited for automatic operation only for loss of main feed pumps. Otherwise, the SFP is handled as a recovery action, which may include repositioning the SFP flow through the EFW header.

E.11 Instrument Air System (Continued)

- Top Event EF - Emergency Feedwater System and Secondary Cooling

  Several containment isolation valves are air-operated, fail closed, on loss of instrument air. In this case, it would be optimistic to assume loss of instrument air given LOSP or S signal. Thus, no credit is given for valves failing closed on loss of IA.

Thus, of all the functions considered, instrument air quantification is used only for SDVs.

Success Criteria

The minimum requirement for successful operation of instrument air is one of three compressors, one of two receivers, and one of two air headers operating for a period of 24 hours.

The system is assumed to be operable prior to the initiating events specified; otherwise, a loss of feedwater event resulting from loss of instrument air would have previously occurred, and this scenario is not considered in this analysis. For either an S signal, T signal, or a LOSP, the system is inoperable due to a loss of cooling to the compressors. Function can only be regained if SWS cooling to the SCC heat exchangers is recovered or fire protection cooling water is aligned.

Analysis Conditions

The IA System was analyzed under the following conditions:

- The system is considered to be operating normally prior to the occurrence of any of the initiating events. It is assumed that one of the compressors is operating, one is in maintenance, and one is in a standby mode. The receivers and dryers are functioning properly.

E.11 Instrument Air System (Continued)

- System functioning in the short term does not require that the air be dry. Therefore, loss of the air dryers does not contribute to system failure unless their malfunctioning obstructs air flow.

- Once air gets to the common supply headers, it is assumed that its path is unobstructed to the equipment it serves due to the very small failure rate attributable to piping and to valves transferring closed.

- Since the SCC System was operating prior to the transient, unless the initiating event is failure or isolation of the Service Water System, SCC continues to provide cooling to the compressors.

- It is very unlikely that the service air header will fail (i.e., pipe rupture), causing depressurization. For this reason, it is assumed that the Service Air System cannot fail, implying that it is immaterial whether Valves SA-V92 and SA-V93 fail to function (close given low pressure signal) since they will not be required to operate.

- No credit is taken for operator actions to recover failed equipment over the time period of this analysis. Therefore, if a loss of SCC occurs, manual Valve SCC-V318 will not be opened to allow fire protection water to serve as a backup.

E.11.3 Results

To be conservative, one of the compressors has been assumed to be undergoing maintenance while one compressor is running and the other is in standby.

Summing the various contributors to system unavailability yields:

$$Q_{SYS} = Q_{Independent\ Hardware} + Q_{Common\ Cause}$$

E.11 Instrument Air System (Continued)

Where:

Q_{SYS} = 3.1E-4

$Q_{Independent\ Hardware}$ = 1.3E-5

$Q_{Common\ Cause}$ = 3.0E-4

Note: **These results are based on input data from the SSPSA, Section 6.0.**

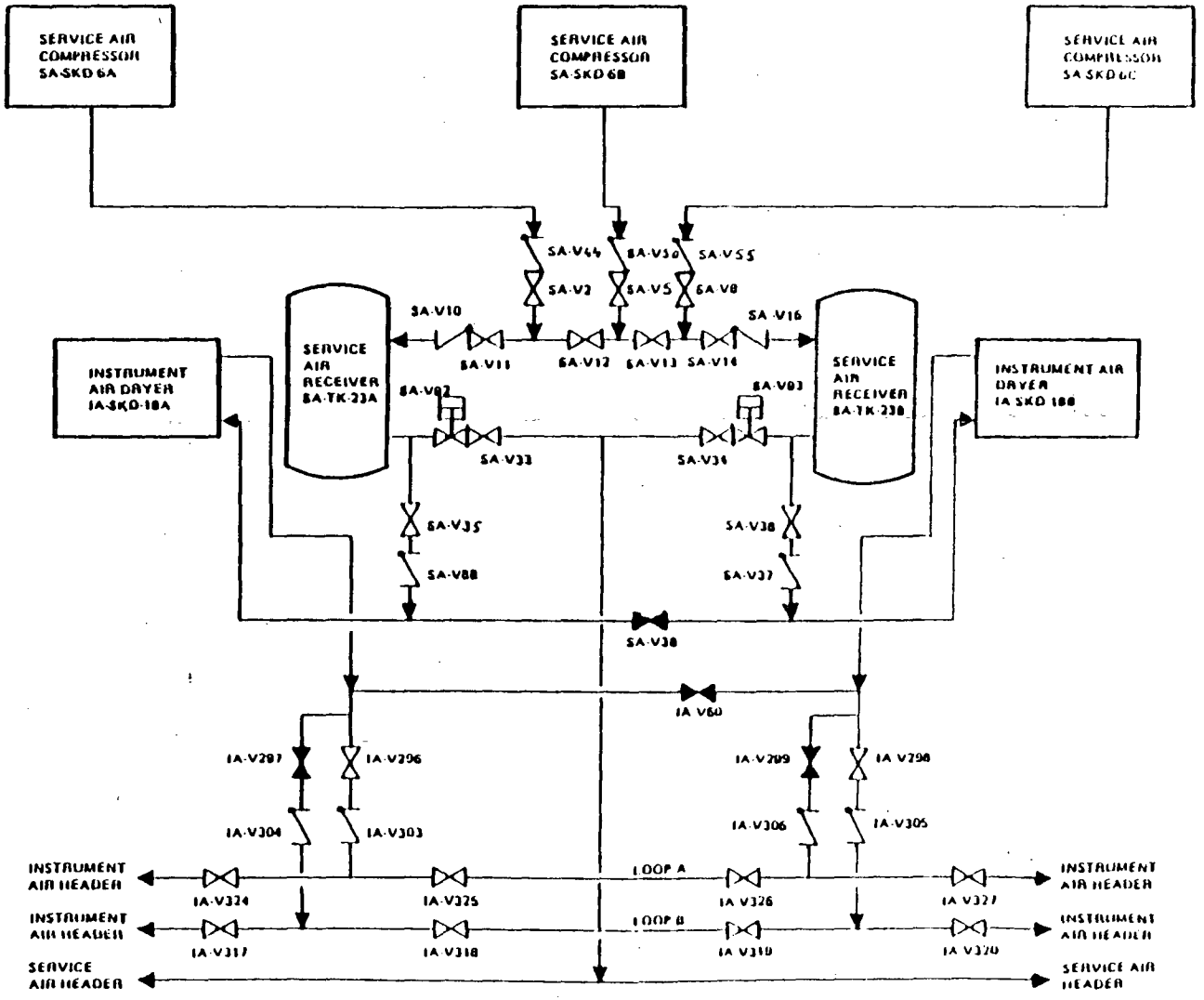

FIGURE E.11-1 INSTRUMENT AIR SYSTEM SCHEMATIC

E.12 Emergency Core Cooling System

E.12.1 System Description

Function

The Emergency Core Cooling System (ECCS) is designed to remove the stored and fission product decay heat from the reactor core following an accident. In addition, the ECCS provides core cooling and shutdown capabilities during the following accident conditions:

- Loss-of-Coolant-Accidents (LOCA).
- Rupture of a control rod drive mechanism causing a Rod Control Cluster Assembly (RCCA) ejection accident.
- Transient events including a steam or feedwater system break.
- A Steam Generator Tube Rupture (SGTR).
- An Anticipated Transient Without Scram (ATWS).

A number of different operation modes are provided by the ECCS over a range of pressures. In terms of injecting water into the RCS, the charging system, safety injection system, the accumulators, the RHR system, and the RWST and containment recirculation sump operate at different levels of pressure and times during the event.

System Configuration

The ECCS consists of three systems which, by using different configurations, perform the functions of high pressure injection, low pressure injection, high pressure recirculation, and low pressure recirculation. The three systems are the SI System (including the accumulators), the CVCS, and the RHR System. The SI System functions solely as part of the ECCS. The CVCS and RHR Systems have normal nonemergency related functions as well. During a plant accident, the CVCS centrifugal charging pumps and RHR pumps assume a role as part of the ECCS. The ECCS simplified flow diagram is presented in Figure E.12-1.

E.12 Emergency Core Cooling System (Continued)

System Dependencies

The SI, RHR, and CVCS Systems are dependent on a number of other systems by support and interface as discussed below:

- Engineered Safety Features Actuation System. This serves to actuate the SI pumps and associated valves. If no signals are available, the associated pumps and valves do not automatically operate.

- Electric Power. The SI System depends on the electric power system for operation, monitoring, and instrumentation and control of its pumps and valves.

- Primary Component Cooling Water. Lube oil cooling for the SI and CVCS pumps (and RHR on recirculation only) depends on the primary component cooling water system which supplies cooling water independently to the two ECCS trains. A failure of the cooling water system is assumed to fail the pumps within five minutes.

- Containment Enclosure Cooling System. Pump operation during HPR depends on successful operation of the containment enclosure cooling and ventilating system for motor cooling. If this system fails, the pumps are assumed to fail at some time longer than six hours.

- RWST/Containment Recirculation Sumps - RHR. The SI System depends on the RWST during the injection phase and on the containment recirculation sumps and the RHR System during the recirculation phase as a source for borated water. Given a failure of a suction source, the pumps will fail within five minutes.

E.12 Emergency Core Cooling System (Continued)

The ECCS water sources (RWST during the injection mode and containment recirculation sumps during the recirculation mode) are shared among the SI, RHR, containment building spray, and the CVCS. The SI System (including the accumulators) and the RHR System share the same piping for injecting into each RCS cold leg. The SI System, excluding the accumulators, and the RHR System share common piping to two of the RCS hot legs.

System Operation

1. Normal Operations

During normal plant operations, most of the ECCS components are in a standby mode. The system is aligned for HPI using the charging and safety injection pumps, and for LPI using the residual heat removal pumps, as described in the previous section.

RCP seal water is provided by either a centrifugal charging pump or a positive displacement pump during normal plant operation. The flow is directed through the seal water injection filters to the RCP seals and it returns to the CVCS suction via the seal water return filter and the seal water heat exchanger. One pump operates continuously to provide this flow.

During plant cooldown, when reactor coolant temperature and pressure are reduced to 350°F and 425 psi, respectively, the RHR shutdown cooling mode takes place using the RHR pumps and heat exchangers. The two normally closed MOVs in series (RH-V22 with RH-V23 and RH-V87 with RH-V88) in each of the lines between the RCS hot legs and the RHR pumps' suctions are opened to allow this mode of cooling.

E.12 Emergency Core Cooling System (Continued)

During refueling, both RHR pumps are utilized to pump borated water from the RWST to the refueling cavity. Following refueling, the RHR pumps are used to drain the refueling cavity to the top of the reactor vessel.

2. Automatic Actuation

ECCS actuation is performed by the Engineered Safety Features Actuation System. The ESFAS is described separately in Section E.8. Injection and RHR recirculation actuation depends only on automatic ESFAS action.

a. Injection

Following an accident and successful ESFAS actuation, all components which are part of the ECCS are aligned (or have their aligned configuration confirmed) to perform their functions. The ESFAS S signal will initiate the following automatic actions:

- All charging (CS-P-2A/CS-P-2B), residual heat removal (RH-P-8A/RH-P-8B), and safety injection pumps (SI-P-6A/SI-P-6B) start.
- The RWST suction valves (LCV-112D/LCV-112E) to the charging pump open.
- The charging pump discharge valves (SI-V138/SI-V139) open.
- Normal charging paths to RCS valves (CS-V142/CS-V143) close.
- Volume control tank outlet valves (LCV-112B/LCV-112C) close.

E.12 Emergency Core Cooling System (Continued)

- Charging pump miniflow valves (CS-V196/CS-V197) close.
- A T signal opens the primary component cooling water RHR heat exchanger discharge valves enabling cooling of the recirculated water.
- The accumulator isolation valves (SI-V3/SI-V17/SI-V32/SI-V47) receive a confirming open signal.
- RWST valves to the RHR (CBS-V2/CBS-V5) System receive a confirming open signal.

The RCS pressure will decrease and as a result of these actions and the prior alignment of the CVCS, SI, and RHR Systems, the CVCS pumps will inject the boric acid solution of the RWST into the reactor vessel via the RCS cold legs. As the pressure decreases, the RWST continues to supply borated water. When the pressure decreases below the SI pump shutoff pressure, the SI pumps will start injecting borated water from the RWST into the RCS cold legs.

When the RCS pressure decreases below the discharge pressure of the pressurized accumulators, their contents discharge through the open MOVs to the RCS cold legs.

While the RCS pressure is decreasing but still higher than RHR pump shutoff pressure, flow is circulated through the RHR pump's miniflow lines, preventing pump overheating. Once the pressure decreases sufficiently so that the flow through the pumps is above 1,000 gpm, the miniflow isolation valves close and all flow is injected to the RCS cold legs.

E.12 Emergency Core Cooling System (Continued)

b. Recirculation

During recirculation, the SI and CVCS pumps must take suction from the Containment Recirculation Sumps (CRSs) rather than the RWST. Having no direct suction path from the CRSs, the SI and CVCS pumps are aligned in series with the RHR pumps. While transfer of the cooling water source from the RWST to the CRSs is done automatically, the alignment of RHR discharge to SI and CVCS suction is carried out manually.

Transfer of the ECCS from the injection mode to the recirculation mode is initiated automatically in response to coincident RWST low-low level signals and an ESFAS S signal. When the transfer signal is received, the CRSs isolation valves, CBS-V8 and CBS-V14 open. At this point, the CBS pumps and the RHR pumps are taking suction from the CRSs while the SI and CVCS pumps are continuing to draw down the RWST. At this time, the plant procedures call for manual operator actions to complete switchover to containment sump suction.

E.12.2 System Models

The ECCS logic models are developed separately for the different subsystems (or parts thereof) of the Emergency Core Cooling System; namely, the Chemical and Volume Control System (CVCS), the Safety Injection (SI) System, and parts of the Residual Heat Removal (RHR) System, as well as the accumulators. Since the RWST, containment recirculation sumps, and the RHR shutdown heat removal function are utilized in conjunction with the ECCS, their logic models are also developed in this section.

E.12 Emergency Core Cooling System (Continued)

The logic models of the ECCS are used for ECCS quantification. In order to quantify the event trees, the relevant part of each system is included in particular ECCS operating mode quantifications.

Top Events

The failure of the ECCS has been defined according to its functions during various operating modes. These failure state top events are used in a number of event trees as shown in Table E.12-1.

Success Criteria

The success criteria for the various ECCS operation modes are as follows:

The HPI success criterion for a transient or small LOCA during injection phase is one SI or CVCS pump (centrifugal charging pump) delivering water to a least two cold legs for six hours.

The HPI success criterion for a medium LOCA is any two of the four SI and CVCS pumps delivering water to at least two cold legs for two hours.

The HPI success criterion for an ATWS is one of two CVCS pumps delivering water to at least two cold legs for two hours.

The HPR success criterion for a transient or small LOCA is one RHR pump and one CVCS or SI pump delivering cooling water to at least two RCS cold legs for 18 hours. Prior to HPR, during the HPI mode, the RHR pumps were in the miniflow recirculation mode for six hours.

The LPI success criterion for a large LOCA is at least one RHR pump delivering cooling water to at least two cold legs for one hour. Three of the accumulators are required for one hour for Accumulator System success. The other accumulator is assumed to discharge into the ruptured leg and is, therefore, unavailable.

E.12 Emergency Core Cooling System (Continued)

The LPI success criterion for a medium LOCA is at least one RHR pump delivering cooling water to at least two cold legs for two hours.

The LPR success criterion for a large LOCA is at least one RHR pump delivering cooling water to at least two cold legs for 23 hours; for a medium LOCA, at least one RHR pump delivering cooling water to at least two cold legs for 22 hours.

Hot leg recirculation during a large LOCA requires one RHR pump to supply flow to one hot leg for four hours.

The RHR shutdown cooling success criterion for a transient or small LOCA is at least one RHR pump delivering cooling water to at least two cold legs for 24 hours.

Analysis Assumptions

The following boundary conditions apply to all the subsystems:

- The unit is considered to be operating at normal power prior to the occurrence of the initiating events. It is assumed for this analysis that the reciprocating charging pump, CS-P-128, is providing normal reactor coolant pump seal injection flow. All other ECCS pumps are in the standby mode. Valve alignment is assumed to be in the normal plant operating mode, except as modified by plant testing, maintenance, or operator errors.

- Since the system is designed to satisfy the single active failure criteria, two injection paths are assumed to be sufficient to deliver a full rated flow from any one pump. One suction path is considered sufficient to supply two high pressure pumps.

E.12 Emergency Core Cooling System (Continued)

- No credit is given for operator actions to recover failed equipment in this analysis.

- Piping failures were found to be a negligible contributor to system unavailability.

- All valves which could isolate interfacing systems from ECCS are in the ECCS logic models. These include the SI, CVCS, and RHR pumps' primary component cooling water inlet and outlet valves.

- Piping sections and their failures are not explicitly modeled in the logic but are taken into account in the quantification section.

- Primary component cooling water is required for the charging and SI pumps during injection and recirculation and for the RHR pumps during recirculation and minimum flow mode.

- The organization of the event trees is such that questions about the suction sources (RWST and containment recirculation sumps) are asked independently. Only if the appropriate suction source is available are the unavailabilities of the systems considered.

- Automatic valves (MOVs) failing by transferring open is not considered as a failure mode.

- Containment recirculation sump plugging and other failure modes (such as vortexing) are not included explicitly as a failure mode due to their small likelihood of occurring.

- Maintenance on more than one high pressure (SI and CVCS) ECCS pump in the same train is allowed by the Technical Specifications; however, due to the low frequency of maintenance attributable to the pumps, only one pump is considered to be out of service at a time.

E.12 Emergency Core Cooling System (Continued)

The failure logic expressions for the ECCS analysis are derived for the following system operability states (boundary conditions):

BC1. Support systems are available for both trains.

BC2. Electrical power or actuation signal is available to only one train.

BC3. Primary component cooling water is available to only one train.

BC4. Only one containment recirculation sump or one RHR pump is available.

For some system functions, all four operability states are not quantified due to a common impact of two or more states.

E.12.3 Results

The dominant contributors to system unavailability for all boundary conditions are provided below:

LI1 1 4.454E-03 LP Injection - LLOCA - BC1, BC3
LI1 2 4.389E-03 NORMAL
LI1 3 4.164E-03 Hardware
LI1 4 2.254E-04 Common Cause
LI1 5 6.463E-05 MAINTENANCE
LI1 6 0.000E+00 TEST

LI2 1 1.536E-02 LP Injection - LLOCA - BC2
LI2 2 1.033E-02 NORMAL
LI2 3 1.033E-02 Hardware
LI2 4 0.000E+00 Common Cause
LI2 5 5.025E-03 MAINTENANCE
LI2 6 0.000E+00 TEST

LP1 1 1.259E-05 LP Recirc - LLOCA - BC1
LP1 2 1.259E-05 NORMAL
LP1 3 4.839E-06 Hardware
LP1 4 7.749E-06 Common Cause
LP1 5 0.000E+00 MAINTENANCE
LP1 6 0.000E+00 TEST

E.12 Emergency Core Cooling System (Continued)

LP2	1	1.194E-03	LP Recirc - LLOCA - BC2,BC3
LP2	2	1.194E-03	NORMAL
LP2	3	1.194E-03	Hardware
LP2	4	0.000E+00	Common Cause
LP2	5	0.000E+00	MAINTENANCE
LP2	6	0.000E+00	TEST
R51	1	7.390E-07	Hot Leg Recirc - LLOCA - BC1
R51	2	7.390E-07	NORMAL
R51	3	7.390E-07	Hardware
R51	4	0.000E+00	Common Cause
R51	5	0.000E+00	MAINTENANCE
R51	6	0.000E+00	TEST
R52	1	1.114E-06	Hot Leg Recirc - LLOCA - BC2,BC3
R52	2	1.114E-06	NORMAL
R52	3	1.114E-06	Hardware
R52	4	0.000E+00	Common Cause
R52	5	0.000E+00	MAINTENANCE
R52	6	0.000E+00	TEST
HB1	1	3.465E-04	RHR Heat Exchange Cooling - LLOCA - BC1
HB1	2	3.465E-04	NORMAL
HB1	3	3.921E-05	Hardware
HB1	4	3.073E-04	Common Cause
HB1	5	0.000E+00	MAINTENANCE
HB1	6	0.000E+00	TEST
HB2	1	4.363E-03	RHR Heat Exchange Cooling - LLOCA - BC2,BC3
HB2	2	4.363E-03	NORMAL
HB2	3	4.363E-03	Hardware
HB2	4	0.000E+00	Common Cause
HB2	5	0.000E+00	MAINTENANCE
HB2	6	0.000E+00	TEST
H11	1	3.452E-05	HPI - MLOCA - BC1
H11	2	2.436E-05	NORMAL
H11	3	1.384E-05	Hardware
H11	4	1.052E-05	Common Cause
H11	5	1.015E-05	MAINTENANCE
H11	6	1.884E-08	TEST
H12	1	3.093E-02	HPI - MLOCA - BC2
H12	2	2.330E-02	NORMAL
H12	3	2.330E-02	Hardware
H12	4	0.000E+00	Common Cause
H12	5	7.618E-03	MAINTENANCE
H12	6	1.268E-05	TEST

E.12 Emergency Core Cooling System (Continued)

H13	1	1.656E-02	HPI - MLOCA - BC3
H13	2	1.002E-02	NORMAL
H13	3	9.825E-03	Hardware
H13	4	1.993E-04	Common Cause
H13	5	6.528E-03	MAINTENANCE
H13	6	1.268E-05	TEST
MM1	1	5.983E-04	RHRM Recirc - MLOCA - BC1
MM1	2	4.783E-04	NORMAL
MM1	3	2.525E-04	Hardware
MM1	4	2.258E-04	Common Cause
MM1	5	1.200E-04	MAINTENANCE
MM1	6	0.000E+00	TEST
MM2	1	1.785E-02	RHRM Recirc - MLOCA - BC2, BC3
MM2	2	1.283E-02	NORMAL
MM2	3	1.283E-02	Hardware
MM2	4	0.000E+00	Common Cause
MM2	5	5.025E-03	MAINTENANCE
MM2	6	0.000E+00	TEST
H31	1	1.127E-03	HPI - ATWS - BC1
H31	2	1.093E-03	NORMAL
H31	3	6.731E-04	Hardware
H31	4	4.201E-04	Common Cause
H31	5	3.428E-05	MAINTENANCE
H31	6	0.000E+00	TEST
H32	1	2.226E-02	HPI - ATWS - BC2
H32	2	1.792E-02	NORMAL
H32	3	1.792E-02	Hardware
H32	4	0.000E+00	Common Cause
H32	5	4.334E-03	MAINTENANCE
H32	6	0.000E+00	TEST
H33	1	8.078E-03	HPI - ATWS - BC3
H33	2	4.809E-03	NORMAL
H33	3	4.610E-03	Hardware
H33	4	1.993E-04	Common Cause
H33	5	3.268E-03	MAINTENANCE
H33	6	0.000E+00	TEST
H21	1	1.168E-06	HPI - SLOCA + TRANS - BC1
H21	2	1.116E-06	NORMAL
H21	3	4.827E-07	Hardware
H21	4	6.334E-07	Common Cause
H21	5	5.221E-08	MAINTENANCE
H21	6	9.434E-11	TEST

E.12 Emergency Core Cooling System (Continued)

H22 1 1.950E-04 HPI - SLOCA + TRANS - BC2
H22 2 1.127E-04 NORMAL
H22 3 1.127E-04 Hardware
H22 4 0.000E+00 Common Cause
H22 5 8.213E-05 MAINTENANCE
H22 6 2.278E-07 TEST

H23 1 7.334E-05 HPI - SLOCA + TRANS - BC3
H23 2 4.129E-05 NORMAL
H23 3 4.025E-05 Hardware
H23 4 1.040E-06 Common Cause
H23 5 3.200E-05 MAINTENANCE
H23 6 5.902E-08 TEST

T1A 1 1.649E-08 HP Recirc - SLOCA - BC1A (ASSA, SI Pump B Failed)
T1A 2 1.649E-08 NORMAL
T1A 3 5.252E-09 Hardware
T1A 4 1.123E-08 Common Cause
T1A 5 0.000E+00 MAINTENANCE
T1A 6 0.000E+00 TEST

T1B 1 1.649E-08 HP Recirc - SLOCA - BC1B (ASSA, SI Pump A Failed)
T1B 2 1.649E-08 NORMAL
T1B 3 5.252E-09 Hardware
T1B 4 1.123E-08 Common Cause
T1B 5 0.000E+00 MAINTENANCE
T1B 6 0.000E+00 TEST

T1C 1 4.046E-08 HP Recirc - SLOCA - BC1C (ASSA, 1 CVCS Pump Failed)
T1C 2 4.046E-08 NORMAL
T1C 3 2.947E-08 Hardware
T1C 4 1.099E-08 Common Cause
T1C 5 0.000E+00 MAINTENANCE
T1C 6 0.000E+00 TEST

T2A 1 1.244E-06 HP Recirc - SLOCA - BC2A (Train A Emergency Bus or Signal Available)
T2A 2 1.244E-06 NORMAL
T2A 3 1.244E-06 Hardware
T2A 4 0.000E+00 Common Cause
T2A 5 0.000E+00 MAINTENANCE
T2A 6 0.000E+00 TEST

T2B 1 1.242E-06 HP Recirc - SLOCA - BC2B (Train B Emergency Bus or Signal Available)
T2B 2 1.242E-06 NORMAL
T2B 3 1.242E-06 Hardware
T2B 4 0.000E+00 Common Cause
T2B 5 0.000E+00 MAINTENANCE
T2B 6 0.000E+00 TEST

E.12 Emergency Core Cooling System (Continued)

T3A	1	1.242E-06	HP Recirc - SLOCA - BC3A (PCC Train B Unavailable)
T3A	2	1.242E-06	NORMAL
T3A	3	1.242E-06	Hardware
T3A	4	0.000E+00	Common Cause
T3A	5	0.000E+00	MAINTENANCE
T3A	6	0.000E+00	TEST
T3B	1	1.240E-06	HP Recirc - SLOCA - BC3B (PCC Train A Unavailable)
T3B	2	1.240E-06	NORMAL
T3B	3	1.240E-06	Hardware
T3B	4	0.000E+00	Common Cause
T3B	5	0.000E+00	MAINTENANCE
T3B	6	0.000E+00	TEST
TA1	1	3.218E-06	HP Recirc - SLOCA - BC4AA (Sump A or RHR Pump A
TA1	2	3.218E-06	NORMAL unavailable; SI Pump B
TA1	3	3.207E-06	Hardware unavailable)
TA1	4	1.123E-08	Common Cause
TA1	5	0.000E+00	MAINTENANCE
TA1	6	0.000E+00	TEST
TA2	1	1.963E-08	HP Recirc - SLOCA - BC4AB (Sump A or RHR Pump A
TA2	2	1.963E-08	NORMAL unavailable; SI Pump A
TA2	3	8.398E-09	Hardware unavailable)
TA2	4	1.123E-08	Common Cause
TA2	5	0.000E+00	MAINTENANCE
TA2	6	0.000E+00	TEST
TA3	1	4.263E-08	HP Recirc - SLOCA - BC4AC (Sump A or RHR Pump A
TA3	2	4.263E-08	NORMAL unavailable; 1 CVCS Pump
TA3	3	3.163E-08	Hardware unavailable)
TA3	4	1.099E-08	Common Cause
TA3	5	0.000E+00	MAINTENANCE
TA3	6	0.000E+00	TEST
TB1	1	1.650E-08	HP Recirc - SLOCA - BC4BA (Sump B or RHR Pump B
TB1	2	1.650E-08	NORMAL unavailable; SI Pump B
TB1	3	5.255E-09	Hardware unavailable)
TB1	4	1.124E-08	Common Cause
TB1	5	0.000E+00	MAINTENANCE
TB1	6	0.000E+00	TEST
TB2	1	1.649E-08	HP Recirc - SLOCA - BC4BB (Sump B or RHR Pump B
TB2	2	1.649E-08	NORMAL unavailable; SI Pump A
TB2	3	5.255E-09	Hardware unavailable)
TB2	4	1.123E-08	Common Cause
TB2	5	0.000E+00	MAINTENANCE
TB2	6	0.000E+00	TEST

E.12 Emergency Core Cooling System (Continued)

TB3	1	4.143E-08	HP Recirc - SLOCA - BC4BC (Sump B or RHR Pump B	
TB3	2	4.143E-08	NORMAL	unavailable; 1 CVCS Pump
TB3	3	3.045E-08	Hardware	unavailable)
TB3	4	1.098E-08	Common Cause	
TB3	5	0.000E+00	MAINTENANCE	
TB3	6	0.000E+00	TEST	
SM1	1	6.084E-04	RHRM Recirc - SLOCA - BC1	
SM1	2	4.860E-04	NORMAL	
SM1	3	2.589E-04	Hardware	
SM1	4	2.271E-04	Common Cause	
SM1	5	1.224E-04	MAINTENANCE	
SM1	6	0.000E+00	TEST	
SM2	1	1.810E-02	RHRM Recirc - SLOCA - BC2,BC3	
SM2	2	1.308E-02	NORMAL	
SM2	3	1.308E-02	Hardware	
SM2	4	0.000E+00	Common Cause	
SM2	5	5.025E-03	MAINTENANCE	
SM2	6	0.000E+00	TEST	
LR1	1	5.167E-04	Long Term Cooling - SLOCA - BC1	
LR1	2	5.167E-04	NORMAL	
LR1	3	2.184E-04	Hardware	
LR1	4	2.983E-04	Common Cause	
LR1	5	0.000E+00	MAINTENANCE	
LR1	6	0.000E+00	TEST	
LR3	1	1.119E-02	Long Term Cooling - SLOCA - BC3	
LR3	2	1.119E-02	NORMAL	
LR3	3	1.119E-02	Hardware	
LR3	4	0.000E+00	Common Cause	
LR3	5	0.000E+00	MAINTENANCE	
LR3	6	0.000E+00	TEST	
SP1	1	8.219E-06	RHR Pumps for HPR - SLOCA - BC1	
SP1	2	8.219E-06	NORMAL	
SP1	3	2.154E-06	Hardware	
SP1	4	6.065E-06	Common Cause	
SP1	5	0.000E+00	MAINTENANCE	
SP1	6	0.000E+00	TEST	
SP2	1	1.121E-03	RHR Pumps for HPR - SLOCA - BC2,BC3	
SP2	2	1.121E-03	NORMAL	
SP2	3	1.121E-03	Hardware	
SP2	4	0.000E+00	Common Cause	
SP2	5	0.000E+00	MAINTENANCE	
SP2	6	0.000E+00	TEST	

E.12 Emergency Core Cooling System (Continued)

SR1 1 7.981E-06 RHR Pumps for LPR - SLOCA - BC1
SR1 2 7.981E-06 NORMAL
SR1 3 1.916E-06 Hardware
SR1 4 6.065E-06 Common Cause
SR1 5 0.000E+00 MAINTENANCE
SR1 6 0.000E+00 TEST

SR2 1 1.011E-03 RHR Pumps for LPR - SLOCA - BC2,BC3
SR2 2 1.011E-03 NORMAL
SR2 3 1.011E-03 Hardware
SR2 4 0.000E+00 Common Cause
SR2 5 0.000E+00 MAINTENANCE
SR2 6 0.000E+00 TEST

RWLL 1 2.762E-08 RWST (1 Hour) - LLOCA
RWML 1 5.524E-08 RWST (2 Hour) - MLOCA
RWSL 1 1.657E-07 RWST (6 Hour) - SLOCA
RALL 1 1.026E-04 RWST Outlet Valve V2 - LLOCA
RBLL 1 1.026E-04 RWST Outlet Valve V5 - LLOCA
RAML 1 1.027E-04 RWST Outlet Valve V2 - MLOCA
RBML 1 1.027E-04 RWST Outlet Valve V5 - MLOCA
RASL 1 1.030E-04 RWST Outlet Valve V2 - SLOCA
RBSL 1 1.030E-04 RWST Outlet Valve V5 - SLOCA

V11 1 3.515E-04 Containment Recirc Sumps - LLOCA - BC1,BC3
V11 2 3.073E-04 Common Cause
V12 1 4.878E-03 Containment Recirc Sumps - LLOCA - BC2
V21 1 3.460E-04 Containment Recirc Sumps - SLOCA - BC1,BC3
V21 2 3.073E-04 Common Cause
V22 1 4.317E-03 Containment Recirc Sumps - SLOCA - BC2

TABLE E.12-1

ECCS Interface With Seabrook Event Trees

System Function	RWST	HPI (SI-CVCS)	RWST		RHR Miniflow		RHR Shutdown Cooling	LPI (RHR)		Recirculation Switchover		LPR-Cold Leg (RHR)				HPR (RHR-SI-CVCS)	LPR Hot Leg	
"Train"/"System"																		
Top Event Identification*	RW	HP	RA	RB	L1	L2	LR	LA	LB	SA	SB	LC	HA	LD	HB	RC	HE	HS
Large LOCA A - Early B - Late	X		X	X				X	X	X	X	X	X	X	X		X	X
Medium LOCA	X	X(HP1)	X	X	X	X												
Small LOCA	X	X(HP2)	X	X	X	X	X											
Steam Generator Tube Rupture	X	X(HP2)	X	X	X	X	X											
Steam Line Break	X	X(HP2)	X	X	X	X												
Transient	X	X(HP2)	X	X	X	X	X											
Long-Term Response										X	X	X		X		X		
Anticipated Transient Without Scram	X	X(HP3)	X	X	X	X	X											

*These represent general classes of initiating events.

1986 Update

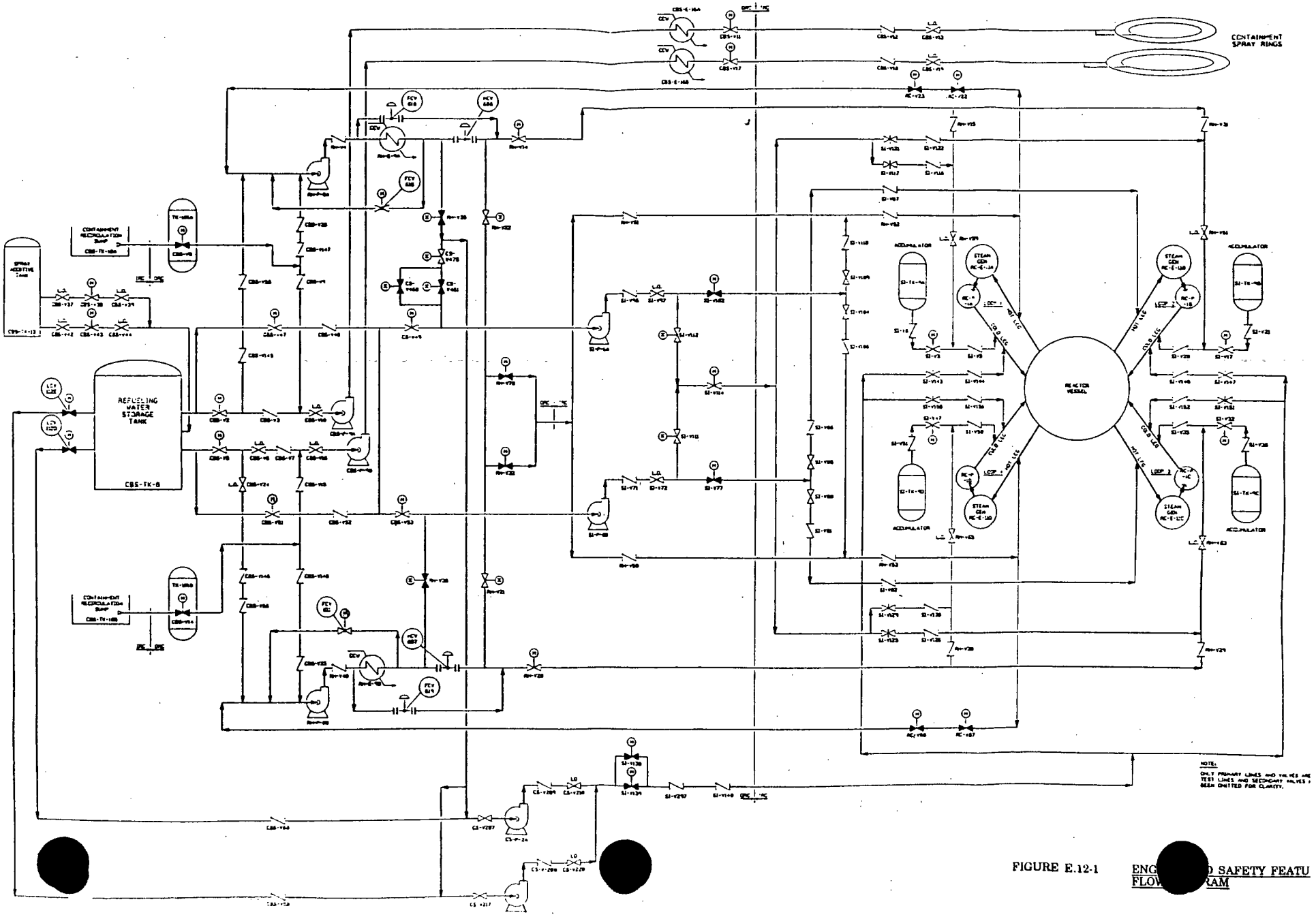

FIGURE E.12-1 ENG[redacted]D SAFETY FEATU
FLOW[redacted]RAM

E.13 Reactor Coolant Pressure Relief System (RCPRS)

E.13.1 System Description

Function

The function of the RCPRS is to provide primary system pressure relief for overpressure transients and cooling in the "feed and bleed" mode for primary system cooldown through the operation of Power-Operated Relief Valves (PORV) and safety valves.

Configuration

The RCPRS (see Figure E.13-1) consists of a pressurizer which is connected to the Reactor Coolant System , three pressurizer safety valves and two pressurizer relief valves, two PORV block valves, a pressurizer relief Tank, and interconnecting piping and instrumentation necessary for operational control. The relief valves (PORV) and block valves are configured such that each set of valves (one PORV and one block valve) is powered from its respective power train.

Dependencies

The RCPRS depends directly on DC power to open the PORVs and indirectly on AC Power (Buses E5/E6) to open the block valves if closed. The safety valves are self actuating and require no support systems.

Operation

During normal plant operation, the relief and safety valves remain closed assuring RCS integrity. In the event of an overpressurization scenario, the PORVs and safety valves provide steam relief to the pressurizer relief tank where the steam is condensed and cooled by mixing with water. During plant cooldown, the PORV setpoint automatically follows RCS temperature below 305°F to protect the system from low temperature

E.13 Reactor Coolant Pressure Relief System (RCPRS) (Continued)

overpressurization events. Also, the PORV block valves receive an auto open signal for those cases where they may have been closed. The PORVs may be manually opened for "feed and bleed" cooling or in those instances where an auto open signal failed to actuate the PORVs. Discharge piping leakage is identified via temperature and acoustic emissions monitoring methods. Alarms are provided in the control room to alert operators of leakage through these valves.

E.13.2 System Model

Top Event

The RCPRS portion of the 'feed and bleed cooling' function is included in Top Event OR where EFW has failed in the following event trees: general transient, small LOCA, steam line break inside/outside containment,and steam generator tube rupture.

The ATWS event tree includes three top events with reactor coolant relief functions:

PS - pressure relief out the safeties and PORVs.

P2 - safeties and PORVs reseating after opening to relieve pressure surge.

PR - PORVs manually opened to perform chemical shutdown or, if EFW has failed and no LOCA exists, to perform 'feed and bleed cooling'.

Success Criteria

Success criteria are summarized as follows:

E.13 Reactor Coolant Pressure Relief System (RCPRS) (Continued)

Event	System Success Criteria
Feed and Bleed Cooling	Two-out-of-two PORVs need to open on demand.
Chemical Shutdown in ATWS	One-out-of-two PORVs need to open on demand.
ATWS	Three-out-of-three safety valves need to open on demand for all RCS overpressurization sequences. The number of PORVs required (none, one, or two), in addition to the safety valves, depends on the time in core life in which the ATWS occurs.
After ATWS	Three-out-of-three safety valves and two-out-of-two PORVs or block valves need to reset on demand.

Analysis Conditions

The following boundary conditions and assumptions are common to the analysis of the RCPR system under all the scenarios evaluated in the plant event tree models.

- The unit is considered to be at normal power operation prior to the occurrence of any initiating event.
- The pressurizer and its relief tank are not included in this analysis.
- No credit is taken for operator actions to recover failed valves or to provide cooling via other means in this analysis.

E.13 Reactor Coolant Pressure Relief System (RCPRS) (Continued)

E.13.3 Results

The dominant contributors to system unavailability for all support states are provided below:

PS1	1.284E-03	RC Pressure Relief - Severe ATWS,1/2 PORV,3/3 Safety
PS1	3.177E-04	Safety Valve Fail To Open on Demand
PS1	4.912E-03	PORV OR Block Valve
PS1	3.938E-03	PORV Fail To Open on Demand - Non Common Cause
PS1	4.367E-04	Block Valve Fail to Open on Demand
PS1	2.202E-06	Block Valve Transfer Closed or Open During Oper.
PS1	5.350E-04	BValve Undiscovered Failure Prior to Initiation
PS1	5.832E+03	18 Month Time*(1-f)/2
PS1	2.973E-04	PORV Common Cause Contribution
PS2	6.162E-03	RC Pressure Relief - Severe ATWS, Single Train AC/DC
PS3	9.822E-04	RC Pressure Relief - ATWS, 3/3 Safety Valves
P21	5.853E-02	Safety and Relief Valves Reseat - ATWS
PR1	1.042E-02	PORV in Feed and Bleed, 2/2 PORVs
PR1	5.209E-03	PORV OR Block Valve
PR1	4.235E-03	PORV Fail to Open on Demand
PR1	4.367E-04	Block Valve Fail to Open on Demand
PR1	2.202E-06	Block Valve Transfer Closed or Open During Oper.
PR1	5.350E-04	BValve Undiscovered Failure Prior to Initiation
PR2	3.306E-04	PORV in Chemical Shutdown - ATWS, 1/2 PORV
PR2	4.912E-03	PORV OR Block Valve
PR2	2.973E-04	PORV Common Cause Contribution
PR3	5.209E-03	PORV in Chemical Shutdown - ATWS, Single Train AC/DC

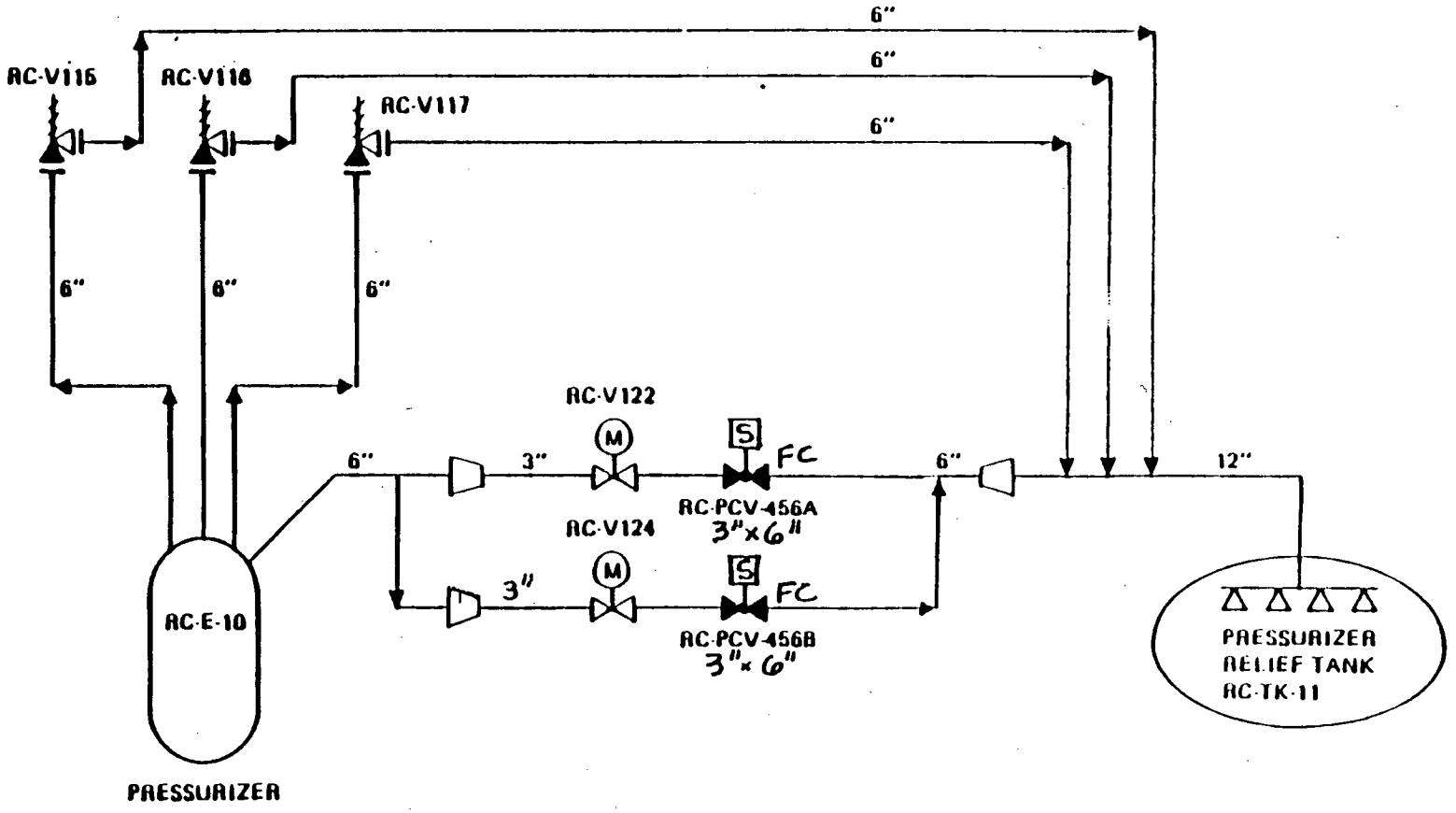

FIGURE E.13-1 REACTOR COOLANT PRESSURE RELIEF SIMPLIFIED VALVE SCHEMATIC

E.14 Emergency Feedwater (EFW) System

E.14.1 System Description

Function

The EFW System supplies water to the steam generators in order to remove heat from the reactor coolant system during emergencies when the main feedwater system is unavailable. In addition, the EFW System is capable of reducing RCS pressure and temperature to enable operation of the RHR System as an alternate means of decay heat removal.

Configuration

The EFW System (see Figure E.14-1) consists of two EFW pumps (one motor-driven and the other turbine- driven), a start-up feed pump (SUFP), and a condensate storage tank (CST). The two EFW pumps are full-sized pumps which take suction from separate lines from the CST. A common EFW pump recirculation line discharges back to the CST. The EFW pumps feed a common discharge header, which in turn supplies the four steam generators. Each steam generator supply line contains two normally open flow control valves, which provide isolation in the event of a pipe break. Each EFW feed line is connected to a main feed line downstream of the feedwater isolation valve.

The SUFP provides additional emergency feedwater capability. The SUFP takes suction from the CST, the condenser hotwell, or the condensate cleaning system and discharges to the main feedwater header. For conditions in which main feedwater is isolated, the operator can manually align the SUFP to the EFW header.

Dependencies

Environmental control of the EFW pumphouse is maintained by the Emergency Feedwater Pumphouse Ventilation System. Electric power for the motor-driven EFW pump is supplied by 4.16 kV emergency ac Bus E6 (Train B), which is supplied by Diesel Generator 1B upon a loss of off-site power. Each pair of MOVs in the EFW supply lines is powered by 460 V Motor Control Centers (MCC); one by Train A and one by Train B.

E.14 Emergency Feedwater (EFW) System (Continued)

Control power for the steam turbine admission valves is supplied from two sources, so that one valve receives power from dc Bus 11A and the other receives power from dc Bus 11B. The steam admission valves are air-operated to close and fail open upon loss of air. Actuation signals are provided by the Solid State Protection System (SSPS) and ATWS Mitigating System Actuation Circuitry (AMSAC). The Main Steam System provides the steam necessary for turbine-driven pump operation.

Electrical power for the SUFP is supplied normally by 4.160 kV Bus 4. Via operator actions, the SUFP can be powered from emergency Bus E5. Air cooling for the pump is provided by the Turbine Building Ventilation System.

Operation

During normal power operation, the EFW System is in standby with the water flow valves aligned for operation. Upon receiving an EFW actuation signal, valves open to admit steam to the turbine-driven EFW pump, and the motor-driven pump starts. Both pumps automatically supply water to all four steam generators through the flow control valves.

The start-up feed pump normally discharges to the main feedwater header. The pump automatically starts if the main feed pumps trip, provided that a safety injection or steam generator high-high level signal is not sensed.

E.14.2 System Model

The system model includes the two pumps in the EFW System and the start-up feed pump.

System unavailability is quantified using the block diagram method and RISKMAN software.

E.14 Emergency Feedwater (EFW) System (Continued)

Top Event

The results from the EFW System analysis are used as Top Event EF and Top Event FR in the frontline event trees (e.g., general transient, ATWS, steam line break, etc.), except for the large LOCA event tree. Top Event EF only quantifies automatic operation of the EFW System; this includes automatic starting of the start-up feed pump for the loss of main feedwater initiating event. Top Event FR models recovery of the turbine-driven EFW pump and/or start-up feed pump.

Top Event EF also includes secondary steam release, which is analyzed in the Main Steam System analysis (see Section E.15).

Success Criteria

Success of the EFW System has been defined in this analysis as success of either one of the two EFW pumps or success of the start-up feed pump (used only for the loss of main feedwater initiating event) to provide at least 470 gpm of water to two out of four operable steam generators for a period of nine hours following transient initiation. The EFW System supplies sufficient water to cool the RCS, allowing operation of the RHR System within nine hours. For ATWS initiators, the success criteria is changed to feeding four-out-of-four steam generators.

Analysis Conditions

The EFW System was analyzed under the following conditions:

- Human intervention is permitted in the case where, during testing of an EFW pump, a system actuation occurs and the testing procedure explicitly directs the operators to realign the system to its normal flow path.

- Failure of the Emergency Feedwater Pumphouse Heating and Ventilation System is not considered to impair EFW pump operation within the considered time frame.

E.14 Emergency Feedwater (EFW) System (Continued)

- A mission time of nine hours is used which allows sufficient time to cool down the RCS to allow RHR shutdown cooling.

- Reverse leakage through all check valves in idle flow paths is not significant enough to affect water flow.

- Since the Turbine Building Ventilation System is operating prior to an event requiring the start-up feed pump, it is assumed that it remains operable. In addition, due to the large size of the turbine building, it is assumed that sufficient time exists for operator action, if necessary, on failure of the Ventilation System.

E.14.3 Results

The results of the EFW System quantification are shown below:

EF1	2.747E-04	EFW1 - MDP and TDP
EF1	1.952E-04	NORMAL Configuration
EF1	1.817E-04	Hardware Total
EF1	1.352E-05	Common Cause Total
EF1	7.696E-05	MAINTENANCE Configuration
EF1	1.174E-06	TEST - OP fails to realign for demand during test
EF1	1.303E-06	HUMAN- OP fails to realign system following test
EF2	4.755E-02	EFW2 - Turbine Driven Pump Only
EF2	4.305E-02	NORMAL Configuration - Hardware Only
EF2	4.297E-02	TD Pump Start and Run
EF2	8.101E-05	Valves
EF2	4.444E-03	MAINTENANCE Configuration
EF2	2.450E-05	TEST - OP fails to realign for demand during test
EF2	2.729E-05	HUMAN- OP fails to realign system following test
EF3	5.288E-03	EFW3 - Motor Driven Feed Pump Only
EF3	4.006E-03	NORMAL Configuration - Hardware Only
EF3	3.925E-03	MD Pump Start and Run
EF3	8.101E-05	Valves
EF3	1.230E-03	MAINTENANCE Configuration
EF3	2.450E-05	TEST - OP fails to realign for demand during test
EF3	2.729E-05	HUMAN- OP fails to realign system following test

E.14 Emergency Feedwater (EFW) System (Continued)

EF4	5.668E-03	EFW4 - Start-Up Feed Pump - Auto Start
EF4	4.386E-03	NORMAL Configuration - Hardware Only
EF4	3.527E-03	Start-Up Feed Pump Start and Run
EF4	8.598E-04	Valves
EF4	1.230E-03	MAINTENANCE Configuration
EF4	2.450E-05	TEST Configuration
EF4	2.729E-05	HUMAN- OP fails to realign system following test
EF5	5.551E-03	EFW1 - Feeding All 4 SGs - ATWS (TDP + MDP)
EF6	5.320E-02	EFW2 - Feeding All 4 SGs - ATWS (TDP only)
EF7	1.094E-02	EFW3 - Feeding All 4 SGs - ATWS (MDP only)
EFRTDP	5.534E-01	TDP Recovery Fraction
EFRTDP	2.636E-02	TDP Recovery
EFRSFP	8.616E-03	SFP Recovery (Manual Actuation)
EFRSFP	3.000E-03	SFP Operator Action

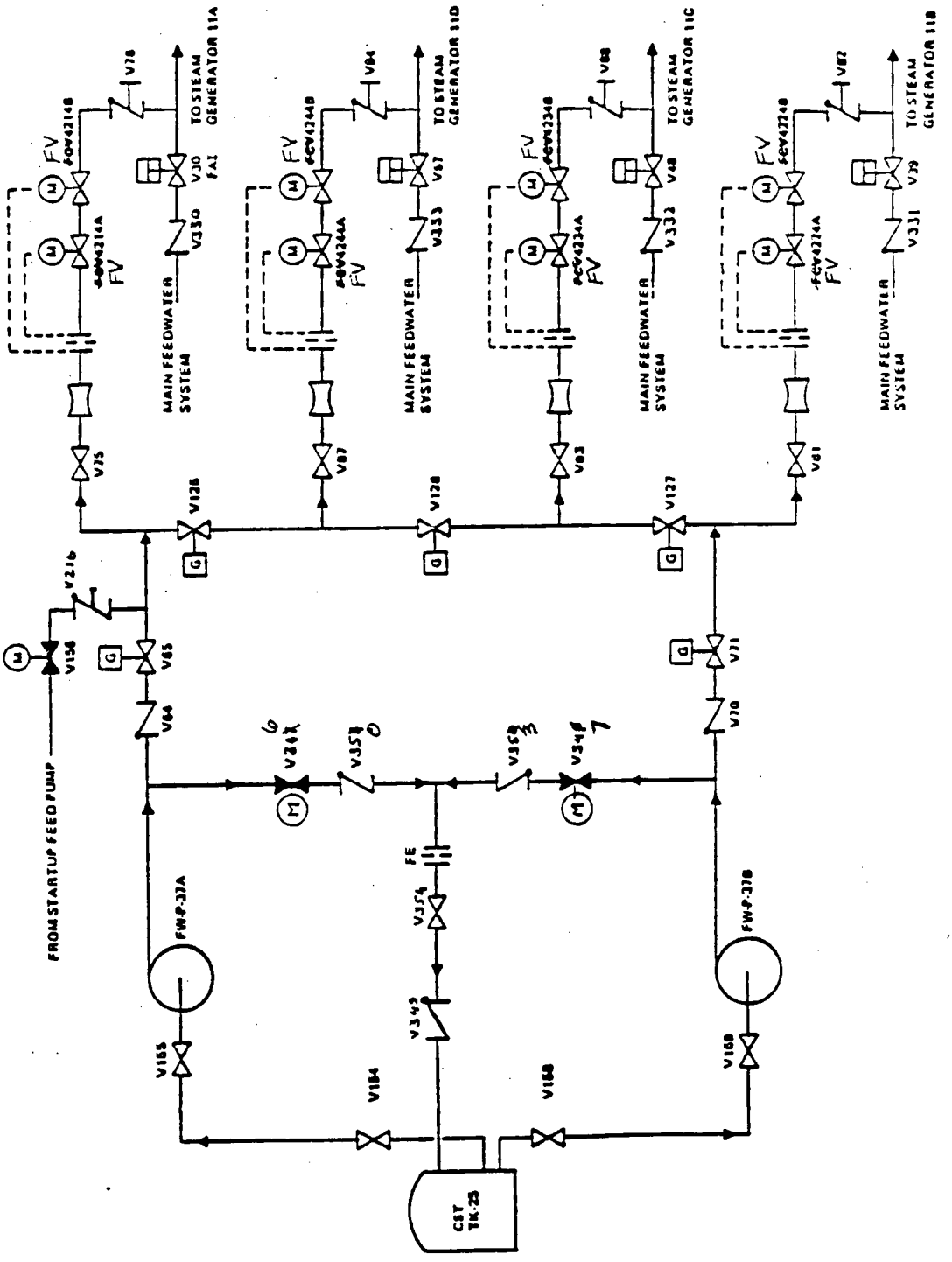

FIGURE E.14-1 SIMPLIFIED SKETCH OF THE EMERGENCY FEEDWATER SYSTEM (Sheet 1 of 3)

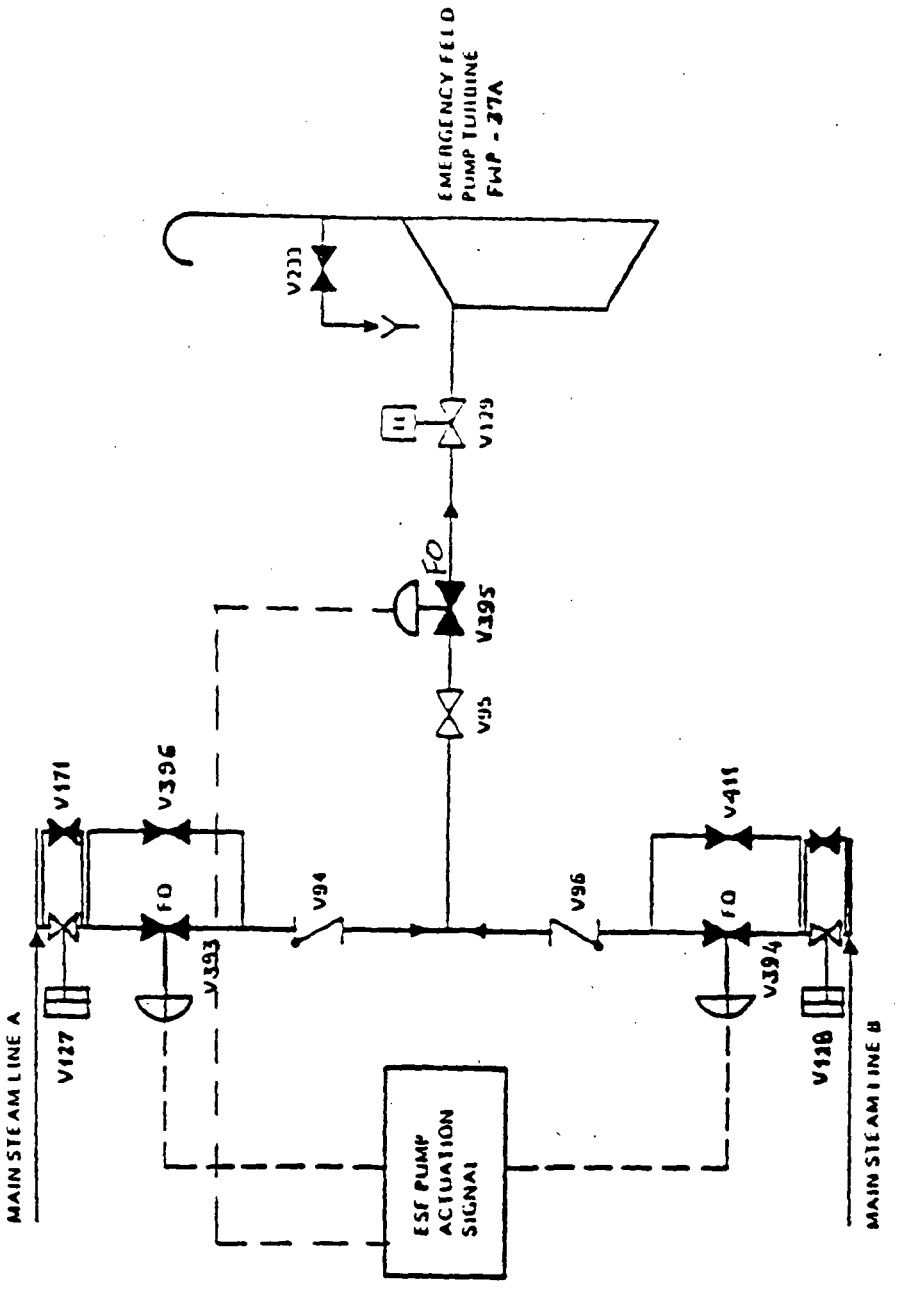

FIGURE E.14-1 SIMPLIFIED SKETCH OF THE EMERGENCY FEEDWATER SYSTEM (Sheet 2 of 3)

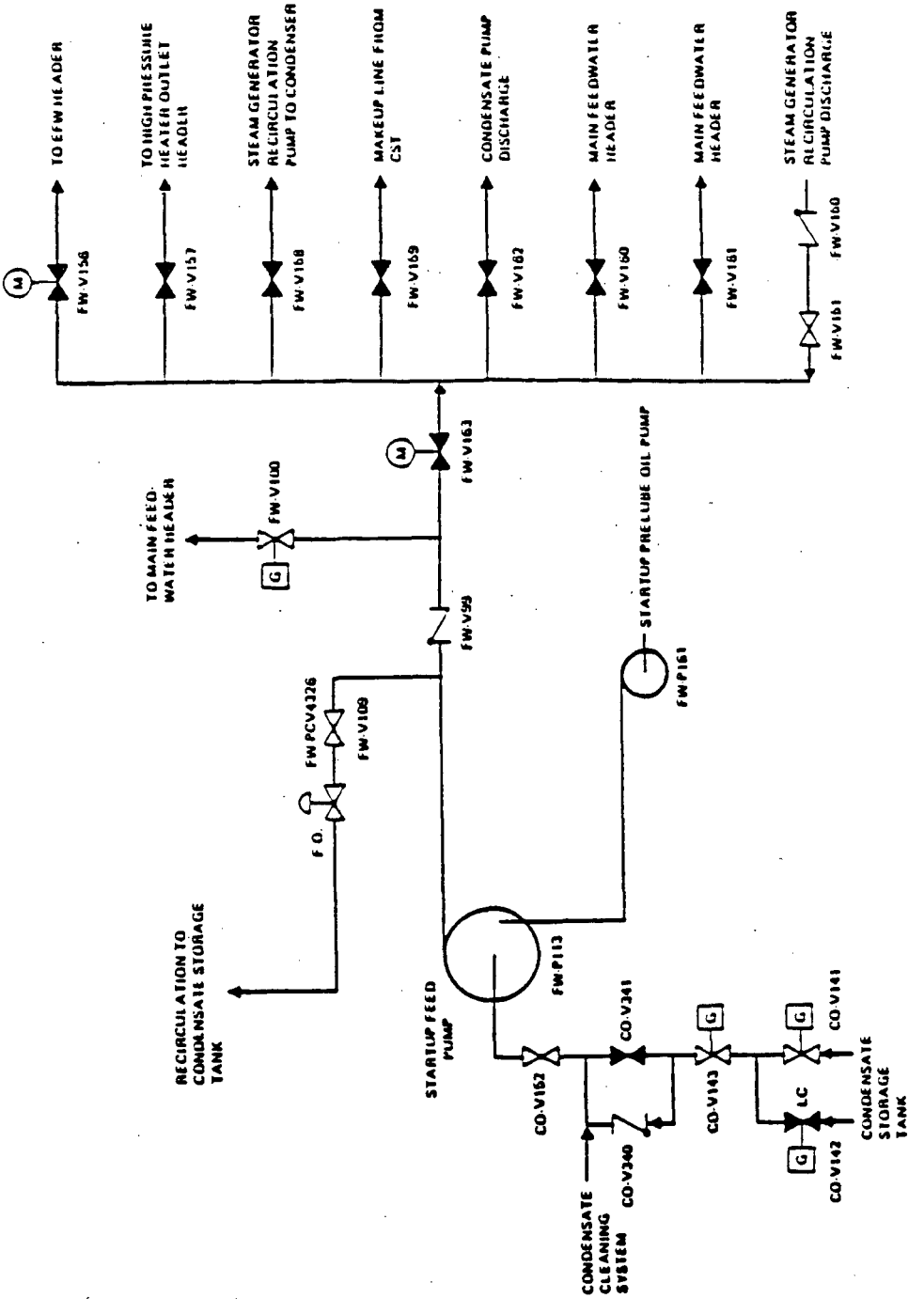

FIGURE E.14-1 SIMPLIFIED SKETCH OF THE EMERGENCY FEEDWATER SYSTEM (Sheet 3 of 3)

E.15 Main Steam System

E.15.1 System Description

Function

The functions of the Main Steam System, with regard to accident mitigation, are to provide for adequate heat removal, to prevent excessive heat removal (i.e., overcooling transients), and to provide overpressure protection of the main steam piping.

Configuration

The Main Steam System consists of the following major components (see Figure E.15-1):

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

Dependencies

The Main Steam Isolation Valves (MSIVs) depend on the Solid State Protection System (SSPS) and the Engineered Safety Features Actuation System (ESFAS) for an automatic isolation signal.

The Atmospheric Relief Valves (ARVs) depend on 120 V ac vital instrument power and instrument air for operation. The ARVs are supplied with backup high pressure nitrogen gas bottles. The ARVs can be operated locally without power.

Automatic operation of the Steam Dump Valves (SDVs) requires condenser vacuum to be available, at least one circulating water pump operating, availability of instrument air, and control signals from primary temperature or steam line pressure and reactor trip. The 120 V AC Instrument System provides the variable signals for valve modulation. With loss of off-site power, the SDVs are assumed to be unavailable due to the loss of the circulating water pumps and instrument air.

E.15 Main Steam System (Continued)

Operation

During normal plant operation, the main steam system automatically transports the steam generated in the steam generators to the turbine. Normally on a turbine trip, the SDVs bypass steam from the steam generators directly to the main condenser. If the SDVs or condenser is unavailable, steam relief is provided through the ARVs or steam generator safety valves. In the event of an overcooling transient, such as a steam line break, the MSIVs automatically isolate (either on low steam line pressure or high rate of change of steam line pressure). MSIV isolation also occurs on Hi-2 containment pressure.

E.15.2 System Model

The model for the Main Steam System includes the atmospheric relief valves, condenser steam dump valves, and the steam generator safety valves. The system analysis also models the turbine trip function of the turbine stop valves and/or turbine control valves and the main steam isolation function of the MSIVs.

The model is quantified using the block diagram method and RISKMAN software to generate results.

Top Event

The results from the Main Steam System quantification of secondary cooling are combined with the results from the EFW System quantification to produce Top Events EF and FR in the early response event trees. Top event EF models automatic operation of the EFW pump(s) and secondary steam relief. Top Event FR models recovery of the turbine-driven and/or start-up feed pump.

Results from main steam line isolation and turbine trip quantification are combined in Top Event TT in the early response trees. Other system models are for Top Events IV, SO, and SS used in he SGTR event tree quantification. Top Event SV is used in the ATWS event tree.

E.15 Main Steam System (Continued)

Success Criteria

The success criteria for the Main Steam System depends on the initiating event being analyzed. The following criteria have been used:

- For secondary cooling, at least two-out-of-four ARVs or six-out-of-twelve SDVs must open on demand.
- For the main steam line isolation function, at least three-out-of-four MSIVs must close on demand.
- For steam generator isolation during a SGTR event, the MSIV on the affected steam generator must close in response to the isolation signal.
- For faulted steam generator isolation for SGTR events, all steam dump valves and three-out-of-three MSIVs on the unaffected steam generators must close.
- For SGTR events, all safety valves are required to open and then reseat.
- For successful turbine trip, all turbine stop valves or all turbine control valves must close on receipt of a trip signal.

Analysis Conditions

The Main Steam System is analyzed under the following assumptions:

- The unit is at normal full power operation prior to the initiating event.

E.15 Main Steam System (Continued)

- Some operator actions have been included regarding operation of the ARVs. For example, on loss of instrument power and depletion of the back-up gas bottles, the ARVs must be operated manually (and locally) either by use of a pneumatic control station or by handwheel.

- In addition, operator action may be necessary in the long term to cool down using the steam dump valves.

E.15.3 Results

ARVSDV	3.243E-08	Atmos Relief Valves & Cond Steam Dump Valves
ARVSDV	3.070E-04	- Steam Dump System Failure Total
ARVSDV	1.288E-13	- Steam Dump Valve Cut Sets (7 of 12 AOVs fail)
ARVSDV	3.070E-04	- Steam Dump Control System
ARVSDV	3.070E-04	- Instrument Air System
ARVSDV	8.686E-08	- ARV Hardware
ARVSDV	1.055E-04	- ARV Common Cause
ARVV	1.056E-04	Atmos Relief Valves Only
ARVV	8.686E-08	- ARV Open on Demand
ARVV	1.055E-04	- ARV Common Cause
MS1	1.323E-04	MSIV Isolation - SLBOC or Turbine Trip Failure
MS1	2.508E-05	- Hardware
MS1	1.072E-04	- Common Cause
IV1	1.523E-03	MSIV and Bypass Isolated - SL Tree - SGTR
SS1	5.361E-03	Steaming SG Isolated - SL Tree - SGTR
SO1	9.578E-03	Safety Valves Open/Close,Steam Relief - SL Tree,SGTR
SO2	2.914E-01	Safety Valves Open/Close,Water Relief - SL Tree,SGTR
TT1	4.456E-06	Turbine Trip for Non - TT Failure Initiating Events
TT1	4.059E-06	- Trip Signal from Control System

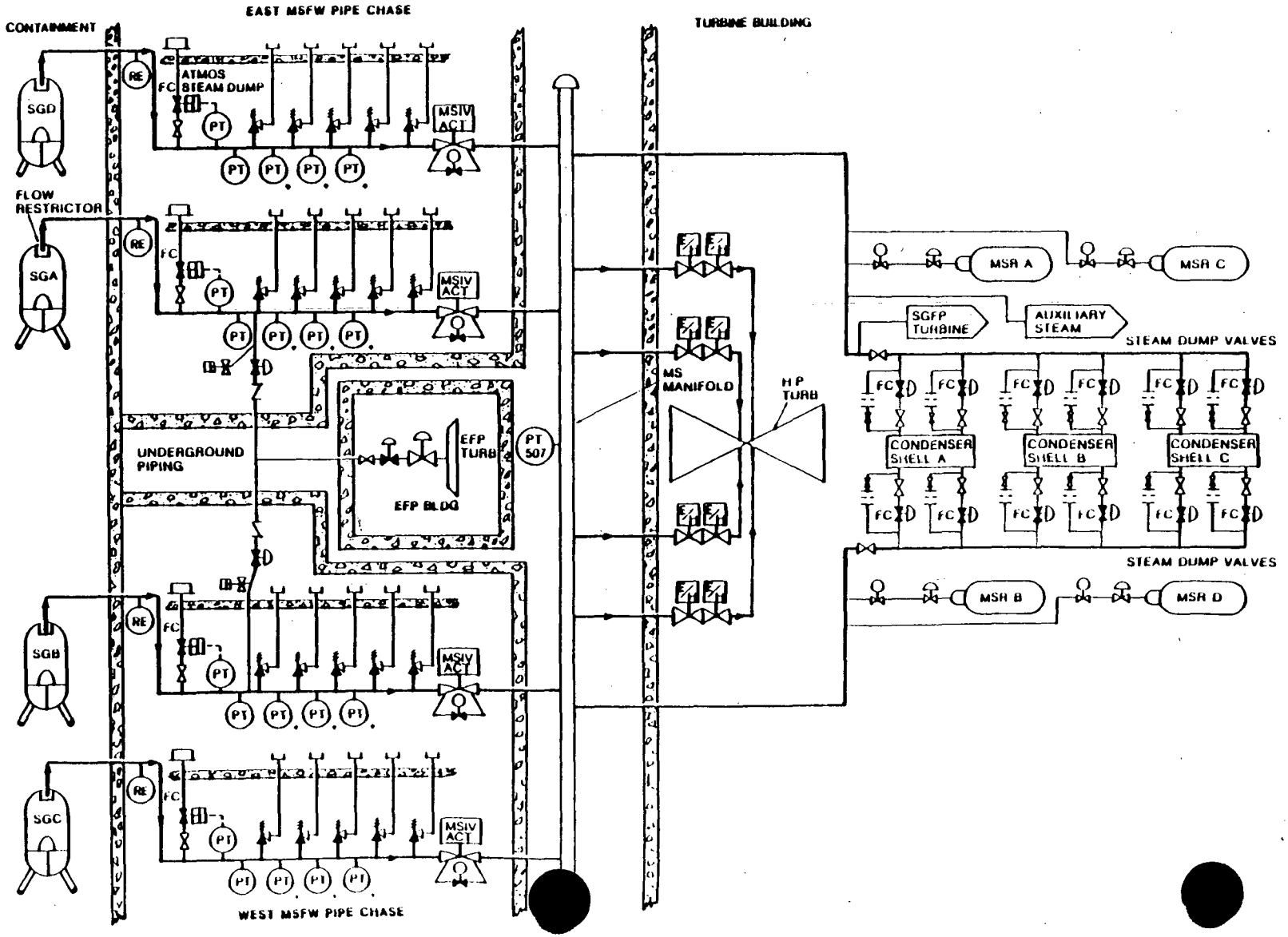

FIGURE E.15-1 MAIN STEAM SYSTEM SCHEMATIC

E.16 Containment Building Spray (CBS) System

E.16.1 System Description

Function

The Containment Building Spray (CBS) System is designed to maintain the containment building pressure and temperature within design limits in the event of a main steam line break or loss of coolant accident (LOCA). It serves as an active heat removal system and provides a fission product, especially radioiodine, removal function following a LOCA.

Configuration

The CBS System (see Figure E.16-1) consists of two trains, each of which consists of a 100% capacity pump, a heat exchanger, two spray headers, and a shared spray additive tank (SAT).

Dependencies

The CBS System depends on the RWST for a water source during the injection mode and on the containment sumps during the recirculation mode. Cooling for the pumps and heat exchangers is provided by the PCC System. Electric power is necessary for the pumps and motor-operated valves. During the recirculation mode, the Enclosure Air Handling (EAH) System is required to maintain the pump room temperature within design specifications.

Operation

During normal plant operation, the CBS System is in standby. The injection phase of CBS operation is automatically initiated by a Containment Spray Actuation Signal (CSAS) which is generated by a Hi-3 containment pressure signal (P signal). The recirculation phase is initiated when a low-low level in the RWST (in conjunction with an S signal) is detected.

E.16 Containment Building Spray (CBS) System (Continued)

E.16.2 System Model

System unavailability is quantified using the block diagram approach and RISKMAN software.

Top Event Definition

The results from the CBS System analysis are used as eight different top events in the event trees, as follows:

Top Events	Description
CA, CB	One train of CBS starts and runs for one hour in the injection mode, drawing water from the RWST. These top events are used in early trees such as large LOCA where CBS is initiated early in the sequence.
XA, XB	One train of CBS runs for seven days (168 hours) in the recirculation mode, drawing water from the containment recirculation sump. The associated CBS heat exchanger is modeled separately (as Top Event VA for XA and Top Event VB for XB).
VA, VB	One train of CBS heat exchanger is cooled for seven days. These top events are used in the long-term response event trees in connection with the start and run in the injection mode.
XC, XD	One train of CBS starts in the recirculation mode and operates for seven days with heat exchanger cooling. These top events are used in the long-term response trees where CBS has not been automatically actuated earlier when the RWST is injected.

E.16 Containment Building Spray (CBS) System (Continued)

Success Criteria

The system success criteria for the injection mode is that at least one out of two trains start and run for one hour. For the recirculation mode, system success requires at least one out of two trains operate for one week. One train of CBS is sufficient to remove the core decay heat and prevent containment overpressure.

Analysis Conditions

The CBS System is analyzed under the following assumptions:

- The CBS System model does not include the RWST, containment recirculation sumps, and their associated valves. These components are included as their own top events in the event trees.

- Failure of PCC cooling to the CBS pump seal coolers or to the CBS heat exchangers is assumed to cause pump failure or containment cooling function failure, respectively, during the recirculation mode but not during the injection phase. (The CBS pumps are self-cooled in the injection mode.)

- The containment spray actuation signal is assumed available for the injection mode of operation.

- Failure of test lines (i.e., open test line valves) is assumed to cause failure of the CBS train due to insufficient flow to the headers.

- Failure of NaOH addition (i.e., the Spray Additive Tank) is not considered to cause system failure for this analysis.

- The injection and recirculation modes of operation are quantified separately.

E.16 Containment Building Spray (CBS) System (Continued)

E.16.3 Results

CBSCA1	6.257E-04	CBS INJECTION - All Support Systems Available (ASSA)
CBSCA1	6.044E-04	NORMAL Configuration
CBSCA1	9.129E-05	Hardware
CBSCA1	5.131E-04	Common Cause
CBSCA1	2.127E-05	MAINTENANCE Configuration
CBSCA1	1.721E-12	TEST Configuration
CBSCA2	9.707E-03	CBS INJECTION - Single Support Train Available
CBSCA2	8.365E-03	NORMAL
CBSCA2	7.852E-03	Hardware
CBSCA2	3.379E-03	P9A-I Block
CBSCA2	4.473E-03	MOV11 Block
CBSCA2	5.131E-04	Common Cause
CBSCA2	2.256E-04	Pump Injection CC
CBSCA2	2.876E-04	MOV CC
CBSCA2	1.343E-03	MAINTENANCE
CBSCA2	1.074E-10	TEST
CBSXA1	1.679E-04	CBS PUMP RECIRC W/O HX COOLING - ASSA
CBSXA1	1.679E-04	NORMAL
CBSXA1	1.091E-04	Hardware
CBSXA1	5.879E-05	Common Cause
CBSXA2	6.377E-03	CBS PUMP RECIRC W/O HX COOLING - SSTA
CBSXA2	6.377E-03	NORMAL
CBSXA2	6.318E-03	Hardware
CBSXA2	6.270E-03	P9A-R Block
CBSXA2	4.826E-05	MOV11T
CBSXA2	5.879E-05	Common Cause
CBSVA1	3.177E-04	CBS HEAT EXCHANGER COOLING DURING RECIRC - ASSA
CBSVA1	3.177E-04	NORMAL
CBSVA1	3.019E-05	Hardware
CBSVA1	2.876E-04	Common Cause
CBSVA2	4.257E-03	CBS HEAT EXCHANGER COOLING DURING RECIRC - SSTA
CBSVA2	4.257E-03	NORMAL
CBSVA2	3.970E-03	Hardware
CBSVA2	2.876E-04	Common Cause
CBSXC1	1.173E-03	CBS RECIRC: START & RUN W/ HX COOLING - ASSA
CBSXC1	1.125E-03	NORMAL
CBSXC1	4.578E-04	Hardware
CBSXC1	6.674E-04	Common Cause
CBSXC1	4.729E-05	MAINTENANCE
CBSXC1	3.751E-12	TEST

E.16 Containment Building Spray (CBS) System (Continued)

CBSXC2	1.981E-02	CBS RECIRC: START & RUN W/ HX COOLING - SSTA
CBSXC2	1.847E-02	NORMAL
CBSXC2	1.780E-02	Hardware
CBSXC2	9.314E-03	P9A-SR Block
CBSXC2	4.521E-03	MOV11R Block
CBSXC2	3.970E-03	MOV137 Block
CBSXC2	6.674E-04	Common Cause
CBSXC2	2.840E-04	Pump Start/Run CC
CBSXC2	3.834E-04	Two MOV Pairs CC
CBSXC2	1.343E-03	MAINTENANCE
CBSXC2	1.074E-10	TEST

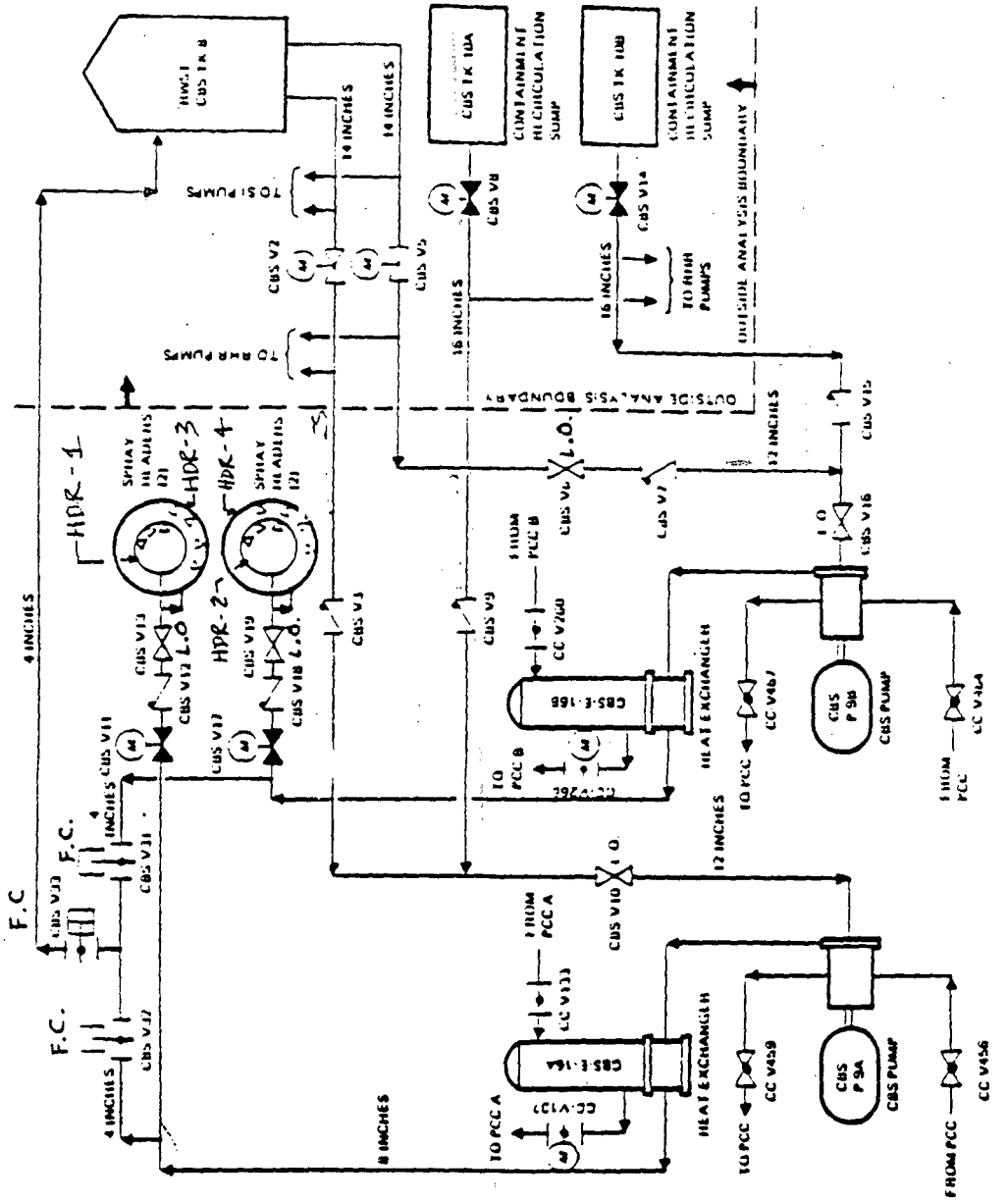

FIGURE E.16-1 CONTAINMENT BUILDING SPRAY SYSTEM SIMPLIFIED PIPING AND INSTRUMENTATION DIAGRAM

E.17 Containment Isolation System

E.17.1 System Description

Function

The function of CI is to guard against the atmospheric release of radioactive material in the event of an accident by isolating those lines penetrating the containment which are not required for the operation of ESF systems.

Configuration

The CI system consists of those valves and other barriers from a number of different systems that serve to isolate the containment on demand. The CI system provides a double barrier protection for lines that penetrate containment. The barriers may consist of valves, a closed system (i.e., blind flanges of locked-closed isolation valves) of diaphragms.

Dependencies

Support for the isolation valves is provided by the AC Power System for motor-operated valves and by SSPS for the generation of a containment isolation signal. The air- and solenoid-operated valves fail to the safe (closed) position on loss of support systems except for loss of the signal.

Operation

The automatic isolation valves in the nonessential process lines, which do not increase the potential for damage to in-containment equipment when isolated, are closed given a T signal derived in conjunction with the automatic safety injection on high containment pressure. The automatic isolation valves in the other process lines (which do not include safety injection lines) are closed given a P signal, derived from high-3 containment pressure.

E.17 Containment Isolation System (Continued)

E.17.2 System Model

The CI System is modeled as two top events in the long-term event trees: CI, which includes all small CI valves (all valves except the Containment On-Line Purge (COP) valves) plus the likelihood for small pre-existing leaks that do not auto isolate; and C2, which includes the two 8-inch COP lines plus the likelihood for large pre-existing leaks that do not auto isolate.

Success Criteria

Success of CI is defined as proper actuation and operation of at least one of the double barriers in all lines modeled.

Analysis Conditions

- Of the some 80 mechanical penetrations, most can be dismissed from consideration because the system is closed, because of multiple isolation valves, etc. Eight penetration lines were chosen for quantification based on a screening analysis.

- Operator action is included in this model to isolate the seal return line MOV (outside containment) given a station blackout. Also, signal recovery is modeled explicitly in the Auxiliary Systems event tree, which includes recovery of containment isolation signal.

- CI System model is analyzed for various combinations of support states including loss of signals and loss of ac power.

E.17.3 Results

Top Event CI - Small Containment Isolation Valves (≤3" Dia.) and Small Pre-existing Leaks

CIA	4.495E-03	All Support Available
CIB	8.449E-03	Loss of Train B Power, All Signals Available
CIC	4.168E-03	Loss of Train A Power, All Signals Available
CID	5.164E-03	Loss of Both Trains of Power, All Signals Available

E.17 Containment Isolation System (Continued)

CIE	1.710E-02	Loss of Train B Signal, Both Buses Available
CIF	1.710E-02	Loss of Train B Signal and Power
CIG	1.000E+00	Loss of Train B Signal and Train A Power
CIH	1.213E-02	Loss of Train B Signal and All Power
CII	4.450E-03	Loss of Train A Signal, All Power Available
CIJ	1.000E+00	Loss of Train A Signal and Train B Power
CIK	1.683E-02	Loss of Train A Signal and Power
CIL	1.000E+00	Loss of Train A Signal and All Power
CIM	1.000E+00	Loss of All Signals, All Power Available
CIN	1.000E+00	Loss of All Signals and Train B Power
CIO	1.000E+00	Loss of All Signals and Train A Power
CIP	1.000E+00	Loss of All Signals and All Power
CIT	1.000E+00	Guaranteed Failure
CI1	4.495E-03	All Support Available - Seismic Event
CI2	8.449E-03	Loss of Train B Power, Signals Available - Seismic Event
CI3	8.449E-03	Loss of Train A Power, Signals Available - Seismic Event
CI4	1.000E+00	Guaranteed Failure - Loss of All Power and Any State of Signals - Seismic Event
CI5	1.683E-02	Loss of Train B Signal, All Power Available - Seismic Event
CI6	1.683E-02	Loss of Train B Signal, and Power - Seismic Event
CI7	1.683E-02	Loss of Train A Signal, All Power Available - Seismic Event
CI8	1.683E-02	Loss of Train A Signal, and Power - Seismic Event

Top Event C2 - Large Containment Isolation Valves (Purge Lines) and Large Pre-Existing Leaks

C2A	1.096E-04	All Support Available
C2B	1.096E-04	Loss of Train B Power, All Signals Available
C2C	1.096E-04	Loss of Train A Power, All Signals Available
C2D	9.837E-05	Loss of Both Trains of Power, All Signals Available
C2E	4.024E-04	Loss of Train B Signal, Both Buses Available
C2F	4.024E-04	Loss of Train B Signal and Power
C2G	4.024E-04	Loss of Train B Signal and Train A Power
C2H	9.837E-05	Loss of Train B Signal and All Power
C2I	4.024E-04	Loss of Train A Signal, All Power Available
C2J	4.024E-04	Loss of Train A Signal and Train B Power
C2K	4.024E-04	Loss of Train A Signal and Power
C2L	9.837E-05	Loss of Train A Signal and All Power
C2M	1.001E-01	Loss of All Signals, All Power Available
C2N	1.001E-01	Loss of All Signals and Train B Power
C2O	1.001E-01	Loss of All Signals and Train A Power
C2P	9.837E-05	Loss of All Signals and All Power
C2T	1.000E+00	Guaranteed Failure
C21	1.096E-04	For Support States Where All Power is not Lost, All Signals Available - Seismic Event
C22	9.837E-05	For Loss of All Power - Seismic Event
C23	4.024E-04	For Support States Where All Power is not Lost, One Train of Signals is Lost - Seismic Event
C24	1.512E-04	Loss of One Signal, All Power Lost - Seismic Event
C25	2.001E-01	All Power is not Lost and Both Trains of Signals Lost - Seismic Event

APPENDIX F

SPLIT FRACTION LOGIC AND BINNING RULES

APPENDIX F

Split Fraction Logic and Binning Rules

This section contains a listing of all split fraction logic rules (Section F.1) and binning logic rules (Section F.2) used in the plant and containment models. With these rules, the event tree structures in Section 3.1 and the master frequency file (Table 3.4-3), any accident sequence in the model can be quantified. The use of logic rules in sequence quantification is explained in Section 3.3.7.

F.1 Split Fraction Logic Rules

This section contains split fraction logic rules for the:

- Plant Response Event Trees - Section F.1.1, and
- Containment Event Tree - Section F.1.2.

F.2 Bining Logic Rules

This section contains binning logic rules as follows:

- Plant Damage State Binning Logic Rules - Section F.2.1, and
- Release Category Binning Logic Rules - Section F.2.2.

F.1.1 SPLIT FRACTION LOGIC RULES FOR THE PLANT RESPONSE EVENT TREES

Split Fraction Logic for Seismic Event Tree: QUAKE

SF..... Split Fraction Logic.....

QSF	SEISMIC
QSS	-SEISMIC
QYC	INIT-E20T + INIT-E20AT + INIT-E20L
QYB	INIT-E14T + INIT-E14AT + INIT-E14L
QYA	INIT-E10T + INIT-E10L + INIT-E10AT
QY7	INIT-E7T + INIT-E7L + INIT-E7AT
QY5	INIT-E5T + INIT-E5AT + INIT-E5L
QY4	INIT-E4T + INIT-E4AT
QY3	INIT-E3T
QY2	INIT-E2T
QY1	INIT-E1T
QKC	INIT-E20T + INIT-E20AT + INIT-E20L
QKB	INIT-E14T + INIT-E14AT + INIT-E14L
QKA	INIT-E10T + INIT-E10L + INIT-E10AT
QK7	INIT-E7T + INIT-E7L + INIT-E7AT
QK5	INIT-E5T + INIT-E5L + INIT-E5AT
QK4	INIT-E4T + INIT-E4AT
QK3	INIT-E3T
QK2	INIT-E2T
QK1	INIT-E1T
QDC	INIT-E20T + INIT-E20AT + INIT-E20L
QDB	INIT-E14T + INIT-E14AT + INIT-E14L
QDA	INIT-E10T + INIT-E10L + INIT-E10AT
QD7	INIT-E7T + INIT-E7L + INIT-E7AT
QD5	INIT-E5T + INIT-E5L + INIT-E5AT
QD4	INIT-E4T + INIT-E4AT
QD3	INIT-E3T
QD2	INIT-E2T
QD1	INIT-E1T
QRC	INIT-E20T + INIT-E20AT + INIT-E20L
QRB	INIT-E14T + INIT-E14AT + INIT-E14L
QRA	INIT-E10T + INIT-E10L + INIT-E10AT
QR7	INIT-E7T + INIT-E7L + INIT-E7AT
QR5	INIT-E5T + INIT-E5L + INIT-E5AT
QR4	INIT-E4T + INIT-E4AT
QR3	INIT-E3T
QR2	INIT-E2T
QR1	INIT-E1T

Split Fraction Logic for General Transient Early Response Event Tree: GT

SF..... Split Fraction Logic.....

```
          TDPIE1:=OG=F*(INIT=LOSP + INIT=RT + INIT=PLMFW + INIT=TT+ INIT=EXFW + INIT=LOPF + INIT=LCV
                 + INIT=MSIV +INIT=TLMFW + INIT=AMSIV + INIT=SI + INIT=CPEXC)

          TDPIE2:=INIT=LDCA + INIT=LDCB + INIT=RT + INIT=TT +INIT=EXFW + INIT=LOPF + INIT=MSIV +
                 INIT=AMSIV + INIT=LCV+ INIT=SI + INIT=CPEXC

          SUFPIE:=OG=S*(INIT=LDCB + INIT=RT + INIT=TT + INIT=EXFW +INIT=LCV + INIT=MSIV + INIT=LOPF
                 + INIT=AMSIV + INIT=SI +INIT=CPEXC) + INIT=LOSP*GA=S*GB=S*WA=S*WB=S

          TS2:= INIT=RT + INIT=EXFW + INIT=CPEXC + INIT=MSIV +INIT=LOPF + INIT=SI + INIT=FET1 +
                INIT=FET3 + INIT=APAB +INIT=FCRSW + INIT=FLSW + INIT= FSRCC +INIT=FCRCC + INIT=FPCC
                +INIT=L1SWA + INIT=L1SWB + INIT=L1CCA + INIT=L1CCB+ INIT=ACR

          TS4:= INIT=LCV + INIT=AMSIV + INIT=LOSP +INIT=FTBLP + INIT=FLLP +INIT=TCTL

          TS6A:= INIT=E3T + INIT=E4T + INIT=E5T +INIT=E7T + INIT=E10T + INIT=E14T + INIT=E20T

          TS6B:= INIT=E1T + INIT=E2T

TT1       ESFASA*ESFASB + OG=F*GA=F*GB=F
TT3       1

EFF       POWERB*ESFASA*ESFASB*(-(INIT=LDCA + INIT=LDCB + INIT=FCRAC+ INIT=FSRAC +INIT=FL1SG
          +INIT=FL2SG)) + TT=F*(POWERB + ESFASB) +(INIT=TT + TS4 + TS2 + TS6A + TS6B)*ESFASA*ESFASB
EFD       (OG=F + TS2 + INIT=TT + TS4)*(POWERB + ESFASB) + INIT=LDCB+ INIT=FCRAC + INIT=FSRAC
EFL       (INIT=TLMFW + INIT=PLMFW)*ESFASA*ESFASB
EFG       TT=F
EFB       OG=F*TT=S + TS4 + TS6A + TS6B
EFA       TS2 + INIT=LDCA + INIT=TT
EFH       INIT=TLMFW + INIT=PLMFW

FRS       EF=S
FRF       -TDPIE1*-TDPIE2*-SUFPIE + (POWERA + EA=F) * TT=F + OS=F
FR5       POWERA*(POWERB+EB=F)*(TDPIE1+TDPIE2) +-SUFPIE*(POWERB + EB=F) * (TDPIE1+TDPIE2)
FR4       -POWERA*(POWERB+EB=F)*SUFPIE*(TDPIE1+TDPIE2)
FR3       TT=F*-POWERA*-POWERB*-EB=F*SUFPIE
FR2       POWERA*-POWERB*-EB=F*(TDPIE1+TDPIE2) +-SUFPIE*-POWERB*-EB=F*(TDPIE1+TDPIE2)
FRA       (INIT=RT+INIT=TT+INIT=LOPF+INIT=EXFW) * OG=S * -EB=F *-POWERA * -POWERB
FR1       -POWERA*-POWERB*-EB=F*SUFPIE*(TDPIE1+TDPIE2) +INIT=LOSP*GA=S*GB=S*-POWERA*-POWERB*-EB=F
FR0       1

OMF       TS6A + OS=F
OM2       POWERA + POWERB
OM1       1

NLF       (POWERA + PCCA)*(POWERB + PCCB)
NL1       1

RWF       QR=F
RW3       1

H2F       (POWERA + PCCA + ESFASA)*(POWERB + PCCB + ESFASB) + TS6A
H22       POWERA + POWERB
H23       PCCA + PCCB
H21       1

OPF       TS6A + OS=F
OP2       POWERA + POWERB + (POWERA + PCCA)*(POWERB + PCCB)
OP1       1

RV2       1

ORF       (POWERA + PCCA)*(POWERB + PCCB) + OG=F*(DA=F + DB=F) +TS6A + INIT=LDCA + INIT=LDCB + OS=F
OR4       1

OQ2       (POWERA + PCCA)*(POWERB + PCCB)
OQ1       1

RA3       1
```

Split Fraction Logic for General Transient Early Response Event Tree: GT

SF..... Split Fraction Logic.....

RB3	1
L1F	POWERA + PCCA + ESFASA + INIT-LDCA
L14	POWERB + PCCB + ESFASB + INIT-LDCB
L13	1
L2F	POWERB + PCCB + ESFASB + INIT-LDCB
L24	POWERA + PCCA + ESFASA + -RA-S + INIT-LDCA
L2C	L1 = F
L23	L1-S
CAF	POWERA + ESFASA + INIT-LDCA
CA2	1
CBF	POWERB + ESFASB + INIT-LDCB
CB2	POWERA + ESFASA + -RA-S + INIT-LDCA
CBA	CA = F
CB1	CA-S

Split Fraction Logic for General Transient Long-Term Response Event Tree: LTGT

SF..... Split Fraction Logic.....

BLEED:= (H2=S * OR=S)

PDSA:= ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

PDSC:= ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

PDSD:= C2=S * CI=S

PDSE:= ((XA=S+XB=S) + (XC=S+XD=S))*(C2=F+CI=F)

PDSF:=C2=F

PDSFP:=C2=S * CI=F

PDS2:=RV=F

PDS3:= ((NL=F*OQ=F)+(EF=F*FR=F+OM=F)*(RW=F+H2=F+OR=F))*-CA=S*-CB=S

PDS4:= ((NL=F*OQ=F)+(EF=F*FR=F+OM=F)*(RW=F+H2=F+OR=F))*(CA=S+CB=S)

PDS6:= RP=F

PDS7:= (EF=S+FR=S)*((NL=S*OQ=F)+(NL=F*OQ=S)+(OM=F*OQ=F))*-CA=S*
-CB=S

PDS8:= (EF=S+FR=S)*((NL=S*OQ=F)+(NL=F*OQ=S)+(OM=F*OQ=F))*(CA=S+CB=S) + RP=S +(H2=S * OR=S
* -L1=S * -L2=S)

SUS (OM=B * NL=S * OQ=S) + (OM=S *(-OP=F + OQ=S)*-NL=F) +(OM=F * OQ=S)
SUF 1

LTF OQ=F + NL=F + RV=F + (OM=F * -OQ=S) + (EF=F*FR=F * -L1=S *-L2=S)
LTS H2=S * OR=S * (L1=S + L2=S)

WST -CA=S*-CB=S*-RV=F*-BLEED
WS1 CA=S+CB=S+RV=F+BLEED

ZAF ESFASA + POWERA
ZA4 1

ZBF ESFASB + POWERB
ZB4 ESFASA+POWERA
ZBB ZA=F
ZB3 ZA=S

XAF CA=F + CA=B*-BLEED + EH=F + PA=F + ESFASA
XA3 BLEED
XA2 1

VA3 BLEED
VA2 1

XBF CB=F + CB=B*-BLEED + EH=F + PB=F + ESFASB
XB3 BLEED*(PA=F+ESFASA)
XBB BLEED*XA=F
XB2 CA=F + CA=B*-BLEED + PA=F + ESFASA + ZA=F
XBA XA=F
XB1 XA=S

VB3 BLEED
VBA VA=F
VB2 VA=B

RPF H2=F
RP1 H2=S

L5F -L1=S+EH=F+PA=F+ESFASA
L54 RP=S
L52 RP=F

Split Fraction Logic for General Transient Long-Term Response Event Tree: LTGT

SF..... Split Fraction Logic.....

L6F	-L2=S+EH=F+PB=F+ESFASB
L64	RP=S*(L5=B+-L1=S+PA=F+ESFASA)
L6B	RP=S*L5=F
L63	RP=S*L5=S
L62	RP=F*(L5=B+-L1=S+PA=F+ESFASA)
L6A	RP=F*L5=F
O31	1
RC4	POWERA*L6=S + POWERB*L5=S
RC5	PA=F*L6=S + PB=F*L5=S
RC2	-L5=S*L6=S
RC3	L5=S*-L6=S
RC1	L5=S*L6=S
XCF	-L5=S+H2=F*RA=F+CA=F+EH=F+PA=F+ESFASA
XC2	1
XDF	-L6=S+H2=F*RB=F+CB=F+EH=F+PB=F+ESFASB
XD2	H2=F*RA=F+CA=F+XC=B + PA=F + ESFASA
XDA	XC=F
C25	QS=F*ESFASA*ESFASB*(-POWERA + -POWERB)
C23	QS=F*(POWERA + POWERB)*(ESFASA + ESFASB)
C24	QS=F*POWERA*POWERB*(ESFASA + ESFASB)
C21	QS=F*(-POWERA*POWERB + POWERA*-POWERB)*-ESFASA*-ESFASB
C22	QS=F*POWERA*POWERB*-ESFASA*-ESFASB
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1
CI4	QS=F*POWERA*POWERB
CI6	QS=F*ESFASB*POWERB*-ESFASA*-POWERA
CI8	QS=F*ESFASA*POWERA*-ESFASB*-POWERB
CI7	QS=F*ESFASA*-ESFASB*-POWERA*-POWERB
CI5	QS=F*ESFASB*-ESFASA*-POWERA*-POWERB
CI3	QS=F*POWERA*-ESFASA*-ESFASB*-POWERB
CI2	QS=F*POWERB*-ESFASA*-ESFASB*-POWERA
CI1	QS=F
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for Small LOCA Early Response Event Tree: SLOCA

SF..... Split Fraction Logic.....

TT2	OG=F
TT1	ESFASA*ESFASB + OG=F*GA=F*GB=F
TT3	1
EFF	ESFASA*ESFASB
EFD	OG=F*(POWERB + ESFASB)
EFC	ESFASB
EFB	OG=F
EFG	TT=F
EFA	1
OMF	OS=F
OM1	1
RWF	QR=F
RW3	1
H2F	(POWERA + ESFASA)*(POWERB + ESFASB) + PCCA*PCCB +PCCB*(POWERA + ESFASA) + PCCA*(POWERB + ESFASB)
H22	POWERA + ESFASA + POWERB + ESFASB
H23	PCCA + PCCB
H21	1
OPF	OS=F
OP1	1
RV2	1
ORF	OS=F
OR4	1
OQ2	HT=F
OQ1	1
RA3	1
RB3	1
L1F	RW=F + RA=F + POWERA + PCCA + ESFASA
L14	RB=F + POWERB + PCCB + ESFASB
L13	1
L2F	RW=F + RB=F + POWERB + PCCB + ESFASB
L24	RA=F + POWERA + PCCA + ESFASA
L2C	L1=F
L23	1
LRF	OG=F + (PCCA + ESFASA)*(PCCB + ESFASB) + L1=F*L2=F +EH=F
LR2	PCCA + PCCB + ESFASA + ESFASB + L1=F + L2=F
LR1	1
CAF	RW=F + RA=F + POWERA + ESFASA
CA2	1
CBF	RW=F + RB=F + POWERB + ESFASB
CB2	RA=F + POWERA
CBA	CA=F
CB1	1

Split Fraction Logic for Small LOCA Long-Term Response Event Tree: LTSLOCA

SF..... Split Fraction Logic.....

```
BLEED:= (H2=S * -OR=F)

PDSA:= ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

PDSC:= ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

PDSD:= C2=S * CI=S

PDSE:= ((XA=S + XB=S) + (XC=S + XD=S)) * (C2=F + CI=F)

PDSF:=C2=F

PDSFP:=C2=S * CI=F

PDS2:=RV=F

PDS3:= ((RW=F*OQ=F)+(EF=F+OM=F)*(RW=F+H2=F)+(EF=F*OR=F))*-CA=S*-CB=S

PDS4:= ((RW=F*OQ=F)+(EF=F+OM=F)*(RW=F+H2=F)+(EF=F*OR=F))*(CA=S+CB=S)

PDS6:= RP=F

PDS7:=(EF=S*-OM=F*H2=F + RW=F*OQ=S + OM=F*OR=F)*-CA=S*-CB=S

PDS8:=(EF=S*-OM=F*H2=F + RW=F*OQ=S + OM=F*OR=F)*(CA=S +CB=S) +-OP=B*OQ=F +
      TT=S*EF=S*H2=S*OQ=F + H2=S*(OQ=S +OR=S) + RP=S

SUS     LR=S
SUF     1

LTF     -(L1=S + L2=S)
LTS     (L1=S + L2=S) * -LR=S

WSF     -CA=S*-CB=S*-RV=F*-BLEED
WS1     CA=S+CB=S+RV=F+BLEED

ZAF     ESFASA+POWERA
ZA4     1

ZBF     ESFASB+POWERB
ZB4     ESFASA+POWERA
ZBB     ZA=F
ZB3     ZA=S

XAF     CA=F + CA=B*-BLEED + EH=F + PA=F + ESFASA
XA3     BLEED
XA2     1

VA3     BLEED
VA2     1

XBF     CB=F + CB=B*-BLEED + EH=F + PB=F + ESFASB
XB3     BLEED*(PA=F+ESFASA)
XBB     BLEED*XA=F
XB2     CA=F + CA=B*-BLEED + PA=F + ESFASA + ZA=F
XBA     XA=F
XB1     XA=S

VB3     BLEED
VBA     VA=F
VB2     VA=B

RPF     H2=F
RP1     H2=S

LSF     -L1=S+EH=F+PA=F+ESFASA
LS4     RP=S
LS2     RP=F
```

Split Fraction Logic for Small LOCA Long-Term Response Event Tree: LTSLOCA

SF..... Split Fraction Logic.....

L6F	-L2=S+EH=F+PB=F+ESFASB
L64	RP=S*(L5=B+-L1=S+PA=F+ESFASA)
L6B	RP=S*L5=F
L63	RP=S*L5=S
L62	RP=F*(L5=B+-L1=S+PA=F+ESFASA)
L6A	RP=F*L5=F
O31	1
RC4	POWERA*L6=S + POWERB*L5=S
RC5	PA=F*L6=S + PB=F*L5=S
RC2	-L5=S*L6=S
RC3	L5=S*-L6=S
RC1	L5=S*L6=S
XCF	H2=F*RA=F+CA=F+EH=F+PA=F+ESFASA
XC2	1
XDF	H2=F*RB=F+CB=F+EH=F+PB=F+ESFASB
XD2	H2=F*RA=F+CA=F+XC=B + PA=F + ESFASA
XDA	XC=F
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for Medium LOCA Early Response Event Tree: MLOCA

SF..... Split Fraction Logic.....

RWF	QR=F
RW2	1
RA2	1
RB2	1
H1F	(POWERA + ESFASA)*(POWERB + ESFASB) + PCCA*PCCB +(POWERA + ESFASA)*PCCB + (POWERB + ESFASB)*PCCA
H12	POWERA + POWERB + ESFASA + ESFASB
H13	PCCA + PCCB
H11	1
EFF	ESFASA*ESFASB
EFD	OG=F*(POWERB + ESFASB)
EFC	ESFASB
EFB	OG=F
EFA	1
OD1	1
L1F	RW=F + RA=F + POWERA + ESFASA + PCCA
L12	1
L2F	RW=F + RB=F + POWERB + ESFASB + PCCB
L22	RA=F + POWERA + ESFASA + PCCA
L2A	L1=F
L21	L1=S
CAF	RW=F + RA=F + POWERA
CA2	1
CBF	RW=F + RB=F + POWERB
CB2	RW=F + POWERA
CBA	CA=F
CB1	CA=S

Split Fraction Logic for Medium LOCA Long-Term Response Event Tree: LTMLOCA

SF..... Split Fraction Logic.....

PDSA:= ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

PDSC:= ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

PDSD:= C2=S * CI=S

PDSE:= ((XA=S + XB=S) + (XC=S + XD=S)) * (C2=F + CI=F)

PDSF:=C2=F

PDSFP:=C2=S * CI=F

PDS2:= H1=S*OD=S*-(L1=S+L2=S)*-(RA=F*RB=F)

PDS3:= RW=F + H1=F*-(L1=S+L2=S+CA=S+CB=S)

PDS4:= (-H1=S + -OD=S)*-(L1=S+L2=S)*(CA=S+CB=S)

PDS6:= RP=F

PDS8:= H1=S*-OD=S*-(L1=S+L2=S+CA=S+CB=S) + H1=S*OD=S*RA=F*RB=F + RP=S

SUF	1
LTF	-(L1=S+L2=S)
LTS	(L1=S + L2=S)
WSF	-(CA=S+CB=S+L1=S+L2=S+H1=S)
WS1	(CA=S+CB=S+L1=S+L2=S+H1=S)
ZAF	ESFASA + POWERA
ZA4	1
ZBF	ESFASB + POWERB
ZB4	ESFASA+POWERA
ZBB	ZA=F
ZB3	ZA=S
XAF	-CA=S + EH=F + PA=F + ESFASA
XA2	1
VA2	1
XBF	-CB=S + EH=F + PB=F + ESFASB
XB2	-CA=S + PA=F + ESFASA + ZA=F
XBA	XA=F
XB1	XA=S
VBA	VA=F
VB2	VA=B
RPF	OD=S
RP1	-OD=S
L5F	-L1=S+EH=F+PA=F+ESFASA
L54	RP=S
L52	RP=F
L6F	-L2=S+EH=F+PB=F+ESFASB
L64	RP=S*(L5=B+-L1=S+PA=F+ESFASA)
L6B	RP=S*L5=F
L63	RP=S*L5=S
L62	RP=F*(L5=B+-L1=S+PA=F+ESFASA)
L6A	RP=F*L5=F
031	1
RC4	POWERA*L6=S + POWERB*L5=S
RC5	PA=F*L6=S + PB=F*L5=S

Split Fraction Logic for Medium LOCA Long-Term Response Event Tree: LTMLOCA

SF..... Split Fraction Logic.....

RC2	-L5=S*L6=S
RC3	L5=S*-L6=S
RC1	L5=S*L6=S
XCF	-CA=S+EH=F+PA=F+ESFASA
XC2	1
XDF	-CB=S+EH=F+PB=F+ESFASB
XD2	-CA=S+XC=B + PA=F + ESFASA
XDA	XC=F
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for Large LOCA Early Response Event Tree: LL1

SF..... Split Fraction Logic.....

RWF	QR=F
RW1	1
RA1	1
RB1	1
LAF	INIT=ELOCA + POWERA + ESFASA
LA2	1
LBF	INIT=ELOCA + POWERB + ESFASB
LB2	POWERA + ESFASA + LA=B
LBA	LA=F
LB1	LA=S
CAF	INIT=APC + POWERA + ESFASA + INIT=TMLL
CA2	1
CBF	INIT=APC + POWERB + ESFASB + INIT=TMLL
CB2	CA=B + POWERA + ESFASA
CBA	CA=F
CB1	CA=S

Split Fraction Logic for Large LOCA Long-Term Response Event Tree: LL2

SF..... Split Fraction Logic.....

PDSA:= (XA=S + XB=S)*(HA=S + VA=S + HB=S + VB=S)*CI=S

PDSC:= (XA=S + XB=S)*-(HA=S + VA=S + HB=S + VB=S)*CI=S

PDSD:= C2=S * CI=S

PDSE:= (XA=S + XB=S) * (C2=F + CI=F)

PDSF:= C2=F * -(INIT=APC)

PDSFP:= C2=S * CI=F

PDSFA:= C2=F * INIT=APC

PDS1:= WS=F

PDS6:= HE=F + HS=F + CV=F*(LD=S+LC=S)

PDS2:= (HE=B * HS=B) * WS=S

WSF	-(LA=S + LB=S + CA=S + CB=S)
WS1	(LA=S + LB=S + CA=S + CB=S)
ZAF	ESFASA + POWERA
ZA2	1
ZBF	ESFASB + POWERB
ZB2	ESFASA + POWERA
ZBA	ZA=F
ZB1	1
CVF	EH=F
CV1	1
LCF	-LA=S + PA=F + EH=F
LC2	1
HAF	EH=F
HA2	1
LDF	-LB=S + PB=F + EH=F
LD2	LC=B + -LA=S + PA=F
LDA	LC=F
LD1	LC=S
HBF	EH=F
HB2	HA=B
HBA	HA=F
HEF	EH=F
HE1	1
XAF	-CA=S + EH=F + PA=F + ESFASA
XA2	1
VAF	QD=F + QK=F
VA2	1
XBF	-CB=S + EH=F + PB=F + ESFASB
XB2	-CA=S + PA=F + ESFASA + ZA=F
XBA	XA=F
XB1	XA=S
VBF	QD=F + QK=F
VBA	VA=F
VB2	VA=B
HSF	EH=F
HS1	1

Split Fraction Logic for Large LOCA Long-Term Response Event Tree: LL2

SF..... Split Fraction Logic.....

C2T	INIT=APC + INIT=TMLL
C25	QS=F*ESFASA*ESFASB*(-POWERA + -POWERB)
C23	QS=F*(POWERA + POWERB)*(ESFASA + ESFASB)
C24	QS=F*POWERA*POWERB*(ESFASA + ESFASB)
C21	QS=F*(-POWERA*POWERB + POWERA*-POWERB)*-ESFASA*-ESFASB
C22	QS=F*POWERA*POWERB*-ESFASA*-ESFASB
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1

CIT	INIT=APC + INIT=TMLL
CI4	QS=F*POWERA*POWERB
CI6	QS=F*ESFASB*POWERB*-ESFASA*-POWERA
CI8	QS=F*ESFASA*POWERA*-ESFASB*-POWERB
CI7	QS=F*ESFASA*-ESFASB*-POWERA*-POWERB
CI5	QS=F*ESFASB*-ESFASA*-POWERA*-POWERB
CI3	QS=F*POWERA*-ESFASA*-ESFASB*-POWERB
CI2	QS=F*POWERB*-ESFASA*-ESFASB*-POWERA
CI1	QS=F
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for Steamline Break Inside Containment Early Response Event Tree: SLBI

SF..... Split Fraction Logic.....

```
MSF     ESFASA*ESFASB + POWERA*POWERB
MS1     1

EFF     ESFASA*ESFASB
EFD     POWERB + ESFASB
EFG     MS=F
EFB     1

OMF     OS=F
OM1     1

RWF     QR=F
RW3     1

H2F     (POWERA + PCCA) * (POWERB + PCCB)
H23     PCCA + PCCB
H22     POWERA + POWERB
H21     1

OPF     OS=F
OP2     POWERA + POWERB + (POWERA + PCCA) * (POWERB + PCCB)
OP1     1

RV2     1

ORF     (POWERA + PCCA) * (POWERB + PCCB) + OG=F * (DA=F + DB=F) +OS=F
OR4     1

OQ1     1

RA3     1

RB3     1

L1F     POWERA + PCCA
L14     1

L2F     POWERB + PCCB
L24     POWERA + PCCA + -RA=S
L2C     L1=F
L23     L1=S

CAF     POWERA + ESFASA
CA2     1

CBF     POWERB + ESFASB
CB2     -RA=S + POWERA + ESFASA
CBA     CA=F
CB1     CA=S
```

Split Fraction Logic for Steamline Break Inside Containment (SLBI) Long-Term Response Event Tree: LTSLBI

SF..... Split Fraction Logic.....

PDSA:= ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

PDSC:= ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

PDSD:= C2=S * CI=S

PDSE:= ((XA=S + XB=S) + (XC=S + XD=S)) * (C2=F + CI=F)

PDSF:=C2=F

PDSFP:=C2=S * CI=F

PDS2:=RV=F

PDS3:= ((OM=B*-OQ=F)+(RW=F*OQ=F)+(OM=S*((-OQ=F*-RV=F)+(H2=F+(RW=F* OQ=F)))+OM=F*(OQ=S+RW=F)))*-(CA=S+CB=S+L1=S+L2=S)

PDS4:= (OM=B*-OQ=F+RW=F*OQ=F+OM=S*(-OQ=F*-RV=F+H2=F+RW=F*OQ=F)+OM=F *(OQ=S+RW=F))*(-L1=S+-L2=S)

PDS6:= RP=F

PDS7:=(RW=F*OQ=S) + ((H2=F*OQ=F)+(OM=F*H2=F))*-(CA=S+CB=S)

PDS8:=(RW=F*OQ=S) + ((H2=F*OQ=F)+(OM=F*H2=F))*(CA=S+CB=S) +(H2=S*OQ=F) + RP=S + H2=S*OR=S*-L1=S*-L2=S

SUF	1
LTF	-(L1=S + L2=S)
LTS	(L1=S + L2=S)
WSF	-CA=S*-CB=S*-RV=F
WS1	CA=S+CB=S+RV=F
ZAF	ESFASA + POWERA
ZA4	1
ZBF	ESFASB + POWERB
ZB4	ESFASA + POWERA
ZBB	ZA=F
ZB3	ZA=S
XAF	-CA=S + EH=F + PA=F + ESFASA
XA2	1
VA2	1
XBF	-CB=S + EH=F + PB=F + ESFASB
XB2	-CA=S + PA=F + ESFASA + ZA=F
XBA	XA=F
XB1	XA=S
VBA	VA=F
VB2	VA=B
RPF	H2=F
RP1	H2=S
L5F	-L1=S+EH=F+PA=F+ESFASA
L54	RP=S
L52	RP=F
L6F	-L2=S+EH=F+PB=F+ESFASB
L64	RP=S*(L5=B+-L1=S+PA=F+ESFASA)
L6B	RP=S*L5=F
L63	RP=S*L5=S
L62	RP=F*(L5=B+-L1=S+PA=F+ESFASA)

Split Fraction Logic for Steamline Break Inside Containment (SLBI) Long-Term Response Event Tree: LTSLBI

SF..... Split Fraction Logic.....

L6A	RP=F*L5=F
031	1
RC4	POWERA*L6=S + POWERB*L5=S
RC5	PA=F*L6=S + PB=F*L5=S
RC2	-L5=S*L6=S
RC3	L5=S*-L6=S
RC1	L5=S*L6=S
XCF	-CA=S+EH=F+PA=F+ESFASA
XC2	1
XDF	-CB=S+EH=F+PB=F+ESFASB
XD2	-CA=S+XC=B + PA=F + ESFASA
XDA	XC=F
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for Steamline Break Outside Containment (SLBO) Early Response Event Tree: SLBO

SF..... Split Fraction Logic.....

```
MSF     ESFASA*ESFASB + POWERA*POWERB
MS1     1

EFF     ESFASA*ESFASB
EFD     POWERB + ESFASB
EFG     MS=F
EFB     1

FRF     (POWERA + EA=F) * MS=F + OS=F
FR5     POWERA*(POWERB+EB=F)
FR4     -POWERA*(POWERB+EB=F)*OG=S
FR3     MS=F*-POWERA*-POWERB*-EB=F*OG=S
FR2     POWERA*-POWERB*-EB=F
FR1     -POWERA*-POWERB*-EB=F*OG=S
FR0     1

OMF     OS=F
OM1     1

NLF     PCCA*PCCB + POWERA*POWERB
NL1     1

RWF     QR=F
RW3     1

H2F     (POWERA + PCCA) * (POWERB + PCCB)
H23     PCCA + PCCB
H22     POWERA + POWERB
H21     1

OP3     OS=F
OP2     POWERA + POWERB + (POWERA + PCCA) * (POWERB + PCCB)
OP1     1

FV2     1

ORF     (POWERA + PCCA) * (POWERB + PCCB) + OG=F * (DA=F + DB=F) +OS=F
OR4     1

OQ2     (POWERA + PCCA) * (POWERB + PCCB)
OQ1     1

RA3     1

RB3     1

L1F     POWERA + PCCA
L14     1

L2F     POWERB + PCCB
L24     POWERA + PCCA + -RA=S
L2C     L1=F
L23     L1=S

CAF     POWERA + ESFASA
CA2     1

CBF     POWERB + ESFASB
CB2     -RA=S + POWERA + ESFASA
CBA     CA=F
CB1     CA=S
```

Split Fraction Logic for Steam Generator Tube Rupture (SGTR) Early Response Event Tree: SGTR

SF..... Split Fraction Logic.....

```
TT2         POWERA
TT1         1

EFF         ESFASA*ESFASB + TT-F*(POWERB+ESFASB)
EFD         OG-F*(POWERB+ ESFASB)
EFG         TT-F
EFB         OG-F*TT-S
EFC         TT-S*(POWERB+ESFASB)
EFA         1

OM2         POWERA + ESFASA*ESFASB + OG-F*(DA-F + GA-F)*(DB-F + GB-F)
OMF         1

NLF         (PCCA + POWERA)*(PCCB + POWERB)
NL1         1

RWF         QR-F
RW3         1

H2F         (POWERA + PCCA + ESFASA)*(POWERB + PCCB + ESFASB)
H23         PCCA + PCCB
H22         POWERA + POWERB
H21         1

OP2         (POWERA + PCCA + ESFASA)*(POWERB + PCCB + ESFASB)
OP1         1

RV1         (POWERA + PCCA + ESFASA)*(POWERB + PCCB + ESFASB)
RV2         1

O4F         (POWERA + PCCA + ESFASA)*(POWERB + PCCB + ESFASB)
O42         EF-F
O41         1

SL7         POWERA*POWERB
SL6         ESFASA*ESFASB
SL4         POWERA*-(POWERB) + -(POWERA)*POWERB + O4-F
SL1         -O4-F

O53         EF-S*SL-F
O51         EF-F
O52         1

OQ2         (POWERA + PCCA)*(POWERB + PCCB)
OQ3         EF-S * O4-F
OQ1         1

RA3         1
RB3         1

L1F         PCCA + ESFASA + POWERA
L14         1

L2F         PCCB + ESFASB + POWERB
L24         PCCA + ESFASA + POWERA + RA-F
L2C         L1-F
L23         L1-S

LRF         POWERA + POWERB + ZH-F + PCCA*PCCB
LR2         -L1-S + -L2-S
LR1         1

CAF         POWERA + ESFASA
CA2         1

CBF         POWERB + ESFASB
CB2         POWERA + ESFASA + CA-B
CBA         CA-F
CB1         CA-S
```

Split Fraction Logic for SGTR Long-Term Response Event Tree: LTSGTR

SF..... Split Fraction Logic.....

SLEAK:= SL=F + O4=F * SL=B

BLEED:= (EF=F * H2=S * O4=S)

PDSA:= ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

PDSC:= ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

PDSD:= C2=S * CI=S

PDSE:= ((XA=S + XB=S) + (XC=S + XD=S)) * (C2=F + CI=F)

PDSF:=C2=F

PDSFP:=C2=S * CI=F

PDS2:=RV=F

PDS3:= ((NL=F*OQ=F)+(EF=F+OM=F)*(RW=F+H2=F)) *-CA=S*-CB=S

PDS4:= ((NL=F*OQ=F)+(EF=F+OM=F)*(RW=F+H2=F))*(CA=S+CB=S)

PDS6:= (EF=F * O5=S * -L1=S * -L2=S) + RP=F

PDS7:= (RW=F * -LR=S) + (EF=F * O4=F) +((-NL=F*OQ=F)+(NL=F*OQ=S)+(H2=F*-LR=S))
*-(CA=S+CB=S)

PDS8:=((-NL=F*OQ=F)+(NL=F*OQ=S)+(H2=F*-LR=S)) *(CA=S+CB=S)+ RP=S + EF=F*O5=F

SUS	(NL=S * OQ=S) + LR=S + (OM=F * OQ=S)
SUF	1
LTS	EF=F * LR=F
LTF	1
WSF	-CA=S*-CB=S*-RV=F*-BLEED
WS1	CA=S+CB=S+RV=F+BLEED
ZAF	ESFASA + POWERA
ZA4	1
ZBF	ESFASB + POWERB
ZB4	ESFASA + POWERA
ZBB	ZA=F
ZB3	ZA=S
XAF	-CA=S + EH=F + PA=F + ESFASA
XA2	1
VA2	1
XBF	-CB=S + EH=F + PB=F + ESFASB
XB2	-CA=S + PA=F + ESFASA + ZA=F
XBA	XA=F
XB1	XA=S
VBA	VA=F
VB2	VA=B
RPF	1
L5F	-L1=S+EH=F+PA=F+ESFASA
L52	RP=F
L6F	-L2=S+EH=F+PB=F+ESFASB
L62	RP=F*(L5=B+-L1=S+PA=F+ESFASA)
L6A	RP=F*L5=F
O31	1

Split Fraction Logic for SGTR Long-Term Response Event Tree: LTSGTR

SF.....	Split Fraction Logic.....
RC6	1
XCF	EF=F*RA=F+CA=F+EH=F+PA=F+ESFASA
XC2	1
XDF	EF=F*RB=F+CB=F+EH=F+PB=F+ESFASB
XD2	EF=F*RA=F+CA=F+XC=B + PA=F + ESFASA
XDA	XC=F
C2T	SLEAK
C2P	ESFASA*ESFASB*POWERA*POWERB
C2O	ESFASA*ESFASB*POWERA
C2N	ESFASA*ESFASB*POWERB
C2H	ESFASB*POWERA*POWERB
C2L	ESFASA*POWERA*POWERB
C2M	ESFASA*ESFASB
C2K	ESFASA*POWERA
C2J	ESFASA*POWERB
C2G	ESFASB*POWERA
C2F	ESFASB*POWERB
C2D	POWERA*POWERB
C2I	ESFASA
C2E	ESFASB
C2C	POWERA
C2B	POWERB
C2A	1
CIT	SLEAK
CIP	ESFASA*ESFASB*POWERA*POWERB
CIO	ESFASA*ESFASB*POWERA
CIN	ESFASA*ESFASB*POWERB
CIH	ESFASB*POWERA*POWERB
CIL	ESFASA*POWERA*POWERB
CIM	ESFASA*ESFASB
CIK	ESFASA*POWERA
CIJ	ESFASA*POWERB
CIG	ESFASB*POWERA
CIF	ESFASB*POWERB
CID	POWERA*POWERB
CII	ESFASA
CIE	ESFASB
CIC	POWERA
CIB	POWERB
CIA	1

Split Fraction Logic for ATWS Early Response Event Tree: ATWS

SF..... Split Fraction Logic.....

SEISATWS:= INIT-E4AT + INIT-E5AT + INIT-E7AT + INIT-E10AT+ INIT-E14AT + INIT-E20AT

PL1	1
MF1	INIT-ATT + INIT-AMFW
MFF	1
AMF	SA-F*SB-F
AM1	1
TT2	INIT-ATT
TTF	AM-F*RT-F
TT1	1
EFF	EA-F*EB-F*AM-F + TT-F*(EB-F+POWERB)
EFT	INIT-ALOMF * EA-F * EB-F
EFS	INIT-ALOMF
EFR	TT-F
EFP	(OG-F + TT-S)*(EB-F+POWERB)
EFN	(OG-F + INIT-AGT) * TT-S
EFM	-TT-F
MRF	SEISATWS
MR2	MT-F
MR1	1
PS4	PL-S + PL-F*MF-S
PSA	PL-F*TT-S*EF-S*MR-S
PSB	PL-F*TT-S*EF-S*MR-F
PSF	1
RV3	1
RWF	QR-F
RV3	1
OHF	MT-F
OH1	INIT-ATT + INIT-ALOMF + INIT-ALOSP + SEISATWS + INIT-AGT +INIT-AMFW
OH3	1
H3F	(ESFASA+POWERA+PCCA)*(ESFASB+POWERB+PCCB)
H3B	PS-F*RV-S*(ESFASA+POWERA+ESFASB+POWERB)
H3C	PS-F*RV-S*(PCCA+PCCB)
H3A	PS-F*RV-S
H32	PS-S*(ESFASA+POWERA+ESFASB+POWERB)
H33	PS-S*(PCCA+PCCB)
H31	1
L1F	EA-F * EB-S
L14	1
L2F	EB-F * EA-S
L24	EA-F + L1-B
L2C	L1-F*EA-S
L23	L1-S
PRF	SEISATWS + POWERA*POWERB
PR2	1
P2F	PS-F*RV-S
P21	1
OQF	OS-F
OQ2	1
RA3	1
RB3	1

Split Fraction Logic for ATWS Early Response Event Tree: ATWS

SF..... Split Fraction Logic.....

LRF	EH-F+POWERA+POWERB+PCCA*PCCB
LR2	-L1-S + -L2-S
LR1	1
CAF	EA-F
CA2	EA-S
CBF	EB-F
CB2	EA-F + CA-B
CBA	CA-F
CB2	CA-S

Split Fraction Logic for ATWS Long-Term Response Event Tree: LTATWS

SF..... Split Fraction Logic.....

```
          LT2Y:=     (-RW=F*-OH=F*-H3=F*-PR=F*-OQ=F*-RA=B*-L1=S*-L2=S)

          PDSA:=     ((XC=S + XD=S) + (VA=S + VB=S))*CI=S

          PDSC:=     ((XA=S + XB=S) * -(VA=S + VB=S))*CI=S

          PDSD:=     C2=S * CI=S

          PDSE:=     ((XA=S + XB=S) + (XC=S + XD=S)) * (C2=F + CI=F)

          PDSF:=     C2=F

          PDSFP:=    C2=S * CI=F

          PDS1 :=    LT=F * (RV=F + RP=F) * (RW=F + -CA=S*-CB=S)

          PDS2 :=    LT=F * (RV=F + RP=F) * (OH=S*H3=S*PR=S + CA=S +CB=S)

          PDS3 :=    LT=F * (-RV=F + RV=S) * (RW=F + -CA=S*-CB=S)

          PDS4 :=    LT=F * (-RV=F + RP=S) * (OH=S*H3=S*PR=S + CA=S +CB=S)

          PDS5 :=    LT=S * (RV=F + RP=F) * (RW=F + -CA=S*-CB=S)

          PDS6 :=    LT=S * (RV=F + RP=F) * (OH=S*H3=S*PR=S + CA=S +CB=S)

          PDS7 :=    LT=S * -RV=F * (RW=F + -CA=S*-CB=S)

          PDS8 :=    LT=S * (-RV=F + RP=S) * (OH=S*H3=S*PR=S + CA=S +CB=S)

SUS       LR=S + P2=S
SUF       1

LTS       L1=S + L2=S
LTF       1

WSF       PDS3 + PDS7
WS1       -(PDS3 + PDS7)

ZAF       (EA=F * EB=S) + POWERA
ZA4       1

ZBF       (EB=F * EA=S) + POWERB
ZB4       EA=F+POWERA
ZBB       ZA=F
ZB3       ZA=S

XAF       CA=F + CA=B*-LT2Y + EH=F + PA=F + EA=F
XA3       LT2Y
XA2       1

VA3       LT2Y
VA2       1

XBF       CB=F + CB=B*-LT2Y + EH=F + PB=F + EB=F
XB3       LT2Y*(PA=F+EA=F)
XBB       LT2Y*XA=F
XB2       CA=F + CA=B*-LT2Y + PA=F + EA=F + ZA=F
XBA       XA=F
XB1       XA=S

VB3       LT2Y
VBA       VA=F
VB2       VA=B

RP1       H3=S

L5F       -L1=S+EH=F+PA=F+(EA=F*EB=S)
L54       RP=S
```

Split Fraction Logic for ATWS Long-Term Response Event Tree: LTATWS

SF..... Split Fraction Logic.....

```
L6F     -L2=S+EH=F+PB=F+(EB=F*EA=S)
L64     RP=S*(L5=B+-L1=S+PA=F+EA=F)
L6B     RP=S*L5=F
L63     RP=S*L5=S

031     1

RC4     POWERA*L6=S + POWERB*L5=S
RC5     PA=F*L6=S + PB=F*L5=S
RC2     -L5=S*L6=S
RC3     L5=S*-L6=S
RC1     L5=S*L6=S

XCF     EH=F+PA=F+(EA=F*EB=S)
XC2     1

XDF     EH=F+PB=F+(EB=F*EA=S)
XD2     CA=F+XC=B + PA=F + EA=F
XDA     XC=F

C25     QS=F*ESFASA*ESFASB*(-POWERA + -POWERB)
C23     QS=F*(POWERA + POWERB)*(ESFASA + ESFASB)
C24     QS=F*POWERA*POWERB*(ESFASA + ESFASB)
C21     QS=F*(-POWERA*POWERB + POWERA*-POWERB)*-ESFASA*-ESFASB
C22     QS=F*POWERA*POWERB*-ESFASA*-ESFASB
C2P     ESFASA*ESFASB*POWERA*POWERB
C2O     ESFASA*ESFASB*POWERA
C2N     ESFASA*ESFASB*POWERB
C2H     ESFASB*POWERA*POWERB
C2L     ESFASA*POWERA*POWERB
C2M     ESFASA*ESFASB
C2K     ESFASA*POWERA
C2J     ESFASA*POWERB
C2G     ESFASB*POWERA
C2F     ESFASB*POWERB
C2D     POWERA*POWERB
C2I     ESFASA
C2E     ESFASB
C2C     POWERA
C2B     POWERB
C2A     1

CI4     QS=F*POWERA*POWERB
CI6     QS=F*ESFASB*POWERB*-ESFASA*-POWERA
CI8     QS=F*ESFASA*POWERA*-ESFASB*-POWERB
CI7     QS=F*ESFASA*-ESFASB*-POWERA*-POWERB
CI5     QS=F*ESFASB*-ESFASA*-POWERA*-POWERB
CI3     QS=F*POWERA*-ESFASA*-ESFASB*-POWERB
CI2     QS=F*POWERB*-ESFASA*-ESFASB*-POWERA
CI1     QS=F
CIP     ESFASA*ESFASB*POWERA*POWERB
CIO     ESFASA*ESFASB*POWERA
CIN     ESFASA*ESFASB*POWERB
CIH     ESFASB*POWERA*POWERB
CIL     ESFASA*POWERA*POWERB
CIM     ESFASA*ESFASB
CIK     ESFASA*POWERA
CIJ     ESFASA*POWERB
CIG     ESFASB*POWERA
CIF     ESFASB*POWERB
CID     POWERA*POWERB
CII     ESFASA
CIE     ESFASB
CIC     POWERA
CIB     POWERB
CIA     1
```

Split Fraction Logic for Interfacing System LOCA (Suction Lines) Event Tree: VS

SF..... Split Fraction Logic.....

LE1	1
VO1	1
PI1	1
SI1	1
LX1	1
LY1	1
LZ1	1
O11	1
CSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
CSA	1
RSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
RSA	1
SSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
SSA	1
VC1	1
O3C	1

Split Fraction Logic for Interfacing System LOCA (Injection Lines) Event Tree: VI

SF..... Split Fraction Logic.....

LE1	1
VO1	1
PI1	1
SI1	1
LX1	1
LY1	1
LZ1	1
O11	1
O22	1
CSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
CSD	LX=F*LY=F
CSC	LX=F
CSB	1
RSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
RSC	LX=F
RSB	1
SSF	(POWERA + ESFASA + PCCA)*(POWERB +ESFASB + PCCB)
SSC	LX=F
SSB	1
VC1	1
O3C	1

Split Fraction Logic for Recovery Event Tree: RECOVERY

SF..... Split Fraction Logic.....

```
          EP1INIT:=OG=F*(INIT=LOSP+INIT=RT+INIT=PLMFW+INIT=TT+INIT=EXFW+
                   INIT=LOPF+ INIT=LCV+ INIT=MSIV+ INIT=TLMFW +INIT=AMSIV + INIT=SI + INIT=CPEXC)

          EPNOSP:=OG=F*(INIT=FLLP+INIT=FTBLP+INIT=TCTL+INIT=LSF6)

ERS       OG=S * GA=S * GB=S
ER1       EP1INIT*DA=S*DB=S*GA=F*GB=F*EF=S
ER2       EP1INIT*DA=S*DB=S*GA=F*GB=F*EF=F
ER3       EP1INIT*DA=S*GA=F*EF=S +EP1INIT*DB=S*GB=F*EF=S +(INIT=L1SWA+INIT=L1CCA+INIT=LDCA) * OG=F
          * DB=S * GB=F * EF=S +(INIT=L1SWB+INIT=L1CCB+INIT=LDCB)*OG=F*DA=S*GA=F*EF=S
ER4       EP1INIT*DA=S*GA=F*EF=F +EP1INIT*DB=S*GB=F*EF=F
ER5       EPNOSP*DA=S*DB=S*GA=F*GB=F*EF=S
ER6       EPNOSP*DA=S*DB=S*GA=F*GB=F*EF=F
ER7       EPNOSP*DA=S*GA=F*EF=S +EPNOSP*DB=S*GB=F*EF=S
ER8       EPNOSP*DA=S*GA=F*EF=F +EPNOSP*DB=S*GB=F*EF=F
ER9       EP1INIT*(DA=F+WA=F+PA=F)*(DB=F+WB=F+PB=F)*EF=S
ERA       EP1INIT*(DA=F+WA=F+PA=F)*(DB=F+WB=F+PB=F)*EF=F +INIT=LOSP*EF=F
ERF       1

RMS       -(INIT=SLOCA)
RMU       INIT=SLOCA * OG=S * GA=S * GB=S * WA=S * WB=S * OS=S *PA=S * PB=S * EH=S * RW=S * H2=S
          * EF=S * L1=F * L2=F
RMF       1
```

F.1.2 SPLIT FRACTION LOGIC RULES FOR THE CONTAINMENT RESPONSE EVENT TREES

Split Fraction Logic for General Transient Containment Response Event Tree: CONTGR (Used for recovered initiating events)

SF..... Split Fraction Logic.....

```
SCS     SU=S + LT=S*((RP=S*RC=S)+(RP=F*(L5=S + L6=S))) +ER=S*(POWERA + POWERB) + RT=F + INIT=SLOCA
          * RM=S
SCF       1

R1S       PDSFP + PDS2 + PDSF
R1F       1

VDF       1

DPF       1

HLF       PDS2 + PDS6 + DP=S
HL1       1

IS1       HL=F * DP=F * (PDS3 + PDS4 + PDS7 + PDS8) * EF=F * FR=F
ISS       1

VHA       (PDS2 + PDS6) * PDSA
VHB       (PDS2 + PDS6) * (PDSC + PDSD)
VHC       (PDS3 + PDS7) * (PDSC + PDSD)
VHD       (PDS4 + PDS8) * PDSA
VHE       (PDS4 + PDS8) * PDSC
VHG       (PDS4 + PDS8) * PDSD
VHG       1

VIA       ((PDS2 + PDS6) * PDSA) * VH=F
VI1       (PDS2 + PDS6) * PDSA
VIB       (PDS2 + PDS6) * (PDSC + PDSD) * VH=F
VI1       (PDS2 + PDS6) * (PDSC + PDSD)
VIC       (PDS3 + PDS7) * PDSD * VH=F
VI3       (PDS3 + PDS7) * PDSD
VID       (PDS4 + PDS8) * PDSA * VH=F
VI3       (PDS4 + PDS8) * PDSA
VIE       (PDS4 + PDS8) * PDSC * VH=F
VI3       (PDS4 + PDS8) * PDSC
VIG       (PDS4 + PDS8) * PDSD * VH=F
VI3       (PDS4 + PDS8) * PDSD
VI3       1

CDF       HL=S
CD2       PDS3 + PDS4 + PDS7 + PDS8
CDF       1

CCA       (PDS2 + PDS4 + PDS6 + PDS8) * CD=F
CC1       PDS2 + PDS4 + PDS6 + PDS8
CC2       PDS3 + PDS7
CC2       1

CHF       (PDS3 + PDS4 + PDS7 + PDS8) * CD=S
CHS       1

CYF       CH=F
CYA       ((PDS2 + PDS6) * VH=F) + ((PDS3 + PDS4 + PDS7 +PDS8) * CD=F * VH=F)
CY2       (PDS2 + PDS6) * PDSA
CY3       (PDS2 + PDS6) * (PDSC + PDSD)
CY7       (PDS3 + PDS7) * PDSD * CH=S
CY4       (PDS4 + PDS8) * PDSA * CH=S
CY5       (PDS4 + PDS8) * PDSC * CH=S
CY6       (PDS4 + PDS8) * PDSD * CH=S
CY1       1

CNA       (PDS2 + PDS6) * PDSA * CY=F
CN1       (PDS2 + PDS6) * PDSA
CNB       (PDS2 + PDS6) * (PDSC + PDSD) * CY=F
CN2       (PDS2 + PDS6) * (PDSC + PDSD)
CND       (PDS3 + PDS7) * PDSD * CH=F
CNC       (PDS3 + PDS7) * PDSD * CY=F
CN3       (PDS3 + PDS7) * PDSD
```

Split Fraction Logic for General Transient Containment Response Event Tree: CONTGR (Used for recovered initiating events)

SF..... Split Fraction Logic.....

```
CNG     (PDS4 + PDS8) * PDSA * CH=F
CNE     (PDS4 + PDS8) * PDSA * CY=F
CN4     (PDS4 + PDS8) * PDSA
CNI     (PDS4 + PDS8) * PDSC * CH=F
CNH     (PDS4 + PDS8) * PDSC * CY=F
CN5     (PDS4 + PDS8) * PDSC
CNK     (PDS4 + PDS8) * PDSD * CH=F
CNJ     (PDS4 + PDS8) * PDSD * CY=F
CN6     (PDS4 + PDS8) * PDSD
CN6     1

REF     1

LHB     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * CC=F
LHA     (PDS2 + PDS4+ PDS6 + PDS8) * PDSA * (VH=F + CH=F)
LH1     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA
LHD     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * CC=F
LHC     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * (VH=F + CH=F)
LH2     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD)
LH2     1

LSC     (PDS2 + PDS6) * PDSA * CC=F * LH=F
LSB     (PDS2 + PDS6) * PDSA * CC=F * LH=S
LSA     (PDS2 + PDS6) * PDSA * CC=S * LH=F
LS1     (PDS2 + PDS6) * PDSA
LSD     (PDS2 + PDS6) * (PDSC + PDSD) * CC=F
LS2     (PDS2 + PDS6) * (PDSC + PDSD)
LSE     (PDS3 + PDS7) * PDSD * CC=F * LH=F
LS3     (PDS3 + PDS7) * PDSD * CC=F * LH=S
LS2     (PDS3 + PDS7) * PDSD * CC=S
LSI     (PDS4 + PDS8) * PDSA * CC=F * LH=F
LSH     (PDS4 + PDS8) * PDSA * CC=F * LH=S
LSG     (PDS4 + PDS8) * PDSA * CC=S * LH=F
LS4     (PDS4 + PDS8) * PDSA
LSJ     (PDS4 + PDS8) * (PDSC + PDSD) * CC=F
LS5     (PDS4 + PDS8) * (PDSC + PDSD)
LS5     1

LM2     (PDS3 + PDS7) * PDSD
LMA     (PDS2 + PDS4 + PDS6 + PDS8) * CC=F
LM1     (PDS2 + PDS4 + PDS6 + PDS8)

SMF     CH=F * CN=F
SMA     (PDS2 + PDS6) * PDSA * LS=F
SM1     (PDS2 + PDS6) * PDSA
SMB     (PDS2 + PDS6) * (PDSC + PDSD) * LS=F
SM2     (PDS2 + PDS6) * (PDSC + PDSD)
SMC     (PDS3 + PDS7) * PDSD * LS=F
SM3     (PDS3 + PDS7) * PDSD
SMD     (PDS4 + PDS8) * PDSA * LS=F
SM4     (PDS4 + PDS8) * PDSA
SME     (PDS4 + PDS8) * (PDSC + PDSD) * LS=F
SM5     (PDS4 + PDS8) * (PDSC + PDSD)
SM5     1
```

Split Fraction Logic for SGTR Containment Response Event Tree: CONTSG

SF..... Split Fraction Logic.....

PDS9A:= ((XC=S + XD=S) +(VA=S + VB=S)) * SLEAK

PDS9C:= ((XC=S + XB=S) + (VA=S + VB=S)) * SLEAK

PDS9D:= C2=F * SLEAK

SCS	SU=S + LT=S * ((RP=S * RC=S)+(RP=F * (L5=S + L6=S))) +RT=F
SCF	1
R1S	PDSFP + PDSE + PDSF + PDS9A + PDS9C + PDS9C
R1F	1
VDF	1
DPS	PDS2 + PDS6
DPF	1
HLF	PDS2 + PDS6 + DP=S
HL1	1
IS1	HL=F * DP=F * (PDS3 + PDS4 + PDS7 + PDS8) * EF=F
ISS	1
VHA	(PDS2 + PDS6) * PDSA
VHB	(PDS2 + PDS6) * (PDSC + PDSD)
VHC	(PDS3 + PDS7) * (PDSC + PDSD)
VHD	(PDS4 + PDS8) * PDSA
VHE	(PDS4 + PDS8) * PDSC
VHG	(PDS4 + PDS8) * PDSD
VHG	1
VIA	((PDS2 + PDS6) * PDSA) * VH=F
VI1	(PDS2 + PDS6) * PDSA
VIB	(PDS2 + PDS6) * (PDSC + PDSD) * VH=F
VI1	(PDS2 + PDS6) * (PDSC + PDSD)
VIC	(PDS3 + PDS7) * PDSD * VH=F
VI3	(PDS3 + PDS7) * PDSD
VID	(PDS4 + PDS8) * PDSA * VH=F
VI3	(PDS4 + PDS8) * PDSA
VIE	(PDS4 + PDS8) * PDSC * VH=F
VI3	(PDS4 + PDS8) * PDSC
VIG	(PDS4 + PDS8) * PDSD * VH=F
VI3	(PDS4 + PDS8) * PDSD
VI3	1
CDF	HL=S
CD2	PDS3 + PDS4 + PDS7 + PDS8
CDF	1
CCA	(PDS2 + PDS4 + PDS6 + PDS8) * CD=F
CC1	PDS2 + PDS4 + PDS6 + PDS8
CC2	PDS3 + PDS7
CC2	1
CHF	(PDS3 + PDS4 + PDS7 + PDS8) * CD=S
CHS	1
CYF	CH=F
CYA	((PDS2 + PDS6)*VH=F) + ((PDS3 + PDS4 + PDS7 +PDS8)* CD=F * VH=F)
CY2	(PDS2 + PDS6) * PDSA
CY3	(PDS2 + PDS6) * (PDSC + PDSD)
CY7	(PDS3 + PDS7) * PDSD * CH=S
CY4	(PDS4 + PDS8) * PDSA * CH=S
CY5	(PDS4 + PDS8) * PDSC * CH=S
CY6	(PDS4 + PDS8) * PDSD * CH=S
CY1	1
CNA	(PDS2 + PDS6) * PDSA * CY=F
CN1	(PDS2 + PDS6) * PDSA

Split Fraction Logic for SGTR Containment Response Event Tree: CONTSG

SF..... Split Fraction Logic.....

```
CNB         (PDS2 + PDS6) * (PDSC+ PDSD) * CY=F
CN2         (PDS2 + PDS6) * (PDSC + PDSD)
CND         (PDS3 + PDS7) * PDSD * CH=F
CNC         (PDS3 + PDS7) * PDSD * CY=F
CN3         (PDS3 + PDS7) * PDSD
CNG         (PDS4 + PDS8) * PDSA * CH=F
CNE         (PDS4 + PDS8) * PDSA * CY=F
CN4         (PDS4 + PDS8) * PDSA
CNI         (PDS4 + PDS8) * PDSC * CH=F
CNH         (PDS4 + PDS8) * PDSC * CY=F
CN5         (PDS4 + PDS8) * PDSC
CNK         (PDS4 + PDS8) * PDSD * CH=F
CNJ         (PDS4 + PDS8) * PDSD * CY=F
CN6         (PDS4 + PDS8) * PDSD
CN6         1

REF         1

LHB         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * CC=F
LHA         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * (VH=F + CH=F)
LH1         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA
LHD         (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * CC=F
LHC         (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * (VH=F + CH=F)
LH2         (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD)
LH2         1

LSC         (PDS2 + PDS6) * PDSA * CC=F * LH=F
LSB         (PDS2 + PDS6) * PDSA * CC=F * LH=S
LSA         (PDS2 + PDS6) * PDSA * CC=S * LH=F
LS1         (PDS2 + PDS6) * PDSA
LSD         (PDS2 + PDS6) * (PDSC+ PDSD) * CC=F
LS2         (PDS2 + PDS6) * (PDSC + PDSD)
LSE         (PDS3 + PDS7) * PDSD * CC=F * LH=F
LS3         (PDS3 + PDS7) * PDSD * CC=F * LH=S
LS2         (PDS3 + PDS7) * PDSD * CC=S
LSI         (PDS4 + PDS8) * PDSA * CC=F * LH=F
LSH         (PDS4 + PDS8) * PDSA * CC=F * LH=S
LSG         (PDS4 + PDS8) * PDSA * CC=S * LH=F
LS4         (PDS4 + PDS8) * PDSA
LSJ         (PDS4 + PDS8) * (PDSC + PDSD) * CC=F
LS5         (PDS4 + PDS8) * (PDSC + PDSD)
LS5         1

LM2         (PDS3 + PDS7) * PDSD
LMA         (PDS2 + PDS4 + PDS6 + PDS8) * CC=F
LM1         (PDS2 + PDS4 + PDS6 + PDS8)
LM1         1

SMF         CH=F * CN=F
SMA         (PDS2 + PDS6) * PDSA * LS=F
SM1         (PDS2 + PDS6) * PDSA
SMB         (PDS2 + PDS6) * (PDSC + PDSD) * LS=F
SM2         (PDS2 + PDS6) * (PDSC + PDSD)
SMC         (PDS3 + PDS7) * PDSD * LS=F
SM3         (PDS3 + PDS7) * PDSD
SMD         (PDS4 + PDS8) * PDSA * LS=F
SM4         (PDS4 + PDS8) * PDSA
SME         (PDS4 + PDS8) * (PDSC + PDSD) * LS=F
SM5         (PDS4 + PDS8) * (PDSC + PDSD)
SM5         1
```

Split Fraction Logic for Medium LOCA Containment Response Event Tree: CONTML

SF..... Split Fraction Logic.....

```
SCS     SU=S + LT=S * ((RP=S * RC=S) + (RP=F * (L5=S + L6=S))) +RT=F
SCF     1

R1S     PDSFP + PDSE + PDSF
R1F     1

VDF     1

DPS     PDS2 + PDS6
DP1     1

HLF     PDS2 + PDS6 + DP=S
HL1     1

IS1     HL=F * DP=F * (PDS3 + PDS4 + PDS8) * EF=F
ISS     1

VHA     (PDS2 + PDS6) * PDSA
VHB     (PDS2 + PDS6) * (PDSC + PDSD)
VHC     (PDS3) * (PDSC + PDSD)
VHD     (PDS4 + PDS8) * PDSA
VHE     (PDS4 + PDS8) * PDSC
VHG     (PDS4 + PDS8) * PDSD
VHG     1

VIA     ((PDS2 + PDS6) * PDSA) * VH=F
VI1     (PDS2 + PDS6) * PDSA
VIB     (PDS2 + PDS6) * (PDSC + PDSD) * VH=F
VI1     (PDS2 + PDS6) * (PDSC + PDSD)
VIC     (PDS3) * PDSD * VH=F
VI3     (PDS3) * PDSD
VID     (PDS4 + PDS8) * PDSA * VH=F
VI3     (PDS4 + PDS8) * PDSA
VIE     (PDS4 + PDS8) * PDSC * VH=F
VI3     (PDS4 + PDS8) * PDSC
VIG     (PDS4 + PDS8) * PDSD * VH=F
VI3     (PDS4 + PDS8) * PDSD
VI3     1

CDF     HL=S
CD2     PDS3 + PDS4 + PDS8
CDF     1

CCA     (PDS2 + PDS4 + PDS6 + PDS8) * CD=F
CC1     PDS2 + PDS4 + PDS6 + PDS8
CC2     PDS3

CHF     (PDS3 + PDS4 + PDS8) * CD=S
CHS     1

CYF     CH=F
CYA     ((PDS2 + PDS6)*VH=F) + ((PDS3 + PDS4 +PDS8)* CD=F * VH=F)
CY2     (PDS2 + PDS6) * PDSA
CY3     (PDS2 + PDS6) * (PDSC + PDSD)
CY7     (PDS3) * PDSD * CH=S
CY4     (PDS4 + PDS8) * PDSA * CH=S
CY5     (PDS4 + PDS8) * PDSC * CH=S
CY6     (PDS4 + PDS8) * PDSD * CH=S
CY1     1

CNA     (PDS2 + PDS6) * PDSA * CY=F
CN1     (PDS2 + PDS6) * PDSA
CNB     (PDS2 + PDS6) * (PDSC+ PDSD) * CY=F
CN2     (PDS2 + PDS6) * (PDSC + PDSD)
CND     (PDS3) * PDSD * CH=F
CNC     (PDS3) * PDSD * CY=F
CN3     (PDS3) * PDSD
CNG     (PDS4 + PDS8) * PDSA * CH=F
CNE     (PDS4 + PDS8) * PDSA * CY=F
```

Split Fraction Logic for Medium LOCA Containment Response Event Tree: CONTML

SF..... Split Fraction Logic.....

```
CN4         (PDS4 + PDS8) * PDSA
CNI         (PDS4 + PDS8) * PDSC * CH-F
CNH         (PDS4 + PDS8) * PDSC * CY-F
CN5         (PDS4 + PDS8) * PDSC
CNK         (PDS4 + PDS8) * PDSD * CH-F
CNJ         (PDS4 + PDS8) * PDSD * CY-F
CN6         (PDS4 + PDS8) * PDSD
CN6         1

REF         1

LHB         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * CC-F
LHA         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * (VH-F + CH-F)
LH1         (PDS2 + PDS4 + PDS6 + PDS8) * PDSA
LHD         (PDS2 + PDS3 + PDS4 + PDS6 + PDS8) *(PDSC + PDSD) * CC-F
LHC         (PDS2 + PDS3 + PDS4 + PDS6 + PDS8) *(PDSC + PDSD) * (VH-F + CH-F)
LH2         (PDS2 + PDS3 + PDS4 + PDS6 + PDS8) *(PDSC + PDSD)
LH2         1

LSC         (PDS2 + PDS6) * PDSA * CC-F * LH-F
LSB         (PDS2 + PDS6) * PDSA * CC-F * LH-S
LSA         (PDS2 + PDS6) * PDSA * CC-S * LH-F
LS1         (PDS2 + PDS6) * PDSA
LSD         (PDS2 + PDS6) * (PDSC+ PDSD) * CC-F
LS2         (PDS2 + PDS6) * (PDSC + PDSD)
LSE         (PDS3) * PDSD * CC-F * LH-F
LS3         (PDS3) * PDSD * CC-F * LH-S
LS2         (PDS3) * PDSD * CC-S
LSI         (PDS4 + PDS8) * PDSA * CC-F * LH-F
LSH         (PDS4 + PDS8) * PDSA * CC-F * LH-S
LSG         (PDS4 + PDS8) * PDSA * CC-S * LH-F
LS4         (PDS4 + PDS8) * PDSA
LSJ         (PDS4 + PDS8) * (PDSC + PDSD) * CC-F
LS5         (PDS4 + PDS8) * (PDSC + PDSD)
LS5         1

LM2         (PDS3) * PDSD
LMA         (PDS2 + PDS4 + PDS6 + PDS8) * CC-F
LM1         (PDS2 + PDS4 + PDS6 + PDS8)
LM1         1

SMF         CH-F * CN-F
SMA         (PDS2 + PDS6) * PDSA * LS-F
SM1         (PDS2 + PDS6) * PDSA
SMB         (PDS2 + PDS6) * (PDSC + PDSD) * LS-F
SM2         (PDS2 + PDS6) * (PDSC + PDSD)
SMC         (PDS3) * PDSD * LS-F
SM3         (PDS3) * PDSD
SMD         (PDS4 + PDS8) * PDSA * LS-F
SM4         (PDS4 + PDS8) * PDSA
SME         (PDS4 + PDS8) * (PDSC + PDSD) * LS-F
SM5         (PDS4 + PDS8) * (PDSC + PDSD)
SM5         1
```

Split Fraction Logic for General Transient Containment Response Event Tree: CONTGT (Used for non-recovered initiating events)

SF..... Split Fraction Logic.....

```
SCS     SU=S + LT=S*((RP=S*RC=S)+(RP=F*(L5=S + L6=S))) + RT=F
SCF     1

R1S     PDSFP + PDSE + PDSF
R1F     1

VDF     1

DPS     PDS2 + PDS6
DPF     1

HLF     PDS2 + PDS6 + DP=S
HL1     1

IS1     HL=F * DP=F * (PDS3 + PDS4 + PDS7 + PDS8) * EF=F * FR=F
ISS     1

VHA     (PDS2 + PDS6) * PDSA
VHB     (PDS2 + PDS6) * (PDSC + PDSD)
VHC     (PDS3 + PDS7) * (PDSC + PDSD)
VHD     (PDS4 + PDS8) * PDSA
VHE     (PDS4 + PDS8) * PDSC
VHG     (PDS4 + PDS8) * PDSD
VHG     1

VIA     ((PDS2 + PDS6) * PDSA) * VH=F
VI1     (PDS2 + PDS6) * PDSA
VIB     (PDS2 + PDS6) * (PDSC + PDSD) * VH=F
VI1     (PDS2 + PDS6) * (PDSC + PDSD)
VIC     (PDS3 + PDS7) * PDSD * VH=F
VI3     (PDS3 + PDS7) * PDSD
VID     (PDS4 + PDS8) * PDSA * VH=F
VI3     (PDS4 + PDS8) * PDSA
VIE     (PDS4 + PDS8) * PDSC * VH=F
VI3     (PDS4 + PDS8) * PDSC
VIG     (PDS4 + PDS8) * PDSD * VH=F
VI3     (PDS4 + PDS8) * PDSD
VI3     1

CDF     HL=S
CD2     PDS3 + PDS4 + PDS7 + PDS8
CDF     1

CCA     (PDS2 + PDS4 + PDS6 + PDS8) * CD=F
CC1     PDS2 + PDS4 + PDS6 + PDS8
CC2     PDS3 + PDS7
CC2     1

CHF     (PDS3 + PDS4 + PDS7 + PDS8) * CD=S
CHS     1

CYF     CH=F
CYA     ((PDS2 + PDS6) * VH=F) + ((PDS3 + PDS4 + PDS7 +PDS8) * CD=F * VH=F)
CY2     (PDS2 + PDS6) * PDSA
CY3     (PDS2 + PDS6) * (PDSC + PDSD)
CY7     (PDS3 + PDS7) * PDSD * CH=S
CY4     (PDS4 + PDS8) * PDSA * CH=S
CY5     (PDS4 + PDS8) * PDSC * CH=S
CY6     (PDS4 + PDS8) * PDSD * CH=S
CY1     1

CNA     (PDS2 + PDS6) * PDSA * CY=F
CN1     (PDS2 + PDS6) * PDSA
CNB     (PDS2 + PDS6) * (PDSC + PDSD) * CY=F
CN2     (PDS2 + PDS6) * (PDSC + PDSD)
CND     (PDS3 + PDS7) * PDSD * CH=F
CNC     (PDS3 + PDS7) * PDSD * CY=F
CN3     (PDS3 + PDS7) * PDSD
```

Split Fraction Logic for General Transient Containment Response Event Tree: CONTGT (Used for non-recovered initiating events)

SF..... Split Fraction Logic.....

```
CNG     (PDS4 + PDS8) * PDSA * CH=F
CNE     (PDS4 + PDS8) * PDSA * CY=F
CN4     (PDS4 + PDS8) * PDSA
CNI     (PDS4 + PDS8) * PDSC * CH=F
CNH     (PDS4 + PDS8) * PDSC * CY=F
CN5     (PDS4 + PDS8) * PDSC
CNK     (PDS4 + PDS8) * PDSD * CH=F
CNJ     (PDS4 + PDS8) * PDSD * CY=F
CN6     (PDS4 + PDS8) * PDSD
CN6     1

REF     1

LHB     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * CC=F
LHA     (PDS2 + PDS4+ PDS6 + PDS8) * PDSA * (VH=F + CH=F)
LH1     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA
LHD     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * CC=F
LHC     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * (VH=F + CH=F)
LH2     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD)
LH2     1

LSC     (PDS2 + PDS6) * PDSA * CC=F * LH=F
LSB     (PDS2 + PDS6) * PDSA * CC=F * LH=S
LSA     (PDS2 + PDS6) * PDSA * CC=S * LH=F
LS1     (PDS2 + PDS6) * PDSA
LSD     (PDS2 + PDS6) * (PDSC + PDSD) * CC=F
LS2     (PDS2 + PDS6) * (PDSC + PDSD)
LSE     (PDS3 + PDS7) * PDSD * CC=F * LH=F
LS3     (PDS3 + PDS7) * PDSD * CC=F * LH=S
LS2     (PDS3 + PDS7) * PDSD * CC=S
LSI     (PDS4 + PDS8) * PDSA * CC=F * LH=F
LSH     (PDS4 + PDS8) * PDSA * CC=F * LH=S
LSG     (PDS4 + PDS8) * PDSA * CC=S * LH=F
LS4     (PDS4 + PDS8) * PDSA
LSJ     (PDS4 + PDS8) * (PDSC + PDSD) * CC=F
LS5     (PDS4 + PDS8) * (PDSC + PDSD)
LS5     1

LM2     (PDS3 + PDS7) * PDSD
LMA     (PDS2 + PDS4 + PDS6 + PDS8) * CC=F
LM1     (PDS2 + PDS4 + PDS6 + PDS8)

SMF     CH=F * CN=F
SMA     (PDS2 + PDS6) * PDSA * LS=F
SM1     (PDS2 + PDS6) * PDSA
SMB     (PDS2 + PDS6) * (PDSC + PDSD) * LS=F
SM2     (PDS2 + PDS6) * (PDSC + PDSD)
SMC     (PDS3 + PDS7) * PDSD * LS=F
SM3     (PDS3 + PDS7) * PDSD
SMD     (PDS4 + PDS8) * PDSA * LS=F
SM4     (PDS4 + PDS8) * PDSA
SME     (PDS4 + PDS8) * (PDSC + PDSD) * LS=F
SM5     (PDS4 + PDS8) * (PDSC + PDSD)
SM5     1
```

Split Fraction Logic for SLBI Containment Response Event Tree: CONTSB

SF..... Split Fraction Logic.....

```
SCS     SU=S + LT=S * ((RP=S * RC=S) + (RP=F * (L5=S + L6=S))) +RT=F
SCF     1

R1S     PDSFP + PDSE + PDSF
R1F     1

VDF     1

DPS     PDS2 + PDS6
DPF     1

HLF     PDS2 + PDS6 + DP=S
HL1     1

IS1     HL=F * DP=F * (PDS3 + PDS4 + PDS7 + PDS8) * EF=F
ISS     1

VHA     (PDS2 + PDS6) * PDSA
VHB     (PDS2 + PDS6) * (PDSC + PDSD)
VHC     (PDS3 + PDS7) * (PDSC + PDSD)
VHD     (PDS4 + PDS8) * PDSA
VHE     (PDS4 + PDS8) * PDSC
VHG     (PDS4 + PDS8) * PDSD

VIA     ((PDS2 + PDS6) * PDSA) * VH=F
VI1     (PDS2 + PDS6) * PDSA
VIB     (PDS2 + PDS6) * (PDSC + PDSD) * VH=F
VI1     (PDS2 + PDS6) * (PDSC + PDSD)
VIC     (PDS3 + PDS7) * PDSD * VH=F
VI3     (PDS3 + PDS7) * PDSD
VID     (PDS4 + PDS8) * PDSA * VH=F
VI3     (PDS4 + PDS8) * PDSA
VIE     (PDS4 + PDS8) * PDSC * VH=F
VI3     (PDS4 + PDS8) * PDSC
VIG     (PDS4 + PDS8) * PDSD * VH=F
VI3     (PDS4 + PDS8) * PDSD

CDF     HL=S
CD2     PDS3 + PDS4 + PDS7 + PDS8
CDF     1

CCA     (PDS2 + PDS4 + PDS6 + PDS8) * CD=F
CC1     PDS2 + PDS4 + PDS6 + PDS8
CC2     PDS3 + PDS7

CHF     (PDS3 + PDS4 + PDS7 + PDS8) * CD=S
CHS     1

CYF     CH=F
CYA     ((PDS2 + PDS6)*VH=F) + ((PDS3 + PDS4 + PDS7 +PDS8)* CD=F * VH=F)
CY2     (PDS2 + PDS6) * PDSA
CY3     (PDS2 + PDS6) * (PDSC + PDSD)
CY7     (PDS3 + PDS7) * PDSD * CH=S
CY4     (PDS4 + PDS8) * PDSA * CH=S
CY5     (PDS4 + PDS8) * PDSC * CH=S
CY6     (PDS4 + PDS8) * PDSD * CH=S
CY1     1

CNA     (PDS2 + PDS6) * PDSA * CY=F
CN1     (PDS2 + PDS6) * PDSA
CNB     (PDS2 + PDS6) * (PDSC+ PDSD) * CY=F
CN2     (PDS2 + PDS6) * (PDSC + PDSD)
CND     (PDS3 + PDS7) * PDSD * CH=F
CNC     (PDS3 + PDS7) * PDSD * CY=F
CN3     (PDS3 + PDS7) * PDSD
CNG     (PDS4 + PDS8) * PDSA * CH=F
CNE     (PDS4 + PDS8) * PDSA * CY=F
CN4     (PDS4 + PDS8) * PDSA
CNI     (PDS4 + PDS8) * PDSC * CH=F
```

Split Fraction Logic for SLBI Containment Response Event Tree: CONTSB

SF..... Split Fraction Logic.....

```
CNH     (PDS4 + PDS8) * PDSC * CY=F
CN5     (PDS4 + PDS8) * PDSC
CNK     (PDS4 + PDS8) * PDSD * CH=F
CNJ     (PDS4 + PDS8) * PDSD * CY=F
CN6     (PDS4 + PDS8) * PDSD

REF     1

LHB     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * CC=F
LHA     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA * (VH=F + CH=F)
LH1     (PDS2 + PDS4 + PDS6 + PDS8) * PDSA
LHD     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * CC=F
LHC     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD) * (VH=F + CH=F)
LH2     (PDS2 + PDS3 + PDS4 + PDS6 + PDS7 + PDS8) *(PDSC + PDSD)

LSC     (PDS2 + PDS6) * PDSA * CC=F * LH=F
LSB     (PDS2 + PDS6) * PDSA * CC=F * LH=S
LSA     (PDS2 + PDS6) * PDSA * CC=S * LH=F
LS1     (PDS2 + PDS6) * PDSA
LSD     (PDS2 + PDS6) * (PDSC+ PDSD) * CC=F
LS2     (PDS2 + PDS6) * (PDSC + PDSD)
LSE     (PDS3 + PDS7) * PDSD * CC=F * LH=F
LS3     (PDS3 + PDS7) * PDSD * CC=F * LH=S
LS2     (PDS3 + PDS7) * PDSD * CC=S
LSI     (PDS4 + PDS8) * PDSA * CC=F * LH=F
LSH     (PDS4 + PDS8) * PDSA * CC=F * LH=S
LSG     (PDS4 + PDS8) * PDSA * CC=S * LH=F
LS4     (PDS4 + PDS8) * PDSA
LSJ     (PDS4 + PDS8) * (PDSC + PDSD) * CC=F
LS5     (PDS4 + PDS8) * (PDSC + PDSD)

LM2     (PDS3 + PDS7) * PDSD
LMA     (PDS2 + PDS4 + PDS6 + PDS8) * CC=F
LM1     (PDS2 + PDS4 + PDS6 + PDS8)


SMF     CH=F * CN=F
SMA     (PDS2 + PDS6) * PDSA * LS=F
SM1     (PDS2 + PDS6) * PDSA
SMB     (PDS2 + PDS6) * (PDSC + PDSD) * LS=F
SM2     (PDS2 + PDS6) * (PDSC + PDSD)
SMC     (PDS3 + PDS7) * PDSD * LS=F
SM3     (PDS3 + PDS7) * PDSD
SMD     (PDS4 + PDS8) * PDSA * LS=F
SM4     (PDS4 + PDS8) * PDSA
SME     (PDS4 + PDS8) * (PDSC + PDSD) * LS=F
SM5     (PDS4 + PDS8) * (PDSC + PDSD)
```

F.2.1 PLANT DAMAGE STATE BINNING LOGIC RULES-PLANT MODEL EVENT TREES

Binning Logic for General Transient Long-Term Response Event Tree: LTGT

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=F
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP
ERROR	1

Binning Logic for Small LOCA Long-Term Response Event Tree: LTSLOCA

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=F
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP

Binning Logic for Medium LOCA Long-Term Response Event Tree: LTMLOCA

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=F
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP
ERROR	1

Binning Logic for Large LOCA Long-Term Response Event Tree: LL2

Bin.....	Binning Rules.....
S	HE=S + HS=S + RT=F
PDS1D	PDS1 * PDSD
PDS1F	PDS1 * PDSF
PDS1FP	PDS1 * PDSFP
PDS1FA	PDS1 * PDSFA
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS2FA	PDS2 * PDSFA
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS6FA	PDS6 * PDSFA

Binning Logic for SLBI Long-Term Response Event Tree: LTSLBI

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=F
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP

Binning Logic for Steam Generator Tube Rupture Long-Term Response Event Tree: LTSGTR

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=F
PDS9A	((XC=S + XD=S)+(VA=S + VB=S))*SLEAK
PDS9C	((XA=S + XB=S)*-(VA=S + VB=S))*SLEAK
PDS9D	C2=F * SLEAK
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP
ERROR	1

Binning Logic for ATWS Long-Term Response Event Tree: LTATWS

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) + RT=S
PDS1D	PDS1 * PDSD
PDS1F	PDS1 * PDSF
PDS1FP	PDS1 * PDSFP
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP

Binning Logic for Interfacing System LOCA (Suction Line) Event Tree: VS

Bin.....	Binning Rules.....
S	O1=S*(O3=S + CS=F*VC=S*-O3=F) + O1=F*CS=F*VC=S*RS=S + RT=F
LOCA	LE=S*VO=S + SI=S
FV1	VO=F + PI=F
FPV7	O1=S*VC=S*O3=F + O1=F*(VC=S*-CS=F + CS=F*RS=F*VC=S) + LZ=F*VC=S
FPV1	VC=F*-O3=S
ERROR	1

Binning Logic for Interfacing System LOCA (Injection Line) Event Tree: VI

Bin..... Binning Rules.....

S	O2=S*-O3=F + O2=F*(O3=S + CS=F*VC=S*-O3=F) +O1=F*CS=F*RS=S*VC=S + RT=F
LOCA	LE=S*VO=S + SI=S
FV1	VO=F + PI=F
PDS8C	O2=S*CS=S*O3=F
PDS7D	O2=S*(CS=F*O3=F + CS=B)
FPV7	VC=S*O3=F + O1=F*(VC=S*-CS=F + VC=S*RS=F*CS=F) + LZ=F*VC=S
FPV1	VC=F*-O3=S

Binning Logic for Recovery Event Tree: RECOVERY

Bin.....	Binning Rules.....
S	SU=S + LT=S * ((RP=S * RC=S)+(RP=F *(L5=S + L6=S))) +ER=S*(POWERA+POWERB) + RT=F + INIT=SLOCA * RM=S
PDS2A	PDS2 * PDSA
PDS2C	PDS2 * PDSC
PDS2D	PDS2 * PDSD
PDS2E	PDS2 * PDSE
PDS2F	PDS2 * PDSF
PDS2FP	PDS2 * PDSFP
PDS3D	PDS3 * PDSD
PDS3F	PDS3 * PDSF
PDS3FP	PDS3 * PDSFP
PDS4A	PDS4 * PDSA
PDS4C	PDS4 * PDSC
PDS4D	PDS4 * PDSD
PDS4E	PDS4 * PDSE
PDS4F	PDS4 * PDSF
PDS4FP	PDS4 * PDSFP
PDS6A	PDS6 * PDSA
PDS6C	PDS6 * PDSC
PDS6D	PDS6 * PDSD
PDS6E	PDS6 * PDSE
PDS6F	PDS6 * PDSF
PDS6FP	PDS6 * PDSFP
PDS7D	PDS7 * PDSD
PDS7F	PDS7 * PDSF
PDS7FP	PDS7 * PDSFP
PDS8A	PDS8 * PDSA
PDS8C	PDS8 * PDSC
PDS8D	PDS8 * PDSD
PDS8E	PDS8 * PDSE
PDS8F	PDS8 * PDSF
PDS8FP	PDS8 * PDSFP
ERROR	1

F.2.2 RELEASE CATEGORY BINNING LOGIC RULES-CONTAINMENT RESPONSE EVENT TREES

Binning Logic for General Transient Containment Response Event Tree: CONTGR (Used for recovered initiating events)

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3 + PDS7) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for General Transient Containment Response Event Tree: CONTGT (Used for non-recovered initiating events)

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3 + PDS7) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for SGTR Containment Response Event Tree: CONTSG

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S7B	(PDS9A + PDS9C + PDS9D) * R1=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S + CN=S + LS=F +((PDS3 + PDS7) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for Small LOCA Containment Response Event Tree: CONTSL

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3 + PDS7) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for Medium LOCA Containment Response Event Tree: CONTML

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for Large LOCA Containment Response Event Tree: CONTLL

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F) + INIT=APC + INIT=TMLL
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP +PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS1) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for ATWS Containment Response Event Tree: CONTAT

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3 + PDS7) * R1=S)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F

Binning Logic for Interfacing System LOCA (Injection Line) Containment Response Event Tree: CONTVI

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S7A	FV1 * R1=S
S7B	(FPV1 + FPV7) * R1=S
LOC	LE=S*VO=S + SI=S
S1A	(VI=F*SM=F) + (CN=F * SM=F)
S2	(CN=F* SM=S) + (VI=F * SM=S)
S3A	VI=S * CN=S * LS=F * PDS7D
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S

Binning Logic for Interfacing System LOCA (Suction Line) Containment Response Event Tree: CONTVS

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
LOC	LE=S*VO=S + SI=S
S1A	(VI=F*SM=F) + (CN=F * SM=F)
S2	(CN=F* SM=S) + (VI=F * SM=S)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	FV1 * R1=S
S7B	(FPV1 + FPV7) * R1=S

Binning Logic for SLBI Containment Response Event Tree: CONTSB

Bin.....	Binning Rules.....
SUC	SC=S + VD=S
S6	PDSF * R1=S
S1A	(VI=F * SM=F) + (CN=F * SM=F)
S2	(CN=F * SM=S) + (VI=F * SM=S) + ((PDSFP + PDSE) * R1=S)
S3A	VI=S * CN=S * LS=F * ((PDS3 + PDS7) * PDSD)
S3B	VI=S * CN=S * LS=F
S4	LM=F
S5	VI=S * CN=S * LS=S
S7A	IS=F