Volume 1

WATTS BAR
NUCLEAR PLANT UNIT 1
PROBABILISTIC RISK ASSESSMENT
INDIVIDUAL PLANT EXAMINATION

PLG

TVA PROBABILISTIC RISK ASSESSMENT TEAM

Project Management

Richard J. McMahon — Corporate Engineering
W. Douglas Stevens — Site Project Manager
Susan D. Ferrell — Site Project Engineer

Task Leaders

Ara H. Djirikian/Raj I. Paul — Systems Analysis
Carla A. Borrelli — Data Analysis
J. Stephen Clements — Level 1 Analysis
William Z. Mims, Jr. — Level 2 Analysis

PLG PROBABILISTIC RISK ASSESSMENT TEAM

Project Management

B. John Garrick — Project Director
Jack W. Read — Project Manager
David H. Johnson — Technical Manager

Task Leaders

James C. Lin — Systems Analysis
Shobha B. Rao — Data Analysis
Donald J. Wakefield — Level 1 Analysis
R. Kenneth Deremer — Level 2 Analysis
Shawn S. Rodgers — Site Representative
Andrew A. Dykes — Human Factors Analysis

Technical Contributors

David A. Bidwell
Kurt Ching
Michael A. Emerson
Vincent S. Ho
Yung-Ming Hou
Eugene S. Kim
Wee Tee Loh
Subodh R. Medhekar
Thomas J. Mikschl
James H. Moody
Kathleen A. Ramp
Grant A. Tinsley

TVA Consultants

James K. Liming — ERIN
D. William Stillwell — ERIN

PLG Consultants

Marc Kenton — GKA
Jeff Gabor — GKA
Donald A. Wesley — EQE

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1. EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This report documents the work performed by Tennessee Valley Authority (TVA) in accordance with the U.S. Nuclear Regulatory Commission (NRC) Generic Letter No. 88-20 (Reference 1-1), which requested each utility to perform an individual plant examination (IPE) to "(1) develop an appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur at its plant, (3) gain a more quantitative understanding of the overall frequencies of core damage and fission product releases, and (4) if necessary, reduce the overall frequencies of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents." To satisfy these requirements, a Level 2 probabilistic risk assessment (PRA) was performed for the Watts Bar Nuclear Plant. The PRA was performed by an integrated team of engineers and PRA specialists from TVA and PLG, Inc., with support from ERIN Engineering, Inc, and Gabor, Kenton and Associates. The PRA models are developed for Unit 1, and include the Unit 2 systems that are shared with Unit 1. The results are applicable to Unit 1 only; Unit 2 is still under construction.

TVA's overall objectives of the PRA program were to

- Meet the NRC requirements for IPEs as set forth in Generic Letter No. 88-20 and in NUREG-1335 (Reference 1-2).
- Develop a plant-specific Level 2 PRA model for Watts Bar based on the plant design as of December 1991*.
- Develop and apply databases for initiating event frequencies, component failure rates, maintenance unavailabilities, common cause failure parameters, and human error rates.
- Develop point estimate and uncertainty distribution results for the frequency of core damage and a full spectrum of radioactive release categories for Watts Bar.
- Determine the underlying risk controlling factors and key sources of uncertainty in developing the risk estimates to identify opportunities for safety enhancement.

The scope of the assessment is classified as a Level 2 PRA in which the accident sequences are developed sufficiently to define a set of radioactive material release categories and a definition of the source terms for radioactive release. This assessment is based on a set of initiating events that addresses internal events and internal plant floods.

The purpose of this summary is to present the results for the Level 2 PRA on Watts Bar. These results include an estimate of the core damage frequency; a quantification of uncertainties in this estimate; and a delineation of the key plant states and release

*Except as noted at the end of Section 1.4.1.

categories as well as the sequences, systems, and sources of uncertainty that are driving the results. In addition, information is provided on the nature, timing, and magnitude of potential releases of radioactive material based on the results of plant-specific analyses and NUREG-1150 results (References 1-3 and 1-4) for a sister plant, Sequoyah.

1.2 PLANT FAMILIARIZATION

The Watts Bar Nuclear Plant is located in Rhea County, Tennessee, approximately 50 miles northeast of Chattanooga and 31 miles north-northeast of TVA's Sequoyah Nuclear Plant. The plant is on the west shore of Chickamauga Lake on the Tennessee River. The plant consists of two units, each with an initial rated power level of 3,411 MWt. Unit 1, which is currently under construction, will be the lead unit for the plant.

Unit 1 is a four-loop pressurized water reactor (PWR) nuclear steam supply system furnished by Westinghouse Electric Corporation. Major structures at Watts Bar include two reactor buildings with ice condenser containments, a turbine building, an auxiliary building, a control building, a service and office building, two diesel buildings, an intake pumping station, and two natural draft cooling towers.

A detailed description of the plant site, facilities, and safety criteria is documented in the Watts Bar Final Safety Analysis Report (Reference 1-5).

1.3 OVERALL METHODOLOGY

The Watts Bar PRA is founded on a scenario-based definition of risk (Reference 1-6). In this application, "risk" is defined as the answers to three basic questions:

1. What can go wrong?
2. What is the likelihood?
3. What are the consequences?

Question 1 is answered with a structured set of scenarios that is systematically developed to account for design and operating features specific to Watts Bar. Question 2 is answered with a prediction or estimate of the frequency of occurrence of each Level 1 scenario identified in the answer to question 1. Since there is uncertainty in that frequency, the full picture of likelihood will be conveyed by a probability curve—a curve that conveys the state of knowledge, or confidence, about that frequency.

The third question is answered in a Level 2 PRA in terms of the key characteristics of radioactive material releases that could result from the scenarios identified. The results currently reported are for a Level 2 PRA, as defined in the IEEE/ANS "PRA Procedures Guide" (Reference 1-7) and Appendix A to Generic Letter No. 88-20.

A large fraction of the effort needed to complete a PRA is spent in the development of a model to define a set of accident sequences that is appropriate for the specific plant. An overview of the accident sequence model for Watts Bar is presented in Figure 1-1. This model contains a large number of scenarios that have been systematically developed from the point of initiation to termination. A series of event trees is used to systematically

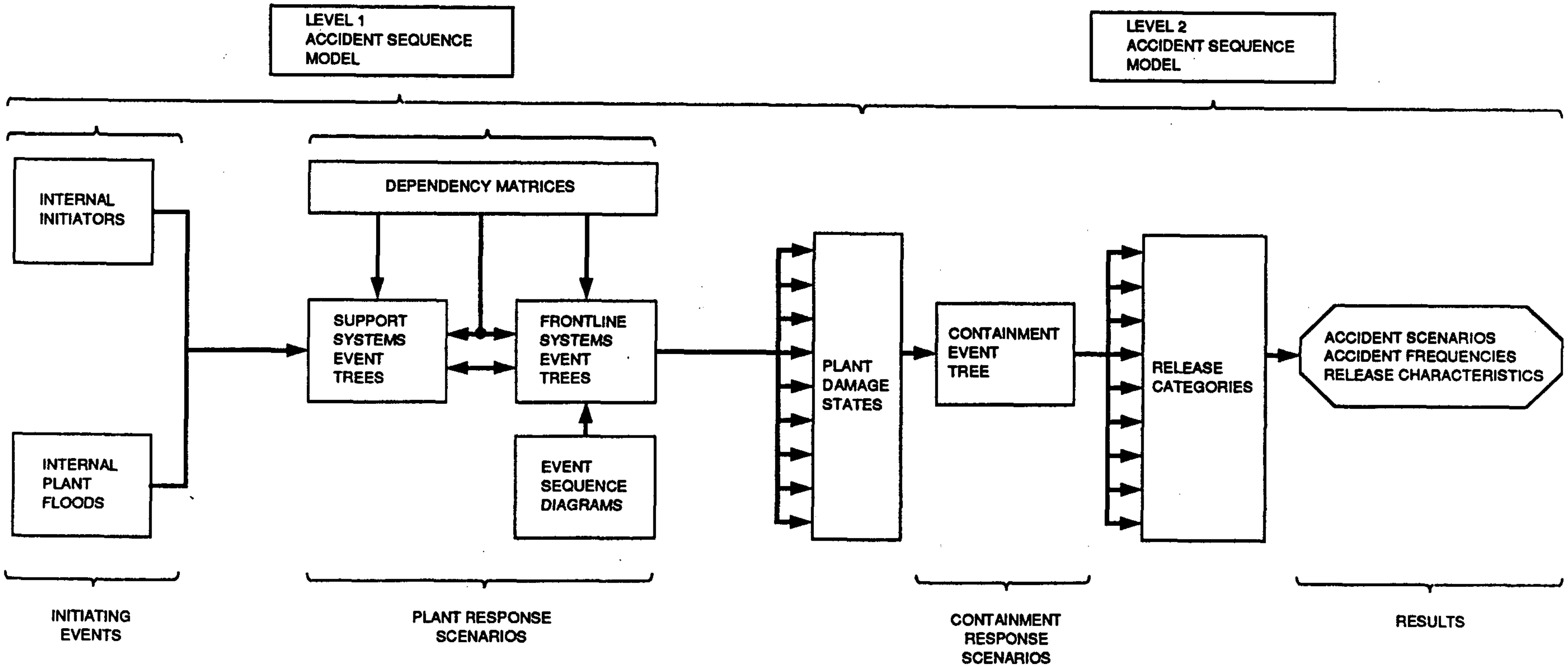

Figure 1-1. Definition of Accident Sequences in Watts Bar PRA

identify the scenarios. Given knowledge of the event tree structures, specific accident sequences can be uniquely identified by specifying:

1. The initiating event.
2. The plant response in terms of combinations of systems and operator responses.
3. The end state of the accident sequence.

As noted in Figure 1-1, a series of linked event trees is used for the analysis. The linking is accomplished within the RISKMAN® PC-based software system that effectively constructs a single, large tree for Level 1 and a second tree for Level 2. The Level I scenarios are linked together without the need for the use of support states or impact vectors to accomplish the linking, as was used in earlier versions of this methodology. The end states that are used to terminate the sequences are the plant damage states for the Level 1 part of the risk model and the release categories at the end of the Level 2 event trees.

The initiating events and the event tree branching frequencies are quantified using different types of models and data. The system failures that contribute to these events are analyzed with the use of fault trees that relate the initiating events and event tree branching frequencies to their underlying causes. These causes are quantified, in turn, by application of models and data on the respective unavailabilities due to hardware failure, common cause failure, human error, and test and maintenance unavailabilities.

Dependency matrices are developed from a detailed examination of the plant systems to account for important interdependencies and interactions that are highly plant specific. To facilitate a clear definition of plant conditions in the scenarios, separate stages of event trees are provided for the response of the support systems (e.g., electric power and cooling water), the frontline systems (e.g., auxiliary feedwater and containment spray), operator recovery actions, and containment phenomena; e.g., containment overpressurization failure. The latter stage of event trees is included in the Level 2 PRA. A detailed definition of plant damage states provides a clean interface between the Level 1 and Level 2 event trees.

The systematic, structured approach that was followed in constructing the accident scenario model provides assurance that plant-specific features will be identified. It also provides for the systematic, top-down development of engineering insights into the key risk controlling factors that drive the results. The current perspective of these results is provided below.

1.4 SUMMARY OF MAJOR FINDINGS

The major findings of the Watts Bar Level 2 PRA are presented in this section. These findings include the results of the Level 2 risk quantification, identification of the principal contributors to risk, and engineering insights into plant and operational features of Watts Bar that have been found to be important to safety.

The Level 2 PRA results were developed in two stages. In the first stage, the Level 1 models were evaluated separately to address the frequency of severe core damage, as described in Section 1.4.1. Contributors to the CDF are identified in Section 1.4.2.

The second stage of the results was obtained by quantification of the Level 2 (containment) event trees for the key sequences identified in the Level 1 models. These results and the major contributors to release frequency are delineated in Section 1.4.3. These results provide the basis for the conclusions regarding the safety significance of plant features and operator actions that are discussed in Section 6. Section 1.4.4 presents a comparison of Watts Bar IPE results with the NRC's NUREG-1150 study of Sequoyah.

1.4.1 RESULTS OF CORE DAMAGE FREQUENCY

The mean point estimate CDF for Watts Bar was found to be 3.3×10^{-4} per reactor-year.* No vulnerabilities were identified. The results for CDF were developed in terms of a mean point estimate, as required in NUREG-1335 (Reference 1-2). The more general format for presenting the CDF is in terms of an uncertainty distribution as shown in Figure 1-2.

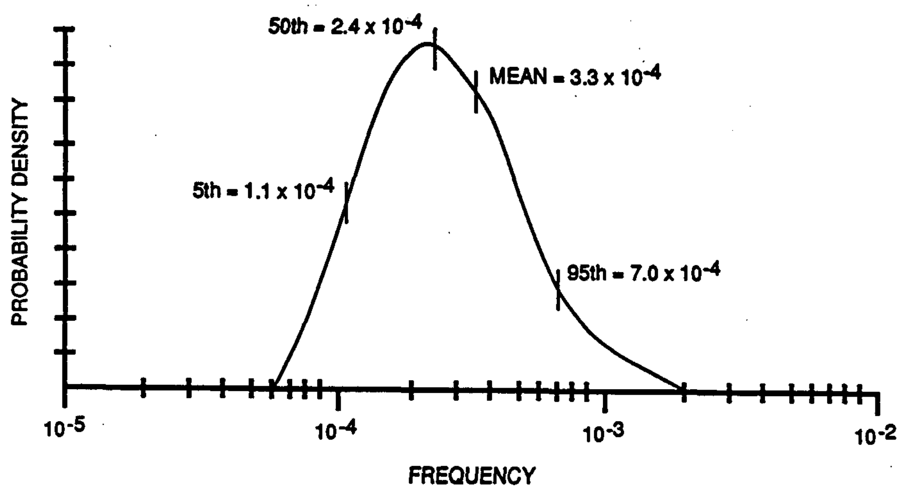

Figure 1-2. Uncertainty Distribution for Watts Bar Core Damage Frequency

The mean CDF for Watts Bar is on the order of corresponding results from PRAs on other plants that were derived from comparable methods, databases, and work scopes. The following table lists the published CDF results for other PWR plants prior to the implementation of a risk management program. These results include an assessment of internal events and internal floods, comparable with the results of this study.

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*The unit for the core damage frequency is events per nuclear-powered electric generating unit per calendar year. This definition is abbreviated to per reactor-year in this presentation.

Plant	Mean CDF (per reactor-year)
Three Mile Island (Reference 1-8)	4.4×10^{-4}
Beaver Valley Unit 2 (Reference 1-9)	1.9×10^{-4}
Seabrook (Reference 1-10)	1.7×10^{-4}
South Texas Project (Reference 1-11)	1.7×10^{-4}
Diablo Canyon (Reference 1-12)	1.3×10^{-4}

Factors that contribute to the nature of the results for Watts Bar are summarized below:

- The accident sequences that were analyzed are limited to those initiated by internal events and internal floods, in accordance with IPE requirements. Sequences initiated by internal fires, seismic events, and other external events have not been considered in this report.

- The current results do not reflect any plant or procedural changes that TVA may decide to make to improve safety after the IPE submittal.

- The results were obtained using industry generic data for failure rates, maintenance unavailabilities, and initiating event frequencies (Reference 1-13). The common cause parameters of the multiple Greek letter model were used in this study and were first estimated with the benefit of a plant-specific screening of industry common cause event data in accordance with NUREG/CR-4780 (Reference 1-14). Since Watts Bar is not yet operating, no plant-specific data were available for use in the development of the database parameters.

- This analysis is of the Watts Bar Nuclear Plant design as of December 1, 1991. One key exception to this cut-off, is the more recent design change (in progress) to have the shutdown boards continually fed from an offsite source, rather than from the unit boards which must transfer to an alternate source every plant trip. One other exception is the revised procedural guidance to check for ERCW flow to the diesel generators earlier then in the guidance as of 1991. This procedure was accounted for in the human factors analysis.

1.4.2 CONTRIBUTORS TO CORE DAMAGE FREQUENCY

In the quantification of the Level 1 event sequence models, the principal contributors to the CDF were identified from several vantage points. The results and contributors are summarized in Section 1.4.2.1. Section 3.4 describes the results in detail, along with the accident sequence screening process for sequences deemed to be reportable to the NRC as part of the IPE process. Causes to individual system failures are listed in each systems analysis notebook.

1.4.2.1 Important Core Damage Sequence Groups

The importance of initiating events was examined by determining the contributions of core damage sequences grouped by initiating event. The ranked results are shown in Figure 1-3 and Table 1-1 for eight major initiating event categories.

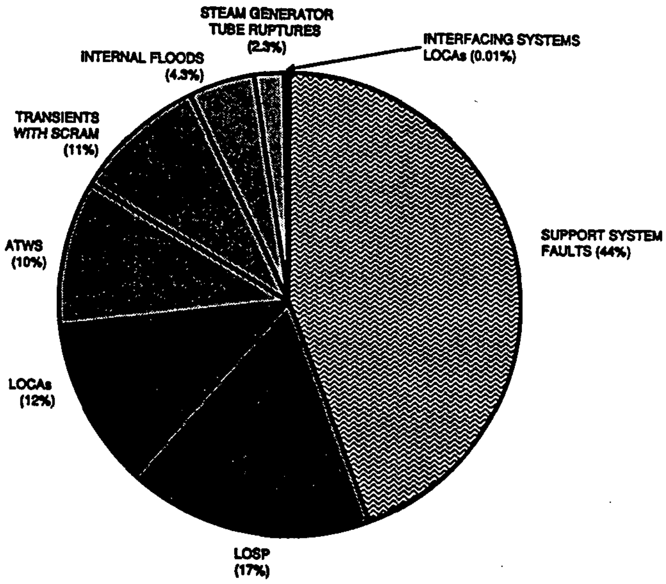

Figure 1-3. Categories of Watts Bar Initiators Contributing to Core Damage

Table 1-1. Sequence Group Contributions to Core Damage Frequency

Accident Sequence Group	Mean Annual CDF (per reactor-year)	Percentage of Total
Support System Faults	1.5×10^{-4}	44
Loss of Offsite Power	5.5×10^{-5}	17
Loss of Coolant Accidents	4.1×10^{-5}	12
Transients	3.4×10^{-5}	11
Anticipated Transient without Scram	3.2×10^{-5}	10
Internal Floods	1.4×10^{-5}	4
Steam Generator Tube Rupture	7.8×10^{-6}	2
Interfacing Systems LOCAs	4.1×10^{-8}	< 1
Total	3.3×10^{-4}	100

The general class of support system faults accounts for more than 44% of the total CDF. Included in this grouping are system and system train failures involving component cooling water (CCS), essential raw cooling water (ERCW), and electrical power boards. Loss of offsite power is separated from other support faults in this presentation. Many of the core damage sequences initiated by transients (e.g., turbine trips, reactor trips, and losses of main feedwater) also involve failure of key support systems.

Sequences initiated by loss of offsite power contribute 17% to the core damage frequency. These include unit blackout sequences and sequences in which only one train of shutdown power is available.

Loss of coolant accidents (LOCA) of different sizes and general transients make up more than another 23% of the CDF. Sequences involving transient-induced LOCAs (e.g., stuck-open pressurizer power-operated relief valve (PORV) in response to a loss of main feedwater initiator) are included with the transient initiating event category.

Sequences without reactor trip (ATWS) contribute 10%. Such sequences may lead to core damage if there is insufficient emergency boration, or if the initial RCS pressure transient is not mitigated.

Internal floods make up about 4%. The most important sources of internal floods are associated with the essential raw cooling water system.

Steam generator tube ruptures and interfacing system LOCAs make up only a small part of the total CDF. These initiators, should they lead to core damage, are significant because of their potential for a release path to bypass the containment.

Table 1-2 summarizes the contribution of RCP seal failures to the CDF at Watts Bar. About 70% of the CDF involves some form of RCP seal failure. RCP seal failures are divided into four categories, as shown in the table. Unit blackouts make up only a relatively small part of the total. The failure of component cooling water also accounts for a large portion of the core damage frequency. Loss of just train A of CCS with failure of the operator to trip the RCPs also makes up a significant fraction of the total.

Table 1-2. Watts Bar Contributors to Core Damage Involving RCP Seal Failures		
Case	Frequency per Reactor-Year	Percent of Total CDF
All RCP Seal Failures	**2.3-04**	**70**
Total Losses of Unit 1 CCS with Shutdown Power Available	1.0-04	30
Unit Blackouts	3.0-05	9
Failure To Trip RCPs, Given CCS Train A Lost, Resulting in an Early Seal LOCA	6.4-05	20
Other RCP Seal Failures Involving Loss of CCS Train A	3.6-05	11
Total Core Damage Frequency: 3.3-04 per reactor-year		
Note: Exponential notation is indicated in abbreviated form; e.g., 2.3-04 = 2.3×10^{-04}.		

1.4.2.2 Analysis of Individual Sequences

No single core damage sequence was found to dominate the total frequency of core damage. A large number of sequences make up the total CDF. Table 1-3 provides information on the distribution of core damage sequences across the frequency range.

Table 1-3. Breakdown of Core Damage Sequences in Each Frequency Range		
Frequency Range (events per year)	Number of Sequences	Percentage of CDF
$> 10^{-5}$	3	11
10^{-6} to 10^{-5}	43	32
10^{-7} to 10^{-6}	355	29
10^{-8} to 10^{-7}	2,081	18
$< 10^{-8}$	Very Large Number	10

The following presents a brief description of the highest ranking sequences to the core damage frequency. Each sequence either begins with a small LOCA or involves a loss of RCP seal integrity resulting in a small LOCA. Core damage then results from a failure of injection or sump recirculation.

- Total loss of CCS with failure to trip the RCPs, resulting in a small LOCA without sump recirculation.
- Total loss of CCS with failure to align backup cooling to centrifugal charging pump 1A-A; consequential RCP seal small LOCA (injection fails).

- Loss of essential raw cooling water; consequential RCP seal small LOCA (injection fails).
- Small LOCA with failure to align for sump recirculation.
- Loss of offsite power with failure of both unit diesel generators and no recovery before core damage; consequential RCP seal small LOCA (injection fails).
- Loss of train A of CCS with failure to trip the RCPs and failure of RHR pump 1B-B, consequential RCP seal LOCA and failure of sump recirculation.
- Total loss of CCS with failure of centrifugal charging pump 1A-A; consequential RCP seal small LOCA (injection fails).
- Small LOCA with failure of both RHR pumps and of makeup to the RWST; loss of sump recirculation.
- Loss of CCS train A with failure to trip the RCPs; consequential RCP seal small LOCA (injection fails) and failure to align for sump recirculation.
- Flood in ERCW train B strainer room with added failure of ERCW header 1A; consequential RCP seal small LOCA (injection fails).

Section 3.4 contains a detailed discussion of the top 10 sequences as well as a listing of the top 100 core damage sequences, in accordance with the NRC screening criteria for IPE reporting.

1.4.2.3 Important Operator Actions

The determination of contributions from sequences grouped by the occurrence of specific operator actions and system failure modes is termed an importance analysis. The importance analysis that was performed for Watts Bar is discussed more fully in Section 3.4.

The importance measure used here is the percentage of the total CDF associated with sequences in which the specific action was not performed in response to an initiating event. Table 1-4 summarizes the important operator action failures that appear in core damage sequences whose frequencies contribute at least 3% of the total CDF.

Table 1-4. Important Operator Actions	
Operator Action	Percentage Importance
1. Stop RCPs Following Loss of CCS Train A	19
2. Align ERCW to Charging Pump Following Loss of CCS Train A	8
3. Trip CRD Motor Generator Power and Initiate Emergency Boration Following ATWS	5
4. Align for High Pressure Recirculation	5
5. Isolate CCS Train A from Spent Fuel Pool Heat Exchanger	5
6. Makeup RWST Inventory Following LOCA without Sump Recirculation	4

The operator actions to recover electric power are not included in Table 1-4 because they are a complex function of the time available and the specific equipment failures involved. For comparison purposes, about 11% of the CDF involves failure to recover electric power in a station blackout before core damage.

Actions not included in this table are relatively unimportant as failures to perform these actions appear in sequences that contribute less than 3% of the total CDF. These importance measures are only meaningful in the context of the specific sequences in which these operator actions are required.

1.4.2.4 Important Plant Hardware Characteristics

An importance analysis of plant system failure modes to the total CDF was also performed. Only hardware failures involving the system itself are considered in Table 1-5.

Table 1-5. Important Systems	
System	Percentage Importance
Component Cooling Water System	37
Essential Raw Cooling Water System	21
Offsite and Common AC Power	17
Reactor Coolant System; i.e., LOCAs	17
Shutdown Boards (6.9-kV and 480V)	15
Auxiliary Feedwater System	13
Emergency Diesel Generators	12
Reactor Trip System	11
Main Feedwater and Condensate Systems	10
125V DC Power System	8

Here, importance means the percentage of the CDF involving failure of part or all of the indicated system. These importance measures are not strictly additive because multiple system failures may occur in the same sequence. The importance rankings account for failures within the systems that lead to a plant trip, or failures that limit the capability of the plant to mitigate the cause of a plant trip. Consequential failures resulting from dependencies on other plant systems (e.g., the loss of CCS due to failure of ERCW) are not included in this importance ranking. In addition, the failures of key operator actions are not included in Table 1-5 totals.

The top systems by importance are listed above. A further discussion of system importance and the results of additional sensitivity analysis is provided in Section 3.4.

1.4.2.5 Plant Damage States

One output from the Level 1 event sequence model is a description of the plant damage state (e.g., RCS pressure, containment isolation) at the time of core damage. This plant damage state strongly influences the performance of the containment and magnitude of fission product release that are assessed in the Level 2 analysis, as described in Section 4.

Table 1-6 states the frequencies of different plant damage states associated with core damage. The results in Table 1-6 only account for the impact on containment integrity of the accident sequence up to the time of core damage. As examples, the Level 1 analysis considers failures to isolate containment penetrations, containment bypass from steam generator tube rupture, and preexisting leaks. Table 1-6 results do not account for challenges to containment from severe accident phenomena; e.g., hydrogen burns, concrete degradation, which are analyzed in Level 2.

As shown in Table 1-6, over 96% of core damage events would be associated with an intact containment, and approximately 4% would be associated with small and large violations of containment integrity. Core damage sequences accompanied by a loss of containment integrity lead to greater fission product releases.

Table 1-6. Core Damage Frequency Breakdown for Watts Bar by Major PDS Group

Containment State	Frequency per Reactor-Year	CDF (percent)
Isolated and Not Bypassed	3.2×10^{-4}	96
Not Isolated or Bypassed — Small Leak	1.5×10^{-5}	4
Not Isolated or Bypassed — Large Leak	6.0×10^{-7}	<1
Total	3.3×10^{-4}	**100**

1.4.3 RESULTS FOR RELEASE FREQUENCY

The purpose of this section is to present the results of the Watts Bar Level 2 IPE in terms of the frequencies of fission product releases into the environment; (release category frequencies). These results are based on the integration of the Level 1 ("front-end" or "plant") model in which the responses of the plant systems and operators are addressed, and the "back-end" model whose containment event tree defines the outcome of the core damage scenarios in terms of the timing of the containment response and the magnitude of the release of radioactive material (referred to as source terms). The extension of the front-end analysis to include back-end analyses is called a Level 2 analysis.

There is a continuum of possible releases that could result from a core damage event. A reasonable treatment of this continuum is to use a representative set of discrete release categories that span the spectrum from relatively large, early releases to ones which are much smaller, occur later, and/or over a long time period. A detailed definition of the Watts Bar release categories is given in section 4.9. Table 1-7 represents a summary of these release categories in terms of general release category groups and percentage of the CDF.

Table 1-7. Definition and Results for General Release Category Groups

General Release Category Group	Description	Percentage of CDF Analyses*
I	Large, Early Containment Failures and Large Bypasses**	2
II	Small, Early Containment Failures and Small Bypasses**	4
III	Late Releases and Long-Term Releases	29
IV	Long-Term, Contained Releases (containment intact following vessel breach)	65

*Group frequency divided by CDF.
**The term "bypass" refers to the consideration when a release path from the RCS bypasses the containment and releases directly to the environment or into the auxiliary building, [e.g., faulted Steam Generator Tube Rupture (SGTR)].

Early fatality risk is dominated by General Release Group I and certain bypasses from Group II. As indicated in Table 1-7, the frequency of Category I releases for Watts Bar is estimated to be 2% of the core damage frequency, or 6.9×10^{-6} per reactor-year.

Release Category Group II is dominated by sequences involving an SGTR or other initiator followed by a stuck-open secondary relief valve. While such events involve a bypass of relatively small flow area, the associated source term can be relatively large. As indicated in Table 1-7, the frequency of Release Category Group II is 4% or 1.5×10^{-5} per reactor-year of the total CDF. SGTR events account for 55% of the Group II frequency and are considered to be a large, early release.

Release Category Group III involves sequences leading to degraded containment performance, but which do not contribute significantly to early health risk. The releases associated with Release Category IV should be comparable to those from unmitigated design bases accidents.

The important benchmark for a Level 2 PRA is the frequency of large, early releases. For this IPE, the frequency of large, early releases is the sum of Release Category Group I plus that fraction (.55) of Release Category Group II that is associated with SGTR initiating events with a stuck-open secondary side relief valve. The Watts Bar frequency of large, early release is approximately 4% of the CDF, or 1.5×10^{-5} per reactor-year. This frequency is low and is dominated by containment bypass results from SGTRs, which accounts for approximately 8.3×10^{-6} per reactor year.

1.4.3.1 Contributors to Release Frequency

Table 1-8 lists the major contributors to large, early release. Included in this table is the type of event and the percentage contribution for each event.

Table 1-8 Major Contributors to Large, Early Release Frequency for Internal Events

Type of Event	Percent Contributions to Large, Early Release*
SGTRs (with bypass to the environment)	55
Containment Failure due to Direct Impingement	24
α-Mode Failure of Vessel/Containment	14
HPME/Hydrogen Burns at Vessel Breach	5
Hydrogen Burns/DDT before and after Vessel Breach	2
Interfacing System LOCAs	<1

*Sum of Release Category Group I and SGTR Bypasses from Group II.

As shown in Table 1-8, SGTR sequences contribute about 55% to the frequency of large early releases. Containment failure due to direct impingement of debris on the containment cylinder wall in the seal table room is the second largest contributor. Steam explosions, caused by the interaction of hot fuel with water (alpha-mode)** is the third largest contributor. Overpressurization of the containment from hydrogen burns, detonations, and high pressure ejection of molten fuel when the pressure vessel fails contribute only 7% to the frequency of large, early releases. Less than 1% of the large, early release is caused by interfacing systems LOCAs.

**Failure of both the vessel and containment due to missiles generated within the vessel due to in-vessel explosion following relocation of core debris to the bottom.

1.4.3.2 Important Plant Hardware Characteristics for Containment Performance

This report has confirmed the importance of the following plant hardware in assuming containment integrity for the low (6%) percentage of core damage which would result in large, early releases of radioactive material to the environment.

- Air Return Fans and Hydrogen Ignitcrs
- Ice Condenser
- Ability to Depressurize and Maintain Secondary Side Cooling
- Long-Term Capability to Quench Core Debris in the Reactor Cavity

Supplement No. 3 to Generic Letter No. 88-20 requests that "licensees with Ice Condenser Containments are expected to evaluate the vulnerability to interruption of power to the hydrogen ignitors as part of their IPE." The Level 2 analysis in Section 4, considered the unavailability of ignitors as a part of the overall assessment of containment performance. As concluded in Section 4.10, it was found that ignitor unavailability made a relatively small contribution to Release Categories III and IV.

1.4.4 COMPARISON OF WATTS BAR IPE WITH SEQUOYAH IPE AND NUREG-1150

A comparison of accident sequence group frequencies for Watts Bar with the corresponding groups evaluated for Sequoyah by TVA and for Sequoyah by the NRC in the NUREG-1150 work, is presented in Table 1-9. The total CDF evaluated in this study for Watts Bar is higher than those for Sequoyah. Relative to the TVA analysis for Sequoyah, the results for Watts Bar are higher for two key reasons. A hardware design difference is that the charging pumps at Watts Bar depends on component cooling for the lube oil coolers, instead of from ERCW cooling, as at Sequoyah. Other differences in the results between Sequoyah and Watts Bar are attributable largely to the difference in the assessment of operator actions, based on input from each station's operations department. Third, the model for Watts Bar was evaluated using generic data, which leads to higher core damage sequence frequencies for Watts Bar because some key Sequoyah-specific failure data were found to be lower than industry generic data.

A comparison between the results for Watts Bar and the NRC's results for Sequoyah is also provided in Table 1-9. The CDF for Watts Bar is again higher, for the reasons mentioned in the preceding paragraph. In addition, the current analysis for Watts Bar includes an assessment of the contribution from support system faults (e.g., loss of CCS, loss of ERCW, and loss of shutdown power) and internal flooding initiators. Support faults and internal floods were not evaluated in the NUREG-1150 study for Sequoyah, but were found to be important for Watts Bar and Sequoyah, in the TVA studies. The current study included a more thorough analysis of common cause failures. In Reference 1-15, with regard to the omission of the loss of CCS and ERCW initiators for Sequoyah, the following comment was noted: "Use of different component failures probabilities or common cause modeling assumption could lead to different conclusions about the importance of loss of CCS." A similar statement was made for ERCW.

Table 1-9. Comparison of Accident Sequence Group Frequencies

Accident Sequence Group*	Watts Bar IPE		Sequoyah (NUREG-1150)		Sequoyah IPE	
	CDF	Fraction of Total	CDF	Fraction of Total	CDF	Fraction of Total
Support System	1.5-04	0.44	--	--	8.2-05	0.48
LOSP	5.5-05	0.17	1.5-05	0.26	1.1-05	0.06
LOCAs	4.1-05	0.12	3.6-05	0.62	3.1-05	0.18
Transients**	3.4-05	0.11	2.6-06	0.05	2.7-05	0.16
ATWS	3.2-05	0.10	1.9-06	0.03	7.1-06	0.04
Floods	1.4-05	0.04	--	--	6.8-06	0.04
SGTR	7.8-06	0.02	1.7-06	0.03	6.8-06	0.04
Interfacing Systems LOCAs	4.1-08	< 0.01	6.5-07	0.01	9.6-09	< 0.01
Total	3.3-04	1.00	5.8-05	1.00	1.7-04	1.00

*Groups defined to be exclusive.
**Transients with successful reactor trip.

Note: Exponential notation is indicated in abbreviated form; e.g., 1.5-04 = 1.5×10^{-04}.

This study concludes that the frequency of interfacing systems LOCA is about an order of magnitude lower than the NUREG-1150 results for Sequoyah. The net difference is attributed mostly to differences in models and assumptions employed in the respective studies and, to a lesser extent, design differences between Watts Bar and Sequoyah. The differences in models and assumptions include the following:

1. This analysis carries a range of sizes of interfacing leaks ranging from within to beyond the capacities of RHR relief valves that could limit the degree of RHR pressurization. NUREG-1150 only considered one leak size beyond the relief valve capacity.

2. This analysis considered the capability of RHR piping to withstand overpressurization.

3. This study included both RHR injection and RHR suction paths for the event (a total of eight paths) whereas NUREG-1150 considered only the four injection paths.

4. This analysis employed a more realistic treatment of operator actions to isolate and terminate the interfacing leak.

The contribution to CDF from the loss of offsite power initiator in the current study is about a factor of four greater than that found in NUREG-1150 for Sequoyah. In large part, this difference is due to the different data and methods employed. In the NUREG-1150 analysis, the CDF initiated by loses of offsite power involve unit blackout. In this study for Watts Bar, core damage sequences initiated by losses of offsite power include unit blackouts and sequences involving loss of just one shutdown board, that combined with other failures, result in loss of RCP seal cooling.

Table 1-10 compares the frequencies of the various major release category groups as determined for the Watts Bar IPE with those calculated for Sequoyah in Reference 1-4.

The sum of the IPE results for Release Category Groups I and II compare favorably with those reported in NUREG-1150 for Sequoyah. Observed differences in frequencies can be attributed primarily to the difference in CDF between the IPE and the NUREG-1150 values.

Table 1-10. Comparison of Release Frequency Results

Release Category Group	Description	Mean Annual Frequency per Reactor-Year (percentage of CDF)	
		WATTS BAR IPE	NUREG-1150*
I	Large, Early Containment Failure and Large Bypasses	6.9×10^{-6} (2)	7.4×10^{-6} (13)**
II	Small, Early Containment Failure and Small Bypasses	1.5×10^{-5} (4)	
III	Late and Long-Term Releases	9.5×10^{-5} (29)	1.2×10^{-5} (21)
IV	Long-Term Contained Releases†	2.2×10^{-4} (65)	3.6×10^{-5} (64)

*Figure S.2, NUREG/CR-4551, Vol. 5, Rev. 1, Part I.
**Large and small failures were not distinguished. Includes no vessel breach, early containment failure (during core damage) contribution.
†Includes sequences arrested prior to vessel breach and containment intact following vessel breach.

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2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The objectives described in Section 1.1 were accomplished by the completion of a Level 2 **probabilistic risk assessment** (PRA) on the Watts Bar Nuclear Plant. Reference 2-1 describes the three levels of PRA work scopes, as follows:

- Level 1 considers the performance of the plant systems to the extent needed to resolve sequences to the point of success or core damage.
- Level 2 includes the Level 1 scope plus the issues of core and containment phenomenology to the extent needed to resolve sequences sufficiently to determine the point of release, timing, and magnitude of radioactive material released.
- Level 3 includes the Level 1 and 2 scope plus an assessment of offsite consequences to public health and property.

The study described in this report represents a Level 2 analysis. It includes the quantification of sequence frequencies that lead to core damage, an assessment of the **frequency** of a spectrum of **release categories**, together with information to describe the timing and magnitude of **source terms**, that could be expanded into a Level 3 PRA at a later date, if desired.

The scope of accident sequences that are included in the PRA is those sequences that are initiated by the internal events and internal floods in conformance with NUREG-1335 (Reference 2-2).

For Watts Bar, the PRA models are developed for Unit 1. Systems shared with Unit 2 are also included.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

NRC Generic Letter No. 88-20 (Reference 2-3), which was issued on November 23, 1988, requested that an individual plant examination (IPE) for severe accident vulnerabilities be performed and that the results of the examination be submitted to the U.S. Nuclear Regulatory Commission (NRC). Supplement No. 1 to Generic Letter No. 88-20 was issued August 29, 1989, announcing the availability of NUREG-1335, and requesting, in accordance with Generic Letter No. 88-20, a submittal, within 60 days, describing proposed programs for completing the IPEs. Supplement No. 2, which is relative to procedures for feed and bleed cooling, makeup to water tanks, has been included to the extent that these items are included in the existing procedures.

The TVA response to the key issues raised in the Generic Letter are as follows:

- **Methods of Examination.** This is a Level 2 PRA using current state-of-the-art methods consistent with NUREG/CR-2300 (Reference 2-1) and severe accident phenomenological issues discussed in Appendix 1 of the Generic Letter. A Level 2

PRA was one of the options identified as acceptable by the NRC in Generic Letter No. 88-20 for performing an IPE.

- **Summary.** This PRA is written according to the NUREG-1335 format and content, and provides a plant-specific examination of Watts Bar for vulnerabilities. TVA is using the PRA to develop an appreciation of severe accidents, understand the most likely sequences, and to gain a more quantitative understanding of core damage and release probabilities.

- **Examination Process.** TVA personnel, who are familiar with the details of design, controls, procedures, and system configurations, have been involved with the analysis and technical reviews, and in the development of the Watts Bar PRA models.

- **Internal Events.** This PRA includes a detailed treatment of internal **initiating events** and internal floods.

- **Resolution of Unresolved Safety/Generic Issues (relationship to USI A-45).** Decay heat removal systems have been included in the models, and the results show no significant vulnerabilities. No other unresolved safety/generic issues are judged to be resolved for Watts Bar as a result of this submittal (discussed in Section 3.4.5).

- **PRA Benefits.** This study is a Level 2 PRA. TVA recognizes the potential benefits of a PRA and its potential use in future evaluations.

- **Severe Accident Sequence Selection.** The results of the accident screening are presented in Section 3.4.2 (as described in Section 2.1.6 of NUREG-1335) for systemic sequences.

- **Use of IPE Results.** TVA has evaluated the results of this PRA, as noted in Section 6.

- **Documentation of Examination Results.** This summary report, with Appendices A through E and the system notebooks, provides the tier 2 documentation. The summary report alone satisfies the requirements for the IPE submittal in accordance with NUREG-1335.

2.3 GENERAL METHODOLOGY

This section summarizes the technical approach and methodology employed in the performance of the IPE for Watts Bar.

The key features of the technical approach are summarized as follows:

- The use of **event sequence diagrams** (ESD) to develop and document insights into plant behavior and operator responses to initiating events. The development of these ESDs with plant operations personnel was considered to be a prerequisite to modeling accident sequences.

- The development of detailed **dependency matrices** that segregate the plant systems into support and frontline systems, and that define dependencies and interactions that need to be accounted for in the PRA.

- The use of large linked **event trees** to model accident sequence progression from initiating event to plant damage states, with explicit representation of support systems, frontline systems, human response, and significant dependencies.

- The use of the Level 1 plant damage states as initiators for the quantification of the Level 2 event tree.

- The use of a front-end (Level 1)/back-end (Level 2) interface boundary that provides for an integrated treatment of the systems in the Level 1 event trees and reserves the Level 2 analysis primarily for the treatment of phenomenological issues associated with severe accidents and post-core damage accident management.

- The use of a moderate-size **containment event tree** (CET) that addresses the severe accident issues considered in NUREG-1150 (Reference 2-4) with a fraction of the number of event tree **top events** that were used in NUREG-1150.

- The use of a systems analysis approach that consistently treats **common cause events** in accordance with NUREG/CR-4780 (Reference 2-5), and includes consideration of test and maintenance alignments.

- The use of a modified version of the success likelihood index methodology (SLIM) to elicit the expert judgment of licensed plant operators to quantify human error probabilities (Reference 2-6).

- The use of a consistent approach to the treatment of uncertainties in the development of a generic PRA database and a quantification of uncertainties in the PRA results.

- The development and quantification of PRA models using the PC-based RISKMAN software, Version 3.0 (References 2-7 and 2-8).

The technical approach followed in this IPE is similar to that employed in several other IPEs, including those on Seabrook (Reference 2-9), South Texas (Reference 2-10), Diablo Canyon (Reference 2-11), Beaver Valley, Hatch, and TVA plants (Browns Ferry and Sequoyah). Several of these studies have already been reviewed by the NRC staff and their contractors. The safety evaluation reports (SER) on the results of these reviews performed recently for the South Texas PRA (Reference 2-12) and Diablo Canyon PRA (Reference 2-13) as well as the Staff Evaluation of the Seabrook IPE submittal (Reference 2-14) reflect a general appreciation by the NRC of the essential distinguishing features of this approach to PRA.

The following sections provide a brief outline of the methods followed in each technical area of the PRA. The key references for the detailed methods, software, and mathematical bases in each of several technical areas of PRA are located in Table 2-1.

2.3.1 BASIC PRA STRUCTURE

2.3.1.1 Questions Addressed in a PRA

The purpose of performing a PRA is to answer the following three basic questions:

1. What can go wrong during operation or maintenance of the plant that could result in an accident sequence that leads to severe core damage?

2. How likely are these sequences to occur during the plant lifetime?

3. What is the level of damage to the plant and to the environment that could result if severe core damage occurs?

The first question is answered in the form of a structured set of scenarios that is systematically developed to account for design and operating features specific to Watts Bar. This set of scenarios is generated using an accident sequence model consisting of a set of initiating events, linked event trees supported by event sequence diagrams, dependency matrices, and other analysis tools that define the progression of accidents. The intersystem dependencies are also accounted for in defining the response of the plant.

The second question is answered in terms of the frequency of occurrence of each scenario identified in the answer to question 1. This involves quantifying the frequencies of events that could initiate an accident sequence and the failure frequencies of the systems and operators that respond to mitigate against core damage or loss of containment integrity in response to these initiators. The frequency results are then combined in the accident sequence model to obtain the frequency of each accident scenario.

The quantification process for WBN is accomplished using data from actual experience and a wide variety of models that interpret and combine the data into failure frequencies of the events questioned in the accident sequence model. These quantification models provide the link between the overall risk and its underlying causes.

An important part of quantification is expressing our confidence in the frequencies calculated by the models for each of the events, which means displaying the uncertainty in each of the frequencies. This is done using **probability** distributions for the measurable quantities and expressing modeling uncertainties explicitly. Many sources of uncertainty are addressed in the development of the parameters used to quantify the frequency of accident sequences. Examples of these uncertainties are the source of generic data, plant-to-plant variability in the application of data, sample size for plant-specific applications, and accident progression phenomenological questions. The resulting state of knowledge or confidence in each frequency parameter is expressed by deriving a probability distribution for that parameter. The basic data distributions and selected modeling uncertainties are then propagated through the systems quantification models and the important event sequences to obtain uncertainties in the overall results.

Within the IPE scope of examination, question 3 is answered by delineating the key characteristics of radioactive material releases that could result from the scenarios that led to core damage. The analysis required to do this addresses the progression of physical containment responses following the onset of core damage. The PRA does not address offsite consequences such as public health effects. However, the containment response analysis provides the basic groundwork and the information needed to address offsite consequences.

2.3.1.2 Basic Structure for Answering the Three Questions

To answer the above questions, a scenario-based approached is employed for the Watts Bar IPE. This systematic framework for the definition of accident sequences is illustrated in Figure 2-1. The following paragraphs provide an overview of the framework. Important terms used in describing the risk assessment methodology are shown in bold letters when they first appear and are defined in Table 2-2.

Each sequence begins with a well-defined **initiating event.** For the initial condition of full-power operation, an initiating event is a failure or external event that requires the reactor or turbine trip function.

Each possible scenario that could result from an initiating event is defined in terms of a specific combination of three general categories of successful and unsuccessful plant responses that must function to bring the plant to a safe and stable shutdown condition.

- Frontline systems perform the basic safety functions of reactivity control, core and reactor coolant heat removal, and reactor coolant pressure and volume control. In addition, active systems needed to cool, isolate, and filtrate the containment as well as those needed to determine the possibilities for a containment bypass are considered to be frontline systems in the Level 1 portion of the scenario-defining event trees.

- Support systems provide the electric power, motive force, cooling functions, and control signals that enable other support systems and ultimately the frontline systems to perform their safety functions over the mission time required to achieve the safe and stable shutdown.

- Operator actions to initiate, align, control, recover, or inhibit any of the above systems as a necessary and integral part of performing the plant safety functions.

These three elements are organized into event sequence models. **Top events** define the functional requirements for the systems and operator actions addressed at each step in the event sequence models. The top events are defined so that success brings the plant closer to a safe and **stable shutdown** condition.

Because reactors are protected by reliable, diverse, and redundant safety systems, severe core damage requires a series of component and system failures, and possibly human failures. Actual operating experience and modeling techniques demonstrate that, although the likelihood of a series of failures is quite small, it is numerically higher than would be estimated solely from a postulated chain of independent failures. This is because physical and human interactions result in **dependent failures** that increase the conditional probability of each successive failure in the chain.

As a consequence of the functional dependence of one system or action on others, the failure frequency of a top event function at some point in the accident sequence may have a variety of values that are conditioned on the failures that preceded it. These conditional failure frequencies are explicitly represented by defining one or more **split fractions** for each top event that relate the results of that function's quantitative analysis under those conditions. The rules for assigning these split fractions to each branch point of every top event are then written in terms of the success or failure of earlier top events in the plant **event trees**. These trees are normally divided into support system event trees and frontline event trees for clarity during analysis. Aspects of the causes and consequences of each sequence can be displayed and accounted for in a way that permits the importance of dependencies between top events to be determined.

Individual Level 1 sequence frequencies are calculated using RISKMAN, which combines the elements of the accident sequence evaluation. The major parts of the integrated model are shown in Figure 2-2. Flow charts called **event sequence diagrams** graphically display how the progression of the accident sequence is being modeled to facilitate communication with plant personnel. Dependency matrices organize the functional dependencies among the systems into one central tool to assist in ensuring that such dependencies are accounted for.

The Level 1 sequences proceed to either a successful safe and stable shutdown or to core damage. The core damage sequences are further assigned to **plant damage states**. These states are defined to categorize the key plant conditions at the time of core damage that can influence the likelihood of success of the containment functions. These states consider the status of active systems that perform a function in the containment system.

The plant damage states form the initial conditions for the Level 2 analysis, which models the sequence of physical phenomena that determines the ultimate consequences with regards to release of radioactive materials from the containment. For some core damage sequences (i.e., those that involve preexisting leaks, bypasses, and failure to isolate the containment), the ultimate release state is already resolved in the Level 1 portion of the scenario definition. End states for the remaining core damage sequences are resolved by considering the phenomenological challenges to the isolated containment in the containment response analysis.

The complete core damage sequences (i.e., which include the Level 1 and the Level 2 containment response models) terminate with a release category end state. The release categories define the ultimate response of the containment and the resulting radioactive material release. In Figure 2-1, these categories are grouped into four major classes listed in decreasing level of severity as:

- I. Large, Early Containment Failure or Bypass
- II. Small, Early Containment Failure or Bypass
- III. Late Containment Degradation
- IV. Long-Term Intact Containment

In the Level 2 PRA performed in this examination, the event sequences include an extension of the Level 1 sequences resulting in core damage through a containment event tree that considers severe accident phenomena that could influence containment integrity and the ultimate severity of release.

The PRA model elements introduced above are described in more detail later in this section.

2.3.2 BASIC STEPS TO PERFORM A PRA

2.3.2.1 Overview

PRAs that are performed to meet the IPE requirements must complete the same basic steps before they are considered to be complete. These basic steps include

1. Develop Basic Plant Familiarization
2. Define Level 1 Event Sequences
3. Develop Models To Support Sequence Quantification
4. Develop PRA Database
5. Determine Severe Accident Progression and Release
6. Assemble and Quantify Accident Sequences
7. Review and Interpret Results

A summary of the basic steps and key products developed in the performance of this PRA that meet the above requirements is provided in Table 2-3. These products are developed within the framework of an integrated event sequence model, as illustrated in Figure 2-2. Although there is considerable iteration among the efforts to develop many of these products, they are listed in this table in the approximate sequence of their development.

The methods employed in the development of the key products are described in the remainder of this section. The treatment of dependent events is first described due to the importance that it has on many of the tasks.

2.3.2.2 Overall Treatment of Dependent Events

The WBN PRA models scenarios in terms of sequences of events. Dependencies among these events must be adequately represented. There are many important dependencies to be considered throughout the PRA process. For these reasons, the concepts for identification and analysis of significant dependencies outlined below must be kept in the forefront of attention while accomplishing the tasks of constructing and quantifying the PRA models.

2.3.2.2.1 Dependent Failure Mechanisms

The first requirement for an adequate treatment of dependent events is to be aware of the spectrum of dependency mechanisms. These mechanisms are discussed fully in Reference 2-5. The particular types of dependencies that have a significant impact on the PRA modeling process are summarized briefly below:

- **Functional Dependencies.** These important dependencies arise from the design and operation of the plant. Many common support systems provide support functions to several frontline systems and other support systems. If the support system fails, some or all of its supported systems may also fail because of the loss of power, cooling, actuation signal, or source of water inventory.

Although the identification of functional dependencies is straightforward and, for the most part, would naturally fall out of a basic understanding of the plant design, they are extremely important to model properly, as they are often found to be significant contributors to risk. For example, accident sequences involving failures of critical support systems such as service water, component cooling water, electric power, and room cooling tend to have a larger risk significance than sequences involving failures in frontline safety systems such as reactor trip, high pressure injection, or low pressure injection. The consequences of critical support system failures are more extensive because of the extent of functional dependencies on these systems.

- **Common Cause Events.** This term describes the situation in which two or more events occur within some short period of time as a result of some shared cause.

  Common cause events are troublesome because they involve unintended dependencies, often hidden, and therefore frequently difficult to find, and yet are often dominant contributors to system failures and accident sequences. Examples of common cause dependencies include

  - Latent and undiscovered design errors shared by two or more components.
  - External and internal environmental stresses (e.g., vibration) that impact two or more components at the same time.
  - Commonality in location of two or more components that can be influenced by one external event; i.e., external to the components.
  - Human errors associated with the calibration of equipment with a wrong parameter or faulty equipment.
  - Human errors in testing or maintaining equipment that unintentionally render two or more components unavailable at the same time.

  Although it is often difficult to identify potential common cause events before they occur, the methods and databases that are currently available (References 2-17 and 2-24) have matured to the point where the analyst can reasonably well predict how often common cause events occur in redundant systems and thereby can quantify their quantitative impacts on system performance. Engineering insights needed to reduce the frequency of these events are available through detailed examination of the underlying database.

- **Dynamic Human Interactions.** Human-dependent failure mechanisms arise from the ability of the plant operations staff to influence multiple systems, should a misconception of a situation be perceived and perpetuated. Examples include:

  - Misdiagnosis of a situation that precludes the operators from taking a series of actions in response to an initiating event.
  - Inability to act due to a previous failure.

As both the Three Mile Island and the Chernobyl accidents arose from dependencies among human actions that served to defeat a variety of redundant safety systems, the human action analysis of operator actions during accident sequences must consider the potential for these types of dependency mechanisms.

It is important to recognize that there is no sharp boundary between common cause events and human interactions. In fact, most common cause events could be considered to be due to some lack of foresight or error regarding human interactions during the design, construction, or operation of the facility. As will be seen below, the various dependency mechanisms are accounted for during the application of PRA tools in different ways. The important point is to be reasonable and complete but not duplicative.

2.3.2.2.2 Identification of Dependencies in the Event Sequence Model

Within an accident sequence model composed of an initiating event and a specific combination of system failures, there are three general categories for explicitly defining dependent events that are defined by the different ways dependent events can cause or contribute to an accident sequence. They are defined and discussed below. In addition, Table 2-4 shows how they relate to the dependency mechanisms discussed above and gives examples of each.

- **Common Cause Initiating Event.** This category of dependent event results when a single event causes both an initiating event and failure or degradation of one or more systems in the sequences following the initiating event. Examples include floods and various support system failures.

  Initiating events that produce a challenge that requires either reactor or turbine trip and also fail or degrade one or more systems important for safe shutdown are explicitly defined and quantified as separate sets of scenarios. Examples include loss of offsite power and loss of service water. Common cause initiating events are processed with the same accident sequence logic as other initiating events, with the exception that the impacted systems are guaranteed to fail. The accident sequence model can then predict how well the plant can respond to these degraded conditions.

- **Intersystem Dependency.** This category of dependent events arises from initiators involving functional dependencies. The linkage created by the dependency is made between two or more systems through the definition of conditional frequencies (i.e., split fractions) for the impacted system that are dependent on the status of the impacting system.

  Intersystem dependencies are modeled explicitly in the event trees and in the logic rules for the assignment of conditional event tree branching ratios, referred to as "split fractions," for event tree quantification. At the system level, **fault trees** are quantified separately for each split fraction to account for functional dependencies on other systems, where success or failure status is resolved in the event trees along the particular sequence paths leading up to the associated branch point.

  In the system fault trees, supporting systems are represented as "house events" that are either certain to occur or certain not to occur, depending on the event tree path. More details on the fault tree analyses approach are provided in Section 2.3.5.1.

- **Intrasystem (intercomponent) Dependency.** The dependent event in this case links together two or more items within a system and thereby increases the system failure probability and that of all affected sequences. These dependencies are in some cases treated in the fault trees explicitly. In other cases, they are treated implicitly through the use of alternative system alignments and parametric models for common cause failures.

2.3.2.2.3 Implementation of Dependency Evaluation throughout the PRA Process

The treatment of dependent events is included throughout virtually all aspects of PRA modeling and quantification. The principal methods employed in each step of the PRA process as practiced in this evaluation are summarized in Table 2-5. A review of this table reveals that the evaluation of dependencies must be an iterative process, where the process of accomplishing one task can bring insights into other tasks. This integration process is critical to the realistic evaluation of plant risk.

A detailed description of the methods used to treat dependent events and other important issues in PRA is found in the sections below that address the methods used for each of the basic steps of a PRA.

2.3.3 PLANT FAMILIARIZATION

The natural starting point to a successful PRA is the acquisition of an in-depth knowledge of the plant. This is performed in the early stages of the project and is accomplished by the completion of key products that document a qualitative evaluation of the plant. These products, which were developed with plant design and operations personnel prior to the actual development of PRA quantification models, include:

- **System Notebooks.** At the plant familiarization stage, the plant systems are screened for applicability to the PRA. Any system that could cause a plant trip or is needed to characterize the plant response to a plant trip is selected for further analysis. System notebooks are prepared for systems that satisfy the screening. These notebooks contain qualitative system descriptions including the details of system operation, test, and maintenance procedures; identification of key system interfaces, system diagrams, and drawings; technical specifications; and other qualitative information needed to begin the PRA. Qualitative system summaries for this evaluation are presented in Section 3.2.1. In later steps, the systems notebooks are augmented with the detailed system model descriptions and results.

- **Plant Walkdowns.** Initial plant walkdowns are performed by the members of the PRA team to visually inspect key areas of the plant and major plant systems and to resolve questions generated during the development of the plant familiarization products. Locations of plant equipment are documented to support the internal flooding analyses. Meetings are held with cognizant engineers and plant operations personnel to review the analyst's understanding of the plant and to obtain agreement on the dependency matrices.

- **System Dependency Matrices.** The significant functional dependencies identified in the system notebooks are organized into system dependency matrices. Dependency matrices are charts that contain supporting systems on a vertical axis and supported systems on a horizontal axis. Interdependencies and interactions between those systems are listed at the node corresponding to the two systems. A set of notes for the matrices along with the systems notebooks provides the necessary details to describe the complex intersystem dependencies and provide references to the appropriate systems notebook. The result is a centralized overview of the intersystems dependencies of the plant.

  Dependency matrices are used to support the following:

  - Document the classification of systems into support versus frontline systems to facilitate the construction of the event trees. Frontline systems perform a critical safety function, whereas the support systems provide a support function to one or more frontline systems.

  - Define important boundary conditions needed by the systems analyst to evaluate the system. These include definition of the cutoffs between supporting and supported functions and identification of intersystem dependencies that must be modeled.

  - Provide a convenient reference of how the plant is actually connected, against which simplifying assumptions made in developing the event trees can be compared.

  - Provide a useful tool in communicating with operations personnel about the analyst's understanding of the plant.

  - Serve as a useful check in the review of the dominant sequences that are produced in the event tree quantification to ensure that dependencies have been properly modeled.

  While, on the surface, these matrices may seem redundant to the information presented in the system descriptions, experience has shown that a proper understanding of plant behavior requires that the combined effects of many dependencies be organized in one place. The dependency matrices for this evaluation are presented in Section 3.2.3.

2.3.4 LEVEL 1 SEQUENCE DEFINITION

A large fraction of the effort needed to complete a PRA is spent in the development of the Level 1 accident sequence model and in the verification that it defines a complete and appropriate set of accident sequences. This section outlines the process used to accomplish this objective. First, however, the basic modeling tool for this development is described (also see Reference 2.1).

An event tree is a means of organizing the answers into a series of questions regarding the progression of plant status towards safe shutdown. An example of a simplified plant event tree is shown in Figure 2-3.

The event tree starts with an initiating event, I, on the left. Within the scope of the PRA, the initiating event perturbs the plant at a normal operating condition in a way that requires either a reactor or turbine trip. This starts a sequence of events that could lead to core damage if the plant safety systems do not function. Examples of initiators are a system failing, loss of offsite power, or a human error.

The tree then splits at branch points corresponding to answers to questions regarding the plant response to the initiating event. Depending on the answers to the questions, the branches terminate on the right with either a safe shutdown condition (designated "OK") or a plant damage state (designated "DS_n").

The branch points are called **top events.** A top event evaluates a question regarding the

- Conditions presented by the initiating event.
- Status of a system function against a set of success criteria and the conditions generated by the sequence of events leading to that point.
- Performance of the operating team within the evolving scenario.
- Occurrences of certain physical phenomena, should core damage occur (for the back-end or Level 2 event trees).

The order in which the top events appear is established by the precedence of initiating event, temporal, and intersystem dependencies. Those that impact the potential for success of others are listed to the left of the event tree. The frequency of each branch point may be dependent on the status of preceding top events in the sequence path.

A favorable outcome to a top event question proceeds to the right. One can associate that answer to progressing a step closer to a safe shutdown condition. An unfavorable outcome branches downward. This answer is normally associated with a step closer to core damage in that additional systems must function or some system asked later in the tree must function under more difficult conditions.

The fraction of time that the downward path is followed, given that the sequence up to that point has occurred, is called a split fraction. The split fraction is equivalent to the frequency that the function defined by the top event fails to be accomplished under the conditions determined by the progression of the scenario to that point. Split fractions are quantified as part of the answer to question 2 of the scenario-based definition of risk, "What is the likelihood?" which is discussed further in Section 2.3.5.

As indicated in Figure 2-3, a top event question may be asked after earlier top events have succeeded or failed. Each answer may have a different failure frequency that is dependent on the conditions presented by those outcomes. Within the sequence model, each set of conditions that is judged to change significantly the failure frequency of the function being questioned is assigned its own split fraction designation, normally delineated by a number attached to the end of the top event designation. For the example in the figure, Top Event B will be quantified under two sets of different conditions. Either the preceding Top Event A has occurred or not. The sequence S involves success of Top Event A and failure of Top Event B. The frequency of event sequence S is given by $\phi(S)$, and is obtained by multiplying the frequency of each branch point by the initiating event frequency.

Sometimes, conditions prior to a top event either guarantee it to succeed or fail, or make it not applicable to a particular sequence, warranting the assignment of a conditional frequency of 1 or 0 to the split fraction of a node. In this case, the event tree would not split into two branches at the top event but would simply pass through, thus reducing the number of sequences in the tree. Although not shown in this figure, this is an important feature that enables one general event tree to be used for a variety of initiating events. Section 3.3 discusses the construction and reduction of event trees in more detail.

An overview of the accident sequence model for Watts Bar is presented in Figure 2-2. A variety of analytical techniques are used to identify and structure the model to give reasonable assurance that it is realistically representing a complete set of possible accident sequences.

The process of defining accident sequences consists of completing the following basic steps:

1. Define initiating events and organize them into groups according to similar impacts on the plant and the systems needed to control the event.

2. Construct event sequence diagrams to cover all major initiating event groups. These diagrams map out alternative plant responses to the initiating events, operator actions that are called for in the emergency operating procedures, and important physical events to characterize the condition of the plant along alternative event sequences. The ESD is a means of documenting the key assumptions made regarding accident progression that are implicit in the event trees constructed in the next step. Therefore, event sequence diagrams are set up to communicate understanding regarding the plant and operational response with those who are not familiar with PRA nomenclature. While the ESDs contain the detail that eventually is described by the frontline event trees, they also contain additional events that are judged to be unimportant for inclusion in the event trees. The reasons for omitting these events are described in the presentation of the ESDs.

3. Construct sets of modularized event trees to define the possible accident sequences, from each initiating event group to a sequence end state. One or more separate event trees is provided for the response of support systems to the initiating events. Separate sets of event trees are developed for each major group of initiating events to model the response of the frontline systems and operator actions.

4. Develop definitions for each event tree top event. These definitions serve as a specification for the systems and human reliability analyses that will be performed to quantify the event tree split fractions. Success criteria are developed for each top event, and boundaries are established to determine which equipment and operator actions are to be modeled in each top event.

5. Define end states for each event tree sequence including a successful end state and a number of plant damage states to identify particular cases of core damage.

These plant damage states provide a key interface with the Level 2 or "back-end" analysis of severe accident phenomenon, as explained more fully in Section 2.3.6.

A more detailed discussion of each major element of the sequence definition process is provided below.

2.3.4.1 Identification of Initiating Events

In PRA evaluations, an initiating event is defined as any event that results in a plant transient condition or otherwise perturbs the normal operation of the plant such that, depending on the response of the plant systems and operations personnel, a sequence of events involving undesirable consequences could result. The undesirable consequence is damage or core melt. For the initial condition full-power operation such as this one, the plant transient condition is equated with a challenge to the reactor trip and/or turbine trip functions. In less sudden transients, such as controlled power reductions that do not induce trips, there is a high probability that plant operations personnel will effect an orderly plant shutdown. If an orderly plant shutdown is not achieved, an eventual reactor or turbine trip condition is assumed.

Initiating events analysis is carried out in the following sequence of steps:

- **Step 1 — Identification of Candidate Events.** A variety of experience-based and analytical approaches are used to identify candidate initiating events. With the increasing amount of operating experience that has accumulated and the large number of PRAs that have been accomplished, the primary source of candidate initiating events is a review of relevant industry experience and plant data. These include:

  - **Review of Reactor Operating Experience.** In this approach, both industry-and plant-specific operating experiences are reviewed and classified to enumerate the plant trips and initiating events that have actually occurred. Both the industry and the NRC have sponsored generic industry surveys of this experience. These surveys as well as plant-specific data are reviewed and factored into the evaluation.

  - **Feedback from other PRA Analysis Tasks.** As important knowledge regarding plant operation, system dependencies, plant behavior, and system failure modes is built up in other PRA tasks, new insights into important and unique initiating events are invariably developed. This feedback is especially useful in the qualitative evaluation of candidate initiators and also provides a useful check on completeness in the candidate events identified using the above techniques.

  - **Failure Modes and Effects Analysis (FMEA).** Failure modes and effects analysis is a bottom-up approach to identifying initiating events. The approach consists of evaluating the impact of major component failures on frontline safety and support system failure modes, and ultimately the impact on normal plant operation. This approach is especially useful for identifying important common cause initiating events in support systems that simultaneously impact other plant systems. The Watts Bar PRA explicitly uses FMEAs to verify that the plant-specific potential for support systems failures to induce plant trips has been adequately considered. Its application is discussed in Section 3.1.1.

- **Step 2 — Grouping of Candidate Initiating Events.** In this step, individual initiating events that were identified in the previous step are classified and categorized to support subsequent evaluation and modeling.

Each initiating event is carefully examined to see which of the systems that must function to mitigate its consequences might also be made unavailable by the initiating event. Such dependence is modeled by grouping those initiating events that require similar mitigating systems, then defining the boundary conditions for each mitigating system to make them specific to those initiators.

Certain initiating events that affect more than one event in a scenario are modeled explicitly if their likelihood and potential consequences are judged to be significant. Examples of such initiating events include steam line breaks, loss of coolant accidents, and internal floods.

Initiating events are grouped into two levels of categories: a coarse grouping and a fine grouping.

The coarse grouping identifies initiating events that require a plant response that is sufficiently different from other initiating events to change the interdependencies of the mitigating systems. For each coarse group, a separate event sequence model is constructed, including a separate set of event trees, success criteria for plant systems, and logic rules for assigning split fractions.

After an event tree is defined for each coarse group, the individual initiators within the group are further reviewed to define a fine structure grouping. A fine structure group is established when the logic rules for reducing the tree to model the dependencies of the plant response (discussed in Section 2.3.5) are similar enough for the initiating events within that group to warrant the same quantification. Thus, the number of fine groups determines the number of sequence frequency quantification runs that need to be made for each event tree.

For example, the coarse group called general transients (i.e., turbine trip, reactor trip, etc.), for which a single set of event trees can be developed, typically has many quantifications needed to account for the unique impacts that each fine group of initiating events has on the plant.

In developing the two-tier initiating event grouping described above, it is recognized that the division of these groups is somewhat arbitrary. In principle, one could develop a single, very general set of event trees that could conceivably be used to analyze all initiators. However, such an event tree would be too complex to analyze practically. At the other extreme of impracticality would be the development of a separate set of event tree models for each initiator having a unique plant impact on mitigating systems. The two-tier grouping process is found to provide a good tradeoff between these two extremes.

- **Step 3 — Quantification of Initiating Event Frequency.** The purpose of this final step of initiating events analysis is to set up the models and database requirements for estimating the frequency of initiating events.

  Some of the initiating events, particularly those of moderate-to-high frequency, are quantified using experience data from plant operating experience, if available, supported by industry data from the experience from other similar plants. The data analysis itself is performed using methods described and applied in Section 2.3.5. Examples of this category include reactor trip, turbine trip, and loss of offsite power.

  Other initiating events are too infrequent to rely solely on experience data to estimate their frequency. These events require the use of models and expert judgment to supplement the statistical data. Applications of Bayes' theorem is used in this study to combine different types and sources of evidence in the estimation of event frequencies.

  Initiating events involving system or subsystem failures are estimated using models that derive the initiating event frequency in terms of combinations of more basic events for which data are usually available. The use of systems models for initiating events is developed in the systems analysis task that is described more fully in Section 2.3.5 (see also Reference 2-7).

2.3.4.2 Event Sequence Diagrams

Event sequence diagrams are graphical depictions of the plant support and frontline systems' response to an initiating event in flow chart format. ESDs are used to document the possible scenarios and courses of action that can be taken by the operators after a specified initiating event has occurred. Such actions include the plant hardware response and the steps taken by the operators to implement Emergency Operating Procedures (EOP). Analysis of ESDs is the first step towards the development of event trees that will subsequently be used to quantify the frequency of all modeled accident sequences.

Although ESDs are easily understood and are useful tools for documenting required plant system and operator actions after an initiating event has occurred, they cannot be directly used for accident sequence quantification. A necessary next step therefore is to convert the ESD into an event tree for the purpose of quantification of event or accident sequences. The event tree represents the transformation of the qualitative details contained in the ESD into a functional logic framework for sequence frequency quantification. Specific actions identified in the ESD are grouped into top events for the corresponding event tree.

The process used in constructing ESDs is described in Section 3.1.2. The driving factor in defining the level of detail in the ESDs is to be able to distinguish the application of the correct EOPs for each scenario. Experience has shown that this approach to ESD development is particularly useful in communicating with operations personnel and soliciting their crucial input to both the sequence definition and the human reliability assessment process. When developed to this level of detail, the ESDs provide the following key inputs to the risk assessment process:

- Document assumptions regarding accident progression for use in event tree development.
- Define top events and boundary conditions for systems analysis.
- Define sequence boundary conditions and dependencies for human actions analysis.
- Support future efforts to develop accident management strategies.

2.3.4.3 Support System Event Trees

The next step in the accident sequence development process is to develop the event trees themselves. The event tree linking methodology employed in this evaluation uses a separate event tree or trees to model the response of the support systems to the initiating events. The motivations for this are several. The most important is that previous PRAs have consistently shown that support systems such as electric power, service water, room cooling, have a higher propensity to be risk significant than all but a few frontline systems. By giving them visibility in the event trees, there is less chance that their role will be overlooked in areas such as human reliability. A second reason is the need to arrange the top events in an event tree so that the sources of dependency are on the left of the tree and the impacts of these sources are on the right. If we were constructing a single large event tree, the support systems would be on the left because many frontline systems are dependent on them. With the modularized technique employed in RISKMAN, as illustrated in Figure 2-2, a separate set of support systems event trees is used for this purpose.

Another motivation for this approach is that support systems tend to be normally operating systems as opposed to standby-type systems. In placing the support system event trees first on the left, we are in position to query the status of the support systems immediately after the occurrence of the initiating event prior to resolving the responses of the frontline systems. In this way, the operability of the frontline systems as well as the dynamic operator actions in the event trees can be conditioned on the status of the support systems as well as the frontline systems. The support systems event trees developed for this evaluation are presented in Section 3.1.4.

The top events defined for the support trees were selected based on the functional intersystem dependencies described in Section 3.2.3. Individual trains of a single support system are often represented by multiple top events (i.e., one top event for each train), so that the affects of different support system train failures can be tracked for each sequence path.

2.3.4.4 Frontline System Event Trees

Once the frontline system requirements have been identified (i.e., once the Watts Bar ESDs have been constructed) for the initiating events that have been grouped according to plant response, the frontline event trees are then constructed. The particular way that frontline event trees are constructed is determined by the characteristics of the RISKMAN software that is used for this purpose. An example event tree developed in the RISKMAN environment for Watts Bar is illustrated in Figure 2-4. This example is developed for a large loss of coolant accident (LOCA) initiating event.

The RISKMAN event tree format uses a subtree technique to represent a large event tree structure, 1,863,680 sequences in this example, on a single page. The left-hand column of sequence numbers identifies whether a subtree is used on each path. Thus, it can be seen that the 9th event tree path ends in subtree X1, whose structure is revealed in the tree above in sequences 1 through 8. In the adjacent column, the sequence number range for the expanded tree is listed. This particular tree is fairly large because not only the core cooling but also the active containment systems are included to support a Level 2 PRA. The frontline event trees developed for Watts Bar are presented in Section 3.1.2.

The frontline top events selected for each tree are a function of the systems required to mitigate each initiating event that will use the tree for accident sequence frequency quantification. The top events are ordered such that the split fractions for each top event depend on the preceding top events in the tree. Typically, this means that the frontline top events are arranged temporally. Only if a system performs multiple functions (e.g., for injection and recirculation) would the order likely deviate from a temporal arrangement.

2.3.4.5 Plant Damage States

The last step in the definition of Level 1 event sequences is to assign end states to the Level 1 sequences. These end states include successful termination and core damage states. The latter states are normally referred to as plant damage states and also serve as the entry states to the Level 2 portion of the scenarios as defined in the containment event tree. The development of the plant damage states is described in Section 2.3.6.

2.3.5 LEVEL 1 SEQUENCE QUANTIFICATION

In the previous section, a full set of accident sequences were defined that cover the progression of the scenarios from the initiating events to either successful termination or core damage as defined by the plant damage states. The purpose of this section is to describe the quantification of these sequences. The principal objectives of this quantification are to obtain a list of sequences, realistic estimates of the frequencies of these sequences as expressed in units of events per reactor-year, a quantification of the range of uncertainty in these estimates, and to understand the key risk-controlling factors that drive the results. Such an understanding is needed to be able to evaluate plant features that produce the risk results and to identify any potential plant vulnerabilities.

When the Level 1 portion of the event sequence model is quantified, results are obtained for core damage frequency as well as the frequency of any special groups of core damage sequences, such as those groups determined by initiating event, plant damage state, or any other sequence parameter that is resolved in the Level 1 event trees. The results of the Level 1 quantification can be used to determine the resolution of potential plant vulnerabilities in the systems needed to protect the reactor core. Such vulnerabilities, if any, would be evident in the results for core damage frequency.

To quantify the Level 1 event sequence model, it is necessary to develop a variety of different types of models that are needed to quantify different elements of the accident sequences and to construct a database that relates the various parameters of these models to the available evidence. This evidence includes various types of data that have accumulated at the plant being assessed and at other similar plants, as well as expert information.

A flow chart illustrating the key elements of Level 1 sequence quantification is presented in Figure 2-5. Although not explicitly shown, these elements include:

- Systems Fault Tree Models
- Human Reliability Models
- Electric Power Recovery Models
- Internal Flood Models
- **Interfacing Systems LOCA (ISLOCA) Models**
- Plant-Specific PRA Database (WBN database is generic)
- Event Sequence Quantification Using the RISKMAN Software
- Quantification of Uncertainty

Each of the above elements of the sequence modeling and quantification process is accomplished by the analyst using the RISKMAN PC-based workstation software that meets 10 CFR 50, Appendix B (Reference 2-25). A more complete description of each of these elements is provided in the sections below and in Reference 2-7, including a description of how RISKMAN is used to implement each step.

2.3.5.1 System Analysis Models

The systems analysis task assesses the likelihood that a system will fail to meet its functional success criteria defined by the plant response event tree model top events. The method by which this is accomplished is fault tree analysis. This task is also where a large fraction of the effort to perform a PRA is devoted. Most of the split fraction values that are needed to quantify the event trees are obtained via the systems analysis task. Other split fractions are obtained directly from data analysis, or from the human reliability analysis task.

System failures may result from independent or common cause equipment hardware failures, human error, or from combinations of equipment failure, human errors, maintenance actions, and testing activities. Specific system failures may affect the **availability** of other systems (e.g., support system failures and, in limited cases, frontline systems), or they may directly affect the ability to mitigate the consequences of accidents or transient events; e.g., frontline system failures. The qualitative systems analysis identifies physical and functional dependencies among the systems, and the qualitative results are used in constructing the plant event tree models. The logical structure of the event trees, in turn, defines sequence-specific success criteria for system performance and boundary conditions within which the system is required to operate. Therefore, the systems analysis task provides:

- Engineering knowledge about the plant systems needed to develop the plant risk model; i.e., dependency matrices and event tree models.

- Input for quantification of the integrated plant event tree models; i.e., failure frequency of each top event split fraction for specified boundary conditions.

Each system analysis contains the components required for system success as defined by the event tree system split fractions. The development of the database used for quantification of the system models is described in Section 2.3.5.6. The plant-specific operating and test procedures are reviewed during the qualitative systems analysis task done with the plant familiarization task. Human errors during testing that could contribute to system unavailability are included in the systems models.

The systems analysis task is carried out in the following steps:

1. Selection of Systems To Be Analyzed
2. Qualitative Analysis
3. Definition of Top Events and Split Fractions
4. Fault Tree Development
5. Common Cause Modeling
6. Quantification of Basic Event Unavailability
7. Specification of System Alignments
8. System Model Qualification

An overview of each step is provided below.

- **Step 1 — Selection of Systems To Be Analyzed.** Plant systems are initially screened to determine whether they need to be considered in the definition or quantification of accident sequences. These include any frontline systems that could cause, influence, or mitigate a sequence of events involving a reactor vessel cooldown and/or pressure/temperature transient in the main or secondary cooling systems or systems that support or interface with these frontline systems. The systems selected fall into the following general categories:

  — Frontline Systems Categories

    - Reactivity Control Systems
    - Main Cooling System Inventory Control
    - Main Cooling System Pressure and Flow Control
    - Main Steam Systems
    - Feedwater, Condensate, and Circulating Water Systems
    - Safety Injection and Shutdown Cooling Systems
    - Auxiliary Feedwater Systems
    - Containment Heat Removal, Isolation, and Filtration Systems

  — Support and Interfacing Systems Categories

    - AC and DC Electric Power Systems
    - Service Water and Component Cooling Systems
    - Room Cooling and Heating, Ventilating, and Air Conditioning (HVAC) Systems
    - Reactor and Plant Protection/Actuation Systems
    - Feedwater and Turbine Control Systems

- **Step 2 — Qualitative Analysis.** After the systems screening, a summary is developed for each of these selected systems and documented in systems notebooks. These summaries briefly describe the system and generally include:

  - System Function
  - System Success Criteria
  - Support Systems Required for System To Perform Function
  - Systems Supported
  - System Operation and Special Features
  - Testing
  - Maintenance
  - Potential for Event Initiation
  - Technical Specification Requirements
  - Modeling Assumptions
  - System Logic Model (Fault Trees)
  - References

- **Step 3 — Definition of Top Events and Split Fractions.** This step involves combining the outputs of a number of tasks into a concise set of systems analysis requirements to focus the quantitative analysis. The requirements are conveyed to the systems analyst in the form of top event and split fraction definitions. The categories of information comprising each and the source of that information are summarized below.

  The top event defines the functional requirement specified in the event sequence model for the system, the success of which will bring the plant closer to a safe, stable shutdown condition. Conversely, the failure of the top event function will place demands on other functions, or may ultimately result in core damage. The top event definitions consists of the following parts:

  - **Definition of the Function To Be Performed.** This is a concise statement of the physical process to be accomplished. This statement is directly based on the use of the top event in the event tree model.

  - **Success Criteria for that Function.** This is a specification of how many items of equipment must operate, the alignment under which they must operate, and their mission time. Physical analysis of the plant design and thermal hydraulic analysis of the accident conditions are evaluated to give the minimum required equipment to provide the function successfully.

  - **Model Boundaries under which the Function is Performed.** This is a summary of the system equipment required to accomplish the function and of any operator actions available to either implement or back up the function. The qualitative analysis of systems provides information as to how the system can perform the function.

    The grouping of system equipment among the top events is performed in the support and frontline event tree development task.

The **split fractions** defined for each top event further delineate the conditions under which the top event function is demanded. These alternate conditions arise from three general sources:

- Support systems necessary for its operation. The qualitative analysis of the systems, as reflected in the dependency matrices, provide this information.

- Conditions established by the specific initiating event being addressed in the accident sequence frequency quantification. The course and fine grouping of initiating events defines these dependencies.

- Functional dependencies created by the success or failure of top events questioned earlier in the event tree. These dependencies are obtained by reviewing the plant conditions generated by the earlier events as related in the event sequence diagrams and event trees.

Definition of split fractions involves two steps:

- Identification of the intersystem dependencies.

- Determination of what combinations of these conditions influence the likelihood that the function can be successfully accomplished in similar ways.

To make the number of split fractions that need to be quantified reasonable, the accident sequence analyst must establish categories of conditions and assign the various combinations of intersystem dependency failures to each of these categories. This is done manually using a combination of truth tables and physical reasoning regarding the influence of the failures on the function being modeled.

- **Step 4 — Fault Tree Development.** The systems of interest for a PRA study are modeled with fault trees. Fault trees provide a logical and convenient mechanism for quantification of the event tree top events by providing a structured format for identifying various combinations of equipment faults and human errors that are both necessary and sufficient to cause system failure. This detailed breakdown of system failure in this way serves four primary purposes:

  - Provides a method for communicating the principal ways in which a system can fail, and provides insight into means by which such failures can be prevented or their impact reduced.

  - Allows a calculation of the probability of system failure by defining failure according to its constituent parts for which statistically relevant failure rate data exist. This is done through the generation of **minimal cutsets** to be used to develop a quantification model.

  - Allows dependencies among and within systems to be explicitly defined. Those "among" systems are incorporated into the sequence quantification via the support system event tree and/or "conditional split fractions."

— Permits basic events associated with common cause failures within a system to be accounted for by adding to the fault trees in accordance with the procedures outlined in Step 5 and in accordance with NUREG/CR-4780 (Reference 2-5).

As with event trees, fault tree modeling is a common practice in PRA development and is well documented throughout the industry, so an extensive discussion of the subject is not included here. Reference 2-1 provides a good review of the basic approach to fault tree development. Certain specifics related to their application in this analysis are discussed below.

Certain aspects of system modeling are plant and component-specific; they must be determined by the analyst and adequately reviewed. These judgments are documented in the system notebooks. Important assumptions or judgments are also discussed in Section 3 for each specific system model. Fault tree construction ground rules were developed and followed for this project regarding:

— Level of Modeling Detail
— Naming of Basic Events
— System Interfaces
— Failure Data Sources
— Human Error Modeling

The fault trees model plant systems to the major component level; examples are pumps failing to start, valves failing to open and heat exchangers rupturing per unit time. The status of required support systems necessary for successful operation of the modeled system, and other intersystem dependencies are modeled as house events in the frontline system fault trees. The house events are then quantified as either guaranteed success or failure in the individual split fractions, to account for the condition of the plant based on top event failures up to that point in the sequence. Using this approach, one fault tree can be used to quantify the failure frequency of the system's top event function for a wide variety of plant conditions.

As two examples of the use of house events, consider the conditions modeled in Figure 2-6. In the first example, consider a check valve in an active flow path that is normally open. In most accident sequences, its only failure mechanism would be plugging. However, during the temporary loss of AC power following a loss of offsite power (LOSP) initiator, the valve will swing shut. When the pump is restarted, the check valve can fail to open, adding a new failure mechanism to the flow path failure mode. The house event LOSP is guaranteed failed when the system is quantified for the loss of offsite power initiator, or a loss of offsite power occurs during an event sequence. It is guaranteed success for all other initiators. For this system model, the AND gate allows the system to fail by a failure of a check valve to open only in those sequences where a loss of offsite power has occurred.

The second example shows how to model most support system dependencies. The house event is placed in the fault tree with an OR gate in series with the components or function it supports. In that way, the function will fail if either the frontline hardware fails or the support system is failed.

Component control circuits are generally not included in fault tree models. However, the circuits are examined to verify system operation and to ensure that interlocks between components are not overlooked. System models are normally developed for major actuation circuits down to the relay/pressure switch level. No attempt is made to extend these models to the contact level, although circuits are reviewed to that level.

- **Step 5 — Common Cause Modeling.** The contribution to system unavailability from common cause dependent failures is treated by the multiple Greek letter (MGL) method for each system analysis, according to the general methodology described in NUREG/CR-4780 (Reference 2-5). To incorporate common cause events into the systems analysis, the analyst determines the populations of components subject to common cause failure mechanisms. These include consideration of:

  - How groups of components are used.
  - The extent of their diversity, design, manufacture, and type (if any).
  - The physical proximity or separation of redundant components.
  - 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 from the common cause analysis and categorize common cause groups of components for systems of interest.

  A common cause group of components has a significant likelihood of experiencing one or more common cause events affecting two or more components in that group. Based on experience in evaluating operating data, the following guidelines are developed to help assign component groups:

  - When identical, nondiverse, and active components are used to provide redundancy, they should be considered for assignment to common cause groups, one group for each identical redundant component.

  - The likelihood of common cause events linking diverse components in the system can be assumed to be negligible compared to identical, nondiverse, and active components that are present in the system.

  - When diverse major components serving the same function have parts that are identically redundant, the components should not be assumed to be fully independent. (One approach is to break down the component boundaries and to identify the parts as a common cause component group.)

  - When each redundant leg of a system contains one or more active components, the contributions due to both independent and common cause events involving passive components are generally insignificant in the calculation of system unavailability.

- In redundant systems in which no identical active components or parts can be identified, no common cause grouping need be attempted.

Typical types of components and failure modes normally considered for common cause modeling are listed in Table 2-6. Due to practical limitations, all of the possible ways that similar components within a system can be grouped for common cause analysis may not be able to be modeled.

Once the common cause groups have been determined, the groups are entered as input to RISKMAN, which automatically generates common cause basic events to the fault tree. Each of these basic events accounts for each unique way in which a causal event can impact a different combination of components within the group in accordance with the procedures set forth in Reference 2-5. For example, if components A, B, and C are in a common cause group, the following new basic events are added to the tree:

C_A = causes that result in failure of component A only.

C_B = causes that result in failure of component B only.

C_C = causes that result in failure of component C only.

C_{AB} = causes that result in failure of components A and B.

C_{BC} = causes that result in failure of components B and C.

C_{AC} = causes that result in failure of components A and C.

C_{ABC} = causes that result in failure of components A, B, and C.

Once a component is added to a common cause group, the component failure in the original fault tree is no longer a basic event. The RISKMAN code develops a subtree for each component basic event that includes the appropriate set of common cause events for that component, as illustrated in Figure 2-4. The common cause events in that group become the new basic events for every component in the group. Because of this, final Boolean reduction of the fault tree cannot be performed until the common cause basic events are added.

- **Step 6 — Quantification of Basic Event Unavailability.** To quantify a split fraction, the basic events appearing in the top event fault tree must be quantified in terms of basic database variables and parameters defined during the event sequence and systems analyses.

  The basic event unavailabilities are determined by identifying the failure modes for the components making up the basic events and by assigning failure rates to the failure modes from the basic event database. The basic event database is developed in the Data Analysis Module of RISKMAN, as documented in Section 3.3. The failure modes applicable to this PRA are listed in the data analysis Section 3.3.2.

The calculation of component unavailability is best explained by a few examples.

- For a standby pump to be unavailable for an emergency mission, it may fail to start on demand or fail during operation.
- For a normally closed, motor-operated valve (MOV) to be unavailable to pass flow, it may fail to open on demand or fail to remain open during the mission time.
- For a normally open, motor-operated valve to be unavailable to pass flow, it may fail to remain open during the mission time or during the period of time between the previous test and the initiating event.

The unavailability for these three components can then be modeled as

Q_{ps} = standby pump unavailability. (Pump must start and run for t_M hours.)

$\simeq q_{ps} + \lambda_p t_M{}^*$

Q_{vc} = normally closed MOV unavailability. (Valve must open and remain open for t_M hours.)

$\simeq q_{vo} + \lambda_v t_M{}^*$

Q_{vo} = normally open MOV unavailability. (Valve must be open and remain open for t_M hours.)

$\simeq \lambda_v T_t{}^*/2 + \lambda_v t_M{}^*$

where the following parameters are selected from the basic event database for the size and type of component and the failure mode being modeled:

q_{ps} = demand failure rate for pump; failure to start per demand.

λ_p = operation failure rate for pump; failures per operating hour.

q_{vo} = demand failure rate for MOV; failure to open per demand.

λ_v = transfer closed failure rate for MOV; failures per operating hour.

Furthermore, the following parameters are obtained from the event sequence analysis and qualitative system analysis as indicated:

T_t = system flow test interval (hours), which is obtained during the qualitative systems evaluation and recorded in the system notebook,

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*Note: λt is an approximation for the exact expression $1 - e^{-\lambda t}$ (Reference 2-15).

t_M = system mission time (hours). This value is specified in the success criteria of the top event.

The RISKMAN Systems Analysis Module provides window formats to enter basic event quantification equations for each basic event in the fault tree. Parameters such as mission time can be given a local variable name so that it need be changed only once in the event of a model change. A mission time of 24 hours was used (Reference 2-2). The set of quantification equations is stored in a common basic event database. A number of reports are available to the user to verify the consistency of basic event modeling assumptions throughout the plant model.

The basic event quantification window makes it possible to reduce the number of system cutsets by the lumping of series component failure modes into "super component." Care must be exercised to ensure that none of the components combined in this manner are subject to common cause grouping. Normally, this technique is only useful for reducing the number of independent failures that need to be processed by the minimal cutset generator.

- **Step 7 — Specification of System Alignments.** Having developed the system logic model, the next step is to convert the logic model into an algebraic model that can be quantified. The logic model defines the combinations of basic events that cause the fault tree top event to occur. The algebraic model permits the quantification of fault tree top event likelihood in terms of the parameters of the database for the conditions established by the event sequence model.

  The logic model discussed in the previous section was developed for the normal alignment case. The initial conditions for the normal system 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 the important causes for the unavailability of components in the system. These include:

  - **Functional Unavailability due to Lack of Required Support.** This is accounted for by specifying the success or failure of the support system house events. These house events are either "true" or "false," as designated by the boundary conditions for each split fraction that models a possible combination of support system states. Minimal cutsets are obtained separately for each split fraction by assuming each basic event failed by the boundary condition of that split fraction to have occurred (i.e., assumed failed) and by re-reducing to provide a different list of minimal cutsets for each split fraction.

  - **Independent and Common Cause Hardware Failures.** These failures include undetected failures 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 may occur due to errors of omission and commission, particularly for periodically performed tests or maintenance.

A large number of dependencies can greatly complicate the logic models and create problems with the software that must perform the steps of Boolean reduction and minimal cutset quantification. Many of these complexities stem from the fact that the system can be in a number of different alignments at the time of the initiating event and the likelihood of occurrence of system failure causes can be highly dependent on the alignment. For example, if one part of the system is down for maintenance, it is much less likely, and perhaps a violation of technical specifications, to have other redundant parts out of service at the same time. In addition, the normal valve positions, breaker positions, or pumps designated for standby can change from one initial alignment to another.

In principle, the above complexities can be modeled directly in the fault trees, but they make the trees very complicated, difficult to check, and create problems with the fault tree software. Many of these dependencies would require the use of complement events, which most fault tree software cannot handle properly.

To tackle these problems, RISKMAN decomposes the models for each split fraction into a number of different alignments.

The alignment concept is used to decompose the model of each split fraction, $F(SF_i)$, into contributions from individual alignments according to the following equation:

$$F(SF_1) = \sum_{1}^{N} F(A_j) * F(SF_i | A_j) \tag{2.1}$$

where

$F(A_j)$ = fraction of time the system is in alignment A_j.

$F(SF_i | A_j)$ = conditional frequency of split fraction SF_i, given that the system is in alignment A_j.

N = total number of different alignments.

Equation (2.1) is exact as long as the set of alignments considered are both mutually exclusive and complete. The approach to implementing the alignment concept in RISKMAN is to first develop a fault tree for the "normal alignment" in which no test or maintenance is in progress. Then, the models for the separate alignments are developed as special cases of the fault tree, and each one is analyzed separately through Boolean reduction and quantification. The separate models are integrated automatically by RISKMAN, which applies Equation (2.1) to construct the split fraction model from the individual alignment models.

- **Step 8 — System Model Quantification with RISKMAN.** RISKMAN generates several versions of the fault tree minimal cutsets: one for the basic model of the top event, one set for each split fraction within each top event, and one set for each alignment within each split fraction. The cutsets for each alignment are constructed with the use of Equation (2.1) in terms of alignments. For each alignment within a split fraction, a separate set of minimal cutsets is generated by a switch on the appropriate house events to reflect the components not in service for that alignment and to change the set of component failure models that apply for that alignment. The minimal cutsets are then converted into algebraic equations that can be used to quantify the frequency of each system split fraction. The alignment contributions are summed to give the total for a particular split fraction.

  The basic algebraic models that are generated by RISKMAN are based on the cutsets, which are, in turn, based on the basic events from the fault tree input. These basic events must be related to failure designators from the database. RISKMAN allows the system analyst to provide an equation that applies to each basic event, and may include database variables, local variables, or constants.

  The fault trees, common cause models, system split fractions, cutsets, and basic events are managed and quantified through application of RISKMAN, which stores the basic events and combines them, as necessary, for quantification of the split fractions. A simplified schematic of the use of RISKMAN for systems analysis is presented in Figure 2-5.

  The top event split fractions are quantified using component failure data, maintenance frequency and duration data, human error rates, and common cause parameter data stored in the RISKMAN database file. RISKMAN uses the Monte Carlo and Latin Hypercube techniques to combine the uncertainty distributions for the database failure parameters modeled in each split fraction equation. This results in a mean or point estimate value and a probability distribution that quantifies uncertainty in the likelihood of each split fraction. The mean values of these uncertainty distributions are used initially to quantify the support and frontline event trees. Subsequently, the probability distribution for each split fraction is used in the plant model uncertainty analysis for the identified important sequences.

  The results of the quantification for each split fraction are stored by the RISKMAN software in the master frequency file (MFF), which is used to quantify the event trees. The MFF includes a split fraction identifier, a description of the split fraction, and the mean value of the split fraction. Multiple MFFs can be used to represent either the point estimates of the split fraction models or the means of the Monte Carlo simulations to quantify uncertainty in the split fraction values. Split fractions are produced by not only the systems analysis but also by human reliability analysis for top events involving operator actions, and, in some cases, values directly from the failure rate database. As illustrated in Section 3.2, RISKMAN produces a number of reports that permit the detailed analysis of system results.

2.3.5.2 Human Reliability Models

The approach to human interaction modeling emphasizes a close coordination with plant operators and a thorough review of their procedures. The following types of human actions are quantified:

- **Routine Actions before an Initiating Event.** Routine actions are considered in the analysis of individual systems. They involve restoring a component or flow path to normal after completing testing, inspection, or maintenance and ensuring that the sensing equipment is correctly aligned and calibrated for automatic response to emergency actuation conditions. Errors that are important to plant risk leave safety-related equipment disabled or in an undetected, misaligned state, causing it to be unavailable to accomplish its function on demand during an event sequence.

  Each system analyst is responsible for reviewing each surveillance procedure accomplished on the system and for identifying the sources of human error that can leave equipment unavailable. If an error is identified as a potentially significant contributor to equipment unavailability, the analyst designates a local system alignment variable for the procedure associated with that error and includes it in the system cause table.

  The numerical estimates for errors during routine actions are evaluated using the methods of Swain and Guttman (Reference 2-26). These methods have been implemented by developing distributions that could apply to a range of surveillance procedures depending on type of potential error, location of the action, and complexity of restoration for restorations having independent verification. The application of this methodology is discussed in more detail in Section 3.3.3.

- **Actions That Can Cause Initiating Events.** Actions that can initiate plant transients are implicitly accounted for in the quantification of initiating event frequencies to the extent that these human actions are the cause of such events.

- **Dynamic Operator Actions That Take Place Following an Initiator.** This type accounts for the operating team's ability to manually align, initiate, and control plant equipment to mitigate against accidents. These tasks are generally guided by plant emergency response procedures.

An adaptation of the SLIM is employed by TVA to quantify the dynamic human actions. This methodology is based on the assumption that the likelihood of operator error in a particular situation depends on the combined effects of performance-shaping factors (PSF) that influence the ability of the operator to accomplish the action successfully.

TVA has adapted SLIM through the use of a set of forms and instructions to explain and expand on the procedure proposed by Embrey (Reference 2-27). These guidelines are described in detail in Section 3.3.3.

First, each dynamic action is qualitatively described in detail and related to the plant event sequence model using an operator response form. Once the actions are adequately described, seven PSFs are used to relate the impact of

- Conditions of the work setting under which the action must be accomplished. The PSFs for this are as follows:
  - Significant preceding and concurrent actions.
  - Plant interface and indications.
  - Adequacy of time to accomplish the action.
- Requirements of the task itself. The PSFs that describe this are as follows:
  - Required actions and procedural guidance.
  - Complexity of the task relative to resources, coordination, and location.
- Psychological and cognitive condition of the operators. The PSFs that consider this are as follows:
  - Training, and experience relative to the action.
  - Stress due to the situation and environmental conditions.

The performance-shaping factors are rated against two criteria:

- A weight that relates the relative influence of each PSF on the likelihood of the success of the action.
- A score that relates whether the PSF helps or hinders the operator to perform the action.

The score addresses the actual conditions under which the specific action must be accomplished. The weight describes the extent to which the operators believe how much the conditions relative to a specific PSF actually impact the potential for success or failure of the action. If it is not a factor that controls the ability of the operator to do the action, it is weighted low or insignificant. Guidelines for rating PSFs enable these ratings to be consistently applied for all actions.

One of the premises of the SLIM methodology is that the evaluation team can rate the weight and score independently. The thought process necessary to distinguish between these two orientations of the rating process is stressed in the initial training of the raters. In addition, the human actions analyst provides feedback to the raters during the evaluation process regarding the broad qualitative interpretation of their ratings.
For this application, the SLIM methodology has been slightly modified so that the operators can scale the degree of difficulty rather than the potential for success when they rate the actions. This change in orientation produces a failure likelihood index (FLI) rather than a success likelihood index. This has the advantage of quantitatively highlighting the causes of operator difficulty. A high degree of difficulty, combined with a large weight, points to the primary area of concern for accomplishing an action.

The human error probability quantification is performed by grouping actions that have similar PSF weights. The error rate of each action is estimated by comparing the overall FLI to a correlation that follows the relationship:

Logarithm (human error rate) = A + B(FLI)

The coefficients of the correlation are obtained from a least squares fit of the FLI of calibration actions. The calibration actions for a particular group are chosen to match the actions in the group using similarity of PSF weights as the selection criteria. A series of spreadsheets are used to accomplish the quantification process.

A key step in the quantification is to calibrate the numerical model using well-defined actions obtained from statistical data, results from simulator studies, evaluations for other PRAs, or other analytical evidence of failure rates for these actions. The calibration procedure should ensure that the numerical error rate estimates are realistic and consistent with available data, observed human behavior, and the results from comparable expert evaluations of similar activities.

Uncertainty distributions are developed for each evaluated human action error rate. The magnitude of the uncertainty of the error rate is determined by the magnitude of the nominal error rate and the variation of the values assigned to the parameters by the individual evaluation groups. If the operator groups differ substantially in their evaluations, a large uncertainty results.

The final results from the evaluation and the quantification are displayed in a tabular format that allows easy review, comparison, and identification of the most important factors influencing each assessment. The application of this methodology is discussed in more detail in Section 3.3.3.

2.3.5.3 Electric Power Recovery Models

Accident sequences involving station blackout are frequently found to make important contributions to risk in PRAs. Because of the importance of these sequences, a realistic treatment of the possibilities for recovery of both onsite and offsite power is needed to obtain realistic frequencies for such sequences leading to core damage. To treat such sequences, time-dependent models are needed to account for important time-dependent interactions. These interactions include:

- The time-dependent nature of the occurrence of station blackout created by the possibilities for one or more diesel generators starting but failing to continue running before offsite power is restored, whose probability is also a function of the amount of time lapsed from the loss of offsite power occurrence.

- The finite time required to deplete the station batteries, after which the major part of the plant instrumentation is lost, as is the capability to operate certain electrical breakers needed to recover electrical power and to start and excite the diesel generators.

- The finite time available to restore electric power before various competing processes such as steam generator dryout, reactor coolant pump seal degradation, and other processes lead to core damage. The specific processes of interest are determined on a plant-specific basis.

To compound the above complexities, many of the parameters of the above processes exhibit large uncertainties. To address these complexities, time-dependent models that consider the interactions between above processes are derived from the following equation for a mission time of 24 hours (Reference 2-2):

$$F(EP,T)=\int_{0}^{24} f(x)[1-G(x+T)][1-H(T)]dx \tag{2.2}$$

where

F(EP,T) = probability that there is no onsite or offsite power T hours after the occurrence of station blackout; i.e., T hours after the loss of onsite power when offsite power has not been recovered.

f(x)dx = probability density function for failure of the onsite power system in the time interval (x, x+dx) after the loss of offsite power at time 0.

G(x+T) = probability that offsite power is restored within x+T hours after loss of offsite power.

H(T) = probability that onsite power is restored within T hours after the beginning of the station blackout.

T = time interval between the beginning of the station blackout and the point of no return for the return of electric power to prevent core damage. (Note that T is a function of x, the time after LOSP when station blackout initiates.)

The function f(x)dx accounts for all combinations of diesel-generator failures to start and to continue running, independent failures, common cause failures, and initial unavailabilities due to maintenance. The recovery functions are also a function of battery life.

The above equation is applied differently to each unique sequences to account for the following types of dependencies:

- Depending on the initiating event, offsite power or one or more diesels may not be recoverable. For example, a flood may damage switchgear, or a loss of cooling water may lead to an overheated diesel generator that cannot be quickly repaired.

- The time available to restore electric power during a station blackout is dependent not only on the timing of the blackout after loss of offsite power but also on the sequence. For example, in a pressurized water reactor (PWR) with the steam-driven feed pump working, a reactor coolant pump seal LOCA may dictate the time available. However, if there is no steam generator heat removal, the steam generator dryout and the resulting primary loss of coolant out the pressurizer power-operated relief valves dictate the time available.

- Depending on the event sequence and the application of the emergency operating procedures, operator actions to depressurize the reactor coolant system lengthen the time available for recovery.

The use of these time dependent models is important because when the diesel generator failure and electric power recovery analyses are performed independently, optimistic results often occur. Detailed application of the offsite power recovery models in this examination is described in Section 3.3.3.

2.3.5.4 Internal Flood Models

Consistent with the requirements of Generic Letter No. 88-20, this examination includes an assessment of sequences caused by internal floods. The principal concern with internal floods is that their occurrence might constitute a common cause initiating event; i.e., a flood might cause a plant trip and damage one or more systems needed to respond to the initiating event. In principle, a flood could occur independently in the same time frame as some other cause of an initiating event. However, the simultaneous, independent occurrences of such events are orders of magnitude less likely than the case in which the flood and the initiating event are causally related.

As discussed in Section 2.3.4, initiating events are defined to allow the event sequence model to account for both internally and externally caused events. Thus, the same integrated event sequence model developed for the Level 1 analysis of internally initiated accidents is used for the analysis of internal flooding events.

In the case of internal floods, a flooding analysis produces a set of flood scenarios, each of which is described by a flooding initiating event frequency and a specification of the flood's impact on the plant. The impact specification includes a general initiating event type that defines the event tree being quantified and a list of the equipment items that are damaged by the flood. The event sequence model is then quantified with this special set of event tree quantification data, with the damaged equipment assigned a conditional failure probability of 1.0 (guaranteed failure).

If a given source of flooding can produce a variable amount of damage, multiple flood scenarios are defined with different frequencies and damage states. As the flood scenarios are processed through the integrated event sequence models, sequences with combinations of flood-induced and normal failures are defined. Finally, the supporting human reliability analysis identifies ways to mitigate against the dominant sequences so that realistic results are obtained.

The separate analysis of internal floods is carried out in the following steps:

- **Plant Familiarization.** This includes a review of key plant design information such as arrangement drawings, equipment locations, PRA models, and system drawings.

- **Flood Experience Review.** Flood data collected from industry experience (Reference 2-18) are reviewed and used in the quantification of internal flood scenario initiating event frequencies. The potential for flooding that results from the initiation of the fire protection system is also considered by reviewing the fire hazards analysis.

- **Flood and Equipment Location.** Major flood sources and equipment are identified for each key plant location. The potential for an initiating event; propagation to other locations including drainage, detection, and isolation potential; and system impact are considered in judging whether a scenario should be postulated.

- **Plant Walk-Through.** A plant walk-through was conducted to collect additional information and to confirm previous judgments.

- **Scenario Quantification.** Based on the above, internal flooding scenarios are identified, evaluated, and quantified as initiating events, with their impact on other plant systems defined.

- **Risk Model Quantification.** The important internal flooding scenarios are included as separate initiating events in the event sequence quantification process.

2.3.5.5 Interfacing Systems LOCA Models

Interfacing systems loss of coolant accidents are events caused by failures at the pressure boundary between the high pressure reactor coolant system and interfacing systems that are not designed to high pressure. The initiation of pressure boundary failures normally involves multiple leaks, ruptures, or mispositioning of valves at the pressure boundary. Such failures can be important if the low pressure system would overpressurize and leak outside containment. If such failures occur, reactor coolant would be lost, and a release path bypassing the containment would be created. Loss of inventory out the break could then lead to eventual core damage. Although these events are typically found to be small contributors to the frequency of core damage, they are often found to be significant contributors to the frequency of core damage accidents resulting in an early release from the containment.

The analysis of interfacing systems LOCAs entails the following steps:

- Systems that interface with the primary coolant system are reviewed to identify potential leak paths.

- Each leak path is screened to identify those with significant potential for an interfacing systems LOCA. For example, low pressure interfacing systems with no more than two normally closed isolation or check valves have significant potential. Pipes with diameters of less than 1 inch that result in a leak less than the charging pump capability are neglected.

- For each system selected in the screening analysis, the pressure capacity and overpressure failure modes are determined either realistically or using conservative assumptions. In this evaluation, the integrity of the piping, heat exchangers, flanges, pump seals, and other components is considered, as is the response of any relief valves. Walkdowns are performed as needed to support this assessment.

- Initiating event frequency models are developed for each release path that accounts for applicable failure modes of the interfacing valves. These models are used to develop point estimate and uncertainty distribution results for the frequency of each initiator.

- Scenarios are developed with appropriate consideration of human recovery actions to isolate the leak path and to prevent the event progression to core damage. In some cases, these scenarios are developed with event sequence diagrams and event trees, and in other cases, the initiating events are mapped directly to a containment bypass plant damage state.

More details on the basic approach to the treatment of these events can be found in Reference 2-22.

2.3.5.6 Failure Rate Database

The principal objective of this task is to develop a database that addresses the relevant parameters of the PRA models and that accounts for the actual operating experience of the plant and equipment being assessed. While this experience is frequently documented in various forms to support both in-house and regulatory reporting requirements, significant effort is required to collect, interpret, and analyze this evidence to put it in a form that could be used in the PRA.

There is insufficient plant-specific operating experience to support a direct prediction of the frequency of such rare events as sequences leading to core damage. Therefore, logic models are constructed to combine more frequent events which together may lead to a core damage sequence. For any event in the model less frequent than about 10^{-1} to 10^{-2} per year, the plant-specific data are statistically insignificant. Thus, these plant-specific data must be supplemented with data from other sources, such as data from other relevant nuclear power stations, and with subjective estimates of experts.

This section describes both the collection of plant-specific and generic data methods for combining these data to produce uncertainty distributions for each PRA model parameter. The collection of the sources of data other than plant-specific data is referred to as "generic data."

A generic database for PRA has been collected (Reference 2-18). For Watts Bar, the plant is not yet operating, so only generic data are used. Future plant-specific data may be collected and used to update the state of knowledge supporting the WBN PRA, using these same methods.

The event sequence modeling parameters in the database are primarily associated with the models needed to quantify the initiating event frequencies and the system fault trees. The specific types of parameters that need to be addressed include:

- Initiating Event Frequencies
  - Internal Events
  - Internal Floods
- Component Failure Rates
  - Failure Rates per Demand for Standby Components
  - Failure Rates per Hour for Operating Components

- Common Cause Failure (MGL model) Parameters
- Component Maintenance Unavailability
  - Maintenance Frequency
  - Maintenance Duration

2.3.5.6.1 Overview of Database Development Process

The database development process was organized into a series of 10 steps, as illustrated in Figure 2-9. A brief description of each step is provided below.

- **Step 1 — Identification of Generic Data Sources.** Development of a generic database requires the identification, collection, and review of a number of generic data sources. In general, each data source is designed for a special purpose, thereby imposing certain limitations on its usefulness. A single data source rarely provides all of the required information on a specific subject.

  A significant number of generic data sources containing nuclear and conventional power plant data currently exist (References 2-28 and 2-29). However, not all of these sources contain data that are usable, either directly or indirectly, for the purpose of a PRA. In this step, some of the more applicable generic data sources that apply to the PRA being conducted are identified.

- **Step 2 — Preparation of an Important "Plant Item" List.** The objective of this step is to select the plant equipment for which data are required. The definition of what failure modes are and are not included in each component variable in the database is a key part of the plant item list. An example of these definitions is given in Table 2-6.

  The plant item list is developed in coordination with qualitative systems analysis and system notebook preparation during plant familiarization. Preparation of a plant item list is important because it reduces the amount of data which could be collected to that which is actually needed.

  The plant item list is used to define the level of detail to which system analysts model the fault tree basic events. Although, in principle, the plant item list defines the scope of plant-specific data collection, in practice, the data collection needs to be started well before the completion of these other PRA tasks. Therefore, to remain practical, a generic list of plant items is developed with input from other available PRA studies for similar plants. This list is modified several times before the finalization of other PRA tasks.

- **Step 3 — Development of a Generic Database.** Generic event frequency distributions are developed based on the following types of generic information:
  - **Type 1.** Data from operating experience at various nuclear power plants.

- **Type 2.** Estimates or distributions contained in various industry compendia, such as WASH-1400 (Reference 2-28) and IEEE STD-500 (Reference 2-29).

Type 1 information is data collected from the performance of similar equipment in various power plants. The PLG generic database (Reference 2-18) contains plant-specific data from about 20 different plants.

Type 2 information, which could be called processed data, consists of estimates based on information ranging from the opinion of experts with engineering knowledge about the design and manufacture of the equipment in question to the observed performance of that equipment in various applications.

The methodology for creating generic failure rate probability distributions uses both types of information. Such distributions represent the variability of the assessed quantities from source to source (for type 2 information) and/or from plant to plant (for type 1 information). In the absence of plant-specific information, these population variability distributions are our state-of-knowledge curves. Methods for developing these curves are discussed in Section 3.3.1.

- **Step 4 — Identification of Plant-Specific Data Sources.** Plant-specific data sources provide information regarding failure events, maintenance outages, test and maintenance frequency and duration, and operational data for various plant items. However, this information is not usually documented in a fashion that can be directly used for probabilistic analysis. Consequently, plant-specific data sources are identified prior to starting the data collection process, and the data analyst becomes familiar with the content of these data sources so that the required information can be retrieved efficiently.

- **Step 5 — Collection of Plant-Specific Raw Information (Part One).** For Watts Bar, only generic data are used. This and the next several sections, however, describe the methodology for plant-specific data collection. This section describes the process of plant-specific information collection for the following categories:

  - Component Failure Data (including any common cause failures)
  - Component Test and Maintenance Data
  - Initiating Event Data

  The required raw data are collected by examining the plant-specific data sources identified in Step 4. Careful documentation of the collected raw data is performed because this is vital to the quality, applicability, traceability, and defendability of the subsequent data analysis and the whole PRA study. When the desired information is not available, engineering judgment is required to enable use of data.

- **Step 6 — Determination of Exposure Data.** The calculation of plant-specific component failure rates requires both the numbers of failures and the frequency of challenge, or exposure, information. This information includes the following:

  - **In-Service Hours.** Two different categories of component in-service hours are distinguished:

    - **Normally Operating Components.** The total number of running hours in different operational modes in the data period.

    - **Standby Components.** The in-service hours are obtained by a review of appropriate plant records and estimates of running durations during surveillance testing and actual demands.

  - **Number of Demands.** The number of demands for both normally operating and standby components are obtained by a review of appropriate records of actual demands and evaluation of test procedures.

- **Step 7 — Processing of Plant-Specific Information (Part One).** A two-part approach is taken to perform the processing of plant-specific raw information. In Part One, the system analysts review and interpret the information collected in Step 5. The purposes of this review are to assess and to reduce the raw information to a form suitable for use by the data analyst. Both the number of failure events and the number of challenges or exposure period must be discerned. Part Two of plant-specific information processing (Step 9) is an ongoing effort to process the additional information gathered in Part Two of the plant-specific information collection (Step 8) in order to modify and/or efficiently complete the result of Step 7 within the constraints of project resources; i.e., time and manpower.

- **Step 8 — Collection of Plant-Specific Raw Information (Part Two).** Part Two of plant-specific raw information collection is an ongoing effort and consists of the following major substeps:

  - Collecting additional, required recorded information.
  - Interviewing key plant personnel.
  - Adding the results of the above substeps to the raw data book.

- **Step 9 — Processing of Plant-Specific Information (Part Two).** In this step, the information gathered in Step 8 is used to modify and/or complete the results of plant-specific information processing (Step 7) and to prepare final plant-specific data in different data categories for use in Step 10.

- **Step 10 — Development of Final Plant-Specific Database.** The objective of this step is to provide a suitable plant-specific database for systems unavailability quantification, as well as initiating event quantification, using the results of previous steps. The required calculations are performed to determine required PRA parameters in the data categories. The appropriate distribution for each parameter is developed using the RISKMAN computer code package as described below.

For those plant items with no plant-specific data, the generic distributions are taken directly from the generic database. For those plant items with plant-specific data, Bayes' theorem is then used to update the generic distribution to obtain a plant-specific posterior distribution. The use of Bayes' theorem to develop both generic and plant-specific distributions is discussed in the next section.

2.3.5.6.2 Bayesian Method for Data Analysis

The methodology used to develop the database for this study is based on the Bayesian interpretation of probability and the concept of "probability of frequency." In this context, component failure rates are treated as measurable quantities whose uncertainty is dependent on the state of knowledge of the investigation. The "state of knowledge" is presented in the form of a probability distribution over the range of possible values of that quantity. The probability associated with a particular numerical value of an uncertain but measurable quantity indicates the likelihood that the numerical value is the correct one.

A key issue in developing state of knowledge for the parameters of the PRA models is to ensure that the information regarding each parameter, its relevance, and its value as viewed by the analyst are presented correctly, and that various pieces of information are integrated coherently. "Coherence" is preserved if the final outcome of the process is consistent with every piece of information used and all assumptions made. This is done by using Bayes' theorem. The concepts behind the use of Bayes' theorem are discussed in Reference 2-19. This section describes its application for combining different types of information.

In the context of a plant-specific PRA, three types of information are available for the frequency of elemental events:

Type 1 = the historical information from other similar plants.

Type 2 = general engineering knowledge such as that of the design and manufacture of equipment, sometimes expressed in terms of expert estimate of parameter values or their uncertainty distributions.

Type 3 = the past experience in the specific plant being studied.

Recall that the information of types 1 and 2 together constitute the "generic" information. Type 3 is the "plant-specific" or "item-specific" information.

It is very important to note that type 1 information brings an element of plant specificity into the generic data developed for a plant-specific PRA. In general, decisions regarding the relevance and applicability of different pieces of type 1 information must be made while developing each generic distribution. A piece of information may be judged as being relevant in developing the generic data in one PRA and not relevant in another. As a result, generic distributions for different plant-specific studies could be significantly different.

2.3.5.6.3 Motor-Operated Valve Example

The application of RISKMAN in the development of a generic distribution (prior) for the failure rate of motor-operated valves is illustrated in Figure 2-10. As shown in the figure, the generic distribution is based on plant-specific data collected at six different plants and covering more than 30,000 actual demands, and three sources of type 2 or expert estimates. Application of the two-stage Bayesian procedures described to this case produces a distribution that is seen to envelope the respective point estimates supported by the data sources. Note that if the 107 failures in 32,380 demands represented in the 6 plants were treated using classical statistical methods, the resulting distributions would exhibit negligible uncertainty because the plant-to-plant and source-to-source variability would not have been preserved.

Application of the second stage updating process for the incorporation of plant-specific evidence (posterior) on the plant being assessed is illustrated in Figure 2-11 for two hypothetical cases of motor-operated valve data. These cases illustrate a very useful property of Bayesian updating in that the weighting of generic and plant-specific evidence is done "automatically" and according to the quantity of evidence. In the case of posterior 1, an update with 1 plant-specific failure in 11,000 demands, the generic distribution appears to be shifted down towards the location of the evidence, 10^{-3}. When five times as much evidence is accumulated for posterior 2, the updated distribution becomes quite peaked (about 10^{-3}), with little apparent influence of the generic prior.

2.3.5.6.4 Treatment of Zero Failures

Another useful property of Bayes' theorem is that it provides a consistent treatment of any type of evidence, even when that evidence is made up from experience data in which no failures were observed; e.g., the failure rate of a valve that has not failed during N demands. Using Bayes' theorem, this plant-specific information can be used to update the generic distribution, as shown in Figure 2-12. As can be seen in this figure, the posterior distribution is heavily influenced by the prior distribution for $N = 10$ demands, indicating rather weak evidence. However, for $N = 1{,}000$ demands, the posterior essentially vanishes for values of demand failure rates in excess of 3×10^{-3} because of the influence of the likelihood that zero failures would result in such a large sample if the failure rate were that large. Thus, zero failures does not pose any problems for the Bayesian approach, and the results are a strong function of the quantity of evidence; i.e., the number of successful demands.

A more complete description of how data handling methods employed in RISKMAN are applied in the examination of Watts Bar is contained in Section 3.3.2.

2.3.5.7 <u>Event Sequence Quantification Using RISKMAN</u>

Sequence quantification is the calculation of complete accident sequence frequencies. It involves the combination of the frequency of equipment operation and operator actions in response to initiating events with the frequency of those initiating events. The sequences of failures that are of most interest here are those that result in inadequate core cooling and eventual plant damage, although the quantification process treats all sequences including those resulting in successful event mitigation. This process requires the assembly of many distinct parts of the PRA model; it can be divided into two major tasks: sequence assembly and sequence quantification.

Sequence assembly requires the linking of

1. The initiating events that have been identified for the analysis.

2. Support system event trees that model the functional relationship among support systems (e.g., electric power, component cooling, and instrument air), the availability of which directly affects the performance of frontline systems (e.g., safety injection) that are needed to respond to the initiating event.

3. Frontline event trees that model the functional relationship among operator actions, equipment, and instrumentation in frontline systems that are important to risk.

Sequence quantification requires the assignment of a split fraction value to each branch in each linked event tree. As described in Section 2.3.4, a split fraction value is the conditional frequency of failure of a given event tree top event. It can be dependent on the status of support systems, the specific initiating event, and on the success or failure of the previous top events of the support and frontline event trees. In other words, each specific split fraction value represents the frequency of failure of the event tree top event based on preceding scenario conditions. The RISKMAN software permits users to specify split fraction assignment logic rules that enable the program to implement the scenario-based selections.

When the appropriate information is provided, the RISKMAN software quantifies each sequence through the tree one at a time. The frequency of each sequence is computed by multiplying the initiating event frequency, expressed in units of expected events per year, by the product of the branch frequencies along that sequence path. Both success and failure branches are considered. To account for intersystem dependencies, the branching frequencies are quantified dependent on the status of preceding top events in the tree. Therefore, the branching frequency for the same top event but along different sequences in the tree may differ. Each top event may have two or more split fraction values for the same event tree.

The result of sequence assembly and quantification is a set of scenarios, each one leading to either success (i.e., as a safe and stable shutdown condition) or one of various plant damage states. The frequency of each sequence is therefore established. Together, the sequences include all of the possible combinations of success and failure of the event tree top events. The individual sequence frequencies are readily summed by plant damage state to determine the annual frequency of each plant damage state (and of successful mitigation). The sum total of all of the sequence frequencies is equal to the sum total of all the plant damage state frequencies (including success), which is also equal to the sum total of all of the initiating event frequencies. This is true because the event tree sequences represent a complete set of mutually exclusive outcomes of the initiating events.

2.3.5.7.1 Detailed Method for Event Tree Quantification

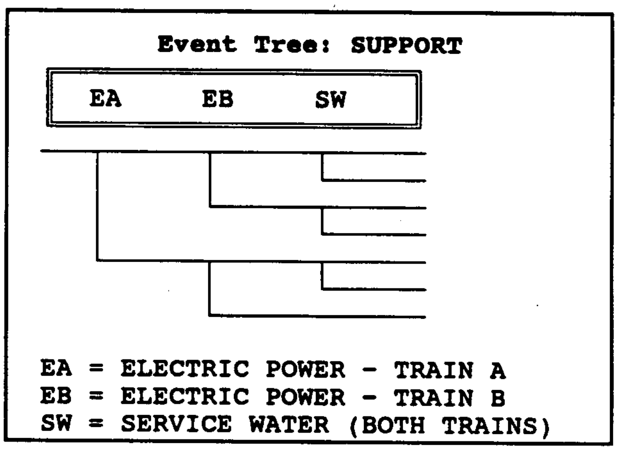
The assembly and quantification process can be best illustrated with a simple example.

For this simplified example, consider a nuclear power plant that has only one potential initiating event called TRIP, which has an annual frequency of 5.0 per year.

The actual event tree development and quantification are carried out by the RISKMAN software, with instructions provided by the user. To ensure traceability, the instructions that are provided are documented in reports that describe exactly which event trees, logic rules, master frequency files, truncation values, and other input values were used to support the quantification. The following figures are reproductions of those reports for this example. For the convenience of the analyst, the event sequences can be broken down into a series of linked event trees that model specific aspects of scenario development. Once inside the program, however, RISKMAN links together these trees to construct a single, large event tree that spans the entire scenario from initiating event to final end state.

For support of the frontline systems response to a trip, the plant has two trains of electric power and a service water system with a common discharge header. The service water system (SW) has two pumps, one powered from each electric power train. Either pump alone will provide sufficient cooling water.

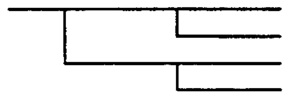
The functional system relationships are modeled in the support system event tree shown here. Note that the likelihood of SW failure depends on the availability of the two trains of electric power (on the success or failure of Top Events EA and EB); thus, the SW split fraction will be different for each path of the support systems event tree. (To keep this example simple, assume, unrealistically, that the failure frequencies of the two trains of electric power are completely independent so that the split fraction value for Top Event EB will be the same for each of its two failure paths; i.e. given success or failure of Top Event EA.)



The support systems provide power and cooling to two frontline systems: FA, which is powered by train A electric power (Top Event EA), and FB, which is powered by train B electric power (Top Event EB); both frontline systems are cooled by service water (SW).

Event Tree: FRONTLINE

FA FB



FA = FRONTLINE SYSTEM - TRAIN A
FB = FRONTLINE SYSTEM - TRAIN B

The frontline event tree structure models any functional relationships among frontline systems but is constructed independent of support system dependencies. Those dependencies are addressed by the split fraction definitions and logic rules.

Based on the analysis of intersystem support dependencies, the following split fractions have been defined and quantified using the split fractions from the master frequency file, which contains all split fraction values for the entire model. These values are determined by the systems models, from results of the human reliability analysis, and in some cases directly from plant data.

Master Frequency File: EXAMPLE

SF.......	SF Value..........	Split Fraction Description....................
EA1	1.0000E-03	
EB1	1.0000E-03	
FA1	1.0000E-02	
FAF	1.0000E+00	REQUIRED SUPPORT NOT AVAILABLE
FB1	9.0000E-03	CONDITIONAL ON FA SUCCESS
FB2	1.0000E-01	CONDITIONAL ON FA FAILURE
FB3	1.0000E-02	FA NOT CHALLENGED
FBF	1.0000E+00	REQUIRED SUPPORT NOT AVAILABLE
SW1	1.0000E-05	ALL SUPPORT AVAILABLE
SW2	5.0000E-03	ONLY TRAIN A POWER AVAILABLE
SW3	7.0000E-03	ONLY TRAIN B POWER AVAILABLE

The RISKMAN software employs split fraction logic rules to specify when a particular split fraction applies. The split fraction rules for the example support system event tree are shown below with an explanation of what each rule means. [For a more complete description of logic options, refer to the RISKMAN Users Manual (Reference 2-7).] For each event tree top event and for each path through the event tree, the first split fraction for which the logic rule is true specifies the split fraction to be used. (Note that the rule "1" is always true.) In this manner, the entire event tree can be quantified with the appropriate split fraction value being used at each event tree branch.

Split Fraction Logic for Event Tree: SUPPORT

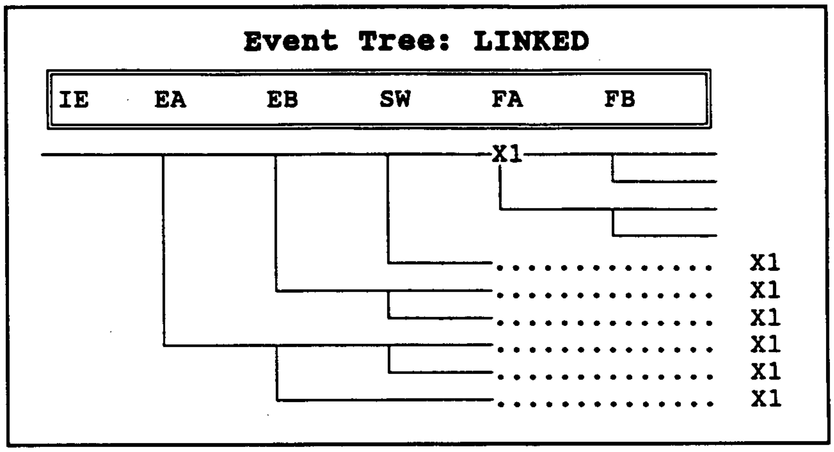
SF	SF Logic	Optional Text
EA1	1	Always use split fraction EA1 for Top Event EA.
EB1	1	Always use split fraction EB1 for Top Event EB. (This implies that Top Event EB is independent of Top Event EA.)
SW1	EA=S * EB=S	Use split fraction SW1 when both trains of electric power are available.
SW2	EA=S * EB=F	Use split fraction SW2 when only train A of electric power is available. (Note that this split fraction rule could have been simply EA=S since the one above - SW1 - captures the case of both trains available and RISKMAN uses the first split fraction that applies for each top event.)
SW3	EA=F * EB=S	Use split fraction SW3 when only train B power is available. Note that this rule could have been simply "1" since "only train B power available" is the only remaining possibility if none of the above (SW1 or SW2) applies.

For each frontline event tree top event in the model, the split fraction rules are written based on previous dependent top events (including the initiating event if appropriate), regardless of whether they occur in the same event tree or in a previous event tree. The minus sign before SW in the logic for the frontline Top Event FA is the RISKMAN symbol for the logic operator .NOT.

Split Fraction Logic for Event Tree: FRONTLINE

SF	SF Logic	Optional Text
FAF	EA=F + -SW=S	Use split fraction FAF (which = 1.0 - or guaranteed failure) whenever Top Event EA is failed or SW is not successful; i.e failed or bypassed.
FA1	1	Use split fraction FA1 whenever FAF does not apply.
FBF	EB=F + -SW=S	Use split fraction FBF (which = 1.0 - or guaranteed failure) whenever Top Event EB is failed or SW is not successful.
FB3	EA=F	Use split fraction FB3 whenever FBF does not apply and Top Event EA is failed. (For the case when Top Event EA is failed, FA is guaranteed failed; FA is effectively not challenged or "not asked.") FB3 is simply the failure probability of FB conditional on its support systems (Top Event EB and SW) being available but unconditional relative to FA.
FB1	FA=S	Use split fraction FB1 whenever neither FBF nor FB3 apply and FA is successful. (Use the failure probability for FB, which is conditional on the success of FA.)
FB2	FA=F	Use split fraction FB2 when neither FBF nor FB3 nor FB1 apply and FA has failed. Note that the logic rule here could have been simply "1" rather than FA=F since FA=F is the only remaining possibility. FB2 is the probability that FB fails, conditional on all support systems being available and FA having failed.

Complete event sequences can then be assembled by simply linking the appropriate combinations of initiating events and event trees (in the proper order) and quantified by simply multiplying initiating event frequencies by conditional split fraction values for each sequence as governed by the logic rules. Note that the value for each success branch is 1 minus the appropriate failure branch value. The linked support and frontline event trees are shown below.



For this simple example, a detailed RISKMAN report for one complete sequence and a summary report of all sequences are provided here. This single sequence is initiated by a turbine trip, involves the failure of train B of electric power, an independent failure of train A frontline system, and a guaranteed failure of train B frontline system due to the loss of train B electric power. For brevity of presentation, the summary report shows only failed top events.

Initiator: TRIP (frequency = 5.0 / year)
Sequence:

Top....	State..	SF.....	SF Value.......	Top Event / SF Description....................................
EA	S	EA1	9.9900E-01	ELECTRIC POWER - TRAIN A
EB	F	EB1	1.0000E-03	ELECTRIC POWER - TRAIN B
SW	S	SW2	9.9500E-01	SERVICE WATER (BOTH TRAINS) /ONLY TRAIN A POWER AVAILABLE
FA	F	FA1	1.0000E-02	FRONTLINE SYSTEM - TRAIN A
FB	F	FBF	1.0000E+00	FRONTLINE SYSTEM - TRAIN B /REQUIRED SUPPORT NOT AVAIL.

Sequence Frequency = 4.9700E-05 per year
End State = FAILURE

Detailed Report of All Sequences

Sequences For Initiator TRIP

Frequency.............	Failed SFs.......................	End State.........................
4.8956E+00		SUCCESS
4.4910E-02	FA1	DEGRADED
4.4460E-02	FB1	DEGRADED
4.9900E-03	FA1*FB2	FAILURE
4.9203E-03	EB1*FBF	DEGRADED
4.9104E-03	EA1*FAF	DEGRADED
4.9900E-05	SW1*FAF*FBF	FAILURE
4.9700E-05	EB1*FA1*FBF	FAILURE
4.9600E-05	EA1*FAF*FB3	FAILURE
3.4965E-05	EA1*SW3*FAF*FBF	FAILURE
2.4975E-05	EB1*SW2*FAF*FBF	FAILURE
5.0000E-06	EA1*EB1*FAF*FBF	FAILURE

Summary Report of all Sequences

The end state of each sequence shows the plant damage states into which it is binned. This binning occurs per binning logic rules, which are very similar to split fraction logic rules. The figure below is the RISKMAN summary of the binning rules used for this example.

Binning Rules for Event Tree: FRONTLINE

Bin.................	Binning Rules / Optional Text..
SUCCESS	FA=S * FB=S When both frontline systems FA and FB are successful, the event TRIP has been mitigated without any plant damage; such sequences are binned to the PDS SUCCESS. These sequences will not be propagated to Level II/III; analysis of them is complete.
DEGRADED	FA=S*(-FB=S) + (-FA=S)*FB=S When either of the two frontline systems (but only one) has been successful, event TRIP has resulted in some plant damage; such sequences are binned to the PDS DEGRADED. (Note that the logic rule for this PDS bin could have been written as simply FA=S + FB=S since the case of both successful would already be captured above.)
FAILURE	FA=F * FB=F When neither FA nor FB succeed, TRIP has not been mitigated. These accident sequences bin to PDS FAILURE.

One unique RISKMAN feature for performing a quantification operation is illustrated in the following figure. The particular initiating event shown in the figure, encoded "AOLOSP," is being analyzed with a series of five linked event trees and a truncation frequency of 1.0×10^{-12}. The RISKMAN program, upon execution, constructs a single, large event tree by exhaustively linking each successive event tree at the end of each sequence from the previous tree. As explained in the previous example, the logic rules are used to select the appropriate split fraction values for computation of each sequence frequency. When the cumulative sequence frequency drops below the user-specified cutoff, the frequency to that point is added to an unaccounted-for bin, which, in this example, totaled to

1.4×10^{-9} per year. This quantification produced results listed by event tree end state under "binning information." (If the analyst forgets the definitions of the various codes, additional reports can be accessed with word definitions of each code.) In this example, the sequence frequencies for a total of 3,560 sequences were computed. All of these sequences have frequencies greater 1.0×10^{-12} per year. Ninety-seven sequences were saved to the sequence database for further analysis. (The remainder were found to be "success.")

```
Results of Last Calculation              Binning Information
---------------------------           --------------------------

Initiating Event      AOLOSP           SUCCESS          8.8538E-04
Initiating Frequency 1.0000E-03        HINOHR           1.0509E-04
                                       SYNOHR           6.4341E-06
Tree Names SUPPORT                     HIWCHR           9.8702E-07
           SUPPORT2                    HISBYP           8.5199E-07
           GENTRANS                    MDWCHR           8.0211E-07
           GTRECIRC                    SYNISO           2.5136E-07
           EPRECOVERY                  MDNOHR           9.6778E-08
                                       HINISO           8.8825E-08
                                       SYWCHR           8.5328E-09
                                       MDNISO           5.0908E-09
                                       MDSBYP           1.9006E-12
                                       LONISO           0.0000E+00
                                       LOSBYP           0.0000E+00
Sequences Quantified  3560             LOLBYP           0.0000E+00
Sequences Saved       97               LOWCHR           0.0000E+00
Seconds     736.710000                 SYSBYP           0.0000E+00
Cutoff      1.0000E-12                 LONOHR           0.0000E+00
Unaccounted For: 1.4006E-09
```

Event Tree Quantification Screen for RISKMAN
Evaluation of Initiator "AOLOSP"

The event tree quantification can be performed on initiating events individually or in a batch mode, with designated initiating events quantified in the same run, but each initiating event set up with a specification for quantification as illustrated above.

2.3.5.7.2 Review and Interpretation of Results

The final steps in sequence quantification and integration are to review and interpret the results. The principal goals of this review are to determine the key risk-controlling factors that determine severe accident frequency and to develop engineering insights needed to control this frequency to adequately low values. There are a variety of ways to dissect and analyze the results to support these goals. These include:

- Review individual top-ranking sequences that comprise the major portion of the core damage frequency.
- Determine the major classes of sequences that comprise the major portion of the core damage frequency. Convenient ways to classify sequences are by common

initiating event, common plant damage state, or common sequence characteristics; e.g., pump seal LOCA.

- Quantify risk importance measures that identify systems, components, and operator actions that are the most important in the risk determination.

The RISKMAN software provides a number of reports that address each of the above three ways to analyze the results. One report is a ranking of sequences by frequency, wherein each sequence satisfies particular logic rules specified by the user; e.g. all sequences involving failure of offsite power. An example of this report is presented in Figure 2-13 for the category of sequences of core damage from a typical PRA study. The sequence is organized to first describe the initiating event and additional failures that are not a direct result of the initiating event but contribute to the determination of the sequence frequency. The second column lists all of the consequential failures or events that are a direct consequence of the events in the first column. These "guaranteed events" result from dependencies of various types; e.g., functional, spatial, and human dependencies. Even though they do not affect the sequence frequencies (i.e., they all have conditional split fraction values of 1.0), they are important to understanding the nature of the accident sequence and to verifying the proper application of human reliability models. When reviewed with the dependency matrices, these sequence reports have proven to be extremely useful in verifying that dependencies have been properly modeled. This report can be developed for each plant damage state as well as for the total core damage frequency for the Level 1 sequence quantification.

RISKMAN provides tables that break down the total accident frequency by end state, initiating event category, and any other grouping for which a logic rule can be defined that relates to information tracked in the sequence database. The user can also select from three different types of importance reports as illustrated in Figures 2-14 through 2-16. These reports ascribe importance to portions of the model at three different levels of detail: top event, split fraction, and basic event.

- The top event level covers all of the plant equipment or operator actions modeled in each event tree top event and distinguishes between independent and dependent failures of equipment.

- The split fraction level addresses several different importance measures that apply to the distinct conditional quantification cases addressed in the event sequence model.

- The risk achievement factor in Figure 2-15 shows the effect of taking equipment associated with that split fraction out of service on the final result.

- The overall importance of individual basic events can be examined to identify very specific causes of the final result as shown in Figure 2-16.

Information provided in these reports is extremely useful in identifying cost-effective risk management strategies from the PRA results by pinpointing the source of the risk. These importance measures are used to interpret the key factors driving the determination of core damage frequency and in the identification and evaluation of vulnerabilities.

2.3.5.8 Quantification of Uncertainties

The event tree computations outlined above must account for a variety of sources of uncertainty that prevent the development of highly accurate estimates of accident sequence frequencies. These sources of uncertainty include the lack or sparsity of data from which to quantify the risk model input parameters (i.e., component failure rates, initiating event frequencies, etc.), plant-to-plant variability in the performance of similar equipment at other plants, modeling uncertainty, equipment behavior in harsh environments, uncertainty in classification of common cause event data, and many other sources.

The overall flow of data associated with the quantification of uncertainty is illustrated in Figure 2-17. This figure shows the four principal modules of the RISKMAN software program and identifies where point estimates and full distribution results are obtained.

The basic approach to quantifying the effects of uncertainties on the PRA results is to determine the appropriate probability distributions for each uncertain parameter in the analysis. Those assignments are made with the use of data analysis software that uses Bayesian updating techniques for incorporating operating experience from other plants, expert opinion, and plant-specific data.

The RISKMAN Data Analysis Module outputs both the actual distribution associated with the parameters and its major characteristics; e.g., the mean value. The propagation of these uncertainties is then done in two stages.

The RISKMAN Systems Analysis Module combines the individual failure rates, maintenance, and common cause parameters into the split fraction frequencies that will be used by the event sequence model. During model development and debugging, it uses the mean values of the individual parameters directly to provide a point estimate of split fraction frequencies. Once the model is finalized, a Monte Carlo routine is used with the complete distributions to calculate the split fraction frequencies. Like the data module, the systems module outputs both the actual distribution associated with the parameters and its major characteristics; e.g., the mean value.

When the event trees are quantified and linked together in the Event Tree Analysis Module, the mean values of the split fractions obtained from the systems Monte Carlo runs are used. These point estimates approximate the mean values of the sequence frequencies because mean values of the split fraction, initiating event, and human error rate distributions are used from the preceding steps of the uncertainty propagation.

It is important that the event trees be quantified using the means of the system-level Monte Carlo results rather than the point estimates used during model development and debugging. Although the systems may be quantified using point estimates for the purpose of reviewing and screening, these estimates frequently underestimate the means of the system-level uncertainty distributions because of the coupling of failure rates. Past experience has shown the use of system unavailability Monte Carlo means for event tree quantification to be important for redundant systems.

The uncertainty in the overall core damage frequency and other risk measures is computed by the Important Sequence Module of RISKMAN. The important sequence module accepts a prioritized set of the most important sequences from the results of the Event Tree Analysis module for predefined groups of accident sequences. These groups include, for example:

- All core damage sequences.
- Each plant damage state or plant damage state group.
- Each release category or release category group.
- Each initiating event.
- Any other groups of interest.

For each group, the equations for the frequency of each sequence within that group are used in another Monte Carlo sampling step to propagate the split fraction uncertainties and obtain the uncertainties in the overall results. An uncertainty propagation accomplished in this manner encompasses the major contributors to risk while permitting efficiency in calculation.

2.3.6 CONTAINMENT PERFORMANCE ANALYSIS (LEVEL 2)

The back-end (or core and containment response) analysis addresses the physical progression of accident sequences from the onset of core damage through the release of radionuclides into the environment. In this section, the terms "back-end," "containment response," "containment performance," and "Level 2 PRA" are used interchangeably. This arises from the fact that a limited Level 2 PRA is performed in this study to meet the IPE requirements for the containment performance analysis of Watts Bar.

The back-end analysis characterizes (in terms of fission product source terms) the impact of each severe accident sequence on the mode, timing, and magnitude of radionuclides released from the plant. This characterization is accomplished through a range of deterministic engineering analyses of the physical processes that determine core melt progression, containment response, containment failure mode (if any), and the transport and release of radionuclides. These analyses determine such physical parameters as the

- Containment pressure and temperature as a function of time.
- Pressure at which the containment will fail.
- Containment failure mode.
- The rate at which the molten debris penetrates into the concrete basemat.
- The rate and quantity of hydrogen produced and released into the containment.

The probabilistic quantification for the containment event tree is a statement of the analyst's confidence about the outcome of a severe accident. A unique CET quantification can be defined for individual sequences or for each group of severe accident sequences having the same plant damage state. Different split fractions for the containment event tree nodes characterize the different plant damage states.

The end products of the back-end analysis include:

- A set of release categories that defines the radionuclide releases into the environment, and a quantification of the frequency of each release category. The release categories constitute the endpoints of this Level 2 PRA, and the associated source terms provide a measure of the potential consequences of severe accidents.

- The identification of individual accident sequences whose frequencies exceed the screening frequency prescribed in NUREG-1335 (Reference 2-2). This product may be the most important of all because it is the key to the development of insights into plant safety characteristics.

The scope of the back-end analysis includes:

1. The definition of plant damage state (PDS) parameters (e.g., reactor coolant system (RCS) pressure or status of containment integrity) applicable to Watts Bar. Various combinations of these parameters are referred to as plant damage states. The thermal-hydraulic response of sequences assigned to a given PDS is expected to be very similar.

2. The selection of key plant damage states (KPDS) and accident sequence(s) to represent these KPDSs.

3. The determination of containment failure modes.

4. The determination of the core and containment response for each important plant damage state.

5. The development and quantification of the containment event tree.

6. The definition of radionuclide release categories as a function of the degree of core damage, and the mode and timing of containment failure.

7. The quantification of the frequency of each release category and a description of the important sequences contributing to each release category and the determination of the uncertainties in the containment response quantification for the risk-significant containment response sequences and release categories. Such uncertainties can also be assessed for risk-significant PDSs.

The following sections provide more details regarding the Level 2 quantification process.

2.3.6.1 Definition of Plant Damage States

Each sequence through the frontline event tree eventually ends at a safe shutdown state or a plant damage state. Plant damage states result from core melt scenarios and are defined in terms of the

- Conditions in the reactor vessel at core damage.

- Type and degree of core damage.

- Status of the containment safety/mitigation systems that result from the failures in the sequence of events making up the Level 1 sequence.

Plant damage states are chosen and defined with sufficient specificity that, once such a state has occurred, the subsequent events in the containment are the same regardless of the path by which that state was reached. As a result of this definition, a coalescence of scenarios occurs at this point that structures the scenarios list and greatly simplifies the computational labor involved in the containment event tree analysis. This coalescence is commonly referred to as establishing a **pinch point**.

The logic by which one selects plant damage states for the end of a Level 1 sequence must be established by logic rules established using techniques similar to those used for the event sequence model. The logic rules are supplemented by knowledge of the physical core damage mechanisms generated by the failures in the scenario. The process for Watts Bar is discussed in more detail in Section 4.3.

2.3.6.2 Key Plant Damage States and Representative Sequences

Level 1 model quantification will identify a significant number of PDSs that will be reached with some frequency. For Level 2 analysis, these PDSs are condensed into a reduced set of KPDSs. This condensation process involves the binning of PDSs with lower frequency and anticipated lower consequences, to PDSs with higher frequency and higher consequences. Thus, in this process, the total core damage frequency of the KPDSs is equal to the total frequency of all of the Level 1 plant damage states. However, some conservatism is introduced since the frequency of those plant damage states that are eliminated in this process is assigned to states of higher consequence.

Once KPDSs are identified, accident sequences are selected to represent each of the KPDSs for detailed analysis with the **Modular Accident Analysis Program** (MAAP Reference 2-30). Since these sequences can result in different source terms, depending on the pathway through the CET (e.g., success or failure of CET top events related to multiple containment failure), different versions of the representative accident sequences are analyzed to address consequence uncertainties.

2.3.6.3 Containment Pressure Capacity Analyses

Containment pressure capacity analyses have been performed for a number of PRAs. In most cases, the emphasis is placed on determining the pressure at which membrane failure of the containment structure occurs. In a broader sense, it is possible that one of several failure modes occurs first, and we cannot say for sure which one will occur at the lowest pressure. In addition, for containment designs that allow large shell deformations before failure, failure modes that result from structural interferences are possible. Large deformations are considered possible in freestanding steel containments and in steel lined reinforced concrete containments. Post-tensioned containment structures are considered to have a lesser potential for large strain because of the major discontinuities in the concrete structure and because of the designed accommodation of slip between the concrete and the tendons, which allows local accumulation of strains. In any containment type, local failure can occur where large strains can locally accumulate. This failure mode was observed in the concrete containment pressure test at Sandia National Laboratory (SNL). However, some containments provide features designed to avoid such local failure modes.

A containment pressure capacity in support of Level 2 PRA should answer the following questions:

- At which locations can failure occur?
- What is the most likely failure pressure for each location?
- What is the most likely failure area (leak area) for each failure location?
- What are the uncertainties in the failure pressures for the lowest pressure failure locations?
- What are the uncertainties in the failure area for the leak-type failure locations?

The PRA Level 2 analyst then uses this information to develop a containment failure pressure uncertainty distribution of the type shown in Figure 2-18. For Watts Bar, the containment pressure capacity was reviewed to determine the applicability of the findings in Reference 2-4 for Sequoyah.

2.3.6.4 Core and Containment Response Analysis

Deterministic analyses of accident progression phenomena form the foundation for the Level 2 analysis. A large number of calculations have been performed in the past under several U.S. and international programs. For example, the IDCOR program, sponsored by the U.S. nuclear power utilities, examined severe accident phenomena. The MAAP code was developed as part of this program. In addition, a number of safety research programs have been sponsored by the NRC at U.S. National Laboratories. These include the SARP and the NUREG-1150 program, both performed at SNL.

Those and other research programs as well as analyses performed for past PRAs have developed a large foundation for analysis that can be useful for any Level 2 PRA in providing the current state of knowledge as well as, in some cases, directly applicable analyses. Some of the major computer codes used for deterministic accident progression analyses in support of Level 2 PRAs are

- MAAP — IDCOR Accident Progression and Source Terms
- STCP — NRC Source Term Code Package
- HECTR — Containment Hydrogen Burn Analysis
- CORCON — Concrete Penetration Analysis
- CONTAIN — Containment Response Analysis
- NAUA — Containment Aerosol Analysis
- LTAS — Boiling Water Reactor Accident Progression Analysis
- THALES — JAERI Accident Progression Analysis

The extent to which these or other codes are used in support of a Level 2 PRA varies greatly. For the Level 2 portion of an IPE, the NRC has specified that no plant-specific calculation of accident progression, containment strength, and source terms is required, as long as the information that is used to support the probabilistic containment event tree quantification can be justified.

Plant-specific analyses of containment strength, accident progression, and source terms must be selected judiciously because such analyses are usually time and labor intensive. A reasonable approach to determine which analyses are needed is outlined below.

Supporting deterministic analyses for a Level 2 PRA usually fall into one of the following categories:

- Containment Pressure Capacity, Failure Locations, Failure Modes, and Failure Area Size

- Accident Progression and Containment Pressure and Temperature Response

- Source Term and Release Calculations

For the Watts Bar IPE submittal, MAAP has been selected as the primary tool for accident progression, including the determination of source terms. For certain phenomena with large uncertainty (e.g., thermally induced failures of the RCS hot leg or steam generator tubes), data will be taken directly from NUREG-1150.

2.3.6.5 Containment Event Tree Development

The CET is the logic model that is used to describe the various scenarios by which an accident sequence involving core damage can lead to the release of radioactivity into the environment. The development and quantification of a containment event tree require the definition of

- Top Events for the CET
- Logic for Developing the CET Structure
- Assignment of Each CET End State to a Release Category

Top events on the CET are selected to represent those physical processes in the containment that can influence the time, quantity, and location of released radionuclides during the progression of core melt and relocation. The number of top events used to describe that process will directly influence the complexity of the CET. While, on the one hand, the important processes must be included in the top events, care must also be taken to keep the tree to a size that permits it to be used as an effective tool for communicating the overall containment response behavior for a given plant.

The CET defines the success and failure states of the containment that can result from core melt accidents. In general, these states are too numerous for a detailed radiological analysis to be feasible for each sequence. Furthermore, such analyses are not necessary because the quantities and timing of radionuclide releases into the environment are similar for many of the containment tree sequences. The CET end states are examined and evaluated to determine the number and type of release categories that are required for the consequence analysis. CET end states with similar release characteristics in quantity and timing are binned into appropriate release categories.

2.3.6.5.1 Selection and Ordering of Top Events

In general, the top events in the CET are time sequenced and consider major phenomenological events that could occur for the formation and relocation of core debris. Processes or events are included as top events in the CET if they are significant in one or more of the five following categories:

1. Processes that establish a safe, stable state; i.e., prevent threats to containment integrity due to vessel breach.

2. Events that isolate significant dependencies between events that appear later in the tree.

3. Containment failure events.

4. Events that define containment leakage (size, path) after failure.

5. Events that influence the magnitude and characterization of the radioactive materials released into the environment; i.e., source terms.

2.3.6.5.2 CET Top Event Dependencies and Combinations

The response to many of the top event questions typically identified for CETs can be dependent on the responses to one or more previous questions. The following are examples of such dependencies:

- If the containment fails early, subsequent top event questions need not be addressed unless more severe containment failure modes are possible.

- Early hydrogen burns will impact the quantity of hydrogen available for later burns, the amount of oxygen available for subsequent burns, and the concentration of diluent in the atmosphere.

- Whether debris is dispersed at the time of vessel failure has a strong influence on whether ex-vessel debris cooling can occur.

There can also be dependencies between the CET top event questions and the PDS. For example, for PDSs characterized by low RCS pressure at the time of reactor vessel breach (e.g., less than 200 psia), significant debris dispersal is unlikely. Thus, for such PDSs, the failure fraction for this top event can be taken as 1.0.

In the Level 2 methodology described in References 2-20 and 2-21, dependencies between top events are modeled in a manner similar to the dependencies between top events in the Level 1 plant model. Multiple split fraction values are defined for those CET top event questions exhibiting dependencies on the response to previous top event questions. The choice of which split fraction is to be used is controlled by the logic of the tree.

The Event Tree Analysis Module of RISKMAN (Reference 2-7) provides the tools and structure to (1) develop the detailed logic for the CET, (2) propagate the assigned branch probabilities for each sequence, and (3) combine the probabilities of all sequences belonging to each end state bin; i.e., release category. For Watts Bar, RISKMAN is used to quantify the CET separately for each PDS that has both significant frequency and distinct branching probabilities. It has the capability to address the conditional branching probabilities (i.e., multiple split fraction values for the same CET top event) discussed earlier in this section.

2.3.6.6 Radionuclide Release Characteristics

To satisfy the requirements of a Level 2 PRA, it is necessary to calculate the quantity of radioactive materials released into the environment for each accident sequence involving significant core damage. Such releases into the environment are affected by

- The timing and failure mode of inherent barriers to such releases, such as the fuel pellet itself, the fuel rod, the reactor coolant system, and the containment.

- Physical-chemical processes, such as plate-out in the reactor and containment, fallout in the containment, revaporization, aerosols produced by pressurized ejection from the reactor vessel, core debris-concrete interactions, direct heating, and other processes that determine the integrity of the containment and the driving forces for release.

- Passive engineered safety features; e.g., an ice condenser.

- Active engineered safety features; e.g., containment sprays.

Source term analysis involves the characterization of the release of radioactive material into the environment accounting for all of the phenomena listed above. Source terms are typically characterized by the fractions of the initial core inventory of radionuclides that are released into the environment as well as the time dependence of the release, size distribution of the aerosols released, elevation of the release, time of containment failure, warning time, and the energy released with the radioactive material. These characterizations are required as input to the codes that assess offsite consequences.

To assess the release of radioactive materials into the environment, an extensive analysis of events that occur in the containment prior to, simultaneous with, and after failure of the reactor vessel is required. The usual approach to organizing such analysis is through the use of a CET. In this study, events that occur prior to core damage (e.g., containment isolation failures and unavailability of containment sprays) are identified in the Level 1 plant model and reflected in the characterization of plant damage states.

Each accident sequence involving core damage has its own probability of occurrence and release characteristics describing the timing, mode, and magnitude of the radionuclides released to the environment. However, as noted earlier, accident sequences in which the plant response is similar are binned into PDSs. The frequencies of these PDSs represents the "end product" of a Level 1 PRA and the entry points to the Level 2 analysis. Thus, instead of quantifying the containment event tree(s) for each accident sequence, the CET is, in principle, quantified for each PDS.

Depending on the number of top events considered in the tree, the number of CET end states is potentially very large. However, a unique analysis of each sequence is unnecessary because the quantities and timing of radionuclide release into the environment are similar for many of the CET sequences. Thus, CET end states are binned into release categories in a manner similar to that in which plant model event tree (Level 1) end states are binned into PDSs. This process of binning CET end states into release categories is an important step in the overall quantification process, and can have a significant impact on the results. Therefore, the binning process must be performed with great care.

CET sequences are mapped to end states according to similarities in the quantity of materials released and the mode and timing of containment failure. These end states are subsequently binned into release categories to simplify the calculation of risk. The most important distinction among the types of releases is whether they are initially released into the atmosphere or to the ground; e.g., basemat melt-through. In general, atmospheric releases require substantially less time to reach the populace, whereas underground releases require greater time, could appear in liquid pathways, and could undergo substantial deposition prior to release into the environment.

Level 2 PRAs generally are concerned only with airborne releases. Given that only airborne releases are being considered, the radionuclides are first binned into groups of elements that exhibit similar chemical behavior. For example, the noble gases (such as xenon and krypton) are binned into a simple group, as are the halogens (such as iodine and bromine). This approach is adopted in this study for Watts Bar.

Release categories are also characterized by the timing and mode of containment failure. Timing is important for a number of reasons. First, the time at which containment fails establishes the time during which material deposition processes, radioactive decay, and actuation of engineered safety features affect the quantity of radioactive materials released into the environment. In addition, the timing for release is important in emergency response, particularly evacuation.

From the standpoint of engineered safety features, the containment is typically the final barrier to the release of fission products into the environment. However, in many reactor designs, secondary containments are provided that can mitigate the release even further, provided that the failure of the containment is not sufficiently energetic to fail the secondary containments as well. In some plants, fission products can be naturally depleted in these secondary containments, and/or the release occurs through filters and the plant stack. Failures of the containment that result in direct releases into the environment should be differentiated according to the type of failure (e.g., catastrophic depressurization versus slow leak), elevation of the failure, and the amount of energy involved. The latter two characteristics are important since structures in the immediate vicinity and the local terrain can have significant impacts on plume trajectory and dispersion. Thus, important containment failure modes and the time of failure are important differentiators in release category definition.

The timing of radionuclide release into the environment has important effects on offsite consequences. In particular, the probability of an effective emergency response increases significantly with an increasing delay prior to release.

The amount of thermal energy associated with a release impacts the effective elevation of the release and therefore the downwind distance at which the plume first contacts the ground. In the absence of precipitation, buoyant plumes usually do not contact the ground until they are substantially dispersed. Thus, the number of early effects is usually lower for buoyant releases than for nonbuoyant releases.

Because of their importance in establishing the chemical form of the release, certain important phenomena, such as steam explosions, vessel debris cooling (i.e., core debris-concrete interactions), pressurized ejection from the reactor vessel, and associated direct heating phenomena, may also be used as differentiators in defining release categories.

In practice, a large number of potential release categories are identified. However, it is expected that many of the defined release categories will have little or no frequency once the containment event tree is quantified. Thus, the generation of source terms is not required for all release categories. Furthermore, some rebinning of release categories into "key release categories" can be accomplished to limit the number of categories considered individually. This rebinning must be performed with caution, however, since some release categories (e.g., those associated with containment bypass) can make significant contributions to risk even though their frequencies are one or more orders of magnitude smaller than those release categories with the greatest frequency.

2.3.6.7 Containment Event Tree Quantification

Quantification of the CET includes the quantification of split fractions for the top events, combining those split fractions to determine the conditional properties for each sequence, the assignment of each CET sequence to a fission product release category, and the summation of all sequence probabilities within each release category.

The calculation of CET split fractions is based on the analysis of potential containment failure modes and the loads imposed on the containment during the course of accident progression. For some PDSs, the split fraction for certain CET top events could take on the values of 0.0 or 1.0 since guaranteed success or failure of the top event is guaranteed by the PDS definition or by the response to earlier questions in the CET. This is, however, only an analytical convenience because the same CET structure is used to quantify all PDSs. In many cases, the values assigned to a split fraction for a given top event will depend on what branches were taken earlier in the tree. For example, the probability of a late hydrogen burn is dependent on whether hydrogen burned previously.

The engineering bases for split fraction quantification consist of containment failure mode analyses and thermal-hydraulic analysis of the conditions inside the containment as the accident progresses. For the most part, these analyses are deterministic and yield point estimates for the parameter of interest. For example, for the CET top event question, "Does the containment remain intact following vessel breach?" the peak pressure that is predicted for this event for the accident sequences representing the PDS is compared with the pressure predicted for containment failure. If the margin between the two pressures is very large, we are confident that the split fraction will be close to 0.0 for failure. If the uncertainties in these pressures can be determined, then the stress-strength interference theory approach can be used to determine the value of the split fraction.

Once dominant CET sequences are determined using point estimates for the split fractions, the top events that are responsible for the importance of the sequence can be identified, and an explicit evaluation of the uncertainties in the split fraction for those top events can be performed using the important sequence model of RISKMAN.

Once estimates are available for the split fractions for each of the CET top events (for each KPDS), quantification of release category frequencies is accomplished using the RISKMAN program in a manner that is analogous to the quantification of PDSs in the Level 1 model.

2.4 INFORMATION ASSEMBLY

2.4.1 PLANT DOCUMENTATION

The Watts Bar PRA is based primarily on the plant-specific information that is contained in the documents identified in Table 2-8. The major exception involves generic issues such as RCP seal LOCA information and the PRA database. The PRA database, based on generic nuclear plant and component data, is discussed further in Section 3.3.2.

2.4.2 REVIEW OF OTHER PRAs AND INSIGHTS

The IPE for Watts Bar is the responsibility of the TVA Risk Assessment Staff (RAS). The group reviews industry and NRC studies and participates in industry groups such as IDCOR and EPRI. Additionally, PRA consultants with extensive experience in PRA were used in the IPE. RAS has reviewed NUREG/CR-4405 (Reference 2-31) for generic insights and consulted the Sequoyah PRA in NUREG/CR-4550 (Reference 2-32) and NUREG/CR-4551 (Reference 2-33) for plant-specific insights (Watts Bar is very similar to Sequoyah.) WCAP-11769 (Reference 2-34), an IDCOR IPE methodology report specific to Sequoyah, was also examined. Additionally, many of the references listed in Attachment 2 to NRC Generic Letter No. 88-20 were examined for insights and lessons learned. Some examples follow.

- Both of the Sequoyah-specific studies above concluded that small LOCAs were significant contributors to risk. One key element to their contribution was evidence that containment spray is initiated shortly after the LOCA. This depletes the RWST and does not allow depressurization and cooldown to RHR before recirculation is required.

- Both of the Sequoyah-specific studies were reviewed for success criteria in establishing the success criteria for the IPE.

- The NUREG-1150 expert opinion information was used in areas such as reactor coolant pump seal LOCAs and direct containment heating.

2.4.3 WALK-THROUGH ACTIVITIES

Walk-throughs were conducted for the internal flood analysis, the ISLOCA analysis, and the Level 2 back-end analysis. The team for the flooding and ISLOCA activities included the task leader, RAS plant model engineer, and an operator. The containment walk-through team for the Watts Bar back-end analysis included the task leader, the RAS Level 2 engineer, a Watts Bar operator, and two individuals responsible for the MAAP Watts Bar model.

2.5 REFERENCES

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Table 2-1. References for Technical Approach to PRA	
Technical Area	**References**
Basic Concepts of Risk, Probability, and Frequency	2-15
Event Sequence Diagrams	2-4, 2-16
Fault Tree Analysis	2-1
Treatment of Dependent Events	2-17
Common Cause Failure Analysis	2-8
PRA Database and Treatment of Uncertainties	2-18, 2-19
Treatment of Human Reliability	2-3, 2-9
Level 2 PRA Methodology	2-2, 2-20
Containment Failure Probability Estimation	2-2, 2-21
Treatment of Interfacing Systems LOCA	2-22, 2-23
Risk Quantification Using RISKMAN	2-10, 2-11
Risk Analysis of Internal Floods	2-16

Table 2-2 (Page 1 of 4). Glossary of Terms Used in Probabilistic Risk Assessment	
Term	**Definition**
Availability	The probability that a component or system will be able to operate successfully at a random point in time.
Common Cause Event	An event that adversely affects the performance of two or more components at the same time due to a single shared cause. When the performance degrades to the extent that component failures result, the event is referred to as a common cause failure. When the consequences of the event include the occurrence of an accident sequence initiating event, the event is called a common cause initiating event.
Containment Event Tree	Addresses the sequence of key phenomenological issues associated with the response of the containment to severe accidents. Each sequence through the CET is assigned to a release category.
Dependency Matrices	Tables describing the impact on systems listed across the top of the tables caused by the failure of other systems defining the rows of the tables.
Dependent Failures	Two or more failures are dependent when the probability of any event in the set is dependent on the occurrence or nonoccurrence of any other event in the set. The two failures A and B are independent events if and only if: $\text{Prob}[A \text{ and } B] = \text{Prob}[A]*\text{Prob}[B]$ This requires that: $\text{Prob}[BA] = \text{Prob}[B]$ and $\text{Prob}[AB] = \text{Prob}[A]$
Event Sequence Diagram (ESD)	A logic block diagram that defines event sequence progression from the initiating event to ultimate end state. The ESD documents the PRA analysts understanding about how event sequences develop in terms of system responses, plant conditions and operator actions. They are used as a communication tool with plant operations personnel to elicit their input into the PRA model development and its quantification.
Event Tree	A logical network of event sequences that is cast in a form to quantify accident frequencies. It contains much of the same information as in the ESD but reflects key modeling assumptions that are needed to support risk quantification.

Table 2-2 (Page 2 of 4). Glossary of Terms Used in Probabilistic Risk Assessment	
Term	**Definition**
Stable Shutdown	State of reactor when it is not neutronically "criticial" and its decay heat is being adequately dissipated under steady state condition.
Fault Tree	A logic diagram that is used to determine the logical combination of causes that will produce an undesirable event. Fault trees are used to determine the logical combination of causes that would result in failure or unavailability of each system that was modeled in the event sequence models.
Frequency	The quantification of expected occurrences per trial. In principle, frequency can be experimentally measured.
Initiating Event	Any event that perturbs the steady state operation of the plant such that a transient is initiated in the plant. Initiating events are the starting points of defining event sequences in the probabilistic risk assessment (PRA) because they directly result in challenges to the plant control and safety systems such that depending on the response to these challenges, accident scenarios could result.
Interfacing Systems LOCA	A breach in a system that interfaces with the RCS could cause a loss of coolant accident, if the breach is not isolated from the RCS. Such a breach could be caused if valves fail to isolate the RCS from an interfacing system not designed for the high RCS pressures. When portions of an interfacing system are located outside the containment, particular concern arises because an unisolated system breach outside containment can result in a release of radionuclides that bypasses the containment. Interfacing system LOCAs that bypass the containment were recognized in WASH-1400, where they were referred to as a V-sequence.
Minimal Cutset	The smallest combination of component failures which, if they all occur, will cause the top event of a fault tree to occur
Modular Accident Analysis Program (MAAP)	A computer code that simulates LWR system response to accident initiation events. MAAP is an integrated accident analysis code that treats all important engineered systems and a wide spectrum of phenomena, ranging from core heatup and cladding oxidation through vessel failure, core-concrete interaction, and ignition of combustible gases, to fission product release, transport and deposition.

Table 2-2 (Page 3 of 4). Glossary of Terms Used in Probabilistic Risk Assessment

Term	Definition
House Event	A conditional event used in quantitative fault tree evaluations. It is used as as switch (having the value of either success or failed) to allow the fault tree to model response to specific sets of accident sequences.
Pinch Point	The grouping of similar sequences through the front-end event trees into bins or plant damage states. All sequences assigned to the same PDS are evaluated as if they were identical, when performing the back-end analysis of containment response. A pinching of the analysis occurs at the interface point between the front-end and back-end analysis.
Plant Damage States (PDS)	Define the condition of the plant at the time of onset of severe core damage. The conditions considered in the definition of these plant damage states are those that determine the capability of the containment to cope with a core damage accident and potential for release of radioactive material. Plant damage states serve as the end states of the Level 1 portion of the accident sequence models that address the performance of all active plant systems and the entry states of the Level 2 portion of the event sequence model that address severe accident phenomena and containment structural response.
Probabilistic Risk Assessment (PRA)	A probabilistic analysis of the risk (consequence per unit time) posed by the system or plant being analyzed. A PRA analyzes the frequency of occurrence and consequence of a particular scenario or set of scenarios, including uncertainties in both.
Probability	A numerical measure of a state of knowledge, a degree of belief, or a state of confidence.
Release Categories	Define major classes of accident sequences in terms of the nature, timing and magnitude of the release of radioactive material from the plant during a core damage accident. The factors addressed in the definition of the release categories include the response of the containment structure, timing and mode of containment failure, timing and magnitude and radionuclide mix of any releases of radioactive material, thermal energy of release, and key factors affecting deposition and filtration of radionuclides. Release categories define the end states of the Level 2 portion of the event sequence model.

Table 2-2 (Page 4 of 4). Glossary of Terms Used in Probabilistic Risk Assessment	
Term	**Definition**
Source Term	The radiological source term for a given accident sequence or release category consists of the release fractions for various radionuclide groups (expressed as fractions of initial core inventory), the timing of the release, the elevation of the release, and the energy of the release.
Split Fraction	A split fraction (also called a branching ratio) is a parameter used in quantifying an event tree. It represents the fraction of the time (or probability) that each possible outcome, or branch, of a particular top event may be expected to occur. Split fractions may be conditional on precursor events. At any branch point, the sum of all the split fractions (i.e., the sum of the probabilities of all possible outcomes), must be unity.
Top Events	Event tree top events are the conditions that are considered at each branch point of an event tree. They may address system behavior or operability, or phenomenological events. A particular event tree sequence can be described in terms of the status of the plant relative to each top event. Fault tree top events are at the very top of the fault tree. These events represent the undesirable outcome of a set of failures involving the system modeled in the fault tree. These events are defined to support split fraction quantification.

Table 2-3 (Page 1 of 2). Basic Steps and Key Products of a PRA

Basic Steps To Perform a Level 2 PRA	Key Products of Each Step
1. Develop Basic Plant Familiarization	• Qualitative systems analysis. • System dependency matrices, component locations, and environmental susceptibilities.
2. Define Level 1 Event Sequences	• Set of initiating events. • Coarse grouping of initiating events with separate event sequence models for each group. • Event sequence diagrams with explicit coverage of emergency operating procedures. • Event trees with all support and frontline system responses and operator actions. • Definitions and success criteria for each event tree top event. • Plant damage states for all sequences ending in core damage.
3. Develop Models To Support Sequence Quantification	• System fault tree models for all systems event tree top events and system initiators. • Component unavailability models for common cause failures, hardware failures, and test and maintenance. • Human reliability models and performance-shaping factors for all dynamic actions. • Internal flood models for all flood-induced scenarios.
4. Develop PRA Database	• Uncertainty distributions incorporating generic and plant-specific data for initiating events, component failure rates, common cause failures, maintenance frequency, maintenance duration, and flood frequencies.

Table 2-3 (Page 2 of 2). Basic Steps and Key Products of a PRA	
Basic Steps To Perform a Level 2 PRA	**Key Products of Each Step**
5. Determine Severe Accident Progression and Release	• CET defining all severe accident phenomena that impact the containment integrity and severe accident consequences. • CET quantification models. • Containment failure analysis. • Severe accident progression and source term analysis.
6. Assemble and Quantify Accident Sequences	• Logic rules for linking all Level 1 and CETs. The Watts Bar PRA did not link the Level 1 and CET by computer at this time. The Level 1 PDSs were used as initiators for the CET. • Point estimates and uncertainty distributions for accident frequency. • Key risk contributors and importance measures.
7. Review and Interpret Results	• Appreciation of absolute risk levels and key factors controlling risk. • Effective strategies for risk and accident management.

Table 2-4. Types of Dependent Failures Encountered in Probabilistic Risk Assessment

Dependent Failure Type	Characteristics	Subtypes	Examples
1. Common Cause Initiating Event	Causes a plant transient and increases unavailability of one or more systems.	1A Functional Interaction	• Loss of service water.
		1B Human Interaction	• Maintenance error shorting out instrument bus.
		1C Physical Interaction	• Flooding
2. Intersystem Dependency	Causes a dependency in the joint failure probability of two or more systems.	2A Functional Dependency	• Coolant charging pump fails because component cooling fails.
		2B Human Interaction	• Operator error causes loss of two systems.
		2C Physical Interactions	• Flooding causes loss of equipment in two systems.
3. Intrasystem (intercomponent) Dependency	Causes a dependency in the joint failure probability of two or more components within a system.	3A Functional Dependency	• Battery loses charge after it is run beyond capacity.
		3B Human Interaction	• Design error present in redundant pump controls.
		3C Physical Interaction	• Flooding causes loss of redundant pumps.

Table 2-5 (Page 1 of 2). Treatment of Dependent Events in PRA

Basic Steps in PRA	How Dependent Events are Treated
1. Develop Basic Plant Familiarization	• Each system is reviewed for support and interfacing systems. • Detailed dependency matrices are developed and reviewed with operations personnel. • Plant walkdowns are performed to identify spatial dependencies.
2. Define Level 1 Event Sequences	• FMEAs are performed on plant systems to identify systemic common cause initiators. • Key internal flood locations and scenarios are selected for event tree analysis. • Functional dependencies are explicitly modeled in the event tree structure and in logic rules for linking the event trees and for assignment of systems fault tree results for event tree quantification. • Support systems are modeled explicitly on the event trees. • Human actions are defined and modeled in the event trees as dependent on the accident sequences in which they are applied.
3. Develop Models To Support Sequence Quantification	• Event tree split fractions are quantified as conditional on the sequence of events preceding the corresponding event tree node on the event tree. • Support systems are explicitly modeled on system fault trees as house events; dependencies due to test and maintenance alignments are modeled explicitly. • Common cause events are systematically placed on the system fault trees prior to Boolean reduction for all active redundant components in the plant. Common cause model reports are generated to summarize the common cause treatment for each system top event.

Table 2-5 (Page 2 of 2). Treatment of Dependent Events in PRA	
Basic Steps in PRA	**How Dependent Events are Treated**
	• Common cause events are quantified using the MGL parametric model and a plant- and system-specific screening of a common cause event database (Reference 2-24) as specified in NUREG/CR-4780 (Reference 2-10). • When two or more human errors appear on the same accident sequence, the error rate for the second and subsequent errors are, if appropriate, increased relative to the case of a single error.
4. Develop PRA Database	• Generic and plant-specific data on common cause events are included in the database.
5. Determine Severe Accident Progression and Release	• Impacts of severe accident phenomena on equipment in the containment are explicitly treated.
6. Assemble and Quantify Accident Sequences	• Sequence results reports explicitly display functional dependencies that contribute to the sequence. • Risk importance measures are assigned to the functional dependencies as well as to the associated equipment. • Risk importance measures are assigned to common cause events; the contributions of common cause events to system level results are explicitly displayed.

Table 2-6. Component Types and Failure Modes Normally Considered for Common Cause Groups

Component	Failure Mode
Pump	Fails To Start
Pump	Fails during Operation
Diesel Generator	Fails To Start
Diesel Generator	Fails during Operation
Ventilation Fan	Fails To Start
Ventilation Fan	Fails during Operation
Motor-Operated Valve	Fails To Open on Demand
Motor-Operated Valve	Fails To Close on Demand
Air-Operated Valve	Fails To Open on Demand
Air-Operated Valve	Fails To Close on Demand
Check Valve	Fails To Close on Demand
Solenoid Valve	Fails To Open on Demand
Solenoid Valve	Fails To Close on Demand
Air Compressor	Fails To Start
Air Compressor	Fails during Operation
Air Conditioning Unit	Fails To Start
Air Conditioning Unit	Fails during Operation
Circuit Breaker	Fails To Open on Demand
Circuit Breaker	Fails To Close on Demand

Table 2-7 (Page 1 of 2). Typical Database Component Boundaries

Component	Includes	Does Not Include
1. Diesel Generators	• Diesel engine. • Electrical generator. • Air start system, if applicable. • Starter motor, if applicable. • Diesel radiator/cooler. • Air inlet filter. • Shaft-driven fuel oil or fuel oil booster pumps. • Diesel exhaust system.	• Diesel generator load sequencers. • Diesel fuel oil transfer system. • Cooling water valves. • Diesel generator output breaker or output bus. • Protection system actuation relays. • Diesel room cooling.
2 Motor-Operated Valves	• Valve body and internal parts. • Valve operator (motor and shaft). • Valve breaker. • Valve control circuitry. • Valve limit switches and torque switches.	• MCCs or other electrical equipment not listed above. • Interlock logic.
3. Check Valves	• Valve body and seat. • Valve disk, pivot shaft, and connecting key/bolts. • Valve counterweights, if applicable. • Valve test mechanism, if applicable.	
4. BWR Safety/Relief Valves	• Valve body and internal parts. • Pilot valve assembly, if applicable.	
5. Motor-Driven Pumps	• Pump casing, impeller, shaft, bearings, and seals. • Pump motor (driver). • Pump-to-motor coupling. • Pump motor circuit breaker. • Pump motor cooler, if applicable. • Pump motor thermal overloads, if applicable. • Pump motor control circuitry. • Relief valves on positive displacement pumps.	• Pump discharge valve or suction valve. • Actuation system relays. • Minimum flow recirculation valves. • Room cooling systems. • Interlock logic.

Table 2-7 (Page 2 of 2). Typical Database Component Boundaries

Component	Includes	Does Not Include
6. Turbine-Driven Pumps	• Pump casing, impeller shaft, bearings, and seals. • Pump driver. • Pump-to-driver coupling. • Oil pumps, if applicable. • Oil pump control circuitry. • Turbine trip/control valve.	• Pump discharge valve or suction valve. • Actuation system relays. • Minimum flow recirculation valves. • Pump seal water valves. • Room cooling systems.
7. HVAC Chillers	• Chiller compressor. • Condenser/evaporator. • Chiller compressor circuit breaker. • Compressor thermal overloads, if applicable. • Compressor control circuitry.	• Chilled water discharge valve or inlet valve. • Actuation system relays. • Condenser cooling water valves. • Room cooling. • Chilled water pump.
8. HVAC Fans	• Fan housing, impeller, shaft, and bearings. • Fan motor (driver). • Fan-to-motor coupling/belt. • Fan motor circuit breaker. • Fan motor thermal overloads, if applicable. • Fan motor control circuitry.	• Fan discharge or inlet dampers. • Actuation system relays. • Fan actuation temperature switches/ transmitters. • Interlock logic circuitry.
9. Circuit Breakers	• Circuit breaker mechanical parts. • Circuit breaker contacts. • Circuit breaker trip coil/mechanism. • Circuit breaker closing coil/mechanism. • Circuit breaker overcurrent protection device. • Circuit breaker control circuitry.	• Actuation system relays.
10. Batteries, Battery Chargers Inverters	• Entire component.	• Bus. • Output breaker.

Table 2-8. Watts Bar-Specific Information Sources
Updated Final Safety Analysis Report
Appendix R Calculations
Design Basis Documents
Flow and Control Diagrams
Logic Diagrams
Electric Single Line Diagrams
Wiring Schematic Diagrams
Plant Operating Procedures
Plant Surveillance Procedures
Emergency Operating Procedures
Abnormal Operating Procedures
PRA-Dedicated Senior Reactor Operator
Operating Crew Surveys
Vendor Manuals
Plant Walk-Throughs

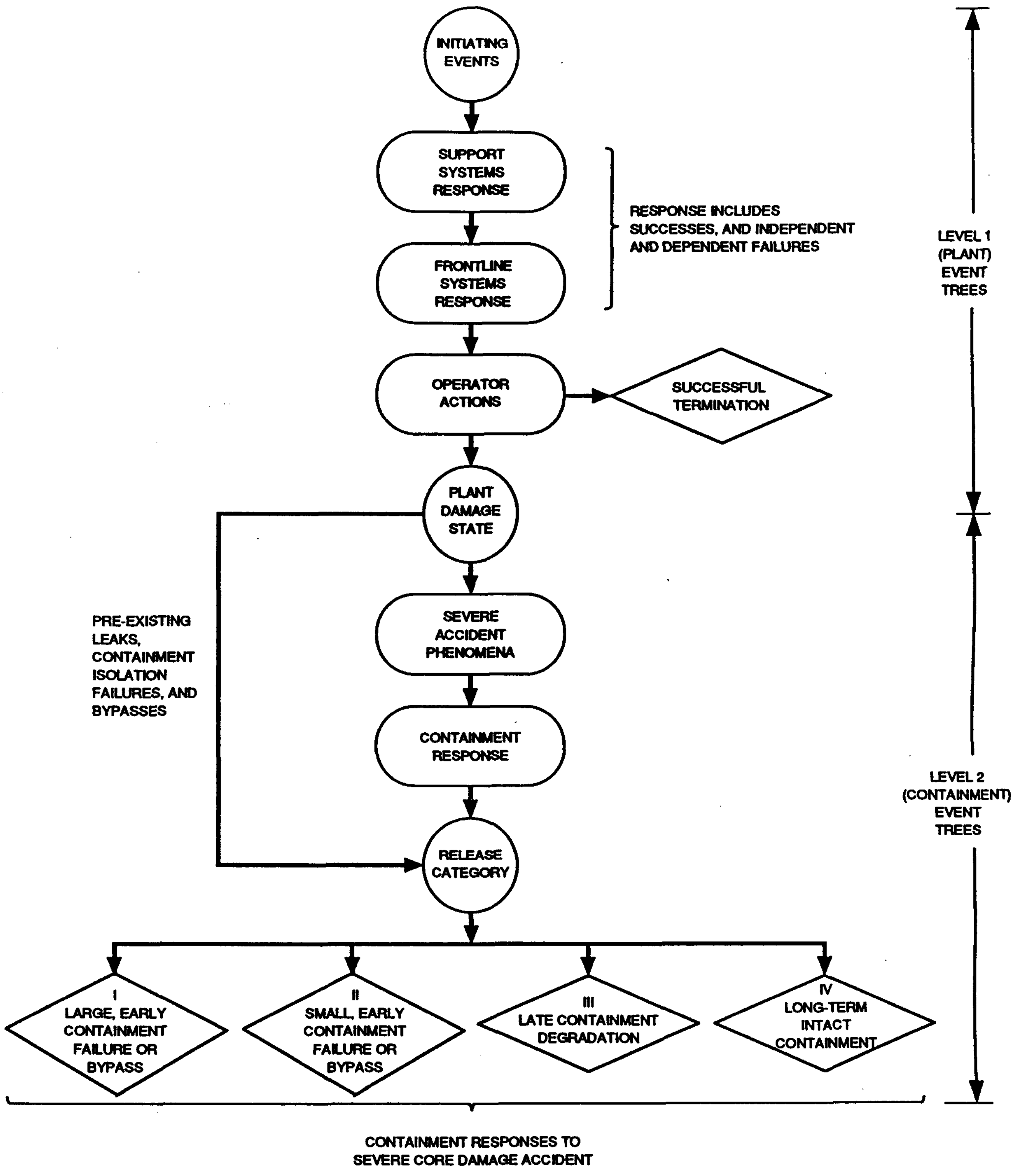


Figure 2-1. Systematic Framework for Defining Accident Sequences

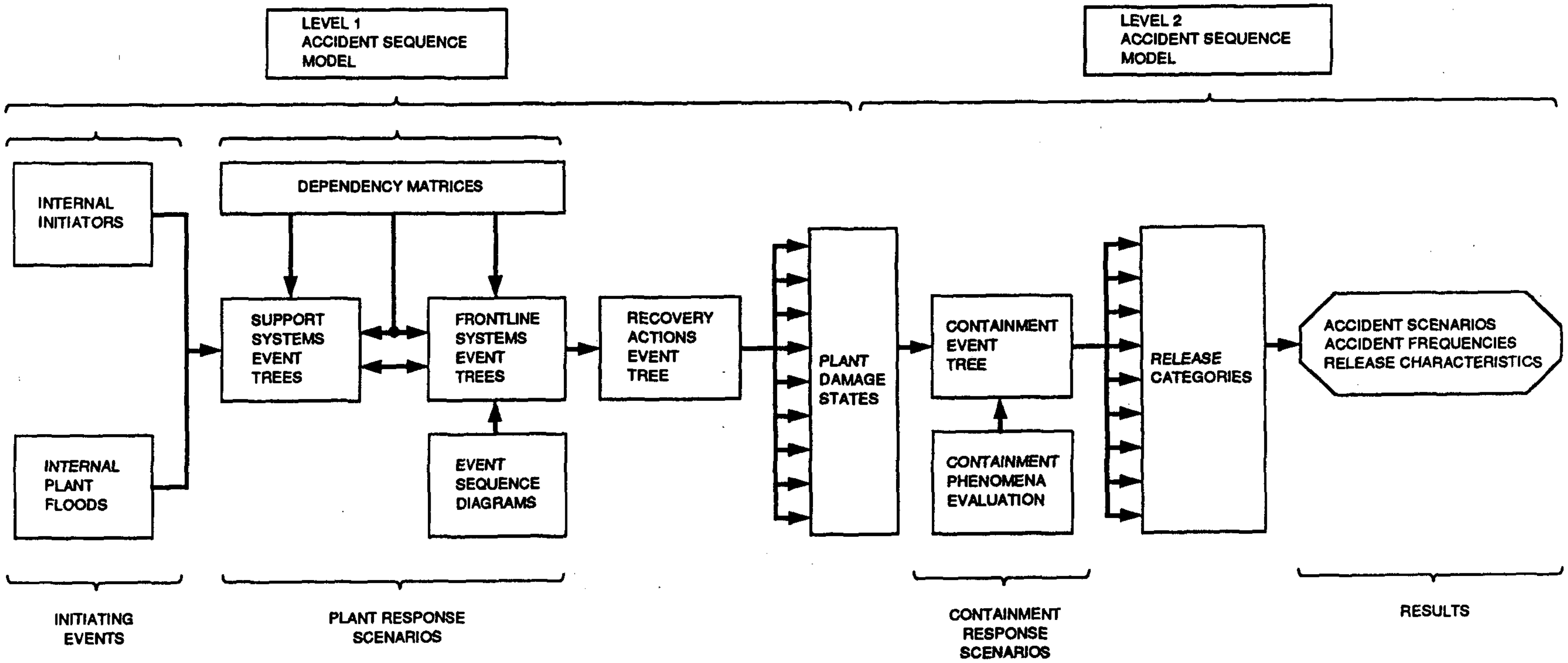


Figure 2-2. Definition of Accident Sequences in a Level 2 PRA

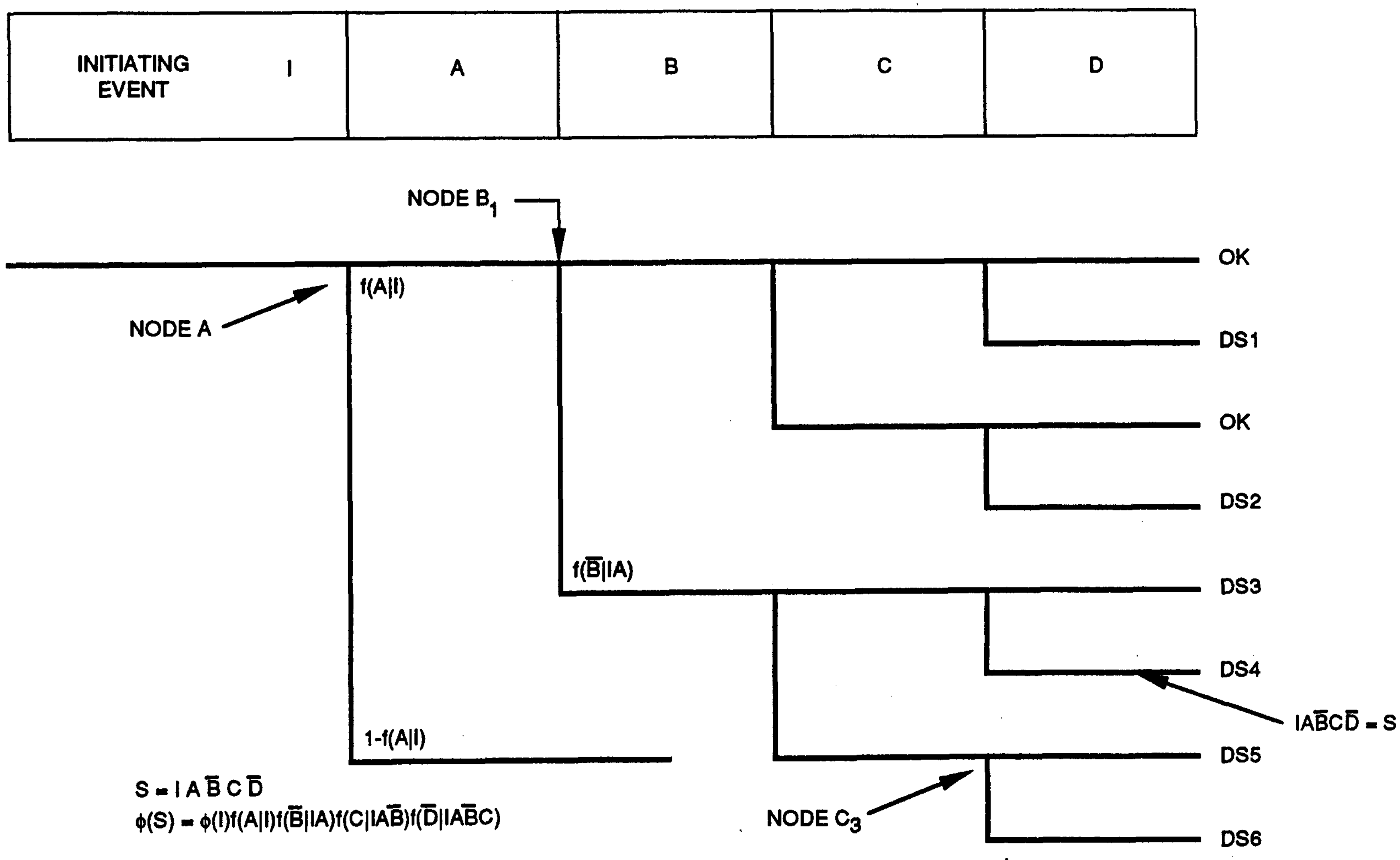
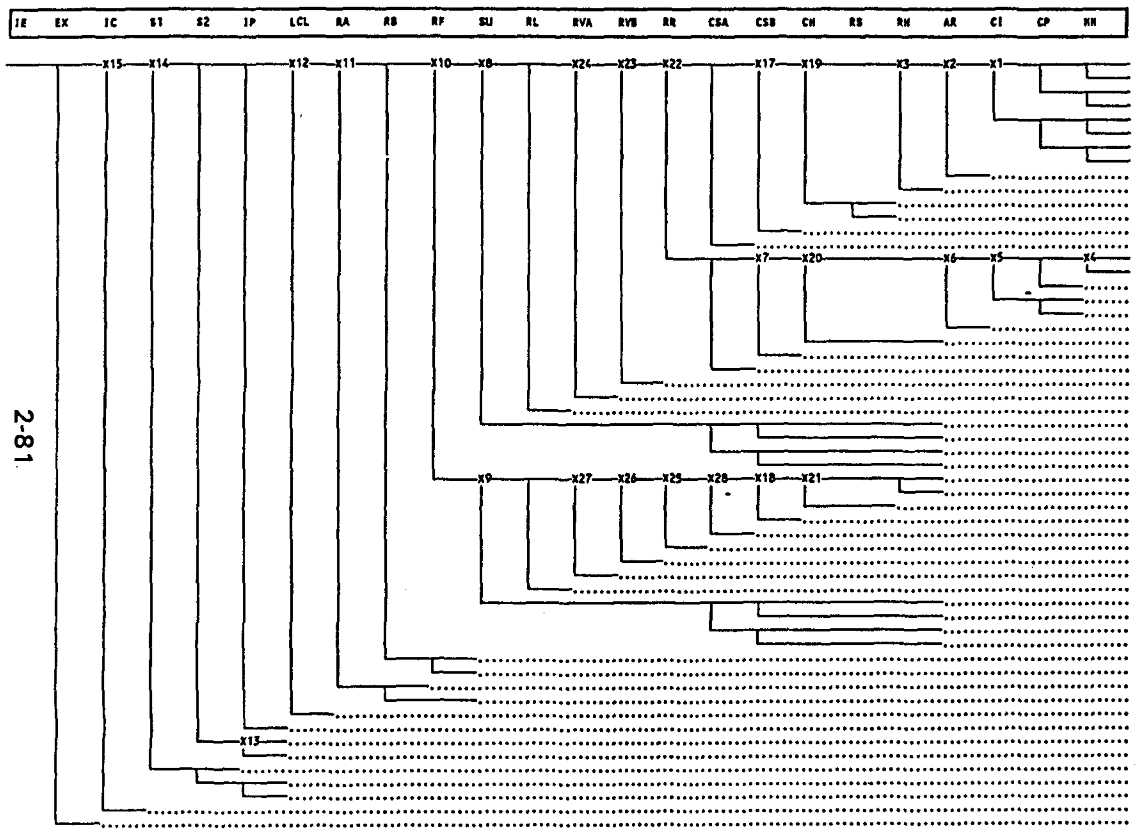


Figure 2-3. Sample Simplified Event Tree

Page No. 1 Event Tree: LARLOC4 14:18:46 01 AUG 1992



1		1
2		2
3		3
4		4
5		5
6		6
7		7
8		8
9	X1	9-16
10	X2	17-32
11	X3	33-64
12	X3	65-96
13	X19	97-192
14	X17	193-384
15		385
16		386
17	X4	387-388
18	X4	389-390
19	X4	391-392
20	X5	393-400
21	X6	401-416
22	X20	417-448
23	X7	449-512
24	X22	513-1024
25	X23	1025-2048
26	X24	2049-4096
27	X6	4097-4112
28	X6	4113-4128
29	X6	4129-4144
30	X6	4145-4160
31	X2	4161-4176
32	X2	4177-4192
33	X3	4193-4224
34	X21	4225-4288
35	X18	4289-4416
36	X28	4417-4672
37	X25	4673-5184
38	X26	5185-6208
39	X27	6209-8256
40	X6	8257-8272
41	X6	8273-8288
42	X6	8289-8304
43	X6	8305-8320
44	X8	8321-12480
45	X9	12481-16640
46	X10	16641-24960
47	X9	24961-29120
48	X11	29121-58240
49	X12	58241-116480
50	X12	116481-174720
51	X12	174721-232960
52	X13	232961-349440
53	X12	349441-407680
54	X12	407681-465920
55	X14	465921-931840
56	X15	931841-1863680

Top Event Designator.....	Top Event Description..............................
IE	INITIATING EVENT
EX	EXCESSIVE LOCA
IC	FAILURE OF ICE CONDENSER
S1	LOSS OF SAFETY INJECTION PUMP 1A-A
S2	LOSS OF SAFETY INJECTION PUMP 1B-B
IP	LOSS OF 2/3 SIS COLD LEG INJECTION PATHS
LCL	LOSS OF 1/3 COLD LEG ACCUMULATORS
RA	LOSS OF RHR PUMP 1A-A
RB	LOSS OF RHR PUMP 1B-B
RF	FAILURE OF 2/3 RHR COLD LEG INJECTION PATHS
SU	LOSS OF CONTAINMENT SUMP
RL	FAILURE OF SWAP SWAPOVER INSTRUMENTS
RVA	FAILURE OF SUMP SWAPOVER VALVE, 1-FCV-63-72
RVB	FAILURE OF SUMP SWAPOVER VALVE 1-FCV-63-73
RR	FAILURE OF AUTOMATIC/MANUAL SWAPOVER FROM THE RWST TO THE CONTAINMENT SUMP
CSA	FAILURE OF TRAIN A CONTAINMENT SPRAY
CSB	FAILURE OF TRAIN B CONTAINMENT SPRAY
CH	FAILURE OF CONTAINMENT SPRAY HEAT EXCHANGERS
RS	FAILURE OF RHR SPRAY
RH	FAILURE OF RHR & SIS HOT LEG RECIRCULATION
AR	FAILURE OF AIR RETURN FANS
CI	FAILURE OF CONTAINMENT ISOLATION
CP	CONTAINMENT PURGE FAILS TO CLOSE
HN	FAILURE OF HYDROGEN IGNITORS

Figure 2-4. Example Large LOCA Event Tree for Watts Bar

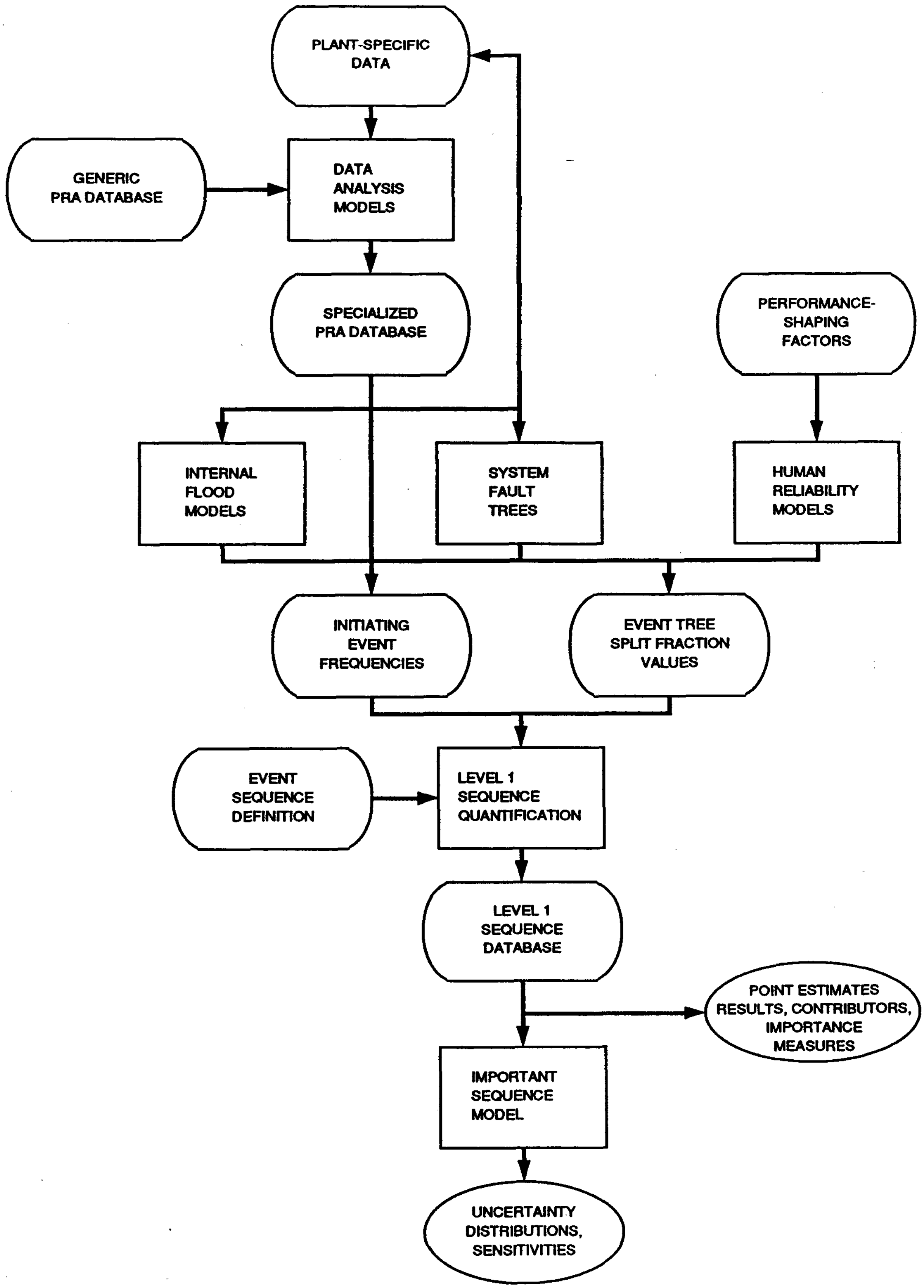
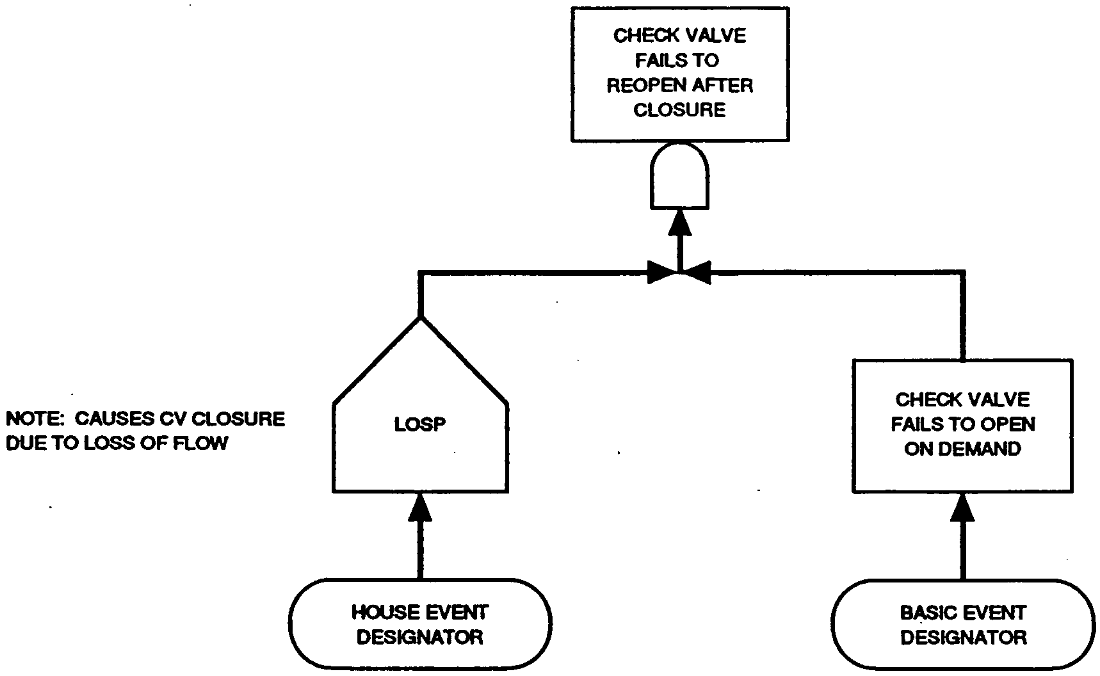
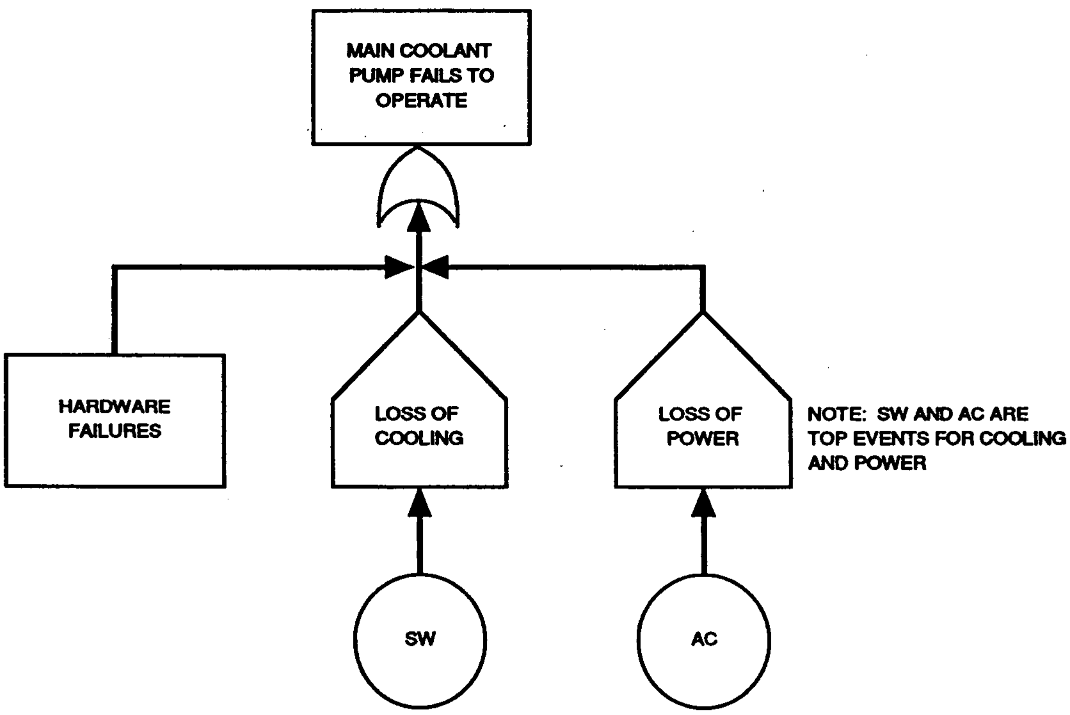


Figure 2-5. Flow Chart for Level 1 Event Sequence Quantification



FAILURE MODES CONDITIONED ON THE SCENARIO



SUPPORT SYSTEM REQUIRED FOR OPERATION

Figure 2-6. Example of House Events Representing Failure Modes Conditioned on the Scenario and Support System Required for Operation

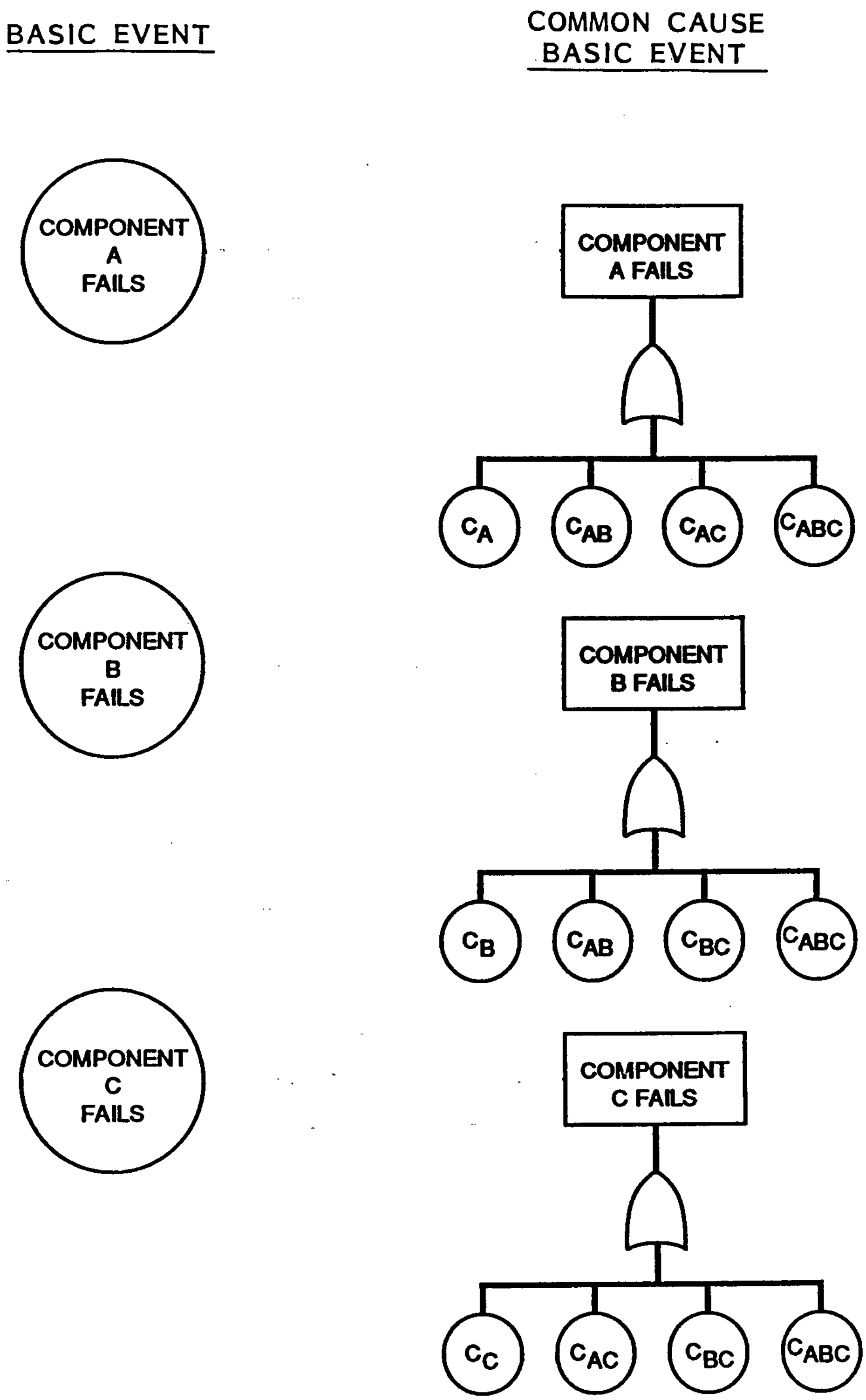


Figure 2-7. RISKMAN Development of Common Cause Event Subtrees for Common Cause Group {A, B, C}

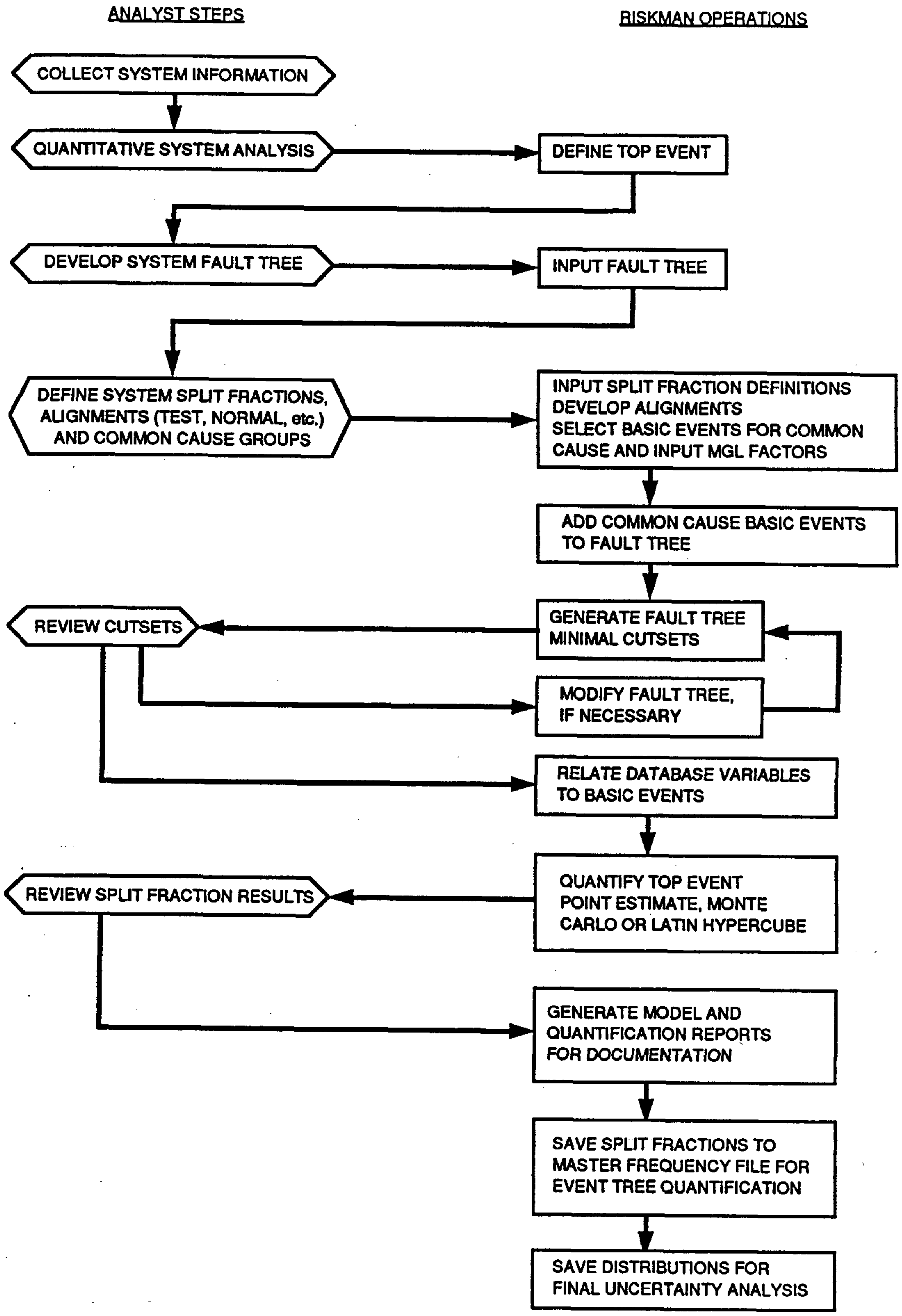


Figure 2-8. Flow Chart of Systems Analysis Steps Using RISKMAN

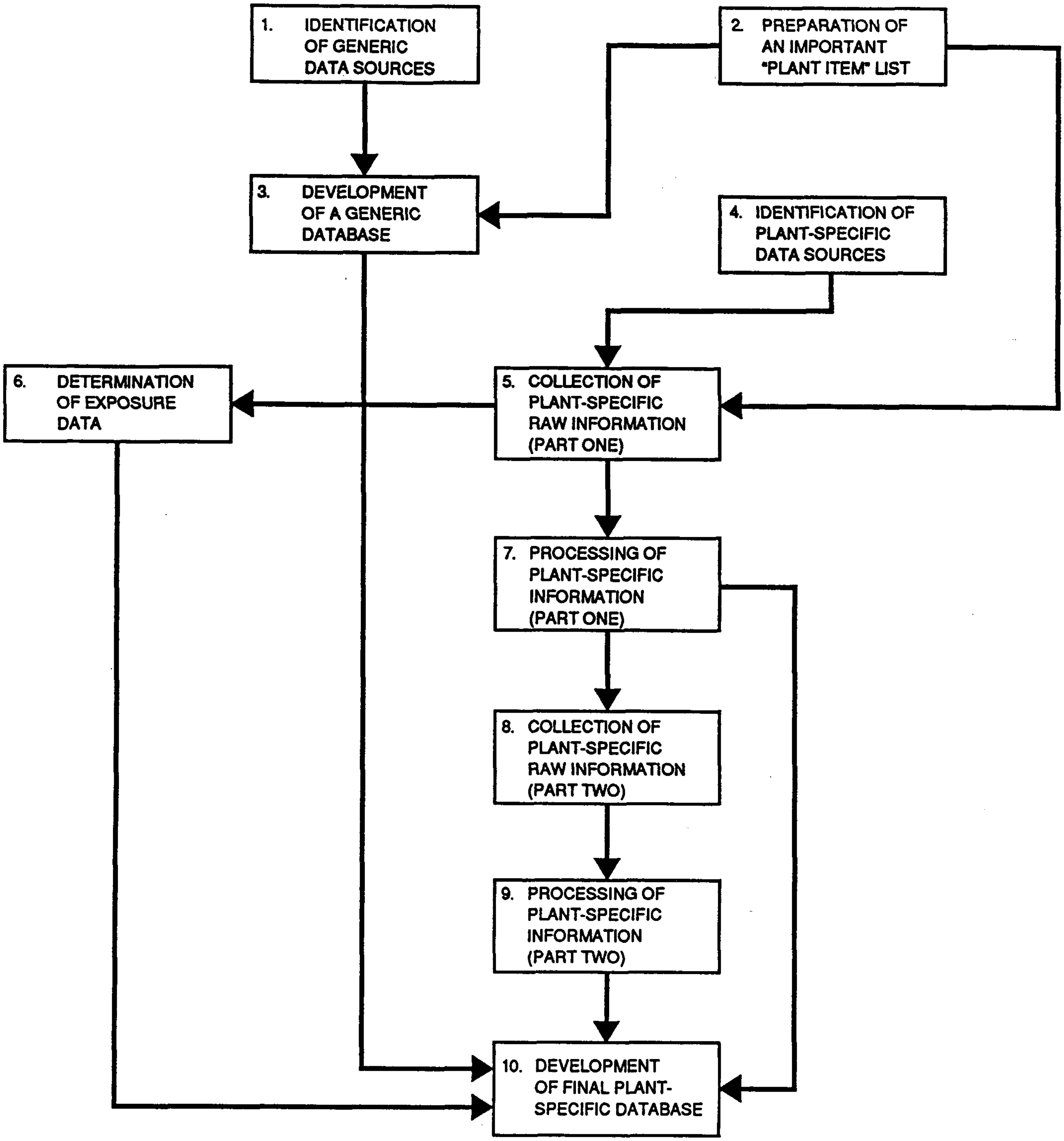


Figure 2-9. PRA Database Development Process Flow Chart

EVIDENCE COLLECTED FOR FAILURE RATE				
SOURCE	DATA			
	NUMBER OF FAILURES	NUMBER OF DEMANDS	ESTIMATE	ASSIGNED RANGE FACTOR
TYPE 1				
PLANT A	10	1.65+3		
PLANT B	14	1.13+4		
PLANT C	7	1.73+3		
PLANT D	42	6.72+3		
PLANT E	3	1.26+3		
PLANT F	31	9.72+3		
TYPE 2				
EXPERT 1			1.00-3	5
EXPERT 2			5.60-3	3
EXPERT 3			1.00-3	10

NOTE: EXPONENTIAL NOTATION IS INDICATED IN ABBREVIATED FORM: e.g., $1.65{+}3 = 1.65 \times 10^{3}$.

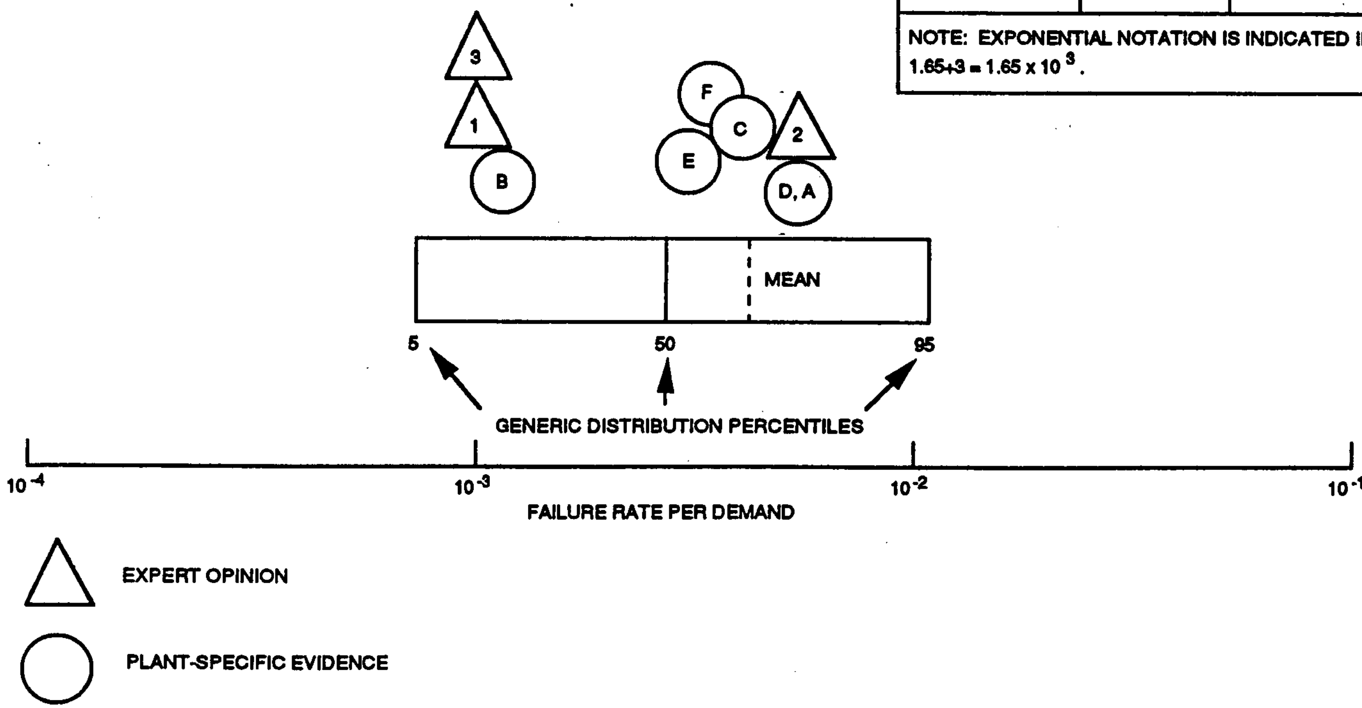


Figure 2-10. Application of RISKMAN To Develop Generic Distributions for MOV Failure Rates

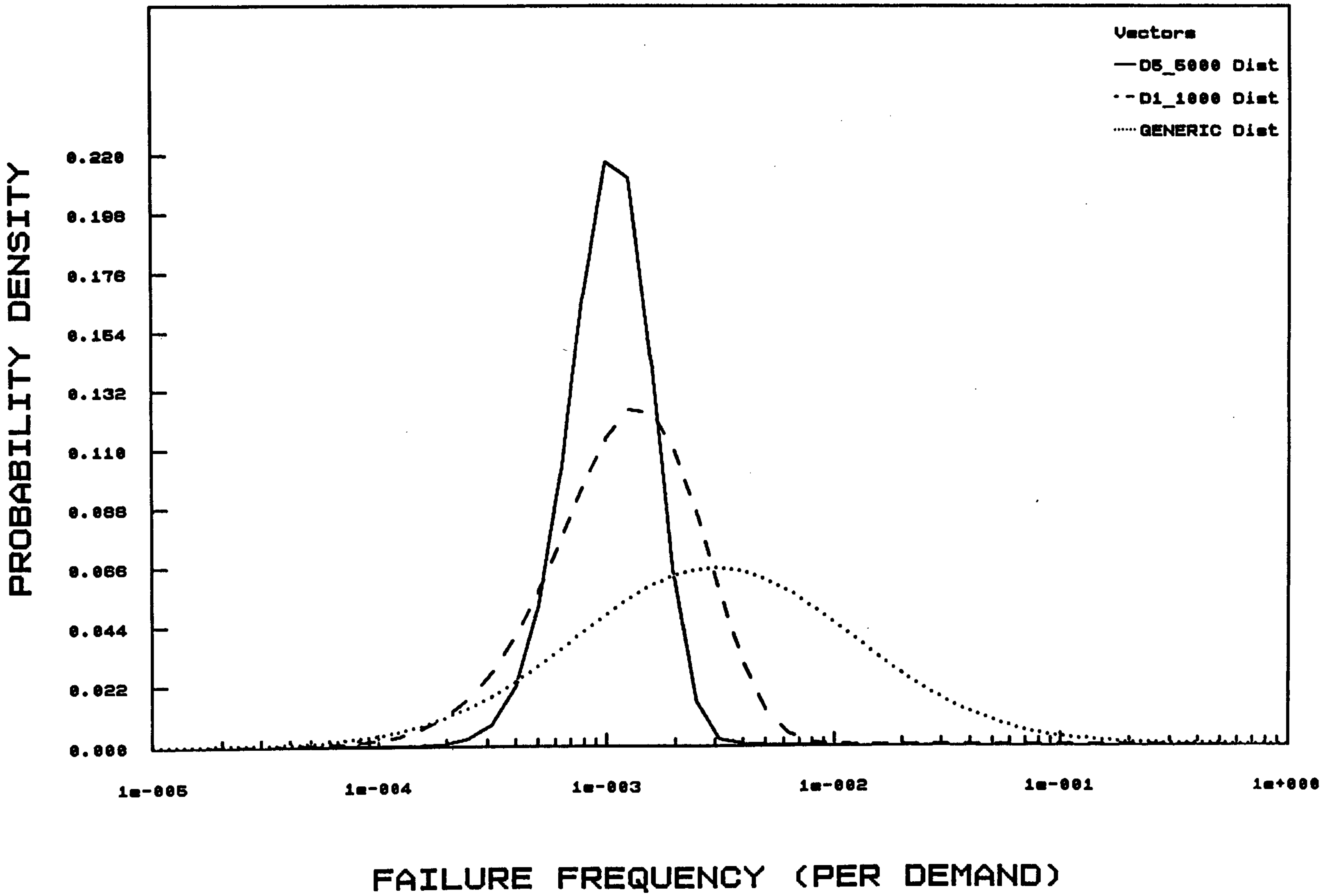


Figure 2-11. Plant-Specific Updates of MOV Failure Rate

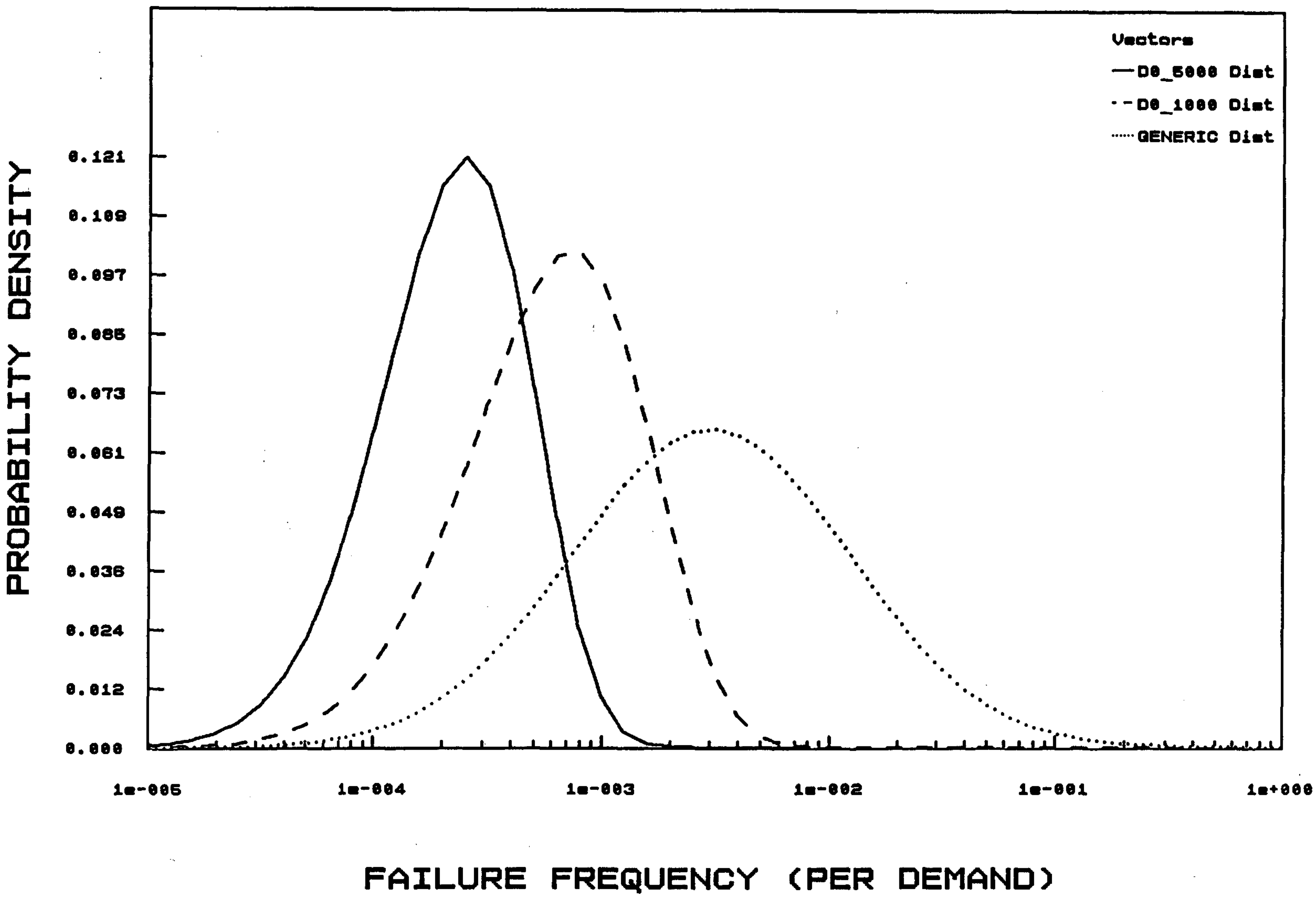


Figure 2-12. Plant-Specific Updates of Valves with Zero Failures

MODEL Name: BV2

Top-Ranking Sequences Contributing to Group : CMELT Frequency
CMELT = CORE MELT SEQUENCES

18:48:30 14 JAN 1992

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
1	LOSS OF OFFSITE GRID & AC ORANGE TRAIN - EMERGENCY AC PURPLE TRAIN	- OFFSITE GRID - EMERGENCY AC ORANGE TRAIN - HHSI SUCTION PATH FROM RWST - HIGH HEAD SAFETY INJECTION PUMPS - RCP SEAL INJECTION/THERMAL BARRIER COOLING - DEPRESSURIZATION OF RCS FOR RHR ENTRY - QUENCH SPRAY PUMPS - LOW HEAD SAFETY INJECTION PUMPS - LHSI COLD LEG INJECTION PATHS - CONTAINMENT SUMP WATER LEVEL, PLUGGING - RECOVERY PRIOR TO CORE UNCOVERY	HINOHR	7.86E-05	32.33
2	LOSS OF EMERGENCY SWITCHGEAR VENTILATION	- EMERGENCY AC ORANGE TRAIN - HHSI SUCTION PATH FROM RWST - AUXILIARY FEEDWATER - MANUAL ACTIONS TO REESTABLISH MFW - BLEED & FEED COOLING - HIGH HEAD SAFETY INJECTION PUMPS - QUENCH SPRAY PUMPS - LOW HEAD SAFETY INJECTION PUMPS - LHSI COLD LEG INJECTION PATHS - CONTAINMENT SUMP WATER LEVEL, PLUGGING - CONTAINMENT ISOLATION - RECOVERY PRIOR TO CORE UNCOVERY	SYNISO	1.79E-05	7.34
3	LOSS OF OFFSITE GRID & AC ORANGE TRAIN - EMERGENCY AC PURPLE TRAIN - ERF BLACK DIESEL GENERATOR	- OFFSITE GRID - EMERGENCY AC ORANGE TRAIN - HHSI SUCTION PATH FROM RWST - HIGH HEAD SAFETY INJECTION PUMPS - RCP SEAL INJECTION/THERMAL BARRIER COOLING - DEPRESSURIZATION OF RCS FOR RHR ENTRY - QUENCH SPRAY PUMPS - LOW HEAD SAFETY INJECTION PUMPS - LHSI COLD LEG INJECTION PATHS - CONTAINMENT SUMP WATER LEVEL, PLUGGING - RECOVERY PRIOR TO CORE UNCOVERY	HINOHR	1.25E-05	5.13
4	SMALL LOCA, ISOLABLE - SERVICE/STANDBY SW HEADER 'A' FLOW PATH - SERVICE WATER/STANDBY SW HEADER 'B' FLOW PATH	- HIGH HEAD SAFETY INJECTION PUMPS - RCP SEAL INJECTION/THERMAL BARRIER COOLING - RESIDUAL HEAT REMOVAL - RSS TRAIN C FOR COLD LEG RECIRCULATION - RSS TRAIN D FOR COLD LEG RECIRCULATION - RECIRCULATION SPRAY FROM PUMPS A OR B - RECOVERY PRIOR TO CORE UNCOVERY	HINOHR	8.00E-06	3.29
5	LOSS OF OFFSITE GRID & AC ORANGE TRAIN - EMERGENCY AC PURPLE TRAIN - PRESSURIZER RELIEF AND RECLOSURE	- OFFSITE GRID - EMERGENCY AC ORANGE TRAIN - HHSI SUCTION PATH FROM RWST - HIGH HEAD SAFETY INJECTION PUMPS - QUENCH SPRAY PUMPS - LOW HEAD SAFETY INJECTION PUMPS - LHSI COLD LEG INJECTION PATHS - CONTAINMENT SUMP WATER LEVEL, PLUGGING - RECOVERY PRIOR TO CORE UNCOVERY	HINOHR	6.50E-06	2.67

Figure 2-13. Example RISKMAN Report of Ranking Sequences to Core Melt

Top Event Importance for Model: 8V2
Sorted by Total Importance
Total Sequence Frequency = 1.2898E-04

16:10:13 02 DEC 1991
Page 1

	Top	[1] Guar. Event	[2] Probabilistic	[3] Total	[4] Frequency
1.	RE	7.5609E-01	1.7678E-01	9.3288E-01	1.2032E-04
2.	NR	9.3238E-01	0.0000E+00	9.3238E-01	1.2026E-04
3.	IC	9.2414E-01	0.0000E+00	9.2414E-01	1.1920E-04
4.	NM	9.1795E-01	0.0000E+00	9.1795E-01	1.1840E-04
5.	CC	7.8413E-01	2.7747E-04	7.8441E-01	1.0117E-04
6.	TB	7.0725E-01	3.0809E-02	7.3806E-01	9.5195E-05
7.	HH	6.5646E-01	5.6621E-02	7.1308E-01	9.1974E-05
8.	WA	5.1118E-01	1.1126E-01	6.2244E-01	8.0283E-05
9.	VL	4.5632E-01	1.1384E-01	5.7017E-01	7.3541E-05
10.	SM	5.6698E-01	1.6153E-03	5.6859E-01	7.3338E-05
11.	WB	4.4747E-01	1.1151E-01	5.5897E-01	7.2097E-05
12.	QS	5.5645E-01	9.8311E-04	5.5743E-01	7.1898E-05
13.	LH	5.3959E-01	2.8526E-03	5.4245E-01	6.9965E-05
14.	LC	5.1477E-01	8.3059E-06	5.1478E-01	6.6397E-05
15.	FA	4.9504E-01	1.6615E-02	5.1166E-01	6.5994E-05
16.	EA	4.9002E-01	2.1163E-02	5.1118E-01	6.5933E-05
17.	AO	2.2263E-01	2.2568E-01	4.4831E-01	5.7823E-05
18.	FB	4.3027E-01	1.7742E-02	4.4801E-01	5.7785E-05
19.	EB	4.2524E-01	2.2225E-02	4.4747E-01	5.7715E-05
20.	BP	2.0891E-01	1.7624E-01	3.8515E-01	4.9677E-05
21.	AF	3.1426E-01	6.4655E-02	3.7891E-01	4.8872E-05
22.	OB	3.7121E-01	1.5957E-03	3.7280E-01	4.8085E-05
23.	RR	3.2071E-01	0.0000E+00	3.2071E-01	4.1365E-05
24.	SE	3.1344E-01	1.3684E-03	3.1481E-01	4.0605E-05
25.	CI	3.1175E-01	1.9694E-03	3.1372E-01	4.0464E-05
26.	OS	2.7665E-01	3.5958E-02	3.1261E-01	4.0321E-05

Notes:

[1] Guaranteed Event Importance is fraction of total (core melt) frequency with this top event quantified with a guaranteed failure split fraction (1.0).

[2] Probabilistic Importance is fraction of total (core melt) frequency with this top event quantified with a split fraction less than 1.0.

[3] Total Importance is sum of [1] and [2].

[4] Frequency is the total frequency of all (core melt) sequences with this top event failed.

Figure 2-14. Top Event Importance Report Generated from RISKMAN

MODEL Name: BV2
Split Fraction Importance for Group : CMELT
Sorted by Importance
Group Frequency = 1.2898E-04

16:24:48 02 DEC 1991
Page 1

......	SF Name...	[1] Importance.....	[2] Achievement..	[3] Reduction...	[4] Derivative..	[5] SF Value.......	[6] Frequency......
1.	NRF	9.3238E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.2026E-04
2.	ICF	9.2414E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.1920E-04
3.	NMF	9.1795E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.1840E-04
4.	CCF	7.8413E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.0114E-04
5.	REF	7.5609E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	9.7522E-05
6.	TBF	7.0725E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	9.1222E-05
7.	HHF	6.5646E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	8.4671E-05
8.	SMF	5.6698E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	7.3129E-05
9.	QSF	5.5645E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	7.1772E-05
10.	LHF	5.3959E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.9597E-05
11.	LCF	5.1477E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.6396E-05
12.	WAF	5.1118E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.5933E-05
13.	FAF	4.9504E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.3851E-05
14.	EAF	4.9002E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.3203E-05
15.	VLF	4.5632E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.8857E-05
16.	WBF	4.4747E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.7715E-05
17.	FBF	4.3027E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.5496E-05
18.	EBF	4.2524E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.4848E-05
19.	OBF	3.7121E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.7879E-05
20.	RRF	3.2071E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.1365E-05
21.	AFF	3.1426E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.0533E-05
22.	CIF	3.1175E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.0210E-05
23.	OFF	3.1025E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.0016E-05
24.	OSF	2.7476E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	3.5438E-05
25.	SEF	2.6833E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	3.4609E-05
26.	AOF	2.2263E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.8715E-05
27.	SAF	2.2142E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.8560E-05
28.	SBF	2.2127E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.8540E-05
29.	RCF	2.1013E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.7103E-05
30.	BPF	2.0891E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.6945E-05
31.	RDF	1.9943E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.5722E-05
32.	IRF	1.9255E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.4836E-05
33.	IWF	1.9240E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.4815E-05
34.	DPF	1.8003E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.3221E-05
35.	DOF	1.7212E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.2201E-05
36.	RSF	1.6787E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1652E-05
37.	IBF	1.6369E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1113E-05
38.	IYF	1.6346E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1083E-05
39.	BVF	1.6311E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1038E-05
40.	HCF	1.5623E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.0151E-05
41.	OGF	1.5394E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9855E-05
42.	PRF	1.1318E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.4597E-05
43.	WB4	1.1044E-01	2.5478E+00	8.8956E-01	2.1388E-04	6.6600E-02	1.4244E-05
44.	WC2	1.1020E-01	1.9328E+02	8.8980E-01	2.4815E-02	5.7280E-04	1.4214E-05
45.	MFF	1.0858E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.4004E-05
46.	BX2	1.0634E-01	8.9012E+00	8.9366E-01	1.0328E-03	1.3280E-02	1.3716E-05
47.	BP5	1.0634E-01	1.8586E+00	8.9366E-01	1.2446E-04	1.1020E-01	1.3716E-05
48.	VL1	9.9115E-02	1.3770E+02	9.0113E-01	1.7645E-02	7.2270E-04	1.2784E-05
49.	IAF	9.3982E-02	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.2122E-05
50.	OOF	8.6471E-02	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.1153E-05

Notes:

[1] Importance is fraction of total core melt frequency with this split fraction failed.

[2] Achievement is ratio:

$$\frac{\text{Core melt frequency SF} = 1.0}{\text{Core melt frequency}}$$

[3] Reduction is ratio:

$$\frac{\text{Core melt frequency SF} = 0.0}{\text{Core melt frequency}}$$

[4] Derivative is:

$$\frac{\partial CMF}{\partial SF}$$

[5] SF = mean split fraction value in master frequency file.

[6] Frequency is total frequency of core melt sequences with this SF failed.

Figure 2-15. Split Fraction Importance Report Generated from RISKMAN

MODEL Name: BV2
Basic Event Importance Report for Group CMELT

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	[1]	[2]	[3]	[4]
Rank.	Basic Event.......	SF Name...	Basic Event Value	Basic Event Importance
1	OPRBV1	BV1	3.0342E-02	6.2868E-03
		BV2	3.0342E-02	8.1250E-02
		BV3	3.0342E-02	4.4601E-04
		BV4	3.0342E-02	2.2122E-02
		BV5	3.0342E-02	1.5571E-03
	Basic Event Total			1.1166E-01
2	XXFRACTION1	PR1	1.0000E+00	2.6669E-03
		PR2	1.0000E+00	0.0000E+00
		PR3	1.0000E+00	0.0000E+00
		PR4	1.0000E+00	1.4507E-03
		PR5	1.0000E+00	0.0000E+00
		PR6	1.0000E+00	2.2607E-03
		PR7	1.0000E+00	1.6920E-02
		PR8	1.0000E+00	5.6863E-03
		PR9	1.0000E+00	5.8191E-02
	Basic Event Total			8.7175E-02
3	DGSR2EGSEG22	BX1	7.3849E-02	9.2934E-03
		BX2	7.3849E-02	6.3422E-02
		BX3	7.3849E-02	0.0000E+00
	Basic Event Total			7.2715E-02
4	DGSR2EGSEG21	BX1	7.3849E-02	9.2934E-03
		BX2	7.3849E-02	6.3403E-02
		BX3	7.3849E-02	0.0000E+00
	Basic Event Total			7.2697E-02
5	OPRHH2	HH1	5.8872E-04	4.9263E-02
		HH2	5.8872E-04	1.5363E-03
		HH3	5.8872E-04	4.9740E-04
		HH4	5.8872E-04	2.3201E-03
		HH5	5.8872E-04	4.9241E-04
		HH7	5.8872E-04	1.3621E-05
	Basic Event Total			5.4122E-02
6	DGSR2EGSEG21	A01	7.4730E-02	1.0444E-02
		A02	7.4730E-02	4.1663E-02
	Basic Event Total			5.2107E-02

Notes:

[1] Basic event name

[2] Split fractions that include basic event in model

[3] Frequency of basic event

[4] Fraction of core melt frequency with this basic event failed (includes breakdown by split fraction)

Figure 2-16. Basic Event Importance Report Generated from RISKMAN

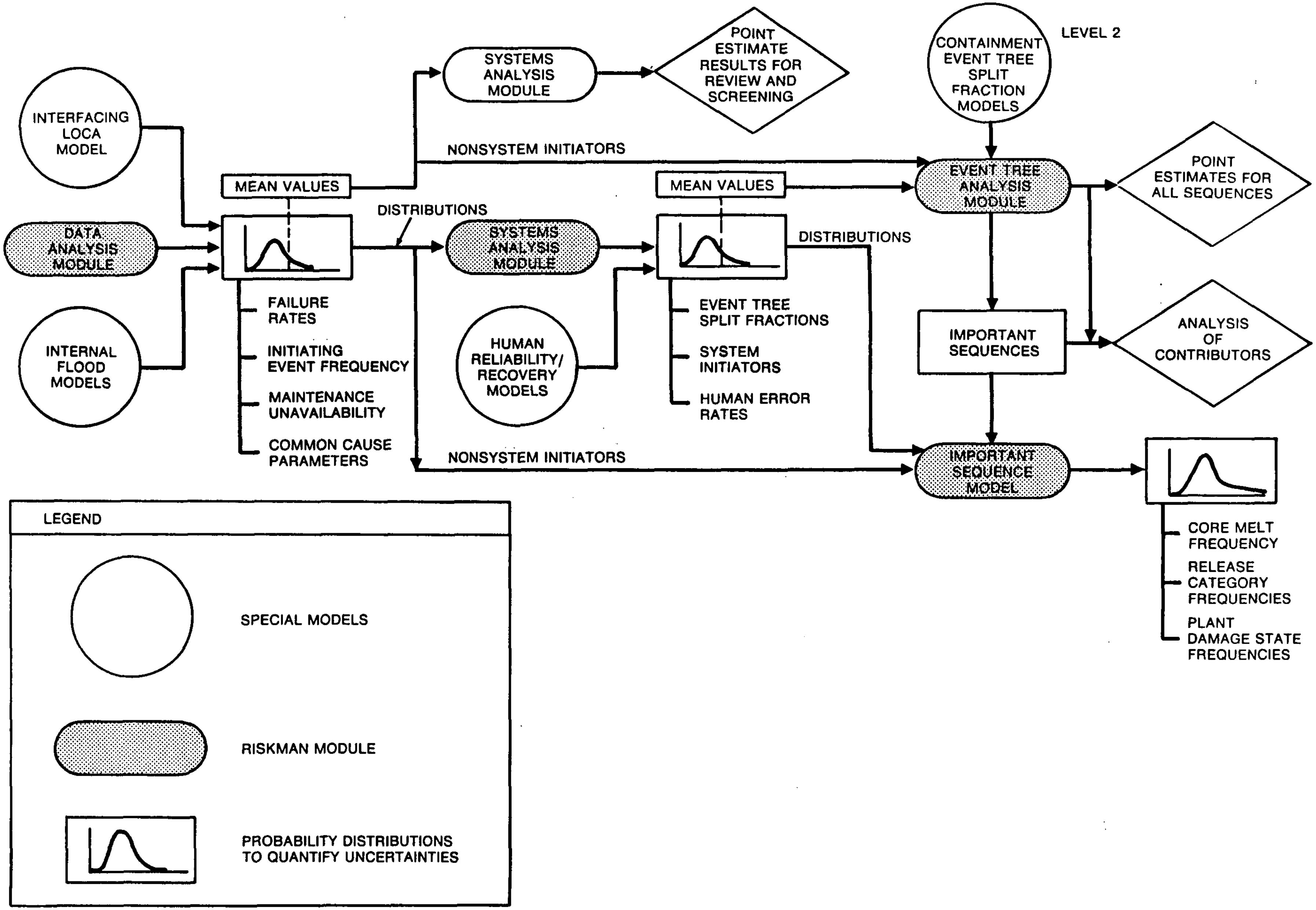

Figure 2-17. PRA Quantification Flow Chart for Point Estimates and Uncertainties

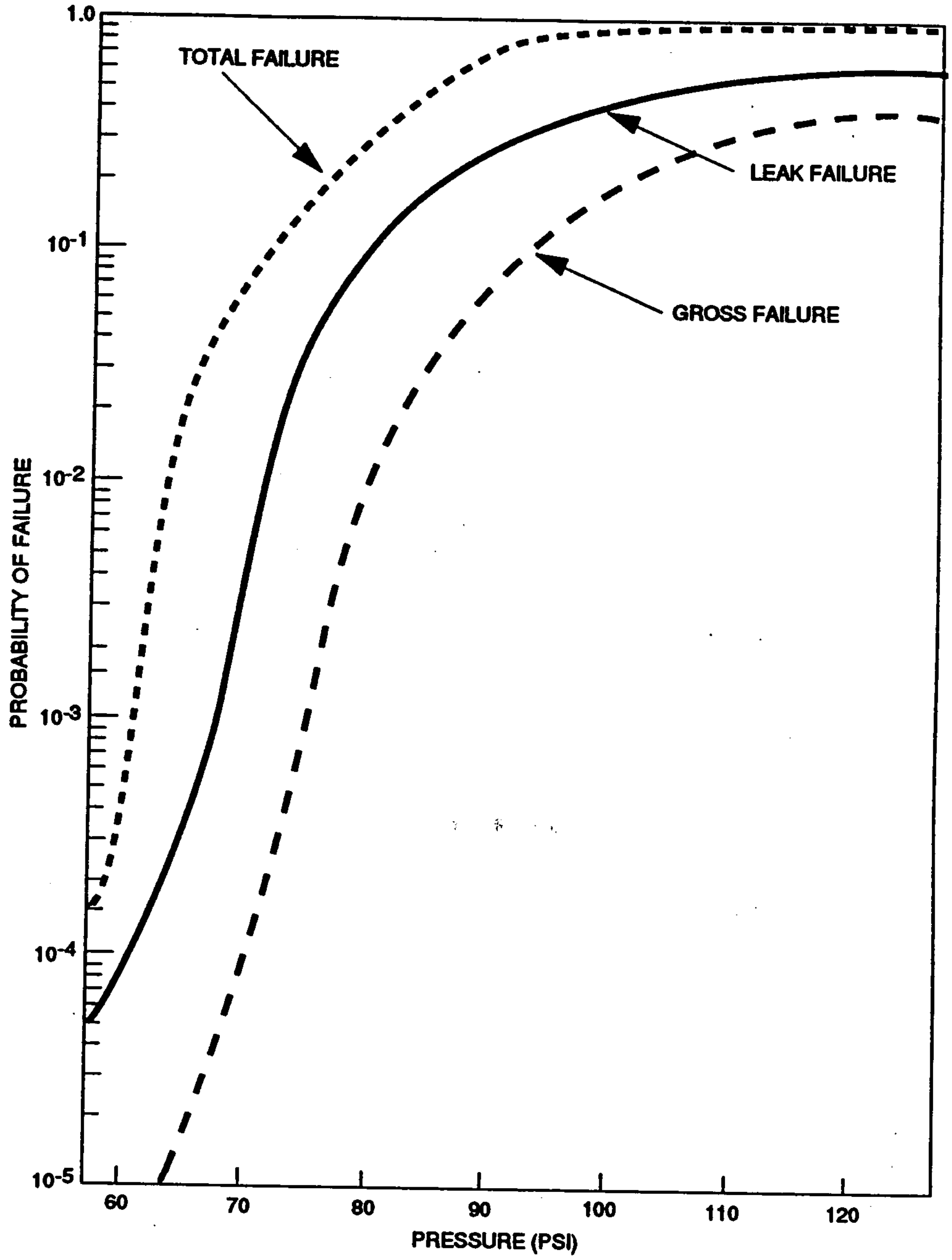

Figure 2-18. Typical Containment Failure Probability Distributions for Leak Failure, Gross Failure, and Total Failure

3. FRONT-END ANALYSIS

3.1 ACCIDENT SEQUENCE DEFINITION

This section describes the accident sequence models that were developed for the front-end analysis. The accident sequence models are used to combine the results of the systems analysis (presented in Section 3.2) in order to perform the front-end sequence quantification, as described in Section 3.3. This section describes the selected initiating event categories, the response of each system needed to mitigate each initiator, and the assignment of end states to each accident sequence.

The purposes of the front-end plant model are to define a set of potential accident scenarios that could result in core damage, and to evaluate the status of containment engineered systems for consideration in the back-end. Accident scenarios are defined by evaluating the plant response to an initiating event. "Plant response" refers to the progression of a wide spectrum of possible event sequences based on the success or failure combinations of plant systems/equipment, and human operator actions that could either prevent core damage or mitigate the accident consequences, should core damage occur.

An initiating event is any event that initiates a plant transient condition or otherwise perturbs the normal operation of the plant, which, together with associated failures evaluated in the front-end plant model, results in a sequence of events that may involve undesirable consequences, such as the release of radioactive material.

The plant model therefore consists of scenarios that begin with initiating events and end states with either stable plant conditions or states of core damage. "Stable plant condition" means that the plant is in either a stable hot shutdown or a cold shutdown condition 24 hours after the initiating event has occurred, with core decay heat being safely rejected or removed.

The first objective of the front-end plant modeling is to construct a listing of the set of accident sequences. The second key objective is to quantify the likelihood and associated uncertainties of these accident sequences.

To accomplish these two objectives, it is first necessary to identify a sufficiently complete and well-defined set of initiating events that is specific to Watts Bar. The identification of the initiating events is performed using two methods: a review of initiating event lists from other studies and the Final Safety Analysis Report, and a failure modes and effects analysis (FMEA) of plant systems and components. Section 3.1.1 describes the selection of initiating events that are specific to Watts Bar.

The next step in the construction of a set of accident sequences is to identify the equipment items and systems that are required to operate, and the operator actions that are necessary to successfully mitigate each initiator. The model for possible event sequences is made up of two key parts: the support systems models and the frontline systems models. The support systems do not directly perform the plant-mitigating functions in response to a plant disturbance. Instead, they provide the necessary control and motive power, cooling water, and actuation signals needed for the frontline systems to perform the plant-mitigating functions. An example of a support system is the electric power system; the auxiliary feedwater system is an example of a frontline system.

A key to the understanding of the systems at Watts Bar are the system dependency tables that show how a failure of each support system (e.g., major electric power bus or vital instrument bus) affects equipment in other support systems, and how a failure of a support system affects frontline system trains or equipment. Information from the dependency tables is used to construct the support system models. The system dependency tables are presented in Section 3.2.3. Section 3.1.4 describes the support system model event trees developed for Watts Bar.

The development of the frontline systems response to each initiator is performed with the aid of event sequence diagrams (ESD). ESDs are logic diagrams that display the analysts' understanding and assumptions about the physical development of accident sequences and the key operator actions. The ESDs for Watts Bar are presented in Section 3.1.2.1. The events in the ESDs are keyed to the steps in the Emergency Operating Procedures to facilitate review and to ensure proper consideration of the operator actions.

Since the ESDs do not easily lend themselves to direct quantification, the ESDs are converted into event tree model logic for sequence quantification. Section 3.1.2 provides a detailed description of the frontline accident sequence models developed for Watts Bar.

A large number of accident sequences are specified by the plant model, each beginning with an initiating event and ending with a plant damage state (PDS). The PDSs define categories of core damage sequences for consideration in the back-end analysis; e.g., whether, in addition to core damage, the containment isolates successfully. The PDSs defined for Watts Bar are described in Section 3.1.5. The active containment systems used for consequence mitigation are considered in the definition of plant damage states. This ensures proper treatment of dependencies between core cooling systems, containment systems, and their support systems. The relationships between the states of top events in the front-end event trees and the physical parameters being tracked for use in assigning PDSs are presented in Section 3.1.3.2.

To quantify the frequency of each sequence defined by the plant model, system-specific logic models are developed for each mitigating system. The development and quantification of the system models are presented in Section 3.2, and the results are presented in Section 3.3.5.

The use of these system results to quantify the plant event tree models is described in Section 3.3. Recovery models are developed based on the results of a preliminary sequence quantification. These recovery models include the development of a new event tree in which recovery actions are explicitly modeled. These recovery models are described in Section 3.1.3.

3.1.1 INITIATING EVENTS

The initiating event categories selected for the Watts Bar Nuclear Plant Unit 1 probabilistic risk assessment (PRA)/individual plant examination (IPE) model are presented in this section. The three main objectives in selecting initiating events are to:

- Provide completeness to account for all possible initiating events.
- Account for unique plant design and operational features.
- Categorize the events in unique ways that the event may impact the rest of the plant.

This process of grouping initiating events by similarity of plant response is common to PRA/IPE models and helps to limit the number of plant event sequence models to be developed. It is necessary and practical to analyze only those initiating events that make appreciable contribution to risk. Given knowledge of the approximate frequency of the initiating events and the relative impact of these events on the plant systems, it is desirable to screen and group the initiating events to simplify the quantification of risk. This is possible without introducing significant errors into the risk estimate.

The list of initiating event categories selected for consideration in the Watts Bar Unit 1 IPE is presented in Table 3.1.1-1. Each initiating event category identified in this table leads to a plant trip; i.e., either a reactor trip or a turbine trip. Events that lead only to an orderly, controlled shutdown are not considered. This is because, during a normal, controlled shutdown, the plant is near equilibrium, the shutdown proceeds at a controlled rate, and the standby systems are started before they are needed. If the standby systems fail, most of the normal systems are available to maintain operation, and the allowed recovery response times are greater. Since the reactor is being shut down, the number of safety functions that must be performed to provide sufficient core cooling is reduced. Therefore, normal, controlled shutdowns are not considered as initiating events for quantification.

Failure of the reactor to trip automatically, called anticipated transient without scram (ATWS), is considered in the IPE model in the course of developing plant response scenarios. Therefore, ATWS events are not defined as a separate initiating event category.

Of the so-called external events such as earthquakes, severe weather conditions, internal plant fires, and internal and external floods, only the internal floods are considered for the current IPE. The other external event initiators are not required for the initial IPE submittal. Therefore, only a task to investigate the scenarios initiated by internal floods was performed for the external events.

The initiating event categories listed in Table 3.1.1-1 were identified using several approaches: a comparison with categories from previous PRAs and other industry studies, a failure modes and effects analysis (FMEA) of the plant systems, a review of the Final Safety Analysis Report (FSAR), discussions with plant operators about specific postulated events, and a review of plant trips that have occurred at TVA's Sequoyah Nuclear Plant.

A very effective approach for identifying initiating event categories and ensuring completeness is to compare similar lists prepared for other Westinghouse reactors. Numerous lists are available and were considered during the preparation of the initiating event categories for Watts Bar Unit 1. In particular, the lists used in this study included those prepared for the Diablo Canyon PRA (Reference 3.1.1-1), the South Texas Project PRA (Reference 3.1.1-2), and the recent core damage frequency analysis from internal events performed for Sequoyah Unit 1 in NUREG/CR-4550, Volume 5 (Reference 3.1.1-3). The Diablo Canyon and South Texas Project PRAs are particularly useful since they include a formal application of the master logic diagram and heat balance fault tree techniques to search for key initiating events in a further effort to ensure completeness. In addition, the lists of events used herein were compared against other published sources, including the NUREG/CR-3862 report (Reference 3.1.1-4), WASH-1400 (Reference 3.1.1-5), NUREG/CR-2300 (Reference 3.1.1-6), the Indian Point Probabilistic Safety Study (Reference 3.1.1-7), and the PLG Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants (Reference 3.1.1-8).

The initiating event categories selected for Watts Bar Unit 1 fall into four broad groups: losses of reactor coolant inventory, transients, loss of support system events, and internal floods.

The loss of coolant inventory initiating event categories are the same as those quantified in earlier studies; i.e., References 3.1.1-1 through 3.1.1-5. Included in this group are the interfacing systems loss of coolant accident (LOCA) events that may lead to release paths that bypass the containment. These initiators receive special consideration. The interfacing systems LOCA analysis, including the development of initiating event probability distribution, is presented in Section 3.3.9.

The list of transient initiating event categories prepared for Watts Bar Unit 1 closely parallels the lists developed for Diablo Canyon and the South Texas Project in that it is more detailed than the list prepared for the analysis of Sequoyah Unit 1 in NUREG/CR-4550; that is, the NUREG/CR-4550 transient categories, events with and without main feedwater (MFW) available, have been further subdivided for more accurate treatment of the plant response to each subcategory.

The support system faults of interest were identified by an FMEA of key plant support systems. This analysis is documented in Table 3.1.1-2. The analysis makes use of information in the intersystem dependency tables presented in Section 3.2.3. Support system faults are of special interest for IPE quantification because they are very plant specific and because they cannot only cause a plant trip but also degrade the systems designed to mitigate such events. As such, they have been found to be important risk contributors.

The support system events that are listed in Table 3.1.1-1 provide a thorough coverage of electrical and other support system faults. The loss of offsite power initiating event is modeled as if power were unavailable at the 161-kV source. Losses of the 500-kV source in which the 161-kV source remains available are treated in the transient category for turbine trip.

Loss of various plant heating, ventilating, and air conditioning (HVAC) systems were carefully considered in this analysis (see Reference 3.1.1-9). While HVAC systems were found to be important support systems in the plant response during some modeled event sequences, the loss of any particular HVAC system was determined to be insignificant as an initiating event for the IPE. This is the case primarily because the loss of important HVAC systems is well annunciated, and heatup calculations show that there is ample time for the operators to restore HVAC or conduct a normal plant shutdown prior to equipment failure causing a plant trip.

Another category of initiating events is internal floods. Several internal flood initiators, each with a different frequency and flood damage impact, were selected for quantification. The approach followed to identify the internal flood initiators and to assign proper initiating event frequencies for specific flood scenarios is documented in Section 3.3.8.

Although the list of initiating event categories for Watts Bar Unit 1 is more detailed than that developed for Sequoyah in NUREG/CR-4550, 1 of the 10 initiating event categories considered for the Sequoyah analysis in NUREG/CR-4550 was not considered in the Watts Bar Unit 1 IPE. The initiating event category for very small LOCA (i.e., less than ½-inch equivalent diameter) was considered in NUREG/CR-4550 but not for the Watts Bar Unit 1 IPE. Instead, such events are assumed to be within the makeup capacity of the normal charging system as described in the FSAR and therefore would not lead to an immediate plant trip.

Table 3.1.1-1 presents the initiating event frequencies used in the IPE quantification. The unit for each initiating event category frequency probability distribution is initiating event per nuclear-powered electric generating unit per calendar year. This definition is referred to as simply per reactor-year in this presentation. The primary source of generic data is Reference 3.1.1-8. The methodology applied in developing the initiating event frequency probability distributions is described in Reference 3.1.1-8 and Section 3.3.1.

To develop system and event sequence models accurately and efficiently for the IPE, it is important to define clearly specific success criteria for each major plant safety function modeled in the IPE with respect to the initiating event categories. The major plant functions modeled in the IPE are reactor criticality control, early core heat removal, reactor coolant system integrity, containment pressure suppression, late core heat removal, and containment atmospheric heat removal. The success criteria for these major plant safety functions with respect to each initiating event defined herein were determined through engineering knowledge of the plant and a careful review of References 3.1.1-3 through 3.1.1-10. A summary of these success criteria is presented in Table 3.1.1-3. A more detailed statement of system success criteria applied in event sequence quantification is presented in the individual system analysis notebooks developed as supporting documents to this report and in Appendix A.

The event sequence models that were developed to consider the plant response to each of these initiating event categories are presented in the following sections.

3.1.1.1 References

3.1.1-1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric Company, Vols. 1 - 9, PLG-0637, July 1988.

3.1.1-2. Pickard, Lowe and Garrick, Inc., "South Texas Project Probabilistic Safety Assessment, Summary Report," prepared for Houston Lighting & Power Company, PLG-0700, Vols. 1 and 2, April 1989.

3.1.1-3. Bertucio, R. C., et al., "Analysis of Core Damage Frequency: Sequoyah, Unit 1 Internal Events," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4550, Vol. 5, Rev. 1, April 1990.

3.1.1-4. EG&G Idaho, Inc., "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3862, May 1985.

3.1.1-5. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risk in U.S. Nuclear Power Plants," WASH-1400, NUREG-75/014, 1975.

3.1.1-6. American Nuclear Society and Institute of Electrical and Electronic Engineers, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, NUREG/CR-2300, April 1983.

3.1.1-7. Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Indian Point Probabilistic Safety Study," prepared for the Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., March 1982.

3.1.1-8. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Vol. 6, PWR Initiators, proprietary, August 1989.

3.1.1-9. Stillwell, D. W., "Memorandum on the Importance of Watts Bar Unit 1 HVAC Systems in the IPE," May 1992.

3.1.1-10. Tennessee Valley Authority, "Watts Bar Unit 1 Final Safety Analysis Report (FSAR)," April 1992.

Table 3.1.1-1 (Page 1 of 2). Watts Bar Initiating Event Categories and Associated Frequencies

Initiating Event		Plant Model Designator	Mean Initiating Event Frequency (per reactor-year)[1]
Group	Category		
Loss of Coolant	1. Excessive LOCA (reactor vessel failure)	ELOCA	2.66-7
	2. Large LOCA (> 6-inch diameter)	LLOCA	2.03-4
	3. Medium LOCA (≥ 2 to ≤ 6-inch diameter)	MLOCA	4.65-4
	4. Small LOCA[2] (nonisolable)	SLOCAN	5.83-3
	5. Small LOCA[3] (isolable)	SLOCAI	2.30-2
	6. Steam Generator Tube Rupture	SGTR	2.84-2
	7. Interfacing Systems LOCA — RHR Injection Path	VI	4.00-6
	8. Interfacing Systems LOCA — RHR Suction Path	VS	7.20-6
Transients	9. Reactor Trips[4]	RTIE	1.35+0
	10. Core Power Excursion	CPEX	2.68-2
	11. Turbine Trip[5]	TTIE	1.07+0
	12. Inadvertent Safety Injection	ISI	2.99-2
	13. Total Loss of All Main Feedwater	TLMFW	1.62-1
	14. Partial Loss of Main Feedwater	PLMFW	1.13+0
	15. Loss of Condenser Vacuum	LOCV	1.18-1
	16. Excessive Feedwater	EXMFW	1.68-1
	17. Inadvertent Closure of One MSIV	MSIV	8.66-2
	18. Inadvertent Closure of All MSIVs	IMSIV	1.93-2
	19. Loss of Primary Flow	LRCP	1.76-1
	20. Steam Line Break outside Containment	SLBOC	6.04-3
	21. Steam Line Break inside Containment	SLBIC	4.65-4
	22. Inadvertent Opening of Main Steam Relief Valves	MSVO	4.19-3

Notes:

1. Exponential notation is indicated in abbreviated form; e.g., 2.66-7 = 2.66×10^{-7}. The unit for the initiating event frequency is events per nuclear-powered electric generating unit per calendar year. This definition is abbreviated to per reactor-year in this presentation.
2. A small break LOCA is defined as any RCS inventory loss greater than the makeup ability of one centrifugal charging pump through normal charging up to a 2-inch-diameter break. These nonisolatable LOCAs are primarily RCP failures.
3. The size of this small LOCA is the same as that in Note 2. These isolable LOCAs are primarily PORV failures.
4. The reactor trip initiator will include all transients that result in an automatic or a manual reactor trip other than those categorized in separate initiating event categories.
5. The turbine trip initiator will include all transients that will cause a turbine trip automatically and manually other than those categorized in separate initiating event categories.

Table 3.1.1-1 (Page 2 of 2). Watts Bar Initiating Event Categories and Associated Frequencies

Initiating Event: Group	Initiating Event: Category	Plant Model Designator	Mean Initiating Event Frequency (per reactor-year)[1]
Loss of Support Initiating Events	23. Loss of Offsite Power	LOSP	8.56-2
	24. Loss of 1A-A, 6.9-kV Shutdown Board	LASD[2]	3.03-3
	25. Loss of 1B-B, 6.9-kV Shutdown Board	LBSD[2]	3.04-3
	26. Loss of 1-I Vital AC Instrument Board	LDAAC[2]	1.19-1
	27. Loss of 1-II Vital AC Instrument Board	LDBAC[2]	1.19-1
	28. Loss of 1-III Vital AC Instrument Board	LDCAC[2]	1.16-1
	29. Loss of 1-IV Vital AC Instrument Board	LDDAC[2]	1.14-1
	30. Loss of Vital Battery Board I	LVBB1[2]	5.98-3
	31. Loss of Vital Battery Board II	LVBB2[2]	5.79-3
	32. Total Loss of CCS	CCSTL[2]	1.11-3
	33. Loss of CCS Train A	CCSA[2]	2.78-2
	34. Total Loss of ERCW	ERCWTL[2]	1.51-5
	35. Loss of ERCW Train A	ERCWA[2]	7.10-4
	36. Loss of ERCW Train B	ERCWB[2]	7.10-4
Internal Flooding Events	37. Turbine Building Flood — Loss of Feedwater, Condenser, and Station Air	FLTB	2.0-2
	38. ERCW Strainer Room A Flood — Loss of All Four "A" Pumps and Headers	FLPHIA	2.3-3
	39. ERCW Strainer Room B Flood — Loss of All Four "B" Pumps and Headers	FLPHIB	2.3-3
	40. ERCW Flood in Auxiliary Building for 30 Minutes — RHR and Containment Spray Unavailable	FLAB2	4.2-6
	41. CST Drained to Auxiliary Building — CST, RHR, Containment Spray, and one AFW Pump Unavailable	FLAB3C	2.8-5
	42. RWST Drained to Auxiliary Building — RWST, RHR, and Containment Spray Unavailable	FLAB3R	3.2-3

Notes:

1. Exponential notation is indicated in abbreviated form; e.g., 2.66-7 = 2.66×10^{-7}. The unit for the initiating event frequency is events per nuclear-powered electric generating unit per calendar year. This definition is abbreviated to per reactor-year in this presentation.
2. The system model described in the systems notebooks was used to quantify initiating event frequencies for these support system failures.

Table 3.1.1-2 (Page 1 of 5). Failure Modes and Effects Analysis of Watts Bar Unit 1 Key Systems

System/Subsystem	Failure Mode	Effect on Safety System(s) or Key Plant Equipment	Initiating Event Category	Code Designator	Comments
Offsite Grid					
500-kV Line	Discontinuity/Loss of Power from	Turbine Trip Reactor Trip	9	TTIE	Results in a generator trip on load rejection and fast transfer to 161-kV line.
161-kV Line	Discontinuity/Loss of Power from	Loss of Power to 6.9-kV Shutdown Boards	--	--	Does not cause a plant trip of offsite power to safety equipment.
Both 161-kV and 500-kV Lines	Discontinuity/Loss of Power from	Reactor Coolant Pump (RCP) Condensate Main Condenser Circulation Water Secondary Component Cooling Water	21	LOSP	Results in a plant trip. Equipment listed is unavailable. Equipment normally operating and powered from emergency buses must restart.
Nonemergency AC Power					
Unit Station Service Transformer CSST RCP Buses 6.9-kV Unit Boards 6.9-kV Common Boards 480V Common Boards	Discontinuity/Loss of Power from	Subset of the Equipment Impacted by a Loss of Both 161-kV and 500-kV Lines	21	LOSP	Loss of these electric power systems is bounded by the loss of offsite power event for both frequency of occurrence and impact. Loss of individual boards results in a loss of balance-of-plant (BOP) equipment but does not usually result in a plant trip. Losses that result in a plant trip are included in trips resulting from equipment loss.
Emergency AC Power					
CSST 6.9-kV Shutdown Boards 480V Shutdown Boards	Discontinuity/Loss of Power from	Loss of Normal Source to 6.9-kV Shutdown Board Battery Chargers for 250V and 125V DC Power Control Air System Compressors Various Motor-Driven Pumps: ERCW, CCS, AFW, CVCS, Safety Injection, RHR, CSS, HPFP, etc.	--	--	Loss of a single train may require the opposite train of a normally operating system to start; e.g., CVCS.

Table 3.1.1-2 (Page 2 of 5). Failure Modes and Effects Analysis of Watts Bar Unit 1 Key Systems

System/Subsystem	Failure Mode	Effect on Safety System(s) or Key Plant Equipment	Initiating Event Category	Code Designator	Comments
125V Vital DC Power					
Battery Board I	Discontinuity/Loss of Power from	Main Steam Isolation Valves Main Feed Regulating Valves Steam Generators 2 and 4 Main Feed Regulating Bypass Valves Primary PORV Steam Generator 2 PORV Reactor Trip Breaker Control Power Steam Dump Valves AFW Motor-Driven Pump 1A-A Breaker Control Power for CCP 1A-A	28	LVBB1	Reactor trip - turbine trip due to MSIVs, feedwater control, and bypass valves failing closed, resulting in loss of feedwater and low-low steam generator water level.
Battery Board II	Discontinuity/Loss of Power from	Main Steam Isolation Valves Main Feed Regulating Valves Steam Generators 1 and 3 Main Feed Regulating Bypass Valves Primary PORV Steam Generator 3 PORV Reactor Trip Breaker Control Power Steam Dump Valves AFW Motor-Driven Pump 1B-B Breaker Control Power to CCP 1B-B	29	LVBB2	Reactor trip - turbine trip due to MSIVs, feedwater control, and bypass valves failing closed, resulting in loss of feedwater and low-low steam generator water level.
120V Vital AC					
Board 1-I	Discontinuity/Loss of Power from	Steam Generator Level Control Main Feed Pumps Go to Minimum Speed Auxiliary Control Air Dryer A RPS and SSPS Room Cooling Train A AFW Train A Actuation RHR Heat Exchanger 1A-A Outlet Valve MFWP Auxiliary Relay Panel	24	LDAAC	Reactor trip assumed due to steam generator level changes.
Board 1-II	Discontinuity/Loss of Power from	SSPS Train B Auxiliary Control Air Dryer B RPS and SSPS Room Cooling Train B AFW Train B Actuation RHR Heat Exchanger 1B-B Outlet Valve MFWP Auxiliary Relay Panel	25	LDBAC	Reactor trip assumed due to steam generator level changes.

Table 3.1.1-2 (Page 3 of 5). Failure Modes and Effects Analysis of Watts Bar Unit 1 Key Systems

System/Subsystem	Failure Mode	Effect on Safety System(s) or Key Plant Equipment	Initiating Event Category	Code Designator	Comments
120V Vital AC (con't.)					
Board 1-III	Discontinuity/Loss of Power from	Steam Generator 3 Feed Flow Demand Signal AFW TDP Flow Controller TDAFWP Steam Generators 3 and 4 Level Control SSPS Train A	26	LDCAC	Reactor trip assumed due to steam generator low-low level resulting from reduction of feedwater flow from steam generator 3 feed flow demand failing low.
Board 1-IV	Discontinuity/Loss of Power from	Steam Generator 4 Feed Flow Demand Signal SSPS Train B MFWP Auxiliary Relay Panel	27	LDDAC	Reactor trip assumed due to steam generator low-low level resulting from reduction of feedwater flow from steam generator 4 feed flow demand failing low.
SSPS	Fault Leading to Inadvertent Safety Function/System Actuation	Actuation for CCS, MSIV, Main Turbine Trip, AFW, Reactor Trip, CVCS, Safety Injection, RHR, CSS, EGTS, Containment Isolation, and Air Return Fans	10	ISI	Spurious signal causes an inadvertent safety injection. Spurious actuation of individual systems is also possible. Loss of one train will lead to an orderly shutdown per Technical Specifications.
Raw Cooling Water	Loss of Cooling Function from	Condensate System Pumps Main Feedwater Pumps	11	TLMFW	Loss results in a failure of the condensate system and feedwater pumps and a plant trip.
Instrument Air					
Essential Air (Auxiliary Control Air)	Inadequate System Pressure or Capacity	EGTS Steam Generator PORVs Steam Generator Level Control Valves for AFW	--	--	Standby system actuated on loss of control air. Failure of this system considered subsequent to failure of control air.

Table 3.1.1-2 (Page 4 of 5). Failure Modes and Effects Analysis of Watts Bar Unit 1 Key Systems

System/Subsystem	Failure Mode	Effect on Safety System(s) or Key Plant Equipment	Initiating Event Category	Code Designator	Comments
Essential Raw Cooling Water Train A	Loss of Cooling Function from	CCS Heat Exchanger B Alternate Cooling to Centrifugal Charging Pump 1 A-A Cooler Diesel Generators Auxiliary Control Air Compressors RCP Motor Coolers 1 and 3 Containment Spray Heat Exchangers AFW Backup Water Source CCS Lube Oil Cooler 1A Alternate Room Coolers for: CSS, CVCS, RHR, Safety Injection, CCS/AFW	33	ERCWA	Plant trip assumed.
Essential Raw Cooling Water Train B	Loss of Cooling Function from	CCS Heat Exchangers C and A Diesel Generators Auxiliary Control Air Compressors RCP Motor Coolers 2 and 4 Containment Spray Heat Exchangers AFW Backup Water Source Room Coolers for: CSS, CVCS, RHR, Safety Injection, CCS/AFW	34	ERCWB	Plant trip assumed.
Essential Raw Cooling Water System	Loss of Cooling Function from	All Equipment Supplied by ERCW Trains A and B	32	ERCWTL	Plant trip assumed.
Refueling Water Storage Tank	Loss of Flow Source from	Primary water source for Safety Injection, RHR, CVCS	-	-	This event is applicable only with concurrent requirement for ECCS systems and therefore is not a separate initiator.

Table 3.1.1-2 (Page 5 of 5). Failure Modes and Effects Analysis of Watts Bar Unit 1 Key Systems

System/Subsystem	Failure Mode	Effect on Safety System(s) or Key Plant Equipment	Initiating Event Category	Code Designator	Comments
Component Cooling Water					
Train A	Loss of Cooling Function from	Normal Cooling to CCP 1A-A Pump Oil Coolers RCP Bearing Oil Coolers RCP Thermal Barrier Coolers Pump Oil Coolers for Train A: CVCS, CSS, Safety Injection Spent Fuel Pit Heat Exchangers Train A RHR Heat Exchangers	31	CCSA	Manual trip to protect RCPs.
Train B	Loss of Cooling Function from	Pump Oil Coolers for Train B: CVCS, CSS, Safety Injection Train B RHR Heat Exchangers	--	CCSB	Not a separate initiator.

Table 3.1.1-3 (Page 1 of 3). Success Criteria Summary

Function	Initiating Event: Group*	Initiating Event: Categories*	Success Criteria	Comments
1. Reactor Criticality Control	Loss of Coolant Inventory	1 and 2 (> 6")	Not required.	Reactor subcriticality is provided by core voiding and borated water injection.
		3 (2" to 6")	Not required.	Reactor subcriticality is provided by borated water injection.
		4 through 6 (< 2")	Reactor trip.	
	Transients	7 through 20	Reactor trip.	
	Loss of Support System	21 through 34	Reactor trip.	
2. Early Core Heat Removal	Loss of Coolant Inventory	1 and 2 (> 6")	One of two low pressure injection trains and three of three intact accumulator trains.	
		3 (2" to 6")	Two of four high pressure injection trains.	
		4 through 6 (< 2")	One of four high pressure injection trains and one of three auxiliary feedwater trains, or one of four high pressure injection trains and two of two power-operated relief valves in feed and bleed operation.	
	Transients	7 through 20	One of three auxiliary feedwater trains, or one main feedwater and condensate booster train, or one of four high pressure injection trains and two of two power-operated relief valves in feed and bleed operation.	
	Loss of Support System	21 through 34	One of three auxiliary feedwater trains, or one main feedwater and condensate booster train, or one of four high pressure injection trains and two of two power-operated relief valves in feed and bleed operation.	
3. Reactor Coolant System Integrity	Loss of Coolant Inventory	1 through 6	Not applicable.	Reactor coolant system integrity is lost as a direct result of the initiating event.
	Transients	7 through 20	All open power-operated relief valves close on demand; one of two centrifugal charging pump trains operating in a reactor coolant pump seal injection mode, or one component cooling water train providing cooling to reactor coolant pump thermal barriers.	

*See Table 3.1.1-1.

Table 3.1.1-3 (Page 2 of 3). Success Criteria Summary

Function	Initiating Event		Success Criteria	Comments
	Group*	Categories*		
3. Reactor Coolant System Integrity (continued)	Loss of Support System	21 through 34	All open power-operated relief valves close on demand; one of two centrifugal charging pump trains operating in a reactor coolant pump seal injection mode, or one component cooling water train providing cooling to reactor coolant pump thermal barriers.	
4. Containment Pressure Suppression	Loss of Coolant Inventory	1 through 3 (≥ 2")	Ice condenser system success.	
		4 through 6 (< 2")	Not required.	Containment pressure suppression is not required during the initial blowdown phase of a small loss of coolant accident.
	Transients	7 through 20	Not required.	Containment pressure suppression is not required for initiating events that do not involve loss of coolant inventory or main steam line break inside containment.
	Loss of Support System	21 through 34	Not required.	Containment pressure suppression is not required for initiating events that do not involve loss of coolant inventory.
5. Late Core Heat Removal	Loss of Coolant Inventory	1 through 3 (≥ 2")	One of two low pressure recirculation trains.	
		4 through 6 (< 2")	One of four high pressure recirculation trains and one of two low pressure recirculation trains.	
	Transients	7 through 20	One of four high pressure recirculation trains and one of two low pressure recirculation trains.	Late core heat removal is only required when feed and bleed is used or when reactor coolant system integrity is lost.
	Loss of Support System	21 through 34	One of four high pressure recirculation trains and one of two low pressure recirculation trains.	Late core heat removal is only required when feed and bleed is used or when reactor coolant system integrity is lost.

*See Table 3.1.1-1.

Table 3.1.1-3 (Page 3 of 3). Success Criteria Summary				
Function	**Initiating Event**		**Success Criteria**	**Comments**
	Group*	**Categories***		
6. Containment Atmospheric Heat Removal	Loss of Coolant Inventory	1 through 3 (≥ 2")	One of two containment spray system trains with heat exchanger.	
		4 through 6 (< 2")	One of two containment spray system trains with heat exchanger, <u>or</u> one of three auxiliary feedwater trains and one of two low pressure recirculation trains with heat exchanger and low pressure recirculation spray.	
	Transients	7 through 20	One of two containment spray system trains with heat exchangers.	Containment atmospheric heat removal is only required when feed and bleed is used or when reactor coolant system integrity is lost.
	Loss of Support System	21 through 34	One of two containment spray system trains with heat exchangers.	Containment atmospheric heat removal is only required when feed and bleed is used or when reactor coolant system integrity is lost.

*See Table 3.1.1-1.

3.1.2 FRONTLINE EVENT TREES

This section describes the development and definition of the frontline event trees. First, the Watts Bar-specific event sequence diagrams (ESD) are presented in Section 3.1.2.1. Insights gained from the ESDs are then used in the definition of the frontline event tree top events. The general transient, medium loss of coolant accident (LOCA), and large LOCA event trees are presented in Section 3.1.2.2. The general transient event tree is used for all initiators other than for the medium and large LOCAs. It includes considerations of small LOCAs, steam generator tube ruptures (SGTR), steam line breaks, feedwater transients, and anticipated transient without scram (ATWS) sequences.

3.1.2.1 Event Sequence Diagrams

Event sequence diagrams are used to document the possible scenarios and courses of actions that can be taken by the operators after a specified initiating event has occurred. Such actions include the plant hardware response and the steps taken by the operators. The ESDs document graphically the probabilistic risk assessment (PRA) team's understanding of how the plant is designed and operated to respond to transient events. Construction of the ESDs is the first step towards the development of event trees. The event trees will subsequently be used to quantify the frequency of all frontline accident sequences.

Although ESDs are easily understood and are useful tools for documenting required plant systems and operator actions after an initiating event has occurred, they do not lend themselves directly to accident sequence quantification. A necessary next step then is to convert the understanding documented in the ESDs into an event tree for the purpose of quantification of accident sequences. The event tree represents the transformation of the qualitative details contained in the ESD into a functional logic framework for quantification. Specific actions identified in the ESD are grouped into top events for the corresponding event tree. For each top event then, the system boundary, boundary conditions, and success criteria are defined for the system or operator actions associated with the top event. The frontline event trees are described in the remainder of this section and in Section 3.1.3.

3.1.2.1.1 ESD Symbology

The symbols used in constructing the ESDs are shown in Figure 3.1.2-1. The starting point, or accident sequence initiating event, is identified by a "waving flag" block. This first event is drawn in the upper left-hand corner of the first page of the ESD. Subsequent operator actions and system responses to the initiator are then presented throughout the remainder of the ESD. The normal, or expected, sequence of events is drawn across to the right, beginning with the initiator. These events are arranged, for the most part, in chronological order. This is not strictly adhered to because sometimes it is easier to follow the events if related actions, dependent on each other, are grouped together even though they may not be closely related in time. Also, the sequence of events depicted is closely related to the steps in the Emergency Operating Procedures (EOP), which is not always the same order in which the plant responds.

Events whose occurrence or nonoccurrence influences the course of a scenario are represented by rectangles. Successful occurrence of the event described within the rectangle is represented by the arrow exiting the right side of the rectangle; failure of the described event is represented by the arrow exiting the base of the rectangle. Rectangles are the only symbols in the ESD having two exit paths. The path sequences leading into rectangles are shown by arrows entering from the left or top of the rectangle. Rectangles describing the same event may be asked more than once along a single sequence path. This is because, like the EOPs, not all paths up to a specified point in the ESD are unique. While the event may have been asked along one path up to that point, it may not have been asked along another path to that point. Therefore, it is sometimes useful to ask the same event in more than one place in the ESD.

The oval symbol is used as a place for describing the status of key plant parameters or the phenomena that would likely occur as a result of an accident sequence described by the events up to that point in the ESD. Since the oval symbol is just a descriptive block, only one exit path is allowed. The exit path may be drawn from the bottom or right-hand side of the oval. The position of the exit path does not signify anything special but is merely to facilitate the linking of the events. Several entry paths are permitted, however. The entry paths may enter from any direction.

Events representing an entire sequence may result in either successful mitigation of the initiating event or in core damage. Success or stable end states are represented by parallelogram-shaped figures. Sequences resulting in at least some core damage are shown ending in diamond-shaped symbols. There are no exit paths from such symbols.

Two additional symbols are used to represent transfers to other places in the ESD logic that cannot be conveniently connected by solid lines. The triangle symbols are used to connect two different places in the ESD. Triangles with the point upward represent exits from the indicated place in the ESD. Triangles with the point facing down indicate transfers into the indicated place in the ESD from another portion of the ESD. These triangles contain numbers so that the exit and entrance points can be readily matched. If the exit to another procedure coincides with the return point from that same procedure, then just the exit triangle is used. In such cases, the arrows into and out of the triangle indicate that the operators are to complete the actions in the procedure transferred to, and then return to the procedure transferred from.

Some paths through the ESD are judged to have so small a likelihood of occurrence that no further modeling is performed. These paths may terminate in elongated hexagons that explicitly say, "Not Developed Further." The preceding event is then assumed to always result in the alternative outcome, which is then developed further.

Throughout the ESD, reference is made to numbered steps in the Watts Bar EOPs. These references indicate the places in the EOPs where the operators would be if the accident sequence had progressed to that point. The referenced procedures may direct the operators to carry out or to verify the actions represented by the nearby event.

3.1.2.1.2 Watts Bar Event Sequence Diagram Overview

The EOPs evaluated for the Watts Bar IPE are those that were effective on December 1, 1991. The EOPs are symptom based. Entry to the EOPs is always through E-0, which then directs the operators to the appropriate steps, depending on the plant indications. Therefore, to facilitate communication with plant operating staff, it was decided for this project to have just one large ESD that covers all initiating events considered in the study. Table 3.1.2-1 identifies the Emergency Operating Procedures modeled explicitly in the ESD. The ESD itself is presented as Figure 3.1.2-2. These procedures cover the automatic plant and directed operator responses to all events. The ESD is organized similar to the EOPs, with one sheet representing a single, separate EOP.

The ESD documents the path through the EOPs that the operator is expected to take for selected accident sequences. For example, in the event of a LOCA, the procedures may direct the operators to enter the post-LOCA cooldown procedure, or to transfer to the residual heat removal (RHR) containment sump before leaving EOP E-1. The decision as to which procedure to follow is symptom based. The ESD, on the other hand, documents the event conditions that are assumed in the development of the IPE models to lead the operators to one path or the other. These judgments are a necessary first step in deciding the proper sequence of actions for the corresponding event trees.

The purpose of the ESD is to document in a fairly detailed way the possible plant response and the procedural guidance provided to the operators for a wide range of events. The intention is to identify plant-specific conditions that may lead to core damage and to relate such conditions to steps in the EOPs that the operators follow up to that point. A key objective is to ensure that the operator actions are properly considered in the study. Once the core damage conditions are identified, the events that affect the response of the containment and the potential for radioactivity release from containment (e.g., containment isolation, heat removal, and spray status) are questioned. The ESD includes more events than are actually incorporated into the final event trees used to quantify the accident sequence frequencies. This level of detail is useful for discussion purposes, particularly with representatives of plant operations who are most familiar with the EOPs. The added detail permits discussions of what is to be omitted, as well as what is to be included, in the individual plant examination (IPE) models.

The logical structure of the ESD is developed so that it can be specialized to model the plant response to many initiating events. It includes various success paths that satisfy the major core protection functions; i.e., core reactivity control, coolant inventory control, and core heat removal. The model also represents the important features that can affect plant and containment response if core damage occurs; i.e., core debris cooling, containment heat removal, containment pressure control, fission product removal, and containment isolation.

The specific plant response to each initiating event is modeled by adjusting the general sequence framework to account for the unique impact of the initiator on each system, operator action, and function. Thus, the ESD can be viewed as the parent for the large family of detailed plant response event trees. Individual sequences in the Level 1 model can be traced from beginning to end through the event sequence diagrams.

3.1.2.1.3 Tracing Sequences through the Watts Bar ESD

Emergency Operating Procedure E-0 is entered if the conditions exist for a reactor trip or safety injection signal. The expected sequence of actions is shown across the top of the diagram. The normal plant response following a reactor trip without a safety injection signal is for the reactor and turbine to trip and for power to be available to the shutdown boards from offsite. With everything successful and no safety injection signal or conditions present, the operators then continue to EOP ES-0.1 (Reactor Trip Response), transfer 2 of the ESD.

For the normal plant response to a simple plant trip, the main feedwater (MFW) regulating valves close when reactor coolant system (RCS) T_{AVG} drops below the required setpoint. The MFW pumps are also tripped on this feedwater isolation signal. Auxiliary feedwater (AFW) then starts automatically, either on low-low steam generator level or on trip of both MFW pumps. Steam generator pressure control is achieved using the condenser dumps in the steam pressure mode. Reactor coolant pump (RCP) seal cooling or injection is maintained to protect the seals, and RCP motor bearing cooling is available so that the RCPs can continue running. Typically, MFW would be restored in anticipation of plant restart. The operators would then follow procedure GOI-2 (Plant Startup from Hot Standby to Minimum Load).

Selected accident sequences are now described with reference to the Watts-Bar specific ESD presented as Figure 3.1.2-2. This should provide the reader with a general feel for the response of Watts Bar systems to selected plant trips, and especially for the role of the operators following plant trips. In the next section, some additional explanation for each sheet of the ESD is presented, including a discussion of what was finally selected for inclusion in the event trees, what was not, and why.

3.1.2.1.3.1 Example Sequence — Station Blackout. The first accident sequence discussed is that of a station blackout. The sequence begins with a loss of offsite power. The loss of offsite power results in the removal of control rod holding power, which allows the control rods to fall into the reactor core. Reactor trip is very likely to be successful for this sequence since the reactor trip breakers need not open for success. The ESD notes that with success of reactor trip, the rod bottom lights should be lit and neutron flux should be decreasing. While the conditions for a turbine trip will certainly be present, the actual turbine trip may or may not occur. For the sequence being discussed, turbine trip is assumed to be successful. The reader should therefore follow across to the right from the rectangle describing "turbine trip," which is the success path. All of the turbine stop valves are closed. Early in E-0, the operators are asked to verify that at least one complete train of shutdown boards is energized. If not, then E-0 transfers the operators to ECA-0.0 (transfer 14 of the ESD).

The operators then follow the steps in ECA-0.0, per transfer 14 of the ESD. The ESD notes that once in this procedure, the functional response guidelines are not to be implemented. ECA-0.0 indicates that the operators are to verify that RCS is isolated and that the turbine-driven AFW pump is operating. At Watts Bar, local manual control of the turbine-driven AFW pump level control valves is required under station blackout conditions. For this sequence, it is assumed that AFW operates successfully, and that the RCS is isolated. ECA-0.0 then directs the operators to try to restore electric power to the shutdown boards from either the diesel generators or from offsite sources. Offsite power is assumed to be lost for an extended time in this sequence, however, so that the operators do not restore power and return to E-0 but, instead, continue with ECA-0.0.

The subsequent actions in ECA-0.0 require the operators to place pumps in pull-to-lock in anticipation of restoring AC power. Also, the operators are to isolate the seal return line locally to avoid a containment bypass path developing by virtue of RCP seal damage.

Additional actions are called out via transfer 14 of the ESD. Secondary-side radiation is assumed to be normal because there is no steam generator tube rupture in this sequence.

The next key action has to do with the availability of DC power for steam generator level instrumentation. At Watts Bar, the availability of diesel generators on the opposite unit means that power for steam generator level instrumentation will still be available even for an extended station blackout on the unit analyzed. In this sequence, the diesel generators of the opposite unit are assumed to operate so that steam generator level instrumentation is available indefinitely. The operators are directed to ensure that adequate makeup for continued heat removal using AFW is available. If no makeup is provided, the eventual loss of AFW could limit the time for electric power recovery.

The operators are then directed to depressurize the steam generators to cool down and depressurize the RCS, thereby minimizing RCP seal leakage. Successful depressurization would likely result in accumulator injection. Successful depressurization with accumulator injection lengthens the time for electric power recovery, although eventual RCP seal damage and increased leakage are expected if seal cooling is not restored. In this sequence, electric power is not restored in time, the RCP seals are damaged by excessive RCS temperatures, and core damage results from failure to replace inventory lost via the seal LOCA. The ESD then sends the reader to transfer 38 (Containment and Plant Response to Core Damage).

Transfer 38 addresses all of the questions of interest for establishing the containment response and radiological release source term for the core damage sequence. Several of the actions noted may have already been addressed by the EOPs prior to core uncovery. Since none of the EOPs were written specifically for core damage sequences, there are no references to steps in the EOPs. In the future, accident management guidance will be documented in this portion of the ESD.

3.1.2.1.3.2 Example Sequence — Small LOCA. The next sequence traced through the ESD is that of a small LOCA, with failure of all recirculation from the containment sump. The sequence begins on the first sheet of the ESD, which represents EOP E-0. Automatic reactor trip and turbine trip are both successful. Both offsite power and power to the shutdown boards are available.

The operators are asked to verify that safety injection is not activated and not required. For the small LOCA sequence discussed, it is assumed that pressurizer pressure drops below the required value and that a safety injection signal is generated by the solid state protection system (SSPS). Reactor trip will have occurred earlier when the reactor coolant pressure falls below the required value. Steam line pressure remains normal so that the main steam isolation valves (MSIV) initially remain open. However, the small LOCA causes continued loss of RCS inventory to containment and pressure rises. Eventually, high and high-high containment pressure setpoints are exceeded, and the MSIVs are then signaled to close by the main steam isolation signal. The operators are then directed to verify that the centrifugal charging pumps (CCP), safety injection pumps (SIP), and RHR pumps are running and injecting to control RCS inventory. Emergency core cooling system (ECCS) flow is assumed to be successful in this example sequence. Containment isolation is also successful. The safety injection signal causes feedwater isolation, trips the MFW pumps, and sends another turbine trip signal. AFW is verified to be operational. The condenser dumps are assumed to be unavailable because the MSIVs were closed due to high-high containment pressure. Secondary steam relief is provided by the steam generator power-operated relief valves (PORV).

The operators are then directed to verify that CCS, essential raw cooling water (ERCW), emergency gas treatment system (EGTS), and auxiliary building gas treatment system (ABGTS) are running. The high-high containment pressure condition causes a Phase B signal. The MSIVs are closed. The operators must then stop the RCPs in response to the Phase B signal and ensure the containment spray pumps are running. They also ensure that the air return fans are running after 10 minutes. The steam generator relief valves are assumed to respond normally to this sequence; e.g., none of the steam generators are faulted. RCS pressure remains above 180 psig so that one RHR pump is stopped. As RCS pressure falls, the pressurizer PORV is not challenged to relieve pressure. Auxiliary building radiation is normal because the break flow is confined to inside containment. Secondary radiation is normal because there is no steam generator tube rupture. The ESD directs the reader to E-1 on transfer 5 due to abnormal containment conditions caused by the LOCA.

Once transition to E-1 has occurred, the operators are directed by an orange path condition into FR-Z.1 (transfer 32). One step beyond those already covered in E-0 is that the hydrogen igniters are to be energized. After completing FR-Z.1, the operators are directed back to the instruction in effect. The ESD assumes that the procedure in effect is E-1, on transfer 5 of the ESD.

The operators reset the Phase B signal and stop the containment spray pumps once containment pressure falls below the Phase B setpoint. This is a key step, as stopping the spray pumps preserves refueling water storage tank (RWST) inventory, delaying the time for switchover to the containment sump. The operators must also stop the remaining RHR pump, which is operating in miniflow to prevent it from overheating. E-1 directs the operators to refill the condensate storage tank (CST) for continued heat removal via AFW.

The safety injection termination criteria are not satisfied for this sequence because of the small LOCA. The ECCS pumps continue operating to control RCS inventory. The operators are directed to energize the hydrogen igniters for hydrogen control after considering hydrogen concentration. RCS pressure remains high so the ESD transfers the reader to ES-1.1 (Post-LOCA Cooldown), as presented in transfer 6.

In this sequence, power is available to the RHR pumps. The operators are directed to reset the safety injection, the Phase A, and the Phase B signals if they have not already done so, and to reestablish control air to the containment. The RHR pumps are stopped. RCS cooldown and depressurization are then initiated. These actions are to first restore pressurizer level, and then are carried out while maintaining pressurizer level within the required range. An RCP is not restarted due to the adverse containment condition. ECCS flow is reduced, and the operators try to establish normal charging while maintaining pressurizer level and subcooling by turning off the SIPs and one CCP.

For this example of ESD use, the LOCA is assumed to be large enough to empty the RWST before closed-loop RHR cooldown mode entry conditions are satisfied, even if containment spray flow is terminated. RWST level drops to less than the low level setpoint, leading to automatic switch to sump recirculation. For this case, the ESD transfers the reader to ES-1.2, transfer 7.

Transfer 7 presents the actions for successful switchover for sump recirculation. For the sequence being discussed, both automatic and manual switchovers fail before the RWST empties. Attempts to align the containment sump valves locally also fail. The ESD then transfers the reader to ECA-1.1 (Loss of RHR Sump Recirculation), presented in transfer 17. RCS pressure is still assumed to be high enough to preclude low pressure RHR injection.

Per ECA-1.1, the operators then provide borated water makeup to the RWST for continued high pressure makeup to the RCS. In this sequence, makeup is provided from the spent fuel pit, or containment spray pumps recirculation flow. Consideration about whether the ECCS flow is sufficient for inventory control must be asked. Ample flow is judged to be available for this sequence. The plant is stable and trending toward cold shutdown.

3.1.2.1.3.3 Example Sequence – Anticipated Transient Without Scram. A third example sequence is that of a loss of all MFW from full reactor power, followed by a failure of reactor trip and insufficient RCS pressure relief. As with the other sequences, the conditions for a plant trip are annunciated in the control room. This is shown as the entry condition to E-0, on the first sheet of the ESD. Both automatic and manual reactor trip fail. Neutron flux is not decreasing, and the power remains above 5%, so the operators are directed to FR-S.1 [Response to Nuclear Power Generation/Anticipated Transient without Scram (ATWS)], as presented in transfer 22 of the ESD.

Manipulation of the panel switches, per FR-S.1, also fails to achieve reactor trip. The loss of MFW places a severe transient on the plant. ATWS mitigation safeguards actuation system (AMSAC) detects the trip of both MFW pumps without reactor trip. The turbine trips as a result of the AMSAC signal. The high initial power level (greater than 40%) places a severe challenge to the pressurizer relief and safety valves for them to limit RCS pressure to less than 3,200 psig. The 3,200 psig is a realistic failure limit, as discussed in References 3.1.7.1 and 3.1.2-2. Although the reactor trip breakers did not open, manual rod insertion is successfully initiated by the operators within the first minutes. AFW successfully starts. Insufficient flow capacity exists from the pressurizer PORVs and safety valves to limit RCS pressure. The increased RCS pressure poses a challenge to the steam generator tubes, but they remain intact. The reactor vessel, however, fails when RCS pressure exceeds 3,200 psig. The loss of vessel integrity is assumed to result in core damage with the RCS pressure low.

3.1.2.1.4 Use of ESD in Event Tree Development

The following paragraphs describe each sheet of the ESD as they interpret the Watts Bar EOPs for application to the PRA model. Each sheet interprets a specific EOP. A cross-reference between the ESD sheets and the EOPs is given in Table 3.1.2-1.

For each sheet of the ESD, a discussion is provided about how each sheet is entered from other places in the ESD. Key operator actions and the objectives of each procedure are summarized. The impacts of all actions in the procedure being either successful or failed are then noted. The reasons for not modeling selected portions of the ESD in the event tree models are then provided. Some procedures are entered via a yellow path on the status tree. While it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, the actions in this procedure may not be included in the IPE model top events.

The particular events that appear in the ESD and are modeled in the associated event trees are identified by the top event designations placed just outside the lower right-hand corner of each rectangle. Some rectangles are designated by "IE," which stands for initiating event. Such events are considered when identifying the impacts of each initiator. For example, individual causes for the occurrence of an inadvertent safety injection condition as the cause of a plant trip are not modeled explicitly. Instead, such causes are modeled implicitly by defining the initiating event categories appropriately to cover each situation. As is standard practice, the frequencies of these initiating event categories are then derived directly from industry experience and plant-specific data.

In some cases, the number of top events related to a specific system function is rather large. In such cases, an abbreviated identifier is used to represent the associated top events, as described below.

- The identifier AFW is used to describe all of the top events that are used to model the AFW system. The top events used to model AFW include CT, MA, MB, TP, and AF.

- The identifier ECCS is used to describe the emergency core cooling systems, centrifugal charging pumps, safety injection pumps, and residual heat removal pumps. The top events used to represent these systems include RW, VA, VB, VF, VC, S1, S2, IP, SI, RA, RB, RI, and RF.

- The event tree top events for the interfacing systems LOCA initiator were represented by the identifier VSEQ, to indicate that such actions are represented by top events in the interfacing LOCA, or V-sequence analysis.

The relationship between the Watts Bar Nuclear Plant EOPs and the different pages of the ESD is summarized in Table 3.1.2-1. The event tree top events covered by each page of the ESD (i.e., one procedure per page of the ESD) are then summarized in Table 3.1.2-2. The applicable procedure guidance for each frontline event tree top event is then cross-referenced in Table 3.1.2-3. These latter two tables will be useful references for the presentation of the event trees in subsequent sections. Finally, every block in the ESDs has a reference number at its upper left-hand corner beginning with "IV." This number has been included for use as a reference to assist the reader to find the block. They are listed on the ESDs in ascending order, but may not always be sequential.

- **E-0 — Reactor Trip or Safety Injection Conditions.** The first page of the ESD considers the immediate action steps (1 to 13) in EOP E-0. As can be seen from the ESD, nearly all of the events shown are represented by specific top events in the event trees.

  In the event of a safety injection signal, MFW is isolated and the MFW pumps are tripped. In the event trees, this action is assumed to be successful. If feedwater isolation fails and the pumps continue operation, excessive cooling would result. As pressurized thermal shock (PTS) concerns are not being modeled in the current study, event N14 is assumed to always be successful (References 3.1.2-1 and 3.1.2-2).

The sequence of events resulting from the failure of the condenser steam dumps, event N16B, is not modeled explicitly. If the condenser steam dumps stick open, the excessive cooling would likely result in a signal to close the MSIVs. This sequence of events is not modeled explicitly; instead, the condenser is simply assumed to be unavailable. The redundancy afforded by the steam generator PORVs and the many steam generator safety valves for steam relief led to a modeling simplification. As long as feedwater is available from AFW or the MFW pumps, credit for only the steam generator safety valves is modeled for adequate steam relief. The probability of all five safety valves on a single steam generator failing to open is negligible. Therefore, event N32 is not considered further in the event trees. Consideration of the availability of steam generator PORVs for depressurization of the steam generators is considered but in a separate top event, DS, specifically for that purpose.

The rest of the actions on the first sheet of the ESD are modeled in the associated event trees. The remainder of the actions in E-0 are displayed in transfer 1 of the ESD.

- **E-0 Continued — Reactor Trip or Safety Injection Conditions.** Transfer 1 of the ESD is considered when a safety injection has occurred, MFW is successfully isolated, and RCS temperature decrease is controlled. Consideration is then given to isolation of the LOCA, if possible, and to establishing the nature of the event (LOCA, SGTR, or steam line or feed line break).

  In the event of a safety injection condition in which RCS pressure remains high enough to preclude injection from the RHR pumps, the operators are instructed to either stop the RHR pumps or, if RCS pressure continues to decrease, to align CCS flow to the RHR heat exchangers and let the RHR pumps run. If the operators fail to protect the RHR pumps (events N42B and N42C), then the pumps may overheat while operating on miniflow. The time to overheating while on miniflow is approximately 100 minutes. In the event that a Phase B condition also occurs and spray actuates successfully, switchover to recirculation would occur before the pumps overheat. This possibility is neglected in the ESD.

- **ES-0.1 — Reactor Trip Response.** The actions shown in transfer 2 of the ESD are considered when a plant trip has occurred, but there is no safety injection condition. Event N68, which considers that isolation of MFW is assumed to be successful, and events N74, N69, N70, and N70A are not considered in the associated event trees.

  As with the actions considered in E-0, if the condenser dumps are unavailable, the steam generator safety valves have sufficient redundancy to make the failure probability of steam relief negligible. Therefore, event N67C, which models the steam generator PORVs, is not considered in the associated event trees.

  All of the other events represented in transfer 2 of the ESD are included in the associated event tree models. If the plant is stable at hot shutdown without a

LOCA, the actions for natural circulation cooldown, startup, or for plant shutdown from minimum load are not modeled.

- **ES-0.2 — Safety Injection Termination Criteria Satisfied.** Transfer 3 of the ESD is entered when there has been a plant trip with a safety injection signal, but the conditions for terminating safety injection have been achieved. The key actions represented by this portion of the ESD are to ensure that safety injection is terminated before the RCS is repressurized sufficiently to lift a pressurizer PORV, and to see to it that no other source of LOCA develops.

  Events N80 and N81 are not considered in the associated event trees. ECCS is already assumed to be required by the specific event tree path before this procedure is entered; e.g., event N22 on transfer 1. Given that safety injection is reset and that no LOCA has yet developed, the operators would normally reset Phase B and reestablish air to containment to align for normal charging and letdown. No significant consequence of failing these actions has been identified.

- **ES-0.3 — Natural Circulation Cooldown.** Transfer 4 of the ESD covers the actions directed by procedure ES-0.3, Natural Circulation Cooldown. This procedure is entered when the RCPs are off and either there has been no safety injection signal or the safety injection signal has been terminated. If a safety injection occurs while in this procedure, the operators are directed back to E-0. The key actions in this procedure are to ensure continued secondary heat removal and RCP seal cooling.

  Most of the actions directed by this procedure are not explicitly modeled in the associated event trees. While in this procedure, the plant is already stabilized at hot shutdown, without a LOCA. This procedure directs the actions for a slow cooldown and depressurization to a cold shutdown condition. For the IPE models, stable at hot standby is considered to be a successful end state. Therefore, only those actions that are necessary to ensure long-term secondary heat removal (event N118) and to ensure seal cooling (event N128) are modeled. For long-term secondary heat removal, the IPE models assume that the CST is normally kept at an inventory in excess of 300,000 gallons, based on Sequoyah operating data. Assuming that Watts Bar will maintain a similar amount of water, there is adequate CST inventory to last 24 hours, except for ATWS sequences, where greater AFW flow is required initially. Therefore, event N118 is only modeled following recovery from an ATWS sequence.

- **E-1 — Loss of Reactor or Secondary Coolant.** Transfer 5 describes the actions in procedure E-1. This procedure is entered from E-0, E-2, ES-0.2, FR-I.2, ECA-0.2, ECA-1.2, and ECA-2.1 when it has been determined that a safety injection condition has occurred; e.g., a stuck-open pressurizer PORV train. By the time that this part of the procedure has been reached, the operators have already verified that the ECCS has functioned properly. Most of the actions depicted in transfer 5 of the ESD are modeled in the associated event trees. Many of the actions are largely determined by the specific initiating event for which the plant response is being evaluated. Those steps that are not modeled are discussed below.

  Event N157, to open the ice condenser air handling unit circuit breakers, is not modeled. Failure to open these circuit breakers just means that there is another ignition source available for hydrogen combustion. However, in the very next step, the operators are instructed to energize the hydrogen igniters. Therefore, the performance of N157 did not seem significant enough to include in the event tree models. Also, the air return fans should be running for all LOCAs large enough to be modeled in the IPE. Therefore, there are ample sources for hydrogen ignition even without the air handling units operating.

Event N165 is also not modeled in the event trees. With successful switchover to RHR sump recirculation during a LOCA, the need to depressurize the intact steam generators is simply a precautionary measure to protect against potential damage resulting from the large secondary to primary pressure drop. The chance of failing a steam generator under such conditions was judged to be small. In the Level 1 event tree models, induced steam generator tube failures resulting from excessive primary to secondary pressure drops are only modeled for ATWS sequences and steam line breaks. The Level 2 analysis considers the potential for temperature-induced failures after core uncovery.

All other events depicted in transfer 5 are considered in the associated event trees.

- **ES-1.1 — Post-LOCA Cooldown.** Transfer 6 considers the actions directed by procedure ES-1.1. This procedure is entered when a small or medium LOCA has developed (e.g., a small enough break that the operators complete the actions through Step 23 of E-1) and RCS pressure is still above 180 psig. The key actions in this procedure are to reset the safety injection signals, and to cool down and depressurize the RCS sufficiently to go on closed-loop RHR by the time that the RWST level reaches the low-level setpoint. By accomplishing these actions, the need for RHR sump recirculation may be avoided, depending on the break size and the response of the containment spray system.

  Most of the events represented in transfer 6 are included in the associated event trees.

  Event N183D (stop one CCP and establish normal charging) is not modeled explicitly in the event trees. Failure to reduce ECCS flow would likely preclude successful cooldown and depressurization before low level is reached in the RWST. Even with ECCS flow reduced, recirculation from the sump is only assumed avoided if the LOCA is very small. In the current plant model, all LOCA sizes are assumed to result in high enough containment pressure to actuate containment spray.

  Therefore, the failure to perform the action to reduce ECCS flow is judged to have only minimal chance of requiring the need for sump switchover when it otherwise would not be required. ECCS flow termination is considered in other parts of the ESD to prevent a challenge to the pressurizer PORVs and to successfully mitigate a steam generator tube rupture. Normal letdown and charging are considered in the mitigation of steam generator tube ruptures.

  Event N173 (isolate the accumulators) is not modeled in the event trees. Failure to isolate the accumulators would likely hold up RCS pressure for a while longer but would probably not be sufficient to prevent a successful RCS depressurization by itself. This is the assumption made in the event trees.

- **ES-1.2 — Transfer to Containment Sump.** Transfer 7 considers the actions directed by procedure ES-1.2. This procedure is entered when there is a LOCA, and safety injection has depleted the RWST to less than the low-level setpoint. The key actions in this procedure are to verify the automatic switchover actions, and to manually complete the crosstie from RHR discharge to the high pressure injection pumps.

  Nearly all of the events in transfer 7 are modeled in the associated event trees. Two exceptions are discussed below.

  Event N198C (stop the corresponding train of RHR pumps when the associated sump valve fails to open) is not modeled because these pumps would be rendered ineffective anyway. The action to open the failed sump valve locally is not included in the base event tree models but is considered as a possible operator recover action.

- **ES-1.3 — Transfer to Hot Leg Recirculation.** Transfer 8 considers the actions directed by procedure ES-1.3. This procedure is entered from ES-1.2, about 15 hours after switchover to the containment sump is completed. The key action in this procedure is to redirect the injection flow from the cold legs to the hot legs. This involves the realignment of one train of RHR injection flow to the hot legs. If train A cannot be realigned, then the operators are instructed to align train B.

  Failure to complete the alignment to hot leg recirculation is conservatively assumed to result in boron precipitation and eventual core flow blockage, and therefore core damage for large and medium LOCAs. For large breaks, the problem of boron precipitation results as the RCS boils off, leaving the precipitate behind. This is assumed not to be the case for the smaller breaks, where cooling is provided solely by heating the RCS fluid without boiling; i.e., the RCS is kept full of water and subcooled.

  Event N213 (safety injection pumps switched to hot leg recirculation) is not modeled in the associated event trees. While the switchover of safety injection flow is also directed by procedures, flow from the RHR via the hot legs is assumed to be sufficient to prevent core flow blockage. Therefore, the switchover of the safety injection pumps is assumed not to be required.

  All of the other events represented by transfer 8 are modeled in the event trees.

- **E-2 — Faulted Steam Generator Isolation.** Transfer 9 of the ESD covers the actions directed by procedure E-2. This procedure is entered when steam generator pressure is low or decreasing uncontrollably, indicating that there is a break or stuck-open valve on the secondary side. The key actions in this procedure are for the MSIVs to close on low steam line pressure coincident with high steam flow and for the operators to stop both AFW and MFW to the affected steam generator(s). If all four steam generators are affected (e.g., turbine trip failure with the MSIVs also failing to close), the operators are transferred to procedure ECA-2.1, whose actions are modeled in transfer 19. Some of the actions shown in transfer 9 are not modeled in the event trees. These events are discussed below.

If successful automatic closure of the MSIVs on low steam line pressure does not isolate the break, the operators are instructed to identify and isolate the faulted steam generator(s) by manually closing the MSIVs, steam generator PORV, and blowdown line on the faulted steam generator(s), or by closing all of the intact steam generator's MSIVs, thereby bottling up the common secondary side (events N216 and N216C). These actions are not modeled explicitly. However, the actions to isolate feedwater to the affected steam generators (events N219, N219A, and N219B) are assumed to be successful, as this limits the number of steam generators available for secondary heat removal. Failure to consider the manual actions explicitly to isolate the break means that, in the event of a steam generator tube rupture, the chances of a release path through the secondary are overstated.

- **E-3 — Steam Generator Tube Rupture.** Transfer 10 of the ESD follows the actions in procedure E-3 for steam generator tube ruptures. This procedure is entered from E-0, E-1, ES-1.1, E-2, ES-3.2, ES-3.3, and FR-H.3 after the operators have recognized that a safety injection condition exists, verified that the ECCS systems are operating, that AFW is available, and that secondary radiation is abnormal (see transfer 1 of the ESD). The key actions covered in this portion of the ESD are to identify which steam generator is ruptured, and to take the necessary steps to isolate it and to cool down and depressurize the RCS so that the relief and safety valves on the ruptured steam generator are not challenged unnecessarily.

  Successful completion of these actions is assumed to lead to the conditions for safety injection termination. Failure to complete these actions is modeled as a release path through a stuck-open valve on the ruptured steam generator to the environment. This leads to the conditions for subcooled recovery, as directed by ECA-3.1 and displayed in transfer 20 of the ESD.

  Most of the events in transfer 10 of the ESD are explicitly modeled. The exceptions are noted here. The turbine-driven AFW pump requires a steam supply from one of two steam generators. If the ruptured steam generator is one of the supplies, and steam from the other generator is not available, the IPE models assume that the turbine-driven AFW pump is unavailable. Procedure E-3 permits the turbine-driven pump to be supplied from the ruptured steam generator if neither motor-driven AFW pump is available, but this is conservatively not modeled.

  If the MSIV or the MSIV bypass valve on the ruptured steam generator is not isolated, the operators are instructed to close the MSIVs, MSIV bypass valves, and other valves on the intact steam generators to isolate the secondary side on the ruptured steam generator. This alternative way to isolate the steam side of the ruptured steam generator is conservatively neglected to simplify the IPE models.

  Following successful cooldown of the RCS (event N237), the operators must then depressurize the RCS using one of three methods. The first two methods represented by events N238 and N238A are included in the IPE models. However, the third priority option, to use auxiliary spray as represented by event N238B, is omitted. This omission is a modeling simplification. It is not expected to be significant because of the options already modeled and the common dependence of all three options on the operator.

Once depressurization is completed, the IPE models consider the event for failure to reclose the pressurizer PORV because failure to do so would cause a LOCA. A similar event for stopping pressurizer spray is not modeled because failure to do so does not lead to a LOCA.

- **ES-3.1 — Safety Injection Termination Following Steam Generator Tube Rupture.** Transfer 11 of the ESD follows the actions in ES-3.1. This procedure is entered from E-3 as indicated in transfer 10, when the ruptured steam generator has been successfully isolated and the RCS depressurized. The key actions in this transfer are to reset the safety injection condition and to reduce ECCS flow before a subsequent steam generator PORV challenge on the ruptured steam generator, while maintaining adequate RCP seal cooling and continuing to depressurize the RCS. Successful completion of these actions leads to the post-steam generator tube rupture cooldown procedures, as represented by transfers 12 and 13. Failure of the actions in transfer 11 means that a loss of RCS coolant continues from a stuck-open pressurizer PORV, an RCP seal leak, or via a release path through the secondary side. With RCS leakage in progress, despite the successful RCS cooldown and depressurization, the operators are directed to ECA-3.1 as represented in transfer 20 of the ESD.

  Selected actions displayed in transfer 11 are not modeled explicitly in the IPE. Event N249 to reestablish normal charging and to isolate the BIT is not modeled explicitly; however, RCP seal injection is modeled.

  Event N252 to reestablish letdown is modeled. Failure to reestablish letdown would complicate pressure control. The model currently assumes that this action is required to cool down and depressurize the RCS sufficiently for closed-loop RHR cooling.

  Event N258 directs the operators to minimize secondary system contamination. Such actions are desirable but not necessary to prevent fuel damage or to determine the extent of release, should fuel damage occur. This action is to simply close up the turbine building. Such actions could be significant for considering the degree of release for Level 2 if the only release is via the turbine building. However, the most likely release path is via a stuck-open or leaking steam generator relief valve for which the release path does not go through the turbine building. Therefore, this event was not modeled.

  Event N260A use of auxiliary spray to depressurize the RCS, is not modeled in the IPE. This is consistent with the treatment discussed earlier for transfer 10. Event N268 is a procedural switch for the operators to determine which long-term procedure to use if the actions in ES-3.1 are completed successfully.

- **ES-3.2 — Post-SGTR Cooldown Using Backfill.** Transfer 12 of the ESD models the actions in ES-3.2 for poststeam generator tube rupture cooldown using backfill. This procedure is entered following successful completion of the actions in ES-3.1, as represented by transfer 11 of the ESD. The key actions in this procedure are to continue the RCS depressurization, establish RHR entry conditions, and then align for closed-loop RHR cooling.

Successful completion of the actions in transfer 12 results in the reactor being stable at cold shutdown. Failure to depressurize the RCS successfully is assumed to lead to a loss of RCS inventory and therefore transfers to 10.

Selected events in transfer 12 are not modeled in the IPE. Event N273 (borate for shutdown margin) is not modeled. It is assumed that recriticality would not occur with the borated water from the RWST injected. Event N275C to restore AFW to the ruptured steam generator is not modeled explicitly. With the large amount of time available for backfill operations, this action is assumed to be successful or, rather, its failure probability is neglected relative to the other failures to complete the RCS cooldown and depressurization. Also, as described earlier for transfer 10, the third option for RCS depressurization, using auxiliary spray, is conservatively omitted from the model.

- **ES-3.3 — Post-SGTR Cooldown by Ruptured Steam Generator Depressurization.** Transfer 13 of the ESD represents the actions in procedure ES-3.3 for post-SGTR cooldown using depressurization via the ruptured steam generator. This procedure is entered from ES-3.1 when safety injection is already terminated following a steam generator tube rupture only if instruction ES-3.2 is inadequate. Therefore, none of the events in this procedure are modeled in the IPE. Transfer 12 actions are used instead.

- **ECA-0.0 — Loss of Shutdown Power.** Transfer 14 is entered from E-0 when there has been a loss of power to both trains of shutdown boards (see the first page of the ESD). The functional response guidelines should not be implemented when this procedure is in effect. The key actions in this procedure are to ensure that secondary heat removal is provided by the turbine-driven AFW pump, depressurize the steam generators to reduce RCS pressure, and attempt to recover electric power from either offsite or the diesel generators. Of particular interest are the actions to maintain long-term DC power. Load shedding is called out in event N331. More importantly, if the diesel generators of the opposite unit are available, power for the instrumentation needed for steam generator level measurement would be maintained. These considerations are addressed in the electric power recovery analysis.

  Successful completion of the actions in this procedure results in a recovery of power prior to core damage. After successful restoration of power, the operators are transferred to either the instruction in effect, ECA-0.1, or to ECA-0.2 for actions to be performed after power is restored, depending on whether a safety injection has occurred; e.g., whether the accident has progressed to the point of RCP seal damage.

  Some of the actions represented by transfer 14 are not modeled explicitly in the IPE. Event N323 (place pump switches in pull-to-lock) is not modeled. If the CCPs are not put in pull-to-lock, the RCP seals may be shocked by cold water once power is recovered. However, the consequences of this action are not so important once flow from the CCPs is established, and the likelihood of failing to perform this step is low, especially relative to the chances of recovering power before core damage. Therefore, the failure to perform this action is omitted from the IPE models.

Event N328 considers the possibility of one or more steam generator valves failing open causing a steam generator to depressurize. This action is not modeled. Should it occur, secondary heat removal would continue. The operators would be directed to isolate the faulted steam generator, but not if all other steam generators are unavailable. If the generator is not isolated, the rate of feedwater flow required would increase, but the increased flow is judged to not be large enough to deplete the CST at a significantly higher rate. Therefore, the time before makeup to the CST is required is still substantial. Therefore, this event was judged to have a negligible influence on the availability of AFW.

Event N334 considers the accumulators. If the steam generators are successfully depressurized, the accumulators should inject, providing additional inventory to delay the time to core uncovery due to RCP seal leakage. In computing the core uncovery time due to RCP seal leakage, the inventory from the accumulators is credited, provided that the steam generators are depressurized. The chances of the accumulators not injecting are judged to be small enough that failure of this event can be neglected.

Event N337 considers additional actions to isolate any containment isolation valves. The specific valves to close are not explicitly defined. Therefore, this action was not modeled in the IPE. The local action to isolate the seal return line is modeled per event N325.

- **ECA-0.1 — Loss of Shutdown Power Recovery without Safety Injection Required.** Transfer 15 represents the actions in ECA-0.1, following recovery of power without a safety injection signal required. This procedure is entered from ECA-0.0 once power is recovered to the shutdown boards. The events in this portion of the ESD are modeled in the IPE if power is recovered by 1 hour. For power recovery after 1 hour, RCP seal damage is assumed to have taken place, resulting in a small LOCA with safety injection always required. The actions for this interpretation are displayed in transfer 16.

- **ECA-0.2 — Loss of All Shutdown Power Recovery with Safety Injection Required.** Transfer 16 represents the actions to be followed once AC power is restored following an initial loss of all AC power. None of these actions are explicitly modeled in the Level 1 IPE models. Successfully completing these actions would lead to the implementation of procedure E-1, Loss of Reactor or Secondary Coolant. Failure to complete these actions could lead to a loss of heat sink, or inadequate inventory control. The failure of these actions is judged to be negligibly small when compared against the initial failure probability to recover electric power. For this reason, all of the events represented in transfer 16 are also not modeled.

  One of the events in this procedure is, however, of interest for Level 2. Event N382 specifically directs the operators to start containment spray if containment pressure is greater than Phase B pressure. Starting spray may influence the potential for hydrogen combustion by condensing steam in the containment, which functions as a diluent.

- **ECA-1.1 — Loss of RHR Sump Recirculation.** Transfer 17 of the ESD considers the events in procedure ECA-1.1, Loss of RHR Sump Recirculation. This procedure is entered from procedure E-1, ES-1.2, or ECA-1.2 (transfer 7) when there is a LOCA but RHR sump recirculation is unavailable. There are two key strategies in this procedure: to provide makeup to the RWST from other existing borated water sources (e.g., spent fuel pool), or to provide makeup to the RWST by using the containment spray pumps to recirculate water back from the containment sump.

Successful completion of the actions in this procedure implies that RCS inventory control is being maintained despite the LOCA and unavailability of RHR sump recirculation, and the reactor is stable and trending toward cold shutdown. Failure of the actions in this procedure implies that the RWST will eventually empty, resulting in inadequate ECCS flow.

Some of the events represented in transfer 17 are not modeled in the IPE. Event N391 (restoration of the RHR pumps or sump valves to service) is not included in the base IPE models. The local alignment of the containment sump valves is considered as a sequence-specific recovery action, after the initial sequence quantification. Event N395 considers the depressurization of the RCS to minimize break flow and, thus, the makeup requirements. These events are conservatively modeled as required, even though adequate makeup flow is believed to be available without RCS depressurization.

A restriction on the first option for makeup to the RWST is that the action be completed before RWST level drops below the low-low level setpoint. If level drops below 5%, the operators are instructed to stop all pumps taking suction from the RWST and to initiate makeup to the RWST and go on closed-loop RHR, events N404 and N410. This makeup strategy is not modeled in the IPE, except for steam generator tube ruptures, because it is not believed to be sufficient for RCS inventory control for the LOCA sizes of interest (LOCAs large enough to cause a safety injection).

Finally, if makeup to the RWST from existing borated water sources fails, then the strategy to use the containment spray pumps in recirculation from the sump back to the RWST is considered. This spray recirculation option is modeled in the IPE via Top Event MU. However, no credit for this option is given for steam generator tube rupture events. For such sequences, the most likely reason that sump recirculation is unavailable is because the RWST inventory was lost outside containment. This would also preclude recirculation using the spray pumps back to the RWST.

- **ECA-1.2 — LOCA Outside Containment.** Transfer 18 considers the actions in ECA-1.2, LOCA Outside Containment. This procedure is entered from E-0, transfer 18, when auxiliary building radiation monitors indicate abnormal levels. The key action in this procedure is to isolate the break by closing selected valves.

The actions in this procedure are not included in the IPE models. For LOCAs outside containment (e.g., via the RHR cold leg injection or hot leg suction lines), it is expected that the operators would be kicked out of the E-0 procedure into E-1, before reaching the transfer to ECA-1.2 at Step 26. During the start of a LOCA outside containment, the pressure rise in the RHR system is assumed to occur gradually so that the RHR system relief valves direct RCS inventory first into containment. Only after the RHR system pressure increases enough to cause failure of the RHR system boundary would the auxiliary building radiation alarms read abnormally. However, by this time, containment conditions would not be normal, satisfying the condition for transferring to E-1.

- **ECA-2.1 — Uncontrolled Depressurization of All Steam Generators.** Transfer 19 represents the actions in ECA-2.1, Uncontrolled Depressurization of All Steam Generators. This procedure is entered from E-2 when all four steam generators are faulted; e.g., when there is a steam line break downstream of the MSIVs or a turbine trip failure, and the MSIVs fail to close. The key actions represented are to respond to the safety injection condition, terminate safety injection, reestablish normal charging, and proceed to cold shutdown.

  Successful completion of the events in this portion of the ESD implies that the reactor is in cold shutdown. Failure of the events in this procedure implies that a LOCA has developed, further complicating the event sequence.

  Some of the events shown in the ESD are not modeled in the IPE. These are discussed below. Success of event N432 (one steam generator is eventually isolated from the break) sends the reader back to procedure E-2. This event switch is not modeled in the IPE because the failures of the MSIVs to close are assumed to not be recovered. Event N435 considers the control of feedwater to avoid a PTS condition. PTS considerations are judged to be insignificant so that this event was not considered further.

  Events N452, N436, and N468 are also not modeled. If no LOCA develops as a result of the uncontrolled depressurization, then the plant is assumed to be stable at hot shutdown. These events simply direct the operators to bring the plant to cold shutdown. Since these events are not required to avoid fuel damage, they are not modeled in the IPE.

- **ECA-3.1 — SGTR and LOCA - Subcooled Recovery.** Transfer 20 represents the events in procedure ECA-3.1, SGTR and LOCA - Subcooled Recovery. This procedure is entered from E-3, ES-3.1, or ECA-3.3 when either there is a steam generator tube rupture that either cannot be isolated on the secondary side or an additional LOCA has occurred. The key actions in this procedure are to reset the safety injection conditions, and to depressurize the RCS while maintaining injection to mitigate the LOCA.

  Successful completion of these actions implies that the reactor is brought to a stable state on recirculation from the sump or stable at cold shutdown on closed-loop RHR. Failure to complete these actions implies that too much water is being lost from the RWST without a corresponding increase in containment sump level, so that the faster depressurization method, ECA-3.2, saturated recovery, should be implemented.

  Event N472 is not modeled in the IPE. Generally, this is not really an omission because a Phase B condition is not expected during a steam generator tube rupture. If an additional LOCA develops, a Phase B condition may occur, however. Event N472 considers the restoration of compressed air to containment for normal pressurizer spray and letdown. Failure to restore compressed air then just complicates subsequent RCS pressure control. Based on the limited impact of failing to restore air to containment, this event is not modeled in the IPE.

Event N478 considers the potential for a faulted steam generator due to a stuck-open valve on the secondary. Procedure E-2 then directs the operators to isolate the faulted steam generator if it is not the only one available as a secondary heat sink. If the faulted steam generator is the only one available, the operators are cautioned not to isolate it. Secondary heat sink can be maintained by feeding the faulted steam generator. Therefore, event N478 is not modeled in the IPE.

Event N489D is also not modeled in the IPE. Auxiliary spray is only the third option for accomplishing RCS depressurization, given that successful cooldown has occurred. This success path is conservatively neglected. This omission is not believed to be significant because all three options require operator action, and the failure of the operators to initiate the action is expected to govern the depressurization function, even without considering this third equipment option.

Event N494 considers the establishment of normal charging. Failure to complete this action would likely require the need for sump recirculation. In the IPE sequence models, all LOCAs (e.g., stuck-open PORV or RCP seal leaks) are assumed to require sump recirculation anyway, ECCS injection cannot be stopped before RWST level drops below the low-level setpoint so that transfer to the containment sump is required.

Event N498 considers the status of the RCPs. If the RCPs are off, natural circulation cool-down is required. The requirement for natural circulation does not significantly impact the plant's ability to mitigate the steam generator tube rupture; rather, it just delays the time it takes to successfully cool down enough to go on closed-loop RHR cooling. Therefore, this event is not considered in the IPE models.

Event N499 considers the establishment of closed-loop RHR cooling. This event is not modeled because the plant is assumed to require recirculation from the containment sump to mitigate the LOCA in progress.

- **ECA-3.2 — SGTR and LOCA - Saturated Recovery.** Transfer 21 represents the actions in ECA-3.2, SGTR and LOCA - Saturated Recovery. This procedure is only entered from ECA-3.1 when RWST level has been lowered and containment sump level is not as high as expected. This procedure is likely to be entered when the secondary side is not successfully isolated so that RCS inventory is lost outside containment. The key actions in this procedure are to reset the safety injection conditions, and to depressurize the RCS while maintaining injection to mitigate the LOCA. The actions governed by this procedure are similar to those in ECA-3.1 except that the operators also are instructed to provide makeup to the RWST.

  Successful completion of these actions implies that the reactor is brought to a stable state on recirculation from the sump or stable at cold shutdown on closed-loop RHR. Failure to complete these actions implies that inventory control cannot be maintained and that inadequate core cooling is imminent.

The events in transfer 21 of the ESD are very similar to those in transfer 20. Therefore, many of the same events in transfer 20 are also not modeled for transfer 21, and for the same reasons. The events not modeled in the IPE include N510E, N513, N521, and N520.

One difference with the events in transfer 20 concerns event N515A. Normal charging and letdown are assumed to be required to depressurize the RCS sufficiently to go on closed-loop RHR during a steam generator tube rupture without secondary-side isolation. Closed-loop RHR cooling is modeled as a success path for this sequence, so the availability of normal charging and letdown is important.

- **ECA-3.3 — SGTR without Pressurizer Pressure Control.** Transfer 40 represents the actions in ECA-3.3, SGTR without Pressurizer Pressure Control. This procedure is entered from E-3 or ES-3.1 when an SGTR is in progress and the pressurizer cannot be depressurized to stop the leak. This procedure is entered when normal spray, a pressurizer PORV, or auxiliary spray is not operable. The key actions in this procedure are to reset the safety injection conditions and to depressurize the RCS while maintaining injection to mitigate the LOCA. The actions governed by this procedure are similar to those in ECA-3.1 except that the operators also are instructed to provide makeup to the RWST.

  This procedure is not modeled in the IPE.

- **FR-S.1 — Nuclear Power Generation/ATWS.** Transfer 22 of the ESD represents the actions in FR-S.1, Nuclear Generation/ATWS. This procedure is entered from E-0, the critical function status tree, or ES-3.3 whenever there has been a demand for reactor trip but the reactor has not been shut down. The key action in this procedure is to add negative reactivity to the core. This is a red path on the status tree.

  Successful completion of the actions in this procedure implies that sufficient negative reactivity has been inserted to shut down the nuclear power generation. Failure to successfully complete the actions in this procedure implies that there will eventually be inadequate core cooling due to the inability to shut down nuclear power generation.

  Most of the events represented in transfer 22 of the ESD are modeled in the IPE. The exceptions are noted below.

  Event N526A is not modeled in the IPE. These actions are manual backup actions to automatic turbine trip. These manual actions are judged to come too late to mitigate the impact of failing to trip the turbine in response to an ATWS on peak RCS pressure. Therefore, if automatic turbine trip fails following a loss of MFW ATWS (e.g., if AMSAC fails to generate an automatic trip signal), then steam generator dryout is assumed.

  Event N574 considers the actions to close off all boron dilution paths. These actions are judged to be insufficient alone to limit nuclear power generation, and the impacts of the dilution could be overcome by the negative reactivity options evaluated, if they were not isolated. Therefore, this event is not considered in the IPE model.

- **FR-S.2 — Loss of Core Shutdown.** Transfer 41 of the ESD represents the actions in FR-S.2, Loss of Core Shutdown. This procedure is entered from the critical function status tree when the source range startup rate is positive or the intermediate range startup rate is more positive than -0.2 decades per minute. The key action in this procedure is to start normal boration to restore core subcriticality.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-C.1 — Inadequate Core Cooling.** Transfer 23 represents the actions considered in FR-C.1, Inadequate Core Cooling. This procedure is entered from the core cooling status tree whenever core exit thermocouples are greater than the required value, subcooling is less than the required value, and RVLIS is less than the required value. This is a red path on the status tree. The purpose of this procedure is to restore core cooling.

  Successful completion of the actions in transfer 23 implies that core cooling is restored. Failure of the actions implies that core damage occurs.

  In the IPE model, core cooling is never assumed to be restored by the actions in this procedure. In the IPE model, this procedure is entered when RCS pressure is high and a small LOCA has occurred. For core cooling to be permanently restored, RCS pressure would have to be reduced sufficiently to permit low pressure injection using the RHR pumps. No analyses have been performed to support the conclusion that RCS pressure could be reduced sufficiently for low pressure injection prior to core damage, given that the operators followed the guidelines in this procedure and that the accident occurred with the plant at 100% power. In other words, it is not expected that RCS pressure could be reduced to less than 180 psig before core damage. Therefore, many of the events in this procedure are not modeled.

  Event N592 is also not modeled in the IPE. Even if the operators open the ice condenser air handling unit circuit breakers, other hydrogen ignition sources in containment are present. Therefore, this event is not modeled since it has little impact on the sequence of events.

- **FR-C.2 — Saturated Core Cooling.** Transfer 24 represents the actions in procedure FR-C.2, Saturated Core Cooling. This procedure is entered from the core cooling status tree when the RCS is at saturation temperature. The key action in this procedure is to restore subcooled core cooling.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-H.1 — Response to Loss of Secondary Heat Sink.** Transfer 25 represents the actions in FR-H.1, Response to Loss of Secondary Heat Sink. This procedure is entered from the heat sink status tree, E-0, or ES-0.1 whenever narrow range steam generator level is less than the required value in all steam generators and total feed flow to the steam generators is less than the required value. This is a red path on the status tree. This instruction is not required when RHR is in shutdown cooling mode or if RCS pressure is below intact steam generator pressure. The purpose of this procedure is to respond to the loss of heat sink by either restoring the heat sink, or initiating feed and bleed cooling. For large or medium LOCAs, the RCS pressure would be less than the intact steam generator's pressure. For these sequences, F-H.1 actions would not be implemented.

  Success of the actions in this tree implies that the heat sink is restored prior to core damage or that feed and bleed cooling is initiated so that transfer to the RHR containment sump is eventually required. Failure to complete the actions in this tree implies that the secondary heat sink is not restored and that feed and bleed cooling is unavailable. Therefore, core damage occurs.

  Event N618 considers the local start of AFW, given failure of automatic AFW actuation. This event is not modeled in the IPE base model.

  Event N620A considers the operator action to stop the RCPs to delay the steam generator dryout time. This event is not required for success of feed and bleed or for successful recovery of MFW. Therefore, it is not modeled separately in the IPE.

  Event N623 considers the depressurization of the steam generators to feed directly with a low pressure water source. This success path requires the same feedwater valves to operate as for restoration of MFW, so that they are not completely redundant to restoration of MFW. Also, the time available for action is limited so that these additional options may not be implemented before the procedures direct the operators to initiate feed and bleed. Therefore, these low pressure heat sink options are conservatively omitted from the IPE models.

  Events N629 and N630 consider the successful restoration of secondary heat sink after initiating feed and bleed cooling. In the IPE models, recovery of the secondary heat sink is not modeled. Therefore, event N629 is always assumed failed, and event N630 is therefore not questioned. This modeling approach assumes that if the secondary heat sink is not restored prior to initiating feed and bleed, it is not likely to be restored before recirculation from the containment sump is required.

- **FR-H.2 — Steam Generator Overpressure.** Transfer 26 models the actions in procedure FR-H.2, Steam Generator Overpressure. The entry for this procedure is satisfied when the heat sink status tree indicates that there is still narrow range level greater than the required value in at least one steam generator, but pressure is greater than the required value in at least one steam generator. This would be the case if all relief and safety valves on at least one isolated steam generator failed to open sufficiently to relieve pressure.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-H.3 — Steam Generator Level High.** Transfer 27 models the actions in procedure FR-H.3, Steam Generator Level High. The entry for this procedure is satisfied when the heat sink status tree indicates that level is greater than the required value in at least one steam generator. This would be the case if a feedwater control valve malfunctioned, diverting too much feed to one or more steam generators.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-H.4 — Loss of Normal Steam Release Capabilities.** Transfer 28 models the actions in procedure FR-H.4, Loss of Normal Steam Release Capabilities. The entry for this procedure is satisfied when the heat sink status tree indicates that pressure is greater than the required value in at least one steam generator. This would be the case if the steam generator PORV on one steam generator failed to open following plant trip.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-H.5 — Steam Generator Low Level.** Transfer 29 models the actions in procedure FR-H.5, Steam Generator Low Level. The entry for this procedure is satisfied when the heat sink status tree indicates that narrow range level is less than the required value in at least one steam generator but not in all. This would be the case if secondary heat sink were lost but steam generator level had not yet been lowered sufficiently to enter FR-H.1.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-P.1 — Pressurized Thermal Shock (PTS).** Transfer 30 considers the actions in FR-P.1, Pressurized Thermal Shock. The entry for this procedure is satisfied by the pressurized thermal shock status tree when the RCS pressure and T_{COLD} temperature points are to the right of limit A in the PTS limits curve. This procedure is a red path or orange path in the status tree.

  This procedure may be entered from a number of plant sequences as called out in the ESD. However, PTS concerns are not modeled in the current IPE because Watts Bar easily satisfies the screening criteria of the U.S. Nuclear Regulatory Commission (NRC) for PTS. Therefore, none of the events in transfer 31 are included in the IPE.

- **FR-P.2 — Cold Overprotection Condition.** Transfer 31 considers the actions in FR-P.2, Cold Overprotection Condition. The entry for this procedure is satisfied by the pressurized thermal shock status tree when T_{COLD} cooldown is greater than the specified limit, or when RCS pressure is greater than the cold overpressure limit and T_{COLD} is greater than a higher limit.

  This procedure is a yellow path in the status tree. This procedure may be entered from a number of plant sequences as called out in the ESD. However, PTS concerns are not modeled in the current IPE because Watts Bar easily satisfies the NRC's screening criteria for PTS. Therefore, none of the events in transfer 31 are included in the IPE.

- **FR-Z.1 — Phase B Containment Pressure.** Transfer 32 considers the actions in FR-Z.1, Phase B Containment Pressure. This procedure directs actions to mitigate a high containment pressure and then returns the operators to the procedure in effect. This procedure is entered from the containment status tree when containment pressure is greater than the required value. The path is red if pressure is greater than the required value, and is orange if pressure is above the high-high containment pressure setpoint but less than the required value for a red path. Typically, this procedure would be entered following a LOCA or high energy line break inside containment.

  The key action in this procedure is to verify that the containment systems respond correctly: the MSIVs close following a Phase B condition, and the operators manually trip the RCPs, and initiate the hydrogen igniters with hydrogen less than the required value. Also, if containment spray is not aligned for sump recirculation, this procedure is the only place that directs the operators to align for RHR spray recirculation. Successful completion of the actions in this procedure implies that containment pressure is limited to an acceptable range. Failure to complete these actions could mean that containment pressure will continue to rise, depending on the sequence being analyzed.

  Event N722 considers the action to open the circuit breakers on the ice condenser air handling units. Disabling these units eliminates chilled air to the ice. The ice chilling capacity is too limited to be of significance during an accident. Disabling these units also reduces the potential for sparks inside containment leading to uncontrolled hydrogen burns. However, there are other potential sources of sparks inside containment; e.g., the air return fans. Therefore, this event is not modeled in the IPE.

- **FR-Z.2 — Containment Flooding.** Transfer 33 considers the actions in FR-Z.2, Containment Flooding. This procedure is entered from the containment status tree if containment sump level is greater than the required value. This condition may be reached if there is a low energy pipe leak into containment for which there may be a concern about sump dilution. This branch of the status tree is an orange path.

  None of the actions in this procedure are explicitly modeled in the IPE. The key actions in this procedure are to identify the unexpected water source and to consider how to stop the leak. The IPE models assume that such actions would not lead the operators to terminate RCS injection in the event of a LOCA, even if the conditions for this instruction occurred. This assumption is believed to be reasonable. During a LOCA, borated water from the RWST and from the ice condensers would be present. Most likely, the control rods would have been inserted. Therefore, it would take substantial unborated water addition to dilute the sump inventory sufficiently to cause concern once the plant was aligned for recirculation. The only unlimited supplies of water to containment are from breaks from the ERCW system or inadvertent actuation of the high pressure fire protection system. However, only small lines penetrate the containment. Therefore, ample time is available for the operators to determine the leak source and isolate it.

- **FR-Z.3 — High Containment Radiation.** Transfer 34 considers the actions in FR-Z.3, High Containment Radiation. This procedure is entered from the containment status tree if containment upper compartment radiation or the lower compartment radiation is greater than specified limits. This condition may be reached if there is an RCS leak into containment.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-I.1 — High Pressurizer Level.** Transfer 35 considers the actions in FR-I.1, High Pressurizer Level. This procedure is entered from the inventory status tree if ECCS is not in service and pressurizer level is greater than the required value. This condition may be reached if there is a failure of pressurizer level control without ECCS in service.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-I.2 — Low Pressurizer Level.** Transfer 36 considers the actions in FR-I.2, Low Pressurizer Level. This procedure is entered from the inventory status tree if ECCS is not in service and pressurizer level is less than the required value. This condition may be reached if there is a failure of pressurizer level control without ECCS in service.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **FR-I.3 — Voids in Reactor Vessel.** Transfer 37 considers the actions in FR-I.3, Voids in the Reactor Vessel. This procedure is entered from the inventory status tree if ECCS is not in service and RVLIS has indications of voids in the reactor vessel.

  This procedure is a yellow path on the status tree. Therefore, while it is recognized by the operating crew, the actions in this procedure would most likely not be followed until all higher priority tasks have been completed. Therefore, none of the events in this procedure are included in the IPE model.

- **Plant and Containment Response after Core Damage.** Transfer 38 considers all of the actions necessary to determine to which plant damage states each core damage sequence should be assigned. For this portion of the ESD, no attempt has been made to list all of the event tree top events corresponding to each event. This is because, in many cases, the status of each event in the transfer is a function of several top events in the Level 1 model; e.g., the status of RCS pressure at the time of core damage.

  This portion of the ESD is intended to just serve as a reminder to the reader that in addition to determining whether core damage occurs, it is also important to identify the status of key parameters that can influence the ability of the containment to mitigate radioactivity release into the environment.

- **AOI-35 — Loss of Offsite Power.** Transfer 39 considers the actions in AOI-35, Loss of Offsite Power. This is the only nonemergency procedure represented by the ESD. This procedure is entered whenever there is a loss of offsite power, but power is available from onsite to at least one of the shutdown boards. This procedure is also called out for implementation in several places of the ESD; e.g., the start of E-0.

  Successful completion of the actions in this portion of the ESD implies that offsite power is restored to the plant and that the operators should continue to take the plant to cold shutdown via GOI-3. If offsite power is not recovered, the plant most likely will remain in a stable state, relying on power from the diesel generators.

  The key actions in this procedure are to verify that the diesels start and load as appropriate and to attempt to recover offsite power. None of the manual equipment actuations are credited in the baseline IPE models. This is just a modeling simplification. Following the initial sequence quantification, these actions are considered on a sequence-specific basis, as recovery actions.

3.1.2.2 Frontline Top Event Descriptions

The Watts Bar ESDs were described in the previous section. Selected events in the ESDs are referenced to top events in the frontline event trees. This section presents the frontline event tree top events for the general transient early and late event trees, the medium LOCA event tree, and the large LOCA event tree. The support system event tree top events are described in Section 3.1.4. The general transient event trees are used to quantify all of the initiating events described in Section 3.1.1 that are not otherwise addressed by an event-specific event tree. The process of tailoring the general transient event trees for quantification of each initiating event is discussed in Section 3.3, Sequence Quantification.

The general transient tree covering the early or injection phase of the plant's response is referred to as the GENTRANS tree and is described in Section 3.1.2.2.1. Figure 3.1.2-3 presents the GENTRANS tree structure. GENTRANS is connected to another frontline event tree, which covers the late or recirculation phase of the plant's response; i.e., the

RECIRC tree. The RECIRC tree top events are presented in Section 3.1.2.2.2. Figure 3.1.2-4 presents the RECIRC tree structure. Together, event trees GENTRANS and RECIRC describe the entire frontline system response to each initiating event that uses them for accident sequence frequency quantification. The event tree top events for the medium LOCA and large LOCA initiating events (i.e., trees MLOCA and LLOCA, respectively) are described in Sections 3.1.2.2.3 and 3.1.2.2.4. Figure 3.1.2-5 presents the medium LOCA event tree structure, and Figure 3.1.2-6 presents the large LOCA event tree structure.

For each top event in the frontline event trees, several topics are covered in the discussion. First, the function evaluated is briefly presented; then, a statement of the success criteria is given. Next, a statement of the top event model boundaries is provided. This paragraph includes a statement of the equipment modeled and whatever operator actions are considered. If the top event must be evaluated conditionally on the status of previous frontline top events because of shared equipment, this is noted.

Another paragraph then describes the conditions when the top event is not questioned in the event tree; i.e., under what conditions there is no branch for this top event. Sequence conditions that would yield guaranteed success or failure states of the top event are also noted. This includes unique dependencies introduced by individual initiators.

Finally, the discussion for each top event concludes with paragraphs describing the impacts on the remainder of the model if the current top event is successful or failed.

3.1.2.2.1 GENTRANS Event Tree (RISKMAN Designator: GTRAN4)

The early response of the Watts Bar Nuclear Plant to transient initiators, small LOCAs, ATWS events, and SGTRs is modeled using the GENTRANS event tree. This event tree is illustrated in Figure 3.1.2-3.

The logical structure of the tree is such that it can be specialized to handle any of the above initiators. The tree includes various success paths that satisfy the major core protection functions; i.e., core reactivity control, coolant inventory control, and core heat removal.

A detailed discussion of the top events in the GENTRANS tree follows. The top events are described in the order in which they appear in the event tree.

- **Top Event RT — Reactor Trip**
  - **Function Evaluated.** This top event models the reactor trip function.
  - **Success Criteria.** This event requires that at least one of two reactor trip breakers open and that not more than one control rod assembly fails to be inserted into the core. Either automatic or manually initiated reactor trip are success paths. If SSPS fails to actuate reactor trip automatically in response to a plant trip condition, Emergency Procedure E-0, Step 1 calls for the operator to trip the reactor manually and then directs the operator to FR-S.1 where Step 1 repeats the call for the manual trip. Based on WCAP-11993, the time available for success of this action is 1 minute.

- **Model Boundaries.** Major pieces of equipment modeled in this top event include the automatic reactor trip signal from SSPS (i.e., the model is dependent on the status of support tree Top Events ZA and ZB), the control rods, the reactor trip breakers and bypass breakers, the undervoltage trip coils, and the shunt trip coils. A backup operator action for manually tripping the reactor in the event of failure of the automatic signal is included. This action is designated DHART1.

- **Conditions when Demanded.** This top event is asked for every sequence that uses this event tree.

  If the initiating event is a reactor trip, this event is then guaranteed successful.

  On loss of offsite power events, an actuation signal is not required due to the loss of control rod holding power. Therefore, the only possible failure for loss of offsite power sequences is mechanical binding of two or more control rod assemblies.

- **Scenario Impact if Successful.** Success of this top event means that reactivity control is established. Subsequent top events required only for ATWS mitigation are no longer of interest.

- **Scenario Impact if Failed.** Failure of this top event means that the following top events in the trees that are required to mitigate ATWS events are of interest. Also, even if the increase in RCS pressure is successfully mitigated, a small LOCA via a stuck-open pressurizer valve is assumed; i.e., recirculation from the sump is eventually required.

- **Top Event MR — Operator Actions for Manual Rod Insertion**

  - **Function Evaluated.** This event models the operator action to insert control rods manually, given failure of reactor trip.

  - **Success Criteria.** FR-S.1 instructs the operators to insert control rods manually if the reactor trip breakers fail to open. Success of this event requires that the operators provide 1 minute of rod cluster control assembly (RCCA) bank insertion at the nominal rate before the RCS pressure peaks; i.e., within about 2 minutes of a postulated loss of main feedwater. Manual rod insertion must therefore be initiated within 1 minute.

  - **Model Boundaries.** This event considers only the operator action to initiate control rod insertion. The action is designated DHAMR1. Failures of system hardware are not modeled. For about 80% of the time, rod control would be left in automatic. No credit is given for automatic rod run in, however; i.e., manual initiation is assumed always required. The procedural guidance also asks the operators to trip the turbine manually (if the turbine stop valves are not closed) and to run back the turbine if the turbine cannot be tripped. The operator is then instructed to close the MSIVs and bypasses if the turbine cannot be run back. No credit is taken to trip the turbine manually, to run back the turbine, or to close the MSIVs/bypasses. These additional actions would all impact the peak pressure experienced during the ATWS. However, no analysis is available to quantify the degree of influence on the peak pressure. Therefore, these additional actions are conservatively omitted. This is consistent with the approach taken in WCAP-11993.

- **Conditions when Demanded.** This event is asked only if Top Event RT is failed, indicating that an ATWS sequence is in progress. If reactor trip fails and offsite power is lost, no credit is given for this action. For such a sequence, the control rods must have mechanically bound to prevent reactor trip. No credit is therefore given for rod run in under such conditions.

- **Scenario Impact if Successful.** Success of Top Event MR relaxes the success criteria for RCS primary relief (i.e., Top Event SR) during ATWS sequences by mitigating the peak RCS pressure. Success of Top Event MR is not redundant to Top Event RT or EB because the total amount of negative reactivity inserted by manual rod run-in, within 10 minutes, is not sufficient to turn over the power level.

- **Scenario Impact if Failed.** Failure of Top Event MR makes the success criteria for RCS primary relief more stringent (i.e., Top Event SR) during ATWS sequences.

• **Top Event TT — Turbine Trip**

- **Function Evaluated.** This event models the turbine trip function.

- **Success Criteria.** Success of turbine trip requires that all four of the steam stop valves or all four of the governor valves close automatically on demand.

- **Model Boundaries.** This event models the BOP hardware required for tripping of the main turbine. The closure signals are not included in this top event. One signal to close comes from the auxiliary contacts on the reactor trip breakers, which goes through the SSPS. This signal is assumed to be working if Top Event RT is successful. An additional trip signal comes from AMSAC, which is modeled in this tree under Top Event AM. No credit is given for manual trip of the turbine trip as a backup prior to a safety injection signal or a signal to isolate the MSIVs. Successful turbine trip requires the success of Top Event TT and at least one of Top Event RT or AM. In the non-ATWS portion of the tree, no credit is taken for Top Event AM.

- **Conditions when Demanded.** Top Event TT is demanded for every sequence that uses the general transient/small LOCA/ATWS/SGTR event tree. If the initiator is a turbine trip, this top event is set to guaranteed success. If the initiator is not a turbine trip, the signal for turbine trip is assumed to be available if the reactor trip breakers have opened or AMSAC is successful.

- **Scenario Impact if Successful.** Tripping the turbine in an ATWS loss of feedwater event causes a rapid reduction in steam flow out of the steam generators, and resultant rapid increase in steam pressure to the steam line safety valve setpoint. Turbine trip extends steam generator inventory and results in an increase in core coolant temperature. The increase in coolant temperature causes a decrease in core power early in the transient before steam generator tubes have begun to uncover. Later, as the steam generator tubes uncover, the rate of increase in RCS pressure is lower because it started at a lower core power level.

For sequences with reactor trip, successful turbine trip limits the steam flow out of the steam generators, precluding overcooling from excessive steaming.

- **Scenario Impact if Failed.** In the portion of the event tree with successful reactor trip, failure to trip the turbine implies that an overcooling event is underway. The overcooling is assumed to lead to a low RCS pressure resulting in a safety injection condition and in the need for main steam line isolation.

  ATWS sequences in which there is no turbine trip, the reactor is initially above 40% power, and main feedwater fails are assumed to proceed directly to core damage, regardless of the response of the pressurizer PORVs or safety valves. Without the turbine trip, the steam generators will continue to boil off their inventory at the same rate as before. After the steam generator tubes are exposed, the heat transfer from the primary to the secondary will decrease dramatically. The resulting temperature rise will result in an RCS pressure above 3,200 psig. For such sequences, the peak RCS pressure is conservatively assumed to be high enough to cause a break in the RCS pressure boundary that cannot be mitigated.

  For ATWS sequences without turbine trip but the initial power level is less than 40%, the peak RCS pressure is less than 3,200 psig. Therefore, for such sequences, the RCS remains intact, but the pressurizer relief valves are challenged and must reclose after opening. Under these circumstances, a small LOCA due to the failure of a PORV or safety valve to close is conservatively assumed to result.

- **Top Event PL — Reactor Power Less Than Forty Percent**

  - **Function Evaluated.** This event is used to determine whether the accident is initiated from a reactor power level less than 40%.

  - **Success Criteria.** Top Event PL is successful if the power is less than 40%. Top Event PL is used to determine if the AMSAC system is armed. Below 40%, AMSAC is not needed to mitigate the consequences of an ATWS event since the peak pressure attained in the primary system is not predicted to exceed 3,200 psig (Reference WCAP-11993, p. 4-3 and WCAP-10858P-A, Rev. 1).

  - **Model Boundaries.** There is no equipment or operator actions modeled in this top event.

- **Conditions when Demanded.** This event is asked whenever reactor trip fails; i.e., when Top Event RT is failed.

- **Scenario Impact if Successful.** Success of this top event implies that the initial power level is less than 40% and that Top Event SR (i.e., ATWS primary relief) is guaranteed successful.

- **Scenario Impact if Failed.** Failure of this top event with failure of main feedwater (Top Event FW) requires that Top Event SR be queried to determine if the peak RCS pressure is mitigated.

- Top Event MS — Main Steam Isolation

  - **Function Evaluated.** Automatic main steam isolation in response to low steam line pressure, high negative steam pressure rate with pressurizer pressure low, or high-high containment pressure.

  - **Success Criteria.** This top event models the successful closing of three of the four MSIVs, given a main steam isolation condition. Main steam line closure conditions may result from LOCAs, steam line breaks, and failures of the turbine to trip.

  - **Model Boundaries.** This event considers the response of the four MSIVs, given an actuation signal from SSPS on high-high containment pressure, or high negative steam pressure rate (< P-11), or low steam line pressure (> P-11). Inadvertent MSIV closures are not considered here but, rather, are considered separately in the definition and frequency quantification of initiating events that involve such events. Therefore, the potential for condenser dump valves sticking open after plant trip leading to a low steam line pressure is not modeled explicitly.

  - **Conditions when Demanded.** This event is always asked in the event tree. Guaranteed success is assumed if the initiator is a closure of all MSIVs. Guaranteed failure is assumed if the conditions for closure are not present or if the SSPS system fails to provide an actuation signal. No credit for manual isolation is given.

  - **Scenario Impact if Successful.** Success implies that three MSIVs have closed and that the condenser steam dumps and flow path are not available; i.e., Top Event CD is guaranteed failed. Once closed, the plant model takes no credit for reopening the MSIVs.

  - **Scenario Impact if Failed.** Failure of this top event implies that at least two MSIVs did not close. For steam line breaks, failure to isolate is conservatively assumed to lead to failure of the turbine-driven AFW pump due to loss of steam pressure. For other initiating events, failure to trip the turbine and failure to isolate main steam lead to failure of the AFW turbine-driven pump.

For large steam line breaks and for turbine trip failures, failure to close three of four MSIVs may potentially result in recriticality due to the rapid RCS cooldown. This concern is not modeled in the event trees, The frequency of such sequences is low, and the impact of a return to criticality is not expected to alter the success criteria for mitigating systems. The same reasoning applies to steam line breaks inside containment with successful MSIV closure but with failure of high head safety injection (HHSI).

- **Top Event CD — Condenser Available for Controlled Cooldown**

  - **Function Evaluated.** Condensing of steam from the main steam lines and return to the condensate booster suction via the condensate system.

  - **Success Criteria.** Success of this event requires that the steam flow path from the steam generators back to the main condenser be available, and that the condensate system be available for 24 hours following a plant trip to permit a controlled cooldown.

  - **Model Boundaries.** This top event models: (1) the turbine bypass valves, (2) the availability of the main condenser and that portion of the condensate system necessary to ensure discharge from a hotwell pump, and (3) a water path to the CST. This event also models the operator actions to verify the steam dumps open and to perform a cooldown using the pressure mode control of the steam dumps.

    The path from the steam generators to the condensers includes the banks of three condenser bypass valves used for cooldown: FCV-1-103, FCV-1-107, and FCV-1-111. The condensate portion models the flow from the condensers to hotwell pump A and out through the discharge check valve and manual isolation valve. The water path from the condensers to the condensate storage tank includes valves 2-521, 2-522, 2-506, and LCV-2-9.

  - **Conditions when Demanded.** This event is asked for all sequences in which the MSIVs remain open, permitting flow to the condenser; i.e., when Top Event MS is failed. This event is guaranteed failed when condenser vacuum is lost as an initiator.

  - **Scenario Impact if Successful.** Success of this event implies that there is no need for the steam generator PORVs or for long-term makeup for feedwater. Success of this top event with Top Events MF and OF implies that long-term secondary cooling and controlled cooldown are available using main feedwater and the turbine bypass valves. Similarly, success of this top event and AFWS success also imply successful long-term secondary cooling and controlled cooldown.

  - **Scenario Impact if Failed.** Failure of this event guarantees the failure of secondary heat removal using the main feedwater system.

- **Top Event FW — Feedwater Availability in ATWS Paths**

  - **Function Evaluated.** Feeding of steam generators with condensate water via the main feedwater pumps when the reactor has not tripped.

  - **Success Criteria.** Success of this event requires that the main feedwater system remain operational following a demand for reactor trip in which the reactor fails to trip.

  - **Model Boundaries.** This event considers the main feedwater system receiving flow from the condensate system via the main feedwater pumps, and through the feedwater control valves into the steam generators.

  - **Conditions when Demanded.** This event is asked whenever there is a failure of reactor trip and the MSIVs remain open (i.e., Top Event MS is failed) permitting a flow path back to the condenser, and the condensate system operates successfully; i.e., Top Event CD is successful. Feedwater is isolated if a safety injection condition occurs; i.e., Top Event FW is then guaranteed failed. This top event is also failed if the initiating event involved the feedwater system, i.e., partial or total loss of main feedwater, steam line breaks downstream of the MSIVs.

    If the reactor tripped successfully, this event is not asked. In this case, the availability of main feedwater for recovery after partial feedwater isolation is modeled via Top Event MF.

  - **Scenario Impact if Successful.** For ATWS events in which main feedwater is available, the peak RCS pressure will not exceed 3,200 psig (see WCAP-11993, page 4-12). Thus, success of this top event guarantees the success of RCS pressure relief during an ATWS; i.e., Top Event SR will be successful. The plant model also assumes that the pressurizer PORVs will not be challenged if main feedwater continues operation.

  - **Scenario Impact if Failed.** For ATWS events above 40% power, failure of main feedwater could result in RCS pressure exceeding 3,200 psig under some conditions. The conditions are dependent on the amount of AFW available, the success of manual rod insertion, and the degree of pressure relief. Failure of top event FW with initial reactor power above 40% requires that AFW be questioned before proceeding to Top Event SR.

- **Top Event AM — ATWS Mitigating System Actuation Circuitry**

  - **Function Evaluated.** Turbine trip and AFW start signals, given ATWS without main feedwater.

  - **Success Criteria.** This top event models the availability of the AMSAC system to provide signals to both trip the main turbine and start auxiliary feedwater, independent of SSPS when reactor power is above 40%.

- **Model Boundaries.** The components of the AMSAC system are not explicitly modeled. The probability of failure, given the availability of all necessary support, is taken from WCAP-11993. Failures of the support systems for AMSAC (250V DC station batteries) that disable AMSAC are considered separately.

- **Conditions when Demanded.** This event is asked only in the ATWS sequences when Top Event PL is failed (power level is greater than 40%) and feedwater is unavailable; i.e., Top Event FW is failed. AMSAC is not asked if the reactor trips, initial power level is less than 40%, or feedwater is successful.

- **Scenario Impact if Successful.** If Top Event AM is successful, credit is taken for AMSAC providing an additional start signal to AFWS and for tripping the turbine. Success of Top Event AM implies that the turbine successfully trips if the valve hardware (Top Event TT) were successful.

- **Scenario Impact if Failed.** The failure of AMSAC, with reactor power above 40% and main feedwater failed, implies that core damage will occur because the turbine fails to trip. Auxiliary feedwater may still be automatically actuated by SSPS or manually.

• **Top Events MA, MB, TP, TPR, and AF — Auxiliary Feedwater**

- **Function Evaluated.** Secondary heat removal via auxiliary feedwater following plant trip.

- **Success Criteria.** This sequence of top events is used to determine the status of auxiliary feedwater and steam relief to provide adequate secondary heat removal for 24 hours, given a plant trip condition. For sequences with successful reactor trip, successful secondary cooling by the AFW system requires at least one pump (Top Event MA, MB, TP, or TPR) feeding two steam generators (Top Event AF).

  For sequences involving failure of the reactor to trip, conditions of 50% flow and less than 50% AFW flow can be tracked. In this model, two motor-driven pumps supplying their respective steam generators or the turbine-driven AFW pump to all four steam generators are assumed to be required for successful secondary heat removal.

- **Model Boundaries.** Top Events MA and MB model the train A and train B motor-driven pump trains, respectively. They include the suction line to the CST, the alternate suction lines to the ERCW, the pumps, and discharge piping up to the header point. The turbine-driven pump is modeled in a similar fashion in Top Event TP. In the event that the turbine-driven pump trips on overspeed, local restart of the pump is modeled in Top Event TPR. Top Event AF models the discharge piping to the steam generators and steam relief. Top Event AF also includes the five safety relief valves for each steam generator.

The components modeled for Top Event MA include:

- The CST suction valves 3-803 and 3-605.
- The ERCW suction valves 3-116A and 3-116B.
- Motor-driven pump 1A-A and discharge valve 3-820.

The components modeled for Top Event MB are analogous and include:

- The CST suction valves 3-804 and 3-606.
- The ERCW suction valves 3-126A and 3-126B.
- Motor-driven pump 1B-B and discharge valve 3-821.

The components modeled for Top Event TP include:

- The CST suction valves 3-809 and 3-810.
- The ERCW suction valves 3-136A, 3-136B, 3-179A, and 3-179B.
- The steam generator No. 1 steam admission valves FCV-1-15 and 3-891.
- The steam generator No. 4 steam admission valves FCV-1-16 and 3-892.
- Downstream steam line valves FCV-1-17 and FCV-1-18.
- The turbine pump and discharge valve 3-864.

Human action TPR1 accounts for the likelihood of successfully restarting the turbine-driven AFW pump, given it tripped on overspeed. Approximately 30 minutes are available to complete the action.

The components modeled for Top Event AF consists of components in the flow paths to the four steam generators. The components for flow to steam generator 4 include:

- Motor-driven 1A discharge path valves 3-829, LCV-3-171, 3-833, and 3-837.
- Turbine-driven pump discharge path valves 3-870, LCV-3-175, 3-874, and 3-878.
- Common check valves 3-644 and 3-645.

The flow paths to the other steam generators are analogous. The five steam generator safety valves on each steam generator are also modeled.

Actuation signals for the three AFW pumps can come from SSPS, from AMSAC, or manually by the operators. The available signals determine the split fraction used for each pump's top event.

For steam line breaks and for steam generator tube rupture sequences, the affected steam generator is conservatively assumed to be unavailable for feeding by auxiliary feedwater.

During a station blackout, the AFW turbine-driven pump level control valves must be opened locally, manually after the control air header has lost pressure. This action is then included in the model for Top Event AF via action HAAF1. One hour is assumed to be available before the steam generators dry out.

- **Conditions when Demanded.** Auxiliary feedwater is asked for all sequences except when both the reactor fails to trip and main feedwater continues to operate. In that case, secondary heat removal via main feedwater is sufficient. These top events are asked if the reactor trips. For such sequences, main feedwater is eventually isolated, and the main feedwater pumps trip when RCS T_{avg} drops below the required value. If none of the pumps operate, Top Event AF, which contains the steam generator flow control valves, is not asked because AFW cannot be successful. Also, if the reactor fails to trip, and Top Events TP and MA both fail, then Top Event MB is not asked. About 50% AFW flow is required for success in such sequences, therefore, the availability of the last, motor-driven pump, which cannot provide that much flow, is inconsequential. No credit for restarting the turbine-driven pump is given for reactor trip failure sequences.

  All of the AFW pumps are guaranteed failed if the long-term suction supplies are unavailable. The turbine-driven pump is failed due to loss of steam pressure if the turbine fails to trip or a steam line break occurs downstream of the MSIVs; however, in either case, the MSIVs fail to close. If a steam generator is faulted (e.g., main steam line break inside containment, or stuck-open steam generator PORV or safety valve), then the possibility of feeding that steam generator for secondary heat removal is omitted.

- **Scenario Impact if Successful.** If one or more of the AFW pump top events are successful, along with success of the pump discharge path (i.e., Top Event AF), then AFW is providing sufficient secondary heat removal.

- **Scenario Impact if Failed.** If sufficient AFW flow is not available, then an alternate means of cooling the core must be provided. Either main feedwater must be restored or the operators must initiate feed and bleed cooling. These alternatives are considered in later top events.

- **Top Event MF — Main Feedwater Restoration (Non-ATWS Sequences Only)**

  - **Function Evaluated.** The availability of main feedwater hardware following successful plant trip.

  - **Success Criteria.** This top event models the hardware required for restoration and continued operation of main feedwater for 24 hours, given that the auxiliary feedwater has failed.

- **Model Boundaries.** Top Event MF models the hardware failure of the equipment involved in feedwater restoration, while the operator actions associated with restoration are modeled in Top Event OF. This top event interfaces with Top Event CD to provide secondary cooling. Typically, main feedwater would be isolated and the pumps tripped following a plant trip, either due to a safety injection signal, or when the RCS T_{avg} temperature is reduced below the required value. Success of Top Event MF requires that the feedwater isolation signal, if present, be removed, and requires the proper operation of the following equipment:

  - The gland steam condenser.
  - The steam generator blowdown second-stage heat exchanger.
  - The bypass line of the gland steam condenser and the steam generator second-stage heat exchanger.
  - One main feedwater pump turbine (MFPT) condenser (assuming that 1A is required).
  - One low pressure heater string A, B, or C.
  - The steam generator blowdown first-stage stacked heat exchangers.
  - The bypass line of the LP heaters and the steam generator blowdown first-stage stacked heat exchangers.
  - One condensate booster pump (assuming that pump A is required).
  - One intermediate pressure heater string A, B, or C.
  - One turbine-driven main feedwater pump (assuming that pump A is required).
  - One high pressure heater A, B, or C.
  - The steam generator 1 feedwater flow control valve bypass line.
  - Standby feedwater pump.

  Failure of Top Event CD guarantees failure of this top event. Success of this top event with Top Events CD and OF implies that long-term secondary cooling and controlled cooldown are available using main feedwater and the turbine bypass valves.

- **Conditions when Demanded.** This event is asked when the reactor trips successfully, but AFW has failed. If the reactor fails to trip and AFW fails, restoration of main feedwater is assumed to come too late to mitigate the accident. This event is guaranteed failed if the initiator is a total loss of main feedwater. Partial loss of main feedwater initiators is assumed to pass enough flow to remove decay heat after plant trip once the system is realigned. Excessive main feedwater initiators are assumed to fail this top event. This event is also failed if the sequence is initiated by a steam line break downstream of the MSIVs.

- **Scenario Impact if Successful.** Success of Top Event MF means that the hardware response needed for restoration of main feedwater is available. If the operators then attempt to initiate main feedwater restoration (i.e., Top Event OF is successful), then secondary heat removal is restored. Feed and bleed cooling is then not required to prevent core damage.

- **Scenario Impact if Failed.** If Top Event MF fails, then main feedwater cannot be restored. For such sequences, the operators must initiate feed and bleed cooling to prevent core damage.

- **Top Event OF — Operator Actions To Restore Main Feedwater**

  - **Function Evaluated.** Operator action to restore main feedwater for secondary heat removal after plant trip.

  - **Success Criteria.** Top Event OF models the operator action to restore main feedwater prior to the need for feed and bleed cooling in the event of a loss of all AFW. The operators must complete this action before the steam generator wide-range level for all steam generators drops below 25% since Step 10 of FR-H.1 then instructs the operators to go to feed and bleed.

  - **Model Boundaries.** This top event models the operator actions required to restore main feedwater after auxiliary feedwater has failed. These actions are identified in the Functional Restoration Guideline FR-H.1. The following is a summary of the actions required:

    - Ensure safety injection signals cleared or blocked.
    - Reset MFW isolation.
    - Start condensate booster pump or DI booster pump.
    - Restart standby MFW pump.
    - Open MFW isolation valve.
    - Control MFW flow using FW flow control bypass to restore steam generator level.

    Two separate conditions for the actions are evaluated:

    - OF1. Following AFW failure during a transient in which a safety injection condition has not occurred, complete before feed and bleed is directed at 45 minutes.

- **OF2.** Following AFW failure during a transient such as a small LOCA, in which a safety injection condition has occurred, complete before transition to feed and bleed is directed at about 30 minutes.

– **Conditions when Demanded.** Top Event OF is asked whenever main feedwater initially isolates but is available for recovery (i.e., Top Event MF is successful) and AFW fails. Top Event OF is not asked, given failure of Top Event MF or CD. Different split fractions are used for Top Event OF, depending on whether a safety injection condition is present.

– **Scenario Impact if Successful.** Success of Top Event OF means that main feedwater, and therefore secondary heat removal, is successfully restored following a plant trip with AFW failed.

– **Scenario Impact if Failed.** If Top Event OF fails, then secondary heat removal is not restored before the procedural guidance calls for the operators to initiate feed and bleed cooling. Feed and bleed cooling must then be initiated to prevent core damage.

- **Top Event SR — Primary Relief**

– **Function Evaluated.** Reactor coolant system pressure control for sequences with failure of reactor trip.

– **Success Criteria.** This top event models the unavailability of RCS pressure relief in ATWS sequences. Adequate pressure relief requires that the peak RCS pressure be less than 3,200 psig. Three safety valves and either 0, 1, or 2 PORVs are required to lift to limit the RCS pressure to less than 3,200 psig. For different fractions of the fuel cycle, the number of PORVs required varies. The number required is also dependent on whether manual rod insertion is successful, and how much feedwater flow is available.

– **Model Boundaries.** The major pieces of equipment modeled in this top event include the three pressurizer safety valves (SFV) and the two pressurizer PORVs.

The number of required valves depends on the moderator temperature coefficient, the status of secondary cooling, and whether manual insertion of control rods (Top Event MR) was successful. The criteria used are recreated from Section B.7.2 of WCAP-11993. Due to the short time available for the operators to respond, no credit is taken for opening an initially closed block valve.

– **Conditions when Demanded.** This top event is always asked for ATWS sequences. In cases in which it is known peak RCS pressure will be less than 3,200 psig, Top Event SR is set to success. The conditions for peak pressure being less than 3,200 psig during an ATWS, without considering the response of the pressurizer relief valves, are that the initial reactor power level is less than 40% (Top Event PL is successful) or that main feedwater were available (Top Event FW is successful).

The number of pressurizer valves required to maintain RCS pressure in an ATWS, is dependent on whether the operators successfully initiate rod step in; i.e., Top Event MR. Therefore, the conditional probability of Top Event SR is a function of the status of Top Event MR. Although the number of PORVs required is also dependent on how much AFW flow is available, this is a second order effect. Therefore, for modeling simplification, AFW flows greater than 50% (e.g., all three pumps operating) were conservatively evaluated as if only 50% AFW flow were available.

If an ATWS sequence initiates with the reactor power level above 40%, main feedwater fails, and the turbine fails to trip; then peak RCS pressure will exceed 3,200 psig. The pressure will exceed 3,200 psig even if auxiliary feedwater is available and the pressurizer relief and safety valves respond correctly. For such ATWS sequences, Top Event SR is then guaranteed failed.

- **Scenario Impact if Successful.** The success of this top event implies that sufficient relief capacity is available for the existing core conditions such that the RCS peak pressure did not exceed 3,200 psig. Success of Top Event SR does, however, imply that a LOCA has been induced. The basis for this is that the pressure spike due to higher temperatures challenges the primary PORVs and safeties to the extent one sticks open.

- **Scenario Impact if Failed.** Failure of this top event means that RCS pressure exceeds 3,200 psig. Reactor vessel failure is then assumed to occur, and ECCS flow is not sufficient to mitigate the loss of coolant. Due to the large break postulated to occur, such sequences are mapped to a low pressure melt.

- **Top Event SL — Secondary Leakage into the Environment**

  - **Function Evaluated.** Isolation of the secondary side in response to a steam generator tube rupture.

  - **Success Criteria.** This event models the operator actions and equipment needed to isolate a ruptured steam generator from the environment, given that the tube rupture initiates the plant trip or occurs in response to a large RCS to secondary-side pressure drop, such as occurs in an ATWS or a steam line break sequence. For these induced tube ruptures, the secondary side is conservatively assumed to be unisolated.

  - **Model Boundaries.** The analysis of actions to isolate the ruptured steam generator is simplified by conservatively assuming that the corresponding MSIV and all steam valves upstream of the MSIV must close. The operator must identify and manually isolate these valves, if open, in accordance with EOP E-3. The valves that must close on the rupture steam generator are the MSIV, the five safety relief valves, the PORV, the blowdown valves, and the steam supply line to the turbine-driven AFW pump. The time window for identifying and isolating the ruptured steam generator and controlling feedwater is based on the time to overfill the steam generator; i.e., estimated as about 50 minutes. The operator action is designated HASL1.

The probability of an ATWS-induced SGTR is dependent on the peak RCS pressure, which is, in turn, dependent on the status of Top Event SR. If an induced SGTR occurs from an ATWS, Top Event SL is guaranteed to fail. If an induced SGTR has not occurred, then Top Event SL is assumed to be successful.

In the case of an induced tube rupture caused by a main steam line break, the rupture occurs on the same steam generator that blew down and, thus, that steam generator cannot be isolated.

– **Conditions when Demanded.** This event is the mechanism by which the impacts of an SGTR initiating event are modeled in the event trees. This top event is asked in every sequence except where Top Event SR fails.

Top Event SR fails in ATWS sequences in which RCS pressure relief is inadequate. For such sequences, the steam generator tubes were assumed not to fail before the reactor vessel. After vessel failure due to overpressure, the RCS pressure was assumed to be inadequate to fail the tubes. For ATWS sequences in which the vessel remains intact, some chance of overpressure-induced steam generator tube ruptures is considered.

There is also a chance of induced steam generator tube ruptures from steam line breaks. These can be significant if the steam line break is outside containment. These events are considered in the assignment of branch point probabilities for Top Event SL.

– **Scenario Impact if Successful.** Success of this event implies that the ruptured steam generator is isolated, if the rupture occurred as the initiator. If an ATWS or a steam line break outside containment occurs, then success of Top Event SL implies that the steam generator tubes remained intact, despite the abnormally large pressure drop across them during the transient.

– **Scenario Impact if Failed.** Failure of Top Event SL means that an SGTR has occurred (either as the initiator or induced by the accident progression due to a large pressure drop across the steam generator tubes) and that there is a release path from the RCS to the environment. Failure of Top Event SL is assumed to mean that the MSIV as well as a safety valve on the ruptured steam generator have failed open. The procedures then direct the operators to close the MSIVs on the intact steam generators, and to close the condenser dumps. Therefore, failure of Top Event SL implies a loss of flow from the intact steam generators to the condenser.

RCS pressure must be reduced and maintained below the ruptured steam generator pressure (Top Events DS and DP) or else a long-term source of makeup to the RCS must be provided (Top Event MU in the recirculation tree). Normally, the operators would be required to place RHR in service after the initial depressurization to proceed to cold shutdown, thereby stopping the leak (Top Event RD). If RCS inventory cannot be controlled (either by continued high pressure makeup from the RWST or sufficient pressure reduction), then eventual core damage is assumed with containment bypass through the ruptured steam generator.

- **Top Event VS — Supply to Centrifugal Charging Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Flow path for water to charging pumps suction.

  - **Success Criteria.** Top Event VS is used to determine the availability of the supply of borated water to the centrifugal charging pumps for 24 hours after a plant trip. The supply may be provided from the volume control tank or the RWST, depending on whether a safety injection signal has been initiated.

  - **Model Boundaries.** The equipment required to operate is dependent on the reactor conditions. Volume control tank (VCT) level control valves, LCV-62-132 and LCV-62-133, are required to remain open during a nonsafety injection general transient. For the cases with a safety injection, LCV-62-132 and LCV-62-133 are required to close and RWST level control valves LCV-62-135 and LCV-62-136 are required to open and remain open for 24 hours. There is an operator action house event used in this top event to align these valves manually if the actuation signal from SSPS fails. This manual action is analyzed in Top Event OS. The automatic opening of LCV-62-135 and LCV-62-136, in the absence of a safety injection signal, on the closure of LCV-62-132 or LCV-62-133 is not modeled.

  - **Conditions when Demanded.** Top Event VS is asked in all sequences not already determined to be core damage sequences. Core damage is already established when Top Event SR fails in response to an ATWS, or when there is no secondary heat removal during an ATWS; i.e., when both Top Events FW and AF have failed. Different split fractions are used to distinguish safety injection conditions where the supply must swap to the RWST versus those that do not involve a safety injection condition, and the supply remains from the VCT.

  - **Scenario Impact if Successful.** If Top Event VS is successful, this implies that a suction source is available for makeup to the RCS, including for RCP seal injection.

  - **Scenario Impact if Failed.** Failure of Top Event VS, the charging pump suction supply, fails both the centrifugal charging pump Top Events VA and VB.

- **Top Events VA and VB — Centrifugal Charging Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Operation of the charging pumps.

  - **Success Criteria.** These top events, VA and VB, model the two centrifugal charging pumps 1A-A and 1B-B, respectively. The success of these top events requires that the modeled pump start on an actuation signal and operate for 24 hours to provide flow to the cold leg injection path and the RCP seal injection header. The centrifugal charging pumps are used for the following modeled functions in this analysis:

- High head safety injection, in both injection and recirculation modes.
- Emergency boration.
- Reactor coolant pump seal injection.

— **Model Boundaries.** The pieces of equipment modeled in each top event include the pump itself, the pump lube oil cooler, the room cooler, and the manual valves and check valve in the pump train. These top events have a mission time of 24 hours. Centrifugal charging pump (CCP) 1A-A is modeled in Top Event VA as the normally running pump to supply normal charging to the reactor coolant system and RCP seal injection at the start of the event. CCP 1B-B in Top Event VB is modeled in standby. These pumps are being normally supplied from the VCT.

For an initiating event that results in a safety injection signal or loss of offsite power, both pumps are given an actuation signal to start and are required to inject borated water from the RWST to the RCS cold legs. Manual operator action to start the CCPs, given automatic actuation system failure, is modeled in Top Event OS. The supply to the CCPs swaps over from the VCT to the RWST as modeled in Top Event VS. The injection path to the RCS cold legs is automatically aligned as modeled in Top Event VC. If sump recirculation is required, the RWST is isolated and the RHR pumps draw from the containment sump to supply the centrifugal charging pumps for high head injection during recirculation mode.

For an initiating event that results in an ATWS, the centrifugal charging pumps are used to inject concentrated boric acid solution from the boric acid tank and boric acid transfer pumps as modeled in Top Event EB.

The CCPs are required during normal operation and any initiating event to supply injection water to the reactor coolant pump seals as modeled in Top Event SE. The seal injection and thermal barrier protection prevents a postulated RCP seal LOCA. For a general transient, the CCPs continue to draw from the VCT; for a transient that results in a safety injection signal, the CCPs draw from the RWST. Both of these cases are modeled in Top Event VS.

— **Conditions when Demanded.** Both Top Events VA and VB are asked each time Top Event VS is successful. They are not asked if Top Event VS fails because the suction supply is unavailable. The status of offsite power is used to determine whether the normally running charging pump is tripped off and must be restarted.

— **Scenario Impact if Successful.** If either Top Event VA or VB is successful, then centrifugal charging is available for injection, recirculation, emergency boration, and for RCP seal injection.

- **Scenario Impact if Failed.** If both Top Event VA and VB fail, then neither centrifugal charging pump is available. Alternate means of makeup to the RCS and for RCP seal cooling may be required.

- **Top Event VC — Centrifugal Charging Pump Cold Leg Injection**

  - **Function Evaluated.** Availability of the flow path from the charging pumps to the cold leg injection lines.

  - **Success Criteria.** This top event models the centrifugal charging pumps discharge piping and valves from the pumps to the cold legs of the RCS. The success of this top event requires that one of the two parallel boric injection tank (BIT) outlet valves open and remain open, the inline check valve opens and remains open, and two of the four cold leg paths are available including the associated check valve, which must open and remain open. This top event has a mission time of 24 hours.

  - **Model Boundaries.** The equipment modeled includes the equipment from the point where the pump discharge lines are joined through the RCS cold leg injection check valves. This includes:

    - Parallel MOVs FCV-63-39 and FCV-63-40 (locked open).

    - The BIT and downstream valves FCV-63-25 and FCV-63-26.

    - The injection line check valve inside containment (1-63-581).

    - The four branch lines into the RCS, each containing a throttling valve (THV-63-582, THV-63-583, THV-63-584, and THV-63-585) and a check valve (1-63-586, 1-63-587, 1-63-588, 1-63-589).

    This top event is required for chemical volume control system (CVCS) high head safety injection success in both the injection and recirculation modes. The pumps are modeled separately in Top Events VA and VB.

    An operator action house event is used in this top events to manually make the valve alignment if the automatic actuation from SSPS has failed. This operator action is modeled in Top Event OS.

  - **Conditions when Demanded.** Top Event VC is asked in all sequences not already determined to be core damage sequences. Core damage is already established when Top Event SR fails in response to an ATWS, or when there is no secondary heat removal during an ATWS; i.e., when both Top Events FW and AF have failed. If both centrifugal charging pumps are failed (i.e., Top Events VA and VB failed) or if a suction path from the RWST is not available (i.e., Top Event VS has failed), then this event is assumed to be guaranteed failed.

  - **Scenario Impact if Successful.** Success of this top event implies that at least one of the centrifugal charging pumps is available to take suction from the RWST and inject into the RCS through the required number of cold leg injection lines to mitigate a loss of RCS inventory.

- **Scenario Impact if Failed.** Failure of this top event implies that CVCS high head injection has failed. High head injection may still be provided by the safety injection system.

• **Top Event EB — Emergency Boration (ATWS Sequences Only)**

- **Function Evaluated.** Reactivity control via emergency boration or deenergization of the control rod motor generator sets in response to an ATWS sequence.

- **Success Criteria.** This top event models the operator actions and equipment required to ensure the reactor is subcritical, given an initial failure of the control rods to insert properly. This includes the operator actions to start emergency boration or to deenergize the control rod motor generator sets. WCAP-11993 indicates that 10 minutes are available to complete either of these actions for successful reactivity control.

- **Model Boundaries.** This top event is employed in ATWS sequences to shut down the reactor. For emergency boration, high concentration boric acid solution is transferred from the boric acid tanks to the centrifugal charging pumps suction by a normally recirculating boric acid pump. Top Events EB, RW, VA or VB, VS, and VC are required for successful emergency boration.

  Deenergizing the control rod latching mechanism is accomplished by opening the input breakers for the control rod drive motor generator sets at 480V unit boards A and B. The model for tripping the motor generator sets also considers the possibility that the control rods may have mechanically bound, preventing insertion; i.e., the failure probability for the motor generator sets tripping considers the conditional probability that the reason Top Event RT failed was because of the control rods sticking. In such cases, tripping the motor generator sets would have no effect. Also, no credit is taken for deenergizing the motor generator sets if the initiating event is an LOSP and a reactor trip has not occurred. Failure of reactor trip with a loss of offsite power indicates that the rods are physically stuck.

  An operator action is modeled within this top event. It is designated HAEB1. The operators open the breakers to the control rod motor generator sets at the 480V unit boards, borate the RCS by speeding up the boric acid pumps, opening the emergency boration valve (1-FCV-62-138), and aligning the centrifugal charging pumps to inject into the RCS. Successful deenergizing of the motor generator sets or emergency boration constitutes successful reactivity control. However, a single action is defined as if both actions are required for success. The procedural guidance for both activities is the same; therefore, the two options are modeled as totally dependent. Numerous actions are required of the operators in response to an ATWS. This model assumes that the actions required in the first minute (i.e., manual trip and rod insertion) are not relevant to this action for longer term reactivity control; i.e., within 10 minutes. Therefore, no dependency is assumed between the action for initiating emergency boration and tripping the motor generator sets on the earlier actions.

- **Conditions when Demanded.** Top Event EB is asked in all ATWS sequences not already determined to be core damage sequences. Core damage is already established when Top Event SR fails in response to an ATWS, or when there is no secondary heat removal during an ATWS; i.e., when both Top Events FW and AF have failed. Top Event EB is not asked if the reactor trips because additional reactivity control is not required.

- **Scenario Impact if Successful.** Success of Top Event EB implies that reactivity control is established using either emergency boration or by tripping the motor generator sets.

- **Scenario Impact if Failed.** Failure of this top event means that attempts at achieving subcriticality have failed and that core damage will eventually occur due to loss of RCS inventory.

• **Top Event WC — RCS Primary Relief Not Water Challenged**

- **Function Evaluated.** The absence of a condition that would challenge the pressurizer relief valves to have to open and then reseat after passing water.

- **Success Criteria.** A water challenge to the pressurizer safety and relief valves is postulated to occur under safety injection conditions with the RCS intact if the operator fails to terminate safety injection in time. This includes responses to inadvertent safety injection and excessive cooldown initiators. Success requires the operators to recognize that the safety injection signal actuation is not necessary and to take action to reduce injection flow in time to prevent a PORV challenge.

- **Model Boundaries.** The model for this top event consists of just the operator action to terminate safety injection. This action is designated as HAWC1. Credit for the operators to prevent the water challenge is only allowed when the operators are instructed to: (1) shut off ECCS pumps, (2) control pressurizer level, or (3) if a blanket action is modeled, that the operators monitor and control pressurizer level during any event all of the time. There are no hardware failure modes considered in this model.

  The initiating events that are evaluated by this top event are Inadvertent SIS (ISI), Steam Line Breaks (SLBOC, SLBIC), Steam Line PORV/SFV Fails Open (MSVO), and Inadvertent MSIV Closure (IMSIV). The selection of ISI is evident. The cooldown events are selected because they eventually require similar action. IMSIV is added on the basis that an inadvertent MSIV closure is most likely caused or associated with an SIS. Combinations of losses of instrument buses that cause an inadvertent safety injection signal are also considered.

Successful prevention of a water challenge is guaranteed if both centrifugal charging pumps fail to provide injection, or if the sequence does not involve conditions that led to the automatic start of the centrifugal charging pumps.

— **Conditions when Demanded.** Top Event WC is asked in all sequences for which secondary heat removal is available. If secondary heat removal is not available, termination of safety injection is not permitted; i.e., feed and bleed is to be performed. If injection flow then exceeds the rate of loss of RCS inventory, pressure relief will then be required. Challenges for water relief are only considered in the one or more of the charging pumps are available and a safety injection condition has occurred.

— **Scenario Impact if Successful.** If Top Event WC is successful, then there is no water challenge to the pressurizer valves. The pressurizer relief valves may still be challenged to relieve steam.

— **Scenario Impact if Failed.** If Top Event WC fails, this means that there has been a safety injection actuation and that the operators failed to terminate safety injection in time to prevent overfilling the pressurizer. The pressurizer valves then are challenged to lift and must reseat after passing water. The failure of the pressurizer valves to reseat is modeled via Top Event PR. An implicit assumption is that the operators do eventually terminate safety injection after the pressurizer valves are challenged.

- **Top Event PR — RCS Pressure Relief**

— **Function Evaluated.** RCS pressure control in response to a plant trip.

— **Success Criteria.** This event models the RCS pressure relief via the pressurizer PORVs and safety valves in response to a pressure relief challenge to pass water or steam. Success of this event requires that either there is no challenge for RCS pressure relief (i.e., RCS pressure remains below the pressurizer PORV's relief setpoint) or the pressurizer relief valves successfully open to relieve pressure and the valves that open then successfully reclose to prevent a LOCA.

— **Model Boundaries.** The pieces of equipment modeled in this top event include the safety valves 68-563, 68-564, and 68-565, the PORVs PCV-68-334-A and PCV-68-340-A, and the PORV block valves FCV-68-332 and FCV-68-333. In the event of a stuck-open pressurizer PORV, the model also considers the operator action to recognize that a PORV has stuck open and attempts to close the associated block valve. Two separate operator actions are considered: one designated HAPR1 and the other HAPR2. HAPR1 is for sequences in which the safety injection signal has occurred. HAPR2 is for sequences in which safety injection actuation has not yet occurred. Both must be completed in less than 5 minutes to avoid failure of the PRT rupture disk, which is assumed to lead to containment spray initiation and an eventual requirement for sump recirculation. For action HAPR2, the model assumes that it must also be completed in less than 30 seconds after the initial opening to prevent a safety injection from occurring.

Top Event PR also considers the probability for a pressure relief challenge given the sequence of interest. The status of Top Event WC serves to separate the water challenges from the steam relief only challenges. Failure of Top Event WC implies a water challenge. Success of Top Event WC precludes a water challenge. The model assumes that a steam relief challenge occurs whenever the support systems for pressurizer spray or the steam generator atmospheric steam dumps are not available.

A nonisolable small LOCA initiating event is modeled as a failure of the RCP seal LOCAs via Top Event SE, which implies the guaranteed failure of this top event.

The isolable small LOCA initiating event requires that the operators close the affected PORV train's block valve. A separate split fraction is modeled for this sequence.

- **Conditions when Demanded.** Top Event PR is asked for all sequences in the Gentrans/small LOCA/ATWS/SGTR event tree. Top Event PR is assumed guaranteed successful if there is no water challenge and both pressurizer spray and steam generator atmospheric relief valves are available; i.e., there is also no challenge for steam relief. The isolable small LOCA initiating event is modeled as requiring the operators to close the PORV block valve to stop the break flow. The loss of primary flow initiator is modeled as disabling all pressurizer spray, resulting in a pressurizer PORV steam challenge.

- **Scenario Impact if Successful.** If Top Event PR is successful, either RCS pressure relief was not required; or if pressure relief was required, it was successful, and the pressurizer valves that opened successfully reclosed. Following the pressure relief, the RCS remains intact.

- **Scenario Impact if Failed.** A failure of Top Event PR is modeled as one valve failing to reclose. If a pressurizer PORV fails to reclose, then credit is taken for the operator isolating the PORVs by closing the block valves. Failure of this event implies that a small LOCA was the initiating event and has not been isolated, or that a small LOCA has been induced by the transient.

- **Top Event TB — RCP Thermal Barrier Cooling/RCP Seal LOCA**

  - **Function Evaluated.** The availability of RCS thermal barrier cooling.

  - **Success Criteria.** This event models the thermal barrier cooling to all four RCPs. Success requires that thermal barrier cooling be provided to all four RCPs for 24 hours following plant trip.

  - **Model Boundaries.** This top event model includes the flow path to the thermal barrier booster pumps from the CCS heat exchanger A, the booster pumps, the associated piping and valves that form the flow path to the thermal barrier of each pump, and flow from the pumps through FCV-70-87, FCV-70-90, and 70-690 for 24 hours following a plant trip. The model boundary starts upstream of the booster pumps at check valve 671 and terminates downstream of the FCVs at manual valve 690.

- **Conditions when Demanded.** Top Event TB is asked for sequences in which no other LOCA source has developed (i.e., Top Event PR is successful) and core damage is known not to already have occurred. Core damage is already established when Top Event SR fails in response to an ATWS, or when there is no secondary heat removal during an ATWS; i.e., when both Top Events FW and AF have failed. This event is guaranteed to fail under conditions of Phase B isolation due to isolation of the booster pumps or if there is a complete loss of CCS.

- **Scenario Impact if Successful.** If Top Event TB is successful, RCP seal cooling is available. Success of Top Event TB means that RCP seal damage will not occur due to loss of cooling.

- **Scenario Impact if Failed.** Failure of this top event is the failure to provide thermal barrier cooling to any RCP. Failure of RCP thermal barrier cooling and seal injection, which is modeled in Top Event SE, is sufficient to cause an RCP seal LOCA.

- **Top Event SE — RCP Seal Injection/RCP Seal LOCA**

  - **Function Evaluated.** The availability of RCP seal integrity; i.e., the absence of a LOCA via the RCP seals.

  - **Success Criteria.** Success of Top Event SE means that either thermal barrier cooling or RCP seal injection is available to cool all of the RCP seals. In the event that cooling to the RCP motor bearings is lost, then the operators must trip the RCPs within 10 minutes to protect the seals from damage caused by excess pump vibration.

  - **Model Boundaries.** Following the loss of RCP bearing cooling by (1) a Phase B isolation signal, (2) loss of CCS train 1A; or (3) RCP bearing cooling path failure, the operators must stop the RCPs to prevent damage to the RCP seals that will result in a seal LOCA. Top Event SE is used to determine if an RCP seal LOCA has occurred. The focus is on the failure of seal injection. Failure of both seal injection and thermal barrier cooling, as modeled in Top Event TB, is sufficient to cause an RCP LOCA.

    RCP seal LOCAs are also postulated to occur under conditions in which CCS flow to the bearings has failed, the RCPs are still running, and the operators fail to trip the pumps before a vibration-induced seal failure occurs. The operator action to trip the affected RCPs under such conditions is included in the model for Top Event SE. The operators' RCP trip response is evaluated under two conditions as indicated below:

    - **HASE1.** There is no LOCA, but Phase B isolation occurs due to a steam line break inside containment.

- **HASE2.** There is no LOCA or steam line break, but CCS train A is unavailable for cooling the RCPs.

The RCP seal injection model interfaces with the centrifugal charging pump model (Top Events VA and VB). The flow path starts where the centrifugal charging pumps are headered (upstream of FCV-62-93 at 62-535), flows through the RCPs, into the seal water return line, through the seal water heat exchanger, and back into the CVCS centrifugal charging pump suction just downstream of 62-650 and 62-653. The CCS cooling to the seal water heat exchanger is also modeled.

- **Conditions when Demanded.** Top Event SE is asked under the same conditions as Top Event TB; i.e., when there is no other LOCA present and core damage is not known to have already occurred. For loss of RCP cooling conditions requiring RCP trip, credit is given for the initiating event being a loss of offsite power, which precludes the need for tripping of the pumps.

- **Scenario Impact if Successful.** Success of Top Event SE means that the RCP seals remain intact. Excess leakage does not develop.

- **Scenario Impact if Failed.** Failure of this event implies that an RCP seal LOCA has occurred and that a small LOCA is in progress. High pressure injection is then required to maintain inventory control.

- **Top Events S1 and S2 — Safety Injection Pumps 1A-A and 1B-B**

  - **Function Evaluated.** The automatic actuation and operation of the safety injection pumps.

  - **Success Criteria.** These top events, S1 and S2, model the safety injection pump trains 1A and 1B, respectively. The success of these top events requires that the modeled pump start automatically on a safety injection actuation signal or manually by the operator and provide flow to the safety injection system cold leg injection paths. One of four charging and safety injection pumps operating for 24 hours is sufficient for inventory control during a small LOCA, but the status of each pump train is tracked individually.

  - **Model Boundaries.** The model boundary for the pump train in each of these top events includes the common pump suction line from the RWST and the safety injection line discharge cross-connect valve FCV 63-152 or FCV 63-153. The pump miniflow lines to the RWST, room cooling and motor cooling are also included. The components within this boundary for train A (train B components shown in parenthesis) include:

    - Normally open suction valve FCV-63-47 (FCV-63-48).

    - Safety injection pump 1A-A (1B-B) and check valve 63-524 (63-526).

- Normally open manual valve 63-525 (63-527) and normally open discharge valve FCV-63-152 (FCV-63-153).

- Miniflow line check valve 63-526 (63-530), motor-operated valve FCV-63-4 (FCV-63-175).

A complete injection path for a safety injection pump requires flow to the pumps [from the RWST (Top Event RW) in the injection phase or from the RHR system in the recirculation phase] and the injection paths defined in Top Event SI. Therefore, successful operation of the safety injection system for a pump in the injection mode requires the pump (Top Events S1 and S2), the suction source (support tree Top Event RW), and the injection path as defined by safety injection.

Top Event S2 is evaluated conditionally on the status of Top Event S1. This permits the intersystem dependencies between trains of the system (i.e., for common cause, test, and maintenance) to be modeled properly.

– **Conditions when Demanded.** Top Events S1 and S2 are asked along almost every sequence in the general transient/small LOCA/ATWS/SGTR event tree. The only exception is for ATWS sequences involving failure of Top Event SR; i.e., failure of RCS pressure relief. For such sequences, reactor vessel failure is assumed, and no amount of flow from the ECCS systems is assumed to be sufficient to permit subsequent inventory control. For sequences involving the unavailability of the RWST, both Top Events S1 and S2 are guaranteed failed.

– **Scenario Impact if Successful.** If either Top Event S1 or S2 is successful, then high pressure injection and high pressure recirculation may be successful. For successful high pressure injection, Top Event SI must also be successful to provide the required cold leg injection paths. Successful high head safety injection in the recirculation mode requires the pump (Top Events S1 and S2), the injection path as defined by Top Event SI, and RHR supply flow to the pump suction (as defined by Top Events RA, RB, RI, and RR).

– **Scenario Impact if Failed.** Failure of both Top Events S1 and S2 means that the safety injection pumps are not available for high pressure injection or recirculation. Successful injection and recirculation may still be available using the charging pumps.

- **Top Event SI — Safety Injection System Discharge Piping**

  – **Function Evaluated.** The availability of a flow path from the discharge of the safety injection pumps to the RCS cold legs.

  – **Success Criteria.** Success of Top Event SI requires that one of the safety injection pumps is available (i.e., Top Event S1 or S2 is successful), that the suction path from the RWST is available, and that at least one of four cold leg injection paths is available for 24 hours.

- **Model Boundaries.** This top event models both the suction and discharge flow path to the safety injection pump trains modeled in Top Events S1 and S2. The discharge portion of the model considers flow from the SIS pump trains to the RCS cold leg loops, from the discharge path of the pumps modeled in top events S1 and S2 to the injection point in the four RCS cold legs downstream of check valves 63-551, 63-553, 63-555, and 63-557. The suction portion of the model considers the flow path from the RWST and through FCV-63-5 and check valve 63-510. Success requires that the suction path and a flow path through one of the four RCS cold legs be available for 24 hours.

  The system boundary is defined so as to complete the flow path between the pump discharge and the RCS cold legs. The components modeled include:

  - The normally open valve FCV-63-22 where the flow from the two pumps is headered before splitting into four injection paths.

  - The four injection paths, each containing a throttle valve (63-550, 63-552, 63-554, and 63-556) and associated check valve (63-551, 63-553, 63-555, and 63-557, respectively).

  - The four cold leg check valves (63-560, 63-561, 63-562, and 63-563).

  The safety injection pump suction path modeled includes:

  - The normally open valve FCV-63-5.
  - Check valve 63-510.

  There is no operator action required or modeled in this top event.

- **Conditions when Demanded.** Top Event SI is asked in the same sequence conditions as for Top Events S1 and S2; i.e., everywhere that core damage is not already ensured. If the RWST is not available, or if both safety injection pumps fail, then Top Event SI is assumed to be guaranteed failed.

- **Scenario Impact if Successful.** Success of Top Event SI, along with success of either or both of Top Events S1 and S2, means that the high head safety injection pumps successfully provide high pressure injection and are available for high pressure recirculation once the RWST is exhausted.

- **Scenario Impact if Failed.** Failure of this top event implies that injection flow from the safety injection system into the RCS during both injection and recirculation phases is not available.

- **Top Event DS — Depressurization of the Secondary**

  - **Function Evaluated.** Depressurization of the secondary side.

  - **Success Criteria.** A source of feedwater and a controllable steam relief path are required to depressurize the secondary and to cool down the RCS. If main feedwater (Top Event MF) or auxiliary feedwater (one of Top Events TP, MA, or MB and success of Top Event AF) is available, then the depressurization/cooldown may be successful if Top Event CD is successful (i.e., the condenser is available for a controlled cooldown through the TBVs) or the steam generator PORVs are available. The success criterion for secondary depressurization and RCS cooldown using the steam generator PORVs is that one of four PORVs open.

  - **Model Boundaries.** This top event models the operator actions and equipment necessary to depressurize the secondary and cooldown the RCS. If Top Event CD is successful, this event models only the operator action to switch to the pressure control mode and initiate the depressurization. The hardware for depressurization using the condenser is modeled in Top Event CD. Depressurization and cooldown using the steam generator PORVs are addressed in this top event if Top Event CD has failed.

    The operator actions included in this model also account for the action to later depressurize the RCS; i.e., for which the hardware only is modeled in Top Event DP. Different operator actions are modeled to account for the different times available to complete the action and for the different procedural guidance in effect as a function of whether the sequence involves a LOCA or a steam generator tube rupture, whether high pressure injection is available; and, if the sequence involves a steam generator tube rupture, whether the secondary side has been initially isolated; i.e., whether Top Event SL is successful. The operator actions considered are listed below:

    - HADS1; no LOCA or steam generator tube rupture, cool down and depressurize to go on closed-loop RHR before exhausting CST and requiring makeup for continued AFW.

    - HADS2; following a steam generator tube rupture with successful isolation of ruptured steam generator, complete to avoid additional challenges to secondary valves and before CST makeup required.

    - HADS3; following a steam generator tube rupture with successful steam generator isolation but failure of all high-head injection, complete in time to prevent additional secondary valve challenges and before CST makeup is required.

    - HADS4; following steam generator tube rupture with a failure to isolate the ruptured steam generator, complete in time to prevent overfilling the steam generator and before makeup to CST is required.

- HADS5; following a steam generator tube rupture when the secondary is not isolated and all high pressure injection is lost, complete before core uncovery.

- HADS6; following a loss of all AC power, complete to limit RCP seal degradation and leakage for extended station blackouts.

- HADS7; following a small LOCA with failure of all high pressure injection, complete cool down and depressurization and establish closed-loop RHR before requiring sump swapover.

— **Conditions when Demanded.** Top Event DS is asked whenever there is a source of secondary heat removal available; i.e., when main feedwater or auxiliary feedwater are available after plant trip. For steam generator tube ruptures and for small LOCAs, it is used as the first step in the reduction of RCS pressure to reduce the break flow. For station blackouts, this action leads to a reduction in RCS pressure, which minimizes the challenges to the RCP seals, which are without cooling due to the station blackout.

— **Scenario Impact if Successful.** Success of Top Event DS means that the primary is cooled down, permitting it to also be depressurized. Success of Top Event DS is required for success of Top Event DP. For station blackouts, RCS pressure would then be reduced enough to permit accumulator injection. For steam generator tube ruptures with the secondary isolated (i.e., when Top Event SL is successful), success of Top Event DS implies that the RCS pressure would be reduced below the steam generator safety valve setpoints, so that no further challenges for them to open and reseat occurs. If Top Event SL fails, success of Top Event DS, together with success of Top Event DP, implies that RCS pressure is reduced enough to allow the operators to establish closed-loop RHR cooling. This then permits a cool down to cold shutdown in time to stop further leakage through the secondary before the RWST empties; i.e., so that makeup to the RWST is not required. If a small LOCA occurs, in addition to Top Event SL failure, this depressurization is assumed not to be in time to avoid the requirement for RWST makeup.

— **Scenario Impact if Failed.** Failure of this top event implies that RCS temperature remains high, precluding depressurization via Top Event DP. For station blackouts, failure of Top Event DS means that RCS pressure remains high, increasing the leakage through the RCPs seals than would otherwise occur. Failure of this top event in steam generator tube rupture sequences implies that RCS pressure remains above the steam generator safety valve setpoints, causing the continued loss of primary coolant into the environment. Therefore, failure of this top event implies that there is a leakage path via a failed-open valve on the ruptured steam generator, requiring continued high pressure injection for RCS inventory control.

- **Top Event DP — Depressurization of the Primary to RHR Entry Conditions**

— **Function Evaluated.** Depressurization of the primary system pressure.

- **Success Criteria.** RCS depressurization is accomplished using normal pressurizer spray or the pressurizer PORVs. No credit is taken for auxiliary spray or RCS head vents. Normal pressurizer spray requires opening a spray valve and successful operation of the associated RCS pump in loop 1 or loop 2. These pumps are assumed to be available if both offsite power and component cooling are available, provided that a Phase B isolation signal is not present. The success criterion for using the pressurizer PORVs to depressurize the RCS is that one of the two PORVs and associated block valves must open. Reclosure of the PORVs on successful depressurization is modeled separately in Top Event PI.

- **Model Boundaries.** Top Event DP models the equipment required for pressurizer spray, and the pressurizer PORVs and block valves. The operator action to initiate RCS pressurization is included in Top Event DS.

- **Conditions when Demanded.** Top Event DP is asked whenever there is a source of secondary heat removal available (i.e., when main feedwater or auxiliary feedwater are available after plant trip), and Top Event DS has been successful. For steam generator tube ruptures and for small LOCAs, the equipment modeled is used in the reduction of RCS pressure to reduce the break flow. For station blackouts, this action is not applicable because it is not called for by procedures, and all of the options to depressurize the RCS require power to operate.

- **Scenario Impact if Successful.** Success of Top Event DP in a steam generator tube rupture sequence implies that the RCS is depressurized below the setpoint of the secondary-side relief valves on the ruptured steam generator so that if the valves initially reclosed (i.e., success of Top Event SL) then there is no leak path through the secondary to the environment. The RCS pressure is also sufficiently reduced to permit RHR to be established.

  For small LOCA sequences, success of Top Event DP limits the break flow. Pressure would not, however, be reduced sufficiently fast to avoid the need for recirculation from the containment sump by going on closed-loop RHR because containment spray operation would empty the RWST in just a couple of hours; i.e., before closed-loop RHR could be established. For small LOCA sequences in which high pressure injection is successful but recirculation eventually fails, success of Top Event DP implies that RCS pressure is low at the time of vessel melt-through; i.e., it is assigned to a low pressure end state.

- **Scenario Impact if Failed.** Failure of this top event implies that RCS pressure remains high. Failure of this top event in steam generator tube rupture sequences implies that RCS pressure remains above the steam generator safety valve setpoints, causing the continued loss of primary coolant into the environment. Thus, failure of this top event implies that there will be a sustained loss of RCS inventory through the secondary side and that core damage will follow.

- **Top Event PI — Pressurizer PORVs are Isolated after RCS Depressurization**
  - **Function Evaluated.** Reclosure of pressurizer PORV after RCS depressurization.
  - **Success Criteria.** This top event models the successful reclosing of the PORV in the event that one was used for RCS depressurization in Top Event DP. If the PORV fails to reseat, manual closure of the associated block valve is also treated as success.
  - **Model Boundaries.** Top Event PI considers both the hardware required to isolate the pressurizer, and if necessary, the operator action to isolate the stuck-open PORV train by closing the block valve. The operator action included in this model is designated HAPI1, and is similar to action PR1. The action is to be completed in time to avoid rupturing the PRT rupture disk, which would initiate containment spray and eventually require recirculation from the containment sump.

    This event is evaluated conditionally on the probability that the PORVs were used in Top Event DP; i.e., the probability that the pressurizer spray was not available due to support system failures. The pressurizer PORV is not the preferred method for RCS depressurization.
  - **Conditions when Demanded.** This top event is asked whenever Top Event DP is successful. A pressurizer PORV is not assumed to have opened if Top Event DP has failed or the PORVs' support systems are unavailable. Top Event PI is guaranteed successful if the support systems for normal pressurizer spray are available and Top Event DP is successful.
  - **Scenario Impact if Successful.** No additional break flow would result. For steam generator tube ruptures, the break flow would be limited to the flow through the broken tube.
  - **Scenario Impact if Failed.** Failure of this top event implies that there is a small LOCA via the stuck-open pressurizer PORV used earlier to depressurize the RCS.

- **Top Event OB — Operators Go to Feed and Bleed Cooling**
  - **Function Evaluated.** Initiation of feed and bleed cooling.
  - **Success Criteria.** One charging or safety injection pump providing flow from the RWST through one cold leg injection path with both PORVs opened by the operators will provide adequate feed and bleed cooling if initiated within about 1 hour from the loss of secondary heat removal. If DC power is available to both PORVs initially, but the loss of an AC train results in the eventual loss of one DC train (i.e., after 4 hours when the battery is discharged), then it is assumed that from 4 hours on, only one pressurizer PORV is needed. Therefore, failure of one AC train does not preclude feed and bleed cooling.

- **Model Boundaries.** This top event models the availability of feed and bleed cooling using the high pressure injection systems and the pressurizer PORVs. The operator action to initiate safety injection and open the PORVs is included; i.e., action HAOB1. The hardware for the PORV is also included. The hardware failure modes for the high pressure injection systems are modeled elsewhere; i.e., as represented by Top Events VA, VB, VC, S1, S2, and SI.

- **Conditions when Demanded.** This event is asked if there is a total loss of secondary cooling; i..e., both main feedwater and auxiliary feedwater are failed. The event is guaranteed failed if either pressurizer PORV is unavailable, or if all four charging and safety injection pumps are unavailable.

- **Scenario Impact if Successful.** Core heat removal is successful. However, the pressurizer PORVs are assumed to be held open long enough that eventually recirculation from the containment sump is required.

- **Scenario Impact if Failed.** Failure of this top event implies that there is a complete loss of core heat removal with no possibility of depressurization for low pressure injection prior to core damage.

• **Top Events RA and RB — Residual Heat Removal Pumps 1A-A and 1B-B**

- **Function Evaluated.** Manual or automatic start and operation of the RHR pumps.

- **Success Criteria.** Success of each of these top events implies that the pump in that train is available to provide flow to the RHR discharge header to the RCS cold legs or to the high head systems during sump recirculation for 24 hours. Success of a train requires that engineered safety features actuation system (ESFAS) provide an automatic start signal, given a safety injection condition, or that the operators provide a manual start signal as a backup to the automatic start signal; i.e., via Top Event OS.

- **Model Boundaries.** These top events model the RHR pump trains 1A and 1B. The boundaries for these top events includes the pump suction valves, FCV-74-3 and FCV-74-21, through the discharge valves, FCV-74-16 and FCV-74-28. The RHR pump miniflow lines, pump room cooling and pump seal cooling are also included. The components within this boundary for train A (train B components are shown in parenthesis) include:

  • The pump suction valve FCV-74-3 (FCV-74-21).

  • Pump 1A-A (1B-B), discharge check valve 74-514, (74-515), and normally open manual valves 74-520 and 74-524 (74-521 and 74-525).

  • Mini flow line valve FCV-74-12 (FCV-74-24).

  • Heat exchanger outlet throttle valve FCV-74-16 (FCV-74-28).

- The heat exchanger's failure because of rupture.

- ERCW train A (B) supply to RHR pump 1A-A room cooler and the associated fan and valves.

- CCS cooling to the RHR pump 1A-A (.B-B) mechanical seals and the associated valves.

The action for the operators to reset the safety injection signal and stop the RHR pumps, if RCS pressure is > 180 psig to prevent pump overheating during extended operation on miniflow, is not modeled explicitly. The action also includes restarting the pumps as necessary for sump recirculation or normal RHR cooldown. This action is directed by the EOPs for safety injection conditions in which RCS pressure remains high; i.e., for all initiating events that use this event tree. However, realistically, for LOCAs, the time of swapover to the containment sump should occur before the time at which the pumps would overheat due to extended operation without CCS aligned to the RHR heat exchangers. The time to pump overheating is estimated as 100 minutes, which is after the time to sump swapover. Therefore, the action to stop the pumps is not required to protect the pumps at Watts Bar.

For feed and bleed sequences, the containment spray pumps are, like for small LOCAs, also assumed to come on early in the transient. Therefore, for feed and bleed sequences, sump recirculation is assumed to be required before any pump problems would develop due to extended operation on miniflow. Therefore, for feed and bleed sequences, this action is also not required.

Top Event RA is evaluated conditionally on the status of Top Event RB. This is to account for the intrasystem dependencies between trains; i.e., for common cause, test, and maintenance dependencies.

— **Conditions when Demanded.** These events are asked on nearly all sequences of the event tree. The status of both trains is asked, even though only one train is required, because the RHR pump trains must work together with the corresponding sump recirculation path.

For ATWS sequences involving a complete loss of secondary heat removal, and consequential failure of RCS pressure control, these top events are not asked; i.e., when Top Event SR fails, implying core damage due to gross failure of the RCS boundary.

If the RWST is unavailable, both trains of RHR are assumed to be failed.

— **Scenario Impact if Successful.** If Top Event RA or RB is successful, the corresponding train of RHR pumps is available. These top events are used in conjunction with the availability of the RWST (Top Event RW), the RHR injection path (i.e., Top Event RI), closed-loop RHR suction (Top Event RD), containment sump recirculation (Top Event RR), and RHR spray (Top Event RS) to evaluate the availability of each of the functions performed by the RHR pumps. For success of close-loop RHR, success of either Top Event RA or RB and of Top Event RI is required.

- **Scenario Impact if Failed.** Failure of both of these top events (RA and RB) implies that neither train of the RHR system is available.

- **Top Event RI — RHR Injection Path**

  - **Function Evaluated.** Availability of a flow path from the RHR pumps to the RCS.

  - **Success Criteria.** Success of Top Event RI requires that the suction line from the RWST to both RHR pumps be available and that a flow path to one of the four RCS cold legs be available for 24 hours.

  - **Model Boundaries.** This top event models the RHR flow paths from the pump discharge path modeled in Top Events RA and RB to the injection point in the four RCS cold legs downstream of check valves 63-551, 63-553, 63-555, and 63-557. The suction line to the RHR pumps from the RWST used only for RHR supply during safety injection (FCV 63-1 and check valve 63-502) is also included here. Therefore, use of this model for the closed-loop RHR function, which does not require suction from the RWST, is conservative The model includes the following components:

    - RWST to RHR pumps isolation valve FCV-63-1 and check valve 63-502.

    - RHR to cold legs isolation valves FCV-63-93 and FCV-63-94.

    - The outer check valve on each RHR cold leg injection path: 63-632, 63-633, 63-634, and 63-635.

    - The four RCS check valves, 63-560, 63-561, 63-562, and 63-563.

    The four check valves listed last above are also included in Top Event SI (i.e., the safety injection pumps suction and cold leg discharge path) since the RHR pumps and the safety injection pumps share common entry paths to the RCS. Therefore, the branch point values for Top Event RI are evaluated conditionally on the status of Top Event SI. If Top Event SI has failed, then a fraction of the time, this will have been due to the failure of the four common RCS check valves, which would also fail Top Event RI.

  - **Conditions When Demanded.** Top Event RI is asked whenever Top Event RA or RB is successful.

  - **Scenario Impact if Successful.** The RHR pump trains are then available to operate in the closed-loop RHR cooling mode.

  - **Scenario Impact if Failed.** Failure of this top event implies that RHR is not available to directly inject into the RCS cold legs. This precludes closed-loop

RHR cooling. Depending on the status of other top events, RHR may still be available to supply the SIS and CVCS pumps for high pressure injection in the sump recirculation mode, or for containment spray recirculation. This event is guaranteed failed if both Top Events RA and RB are failed or if Top Event RW is failed.

- **Top Event RD — Normal RHR Cooldown and Charging**

  - **Function Evaluated.** Normal, or closed-loop, RHR cooling.

  - **Success Criteria.** Given that the RCS has been cooled down and depressurized (i.e., success of Top Events DS and DP), successful entry to closed-loop RHR cooling requires that the RCS be further cooled and RCS pressure reduced. Normal charging and letdown from the RHR system are required to achieve these conditions promptly. Closed-loop RHR cooling then requires that the common RHR hot leg suction line be available, that at least one RHR pump train take suction, with cooling to its associated heat exchanger, and that the pump discharge be directed to the cold legs via at least one of the four lines. Two hours are assumed available to establish RHR before the RWST is emptied, once RHR entry conditions are reached.

  - **Model Boundaries.** Top Event RD models the equipment and operator actions necessary for a continued normal RHR cooldown once the RCS has been cooled down and depressurized sufficiently to allow RHR to be placed in service.

    The model includes the equipment and operator actions used for (1) the common RHR suction line hot leg isolation valves that allow flow to the RHR pump trains, (2) the heat transfer function of the RHR heat exchangers, and (3) the equipment and operator action required to establish normal charging and letdown to achieve the RHR entry conditions following the rapid RCS cooldown and depressurization modeled in Top Events DS and DP. Top Event RI models the return of the coolant to the four RCS cold legs, and Top Events RA and RB model the availability of the RHR pump trains themselves.

    This model includes the following equipment:

    - The series RHR suction isolation valves FCV-74-1 and FCV-74-2.

    - Suction isolation bypass valves FCV-74-8 and FCV-74-9.

    - The RWST isolation valves FCV-63-1 and 63-504, which must remain close.

    - The CCS cooling water and associated valves to the RHR heat exchangers.

    - Letdown from the RHR cleanup line 1-74-530 or 1-74-531 and 1-FCV-62-83.

- The letdown path from the letdown heat exchanger to the volume control tank.

- The charging path from the intersection of the charging line and the seal injection line to the RCS loop 1.

This top event also contains the operator actions (i.e., designated HARD1) required to align for normal RHR cooldown.

— **Conditions when Demanded.** Top Event RD is asked when there is a break in the RCS boundary, the RCS has been successfully cooled down and depressurized (i.e., Top Events DS and DP are successful), at least one RHR train is available (i.e., Top Event RA or RB is successful), and the cold leg injection path is available; i.e., Top Event RI is successful. RHR cooldown is required for sequences involving a steam generator tube rupture in which the leakage to the secondary has not been isolated; i.e., Top Event SL is failed.

Closed-loop RHR cooling during a small LOCA (i.e., if Top Event PR, PI, or SE fails) is assumed to be insufficient at limiting the RCS break flow to avoid the need for recirculation from the containment sump. This is because the containment spray pumps will have emptied the RWST even before the RHR entry conditions are reached. Therefore, closed-loop RHR is assumed to be unsuccessful for small LOCAs. Similarly, close-loop RHR is assumed ineffective for mitigation of ATWS sequences that result in a small LOCA or induced steam generator tube rupture. Currently, credit for Top Event RD is only taken for steam generator tube rupture initiating events.

— **Scenario Impact if Successful.** Success of this event implies that cooldown to cold shutdown conditions is successful. In steam generator tube rupture sequences, this minimizes and controls the leakage from the RCS, avoiding the need for recirculation from the sump due to loss of RCS inventory through the secondary side.

— **Scenario Impact if Failed.** Failure to go to normal RHR, given an unisolated steam generator tube rupture, means that continuous makeup must be supplied to the RWST and injected to the RCS to maintain inventory.

3.1.2.2.2 RECIRC Event Tree (RISKMAN Designator: RECIR2)

The late or recirculation phase responses of the Watts Bar Nuclear Plant to transient initiators, small LOCAs, ATWS events, steam line breaks, and SGTRs are modeled using the RECIRC event tree. This event tree is illustrated in Figure 3.1.2-4.

The logical structure of the tree is such that it can be specialized to handle any of the above initiators. The tree includes various success paths that satisfy the major core protection functions not already covered in the GENTRANS event tree: i.e., long-term inventory control, core heat removal, containment radioactivity removal, containment pressure suppression and heat removal, and containment isolation.

A detailed discussion of the top events in the RECIRC tree follows. The top events are described in the order in which they appear in the event tree. The top events in this tree are separated from those in the GENTRANS tree for presentation purposes only. For every initiating event quantified using the GENTRANS tree, this second tree is also used.

A detailed discussion of each of the top events in the RECIRC tree follows.

- **Top Event CM — Core is Not Damaged at Entry into the Recirculation Event Tree**

  - **Function Evaluated.** The occurrence of core damage during the early part of the accident sequence; i.e., during the injection phase.

  - **Success Criteria.** The initiator and sequence of events up to this point in the trees are such that core damage has not yet occurred.

  - **Model Boundaries.** This top event is a switch, whose success or failure is based solely on the status of earlier events in the sequence. There are no equipment failure modes or operator actions modeled in this top event. The branch probability takes on values of zero or one.

    This event is failed if the early part of the accident sequence involves (1) a failure of feed and bleed cooling in response to a loss of all secondary heat removal, (2) a LOCA with failure of high pressure injection from both charging and safety injection, (3) a reactor trip failure with inadequate RCS pressure relief or failure of emergency boration, or (4) an SGTR with either RCS depressurization or closed loop RHR failed and makeup to the RWST failed.

  - **Conditions when Demanded.** This event is asked for all sequences entering the RECIRC tree.

  - **Scenario Impact if Successful.** Success implies that core damage did not occur in the early phases of the accident sequence. Many of the same questions are then asked, as if core damage were suffered in the early phases. However, it is possible that no core damage would occur, and that, therefore the status of containment isolation (i.e., Top Event CI) and of the hydrogen igniters (i.e., Top Event HH) may not be significant. Additional failures must occur for this sequence to end in core damage.

  - **Scenario Impact if Failed.** Failure implies that core damage did occur in the early phases of the accident sequence. All sequences involving failure of this top event must be assigned to a core damage end state. The status of containment isolation and of the hydrogen igniters is therefore of significance.

- **Top Event RQ — Sump Recirculation is Not Required**

  - **Function Evaluated.** The need for recirculation from the containment sump.

- **Success Criteria.** The initiator and the sequence of events up to this part of the trees are such that there is no need for recirculation from the sump. The plant is in a stable configuration, with no breach of the RCS boundary and secondary heat removal available.

- **Model Boundaries.** This top event is a switch, whose status is based solely on the status of earlier events in the sequence. There are no equipment failure modes or operator actions modeled in this top event. The branch probability takes on values of zero or one.

  This event is guaranteed failed if in the earlier tree, (1) a LOCA occurred, either as the initiator or due to the plant response (e.g., Top Event PR, PI, or SE fails), (2) reactor trip fails in which case a LOCA through a stuck-open pressurizer relief valve is assumed to occur, or (3) there is a loss of all secondary cooling, requiring feed and bleed cooling. Core damage sequences involving a small LOCA with failure of high pressure injection are a subset of the above-defined conditions, and would also result in failure of this top event.

- **Conditions when Demanded.** This event is asked for all sequences coming from the GENTRANS event tree that have not already resulted in core damage.

- **Scenario Impact if Successful.** Success implies that the plant is stable, with no breach of the RCS boundary and secondary heat removal is available. Such sequences are mapped to the success end state. No other events in the recirculation event tree then need be asked.

- **Scenario Impact If Failed.** Failure implies that the status of containment systems and of those systems and actions needed for recirculation from the sump are of interest in terms of mitigating a LOCA or in response to a core melt.

• **Top Event IC — Ice Condenser**

- **Function Evaluated.** Ice condenser availability.

- **Success Criteria.** The primary function of the ice condenser is, along with the containment spray system, the removal of the initial energy and pressure suppression during a LOCA and long-term pressure relief events. Pressure increases in the lower containment will force open the inlet doors to the ice condenser. This allows the heated steam/air mixture up through the ice. For successful energy removal and pressure suppression, the break flow must open the doors and the steam in the lower compartment then be directed through the ice.

- **Model Boundaries.** The availability of the ice condenser under accident conditions is governed by the possibility of bypass of the ice condenser. This is similar to the method of analysis used in NUREG/CR-4551 for the Sequoyah Nuclear Plant. No analysis of the systems for the formation and maintenance of the ice bed is considered. Plant Technical Specifications preclude plant operation without adequate ice.

- **Conditions when Demanded.** The ice condenser top event is asked everywhere in the recirculation event tree in which recirculation from the containment sump is required.

- **Scenario Impact if Successful.** The ice condenser is available to remove energy transferred to containment by the break flow, limiting containment pressure.

- **Scenario Impact if Failed.** The ice condenser is not available to limit containment pressure.

- **Top Event CP — Containment Purge Isolation**

  - **Function Evaluated.** Isolation of the containment purge lines.

  - **Success Criteria.** Success of this top event requires that either (a) the purge system was not in use when required or, (b) it was in use and that at least one valve in each penetration line closed.

  - **Model Boundaries.** This top event models the isolation of the containment purge penetrations, which are allowed to be opened during power operation. The Plant Technical Specifications allow these penetrations to be opened up to 1,000 hours per year with the plant at power. The penetrations modeled are as follows:

    - Lower compartment purge air exhaust (X-4).
    - Instrument room purge air exhaust (X-5).
    - Upper compartment purge air exhaust (X-6).
    - Upper compartment purge air exhaust (X-7).
    - Upper compartment purge air supply (X-9A).
    - Upper compartment purge air supply (X-9B).
    - Lower compartment purge air supply (X-10A).
    - Lower compartment purge air supply (X-10B).
    - Instrument room purge air supply (X-11).

    The penetrations modeled by CP are treated separately from those in Top Event CI due to the larger size of the purge penetrations. Even though the large size of the purge penetrations may limit the containment pressure rise during a LOCA, the Phase A and Phase B isolation signals setpoints are low enough that these isolation signals should occur even if the purge lines are initially open.

    A backup manual action to isolate these penetrations is also considered, per the status of Top Event OS.

  - **Conditions when Demanded.** Top Event CP is asked in all sequences of the recirculation event tree when recirculation from the sump is required. Top

Event CP may still be successful, even if there no automatic or manual closure signals, because the penetrations included in the model are usually closed anyway.

- **Scenario Impact if Successful.** Success of Top Event CP implies that the containment has at most a small hole in it. If Top Event CI is also successful, then the containment is isolated.

- **Scenario Impact if Failed.** Failure of this top event implies that containment isolation has failed and that a large hole in the containment boundary is present.

- **Top Event AR — Containment Air Return Fans**

  - **Function Evaluated.** Performance of the containment air return fans.

  - **Success Criteria.** The success criterion for the air return fans is that one of the two fans function for 24 hours. The fans are automatically actuated on a high-high containment pressure signal from ESFAS.

  - **Model Boundaries.** This top event models the containment air return fans. The air return fans hot circulate saturated air from the upper compartment after a LOCA (10 minutes after high-high containment pressure). The air return fans enhance heat removal from the lower compartment to the ice condenser to help lower containment pressure. All portions of the air return fan functions are modeled. A manual start action to back up the ESFAS actuation is not modeled.

  - **Conditions when Demanded.** Top Event AR is asked for all sequences in the recirculation event tree when recirculation from the sump is required. All LOCAs are assumed to increase containment pressure initially to a high-high pressure condition.

  - **Scenario Impact if Successful.** The lower compartment containment pressure rise is mitigated by the successful operation of the air return fans.

  - **Scenario Impact if Failed.** The lower compartment containment pressure is not mitigated, and local hydrogen pockets may develop due to poor mixing.

- **Top Events CSA and CSB — Containment Spray Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Containment spray injection and pump availability for recirculation.

  - **Success Criteria.** The associated containment spray pump train is automatically or manually actuated, and provides spray injection. The pump operates for 24 hours. Manual actuation must be completed within 20 minutes of reaching the automatic actuation setpoint.

- **Model Boundaries.** This top event models the availability of the containment spray pump trains 1A-A and 1B-B to deliver flow into containment, given a suction source. In the injection mode, suction is from the RWST (Top Event RW), and during recirculation, suction is from the containment sump (Top Event CH). Heat removal from the containment spray heat exchangers is modeled separately in Top Event CH. Top Events CSA and CSB are both actuated automatically by a Phase B isolation signal, which is assumed to be reached in this analysis for all small LOCAs.

  The equipment modeled in Top Event CSA (CSB equipment in parenthesis) includes:

  - The containment isolation valve FCV-72-39 (FCV-72-2).
  - Containment spray pump 1A-A (1B-B).
  - Miniflow valve 1-72-34 (1-72-13).
  - The RWST suction valve FCV-72-22 (FCV-72-21).
  - Normally open manual valve 1-72-528 (1-72-529) and check valves 1-72-506 (1-72-507), 1-72-524 (1-72-525), and 1-72-547 (1-72-548).
  - Containment sump recirculation valve 1-FCV-72-44 (1-FCV-72-45), which must remain closed.
  - The physical integrity of the containment spray heat exchanger 1A (1B), CCS supply to the containment spray pump 1A-A (1B-B) oil coolers including the associated valves.
  - ERCW supply to the containment spray pump 1A-A (1B-B) room coolers including the associated valves.
  - The 263 nozzles of the spray header 1A (1B).

  The operator action to manually initiate containment spray, given a high-high containment pressure condition but failure of the automatic actuation signal from ESFAS, is included in the models for both top events; i.e., via action HACS1.

  Train B of containment spray, as represented by Top Event CSB, is evaluated conditionally on the status of Top Event CSA, to reflect the potential for common cause failures affecting both trains and the maintenance limits imposed by Plant Technical Specifications.

- **Conditions when Demanded.** Top Events CSA and CSB are asked for all sequences where recirculation is required in the recirculation tree. Both trains of containment spray are unavailable if the RWST is not available.

- **Scenario Impact if Successful.** The containment spray system acts with the ice condenser system to provide containment heat removal and to limit the containment pressure increase. Containment spray provides a long-term source of containment heat removal while in the sump recirculation mode. The long-term function of containment spray (i.e., in the recirculation mode) requires the success of at least one of CSA or CSB and of Top Event CH, which is considered next.

- **Scenario Impact if Failed.** Failure of Top Event CSA or CSB implies that the associated train of containment spray is not available in either the injection mode or the subsequent recirculation mode. Even if both trains fail, the containment would still not overpressurize during the injection phase for any size LOCA.

- **Top Event OT — Operator Controls/Terminates Containment Spray**

  - **Function Evaluated.** Manual control/termination of containment spray.

  - **Success Criteria.** The operators must terminate containment spray once the containment pressure drops below the required value. Five minutes are assumed to be available to complete the action once pressure is reduced to permissible levels.

  - **Model Boundaries.** The operators are directed to stop containment spray after the containment pressure is reduced to less than the required value. This is accomplished by

    - Resetting the containment spray signal.
    - Stopping the containment spray pumps and placing them in automatic.
    - Closing the discharge valves (FCV-72-2 and FCV-72-39) and placing them in automatic.
    - Ensuring the miniflow lines (FCV-72-13A and FCV-72-34A) are in automatic.

    This action is included in the model for small LOCAs with a designation of HAOT1. The action is not modeled for larger break sizes because, once spray is terminated, pressure would increase and the pumps would just restart once the high-high containment pressure setpoint was again reached.

  - **Conditions when Demanded.** This event is asked for all sequences in the recirculation event tree that require recirculation from the containment sump; i.e., when Top Event RQ fails. This event is assumed to be guaranteed successful if both containment spray pumps have failed; i.e., if both Top Events CSA and CSB are failed.

- **Scenario Impact if Successful.** Control of containment spray, once containment pressure is reduced, is an important consideration in the determination of the time to empty the RWST, which, in turn, affects the time available to accomplish swapover to recirculation from the containment sump. In the current model, the operator action to complete the recirculation swapover is evaluated conservatively; i.e., no credit for this extended time and reduced RWST drawdown rate is assumed.

- **Scenario Impact if Failed.** The containment spray pumps continue to operate, drawing water from the RWST and injecting it into the containment. Failure to terminate spray reduces the time needed to empty the RWST, and also the time available for completing the swapover to recirculation. In the event of a plugging of the upper compartment drain plug, continued operation of the spray pumps will also fail recirculation from the sump because all of the water will eventually be transferred to the upper compartment.

- **Top Event SU — Containment Sump Available**

  - **Function Evaluated.** The availability of water in the containment sump for recirculation.

  - **Success Criteria.** Given a loss of RCS inventory, success of Top Event SU requires that water is available for recirculation from the containment sump at the time that the RWST empties. A flow path for draining the containment spray water from the upper to the lower flow compartment must be available, and the containment sump itself must not be plugged. Water must have been injected directly or indirectly into the containment by one or more of the ECCS pumps; i.e., by the charging, safety injection, RHR, or containment spray pumps. Melting of the ice alone is not sufficient to permit recirculation from the containment sump.

  - **Model Boundaries.** This top event models the availability of the containment sump for recirculation. Failure of the sump can occur by the plugging of the drain between the upper and lower containment compartments or by plugging of the sump with containment debris. Containment sump debris plugging is evaluated conditionally on whether core damage has occurred.

    The availability of the sump is modeled as the probability that the drain plugs were not removed during the last shutdown or that debris has blocked the sump. The containment spray sumps discharge into the upper containment. Melted ice and discharge from breaks in the RCS boundary are initially directed into the lower compartment. However, functioning spray sumps with failure of the drain plugs or of containment sump plugging will guarantee failure of recirculation.

  - **Conditions when Demanded.** The containment sump Top Event, SU, is asked everywhere in the recirculation event tree when recirculation from the containment sump is required.

- **Scenario Impact if Successful.** Water is then assumed to be available for recirculation from the sump.

- **Scenario Impact if Failed.** RHR recirculation from the sump and containment spray recirculation from the sump are not available if Top Event SU fails. Therefore, Top Events RL, RVA, RVB, RR, CH, RS, and RH are not asked if Top Event SU fails. Containment spray may still operate in the injection mode, taking suction from the RWST until it empties. Continued RCS inventory control may be provided by providing makeup to the RWST and staying on high head injection.

- **Top Event RL — Recirculation Level Instrumentation**

  - **Function Evaluated.** Instrumentation for swapover from RWST to containment sump suction.

  - **Success Criteria.** Two of the four channels of level instrumentation must be available to make up the low level swapover signal. The swapover is initiated when RWST level is less than the low-level setpoint, containment sump level greater than the required value during a LOCA, and a safety injection signal is present.

  - **Model Boundaries.** This top event models the instrumentation necessary for switchover from the injection mode to the recirculation mode. The instrumentation modeled includes:

    - The RWST level switches LS-63-50D, LS-63-51D, LS-63-52D, and LS-63-53D.

    - The containment sump level switches LS-63-180, LS-63-181, LS-63-182, and LS-63-183.

    An operator action is modeled to back up the level transmitters; i.e., action HARL1. This action is applicable for conditions when automatic swapover instrumentation fails due to conditions that can be easily detected; e.g., freezing of the sensor lines, a common relay failure, or if the operators had previously reset the safety injection signal, preventing the automatic swapover signal.

  - **Conditions when Demanded.** Top Event RL is asked for all sequences in the recirculation event tree when the containment sump is available; i.e., when Top Event SU is successful.

  - **Scenario Impact if Successful.** This implies that the automatic actuation signal for pump suction swapover to the containment sump is available to both trains of valves.

  - **Scenario Impact if Failed.** Failure of this top event implies that automatic and manual swapover has failed. Credit is only given for initiation of manual swapover if automatic swapover fails to actuate due to relay failure or gross transmitter failure; i.e., RWST level transmitter freezes. Therefore, failure of Top Event RL implies that recirculation from the containment sump for both core cooling and RHR spray recirculation are unavailable.

- **Top Events RVA and RVB — Train A and B RHR Sump Swapover Valves**

  - **Function Evaluated.** Response of ECCS valves to a pump suction swapover demand from the RWST to the containment sump.

  - **Success Criteria.** The containment sump suction valve must open, and the RWST suction valve on the associated train must close for success of one train.

  - **Model Boundaries.** These top events model the train A and B swapover valves of the RHR system which realign automatically to allow pump suction from the containment sump once the RWST level is low. The swapover is initiated by the instrumentation modeled in Top Event RL on RWST level less than the low-level setpoint and containment sump level greater than the required value during a LOCA. The equipment modeled includes the motor operated sump swapover valves 1-FCV-63-72 for train A and 1-FCV-63-73 for train B, which must open, and 1-FCV-74-3 for train A and 1-FCV-74-21 for train B, which must close. The valve changes necessary for high pressure recirculation are not modeled in this top event. They are instead modeled in Top Event RR.

    Top Event RVB is evaluated conditionally on the status of Top Event RVA. This is because the suction valves in the different trains, both to the RWST and to the containment sump, are in common cause groups. Therefore, the failure probability of the second train is higher if the first train fails than would otherwise be expected, assuming that the trains were independent. These dependencies are captured in the systems models.

  - **Conditions when Demanded.** These events are asked for sequences in the recirculation event tree in which Top Event SU is successful. Failure of the containment sump, makes the question of valve swapover immaterial. If the swapover instrumentation is unavailable (i.e., Top Event RL fails), then both of these events are set to guaranteed failed.

  - **Scenario Impact if Successful.** Success of these valves means that the containment sump suction path is aligned for recirculation on the associated train. For mitigation of medium LOCAs, only low pressure recirculation is required for core cooling; i.e., high pressure recirculation using the safety injection or charging pumps is assumed to be unnecessary because the RCS pressure should be low by the time that the RWST reaches low level.

  - **Scenario Impact if Failed.** Failure of these valves to go to the correct position will fail the corresponding RHR pump train during the recirculation mode.

- **Top Event RR — RHR Recirculation**

  - **Function Evaluated.** This top event models the automatic/manual swapover of the RHR pumps' suction to the containment sump for high pressure recirculation.

  - **Success Criteria.** Successful RHR recirculation mode requires: (1) the automatic transfer of the suction of RHR from the RWST to the containment sump (Top Events RL, RVA, and RVB), (2) manual transfer of the suction of the safety injection pumps and centrifugal charging pumps from the RWST to the discharge of the RHR pumps, (3) manual restart of the RHR pumps, if they were shut down due to RCS high pressure during injection mode, and (4) use of the RHR heat exchangers to remove heat from the containment sump recirculation coolant. Either of two RHR trains supplying flow to at least one operable charging or safety injection pump is considered success. The pumps taking suction from the RWST must be stopped when RWST level drops to the low-low level setpoint, which is estimated to be within 10 minutes of the low-level setpoint condition, and the swapover must be completed within 20 minutes to prevent core uncovery.

  - **Model Boundaries.** The RHR configuration during recirculation from the sump allows: (1) the RHR pumps to draw coolant from the containment sump, (2) the RHR heat exchangers to transfer heat from the coolant to the CCS, (3) and the coolant to be pumped to four discharge paths: (1) the cold legs of the RCS, (2) the inlet of the centrifugal charging pumps, (3) the inlet of the safety injection pumps, and (4) for RHR containment spray.

    The recirculation mode is initiated automatically when the RWST reaches a low level coincident with a high containment sump level, as modeled in Top Event RL. The RHR pumps will then automatically switch their suction from the RWST to the containment sump as modeled in Top Events RVA and RVB.

    During recirculation, component cooling water flow is manually established to the RHR heat exchangers to cool the flow from the containment sump before being discharged into the RCS, the suction of the safety injection or charging pumps, or for RHR.

    RHR valves (FCV-63-8 and FCV-63-11) are manually opened to establish a flow path from the RHR pump discharge to the suction of the centrifugal charging pumps and safety injection pumps for high pressure injection. These valves (FCV-63-8 and FCV-63-11) are interlocked with the closed position of the safety injection pump miniflow valves and the safety injection discharge path to the RWST (isolation valves FCV-63-3, FCV-63-4, and FC-63-175) to ensure that sump coolant is not diverted to the RWST.

    The model for this top event includes the following equipment:

    - The safety injection pump miniflow valves (FCV-63-3, FCV-63-4, and FCV-63-175).

- The charging, safety injection and RHR suction valves from the RWST (LCV-62-135, LCV-62-136, FCV-63-1, and FCV-63-5).

- The component cooling water and control valves to the RHR heat exchangers (FCV-70-153, and FCV-70-156).

The recirculation valves for the swapover of the containment spray pumps are modeled separately in Top Event CH.

Top Event RR is used with the containment sump availability Top Event SU, the safety injection system Top Events S1, S2, and IP; the charging system Top Events VA, VB, and VF; and the RHR pump, flow path, and RHR spray Top Events RA, RB, RF, and RS to determine that flow is available to the RCS from one of four high pressure pumps (centrifugal charging pumps or safety injection pumps) during high pressure recirculation or one of two RHR pumps during low pressure recirculation.

High pressure recirculation to avoid core damage is modeled as being required for all sump recirculation sequences in the recirculation tree. Rapid RCS cooldown and depressurization (Top Events DS and DP successful), as called for by procedures, might reduce RCS pressure prior to emptying the RWST. However, no thermal-hydraulic analysis is available to demonstrate this, so the conservative assumption is made that all sequences in this tree that do not involve core damage during the injection phase require high pressure recirculation. After core damage and eventual vessel penetration, RCS pressure should be low. Therefore, only low pressure recirculation is required for containment heat removal after core damage.

The operator action to align the CCS to the RHR heat exchangers and to make the valve alignments from the RHR pump train discharge to the high pressure pumps is modeled in this top event. Power must also be restored to FCV-63-1, so that this RWST suction valve may be isolated. The action is designated as HARR1. Credit for this action is only modeled if RWST level instrumentation is available; i.e., Top Event RL succeeds. The action evaluated conditionally on the failure of Top Event RL (i.e., action RR2) is not used.

— **Conditions when Demanded.** Whenever the containment sump is available for sump recirculation (i.e., Top Event SU succeeds), this event is asked. If both RHR pumps (RA and RB), or both sump swapover valve trains fail (i.e., Top Events RVA and RVB fail), or if high pressure recirculation is required and none of the high pressure pumps are available, then this event is guaranteed failed.

— **Scenario Impact if Successful.** Implies that there has been a successful transition to high pressure recirculation from the containment sump. Discharge from one or both RHR pumps, taking suction from the sump, is directed to the suction of at least one operable high pressure injection pump, which then injects into the RCS.

- **Scenario Impact if Failed.** Failure of this top event implies that there is no recirculation cooling of the core, and that RHR spray recirculation is not possible.

- **Top Event CH — Containment Spray in Recirculation Mode**

  - **Function Evaluated.** Recirculation mode of containment spray.

  - **Success Criteria.** Success of this event requires that at least one train of containment spray is successful. Success of the containment spray train requires successful swapover of the containment spray swapover valves, and of the associated spray pumps to operate in the injection mode. ERCW cooling must be aligned to the containment spray heat exchangers. The action to accomplish the swapover (action HACH1) is initiated when RWST is less than the low-low level setpoint and containment pressure is greater than the high-high pressure setpoint. It must be completed before the pumps lose suction due to the RWST emptying, assumed to be about 5 minutes after the low-low level setpoint is reached.

  - **Model Boundaries.** This top event models the operator actions and equipment required for containment spray to successfully operate in the recirculation mode.

    The equipment modeled in Top Event CH includes:

    - The containment isolation valves FCV-72-39 and FCV-72-2.
    - The train A and train B containment spray pumps (CSA and CSB).
    - The RWST suction valves FCV-72-22 and FCV-72-21.
    - The containment sump suction valves FCV-72-44 and FCV-72-45.
    - ERCW cooling to the containment spray heat exchangers.
    - ERCW heat exchanger inlet isolation valves FCV-67-125 and FCV-67-123.
    - ERCW heat exchanger outlet check valves 67-537A and 67-537B.
    - ERCW heat exchanger outlet isolation valves FCV-67-126 and FCV-67-124.

    The operator action to align the ERCW supply to the containment spray heat exchangers and to transfer the suction of the containment spray system from the RWST to the containment sump is modeled with this top event. The operators must stop the operating containment spray pumps, close the pump suction valves from the RWST, and open the pump suction valves from the sump. The operator then checks for adequate ERCW flow and cooling to the core spray heat exchangers, and restarts the spray pumps.

These actions are initiated when the RWST level reaches the low-low level setpoint.

- **Conditions when Demanded.** Top Event CH is asked for all sequences in the recirculation event tree for which the containment sump is available; i.e., when Top Event SU succeeds. The containment spray pumps in the injection mode (i.e., Top Events CSA and CSB) must be successful for the corresponding train of spray to operate in the recirculation mode.

- **Scenario Impact if Successful.** Success of this top event implies that at least one train of containment spray is operating, taking suction from the containment sump, with its associated heat exchanger being properly supplied with ERCW. Containment heat removal is available. Spray recirculation from the RHR pumps discharge is not required.

- **Scenario Impact if Failed.** If this top event is not successful, containment heat removal requires successful operation of RHR spray, as modeled in Top Event RS.

• **Top Event RS — RHR Spray**

- **Function Evaluated.** Spray recirculation using RHR pump discharge.

- **Success Criteria.** One of the two RHR pump trains must operate in the recirculation mode. If containment pressure exceeds the required value more than 1 hour into the accident, the operators then align one train of RHR for containment spray recirculation. The alignment must be made before the containment overpressurizes, estimated to be several hours into the accident.

- **Model Boundaries.** This event models the containment spray function of the RHR system. It includes the operator action and the opening of the RHR ring header inlet motor-operated valve. The operators establish one (and only one) train of RHR spray if containment pressure is greater than the required value and the accident is at least 1 hour old. The action included in the model is designated HARS1.

  The equipment modeled in this top event includes:

  - RHR crosstie valves FCV-74-33 and FCV-74-35.
  - The RHR injection path isolation valves FCV-74-93 and FCV-74-94.
  - The RHR spray control valves FCV-72-40 and FCV-72-41.

- **Conditions when Demanded.** This event is asked only if normal containment spray recirculation, as represented by Top Event CH, has failed. It is used with the containment sump availability Top Event SU, the containment sump recirculation Top Event RR, and the RHR pump train Top Events RA and RB. It is not asked if Top Event RR has failed. Top Event RR asks about the valves for alignment of RHR recirculation, and of the alignment of valves to establish CCS flow to the RHR heat exchangers for heat removal.

Credit for this action is taken if at least two RHR trains, at least one charging pump, and at least one safety injection pump are available. Procedural guidance only instructs the operators to use RHR spray if all of these conditions are met.

- **Scenario Impact if Successful.** Thus success of this top event, RS, implies that at least one train of RHR is providing containment spray (provided that Top Events SU, RR, and RA and RB are successful).

- **Scenario Impact if Failed.** Failure of this top event implies that containment heat removal from spray systems is not available.

- **Top Event CI — Containment Isolation**

  - **Function Evaluated.** Isolation of small containment penetrations.

  - **Success Criteria.** Each of the nonessential, small containment penetrations listed below must be either closed at the time of the accident and remain closed, or close by a signal from ESFAS based on a safety injection condition, or on Phase B isolation. For station blackout sequences, the time assumed to be available to locally isolate the seal return line is 3 hours.

  - **Model Boundaries.** This top event models containment isolation of nonessential penetrations during accident conditions. The containment penetrations explicitly modeled are as follows:

    - Containment major vents and drains.

    - Connections to the RCS.

    - Connections to containment atmosphere, with the exception of the large penetrations modeled in Top Event CP.

    This containment isolation top event models only those containment penetrations whose failure to isolate would result in a release path that would bypass containment. The following questions were asked about each penetration to determine the need for inclusion. Only those penetrations not covered by other system analyses in the PRA were considered.

    - Does the penetration communicate directly with the outside environment?

    - Does the penetration communicate with the environment via a low pressure system or a tank with a relief valve?

    - Will the relief valve lift at a pressure below the ultimate containment pressure?

    - Is the system or tank design pressure below the ultimate containment pressure?

Based on these questions, the following penetrations are included in this top event:

- Floor sump pump discharge (X-41).
- RC drain tank and pressurizer vent to VH (X-45).
- RC drain tank pump discharge (X-46).
- Lower compartment pressure relief (X-80).
- RC drain tank to gas analyzer (X-81).
- Upper compartment air monitor intake (X-94A/B).
- Upper compartment air monitor return (X-94C).
- Lower compartment air monitor intake (X-95A/B).
- Lower compartment air monitor return (X-95C).
- RCP seal return line.

All of these penetrations receive a signal to isolate, given a safety injection Phase A or CVI condition, except for the RCP seal return line. The RCP seal return line is automatically signaled to close on high-high (Phase B) containment pressure. During station blackout conditions, the operator is required to locally isolate the motor-operated seal injection and return valves in the RCP seal return line. This action (i.e., action CI1) is included in the model.

– **Conditions when Demanded.** Top Event CI is asked for every sequence in the recirculation event tree that involves core damage. The status of containment isolation is needed for the containment analysis. Top Event CI is not asked if recirculation from the containment sump provides core cooling, or if makeup to the RWST allows continued high pressure injection via the charging or safety injection pumps.

– **Scenario Impact if Successful.** The smaller containment penetrations modeled in Top Event CI are isolated.

– **Scenario Impact if Failed.** One or more of the smaller containment penetrations listed above must have been opened initially and failed to close.

- **Top Event HH — Hydrogen Igniters**

  – **Function Evaluated.** Hydrogen control using the hydrogen igniters.

  – **Success Criteria.** Success of this top event implies that all 34 igniters in one of two trains have functioned.

  – **Model Boundaries.** This top event models the hydrogen igniters of the hydrogen mitigation system. During events that involve fuel cladding damage, the hydrogen igniters are used to burn away the hydrogen before it reaches explosive concentrations when it mixes with the containment atmosphere.

  The system consists of two trains of hydrogen igniters and the associated control circuitry. The system is manually initiated from the control room upon receipt of a Phase B signal and hydrogen concentration as indicated by the hydrogen analyzer is within the required range. The operator is also required to place the hydrogen analyzer in service. The action modeled is designated DHAHH1. There is no time pressure to complete this action; i.e., many hours are assumed to be available. Use of the hydrogen analyzer is included as part of the operator action to initiate the hydrogen igniters. During recovery from an initial station blackout, the hydrogen igniters are not to be placed in service if the hydrogen analyzers indicate that the hydrogen concentration exceeds the required value.

- **Conditions when Demanded.** The hydrogen igniters are asked for in all sequences of the recirculation event tree involving core damage. The status of Top Event HH is used in the evaluation of containment performance. Top Event HH is not asked if recirculation from the containment sump provides core cooling, or if makeup to the RWST allows continued high pressure injection via the charging or safety injection pumps.

- **Scenario Impact if Successful.** The hydrogen igniters are available to continuously burnoff the hydrogen which collects in the containment, prior to the concentration of hydrogen reaching explosive concentrations.

- **Scenario Impact if Failed.** The hydrogen igniters are not available to reduce the concentration of hydrogen within containment.

3.1.2.2.3 MLOCA Event Tree (RISKMAN Designator: MEDLOC2)

The response of the Watts Bar plant to a medium LOCA is quantified using a single event tree constructed specifically for the medium LOCA initiating event. This tree contains all of the top events required to mitigate the effects of the medium LOCA.

A medium LOCA is defined in this analysis as a break in the primary system with an equivalent break size between a 2- and 6-inch-diameter hole. This size is sufficiently large that injection into the RCS is the primary concern with the status of the ability of the steam generators to remove decay heat of no consequence.

The MLOCA event tree models the plant response to a medium LOCA. This includes RCS injection by the centrifugal charging pumps, safety injection pumps, cold leg accumulators, and the residual heat removal pumps. Recirculation from the containment sump and hot leg recirculation are also modeled. This tree also determines the availability of the containment spray systems, the ice condenser, the air return fans used to circulate air in the containment during a LOCA event, and the hydrogen igniters. This event tree is illustrated in Figure 3.1.2-5.

A detailed discussion of the top events in the MLOCA tree follow. The top events are described in the order in which they appear in the MLOCA event tree.

- **Top Event IC — Ice Condenser**

  - **Function Evaluated.** Ice condenser availability.

- **Success Criteria.** The primary function of the ice condenser is, along with the containment spray system, the removal of the initial energy and pressure suppression during a LOCA and long-term pressure relief events. Pressure increases in the lower containment will force open the inlet doors to the ice condenser. This allows the heated steam/air mixture up through the ice. For successful energy removal and pressure suppression, the break flow must open the doors and be directed through the ice.

- **Model Boundaries.** The availability of the ice condenser under accident conditions is governed by the possibility of bypass of the ice condenser. This is similar to the method of analysis used in NUREG/CR-4551 for the Sequoyah Nuclear Plant. No analysis of the systems for the formation and maintenance of the ice bed is considered. Plant Technical Specifications preclude plant operation without adequate ice.

- **Conditions when Demanded.** The ice condenser top event is asked everywhere in the medium LOCA event tree.

- **Scenario Impact if Successful.** The ice condenser is available to remove energy from the break flow, limiting containment pressure.

- **Scenario Impact if Failed.** The ice condenser is not available to limit containment pressure.

- **Top Event VS — Supply to Centrifugal Charging Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Flow path for water to charging pumps suction.

  - **Success Criteria.** Top Event VS is used to determine the availability of the supply of borated water to the centrifugal charging pumps for 24 hours after a plant trip. For medium break LOCAs, the water supply must come from the RWST since a safety injection signal has been initiated.

  - **Model Boundaries.** The equipment required to operate is dependent on the reactor conditions. For medium break LOCAs, volume control tank (VCT) level control valve LCV-62-132 or LCV-62-133 is required to close, and RWST level control valve LCV-62-135 or LCV-62-136 is required to open and remain open for 24 hours. There is an operator action house event used in this top event to align these valves manually if the actuation signal from ESFAS fails. This manual action is analyzed in Top Event OS. The automatic opening of LCV-62-135 and LCV-62-136 in the absence of a safety injection signal (i.e., on the closure of LCV-62-132 or LCV-62-133) is not modeled.

  - **Conditions when Demanded.** Top Event VS is asked in all sequences in the medium LOCA event tree.

  - **Scenario Impact if Successful.** If Top Event VS is successful, this implies that a suction source is available for makeup to the RCS via the centrifugal charging pumps.

- **Scenario Impact if Failed.** Failure of Top Event VS, the charging pump suction supply, fails both the centrifugal charging pump Top Events VA and VB.

- **Top Events VA and VB — Centrifugal Charging Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Operation of the centrifugal charging pumps.

  - **Success Criteria.** These top events, VA and VB, model the two centrifugal charging pumps 1A-A and 1B-B, respectively. The success of these top events requires that the modeled pump start on an actuation signal and operate for 24 hours to provide flow to the cold leg injection path. The centrifugal charging pumps are used for both injection and recirculation modes.

  - **Model Boundaries.** The equipment modeled in each top event include the pump itself, the pump lube oil cooler, the room cooler, and the manual valves and check valve in the pump train. These top events have a mission time of 24 hours. CCP 1A-A is modeled in Top Event VA as the normally running pump to supply normal charging to the reactor coolant system and RCP seal injection at the start of the event. CCP 1B-B in Top Event VB is modeled in standby. These pumps are being normally supplied from the VCT.

    For a medium LOCA, both pumps are given an actuation signal to start and are required to inject borated water from the RWST to the four RCS cold legs. Manual operator action to start the CCPs, given automatic actuation system failure, is modeled in Top Event OS. The supply to the CCPs swaps over from the VCT to the RWST as modeled in Top Event VS. The injection path to the four RCS cold legs is automatically aligned as modeled in Top Event VC. Later, when sump recirculation is required, the RWST is isolated and the RHR pumps draw from the containment sump to supply the centrifugal charging pumps for high head injection during recirculation.

  - **Conditions when Demanded.** Both Top Events VA and VB are asked for every sequence in the medium LOCA event tree.

  - **Scenario Impact if Successful.** If either Top Event VA or VB is successful, then the centrifugal charging pumps are available for injection and recirculation. Emergency boration and RCP seal injection are not of interest for mitigation of medium LOCAs.

  - **Scenario Impact if Failed.** If both Top Events VA and VB fail, then neither centrifugal charging pump is available. Since at least two high head pumps (CCPs or SIPs) must operate for successful injection during a medium LOCA, failure of both Top Events VA and VB implies that both safety injection pumps must operate to prevent core damage.

- **Top Event VF — 2/3 Centrifugal Charging Pump Cold Leg Injection Paths**

  - **Function Evaluated.** Availability of the flow path from the charging pumps to the cold leg injection lines.

  - **Success Criteria.** This top event models the centrifugal charging pumps' discharge piping and valves from the pumps to the cold legs of the RCS. The success of this top event requires that the inline check valve opens and remains open, and two of the three intact cold leg paths are available including the associated check valves that must open and remain open. The fourth cold leg injection line (i.e., RCS loop 4) is postulated as the source of the LOCA, and is therefore assumed to be unavailable. This top event has a mission time of 24 hours.

  - **Model Boundaries.** The equipment modeled includes the equipment from the point where the pump discharge lines are joined through the RCS cold leg injection check valves. This includes:

    - Parallel MOVs FCV-63-39 and FCV-63-40 (locked open).

    - The BIT and downstream valves FCV-63-25 and FCV-63-26.

    - The injection line check valve inside containment (1-63-581).

    - The four branch lines into the RCS, each containing a throttling valve (THV-63-582, THV-63-583, THV-63-584, and THV-63-585) and a check valve (1-63-586, 1-63-587, 1-63-588, and 1-63-589).

    This top event is required for CVCS high head safety injection success in both the injection and recirculation modes. The pumps are modeled separately in Top Events VA and VB and the suction path to the pumps is modeled in Top Event VS.

    An operator action house event is used in these top events to start the pumps manually if automatic actuation from ESFAS has failed. This operator action is modeled in Top Event OS.

  - **Conditions when Demanded.** Top Event VF is asked in all sequences of the medium LOCA event tree, except when both centrifugal charging pumps are unavailable; i.e., when both Top Events VA and VB are failed.

  - **Scenario Impact if Successful.** Success of this top event implies that at least one of the centrifugal charging pumps is available to take suction from the RWST and inject into the RCS through the required number of cold leg injection lines to mitigate a loss of RCS inventory.

  - **Scenario Impact if Failed.** Failure of this top event implies that CVCS high head injection fails to provide inventory control. Sufficient high head injection may still be provided by the safety injection system.

- **Top Events S1 and S2 — Safety Injection Pumps 1A-A and 1B-B**

  - **Function Evaluated.** The automatic actuation and operation of the safety injection pumps.

  - **Success Criteria.** These top events, S1 and S2, model the safety injection pump trains 1A and 1B, respectively. The success of these top events requires that the modeled pump start automatically on a safety injection actuation signal or manually by the operator and provide flow to the safety injection system cold leg injection paths. Two of four charging and safety injection pumps operating for 24 hours is sufficient for inventory control during a medium LOCA, but the status of each pump train is tracked individually. If both charging pumps are operating, the safety injection pumps are not required. If both charging pumps fail, then both safety injection pumps are required.

  - **Model Boundaries.** The model boundary for the pump train in each of these top events includes the common pump suction line from the RWST to the safety injection line discharge cross-connect valves FCV 63-152 or FCV 63-153. The pump miniflow lines to the RWST, room cooling and motor cooling, are also included. The components within this boundary for train A (train B components are shown in parentheses) include:

    - Normally open suction valve FCV-63-47 (FCV-63-48).
    - Safety injection pump 1A-A (1B-B) and check valve 63-524 (63-526).
    - Normally open manual valve 63-525 (63-527) and normally open discharge valve FCV-63-152 (FCV-63-153).
    - Miniflow line check valve 63-526 (63-530), motor-operated valve FCV-63-4 (FCV-63-175).
    - Common miniflow valve FCV-63-3.

    A complete injection path for a safety injection pump requires flow to the pumps [from the RWST (Top Event RW) in the injection phase or from the RHR system in the recirculation phase] and the injection paths defined in Top Event IP. Therefore, successful operation of the safety injection system for a pump in the injection mode requires the pump (Top Events S1 and S2), the suction source (support tree Top Event RW), and the injection path as defined by Top Event IP.

    Top Event S2 is evaluated conditionally on the status of Top Event S1. This permits the intersystem dependencies between trains of the system (i.e., for common cause, test, and maintenance) to be modeled properly.

  - **Conditions when Demanded.** Top Events S1 and S2 are asked along every sequence in the medium LOCA event tree. If the RWST is unavailable, these events are guaranteed failed.

- **Scenario Impact if Successful.** If either Top Events S1 or S2 is successful, then high pressure injection and high pressure recirculation may be successful. For successful high pressure injection, Top Event SI must also be successful to provide the required cold leg injection paths. Successful high head safety injection in the recirculation mode requires the pump (Top Events S1 and S2), the injection path as defined by Top Event SI, and RHR supply flow to the pump suction (as defined by Top Events RA, RB, RI, and RR).

- **Scenario Impact if Failed.** Failure of both Top Events S1 and S2 means that the safety injection pumps are not available for high pressure injection or recirculation. In this case, both charging pumps must be available for inventory control during a medium LOCA.

• **Top Event IP — Safety Injection System Discharge Piping**

- **Function Evaluated.** The availability of a flow path from the discharge of the safety injection pumps to the RCS cold legs.

- **Success Criteria.** Success of Top Event IP requires that one of the safety injection pumps is available (i.e., Top Event S1 or S2 is successful), that the suction path from the RWST is available, and that at least two of the three intact, cold leg injection paths are available for 24 hours. The fourth injection path via RCS loop 4 is assumed to be the source of the LOCA and therefore is unavailable.

- **Model Boundaries.** This top event models both the suction and discharge flow path to the safety injection pump trains modeled in Top Events S1 and S2. The discharge portion of the model considers flow from the safety injection system pump trains to the RCS cold leg loops, from the discharge path of the pumps modeled in Top Events S1 and S2 to the injection point in the four RCS cold legs downstream of check valves 63-551, 63-553, 63-555, and 63-557. The suction portion of the model considers the flow path from the RWST and through FCV-63-5 and check valve 63-510.

  The system boundary is defined so as to complete the flow path between the pump discharge and the RCS cold legs. The components modeled include:

  - The normally open valve FCV-63-22 where the flow from the two pumps is headered before splitting into four injection paths.

  - The four injection paths, each containing a throttle valve (FCV-63-550, FCV-63-552, FCV-63-554, and FCV-63-556) and associated check valve (63-551, 63-553, 63-555, and 63-557, respectively).

  - The four cold leg check valves (63-560, 63-561, 63-562, and 63-563).

The safety injection pump suction path modeled includes:

- Normally open valve FCV-63-5.
- Check valve 63-510.

There is no operator action required or modeled in this top event.

- **Conditions when Demanded.** Top Event IP is asked in all sequences of the medium LOCA event tree. If the RWST is not available, or if both safety injection pumps fail, then Top Event IP is guaranteed failed as determined by the assignment of split fractions.

- **Scenario Impact if Successful.** Success of Top Event IP, along with success of either or both of Top Events S1 and S2, means that the high head safety injection pumps successfully provide some high pressure injection and are available for high pressure recirculation once the RWST is exhausted. Flow from at least two of the four charging and safety injection pumps is required for RCS inventory control during a medium LOCA.

- **Scenario Impact if Failed.** Failure of this top event implies that injection flow from the safety injection system into the RCS during both injection and recirculation phases is not available. For successful inventory control, both charging pumps must then be operable.

- **Top Event CL — Two of Three Cold Leg Accumulators Discharge**

  - **Function Evaluated.** Reflooding via the accumulators.

  - **Success Criteria.** The accumulator connecting to RCS loop 4 is assumed to spill out the break. Success of this top event requires successful injection from two of the three remaining accumulators. This success criterion is believed to be conservative for medium LOCAs, but no analysis is available to justify a relaxation. In the NUREG/CR-4550 analysis for the Surry plant, the accumulators were not assumed to be required for mitigation of medium LOCAs.

  - **Model Boundaries.** This top event models the injection of the cold leg accumulators into the vessel. The model for each of the three accumulators on the intact lines includes the normally open isolation valve (FCV-63-118, FCV-63-98, or FCV-63-80), a check valve (63-622, 63-623, or 63-624), and one of the cold leg check valves (63-560, 63-561, or 63-562). The check valves closest to the RCS are shared with the cold injection flow path from the safety injection pumps; i.e., check valves 63-560,63-561, and 63-562 appear in both this top event and in Top Event IP. Therefore, the availability of the accumulators is evaluated conditional on the status of Top Event IP.

  - **Conditions When Demanded.** Top Event CL is asked for each sequence in the medium LOCA event tree.

- **Scenario Impact if Successful.** The initial period of injection is successful. Continued inventory control may then be provided using the charging and safety injection pumps.

- **Scenario Impact if Failed.** Core damage is assumed due to failure of inventory control. High head injection via the charging and safety injection pumps would likely prevent subsequent vessel breach, but this is not modeled explicitly.

• **Top Events RA and RB — Residual Heat Removal Pumps 1A-A and 1B-B**

- **Function Evaluated.** Manual or automatic start and operation of the RHR pumps.

- **Success Criteria.** Success of each of these top events implies that the pump in that train is available to provide flow to the RHR discharge header to the RCS cold legs or to the high head systems during sump recirculation for 24 hours. Success of a train requires that ESFAS provide an automatic start signal, given a safety injection condition, or that the operators provide a manual start signal as a backup to the automatic start signal; i.e., via Top Event OS. Only one train of RHR is required for low pressure injection and for recirculation from the sump.

- **Model Boundaries.** These top events model the RHR pump trains 1A and 1B. The boundaries for these top events include the pump suction valves, FCV-74-3 and FCV-74-21 through the discharge valves, FCV-74-16 and FCV-74-28. The RHR pump miniflow lines, pump room cooling, and pump seal cooling are also included. The components within this boundary for train A (train B components are shown in parentheses) include:

  • The pump suction valve FCV-74-3 (FCV-74-21).

  • Pump 1A-A (1B-B), discharge check valve 74-514, (74-515), and normally locked open manual valves 74-520 and 74-524 (74-521 and 74-525).

  • Miniflow line valve FCV-74-12 (FCV-74-24).

  • Heat exchanger outlet throttle valve FCV-74-16 (FCV-74-28).

  • Heat exchanger outlet check valves 74-544 and 74-545.

  • The heat exchanger's failure because of rupture.

  • ERCW train A (B) supply to RHR pump 1A-A room cooler and the associated fan and valves.

  • CCS cooling to the RHR pump 1A-A (1B-B) mechanical seals and the associated valves.

The action for the operators to reset the safety injection signal and to stop the RHR pumps if RCS pressure is > 180 psig to prevent pump overheating during extended operation on miniflow is not explicitly modeled for medium LOCAs. Instead, pump suction swapover to the containment sump from the RWST should occur before the pumps overheat due to the actuation of containment spray; i.e., before the 100 minutes estimated to overheat the pumps while operating on miniflow without CCS to the RHR heat exchangers. The model assumes that the RHR pumps, once actuated by a safety injection signal, would not be stopped by the operators prior to going to recirculation from the containment sump.

Top Event RA is evaluated conditionally on the status of Top Event RB. This is to account for the intrasystem dependencies between trains; i.e., for common cause, test, and maintenance dependencies.

- **Conditions when Demanded.** These events are asked for all sequences of the event tree. The status of both trains is asked, even though only one train is required, because the RHR pump trains must work together with the corresponding sump recirculation path.

  If the RWST is unavailable, both trains of RHR pumps are guaranteed failed.

- **Scenario Impact if Successful.** If Top Event RA or RB is successful, the corresponding train of RHR pumps is available. These top events are used with the availability of the RWST (Top Event RW), the RHR injection path (i.e., Top Event RF), containment sump recirculation (Top Event RR), hotleg recirculation (Top Event RH), and RHR spray (Top Event RS) to evaluate the availability of each of the functions performed by the RHR pumps.

- **Scenario Impact if Failed.** Failure of both of these top events (RA and RB) implies that neither train of the RHR system is available.

- **Top Event RF — RHR Injection Path**

  - **Function Evaluated.** Availability of the flow paths from the RWST to the RHR pumps and from the RHR pumps to the RCS.

  - **Success Criteria.** Success of Top Event RF requires that the suction line from the RWST to both RHR pumps be available and that a flow path to two of three RCS cold legs be available for 24 hours. The fourth RCS cold leg injection path (i.e., RCS loop 4) is assumed to be where the break occurs and therefore cannot be used for injection.

  - **Model Boundaries.** This top event models the RHR flow paths from the pump discharge path modeled in Top Events RA and RB to the injection point in the three RCS cold legs downstream of check valves 63-551, 63-553, and 63-555. The suction line to the RHR pumps from the RWST used only for RHR supply during safety injection (FCV 63-1 and check valve 63-502) is also included here. The model includes the following components:

- RWST to RHR pumps isolation valve FCV-63-1 and check valve 63-502.
- RHR to cold legs isolation valves FCV-63-93 and FCV-63-94.
- The outer check valve on each RHR cold leg injection path: 63-632, 63-633, and 63-634.
- Three RCS check valves 63-560, 63-561, and 63-562.

The three RCS check valves last listed above are also included in Top Event IP (i.e., the safety injection pumps suction and cold leg discharge path) since the RHR pumps and the safety injection pumps share common entry paths to the RCS. Therefore, the branch point values for Top Event RF are evaluated conditionally on the status of Top Event IP. If Top Event IP has failed, then a fraction of the time, this will have been due to the failure of the four common RCS check valves, which would also fail Top Event RF.

— **Conditions when Demanded.** Top Event RF is asked whenever Top Event RA or RB is successful. Top Event RF is guaranteed failed if Top Event RA fails. Failure of train A of RHR precludes low pressure injection via cold legs 1 and 2. Since injection via cold leg line 4 is precluded by the break location, only one injection line could possibly be used for injection from RHR pump B. Two cold leg lines are required for success.

— **Scenario Impact if Successful.** The RHR pump trains are then available to operate for low head injection and for low pressure recirculation from the containment sump.

— **Scenario Impact if Failed.** Failure of this top event implies that RHR is not available to directly inject into the RCS cold legs. Core damage results due to inadequate injection flow. This event is guaranteed failed if both Top Events RA and RB are failed or if Top Event RW is failed.

- **Top Event SU — Containment Sump Available**

— **Function Evaluated.** The availability of water in the containment sump for recirculation.

— **Success Criteria.** Given a loss of RCS inventory, success of Top Event SU implies that water is available for recirculation from the containment sump at the time that the RWST empties. A flow path for draining the containment spray water from the upper to the lower flow compartment must be available, and the containment sump itself must not be plugged. Water must have been injected directly or indirectly into the containment by one or more of the ECCS pumps; i.e., by the charging, safety injection, RHR, or containment spray pumps. Melting of the ice alone is not sufficient to permit recirculation.

- **Model Boundaries.** This top event models the availability of the containment sump for recirculation. Failure of the sump can occur by the plugging of the drain between the upper and lower containment compartments or by plugging of the sump with containment debris.

  The availability of the sump is modeled as the probability that the drain plugs were not removed during the last shutdown or that debris has blocked the sump. The containment sprays discharge into the upper containment. Melted ice and discharge from breaks in the RCS boundary are initially directed into the lower compartment. However, functioning sprays with failure of the drain plugs or of containment sump plugging will guarantee failure of recirculation.

- **Conditions when Demanded.** The containment sump top event, SU, is asked everywhere in the medium LOCA event tree.

- **Scenario Impact if Successful.** Water is then assumed to be available for recirculation from the sump.

- **Scenario Impact if Failed.** RHR recirculation from the sump and containment spray recirculation from the sump are not available if Top Event SU fails. Therefore, Top Events RL, RVA, RVB, RRCH, RS, and RH are not asked if Top Event SU fails. Containment spray may still operate in the injection mode, taking suction from the RWST until it empties.

- **Top Event RL — Recirculation Level Instrumentation**

  - **Function Evaluated.** Instrumentation for swapover from RWST to containment sump suction.

  - **Success Criteria.** Two of the four channels of level instrumentation must be available to make up the low level swapover signal. The swapover is initiated when RWST level is less than the low-level setpoint, and containment sump level greater than the required value during a LOCA.

  - **Model Boundaries.** This top event models the instrumentation necessary for switchover from the injection mode to the recirculation mode. The instrumentation modeled includes:

    - The containment sump level switches, LS-63-50D, LS-63-51D, LS-63-52D, and LS-63-53D.

    - The RWST level switches, LS-63-180, LS-63-181, LS-63-182, and LS-63-183.

    An operator action is modeled to back up the level transmitters; i.e., action HARL1. This action is applicable for conditions when automatic swapover instrumentation fails due to conditions that can be easily detected; e.g., freezing of the sensor lines, a common relay failure, or if the operators had previously reset the safety injection signal associated with the swapover valves, preventing the automatic swapover signal.

- **Conditions when Demanded.** Top Event RL is asked for all sequences in the medium LOCA event tree when the containment sump is available; i.e., when Top Event SU is successful.

- **Scenario Impact if Successful.** This implies that the automatic actuation signal for pump suction swapover to the containment sump is available to both trains of valves.

- **Scenario Impact if Failed.** Failure of this top event implies that automatic swapover has failed. Credit is given for initiation of manual swapover if automatic swapover fails to actuate due to relay failure or gross transmitter failure; e.g., RWST level transmitter freezes. Therefore, failure of Top Event RL implies that recirculation from the containment sump for both core cooling and RHR spray recirculation are unavailable.

- **Top Events RVA and RVB — Trains A and B RHR Sump Swapover Valves**

  - **Function Evaluated.** Response of ECCS valves to a pump suction swapover demand from the RWST to the containment sump.

  - **Success Criteria.** The containment sump suction valve must open, and the RWST suction valve on the associated train must close for success of one train.

  - **Model Boundaries.** These top events model the trains A and B swapover valves of the RHR system, which realign automatically to allow pump suction from the containment sump once the RWST level is low. The swapover is initiated by the instrumentation modeled in Top Event RL on RWST level less than the low-level setpoint and containment sump level greater than the required value during a LOCA. The equipment modeled includes the motor-operated sump swapover valves 1-FCV-63-72 for train A and 1-FCV-63-73 for train B, which must open, and 1-FCV-74-3 for train A and 1-FCV-74-21 for train B, which must close. The valve changes necessary for high pressure recirculation are not modeled in this top event. They are instead modeled in Top Event RR.

    Top Event RVB is evaluated conditionally on the status of Top Event RVA. This is because the suction valves in the different trains, both to the RWST and to the containment sump, are in common cause groups. Therefore, the failure probability of the second train is higher if the first train fails than would otherwise be expected, assuming that the trains were independent. These dependencies are captured in the systems models.

  - **Conditions when Demanded.** These events are asked for sequences in the medium LOCA event tree in which Top Event SU is successful. Failure of the containment sump makes the question of valve swapover immaterial. If the swapover instrumentation is unavailable (i.e., Top Event RL fails), then both of these events are set to guaranteed failed.

- **Scenario Impact if Successful.** Success of these valves means that the containment sump suction path is aligned for recirculation on the associated train. For mitigation of medium LOCAs, high pressure recirculation is required for core cooling; i.e., high pressure recirculation using the safety injection or charging pumps is assumed to be unnecessary because the RCS pressure should be low by the time that the RWST reaches low level.

- **Scenario Impact if Failed.** Failure of these valves to go to the correct position will fail the corresponding RHR pump train during the recirculation mode.

- **Top Event RR — RHR Recirculation**

  - **Function Evaluated.** This top event models the automatic/manual swapover of the RHR suction to the containment sump for high pressure recirculation.

  - **Success Criteria.** Successful RHR recirculation mode requires: (1) the automatic transfer of the suction of RHR from the RWST to the containment sump, (Top Events RL, RVA, and RVB), (2) manual transfer of the suction of the safety injection pumps and centrifugal charging pumps from the RWST to the discharge of the RHR pumps if high pressure recirculation is required, and (3) use of the RHR heat exchangers to remove heat from the containment sump recirculation coolant. Either of two RHR trains supplying flow to at least one operable charging or safety injection pump is considered success. The swapover must be completed within 20 minutes of reaching the low-level setpoint to prevent core uncovery. High pressure recirculation from the containment sump is conservatively assumed to be required for the entire spectrum of medium break LOCAs, even though only the smaller size breaks (i.e., those just greater than 2 inches in diameter equivalent) may actually still have RCS pressure greater than the RHR shutoff head at the time of recirculation swapover.

  - **Model Boundaries.** The RHR configuration during recirculation from the sump allows: (1) the RHR pumps to draw coolant from the containment sump, (2) the RHR heat exchangers to transfer heat from the coolant to the CCS, and (3) the coolant to be pumped to four discharge paths: (a) the cold legs of the RCS, (b) the inlet of the centrifugal charging pumps, (c) the inlet of the safety injection pumps, and (d) for RHR containment spray.

    The recirculation mode is initiated automatically when the RWST reaches a low level coincident with a high containment sump level, as modeled in Top Event RL. The RHR pumps will then automatically switch their suction from the RWST to the containment sump as modeled in Top Events RVA and RVB.

During recirculation, component cooling water flow is manually established to the RHR heat exchangers to cool the flow from the containment sump before being discharged into the RCS, the suction of the safety injection or charging pumps, or for RHR.

RHR valves (FCV-63-8 and FCV-63-11) are manually opened to establish a flow path from the RHR pump discharge to the suction of the centrifugal charging pumps and safety injection pumps for high pressure injection. These valves (FCV-63-8 and FCV-63-11) are interlocked with the closed position of the safety injection pump miniflow valves and the safety injection discharge path to the RWST (isolation valves FCV-63-3, FCV-63-4, and FCV-63-175) to ensure that sump coolant is not diverted to the RWST.

The model for this top event includes the following equipment:

- The safety injection pump miniflow valves (FCV-63-3, FCV-63-4, and FCV-63-175).
- The charging, safety injection, and RHR suction valves from the RWST LCV-62-135, LCV-62-136, FCV-63-1, and FCV-63-5.
- The component cooling water and control valves to the RHR heat exchangers (FCV-70-153 and FCV-70-156).

The recirculation valves for the swapover of the containment spray pumps are modeled separately in Top Event CH.

Top Event RR is used with the containment sump availability Top Event SU, the safety injection system Top Events S1, S2, and IP; the charging system Top Events VA, VB, and VF, and the RHR pump, flow path, and RHR spray Top Events RA, RB, RF, and RS to determine that flow is available to the RCS from one of four high pressure pumps (centrifugal charging pumps or safety injection pumps) during high pressure recirculation or one of two RHR pumps during low pressure recirculation. High pressure recirculation is modeled as being required for all sump recirculation sequences initiated by medium LOCAs unless the actions for rapid depressurization (Top Events DS and DP) are successful.

The operator action to align the CCS to the RHR heat exchangers and to make the valve alignments from the RHR pump train discharge to the high pressure pumps is modeled in this top event. Power must also be restored to FCV-63-1, so that this RWST suction valve may be isolated. The action is designated RR1. Credit for this action is only modeled if RWST level instrumentation is available; i.e., Top Event RL succeeds. The action evaluated conditionally on the failure of Top Event RL (action HARR2) is not used.

- **Conditions when Demanded.** Whenever the containment sump is available for sump recirculation (i.e., Top Event SU succeeds), this event is asked. If both RHR sump swapover valve trains fail (i.e., Top Events RVA and RVB fail), or if high pressure recirculation is required and none of the high pressure pumps are available, then this event is assumed to be failed.

- **Scenario Impact if Successful.** Implies that there has been a successful transition to high pressure recirculation from the containment sump. Discharge from one or both RHR pumps, taking suction from the sump, is directed to the suction of at least one operable high pressure injection pump, which then injects into the RCS.

- **Scenario Impact if Failed.** Failure of this top event implies that there is no recirculation cooling of the core, and that RHR spray recirculation is not possible.

- **Top Events CSA and CSB - Containment Spray Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Containment spray pump availability for injection and recirculation.

  - **Success Criteria.** The associated containment spray pump train is automatically or manually actuated and provides spray injection. The pump operates for 24 hours. Manual actuation must be completed within 20 minutes of reaching the automatic actuation setpoint. Only one train of containment spray is required in the injection mode.

  - **Model Boundaries.** This top event models the availability of the containment spray pump trains 1A-A and 1B-B to deliver flow into containment, given a suction source. In the injection mode, suction is from the RWST (Top Event RW), and during recirculation, suction is from the containment sump (Top Event CH). Heat removal from the containment spray heat exchangers is modeled separately in Top Event CH. Top Events CSA and CSB are both actuated automatically by a high-high containment pressure signal, which is assumed reached in this analysis for medium LOCAs.

    The equipment modeled in Top Event CSA (Top Event CSB equipment is shown in parentheses) includes:

    - The containment isolation valve FCV-72-39 (FCV-72-2).
    - Containment spray pump 1A-A (1B-B).
    - Miniflow valve 1-72-34 (1-72-13).
    - The RWST suction valve FCV-72-22 (FCV-72-21).
    - Normally open manual valve 1-72-528 (1-72-529) and check valves 1-72-506 (1-72-507), 1-72-524 (1-72-525), and 1-72-547 (1-72-548).
    - Containment sump recirculation valve 1-FCV-72-44 (1-FCV-72-45), which must remain closed.

- The physical integrity of the containment spray heat exchanger 1A (1B), CCS supply to the containment spray pump 1A-A (1B-B) oil coolers including the associated valves.
- ERCW supply to the containment spray pump 1A-A (1B-B) room coolers including the associated valves.
- The 263 nozzles of the spray header 1A (1B).

The operator action to initiate containment spray manually, given a high-high containment pressure condition but failure of the automatic actuation signal from ESFAS, is included in the models for both top events; i.e., via action HACS1.

Train B of containment spray, as represented by Top Event CSB, is evaluated conditionally on the status of Top Event CSA to reflect the potential for common cause affecting both trains and the maintenance limits imposed by plant technical specifications.

- **Conditions when Demanded.** Top Events CSA and CSB are asked for all sequences in the medium LOCA event tree. Containment spray is unavailable if the RWST is not available.
- **Scenario Impact if Successful.** The containment spray system acts with the ice condenser system to provide short-term containment heat removal and to limit the containment pressure increase. Containment spray provides a long-term source of containment heat removal while in the sump recirculation mode. The long-term function of containment spray (i.e., in the recirculation mode) requires the success of at least one of Top Event CSA or CSB and of Top Event CH, which is considered next.
- **Scenario Impact if Failed.** Failure of Top Event CSA or CSB implies that the associated train of containment spray is not available in either the injection mode or the subsequent recirculation mode. Even if both trains fail, the containment would still not overpressurize during the injection phase for any size LOCA.

- **Top Event CH — Containment Spray in Recirculation Mode**
  - **Function Evaluated.** Recirculation mode of containment spray.
  - **Success Criteria.** Success of this event requires that at least one train of containment spray is successful and of the associated spray pumps to operate in the injection mode. ERCW cooling must be aligned to the containment spray heat exchangers. The action to accomplish the swapover (action HACH1) is initiated when RWST is less than the low-low level setpoint, and containment pressure is greater than the high-high pressure setpoint. It must be completed before the pumps lose suction due to the RWST emptying, assumed to be about 5 minutes after the RWST low-low level setpoint is reached.

– **Model Boundaries.** This top event models the operator actions and equipment required for containment spray to operate successfully in the recirculation mode.

The equipment modeled in Top Event CH includes:

- The containment isolation valves FCV-72-39 and FCV-72-2.
- The train A and train B containment spray pumps (CSA and CSB).
- The RWST suction manual valves FCV-72-22 and FCV-72-21.
- The containment sump suction valves FCV-72-44 and FCV-72-45.
- ERCW cooling to the containment spray heat exchangers.
- ERCW heat exchanger inlet isolation valves FCV-67-125 and FCV-67-123.
- ERCW heat exchanger outlet check valves 67-537A and 67-537B.
- ERCW heat exchanger outlet isolation valves FCV-67-126 and FCV-67-124.

The operator action to align the ERCW supply to the containment spray heat exchangers and to transfer the suction of the containment spray system from the RWST to the containment sump is modeled with this top event. The operators must stop the operating containment spray pumps, close the pump suction valves from the RWST, and open the pump suction valves from the sump. The operator then checks for adequate ERCW flow and cooling to the CS heat exchangers, and restarts the spray pumps. These actions are initiated when the RWST level low-low setpoint is reached.

– **Conditions when Demanded.** Top Event CH is asked for all sequences in the medium LOCA event tree for which the containment sump is available; i.e., when Top Event SU succeeds. The automatic swapover on low RWST level and high sump level (i.e., Top Event RL) must be successful. The containment spray pumps in the injection mode (i.e., Top Events CSA and CSB) must be successful for the corresponding train of spray to operate in the recirculation mode.

– **Scenario Impact if Successful.** Success of this top event implies that at least one train of containment spray is operating, taking suction from the containment sump, with its associated heat exchanger being properly supplied with ERCW. Containment heat removal is available. Spray recirculation from the RHR pumps discharge is not required.

– **Scenario Impact if Failed.** If this top event is not successful, containment heat removal requires successful operation of RHR spray, as modeled in Top Event RS.

- **Top Event RS – RHR Spray**

– **Function Evaluated.** Spray recirculation using RHR pump discharge.

– **Success Criteria.** One of the two RHR pump trains must operate in the recirculation mode, containment pressure must exceed the required value more than 1 hour into the accident, and the operators then align one train of RHR for containment spray recirculation. The alignment must be made before the containment overpressurizes, estimated to be several hours into the accident.

– **Model Boundaries.** This event models the containment spray function of the RHR system. It includes the operator action and the opening of the RHR ring header inlet motor-operated valve. The operators establish one (and only one) train of RHR spray if containment pressure is greater than the required value and the accident is at least 1 hour old. The action modeled is designated HARS1.

  The equipment modeled in this top event includes:

  - RHR crosstie valves FCV-74-33 and FCV-74-35.
  - The RHR injection path isolation valves FCV-63-93 and FCV-63-94.
  - The RHR spray control valves FCV-72-40 and FCV-72-41.

– **Conditions when Demanded.** This event is asked only if normal containment spray recirculation, as represented by Top Event CH, has failed. It is used with the containment sump availability Top Event SU, the containment sump recirculation Top Event RR, and the RHR pump train Top Events RA and RB. It is not asked if Top Event RR has failed. Top Event RR asks about the valves for alignment of RHR recirculation, and of the alignment of valves to establish CCS flow to the RHR heat exchangers for heat removal.

  Credit for this action is taken if at least two RHR trains are available, at least one charging pump, and at least one safety injection pump are available. Procedural guidance only instructs the operators to use RHR spray if all of these conditions are met. Note that when the RHR trains are aligned for hot leg recirculation, one of the RHR spray paths will be unavailable.

– **Scenario Impact if Successful.** Thus success of this top event, RS, implies that at least one train of RHR is providing containment spray (provided that Top Events SU, RR, and RA or RB is successful).

– **Scenario Impact if Failed.** Failure of this top event implies that containment heat removal from spray systems is not available.

Top Event RH — RHR Hot Leg Injection

– **Function Evaluated.** Alignment of RHR recirculation for injection to the hot legs.

– **Success Criteria.** Realignment of one operating RHR recirculation path for injection, via the hot legs rather than cold legs, 15 hours after break occurs. Must be initiated before flow blockage due to boron precipitation occurs, estimated to be many hours later. Realignment of the safety injection pumps for hot leg recirculation is assumed to be unnecessary.

- **Model Boundaries.** This top event models the use of RHR for RCS hot leg recirculation. Hot leg recirculation is used to limit the amount of boron precipitating out on the reactor internals, during a LOCA in which the RCS cannot be kept full, so that heat removal is accomplished by boiling. Excessive boron precipitation is postulated to interfere with heat transfer and reactivity control. The operators are instructed to divert the cold leg injection flow to hot leg recirculation 15 hours after transferring to containment sump recirculation. This action is designated as RH1.

  The model includes the following equipment:

  - The RHR spray isolation valves FCV-72-40 and FCV-72-41.
  - The RHR cold leg isolation valves FCV-63-93 and FCV-63-94.
  - The RHR crosstie valves FCV-74-33 and FCV-74-35.
  - The hot leg injection isolation valve FCV-63-172.

- **Conditions when Demanded.** This event is asked for all sequences in the medium LOCA event tree in which RHR containment sump recirculation is successful. Top Events SU, RR, and at least one of Top Event RA or RB must be available.

- **Scenario Impact if Successful.** Successful switchover to hot leg recirculation implies that the problems associated with boron precipitation are not of concern.

- **Scenario Impact if Failed.** Failure of this top event implies that hot leg recirculation has failed. This is conservatively modeled as eventual fuel damage. Medium LOCA-initiated sequences involving failure of Top Event RH are assigned to core damage end states.

- **Top Event AR — Containment Air Return Fans**

  - **Function Evaluated.** Performance of the containment air return fans.

  - **Success Criteria.** The success criterion for the air return fans is that one of the two fans function. The fans are automatically actuated on a high-high containment pressure signal from ESAS.

  - **Model Boundaries.** This top event models the containment air return fans. The air return fans hot circulate saturated air from the upper compartment after a LOCA (10 minutes after high-high containment pressure). The air return fans enhance heat removal from the lower compartment to the ice condenser to help lower containment pressure. All portions of the air return fan functions are modeled. A manual start action to backup the ESFAS actuation is not modeled.

  - **Conditions when Demanded.** Top Event AR is asked for each sequence in the medium LOCA event tree because a high-high pressure condition should always be reached.

- **Scenario Impact if Successful.** The lower compartment containment pressure rise is mitigated by the successful operation of the air return fans.

- **Scenario Impact if Failed.** The lower compartment containment pressure is not mitigated, and local hydrogen pockets may develop due to poor mixing.

- **Top Event CI — Containment Isolation**

  - **Function Evaluated.** Isolation of small containment penetrations.

  - **Success Criteria.** Each of the small containment penetrations listed below must be either closed at the time of the accident and remain closed, or close by a signal from ESFAS based on a safety injection condition, or on Phase B; i.e., high-high containment pressure. For station blackout sequences, the time assumed available for local isolation of the seal return line is 3 hours.

  - **Model Boundaries.** This top event models containment isolation of nonessential penetrations during accident conditions. The containment penetrations explicitly modeled are as follows:

    - Containment major vents and drains.

    - Connections to the RCS.

    - Connections to containment atmosphere, with the exception of the large penetrations modeled in Top Event CP.

    This containment isolation top event models only those containment penetrations whose failure to isolate would result in a release path that would bypass containment. The following questions were asked about each penetration to determine the need for inclusion. Only those penetrations not covered by other system analyses in the PRA were considered.

    - Does the penetration communicate directly with the outside environment?

    - Does the penetration communicate with the environment via a low pressure system or a tank with a relief valve?

    - Will the relief valve lift at a pressure below the ultimate containment pressure?

    - Is the system or tank design pressure below the ultimate containment pressure?

    Based on these questions, the following penetrations are included in this top event:

    - Floor sump pump discharge (X-41).
    - RC drain tank and pressurizer vent to VH (X-45).

- RC drain tank pump discharge (X-46).
- Lower compartment pressure relief (X-80).
- RC drain tank to gas analyzer (X-81).
- Upper compartment air monitor intake (X-94A/B).
- Upper compartment air monitor return (X-94C).
- Lower compartment air monitor intake (X-95A/B).
- Lower compartment air monitor return (X-95C).
- RCP seal return line.

All of these penetrations receive a signal to isolate, given a safety injection, Phase A, or CVI signal, except for the RCP seal return line. The RCP seal return line is automatically signaled to close on high-high containment pressure; i.e., Phase B. During station blackout conditions, the operator is required to isolate locally the motor-operated seal injection and return valves in the RCP seal return line. This action (i.e., action CI1) is included in the model.

- **Conditions when Demanded.** Top Event CI is asked for every sequence in the medium LOCA event tree. The status of containment isolation is needed for the containment analysis.

- **Scenario Impact if Successful.** The smaller containment penetrations modeled in Top Event CI are isolated.

- **Scenario Impact if Failed.** One or more of the smaller containment penetrations listed above must have been opened initially and failed to close.

- **Top Event CP — Containment Purge Isolation**

  - **Function Evaluated.** Isolation of the containment purge lines.

  - **Success Criteria.** Success of this top event requires that either (1) the purge system was not in use when required, or (2) it was in use and that at least one valve in each penetration line closed.

  - **Model Boundaries.** This top event models the isolation of the containment purge penetrations, which are allowed to be opened during power operation. The Plant Technical Specifications allow these penetrations to be opened up to 1,000 hours per fuel cycle with the plant at power. The penetrations modeled are as follows:

    - Lower compartment purge air exhaust (X-4).
    - Instrument room purge air exhaust (X-5).
    - Upper compartment purge air exhaust (X-6).
    - Upper compartment purge air exhaust (X-7).
    - Upper compartment purge air supply (X-9A).
    - Upper compartment purge air supply (X-9B).
    - Lower compartment purge air supply (X-10A).
    - Lower compartment purge air supply (X-10B).
    - Instrument room purge air supply (X-11).

The penetrations modeled by Top Event CP are treated separately from those in Top Event CI due to the larger size of the purge penetrations. Even though the large size of the purge penetrations may limit the containment pressure rise during a LOCA, the Phase A and Phase B isolation signals setpoints are low enough that these isolation signals should occur even if the purge lines are initially open.

A backup manual action to isolate these penetrations is also considered, per the status of Top Event OS.

- **Conditions when Demanded.** Top Event CP is asked in all sequences of the medium LOCA event tree. Even if the automatic and manual isolation signals fail, this event may still be successful because the penetrations included in the model are normally closed anyway.
- **Scenario Impact if Successful.** Success of Top Event CP implies that the containment has at least a small hole in it. If Top Event CI is also successful, then the containment is isolated.
- **Scenario Impact if Failed.** Failure of this top event implies that containment isolation has failed and that a large hole in the containment boundary is present.

- **Top Event HH — Hydrogen Igniters**
  - **Function Evaluated.** Hydrogen control using the hydrogen igniters.
  - **Success Criteria.** Success of this top event implies that all 34 igniters in a single train have functioned.
  - **Model Boundaries.** This top event models the hydrogen igniters of the hydrogen mitigation system. During events that involve fuel cladding damage, the hydrogen igniters are used to burn away the hydrogen before it reaches explosive concentrations when it mixes with the containment atmosphere.

    The system consists of two trains of hydrogen igniters and the associated control circuitry. The system is manually initiated from the control room upon receipt of a Phase B signal and hydrogen concentration, as indicated by the hydrogen analyzer is within the required range. The action modeled is designated HH1. There is no time pressure to complete this action; i.e., many hours are assumed to be available. The operator is also required to place the hydrogen analyzer in service before the igniters. Use of the hydrogen analyzer is included as part of the operator action to initiate the hydrogen igniters. During recovery from an initial station blackout, the hydrogen igniters are not to be placed in service if the hydrogen analyzers indicate that the hydrogen concentration exceeds the required values.

- **Conditions when Demanded.** The hydrogen igniters are asked for in all sequences of the medium LOCA event tree. The status of Top Event HH is used in the evaluation of containment performance.

- **Scenario Impact if Successful.** The hydrogen igniters are available to continuously burnoff the hydrogen that collects in the containment prior to the concentration of hydrogen reaching significant explosive concentrations.

- **Scenario Impact if Failed.** The hydrogen igniters are not available to reduce the concentration of hydrogen within containment.

3.1.2.2.4 LLOCA Event Tree (RISKMAN Designator: LARLOC4)

The plant frontline system response to a large LOCA is quantified using a single event tree constructed specifically for the large LOCA initiating events. The event tree contains all of the top events required to mitigate the effects of the large LOCA. The tree models both the injection phase, and the recirculation phase including the response of the containment systems.

This event tree quantifies two classes of large LOCAs: a large LOCA greater in equivalent diameter than medium LOCAs but limited to the design basis large LOCA, and an excessive LOCA. The large LOCA event is characterized by a rapid RCS blowdown followed by injection from the cold leg accumulators, safety injection pumps, and RHR pumps. Long-term containment sump recirculation and hot leg recirculation are also modeled.

An excessive LOCA is characterized by a vessel break positioned such that it is impossible to keep the core covered to prevent core damage. For excessive LOCAs, the injection and recirculation top events are questioned to determine the status of containment systems that act to mitigate the event. Core damage is assumed to result.

The LLOCA event tree models the containment systems used for containment pressure and fission product control. These systems include the containment spray and RHR spray, the ice condenser, the air return fans used to circulate air in the containment, and the hydrogen igniters. This event tree is illustrated in Figure 3.1.2-6.

A detailed discussion of the top events in the MLOCA tree follows. The top events are described in the order in which they appear in the LLOCA event tree.

- **Top Event EX — Excessive LOCA**

  - **Function Evaluated.** The size of the large break LOCA.

  - **Success Criteria.** This top event is successful if the initiator is not an excessive LOCA. An excessive LOCA is defined as a vessel break positioned such that it is impossible to keep the core covered to prevent core melt.

  - **Model Boundaries.** This is a top event used as a switch in the event tree to differentiate between large and excessive LOCAs.

- **Conditions when Demanded.**

- **Scenario Impact if Successful.** The size and location of the break are such that it can be mitigated by the ECCS systems. For injection, only one of two RHR pumps taking suction from the RWST, together with accumulator injection, is required.

- **Scenario Impact if Failed.** The size and position of the break is excessive. No combination of ECCS systems can keep the core covered to prevent core damage. The active systems questioned are simply to determine the containment's response.

- **Top Event IC — Ice Condenser**

  - **Function Evaluated.** Ice condenser availability.

  - **Success Criteria.** The primary function of the ice condenser is, along with the containment spray system, the removal of the initial energy and pressure suppression during a LOCA and long-term pressure relief events. Pressure increases in the lower containment will force open the inlet doors to the ice condenser. This allows the heated steam/air mixture to flow up through the ice. For successful energy removal and pressure suppression, the break flow must open the doors and be directed through the ice.

  - **Model Boundaries.** The availability of the ice condenser under accident conditions is governed by the possibility of bypass of the ice condenser. This is similar to the method of analysis used in NUREG/CR-4551 for the Sequoyah Nuclear Plant. No analysis of the systems for the formation and maintenance of the ice bed is considered. Plant Technical Specifications preclude plant operation without adequate ice.

  - **Conditions when Demanded.** The ice condenser top event is asked everywhere in the large LOCA event tree.

  - **Scenario Impact if Successful.** The ice condenser is available to remove energy from the break flow, limiting containment pressure.

  - **Scenario Impact if Failed.** The ice condenser is not available to limit containment pressure.

- **Top Events S1 and S2 — Safety Injection Pumps 1A-A and 1B-B**

  - **Function Evaluated.** The automatic actuation and operation of the safety injection pumps.

  - **Success Criteria.** These top events, S1 and S2, model the safety injection pump trains 1A and 1B, respectively. The success of these top events requires that the modeled pump start automatically on a safety injection actuation signal or manually by the operator and provide flow to the safety injection system cold leg injection paths. One of two safety injection pumps operating for 24 hours is sufficient to transfer water from the RWST to the RCS during a large LOCA, but the status of each pump train is tracked individually. The safety injection pumps cannot alone provide sufficient injection to prevent fuel damage during a large or excessive LOCA.

The status of the charging pumps is not questioned in the large LOCA event tree. The charging pumps could also be used to transfer water to containment, but the PRA team chose to simplify the model by leaving out these pumps. The safety injection pumps were chosen to be included because they share common check valves with the accumulators and RHR injection pathways. The models for the accumulators and RHR injection are dependent on the status of the safety injection pumps and injection flow path; i.e., for modeling purposes, it was more convenient to model the safety injection pumps.

— **Model Boundaries.** The model boundary for the pump train in each of these top events includes the common pump suction line from the RWST to the safety injection line discharge cross-connect valve FCV 63-152 or FCV 63-153. The pump miniflow lines to the RWST, room cooling and motor cooling are also included. The components within this boundary for train A (train B components are shown in parentheses) include:

- Normally open suction valve FCV-63-47 (FCV-63-48).
- Safety injection pump 1A-A (1B-B) and check valve 63-524 (63-526).
- Normally open manual valve 63-525 (63-527) and normally open discharge valve FCV-63-152 (FCV-63-153).
- Miniflow line check valve 63-526 (63-530), motor-operated valve FCV-63-4 (FCV-63-175).
- Common minifow valve FCV-63-3.

A complete injection path for a safety injection pump requires flow to the pumps [from the RWST (Top Event RW) in the injection phase or from the RHR system in the recirculation phase)] and the injection paths defined in Top Event IP. Therefore, successful operation of the safety injection system for a pump in the injection mode requires the pump (Top Events S1 and S2), the suction source (support tree Top Event RW), and the injection path as defined by Top Event IP.

Top Event S2 is evaluated conditionally on the status of Top Event S1. This permits the intersystem dependencies between trains of the system (i.e., for common cause, test, and maintenance) to be modeled properly.

— **Conditions when Demanded.** Top Events S1 and S2 are asked for every sequence in the large LOCA event tree. Since both pumps depend on the RWST for a water source, both events are failed if the RWST is unavailable.

- **Scenario Impact if Successful.** If either Top Event S1 or S2 is successful, and the injection flow path is available (i.e., Top Event IP is successful), then RWST water is transferred to the RCS and eventually out the break to the containment.

- **Scenario Impact if Failed.** Failure of both Top Events S1 and S2 means that the safety injection pumps do not transfer the RWST inventory to the RCS and eventually to containment. This requires that the RHR or containment spray pumps function to transfer the RWST contents to containment.

- **Top Event IP — Safety Injection System Discharge Piping**

  - **Function Evaluated.** The availability of a flow path from the discharge of the safety injection pumps to the RCS cold legs.

  - **Success Criteria.** Success of Top Event IP requires that one of the safety injection pumps is available (i.e., Top Event S1 or S2 is successful), that the suction path from the RWST is available, and that at least one of three cold leg injection paths is available for 24 hours. The fourth injection path via RCS loop 4 is assumed to be the source of the LOCA and therefore is unavailable.

  - **Model Boundaries.** This top event models both the suction and discharge flow path to the safety injection pump trains modeled in Top Events S1 and S2. The discharge portion of the model considers flow from the safety injection system pump trains to the RCS cold leg loops, from the discharge path of the pumps modeled in Top Events S1 and S2 to the injection point in the four RCS cold legs downstream of check valves 63-551, 63-553, 63-555, and 63-557. The suction portion of the model considers the flow path from the RWST and through FCV-63-5 and check valve 63-510.

    The system boundary is defined so as to complete the flow path between the pump discharge and the RCS cold legs. The components modeled include:

    - The normally open valve FCV-63-22 where the flow from the two pumps is headered before splitting into four injection paths.

    - The four injection paths, each containing a throttle valve (FCV-63-550, FCV-63-552, FCV-63-554, and FCV-63-556) and associated check valve (63-551, 63-553, 63-555, and 63-557, respectively).

    - The four cold leg check valves (63-560, 63-561, 63-562, and 63-563).

The safety injection pump suction path modeled includes:

- The normally open valve FCV-63-5.
- Check valve 63-510.

There is no operator action required or modeled in this top event.

- **Conditions when Demanded.** Top Event IP is asked in all sequences of the large LOCA event tree. If the RWST is not available, or if both safety injection pumps fail, then Top Event IP is guaranteed failed.

- **Scenario Impact if Successful.** Success of Top Event IP, along with success of either or both of Top Events S1 and S2, means that the high head safety injection pumps inject RWST water into the RCS, which eventually flows into the containment via the break.

- **Scenario Impact if Failed.** Failure of this top event implies that injection flow from the safety injection system into the RCS during both injection and recirculation phases is not available.

- **Top Event LCL — Three of Three Cold Leg Accumulators Discharge**

  - **Function Evaluated.** Reflooding via the accumulators.

  - **Success Criteria.** The accumulator connecting to RCS loop 4 is assumed to spill out the break. Success of this top event requires successful injection from all three of the three remaining accumulators. This success criterion is believed to be conservative for large LOCAs, but no analysis is available to justify a relaxation. In the NUREG/CR-4550 analysis for the Surry plant, the three of three accumulators were assumed to be required for mitigation of large LOCAs.

  - **Model Boundaries.** This top event models the injection of the cold leg accumulators into the vessel. The model for each of the three accumulators on the intact lines includes the normally open isolation valve (FCV-63-118, FCV-63-98, or FCV-63-80), a check valve (63-622, 63-623, or 63-624), and one of the cold leg check valves (63-560, 63-561, or 63-562). The check valves closest to the RCS are shared with the cold injection flow path from the safety injection pumps; i.e., check valves 63-560, 63-561, and 63-562 appear in both this top event and in Top Event IP. Therefore, the availability of the accumulators is evaluated conditional on the status of Top Event IP. If Top Event IP is guaranteed failed because of a failure of the RWST or because both safety injection pumps fail, then a different split fraction is computed. For the latter case, even though Top Event IP is failed, no information about the status of the shared check valves is available.

  - **Conditions when Demanded.** Top Event LCL is asked for each sequence in the large LOCA event tree.

- **Scenario Impact if Successful.** The initial period of injection is successful. Continued inventory control may then be provided using the RHR pumps.

- **Scenario Impact if Failed.** Core damage is assumed due to failure of inventory control. RHR or high head injection via the charging and safety injection pumps would likely prevent subsequent vessel breach, but this is not modeled explicitly. Failure of sufficient accumulators to inject is conservatively modeled as a full core damage sequence.

- **Top Events RA and RB — Residual Heat Removal Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Manual or automatic start and operation of the RHR pumps.

  - **Success Criteria.** Success of each of these top events implies that the pump in that train is available to provide flow to the RHR discharge header to the RCS cold legs or to the high head systems during sump recirculation for 24 hours. Success of a train requires that ESFAS provide an automatic start signal, given a safety injection condition, or that the operators provide a manual start signal as a backup to the automatic start signal; i.e., via Top Event OS. Only train of RHR is required for successful low pressure injection and for recirculation from the containment sump.

  - **Model Boundaries.** These top events model the RHR pump trains 1A and 1B. The boundaries for these top events include the pump suction valves, FCV-74-3 and FCV-74-21, through the discharge valves, FCV-74-16 and FCV-74-28. The RHR pump miniflow lines, pump room cooling, and pump seal cooling are also included. The components within this boundary for train A (train B components are shown in parenthesis) include:

    - The pump suction valve FCV-74-3 (FCV-74-21).

    - Pump 1A-A (1B-B), discharge check valve 74-514, (74-515), and normally locked open manual valves 74-520 and 74-524 (74-521 and 74-525).

    - Miniflow line valve FCV-74-12 (FCV-74-24).

    - Heat exchanger outlet throttle valve FCV-74-16 (FCV-74-28).

    - Heat exchange outlet check valves 74-544 and 74-545.

    - The heat exchanger's failure because of rupture.

    - ERCW train A (B) supply to RHR pump 1A-A room cooler and the associated fan and valves.

    - CCS cooling to the RHR pump 1A-A (1B-B) mechanical seals and the associated valves.

The action for the operators to reset the safety injection signal and stop the RHR pumps if RCS pressure is > 180 psig to prevent pump overheating during extended operation on miniflow is not modeled for large LOCAs. Instead, RCS pressure should be low enough to permit low pressure injection before the operators complete this action.

Top Event RA is evaluated conditionally on the status of Top Event RB. This is to account for the intrasystem dependencies between trains; i.e., for common cause, test, and maintenance dependencies.

- **Conditions when Demanded.** These events are asked for all sequences of the event tree. The status of both trains is asked, even though only one train is required, because the RHR pump trains must work together with the corresponding sump recirculation path or train of containment spray.

  If the RWST is unavailable, both trains or RHR pumps are guaranteed failed.

- **Scenario Impact if Successful.** If Top Event RA or RB is successful, the corresponding train of RHR pumps is available. These top events are used with the availability of the RWST (Top Event RW), the RHR injection path (i.e., Top Event RF), containment sump recirculation (Top Event RR), hotleg recirculation (Top Event RH), and RHR spray (Top Event RS) to evaluate the availability of each of the functions performed by the RHR pumps.

- **Scenario Impact if Failed.** Failure of both of these top events (RA and RB) implies that neither train of the RHR system is available.

• **Top Event RF — RHR Injection Path**

- **Function Evaluated.** Availability of the flow paths from the RWST to the RHR pumps and from the RHR pumps to the RCS.

- **Success Criteria.** Success of Top Event RF requires that the suction line from the RWST to both RHR pumps be available and that a flow path to two of three RCS cold legs be available for 24 hours. The fourth RCS cold leg injection path (i.e., RCS loop 4) is assumed to be where the break occurs and therefore cannot be used for injection.

- **Model Boundaries.** This top event models the RHR flow paths from the pump discharge path modeled in Top Events RA and RB to the injection point in the three RCS cold legs downstream of check valves 63-551, 63-553, and 63-555. The suction line to the RHR pumps from the RWST used only for RHR supply during safety injection (FCV-63-1 and check valve 63-502) is also included here. The model includes the following components:

  - RWST to RHR pumps isolation valve FCV-63-1 and check valve 63-502.

  - RHR to cold legs isolation valves FCV-63-93 and FCV-63-94.

- The outer check valve on each RHR cold leg injection path: 63-632, 63-633, and 63-634.
- The three RCS check valves, 63-560, 63-561, and 63-562.

The last three RCS check valves listed above are also included in Top Events IP (safety injection pumps suction and cold leg discharge path) and CL (cold leg accumulators) since the RHR pumps and the safety injection pumps share common entry paths to the RCS. Therefore, the branch point values for Top Event RF are evaluated conditionally on the status of Top Event IP. If Top Event IP has failed, then a fraction of the time, this will have been due to the failure of the four common RCS check valves, which would also fail Top Event RF.

— **Conditions when Demanded.** Top Event RF is asked whenever Top Event RA or RB is successful. Top Event RF is guaranteed failed if Top Event RA fails. Failure of train A of RHR precludes injection via cold legs 1 and 2. Since injection via cold leg line 4 is precluded by the break location, only one injection line could possibly be used for injection from RHR pump B. Two cold leg lines are required for success.

— **Scenario Impact if Successful.** The RHR pump trains are then available to operate for low head injection and for low pressure recirculation from the containment sump.

— **Scenario Impact if Failed.** Failure of this top event implies that RHR is not available to directly inject into the RCS cold legs. Core damage results due to inadequate injection flow. This event is guaranteed failed if both Top Events RA and RB are failed or if Top Event RW is failed.

- **Top Event SU — Containment Sump Available**

— **Function Evaluated.** The availability of water in the containment sump for recirculation.

— **Success Criteria.** Given a loss of RCS inventory, success of Top Event SU implies that water is available for recirculation from the containment sump at the time that the RWST empties. A flow path for draining the containment spray water from the upper to the lower flow compartment must be available, and the containment sump itself must not be plugged. Water must have been injected directly or indirectly into the containment by one or more of the ECCS pumps; i.e., by the safety injection, RHR, or containment spray pumps. Melting of the ice alone is not sufficient to permit recirculation.

— **Model Boundaries.** This top event models the availability of the containment sump for recirculation. Failure of the sump can occur by the plugging of the drain between the upper and lower containment compartments or by plugging of the sump with containment debris.

The availability of the sump is modeled as the probability that the drain plugs were not removed during the last shutdown or that debris has blocked the sump. The containment spray pumps discharge into the upper containment. Melted ice and discharge from breaks in the RCS boundary are initially directed into the lower compartment. However, functioning spray pumps with failure of the drain plugs or of containment sump plugging will guarantee failure of recirculation.

- **Conditions when Demanded.** The containment sump top event, SU, is asked everywhere in the large LOCA event tree.

- **Scenario Impact if Successful.** Water is then assumed available for recirculation from the sump.

- **Scenario Impact if Failed.** RHR recirculation from the sump and containment spray recirculation from the sump are not available if Top Event SU fails. Therefore, Top Events RL, RVA, RVB, CH, RS, and RH are not asked if Top Event SU fails. Containment spray may still operate in the injection mode, taking suction from the RWST until it empties.

- **Top Event RL — Recirculation Level Instrumentation**

  - **Function Evaluated.** Instrumentation for swapover from the RWST to containment sump for pump suction.

  - **Success Criteria.** Two of the four channels of level instrumentation must be available to makeup the low-level swapover signal. The swapover is initiated when RWST level is less than the low-level setpoint, and containment sump level is greater than the required value during a LOCA.

  - **Model Boundaries.** This top event models the instrumentation necessary for switchover from the injection mode to the recirculation mode. The instrumentation modeled includes:

    - Containment sump level switches LS-63-50D, LS-63-51D, LS-63-52D, and LS-63-53D.

    - RWST level switches LS-63-180, LS-63-181, LS-63-182, and LS-63-183.

    An operator action is modeled to back up the level transmitters; i.e., action HARL1. This action is applicable for conditions when automatic swapover instrumentation fails due to conditions that can be easily detected; e.g., freezing of the sensor lines, a common relay failure, or if the operators had previously reset the safety injection signal, preventing the automatic swapover signal.

  - **Conditions when Demanded.** Top Event RL is asked for all sequences in the large LOCA event tree when the containment sump is available; i.e., when Top Event SU is successful.

- **Scenario Impact if Successful.** This implies that the automatic actuation signal for pump suction swapover to the containment sump is available to both trains of valves.

- **Scenario Impact if Failed.** Failure of this top event implies that automatic swapover has failed. Credit is given for initiation of manual swapover if automatic swapover fails to actuate due to relay failure or gross transmitter failure; e.g., RWST level transmitter freeze. Therefore, failure of Top Event RL implies that recirculation from the containment sump for both core cooling, and for spray recirculation are unavailable.

- **Top Events RVA and RVB — Trains A and B RHR Sump Swapover Valves**

  - **Function Evaluated.** Response of ECCS valves to a pump suction swapover demand from the RWST to the containment sump.

  - **Success Criteria.** The containment sump suction valve must open and the RWST suction valve on the associated train must close for success of the train.

  - **Model Boundaries.** These top events model the train A and B swapover valves of the RHR system, which realign automatically to allow pump suction from the containment sump once the RWST level is low. The swapover is initiated by the instrumentation modeled in Top Event RL on RWST level less than the low-level setpoint and containment sump level greater than the required value during a LOCA. The equipment modeled includes:

    - The motor-operated sump swapover valves 1-FCV-63-72 for train A and 1-FCV-63-73 for train B, which must open.

    - Valves 1-FCV-74-3 for train A and 1-FCV-74-21 for train B, which must close.

    Top Event RVB is evaluated conditionally on the status of Top Event RVA. This is because the suction valves in the different trains, both to the RWST and to the containment sump, are in common cause groups. Therefore, the failure probability of the second train is higher if the first train fails than would otherwise be expected, assuming that the trains were independent. These dependencies are accounted for in the systems models.

  - **Conditions when Demanded.** These events are asked for sequences in the large LOCA event tree in which Top Event SU is successful. Failure of the containment sump makes the question of valve swapover immaterial. If the swapover instrumentation is unavailable (i.e., Top Event RL fails), then both of these events (i.e., the two trains of sump swapover valves) are set to guaranteed failed.

  - **Scenario Impact if Successful.** Success of these valves means that the containment sump suction path is aligned for recirculation on the associated train. For mitigation of large LOCAs, only low pressure recirculation is required for core cooling; i.e., high pressure recirculation using the safety injection or charging pumps is assumed to be unnecessary because the RCS pressure is low by the time that the RWST reaches low level.

— **Scenario Impact if Failed.** Failure of these valves to go to the correct position will fail the corresponding RHR pump train during the recirculation mode. The physical arrangement of these valves at Watts Bar also causes the associated train of containment spray to be unavailable, given the failure of this top event.

- **Top Event RR — RHR Recirculation**

— **Function Evaluated.** This top event models the automatic/manual swapover of the RHR suction to the containment sump for recirculation.

— **Success Criteria.** Successful low pressure recirculation mode requires: (1) the automatic transfer of the suction of RHR from the RWST to the containment sump (Top Events RL, RVA, and RVB), and (2) use of the RHR heat exchangers to remove heat from the containment sump recirculation coolant. Either of two RHR trains with cooling to the associated heat exchanger is considered success. The swapover must be completed within 20 minutes of the low-level setpoint being reached to prevent core uncovery.

— **Model Boundaries.** The RHR configuration during recirculation from the sump allows: (1) the RHR pumps to draw coolant from the containment sump, (2) the RHR heat exchangers to transfer heat from the coolant to the CCS, (3) the coolant to be pumped to the cold legs of the RCS, and (4) for RHR containment spray in the event containment spray recirculation is not available.

The recirculation mode is initiated automatically when the RWST reaches a low level coincident with a high containment sump level, as modeled in Top Event RL. The RHR pumps will then automatically switch their suction from the RWST to the containment sump as modeled in Top Events RVA and RVB.

During recirculation, component cooling water flow is manually established to the RHR heat exchangers to cool the flow from the containment sump before being discharged into the RCS or for RHR.

The model for this top event includes the following equipment:

- The RHR suction valves from the RWST (LCV-62-135, LCV-62-136, and FCV-63-1).
- The component cooling water and control valves to the RHR heat exchangers (FCV-70-153 and FCV-70-156).

The recirculation valves for the swapover of the containment spray pumps is modeled separately in Top Event CH.

Top Event RR is used with the containment sump availability Top Event SU, the RHR pump, flow path, and RHR spray top events (i.e., RA, RB, RF, and RS) to determine whether flow is available to the RCS from one of two RHR pumps during low pressure recirculation.

The operator action to align the CCS to the RHR heat exchangers is modeled in this top event. Power must also be restored to FCV-63-1, so that this RWST suction valve may be isolated. The action is designated as RR1. Credit for this action is only modeled if RWST level instrumentation is available; i.e., Top Event RL succeeds. The action evaluated conditionally on the failure of Top Event RL (action HARR2) is not used.

- **Conditions when Demanded.** Whenever the containment sump is available for sump recirculation (i.e., Top Event SU succeeds), this event is asked in the large LOCA event tree. If a combination of RHR sump swapover valve trains or the RHR pumps themselves lead to the unavailability of both trains, then this event is guaranteed failed.

- **Scenario Impact if Successful.** Implies that there has been a successful transition to low pressure recirculation from the containment sump. Discharge from one or both RHR pumps, taking suction from the sump, is directed to the cold leg injection lines of the RCS.

- **Scenario Impact if Failed.** Failure of this top event implies that there is no recirculation cooling of the core, and that RHR spray recirculation is not possible.

- **Top Events CSA and CSB — Containment Spray Pumps 1A-A and 1B-B**

  - **Function Evaluated.** Containment spray pump availability for injection and recirculation.

  - **Success Criteria.** The associated containment spray pump train is automatically or manually actuated and provides spray injection. The pump operates for 24 hours. Manual actuation must be completed within 20 minutes of reaching the automatic actuation setpoint. Only one train of containment spray is required for containment spray in the injection mode.

  - **Model Boundaries.** This top event models the availability of the containment spray pump trains 1A-A and 1B-B to deliver flow into containment, given a suction source. In the injection mode, suction is from the RWST (Top Event RW), and during recirculation, suction is from the containment sump. Heat removal from the containment spray heat exchangers is modeled separately in Top Event CH. Top Events CSA and CSB are both actuated automatically by a high-high containment pressure signal, which is assumed reached for all large LOCAs.

The equipment modeled in Top Event CSA (CSB equipment is shown in parentheses) includes:

- The containment isolation valve FCV-72-39 (FCV-72-2).
- Containment spray pump 1A-A (1B-B).
- Miniflow valve 1-72-34 (1-72-13).
- The RWST suction valve FCV-72-22 (FCV-72-21).
- Normally open manual valve 1-72-528 (1-72-529) and check valves 1-72-506 (1-72-507), 1-72-524 (1-72-525), and 1-72-547 (1-72-548).
- Containment sump recirculation valve 1-FCV-72-44 (1-FCV-72-45), which must remain closed.
- The physical integrity of the containment spray heat exchanger 1A (1B), CCS supply to the containment spray pump 1A-A (1B-B) oil coolers including the associated valves.
- ERCW supply to the containment spray pump 1A-A (1B-B) room coolers including the associated valves.
- The 263 nozzles of the spray header 1A (1B).

The operator action to initiate containment spray manually, given a high-high containment pressure condition but failure of the automatic actuation signal from ESFAS, is included in the models for both top events; i.e., via action HACS1.

Containment spray is unavailable if the RWST is not available. Train B of containment spray, as represented by Top Event CSB, is evaluated conditionally on the status of Top Event CSA to reflect the potential for common cause affecting both trains and the maintenance limits imposed by plant technical specifications.

– **Conditions when Demanded.** Top Events CSA and CSB are asked for all sequences in the large LOCA event tree.

– **Scenario Impact if Successful.** The containment spray system acts with the ice condenser system to provide short-term containment heat removal and to limit the containment pressure increase. Containment spray provides a long-term source of containment heat removal while in the sump recirculation mode. The long-term function of containment spray (i.e., in the recirculation mode) requires the success of at least one of Top Event CSA or CSB and of Top Event CH, which is considered next.

- **Scenario Impact if Failed.** Failure of Top Event CSA or CSB implies that the associated train of containment spray is not available in either the injection mode or the subsequent recirculation mode. Even if both trains fail, the containment would still not overpressurize during the injection phase for any size LOCA.

- **Top Event CH — Containment Spray in Recirculation Mode**

  - **Function Evaluated.** Recirculation mode of containment spray.

  - **Success Criteria.** Success of this event requires that at least one train of containment spray is successful and of the associated spray pumps to operate in the injection mode. ERCW cooling must be aligned to the containment spray heat exchangers. The action to accomplish the swapover (action HACH1) is initiated when RWST is less than the low-low level setpoint, and containment pressure is greater than the high-high pressure setpoint. It must be completed before the pumps lose suction due to the RWST emptying, assumed to be about 5 minutes after the low-low level setpoint is reached.

  - **Model Boundaries.** This top event models the operator actions and equipment required for containment spray to successfully operate in the recirculation mode.

    The equipment modeled in Top Event CH includes:

    - The containment isolation valves FCV-72-39 and FCV-72-2.
    - The train A and train B containment spray pumps (CSA and CSB).
    - The RWST suction manual valves FCV-72-22 and FCV-72-21.
    - The containment sump suction valves FCV-72-44 and FCV-72-45.
    - ERCW cooling to the containment spray heat exchangers.
    - ERCW heat exchanger inlet isolation valves FCV-67-125 and FCV-67-123.
    - ERCW heat exchanger outlet check valves 67-537A and 67-537B.
    - ERCW heat exchanger outlet isolation valves FCV-67-126 and FCV-67-124.

    The operator action to align the ERCW supply to the containment spray heat exchangers and to transfer the suction of the containment spray system from the RWST to the containment sump is modeled with this top event. The operators must stop the operating containment spray pumps, close the pump suction valves from the RWST, and open the pump suction valves from the RWST. The operator then checks for adequate ERCW flow and cooling to the containment spray heat exchangers, and restarts the spray pumps. These actions are initiated when the RWST low-low level setpoint is reached.

The automatic swapover on low RWST level and high sump level (i.e., Top Event RL) must be successful. The sump swapover valves (Top Events RVA and RVB) and the containment spray pumps in the injection mode (i.e., Top Events CSA and CSB) must also be successful for the corresponding train of spray to operate in the recirculation mode.

– **Conditions when Demanded.** Top Event CH is asked for all sequences in the large LOCA event tree for which the containment sump is available; i.e., when Top Event SU succeeds.

– **Scenario Impact if Successful.** Success of this top event implies that at least one train of containment spray is operating, taking suction from the containment sump, with its associated heat exchanger being properly supplied with ERCW. Containment heat removal is available. Spray recirculation from the RHR pumps discharge is not required.

– **Scenario Impact if Failed.** If this top event is not successful, containment heat removal requires successful operation of RHR spray, as modeled in Top Event RS.

- **Top Event RS — RHR Spray**

– **Function Evaluated.** Spray recirculation using RHR pump discharge.

– **Success Criteria.** One of the two RHR pump trains must operate in the recirculation mode, containment pressure must exceed the required value more than 1 hour into the accident, and the operators then align one train of RHR for containment spray recirculation. The alignment must be made before the containment overpressurizes, estimated to be several hours into the accident.

– **Model Boundaries.** This event models the containment spray recirculation function of the RHR system. It includes the operator action and the opening of the RHR ring header inlet motor-operated valve. The operators establish one (and only one) train of RHR spray if containment pressure is greater than the required value and the accident is at least 1 hour old. The action modeled is designated HARS1.

The equipment modeled in this top event includes:

- RHR crosstie valves FCV-74-33 and FCV-74-35.
- The RHR injection path isolation valves FCV-74-93 and FCV-74-94.
- The RHR spray control valves FCV-72-40 and FCV-72-41.

– **Conditions when Demanded.** This event is asked only if normal containment spray recirculation, as represented by Top Event CH, has failed. It is used with the containment sump availability, Top Event SU, the containment sump recirculation Top Event RR, and the RHR pump train Top Events RA and RB. It is not asked if Top Event RR has failed. Top Event RR asks about the valves for alignment of RHR recirculation, and of the alignment of valves to establish CCS flow to the RHR heat exchangers for heat removal.

Credit for this action is taken if at least one RHR pump train is available, at least one charging pump, and at least one safety injection pump are available. Procedural guidance only instructs the operators to use RHR spray if all of these conditions are met. Note that when the RHR trains are aligned for hot leg recirculation, one of the RHR spray paths will be unavailable.

- **Scenario Impact if Successful.** Thus success of this top event, RS, implies that at least one train of RHR is providing containment spray (provided that Top Events SU, RR, and RA or RB are successful).

- **Scenario Impact if Failed.** Failure of this top event implies that containment heat removal from any containment spray system is not available.

- **Top Event RH — RHR Hot Leg Recirculation**

  - **Function Evaluated.** Alignment of RHR recirculation for injection to the hot legs.

  - **Success Criteria.** The realignment of one operating RHR recirculation path for injection, via the hot legs rather than the cold legs, about 15 hours after plant trip, is required for success of Top Event RH. Common hot leg injection valve FCV-63-172 must open. Realignment of the safety injection pumps for hot leg recirculation is assumed unnecessary.

  - **Model Boundaries.** This top event models the use of RHR for RCS hot leg recirculation. Hot leg recirculation is used to limit the amount of boron precipitating out, during a LOCA in which the RCS cannot be kept full, so that heat removal is accomplished by boiling. Excessive boron precipitation is postulated to interfere with heat transfer and reactivity control. The operators are instructed to divert the cold leg injection flow to hot leg recirculation 15 hours after transferring to containment sump recirculation.

    The model includes the following equipment:

    - The RHR spray isolation valves FCV-72-40 and FCV-72-41.
    - The RHR cold leg isolation valves FCV-74-93 and FCV-74-94.
    - The RHR crosstie valves FCV-74-33 and FCV-74-35.
    - The hot leg injection isolation valve FCV-63-172.

  - **Conditions when Demanded.** This event is asked for all sequences in the large LOCA event tree in which RHR containment sump recirculation is successful, but normal containment spray recirculation is unavailable; i.e., Top Event CH has failed. Top Events SU, RR, and at least one of RA or RB must be available.

- **Scenario Impact if Successful.** Successful switchover to hot leg recirculation implies that the problems associated with boron precipitation are not of concern.

- **Scenario Impact if Failed.** Failure of this top event implies that hot leg recirculation has failed. This is conservatively modeled as leading to fuel damage due to eventual flow blockage. Large LOCA-initiated sequences involving failure of Top Event RH are assigned to core damage end states.

• **Top Event AR — Containment Air Return Fans**

- **Function Evaluated.** Performance of the containment air return fans.

- **Success Criteria.** The success criterion for the air return fans is that one of the two fans function. The fans are automatically actuated on a high-high containment pressure signal from ESFAS.

- **Model Boundaries.** This top event models the containment air return fans. The air return fans circulate hot saturated air from the upper compartment after a LOCA (10 minutes after high-high containment pressure). The air return fans enhance heat removal from the lower compartment to the ice condenser to help lower containment pressure. All portions of the air return fan functions are modeled. A manual start action to backup the ESFAS actuation is not modeled.

- **Conditions when Demanded.** Top Event AR is asked for each sequence in the large LOCA event tree because a high-high pressure condition should always be reached.

- **Scenario Impact if Successful.** The lower compartment containment pressure rise is mitigated by the successful operation of the air return fans.

- **Scenario Impact if Failed.** The lower compartment containment pressure is not mitigated, and local hydrogen pockets may develop due to poor mixing.

• **Top Event CI — Containment Isolation**

- **Function Evaluated.** Isolation of small containment penetrations.

- **Success Criteria.** Each of the small containment penetrations listed below must be either closed at the time of the accident and remain closed, or close by a signal from ESFAS based on a safety injection, Phase A, or CVI signal, or on Phase B isolation. For station blackout sequences, the time assumed to be available for locally isolating the seal return line is 3 hours.

- **Model Boundaries.** This top event models containment isolation of nonessential penetrations during accident conditions. The containment penetrations explicitly modeled are as follows:

    • Containment major vents and drains.

- Connections to the RCS.
- Connections to containment atmosphere, with the exception of the large penetrations modeled in Top Event CP.

This containment isolation top event models only those containment penetrations whose failure to isolate would result in a release path which would bypass containment. The following questions were asked about each penetration to determine the need for inclusion. Only those penetrations not covered by other system analyses in the PRA were considered.

- Does the penetration communicate directly with the outside environment?
- Does the penetration communicate with the environment via a low pressure system or a tank with a relief valve?
- Will the relief valve lift at a pressure below the ultimate containment pressure?
- Is the system or tank design pressure below the ultimate containment pressure?

Based on these questions, the following penetrations are included in this top event:

- Floor sump pump discharge (X-41).
- RC drain tank and pressurizer vent to VH (X-45).
- RC drain tank pump discharge (X-46).
- Lower compartment pressure relief (X-80).
- RC drain tank to gas analyzer (X-81).
- Upper compartment air monitor intake (X-94A/B).
- Upper compartment air monitor return (X-94C).
- Lower compartment air monitor intake (X-95A/B).
- Lower compartment air monitor return (X-95C).
- RCP seal return line.

All of these penetrations receive a signal to isolate, given a safety injection, Phase A, or CVI signal, except for the RCP seal return line. The RCP seal return line is automatically signaled to close on high-high containment pressure; i.e., Phase B isolation. During station blackout conditions, the operator is required to isolate locally the motor-operated seal injection and return valves in the RCP seal return line. This action (i.e., action CI1) is included in the model.

— **Conditions when Demanded.** Top Event CI is asked for every sequence in the large LOCA event tree. The status of containment isolation is needed for the containment analysis.

- **Scenario Impact if Successful.** The smaller containment penetrations modeled in Top Event CI are isolated.

- **Scenario Impact if Failed.** One or more of the smaller containment penetrations listed above must have been opened initially and failed to close.

• **Top Event CP — Containment Purge Isolation**

- **Function Evaluated.** Isolation of the containment purge lines.

- **Success Criteria.** Success of this top event requires that either (1) the purge system was not in use when required, or (2) it was in use and that at least one valve in each penetration line closed.

- **Model Boundaries.** This top event models the isolation of the containment purge penetrations, which are allowed to be opened during power operation. The plant Technical Specifications allow these penetrations to be opened up to 1,000 hours per year with the plant at power. The penetrations modeled are as follows:

  - Lower compartment purge air exhaust (X-4).
  - Instrument room purge air exhaust (X-5).
  - Upper compartment purge air exhaust (X-6).
  - Upper compartment purge air exhaust (X-7).
  - Upper compartment purge air supply (X-9A).
  - Upper compartment purge air supply (X-9B).
  - Lower compartment purge air supply (X-10A).
  - Lower compartment purge air supply (X-10B).
  - Instrument room purge air supply (X-11).

  The penetrations modeled by Top Event CP are treated separately from those in Top Event CI due to the larger size of the purge penetrations. Even though the large size of the purge penetrations may limit the containment pressure rise during a LOCA, the Phase A and Phase B isolation signal setpoints are low enough that these isolation signals should occur even if the purge lines are initially open.

  A backup manual action to isolate these penetrations is also considered, per the status of Top Event OS.

- **Conditions when Demanded.** Top Event CP is asked in all sequences of the large LOCA event tree. Even if the automatic and manual isolation signals fail, this event may still be successful because the penetrations included in the model are normally closed anyway.

- **Scenario Impact if Successful.** Success of Top Event CP implies that the containment has at most a small hole in it. If Top Event CI is also successful, then the containment is isolated.

- **Scenario Impact if Failed.** Failure of this top event implies that containment isolation has failed and that a large hole in the containment boundary is present.

- **Top Event HH — Hydrogen Igniters**

  - **Function Evaluated.** Hydrogen control using the hydrogen igniters.

  - **Success Criteria.** Success of this top event implies that all 34 igniters in one of two trains functioned.

  - **Model Boundaries.** This top event models the hydrogen igniters of the hydrogen mitigation system. During events that involve fuel cladding damage, the hydrogen igniters are used to burn away the hydrogen before it reaches explosive concentrations when it mixes with the containment atmosphere.

    The system consists of two trains of hydrogen igniters and the associated control circuitry. The system is manually initiated from the control room upon receipt of a Phase B signal and hydrogen concentration as indicated by the hydrogen analyzer is within the required range. The action modeled is designated HH1. There is no time pressure to complete this action; i.e., many hours are assumed to be available. The operator is also required to place the hydrogen analyzer in service before the igniters are initiated. Use of the hydrogen analyzer is included as part of the operator action to initiate the hydrogen igniters. During recovery from an initial station blackout, the hydrogen igniters are not to be placed in service if the hydrogen analyzers indicate that the hydrogen concentration exceeds the required value.

  - **Conditions when Demanded.** The hydrogen igniters are asked for in all sequences of the large LOCA event tree. The status of Top Event HH is used in the evaluation of containment performance.

  - **Scenario Impact if Successful.** The hydrogen igniters are available to continuously burn off the hydrogen, which collects in the containment prior to the concentration of hydrogen reaching explosive concentrations.

  - **Scenario Impact if Failed.** The hydrogen igniters are not available to reduce the concentration of hydrogen within containment.

3.1.2.3 References

3.1.2-1. TVA DNE Calculation, "Pressure-Temperature Limits Based on Regulatory Guide 1.99,R2, For Submittal to the NRC," QN-RC-DO53-MTB-WAP-052589, for Sequoyah Units 1 and 2, Revision 1, June 27, 1989.

3.1.2-2. Cheung, A. C., et al., "A Generic Assessment of Significant Flaw Extension, including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants," Westinghouse Electric Corporation, WCAP-10319, December 1983.

Table 3.1.2-1 (Page 1 of 2). Relationship of EOPs to Watts Bar Nuclear Plant ESDs		
Procedure	**Number**	**ESD Transfer Number**
Reactor Trip or Safety Injection	E-0 (4)	Start, 1
Reactor Trip Response	ES-0.1 (3)	2
Safety Injection Termination	ES-0.2 (3)	3
Natural Circulation Cooldown	ES-0.3 (3)	4
Loss of Reactor or Secondary Coolant	E-1 (3)	5
Post LOCA Cooldown	ES-1.1 (3)	6
Transfer to Containment Sump	ES-1.2 (4)	7
Transfer to Hot Leg Recirculation	ES-1.3 (2)	8
Faulted Steam Generator Isolation	E-2 (3)	9
Steam Generator Tube Rupture (SGTR)	E-3 (3)	10
Safety Injector Termination Following SGTR	ES-3.1 (3)	11
Post SGTR Cooldown Using Backfill	ES-3.2 (3)	12
Post SGTR Cooldown By Ruptured S/G Depressurization	ES-3.3 (3)	13
Loss of Shutdown Power	ECA-0.0 (1)	14
Loss of Shutdown Power Recovery without Safety Injection Required	ECA-0.1 (1)	15
Loss of Shutdown Power Recovery with Safety Injection Required	ECA-0.2 (1)	16
Loss of RHR Sump Recirculation	ECA-1.1 (0)	17
LOCA Outside Containment	ECA-1.2 (0)	18
Uncontrolled Depressurization of all Steam Generators	ECA-2.1 (0)	19
SGTR and LOCA - Subcooled Recovery	ECA-3.1 (0)	20
SGTR and LOCA - Saturated Recovery	ECA-3.2 (0)	21
SGTR without Pressurizer Pressure Control	ECA-3.3 (0)	40
Critical Safety Function Status Trees:	FR-0 (4)	N/A
Subcriticality	1FRS-F.0-1	N/A
Core Cooling	1FRC-F.0-2	N/A
Heat Sink	1FRH-F.0-3	N/A

Table 3.1.2-1 (Page 2 of 2). Relationship of EOPs to Watts Bar Nuclear Plant ESDs		
Procedure	**Number**	**ESD Transfer Number**
Pressurized Thermal Shock	1FRP-F.0-4	N/A
Containment	1FRZ-F.0-5	N/A
Inventory	1FRI-F.0-6	N/A
Nuclear Power Generation/ATWS (red)	FR-S.1 (3)	22
Loss of Core Shutdown	FR-S.2 (1)	41
Inadequate Core Cooling (red)	FR-C.1 (4)	23
Saturated Core Cooling	FR-C.2 (1)	24
Response to Loss of Secondary Heat Sink (red)	FR-H.1 (2)	25
Steam Generator Overpressure	FR-H.2 (1)	26
Steam Generator High Level	FR-H.3 (1)	27
Loss of Normal Steam Release Capabilities	FR-H.4 (1)	28
Steam Generator Low Level	FR-H.5 (1)	29
Pressurized Thermal Shock (red)	FR-P.1 (1)	30
Cold Overpressure Condition	FR-P.2 (1)	31
Phase B Containment Pressure (red)	FR-Z.1 (3)	32
Containment Flooding	FR-Z.2 (1)	33
High Containment Radiation	FR-Z.3 (1)	34
High Pressurizer Level	FR-I.1 (1)	35
Low Pressurizer Level	FR-1.2 (1)	36
Voids in Reactor Vessel	FR-1.3 (1)	37
Loss of Offsite Power	AOI-35	39

Table 3.1.2-2 (Page 1 of 2). Relationship of Event Tree Top Events to Watts Bar Nuclear Plant EOPs

Procedure Number (Revision)	Top Events Modeled
E-0 (4)	RT, TT, MS, AFW, CD, OG, AA, BA, ECCS, CI, CP, OS, MF, IE, SE, RA, RB, RR, WC, PR, AC, SL, AR, CSA, CSB
ES-0.1 (3)	MF, AFW, OS, OF, CD, SE, IE, PR, WC, WC, PR, CD, CT, AFW, SE, IE, CT, AFW, SE
ES-0.2 (3)	MF, AFW, OS, OF, CD, SE, IE, PR, WC, WC, PR, CD, CT, AFW, SE, IE, CT, AFW, SE
ES-0.3 (3)	MF, AFW, OS, OF, CD, SE, IE, PR, WC, WC, PR, CD, CT, AFW, SE, IE, CT, AFW, SE
E-1 (3)	IE, SE, CTMU, AFW, OT, RA, RB, RI, RR, OG, PR, ECCS, HH, VSEQ, CSA, CSB, CH, RS, RH, CL
ES-1.1 (3)	RR, IE, DS, DP, CD, OG, RR, RA, RB, PR, SE, RD
ES-1.2 (4)	SU, RR, RA, RB, RL, RVA, RVB, RH, IE, CH
ES-1.3 (2)	RH, IE, SE, PR
E-2 (3)	IE, TT, MS, SL, AFW
E-3 (3)	SL, TP, DS, CD, DP, PI
ES-3.1 (3)	WC, PR, SE, IE, SL, DS, DP, PI
ES-3.2 (3)	PR, SL, SE, DS, CD, DP, RD, PI
ES-3.3 (3)	-
ECA-0.0 (1)	PR, AFW, RE, CI, SL, IE, CTMU, DS, SE
ECA-0.1 (1)	-
ECA-0.2 (1)	-
ECA-1.1 (0)	RE, MU, DS, DP
ECA-1.2 (0)	-
ECA-2.1 (0)	SL, IE, SE, CTMU, PR, RA, RB, OT, CSA, CSB, ECCS, WC
ECA-3.1 (0)	DP, OG, RA, RB, OT, CTMU, AFW, SL, DS, CD, IE
ECA-3.2 (0)	MU, RA, RB, CT, AFW, DS, DP, CD, ECCS, RD
ECA-3.3 (0)	-
FR-0 (4)	-
FRS-F.0-1	-

Table 3.1.2-2 (Page 2 of 2). Relationship of Event Tree Top Events to Watts Bar Nuclear Plant EOPs

Procedure Number (Revision)	Top Events Modeled
FRC-F.0-2	-
FRH-F.0-3	-
FRP-F.0-4	-
FRZ-F.O-5	-
FRI-F.O-6	-
FR-S.1 (3)	RT, FW, AM, TT, PL, MR, AFW, OS, SR, SL, EB
FR-S.2 (1)	-
FR-C.1 (4)	-
FR-C.2 (1)	ECCS, RR, HH
FR-H.1 (2)	-
FR-H.2 (1)	AFW, PR, CT, AFW, OF, MF, OB
FR-H.3 (1)	-
FR-H.4 (1)	-
FR-H.5 (1)	-
FR-P.1 (1)	-
FR-P.2 (1)	-
FR-Z.1 (3)	-
FR-Z.2 (1)	CSA, CSB, CI, OS, MS, SE, AR, HH, RS
FR-Z.3 (1)	-
FR-I.1 (1)	-
FR-I.2 (1)	-
FR-I.3 (1)	-
CORE DAMAGE	-
AOI-35 ()	IE, PR, CSA, CSB, R

Note: N/A = Not applicable; not explicitly identified in the ESDs.

Table 3.1.2-3 (Page 1 of 2). Procedural Guidance Associated with Each Top Event	
Top Event Identifier	Related Procedures
AA, AB	E-0
AC	E-0
AE	E-0, AOI-35
AFW (MA, MB, TP, AF)	E-0, ES-0.1, ES-0.2, ES=0.3, E-1, E-2, E-3, ECA-0.0, ECA-3.1, ECA-3.2, FR-S.1, FR-H.1
AM	FR-S.1
AR	E-0, FR-Z.1
CD	E-0, ES-0.1, ES-0.2, ES-1.1, E-3, ES-3.2, ECA-3.1, ECA-3.2
CH	E-1, ES-1.2
CI	E-0, ECA-0.0, FR-Z.1
CL	E-1
CP	E-0
CSA/CSB	E-0, E-1, ECA-2.1, FR-Z.1, AOI-35
CTMU	ES-0.2, ES-0.3, E-1, ECA-0.0, ECA-2.1, ECA-3.1, ECA-3.2, FR-H.1
CV/CB	-
DS/DP	ES-1.1, E-3, ES-3.1, ES-3.2, ECA-0.0, ECA-1.1, ECA-3.1, ECA-3.2
EB	FR-S.1
FW	FR-S.1
HH	E-1, FR-C.1, FR-Z.1
IE	E-0, ES-0.1, ES-0.2, E-1, ES-1.1, ES-1.2, ES-1.3, E-2, ES-3.1, ECA-0.0, ECA-2.1, ECA-3.1, AOI-35
MF	E-0, ES-0.1, FR-H.1
MR	FR-S.1
MS	E-0, E-2, FR-Z.1
MU	ECA-1.1, ECA-3.2
OB	FR-H.1
OF	ES-0.1, FR-H.1
OG	E-0, E-1, ES-1.1, ECA-3.1
OS	E-0, ES-0.1, FR-S.1, FR-Z.1
OT	E-1, ECA-2.1, ECA-3.1
PI	E-3, ES-3.1, ES-3.2
PL	FR-S.1

Table 3.1.2-3 (Page 2 of 2). Procedural Guidance Associated with Each Top Event

Top Event Identifier	Related Procedures
PR	E-0, ES-0.1, ES-0.2, E-1, ES-1.1, ES-1.3, ES-3.1, ES-3.2, ECA-0.0, ECA-2.1, FR-H.1, AOI-35
RA/RB	E-0, E-1, ES-1.1, ES-1.2, ECA-2.1, ECA-3.1, ECA-3.2, FR-C.1
RD	ES-1.1, ES-3.2, ECA-3.2
RE	ECA-0.0, ECA-1.1, AOI-35
RF	E-1
RH	E-1, ES-1.2, ES-1.3
RI	E-1
RL	ES-1.2
RR	E-0, E-1, ES-1.1, ES-1.2, FR-C.1
RS	E-1, FR-Z.1
RT	E-0, FR-S.1
RVA/RVB	ES-1.2
S1/S2/SP/SI/IP	E-0, E-1, ECA-2.1, ECA-3.2, FR-C.1
SE	E-0, ES-0.1, ES-0.2, ES-0.3, E-1, ES-1.1, ES-1.3, ES-3.1, ES-3.2, ECA-0.0, ECA-2.1, FR-Z.1
SL	E-0, E-2, E-3, ES-3.1, ES-3.2, ECA-0.0, ECA-2.1, ECA-3.1, FR-S.1
SR	FR-S.1
SS	-
SU	ES-1.2
TB	-
TT	E-0, E-2, FR=S.1
VP, VC, VA, VB, VF	E-0, E-1, ECA-2.1, ECA-3.2, FR-C.1
VSEQ	E-1
WC	E-0, ES=0.1, ES-0.2, ES-3.1, ECA-2.1

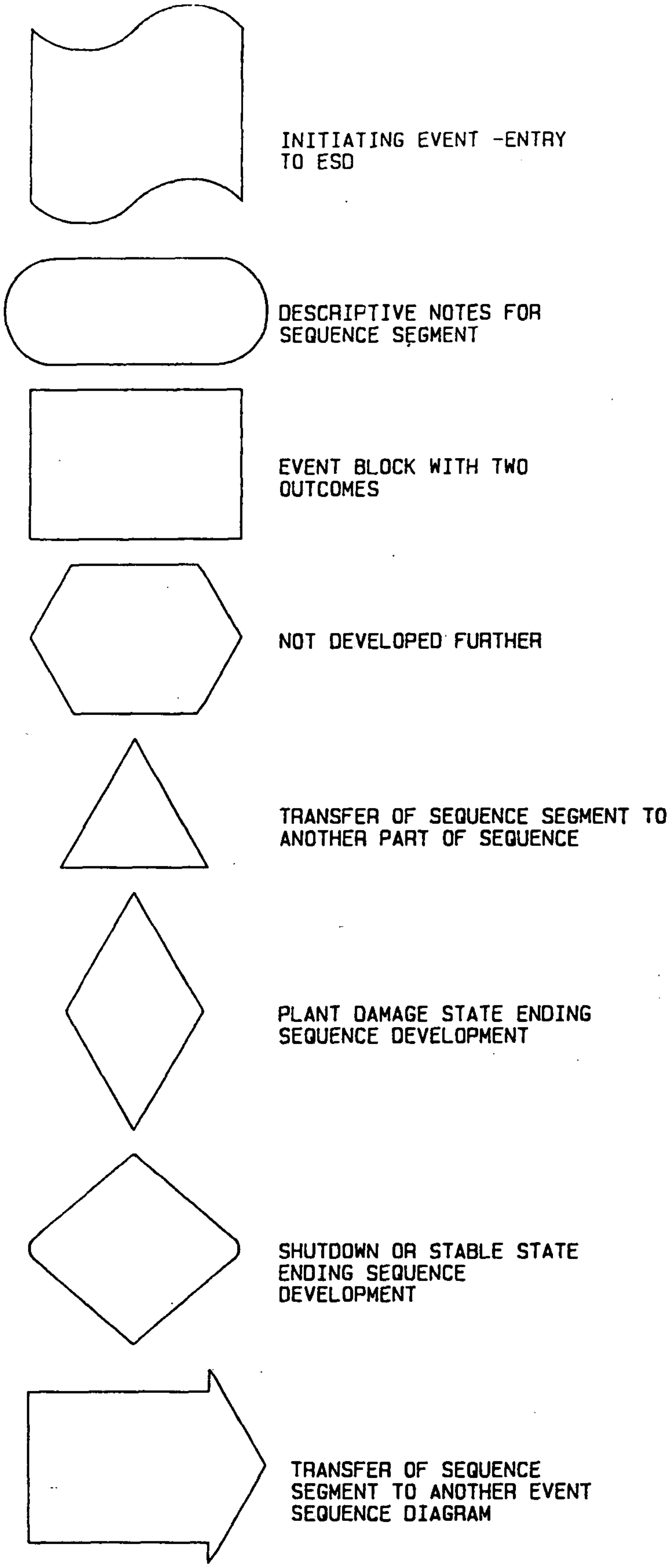

Figure 3.1.2-1. Event Sequence Diagram Symbology

SI APERTURE CARD

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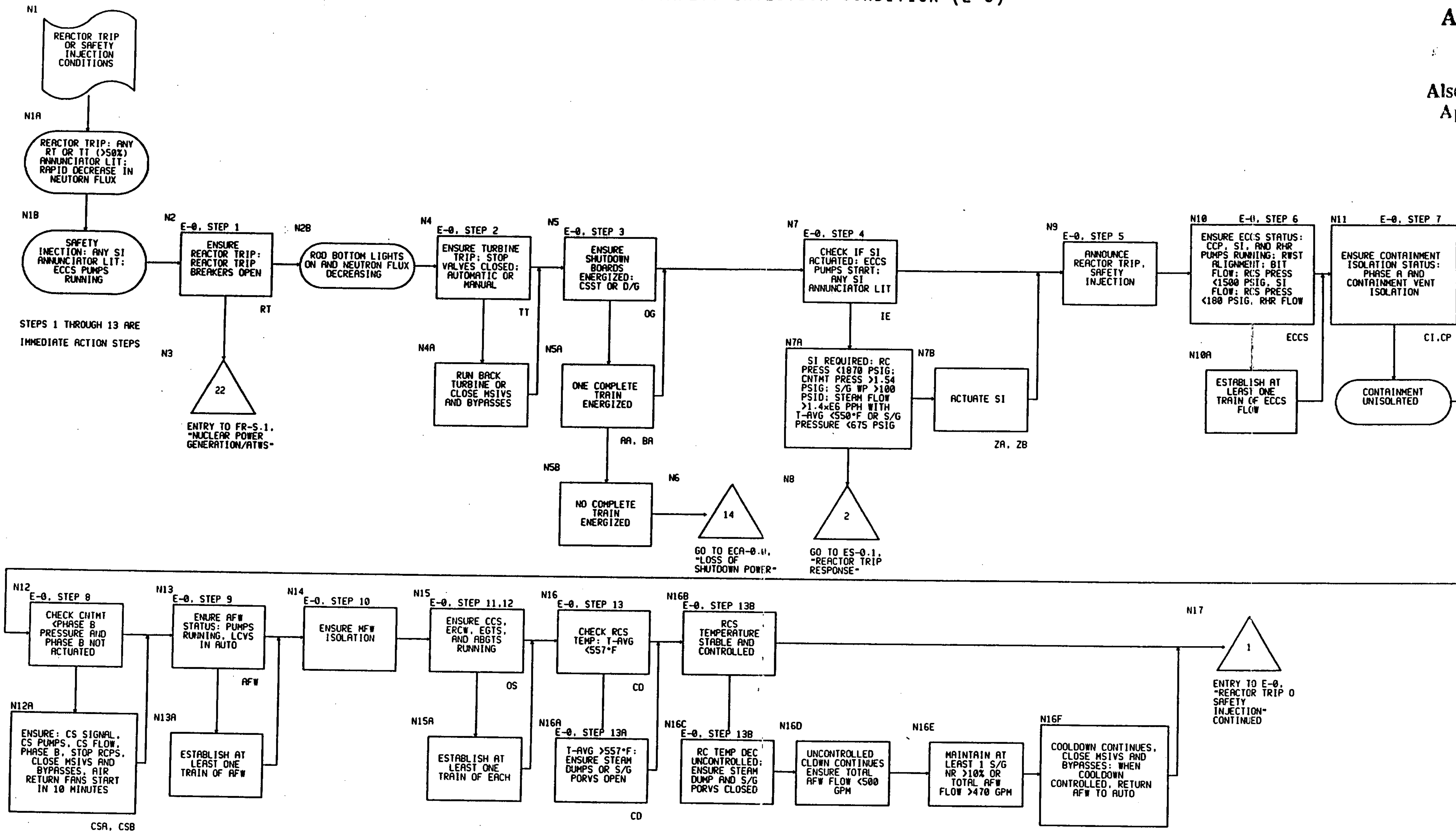
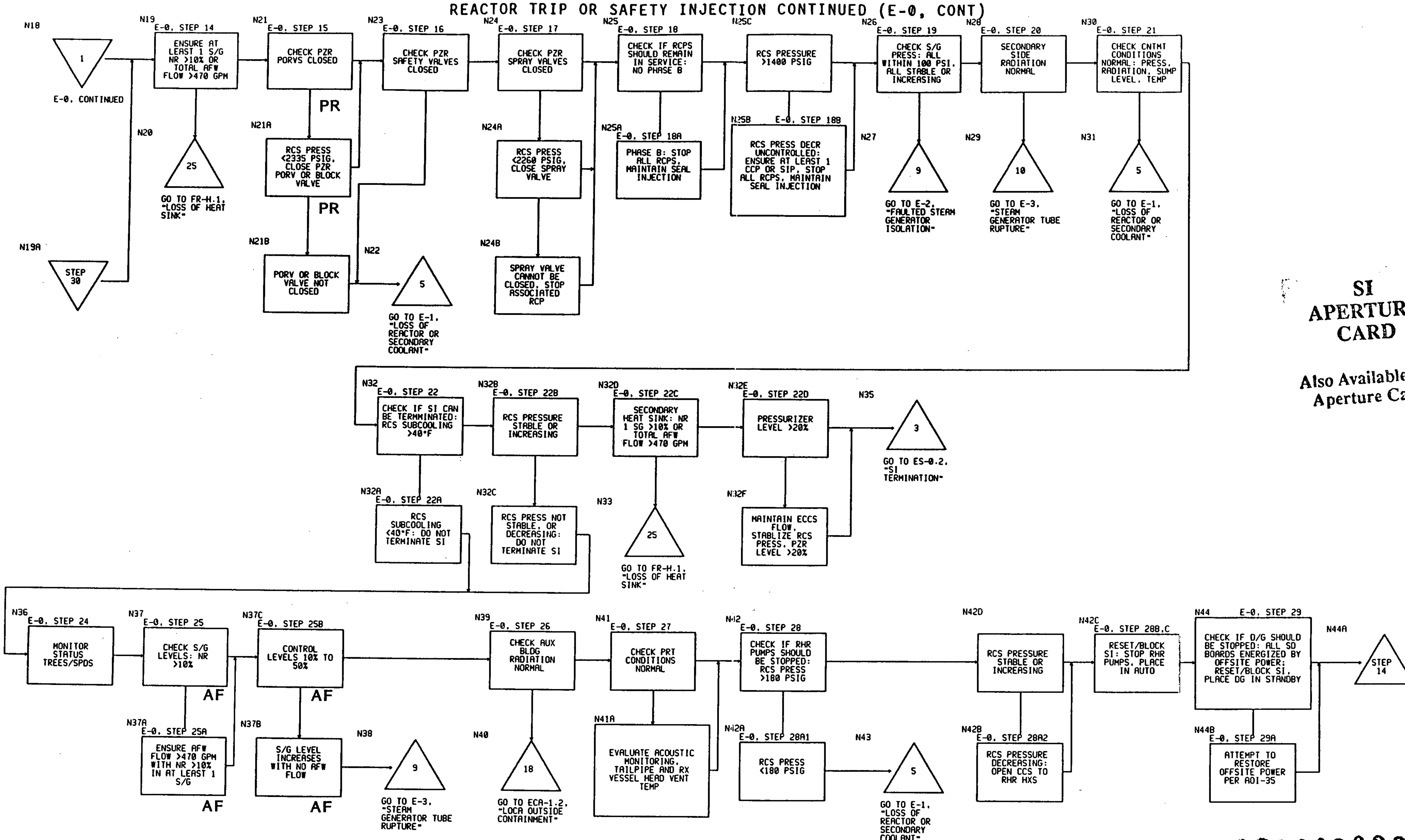
REACTOR TRIP OR SAFETY INJECTION CONDITION (E-0)



9209030229-01

Figure 3.1.2-2 (Page 1 of 44). Watts Bar ESD



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9209030229-02

Figure 3.1.2-2 (Page 2 of 44). Watts Bar ESD

REACTOR TRIP RESPONSE (ES-0.1)

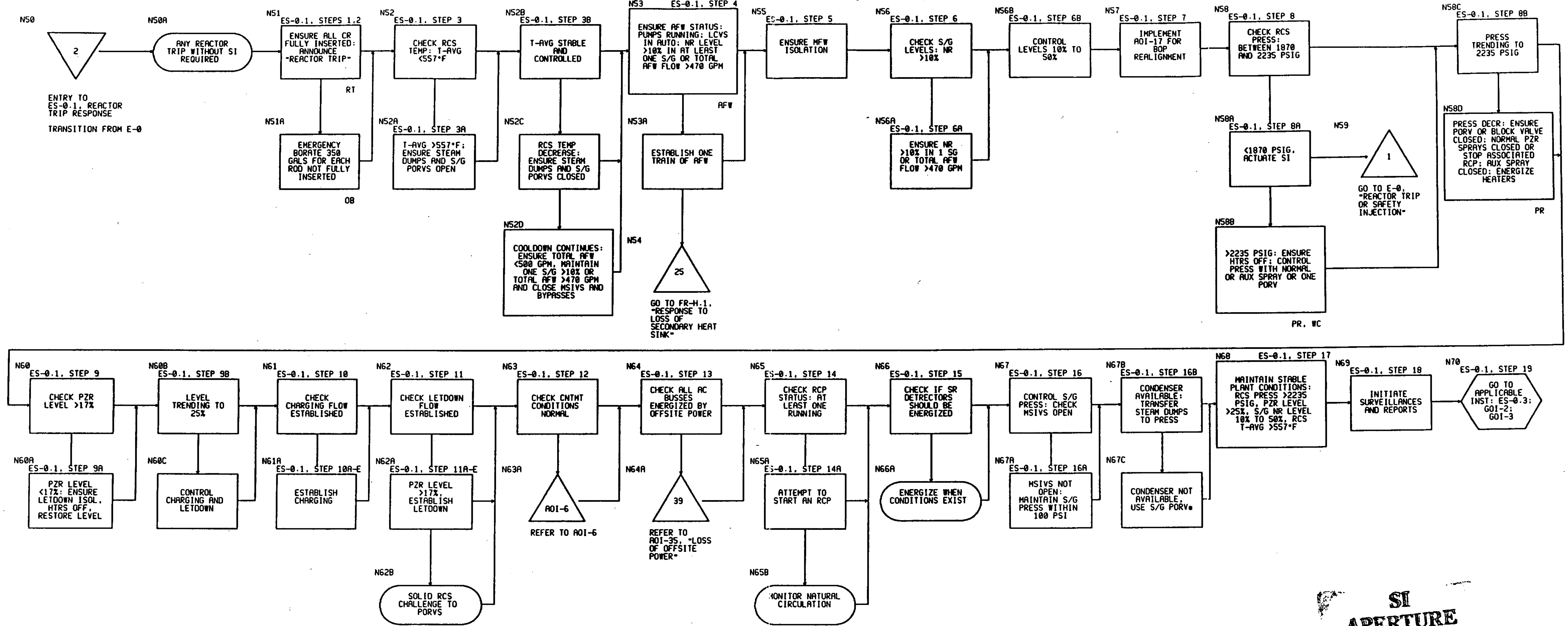


Figure 3.1.2-2 (Page 3 of 44). Watts Bar ESD

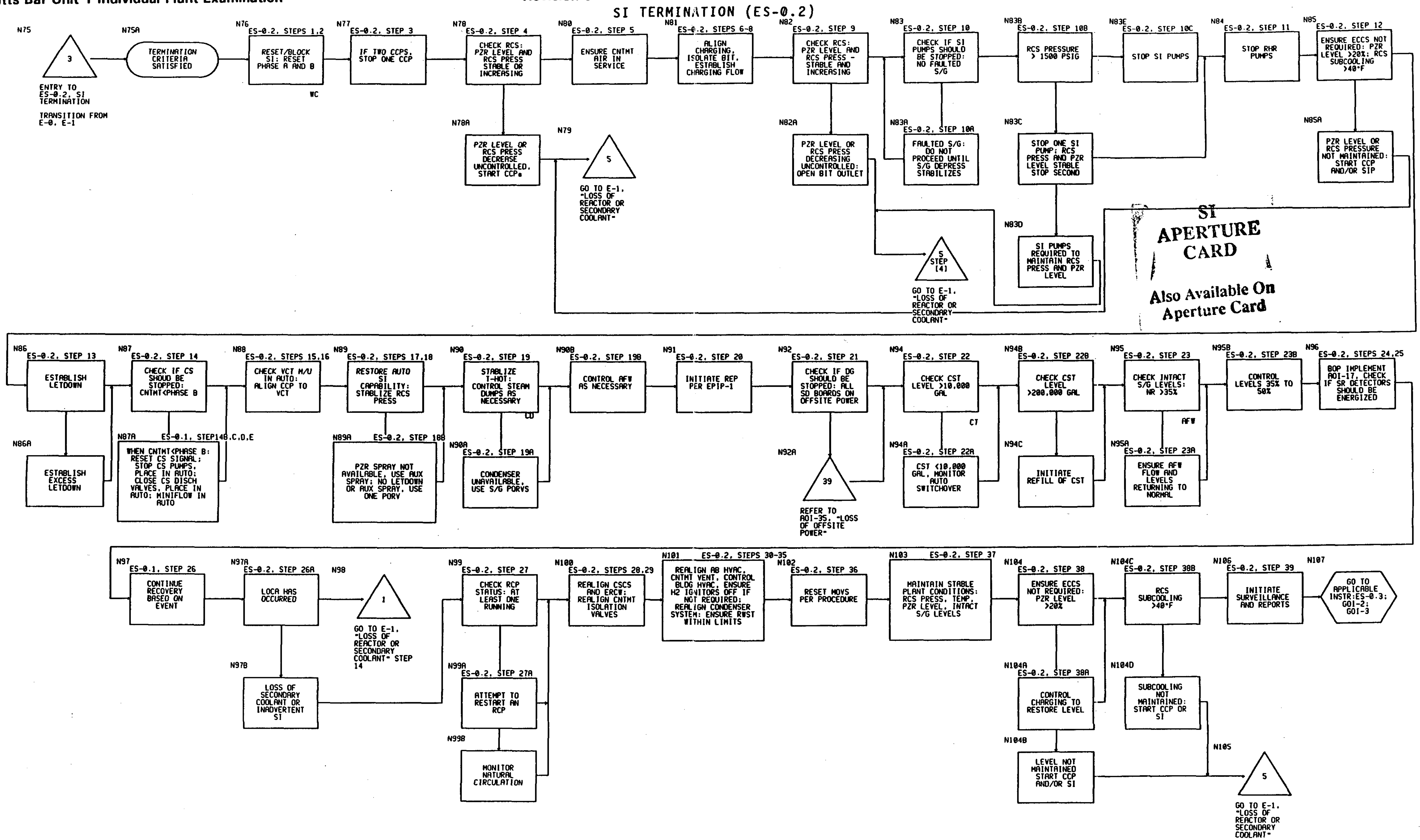
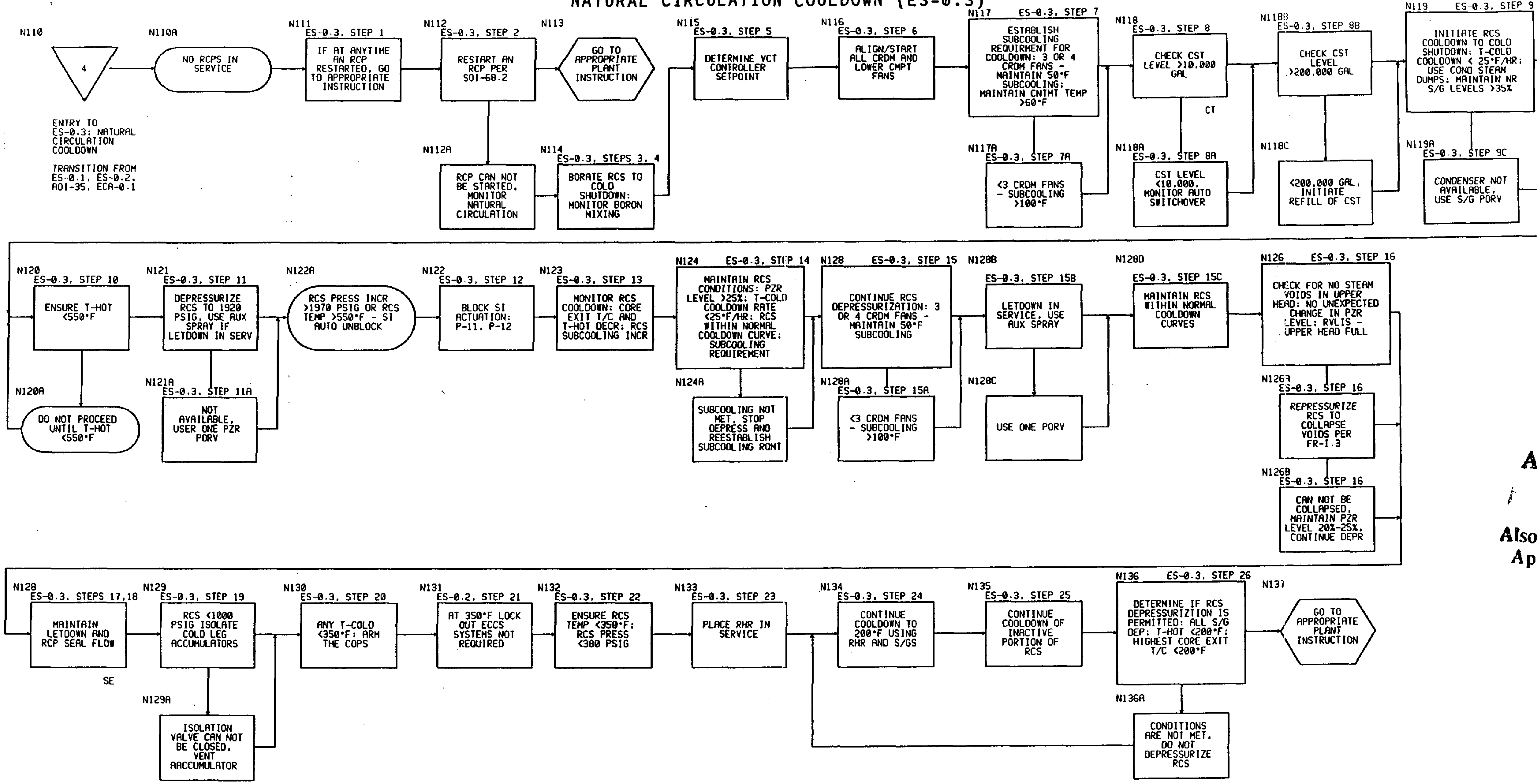


Figure 3.1.2-2 (Page 4 of 44). Watts Bar ESD

9209030229-04

NATURAL CIRCULATION COOLDOWN (ES-0.3)



SI APERTURE CARD

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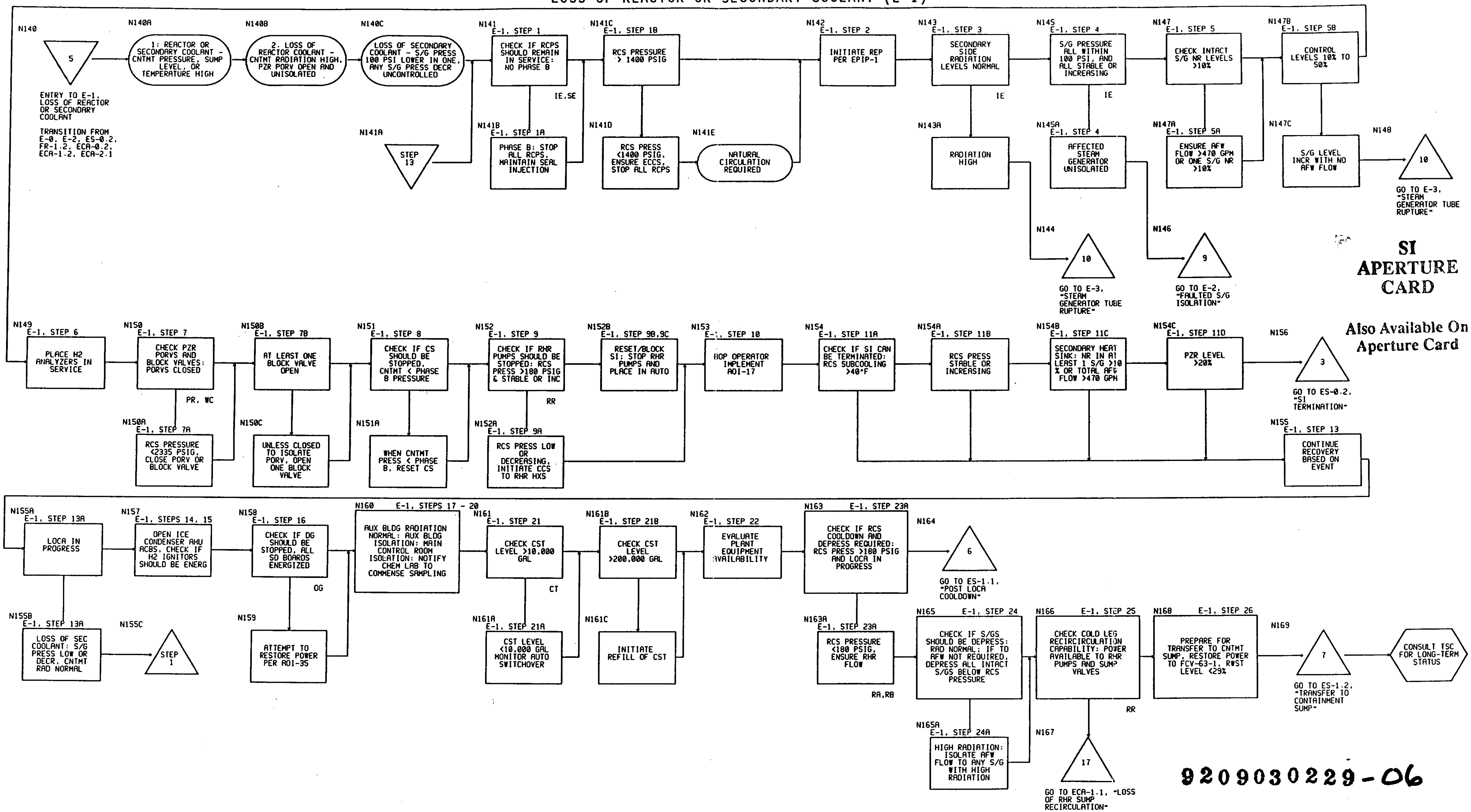
Figure 3.1.2-2 (Page 5 of 44). Watts Bar ESD

9209030229-05

LOSS OF REACTOR OR SECONDARY COOLANT (E-1)



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9209030229-06

Figure 3.1.2-2 (Page 6 of 44).
Watts Bar ESD

POST LOCA COOLDOWN (ES-1.1)

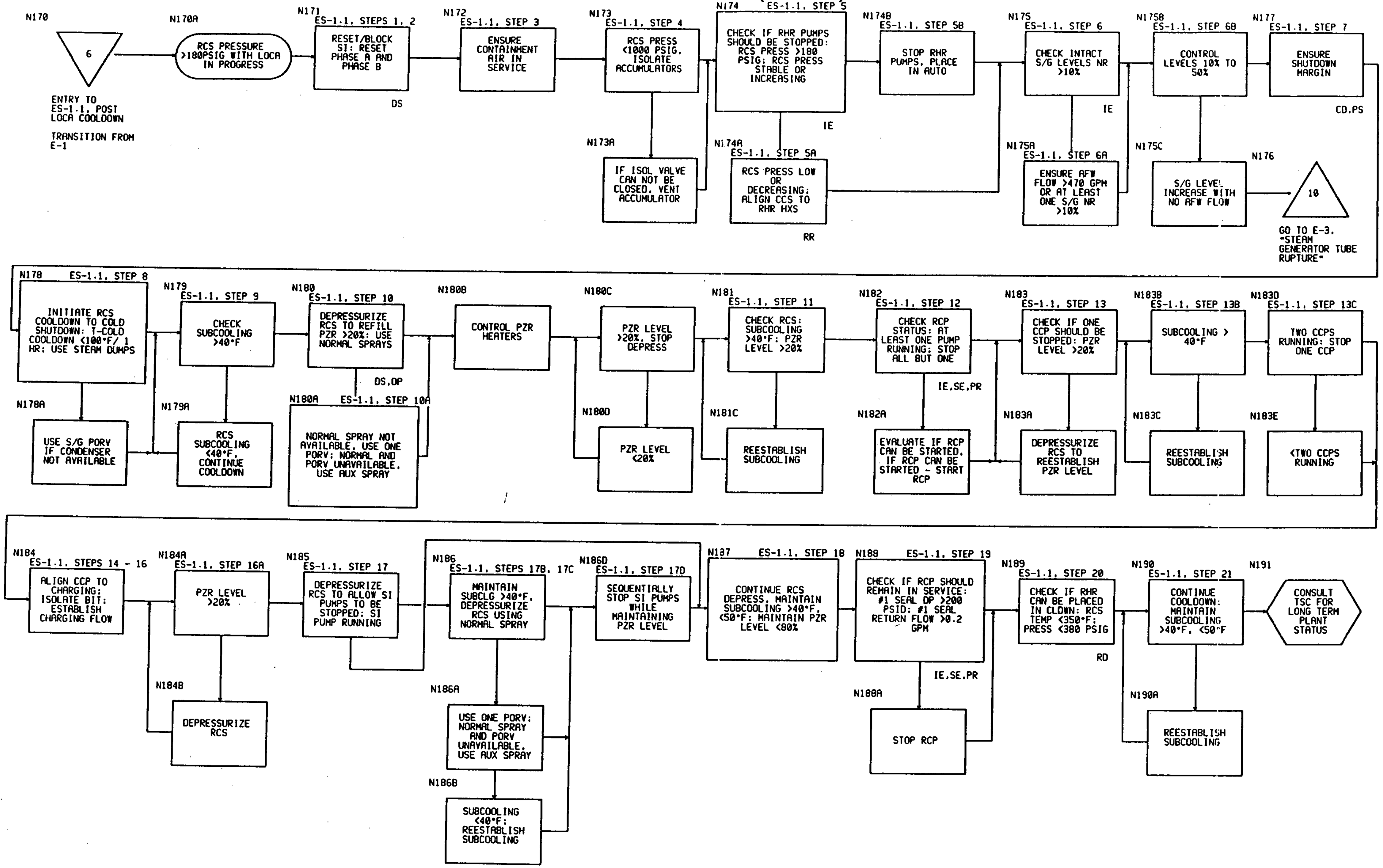

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Figure 3.1.2-2 (Page 7 of 44).
Watts Bar ESD

9209030229 - 07

TRANSFER TO CONTAINMENT SUMP (ES-1.2)

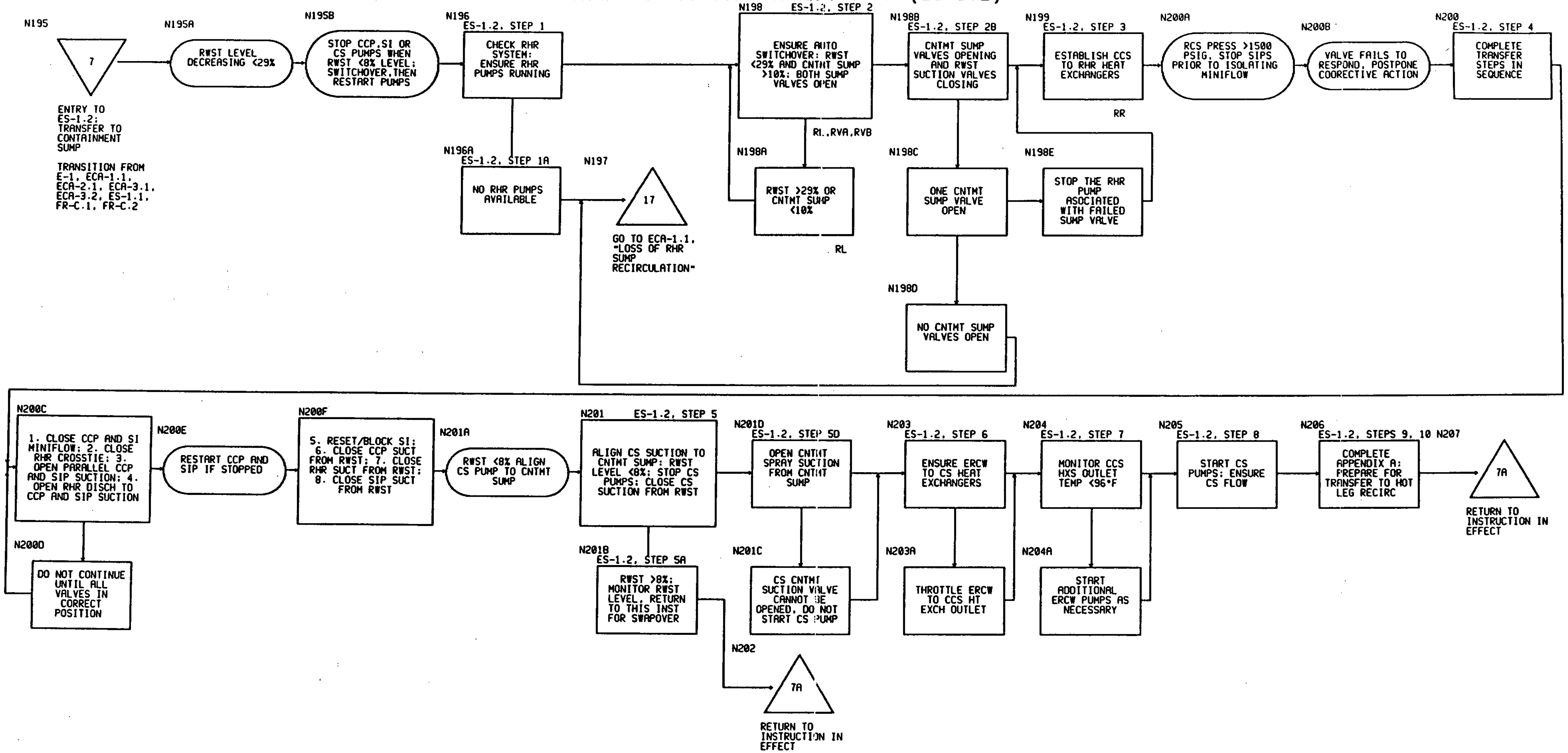
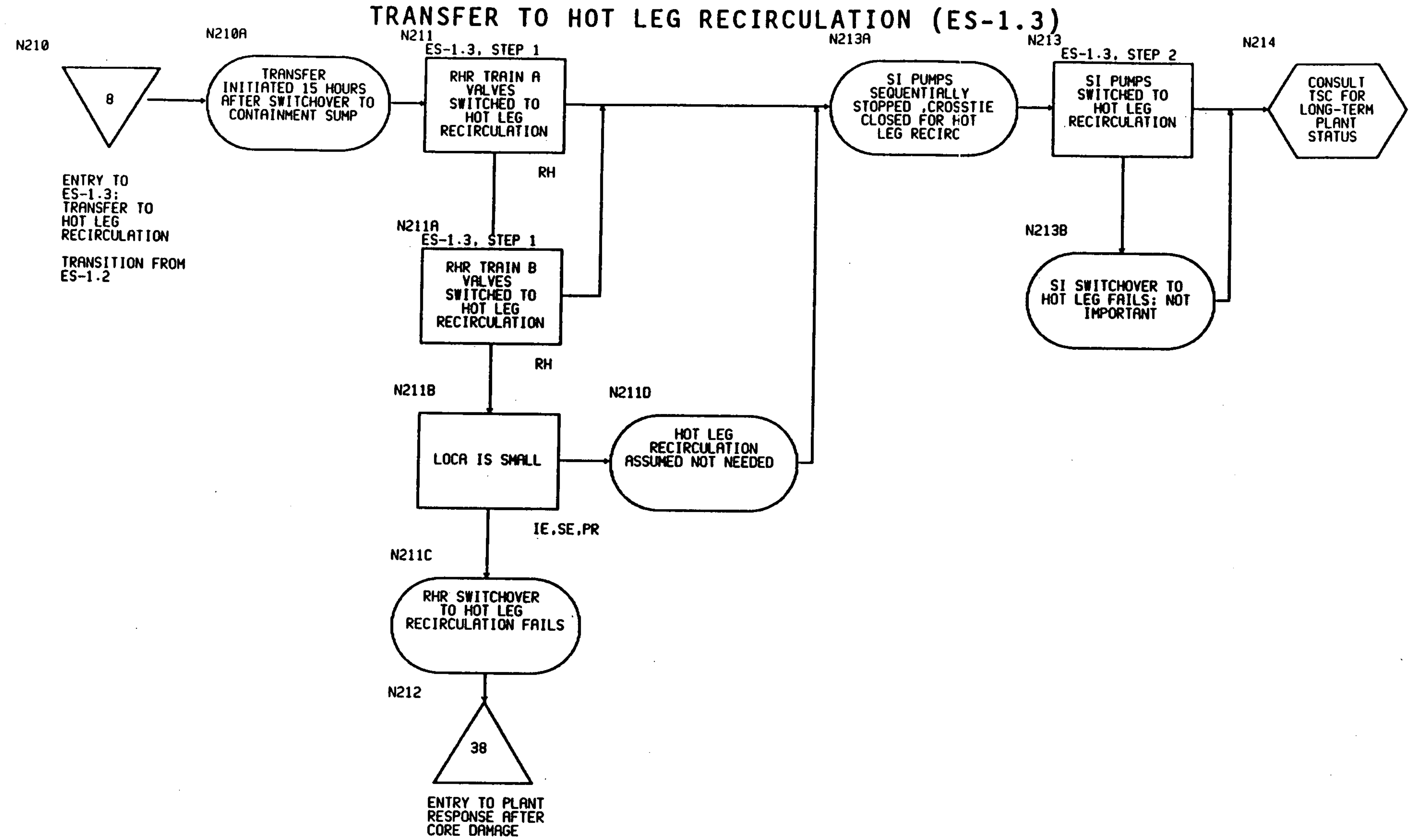


Figure 3.1.2-2 (Page 8 of 44). Watts Bar ESD



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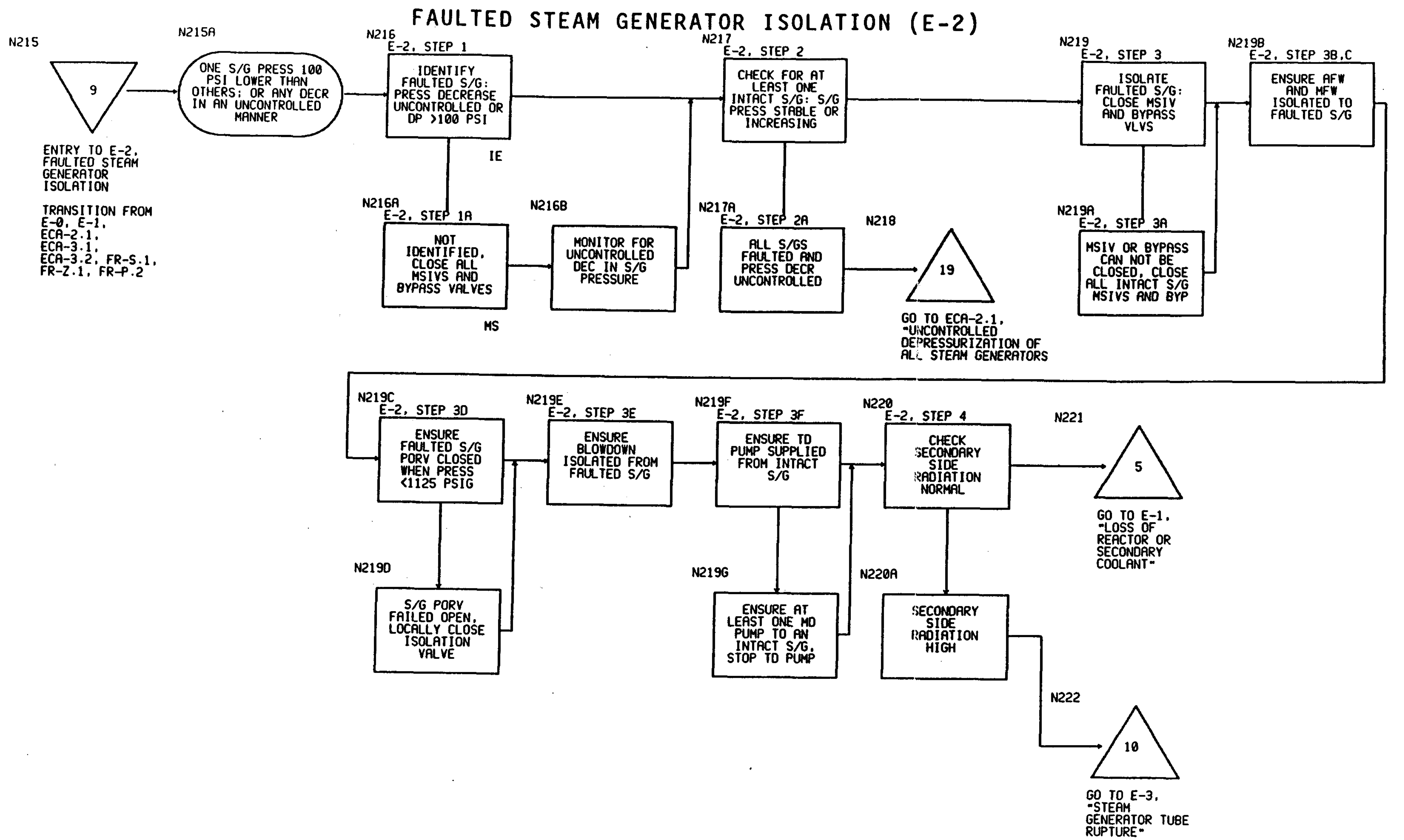
9209030229-07

Figure 3.1.2-2 (Page 9 of 44). Watts Bar ESD



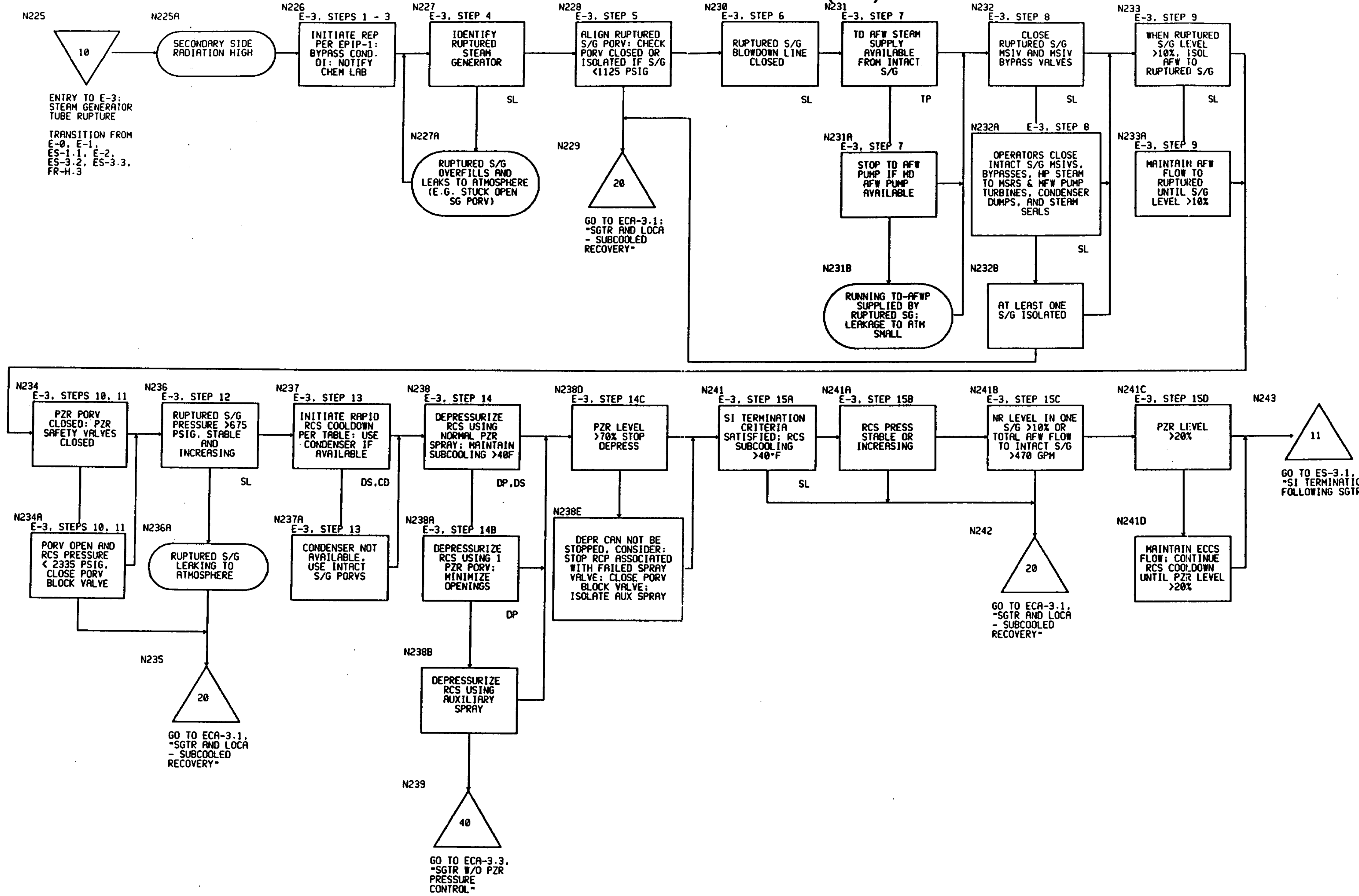
SI APERTURE CARD

Also Available On Aperture Card

9209030229-10

Figure 3.1.2-2 (Page 10 of 44). Watts Bar ESD

STEAM GENERATOR TUBE RUPTURE (E-3)



SI APERTURE CARD

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9209030229-11

Figure 3.1.2-2 (Page 11 of 44). Watts Bar ESD

SI TERMINATION FOLLOWING SGTR (ES-3.1)

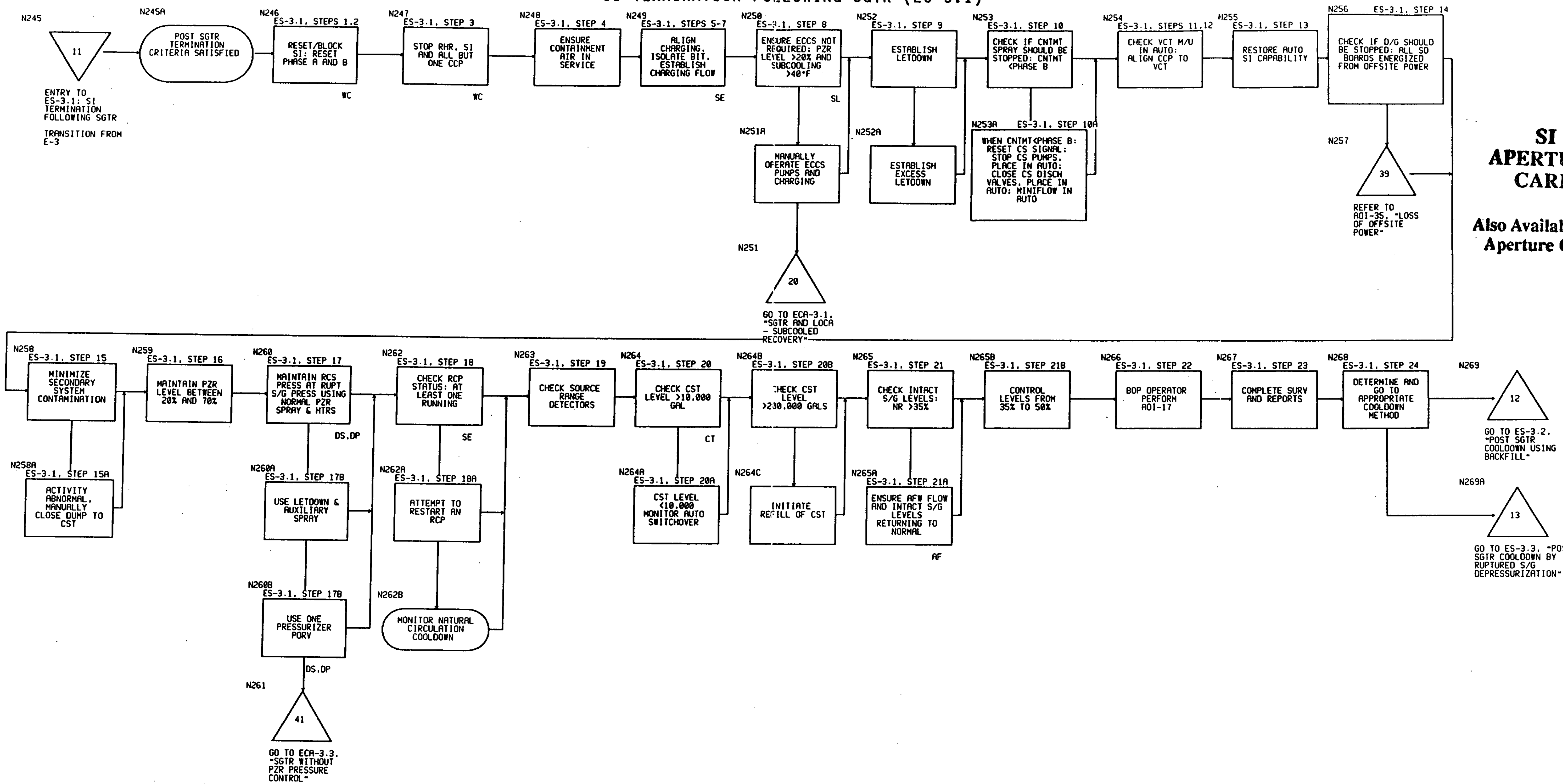
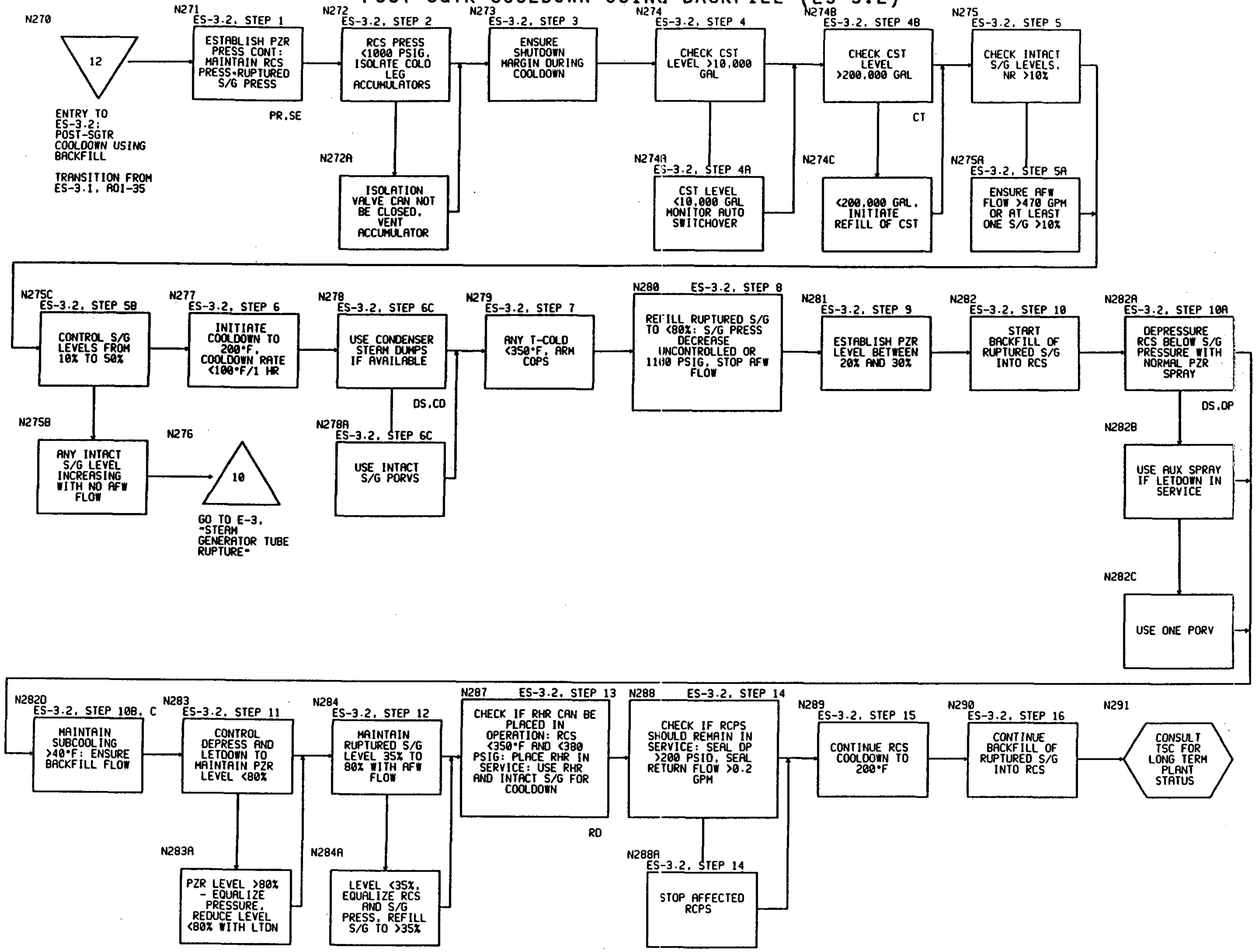


Figure 3.1.2-2 (Page 12 of 44).
Watts Bar ESD

POST-SGTR COOLDOWN USING BACKFILL (ES-3.2)



SI APERTURE CARD

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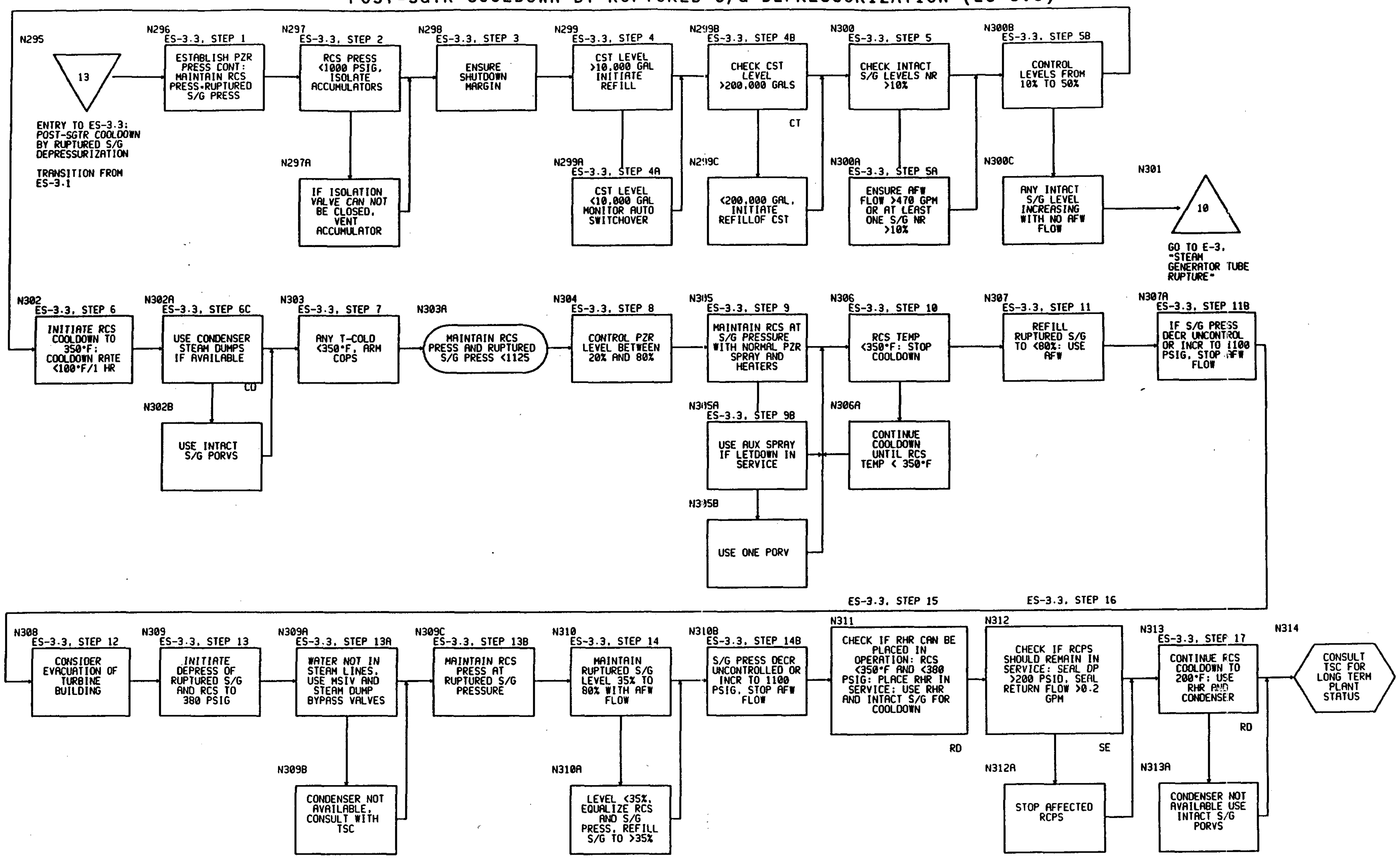
9209030229 -13

Figure 3.1.2-2 (Page 13 of 44). Watts Bar ESD

POST-SGTR COOLDOWN BY RUPTURED S/G DEPRESSURIZATION (ES-3.3)



SI APERTURE CARD

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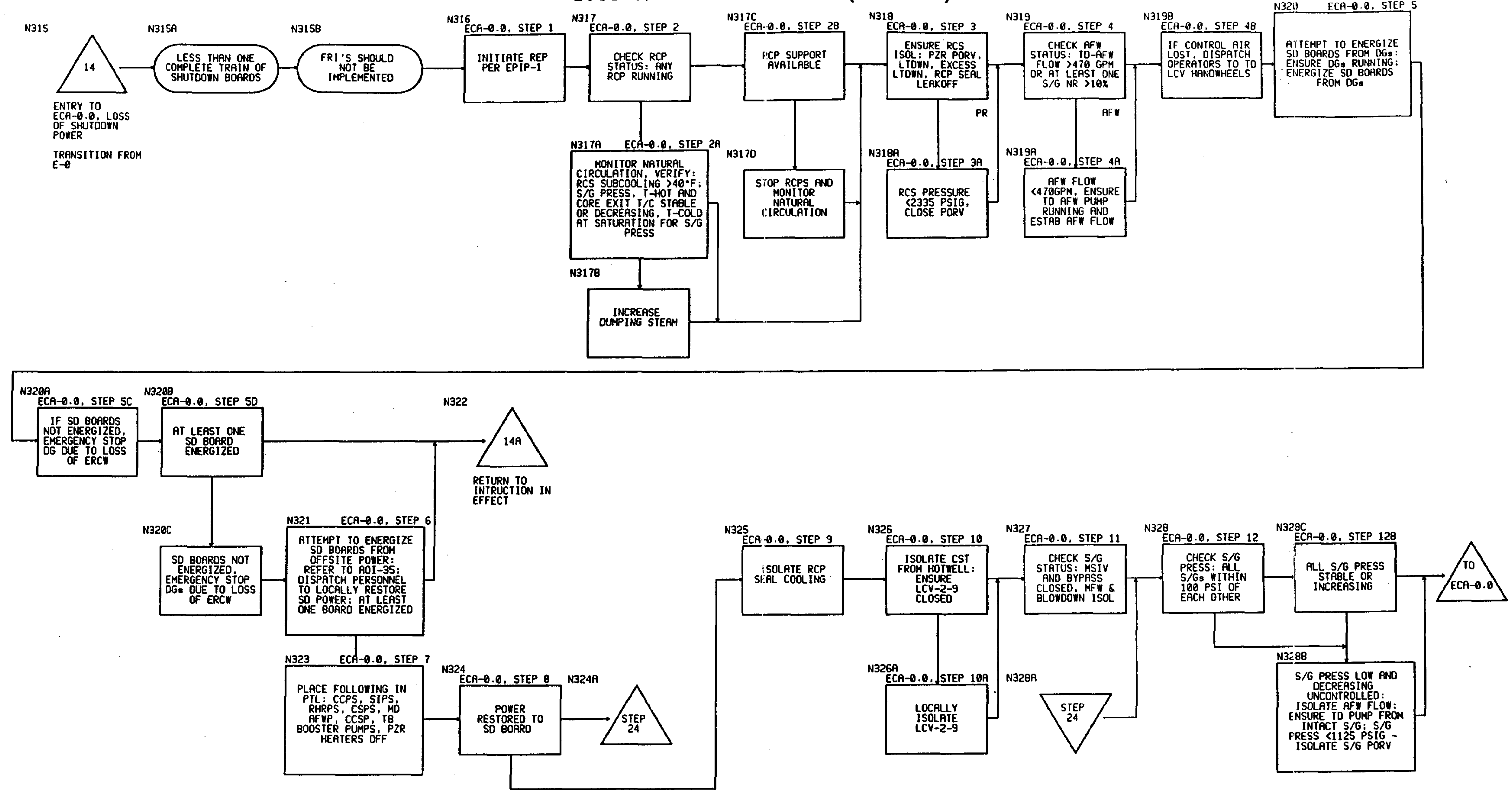
9209030229 -14

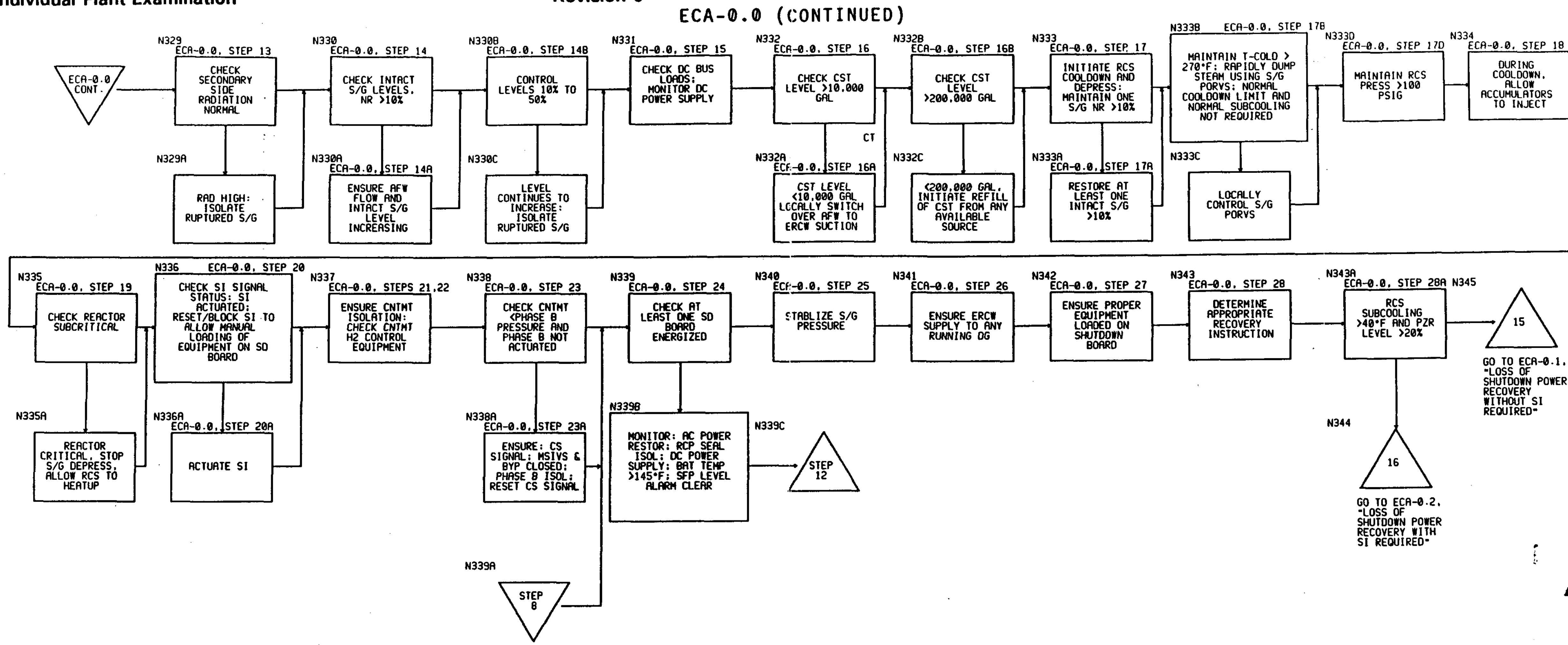
Figure 3.1.2-2 (Page 14 of 44). Watts Bar ESD

LOSS OF SHUTDOWN POWER (ECA-0.0)



9209030229 -15

Figure 3.1.2-2 (Page 15 of 44).
Watts Bar ESD



SI APERTURE CARD

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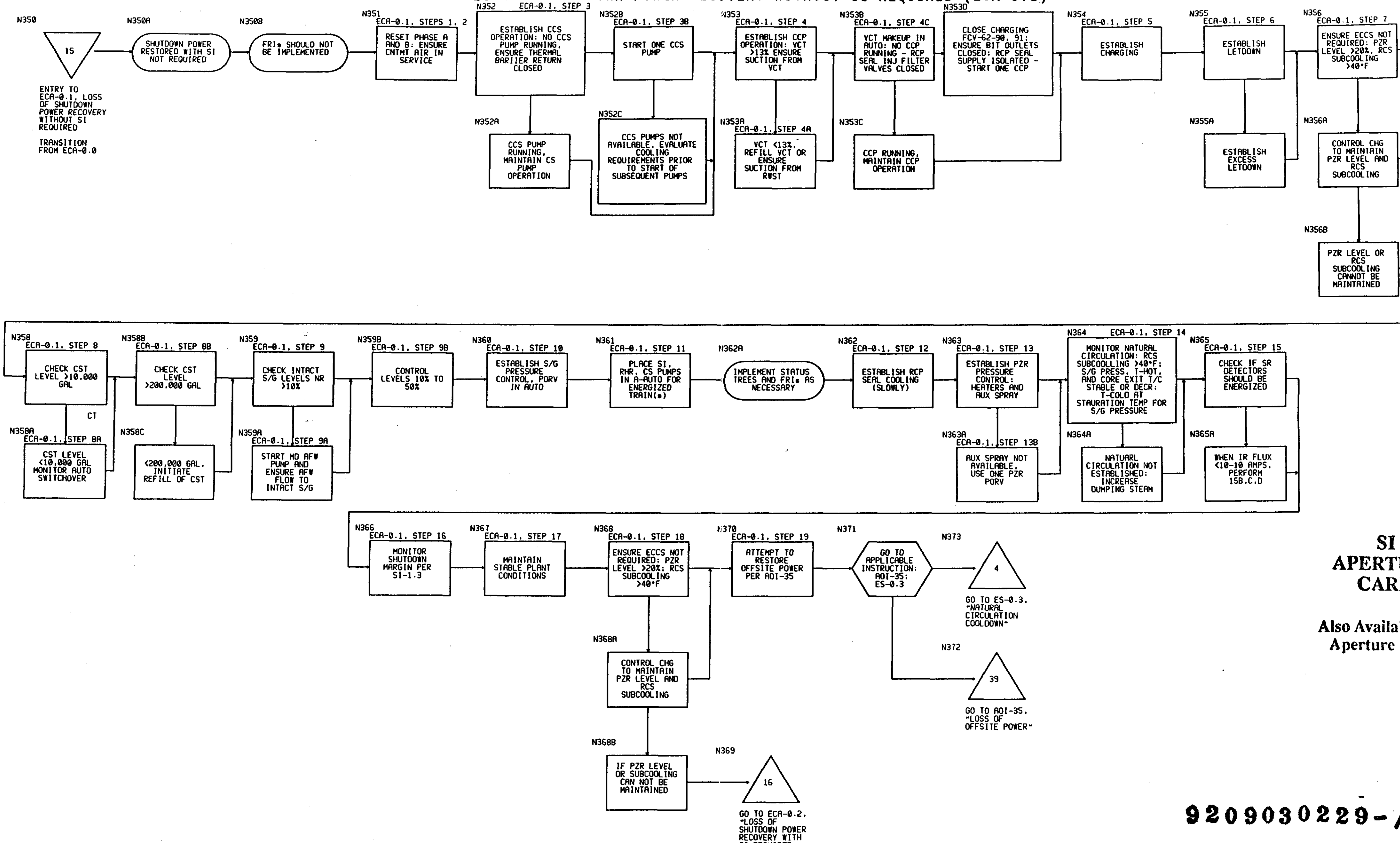
9209030229-16

Figure 3.1.2-2 (Page 16 of 44). Watts Bar ESD

LOSS OF SHUTDOWN POWER RECOVERY WITHOUT SI REQUIRED (ECA-0.1)

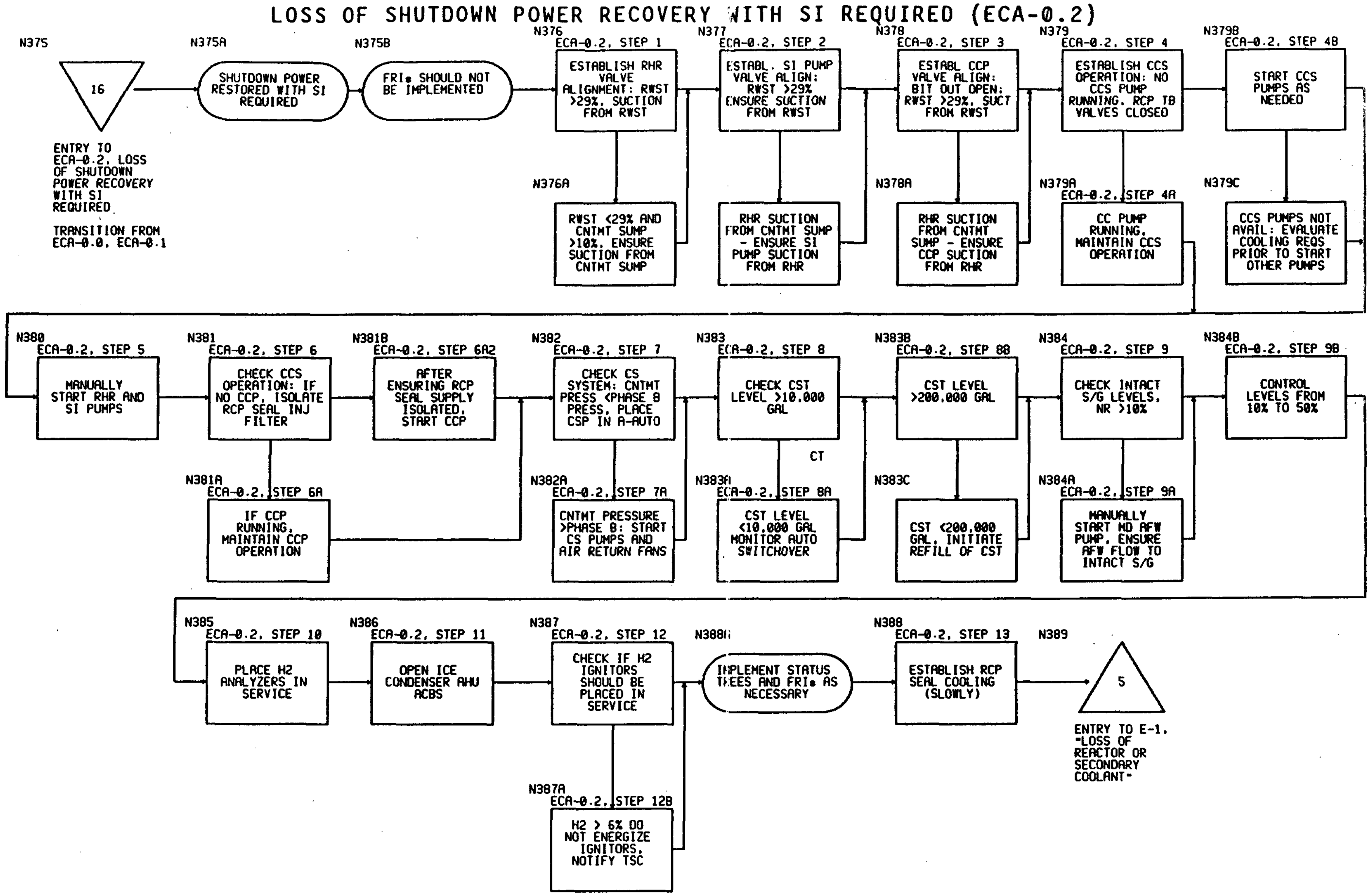


SI APERTURE CARD

Also Available On Aperture Card

9209030229-17

Figure 3.1.2-2 (Page 17 of 44).
Watts Bar ESD



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9209030229 -18

Figure 3.1.2-2 (Page 18 of 44).
Watts Bar ESD

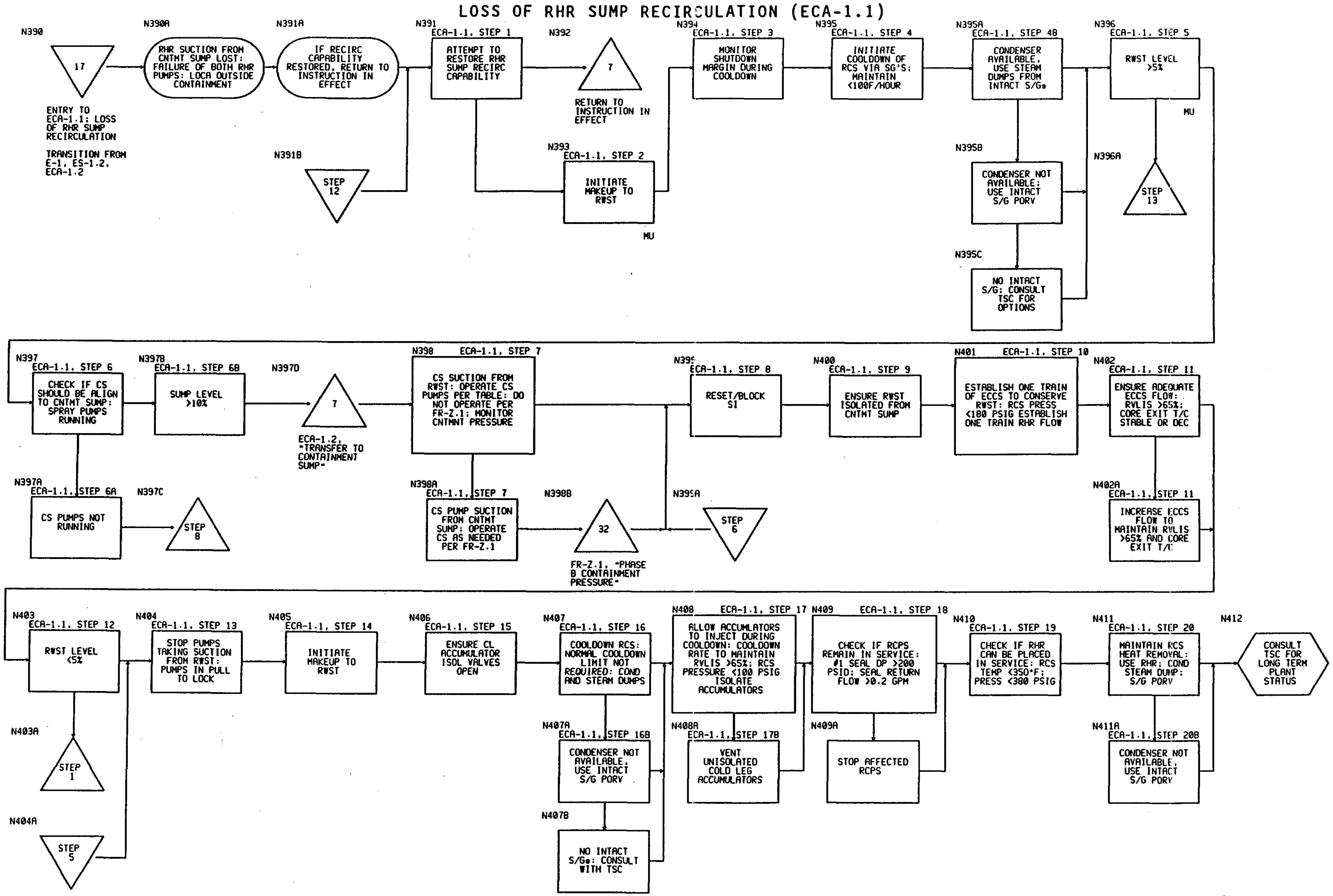
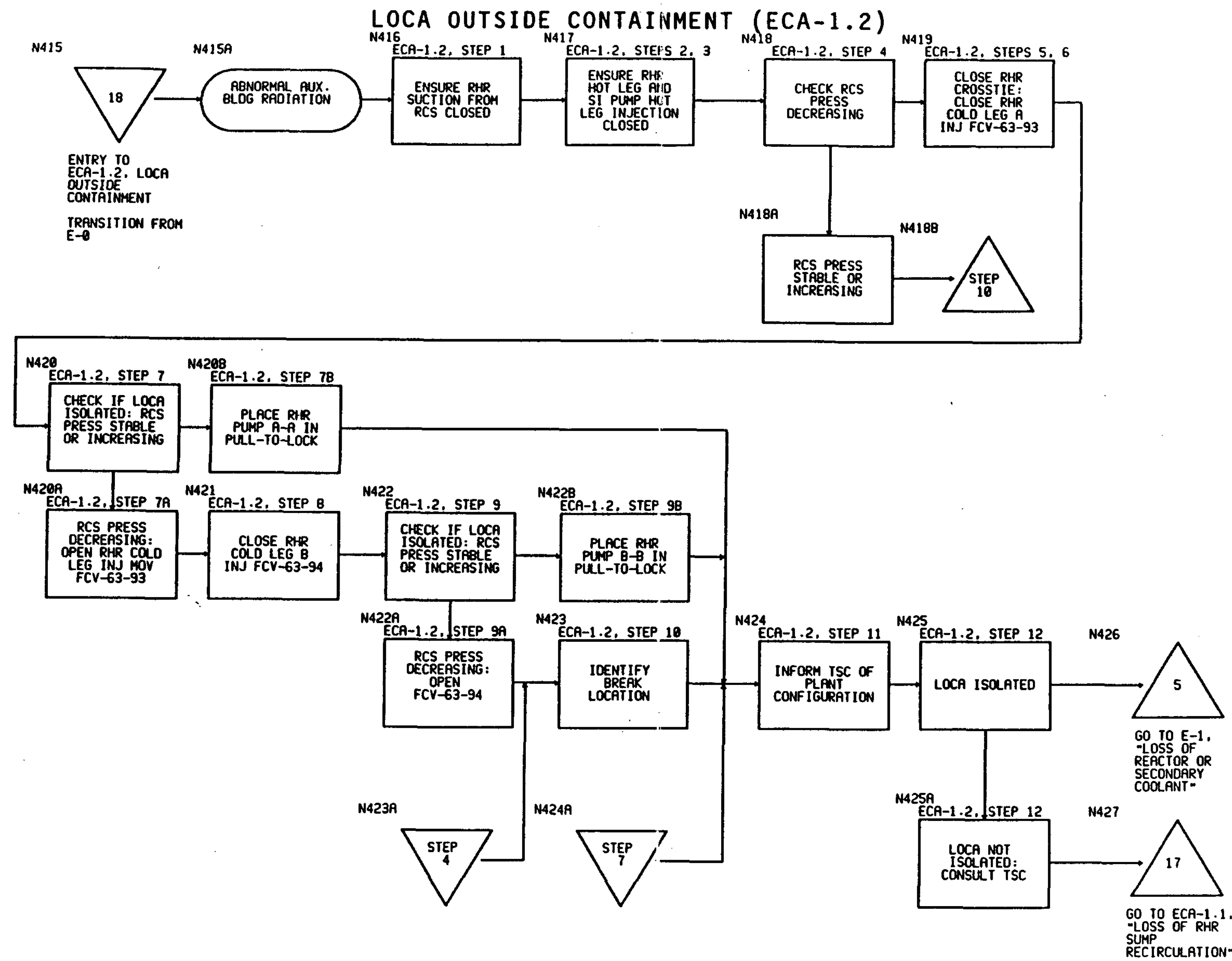


Figure 3.1.2-2 (Page 19 of 44).
Watts Bar ESD



SI APERTURE CARD

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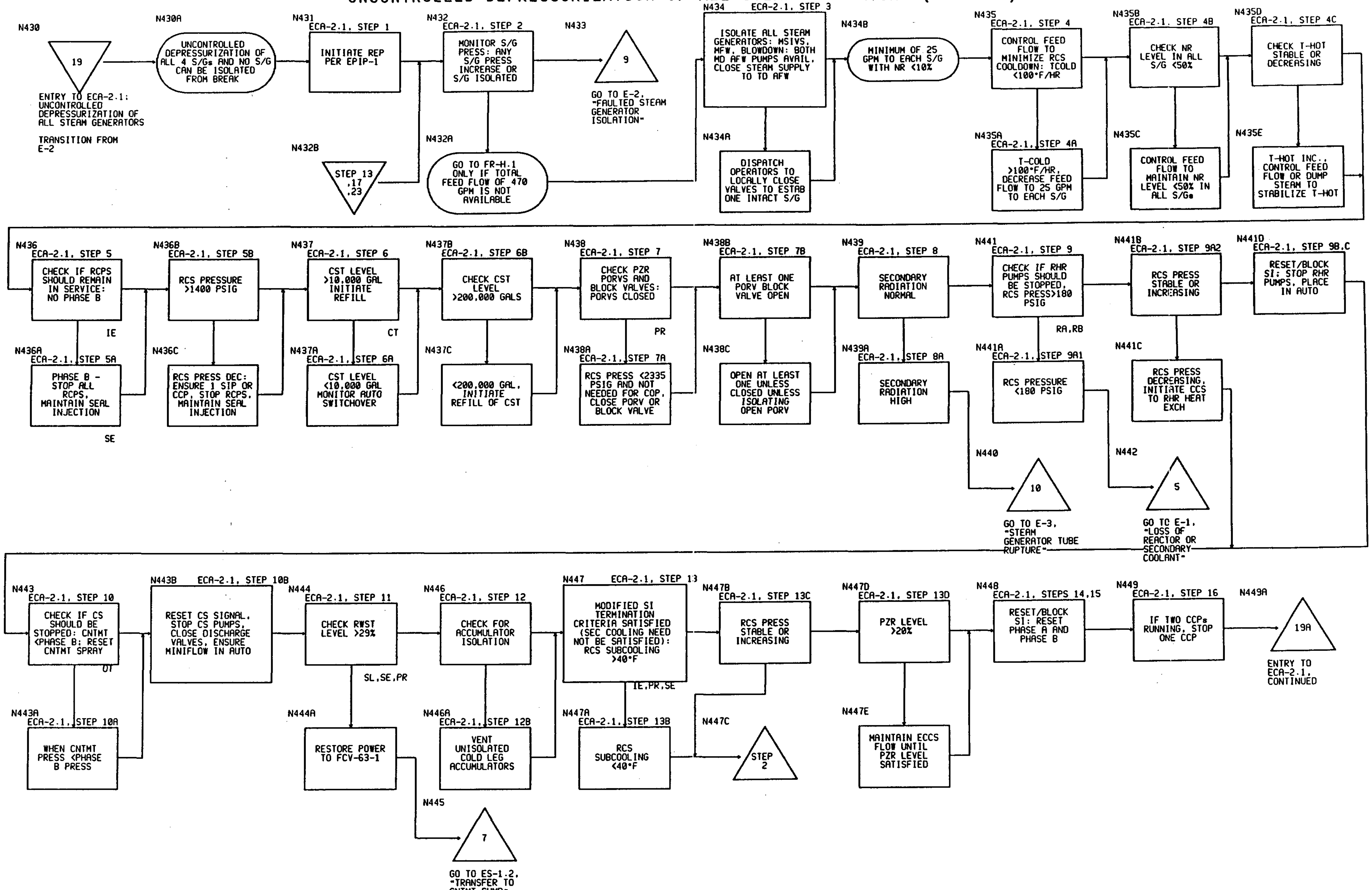
9209030229-20

Figure 3.1.2-2 (Page 20 of 44). Watts Bar ESD

UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS (ECA-2.1)



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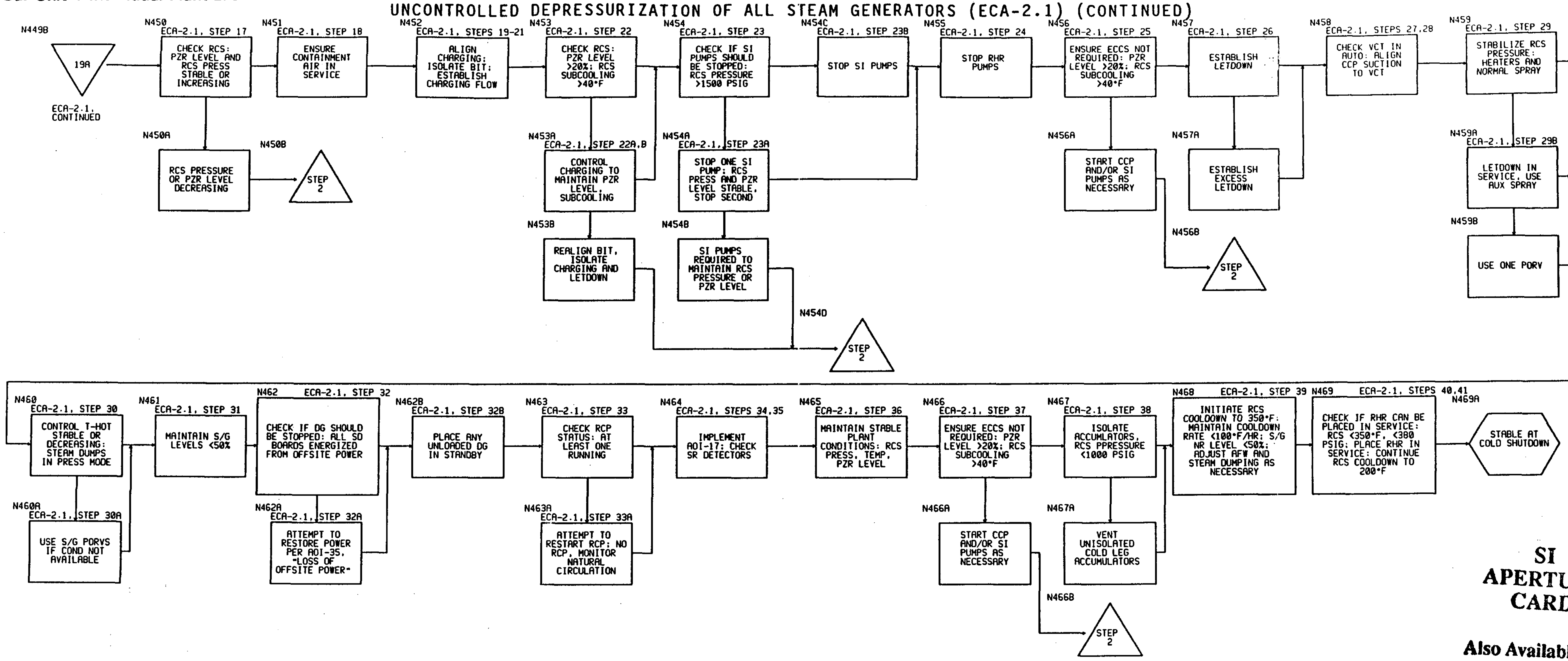
9209030229 -21

Figure 3.1.2-2 (Page 21 of 44).
Watts Bar ESD

UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS (ECA-2.1) (CONTINUED)



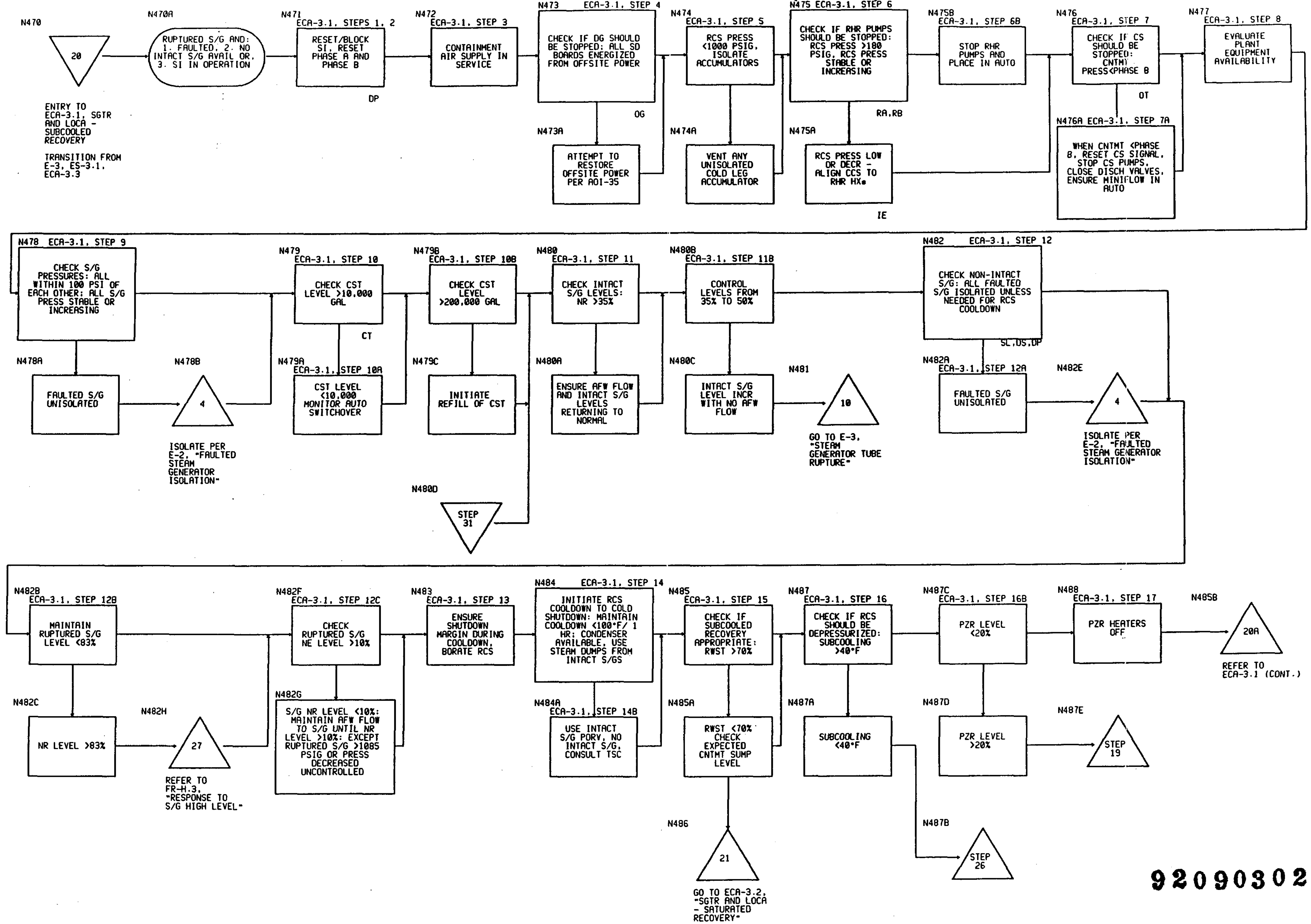
SI APERTURE CARD

Also Available On Aperture Card

9209030229 - 22

Figure 3.1.2-2 (Page 22 of 44). Watts Bar ESD

SGTR AND LOCA - SUBCOOLED RECOVERY (ECA-3.1)



SI APERTURE CARD

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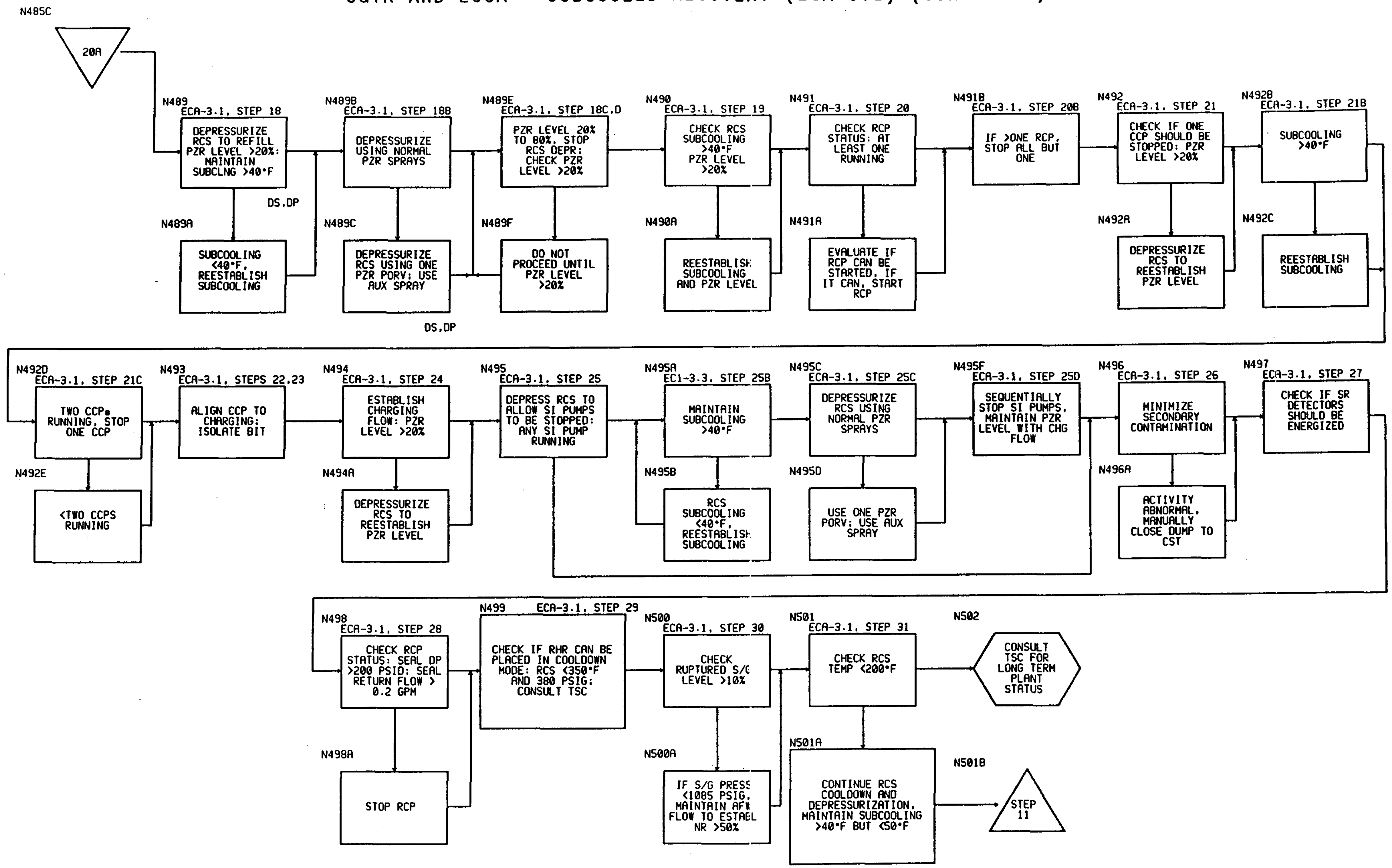
9209030229 - 23

Figure 3.1.2-2 (Page 23 of 44). Watts Bar ESD

SGTR AND LOCA - SUBCOOLED RECOVERY (ECA-3.1) (CONTINUED)



SI APERTURE CARD

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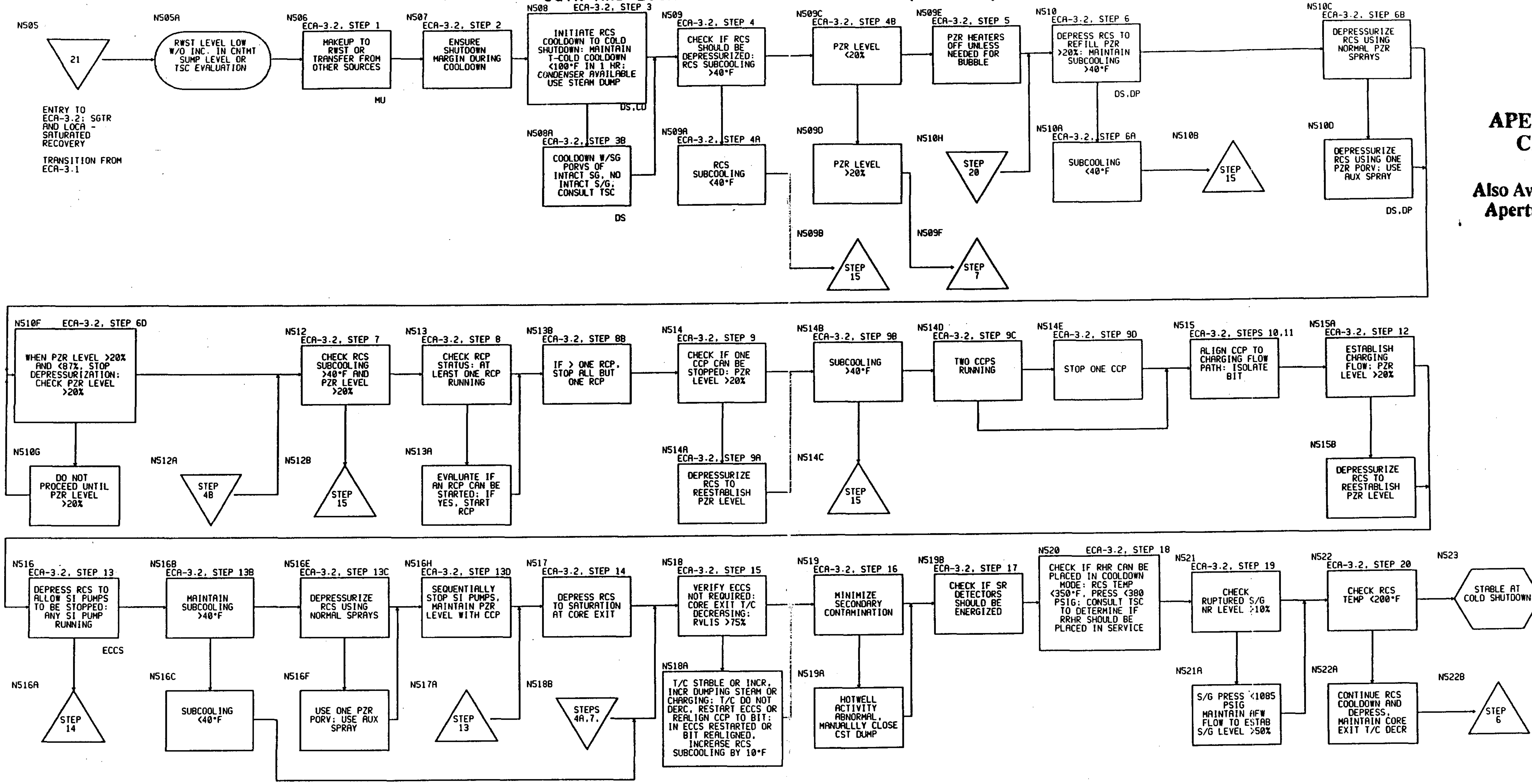
9209030229 -24

Figure 3.1.2-2 (Page 24 of 44).
Watts Bar ESD

SGTR AND LOCA - SATURATED RECOVERY (ECA-3.2)



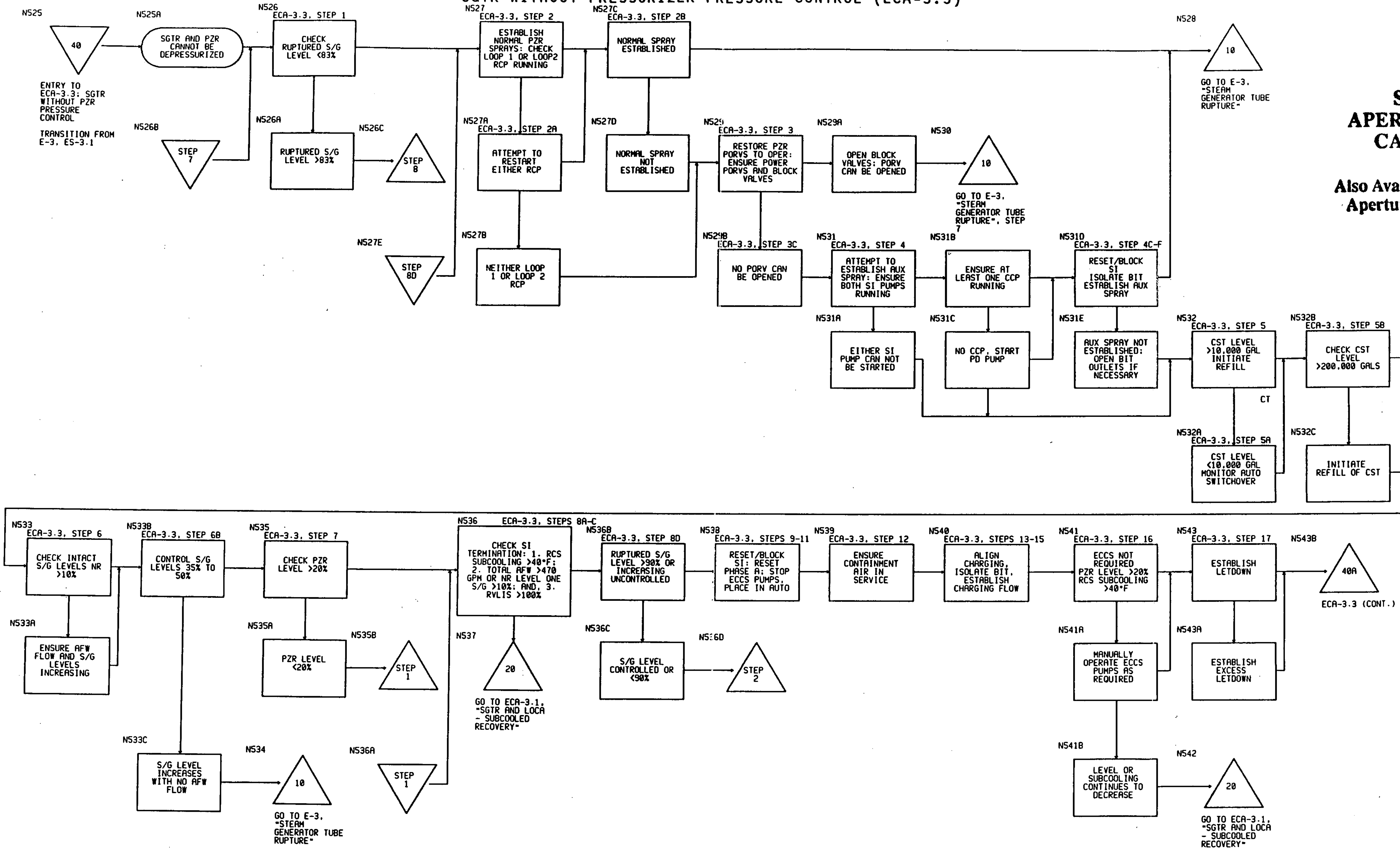
SI APERTURE CARD

Also Available On Aperture Card

920903O229-25

Figure 3.1.2-2 (Page 25 of 44). Watts Bar ESD

SGTR WITHOUT PRESSURIZER PRESSURE CONTROL (ECA-3.3)



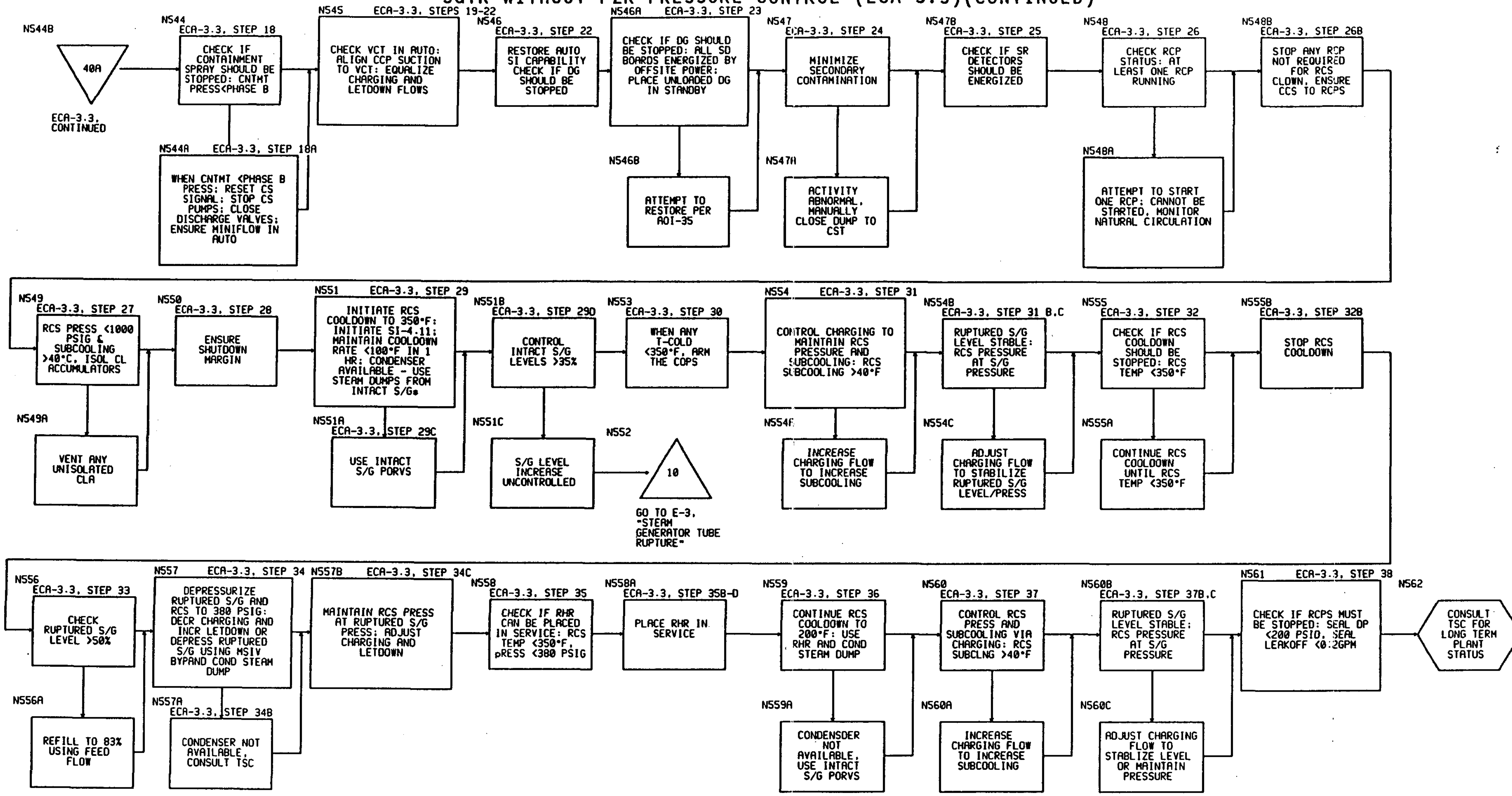
SI APERTURE CARD

Also Available On Aperture Card

9209030229 - 26

Figure 3.1.2-2 (Page 26 of 44). Watts Bar ESD

SGTR WITHOUT PZR PRESSURE CONTROL (ECA-3.3)(CONTINUED)



SI APERTURE CARD

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9209030229-27

Figure 3.1.2-2 (Page 27 of 44). Watts Bar ESD

INADEQUATE CORE COOLING (FR-C.1)

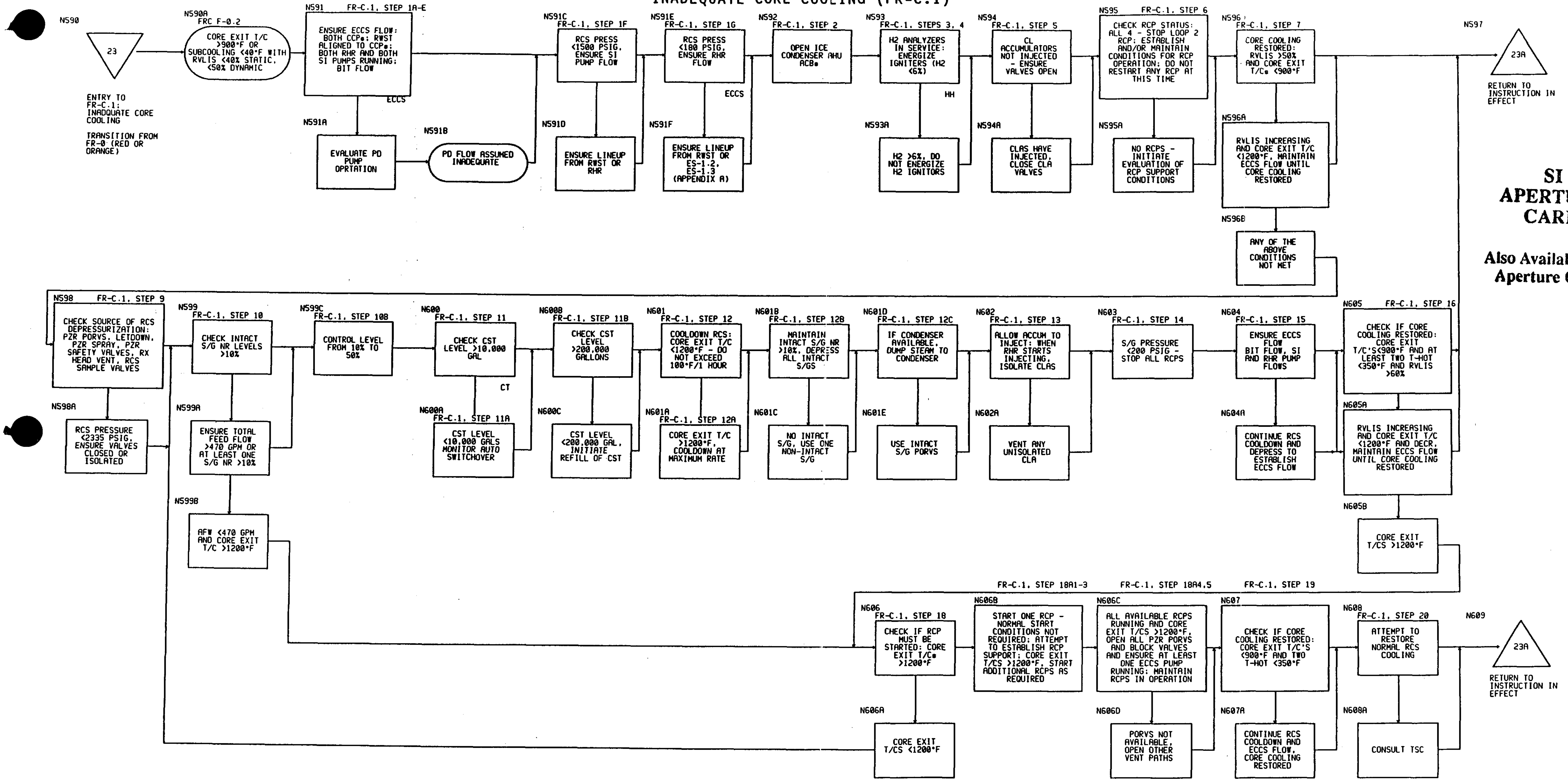
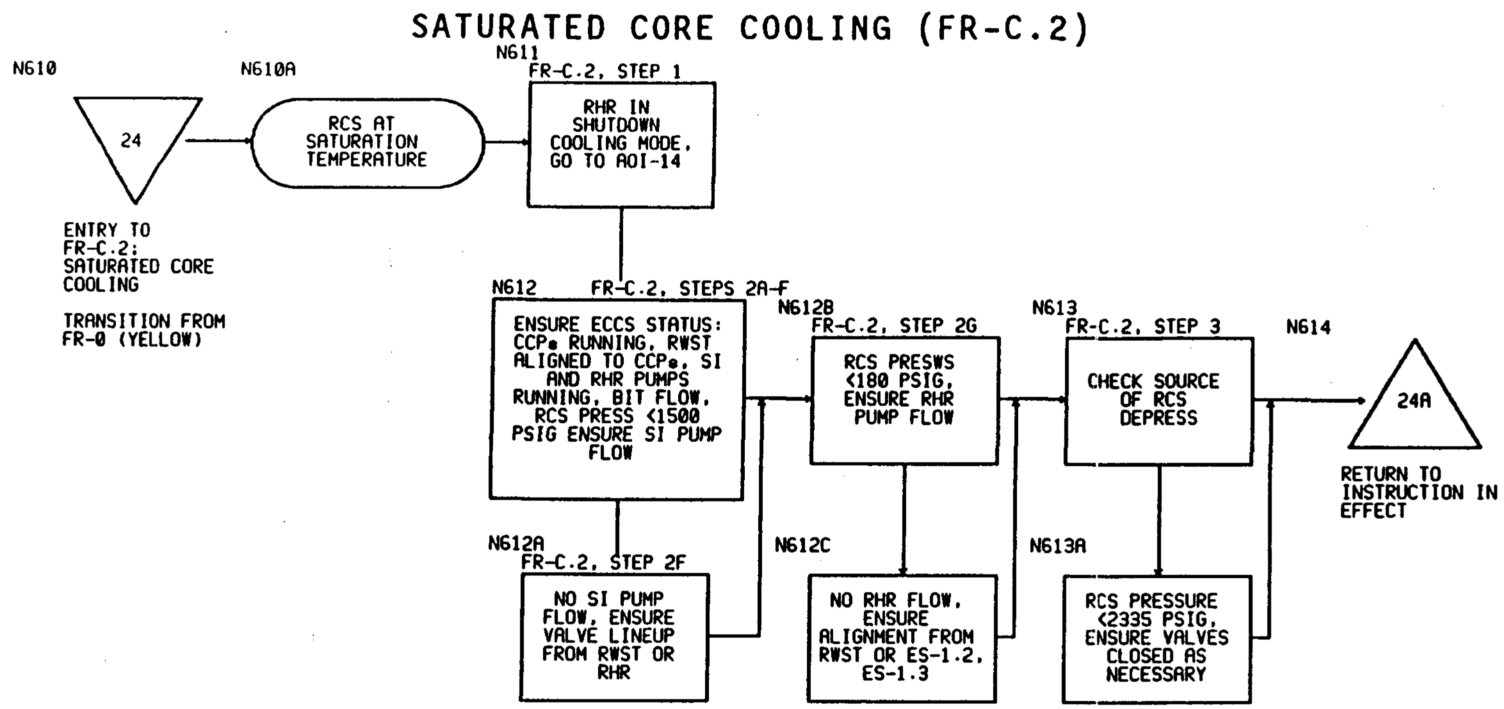


Figure 3.1.2-2 (Page 28 of 44).
Watts Bar ESD



SI APERTURE CARD

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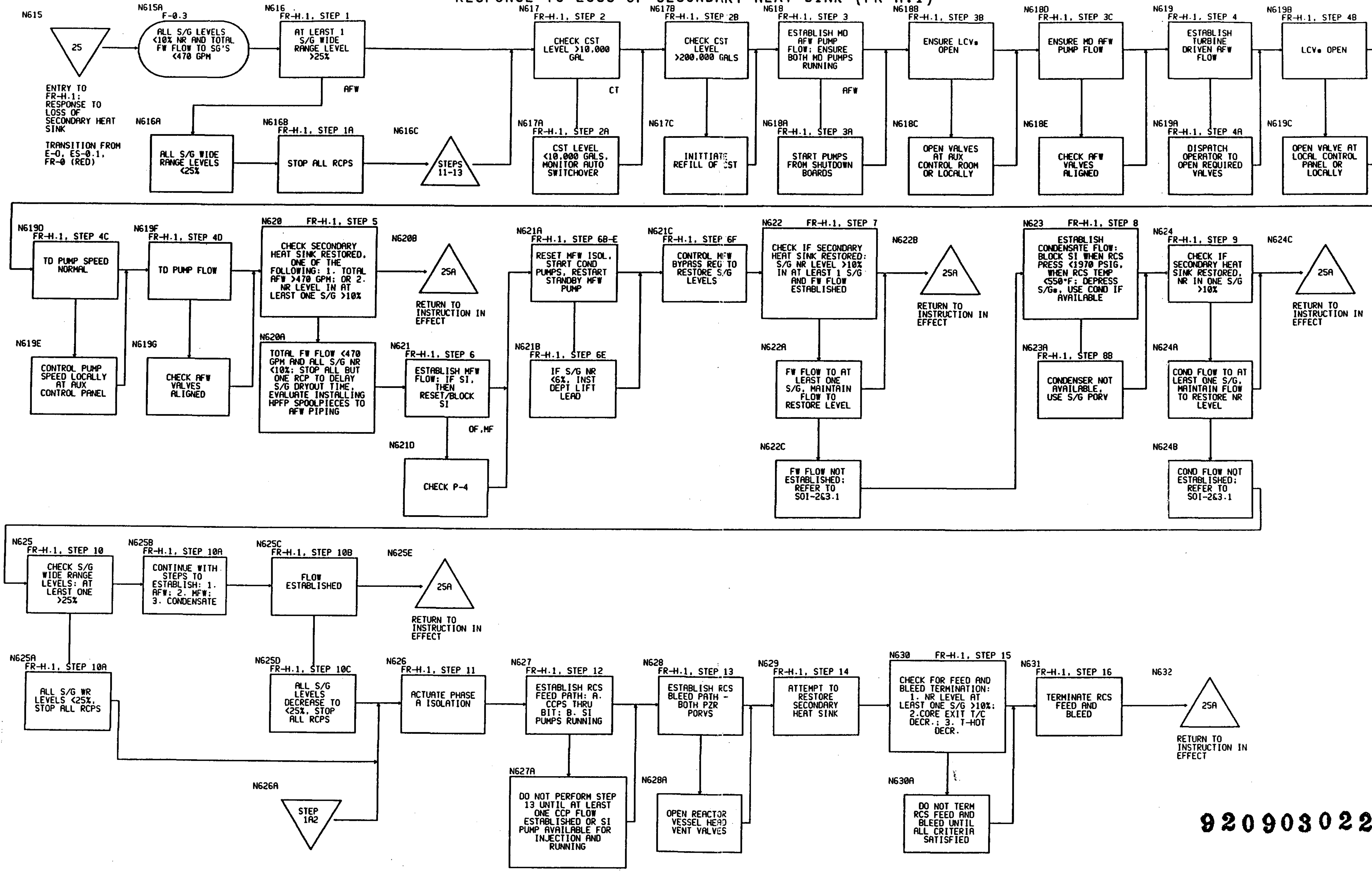
9209030229-29

Figure 3.1.2-2 (Page 29 of 44). Watts Bar ESD

RESPONSE TO LOSS OF SECONDARY HEAT SINK (FR-H.1)

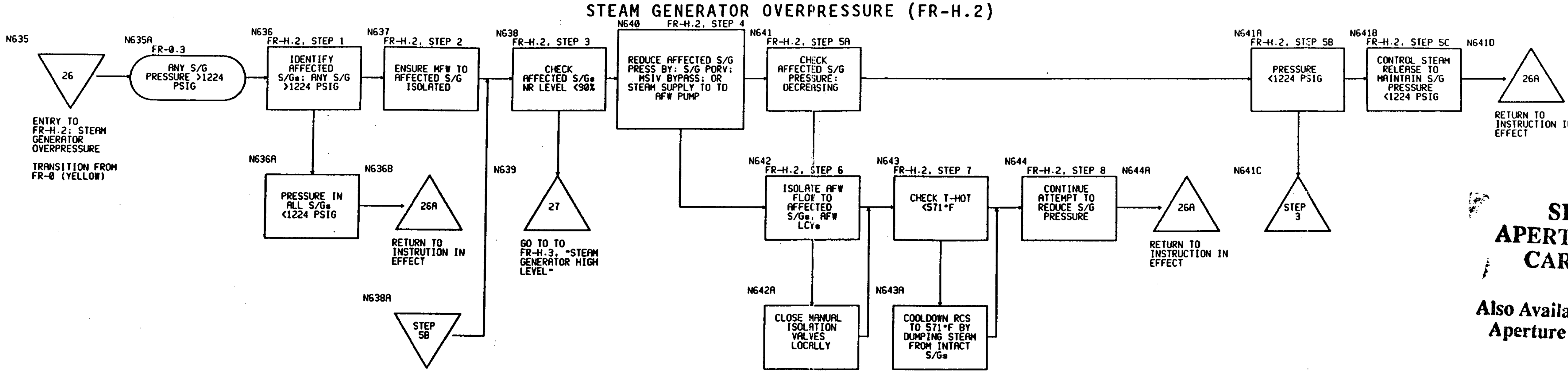


SI APERTURE CARD

Also Available On Aperture Card

9209030229-30

Figure 3.1.2-2 (Page 30 of 44). Watts Bar ESD



SI APERTURE CARD

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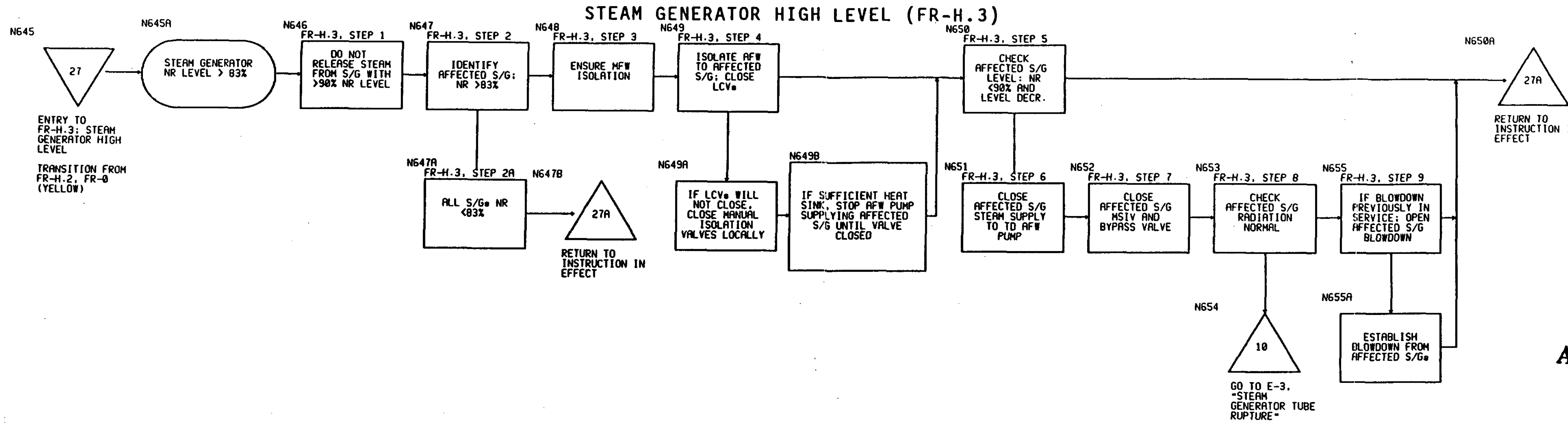
9209030229-31

Figure 3.1.2-2 (Page 31 of 44). Watts Bar ESD



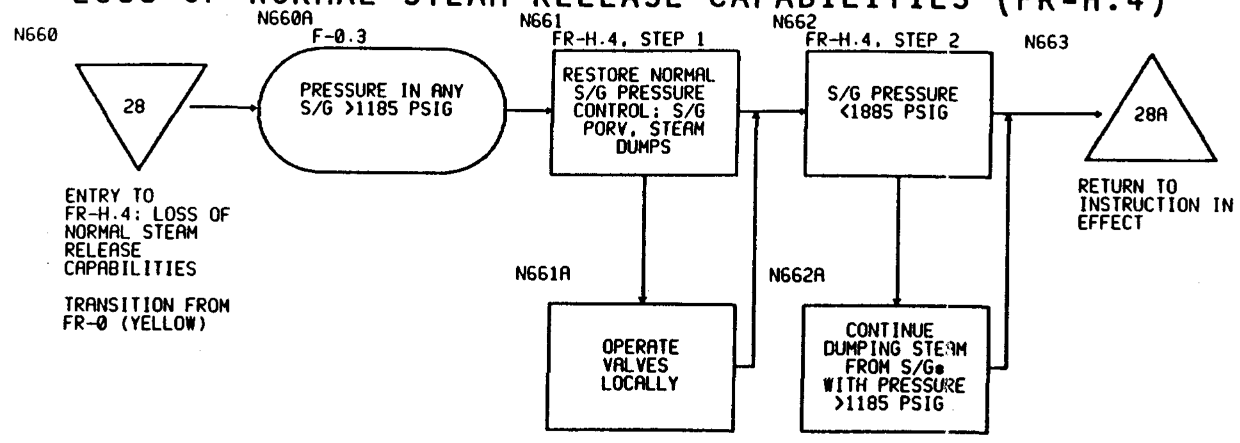
SI APERTURE CARD

Also Available On Aperture Card

9209030229-32

Figure 3.1.2-2 (Page 32 of 44).
Watts Bar ESD

LOSS OF NORMAL STEAM RELEASE CAPABILITIES (FR-H.4)

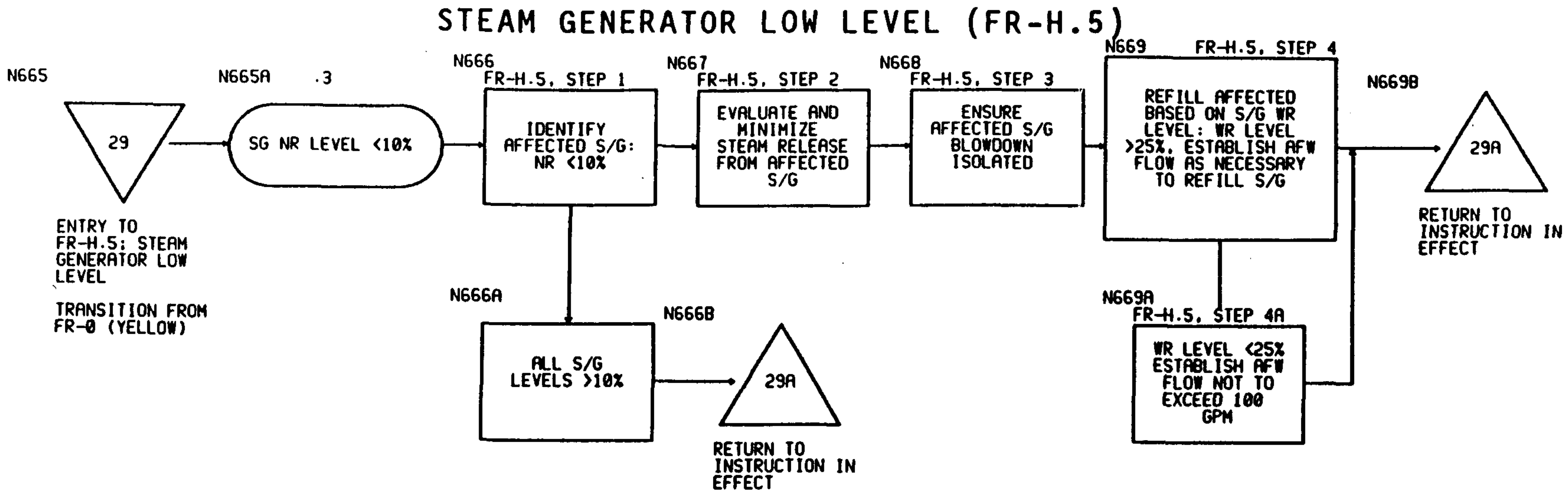


SI APERTURE CARD

Also Available On Aperture Card

9209030229-33

Figure 3.1.2-2 (Page 33 of 44). Watts Bar ESD



SI APERTURE CARD

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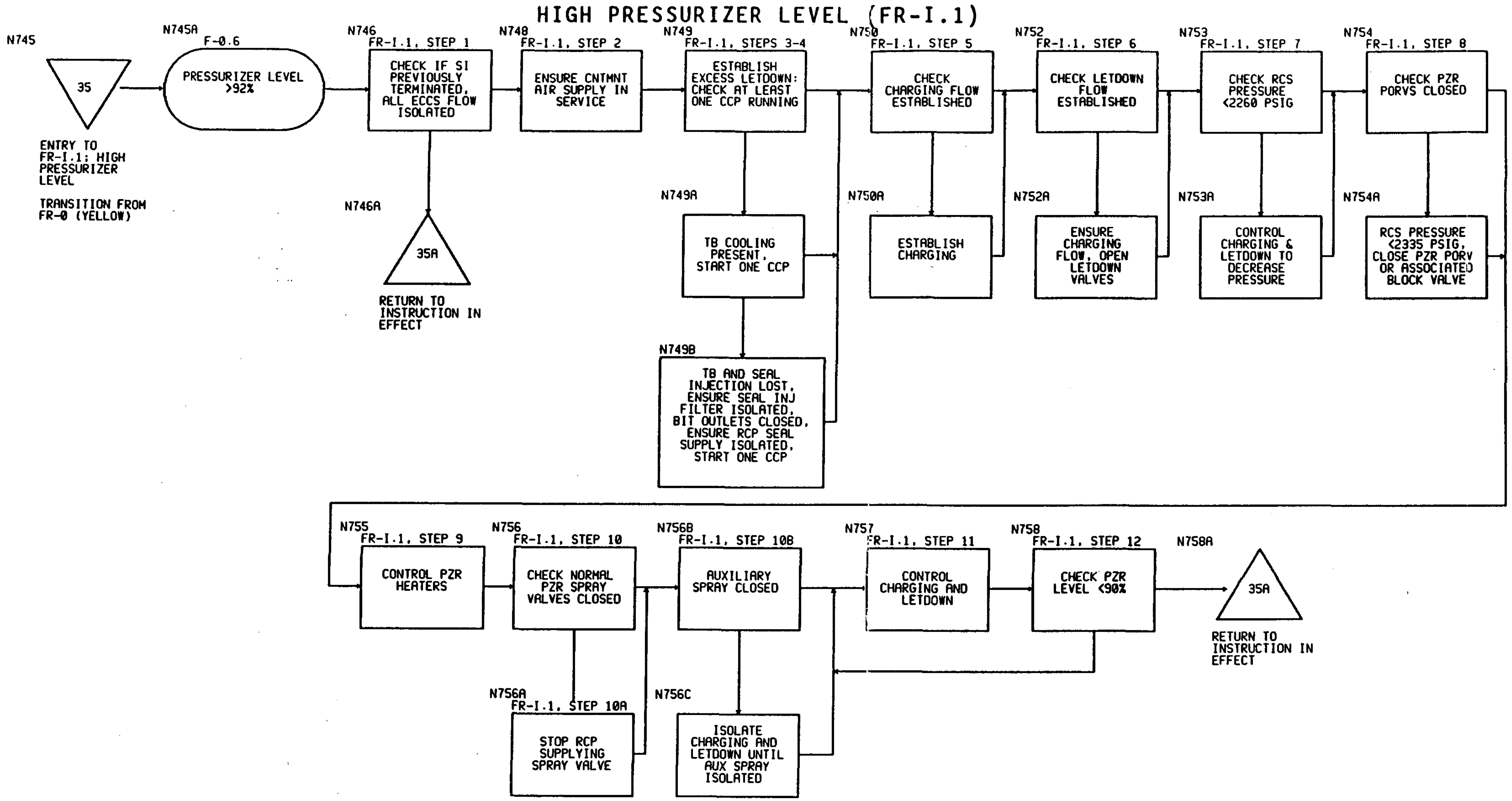
9209030229-34

Figure 3.1.2-2 (Page 34 of 44). Watts Bar ESD

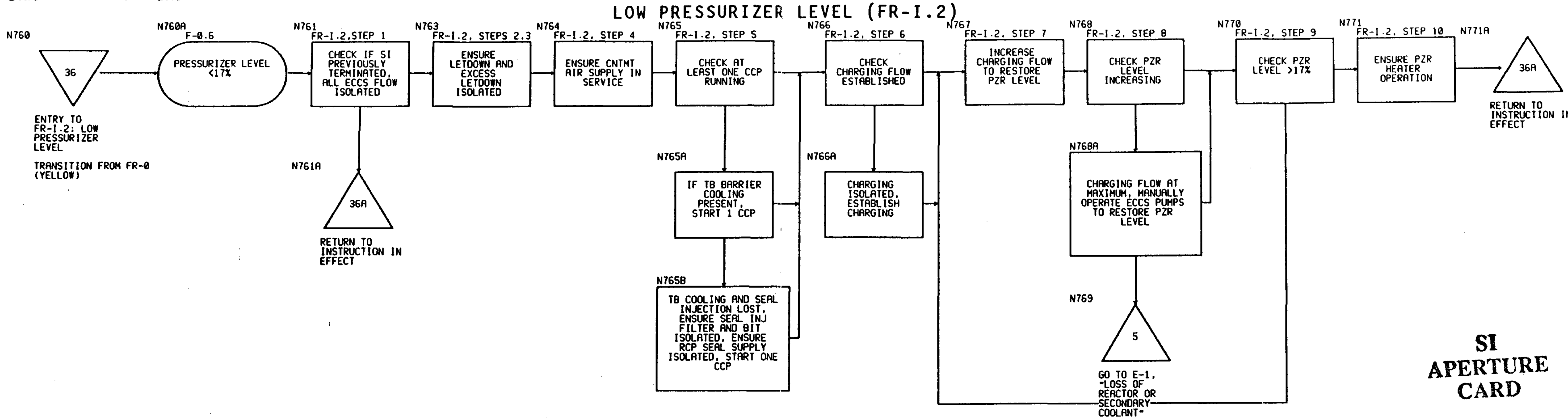


SI APERTURE CARD

Also Available On Aperture Card

9209030229-35

Figure 3.1.2-2 (Page 35 of 44).
Watts Bar ESD



SI APERTURE CARD

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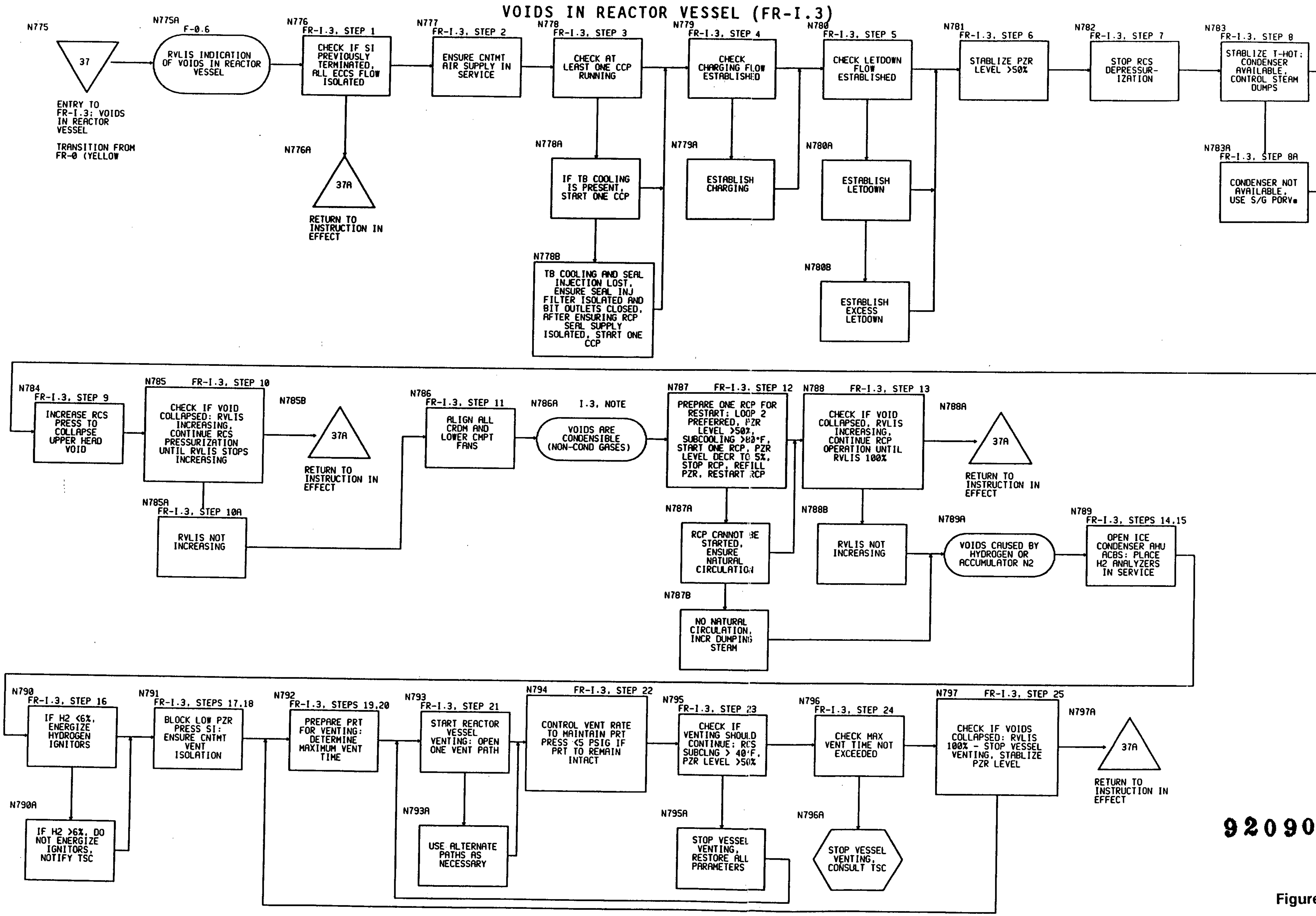
9209030229-36

Figure 3.1.2-2 (Page 36 of 44). Watts Bar ESD

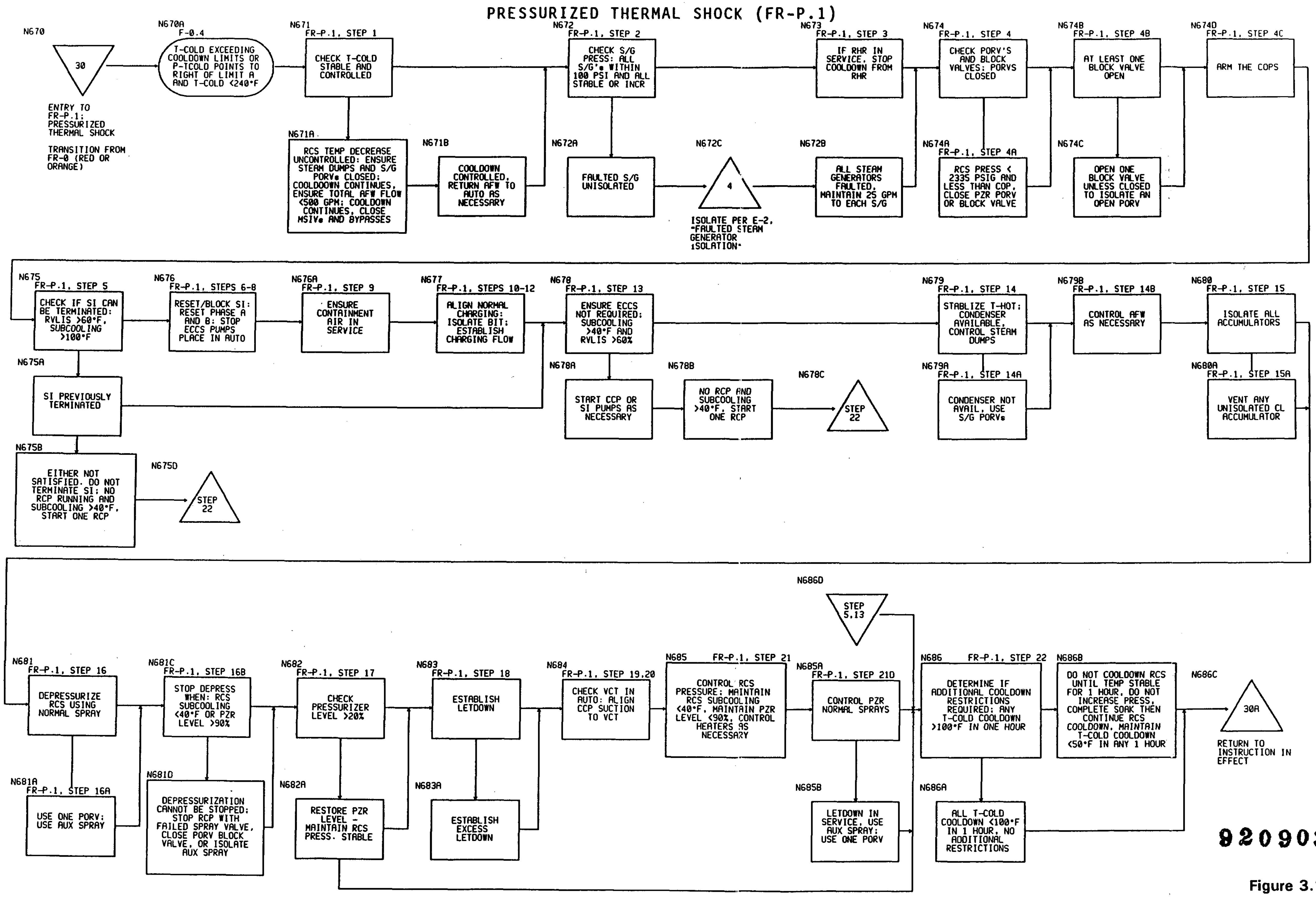


SI APERTURE CARD

Also Available On Aperture Card

9209030229 -37

Figure 3.1.2-2 (Page 37 of 44). Watts Bar ESD



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9209030229-38

Figure 3.1.2-2 (Page 38 of 44). Watts Bar ESD

COLD OVERPRESSURE CONDITION (FR-P.2)

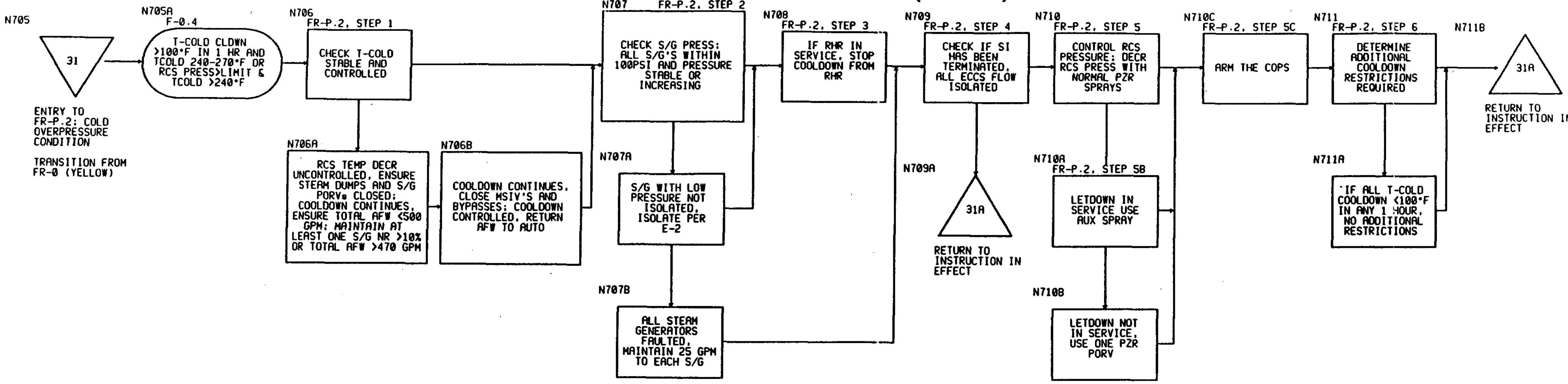

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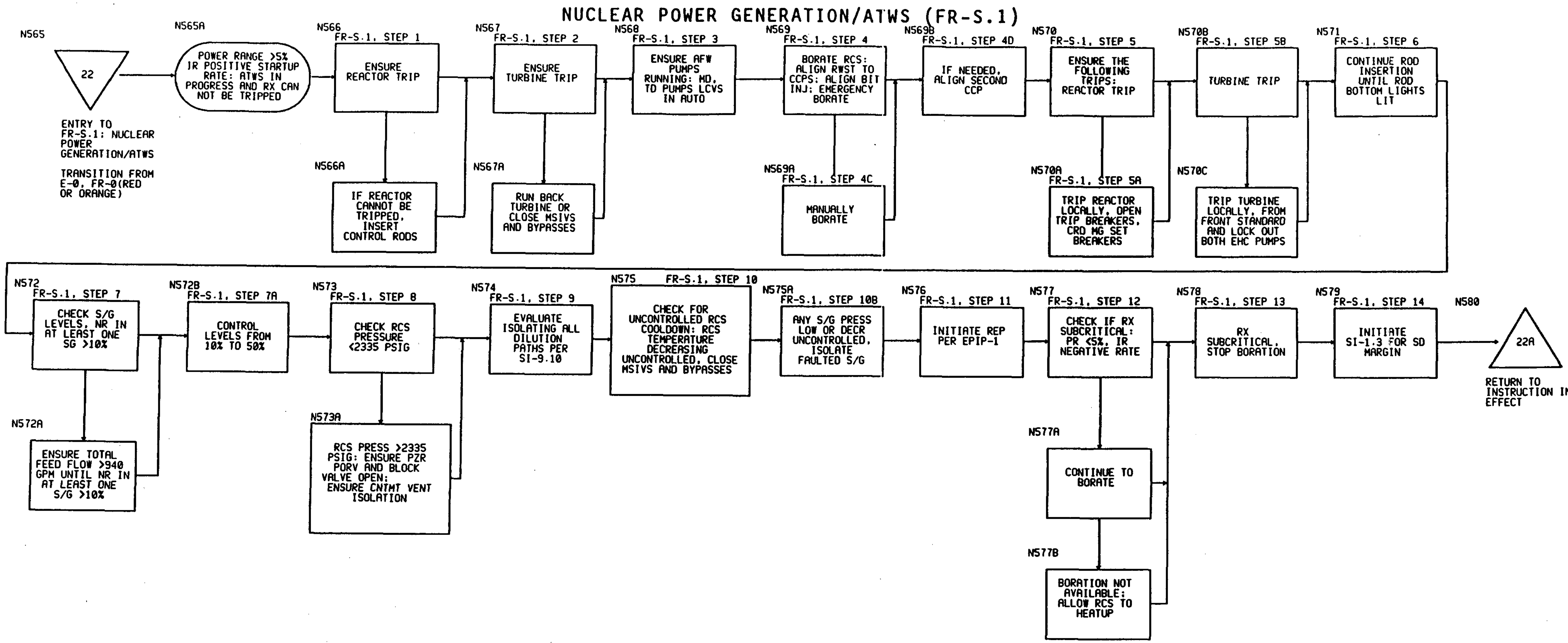
Figure 3.1.2-2 (Page 39 of 44).
Watts Bar ESD



SI APERTURE CARD

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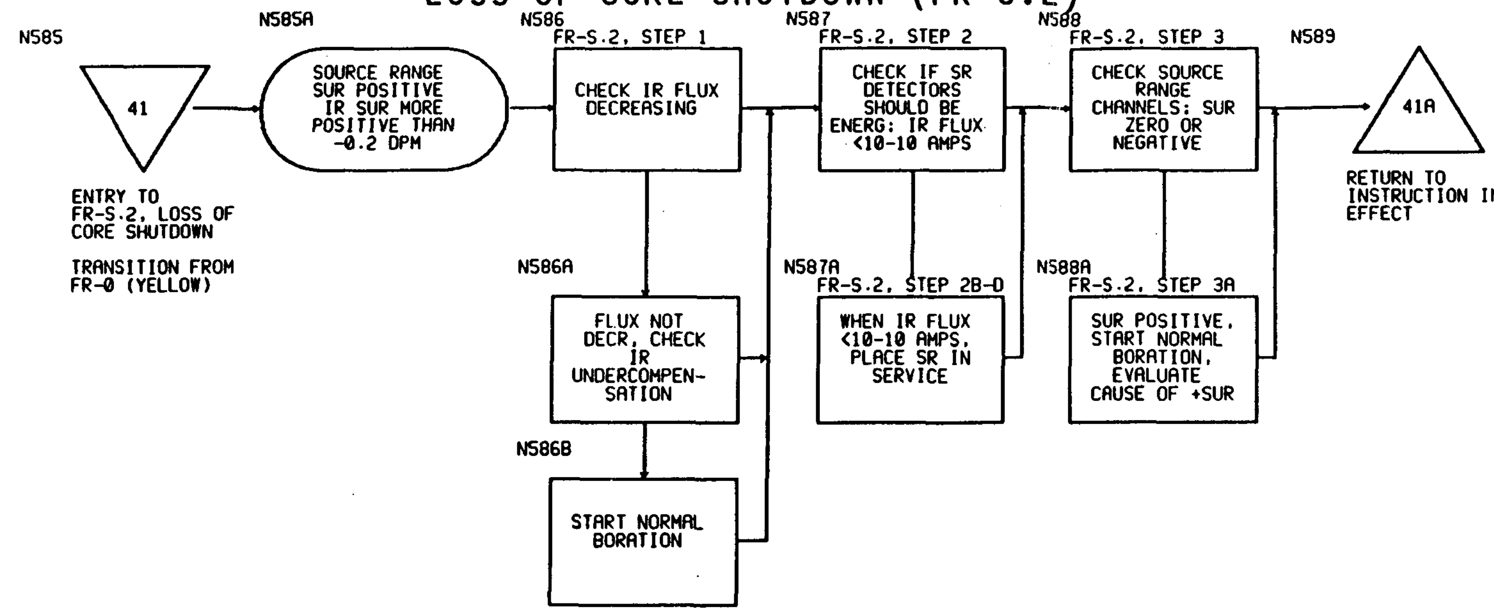
9209030229-40

Figure 3.1.2-2 (Page 40 of 44). Watts Bar ESD

LOSS OF CORE SHUTDOWN (FR-S.2)



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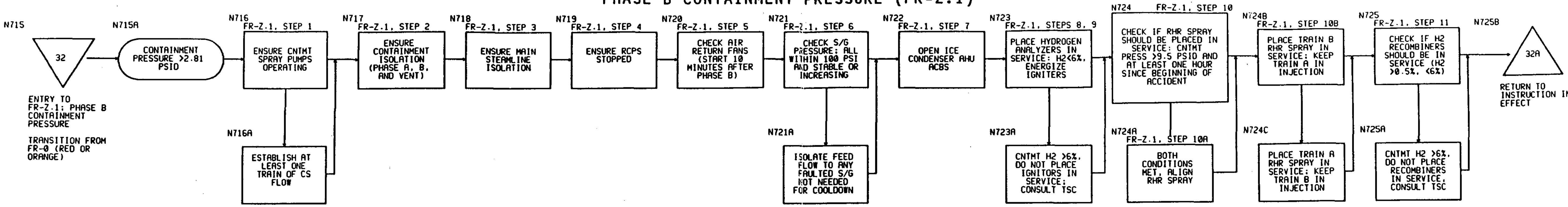
9209030229-41

Figure 3.1.2-2 (Page 41 of 44). Watts Bar ESD

PHASE B CONTAINMENT PRESSURE (FR-Z.1)

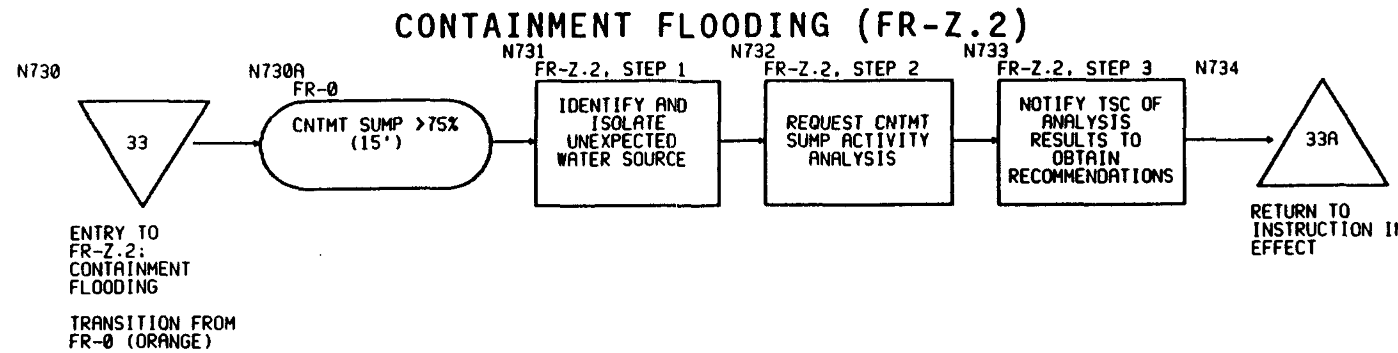


SI APERTURE CARD

Also Available On Aperture Card

9209030229-42

Figure 3.1.2-2 (Page 42 of 44). Watts Bar ESD



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CARD

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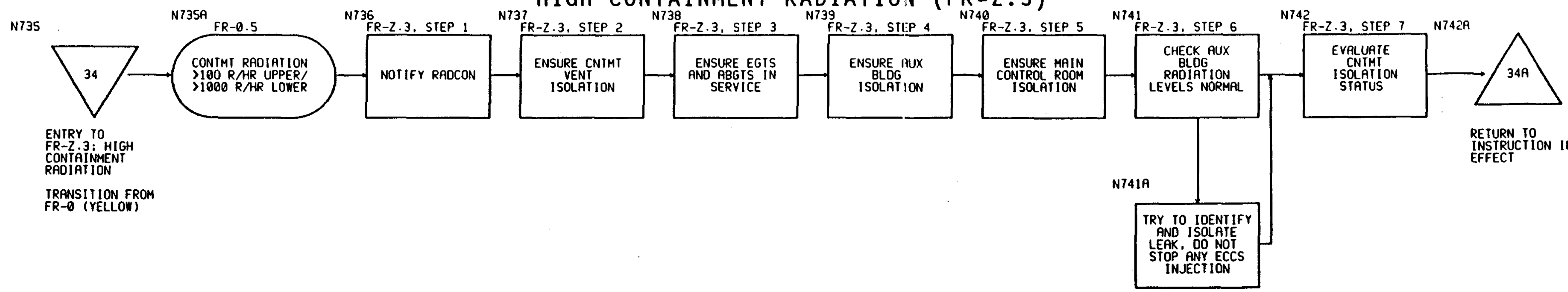
Aperture Card

9209080229 -43

Figure 3.1.2-2 (Page 43 of 44).
Watts Bar ESD

HIGH CONTAINMENT RADIATION (FR-Z.3)



SI APERTURE CARD

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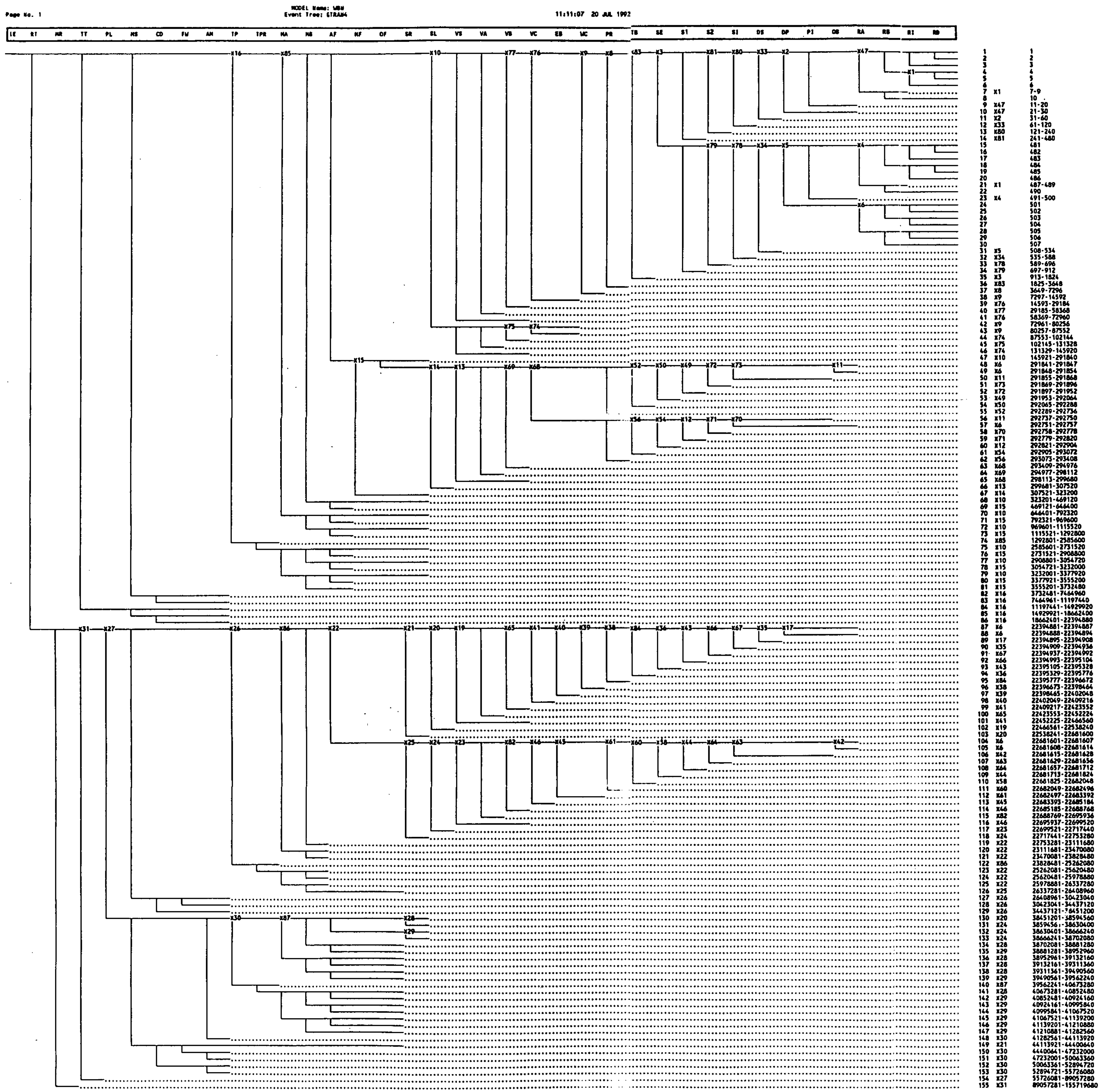
9209030229 -44

Figure 3.1.2-2 (Page 44 of 44). Watts Bar ESD



MODEL Name: WBN
Top Event Legend for Tree: GTRAN4

11:12:01 20 JUL 1992
Page 1

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
RT	REACTOR TRIPS, CONTROL RODS INSERT
MR	MANUAL ROD INSERTION, ATWS ONLY, OPERATOR ACTION DURING FIRST MINUTE
TT	TURBINE TRIP, INCLUDES TURBINE STOP & GOVERNOR VALVES & MOISTURE SEPERATOR REHEATER BYPASS VALVES
PL	POWER LEVEL IS LESS THAN 40%, ATWS ONLY
MS	MSIVS CLOSE, THREE OUT OF FOUR (3/4)
CD	COOLDOWN USING STEAM DUMPS, CONDENSER, & HOTWELL PUMPS TO SUPPLY MFW OR CST
FW	MAIN FEEDWATER CONTINUES DURING AN ATWS EVENT
AM	AMSAC TRIPS TURBINE & STARTS AFW
TP	TURBINE DRIVEN AFW PUMP
TPR	RECOVERY OF TD AFW PUMP START FAILURES IN 30 MINUTES
MA	MOTOR DRIVEN AFW PUMP 1A-A
MB	MOTOR DRIVEN AFW PUMP 1B-B
AF	DISCHARGE PATH FROM THE AFW PUMPS TO THE STEAM GENERATORS
MF	EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER
OF	OPERATOR ACTIONS TO RECOVER MAIN FEEDWATER
SR	STEAM RELIEF, ATWS ONLY, REACTOR PRESSURE IS LESS THAN 3200 PSIG
SL	OPERATOR IDENTIFIES & ISOLATES RUPTURES STEAM GENERATOR
VS	SUPPLY TO CVCS
VA	CENTRIFUGAL CHARGING PUMP 1A-A
VB	CENTRIFUGAL CHARGING PUMP 1B-B
VC	1/4 COLD LEG INJECTION PATH FROM CCP
EB	EMERGENCY BORATION, OPERATOR ACTIONS & EQUIPEMENT
WC	NO WATER CHALLENGE TO PRESSURIZER PORVS
PR	PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOSE
TB	THERMAL BARRIERS TO THE RCPS
SE	RCP SEAL INTEGRITY, SEAL COOLING IS MAINTAINED OR THERMAL BARRIERS PREVENT SEAL DAMAGE & OPERATOR SHUTS DOWN RCP ON LOSS OF BEARING COOLING
S1	SAFETY INJECTION PUMP 1A-A
S2	SAFETY INJECTION PUMP 1B-B
SI	1/4 COLD LEG INJECTION PATHS
DS	OPERATOR DEPRESSURIZES THE RCS USING THE STEAM GENERATOR PORVS
DP	OPERATOR DEPRESSURIZES THE RCS USING THE PZR SPRAYS AND PORVS
PI	PORVS RECLOSED IF OPENED IN DP
OB	OPERATOR ACTION TO FEED & BLEED RCS
RA	RHR PUMP 1A-A
RB	RHR PUMP 1B-B
RI	1/4 COLD LEG INJECTION PATH
RD	RHR NORMAL DECAY HEAT REMOVAL

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9209030229-45

Figure 3.1.2-3.
Watts Bar GENTRANS Event Tree

SI APERTURE CARD

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Page No. 1

MODEL Name: WBN
Event Tree: RECIR2

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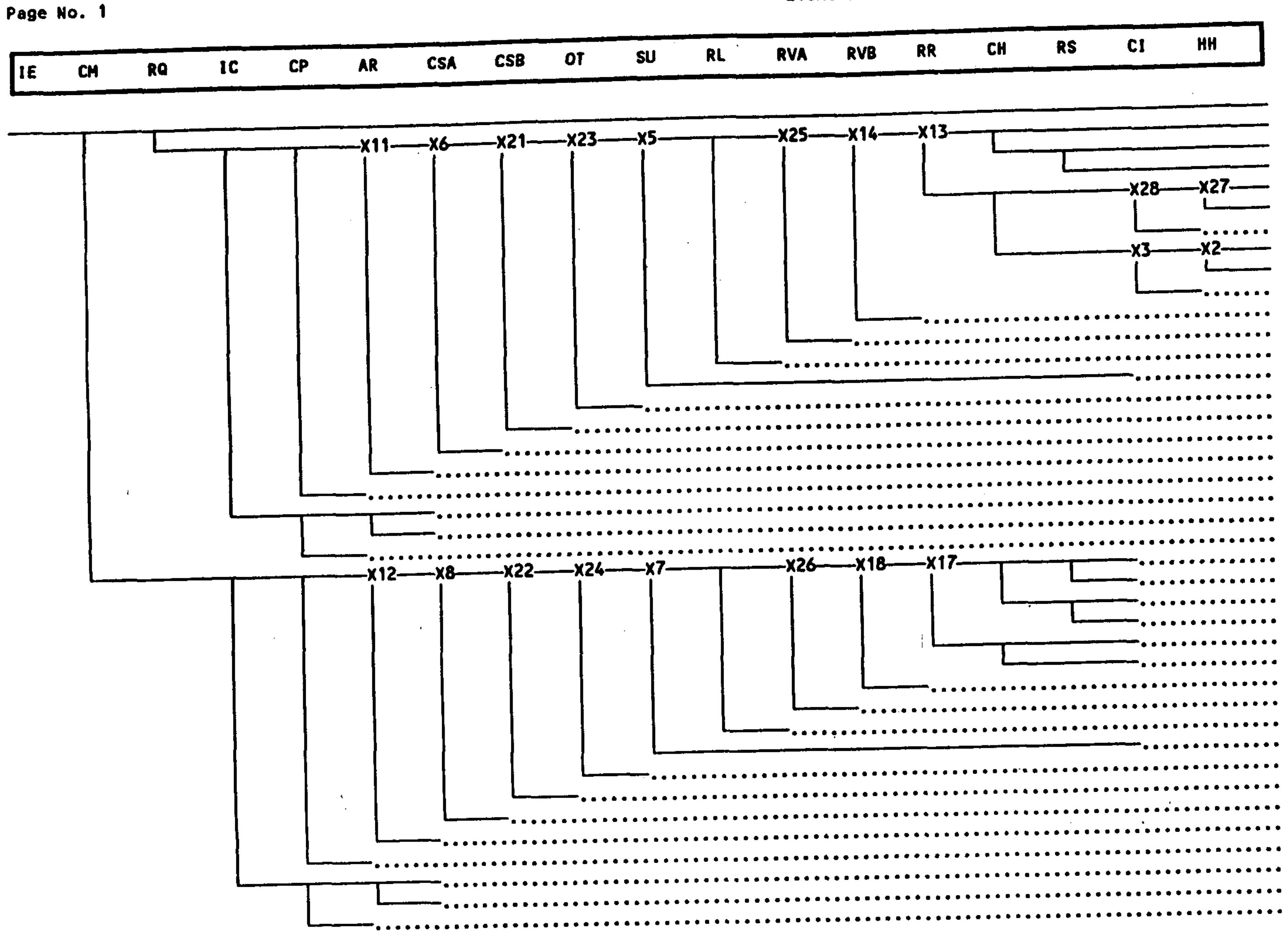

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7	X27	7-8
8		9
9		10
10	X2	11-12
11	X13	13-23
12	X14	24-45
13	X25	46-89
14	X3	90-93
15	X5	94-185
16	X23	186-369
17	X21	370-737
18	X6	738-1473
19	X11	1474-2945
20	X6	2946-3681
21	X6	3682-4417
22	X11	4418-5889
23	X3	5890-5893
24	X3	5894-5897
25	X3	5898-5901
26	X3	5902-5905
27	X28	5906-5909
28	X3	5910-5913
29	X17	5914-5937
30	X18	5938-5985
31	X26	5986-6081
32	X3	6082-6085
33	X7	6086-6281
34	X24	6282-6673
35	X22	6674-7457
36	X8	7458-9025
37	X12	9026-12161
38	X8	12162-13729
39	X8	13730-15297
40	X12	15298-18433

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
CM	CORE IS MELTED DURING INJECTION MODE
RQ	SUMP RECIRCULATION IS NOT REQUIRED
IC	ICE CONDENSER
CP	CONTAINMENT PURGE ISOLATION
AR	AIR RETURN FANS
CSA	TRAIN A CONTAINMENT SPRAY PUMP & VALVES
CSB	TRAIN B CONTAINMENT SPRAY PUMP & VALVES
OT	OPERATOR CONTROLS CONTAINMENT SPRAY
SU	CONTAINMENT SUMP IS AVAILABLE
RL	LEVEL CONTROL SWITCHES FOR RHR SWAPOVER
RVA	TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72
RVB	TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73
RR	AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR & SUPPLY TO SI & CVCS PUMPS
CH	CONTAINMENT SPRAY HEAT EXCHANGERS
RS	RHR SPRAY
CI	CONTAINMENT ISOLATION
HH	HYDROGEN IGNITORS

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Figure 3.1.2-4.
Watts Bar RECIRC Event Tree

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Page No. 1

MODEL Name: WBN
Event Tree: MEDLOC2

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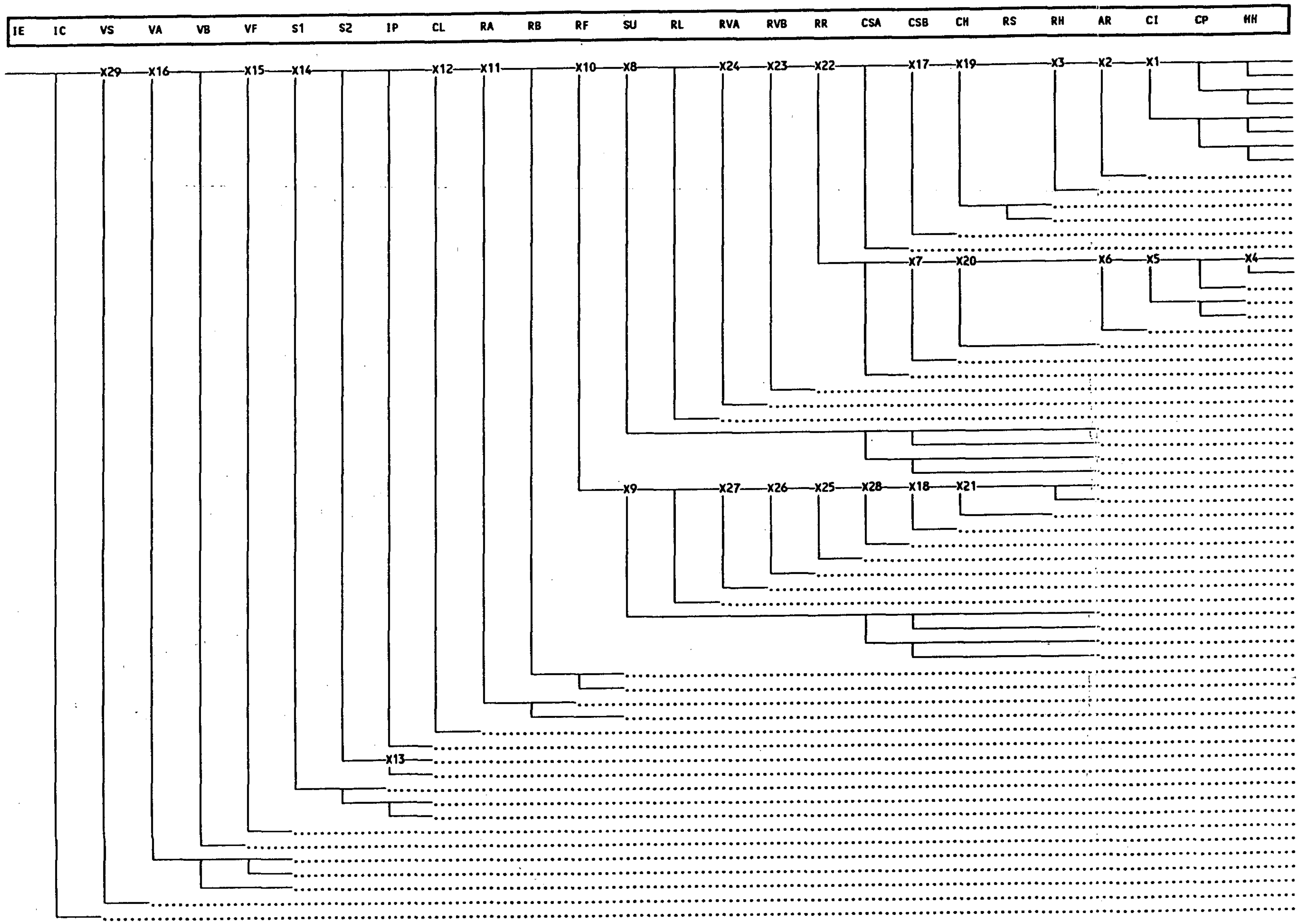

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9	X1	9-16
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11	X3	33-64
12	X3	65-96
13	X19	97-192
14	X17	193-384
15		385
16		386
17	X4	387-388
18	X4	389-390
19	X4	391-392
20	X5	393-400
21	X6	401-416
22	X20	417-448
23	X7	449-512
24	X22	513-1024
25	X23	1025-2048
26	X24	2049-4096
27	X6	4097-4112
28	X6	4113-4128
29	X6	4129-4144
30	X6	4145-4160
31	X2	4161-4176
32	X2	4177-4192
33	X3	4193-4224
34	X21	4225-4288
35	X18	4289-4416
36	X28	4417-4672
37	X25	4673-5184
38	X26	5185-6208
39	X27	6209-8256
40	X6	8257-8272
41	X6	8273-8288
42	X6	8289-8304
43	X6	8305-8320
44	X8	8321-12480
45	X9	12481-16640
46	X10	16641-24960
47	X9	24961-29120
48	X11	29121-58240
49	X12	58241-116480
50	X12	116481-174720
51	X12	174721-232960
52	X13	232961-349440
53	X12	349441-407680
54	X12	407681-465920
55	X14	465921-931840
56	X15	931841-1863680
57	X14	1863681-2329600
58	X14	2329601-2795520
59	X14	2795521-3261440
60	X16	3261441-6522880
61	X29	6522881-13045760

Top Event Designator.....	Top Event Description.............................
IE	INITIATING EVENT
IC	ICE CONDENSER
VS	SUPPLY TO CVCS
VA	CENTRIFUGAL CHARGING PUMP 1A-A
VB	CENTRIFUGAL CHARGING PUMP 1B-B
VF	2/3 CVCS COLD LEG INJECTION PATHS
S1	SAFETY INJECTION PUMP 1A-A
S2	SAFETY INJECTION PUMP 1B-B
IP	2/3 SIS COLD LEG INJECTION PATHS
CL	2/3 COLD LEG ACCUMULATORS
RA	RHR PUMP 1A-A
RB	RHR PUMP 1B-B
RF	2/3 RHR COLD LEG INJECTION PATHS
SU	CONTAINMENT SUMP
RL	SWAP SWAPOVER INSTRUMENTS
RVA	SUMP SWAPOVER VALVE, 1-FCV-63-72
RVB	SUMP SWAPOVER VALVE 1-FCV-63-73
RR	AUTOMATIC/MANUAL SWAPOVER FROM THE RWST TO THE CONTAINMENT SUMP
CSA	TRAIN A CONTAINMENT SPRAY PUMP & VALVES
CSB	TRAIN B CONTAINMENT SPRAY PUMP & VALVES
CH	CONTAINMENT SPRAY HEAT EXCHANGERS
RS	RHR SPRAY
RH	RHR & SIS HOT LEG RECIRCULATION
AR	AIR RETURN FANS
CI	CONTAINMENT ISOLATION
CP	CONTAINMENT PURGE IS CLOSED
HH	HYDROGEN IGNITORS

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Figure 3.1.2-5.
Watts Bar MLOCA Event Tree

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Page No. 1

MODEL Name: WBN
Event Tree: LARLOC4

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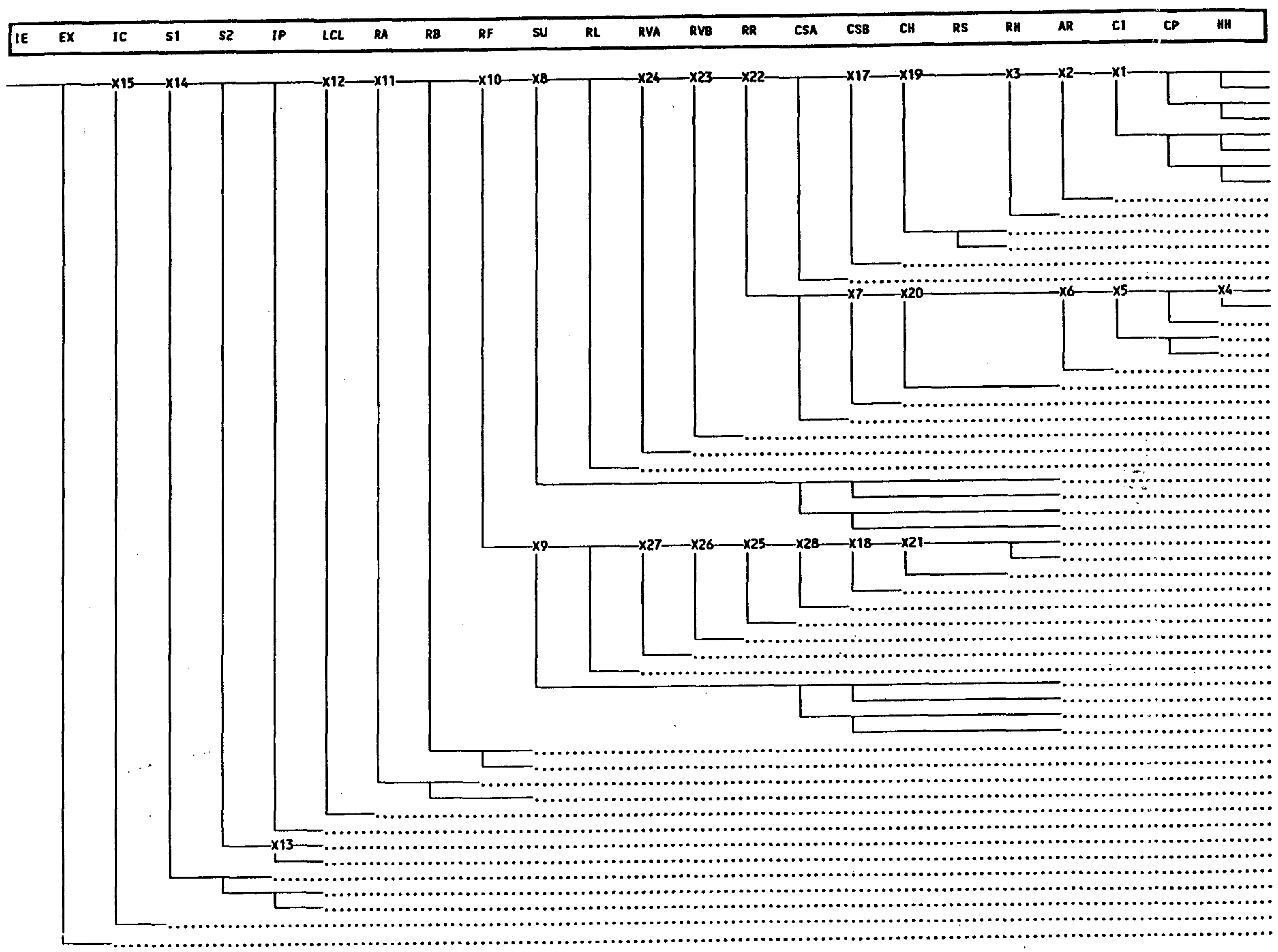

1		1
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8		8
9	X1	9-16
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11	X3	33-64
12	X3	65-96
13	X19	97-192
14	X17	193-384
15		385
16		386
17	X4	387-388
18	X4	389-390
19	X4	391-392
20	X5	393-400
21	X6	401-416
22	X20	417-448
23	X7	449-512
24	X22	513-1024
25	X23	1025-2048
26	X24	2049-4096
27	X6	4097-4112
28	X6	4113-4128
29	X6	4129-4144
30	X6	4145-4160
31	X2	4161-4176
32	X2	4177-4192
33	X3	4193-4224
34	X21	4225-4288
35	X18	4289-4416
36	X28	4417-4672
37	X25	4673-5184
38	X26	5185-6208
39	X27	6209-8256
40	X6	8257-8272
41	X6	8273-8288
42	X6	8289-8304
43	X6	8305-8320
44	X8	8321-12480
45	X9	12481-16640
46	X10	16641-24960
47	X9	24961-29120
48	X11	29121-58240
49	X12	58241-116480
50	X12	116481-174720
51	X12	174721-232960
52	X13	232961-349440
53	X12	349441-407680
54	X12	407681-465920
55	X14	465921-931840
56	X15	931841-1863680

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
EX	EXCESSIVE LOCA
IC	ICE CONDENSER
S1	SAFETY INJECTION PUMP 1A-A
S2	SAFETY INJECTION PUMP 1B-B
IP	2/3 SIS COLD LEG INJECTION PATHS
LCL	3/3 COLD LEG ACCUMULATORS
RA	RHR PUMP 1A-A
RB	RHR PUMP 1B-B
RF	2/3 RHR COLD LEG INJECTION PATHS
SU	CONTAINMENT SUMP
RL	SWAP SWAPOVER INSTRUMENTS
RVA	SUMP SWAPOVER VALVE, 1-FCV-63-72
RVB	SUMP SWAPOVER VALVE 1-FCV-63-73
RR	AUTOMATIC/MANUAL SWAPOVER FROM THE RWST TO THE CONTAINMENT SUMP
CSA	TRAIN A CONTAINMENT SPRAY PUMP & VALVES
CSB	TRAIN B CONTAINMENT SPRAY PUMP & VALVES
CH	CONTAINMENT SPRAY HEAT EXCHANGERS
RS	RHR SPRAY
RH	RHR & SIS HOT LEG RECIRCULATION
AR	AIR RETURN FANS
CI	CONTAINMENT ISOLATION
CP	CONTAINMENT PURGE IS CLOSED
HH	HYDROGEN IGNITORS

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Figure 3.1.2-6.
Watts Bar LLOCA Event Tree

3.1.3 SPECIAL EVENT TREES

This section describes the special event trees used in the Level 1 sequence frequency quantification. These trees are used together with the support and frontline event trees to form a complete accident sequence model. First, the containment interface trees are described; then, the recovery event tree; and, finally, references to other report sections for unique event tree considerations are listed.

3.1.3.1 Containment Interface Trees

Separate containment interface trees were developed for each of the major frontline event trees; i.e., for the GENTRANS/RECIRC combination, for medium loss of coolant accident (MLOCA), and for large loss of coolant accident (LLOCA). These containment interface trees are used solely to assign each accident sequence through the frontline event trees to an end state that reflects the plant conditions at the end of the Level 1 event tree models.

The top events of the containment interface trees are simply switches that have probabilities of zero or one, depending only on the sequence of events up to that portion of the event tree models.

Use of the containment interface trees simplifies the top event rule logic used for assigning end states. The status of top events in the containment interface trees replaces portions of the logic that would otherwise have to be used to describe when the events are true or not. Use of the containment interface trees simplifies the end state assignment logic enough to permit RISKMAN to make the assignments automatically.

The Level 2 containment response event tree is quantified uniquely for each Level 1 end state with significant frequency. The containment response event tree quantification is discussed in Section 4.8. The development of the Level 1 end states is described in Section 4.3.

The three containment interface trees are described in the following sections. The top events in each containment interface tree are first presented. Section 3.1.3.1.4 then describes how the plant damage states are assigned to individual sequences using the status of top events in the containment interface trees.

3.1.3.1.1 GENTRANS/RECIRC Containment Interface Tree

The GENTRANS/RECIRC containment interface tree is presented as Figure 3.1.3-1. The top events are described below. For convenient reference, Table 3.1.3-1 summarizes the top events that are modeled in the GENTRANS/RECIRC containment interface tree.

- **Top Event MELT — No Core Melt**
  - **Function Evaluated.** This top event tracks whether the sequence results in core damage. If the sequence leads to core damage during the injection phase, or if there is a need for recirculation from the containment sump and recirculation fails, then this top event is guaranteed failed. To fail recirculation requires (1) the failure of both residual heat removal (RHR) pump trains taking suction from the sump, (2) the containment sump

plugging or otherwise not having water from the refueling water storage tank (RWST), (3) failure of the RWST level instrumentation to initiate automatic swapover to the sump, or (4) failure to align for high pressure recirculation. Otherwise, this top event is guaranteed successful.

- **Conditions when Demanded.** This top event is asked for all sequences that exit the frontline event tree models and enter the containment interface tree.

- **Scenario Impact if Successful.** Success implies that the Watts Bar plant did not sustain core damage during the accident sequence. The sequence is then mapped to a success end state.

- **Scenario Impact if Failed.** Failure implies that core damage did occur. Additional questions must then be asked to determine the conditions imposed on the containment by the core damage sequence.

- **Top Event LOWPR — RCS Pressure Not Low (> 200 psia)**

  - **Function Evaluated.** This top event tracks whether the reactor coolant system (RCS) pressure at the onset of core damage is low; i.e., less than about 200 psia. For the GENTRANS/RECIRC set of frontline events trees, the only way for the RCS pressure to be low at the time of core damage is to have the RCS fail due to system overpressure during an anticipated transient without scram (ATWS) sequence; i.e., to have the RCS pressure exceed the assumed RCS boundary limit of 3,200 psig. This overpressure may result from sequences involving failure of reactor trip from greater than 40% power initially and (1) both main feedwater and auxiliary feedwater are unavailable, (2) main feedwater is unavailable and the main turbine fails to trip, or (3) a sufficient combination of pressurizer relief and safety valves fails to lift to limit RCS pressure to less than 3,200 psig. For all other core damage sequences, the RCS pressure will not be low at the time of core damage.

  - **Conditions when Demanded.** This top event is asked for all sequences involving core damage; i.e., when Top Event MELT has already failed.

  - **Scenario Impact if Successful.** Success implies that the RCS is at greater than 200 psia at the onset of core damage.

  - **Scenario Impact if Failed.** Failure implies that the RCS is at low pressure at the time of core damage.

- **Top Event INTPR — RCS Pressure < 2,000 psia**

  - **Function Evaluated.** This top event tracks whether the RCS is at an intermediate pressure at the onset of core damage (i.e., between 200 and 2,000 psia), given that RCS pressure was not initially low. Pressure is assumed to be between 200 and 2,000 psia for sequences involving a small loss of coolant accident (LOCA) via (1) the pressurizer via a failed-open valve or because bleed and feed cooling is in progress, (2) the reactor coolant pump (RCP) seals, or (3) a ruptured steam generator tube. The operators may take steps to cool down and depressurize the RCS during the injection phase with high pressure injection operating successfully, but such sequences are not expected to bring the RCS to pressures less than 200 psia before the onset of core damage during the recirculation phase.

- **Conditions when Demanded.** This event is asked only for core damage sequences in which the RCS pressure is not low; i.e., when Top Event MELT fails, but Top Event LOWPR is successful.

- **Scenario Impact if Successful.** Success implies that RCS pressure is greater than 2,000 psia at the onset of core damage. RCS pressure may be high if there is no LOCA, but all secondary heat removal is lost and bleed and feed cooling fails.

- **Scenario Impact if Failed.** Failure implies that the RCS pressure is between 200 and 2,000 psia at the onset of core damage.

• **Top Event MELTB — Melt with Containment Bypassed**

- **Function Evaluated.** This top event tracks whether the sequence results in core damage with a release path that bypasses containment. For the GENTRANS/RECIRC frontline event tree set, the only bypass path modeled is via a ruptured steam generator tube and a coincident failed-open valve on the secondary side of the ruptured steam generator. This may result when the plant trip is caused by a steam generator tube rupture, or a tube rupture occurs in response to an increased pressure drop across the steam generator tubes (e.g., in an ATWS or steam line break sequence), and either the secondary valves on the ruptured steam generator fail open initially or in response to repeated cycling later in the transient. The secondary valves are assumed to fail open if the reactor fails to trip (i.e., when the RCS pressure would get very high) or if the operators fail to cool down and depressurize the RCS so that continued leakage from the primary to the secondary side of the ruptured steam generator occurs.

- **Conditions when Demanded.** This top event is asked for sequences involving core damage with the RCS pressure at the time of core damage being greater than 200 psia. For low pressure sequences, reactor vessel overpressure due to the early pressure increase in an ATWS sequence must have occurred; i.e., Top Event LOWPR failed. For such sequences, the overpressure is assumed to have occurred too fast to repeatedly cycle a secondary valve, even if a ruptured steam generator tube initiated the sequence. Therefore, for these sequences, the secondary valves are assumed to be isolated.

- **Scenario Impact if Successful.** Success implies that there is no bypass path that bypasses the containment.

- **Scenario Impact if Failed.** Failure implies that a release path through a ruptured steam generator and a failed-open secondary valve is available.

- **Top Event SGCLG — Steam Generator Cooling**

  - **Function Evaluated.** This top event tracks whether the accident sequence involves successful cooling of at least two of the steam generators. Steam generator cooling is successful if at least one auxiliary feedwater (AFW) pump provides cooling to two steam generators, or main feedwater operates successfully. For ATWS sequences, flow from two motor-driven AFW pumps may be required for sufficient heat removal to prevent core damage. However, given that core damage has occurred, only one pump is required to protect the steam generator tubes from potential thermal degradation.

  - **Conditions when Demanded.** This top event is asked for all sequences entering the containment interface tree, except for those core damage sequences in which the RCS pressure at the onset of core damage is low. The RCS pressure is considered low when it is less than 200 psia at the onset of core damage as tracked by the status of Top Event LOWPR. When RCS pressure is low, the heat transfer to the steam generators during a core damage sequence is minimized.

  - **Scenario Impact if Successful.** Success implies that cooling via the secondary is available. The steam generator tubes are thereby protected from thermal degradation resulting from the progression of the accident.

  - **Scenario Impact if Failed.** Failure implies that steam generator cooling is not available. The potential for thermal degradation of the steam generator tubes still exists.

- **Top Event MELTI — Melt without Containment Isolated**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with an isolated containment. For the GENTRANS/RECIRC frontline event tree set, containment isolation failures are modeled by Top Events CI and CP. For successful containment isolation, both of these top events must be successful.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences that enter the containment interface tree that do not involve containment bypass; i.e., Top Event MELTB must not be failed.

  - **Scenario Impact if Successful.** Success implies that there is a core damage sequence and the containment is not isolated. Either a large or small hole is present. The size of the hole must then be determined via Top Events MELTS and MELTL.

  - **Scenario Impact if Failed.** Failure implies that the containment is isolated and no bypass flow path is present.

- **Top Event MELTS — Melt with Large Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the GENTRANS/RECIRC frontline event tree set, containment isolation failures are modeled by Top Events CI and CP. For only a small penetration isolation failure, Top Event CP must be successful, and Top Event CI must be failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree that do not involve a containment bypass path but do involve failure of containment isolation.

  - **Scenario Impact if Successful.** Success implies that the containment penetration that failed to isolate must be large.

  - **Scenario Impact if Failed.** Failure implies that there is just a small penetration line that failed to isolate.

- **Top Event MELTL — Melt with Small Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the GENTRANS/RECIRC frontline event tree set, containment isolation failures are modeled by Top Events CI and CP. For a large penetration isolation failure, Top Event CP must be failed. Top Event CI may be successful or failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for core damage sequences involving a failure of containment isolation (i.e., Top Event MELTI is successful) but not involving a small penetration isolation failure; i.e., Top Event MELTS is successful.

  - **Scenario Impact if Successful.** Success implies that the containment isolation failure involved only a small containment penetration.

  - **Scenario Impact if Failed.** Failure implies that the containment isolation failure involved a large size penetration.

- **Top Event CSI — Containment Spray Injection**

  - **Function Evaluated.** This top event tracks the status of containment spray injection. The spray pump trains are modeled via Top Events CSA and CSB. Success of either Top Event CSA or CSB constitutes successful injection.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

- **Scenario Impact if Successful.** Success implies that at least one of the containment spray pumps operate in the injection mode in response to the high containment pressures experienced in a core damage sequence.

- **Scenario Impact if Failed.** Failure implies that containment spray is not available in the injection mode for containment pressure suppression.

• **Top Event CSR — Containment Spray Recirculation**

- **Function Evaluated.** This top event tracks the availability of containment spray in the recirculation mode. Success requires that at least one of the containment spray pumps operates (i.e., Top Event CSA or CSB, and therefore Top Event CSI must be successful), that the sump suction valves for the operable pump train be aligned for sump recirculation, and that the operators align cooling water to the associated containment spray heat exchanger.

- **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

- **Scenario Impact if Successful.** Success implies that containment spray is available in the recirculation mode.

- **Scenario Impact if Failed.** Failure implies that containment spray is not available for recirculation.

• **Top Event RHRS — RHR Spray Recirculation**

- **Function Evaluated.** This top event tracks the availability of containment spray via the RHR pumps, operating in the recirculation mode. Success requires that at least one RHR pump train operate, with its associated sump suction valve aligned, that swapover for recirculation from the sump be completed including the alignment of the component cooling water system (CCS) to the RHR heat exchangers, and that the operators manually align for RHR spray.

- **Conditions when Demanded.** This top event is asked for all sequences in the containment interface tree.

- **Scenario Impact if Successful.** Success implies that RHR spray is available for containment heat removal in the recirculation mode.

- **Scenario Impact if Failed.** Failure implies that RHR spray is not available for containment heat removal in the recirculation mode.

• **Top Event CAV — Water in Reactor Cavity**

- **Function Evaluated.** This top event tracks the availability of water in the reactor cavity for long-term debris bed cooling after core damage. Success requires both the melted ice from the ice condenser and the RWST contents.

Melted ice from the ice condenser is available as long as the ice condenser doors open as required; i.e., success of Top Event IC. The RWST water must be injected to containment by one of the safety injection pumps, the RHR pumps, the containment spray pumps, or the charging pumps. Injection of the RWST inventory into the RCS following penetration of the fuel debris through the RCS boundary (i.e., most likely through the reactor vessel bottom head) is counted as a successful injection of the inventory into containment since that is where it ends up. The upper to lower compartment drain lines must also not be plugged so that containment spray flow does not transfer the RWST inventory to the upper containment compartment, making it unavailable to the lower compartment.

- **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree.
- **Scenario Impact if Successful.** Success implies that sufficient water is in the reactor cavity for debris bed cooling.
- **Scenario Impact if Failed.** Failure implies that there is insufficient water in the reactor cavity for debris bed cooling.

3.1.3.1.2 MLOCA Containment Interface Tree

The MLOCA containment interface tree is presented as Figure 3.1.3-2. The MLOCA containment interface tree is similar to that for GENTRANS/RECIRC except that three top events are left out. For medium LOCAs, it is not possible to have the RCS at greater than 2,000 psia at the onset of core damage. Also, the potential for a containment bypass is insignificant with the RCS pressure so low. The availability of steam generator cooling is also not significant because RCS pressure is too low to thermally induce steam generator tube failure during the core damage progression. Therefore, Top Events INTPR, MELTB, and SGCLG are not included in this tree. The top events that are included are described below. For convenient reference, Table 3.1.3-2 summarizes the top events that are modeled in the MLOCA containment interface tree.

- **Top Event MELT — No Core Melt for Medium LOCA**
  - **Function Evaluated.** This top event tracks whether the sequence results in core damage. If the sequence leads to core damage during the injection phase, or if recirculation from the containment sump fails, then this top event is guaranteed failed. Injection could fail due to insufficient accumulator injection, failure of both RHR pumps to provide cold leg injection, or failure of three of four high pressure injection pumps; i.e., three of the four safety and charging pumps. To fail recirculation requires (1) the failure of both RHR pump trains taking suction from the sump, (2) the containment sump plugging or otherwise not having water from the RWST, (3) failure of the RWST level instrumentation to initiate automatic swapover to the sump, or (4) failure to align for high pressure recirculation. Otherwise, this top event is guaranteed successful.

- **Conditions when Demanded.** This top event is asked for all sequences that exit the frontline event tree model and enter the containment interface tree.

- **Scenario Impact if Successful.** Success implies that the Watts Bar plant did not sustain core damage during the accident sequence. The sequence is then mapped to a success end state.

- **Scenario Impact if Failed.** Failure implies that core damage did occur. Additional questions must then be asked to determine the conditions imposed on the containment by the core damage sequence.

• **Top Event LOWPR — RCS Pressure Not Low (> 200 psia)**

- **Function Evaluated.** This top event tracks whether the RCS pressure at the onset of core damage is low; i.e., less than about 200 psia. For the MLOCA frontline event tree, RCS pressure is assumed to be low at the onset of core damage only if three or more of the high pressure injection pumps fail; i.e., three of the four safety injection and charging pumps. For all other core damage sequences, the RCS pressure will not be low at the time of core damage.

- **Conditions when Demanded.** This top event is asked for all sequences involving core damage; i.e., when Top Event MELT has already failed.

- **Scenario Impact if Successful.** Success implies that the RCS is at greater than 200 psia at the onset of core damage.

- **Scenario Impact if Failed.** Failure implies that the RCS is at low pressure at the time of core damage.

• **Top Event MELTI — Melt without Containment Isolated**

- **Function Evaluated.** This top event tracks whether the sequence results in core damage with an isolated containment. For the MLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For successful containment isolation, both of these top events must be successful.

- **Conditions when Demanded.** This top event is asked for all core damage sequences that enter the containment interface tree.

- **Scenario Impact if Successful.** Success implies that there is a core damage sequence and that the containment is not isolated. Either a large or small hole is present. The size of the hole must then be determined via Top Events MELTS and MELTL.

- **Scenario Impact if Failed.** Failure implies that the containment is isolated.

- **Top Event MELTS — Melt with Large Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the MLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For only a small penetration isolation failure, Top Event CP must be successful, and Top Event CI must be failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences in which the containment is not isolated.

  - **Scenario Impact if Successful.** Success implies that the containment penetration that failed to isolate must be large.

  - **Scenario Impact if Failed.** Failure implies that there is just a small penetration line that failed to isolate.

- **Top Event MELTL — Melt with Small Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the MLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For a large penetration isolation failure, Top Event CP must be failed. Top Event CI may be successful or failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for core damage sequences involving a failure of containment isolation (i.e., Top Event MELTI is successful) but not involving a small penetration isolation failure; i.e., Top Event MELTS is successful.

  - **Scenario Impact if Successful.** Success implies that the containment isolation failure involved only a small containment penetration.

  - **Scenario Impact if Failed.** Failure implies that the containment isolation failure involved a large size penetration.

- **Top Event CSI — Containment Spray Injection**

  - **Function Evaluated.** This top event tracks the status of containment spray injection. The spray pump trains are modeled via Top Events CSA and CSB. Success of either Top Event CSA or CSB constitutes successful spray injection.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

- **Scenario Impact if Successful.** Success implies that at least one of the containment spray pumps operate in the injection mode in response to the high containment pressures experienced in a medium LOCA sequence.

- **Scenario Impact if Failed.** Failure implies that containment spray is not available in the injection mode for containment pressure suppression.

• **Top Event CSR — Containment Spray Recirculation**

- **Function Evaluated.** This top event tracks the availability of containment spray in the recirculation mode. Success requires that at least one of the containment spray pumps operates (i.e., Top Event CSA or CSB, and therefore Top Event CSI must be successful), that the sump suction valves for the operable pump train be aligned for sump recirculation, and that the operators align cooling water to the associated containment spray heat exchanger.

- **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

- **Scenario Impact if Successful.** Success implies that containment spray is available in the recirculation mode.

- **Scenario Impact if Failed.** Failure implies that containment spray is not available for recirculation.

• **Top Event RHRS — RHR Spray Recirculation**

- **Function Evaluated.** This top event tracks the availability of containment spray via the RHR pumps, operating in the recirculation mode. Success requires that at least one RHR pump train operate, with its associated sump suction valve aligned, that swapover for recirculation from the sump be completed including the alignment of CCS to the RHR heat exchangers, and that the operators manually align for RHR spray.

- **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree.

- **Scenario Impact if Successful.** Success implies that RHR spray is available for containment heat removal in the recirculation mode.

- **Scenario Impact if Failed.** Failure implies that RHR spray is not available for containment heat removal in the recirculation mode.

• **Top Event CAV — Water in Reactor Cavity**

- **Function Evaluated.** This top event tracks the availability of water in the reactor cavity for long-term debris bed cooling after core damage. Success requires both the melted ice from the ice condenser and the RWST contents. Melted ice from the ice condenser is available as long as the ice condenser doors open as required; i.e., success of Top Event IC. The RWST water must be injected to containment by one of the safety injection pumps, the RHR pumps, the containment spray pumps, or the charging pumps. Injection of the RWST inventory into the RCS following penetration of the fuel debris through the RCS (i.e., most likely through the reactor vessel bottom head) is counted as a successful injection of the inventory into containment since that is where it ends up. The containment sump must also not be plugged so that containment spray flow does not transfer the RWST inventory to the upper containment compartment, making it unavailable to the lower compartment.

- **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree.
- **Scenario Impact if Successful.** Success implies that sufficient water is in the reactor cavity for debris bed cooling.
- **Scenario Impact if Failed.** Failure implies that there is insufficient water in the reactor cavity for debris bed cooling.

3.1.3.1.3 LLOCA Containment Interface Tree

The LLOCA containment interface tree is presented as Figure 3.1.3-3. The LLOCA containment interface tree is similar to that for GENTRANS/RECIRC except that four top events are left out. For large LOCAs, it is not possible to have the RCS at greater than 200 psia at the onset of core damage. Also, the potential for a containment bypass is insignificant with the RCS pressure so low. The availability of steam generator cooling is also not significant because RCS pressure is too low to thermally induce steam generator tube failure during the core damage progression. Therefore, Top Events LOWPR, INTPR, MELTB, and SGCLG are not included in this tree. The top events that are included are described below. For convenient reference, Table 3.1.3-3 summarizes the top events that are modeled in the LLOCA containment interface tree.

- **Top Event MELT — No Core Melt during Large LOCA**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage. Core damage is assumed guaranteed to occur for the fraction of large LOCAs of sufficient size to be termed excessive; i.e., large enough and positioned such that the safety systems cannot keep the core covered. If the sequence leads to core damage during the injection phase, or if recirculation from the containment sump fails, then this top event is guaranteed failed.

    Injection could fail due to insufficient accumulator injection or failure of both RHR pumps to provide cold leg injection. To fail recirculation requires (1) the failure of both RHR pump trains taking suction from the sump, (2) the containment sump plugging or otherwise not having water from the RWST, or (3) failure of the RWST level instrumentation to initiate automatic swapover to the sump. Otherwise, this top event is guaranteed successful.

- **Conditions when Demanded.** This top event is asked for all sequences that exit the LLOCA frontline event tree model and enter the containment interface tree.

- **Scenario Impact if Successful.** Success implies that the Watts Bar plant did not sustain core damage during the accident sequence. The sequence is then mapped to a success end state.

- **Scenario Impact if Failed.** Failure implies that core damage did occur. Additional questions must then be asked to determine the conditions imposed on the containment by the core damage sequence.

- **Top Event MELTI — Melt with Successful Containment Isolation**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with an isolated containment. For the LLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For successful containment isolation, both of these top events must be successful.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences that enter the containment interface tree.

  - **Scenario Impact if Successful.** Success implies that there is a core damage sequence and the containment is not isolated. Either a large or small hole is present. The size of the hole must then be determined via Top Events MELTS and MELTL.

  - **Scenario Impact if Failed.** Failure implies that the containment is isolated.

- **Top Event MELTS — Melt with Large Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the LLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For only a small penetration isolation failure, Top Event CP must be successful, and Top Event CI must be failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree, and involves failure of containment isolation.

  - **Scenario Impact if Successful.** Success implies that the containment penetration that failed to isolate must be large.

  - **Scenario Impact if Failed.** Failure implies that there is just a small penetration line that failed to isolate.

- **Top Event MELTL — Melt with Small Penetration Isolation Failure**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage with a small or large containment penetration not isolated. For the LLOCA event tree, containment isolation failures are modeled by Top Events CI and CP. For a large penetration isolation failure, Top Event CP must be failed. Top Event CI may be successful or failed. Top Event CP models the larger purge lines, and Top Event CI models the isolation of all of the smaller lines.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for core damage sequences involving a failure of containment isolation (i.e., Top Event MELTI is successful) but not involving a small penetration isolation failure; i.e., Top Event MELTS is successful.

  - **Scenario Impact if Successful.** Success implies that the containment isolation failure involved only a small containment penetration.

  - **Scenario Impact if Failed.** Failure implies that the containment isolation failure involved a large size penetration.

- **Top Event CSI — Containment Spray Injection**

  - **Function Evaluated.** This top event tracks the status of containment spray injection. The spray pump trains are modeled via Top Events CSA and CSB. Success of either Top Event CSA or CSB constitutes successful spray injection.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

  - **Scenario Impact if Successful.** Success implies that at least one of the containment spray pumps operate in the injection mode in response to the high containment pressures experienced in a medium LOCA sequence.

  - **Scenario Impact if Failed.** Failure implies that containment spray is not available in the injection mode for containment pressure suppression.

- **Top Event CSR — Containment Spray Recirculation**

  - **Function Evaluated.** This top event tracks the availability of containment spray in the recirculation mode. Success requires that at least one of the containment spray pumps operates (i.e., Top Event CSA or CSB, and therefore Top Event CSI must be successful), that the sump suction valves for the operable pump train be aligned for sump recirculation, and that the operators align cooling water to the associated containment spray heat exchanger.

  - **Conditions when Demanded.** This top event is asked in the containment interface tree for all core damage sequences.

- **Scenario Impact if Successful.** Success implies that containment spray is available in the recirculation mode.

- **Scenario Impact if Failed.** Failure implies that containment spray is not available for recirculation.

- **Top Event RHRS — RHR Spray in Recirculation**

  - **Function Evaluated.** This top event tracks the availability of containment spray via the RHR pumps, operating in the recirculation mode. Success requires that at least one RHR pump train operate, with its associated sump suction valve aligned, that swapover for recirculation from the sump be completed including the alignment of CCS to the RHR heat exchangers, and that the operators manually align for RHR spray.

  - **Conditions when Demanded.** This top event is asked for all sequences in the containment interface tree.

  - **Scenario Impact if Successful.** Success implies that RHR spray is available for containment heat removal in the recirculation mode.

  - **Scenario Impact if Failed.** Failure implies that RHR spray is not available for containment heat removal in the recirculation mode.

- **Top Event CAV — Water in Reactor Cavity**

  - **Function Evaluated.** This top event tracks the availability of water in the reactor cavity for long-term debris bed cooling after core damage. Success requires both the melted ice from the ice condenser and the RWST contents. Melted ice from the ice condenser is available as long as the ice condenser doors open as required; i.e., success of Top Event IC. The RWST water must be injected to containment by one of the safety injection pumps, the RHR pumps, or the containment spray pumps. Conservatively, no credit is given for injection by the centrifugal charging pumps because these are not modeled in the LLOCA frontline event tree.

    Injection of the RWST inventory into the RCS following penetration of the fuel debris through the RCS (i.e., most likely through the reactor vessel bottom head) is counted as a successful injection of the inventory into containment since that is where it ends up. The upper to lower compartment drain lines must also not be plugged so that containment spray flow does not transfer the RWST inventory to the upper containment compartment, making it unavailable to the lower compartment.

  - **Conditions when Demanded.** This top event is asked for all core damage sequences in the containment interface tree.

  - **Scenario Impact if Successful.** Success implies that sufficient water is in the reactor cavity for debris bed cooling.

- **Scenario Impact if Failed.** Failure implies that there is insufficient water in the reactor cavity for debris bed cooling.

3.1.3.1.4 VIBIN Containment Interface Tree

The VIBIN containment interface is presented in Figure 3.1.3-4. The top events are described below. For convenient reference, Table 3.1.3-4 summarizes the top events that are modeled in the VIBIN containment interface tree.

- **Top Event MELT — No Core Melt**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage. If the sequence leads to core damage during the injection phase, or if there is a need for recirculation from the containment sump and recirculation fails, then this top event is guaranteed failed. To fail recirculation requires (1) the failure of both residual heat removal (RHR) pump trains taking suction from the sump (2) the containment sump plugging or otherwise not having water from the refueling water storage tank (RWST), (3) failure of the RWST level instrumentation to initiate automatic swapover to the sump, or (4) failure to align for high pressure recirculation. Otherwise, this top event is guaranteed successful.

  - **Conditions when Demanded.** This top event is asked for all sequences that exit the frontline event tree models and enter the containment interface tree.

  - **Scenario Impact if Successful.** Success implies that the Watts Bar plant did not sustain core damage during the accident sequence. The sequence is then mapped to a success end state.

  - **Scenario Impact if Failed.** Failure implies that core damage did occur. Additional questions must then be asked to determine the conditions imposed on the containment by the core damage sequence.

- **Top Event CDB — Containment Bypass — V Sequence**

  - **Function Evaluated.** Given an interfacing system LOCA initiating event, this top event determines whether the operators recognize the event, and either isolate the LOCA or preserve the RWST level before it reaches 8%, and then cooldown.

  - **Success Criteria.** The success criterion for this top event is that either (1) the operators recognize the interfacing LOCA and isolate it, or (2) that the operators enter ECA-1.2 or ECA-1.1 before the RWST reaches 8% level. In the latter case, the LOCA does not have to be recognized, but the unavailability of the RHR pumps must be realized and the RWST level must be preserved through makeup. Cooldown to cold shutdown must continue in the either case.

  - **Model Boundaries.** The top event boundaries are the operator actions necessary to diagnose and respond to an interfacing LOCA.

- **Condition when Demanded.** This top event is questioned for all sequences in the VIBIN event tree where Top Event MELT is failed to determine the operator response to the interfacing system LOCA.

- **Scenario Impact if Successful.** If the event diagnosis is correct the operators will isolate LOCA outside containment. Cooldown will progress using the RWST and auxiliary feedwater. If the diagnosis is incorrect, the operators may still succeed by maintaining RWST level. The end result is that cooldown and depressurization are still possible.

- **Scenario Impact if Failed.** Incorrect diagnosis will have the LOCA outside the containment unisolated and the RWST low level will be reached. Core damage will then result with containment bypass assumed.

- **Top Event AR — Containment Air Return Fans**

  - **Function Evaluated.** Performance of the containment air return fans.

  - **Success Criteria.** The success criterion for the air return fans is that one of the two fans function. The fans are automatically actuated on a high-high containment pressure signal from the engineered safety features actuation system (ESFAS).

  - **Model Boundaries.** This top event models the containment air return fans. The air return fans circulate hot saturated air from the upper compartment after a LOCA (10 minutes after high-high containment pressure). The air return fans enhance heat removal from the lower compartment to the ice condenser to help lower containment pressure. All portions of the air return fan functions are modeled. A manual start action to back up the ESFAS actuation is not modeled.

  - **Conditions when Demanded.** Top Event AR is asked for each sequence in the VIBIN event tree in which Top Event MELT has failed and Top Event CDB is successful. These are sequences in which the air return fans can be of use in mitigating the containment pressure rise.

  - **Scenario Impact if Successful.** The lower compartment containment pressure rise is mitigated by the successful operation of the air return fans.

  - **Scenario Impact if Failed.** The lower compartment containment pressure is not mitigated, and local hydrogen pockets may develop due to poor mixing.

- **Top Event CI — Containment Isolation**

  - **Function Evaluated.** Isolation of small containment penetrations.

  - **Success Criteria.** Each of the small containment penetrations listed below must be either closed at the time of the accident and remain closed, or close by a signal from ESFAS based on a safety injection condition or on Phase B isolation. For station blackout sequences, the time assumed available for locally isolating the seal return line is 3 hours.

— **Model Boundaries.** This top event models containment isolation of nonessential penetrations during accident conditions. The containment penetrations explicitly modeled are as follows:

- Containment major vents and drains.
- Connections to the RCS.
- Connections to containment atmosphere, with the exception of the large penetrations modeled in Top Event CP.

This containment isolation top event models only those containment penetrations whose failure to isolate would result in a release path that would bypass containment. The following questions were asked about each penetration to determine the need for inclusion. Only those penetrations not covered by other system analyses in the probabilistic risk assessment (PRA) were considered.

- Does the penetration communicate directly with the outside environment?
- Does the penetration communicate with the environment via a low pressure system or a tank with a relief valve?
- Will the relief valve lift at a pressure below the ultimate containment pressure?
- Is the system or tank design pressure below the ultimate containment pressure?

Based on these questions, the following penetrations are included in this top event:

- Floor Sump Pump Discharge (X-41)
- RC Drain Tank and Pressurizer Vent to VH (X-45)
- RC Drain Tank Pump Discharge (X-46)
- Lower Compartment Pressure Relief (X-80)
- RC Drain Tank to Gas Analyzer (X-81)
- Upper Compartment Air Monitor Intake (X-94A/B)
- Upper Compartment Air Monitor Return (X-94C)
- Lower Compartment Air Monitor Intake (X-95A/B)
- Lower Compartment Air Monitor Return (X-95C)
- RCP Seal Return Line

All of these penetrations receive a signal to isolate given a safety injection condition, except for the reactor coolant pump (RCP) seal return line. The RCP seal return line is automatically signaled to close on high-high containment pressure; i.e., Phase B isolation. During station blackout conditions, the operator is required to locally isolate the motor-operated seal injection and return valves in the RCP seal return line. This action (i.e. action CI1) is included in the model.

The probability of small preexisting leaks is included in this top event model.

- **Conditions when Demanded.** Top Event CI is asked for every sequence in the VIBIN event tree in which Top Event MELT has failed and Top Event CDB is successful. The status of containment isolation is needed for sequences leading to core damage in which the containment is not already bypassed.

- **Scenario Impact if Successful.** The smaller containment penetrations modeled in Top Event CI are isolated.

- **Scenario Impact if Failed.** One or more of the smaller containment penetrations listed above must have been opened initially and failed to close.

  Failure of this top event combined with failure of the RCP seals is modeled as a bypass path.

- **Top Event CP — Containment Purge Isolation**

  - **Function Evaluated.** Isolation of the containment purge lines.

  - **Success Criteria.** Success of this top event requires that either (a) the purge system was not in use when required, or (b) it was in use and that at least one valve in each penetration line closed.

  - **Model Boundaries.** This top event models the isolation of the containment purge penetrations that are allowed to be opened during power operation. The plant Technical Specifications allow these penetrations to be opened up to 1,000 hours with the plant at power. The penetrations modeled are as follows:

    - Lower Compartment Purge Air Exhaust (X-4)
    - Instrument Room Purge Air Exhaust (X-5)
    - Upper Compartment Purge Air Exhaust (X-6)
    - Upper Compartment Purge Air Exhaust (X-7)
    - Upper Compartment Purge Air Supply (X-9A)
    - Upper Compartment Purge Air Supply (X-9B)
    - Lower Compartment Purge Air Supply (X-10A)
    - Lower Compartment Purge Air Supply (X-10B)
    - Instrument Room Purge Air Supply (X-11)

    The penetrations modeled by CP are treated separately from those in Top Event CI due to the larger size of the purge penetrations. Even though the large size of the purge penetrations may limit the containment pressure rise during a LOCA, the Phase A (i.e., 1.5 psid) and Phase B (i.e., 2.8 psid) isolation signal setpoints are low enough that these isolation signals should occur even if the purge lines are initially open.

A backup manual action to isolate these penetrations is also considered in accordance with the status of Top Event OS.

The probability of large preexisting leaks is included in this top event model.

- **Conditions when Demanded.** Top Event CP is asked in all sequences of the VIBIN event tree in which Top Event MELT is failed and Top Event CDB is successful. The status of the purge lines is necessary only for sequences leading to core damage and an interfacing LOCA not present. Even if the automatic and manual isolation signals fail, this event may still be successful because the penetrations included in the model are normally closed anyway.

- **Scenario Impact if Successful.** Success of Top Event CP implies that the containment has at most a small hole in it. If Top Event CI is also successful, then the containment is isolated.

- **Scenario Impact if Failed.** Failure of this top event implies that containment isolation has failed and that a large hole in the containment boundary is present.

- Top Event HH — Hydrogen Igniters

  - **Function Evaluated.** Hydrogen control using the hydrogen igniters.

  - **Success Criteria.** Success of this top event implies that all 34 igniters in 1 of 2 trains functioned.

  - **Model Boundaries.** This top event models the hydrogen igniters of the hydrogen mitigation system. During events that involve fuel cladding damage, the hydrogen igniters are used to burn away the hydrogen before it reaches combustible concentrations when it mixes with the containment atmosphere.

    The system consists of two trains of hydrogen igniters and the associated control circuitry. The system is manually initiated from the control room upon receipt of a Phase B signal, and hydrogen concentration, as indicated by the hydrogen analyzer, is between 0.5% and 6%. The action modeled is designated HH1. There is no time-related pressure to complete this action; i.e., many hours are assumed available. The operator is also required to place the hydrogen analyzer in service. Use of the hydrogen analyzer is included as part of the operator action to initiate the hydrogen igniters. During recovery from an initial station blackout, the hydrogen igniters are not to be placed in service if the hydrogen analyzers indicate that the hydrogen concentration exceeds 6%.

  - **Conditions when Demanded.** The hydrogen igniters are asked for in all sequences of the VIBIN event tree in which Top Event MELT is failed and

Top Event CDB is successful. The status of Top Event HH is used in the evaluation of containment performance only for sequences leading to core damage without an interfacing LOCA present.

- **Scenario Impact if Successful.** The hydrogen igniters are available to continuously burn off the hydrogen that collects in the containment, prior to the concentration of hydrogen reaching combustible concentrations.

- **Scenario Impact if Failed.** The hydrogen igniters are not available to reduce the concentration of hydrogen within containment.

3.1.3.1.5 VSBIN Containment Interface Tree

The VSBIN containment interface is presented in Figure 3.1.3-5. The top events are described below. For convenient reference, Table 3.1.3-5 summarizes the top events that are modeled in the VSBIN containment interface tree.

- **Top Event MELT — No Core Melt**

  - **Function Evaluated.** This top event tracks whether the sequence results in core damage. If the sequence leads to core damage during the injection phase, or if there is a need for recirculation from the containment sump and recirculation fails, then this top event is guaranteed failed. To fail recirculation requires (1) the failure of both RHR pump trains taking suction from the sump (2) the containment sump plugging or otherwise not having water from the RWST, (3) failure of the RWST level instrumentation to initiate automatic swapover to the sump, or (4) failure to align for high pressure recirculation. Otherwise, this top event is guaranteed successful.

  - **Conditions when Demanded.** This top event is asked for all sequences that exit the frontline event tree models and enter the containment interface tree.

  - **Scenario Impact if Successful.** Success implies that the Watts Bar plant did not sustain core damage during the accident sequence. The sequence is then mapped to a success end state.

  - **Scenario Impact if Failed.** Failure implies that core damage did occur. Additional question must then be asked to determine the conditions imposed on the containment by the core damage sequence.

- **Top Event AR — Containment Air Return Fans**

  - **Function Evaluated.** Performance of the containment air return fans.

  - **Success Criteria.** The success criterion for the air return fans is that one of the two fans function. The fans are automatically actuated on a high-high containment pressure signal from the ESFAS.

  - **Model Boundaries.** This top event models the containment air return fans. The air return fans circulate hot saturated air from the upper compartment after a LOCA (10 minutes after high-high containment pressure). The air return fans enhance heat removal from the lower compartment to the ice condenser to help lower containment pressure. All portions of the air return fan functions are modeled. A manual start action to back up the ESFAS actuation is not modeled.

- **Conditions when Demanded.** Top Event AR is asked for each sequence in the VSBIN event tree in which Top Event MELT has failed. Only sequences leading to core damage need to have the status of the containment return fans known.

- **Scenario Impact if Successful.** The lower compartment containment pressure rise is mitigated by the successful operation of the air return fans.

- **Scenario Impact if Failed.** The lower compartment containment pressure is not mitigated, and local hydrogen pockets may develop due to poor mixing.

- **Top Event CI — Containment Isolation**

  - **Function Evaluated.** Isolation of small containment penetrations.

  - **Success Criteria.** Each of the small containment penetrations listed below must be either closed at the time of the accident and remain closed, or close by a signal from ESFAS based on a safety injection condition or on Phase B isolation. For station blackout sequences, the time assumed available for locally isolating the seal return line is 3 hours.

  - **Model Boundaries.** This top event models containment isolation of nonessential penetrations during accident conditions. The containment penetrations explicitly modeled are as follows:

    - Containment major vents and drains.

    - Connections to the RCS.

    - Connections to containment atmosphere, with the exception of the large penetrations modeled in Top Event CP.

    This containment isolation top event models only those containment penetrations whose failure to isolate would result in a release path that would bypass containment. The following questions were asked about each penetration to determine the need for inclusion. Only those penetrations not covered by other system analyses in the PRA were considered.

    - Does the penetration communicate directly with the outside environment?

    - Does the penetration communicate with the environment via a low pressure system or a tank with a relief valve?

- Will the relief valve lift at a pressure below the ultimate containment pressure?

- Is the system or tank design pressure below the ultimate containment pressure?

Based on these questions, the following penetrations are included in this top event:

- Floor Sump Pump Discharge (X-41)
- RC Drain Tank and Pressurizer Vent to VH (X-45)
- RC Drain Tank Pump Discharge (X-46)
- Lower Compartment Pressure Relief (X-80)
- RC Drain Tank to Gas Analyzer (X-81)
- Upper Compartment Air Monitor Intake (X-94A/B)
- Upper Compartment Air Monitor Return (X-94C)
- Lower Compartment Air Monitor Intake (X-95A/B)
- Lower Compartment Air Monitor Return (X-95C)
- RCP Seal Return Line

All of these penetrations receive a signal to isolate, given a safety injection condition, except for the RCP seal return line. The RCP seal return line is automatically signaled to close on high-high containment pressure; i.e., Phase B isolation. During station blackout conditions, the operator is required to locally isolate the motor-operated seal injection and return valves in the RCP seal return line. This action (i.e. action CI1) is included in the model.

— **Conditions when Demanded.** Top Event CI is asked for every sequence in the VSBIN event tree that leads to core damage; i.e., Top Event MELT is failed. The status of containment isolation is needed for the containment analysis.

— **Scenario Impact if Successful.** The smaller containment penetrations modeled in Top Event CI are isolated.

— **Scenario Impact if Failed.** One or more of the smaller containment penetrations listed above must have been opened initially and failed to close.

- **Top Event CP — Containment Purge Isolation**

— **Function Evaluated.** Isolation of the containment purge lines.

— **Success Criteria.** Success of this top event requires that either (a) the purge system was not in use when required, or (b) it was in use and that at least one valve in each penetration line closed.

— **Model Boundaries.** This top event models the isolation of the containment purge penetrations that are allowed to be opened during power operation. The plant Technical Specifications allow these penetrations to be opened up to 1,000 hours a fuel cycle with the plant at power. The penetrations modeled are as follows:

- Lower Compartment Purge Air Exhaust (X-4)
- Instrument Room Purge Air Exhaust (X-5)
- Upper Compartment Purge Air Exhaust (X-6)
- Upper Compartment Purge Air Exhaust (X-7)
- Upper Compartment Purge Air Supply (X-9A)
- Upper Compartment Purge Air Supply (X-9B)
- Lower Compartment Purge Air Supply (X-10A)
- Lower Compartment Purge Air Supply (X-10B)
- Instrument Room Purge Air Supply (X-11)

The penetrations modeled by CP are treated separately from those in Top Event CI due to the larger size of the purge penetrations. Even though the large size of the purge penetrations may limit the containment pressure rise during a LOCA, the Phase A (i.e., 1.5 psid) and Phase B (i.e., 2.8 psid) isolation signal setpoints are low enough that these isolation signals should occur even if the purge lines are initially open.

A backup manual action to isolate these penetrations is also considered in accordance with the status of Top Event OS.

- **Conditions when Demanded.** Top Event CP is asked in all sequences of the VSBIN event tree that lead to core damage, i.e., Top Event MELT is failed. Even if the automatic and manual isolation signals fail, this event may still be successful because the penetrations included in the model are normally closed anyway.

- **Scenario Impact if Successful.** Success of Top Event CP implies that the containment has at most a small hole in it. If Top Event CI is also successful, then the containment is isolated.

- **Scenario Impact if Failed.** Failure of this top event implies that containment isolation has failed and that a large hole in the containment boundary is present.

- **Top Event HH — Hydrogen Igniters**

  - **Function Evaluated.** Hydrogen control using the hydrogen igniters.

  - **Success Criteria.** Success of this top event implies that all 34 igniters in 1 of 2 trains functioned.

  - **Model Boundaries.** This top event models the hydrogen igniters of the hydrogen mitigation system. During events that involve fuel cladding damage, the hydrogen igniters are used to burn away the hydrogen before it reaches combustible concentrations when it mixes with the containment atmosphere.

The system consists of two trains of hydrogen igniters and the associated control circuitry. The system is manually initiated from the control room upon receipt of a Phase B signal, and hydrogen concentration, as indicated by the hydrogen analyzer, is between 0.5% and 6%. The action modeled is designated HH1. There is no time-related pressure to complete this action; i.e., many hours are assumed available. The operator is also required to place the hydrogen analyzer in service. Use of the hydrogen analyzer is included as part of the operator action to initiate the hydrogen igniters. During recovery from an initial station blackout, the hydrogen igniters are not to be placed in service if the hydrogen analyzers indicate that the hydrogen concentration exceeds 6%.

- **Conditions when Demanded.** The hydrogen igniters are asked for in all sequences of the VSBIN event tree in which Top Event MELT has failed. The status of Top Event HH is used in the evaluation of containment performance.

- **Scenario Impact if Successful.** The hydrogen igniters are available to continuously burn off the hydrogen that collects in the containment, prior to the concentration of hydrogen reaching combustible concentrations.

- **Scenario Impact if Failed.** The hydrogen igniters are not available to reduce the concentration of hydrogen within containment.

3.1.3.1.6 Assignment of Containment Interface Tree Sequences to Plant Damage States

The sequences through the containment interface event trees are each assigned to one of the plant damage state (PDS) matrix entries in Figure 4.3-1 of Section 4.3. The status of top events in the containment interface trees permits a one-to-one correspondence between the PDS entries and the sequences.

Since all of the top events in the containment interface trees are just switches, each sequence from the frontline event tree models follows only one path through the associated containment interface tree. Each path in the interface trees is then assigned to one end state. The containment interface trees link to the end of the Level 1 frontline event trees so that each sequence through the frontline trees, which must then pass through the associated containment interface tree, is also assigned to one of the end states developed in Section 4.3 and presented as Figure 4.3-1. The Level 1 PDSs are also discussed in Section 3.1.5.

Not included in the containment interface trees is the status of ice condenser availability and of hydrogen control. The Level 1 probabilistic risk assessment (PRA) team decided that the number of PDSs that would be generated by discriminating the status of these additional features would be excessive. Therefore, the status of ice condenser availability and of hydrogen control is not distinguished by the containment interface trees. However, the status of both of these items can be determined by examining the top event failures along each individual sequence assigned to the current plant damage states. Sequences involving failure of the ice condenser have Top Event IC failed. Hydrogen control requires the availability of the air return fans (i.e., as represented by Top Event AR) and of the hydrogen ignitors, which is determined by the status of Top Event HH in the Level 1 models.

Consider a core damage sequence initiated by a medium LOCA. For this example, high and low pressure injection are successful, but core damage occurs because of a failure to swapover to recirculation when the sump suction valves both fail to open. The containment isolates successfully. The path followed for this accident sequence through the MLOCA containment interface tree involves:

1. A melt (Top Event MELT=F).

2. RCS pressure is intermediate at the onset of core damage because the high pressure injection pumps function (Top Event LOWPR=S).

3. The containment is isolated. (Top Event MELTI=F and Top Events MELTS and MELTL are not asked.)

4. Containment spray injection is successful (Top Event CSI=S).

5. Containment spray recirculation is unavailable due to the failure of the sump valves to open (Top Event CSR=F).

6. RHR spray is unavailable, also due to the failure of the sump valves to open (Top Event RHRS=F).

7. Water is injected to the cavity via the safety injection and charging pumps (Top Event CAV=S).

By comparison with Figure 4.3-1, the PDS code for this example sequence then becomes EYYGI, in which it has been assumed that sequences with RCS pressures greater than 200 psia at the onset of core damage but not greater than 2,000 psia are assigned to the E pressure state. In other words, all sequences with RCS pressure between 200 and 2,000 psia were conservatively assigned to the 400- to 2,000-psia state; i.e., state E. For purposes of the Level 1 sequence quantification, the nomenclature for plant damage matrix rows was relabeled. The rows in the plant damage matrix were labeled alphabetically so that the plant damage state EYYGI then becomes FGI; i.e., the characters tracking the status of containment isolation and spray remain the same as those identified in Figure 4.3-1.

When evaluating the containment response to this example sequence, the Level 2 analysts would also need to know the status of the ice condenser and of hydrogen control. If this PDS were to have significant frequency, then the Level 2 analysts would review the individual sequences assigned to this PDS and determine what the state of the ice condenser, air return fans, and hydrogen ignitors is to be assumed for the plant damage state.

3.1.3.2 Recovery Event Tree

Several operator recovery actions are considered as separate top events within the frontline and support system event trees. These recovery actions are discussed at the appropriate place along with the corresponding top events in the event tree descriptions. The specific actions quantified are presented in the human reliability analysis; i.e., Section 3.3.3.

Recovery actions that are used only for selected core damage sequences are accounted for in a separate event tree; i.e., the recovery event tree. This tree is linked to the end of the frontline event trees but prior to the containment interface trees. The recovery event tree has just a single top event; i.e., Top Event RE.

- **Top Event REC — Recovery Action**
  - **Function Evaluated.** The specific recovery action is dependent on the accident sequence of interest. Currently, the recovery actions considered in the recovery tree are for electric power recovery, particularly for station blackout sequences. Consideration is given to recovery of both offsite power and the onsite emergency diesel generators. The electric power recovery analysis is described along with the other human actions in Section 3.3.8. Separate recovery fractions are computed for each accident sequence considered.
  - **Conditions when Demanded.** This top event is asked for all sequences.
  - **Scenario Impact if Successful.** Success of Top Event RE implies that the accident sequence that would have resulted in core damage, if not for the recovery action, is instead mitigated before core damage occurs. Therefore, success of Top Event RE implies that the sequence is mapped to a successful end state.
  - **Scenario Impact if Failed.** Failure of Top Event RE implies that the recovery action was not successfully performed in time to prevent core damage. Such sequences, with failure of Top Event RE, are then assigned to the appropriate plant damage state.

3.1.3.3 Cross-Reference of Other Special Topics to Event Tree Models

Other topics sometimes require special event trees for accident sequence modeling. This section directs the reader to the appropriate sections of the report for a discussion of the topics below.

- **Interfacing Systems LOCAs.** A summary of the interfacing systems LOCAs analysis is provided in Section 3.3.9. A detailed discussion of interfacing systems LOCAs is then provided in Appendix E.

- **ATWS Sequences.** The plant response to initiating events followed by failure of the reactor to trip in some PRAs is modeled with a separate event tree. However, for this model, ATWS sequences are already included within the GENTRANS/RECIRC frontline event tree; i.e., Section 3.1.2.2.1. The plant responses with and without reactor trip are included within the same tree. The reader should review Section 3.1.2.2.1 for a discussion of the top events associated with ATWS sequences.

- **Internal Flooding Analysis.** The internal flooding event sequence models are described in detail in Appendix E. A summary of the flooding analysis is also provided in Section 3.3.8. Briefly, the GENTRANS/RECIRC frontline event trees were used for the analysis of internal floods; i.e., separate frontline event trees were not needed.

Table 3.1.3-1. Top Events Modeled in the GENTRANS/RECIRC Containment Interface Tree

Designator	Description
IE	Initiating Event
MELT	No Core Melt
LOWPR	RCS Pressure Not Low (> 200 psia)
INTPR	RCS Pressure > 2,000 psia
MELTB	Melt with Containment Bypassed
SGCLG	Steam Generator Cooling
MELTI	Melt without Containment Isolated
MELTS	Melt with Large Penetration Isolation Failure
MELTL	Melt with Small Penetration Isolation Failure
CSI	Containment Spray Injection
CSR	Containment Spray Recirculation
RHRS	RHR Spray Recirculation
CAV	Water in Reactor Cavity

Table 3.1.3-2. Top Events Modeled in the MLOCA Containment Interface Tree

Designator	Description
IE	Initiating Event
MELT	No Core Melt for Medium LOCA
LOWPR	RCS Pressure Not Low (> 200 psia)
MELTI	Melt without Containment Isolated
MELTS	Melt with Large Penetration Isolation Failure
MELTL	Melt with Small Penetration Isolation Failure
CSI	Containment Spray Injection
CSR	Containment Spray Recirculation
RHRS	RHR Spray Recirculation
CAV	Water in Reactor Cavity

Table 3.1.3-3. Top Events Modeled in the LLOCA Containment Interface Tree	
Designator	**Description**
IE	Initiating Event
MELT	No Core Melt during Large LOCA
MELTI	Melt with Successful Containment Isolation
MELTS	Melt with Large Penetration Isolation Failure
MELTL	Melt with Small Penetration Isolation Failure
CSI	Containment Spray Injection
CSR	Containment Spray Recirculation
RHRS	RHR Spray in Recirculation
CAV	Water in Reactor Cavity

Table 3.1.3-4. Top Events Modeled in the VIBIN Containment Interface Tree	
Designator	**Description**
IE	Initiating Event
MELT	No Core Melt
CDB	Containment Bypass - V Sequence
AR	Containment Air Return Fans
CI	Containment Isolation
CP	Containment Purge Isolation
HH	Hydrogen Igniters

Table 3.1.3-5. Top Events Modeled in the VSBIN Containment Interface Tree

Designator	Description
IE	Initiating Event
MELT	No Core Melt
AR	Containment Air Return Fans
CI	Containment Isolation
CP	Containment Purge Isolation
HH	Hydrogen Igniters

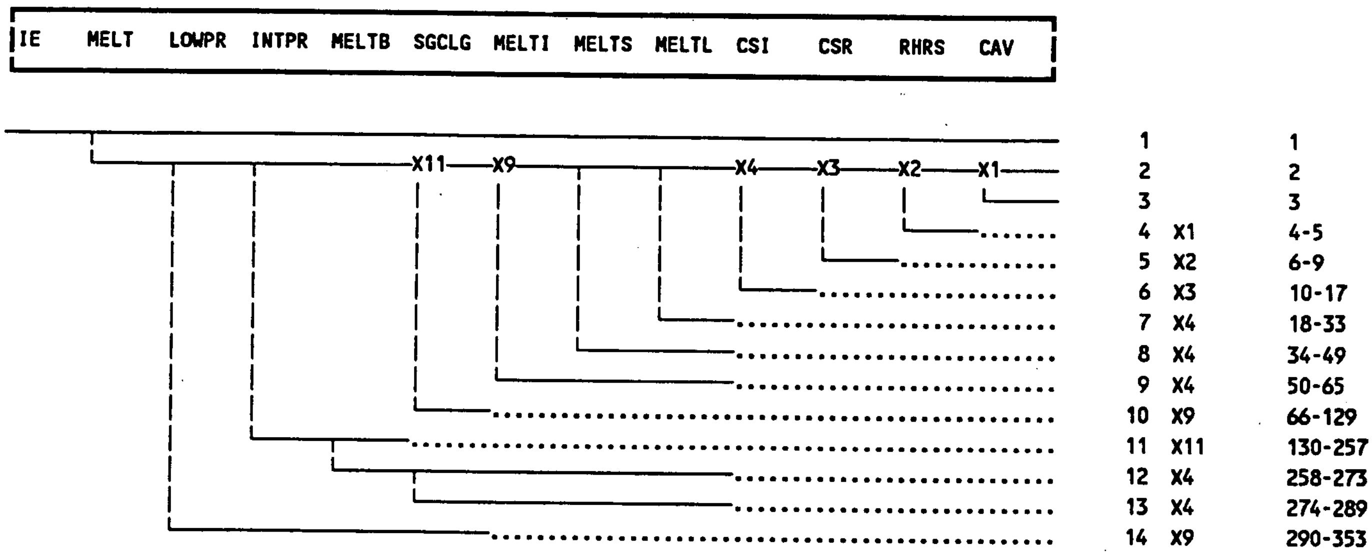

Figure 3.1.3-1. GENTRANS/RECIRC Containment Interface Tree

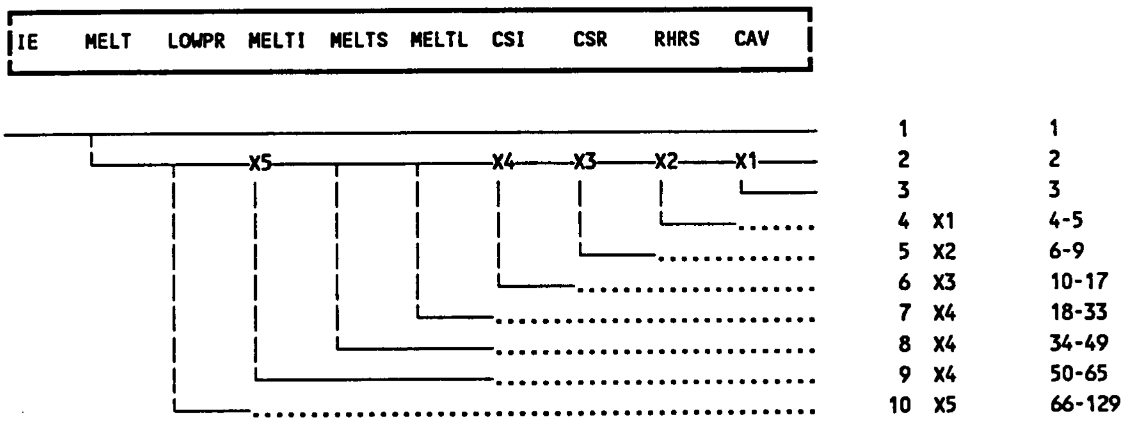

Figure 3.1.3-2. MLOCA Containment Interface Tree

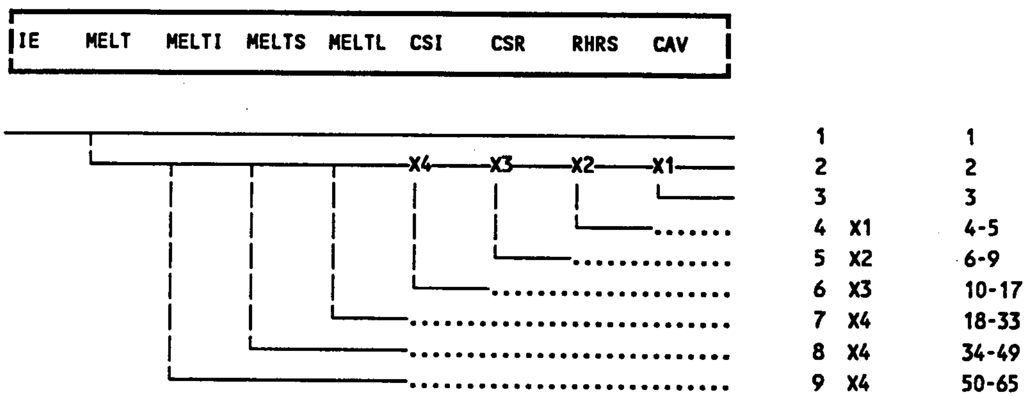

Figure 3.1.3-3. LLOCA Containment Interface Tree

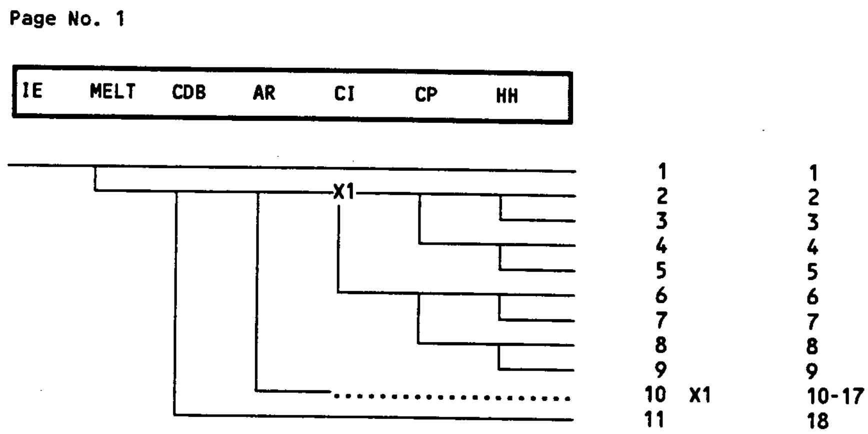

Figure 3.1.3-4. WBN VIBIN Containment Interface Tree

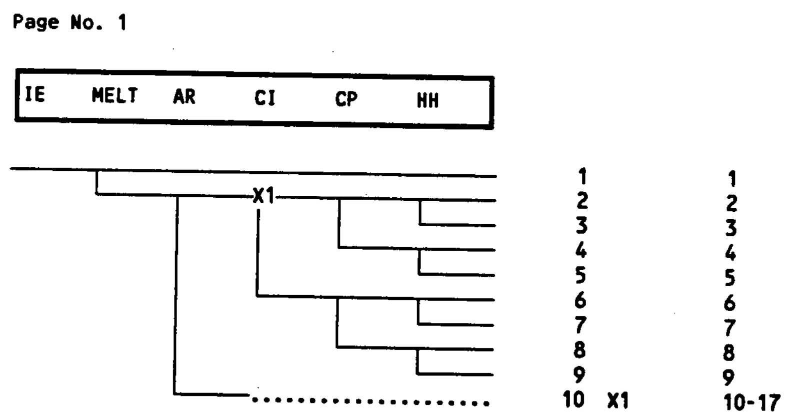

Figure 3.1.3-5. WBN VSBIN Containment Interface Tree

3.1.4 SUPPORT SYSTEM EVENT TREES

The status of support systems is included in the Watts Bar Nuclear Plant event sequence models via three support system event trees. The first two support system event trees include top events describing the status of the electric power systems at Units 1 and 2 (ELECT1 and ELECT2). A third support system event tree considers the mechanical support systems (MECH). In linking the event trees for a particular initiating event, these three event trees follow the initiating event, and precede the frontline event trees.

The top events in these three event trees are described in the following sections. The three support event trees are presented in Figures 3.1.4-1 through 3.1.4-3. Dependencies between the support system top events within a tree are accounted for by the split fraction assignment logic, rather than by the decision to branch or not branch dependent on the status of earlier top events. The order of the top events is important because they must be arranged so that the split fraction assignment logic for one top event can be expressed entirely in terms of the preceding top events. The top events included in each event tree are listed in Tables 3.1.4-1 through 3.1.4-3.

3.1.4.1 Support System Event Tree for Unit 1 Electrical Systems

The electrical support system event tree (ELECT1) is used to represent the various responses of the Watts Bar Unit 1 electrical systems. The top events for this event tree are summarized in Table 3.1.4-1 and represent the different Unit 1 sources of electrical power to the mechanical support and frontline systems.

The Unit 1 electrical support system top events were identified through the system analysis portion of the probabilistic risk assessment (PRA). Top events were classified as such because of the distinct support they provide, the dependencies shared with other electrical top events, and/or the dependencies of frontline systems on specific trains of the electrical support system. The intersystem dependency matrices found in Section 3.2.3 are a tabular representation of the intersystem dependency relationships.

The Unit 1 electrical support system top events are described below in roughly the same order as they appear in the ELECT1 event tree. The exception to this ordering is for top events representing redundant trains of the types of electrical equipment. Rather than repeat the descriptions for the second train, the presentation for both trains is combined into one discussion.

- **Top Event OG — Offsite Grid**
  - **Function Evaluated.** This top event models the supply of AC power from the 161-kV switchyard following a plant trip.
  - **Success Criteria.** Success requires that power from the switchyard to the common station service transformers (CSST) C and D remain energized for 24 hours after plant trip.
  - **Model Boundaries.** Only the unit boards are powered from the 500-kV grid via the unit station service transformers (USSTs) during normal power operation. Given a plant trip, the unit boards automatically transfer to the

common station service transformers (CSST) A and B, which are powered from the 161-kV grid. The shutdown boards are powered from CSSTs C and D during normal power operation and following a plant trip. Therefore, only the availability of the 161-kV lines from the 161-kV grid is modeled in this top event to supply power to the CSSTs. The equipment modeled includes the switchyard and CSSTs C and D. Failure of any one CSST to allow 161-kV power into the plant is conservatively assumed to fail Top Event OG. CSSTs A and B are modeled in the top events that model the unit boards; i.e., Top Events UB1A, UB1B, UB1C, and UB1D. CSSTs A and B were grouped with the unit boards because the unit boards are all that depend on these CSSTs.

- **Important Intersystem Dependencies.** Failure of this top event will necessitate operation of the emergency diesel generators modeled in Top Events GA, GB, GC, and GD.

  For a loss of offsite power initiating event, both the 161-kV and the 500-kV grids are assumed to have lost power; thus, this top event is assumed failed.

- **Top Event OGR1 — Recovery of Offsite Power in 1 Hour**

  - **Function Evaluated.** This top event models the recovery of offsite power.

  - **Success Criteria.** Success of this top event requires that, given an initial loss of power from both offsite grids, the operators restore power to the plant within 1 hour from offsite.

  - **Model Boundaries.** The analysis is restricted to the recovery of power from offsite, and only during the first hour after it is lost. The 1-hour time is the earliest possible that core damage may occur in the absence of a medium or large break loss of coolant accident (LOCA). Subsequent power recovery for later times is considered separately in the recovery tree; i.e., Section 3.1.3.3.

  - **Important Intersystem Dependencies.** This event is guaranteed successful if the offsite grid was never lost; i.e., if Top Event OG is successful.

- **Top Events FA and FB — Fuel Oil Transfer and Supply for Unit 1 Diesel Generators**

  - **Function Evaluated.** Each of these top events models the fuel oil transfer and supply system for its respective Unit 1 diesel generator.

  - **Success Criteria.** Success of this top event requires that one of two transfer pumps for each diesel generator is available to refill the day tanks.

  - **Model Boundaries.** Included in each top event is a fuel oil supply tank, two motor-driven fuel transfer pumps, two 550-gallon day tanks, and associated piping, valves, and instrumentation. The transfer pumps, powered from the

480V diesel auxiliary boards, function to renew each day tank's inventory from the main supply tank. The fuel oil transfer pumps must start and stop as needed to maintain day tank level.

- **Important Intersystem Dependencies.** Failure of a fuel oil transfer and supply system top event will result in the unavailability of the associated diesel generator. Top Events FA and FB are not necessary and therefore are assumed to be guaranteed successful if power is available from the offsite grid; i.e., if Top Event OG is successful. Power is assumed to be available to the transfer pumps for the purpose of quantifying the failure frequencies of these top events. This will be the case if the associated diesel generator is operating and requires fuel for continued operation.

• **Top Events GA and GB — Diesel Generators 1A-A and 1B-B**

- **Function Evaluated.** Each diesel generator top event questions the ability of that diesel generator to supply power to its respective 6.9-kV shutdown board following a loss of the offsite power grid.

- **Success Criteria.** For success, each diesel generator must start and run for 24 hours.

- **Model Boundaries.** Top Event GA supplies train A power to 6.9-kV shutdown board 1A-A, and Top Event GB supplies train B power to 6.9-kV shutdown board 1B-B. As an example, Top Event GA models the availability of diesel generator 1A-A and its auxiliaries; the generator sequencer; dedicated DC control power; electrical board room and diesel room ventilation; cooling water from essential raw cooling water (ERCW); and associated piping and valves. Success of the diesel generator top events depends on successful operation of all of the aforementioned equipment. Top Event GB models diesel generator 1B-B in a similar fashion.

- **Important Intersystem Dependencies.** The diesel generators require success of fuel oil transfer and supply to each diesel engine. If power is available from the offsite grid (i.e., Top Event OG is successful), then Top Events GA and GB are not required and therefore are assumed to be guaranteed successful. For purposes of quantifying the unavailability of these top events, ERCW cooling water and DC control power required for closing the output breakers are assumed to be available. If the later top events that track these support systems are found to fail, then the logic rules for the assignment of split fractions to the systems powered from these loads account for the dependence of the diesel generators on these support systems and will fail emergency power as necessary. This will become clear in the discussion of the mechanical support system event tree.

• **Top Events AA and BA — Unit 1 6.9-kV Shutdown Boards 1A-A and 1B-B**

- **Function Evaluated.** These top events model the availability of shutdown power from the respective 6.9-kV shutdown board during normal and transient conditions, including losses of offsite power.

- **Success Criteria.** Success of these events requires the 6.9-kV shutdown boards to provide power for 24 hours after a plant trip.

- **Model Boundaries.** The equipment included in these events is the respective 6.9-kV shutdown board and associated breakers.

  During normal operations, the 6.9-kV AC power subsystem receives power from the 161-kV grid via CSSTs C and D modeled via Top Events OG and OGR1. There is no need to transfer power on a plant trip. The diesel generators modeled in Top Events GA and GB receive a start signal from their associated 6.9-kV shutdown board when Top Event OG fails to supply offsite power.

- **Important Intersystem Dependencies.** Success of these top events depends on the availability of power from either the offsite grid (Top Events OG and OGR1) or the associated diesel generator, the availability of 125V control power to the board transfer breakers, and board room cooling modeled in Top Event V1. Power from the associated shutdown board is unavailable if the initiator involved loss of the shutdown board, or if power from both offsite and from the associated diesel generator is lost. Consideration is given to recovery of offsite power within the first hour via Top Event OGR1.

  The top events for DC control power and shutdown board room ventilation are asked later in the event tree. For evaluation of the 6.9-kV shutdown boards, they are assumed to be available. If, later in the tree, these support systems are found not to be available, then power from the associated 6.9-kV shutdown board is also assumed to be lost.

- **Top Events A1, A2, B1, and B2 — 480V Shutdown Boards 1A1-A, 1A2-A, 1B1-B, and 1B2-B**

  - **Function Evaluated.** These top events question the availability of power at the Unit 1 480V Class 1E shutdown boards for all conditions.

  - **Success Criteria.** Success of these top events requires that the respective 480V shutdown boards be energized for 24 hours after plant trip.

  - **Model Boundaries.** Each of these top events models the associated 480V shutdown board and 6.9-kV/480V transformers. Also included is the ability to supply power to the associated reactor motor-operated valve (RMOV) board, diesel auxiliary board, reactor vent board, and control and auxiliary (C&A) vent board through their buses and supply breakers. When a transient occurs, the 480V boards continue to receive power from their associated 6.9-kV shutdown board. The shed boards are all of the reactor ventilation boards and the control and auxiliary building ventilation boards 1A2-A, 1B2-B, 2A2-A, and 2B2-B.

- **Important Intersystem Dependencies.** Failure of one of these top events will result in the failure to supply power to the loads fed from the bus. The 480V shutdown boards are unavailable if the associated 6.9-kV bus (i.e., if Top Event AA or BA fails) is not available.

• **Top Events VT1A and VT1B — 480V Shutdown Transformer Rooms 1A and 1B Ventilation**

- **Function Evaluated.** These top events model the availability of ventilation to the Unit 1 480V shutdown transformer rooms 1A and 1B.

- **Success Criteria.** Success of these top events requires that at least one of four fans for room 1A and one of three fans for room 1B are available to provide room ventilation to the associated rooms for 24 hours after plant trip.

- **Model Boundaries.** Separate ventilation systems are available for each of the two shutdown transformer rooms. Four fans draw from room 1A. Three of these fans are powered from 480V C&A ventilation board 1A1-A; i.e., included with the model for Top Event A1. The fourth fan is powered from 480V common board A; i.e., Top Event A3.

  Three fans draw from room 1B. All three of these fans are supplied power by 480V C&A ventilation board 1B1-B; i.e., included with the model for Top Event B1.

- **Important Intersystem Dependencies.** Failure of these top events, unless recovered before temperatures become excessive, will result in failure of the Unit 1 480V shutdown boards. Loss of all power to the fans will fail the shutdown transformer room ventilation for the associated train. This means that the other 480V shutdown board on the same train (i.e., 1A2-A or 1B2-B modeled via Top Events A2 and B2, respectively) would then fail due to excessive room temperatures.

• **Top Events VT1AR and VT1BR — Recovery of Transformer Rooms 1A and 1B Ventilation**

- **Function Evaluated.** These top events model the recovery of room ventilation to the associated transformer rooms prior to overheating of the 6.9-kV to 480V shutdown transformers contained in the affected rooms.

- **Success Criteria.** The establishment of portable ventilation to the rooms affected by the failed ventilation system must be completed within 10 hours for room 1B, and 5 hours for room 1A.

- **Model Boundaries.** These top events consider the recovery actions necessary to restore room cooling to the affected transformer rooms, given that the redundant fans of the normally operating ventilation system fail. The human action identifier is HVT1AR. A single action is used for both rooms. Therefore, 5 hours, which is the shorter of the two recovery times, is used as the limiting time available for both rooms.

- **Important Intersystem Dependencies.** These top events are assumed to be guaranteed successful if the normally operating ventilation system functions normally, or if the transformers in the serviced rooms are deenergized (i.e., unavailable), and therefore the need for room cooling is obviated.

- **Top Events DA and DB — 125V DC Battery Boards I and II**

  - **Function Evaluated.** These top events model the availability of power at the 125V DC battery boards I and II.

  - **Success Criteria.** Success of these top events requires power at the associated battery boards for 24 hours after a plant trip assuming that the associated shutdown board is energized, and for 4 hours following loss of the supplying board. The associated battery charger must also operate to supply charging to the battery for the 24-hour mission time.

  - **Model Boundaries.** Each top event includes a 125V DC bus and the availability of 125V DC power from a battery and charger.

  - **Important Intersystem Dependencies.** Failure of one of these top events will result in the failure to supply DC power to the loads fed from the associated battery board. The 125V DC battery board I requires power from 480V shutdown board* 1A1-A (Top Event A1), and 125V DC battery board II requires power from 480V shutdown board 1B2-B (Top Event B2). The associated battery board is also failed if the initiating event involves failure of the battery board.

- **Top Event DG — 120V AC Instrument Power Board 1A**

  - **Function Evaluated.** This top event models the availability of Unit 1, 120V AC instrument power board 1A.

  - **Success Criteria.** Success requires the availability of power at 120V AC instrument power board 1A for 24 hours following an initiating event.

  - **Model Boundaries.** This top event models the availability of 120V AC from 120V AC bus 1A, proper operation of transformer 1A, and fused disconnect switch 1A.

  - **Important Intersystem Dependencies.** Success of this top event depends on the availability of power from 480V shutdown board* 1A1-A, modeled in Top Event A1.

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*After the production of this model, the power supply to this board was changed to 480V shutdown board 1A2-A.

- **Top Event V1 — Train A Shutdown Board Ventilation**

  - **Function Evaluated.** This top event models the availability of ventilation to the train A shutdown board rooms.

  - **Success Criteria.** Success of this top event requires that one of two air handling units (AHU) is available to provide room ventilation for 24 hours after plant trip. The chiller units associated with the AHUs are not required for success.

  - **Model Boundaries.** Ventilation for the Unit 1 and Unit 2 train A 6.9-kV shutdown boards and Unit 1 train A and train B 480V shutdown boards are supplied by air handling units A-A and C-B. For each AHU to operate properly, its associated breaker, fuses, and damper must also function effectively. The AHUs receive 480V shutdown power to start and run.

  - **Important Intersystem Dependencies.** Failure of this top event, unless recovered before temperatures become excessive, will result in failure of the Unit 1 and Unit 2 train A 6.9-kV shutdown boards and the Unit 1 480V shutdown boards. The AHUs require success of power from the associated 480V shutdown board 1A2-A or 1B2-B; i.e., Top Events A2 and B2, respectively.

  - The present model shows the Unit 1 shutdown boards dependent on Top Event V1. The actual dependency as described above was determined after the quantification of the model. Correcting the alignment in the model is not expected to significantly impact the results.

- **Top Event V1R — Recovery of Unit 1 Shutdown Board Room Ventilation**

  - **Function Evaluated.** This top event models the recovery of room ventilation to the 480V board shutdown boards prior to overheating of the 480V shutdown boards that the rooms serviced contain.

  - **Success Criteria.** The restoration of ventilation to the shutdown boards must be completed within 12 hours.

  - **Model Boundaries.** This top event considers the recovery actions necessary to restore room cooling to 480V shutdown board rooms, given that the redundant fans of the normally operating ventilation system fail; i.e., given Top Event V1 failed. The human action identifier is termed HV1R1.

  - **Important Intersystem Dependencies.** This top event is assumed to be guaranteed successful if the normally operating ventilation system functions normally.

- **Top Event VINV1 — Unit 1 480V Board Room B Ventilation**

  - **Function Evaluated.** This top event models the availability of ventilation to the Unit 1 480V board room B, which contains one-half of the unit inverters.

  - **Success Criteria.** Success of this top event requires that the fan associated with the AHU operate for 24 hours to provide cool outside air to the Unit 1 480V board room after plant trip. The air conditioning unit with the associated AHU is not required for success.

  - **Model Boundaries.** The model for this top event considers both AHUs of the 480V board room ventilation system. The two fans are supplied power by 480V shutdown board 1B1-B; i.e., as modeled in Top Event B1.

  - **Important Intersystem Dependencies.** Failure of this top event, unless recovered before temperatures become excessive, will result in failure of 120V vital instrument channels 1-I and 1-II when the inverters overheat. This ventilation system is failed if power from 480V shutdown board 1B1-B (i.e., Top Event B1) has failed.

- **Top Event VNV1R — Recovery of Unit 1 480V Board Room B Ventilation**

  - **Function Evaluated.** This top event models the recovery of room ventilation to the 480V board room B prior to overheating of the inverters it contains.

  - **Success Criteria.** The restoration of ventilation to room B must be completed within 6 hours.

  - **Model Boundaries.** This top event considers the recovery actions necessary to restore room cooling to 480V board room B, given the normally operating ventilation system fail; i.e., given that Top Event VINV1 failed. The human action is termed HVNVR1.

  - **Important Intersystem Dependencies.** This top event is assumed to be guaranteed successful if the normally operating ventilation system functions normally.

- **Top Events DAAC and DBAC — 120V AC Power Subsystems 1-I and 1-II**

  - **Function Evaluated.** These top events model the availability of power at the 120V AC vital distribution panels for channels 1-I and 1-II.

  - **Success Criteria.** Success of each of these top events requires that power is available at the associated 120V AC panel for 24 hours after plant trip if the associated 480V shutdown board is available, and for 4 hours if it is not.

  - **Model Boundaries.** Each top event includes one 125V DC to 120V AC invertor, a 120V AC bus, and associated breakers and fuses.

- **Important Intersystem Dependencies.** The 120V AC panels are supplied power from two sources each. Top Event DAAC (i.e., 120V AC instrument channel 1-I) is failed if both 125V DC battery board 1 (Top Event DA) and 480V shutdown board* 1A1-A (Top Event A1) are failed. If Top Event A1 alone fails, 120V AC instrument channel 1-I can still be powered for 4 hours from the 125V DC battery board 1. Top Event DAAC is also failed if Unit 1 480V board room B ventilation is lost (Top Event VINV1 fails) and is not recovered, or if channel I is lost as an initiator.

  Top Event DBAC (i.e., 120V AC instrument channel 1-II) is failed if both 125V DC battery board 2 (Top Event DB) and 480V shutdown board 1B2-B (Top Event B1) are failed. If Top Event B1 alone fails, 120V AC instrument channel 1-II can still be powered for 4 hours from the 125V DC battery board 2. Top Event DBAC is also failed if Unit 1 480V board room B ventilation is lost (Top Event VINV1 fails) and is not recovered, or if channel II is lost as an initiator.

- **Top Events A3 and B3 — 6.9-kV Common Boards A and B and 480V Auxiliary Building Common Board Buses A and B**

  - **Function Evaluated.** These top events model the availability of the 6.9-kV and 480V common board buses to supply power to their loads.

  - **Success Criteria.** Success of these top events requires the associated 6.9-kV common boards and 480V auxiliary building common boards to provide power for 24 hours following a plant trip.

  - **Model Boundaries.** The models for these top events include the 6.9-kV common board buses, the 480V auxiliary building common board buses, and the associated breakers and transformers.

  - **Important Intersystem Dependencies.** These top events both fail on a loss of offsite power; i.e., if Top Event OG fails. These boards are conservatively assumed to be unavailable, even if power is restored from the offsite grid within 1 hour.

- **Top Events D1 and D2 — 250V DC Boards 1 and 2**

  - **Function Evaluated.** These top events model the availability of the 250V DC power boards.

  - **Success Criteria.** Success of Top Events D1 and D2 require that 250V DC control power be available from the associated 250V DC boards for 24 hours following a plant trip and 4 hours after station blackout.

  - **Model Boundaries.** These top events model the availability of the respective 250V DC bus, 250V DC turbine building board breakers, 250V DC electric

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*After production of this model, the power supply to the board was changed to 480V shutdown Board 1A2-A.

control board distribution panel breakers, and proper operation of each 250V DC battery and charger. The chargers for boards 1 and 2 are dependent on 480V power from the auxiliary building common boards modeled in Top Events A3 and B3, respectively. The batteries will supply DC power for 4 hours, following a loss of all offsite power.

- **Important Intersystem Dependencies.** Control power for the 6.9-kV unit boards modeled in later top events UB1A, UB1B, UB1C, and UB1D is provided by the Unit 1 250V DC system. Unit boards 1A and 1C are normally provided from 250V DC board 1, and unit boards 1B and 1D are normally powered from 250V DC board 2.

- **Top Events UB1A, UB1B, UB1C, and UB1D — Unit 1 6.9-kV Unit Boards 1A, 1B, 1C, and 1D**

  - **Function Evaluated.** These events model the availability of power at the respective 6.9-kV unit boards during normal and transient conditions, including losses of offsite power.

  - **Success Criteria.** Success of these events requires the 6.9-kV unit boards provide power for 24 hours following a plant trip.

  - **Model Boundaries.** The equipment included in these events is the respective 6.9-kV unit board, CSSTs A and B, and associated transfer breakers.

    During normal operations, the 6.9-kV AC unit boards receive power from the main unit generator via the USSTs. When a unit trip occurs, the unit boards automatically transfer from the USSTs to CSSTs A and B where they receive power from the 161-kV grid, modeled via Top Events OG and OGR1.

  - **Important Intersystem Dependencies.** Success of these events depends on the availability of power from the offsite grid (Top Event OG or OGR1) and the availability of 250V control power to the board transfer breakers. All four 6.9-kV unit boards are failed if power from the offsite grid is not available. Boards 1A and 1C also require 250V DC control power from bus 1; i.e. modeled via Top Event D1. Boards 1B and 1D also require 250V DC control power from bus 2; i.e., modeled via Top Event D2.

3.1.4.2 Support System Event Tree for Unit 2 Electrical Systems

- **Top Events FC and FD — Fuel Oil Transfer and Supply for Unit 2 Diesel Generators**

  - **Function Evaluated.** Each of these top events models the fuel oil transfer and supply system for its respective Unit 2 diesel generators.

  - **Success Criteria.** Success of this top event requires that one of two transfer pumps for each diesel generator is available to refill the day tanks.

- **Model Boundaries.** Included in each top event is a fuel oil supply tank, two motor-driven fuel transfer pumps, two 550-gallon day tanks, and associated piping, valves, and instrumentation. The transfer pumps, powered from the 480V diesel auxiliary boards, function to renew the inventory of each day tank from the main supply tank. The fuel oil transfer pumps must start and stop as needed to maintain day tank level.

  The unavailability calculation for the Unit 2 fuel oil transfer pumps is somewhat more complicated than that for Unit 1. Since the Unit 2 pumps trains are asked second, their unavailability is conditional on the status of the Unit 1 pumps so that the potential for common cause failures between the four sets of pumps can be properly accounted for.

- **Important Intersystem Dependencies.** Failure of a fuel oil transfer and supply system top event will result in the unavailability of the associated diesel generator. These top events are not necessary and therefore are assumed to be guaranteed successful if power is available from the offsite grid; i.e., if Top Event OG is successful. Power is assumed to be available to the transfer pumps for the purposes of quantifying the failure frequencies of these top events. This will be the case if the associated diesel generator is operating and requires fuel for continued operation.

- **Top Events GC and GD — Diesel Generators 2A-A and 2B-B**

  - **Function Evaluated.** Each diesel generator top event questions the ability of that diesel generator to supply power to its respective 6.9-kV shutdown board following a loss of the offsite power grid modeled in Top Event OG.

  - **Success Criteria.** Each diesel generator must start and run for 24 hours.

  - **Model Boundaries.** Top Event GC supplies train A power, and Top Event GD supplies train B power. As an example, Top Event GC models the availability of diesel generator 2A-A and its auxiliaries, the generator sequencer, dedicated DC control power, electrical board room and diesel room ventilation, air start system, cooling water from ERCW, and associated piping and valves. Success of the diesel generator top events depends on successful operation of all of the aforementioned equipment. Top Event GD models diesel generator 2B-B in a similar fashion.

  - **Important Intersystem Dependencies.** The diesel generators require success of fuel oil transfer and supply to each diesel engine. If power is available from the offsite grid (i.e., Top Event OG is successful), then these top events are not required and therefore are assumed to be guaranteed successful. For purposes of quantifying the unavailability of these top events, ERCW cooling water and DC control power required for closing the output breakers are assumed to be available. If the later top events that track these supports are found to fail, then the logic rules for assigning split fractions for the systems powered by the diesel generators account for the dependence of the diesel generators on these support systems. This will become clear in the discussion of the mechanical systems support tree.

- **Top Events AB and BB — Unit 2 6.9-kV Shutdown Boards 2A-A and 2B-B**

  - **Function Evaluated.** These events model the availability of shutdown power from the respective 6.9-kV shutdown board during normal and transient conditions, including losses of offsite power.

  - **Success Criteria.** Success of these events requires the 6.9-kV shutdown boards to provide power for 24 hours after plant trip.

  - **Model Boundaries.** The equipment included in these events is the respective 6.9-kV shutdown board and associated breakers.

    During normal operations, the 6.9-kV AC power subsystem receives power from CSSTs C and D where they receive power from the offsite grid; modeled via Top Events OG and OGR1. The diesel generators modeled in Top Events GC and GD receive a start signal from their associated 6.9-kV shutdown board when Top Event OG fails to supply offsite power.

  - **Important Intersystem Dependencies.** Success of these events depends on the availability of power from either the offsite grid (Top Event OG or OGR1) or the associated diesel generator, the availability of 125V control power to the board transfer breakers, and board room cooling modeled in Top Event V2. Power from the associated shutdown board is unavailable if power from both offsite and the associated diesel generator are lost. Recovery of offsite power is considered via Top Event OGR1.

    The top events for DC control power and shutdown board room ventilation are asked later in the event tree. For evaluation of the 6.9-kV shutdown boards, they are assumed to be available. If, later in the tree, these support systems are found not to be available, then power from the associated 6.9-kV shutdown board is also assumed to be lost.

- **Top Events A1U2, A2U2, B1U2, and B2U2 — 480V Shutdown Boards 2A1-A, 2A2-A, 2B1-B, and 2B2-B**

  - **Function Evaluated.** These events question the availability of power at the Unit 2 480V Class 1E shutdown boards for all conditions.

  - **Success Criteria.** The 480V shutdown boards must all be available for 24 hours after plant trip.

  - **Model Boundaries.** Each of these top events models the associated 480V shutdown board and 6.9-kV/480V transformers. Also included is the ability to supply power to the associated RMOV board, diesel auxiliary board, reactor vent board, and C&A vent board through their buses and supply breakers. When a transient occurs, the 480V boards continue to receive power from their associated 6.9-kV shutdown board. For undervoltage conditions, selected boards are shed. The shed boards are all reactor ventilation and control and auxiliary building ventilation boards 1A2-A, 1B2-B, 2A2-A, and 2B2-B.

- **Important Intersystem Dependencies.** Failure of one of these 480V top events will result in the failure to supply power to the loads fed from the bus. The 480V shutdown boards are unavailable if the associated 6.9-kV bus is not available; i.e., if Top Event AB or BB fails. Success of a 480V shutdown board top event requires the availability of power from its associated 6.9-kV shutdown board, board room and transformer room ventilation, and proper functioning of the transformers, buses, and breakers mentioned above for the 6.9-kV shutdown boards.

• **Top Events VT2A and VT2B — 480V Shutdown Transformer Rooms 2A and 2B Ventilation**

- **Function Evaluated.** These events model the availability of ventilation to the Unit 2 480V shutdown transformer rooms 2A and 2B.

- **Success Criteria.** Success of these events requires that at least one of four fans for room 2B and one of three fans for room 2A are available to provide room ventilation to the associated rooms for 24 hours after plant trip.

- **Model Boundaries.** Separate ventilation systems are available for each of the two shutdown transformer rooms. Four fans draw from room 2B. Three of these fans are powered from 480V C&A ventilation board 2B1-B; i.e., included in the model for Top Event B1U2. The fourth fan is powered from 480V common board A; i.e., Top Event A3.

  Three fans draw from room 2A. All three of these fans are supplied power by 480V C&A ventilation board 2A1-A; i.e., included in the model for Top Event A1U2.

- **Important Intersystem Dependencies.** Failure of this event, unless recovered before temperatures become excessive, will result in failure of the Unit 2 480V shutdown boards. Loss of all 480V shutdown board power to the fans will fail the shutdown transformer room ventilation for the associated train. This means that the other 480V shutdown board on the same train (i.e., 2A2-A or 2B2-B modeled via Top Events A2U2 and B2U2, respectively) would then fail due to excessive room temperatures.

• **Top Events VT2AR and VT2BR — Recovery of Transformer Rooms 2A and 2B Ventilation**

- **Function Evaluated.** These top events model the recovery of room ventilation to the associated transformer rooms prior to overheating of the 6.9-kV to 480V shutdown transformers contained in the affected rooms.

- **Success Criteria.** The establishment of portable ventilation to the rooms affected by the failed ventilation system must be completed within 10 hours for room 2A, and 5 hours for room 2B.

- **Model Boundaries.** This top event considers the recovery actions necessary to restore room cooling to the affected transformer rooms, given that the redundant fans of the normally operating ventilation system fail. The human action identifier is HVT1AR.

- **Important Intersystem Dependencies.** These recovery actions are assumed to be guaranteed successful if the normally operating ventilation system functions normally, or if the transformers in the serviced rooms are deenergized (i.e., unavailable), and therefore the need for room cooling is obviated.

• **Top Events DC and DD — 125V DC Battery Boards III and IV**

- **Function Evaluated.** These top events model the availability of power at the 125V DC Battery Boards III and IV.

- **Success Criteria.** Success of these top events requires power at the associated battery boards for 24 hours after a plant trip assuming that the associated shutdown board is energized, and for 4 hours following loss of the supplying board. The associated battery charger must also operate to supply charging to the battery for the 24-hour mission time.

- **Model Boundaries.** Each top event includes a 125V DC bus and the availability of 125V DC power from a battery and charger.

- **Important Intersystem Dependencies.** Failure of one of these events will result in the failure to supply DC power to the loads fed from the associated battery board. The 125V DC battery board III requires power from 480V shutdown board* 2A1-A (i.e., Top Event A1U2), and 125V DC battery board IV requires power from 480V shutdown board 2B2-B; i.e., Top Event B2U2. The associated battery board is also failed if the initiating event involved failure of the battery board.

• **Top Event DH — 120V AC Instrument Power Board 2A**

- **Function Evaluated.** This event models the availability of Unit 2 120V AC power board 2A.

- **Success Criteria.** Success requires the availability of power at 120V AC instrument power board 2A to be energized for 24 hours following an initiating event.

*After the production of this model, the power supply to this board was changed to 480V shutdown board 2A2-A.

- **Model Boundaries.** This event models the availability of 120V AC from 120V AC instrument bus 2A, proper operation of transformer 2A, and fused disconnect switch 2A. Currently, this equipment is not modeled explicitly. It is included in the event trees for future modeling development.

- **Important Intersystem Dependencies.** Success of this event depends on the availability of power from 480V shutdown board* 2A1-A, modeled in Top Event A1U2.

- **Top Event V2 — Train B Shutdown Board Ventilation**

  - **Function Evaluated.** This event models the availability of ventilation to the train B shutdown board rooms.

  - **Success Criteria.** Success of this event requires that one of two AHUs is available to provide room ventilation for 24 hours after plant trip. The associated chiller units are not required for success.

  - **Model Boundaries.** Ventilation and cooling for the Unit 1 and Unit 2 train B 6.9-kV shutdown boards and Unit 2 train A and train B 480V shutdown boards are supplied by air handling units B-A and D-B. For each AHU to operate properly, its associated breaker, fuses, and damper must also function effectively. The AHUs receive 480V shutdown power to start and run.

  - **Important Intersystem Dependencies.** Failure of this event, unless recovered before temperatures become excessive, will result in failure of the Unit 1 and Unit 2 train B 6.9-kV shutdown boards and the Unit 2 480V shutdown boards. The AHUs require success of power from the associated 480V shutdown board 2A2-A or 2B2-B; i.e., Top Events A2U2 and B2U2, respectively.

  - The present model shows the Unit 2 shutdown boards dependent on Top Event V2. The actual dependency as described above was determined after the quantification of the model. Correcting the alignment in the model is not expected to significantly impact the model.

- **Top Event V2R — Recovery of Unit 2 Shutdown Board Room Ventilation**

  - **Function Evaluated.** This top event models the recovery of room ventilation to the 480V board shutdown board rooms prior to overheating of the 480V shutdown boards.

  - **Success Criteria.** The restoration of ventilation to the shutdown boards must be completed within 12 hours.

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*After the production of this model, the power supply to this board was changed to 480V shutdown board 2A2-A.

- **Model Boundaries.** This top event considers the recovery actions necessary to restore room cooling to 480V shutdown board rooms, given that the redundant fans of the normally operating ventilation system fail; i.e., given that Top Event V2 failed. The human action identifier is termed HV1R1.

- **Important Intersystem Dependencies.** This top event is assumed to be guaranteed successful if the normally operating ventilation system functions normally.

- **Top Event VINV2 — Unit 2 480V Board Room B Ventilation**

  - **Function Evaluated.** This event models the availability of ventilation to the Unit 2 480V board room B, which contains the unit inverters.

  - **Success Criteria.** Success of this event requires that the fan associated with the AHU operate for 24 hours to provide cool outside air to the Unit 2 480V board room after plant trip. The air conditioning unit is not required.

  - **Model Boundaries.** The model for this top event considers the AHU of the 480V board room ventilation system. The fan is supplied power by 480V C&A ventilation board 2B1-B; i.e., included in the model for Top Event B1U2.

  - **Important Intersystem Dependencies.** Failure of this event, unless recovered before temperatures become excessive, will result in failure of 120V vital instrument channels 1-III and 1-IV when the inverters overheat. This ventilation system is failed if power from 480V shutdown board 2B1-B (i.e., Top Event B1U2) has failed, which deenergizes the C&A ventilation board.

- **Top Event VNV2R — Recovery of Unit 2 480V Board Room B Ventilation**

  - **Function Evaluated.** This top event models the recovery of room ventilation to the Unit 2 480V board room B prior to overheating of the inverters it contains.

  - **Success Criteria.** The restoration of ventilation to room B must be completed within 6 hours.

  - **Model Boundaries.** This top event considers the recovery actions necessary to restore room cooling to 480V board room B, given that the normally operating ventilation system fails; i.e., given that Top Event VINV2 failed. The human action is termed HVNVR1.

  - **Important Intersystem Dependencies.** This top event is assumed to be guaranteed successful if the normally operating ventilation system functions normally.

- **Top Events DCAC and DDAC — 120V AC Power Subsystems 1-III and 1-IV**

  - **Function Evaluated.** These top events model the availability of power at the 120V AC vital distribution panels for channels 1-III and 1-IV.

  - **Success Criteria.** Success of each of these top events requires that power is available at the associated 120V AC panel for 24 hours after plant trip if the associated 480V shutdown board is available, and for 4 hours if it is not.

  - **Model Boundaries.** Each top event includes one 125V DC to 120V AC invertor, a 120V AC bus, and associated breakers and fuses.

  - **Important Intersystem Dependencies.** The 120V AC panels are supplied power from two sources each. Top Event DCAC (i.e., 120V AC instrument channel 1-III) is failed if both 125V DC battery board III (i.e., Top Event DC) and 480V shutdown board* 2A1-A (i.e., Top Event A1U2) are failed. If A1U2 alone fails, 120V AC instrument channel 1-III can still be powered for 4 hours from the 125V DC battery board III. Top Event DCAC is also failed if Unit 2 480V board room B ventilation is lost (i.e., Top Event VINV2 fails) and is not recovered, or if channel III is lost as an initiator.

    Top Event DDAC (i.e., 120V AC instrument channel 1-IV) is failed if both 125V DC battery board IV (i.e., Top Event DD) and 480V shutdown board 2B1-B (i.e., Top Event B1U2) are failed. If Top Event B1U2 alone fails, 120V AC instrument channel 1-IV can still be powered for 4 hours from the 125V DC battery board IV. Top Event DDAC is also failed if Unit 2 480V board room B ventilation is lost (i.e., Top Event VINV2 fails) and is not recovered, or if channel IV is lost as an initiator.

3.1.4.3 Support System Event Tree for Units 1 and 2 Mechanical Support Systems

The mechanical support systems event tree (MECH) is used to represent the various responses of the Watts Bar mechanical support systems following an initiating event. Each response is depicted by event sequences that model different combinations of mechanical support system successes and failures. The systems included in this event tree are those that do not provide electric power or directly perform accident-mitigating functions in response to a plant transient. Rather, the mechanical support systems found here provide the actuation signals, cooling water, and compressed air to allow the frontline systems to perform their functions. Mechanical support systems in the event tree were identified through the system analysis portion of this PRA.

The ordering of top events in the MECH event tree is presented in Table 3.1.4-3.

In addition to the mechanical support systems modeled in this event tree, some additional top events are included in the middle of the event tree. These top events summarize the status of electric power availability at both Units 1 and 2; i.e., Top Events AAL

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*After the production of this model, the power supply to this board was changed to 480V shutdown board 2A2-A.

through B2UL. These events account for the circular intersystem dependency logic exhibited by the electrical top events modeled in the first two support system event trees.

For example, the onsite diesel generators depend on cooling water from ERCW, but the ERCW pumps also depend on AC power provided by the diesel generators under loss of offsite power conditions. In evaluating the status of the diesel generators, the availability of cooling water from ERCW was assumed. If ERCW is found in this tree to have failed, the loss of ERCW cooling water on the availability of the affected diesel generators must be accounted for. The logic rules for Top Events AAL through B2U2L in this tree are used to reflect these circular logic dependencies.

These top events are positioned in the middle of the mechanical support systems event tree so that the complete status of power at each of these electrical top events can be used when evaluating the availability of the mechanical support systems that follow these top events but are not involved in the circular logic dependencies; e.g., component cooling water, instrument air, and the refueling water storage tank (RWST).

The top events in the MECH event tree are described below.

- **Top Event ZA — ESFAS Train A**

  - **Function Evaluated.** This top event considers the production of actuation signals by train A of the engineered safety features actuation system (ESFAS) portion of the solid state protection system (SSPS).

  - **Success Criteria.** The top event is successful if all required automatic actuation signals for train A of ESFAS are produced for a specific accident sequence.

  - **Model Boundaries.** Equipment modeled in this top event includes pressure, level, and temperature transmitters, relays, and bistables, their relationship to signal channel operation, and master and slave relays related to SSPS relay operation.

    The ESFAS provides actuation signals for (1) automatic reactor shutdown through the reactor protection system and (2) control of the following modeled safeguard equipment: safety injection, containment isolation, main feedwater isolation, and auxiliary feedwater operation. ESFAS monitors such parameters as reactor coolant system pressure and temperature, containment pressure, and steam generator pressure, level, and flow rate in comparison with preset values for these parameters. When these preset values are exceeded, ESFAS produces signals that lead to the appropriate safety equipment actuation.

    The ESFAS consists of two discrete portions of circuitry: (1) an analog portion consisting of three to four redundant channels per monitored parameter, and (2) a digital portion made up of two redundant logic trains (train A and train B) that receive the analog inputs and perform the logic necessary to actuate the appropriate engineered safety features equipment. Each digital train is capable of actuating the required safety equipment.

The model considers the actuation signals required for a particular accident sequence. Separate evaluations of actuation signal unavailability are computed for large LOCAs, small LOCAs, steam line breaks inside or outside containment, and for general transients that do not involve a safety injection condition but for which actuation of auxiliary feedwater (AFW) is still required. Steam generator tube ruptures and sequences involving an inadvertent safety injection are evaluated under the same conditions as for small LOCAs.

The model conservatively assumes that if any actuation signal required for a given sequence fails, all of the automatic actuation signals for that train are also failed.

— **Important Intersystem Dependencies.** Failure of this top event implies that the train A ESFAS signal was not produced to trip the reactor and that necessary emergency core cooling equipment, containment isolation valves, and containment integrity equipment did not receive an automatic actuation signal.

  This event is guaranteed failed if 120V AC instrument channel 1-I has failed so that the actuation relays for this train cannot be energized; i.e., if Top Event DAAC fails.

- **Top Event ZB — ESFAS Train B**

  — **Function Evaluated.** This top event considers the production of actuation signals by train B of the ESFAS portion of the SSPS.

  — **Success Criteria.** The top event is successful if all required automatic actuation signals for train B of ESFAS are produced for a specific accident sequence.

  — **Model Boundaries.** Equipment modeled in this top event includes pressure, level, and temperature transmitters, relays, and bistables, their relationship to signal channel operation, and master and slave relays related to SSPS relay operation.

  The ESFAS provides actuation signals for (1) automatic reactor shutdown through the reactor protection system, and (2) control of the following modeled safeguard equipment: including safety injection, containment isolation, main feedwater isolation, and auxiliary feedwater operation. ESFAS monitors such parameters as reactor coolant system pressure and temperature, containment pressure, steam generator pressure level and flow rate in comparison with preset values for these parameters. When these preset values are exceeded, ESFAS produces signals that lead to the appropriate safety equipment actuation.

  The ESFAS consists of two discrete portions of circuitry: (1) an analog portion consisting of three to four redundant channels per monitored parameter; and (2) a digital portion made up of two redundant logic trains (train A and train B) that receive the analog inputs and perform the logic necessary to actuate the appropriate engineered safety features equipment. Each digital train is capable of actuating the required safety equipment.

The model considers the actuation signals required for a particular accident sequence. Separate evaluations of actuation signal unavailability are computed for large LOCAs, small LOCAs, steam line breaks inside or outside containment, and for general transients that do not involve a safety injection condition but for which actuation of AFW is still required. Steam generator tube ruptures and sequences involving an inadvertent safety injection are evaluated under the same conditions as for small LOCAs.

The model conservatively assumes that if any actuation signal required for a given sequence fails, all of the automatic actuation signals for that train are also failed. The model is evaluated conditionally on the status of Top Event ZA to ensure that common cause failures involving relay failures in both trains are properly accounted for.

- **Important Intersystem Dependencies.** Failure of this top event implies that the train B ESFAS signals were not produced to trip the reactor and that necessary emergency core cooling equipment, containment isolation valves, and containment integrity equipment did not receive an automatic actuation signal.

  This top event is guaranteed failed if 120V AC instrument channel 1-II has failed so that the actuation relays for this train cannot be energized; i.e., if Top Event DBAC fails.

  This top event models the actuation signals produced by ESFAS train B. The configuration and success criteria for this event are similar to those for Top Event ZA but involve ESFAS train B.

- **Top Event OS — Manual Actions To Back Up ESFAS**

  - **Function Evaluated.** This top event models the operator actions to verify the automatic actuation and alignment of emergency equipment. The term "verify" implies "correctly position or actuate any incorrect alignment."

  - **Success Criteria.** Success requires that either both trains of automatic signals from ESFAS function properly or the operators intervene to actuate the required equipment manually in time for it to perform its intended function. The allowable times for action depend on the accident sequence as follows:

    - Align ECCS, given medium or large LOCA — 1 minute.

    - Align ECCS, given small LOCA, steam generator tube rupture, or steam line break — 30 minutes.

    - Start AFW, given no safety injection condition — 1 hour.

- **Model Boundaries.** The operator actions modeled include:

  - Verify ECCS Status
  - Verify Containment Isolation
  - Verify Auxiliary Feedwater Status
  - Verify Component Cooling System Flow
  - Verify ERCW Pumps Running
  - If Containment Pressure ≥ 2.81 psig, Verify Spray Pumps Running

  Manual intervention is required whether only one or both trains of ESFAS fail, as modeled in Top Events ZA and ZB. The manual actuation to trip the reactor is not modeled in this top event. Instead, it is included in Top Event RT of the frontline event tree models.

- **Important Intersystem Dependencies.** Failure of Top Event OS is treated as a failure to manually initiate one or more trains of high pressure injection, containment spray, auxiliary feedwater, component cooling water, ERCW, and containment isolation if the automatic actuation has failed. Different operator error rates are used, depending on the specific initiating event being evaluated; e.g., small LOCA versus a large LOCA. There are no support dependencies modeled for the manual actuation signal. The control power needed to actuate each piece of equipment is modeled with the top event representing the equipment actuated.

  Top Event OS is guaranteed successful if both Top Events ZA and ZB succeeded.

- **Top Event AE — Essential Raw Cooling Water Train A Pumps**

  - **Function Evaluated.** This top event models the availability of the train A ERCW pumps for providing flow to the two train A ERCW headers (1A and 2A) for Units 1 and 2 as modeled in Top Events CE and EE.

  - **Success Criteria.** Success of this top event requires that (1) two of the four train A pumps supply sufficient cooling water flow for proper unit cooldown or postaccident operations for 24 hours, and (2) one train A traveling screen operates properly; i.e., does not plug during operation.

    For loss of offsite power conditions with failure of one diesel generator that supplies one of the pumps on this train, the success criterion is different. The loss of one train of AC power means that fewer loads need to be supplied. Under these conditions, just one ERCW pump is required to remain at hot standby. Two ERCW pumps are still required, however, if recirculation from the containment pump is needed.

  - **Model Boundaries.** This model considers the four train ERCW pumps that supply the train A Unit 1 and Unit 2 headers. The associated traveling screen wash pumps are not analyzed since they are assumed not required. Failure of the screen wash pumps, which operate intermittently, could lead to eventual blockage of ERCW flow, but this is assumed to occur only gradually over the 24-hour mission time, or until these pumps are restored.

Under normal operating conditions, the two normally operating pumps (C-A and A-A) modeled in this top event receive control power from separate 125V DC boards (I and III, Top Events DA and DC) and motive power from separate 6.9-kV shutdown boards (1A-A and 2A-A, modeled via Top Events AA and AB, respectively). The other two pumps that supply the train A headers, pumps D-A and B-A, are in standby. The normally running pumps are assumed selected for restart on a loss of offsite power.

Following a loss of offsite power, the two running ERCW pumps will stop, and the two selected ERCW pumps, one per shutdown board, will receive ESFAS start actuation signals. Following a safety injection signal, the selected pumps, if different from those already running, will start, while the currently running pumps continue operating.

If one of the running pumps fails during operation, the operator must act to recover lost flow by manually starting a redundant, selected ERCW pump. The operator action to start one of the selected pumps manually during a nonsafety injection condition is HAAE1. This action is not sensitive to time. Tens of minutes are assumed to be available to start the standby pump. Another action to initiate the standby pump is for loss of offsite power conditions in which one diesel generator is also failed. This action is identified as HAAE2. Tens of minutes are also assumed available for this action.

– **Important Intersystem Dependencies.** Failure of the train A ERCW pumps modeled in Top Event AE results in failure of CE and EE, the top events for ERCW header 1A and ERCW header 2A. Failure of Top Event AE will ultimately result in failure or degraded operation of numerous pumps and heat exchangers that are supplied cooling water by these headers.

  Top Event AE is guaranteed failed if a combination of Unit 1 and Unit 2 control and motive power supplies combine to preclude support to at least two of the four train A pumps. The top event is also failed if there is a safety injection condition in a sequence involving loss of one train of AC power (i.e., Top Event AA or AB) and neither ESFAS nor the operators actuate one of the standby pumps for train A. Top Event AE is also failed by initiating events that directly impact the train A ERCW pumps; e.g., initiator ERCWA and selected floods that originate from the ERCW system.

- **Top Event BE — Essential Raw Cooling Water Train B Pumps**

  – **Function Evaluated.** This event models the availability of the train B ERCW pumps for providing flow to the two train B ERCW headers (1B and 2B) for Units 1 and 2 as modeled in Top Events DE and FE.

  – **Success Criteria.** Success of this event requires that (1) two of the four train B pumps supply sufficient cooling water flow for proper unit cooldown or postaccident operations for 24 hours, and (2) one train B traveling screen operates properly; i.e., does not plug during operation.

For loss of offsite power conditions with failure of one diesel generator that supplies one of the pumps on this train, the success criterion is different. The loss of one train of AC power means that fewer loads need to be supplied. Under these conditions, just one ERCW pump is required to remain at hot standby. Two ERCW pumps are still required, however, if recirculation from the containment pump is needed.

— **Model Boundaries.** This model considers the four train ERCW pumps that supply the train B Unit 1 and Unit 2 headers. The associated traveling screen wash pumps are not analyzed since they are assumed not required. Failure of the screen wash pumps that operate intermittently could lead to eventual blockage of ERCW flow, but this is assumed to occur only gradually over the 24-hour mission time, or until these pumps are restored.

Under normal operating conditions, the two normally operating pumps (G-B and E-B) modeled in this top event receive control power from separate 125V DC boards (II and IV, Top Events DB and DD) and motive power from separate 6.9-kV shutdown boards (1B-B and 2B-B, modeled via Top Events BA and BB, respectively). The other two pumps that supply the train B headers, pumps H-B and F-B, are in standby. The normally running pumps are assumed selected for restart on a loss of offsite power.

Following a loss of offsite power, the two running ERCW pumps will stop, and the two selected ERCW pumps, one per shutdown board and assumed to be the normally running pumps, will receive ESFAS start actuation signals. Following a safety injection signal, the selected pumps, if different from those already running, will start, while the currently running pumps continue operating.

If an MCR evacuation occurs, the MCR ERCW pump selector switch is unavailable for use. In this situation, two ERCW train B pumps will automatically restart, and the other pumps must be manually restarted by the operator from the 6.9-kV shutdown board rooms.

If one of the running pumps fails during operation, the operator must act to recover lost flow by manually starting a redundant, selected ERCW pump. The operator action to manually start one of the selected pumps during a nonsafety injection condition is HAAE1. This action is not sensitive to time. Tens of minutes are assumed to be available to start the standby pump. Another action to initiate the standby pump is for loss of offsite power conditions in which one diesel generator is also failed. This action is identified as HAAE2. Tens of minutes are also assumed to be available for this action.

The model for Top Event BE is somewhat more complicated than for Top Event AE. The model for Top Event BE is evaluated conditionally on the status of Top Event AE to account for the common cause dependencies modeled among the four pumps.

- **Important Intersystem Dependencies.** Failure of the train B ERCW pumps modeled in Top Event BE results in failure of Top Events DE and FE, the Top Events for ERCW header 1B and ERCW header 2B. Failure of Top Event BE will ultimately result in failure or degraded operation of numerous pumps and heat exchangers that are supplied cooling water by these headers.

  Top Event BE is guaranteed failed if a combination of Unit 1 and Unit 2 control and motive power supplies combine to preclude support to at least two of the four train B pumps. The top event is also failed if there is a safety injection condition in a sequence involving loss of one train of AC power (i.e., Top Event BA or BB) and neither ESFAS nor the operators actuate one of the standby pumps for train B. Top Event BE is also failed by initiating events that directly impact the train B ERCW pumps; e.g., initiator ERCWB and selected floods that originate from the ERCW system.

- **Top Event MDE — Maintenance on ERCW Header 1B**

  - **Function Evaluated.** This event models the fraction of time that maintenance is being performed on ERCW header 1B.

  - **Success Criteria.** Success requires that ERCW header 1B not be in maintenance at the time of the plant trip.

  - **Model Boundaries.** The model for this event is simply the fraction of time that ERCW header 1B is in maintenance. No other operator actions or hardware failure modes are included.

  - **Important Intersystem Dependencies.** When Top Event MDE is failed, ERCW header 1B is in maintenance. For the maintenance duration, the CCS train A heat exchanger is aligned for cooling via ERCW header 2A, i.e. Top Event EE.

- **Top Event CE — ERCW Header 1A-A**

  - **Function Evaluated.** This event models the availability of ERCW header 1A to supply cooling water to its serviced loads.

  - **Success Criteria.** Success of this event requires that ERCW header 1A provide cooling water to its serviced loads for 24 hours following a plant trip.

  - **Model Boundaries.** This event models the flow path from the train A ERCW pumps via station discharge valve 1-FCV-67-22 through strainer 1A-A and valve 1-FCV-67-81 to the various Unit 1 coolers and heat exchangers for 24 hours.

Normal heat loads supplied by ERCW header 1A include the diesel generator heat exchangers for 1A-A and 2A-A, upper and lower containment vent coolers, various room coolers, space coolers for the motor-driven auxiliary feedwater pumps, air conditioning equipment/chillers, the service air compressor, and aftercoolers as well as containment spray heat exchanger 1A when necessary. These loads are modeled with the system served.

Operator actions to crosstie the ERCW header 1A-A with ERCW header 2B-B, in the event that cooling water to header 1A-A is lost, is conservatively not modeled.

- **Important Intersystem Dependencies.** Loss of cooling water to header 1A-A (i.e., failure of Top Event CE) will result in failure or degradation of numerous systems that rely on header 1A for cooling water. This event is failed if flow from the train A ERCW pumps, modeled via Top Event AE, is unavailable. Top Event CE is also assumed to be failed if this header is the source of a flood initiator in the auxiliary building.

• **Top Event EE — ERCW Header 2A-A**

- **Function Evaluated.** This event models the availability of ERCW header 2A to supply cooling water to its serviced loads.

- **Success Criteria.** Success of this event requires that ERCW header 2A provide cooling water to its serviced load for 24 hours following a plant trip.

- **Model Boundaries.** This event models the flow path from the train A ERCW pumps via station discharge valve 2-FCV-67-22 through strainer 2A-A and valve 1-FCV-67-81 to the various Unit 2 coolers and heat exchangers for 24 hours. The cooling loads serviced by header 2A (e.g., CCS heat exchanger B on Unit 1) are modeled with the system served.

  Operator actions to crosstie the ERCW header 2A-A with ERCW header 1B-B, in the event that cooling water to header 2A-A is lost, is conservatively not modeled.

- **Important Intersystem Dependencies.** Loss of cooling water to header 2A-A (i.e., failure of Top Event EE) will result in failure or degradation of numerous systems that rely on header 2A for cooling water. This event is failed if flow from the train A ERCW pumps, modeled via Top Event AE, is unavailable.

• **Top Event DE — ERCW Header 1B-B**

- **Function Evaluated.** This event models the availability of ERCW header 1B to supply cooling water to its serviced loads.

- **Success Criteria.** Success of this event requires that ERCW header 1B provide cooling water to its serviced loads for 24 hours following a plant trip.

- **Model Boundaries.** This event models the flow path from the train B ERCW pumps via station discharge valve 1-FCV-67-24 through strainer 1B-B and valve 1-FCV-67-82 to the various Unit 1 coolers and heat exchangers for 24 hours.

  Normal heat loads supplied by ERCW header 1B include the diesel generator heat exchangers for 1B-B and 2B-B, upper and lower containment vent coolers, various room coolers, spare coolers for the motor-driven auxiliary feedwater pumps, air conditioning equipment/chillers, CCS heat exchanger A, the service air compressors, and aftercoolers as well as containment spray heat exchanger 1B when necessary. These loads are modeled with the system served.

  Operator actions to crosstie the ERCW header 1B-B with ERCW header 2A-A, in the event that cooling water to header 1B-B is lost, is conservatively not modeled.

- **Important Intersystem Dependencies.** Loss of cooling water to header 1B-B (i.e., failure of Top Event DE) will result in failure or degradation of numerous systems that rely on header 1B for cooling water. This event is failed if flow from the train B ERCW pumps, modeled via Top Event BE, is unavailable.

• **Top Event FE — ERCW Header 2B-B**

- **Function Evaluated.** This top event models the availability of ERCW header 2B to supply cooling water to its serviced loads.

- **Success Criteria.** Success of this top event requires that ERCW header 2B provide cooling water to its serviced load for 24 hours following a plant trip.

- **Model Boundaries.** This top event models the flow path from the train B ERCW pumps via station discharge valve 2-FCV-67-24 through strainer 2B-B and valve 1-FCV-67-82 to the various Unit 2 coolers and heat exchangers for 24 hours. The cooling loads serviced by header 2B (e.g., CCS heat exchanger C on Unit 1) are modeled with the system served.

  Operator actions to crosstie the ERCW header 2B-B with ERCW header 1A-A, in the event that cooling water to header 2B-B is lost, are conservatively not modeled.

- **Important Intersystem Dependencies.** Loss of cooling water to header 2B-B (i.e., failure of Top Event FE) will result in failure or degradation of numerous systems that rely on header 2B for cooling water. This event is failed if flow from the train B ERCW pumps, modeled via Top Event BE, is unavailable.

• **Top Event DSLR — Recovery of ERCW to Diesel from Opposite Side**

- **Function Evaluated.** This top event models the realignment of ERCW cooling flow to a diesel generator that has insufficient cooling water during a loss of offsite power.

- **Success Criteria.** Success requires that the operators trip the diesel generator operating without sufficient cooling water flow within 5 minutes, and then realign cooling from the opposite train of ERCW. If ERCW is aligned quickly, the diesel need not be tripped.

- **Model Boundaries.** This top event models just the operator action to trip the affected diesel generator, realign for alternate cooling by repositioning the valves to the diesel generator's heat exchanger and then restarting the affected diesel generator. The action identifier is HERCW1.

- **Important Intersystem Dependencies.** This action is guaranteed successful if offsite power is available for the entire period. Top Event DSLR is guaranteed failed if headers 1A and 2B or 2A and 1B are both failed.

- **Top Event GE — ERCW Discharge Header A**

  - **Function Evaluated.** This top event models the ERCW discharge flow path for header A.

  - **Success Criteria.** Success requires that the flow path for the ERCW train A discharge line remain open for 24 hours following a plant trip.

  - **Model Boundaries.** This top event models the ERCW discharge header A valve FCV-67-360, and, in the event that it transfers closed or plugs, the hydraulic gradient flow path. Since the second, redundant path is just an open pipe, this event is modeled as always successful.

  - **Important Intersystem Dependencies.** Failure of the discharge path for train A ERCW fails the cooling function for this train. There are no dependencies on other systems for this event.

- **Top Event HE — ERCW Discharge Header B**

  - **Function Evaluated.** This top event models the ERCW discharge flow path for header B.

  - **Success Criteria.** Success requires that the flow path for the ERCW train B discharge line remain open for 24 hours following a plant trip.

  - **Model Boundaries.** This top event models the ERCW discharge header B valve FCV-67-362, and, in the event that it transfers closed or plugs, the hydraulic gradient flow path. Since the second, redundant path is just an open pipe, this event is modeled as always successful.

  - **Important Intersystem Dependencies.** Failure of the discharge path for train B ERCW fails the cooling function for this train. There are no dependencies on other systems for this event.

- **Top Event AAL — ERCW/Diesel 1A/6.9-kV Shutdown Board 1A-A Dependency**

  — **Function Evaluated.** This top event models the status of electric power at the 6.9-kV shutdown board for Unit 1 train A, including all of the effects of circular logic intersystem dependencies.

  — **Circular Logic Intersystem Dependencies.** This event tracks the circular logic dependencies between the diesel generator that supplies an alternate source of power to the Unit 1 6.9-kV shutdown board in the event that offsite power is not recovered within 1 hour, and four other actions that are questioned after Top Event AA, which tracks the direct failures of this same board.

  The four actions are as follows:

  - The ERCW trains that must operate to supply cooling water for the diesel generator 1A-A during a loss of offsite power. The logic is circular in that the ERCW pumps that supply cooling water also require AC power from this shutdown board. The model assumes that for sufficient cooling water to a diesel generator, flow from two ERCW pumps must be available to the associated train of ERCW supplying that diesel generator. Each ERCW pump of a train is supplied AC power from a separate diesel during losses of offsite power but each diesel is on ERCW header 1A. If either diesel generator on a train is insufficiently cooled, both may be eventually lost.

    However, if one diesel generator on a train fails to operate, the success criterion for adequate ERCW flow to the remaining diesel is then only one ERCW pump unless recirculation from the containment sump is required. This dependency on the number of diesel generators initially available is tracked in the model.

  - The dependence of this shutdown board on room ventilation provided by the Unit 1 shutdown board ventilation system and the action to recover the ventilation system if it is initially lost; i.e., Top Events V1 and V1R, which appear in the ELECT1 event tree.

  - The DC control power needed to load the diesel generator; i.e., Top Events DA and DC.*

  - The recovery of ERCW to header 1A for use in cooling diesel generator 1A-A by crosstieing cooling water from the Unit 2 header 2B, as modeled via Top Events DSLR and FE. The availability

*Note: The additional requirement for Top Event DC was discovered after the preparation of this model, but it was felt that the probability of battery failure was significantly less than the probability of the diesel generator to start and run, therefore no change was made to the model at this time.

of the associated ERCW discharge header is also considered; i.e., Top Events GE and HE.

Power to the 6.9-kV Unit 1 shutdown board train A may be lost due to any of the above reasons, even if Top Event AA is successful. The dependencies of other systems on this shutdown board for AC power are therefore evaluated based on the status of Top Event AAL, which considers these additional failure considerations, rather than of Top Event AA, which does not. Two top events were required to consider these phenomena due to the circular logic involved.

- **Top Event A1L — ERCW/Diesel 1A/480V Shutdown Board 1A2-A Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 1A1-A, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 1 shutdown board 1A1-A. All of the dependencies described above for Top Event AAL apply. Therefore, this top event is failed if Top Event AAL fails, or if Top Event A1, which tracks the direct failures of this shutdown board, fails.

    Additionally, this event may be failed due to the loss of 480V shutdown transformer room 1A ventilation, and of recovery of room cooling, as modeled in Top Events VT1A and VT1AR in the ELECT1 event tree for electrical support systems.

- **Top Event A2L — ERCW/Diesel 1A/480V Shutdown Board 1A2-A Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 1A2-A, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 1 shutdown board 1A2-A. All of the dependencies described above for Top Event AAL apply. Therefore, this top event is failed if Top Event AAL fails or if Top Event A2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 1A ventilation, and of recovery of room cooling, as modeled in Top Events VT1A and VT1AR in the ELECT1 event tree for electrical support systems.

- **Top Event ABL — ERCW/Diesel 2A/6.9-kV Shutdown Board 2A-A Dependency**

  — **Function Evaluated.** This top event models the status of electric power at the 6.9-kV shutdown board for Unit 2 train A, including all of the effects of circular logic intersystem dependencies.

  — **Circular Logic Intersystem Dependencies.** This event tracks the circular logic dependencies between the diesel generator that supplies an alternate source of power to the Unit 2 6.9-kV shutdown board in the event that offsite power is not recovered within 1 hour, and four other actions that are questioned after Top Event AB, which tracks the direct failures of this same board.

  The four actions are as follows:

  - The ERCW trains that must operate to supply cooling water for the diesel generator 2A-A during a loss of offsite power. The logic is circular in that the ERCW pumps that supply cooling water also require AC power from this shutdown board. The model assumes that for sufficient cooling water to a diesel generator, flow from two ERCW pumps must be available to the associated train of ERCW supplying that diesel generator. Each ERCW pump of a train is supplied AC power from a separate diesel during losses of offsite power but each diesel is on ERCW header 1A. If either diesel generator on a train is insufficiently cooled, both are eventually lost.

    However, if one diesel generator on a train fails to operate, the success criterion for adequate ERCW flow to the remaining diesel is then only one ERCW pump unless recirculation from the containment sump is required. This dependency on the number of diesel generators initially available is tracked in the model.

  - The dependence of this shutdown board on room ventilation provided by the Unit 2 shutdown board ventilation system and the action to recover the ventilation system if it is initially lost; i.e., Top Events V2 and V2R, which appear in the ELECT2 event tree.

  - The DC control power needed to load the diesel generator; i.e., Top Events DC and DA.*

  - The recovery of ERCW to header 2A for use in cooling diesel generator 2A-A by crosstieing cooling water from the Unit 1 header 2B as modeled via Top Events DSLR and FE. The availability of the associated ERCW discharge header is also considered; i.e., Top Events GE and HE.

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*Note: The additional requirement for Top Event DA was discovered after the preparation of this model, but it was felt that the probability of battery failure was significantly less than the probability of the diesel generator to start and run. Therefore no change was made to the model at this time.

Power to the 6.9-kV Unit 2 shutdown board train A may be lost due to any of the above reasons, even if Top Event AB is successful. The dependencies of other systems on this shutdown board for AC power are therefore evaluated based on the status of Top Event AAL, which considers these additional failure considerations, rather than of Top Event AB, which does not. Two top events were required to consider these phenomena due to the circular logic involved.

- **Top Event A1U2L — ERCW/Diesel 2A/480V Shutdown Board 2A1-A Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 2A1-A, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 1 shutdown board 2A1-A. All of the dependencies described above for Top Event ABL apply. Therefore, this top event is failed if Top Event ABL fails or if Top Event A1U2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 2A ventilation, and of recovery of room cooling, as modeled in Top Events VT2A and VT2AR in the ELECT2 event tree for electrical support systems. This dependence on room ventilation accounts for the circular logic that finds that the fans in this ventilation systems also depend on AC power from this same shutdown board.

- **Top Event A2U2L — ERCW/Diesel 2A/480V Shutdown Board 2A2-A Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 2A2-A, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 2 shutdown board 2A2-A. All of the dependencies described above for Top Event ABL apply. Therefore, this top event is failed if Top Event ABL fails or if Top Event A2U2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 2A ventilation, and of recovery of room cooling, as modeled in Top Events VT2A and VT2AR in the ELECT2 event tree for electrical support systems.

- **Top Event BAL — ERCW/Diesel 1B/6.9-kV Shutdown Board 1B-B Dependency**

  - **Function Evaluated.** This top event models the status of electric power at the 6.9-kV shutdown board for Unit 1 train B, including all of the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This event tracks the circular logic dependencies between the diesel generator that supplies an alternate source of power to the Unit 1 6.9-kV shutdown board in the event that offsite power is not recovered within 1 hour, and four other actions that are questioned after Top Event AA, which tracks the direct failures of this same board.

    The four actions are as follows:

    - The ERCW trains that must operate to supply cooling water for the diesel generator 1B-B during a loss of offsite power. The logic is circular in that the ERCW pumps that supply cooling water also require AC power from this shutdown board. The model assumes that for sufficient cooling water to a diesel generator, flow from two ERCW pumps must be available to the associated train of ERCW supplying that diesel generator. Each ERCW pump of a train is supplied AC power from a separate diesel during losses of offsite power but each diesel is on ERCW header 1B. If either diesel generator on a train is insufficiently cooled, both may be eventually lost.

      However, if one diesel generator on a train fails to operate, the success criterion for adequate ERCW flow to the remaining diesel is then only one ERCW pump unless recirculation from the containment sump is required. This dependency on the number of diesel generators initially available is tracked in the model.

    - The dependence of this shutdown board on room ventilation provided by the Unit 1 shutdown board ventilation system and the action to recover the ventilation system if it is initially lost; i.e., Top Events V1 and V1R, which appear in the ELECT1 event tree.

    - The DC control power needed to load the diesel generator; i.e., Top Events DB and DD.*

    - The recovery of ERCW to header 1B for use in cooling diesel generator 1B-B by crosstieing cooling water from the Unit 2 header 2A, as modeled via Top Events DSLR and EE. The availability

*Note: The additional requirement for Top Event DD was discovered after the preparation of this model, but it was felt that the probability of battery failure was significantly less than the probability of the diesel generator to start and run. Therefore no change was made to the model at this time.

of the associated ERCW discharge header is also considered; i.e., Top Events GE and HE.

Power to the 6.9-kV Unit 1 shutdown board train B may be lost due to any of the above reasons, even if Top Event BA is successful. The dependencies of other systems on this shutdown board for AC power are therefore evaluated based on the status of Top Event BAL, which considers these additional failure considerations, rather than of Top Event BA, which does not. Two top events were required to consider these phenomena due to the circular logic involved.

- **Top Event B1L — ERCW/Diesel 1B/480V Shutdown Board 1B1-B Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 1B1-B, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 1 shutdown board 1B1-B. All of the dependencies described above for Top Event BAL apply. Therefore, this event is failed if Top Event BAL fails or if Top Event B1, which tracks the direct failures of this shutdown board, fails.

    Additionally, this event may be failed due to the loss of 480V shutdown transformer room 1B ventilation, and of recovery of room cooling, as modeled in Top Events VT1B and VT1BR in the ELECT1 event tree for electrical support systems. This dependence on room ventilation accounts for the circular logic that finds that the fans in this ventilation systems also depend on AC power from this same shutdown board.

- **Top Event B2L — ERCW/Diesel 1B/480V Shutdown Board 1B2-B Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 1B2-B, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 1 shutdown board 1B2-B. All of the dependencies described above for Top Event BAL apply. Therefore, this top event is failed if Top Event BAL fails or if Top Event B2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 1B ventilation, and of recovery of room cooling, as modeled in Top Events VT1B and VT1BR in the ELECT1 event tree for electrical support systems.

- **Top Event BBL — ERCW/Diesel 2B/6.9-kV Shutdown Board 2B-B Dependency**

  - **Function Evaluated.** This top event models the status of electric power at the 6.9-kV shutdown board for Unit 2 train B, including all of the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies between the diesel generator that supplies an alternate source of power to the Unit 2 6.9-kV shutdown board in the event that offsite power is not recovered within 1 hour, and four other actions that are questioned after Top Event BB, which tracks the direct failures of this same board.

    The four actions are as follows:

    - The ERCW trains that must operate to supply cooling water for the diesel generator 2B-B during a loss of offsite power. The logic is circular in that the ERCW pumps that supply cooling water also require AC power from this shutdown board. The model assumes that for sufficient cooling water to a diesel generator, flow from two ERCW pumps must be available to the associated train of ERCW supplying that diesel generator. If either diesel generator on a train is insufficiently cooled, both are eventually lost.

      However, if one diesel generator on a train fails to operate, the success criterion for adequate ERCW flow to the remaining diesel is then only one ERCW pump unless recirculation from the containment sump is required. This dependency on the number of diesel generators initially available is tracked in the model.

    - The dependence of this shutdown board on room ventilation provided by the Unit 2 shutdown board ventilation system and the action to recover the ventilation system if it is initially lost; i.e., Top Events V2 and V2R, which appear in the ELECT2 event tree.

    - The DC control power needed to load the diesel generator; i.e., Top Events DD and DB.*

    - The recovery of ERCW to header 2B for use in cooling diesel generator 2B-B by crosstieing cooling water from the Unit 2 header 2A, as modeled via Top Events DSLR and EE. The availability of the associated ERCW discharge header is also considered; i.e., Top Events GE and HE.

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*Note: The additional requirement for Top Event DB was discovered after the preparation of this model, but it was felt that the probability of battery failure was significantly less than the probability of the diesel generator to start and run. Therefore no change was made to the model at this time.

Power to the 6.9-kV Unit 2 shutdown board train B may be lost due to any of the above reasons, even if Top Event BB is successful. The dependencies of other systems on this shutdown board for AC power are therefore evaluated based on the status of Top Event BBL, which considers these additional failure considerations, rather than of Top Event BB, which does not. Two top events were required to consider these phenomena due to the circular logic involved.

- **Top Event B1U2L — ERCW/Diesel 2B/480V Shutdown Board 2B1-B Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 2B1-B, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 2 shutdown board 2B1-B. All of the dependencies described above for Top Event BBL apply. Therefore, this top event is failed if Top Event BBL fails or if Top Event B1U2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 2B ventilation, and of recovery of room cooling, as modeled in Top Events VT2B and VT2BR in the ELECT2 event tree for electrical support systems.

- **Top Event B2U2L — ERCW/Diesel 2B/480V Shutdown Board 2B2-B Dependency**

  - **Function Evaluated.** This top event tracks the status of 480V shutdown board 2B2-B, including the effects of circular logic intersystem dependencies.

  - **Circular Logic Intersystem Dependencies.** This top event tracks the circular logic dependencies that dictate the status of AC power at the 480V Unit 2 shutdown board 2B2-B. All of the dependencies described above for Top Event BBL apply. Therefore, this top event is failed if Top Event BBL fails or if Top Event B2U2, which tracks the direct failures of this shutdown board, fails.

    Additionally, this top event may be failed due to the loss of 480V shutdown transformer room 2B ventilation, and of recovery of room cooling, as modeled in Top Events VT2B and VT2BR in the ELECT2 event tree for electrical support systems. This dependence on room ventilation accounts for the circular logic that finds that the fans in this ventilation systems also depend on AC power from this same shutdown board.

- **Top Event PE — ERCW Cooling to Control Air System Compressors**

  - **Function Evaluated.** This event models the availability of cooling water flow from ERCW headers 1A and 1B to the four control air system (CAS) station air compressors.

  - **Success Criteria.** Success of this top event requires the availability of one Unit 1 header of ERCW (i.e., 1A or 1B) to supply cooling water to the compressor header for 24 hours.

  - **Model Boundaries.** ERCW cooling water flow from header 1A or 1B normally provides cooling through a header that supplies individual CAS compressor intercoolers and aftercoolers. As a compressor starts, a solenoid valve (0-FSV-32-32, 0-FSV-32-37, 0-FSV-32-42, and 0-FSV-32-137) opens to allow cooling water through the intercooler and cylinder water jackets. Cooling water is continually flowing to the aftercoolers whether a compressor is operational or in standby. Two CAS compressors are loaded during normal plant operation. Following loss of the offsite grid, they must be manually restarted. The DC battery power for the solenoid valves is modeled with the individual compressor in Top Event PD. Compressors are supplied cooling water from ERCW header 1A-A, and by ERCW header 1B-B.

  - **Important Intersystem Dependencies.** Failure of Top Event PE results in failure of the control air system modeled next in Top Event PD. This event is guaranteed failed if both ERCW headers 1A-A (Top Event CE) and 1B-B (Top Event DE) are unavailable.

- **Top Event PD — Nonessential Control Air System**

  - **Function Evaluated.** This event questions the availability of the control air system to supply cooled, dry, oil-free, filtered air to all pneumatic equipment required to function for normal and accident plant operation.

  - **Success Criteria.** Success of this event requires that two of four CAS compressors, one of two receiver tanks, one of three control air dryers, and the associated filters, piping, and valves function to provide the total plant essential and nonessential air requirements for 24 hours.

  - **Model Boundaries.** All nonessential (condensate, makeup water, etc.) control air loads are modeled with the respective system supplied. Success of this top event implies that compressed air is being supplied at or above 78 psig to auxiliary compressed air system (ACAS) train A and train B headers through CAS isolation valves FCV 32-82 and FCV 32-85.

    The CAS dryers use cam-operated chamber switching valves to alternate between two desiccant-filled towers to remove moisture from the air after it leaves the receivers. Each dryer is sized to fully handle all control air requirements for one unit.

The CAS also supplies compressed air to the service air system. Because the air supply from the compressors to service air isolates when pressure decreases below 80 psig, only the closure of 0-PCV 33-4, the service/control air-operated isolation valve, is considered.

- **Important Intersystem Dependencies.** Failure of this top event results in CAS isolation from the ACAS headers, start signals being sent to the ACAS compressors, and a failure to supply compressed air to several loads not essential to emergency plant operation.

  The compressors depend directly on the availability of ERCW cooling water, Top Event PE, to prevent failure due to overheating. CAS compressors A and B are powered from 480V shutdown boards; i.e., Top Events B1L and A2L. The compressors must be manually restarted for success of Top Event PD following a loss of offsite power. Compressors C and D are powered from the 480V auxiliary building common board A and B (i.e., Top Events A3 and B3), and are sequenced on when demand exists, during normal plant operations only. DC control power for all four compressors is provided by 125V DC battery board II.

  This top event is also guaranteed failed if nonessential is lost as an initiating event, or if there is a flood in the turbine building.

- **Top Event PA — Auxiliary Control Air System Train A**

  - **Function Evaluated.** This top event models the availability of train A ACAS to supply train A essential air demands.

  - **Success Criteria.** Success requires the proper operation of the train A essential air components (i.e., air compressor, aftercooler, dryer A-A, and CAS isolation valve FCV 32-82) for 24 hours in the event that nonessential control air (i.e., Top Event PD) fails.

  - **Model Boundaries.** Operation of the ACAS compressors only occurs during abnormal or emergency conditions as the result of inadequate CAS supply pressure or via manual startup. The ACAS compressors receive a start signal when system pressure decreases below 80 psig. Isolation from the CAS occurs when pressure decreases below 78 psig with the closing of solenoid-operated isolation valve FCV 32-82. Following a Phase B containment isolation, valve FCV 32-80 will close, preventing air to the pressurizer spray valves. If nonessential control air is successful, then just the flow path from CAS to the essential loads on train B must be operable for success of this top event.

    The ACAS provides support to the AFW steam generator level control and pressure control valves, main steam atmospheric relief valves, pressurizer spray line pressure control valves, and other loads essential to proper plant operation. These air requirements are modeled with the system supplied.

In addition to proper ACAS compressor operation, this model includes consideration of the aftercooler, the ACAS dryer A-A, and achievement of isolation from the CAS by FCV 32-82. Transfer of any train A ACAS header valve from open to closed is conservatively assumed to fail Top Event PA.

- **Important Intersystem Dependencies.** Unavailability of Top Event PA would result in the failure to meet essential air demands. The supply of compressed air from CAS to the auxiliary control air system requires that the isolation valve remain open. Therefore, loss of 120V vital instrument power channel 1-I (i.e., Top Event DAAC) to the train A isolation valve or failure of CAS itself then requires that the ACAS compressors operate to provide essential compressed air. The train A auxiliary control air compressors require ERCW from header 1A (i.e., modeled via Top Events CE and GE) and of 480V shutdown board 2A1-A, modeled via Top Event A1U2L.

- **Top Event PB — Auxiliary Control Air System Train B**

  - **Function Evaluated.** This top event models the availability of train B ACAS to supply train B essential air demands.

  - **Success Criteria.** Success requires the proper operation of the train B essential air components (i.e., air compressor, aftercooler, dryer B-B, and CAS isolation valve FCV 32-85) for 24 hours in the event that nonessential control air (i.e., Top Event PD) fails.

  - **Model Boundaries.** Operation of the ACAS compressors only occurs during abnormal or emergency conditions as the result of inadequate CAS supply pressure or via manual startup. The ACAS compressors receive a start signal when system pressure decreases below 80 psig. Isolation from the CAS occurs when pressure decreases below 78 psig with the closing of solenoid-operated isolation valve FCV 32-85. Following a Phase B containment isolation, valve FCV 32-102 will close, preventing air to the pressurizer spray valves. If nonessential control air is successful, then just the flow path from CAS to the essential loads on train B must be operable for success of this top event.

    The ACAS provides support to the AFW steam generator level control and pressure control valves, main steam atmospheric relief valves, pressurizer spray line pressure control valves, and other loads essential to proper plant operation. These air requirements are modeled with the system supplied.

    In addition to proper ACAS compressor operation, this model includes consideration of the aftercooler, the ACAS dryer B-B, and achievement of isolation from the CAS by FCV 32-85. Transfer of any train B ACAS header valve from open to closed is conservatively assumed to fail Top Event PB.

  - **Important Intersystem Dependencies.** Unavailability of Top Event PB would result in the failure to meet essential air demands. The supply of compressed air from CAS to the auxiliary control air system requires that the isolation valve remain open. Therefore, loss of 120V vital instrument power channel 1-II (i.e., Top Event DBAC) to the train B isolation valve or failure of CAS itself then requires that the ACAS compressors operate to provide essential compressed air. The train B auxiliary control air compressors require ERCW from header 2B (i.e., modeled via Top Events FE and HE) and of 480V shutdown board 2B1-B, modeled via Top Event B1U2L.

- **Top Event V3 — Component Cooling Water System (CCS) Pumps and Motor-Driven Auxiliary Feedwater Pump Ventilation**

  - **Function Evaluated.** This event models that portion of the auxiliary building safety features equipment ventilation system that can provide cooling for the Unit 1 auxiliary feedwater and component cooling water pumps. The model also includes portions of the normal (non-ESF) building ventilation, which provides cooling these pumps.

  - **Success Criteria.** Success requires that two of the four ESF area coolers operate on demand for 24 hours or that both non-ESF Elevation 713' air handling units and associated auxiliary building cooling water operate for 24 hours after a plant trip.

  - **Important Intersystem Dependencies.** Failure of this ventilation system is assumed to eventually lead to failure of the operating CCS and motor-driven AFW pumps, due to overheating. ERCW headers 1A, 1B, 2A, and 2B (i.e., Top Events CE, DE, EE, and FE) provide cooling water to the associated coolers. The 480V shutdown boards 1A1-A, 1B1-B, 2A1-A, and 2B2-B (i.e., Top Events A1L, B1L, A1U2L, and B1U2L) provide power to the associated coolers. The non-ESF cooling is dependent on raw cooling water as a heat sink and is powered from the 6.9-kV common power distribution system (Top Events A3 and B3). The non-ESF cooling is not available following a loss of offsite power.

- **Top Event CCSR — Recovery of CCS by Crosstieing**

  - **Function Evaluated.** This top event models the actions for recovery of the CCS train A, given that it is initially unavailable.

  - **Success Criteria.** The operators must manually realign for successful CCS within 5 to 10 minutes; i.e., prior to charging pump failure.

  - **Model Boundaries.** This analysis models the action to realign CCS, given that CCS train A is initially unavailable. Action HCCSR1 considers the action to realign the C-S CCS pump from train B to the Unit 1 train A heat exchanger, given that both pumps initially aligned to train A (i.e., pumps 1A-A and 1B-B) have failed. Valve 0-70-510 to heat exchanger C must be closed as would be valves 1-FCV-70-64 and 1-FCV-70-74, and valves 1-FCV-70-13, 1-FCV-70-23, 1-FCV-70-75, and 2-FCV-70-75 must be opened.

  - **Important Intersystem Dependencies.** Action HCCSR1 is only used when CCS train B is successful.

- Top Event AC — Component Cooling System Train A

  - **Function Evaluated.** This top event models the availability of CCS train A to serve as an intermediate heat sink for the removal of heat from potentially radioactive heat loads during normal and accident conditions.

  - **Success Criteria.** Success of Top Event AC requires either the availability of both of the train A CCS pumps (1A-A or 1B-B) or one of these pumps and the operator action to reduce heat load by isolating the spent fuel pit heat exchangers, CCS heat exchanger A, and the train A flow path to the associated Unit 1 loads. CCS cooling must be provided for 24 hours following an initiating event.

  - **Model Boundaries.** The success criterion listed above is based on the assumption that the spent fuel pit cooling loads are initially being served by this train. The reactor coolant pumps and all safety-related pumps (i.e., containment spray, centrifugal charging, safety injection, and residual heat removal), with the exception of ERCW pumps, are cooled with water from CCS train A.

    The system is also used for emergency heat removal from the residual heat removal (RHR) heat exchangers as well as flow to the reactor coolant pump thermal barriers to maintain reactor coolant pump (RCP) seal integrity. Individual cooling water loads are modeled with the respective system served.

    Two operator actions are included in the analysis of this top event: i.e., actions HAAC1 and HAAC2. Action HAAC1 models the manual starting of the standby CCS pump, given that the running pump fails. The allowable recovery time is assumed to be 2 minutes, which is the estimated time to avoid overheating the RCPs. This action is only credited for sequences when there is no safety injection condition present.

    Action HAAC2 models the manual isolation of the spent fuel pool heat exchanger from CCS train A, given that there is insufficient cooling available to all of the loads. Isolation of the spent fuel pool heat exchanger changes the required number of pumps to just one. The time available to complete this action is estimated to be on the order of tens of minutes.

  - **Important Intersystem Dependencies.** Failure of AC will result in loss of RCP oil cooling and RCP thermal barrier cooling to the pumps supplied. Loss of the RCPs during power operation will result in a plant transient.

    Train A of CCS (i.e., Top Event AC) is guaranteed failed if the initiator involves a loss of CCS or if the CCS pump room ventilation system modeled via Top Event V3 is failed. ERCW header 1B supplies train A of CCS; i.e., AC fails if Top Event HE or DE fails. Failures of AC power from the two trains of 480V shutdown boards and 125V DC control power from battery boards I and II can also lead to failure of train A of CCS.

- **Top Event BC — CCS Train B**

  - **Function Evaluated.** This event models the availability of pump C-S, CCS heat exchanger C, and the train 1B flow path to meet Unit 1 and Unit 2 train B engineered safety features cooling requirements.

  - **Success Criteria.** Success of Top Event BC requires the availability of pump C-S, CCS heat exchanger C, and the train 1B flow path to the associated Unit 1 loads. CCS train 1B cooling must be provided for 24 hours following an initiating event.

  - **Model Boundaries.** This event models the availability of CCS pump C-S, CCS heat exchanger C, and the train 1B and 2B flow path to the associated Unit 1 and Unit 2 loads. Pump C-S is assumed to be normally running. It also receives an ESFAS signal to start on a safety injection condition. Individual cooling water loads are modeled with the respective system served.

  - **Important Intersystem Dependencies.** Train B of CCS (i.e., Top Event BC) is guaranteed failed if the initiator involves a loss of CCS or, if the CCS pump area ventilation system modeled via Top Event V3, is failed. ERCW header 2B supplies train B of CCS; i.e., Top Event BC fails if Top Event HE or FE fails. Loss of AC power from the Unit 2 480V shutdown board 2B2-B, or of 125V DC power battery board IV can also cause failure of train B of CCS.

- **Top Event CCPR — Align CCP A to ERCW Train A on Loss of CCS Train A**

  - **Function Evaluated.** This top event models the action for supplying alternate lube oil cooling to centrifugal charging pump A in the event of a loss of CCS train A cooling.

  - **Success Criteria.** The operators must manually trip CCP A prior to it overheating on loss of lube oil cooling, align ERCW header 1A flow to the lube oil heat exchanger, and then restart the pump. Ten minutes are available to trip the pump.

  - **Model Boundaries.** This analysis models the action to align alternate cooling to centrifugal charging pump A in the event that cooling from CCS train A, as modeled via Top Event AC, is unavailable. CCP A is the only high pressure injection with this backup cooling capability. The action is HCCSR2. To perform the alignment, ERCW flow from header 1A (Top Event CE) must be available. Three MOVs must be closed and one opened to complete the alignment.

  - **Important Intersystem Dependencies.** This action is only effective if ERCW header 1A is available (i.e., Top event CE is successful), and power is available to operate centrifugal charging pump A.

- **Top Event RW — Refueling Water Storage Tank**

  - **Function Evaluated.** Top Event RW models the availability of the RWST to supply borated water to portions of the emergency core cooling system (ECCS) and to the containment spray system during its injection phase following a LOCA initiating event.

  - **Success Criteria.** Success of Top Event RW requires the RWST to meet any demands for high head core injection through the centrifugal charging pumps, intermediate injection through the safety injection pumps, and low head core injection via the residual heat removal pumps during the first 24 hours after plant trip. Success is assumed if the RWST supplies its Technical Specification required inventory, even if the pump requirements exceed this amount of inventory in the first 24 hours.

  - **Model Boundaries.** Success of Top Event RW depends on the structural integrity of the RWST. A human action to terminate a safety injection signal prematurely is included in Top Event RW as a potential cause of failure and is evaluated in Section 3.3.3. While this action itself does not affect the RWST, it is conservative to model the effect of this action on the other safety injection systems by including it in the analysis for the RWST because all of the other safety injection systems also depend on the RWST.

    Individual demands on RWST inventory are modeled with the system served; e.g., the centrifugal charging pump suction line to the RWST is modeled with frontline Top Event VS. Inventory makeup to the RWST is modeled in Top Event MU.

  - **Important Intersystem Dependencies.** There are no supporting top events to RW. Failure of Top Event RW results in failure of the ECCS injection functions as well as the injection phase of containment spray. This event is assumed failed if the initiating event is a flood into the auxiliary building caused by failure of the RWST.

- **Top Event MU — Makeup to the RWST**

  - **Function Evaluated.** Makeup to the RWST, given leakage to the secondary side of a ruptured steam generator or given a small LOCA.

  - **Success Criteria.** Success requires the addition of borated water to the RWST for continued high pressure injection before the RWST empties.

  - **Model Boundaries.** This event models the operator action and equipment necessary to supply borated water makeup to the RWST during selected steam generator tube rupture and LOCA sequences. The makeup actions are directed by procedure when RWST level drops below 70% and the containment sump level is less than expected, indicating a loss of RCS inventory outside containment.

Currently, credit for makeup to the RWST is taken for steam generator tube rupture events in which there is no other LOCA, a secondary valve on the ruptured steam generator fails to be isolated, but the operators successfully cool down and depressurize the RCS. In this case, recirculation from the sump is not available due to the loss of primary fluid to the environment. Closed-loop RHR may have failed due to the loss of both RHR pumps. The action considered in this model is identified as MU2. The analysis is documented in Section 3.3.3.

The makeup source is the primary water tank. Boron can be added to this tank via the boric acid tank. No other sources of water (e.g., other unit's RWST, spent fuel pit, holdup tank, or recirculation from the containment sump using containment spray) are considered in the current analysis.

Credit is also given for makeup to the RWST during small LOCAs with success of containment spray recirculation. In this case, flow from the spray pumps taking suction from the containment sump is, in part, diverted via a test line back to the RWST. This flow provides a continuous source of borated inventory for high pressure injection in the event that the RHR pumps are unavailable.

– **Important Intersystem Dependencies.** If makeup to the RWST is successful, continued RCS inventory control is available to makeup for the leakage of RCS out the unisolated, ruptured steam generator. Failure of this event implies that the RWST has been depleted, and that recirculation from the sump is unavailable due to the loss of RCS out the ruptured steam generator or the failure of both RHR pumps. Core damage then results.

  Success of makeup requires that the RWST retain its integrity for the 24-hour mission time; i.e., requires that Top Event RW is successful. Providing borated makeup from the primary water tank to the RWST requires the availability of 480V shutdown board 1A1-A (Top Event A1I), 125V DC battery board I (Top Event DA), and nonessential control air; i.e., Top Event PD. Failure of any of these support systems precludes successful makeup.

- **Top Event CT — Condensate Storage Tank A**

  – **Function Evaluated.** Top Event CT models the availability of condensate storage tank (CST) A as the water source for AFW.

  – **Success Criteria.** Success requires that the CST have greater than the technical specification limit of 200,000 gallons at the time of plant trip, and that the tank maintain its integrity for 24 hours.

  – **Model Boundaries.** The model for this top event includes the tank itself and locked-open manual valve 0-2-504 in the common suction line to the auxiliary feedwater pumps.

- **Important Intersystem Dependencies.** Failure of this top event implies that the alternate AFW suction paths from ERCW must function or AFW will fail due to a lack of water. This top event is guaranteed failed for auxiliary building floods in which the CST is the water source. Since alternate water sources are modeled in separate top events, there are no intersystem dependencies involving this top event. The ability to supply makeup to the CST for long-term operation is accounted for in Top Event CTMU.

• **Top Event CTMU — Long-Term Makeup to the Condensate Storage Tank**

- **Function Evaluated.** This top event questions the ability of the operators to provide makeup inventory to the CST, given that the normal steam dump path through the condenser hot well is unavailable.

- **Success Criteria.** Success requires that the operators align for long-term makeup to the CST for continued operation of AFW following a plant trip. The time available is 6 hours from the initial plant trip.

- **Model Boundaries.** This model considers makeup to the condensate storage tank from the 500,000-gallon demineralized water storage tank. This top event conservatively takes no credit for makeup from other water sources. The operator action modeled is identified as HACT1. The operator will perform a valve alignment between the demineralized water storage tank (DWST) and the CST. A booster pump is then used to transfer DWST water to the CST at a flow rate between 150 gpm and 500 gpm. The time available to accomplish this action is the time required to empty the CST, with allowances to negate vortexing to the AFW pumps, at which time an automatic switch to ERCW suction would occur. The backup action to align to the ERCW system for suction is considered separately in the frontline top events for the AFW pumps; i.e., in Top Events TP, MA, and MB.

  During AFW operation, makeup to the CST is accomplished via a path through the steam dumps to the condenser hotwell. The hotwell pumps then transfer water to the CST through the condenser level control valve (1-LCV-2-3). The CST has a storage capacity of 395,000 gallons of primary grade water, 210,533 of which is reserved for AFW suction by means of a standpipe. If the main steam isolation valves go closed, then water is not returned to the hotwell.

- **Important Intersystem Dependencies.** The operators are precluded from providing makeup to the CST if power from offsite is not available to the transfer pumps. No other intersystem dependencies are of interest. Failure to provide makeup to the CST, in the event that flow from the condenser hotwell is unavailable, may result in the eventual loss of AFW within the 24-hour mission time. However, the plant model assumes that the CST would not last 24 hours only if there is a failure of the reactor trip, which would require increased flows from AFW. The need for makeup to the CST within the 24-hour mission time assigned to AFW is considered in the frontline event trees as part of the split fraction assignment rules for AFW.

If CST makeup is required and not successful, consideration is then given to the alignment for ERCW as a suction source.

Table 3.1.4-1. Top Events Modeled in ELECT1 Support System Event Tree	
Name	**Description**
OG	Offsite Grid
OGR1	Recovery of Offsite Power in 1 Hour
FA/FB	Fuel Oil Transfer and Supply for Unit 1 Diesel Generators
GA/GB	Diesel Generators 1A-A and 1B-B
AA/BA	Unit 1 6.9-kV Shutdown Boards 1A-A and 1B-B
A1/A2/B1/B2	480V Shutdown Boards 1A1-A, 1A2-A, 1B1-B, and 1B2-B
VT1A/VT1B	480V Shutdown Transformer Rooms 1A and 1B Ventilation
VT1AR/VT1BR	Recovery of Transformer Rooms 1A and 1B Ventilation
DA/DB	125V DC Battery Boards I and II
DG	120V AC Instrument Power Board 1A
V1	Unit 1 Shutdown Board Ventilation
V1R	Recovery of Unit 1 Shutdown Board Room Ventilation
VINV1	Unit 1 480V Board Room B Ventilation
VNV1R	Recovery of Unit 1 480V Board Room B Ventilation
DAAC/DBAC	120V AC Power Subsystems 1-I and 1-II
A3/B3	6.9-kV Common Boards A and B and 480V Auxiliary Building Common Board Buses A and B
D1/D2	250V DC Boards 1 and 2
UB1A/UB1B/UB1C/ UB1D	Unit 1 6.9-kV Unit Boards 1A, 1B, 1C, and 1D

Table 3.1.4-2. Top Events Modeled in ELECT2 Support System Event Tree	
Name	**Description**
FC/FD	Fuel Oil Transfer and Supply for Unit 2 Diesel Generators
GC/GD	Diesel Generators 2A-A and 2B-B
AB/BB	Unit 2 6.9-kV Shutdown Boards 2A-A and 2B-B
A1U2/A2U2/B1U2/ B2U2	480V Shutdown Boards 2A1-A, 2A2-A, 2B1-B, and 2B2-B
VT2A/VT2B	480V Shutdown Transformer Rooms 2A and 2B Ventilation
VT2AR/VT2BR	Recovery of Transformer Rooms 2A and 2B Ventilation
DC/DD	125V DC Battery Boards III and IV
DH	120V AC Instrument Power Board 2A
V2	Unit 2 Shutdown Board Ventilation
V2R	Recovery of Unit 2 Shutdown Board Room Ventilation
VINV2	Unit 2 480V Board Room B Ventilation
VNV2R	Recovery of Unit 2 480V Board Room B Ventilation
DCAC/DDAC	120V AC Power Subsystems 1-III and 1-IV

Table 3.1.4-3 (Page 1 of 2). Top Events Modeled in MECH Support System Event Tree	
Name	**Description**
ZA	ESFAS Train A
ZB	ESFAS Train B
OS	Manual Actions To Back Up ESFAS
AE	Essential Raw Cooling Water Train A Pumps
BE	Essential Raw Cooling Water Train B Pumps
MDE	Maintenance on ERCW Header 1B
CE	ERCW Header 1A-A
EE	ERCW Header 2A-A
DE	ERCW Header 1B-B
FE	ERCW Header 2B-B
DSLR	Recovery of ERCW to Diesel from Opposite Side
GE	ERCW Discharge Header A
HE	ERCW Discharge Header B
AAL	ERCW/Diesel 1A/6.9-kV Shutdown Board 1A-A Dependency
A1L	ERCW/Diesel 1A/480V Shutdown Board 1A1-A Dependency
A2L	ERCW/Diesel 1A/480V Shutdown Board 1A2-A Dependency
ABL	ERCW/Diesel 2A/6.9-kV Shutdown Board 2A-A Dependency
A1U2L	ERCW/Diesel 2A/480V Shutdown Board 2A1-A Dependency
A2U2L	ERCW/Diesel 2A/480V Shutdown Board 2A2-A Dependency
BAL	ERCW/Diesel 1B/6.9-kV Shutdown Board 1B-B Dependency
B1L	ERCW/Diesel 1B/480V Shutdown Board 1B1-B Dependency
B2L	ERCW/Diesel 1B/480V Shutdown Board 1B2-B Dependency
BBL	ERCW/Diesel 1B/6.9-kV Shutdown Board 2B-B Dependency
B1U2L	ERCW/Diesel 2B/480V Shutdown Board 2B1-B Dependency
B2U2L	ERCW/Diesel 2B/480V Shutdown Board 2B2-B Dependency
PE	ERCW Cooling to Control Air System Compressors
PD	Nonessential Control Air System
PA	Auxiliary Control Air System Train A

Table 3.1.4-3 (Page 2 of 2). Top Events Modeled in MECH Support System Event Tree	
Name	**Description**
PB	Auxiliary Control Air System Train B
V3	Component Cooling Water System (CCS) Pumps and Motor-Driven Auxiliary Feedwater Pump Ventilation
CSSR	Recovery of CSS by Crosstieing
AC	Component Cooling System Train A
BC	CCS Train B
CCPR	Align CCPH to ERCW Train A on Loss of CCS Train A
RW	Refueling Water Storage Tank
MU	Makeup to the RWST
CT	Condensate Storage Tank A
CTMU	Long-Term Makeup to the Condensate Storage Tank

Page No. 1

MODEL Name: WBN
Event Tree: ELECT1

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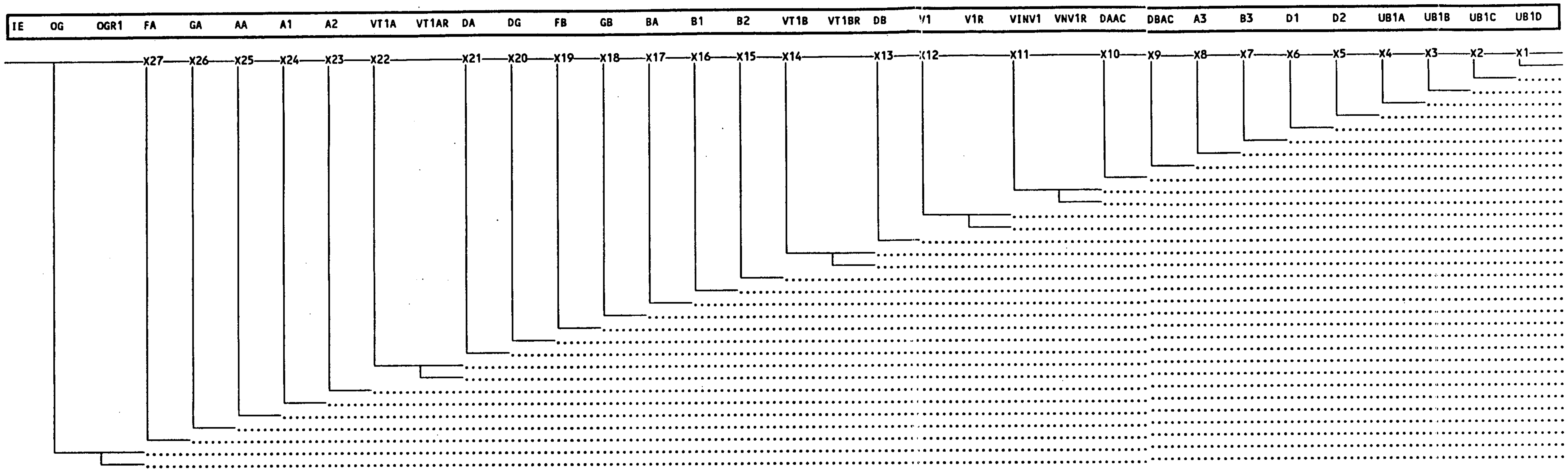

1		1
2		2
3	X1	3-4
4	X2	5-8
5	X3	9-16
6	X4	17-32
7	X5	33-64
8	X6	65-128
9	X7	129-256
10	X8	257-512
11	X9	513-1024
12	X10	1025-2048
13	X10	2049-3072
14	X11	3073-6144
15	X11	6145-9216
16	X12	9217-18432
17	X13	18433-36864
18	X13	36865-55296
19	X14	55297-110592
20	X15	110593-221184
21	X16	221185-442368
22	X17	442369-884736
23	X18	884737-1769472
24	X19	1769473-3538944
25	X20	3538945-7077888
26	X21	7077889-14155776
27	X21	14155777-21233664
28	X22	21233665-42467328
29	X23	42467329-84934656
30	X24	84934657-169869312
31	X25	169869313-339738624
32	X26	339738625-679477248
33	X27	679477249-1358954496
34	X27	1358954497-2038431744

Top Event Designator.....	Top Event Description............................
IE	Initiating Event
OG	161KV OFFSITE POWER
OGR1	RECOVERY OF OFFSITE POWER IN 1 HOUR
FA	UNIT 1 TRAIN A DIESEL FUEL OIL
GA	UNIT 1 DIESEL 1A-A
AA	6.9KV SD BD UNIT 1 TRAIN A
A1	480V SHUTDOWN BOARD 1A1-A
A2	480V SHUTDOWN BOARD 1A2-A
VT1A	480V SD TRANSFORMER ROOM 1A VENTILATION
VT1AR	RECOVERY OF TRANSFORMER ROOM 1A VENTILATION IN 10 HOURS
DA	125V DC BATTERY BD I
DG	120V AC INSTRUMENT UNIT 1 POWER
FB	UNIT 1 TRAIN B DIESEL FUEL OIL
GB	UNIT 1 DIESEL 1B-B
BA	6.9KV SD BD UNIT 1 TRAIN B
B1	480V SHUTDOWN BOARD 1B1-B
B2	480V SHUTDOWN BOARD 1B2-B
VT1B	SD TRANSFORMER ROOM 1B VENTILATION
VT1BR	RECOVERY OF TRANSFORMER ROOM 1B VENTILATION IN 5 HOURS
DB	125V DC BATTERY BD II
V1	UNIT 1 SHUTDOWN BOARD VENTILATION
V1R	RECOVERY OF UNIT 1 SHUTDOWN BD ROOM VENTILATION IN 12 HOURS
VINV1	480V SHUTDOWN BOARD ROOM VENTILATION B
VNV1R	RECOVERY OF 480 V BD RM VENTILATION B IN 6 HOURS
DAAC	120V VITAL INST. CHANNEL 1-I
DBAC	120V VITAL INST. CHANNEL 1-II
A3	480V COMMON BOARD A
B3	480V COMMON BOARD B
D1	250V DC BUS I
D2	250V DC BUS II
UB1A	6.9KV UNIT BOARD 1A
UB1B	6.9KV UNIT BOARD 1B
UB1C	6.9KV UNIT BOARD 1C
UB1D	6.9KV UNIT BOARD 1D

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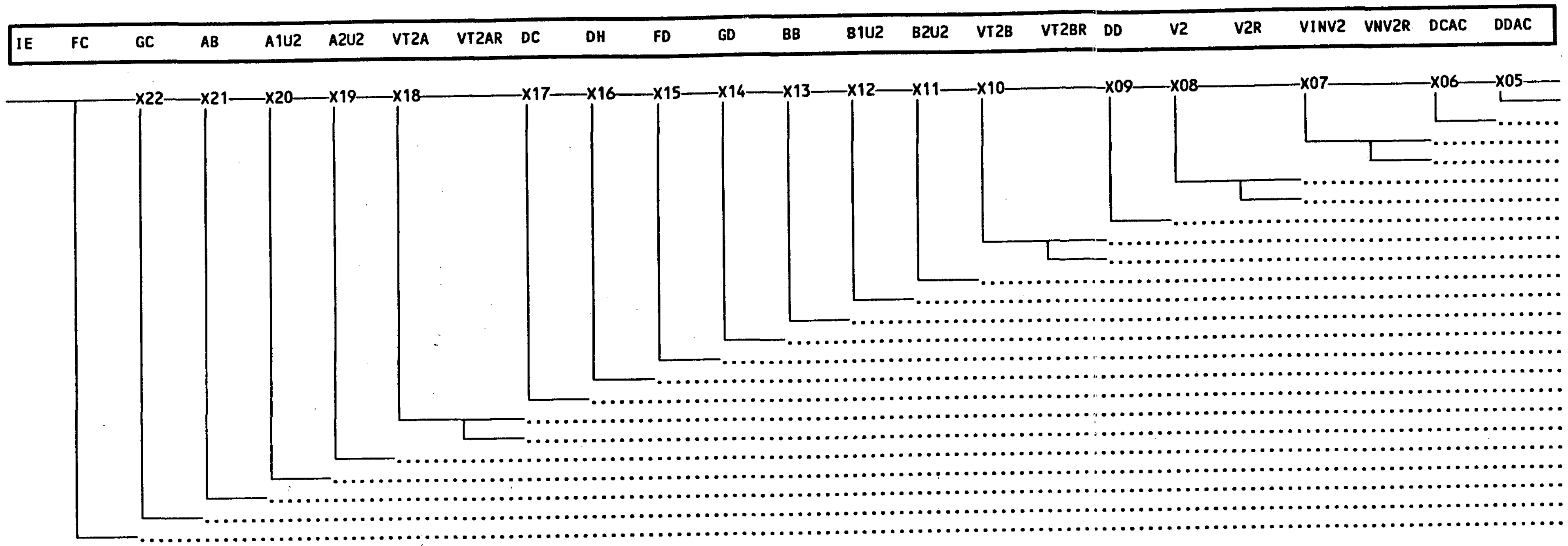
Figure 3.1.4-1.
Watts Bar ELECT1 Event Tree

Page No. 1

MODEL Name: WBN
Event Tree: ELECT2

11:15:56 20 JUL 1992



1		1
2		2
3	X05	3-4
4	X06	5-8
5	X06	9-12
6	X07	13-24
7	X07	25-36
8	X08	37-72
9	X09	73-144
10	X09	145-216
11	X10	217-432
12	X11	433-864
13	X12	865-1728
14	X13	1729-3456
15	X14	3457-6912
16	X15	6913-13824
17	X16	13825-27648
18	X17	27649-55296
19	X17	55297-82944
20	X18	82945-165888
21	X19	165889-331776
22	X20	331777-663552
23	X21	663553-1327104
24	X22	1327105-2654208

APERTURE CARD

Also Available On Aperture Card

Top Event Designator.....	Top Event Description............................
IE	Initiating Event
FC	UNIT 2 TRAIN A DIESEL FUEL OIL
GC	UNIT 2 DIESEL 2A-A
AB	6.9KV SD BD TRAIN A UNIT 2
A1U2	UNIT 2 480V SD BD 2A1-A
A2U2	UNIT 2 480V SD BD 2A2-A
VT2A	480V SD TRANSFORMER ROOM 2A VENTILATION
VT2AR	RECOVERY OF 480V SD TRANSFORMER 2A VENTILATION IN 10 HOURS
DC	125V DC BATTERY BD III
DH	UNIT 2 120V AC INSTRUMENT POWER
FD	UNIT 2 TRAIN B DIESEL FUEL OIL
GD	UNIT 2 DIESEL 2B-B
BB	6.9KV UNIT 2 SHUTDOWN BOARD 2BB
B1U2	UNIT 2 480V SD BD 2B1-B
B2U2	UNIT 2 480V SD BD 2B2-B
VT2B	480V SD TRANSFORMER ROOM 2B VENTILATION
VT2BR	RECOVERY OF 480 V TRANSFORMER ROOM VENTILATION IN 5 HOURS
DD	125V DC BATTERY BD IV
V2	UNIT 2 SHUTDOWN BOARD VENTILATION SYSTEM
V2R	RECOVERY OF UNIT 2 SHUTDOWN BD VENTILATION IN 12 HOURS
VINV2	480V SDBR VENTILATION U-2 B
VNV2R	RECOVERY OF 480 V SDBR U-2 B VENTILATION IN 6 HOURS
DCAC	120 V AC VITAL BD 1-III
DDAC	120 V AC VITAL BD 1-IV

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Figure 3.1.4-2.
Watts Bar ELECT2 Event Tree

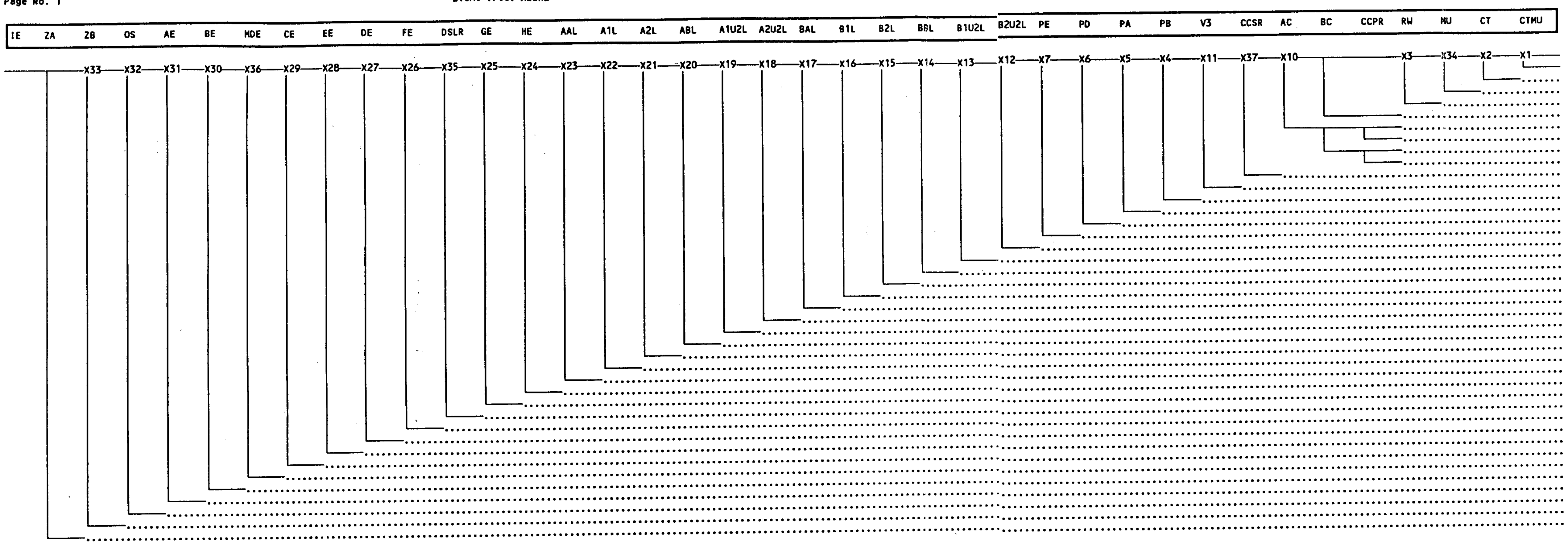
SI
APERTURE
CARD

Also Available On
Aperture Card

Page No. 1

MODEL Name: WBN
Event Tree: MECH2

11:23:19 20 JUL 1992



1		1
2		2
3	X1	3-4
4	X2	5-8
5	X34	9-16
6	X3	17-32
7	X3	33-48
8	X3	49-64
9	X3	65-80
10	X3	81-96
11	X10	97-192
12	X37	193-384
13	X11	385-768
14	X4	769-1536
15	X5	1537-3072
16	X6	3073-6144
17	X7	6145-12288
18	X12	12289-24576
19	X13	24577-49152
20	X14	49153-98304
21	X15	98305-196608
22	X16	196609-393216
23	X17	393217-786432
24	X18	786433-1572864
25	X19	1572865-3145728
26	X20	3145729-6291456
27	X21	6291457-12582912
28	X22	12582913-25165824
29	X23	25165825-50331648
30	X24	50331649-100663296
31	X25	100663297-201326592
32	X35	201326593-402653184
33	X26	402653185-805306368
34	X27	805306369-1610612736
35	X28	1610612737-3221225472
36	X29	3221225473-6442450944
37	X36	6442450945-12884901888
38	X30	12884901889-25769803776
39	X31	25769803777-51539607552
40	X32	51539607553-103079215104
41	X33	103079215105-206158430208

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
ZA	TRAIN A ESFAS
ZB	TRAIN B ESFAS
OS	MANUAL OPERATOR BACKUP OF ESFAS ALIGNMENTS
AE	ERCW TRAIN A PUMPS
BE	ERCW TRAIN B PUMPS
MDE	MAINTENANCE ON ERCW HEADER 1B
CE	ERCW HEADER 1A
EE	ERCW HEADER 2A
DE	ERCW HEADER 1B
FE	ERCW HEADER 2B
DSLR	RECOVERY OF ERCW TO DIESEL FROM OPPOSITE SIDE
GE	ERCW DISCHARGE HEADER A
HE	ERCW DISCHARGE HEADER B
AAL	ERCW/DIESEL 1A/6.9 SHUTDOWN BD 1A-A DEPENDENCY TOP
A1L	ERCW/DIESEL 1A/480V SHUTDOWN BD 1A1-A DEPENDENCY
A2L	ERCW/DIESEL 1A/480V SHUTDOWN BD 1A1-A DEPENDENCY
ABL	ERCW/DIESEL 2A/6.9 SHUTDOWN BD 2A-A DEPENDENCY
A1U2L	ERCW/DIESEL 2A/480V SHUTDOWN BD 2A1-A DEPENDENCY
A2U2L	ERCW/DIESEL 2A/480V SHUTDOWN BD 2A2-A DEPENDENCY
BAL	ERCW/DIESEL 1B/6.9 SHUTDOWN BD 1B-B DEPENDENCY
B1L	ERCW/DIESEL 1B/480V SHUTDOWN BD 1B1-B DEPENDENCY
B2L	ERCW/DIESEL 1B/480V SHUTDOWN BD 1B2-B DEPENDENCY
BBL	ERCW/DIESEL 2B/6.9 SHUTDOWN BD 2B-B DEPENDENCY
B1U2L	ERCW/DIESEL 2B/480V SHUTDOWN BD 2B1-B DEPENDENCY
B2U2L	ERCW/DIESEL 2B/480V SHUTDOWN BD 2B2-B DEPENDNECY
PE	ERCW COOLING TO CAS COMPRESSORS
PD	CONTROL AIR (NON-ESSENTIAL AIR)
PA	TRAIN A ESSENTIAL AIR
PB	TRAIN B ESSENTIAL AIR
V3	CCS PUMPS AND MOTOR DRIVEN AUXILIARY FEEDWATER VENTILATION
CCSR	RECOVER CCS HTX BY REALIGNING ERCW
AC	TRAIN A COMPONENT COOLING WATER SYSTEM
BC	TRAIN B COMPONENT COOLING WATER SYSTEM
CCPR	RECOVER CCP A BY ALIGNING ERCW A TRAIN ON LOSS OF CCS A
RW	REFUELING WATER STORAGE TANK (RWST)
MU	MAKEUP TO RWST
CT	CONDENSATE STORAGE TANK (CST)
CTMU	MAKEUP TO THE CST

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Figure 3.1.4-3.
Watts Bar MECH2 Event Tree

3.1.5 SEQUENCE GROUPING AND BACK-END INTERFACES

The sequence grouping and back-end interfaces are provided as part of Section 4.3, Plant Damage States.

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SEE APERTURE CARDS

NUMBER OF OVERSIZE PAGES FILMED ON APERTURE CARDS 2

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TABLE 3.2-3
WATTS BAR SUPPORT TO SUPPORT DEPENDENCY MATRIX

Column groups (upper axis, left to right): OFF SITE PWR SOURCES 161 KV (OG); 6.9 KV UNIT BOARDS (1A UB1A, 1B UB1B, 1C UB1C, 1D UB1D); 480V UNIT BOARDS (1A UB1A, 1B UB1D); 6.9KV & 480V AUX BLDG COM BOARDS (BUS A A3, BUS B B3); 6.9 KV SHUTDOWN BOARDS (1A-A AA, 1B-B BA, 2A-A AB, 2B-B BB); 480V SHUTDOWN BOARDS (1A1-A A1, 1A2-A A2, 1B1-B B1, 1B2-B B2, 2A1-A A1U2, 2A2-A A2U2, 2B1-B B1U2, 2B2-B B2U2); 480V REACTOR MOV BOARD (1A1-A A1, 1A2-A A2, 1B1-B B1, 1B2-B B2, 2A1-A A1U2, 2A2-A A2U2, 2B1-B B1U2, 2B2-B B2U2); 480V CONTROL AND AUXILIARY BUILDING VENT BOARD (1A1-A A1, 1A2-A A2, 1B1-B B1, 1B2-B B2, 2A1-A A1U2, 2A2-A A2U2, 2B1-B B1U2, 2B2-B B2U2); 480V DIESEL AUX BOARD (1A1-A A1, 1A2-A A2, 1B1-B B1, 1B2-B B2, 2A1-A A1U2, 2A2-A A2U2, 2B1-B B1U2, 2B2-B B2U2); 480V REACTOR VENT BOARD (1A-A A1, 1B-B B1, 2A-A A1U2, 2B-B B1U2); 250 VDC BATTERY BOARD (1 D1, 2 D2); 125 VDC VITAL BATTERY BOARD (I DA, II DB, III DC, IV DD); 120 VAC VITAL INSTR POWER BOARD (I-I DAAC, I-II DBAC, I-III DCAC, I-IV DDAC); 120 VAC INSTR POWER (1A DG); DIESELS GENERATORS (1A GA, 1B GB, 2A GC, 2B GD); FUEL OIL SYSTEM (1A-A FA, 1B-B FB, 2A-A FC, 2B-B FD); SSPS/ESFAS (TRAIN A ZA, TRAIN B ZB); SHUTDOWN BD ROOM VENTILATION (V1, V2); 6.9 KV SD TO 480V SD TRANS ROOM VENT (1A VT1A, 2A VT2A, 1B VT1B, 2B VT2B); 480V BD ROOM VENT (1B VINV1, 2B VINV2); CCS/AFW EQPT VENT (V3); CONTROL AIR SYSTEM (ESS A PA, ESS B PB, NON ESS PD); ESSENTIAL RAW CLG WATER (TR-A PUMPS AE, TR-B PUMPS BE); COMPONENT COOLING SYSTEM (PUMP 1A-A 1B-B AC, PUMP C-S BC).

Marks read along each row (top event, left axis):

Top event	Top event name	Marked columns
161 KV GRID SWYD	OG	OG X; UB1A X; UB1B X; UB1C X; UB1D X; A3 X; B3 X; AA X; BA X; AB X; BB X
START BUS A		UB1A X; UB1C X; A3 X
START BUS B		UB1B X; UB1D X; B3 X
CSST A		OG X; UB1A X; UB1C X; A3 X; B3 X
CSST B		OG X; UB1B X; UB1D X; A3 X; B3 X
CSST C		OG X; AA X; AB X
CSST D		OG X; BA X; BB X
USST 1A		UB1A X; UB1B X
USST 1B		UB1C X; UB1D X
6.9 KV UNIT BD 1A	UB1A	UB1A X; 480V UB1A X
6.9 KV UNIT BD 1B	UB1B	UB1B X
6.9 KV UNIT BD 1C	UB1C	UB1C X
6.9 KV UNIT BD 1D	UB1D	UB1D X; 480V UB1D X
480V UNIT BD 1A	UB1A	480V UB1A X
480V UNIT BD 1B	UB1D	480V UB1D X
6.9 KV COMMON BD A	A3	A3 X; V3 5
6.9 KV COMMON BD B	B3	B3 X; V3 5
480 V AUX COM BD. BUS A	A3	A3 X; PD 6
480 V AUX COM BD. BUS B	B3	B3 X; PD 6; VT1A X; VT2B X
6.9 KV SD BD 1A-A	AA	AA X; SD A1 X; SD A2 X; AE 7
6.9 KV SD BD 1B-B	BA	BA X; SD B1 X; SD B2 X; BE 7
6.9 KV SD BD 2A-A	AB	AB X; SD A1U2 X; SD A2U2 X; AE 7
6.9 KV SD BD 2B-B	BB	BB X; SD B1U2 X; SD B2U2 X; DG X; BE 7
480 V SD BD 1A1-A	A1	SD A1 X; MOV A1 X; C&A A1 X; V1 X; AC 5
480 V SD BD 1A2-A	A2	SD A2 X; MOV A2 X; C&A A2 X; PD 6
480 V SD BD 1B1-B	B1	SD B1 X; MOV B1 X; DBAC X; VINV1 X; PD 6; AC 5
480 V SD BD 1B2-B	B2	SD B2 X; MOV B2 X; DCAC X
480 V SD BD 2A1-A	A1U2	SD A1U2 X; MOV A1U2 X
480 V SD BD 2A2-A	A2U2	SD A2U2 X; MOV A2U2 X; V2 X
480 V SD BD 2B1-B	B1U2	SD B1U2 X; MOV B1U2 X; DDAC X; VINV2 X
480 V SD BD 2B2-B	B2U2	SD B2U2 X; MOV B2U2 X; V2 X; BC 5
480 V RMOV BD 1A1-A	A1	MOV A1 X
480 V RMOV BD 1A2-A	A2	MOV A2 X; AC 5
480 V RMOV BD 1B1-B	B1	MOV B1 X
480 V RMOV BD 1B2-B	B2	MOV B2 X
480 V RMOV BD 2A1-A	A1U2	MOV A1U2 X
480 V RMOV BD 2A2-A	A2U2	MOV A2U2 X; AC 5
480 V RMOV BD 2B1-B	B1U2	MOV B1U2 X
480 V RMOV BD 2B2-B	B2U2	MOV B2U2 X
480 V C&A VENT BD 1A1-A	A1	C&A A1 X; VT1A X; V3 5
480 V C&A VENT BD 1A2-A	A2	C&A A2 X
480 V C&A VENT BD 1B1-B	B1	C&A B1 X; VT1B X; V3 5
480 V C&A VENT BD 1B2-B	B2	C&A B2 X
480 V C&A VENT BD 2A1-A	A1U2	C&A A1U2 X; VT2A X; V3 5; PA 6
480 V C&A VENT BD 2A2-A	A2U2	C&A A2U2 X
480 V C&A VENT BD 2B1-B	B1U2	C&A B1U2 X; VT2B X; V3 5; PB 6
480 V C&A VENT BD 2B2-B	B2U2	C&A B2U2 X
480 V DSL AUX BD 1A1-A	A1	DSL A1 X; GA X
480 V DSL AUX BD 1A2-A	A2	DSL A2 X; GA X
480 V DSL AUX BD 1B1-B	B1	DSL B1 X; GB X
480 V DSL AUX BD 1B2-B	B2	DSL B2 X; GB X
480 V DSL AUX BD 2A1-A	A1U2	DSL A1U2 X; GC X
480 V DSL AUX BD 2A2-A	A2U2	DSL A2U2 X; GC X
480 V DSL AUX BD 2B1-B	B1U2	DSL B1U2 X; GD X
480 V DSL AUX BD 2B2-B	B2U2	DSL B2U2 X; GD X
480 V RV BD 1A-A	A1	RV A1 X
480 V RV BD 1B-B	B1	RV B1 X
480 V RV BD 2A-A	A1U2	RV A1U2 X
480 V RV BD 2B-B	B1U2	RV B1U2 X
250 VDC BATT BD 1	D1	[illegible]; D1 X
250 VDC BATT BD 2	D2	[illegible]; D2 X
125 VDC BATT BD I	DA	[illegible]; DA X; AE 7; AC 5
125 VDC BATT BD II	DB	[illegible]; DB X; PD 6; BE 7; AC 5
125 VDC BATT BD III	DC	[illegible]; DC X
125 VDC BATT BD IV	DD	[illegible]; DD X
120 VAC INST PWR I-I	DAAC	[illegible]; DAAC X
120 VAC INST PWR I-II	DBAC	[illegible]; DBAC X
120 VAC INST PWR I-III	DCAC	[illegible]; DCAC X
120 VAC INST PWR I-IV	DDAC	[illegible]; DDAC X
120 VAC INST PWR 1A	DG	[illegible]; DG X
DIESEL 1A	GA	[illegible]; GA X
DIESEL 1B	GB	[illegible]; GB X
DIESEL 2A	GC	[illegible]; GC X
DIESEL 2B	GD	[illegible]; GD X
FUEL OIL SYS 1A-A	FA	FA X
FUEL OIL SYS 1B-B	FB	FB X
FUEL OIL SYS 2A-A	FC	FC X
FUEL OIL SYS 2B-B	FD	FD X
SSPS/ESFAS A	ZA	ZA X
SSPS/ESFAS B	ZB	ZB X
UNIT 1 SHUTDOWN BD VENTILATION	V1	[illegible]; V1 X
UNIT 2 SHUTDOWN BD VENTILATION	V2	[illegible]; V2 X
480V TRANS ROOM VENTILATION 1A	VT1A	[illegible]; VT1A X
480V TRANS ROOM VENTILATION 2A	VT2A	[illegible]; VT2A X
480V TRANS ROOM VENTILATION 1B	VT1B	[illegible]; VT1B X
480V TRANS ROOM VENTILATION 2B	VT2B	[illegible]; VT2B X
480V BD ROOM VENTILATION 1B	VINV1	[illegible]; VINV1 X
480V BD ROOM VENTILATION 2B	VINV2	[illegible]; VINV2 X
CCS/AFW EQUIP VENTILATION	V3	V3 5
ESSENTIAL AIR A	PA	PA 6
ESSENTIAL AIR B	PB	PB 6
NONESSENTIAL AIR	PD	[illegible]; PD 6
ERCW TR-A PUMPS	AE	[illegible]; AE 6
ERCW TR-B PUMPS	BE	[illegible]; BE 7
COMP COOL SYS A	AC	AC 5
COMP COOL SYS B	BC	BC 5

WBN/SUPTOSUP/D.R00

NOTES

1. TO READ, LOCATE THE SYSTEM/COMPONET ON THE UPPER AXIS AND READ DOWN TO IDENTIFY THE REQUIRED SUPPORTS ON THE LEFT AXIS
2. ALTERNATE POWER/CONTROL POWER FEEDS(AC, DC, CHARGERS, BATTERY BOARDS, BATTERIES) THAT ARE NOT AUTOMATICALLY SWITCHED ARE NOT DESIGNATED ON THIS MATRIX. THE ALTERNATE FEEDS ARE HOWEVER DISCUSSED IN THE APPROPRIATE MATRIX NOTE
3. THIS IS NOT A DESIGN OUTPUT DRAWING AND IS NOT TO BE USED AS DESIGN INPUT
4. REFER TO TABLE 1 OF THE EMERGENCY SAFETY FEATURES ACTUATION SYSTEM NOTEBOOK FOR FURTHER INFORMATION
5. REFER TO TABLE 1 OF THE COMPONENT COOLING SYSTEM NOTEBOOK FOR FURTHER INFORMATION
6. REFER TO TABLE 1 OF THE CONTROL AIR SYSTEM NOTEBOOK FOR FURTHER INFORMATION
7. REFER TO TABLE 1 OF THE ESSENTAIL RAW COOLING WATER SYSTEM NOTEBOOK FOR FURTHER INFORMATION
X REFER TO TABLE 1 OF THE ELECTIC POWER SYSTEM NOTEBOOK FOR FURTHER INFORMATION

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TABLE 3.2-4

WATTS BAR SUPPORT TO FRONTLINE DEPENDENCY MATRIX

SECONDARY SIDE CORE COOLING

SUPPORT SYSTEM FAILED	CONDENSATE/FEEDWATER				STEAM GENERATOR PORVS				STEAM GENERATOR MSIVS/BYPASS VALVES				TURBINE TRIP/RUNBACK	STEAM GEN COND DUMPS	AUXILIARY FEEDWATER					
	CNDS PMPS & VALVES	STANDBY MFWP	FEEDWATER VALVES	HOTWELL PUMPS AND VALVES	LOOP 1 PCV-1-5	LOOP 2 PCV-1-12	LOOP 3 PCV-1-23	LOOP 4 PCV-1-30	LOOP 1 FCV-1-4 FCV-1-147	LOOP 2 FCV-1-11 FCV-1-148	LOOP 3 FCV-1-22 FCV-1-149	LOOP 4 FCV-1-29 FCV-1-150	TURBINE VALVES	STEAM DUMPS	1A-S TDAFW PUMP	AFW VALVES	1A-A MDAFW PUMP	MDAFW PUMP 1A-A HDR/VLVS	1B-B MDAFW PUMP	MDAFW PUMP 1B-B HDR/VLVS
6.9 KV UNIT BD 1A	8			10																
6.9 KV UNIT BD 1B																				
6.9 KV UNIT BD 1C																				
6.9 KV UNIT BD 1D		8																		
6.9 KV SD BD 1A-A																	9			
6.9 KV SD BD 1B-B																			9	
480 V SD BD 1A1-A																				
480 V SD BD 1A2-A																				
480 V SD BD 1B1-B																				
480 V SD BD 1B2-B																				
480 V RX MOV BD 1A1-A																				
480 V RX MOV BD 1A2-A																9		9		
480 V RX MOV BD 1B1-B																				
480 V RX MOV BD 1B2-B																9				9
480 V RX MOV BD 2A1-A																				
480 V RX MOV BD 2A2-A																				
480 V RX MOV BD 2B1-B																				
480 V RX MOV BD 2B2-B																				
480 V C&A VENT BD 1A1-A																				
480 V C&A VENT BD 1A2-A																				
480 V C&A VENT BD 1B1-B																				
480 V C&A VENT BD 1B2-B																				
480 V C&A VENT BD 2A1-A																				
480 V C&A VENT BD 2A2-A																				
480 V C&A VENT BD 2B1-B																				
480 V C&A VENT BD 2B2-B																				
480 V UNIT BD 1A - #19	8		8																	
480 V UNIT BD 1B - #19	8	8	8																	
480 V UNIT BD 2A																				
480 V UNIT BD 2B																				
250 VDC BATT BD I	8												10							
250 VDC BATT BD II		8											10							
125 VDC BATT BD I		8	8				9		10	10	10	10		10			9			
125 VDC BATT BD II			8			9			10	10	10	10		10					9	
125 VDC BATT BD III					9										9	9				
125 VDC BATT BD IV								9								9				
120 VAC INST PWR 1-I					9		9							10	9	9	9			
120 VAC INST PWR 1-II						9		9						10	9	9			9	
120 VAC INST PWR 1-III														10		9				
120 VAC INST PWR 1-IV														10		9				
120 VAC INST PWR DIST PNL 1A															9					
120 VAC INST PWR DIST PNL 1B																				
ESFAS A/SSPS S									10	10	10	10	10		9		9			
ESFAS B/SSPS B									10	10	10	10	10		9				9	
CCS/AFW EQUIP COOLERS																	9		9	
CONT BLDG ELECT BD ROOM COOLING																				
ESSENTIAL AIR A					9		9											9		
ESSENTIAL AIR B						9		9												9
NONESSENTIAL AIR	8			10					10	10	10	10		10						
ERCW 1A-A															9		9			
ERCW 1B-B															9				9	
ERCW 2A																				
ERCW 2B																				
COMP COOL SYS A																				
COMP COOL SYS B																				
UNIT 1 CST															9		9		9	
RCW	8			10																
RWST																				
CONT SUMP																				
AMSAC													10		9		9		9	

REACTOR SHUTDOWN / PRIMARY SIDE CORE COOLING

SUPPORT SYSTEM FAILED	REACTOR TRIP BREAKERS				EMERGENCY BORATION			PRESSURIZER PORVS/BLOCKS/SPRAY			REACTOR COOLANT PUMP AND SEALS	
	RTA	BYA	RTB	BYB	BAT A HEATERS HT TRACE	BA TRANSFER PUMP 1A-A	B A HDR/VLVS	PCV-68-340A PCV-68-333	PCV-68-334 PCV-68-332	SPRAY VALVES PCV-68-340B PCV-68-340D	THERMAL BARRIERS	RCP SEALS
6.9 KV UNIT BD 1A												
6.9 KV UNIT BD 1B												
6.9 KV UNIT BD 1C												
6.9 KV UNIT BD 1D												
6.9 KV SD BD 1A-A												
6.9 KV SD BD 1B-B												
480 V SD BD 1A1-A												
480 V SD BD 1A2-A												
480 V SD BD 1B1-B												
480 V SD BD 1B2-B												
480 V RX MOV BD 1A1-A					12	12		13			14	
480 V RX MOV BD 1A2-A											14	
480 V RX MOV BD 1B1-B					12		12		13		14	
480 V RX MOV BD 1B2-B											14	
480 V RX MOV BD 2A1-A												
480 V RX MOV BD 2A2-A												
480 V RX MOV BD 2B1-B												
480 V RX MOV BD 2B2-B												
480 V C&A VENT BD 1A1-A											14	
480 V C&A VENT BD 1A2-A												
480 V C&A VENT BD 1B1-B											14	
480 V C&A VENT BD 1B2-B												
480 V C&A VENT BD 2A1-A					12							
480 V C&A VENT BD 2A2-A												
480 V C&A VENT BD 2B1-B					12							
480 V C&A VENT BD 2B2-B												
480 V UNIT BD 1A - #19												
480 V UNIT BD 1B - #19												
480 V UNIT BD 2A												
480 V UNIT BD 2B												
250 VDC BATT BD I												14
250 VDC BATT BD II												14
125 VDC BATT BD I	11	11						13				
125 VDC BATT BD II			11	11					13			
125 VDC BATT BD III												
125 VDC BATT BD IV												
120 VAC INST PWR 1-I					12			13		13		
120 VAC INST PWR 1-II					12				13	13		
120 VAC INST PWR 1-III								13				
120 VAC INST PWR 1-IV									13			
120 VAC INST PWR DIST PNL 1A												
120 VAC INST PWR DIST PNL 1B												
ESFAS A/SSPS S	11			11								
ESFAS B/SSPS B		11	11									
CCS/AFW EQUIP COOLERS												
CONT BLDG ELECT BD ROOM COOLING												
ESSENTIAL AIR A										13		
ESSENTIAL AIR B										13		
NONESSENTIAL AIR												
ERCW 1A-A											14	
ERCW 1B-B											14	
ERCW 2A												
ERCW 2B												
COMP COOL SYS A											14	14
COMP COOL SYS B												
UNIT 1 CST												
RCW												
RWST						12						
CONT SUMP												
AMSAC												

PRIMARY SIDE CORE COOLING / CONTAINMENT SYSTEMS

SUPPORT SYSTEM FAILED	CHEMICAL VOLUME CONTROL SYSTEM				SAFETY INJECTION SYSTEM				RESIDUAL HEAT REMOVAL SYSTEM						COLD LEG SUMP RECIRC		RHR SPRAY		HOT LEG RECIRC		CONTAINMENT ISOLATION		AIR RETURN FANS		HYDROGEN MITIGATION		CONTAINMENT SPRAY SYSTEM			
	CCP 1A-A/FAN	CCP 1B-B/FAN	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	1A-A SIP/FAN	1B-B SIP/FAN	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	1A-A RHR PUMP/FAN	1B-B RHR PUMP/FAN	1A HTX/VLVS	1B HTX/VLVS	A-TRAIN HDR/VLVS ACCESSORIES	B-TRAIN HDR/VLVS ACCESSORIES	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	A TRAIN HDR/VLVS	B TRAIN HDR/VLVS	A FANS	B FANS	A IGNITORS	B IGNITORS	CS PUMP 1A-A/HTX A	CS PUMP 1B-B/HTX B	A HDR/VLVS EQPT	B HDR/VLVS EQPT
6.9 KV UNIT BD 1A																														
6.9 KV UNIT BD 1B																														
6.9 KV UNIT BD 1C																														
6.9 KV UNIT BD 1D																														
6.9 KV SD BD 1A-A	12				15				16																		18			
6.9 KV SD BD 1B-B		12				15				16																		18		
480 V SD BD 1A1-A																							16							
480 V SD BD 1A2-A																														
480 V SD BD 1B1-B																								16						
480 V SD BD 1B2-B																														
480 V RX MOV BD 1A1-A	12		12										16		16		16		16		17								18	
480 V RX MOV BD 1A2-A											16		16																18	
480 V RX MOV BD 1B1-B		12		12										16		16		16		16		17								18
480 V RX MOV BD 1B2-B												16		16																18
480 V RX MOV BD 2A1-A																														
480 V RX MOV BD 2A2-A																														
480 V RX MOV BD 2B1-B																														
480 V RX MOV BD 2B2-B																														
480 V C&A VENT BD 1A1-A	12				15				16																16		18			
480 V C&A VENT BD 1A2-A																														
480 V C&A VENT BD 1B1-B		12				15				16																16		18		
480 V C&A VENT BD 1B2-B																														
480 V C&A VENT BD 2A1-A																														
480 V C&A VENT BD 2A2-A																														
480 V C&A VENT BD 2B1-B																														
480 V C&A VENT BD 2B2-B																														
480 V UNIT BD 1A - #19																														
480 V UNIT BD 1B - #19																														
480 V UNIT BD 2A																														
480 V UNIT BD 2B																														
250 VDC BATT BD I																														
250 VDC BATT BD II																														
125 VDC BATT BD I	12				15				16														17				18			
125 VDC BATT BD II		12				15				16														17				18		
125 VDC BATT BD III																														
125 VDC BATT BD IV																														
120 VAC INST PWR 1-I													16		16															
120 VAC INST PWR 1-II														16		16														
120 VAC INST PWR 1-III													16		16															
120 VAC INST PWR 1-IV														16		16														
120 VAC INST PWR DIST PNL 1A																														
120 VAC INST PWR DIST PNL 1B																														
ESFAS A/SSPS S	12		12		15				16				16								17		17				18		18	
ESFAS B/SSPS B		12		12		15				16				16								17		17				18		18
CCS/AFW EQUIP COOLERS																														
CONT BLDG ELECT BD ROOM COOLING																														
ESSENTIAL AIR A																														
ESSENTIAL AIR B																														
NONESSENTIAL AIR																														
ERCW 1A-A	12				15				16																		18			
ERCW 1B-B		12				15				16																		18		
ERCW 2A																														
ERCW 2B																														
COMP COOL SYS A	12				15				16		16																18			
COMP COOL SYS B		12				15				16		16																18		
UNIT 1 CST																														
RCW																														
RWST	12	12							16	16																	18	18		
CONT SUMP																	16	16									18	18		
AMSAC																														

WBN/SUPTOFRLN/0,R00

NOTES:

1. TO READ, LOCATE THE SYSTEM/COMPONET ON THE UPPER AXIS AND READ DOWN TO IDENTIFY THE REQUIRED SUPPORTS ON THE LEFT AXIS
2. ALTERNATE POWER/CONTROL POWER FEEDS(AC,DC, CHARGERS, BATTERY BOARDS, BATTERIES) THAT ARE NOT AUTOMATICALLY SWITCHED ARE NOT DESIGNATED ON THIS MATRIX. THE ALTERNATE FEEDS ARE HOWEVER DISCUSSED IN THE APPROPRIATE MATRIX NOTE
3. THIS IS NOT A DESIGN OUTPUT DRAWING AND IS NOT TO BE USED AS DESIGN INPUT
4. REFER TO TABLE 1 OF THE EMERGENCY SAFETY FEATURES ACTUATION SYSTEM NOTEBOOK FOR FURTHER INFORMATION
5. REFER TO TABLE 1 OF THE COMPONENT COOLING SYSTEM NOTEBOOK FOR FURTHER INFORMATION
6. REFER TO TABLE 1 OF THE CONTROL AIR SYSTEM NOTEBOOK FOR FURTHER INFORMATION
7. REFER TO TABLE 1 OF THE ESSENTAIL RAW COOLING WATER SYSTEM NOTEBOOK FOR FURTHER INFORMATION
8. REFER TO TABLE 1 OF THE CONDENSATE AND FEEDWATER SYSTEM NOTEBOOK FOR FURTHER INFORMATION
9. REFER TO TABLE 1 OF THE AUXILIARY FEEDWATER SYSTEM NOTEBOOK FOR FURTHER INFORMATION
10. REFER TO TABLE 1 OF THE MAIN STEAM SYSTEM NOTEBOOK FOR FURTHER INFORMATION
11. REFER TO TABLE 1 OF THE REACTOR PROTECTION SYSTEM NOTEBOOK FOR FURTHER INFORMATION
12. REFER TO TABLE 1 OF THE CHEMICAL AND VOLUME CONTROL NOTEBOOK FOR FURTHER INFORMATION
13. REFER TO TABLE 1 OF THE PRESSURIZER PORVS NOTEBOOK FOR FURTHER INFORMATION
14. REFER TO TABLE 1 OF THE REACTOR COOLANT PUMP SEALS AND THERMAL BARRIERS NOTEBOOK FOR FURTHER INFORMATION
15. REFER TO TABLE 1 OF THE SAFETY INJECTION SYSTEM NOTEBOOK FOR FURTHER INFORMATION
16. REFER TO TABLE 1 OF THE RESIONAL HEAT REMOVAL SYSTEM NOTEBOOK FOR FURTHER INFORMATION
17. REFER TO TABLE 1 OF THE CONTAINMENT SYSTEM NOTEBOOK FOR FURTHER INFORMATION
18. REFER TO TABLE 1 OF THE CONTAINMENT SPRAY SYSTEM NOTEBOOK FOR FURTHER INFORMATION
19. 480 V UNIT BOARD INCLUDES THE 480 VAC TURBINE MOV BOARDS

X REFER TO TABLE 1 OF THE ELECTIC POWER SYSTEM NOTEBOOK FOR FURTHER INFORMATION

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CHBP1EHB--J0B01807----RM210PU1--PUN2540P
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DATE= 8/22/92 TIME=17 05 ACCESS=NONE
LOGON ID=$PBCAD9
PLOT= 1

Volume 2

WATTS BAR NUCLEAR PLANT UNIT 1 PROBABILISTIC RISK ASSESSMENT INDIVIDUAL PLANT EXAMINATION

PLG

Volume 2

TVA

WATTS BAR NUCLEAR PLANT UNIT 1 PROBABILISTIC RISK ASSESSMENT INDIVIDUAL PLANT EXAMINATION

PLG

TVA PROBABILISTIC RISK ASSESSMENT TEAM

Project Management

Richard J. McMahon — Corporate Engineering
W. Douglas Stevens — Site Project Manager
Susan D. Ferrell — Site Project Engineer

Task Leaders

Ara H. Djirikian/Raj I. Paul — Systems Analysis
Carla A. Borrelli — Data Analysis
J. Stephen Clements — Level 1 Analysis
William Z. Mims, Jr. — Level 2 Analysis

PLG PROBABILISTIC RISK ASSESSMENT TEAM

Project Management

B. John Garrick — Project Director
Jack W. Read — Project Manager
David H. Johnson — Technical Manager

Task Leaders

James C. Lin — Systems Analysis
Shobha B. Rao — Data Analysis
Donald J. Wakefield — Level 1 Analysis
R. Kenneth Deremer — Level 2 Analysis
Shawn S. Rodgers — Site Representative
Andrew A. Dykes — Human Factors Analysis

Technical Contributors

David A. Bidwell
Kurt Ching
Michael A. Emerson
Vincent S. Ho
Yung-Ming Hou
Eugene S. Kim
Wee Tee Loh
Subodh R. Medhekar
Thomas J. Mikschl
James H. Moody
Kathleen A. Ramp
Grant A. Tinsley

TVA Consultants

James K. Liming — ERIN
D. William Stillwell — ERIN

PLG Consultants

Marc Kenton — GKA
Jeff Gabor — GKA
Donald A. Wesley — EQE

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3.2 SYSTEMS ANALYSIS

The probabilistic risk assessment (PRA) systems analyses calculate and document the unavailability of specific top events as required by the plant Level 1 analysis. This section provides a summary of the systems analyses done to support the Watts Bar Nuclear Plant Unit 1 PRA. Section 3.2.1 contains a qualitative description of the systems and the top events modeled within each system. This qualitative summary of each system can also be found in the system notebooks, prepared in support of this document.

The system notebooks, as described in Section 3.2.2, contain detailed system information, reference material, and the quantitative results of the systems analyses. Also presented in Section 3.2.2 are tables that cross-reference the systems, top events, event trees, and the system notebooks.

The intersystem dependencies are presented in Section 3.2.3. This section includes the system dependency matrices and supporting documentation.

3.2.1 SYSTEMS DESCRIPTIONS

This section contains a qualitative description of each system modeled in this PRA. The systems descriptions are presented in order of their appearance in the linked event tree model. The simplified drawings for each of the systems modeled are described in Section 3.2.1.18, and are provided at the end of this section.

3.2.1.1 Electric Power System

3.2.1.1.1 System Function

The function of the electric power system is to provide power for operation of plant loads during normal plant operation and to loads required for safe shutdown during accident conditions including loss of offsite power. The electric power system consists of the unit main generator, two unit station service transformers (USST) per unit, four common station service transformers (CSST) shared by both units, two diesel generators per unit plus one spare, the station and vital batteries, and a two-train electrical distribution system. The electric power system model is broken into six subsystems: offsite grid, 6.9-kV AC power and diesel generators, 480V AC power, 250V DC power, 125V DC power, and 120V AC power. In addition, there are three ventilation systems modeled: the shutdown board room ventilation system, 480V board room ventilation, and the 480V transformer room ventilation system. The ventilation for the diesel generators is modeled with the system itself.

3.2.1.1.2 System Operation

3.2.1.1.2.1 Normal Operation. During normal plant operation, the unit main generators supply electric power for operation of plant equipment via the USSTs. The USSTs connect to the main generator between the generator terminals and the main step-up transformers; they supply power to the 6.9-kV unit boards and the reactor coolant pump (RCP) boards. The 6.9-kV shutdown boards are normally fed from CSSTs C and D. During a plant transient, the shutdown boards are not realigned; however, the unit boards automatically realign to CSSTs A and B. The normal power operation alignment of each of the subsystems is described below and shown in the simplified system diagrams.

3.2.1.1.2.1.1 Offsite Grid. Two offsite power grids are connected to Watts Bar: a 500-kV grid via a 500-kV switchyard and a 161-kV grid via the 161-kV switchyard. The 500-kV grid is supplied power from the unit 1 main generator during normal power operation. When a unit trip occurs, the unit is separated from the 500-kV grid, and all offsite power is supplied by the 161-kV grid. The 161-kV grid receives power from the Watts Bar hydro plant and supplies power to the CSSTs. Top Event OG models the connections from the 161-kV grid to the CSSTs C and D, including their secondary-side breakers. CSSTs A and B are modeled in Top Events UB1A, UB1B, UB2A, and UB2B.

3.2.1.1.2.1.2 6.9-kV AC Power and Diesel Generators. The Unit 1 6.9-kV power subsystem consists of shutdown boards 1A-A (Top Event AA) and 1B-B (Top Event BA), and unit boards 1A (Top Event UB1A), 1B (Top Event UB1B), 1C (Top Event UB1C), and 1D (Top Event UB1D). The Unit 2 6.9-kV power subsystem consists of shutdown boards 2A-A (Top Event AB) and 2B-B (Top Event BB).

During power operation (as well as any nonmaintenance mode of operation), the 6.9-kV shutdown boards are powered from the 161-kV offsite grid via CSSTs C and D. CSST C powers the Unit 1 train A shutdown board, and CSST D powers the Unit 1 train B shutdown board.

Ventilation and cooling for the Unit 1 and Unit 2 Train A 6.9-kV shutdown boards and the Unit 1 480V shutdown boards is supplied by air handling units A-A and C-B modeled in Top Event V1; the Unit 1 and Unit 2 Train B 6.9-kV shutdown boards and the Unit 2 480V shutdown boards are cooled by air handling units B-A and D-B modeled in Top Event V2. During normal operation, one fan in each unit is operating with the other in standby.

The present model shows the Unit 1 shutdown boards dependent on Top Event V1 and the Unit 2 shutdown boards dependent on Top Event V2. The actual dependencies as described above were determined after the quantification of the model. Correcting the alignment in the model is not expected to significantly impact the results.

When a unit trip occurs, the unit boards will transfer to CSSTs A and B, and the shutdown boards will be unaffected. The diesel generators are normally in standby and will automatically start when a loss of power is sensed at their associated 6.9-kV shutdown board. Each diesel generator is modeled by two top events, Gx and Fx, which stand for the diesel generator and fuel oil, respectively. Note: The "x" represents the second letter of the top event for each diesel; i.e., A for diesel generator 1A-A, B for diesel generator 1B-B, C for diesel generator 2A-A, and D for diesel generator 2B-B.

3.2.1.1.2.1.3 480V AC Power. The 480V shutdown boards power subsystems receive power from their associated 6.9-kV shutdown boards. 6.9-kV board 1A-A supplies 480V boards 1A1-A (Top Event A1) and 1A2-A (Top Event A2). 6.9-kV board 1B-B supplies 480V boards 1B1-B (Top Event B1) and 1B2-B (Top Event B2).

During normal operation, Unit 2 480V shutdown boards 2A1-A (Top Event A1U2), 2A2-A (Top Event A2U2), 2B1-B (Top Event B1U2), and 2B2-B (Top Event B2U2) receive power from their associated 6.9-kV shutdown boards. Namely, 2A1-A and 2A2-A receive power from 6.9-kV shutdown board 2A-A, and 2B1-B and 2B2-B receive power from 6.9-kV shutdown board 2B-B.

The 480V auxiliary building common boards are not supplied from the 6.9-kV shutdown boards but, rather, from the 6.9-kV common boards. This relationship as well as the associated 6.9-kV common boards is modeled in Top Event A3 for 480V common board A and Top Event B3 for 480V board B.

Top Events A1, A2, B1, and B2 also include the following sub-boards: reactor motor-operated valve (RMOV), reactor ventilation, control and auxiliary (C&A) ventilation, and diesel auxiliary. Top Events A2 and B2, however, do not have reactor ventilation boards associated with them.

480V transformer room ventilation is supplied by four fans for rooms 1A and 2B and by three fans for rooms 1B and 2A. The system is normally in standby until a fan is activated by a temperature switch. Each switch is set differently so that the fans are started on a staggered basis as the room temperature rises. The ventilation system in each room includes two motor-operated inlet dampers. Each fan has a mechanical backdraft damper to prevent reverse flow while not in operation.

3.2.1.1.2.1.4 250V DC Power. The 250V DC battery boards power subsystem I (Top Event D1) is powered from a normal battery charger that receives power from the 480V auxiliary common board A and a standby charger supplied from 480V auxiliary common board B. A 250V station battery provides emergency backup power to the bus. The 250V DC battery boards power subsystem II (Top Event D2) is comparable to subsystem I except that it is powered from 480V auxiliary common board B. The 250V DC battery boards provide control power to 6.9-kV unit boards.

3.2.1.1.2.1.5 125V DC and 120V AC Power. For the 125V DC top events modeled here (DA, DB, DC, and DD), each contains one 125V DC battery, one battery charger, fuses, breakers, and the associated distribution boards. During normal operation, the battery chargers supply all DC loads including the Uninterruptable Power Supplies (UPSs). The battery chargers are powered from 480V shutdown boards 1A1-A for Top Event DA, 1B2-B for Top Event DB, 2A1-A for Top Event DC, and 2B2-B for Top Event DD. The 125V DC battery is maintained on a float charge by virtue of its connection to the bus and acts as an emergency DC supply, should the charger fail.

There is a spare battery and charger available that is used to replace a channel that requires maintenance; it is modeled in the system analysis for the case when the normal charger or battery is taken out of service for maintenance.

For the 120V AC top events modeled here (DAAC, DBAC, DCAC, and DDAC), each contains one inverter, an alternate supply transformer, breakers, fuses, and distribution boards. During normal operation, the inverter provides power to the 120V AC vital instrument bus. The inverter is normally supplied by the associated 480V shutdown board. As an alternate source of power, the inverter can be directly supplied from the associated 125V DC battery board.

The 120V AC instrument power bus 1A modeled in Top Event DG consists of a bus, a transformer, a fused disconnect switch, and a 480V supply breaker. The transformer is normally powered from 480V shutdown board 1A1-A with alternate supply 480V shutdown board 1B1-B. The manual breaker that can align either supply is physically wired in the normal position. The transformer reduces the voltage from 480V to 120V for use in instrumentation.

The 120V AC instrument power system requires ventilation for success. The 480V board room ventilation system supplies rooms 1B and 2B in which the 120V AC inverters reside. One air handling unit (AHU) supplies each room, and there is no cross connection of ducting.

3.2.1.1.2.2 Accident/Transient Operation. This model describes the availability of the electric power system under all transient conditions. For the electric power system, these transients fall into two categories: loss of offsite power (including station blackout), and all other transients (nonloss of offsite power events). The following describes the expected operation of the electric power subsystems for these two transient categories.

3.2.1.1.2.2.1 Offsite Grid. Top Event OG models the 161-kV grid and the equipment between the Watts Bar hydro plant 161-kV switchyard and the connection to the Watts Bar Nuclear Plant at the CSSTs. This equipment includes 161-kV switchyard, the CSSTs, and their associated secondary-side breakers. During nonloss of offsite power transients, this top event models the availability of offsite power. During loss of offsite power events, this top event is guaranteed to fail.

3.2.1.1.2.2.2 6.9-kV AC Power and Diesel Generators. Top Events AA and BA (Top Events AB and BB for Unit 2) describe the availability of power at the 6.9-kV shutdown boards for all transients. The equipment included in Top Event AA is 6.9-kV shutdown board 1A-A and the associated circuit breakers. The equipment in Top Event BA is 6.9-kV shutdown board 1B-B and the associated circuit breakers.

When a nonloss of offsite power transient causes a unit trip to occur on Unit 1, unit boards 1A, 1B, 1C, and 1D (Top Events UB1A, UB1B, UB1C, and UB1D, respectively) will transfer from the USSTs to the CSSTs A and B as appropriate. The 6.9-kV shutdown boards are unaffected. If the 6.9-kV shutdown boards fail to receive power from the associated CSST, the 6.9-kV shutdown boards will transfer to their associated diesel generator. If the unit boards fail to transfer, the boards become deenergized.

When the transient is a loss of offsite power, the unit boards will not transfer to the CSSTs. The diesel generators will start and supply power to their associated 6.9-kV shutdown board. If all of the diesel generators fail to start during this transient, the transient is classified as a station blackout (SBO).

Each diesel generator is modeled in two top events; e.g., GA and FA for diesel generator 1A-A. Top events for the other three diesel generators are identical to those described here. Top Event GA models the diesel generator and its associated electrical equipment. It includes the following equipment: diesel generator 1A, sequencer, dedicated DC control power, diesel electrical room ventilation, essential raw cooling water (ERCW) cooling systems, output breaker, and associated piping and valves. The Watts Bar diesel generators are each a single generator with a 16-cylinder engine mounted on each end connected to a single shaft. The diesel is started by a start air system unique to each diesel. The unavailability of the start air system is included with the diesel unavailability.

Top Event FA models the fuel oil transfer system for the diesel. The system as modeled consists of the following equipment: a 7-day tank, a day tank, fuel oil transfer pumps 1 and 2, and associated piping, valves, and instrumentation. The fuel oil transfer pumps are started at different levels by level switches such that if the first pump does not start, the level will drop and the second will start. Both pumps will stop as the day tank level reaches a "high level" and actuates a level switch.

3.2.1.1.2.2.3 480V AC Power. The 480V power subsystem is further divided into Class 1E power and balance-of-plant (BOP) power. Top Events A1, A2, B1, and B2 make up the 480V Class 1E power for Unit 1; A1U2, A2U2, B1U2, and B2U2 model the Class 1E 480V power for Unit 2; and the BOP power is described by Top Events A3 and B3. The train A equipment for these top events is described below. Train B equipment is similar in both units; where differences exist, they are described.

Train A Top Events A1 and A2 for Unit 1 and A1U2 and A2U2 for Unit 2 (Top Events B1 and B2 for Unit 1 and B1U2 and B2U2 for Unit 2 train B) describe the availability of power at the 480V Class 1E shutdown boards for all transients.

The equipment modeled in Top Event A1 includes 480V shutdown board 1A1-A, transformer 1A1-A, RMOV board 1A1-A, diesel auxiliary board 1A1-A, reactor ventilation board 1A-A, C&A Vent board 1A1-A, and their associated circuit breakers. Top Event A2 contains similar equipment and boards except no reactor ventilation board. The Unit 2 equipment modeled in A1U2 and A2U2 is equivalent. Top Event B1 includes 480V shutdown board 1B1-B, transformer 1B1-B, RMOV valve board 1B1-B, diesel auxiliary board 1B1-B, control and auxiliary ventilation board 1B1-B, reactor ventilation board 1B-B, and their associated circuit breakers. Top Event B2 contains similar equipment but has no reactor ventilation board. Although transformers 1A-A and 1B-B can be aligned to provide power to the buses (1A1-A and 1B1-B respectively), the transfer is manual on the loss of power.

When a transient occurs, whether or not it is a loss of offsite power, the 480V Class 1E shutdown boards continue to receive power from their associated 6.9-kV shutdown boards. Manual transfer of the 480V Class 1E boards is possible if both units are in cold shutdown, but this action is not modeled. Several sub-boards, however, are shed from the 480V shutdown boards on a loss of offsite power. The affected boards include all of the reactor ventilation boards and half of the control and auxiliary boards (1A2-A, 1B2-B, 2A2-A, and 2B2-B).

Top Event A3 (Top Event B3 for train B) describes the availability of power at 480V auxiliary building common board A (auxiliary building common board B in Top Event B3). The equipment modeled in Top Event A3 includes the 480V and 6.9-kV auxiliary building common boards A, start bus A, CSSTs A and B, and the associated transformers and circuit breakers.

When a nonloss of offsite power transient occurs, the auxiliary building common boards (Top Events A3 and B3) will continue to receive power, provided that power is available to common station service transformers A and B (Top Event OG is successful). If the transient is a loss of offsite power, Top Events A3 and B3 are guaranteed to fail.

The 480V transformer room ventilation system is unaffected by a plant transient but is required to restart on loss of offsite power. All of the fans in rooms 1B and 2A receive power from buses that will become reenergized after a loss of offsite power. However, one fan in room 1A and one fan in room 2B will lose power because they are supplied from the 480V common board power system.

3.2.1.1.2.2.4 250V DC Power. Top Event D1, 250V DC power subsystem I (Top Event D2 for 250V DC subsystem II), describes the availability of power at 250V DC turbine building board I (and board II for D2). The equipment modeled in Top Event D1 includes the 250V DC bus, 250V station battery I, 250V battery charger I, the 250V DC electrical control board, and the associated fuses and breakers. The equipment modeled in Top Event D2 is similar except that it does not include the 250V DC electrical control board distribution panel.

When a nonloss of offsite power transient occurs, the 250V battery chargers will continue to supply power to the 250V DC distribution system, provided that Top Event OG is successful. If the transient is a loss of offsite power, the 250V battery chargers will be unavailable but power to the 250V DC distribution system will continue to be supplied by the station batteries for 4 hours after which the top event is guaranteed to fail.

3.2.1.1.2.2.5 125V DC and 120V AC Power. Top Event DA, 125V DC power subsystem I (Top Events DB, DC, and DD for subsystems II, III, and IV, respectively) describes the availability of power at the 125V DC vital battery boards. Each top event model includes a 125V DC bus, a 125V battery and charger, and their associated fuses, breakers, and transformers.

Top Event DAAC, 120V AC power subsystem I (Top Events DBAC, DCAC, and DDAC for subsystems II, III, and IV, respectively) describes the availability of power at the 120V AC vital instrument power boards. Each top event model includes a 125V DC to 120V AC inverter (UPS), a 120V distribution bus, and their associated fuses and breakers.

A nonloss of offsite power transient has no effect on these top events, provided that power remains available to their associated 480V supply buses. This is also true during a loss of offsite power event, provided that the associated diesel generator is successful. When the diesel generator for that train of power fails, the 125V DC battery charger will lose power, and the DC power will be supplied by the attached 125V DC vital battery. This battery will continue to supply DC power for at least 4 hours based on an SBO calculation (see Reference 1.7.6.2 of the system notebook).

Ventilation for the 480V board rooms 1B and 2B (Top Events VINV1 and VINV2) that contain the 120V AC vital inverters is unaffected by plant transients. However, on a loss of offsite power, the AHUs will restart after power is restored to the shutdown buses by the emergency diesel generators.

The 120V AC instrument power boards 1-I, 1-II, 1-III, and 1-IV are normally supplied from the 480V shutdown boards 1A1-A, 1B2-B, 2A1-A, and 2B2-B, respectively. During accidents, power remains supplied from the 480V bus even during loss of offsite power events. During loss of offsite power events, the 480V shutdown board will receive power from the emergency diesels via the 6.9-kV shutdown boards.

3.2.1.2 Essential Raw Cooling Water System

3.2.1.2.1 System Function

The primary function of the essential raw cooling water (ERCW) system during normal operation is to provide cooling water to primary and secondary components. During normal/accident conditions, ERCW provides an ultimate heat sink function for dissipating heat from essential plant equipment, room ventilation systems, and the component cooling water system. A simplified flow diagram of the ERCW system is shown in Section 2 of the system notebook.

ERCW is the alternate water supply for the auxiliary feedwater system and the component cooling system (CCS) surge tank. The auxiliary feedwater function is modeled in the auxiliary feedwater system notebook. During site flood, which is an external event, the ERCW provides water for the component cooling and raw cooling water systems. External events are not modeled in the system analysis.

3.2.1.2.2 System Operation

The ERCW system consists of redundant and independent full-capacity trains, A and B. Each train contains four ERCW pumps, two traveling screens, two screen wash pumps, two strainers, and one common header located at the intake pumping station. For each train, two pumps, one traveling screen, one screen washpump, and one strainer are powered from Unit 1 and similarly for Unit 2. The pumps and screens from one unit can supply either or both units (see the system notebook, Section 1.7.5, Design Criteria references).

3.2.1.2.2.1 Normal Operation. During normal power operation, the station operators select the number of pumps per station train required to supply flow. The ERCW pumps are manually started by the operator. The automatic features of the ERCW pumps are to load to the 6.9-kV shutdown boards during load sequencing (normally, the running pumps are selected by the operators to start automatically during load sequencing) and to start when a safety injection signal is received (see assumption 3 in Section 3.2.2 of the system notebook).

The ERCW pumps draw suction from the Tennessee River via the ERCW intake pumping station, through the traveling screens, and discharge through a trained manifold system to the ERCW strainers. Unit separation begins at the four ERCW strainers. Two strainers are provided for each station train, one for each unit.

The traveling screens are aligned to remove large debris from the water entering the system. When the screens become sufficiently clogged between the downstream and upstream sides of the screen, the screen wash pumps start on differential level across the strainer. The screen wash cycle can also be started by a timer or by manual initiation from the control room. When the screen wash pump header reaches a preset pressure, the traveling screen motor is automatically started to rotate the screen, washing debris off the screen. The screen wash function was not modeled.

The ERCW strainers filter small debris that is not blocked by the traveling screen. Capability is provided to automatically backwash the strainer. There are two strainers for each common header, one supplying each unit. The strainer flushing operation is not modeled. A plugging failure frequency was used to model the strainer failures.

The normal operating loads for the ERCW system are the CCS heat exchangers, the coolers for the reactor coolant pump motor, the control rod drive ventilation coolers, room coolers, and the air compressors. These loads are modeled in their respective systems where the cooling function is required.

3.2.1.2.2.2 Accident/Transient Operation

3.2.1.2.2.2.1 General Transient with Normal Power Available. The ERCW system is common to WBN Unit 1 and Unit 2. The minimum combined safety requirements for one accident unit and one nonaccident unit or two nonaccident units are met by two pumps in one plant train. Loss of either header or the loss of an entire emergency power train will not prevent safe shutdown. For this model, the screen wash system is not required to function under accident conditions.

If one of the running pumps fails during operation, the operator must act to recover lost flow by manually starting a redundant standby ERCW pump. The operator action to manually start the standby pump (HAAE1) is modeled in the fault tree for the pump train top event.

3.2.1.2.2.2.2 Safety Injection with Offsite Power Available. Following a safety injection signal, the standby pumps will start. Normally running pumps will continue to operate.

3.2.1.2.2.2.3 Loss of Offsite Power. Two ERCW pumps are supplied power from each 6.9-kV shutdown board. Each 6.9-kV shutdown board is supplied power from an associated diesel generator. Following a loss of offsite power (LOSP) event, all running ERCW pumps will stop. Only one pump (the selected pump) on each 6.9-kV shutdown board is automatically started during load sequencing. ERCW pumps supplied from the same 6.9-kV shutdown board are interlocked so that only one pump can be started when the shutdown board is supplied from diesel power. An interlock switch in the main control room (MCR) selects the pump that will be allowed to start. Normally, the running pump is selected for restart.

If a main control room evacuation occurs, the MCR ERCW pump selector switch is unavailable for use. In this situation (LOSP and MCR evacuation), two ERCW pumps per train will automatically restart. Should any of these pumps fail to start, another pump can be manually started by the operator from the 6.9-kV shutdown board rooms. The starting of an ERCW pump from the 6.9-kV shutdown board is not modeled.

3.2.1.2.2.2.4 Flow to Station Air Compressors. The ERCW system supplies flow to the station air compressors in the turbine building. The piping in the turbine building is not seismically qualified. Flow and pressure instrumentation and valves are provided so that if a high flow signal coincident with a low pressure signal are present (indicating a line break), the lines isolate from the ERCW header.

3.2.1.3 Component Cooling System

3.2.1.3.1 System Function

The function of the component cooling system (CCS) is to serve as an intermediate heat sink for the removal of heat from potentially radioactive heat loads during normal and accident conditions. The CCS acts as a barrier between potentially or normally radioactive fluid flowing in the various coolers and the essential raw cooling water (ERCW) system to avoid release of radioactivity into the environment. This function is accomplished through the use of a closed-loop system in which the CCS removes heat from the various components (CCS loads) and transfers it to the CCS heat exchangers where the heat is transferred to the ERCW system. The CCS (for Units 1 and 2) consists of five pumps, four thermal barrier booster pumps, three heat exchangers, two surge tanks, associated valves, piping, and instrumentation serving both Watts Bar units. The scope of the CCS model (i.e., analysis boundary) in this analysis is limited to the CCS equipment that supports shutdown of Watts Bar Unit 1 during accidents or plant transients. The initial conditions of the CCS equipment being modeled are based on the configuration of the system during normal plant operations. A simplified flow diagram of the CCS system model depicting the analysis boundary is shown in Section 2 as Figure 1 in the system notebook. Figure 2 in Section 2 shows the portion of the ERCW system that provides cooling water to the CCS heat exchangers A and C. This portion of the ERCW system is modeled in this analysis.

The ERCW provides cooling water to the area cooling units for the CCS and auxiliary feedwater system (AFW) pump area in the auxiliary building. This area is normally served by the auxiliary building general ventilation and air conditioning system. The CCS/AFW coolers provide the necessary cooling whenever the area temperature exceeds the setpoint. A simplified diagram of the CCS pump coolers is shown in Section 2 (Figure 3) of the system notebook.

3.2.1.3.2 System Operation

3.2.1.3.2.1 Normal Operation. During normal plant operation, with all CCS equipment available, pumps 1A-A and 1B-B and heat exchanger A are aligned with CCS train 1A. Pump C-S and heat exchanger C are aligned with CCS train B. CCS train 1A is aligned to both train 1A ESF equipment and miscellaneous equipment (including equipment required for normal plant operation) for Unit 1. Both pumps in CCS train 1A are normally running. CCS train B is aligned, when required, to the waste disposal system (WDS) condensate demineralizer waste evaporator (CDWE) package and to train 1B ESF equipment.

The ERCW system provides the heat sink for CCS heat exchangers A and C. ERCW supply header 1B is normally aligned to CCS heat exchanger A. One of the two ERCW discharge isolation valves at the outlet of heat exchanger A is normally open. ERCW supply header 2B is normally aligned to heat exchanger C. Flow from ERCW supply header 1A is conservatively modeled as normally isolated from CCS heat exchanger C via isolation valve 1-FCV-67-147. The heat exchanger discharge is isolated from ERCW discharge header B by isolation valve 0-FCV-67-152. One of the two outlet paths to ERCW discharge header A is isolated by isolation valve 0-FCV-67-151 (normally closed with power removed) downstream from the heat exchanger. Valves 0-FCV-67-152 and 0-FCV-67-151 are normally closed and will be infrequently opened (manually) for cooling of the WDS CDWE package.

The CCS/AFW pump area coolers (see Figure 3 in Section 2 of the system notebook) are normally in automatic mode. During normal operation, they are not in use since the cooling loads are being handled by the auxiliary building air handling units 1B and 2B. They are not interlocked with the pumps but start on high temperature or auxiliary building isolation (ABI) signal. A cooler can also be started manually or placed in a standby mode. In the standby mode, the cooler will start when low air flow is detected in the other cooler. Operation of either cooler requires that its associated cooler fan starts and runs and the availability of the ERCW system water as the heat sink. ERCW flow is controlled via a normally closed, fail open valve in the ERCW supply line. The redundant cooler will start up automatically if the pump room area temperature increases above setpoint; i.e., fan will start and ERCW flow control valve will open upon receipt of high temperature signal.

Trains 1A and B CCS header isolation valves are normally open during normal plant operations such that CCS flow is established (upon system startup) to the branches associated with Unit 1 essential safety feature (ESF) and non-ESF equipment. A list of ESF and non-ESF equipment served by the CCS is provided in Table 2 of the system notebook, prepared in support of this document.

The branches off the CCS headers that contain the isolation valves and coolers associated with the above equipment are modeled if that equipment supports mitigation of plant transients subject to the analysis (includes all ESF equipment and non-ESF equipment). Each branch subject to the PRA analysis is included within the model boundary of the system in which its associated cooler is located.

Unit 1 surge tank is provided to serve the following purposes:

- To allow for volumetric changes in the cooling water during operation.
- To serve as monitoring for system water inventory.
- To provide a point for adding makeup water, corrosion inhibitors, etc.
- To provide net positive suction head (NPSH) for the associated CCS pumps.

3.2.1.3.2.2 Accident/Transient Operation. The operational features of the CCS during normal plant operation (see the system notebook, Section 1.3.1) are also applicable during accident or transient conditions. The following is supplemental to the information provided in Section 1.3.1 of the system notebook for identification of the operational features of the system during transient or accident conditions.

A priority of an operator during a transient will be to maintain cooling water supply to the ESF train 1A and 1B headers. The CCS supply to the excess letdown heat exchanger is isolated on a Phase A containment isolation. The CCS supply to the RCP thermal barriers and oil coolers is isolated on a Phase B containment isolation. Additional CCS loads not associated with either of these two headers may be isolated with the CCS still meeting the minimum safe shutdown requirements for the plant. The only non-ESF loads being included in the model are the heat exchangers and thermal barriers associated with the RCS pumps. The RCS system notebook, "RCP Seal Injection and Thermal Barrier Cooling," contains information on the thermal barrier heat exchangers that are included in this PRA model. If the operator successfully isolates train 1A from the spent fuel pit heat exchanger load, only one of the two pumps in train 1A is required to be operational. The operator actions to isolate CCS train A from the spent fuel pit heat exchanger require the operator to align the Unit 2 CCS train A to the spent fuel pit cooling heat exchanger and isolate the Unit 1 CCS train A to the spent fuel pit heat exchanger while monitoring the CCS surge tank levels.

With the exception of the RHR heat exchangers, the CCS loads associated with the ESF headers are normally aligned and supplied with CCS water during normal plant conditions and will automatically continue to be supplied during transient and accident conditions. The RHR heat exchangers are usually valved out during the injection phase of safety injection since residual heat removal is not required during this phase. During the switchover from the injection to the recirculation phase of safety injection, it will be necessary for the operator to manually align the RHR heat exchangers of the accident unit in order to supply these heat exchangers with cooling water. The RHR system notebook contains further information associated with the operational procedures for performing the manual actions required when aligning the RHR heat exchangers during transient and accident conditions.

Heat exchanger A normally receives cooling water from ERCW supply header 1B via valve 1-FCV-67-458 (see Figure 2 in Section 2 of the system notebook). Following a loss of both 6.9-kV shutdown boards 1B-B and 2B-B, event valves 1-FCV-67-223 and 2-FCV-67-223 will open and valve 1-FCV-67-458 will close. Cooling water for heat exchanger A will then be supplied by ERCW header 2A.

The ERCW outlet valve (1-FCV-67-143) from CCS heat exchanger A is normally open and is included in the model. No automatic or manual valve operations are required to align ERCW cooling to this heat exchanger following an initiating event. Another ERCW discharge path from heat exchanger A (via the normally closed valve 1-FCV-67-146) is conservatively excluded from the quantification model (see the system notebook, Figure 2 in Section 2).

CCS heat exchanger C is modeled to normally discharge water to ERCW discharge header A via valve 0-FCV-67-144. The other two of the three ERCW outlet paths from CCS heat exchanger C (see Figure 2 in Section 2 of the system notebook) are normally isolated from the corresponding discharge headers via normally closed valves 0-FCV-67-151 and 0-FCV-67-152. The ERCW outlet from heat exchanger C via valve 0-FCV-67-151 must be manually aligned to the ERCW discharge headers and is excluded from the model. Valve 0-FCV-67-152 will open following a loss of offsite power event and allows train B to discharge to ERCW discharge header B. This path is modeled in the analysis.

In addition to the auxiliary building air handling units, the CCS/AFW pump area coolers will start (in auto-mode), given a high temperature setpoint of 95°F or an ABI signal. The temperature switches are TS-30-190 for train A and TS-30-191 for train B of Unit 1, and TS-30-184 for train A and TS-30-185 for train B of Unit 2. The cooler start will deenergize a solenoid to open the associated flow control valve on the ERCW side (FCV-67-162 for train A and FCV-67-164 for train B of Unit 1, and FCV-67-217 for train A and FCV-67-219 for train B of Unit 2). The auxiliary building general ventilation can also provide cooling to the CCS/MD-AFW pumps.

Automatic isolation of the reactor coolant pump thermal barrier lines is not addressed in this analysis since the components involved with this function are addressed in the notebook for the reactor coolant system seal injection and thermal barrier cooling.

3.2.1.4 Plant Compressed Air System

3.2.1.4.1 System Function

Watts Bar compressed air systems are common to both units and include the service air system, the control air system (CAS), and the auxiliary control air system (ACAS). The compressed air system is a source of motive air for pneumatic equipment required to function for plant operations. The CAS provides cool, oil-free, filtered, dry compressed air to equipment that requires instrument-grade air during normal plant conditions. Two of the CAS compressors may be manually started during a loss of offsite power event. The ACAS is designed to ensure that all vital equipment will have an adequate instrument-grade air supply during both normal and accident conditions. During normal plant operation, the ACAS air receivers are charged by the CAS. The ACAS starts automatically upon loss of air supply from the CAS. Service air provides noninstrument-grade air to miscellaneous equipment throughout the plant and is isolated from the compressed air system if control air pressure decreases. The isolation of the service air system is the only portion of this system modeled.

3.2.1.4.2 System Operation

3.2.1.4.2.1 Normal Operation. The CAS is supplied by four motor-driven, nonlubricated, two-stage reciprocating compressors. Two compressors are normally operating and considered to be sufficient to meet the total plant control air demand under normal conditions. Supply to service air is terminated on low CAS pressure (as sensed by 0-PS-33-4) to ensure that control air requirements are a higher priority. The failure of the isolation valve 0-PCV-33-4 to close and remain closed is included in this analysis. A simplified diagram of the CAS is shown in Section 2, Figure 2, of the system notebook.

The four two-staged compressors discharge into two redundant headers. These headers feed the two control air receivers that, in turn, supply air through two redundant headers to the control air station. The control air station is composed of three trains of prefilter, regenerative-type desiccant dryer, afterfilter, and associated valves.

When CAS is in automatic mode, the control air compressors will trip on loss of board voltage, low oil pressure, high oil temperature, and high air temperature. The CAS air compressors are normally placed in automatic mode and sequentially loaded on decreasing air receiver pressure and unloaded on increasing air receiver pressure to maintain a minimum system pressure. The ACAS compressors are maintained in the automatic mode but are normally idle when adequate pressure is maintained in the essential headers by the CAS system. The ACAS air receivers are charged by the CAS. The ACAS compressors start automatically upon loss of air supply from the CAS as sensed by 0-PS-32-62 for compressor A-A and by 0-PS-32-88 for compressor B-B. The ACAS also has relief valves on the accumulators and air receivers for the purpose of system protection.

The primary function of the CAS air receiver is to provide sufficient volume of air at the compressor discharge to dampen out pressure pulses and to minimize compressor cycling.

A secondary function of the CAS air receivers is to act as a moisture separator (gravity) for removal of entrained water that can be carried over from the aftercooler. The two CAS air receivers are arranged in parallel and are supplied with compressed air via separate supply headers from a common air compressor discharge header.

Both the CAS air dryers and the ACAS air dryers operate continuously during normal plant operations. Normally, all three CAS air dryers are aligned for service. The CAS and ACAS dryers are two-stage regenerative-type that operate automatically and independently of their respective air compressors. The dual dryer towers are alternately regenerated (purged) and fully pressurized for service via cam-operated cycling valves controlled by timers. Air supply pressure to air dryers of CAS and ACAS should be maintained to obtain proper performance of the dryers. In the event that high moisture air passes beyond the dryers, it may cause the eventual failure of air valves. Failure of these valves are modeled in their system notebooks. This would be alarmed by moisture sensors located in the dryer discharge headers. The control air supply would then be diverted to the remaining dryer units.

The CAS supplies no safety-related equipment. However, portions of systems that penetrate the containment are safety related and are analyzed in the containment system model.

3.2.1.4.2.2 Accident/Transient Operation. During accident or abnormal conditions when CAS air flow is degraded or lost, the ACAS supplies air of adequate quality and pressure to plant essential air loads under all conditions including safe shutdown earthquake and maximum possible flood. The ACAS located at Elevation 757.0' in the auxiliary building is a seismic Class I structure and above maximum possible flood elevation. Details for flood protection are given in WB-DC-40-29, "Flood Protection Design Criteria." Redundancy and train separation are provided in the ACAS to the extent that a single failure cannot render both trains inoperable.

Each ACAS train contains a compressor, receiver, accumulator, dryer, and filter (Figure 3 in the system notebook). Each ACAS compressor is sized to supply the total safety-related control air requirements for its respective train in the event of an accident, flood, or loss of CAS. The piping is arranged so that auxiliary receivers are charged from the nonqualified station control air system during normal operation. Use of the ACAS during normal operation protects system components from high moisture resulting from a possible control air dryer malfunction. Isolation valves and bypass lines are provided around components to permit bypass operation for maintenance.

The auxiliary control air is supplied by two identical, electric motor-driven, nonlubricated, single-stage reciprocating compressors that start automatically upon loss of air from CAS. ACAS is automatically isolated from the CAS whenever the system pressure drops below a setpoint by closing air-operated valves (FCV 32-82 and FCV 32-85). Check valves 32-264 and 32-256 provide redundant isolation function by preventing backleakage of air from ACAS into nonessential air lines. The pressurized air volume of the auxiliary air accumulator and air receiver is sufficient to dampen pressure pulses generated by the ACAS compressors and to maintain system air pressure to pneumatically open valves following a loss of offsite power.

ACAS compressors A-A and B-B are cooled by ERCW trains 1A and 2B, respectively. Solenoid valves FSV 32-61 and FSV 32-87 open on compressor start to allow cooling water to both compressor cylinders and the aftercoolers. Pressure and temperature control valves control the amount of flow through each heat exchanger, and a heater warms water before it enters the cylinder water jacket to prevent condensation on cylinder walls when the compressor is not in operation.

Electric power for the auxiliary control air systems is provided from both offsite grid and emergency sources. The two independent auxiliary air systems are powered from separate emergency power sources to prevent a total loss of ACAS due to single electric power failure.

3.2.1.5 Chemical and Volume Control System

3.2.1.5.1 System Function

The chemical and volume control system (CVCS) is designed to maintain the required water inventory and water chemistry control of the reactor coolant system (RCS) during normal and emergency operation. Specifically, the CVCS functions during normal operation to:

1. Maintain the programmed water level in the pressurizer.

2. Maintain seal water flow to the reactor coolant pumps.

3. Control the reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentrations, and makeup.

4. Process excess reactor coolant to recover and reuse the boron and primary water.

During emergency operation, the CVCS functions as a source of high pressure injection into the RCS and as an emergency boration system under anticipated transient without scram (ATWS) conditions. The CVCS is also used during certain events such as an unisolated steam generator tube rupture (SGTR) or loss of sump recirculation to refill the refueling water storage tank (RWST).

Three CVCS functions are modeled in this analysis. These are the ability of the CVCS to:

1. Provide high pressure injection to the RCS during the injection and sump recirculation phases.

2. Provide injection of high concentration boric acid solution into the RCS from the boric acid tanks for emergency boration.

3. Refill the RWST using the boric acid blender and the flow path through containment spray to the RWST.

The function of the reactor coolant pump seal injection is modeled in the "Reactor Coolant Pump Seal Injection and Thermal Barrier Cooling" notebook prepared in support of this document. The functions of maintaining the programmed water level in the pressurizer with normal charging and letdown, controlling the RCS water chemistry, and recovering boron are not modeled in this analysis.

3.2.1.5.2 System Operation

The chemical and volume control system is a safety-related system designed to perform functions during normal operations and accident conditions.

The CVCS operates during the following modes of operation: reactor startup, normal operation, hot standby operation, hot shutdown, cold shutdown, refueling, and emergency shutdown. During reactor startup, the RCS pressure and temperature are increased, and the boron concentration is decreased from shutdown concentration so that criticality can be achieved. During RCS heatup, excess reactor coolant is stored in the holdup tanks and is later processed to recover the boric acid and primary makeup water. During normal operation, the CVCS adjusts the RCS boron concentration to account for core burnup and xenon buildup during the core lifetime. During cold shutdown, the RCS boron concentration is increased to cold shutdown concentration, and the RCS pressure and temperature are decreased. A portion of the CVCS is shared with the emergency core cooling system (ECCS) and is required for emergency shutdown following a LOCA or secondary system line break. Portions of the CVCS are required for RCS filling, RCS hydrostatic testing, and various draining operations.

3.2.1.5.2.1 Normal Operation. During normal operation, the CVCS principally maintains the RCS water inventory and chemistry with normal charging and letdown and provides reactor coolant pump seal flow. Normal charging and letdown are not modeled in this analysis because their unavailability does not affect significantly the overall ability of the plant to mitigate the types of events considered. Normal seal injection to the reactor coolant pumps is modeled in the "Reactor Coolant System Seal Injection and Thermal Barrier Cooling" notebook as Top Event SE. Seal injection or thermal barrier cooling is required by the RCPs to prevent a seal LOCA.

3.2.1.5.2.2 Accident/Transient Operation. The chemical and volume control system functions as part of the ECCS for safety injection during a LOCA or major overcooling transient such as a secondary-side steam line break. The CVCS is primarily used for high pressure injection of borated water from the RWST or containment sump into the RCS via the cold legs to prevent damage to the core.

During a transient that does not result in safety injection, CVCS is used to supply RCP seal injection. RCP seal integrity is modeled with two CVCS top events (VS and VA or VB) and two reactor coolant system top events (TB and SE). This function is described in the "Reactor Coolant System Seal Injection and Thermal Barrier Cooling" notebook.

During the modeled injection mode of ECCS, one of two trains of centrifugal charging pumps is required to start on a safety injection signal and provide high pressure flow to two of four RCS cold legs. The CCPs draw borated water from the RWST for about 1 hour during ECCS injection mode. After 1 hour, when it is assumed that the RWST level has dropped to below 29% and the containment sump level is above 10%, the automatic sump swapover occurs, and the suction path is manually switched from the RWST to the RHR discharge path. This realignment of the suction of the CCPs is modeled in the RHR system analysis. The CCPs are required in the model to operate for 24 hours.

The emergency boration function (modeled in Top Event EB) of CVCS is used during an ATWS event to pump additional borated solution from the BAT to the RCS. The BAT has a boron concentration of 20,000 ppm, which is transferred to the suction of the CCPs by the boric acid pumps. One BAT and one boric acid pump are required to function for emergency boration to be successful. Three BATs are shared between the two Watts Bar units, and two boric acid pumps are available for each unit. Only one BAT and one transfer pump that are normally aligned are modeled in this analysis. Use of the other tanks and pumps would require manual local actions to realign the system.

3.2.1.5.2.2.1 Safety Injection Phase. During normal operation, one of the CCPs is running to maintain normal charging and letdown of the RCS. Upon receipt of a safety injection signal, the standby CCP is started, and the normal charging path in the CVCS is isolated by closing valves FCV-62-90 and FCV-612-91. At the same time, two parallel RWST suction valves (LCV-62-135 and LCV-62-136) are automatically opened to admit borated water from the refueling water storage tank to the suction of CCPs. As soon as these valves are opened, the CCP normal suction of the CCP from the volume control tank is automatically isolated by the VCT isolation valves (LCV-62-132 and LCV-62-133). In addition to the CCP suction transfer, two normally closed valves (FCV-63-25 and FCV-63-26) at the outlet of the BIT open automatically to complete the ECCS injection path. The two CCPs can thus deliver the borated water from the RWST through the BIT into the reactor vessel. These CCPs continue to inject borated water into the RCS. The RWST water is directed to the reactor vessel through a 4-inch header, which reduces to a 3-inch header and is injected into each RCS cold leg through a 1.5-inch branch line. Four preset throttling valves (63-582, 63-583, 63-584, and 63-585) are installed in each of the four 1.5-inch branch lines to prevent CCP runout and to equalize flow through all four lines so that the amount of coolant loss is minimized if one of the injection lines should rupture and spill into the containment. Each branch line or injection line is equipped with a check valve to prevent the reactor coolant from entering into the CVCS.

During automatic valve realignment, the CCP miniflow lines provide pump protection by diverting a portion of charging pump discharge to the volume control tank through the seal water heat exchanger. Each miniflow line is provided with an orifice to limit the maximum recirculation flow to 60 gpm. The miniflow recirculation valves (FCV-62-98 and FC-62-99) are set in the open position with their power removed.

3.2.1.5.2.2.2 Recirculation Phase. The injection mode of ECCS will continue until the RWST water level drops to the low level setpoint ($\leq$29%) coincident with a high level ($\geq$10%) in the containment sump. When these conditions occur, the cold leg recirculation mode will be initiated automatically. The containment sump isolation valves (FCV-63-72 and FCV-63-73) are automatically opened to align with the RHR pump suction, and the RHR pump suction from the RWST is isolated automatically by closing FCV-74-3 and FCV-74-21. The centrifugal charging pumps will continue to take suction from the RWST until the operator manually realigns the CCP suction from the RWST to the RHR pump discharge. This is accomplished by opening FCV-63-8, which is downstream of RHR pump 1A-A, providing CCP suction and closing LCV-62-135 and LCV-62-136, which isolates the CCPs from the RWST. If the normal suction path for the CCP from the RHR system fails, an alternate path from the RHR pump 1B-B (through valves FCV-63-11, FCV-63-48, FCV-63-47, FCV-63-6/FCV-63-7, and FCV-63-177) is available. This automatic and manual realignment for containment sump recirculation is modeled in Top Events RVA, RVB, RL, and RR, which are described in the RHR system analysis.

3.2.1.5.2.2.3 Emergency Boration for ATWS. Emergency boration is necessary during an ATWS event to inject high concentration of boron into the reactor vessel to reduce the reactivity in the core. The CCPs are manually aligned to the RWST with the cold leg injection path open through the BIT. The borated RWST water is supplemented with the boron solution from the boric acid tank (with higher concentration). The operator is instructed in FR-S.1 Step 4c to open the emergency boration valve (FCV-62-138) and to switch the boric acid pumps to the fast speed. This function is modeled as part of the CVCS system analysis (Top Event EB). Successful completion of emergency boration during ATWS requires success of Top Events VS, VA or VB, EB, and VC.

Top Event EB also includes the operator action to open the breakers to the control rod motor generator sets. This action is given in Steps 1 and 5 of FR-S.1. The operator is instructed to go to the 480V unit boards A and B. The human reliability analysis for both the operator actions to open FCV-62-138 and to open the motor generator set breakers is given in the human reliability analysis section.

3.2.1.5.2.2.4 Makeup to the RWST. The modeled RWST makeup function is needed during a small LOCA event with sump recirculation unavailable or an unisolated steam generator tube rupture event to refill the RWST. The RWST is the borated water source for the injection pumps to maintain the RCS inventory. For a LOCA event, makeup to the RWST can be provided by the containment sump, the holdup tank, the spent fuel pit, or the RWST of Unit 2, as instructed in ECA-1.1. In this analysis, the makeup from the containment sump through the containment spray test lines is modeled. The operator action to open the manual isolation valves in the test lines (72-502, 72-503, and 72-504) is also included in the model. The makeup function for an SGTR is described in WBN ECA-3.2, Step 1a, and SOI-62.2, Section 8.1. A flow path to the blender is established for the primary water and the boric acid solution. The primary water supply requires the primary water storage tank and one primary water makeup pump. During plant power operation, one primary water makeup pump is running continuously. The boric acid path to the blender consists of the BAT and the normally aligned boric acid pump and the associated piping and valves. Downstream of the blender, a flow path is established to the RWST through the containment spray system valve 72-501. An operator action is required to perform the alignment for this makeup function.

3.2.1.6 Condensate and Feedwater System

3.2.1.6.1 System Function

The condensate and feedwater system is designed to supply feedwater to the steam generator secondary side during all normal operating conditions. The condensate and feedwater system pumps take suction from the condenser hotwell and deliver water to the steam generators via the feedwater heaters. On receipt of a main feedwater (MFW) isolation signal, the MFW system isolates so that feedwater will not be delivered to the steam generators. Main feedwater isolation is not modeled in the system analysis and is guaranteed successful in cases without anticipated transient without scram (ATWS). In the accident scenarios in which auxiliary feedwater is lost, restoration of main feedwater is necessary to preserve the heat sink for the reactor.

3.2.1.6.2 System Operation

The steam discharged from the low pressure turbines is condensed in the condenser and collected in the hotwell. The condensate is taken from the condenser hotwell by three vertical, centrifugal, motor-driven hotwell pumps. Three horizontal, centrifugal, motor-driven condensate booster pumps are used to provide the condensate, with sufficient net positive suction head (NPSH), to the suction of the main feedwater pumps. The two turbine-driven, variable speed main feedwater pumps are designed to supply feedwater to the feedwater header and then distribute to the four steam generators under all expected operating conditions. Feedwater regulating valves (FCV-3-35/FCV-3-48/ FCV-3-90/FCV-3-103) are put on automatic controls at about 22% power. Bypass regulating valves are then manually controlled. Feedwater control to the individual steam generator is regulated automatically above 22% power by the adjustment of feedwater regulating valves (FCV-3-35/FCV-3-48/FCV-3-90/FCV-3-103) in the piping to each steam generator. The position of the feedwater regulating valve is determined by a three-element controller that uses steam generator water level, steam flow, and feedwater flow as the control variables. During startup or operation below 22% power, the bypass regulating valves operate automatically with the feedwater regulating valves in manual. If operating above 22% power when the flow through the feedwater regulating valves drop below 14%, then the flow automatically transfers to the bypass lines. The MFW bypass regulating valves (FCV-3-35A/FCV-3-48A/FCV-3-90A/FCV-3-103A) are controlled in auto by (1) steam generator level program, (2) actual steam generator level, and (3) auctioneered high nuclear power.

The motor-driven standby main feedwater pump (SBMFP) can be started locally or from the main control room. Lube oil is supplied to the bearings by a shaft-driven oil pump. The SBMFP will be started automatically when either MFPT is tripped and the plant load is greater than 67%. The SBMFP will trip upon receiving a water hammer signal. In addition, low injection water pressure, low oil pressure, a stop signal, or a turbine trip and a main feedwater isolation signal will cause the SBMFP to trip automatically.

The heating of the condensate/feedwater is achieved by directing the condensate and feedwater flow through (1) gland steam condenser, (2) main feedwater pump turbine condensers, and (3) feedwater heaters. The gland steam condenser condenses the steam leakoff from all turbine shaft seals and removes the noncondensibles from the steam. A weighted check valve (FCV-2-260) is provided in a bypass around the gland steam condenser to ensure minimum required flow through the condenser at low condensate flow conditions. The feedwater heaters contain three parallel strings of heaters, each string consisting of three low pressure feedwater heaters, three intermediate pressure feedwater heaters, and one high pressure feedwater heater.

Minimum flow bypasses are provided for equipment protection in the condensate and feedwater system. The condensate system minimum flow bypass (through FCV-2-35A) is located immediately upstream of the No. 7 heaters. The valve position is modulated to maintain the flow through the nozzle located upstream of the gland steam condenser, which will protect the hotwell and condensate demineralizer pumps and provide adequate cooling water to the gland steam condenser at all times. An automatic recirculation control valve (2-604/2-605/2-606) is installed at the discharge of each condensate booster pump to maintain a minimum flow through the pump. A minimum flow bypass line located downstream of each feedwater pump is designed to permit direct recirculation back to the condenser. The miniflow bypass control valve (FCV-3-70/FCV-3-84) modulates in a manner similar to the condensate minimum flow valve and maintains a minimum flow through each operating main feedwater pump. FCV-3-208 maintains a minimum flow for the standby main feed pump. Minimum flow bypass lines are provided for numerous components; e.g., hotwell pumps, gland steam condenser, SGBD heat exchanger, MFPT condenser, and condensate demineralizers.

The feedwater flow to the secondary side of the steam generators provides a heat sink for the heat generated in the reactor core, which is carried by the RCS to the primary side of the steam generators. A loss of feedwater flow to the steam generators will result in the reduction of reactor core heat removal capability. If the reactor is not tripped and a coolant supply provided to the secondary side of the steam generators is unavailable, core damage could occur.

3.2.1.6.2.1 Normal Operation. The condensate/feedwater system normally operates at full load with three hotwell, three condensate demineralizer (in the condensate polisher portion of the system, not shown in the simplified drawing), three condensate booster, and two main feedwater pumps in service. With only one main feedwater pump running, the unit load can be continuously maintained at 67%. The motor-driven main feedwater pump is normally in standby, following a plant startup.

During plant startup, the low-load automatic feedwater control system is used up to approximately 22% of full feedwater flow. Before switching to the lower preheat feedwater nozzle, the feedwater in the line to the lower preheat feedwater nozzle must be warmed to not less than 250°F to minimize the potential for waterhammer (steam bubble collapse) in the steam generator preheater. Waterhammer could occur if cold feedwater were injected into the steam generators. Waterhammer pressure pulses can be generated in the feedwater system as a result of (1) feedwater isolation valve and control valve opening or closing, (2) check valve closing, (3) pumps starting and stopping, and (4) steam bubble collapsing at the steam generator feedwater nozzles. Feedwater line warming is accomplished by flushing the cold water in the line downstream of the main feedwater regulating valves through the deaeration line to the condenser. This forward flush operation is accomplished at 20% to 22% feedwater flow by flowing hot feedwater through this flow path to the condenser until temperature instrumentation in the feedwater line upstream of the junction with the deaeration line indicates a temperature greater than 250°F. During forward flush, the isolation valves upstream of each main feedwater regulating valve are closed, and the bypass around each of these valves is opened. This procedure will cause the pressure upstream of the closed main feedwater isolation valves to be lower than the steam generator pressure. Consequently, no cold water in the main feedwater line can leak into the steam generator and cause waterhammer (bubble collapse) in the steam generator. The Watts Bar units use the model D3 steam generator on which the feedwater inlet is located close to the tube-sheet. It is therefore very unlikely that steam generator level will drop below the inlet nozzle. Nevertheless, the feedwater line layout is such that the portion of feedline susceptible to drainage is minimized. Therefore, in the unlikely event that a waterhammer in the feedwater lines does occur, the resulting force will be minimal.

If condensate demineralizers are in service at higher loads, additional pumps are needed to provide sufficient NPSH to the main feedwater pumps. The condensate demineralizer pumps provide this capability for all normal operating conditions by automatically starting at approximately 80% of full feedwater flow.

3.2.1.6.2.2 <u>Accident/Transient Operation</u>

3.2.1.6.2.2.1 Feedwater Isolation. Receipt of any of the following signals from the reactor protection system, results in a complete isolation of feedwater to all of the steam generators

- High-High Steam Generator Level in Any Steam Generator (2/3 channels to trip)
- Safety Injection Signal
- Reactor Trip Coincident with Reactor Coolant Low-Low Average Temperature

Note: The MFW isolation signal is modeled in the engineered safety features actuation system (ESFAS).

The feedwater isolation signals initiate the closure of the feedwater isolation valves FCV-3-33/FCV-3-47/FCV-3-87/FCV-3-100, feedwater regulating valves FCV-3-35/FCV-3-48/FCV-3-90/FCV-3-103, and feedwater bypass control valves FCV-3-35A/ FCV-3-48A/FCV-3-90A/FCV-3-103A, and feedwater bypass isolation valves (FCV-3-236/ FCV 3-239/FCV 3-242/FCV 3-245) for the steam generators and trips both the main feedwater pumps. Trip of the main feedwater pumps or a main feedwater isolation will shut down the remainder of the condensate and feedwater system pumps except the hotwell pumps (i.e., the condensate booster pumps and demineralized condensate pumps) to maintain the steam seals, gland steam condensers, and other components. High level in any one steam generator initiates a rapid closure of the FCV pertaining to the affected steam generator.

When operating at above 67% load and when loss of one main feedwater pump occurs, the standby main feedwater pump starts automatically, and load runback is initiated. Unit load is decreased to 85%.

3.2.1.6.2.2.2 Feedwater Restoration. As described above, main feedwater is to be restored on loss of AFW. Therefore, two modes of condensate and feedwater restoration are described as follows:

The short-cycle recirculation of the condensate system is provided with a minimum flow bypass valve (FCV 2-35A) located downstream of FCV 2-275. The flow path is started from the hotwell pumps to the condenser through the condensate demineralizers (if applicable), gland steam condensers, steam generator blowdown second-stage heat exchanger, and MFPT condensers. The bypass valve (FCV 2-35A) receives its operating signal from a flow device located in the hotwell pump discharge line. The purposes of this bypass valve are to protect the condensate system pumps and to provide cooling water (condensate) flow to the gland steam condensers, steam generator blowdown second-stage heat exchanger, and MFPT condensers at all times.

The long-cycle recirculation of the feedwater system is provided with a minimum flow valve (PCV-3-40 and FCV-3-195) located downstream of the regulator valve (FCV-3-35, FCV-3-48, FCV-3-90, or FCV-3-103). It is used when the steam generator temperature is 200°F or less, the condensate pressure is less than 350 psig, the steam generator pressure is atmosphere or greater (nitrogen overpressure), and the condensate temperature is greater than 100°F. Two of the three hotwell pumps are needed to circulate condensate and feedwater for long-cycle recirculation. The flow path is started from the hotwell pumps to the downstream of the feedwater regulator valve through condensate system, low pressure heaters, condensate booster pump, intermediate pressure heaters, standby MFW pump, and high pressure heaters.

Note that the long-cycle valves are not modeled because it is assumed that the operator will isolate them during a plant startup.

3.2.1.7 Engineered Safety Features Actuation System

3.2.1.7.1 System Function

The reactor protection system (RPS) is composed of two major systems: the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS). The function of the RTS is to ensure that the reactor operates within the safe operating region identified by Westinghouse and the U.S. Nuclear Regulatory Commission (NRC). The ESFAS is provided to sense accident situations and initiate the operation of necessary engineered safety features (ESF).

ESFAS senses selected plant parameters, determines whether established safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of a primary or secondary system boundary rupture.

The specific plant parameters considered in this analysis are as follows:

- Low Pressurizer Pressure
- High and High-High Containment Pressure
- Low-Low Steam Generator Level
- Low Steam Line Pressure
- High-High Steam Generator Level
- High Negative Steam Pressure Rate

Figure 1 in Section 2 of the system notebook shows the safety features included in the ESFAS model and the parameters used to generate the actuation for each feature.

The ESFAS generates a multitude of signals. Only the following safety features are modeled:

- Safety Injection
- Containment Isolation, Phase A and Phase B
- Containment Spray Actuation
- Main Feedwater Isolation
- Main Steam Isolation
- Auxiliary Feedwater Actuation
- Containment Ventilation Isolation
- Containment Air Return Fans Actuation

The ESFAS is composed of the following systems:

1. Process Protection Set Racks (EAGLE 21)
2. Solid State Protection System (SSPS)
3. Safeguards Test Cabinet (STC)
4. Manual Actuation Circuits

The process protection system and the SSPS are the only systems modeled in this analysis. These two systems

- Generate all necessary process protection signals, combine them into logic matrices, and initiate a reactor trip or actuate necessary ESF equipment.

- Maintain physical and electrical separation by providing four sets of process protection system (EAGLE 21) cabinets and two sets of SSPS cabinets (racks), one for each protection train (A and B).

The EAGLE 21 process protection system is a microprocessor-based system housed in four instrumentation cabinets corresponding to the four protection channel sets. In each protection channel set, the process electronics power the sensors and perform signal conditioning, calculation, and isolation operations on the input signals. The analog input module of the system powers the field sensor(s) and performs signal conditioning. All calculations for the process channel functions are performed by the loop calculation processor (LCP), and channel trip signals are provided through the partial trip output modules to the protection logic circuits of the SSPS. The EAGLE 21 process protection system provides for continuous online self-calibration of all analog input signals, and the system continuously monitors itself for malfunctions.

The SSPS is a dual-train, redundant protection system housed in two 3-bay cabinets, one single bay control board demultiplexer cabinet, and a computer monitored demultiplexer assembly (see Figure 2 in Section 2 of the system notebook for a simplified diagram). Each 3-bay cabinet contains an input relay bay, a logic bay, and an output relay bay. The inputs are transmitted through AC-operated relays that separate SSPS logic circuits from the protection set inputs. The output relays consist of master and slave relays with the slave relays driven by the master relays.

3.2.1.7.2 System Operation

3.2.1.7.2.1 Normal Operation. During normal plant operation, the ESFAS process protection channels are energized (except for the high-high containment pressure) and the parameters listed in Table 1 in the system notebook are continuously monitored. The process protection system (EAGLE 21) sends "keep alive" pulse signals every 100 milliseconds to the comparators via the digital signal converter. If the timer (deadman) in the comparators senses a delay of these pulse signals by more than 20 milliseconds, the comparators will trip. This will be alarmed or indicated in the control room. The process protection system ensures continuity of circuitry.

3.2.1.7.2.2 Accident/Transient Operation. When a transient or an accident occurs, various signals are generated, depending on the event that initiated the transient or accident. These signals provide actuation for the equipment that is expected to operate automatically during the transient. The signals generated for each initiating event category are listed in Section 1.2 and described in Table 2 in the system notebook. Manual actuation of the ESFAS signals is also available. This action is modeled in Top Event OS.

3.2.1.8 Reactor Protection System

3.2.1.8.1 System Function

The reactor protection system (RPS) comprises two subsystems: (1) reactor trip subsystem (RTS), and (2) the engineered safety features actuation subsystem (ESFAS). RTS provides automatic protection against unsafe reactor operation. ESFAS uses selected plant parameters to determine any rupture in the primary or secondary plant boundary conditions and sends signals to mitigate the possible consequences of faulted conditions. In the present notebook, the analysis will be confined to the functions of the RTS.

The major equipment lying within the boundary lines of the RPS model are the reactor trip breakers and anticipated transient without scram (ATWS) mitigating system actuation circuitry (AMSAC). ESFAS is composed of

1. Process and control instrumentation protection racks (Eagle 21).
2. Solid state protection system (SSPS).
3. Engineered safety features (ESF).
4. Manual actuation circuits.

The boundary conditions and functions of these subsystems of ESFAS are discussed in the ESFAS notebook.

ESFAS monitors the safe operational boundary conditions of the reactor. It generates and transmits signals to the appropriate equipment whenever the safe operating limits of the reactor are nearing a breach point. On receipt of the aforementioned signals from ESFAS, the RTS shuts down the reactor to protect it against either gross damage to fuel cladding or loss of system integrity, which could lead to release of radioactive fission products into the containment. This protection function is being modeled under Top Event RT, wherein the hardware required to accomplish the trip and the actions of the operators are analyzed.

Several other events relating to the failure of the RTS to trip the reactor are included in the model. If the reactor fails to trip, the Emergency Operating Procedures instruct the operators to manually trip the reactor. If the manual reactor trip fails, then the operators are instructed to insert the control rods. The manual insertion of the rods is modeled as Top Event MR.

Top Event AM models the availability of the AMSAC (ATWS mitigating system actuation circuitry) to actuate auxiliary feedwater and trip the turbine. AMSAC is actuated if an ATWS event is initiated at or above an initial power level of 40%. Top Event PL represents the fraction of transients that occur above an initial power level of 40%.

3.2.1.8.2 System Operation

Details of system operation applicable to Top Events RT and AM are given below. Operational details applicable to Top Events PL and MR will not be discussed.

3.2.1.8.2.1 Normal Operation. RPS is a safeguard system. During normal operating conditions, RPS monitors the specific variables considered to be essential for reactor protection (functions carried out by RTS) or variables required to mitigate the consequences of faulted conditions (functions carried out by ESFAS). A simplified drawing of the equipment required for reactor trip is shown in the system notebook as Figure 1.

AMSAC is not required during normal operation of the plant. AMSAC is a nonsafety-related system designed to provide backup to the RPS as an alternate means of tripping the turbine and actuating AFW. It is not addressed in the Technical Specifications. Normally, the RPS/ESFAS would actuate turbine trip and auxiliary feedwater flow before the conditions that cause AMSAC actuation are reached. If a common mode failure in the RPS were to fail to initiate auxiliary feedwater flow or turbine trip in addition to prohibiting a reactor trip, then AMSAC is an alternative means of providing auxiliary feedwater flow and turbine trip. A simplified drawing of the AMSAC circuitry is shown in Figure 2 in the system notebook.

3.2.1.8.2.2 Accident/Transient Operation

3.2.1.8.2.2.1 **Reactor Trip System.** The reactor trip system functions by interrupting power to the rod control system, causing the control rods to drop into the core. The two reactor trip breakers, 52/RTA and 52/RTB, are arranged in series between the control rod motor generator set switchgear and the control rod drive power cabinets. A breaker can be tripped electrically by either of two methods: operation of the undervoltage trip attachment (UVTA) and/or the shunt trip attachment (STA). The UVTA is energized with a DC voltage (+48V: the SSPS normal output) when the breaker is closed, causing the moving core on the UVTA to restrain the trip spring. A loss of DC voltage (0V: SSPS trip signal) to the UVTA removes the restraint and allows the trip spring to trip the breaker. The STA is normally deenergized. A trip signal energizes a coil on the STA, causing the armature to push in a trip lever that causes the breaker to trip. An auxiliary relay, the auto shunt relay, is continuously energized from the same voltage as the UVTA coil. When the voltage is switched off, both the UVTA coil and the auto shunt trip relay are deenergized. The UVTA trip lever is released while a contact of the auto shunt trip relay closes to energize the STA coil.

During normal operation, the main breakers (RTA and RTB) are closed, and the bypass breakers (BYA and BYB) are open. During surveillance testing of a main breaker, the bypass is closed. Thus, the power to the rod control switchgear is through the main breaker of one train and the bypass breaker of the opposite train.

3.2.1.8.2.2.2 **ATWS Mitigating System Actuation Circuitry (AMSAC).** Tripping of the turbine on a loss of feedwater ATWS event causes a rapid reduction in steam flow out of the steam generators and a resultant rapid increase in steam pressure to the steam line safety valve setpoint pressure. Turbine trip extends steam generator inventory and results in an increase in core coolant temperature. The increase in coolant temperature causes a decrease in core power early in the transient before steam generator tubes have begun to uncover. Later, as the steam generator tubes uncover, the rate of increase in reactor coolant system (RCS) pressure is lower because it started at a lower core power level. Without the turbine trip, the steam generators will continue to boil off their inventory at the same rate as before. After the steam generator tubes are exposed, the heat transfer from the primary to the secondary will decrease dramatically. The resulting temperature rise can result in an RCS pressure above 3,200 psig.

AMSAC actuation is dependent on turbine load and steam generator level. The turbine load is obtained from turbine first stage pressure signals as indicated by PT-1-314 and PT-1-315. The steam generator levels are provided by LT-3-172, LT-3-173, LT-3-174, and LT-3-175 [AFW turbine-driven level transmitters to level control valves (LCV)]. The AMSAC system is armed if both PT-1-314 and PT-1-315 indicate turbine load is above 40% (AMSAC signal blocked below 40% turbine load). The AMSAC signal is generated when three of the level transmitters indicate low water level.

The design uses dual processors in a one-out-of-two voting configuration to issue the actuating signal. Each processor scans the four signals of steam generator level and two signals of turbine pressure. If three of the four levels are below the trip setpoint, which depends on turbine power, the logic closes the contacts that actuate auxiliary feedwater and trip the turbine. There is an actuation delay to allow the RPS to respond first.

3.2.1.9 Auxiliary Feedwater System

3.2.1.9.1 System Function

The safety function of the auxiliary feedwater (AFW) system is to supply, in the event of a loss of main feedwater (MFW), a sufficient feedwater flow to the steam generators to remove primary system stored and residual core energy. It may also be required in circumstances such as the evacuation of the main control room, cooldown after a loss of coolant accident for a small break, maintaining a water head in the steam generators following a loss of coolant accident, or a flood above plant grade.

The system is designed to start automatically in the event of a loss of offsite electrical power, safety injection, low-low steam generator water level, or a trip of both main feedwater pumps, any of which will result in, may be coincident with, or may be caused by a reactor trip. It will supply sufficient feedwater to prevent the relief of primary coolant through the pressurizer safety relief valves and the uncovering of the core. It has adequate capacity to maintain the reactor at hot standby and then to cool the reactor coolant system (RCS) to the temperature at which the residual heat removal system may be placed in operation, but it cannot supply sufficient feedwater for power generation.

3.2.1.9.2 System Operation

The AFW system, as modeled, consists of two motor-driven pumps (410-gpm rated flow each), a turbine-driven pump (720-gpm rated flow), and the necessary level control valves to maintain steam generator level.

3.2.1.9.2.1 Normal Operation. The AFW system has no normal power operation function. However, it may be used during plant startup and normal plant cooldown when the preferred supply from the MFW is unavailable. During normal power operation, the AFW system is aligned for automatic delivery of flow from the Unit 1 CST to the steam generators.

Motor-driven pump 1A-A delivers flow to steam generators 1 and 2, and motor-driven pump 1B-B services steam generators 3 and 4. Both pumps are aligned for autostart. Turbine-driven pump 1A-S provides flow to all four steam generators. Like the motor-driven pumps, the turbine-driven pump is aligned in autostart mode during normal power operation. The normal steam source for the turbine is main steam piping from steam generator number 1 upstream of the main steam isolation valve (MSIV). The motor-operated FCV controlling this steam supply is maintained in the open position. In the event that steam is unavailable from this source, the turbine is automatically aligned to the backup source, main steam piping from steam generator number 4 upstream of the MSIV.

Both turbine steam supplies provide steam to a common turbine inlet header containing two motor-operated FCVs. Each steam generator inlet line serviced by turbine-driven pump 1A-S contains a 3-inch level control valve that modulates to control pump 1A-S flow so that steam generator level is maintained. These normally closed air-operated valves are aligned to control steam generator level when turbine-driven pump 1A-S operates.

The DWST serves as the makeup water supply for the condensate storage tank, delivering flow through one of the two motor-driven pumps (MPMP31 and MPMP32) if the water inventory needs to be replenished. During normal operation, the line from the DWST to the CST is closed. In addition, the essential raw cooling water (ERCW) system serves as an unlimited backup water supply to the AFW pumps. Main ERCW discharge header 1A provides an alternate supply to motor-driven pump 1A-A (through FCV-3-116A,B) and to turbine-driven pump 1A-S (through FCV-3-136A,B). Similarly, main ERCW discharge header 1B provides backup water to motor-driven pump 1B-B (through FCV-3-126A,B) and to turbine-driven pump 1A-S (through FCV-3-179A,B). There are two motor-operated FCVs in series in each supply line from the main ERCW header to the pump suction (eight total). During normal operation, these valves are closed to isolate the AFW system from the low quality ERCW. The normal operation alignment of the valves allows for automatic switchover of the suction of a running pump to ERCW in the event of low suction pressure in the pump supply line coincident with a time delay.

3.2.1.9.2.2 Accident/Transient Operation. Section 1.3.2.1 of the system notebook provides a description of the equipment and components in each top event model. Section 1.3.2.2 of the notebook provides a detailed description of the operation of the AFW system during accident conditions.

3.2.1.9.2.2.1 Top Event Boundaries. Top Event CT models the immediate and extended availability of the CST as a supply of feedwater to the AFW system. Top Event CT models the ability of the condensate storage tank to deliver required flow to the suction of AFW pumps. The major components include the condensate storage tank A itself and two discharge valves, 1-2-504 and 1-3-800.

Top Event CTMU models the long-term operability and availability of the demineralized water supply as a makeup to the condensate storage tank. This top event models the operator action and the ability of the makeup pumps to deliver flow to the CST. The major components include the DWST itself, associated valves, and the two motor-driven pumps.

Top Event MA models the ability of motor-driven pump A to take suction from the CST and deliver it to the header that provides flow to steam generators 1 and 2. The major components include the pump itself, discharge check valve 3-820, discharge pressure control valve 3-122, manual suction valve 3-803, suction check valve 3-805, and the alternate ERCW suction line FCV 3-116A and 3-116B. All relays and time delays for the swapover to ERCW are implicitly included in the ERCW valves (with the exception of the pressure switches 139A, 139B, and 139D, which are explicitly modeled).

Top Event MB is the train B analogy to Top Event MA. It models the ability of motor-driven pump B to take suction from the CST and deliver it to the header that provides flow to steam generators 3 and 4. The major components include the pump itself, discharge check valve 3-821, discharge pressure control valve 3-132, manual suction valve 3-804, suction check valve 3-806, and the alternate ERCW suction line FCVs 3-126A and 3-126B. All relays and time delays for the swapover to ERCW are implicitly included in the ERCW valves (with the exception of the pressure switches 144A, 144B, and 144D, which are explicitly modeled).

Top Event TP models the ability of the turbine-driven pump to take suction from the CST and deliver flow to the header that provides flow to all four steam generators. Major equipment includes the steam generator 1 supply valves FCV-1-15 and CKV-3-891, the steam generator 4 supply valves FCV-1-16 and CKV-3-892, steam supply motor-operated valves (MOV) FCV-1-17 and FCV-1-18, the pump 1A-S, discharge check valve 3-864, and valves 1-ISV-3-809 and 1-CKV-3-810. The relays and time delays for steam supply swapover are implicitly included with the steam supply valves. The trip and throttle valve FCV-1-51, the governor valve 1-FCV-52, and the instrumentation required for the operation of the governor valve (FE-3-142, FT-3-142, FIC-3-142, etc.) are implicitly included with the turbine pump. The alternate ERCW suction line FCV 3-136A, 3-136B, 3-179A, and 3-179B are included and modeled; however, all relays and time delays for the swapover to ERCW are implicitly included in these ERCW valves (with the exception of the pressure switches 121A, 121B, and 121D; and 125A, 125B, and 125D, which are explicitly modeled).

Top Event AF models the ability to accept flow from the AFW pumps and deliver flow to the steam generators. It also implicitly includes the ability to provide steam relief through the steam generator safety valves. Flow paths to all four steam generators are included. A flow path to a steam generator includes flow paths from both the motor-driven pump and the turbine-driven pump, and it includes the common line to the steam generator where the flows from the motor-driven pump and the turbine-driven pump share a common path. Using the path for steam generator 1 as an example, the turbine-driven pump flow path consists of valves 3-869, 3-174, 3-873, and 3-877. The motor-driven pump flow path consists of valves 3-828, 3-164, 3-832, and 3-836. The common flow path consists of check valves 3-655 and 3-656. (A simplified drawing can be found in the system notebook.) The flow paths for the other steam generators are defined in a similar fashion. The operator action to manually open and control the LCVs on the AFW pump lines in the event of complete loss of AC power is also included in this top event.

Top Event DS models the human action and the ability of steam generator PORV to depressurize the secondary side and to cool down the RCS. The major components include manual isolation valve 1-619, the PORV itself (1-5, only one PORV is modeled, see modeling assumptions) and the pressure switch 1-6. The operator action to align the system properly is also included in this top event. Six dynamic human action values for different scenarios are modeled in the fault tree for this top event with the capability of switching each one on or off.

The intermediate Top Events MAMB, TPMA, and TOT have the same boundaries as the individual Top Events MA, MB, and TP.

3.2.1.9.2.2.2 Description for Operation under Accident Conditions. Motor-driven pumps 1A-A and 1B-B start automatically in the event of a two out of three low-low level signal for any steam generator, a safety injection signal, a trip of both MFW pump turbines, or a loss of offsite power. The AFW system is also started by the AMSAC system during ATWS conditions. The motor-driven pumps may also be manually started via hand switches in the main control room.

The cooling for the motor-driven pump spaces is provided by the component cooling system (CCS) and AFW pumps space coolers 1A-A and 1B-B. The room cooling equipment is modeled in Top Event V3.

As the motor-driven pump breakers close to start the pumps, contacts off of the breakers open to allow the large LCVs in the steam generator inlet lines to open. These LCVs (1-LCV-3-164 and 1-LCV-3-156 from pump 1A-A and 1-LCV-3-148 and 1-LCV-3-171 from pump 1B-B) are then modulated to a preset open position by their respective level control circuits.

In addition, the LCVs can be locally controlled by isolating and depressurizing air to fail the valve open (if the valve has not already failed open on loss of air) and then modulating flow with either the upstream or downstream isolation valve (1-3-828 or 1-3-836 for 1-LCV-3-164). This local operation requires recovery of the isolation valves from their locked-open position and direct communication with the unit operator.

Steam generator level control (i.e., control of AFW flow into the steam generators) during cooldown is assumed to be accomplished by modulating the large LCVs only. The smaller bypass LCVs in the motor driven pump inlet lines to the steam generators are designed to open during a cooldown to RHR conditions. However, with the conservative model used for cooldown (only 1 PORV assumed available) it is not necessary to model the bypass LCVs

Turbine-driven pump 1A-S starts automatically in the event of a two out of three low-low level signal for any two steam generators (train A or B), a train A or B safety injection signal, a start signal, a trip of both main feedwater pump turbines, or a loss of offsite power. Turbine-driven pump 1A-S can be manually started and operated from the main control room (MCR) or from a local control station.

Should the normal turbine steam supply from steam generator number 1 via MOV 1-FCV-1-15 fail, an automatic swapover to the alternate supply from steam generator number 4 via MOV 1-FCV-1-16 is initiated. The steam supply swapover can also be accomplished manually from the MCR or from the associated 480V reactor motor-operated valve (RMOV) board.

The ventilation for the turbine-driven pump room is provided by a DC ventilation fan and an AC ventilation fan.

Following start of turbine-driven pump 1A-S, pressure switches 1-PS-3-138A and 1-PS3-138B on the pump discharge close on increasing pressure to allow an associated pair of LCVs in the steam generator inlet lines of the pump to open. These LCVs (1-LCV-3-172, 1-LCV-3-175, 1-LCV-3-173, and 1-LCV-3-174) are then modulated to a preset open position by dedicated level control circuits similar to those for the motor-driven pump LCVs.

The Unit 1 CST serves as the normal water supply for the AFW pumps with ERCW as an automatic backup source. The motor-operated ERCW supply valves servicing the motor-driven pumps (1-FCV-3-116A and 1-FCV-3-116B for pump 1A-A and 1-FCV-3-126A and 1-FCV-3-126B for pump 1B-B) open automatically when the pump is running and the suction pressure drops below a predetermined setpoint psig as sensed by two out of three pressure switches (1-PS-3-139A, 1-PS-3-139B, and 1-PS-3-139D for pump 1A-A and 1-PS-3-144A, 1-PS-3-144B, and 1-PS-3-144D for pump 1B-B) for a specified duration.

There are two sets of motor-operated ERCW supply valves for turbine-driven pump 1A-S. The first set (1-FCV-3-136A and 1-FCV-3-136B) opens automatically when suction pressure drops below a predetermined setpoint as sensed by two out of three pressure switches (1-PS-3-121A, 1-PS-3-121B, and 1-PS-3121D) for a specified duration. This set returns closed if suction pressure does not return in another 4 seconds. The second set (1-FCV-3-179A and 1-FCV-3-179B) opens automatically when suction pressure has remained below a predetermined setpoint as sensed by two out of three pressure switches (1-PS-3-125A, 1-PS-3-125B, and 1-PS-3-125D) for a specified duration.

There are other options for aligning AFW to the backup ERCW supply. The ERCW supply valves can be manually operated from the MCR by turning appropriate hand switches to OPEN or CLOSE. Each valve can also be similarly operated via a hand switch at a local control station. In addition, handwheels are mounted on the valves for local operation.

In addition to the automatic ERCW backup supply, the Unit 1 and Unit 2 CSTs can be manually crosstied.

3.2.1.10 Main Steam System

3.2.1.10.1 System Function

3.2.1.10.1.1 Normal Conditions. The main steam system (MSS) is designed to conduct steam from the steam generator outlets to the high-pressure turbine or to the condenser turbine bypass (steam dump) system. This system also supplies steam to the feedwater pump turbines, auxiliary feedwater pump turbines, moisture separator reheater, and turbine seals. The MSS includes self-actuating safety valves to provide emergency pressure relief for steam generators and atmospheric relief valves to provide the means for plant cooldown by steam discharge to the atmosphere if the turbine bypass system is not available. The normal functions of the MSS can be summarized as follows: (1) transport main steam from the steam generators to the final steam users, including the turbine generator and main feedwater pump turbines (MFPT), (2) provide steam dump to the condenser for control of nuclear steam supply system (NSSS) temperature and steam generator pressure during all phases of operation (startup, shutdown, and load rejection), and (3) drain condensate, which accumulates in the main steam piping during initial heatup and normal operation.

The primary functions for the condenser are to condense the steam from the turbine exhaust and turbine bypass and to remove the dissolved gases from the condensate. The steam generator blowdown valves control secondary plant water inventory. The isolation function of these valves during a steam generator tube rupture (SGTR) is modeled under Top Event SL in the Steam Generator system notebook.

The equipment required for successful turbine trip are also modeled as a part of this system analysis.

3.2.1.10.1.2 Accident Conditions. The MSS is required to perform the following safety functions during accident conditions:

1. To provide main steam isolation in the event of a break in the steam line upstream of the MSIV or auxiliary feedwater pump turbine steam supply line from one steam generator, a break in the steam header downstream of the main steam isolation valve, a steam generator tube rupture, and in feedwater line rupture situations.

2. To provide steam generator overpressure control during accident/emergency transients.

3. To control steam cooldown of the NSSS to the point that the residual heat removal (RHR) system can be used for closed-loop cooling following accidents or emergency transients.

4. To provide steam to the auxiliary feedwater pump turbine to ensure removal of core decay heat in the event that the motor-driven auxiliary feedwater pumps are not available.

5. To provide isolation of the steam lines for events in which the containment isolation is required.

In this system analysis, the main steam isolation functions for steam line breaks and turbine trip functions are modeled. The main steam isolation function for steam generator tube ruptures is modeled in top event SL in the steam generator isolation system notebook.

The condenser serves as a heat sink for the secondary cooling via turbine bypass valves during accident conditions. The turbine bypass system provides an alternate steam path to the condenser following a large turbine load reduction and thus reduces the magnitude of the nuclear system transients. If the condenser is not available, secondary pressure control and cooling is provided by steam relief through the steam generator power-operated relief valves (PORV) to the atmosphere. Cooldown using the steam generator PORVs is modeled in Top Event DS of the auxiliary feedwater system notebook. Twelve turbine bypass valves (FCV-1-103 through FCV-1-114), connected to the condenser, are provided in Watts Bar. Three of them (FCV-1-103, FCV-1-107, and FCV-1-111; two to the LP-zone, one to the IP-zone) are designated as "cooldown" valves, and they are able to bypass manually the interlock generated by the low-low RCS average temperature condition. This will block the air supply to each turbine bypass valve operator, thereby preventing the valve from opening.

The turbine generator is protected by the turbine protection system. The turbine protection system automatically trips the turbine by closing the turbine stop valve (FCV-1-61, FCV-1-64, FCV-1-67, and FCV-1-70) and the turbine control valve (FCV-1-62, FCV-1-65, FCV-1-68, and FCV-1-71) in each steam supply line to the high-pressure turbine on evidence of low condenser vacuum, excessive shaft vibration, abnormal thrust bearing wear, or low bearing oil pressure. The turbine trip system is also equipped with solenoid-operated trip devices, which provide means to initiate direct tripping of the turbine upon receipt of appropriate electrical signals. The turbine can be tripped manually on detection of a high temperature or pressure difference between condenser shells, high back pressure on the main condenser, high journal or thrust bearing metal temperature, high bearing discharge temperatures, and high differential expansion. The turbine trip isolates the main turbine from the steam generator and thus prevents the reactor coolant system (RCS) from overcooling.

3.2.1.10.2 System Operation

3.2.1.10.2.1 Normal Operation. The MSS is designed to transport steam from each of the four steam generators through MSIVs to a main steam supply header from which the steam is distributed to the high pressure turbine, moisture separator reheaters, main feedwater pump turbines, or the steam dump system. The steam generator safety valves and atmospheric relief valves are located upstream of the MSIVs. The steam generator safety valves provide emergency pressure relief for the steam generators and the MSS piping upstream of the MSIVs in the event that the rate of steam generation exceeds that of steam consumption. The atmospheric relief valves provide a means for plant cooldown and steam generator pressure control by discharging steam into the atmosphere, thereby avoiding unnecessary lifts of the steam generator safety valves if the condenser steam dump is not available. Steam supply to the auxiliary feedwater pump turbines is provided from either main steam lines number 1 or number 4 upstream of the MSIVs.

The main steam isolation bypass valve in each steam line is used to provide steam for downstream pipe warming and to equalize the pressure across the MSIV prior to opening it during plant startup. The main steam isolation bypass valve is a 2-inch, air-to-open, spring-to-close, and fail-closed globe valve that is normally closed during plant operation and must be closed on a main steam isolation (MSI) signal.

The turbine bypass system, including 12 turbine bypass (or steam dump) valves, is designed to reduce the magnitude of nuclear system transients following large turbine load reductions by dumping steam directly to the condenser. Piping for turbine bypass is run from the main steam header through the 12 turbine bypass valves and is then connected to the condenser; i.e., 8 turbine bypass valves for condenser zone A, 3 for condenser zone B, and 1 for condenser zone C. The turbine bypass system is designed to permit a direct bypass steam flow at 40% of rated turbine flow without causing a reactor trip.

The high pressure turbine receives steam from the main steam header through the four main steam supply lines. Each of the four main steam supply lines includes a turbine stop valve and a turbine control valve. The turbine control valve regulates the turbine speed and load, while the turbine stop valve provides turbine protection by shutting off the steam flow during abnormal situations.

3.2.1.10.2.2 Accident/Transient Operation. The MSIVs are provided to protect the plant during accident conditions such as main steam line breaks and steam generator tube rupture. The MSIVs are 32-inch, failed-closed, air-operated valves located in the main steam line downstream of the main steam safety valves and atmospheric valves. Each MSIV is designed to be capable of closing within 5 seconds upon receipt of an isolation signal with design steam flow in either the forward or the reverse direction.

An electrohydraulic control system is used for the controls of turbine stop valves and control valves. Each turbine stop valve is normally in the full open position. Each turbine control valve regulates the steam flow to the high pressure turbine. The stop valve and the control valve are opened by the EHC high pressure fluid and closed by spring action when EHC high pressure fluid is dumped. When a turbine trip is activated, energization of solenoid-operated valves FSV-47-24 and FSV-47-27 will dump the auto stop oil and the EHC fluid to the oil reservoirs and cause all stop valves and the control valves to close. In addition, the turbine trip signal will energize solenoids FSV-47-26A and FSV-47-26B in parallel to dump the EHC fluid and result in closure of the turbine control valves.

Each air-operated fail-closed turbine bypass valve is provided with three solenoid valves (e.g., FSV-1-103A, FSV-1-103B, and FSV-1-103D for FCV-1-103) in series in the air supply line to the valve operator. To open the turbine bypass valve, two methods are provided: (1) solenoids A and B must be energized for modulating control of bypass valves during normal operation while in the "steam pressure" or "T_{avg}" mode, or (2) all three solenoids must be energized to trip open the bypass valves while operating in the "T_{avg}" mode. The solenoids are powered from 125V DC battery boards I and II and 120V AC vital boards 1-I, 1-II, 1-III, and 1-IV. Loss of the air supply or the DC or AC power supply to the turbine bypass valves will prevent the valves from opening. If the valves are open, the loss of air or power supplies will trip them closed. The control system of the turbine bypass valves will block the air supply to the valves when the following conditions occur: (1) inadequate condenser circulating water, (2) high condenser pressure, or (3) low-low T_{avg}. (The signal can be bypassed for cooldown valves.)

As part of the steam generator outlet nozzle, the flow restrictor acts to limit the maximum steam flow and the resulting thrust forces on the main steam line caused by the cooldown of the RCS or by a main steam line break. In the event of a high-energy line break requiring steam generator isolation, an inappropriate opening of an MSIV bypass valve would defeat steam generator isolation. Therefore, the operator is required to reset these valves once they are closed subsequent to their use during plant startup.

3.2.1.11 Residual Heat Removal System

3.2.1.11.1 System Function

Five functions of the residual heat removal system (RHR) are modeled in this analysis:

1. The ability of RHR to provide low pressure injection to the reactor coolant system (RCS).

2. The automatic/manual switchover of the RHR suction from the refueling water storage tank (RWST) to the containment sump, including the valve realignment to

supply suction to the safety injection and centrifugal charging pumps during the recirculation mode.

3. Normal cooldown for decay heat removal with suction from the RCS loop 4 hot leg.

4. RHR spray as part of the containment heat removal spray system.

5. RHR hot leg recirculation.

The primary modeled functions of the RHR are those required during a loss of coolant accident (LOCA). During a LOCA, the RHR system operates as a subsystem of the emergency core cooling system (ECCS) and is required to function in the injection mode and the recirculation mode. During the injection phase, the RHR pumps take suction from the RWST and discharge to the RCS through the cold leg injection lines. During the recirculation phase, the RHR pumps take suction from the containment sump and pump through the RHR heat exchangers for cooling prior to returning the fluid to the RCS through the cold or hot leg injection lines. If RCS pressure is greater than the shutoff head of the RHR pumps, the RHR pumps maintain recirculation by providing suction to the safety injection pumps or the centrifugal charging pumps via the RHR heat exchangers. The RHR system may also be used as part of the containment heat removal spray system (CHRSS) 1 hour after a LOCA. In this mode, coolant is diverted from the low head injection path to the two RHR spray headers.

The normal plant cooldown function of the residual heat removal system, which is to transfer decay heat from the reactor coolant system to the component cooling system (CCS) when the RCS pressure and temperature are below 380 psig and 350°F, respectively, is also modeled for events in which depressurization is successful and further cooldown is required.

The simplified drawings 1 through 5 in Section 2 of the system notebook illustrate the RHR system configurations in different operation modes.

3.2.1.11.2 System Operation

The residual heat removal system is a safety-related system designed to perform functions during normal operations and accident conditions. The RHR consists of two independent pump trains in each unit. With the exception of the common piping described below, each loop is capable of performing the safety-related functions and normal operating functions of the system. Each loop consists of a pump, pump miniflow loop, a heat exchanger, and flow control and isolation valves. Both loops share a common heat exchanger bypass line, suction piping from the RCS, suction and discharge to the RWST, and a spool piping connection to the spent fuel pool cooling system (SFPCS).

3.2.1.11.2.1 Normal Operation. The normal functions of the RHR system are used during reactor startup, cooldown, and refueling. These normal functions of the RHR are (1) to transfer decay heat from the RCS to the component cooling system when the RCS pressure and temperature are below 380 psig and 350°F, (2) to maintain adequate RCS flow with the reactor coolant pumps off to ensure adequate chemical mixing, and (3) to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations. The RHR functions during startup and refueling are not modeled.

The RHR system is in standby during normal power operation. It is aligned for its ECCS function of low pressure injection. See Section 2 of the system notebook for a simplified drawing of the ECCS standby alignment. No valves are required to be realigned for injection to begin.

3.2.1.11.2.2 Accident/Transient Operation. The RHR system is designed to perform several safety functions during accident conditions which are modeled in this analysis. The system may be required to:

1. Provide low pressure injection to the RCS.

2. Provide normal cooldown for decay heat removal with suction from the RCS loop 4 hot leg.

3. Switch the RHR suction from the RWST to the containment sump, and, if necessary, provide suction to centrifugal charging pumps and SIS pumps.

4. Provide hot leg recirculation.

5. Provide RHR spray as part of the containment heat removal spray system.

3.2.1.11.2.2.1 RHR Low Pressure Injection Mode. The RHR cold leg injection components automatically activate when a safety injection signal is generated. The conditions that generate a safety injection signal are as follows:

- Low RCS pressure.
- High containment pressure.
- High steam generator differential pressure.
- High steam flow coincident with low T_{avg} or low steam generator pressure.

Upon receipt of a safety injection signal, the RHR pumps start; the RWST to RHR pump flow control valve (FCV 63-1), normally aligned open, provides a suction path from the RWST; and normally open RHR heat exchanger outlet valves (FCV-74-16 and FCV-74-28) provide a discharge path to the four RCS cold legs (two cold legs per RHR loop). Miniflow valves (FCV-74-12 and FCV-74-24) are opened or closed (depending on RHR injection flow into the RCS) by the respective flow switches (FS-74-12A/B or FS-74-24A/B). The borated water from the RWST will be at the outside ambient temperature, and thus will not require heat exchanger CCS flow. When the RWST level is low (29%) and the containment sump level is high (10%) and increasing, the RHR supply valves automatically swap and the recirculation mode begins. The alignment of the RHR system for low pressure injection is shown in Section 2 of the system notebook as Figure 1.

The successful completion of the RHR low pressure injection requires the following:

- One of two RHR pump trains.
- A suction path from either the RWST during the injection mode or the containment sump during the recirculation mode.
- A discharge path to the RCS cold leg injection paths.

Five RHR system top events are used to model low pressure cold leg injection. Top Event RW models the availability of the RWST. Top Event RI models the suction path from the RWST and the RHR cold leg injection path. Top Event RF models the suction path from the RWST and the RHR cold leg injection path for medium break LOCA and large break LOCA sequences in which it is assumed that only three of the RCS cold legs are effective for injection. Top Events RA and RB model the RHR pump trains A and B, respectively.

Top Event RW is also used to model SIS injection, CVCS injection, and containment spray supply. There is no other dependency between Top Event RW and other top events.

Top Events RA and RB model the ability of the RHR pump trains to deliver flow to the (1) RCS cold legs, (2) RCS hot legs, or (3) high head injection systems during containment sump recirculation for 24 hours. Top Events RA and RB are used with the RHR cold leg injection path RI, the large or medium LOCA cold leg injection path RF, the containment sump recirculation Top Event RR, the normal cooldown Top Event RD, the RHR spray Top Event RS, and the hot leg recirculation Top Event RH.

Top Event RI models the cold leg injection path for small break LOCA and non-LOCA initiating events in which the four RCS loops are available for injection. One of the four cold leg injection paths is required for 24 hours. This includes drawing suction from the RWST for 1 hour and from the containment sump for 23 hours. Top Event RI is used with the RHR pump train Top Events RA and RB, the containment sump recirculation Top Event RR, and the normal cooldown Top Event RD.

Top Event RF models the cold leg injection path for medium break LOCA and large break LOCA initiating events in which two of the three RCS loops are required for injection. The cold leg injection path is required for 24 hours. This includes drawing from the RWST for 1 hour and from the containment sump for 23 hours. RCS loop 4 is assumed to be unavailable due to the LOCA. Top Event RF is used with the RHR pump train Top Events RA and RB, and the containment sump recirculation Top Event RR.

3.2.1.11.2.2.2 Normal Cooldown Using RHR. In addition to the normal decay heat removal function of RHR during planned refueling shutdowns, the RHR system is also used to remove decay heat from the RCS for a steam generator tube rupture (SGTR) initiating event in which plant cooldown to RHR entry conditions is achieved, and it is not required to recirculate from the containment sump. Cooldown is performed for these events by drawing coolant from the hot leg of loop 4, removing heat with the RHR heat exchangers, and returning the coolant to the four cold leg injection lines (as shown in Section 2 of the system notebook as Figure 2). This is the normal path used during plant shutdown and can be used when the RCS temperature is less than 350°F and RCS pressure is below

380 psig. The alignment of the RHR decay heat removal requires opening of the two series RHR suction (from loop 4 hot leg) isolation valves FCV-74-1 and FCV-74-2. The two valves can be opened only when the RWST supply isolation valve FCV-63-1 and containment sump supply isolation valves FCV-63-72 and FCV-63-73 are fully closed, and the RCS pressure is below 380 psig.

Top Event RD models the equipment and operator actions used for normal RHR cooldown. Normal RHR decay heat removal has a mission time of 20 hours. As with the other top events, Top Event RD is used in combination with other top events to model the RHR normal cooldown. Top Event RD models the equipment and operator actions used for normal RHR cooldown, which includes the heat transfer function of the RHR heat exchangers and the RHR suction isolation valves that provide flow to the RHR pump trains (Top Events RA and RB). Top Event RI models the return of the coolant to the four RCS cold legs.

3.2.1.11.2.2.3 RHR Recirculation Mode. The RHR recirculation mode begins with the automatic opening of the containment sump supply valves (FCV-63-72 and FCV-63-73) and closing of the RWST supply valves (FCV-74-3 and FCV-74-21). The valve alignment for recirculation mode is shown in Section 2 of the system notebook as Figure 3. If automatic switchover does not occur, the operators are instructed by Emergency Instruction ES-1.2 to complete the switchover manually.

Component cooling water flow is manually established to the RHR heat exchangers to cool the flow from the containment sump before being discharged into the RCS or the suction of the safety injection or CVCS pumps.

RHR supply valves to the CVCS and safety injection valves (FCV-63-8 and FCV-63-11) are manually opened to establish a flow path from the containment sump through the RHR pumps and heat exchangers to the suction of the centrifugal charging pumps and safety injection pumps if required for high pressure recirculation. These valves (FCV-63-8 and FCV-63-11) are interlocked with the open position of the containment sump supply valves (FCV-63-72 and FCV-63-73) and the closed position of the safety injection pump miniflow valves and the safety injection discharge to the RWST isolation valves (FCV-63-3, FCV-63-4, and FCV-63-175) to ensure that the sump coolant is not pumped to the RWST. The two RHR crosstie valves (FCV-74-33 and FCV-74-35) are manually closed during the recirculation mode to separate the RHR train A and B flow paths for protection against a passive failure.

Switchover of the RHR pump suction to the containment sump from the RWST is required during an event when the RWST level is $\leq$29% and the containment sump level is $\geq$10%. Top Event RL models the RWST and the containment sump water level signals required to activate the automatic switchover. This automatic switchover provides the RHR pumps with suction from the containment sump, and additional manual valving provides suction to the safety injection pumps and centrifugal charging pumps from the RHR pump discharge. Top Events RVA and RVB model the containment sump swapover valves FCV-63-72, FCV-74-3, FCV-63-73, and FCV-74-21. These valves are required to open (or close) to align the containment sump to suction of the RHR pump trains.

Top Event RR models the remaining components required for the automatic or manual switchover to the containment sump. This model requires: (1) manual transfer of the suction of the safety injection pumps and centrifugal charging pumps from the RWST to the discharge of the RHR pumps, and (2) use of the RHR heat exchangers to remove heat from the containment sump recirculation flow. This top event has a mission time of 23 hours. This model can be used for low pressure recirculation by the RHR system or high pressure recirculation using the safety injection and CVCS pumps. One train of RHR, including the supply and discharge alignment and RHR heat exchanger, is required. Top Event RR is used with the containment sump availability Top Event SU (containment systems); the SIS Top Events S1, S2, and SI; the CVCS injection Top Events VA, VB, and VC; and the RHR pump, RHR spray, and hot leg recirculation Top Events RA, RB, RS, and RH.

The manual valve alignment is performed by the control room operators in accordance with ES-1.2, "Transfer to Containment Sump." This operator action is evaluated in the Human Reliability section of this PRA.

3.2.1.11.2.2.4 RHR Hot Leg Recirculation. Hot leg recirculation is initiated by the control room operators approximately 15 hours after transferring to cold leg recirculation. The valve alignment for hot leg recirculation is shown in Section 2 of the system notebook as Figure 4. This alignment is used to prevent boron precipitation, due to core exit boiling in the core, by recirculating back through the RCS hot legs. One RHR pump train is required for hot leg recirculation. The cold leg recirculation paths and RHR spray are isolated during hot leg recirculation.

Top Event RH models the operator actions and equipment required for hot leg recirculation. This top event models opening of one RHR pump train path and isolating the other train. This top event is used with the containment sump availability Top Event SU (containment systems), the containment sump recirculation Top Event RR, and the RHR pump train Top Events RA and RB. The mission time for Top Event RH is 8 hours to allow for the procedural waiting time of 15 hours after initiation of containment sump recirculation and a 1-hour delay after accident initiation before switching to sump recirculation.

The operator actions required to transfer to the hot leg recirculation (ES-1.3) are evaluated in the Human Reliability section of this PRA.

3.2.1.11.2.2.5 RHR Containment Spray. The RHR system can be used in the recirculation mode to supply part of its flow to the two parallel RHR spray headers. The valve alignment for RHR spray is shown as Figure 5 in Section 2 of the system notebook. One of the RHR spray header isolation valves (FCV-72-40 or FCV-72-41) must be opened by the operators to establish the flow path from the RHR pumps to the headers. The RHR containment spray system is a subsystem of the containment heat removal spray system that is used to control the pressure and temperature inside containment following a LOCA.

RHR spray is initiated by the control room operators when containment pressure is higher than 9.5 psi and at least 1 hour has passed since the beginning of a LOCA. As part of the containment spray heat removal system, the RHR spray, along with the containment spray system (CSS), functions to control the containment pressure following a LOCA. RHR spray is used during containment sump recirculation mode by diverting the injection flow from one RHR pump train to its associated RHR spray header.

Top Event RS models the equipment and operator actions required for RHR spray. This top event models isolating one of the two RHR injection paths and opening one of the two RHR spray paths. This top event is used with the containment sump availability Top Event SU (Containment Systems), the containment sump recirculation Top Event RR, and the RHR pump train Top Events RA and RB. The mission time for Top Event RS is 23 hours, which allows for the 1-hour postaccident procedural waiting requirement.

The operator actions required to activate the RHR containment spray (FR-Z.1) are evaluated in the Human Reliability section of this PRA.

3.2.1.11.2.2.6 Miscellaneous Top Events. Top Event RQ is presented with the RHR system as a "switch" top event used to determine the path in the recirculation event tree if RHR sump recirculation is required. Top Event RQ is successful if sump recirculation is not required. No fault tree is developed for this top event; rather, the success of this top event is controlled by the logic rules based on the initiating event and the status of other top events. The specific initiating events and sequence description for Top Event RQ is given in the plant model section for the Recirculation Event Tree Notes and Assumptions.

Top Event CM is presented with the RHR system as a "switch" top event used to determine the path in the recirculation event tree if the reactor core is damaged as a result of top event failures in the general transient event tree. Top Event CM is successful if the reactor core is not damaged. No fault tree is developed for this top event; rather, the success of this top event is controlled by the logic rules based on the initiating event and the status of other top events. The specific initiating events and sequence description for Top Event CM is given in the plant model section for the Recirculation Event Tree Notes and Assumptions.

3.2.1.12 Safety Injection System

3.2.1.12.1 System Function

The safety injection system (SIS) is part of the Watts Bar emergency core cooling system (ECCS). It is a high pressure system and can inject borated water into the reactor vessel when the primary system pressure decreases to approximately 1,520 psig. The SIS pumps provide water inventory makeup to the reactor coolant system (RCS) for small to medium size breaks that require relatively high head and low flow injection. The SIS accumulators are designed to flood the reactor core during the low pressure large break conditions. The SIS is therefore designed to supply RCS inventory makeup, to cool the reactor core, and to provide additional shutdown margin by the addition of negative reactivity in the form of borated water following a loss of coolant accident (LOCA). Injection of borated water into the RCS thus prevents core uncovery, provides core cooling, and ensures that recriticality does not occur due to the addition of cold water following an accident. The purpose of core cooling is to maintain the fuel temperature within the predetermined limits to prevent fuel damage and the subsequent release of radioactive material into the environment.

3.2.1.12.2 System Operation

The SIS consists of two independent pump trains per unit. The two pump trains discharge to a common header before splitting into four injection paths to provide flow to each of the four RCS cold legs. Separate injection paths are used for hot leg recirculation. Miniflow recirculation is provided for the SIS pumps. The cold leg accumulators inject their contents of borated water into the RCS during LOCA events when the RCS pressure decreases below the accumulator pressure of 600 psig. The SIS is used for safety injection during both the injection and sump recirculation phases.

3.2.1.12.2.1 Normal Operation. During normal plant operation, the SIS is in standby alignment for accident mitigation. The SIS pumps are used to fill and top-off the cold leg accumulators. This function of the SIS is not modeled as part of Top Event CL (or LCL).

The normal standby lineup of the SIS (shown in Section 2 of the system notebook) is such that the SIS pumps 1A-A and 1B-B are aligned to take suction from the refueling water storage tank (RWST) through the normally open inlet valve (FCV-63-5) and check valve (63-510). The train A path includes normally open pump suction valve (FCV-63-47), safety injection pump 1A-A, pump discharge check valve (63-524), locked-open pump isolation valve (63-525), and normally open discharge valve (FCV-63-152). The train B path includes normally open pump suction valve (FCV-63-48), safety injection pump 1B-B, pump discharge check valve (63-526), locked-open pump isolation valve (63-527), and normally open discharge valve (FCV-63-153). The two discharge paths join and flow through the normally open valve (FCV-63-22) to the four RCS cold leg injection paths. Each path includes a throttling valve (63-550/63-552/63-554/63-556) and a check valve (63-551/63-553/63-555/ 63-557), and discharges to the RCS cold leg through the common check valve (63-560/63-561/63-562/63-563). Operation of the SIS pumps requires cooling from the component cooling system (CCS) for the pump lube oil coolers and cooling water supplied from the ERCW for the pump room coolers. The pump room cooling fan is also required for successful operation of each SIS pump. The ERCW cooling flow control valve (FCV-67-176/FCV-67-182) is normally closed and is required to open when the SIS pump is running.

3.2.1.12.2.2 Accident/Transient Operation. Each SIS pump will start upon receipt of the "SI" signal (see the Engineered Safety Features Actuation System notebook). The minimum flow line for each pump is normally open so the recirculation flow is available to protect the pumps before the RCS pressure drops to below the shutoff head of the pumps.

3.2.1.12.2.2.1 Injection Phase. The injection phase of the SIS is initiated by an "SI" signal. The "SI" signal automatically starts the SIS pumps. The SIS pumps will recirculate through the minimum flow line back to the RWST until reactor pressure falls below the shutoff head of 1,520 psig. As pressure decreases in the reactor, the SIS pumps will start injecting into the RCS. The SIS pumps are rated at 400 gpm at 1170 psig and are designed to supply borated water during small or medium LOCA conditions.

Three safety injection system top events and one RHR top event are used to model the SIS cold leg injection in the general transient/small break LOCA event tree. These top events are RW, S1, S2, and SI. Top Event RW models the RWST and is discussed in the RHR system analysis. Top Events S1 and S2 model the safety injection pumps. Top Event SI models the suction path from the RWST to the pumps and discharge path from the pumps to the four RCS cold legs including the common check valves (63-560, 63-561, 63-562, and 63-563).

Three safety injection system top events and one RHR top event are used to model the cold leg injection in the medium break and large break LOCA event trees. These are RW, S1, S2, and IP. Safety injection pump 1A-A is modeled in Top Event S1 and safety injection pump 1B-B is modeled in Top Event S2. The injection path Top Event IP is similar to Top Event SI except that three injection paths are modeled rather than four. This is to account for the loss of one of the injection paths due to the LOCA. The injection path to RCS loop 4 is modeled as being unavailable due to the LOCA.

3.2.1.12.2.2.2 Recirculation Phase. The recirculation phase is automatically initiated for ECCS when the RWST reaches low level (≤ 29%), with the containment sump high level (≥ 10%) and an "SI" signal present. In this phase, coolant is pumped into the reactor vessel from the containment sump. Coolant collected in the sump is pumped by the RHR pumps, cooled by the RHR heat exchangers, and discharged to either the RCS cold legs or the SIS and chemical and volume control system (CVCS) pump suction. The SIS and CVCS pumps inject the recirculated coolant into the reactor vessel when the RCS pressure remains high.

Upon receipt of the required level signal, the containment sump isolation valves (FCV-63-72 and FCV-63-73) automatically open and the RHR normal suction valves (FCV-74-3 and FCV-74-21) automatically close to isolate the RWST. The operator is instructed to manually complete the alignment of the SIS for the recirculation phase. This is accomplished by closing the RWST isolation valve (FCV-63-5), closing the SIP minimum flow valves (FCV-63-3, FCV-63-4, FCV-63-175), and opening the supply path from the RHR system to the SIS (FCV-63-11) and the alternate flow path (FCV-63-8, FCV-63-177, FCV-63-7, and FCV-63-6). The minimum flow valves are interlocked with the two supply valves (FCV-63-8 and FCV-63-11) so that the supply valves will not open unless either FCV-63-3 is closed, or FCV-63-4 and FCV-63-175 are closed.

The SIS pumps continue to take suction from the RWST until the realignment is complete. The volume of water remaining in the RWST is sized to allow the operator to make this realignment before the SIS pumps lose suction due to low level in the RWST. The operator must also stop the SIS pumps if the level in the RWST drops to 8% or less before the realignment is complete.

Containment sump recirculation is modeled as Top Events RL, RVA, RVB, and RR, and is described in the RHR system analysis. High pressure recirculation using the safety injection system is modeled by Top Events RL, RVA, RVB, RR, S1, S2, and SI for small LOCAs. The CVCS high pressure injection Top Events VA, VB, and VC are also used for high pressure recirculation.

3.2.1.12.2.2.3 Hot Leg Recirculation. After 15 hours of operation in the sump recirculation mode, the operator is instructed to manually align the RHR and SIS for "hot leg" recirculation. The SIS is aligned by stopping SIS pump 1A-A, closing train A crosstie (FCV-63-152), opening train A hot leg valve (FCV-63-156), restarting pump 1A-A, verifying flow, stopping SIS pump 1B-B, closing train B crosstie (FCV-63-153), opening train B hot leg valve (FCV-63-157), starting pump 1B-B, verifying flow, and closing SIS cold leg injection valve (FCV-63-22). The hot leg recirculation model is discussed in the RHR system notebook.

Placing the unit in the "hot leg" recirculation mode will provide long-term cooling for the reactor, and prevent boric acid "plateout" in the core by reversing flow in the reactor. This will prevent boric acid from blanketing the fuel rods that may degrade the heat transfer from the core.

The top event used to model hot leg recirculation, Top Event RH, is modeled and described in the RHR system analysis. In this analysis, hot leg recirculation is modeled by using either RHR pump train.

3.2.1.12.2.2.4 Cold Leg Accumulators. The cold leg accumulators are pressurized to approximately 600 psig by nitrogen gas and have an injection volume of approximately 8,000 gallons of borated water. Each accumulator injects its coolant through a normally open motor-operated isolation valve and two check valves to the RCS cold leg through a 10-inch line. The four cold leg accumulators function independently of the rest of the SIS and inject into the RCS solely on the basis of pressure differential between the accumulators and the RCS.

The accumulators are designed to inject during large LOCA conditions when RCS pressure rapidly decreases and a large volume of water is needed to flood and cool the reactor in a relatively short period of time.

The cold leg accumulators are modeled in Top Event CL or LCL. The cold leg common check valves (63-560, 63-561, and 63-562) that are required for the accumulator injection are also included in this top event. The accumulators are asked in the medium break and large break LOCA event trees.

3.2.1.13 RCP Seal Injection and Thermal Barrier Cooling

3.2.1.13.1 System Function

The high pressure seal injection water provided to each reactor coolant pump (RCP) acts to cool both the lower and upper parts of the pump and seal assembly as well as to provide clean water for lubrication of the lower radial bearing and the seal system. The presence of borated seal injection water [chemical and volume control system (CVCS) water] also prevents reactor coolant water from escaping into the containment atmosphere. The remainder of the high pressure seal injection water that does not pass through the seal is diverted along the RCP pump shaft through the thermal barriers. This high pressure seal injection then acts as a buffer and pressure boundary to prevent reactor coolant system (RCS) water from entering the radial bearing and seal section of the pump. The thermal barrier heat exchanger provides a means of cooling the system water to an acceptable level prior to reaching the seals. Flow to the thermal barrier heat exchanger is provided by the component cooling system (CCS) thermal barrier booster pumps. CCS flow to the RCP thermal barriers is required to ensure seal and pressure boundary integrity in the event that normal seal injection flow is lost.

This analysis models RCP seal integrity, which includes the functions of the RCP seal injection and the RCP thermal barrier cooling. In addition, this analysis models the operator action to shut off the RCPs on a loss of pump bearing oil cooling.

3.2.1.13.2 System Operation

3.2.1.13.2.1 Normal Operation. During normal power operation, RCP seals are protected by both seal injection and thermal barrier cooling. The following describes the modeled system alignments of these two alternatives for seal integrity maintenance as well as the seals themselves.

3.2.1.13.2.1.1 Seal Injection. A portion of the CVCS charging flow from either the volume control tank (VCT) or the refueling water storage tank (RWST) is directed to the seals of each RCP. Seal injection should remain in service whenever the reactor coolant system water level is above the seals or the RCS pressure is greater than the atmospheric pressure. A simplified diagram of this system can be found in Section 2 of the system notebook (also see drawings 1-47W809-1 and 1-47W859-2).

The modeled flow path begins at the VCT and passes through three valves (LCV 62-132, LCV 62-133, and check valve 62-697) before passing through the centrifugal charging pump lines. The charging pumps and their discharge lines are modeled in the chemical and volume control system. The level control valves are normally open but receive a signal to close under a safety injection condition. The flow path modeled in Top Event SE begins downstream of the charging pumps with normally open manual valve 62-535, air-operated flow control valve (FCV) 62-93, and normally open manual valve 62-536. After valve 62-536, the line branches with a portion of the flow directed to the regenerative heat exchanger and the remainder to seal injection. The flow balancing is performed by charging flow control valve.

Water for seal injection first passes through one of the seal water injection filters and the associated manual valves 62-548 and 62-550. The seal water injection filter collects particulate matter larger than 5 microns that could be harmful to the seal faces. After being filtered, the injection water enters a header where it is divided into four separate paths, one for each RCP.

Flow to each RCP passes through a series of four valves, the first of which is the FCV for that pump. The seal water injection FCVs (62-556, 62-557, 62-558, and 62-559) are manually operated needle valves used with the flow indicators to adjust the seal water flow to each RCP.

Injection water enters each RCP at a point between the lower radial bearing and thermal barrier cooler coil. Upon entering the RCP, the flow splits and a portion (5 gallons per minute per pump) pass downward through the cooler assembly and along the shaft into the casing, removing heat conducted by the shaft and parts of the thermal barrier before passing on into the RCS. This downward flowing water also prevents hot primary water from reaching the pump internals. The remaining 3 gpm of flow is directed to the shaft, cooling the lower radial bearing, and to the No. 1 seal. Approximately 2.9 gpm of seal injection water (with 1 to 5 gpm as an acceptable range) exits through the No. 1 seal as leakoff. A small portion of the seal flow (approximately 3 gallons an hour per pump) passes across the No. 2 seal. Leakoff flow from the No. 2 seal is discharged to the reactor coolant drain tank (RCDT) in the waste disposal system. The volume of water maintained in the standpipe provides sufficient head to direct a minute flow (100 cc per hour) from seal No. 2 to seal No. 3. The No. 3 seal leakoff flow is discharged to the RCDT.

The No. 1 seal leakoff flow discharges to a common manifold and exits from containment prior to entering the seal water return filter. After passing through the return filter, flow enters the seal water heat exchanger.

The seal water heat exchanger uses component cooling water to cool fluid from three sources: (1) RCP seal leakoff returning to the outlet of the VCT, (2) reactor coolant discharged from the excess letdown heat exchanger, and (3) miniflow from the centrifugal charging pumps. The unit is single-shell and multi-pass, and is designed to cool the flow to the temperature normally maintained in the VCT (122°F to 127°F). If the heat removal is degraded and the injection water temperature is high, an alarm is received in the main control room alerting the operator to reduce injection water temperature.

There are two relief valves on the return line. Relief valve 62-636 is stationed prior to the inboard containment isolation valve and discharges to the pressurizer relief tank on closure of either containment isolation valve or other downstream blockage. The capacity of this relief valve equals the maximum flow from the four RCP No. 1 seals and the excess letdown flow. Relief valve 62-649 is located just upstream from the seal water heat exchanger and discharges directly into the volume control tank. This valve has sufficient capacity for overpressure protection in the event that either heat exchanger isolation valve is closed.

3.2.1.13.2.1.2 Thermal Barrier Cooling. The CCS supplies coolant to the RCP thermal barriers during normal operation with a redundant set (one of two) of thermal barrier booster pumps. Each booster pump is capable of supplying adequate flow to the RCP thermal barriers to maintain the integrity of the seals. A simplified diagram of this system as modeled is shown in Section 2 of the system notebook (also see drawing 1-47W859-2).

The CCS thermal barrier booster pumps are located in the auxiliary building. Each pump is a single-stage centrifugal, motor-driven pump with a rated capacity of 160 gpm at a head of 135 feet. Each pump is powered by a 15-hp electric motor of standard commercial design. The motor is powered from the 480V shutdown board.

Both pumps discharge through their own series of check and manual valves to a common injection line to the RCPs. Manual valves are used to isolate a pump for maintenance or to otherwise take it out of service. Check valves ensure that there is no back flow from one pump to the other.

Redundant motor-operated containment isolation valves exist, powered from separate trains, to allow for isolation when required in the event of a single train power failure. These valves will automatically close on a high differential flow signal that is comparing the supply flow to the return flow from containment. Containment isolation valves for the thermal barrier heat exchangers can also be operated from the auxiliary control room. The positions of these valves are indicated in the main control room.

These containment isolation valves are followed by check valve 70-679, also located inside containment, for further protection. The injection supply line then enters a header that supplies the thermal barrier heat exchanger for each RCP. Each RCP thermal barrier heat exchanger is preceded by two check valves. Cooling water at 32 psi and 40 gpm passes through each thermal barrier heat exchanger.

The downstream line includes a spring-loaded relief valve and a manual globe valve that enters a common discharge header flowing to the suction side of CCS pump 1B. The relief valve alleviates the thermal volumetric expansion occurring with each heat exchanger and discharges to the waste disposal system.

The CCS return containment isolation valves with single failure capability are FCV 70-87 and FCV 70-90. FCV 70-87 is paralleled with a containment isolation check valve (check valve 70-687) in the reverse direction to relieve the thermal volumetric expansion pressure for the portion of the piping between containment isolation valves FCV 70-87 and FCV 70-90. Check valve 70-687 remains seated when containment isolation is established.

The component cooling water could become contaminated with radioactive water due to a leak in any cooling coil of the thermal barrier cooler. RCP thermal barrier leak detection is provided for the CCS thermal barrier lines to prevent serious system overpressurization. This is done by measuring the flow rates in both the supply and discharge headers. The two are compared; should a mismatch occur due to leakage, the lines will be isolated. The comparison is made in each electrical power train so that the isolation function is completely redundant. As long as seal injection and RCP motor bearing oil cooling are available, the RCP can continue operating, even though the thermal barrier has been isolated.

3.2.1.13.2.1.3 RCP Seals. Section 2 of the system notebook contains a cutaway view of an RCP with seal injection paths exposed. The No. 1 seal is located above the lower radial bearing and constitutes the most important element in the seal system. It is basically a film-riding face seal. The film is produced by the system pressure drop across the seal. Normal leakage rate for this seal is 1 to 5 gallons per minute (gpm) at system operating pressure. Leakage is radially inward toward the shaft, and the design is such that the axial pressure forces are balanced. The No. 1 seal leakoff channels approximately 2.9 gpm to the seal water return filter. Since this seal rides on a thin film, the seal ring does not mechanically contact the seal runner.

The No. 2 seal normally accepts 3-gpm leakage from the No. 1 seal at a pressure of approximately 50 psi and seals it against a back pressure provided by the RCP standpipe. Normal leakage rate across the No. 2 seal is 3 gallons per hour. This rubbing-face type seal is of conventional design and employs a rotating runner and a stationary carbon member. It is pressure balanced and spring loaded. Although the seal normally handles only 50 psi on the high pressure side, it is designed so that, in an emergency, it can operate with the full system pressure across its face in either the rotating or stationary state. Although its life under these latter emergency conditions will be reduced, it will permit limited pump operation or an orderly shutdown without gross leakage.

The third seal is a smaller, low pressure, rubbing-face type seal designed to limit leakage into the reactor coolant drain tank to 100 cc per hour. This small leakage lubricates and cools the seal faces. The No. 3 seal is similar in materials and design to the No. 2 seal.

3.2.1.13.2.2 <u>Accident/Transient Operations.</u> If operation of the RCPs is required during a specific event, then injection water to the pump seals is also required. When an RCP is idle, either CCS water to the thermal barrier or seal injection water flow must be supplied if the RCS temperature is greater than 150°F. In the event of loss of RCP seal cooling, failure of the seals could occur and result in depressurization and loss of coolant of the RCS through the seals.

Loss of component cooling system train A supply to the upper and lower RCP bearing coolers is postulated to result in a seal LOCA if the operators fail to trip the reactor and turn off the RCPs. Ten minutes are allowed for the operators to turn off the pumps to prevent this type of seal LOCA.

During a Phase A containment isolation (CIA), seal water return containment isolation valves FCV-62-61 and FCV-62-63 will receive a signal to close. Seal injection flow will still be maintained through a 2-inch relief valve (62-636) that relieves to the pressurizer relief tank. The seals will also continue to be protected during a CIA by thermal barrier cooling.

Following a Phase B containment isolation (CIB), the thermal barrier containment isolation valves (FCV-70-133, FCV-70-134, FCV-70-90, and FCV-70-87) receive a signal to close, thus removing that method of seal integrity protection. The Phase B containment isolation also stops flow to the RCP motor bearing oil coolers by closing FCV-70-139, FCV-70-140, FCV-70-89, and FCV-70-92.

During a loss of offsite power, the RCP motor is deenergized, and both cooling supplies (CCS and CVCS) are terminated. However, the diesel generators are started automatically, and either seal injection flow from the CVCS or CCS water to the thermal barrier heat exchanger is restored. Either cooling supply is adequate to provide seal cooling and prevent seal failure following a loss of offsite power. If the station blackout is sustained, a LOCA due to seal failure will result.

3.2.1.14 Pressurizer Power-Operated Relief Valves and Safety Valves

3.2.1.14.1 System Function

Both the pressurizer power-operated relief valves (PORV) and the safety valves function to limit reactor coolant system pressure. The safety valves have a higher setpoint than the PORVs. The combined capacity of the pressurizer safety valves is equal to, or greater than, the maximum surge rate resulting from a complete loss of load without reactor trip or any other control. The pressurizer PORVs prevent actuation of the fixed high pressure trip for a large power mismatch and also limit the necessity for opening the safety valves. The PORVs are also used to establish a bleed and feed heat removal path for loss of all feedwater transients.

The individual plant examination (IPE) models the following:

1. Water and steam pressure challenges to the reactor coolant system (RCS) that could result in a PORV or safety valve failing to reclose, thus inducing a loss of coolant accident (LOCA).

2. Pressure relief during an anticipated transient without scram (ATWS).

3. Feed and bleed operation.

4. Depressurization of the RCS in sequences in which secondary heat removal is available.

3.2.1.14.2 System Operation

3.2.1.14.2.1 Normal Operation

3.2.1.14.2.1.1 Pressurizer Operation. The pressurizer, which acts as a surge volume for the reactor coolant system, provides a point in the RCS at which liquid and steam can be maintained in equilibrium under saturated conditions for control of pressure. Under normal operating conditions approximately 60% of the pressurizer internal volume is occupied by water and the rest by steam. Electric immersion heaters, installed through the bottom head of the pressurizer, keep the water at saturation temperature.

The pressurizer is designed to accommodate positive and negative volume surges caused by load transients. The surge line, attached to the bottom of the pressurizer, connects the pressurizer to a reactor coolant hot leg.

Two separate, automatically controlled spray valves (1-PCV-68-340B and 1-PCV-68-340D) with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve that permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open. The pressurizer spray lines and valves are large enough to provide adequate spray using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray is selected to prevent the pressurizer pressure from reaching the setpoint of the PORVs following a step load reduction in power of 10% of full load with reactor control.

A temperature decrease in reactor coolant, caused by an increase in electrical load, results in a decrease in coolant volume. Coolant flows out from the pressurizer into loops to reduce water level and pressure of the pressurizer. The reduction of pressure energizes the immersion heaters to bring the pressure back to normal.

3.2.1.14.2.1.2 Power-Operated Relief Valves. The PORVs (1-PCV-68-334 and 1-PCV-68-340A), normally in the closed position, are used to limit system pressure increases resulting from a large power mismatch. The PORVs are designed to limit pressurizer pressure to a value below the reactor high pressure trip setpoint (2,385 psig). This feature prevents actuation of the reactor high pressure trip for all design transients including a 50% step load decrease with steam dump but without reactor trip. Also, each PORV is designed to relieve 210,000 pounds per hour of saturated steam at 2,265 psia (see the system notebook, Reference 31, Table 5.5-14).

The PORVs are operated automatically or by remote manual control. One motor-operated block valve per PORV is installed between the PORV and the pressurizer to isolate the solenoid-operated relief valve for repair and other purposes. The block valve also helps to ensure that a stuck-open PORV incident will be mitigated. Positions of the PORVs and block valves are indicated in the main control room.

The PORVs are solenoid pilot-actuated valves that respond to a signal from pressure instrumentation loops connected to the pressurizer. A coincident high pressure signal from two independent channels is needed for the actuation of each PORV.

When the reactor is not critical, these limits are in terms of pressure/temperature limit curves. The RCS low temperature overpressurization protection system continuously monitors the RCS temperature and pressure conditions whenever plant operation is below a predetermined temperature. An auctioneered system temperature will be continuously converted to an allowable pressure and then compared to the actual RCS pressure. This comparison would provide an actuation signal to the PORVs when required to prevent pressure temperature conditions from exceeding allowable limits.

PORV 1-PCV-68-340A uses auctioneered high cold leg temperature compared to RCS pressure to derive its opening signal. This valve is also interlocked with auctioneered high hot leg temperature, which must also be low to allow opening of the PORV.
PORV 1-PCV-68-334 uses auctioneered high hot leg temperature to derive its opening signal with an interlock from auctioneered high cold leg temperature. This provides cold-overpressurization protection during startup operation of the plant.
Cold-overpressurization protection is not considered in this system analysis.

The PORVs are designed with sufficient capacity to preclude actuation of the safety valves during normal operational transients. They are also designed to fail closed on a loss of electric power supply.

If one or more PORV is inoperable due to excessive seat leakage, Technical Specification 3.4.4 requires that within 1 hour the PORV(s) be restored to operable status or that the associated block valve(s) be closed. If these actions are not taken, the plant is to be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

3.2.1.14.2.1.3 Safety Valves. The pressurizer safety valves (68-563, 68-564, and 68-565), connected to the top of the pressurizer, prevent RCS pressure from exceeding 110% of the system design pressure (i.e., ~2,735 psig), in compliance with the ASME Nuclear Power Plant Components Code, Section III. The safety valves are totally enclosed, pop-type spring-loaded valves, and are self-actuated by direct fluid pressure action and back-pressure compensation. Design of the safety valves accounts for the pressure drop between the RCS pump discharge and the most remote safety valves. These valves attain full lift prior to reaching 3% above set pressure. The combined capacity of two of the safety valves is designed to accommodate the maximum surge resulting from a complete loss of load event without a direct reactor trip. Each safety valve is capable of relieving 420,000 pounds per hour of saturated steam.

3.2.1.14.2.2 Accident/Transient Operation. During specific transient scenarios, RCS temperature and pressure will rise to the point at which pressure relief via the pressurizer PORVs or the safety valves is required. Once the PORVs/safety valves have opened, they are required to reseat or be isolated to prevent a LOCA. This function is modeled in the pressure relief (Top Event PR) portion of this system analysis. In the event that the PORVs are used for RCS depressurization (e.g., during a steam generator tube rupture), they are required to reclose. This function is modeled in Top Event PI.

3.2.1.14.2.2.1 Pressure Relief. The primary coolant relief system is challenged when RCS pressure increases to the PORV opening setpoint of 2,335 psig. If the PORVs fail to open, the primary pressure would continue to increase to the RCS design pressure of 2,485 psig, which is the opening setpoint of the safety valves.

Pressurizer pressure transmitters PT 68-323 and PT 68-340 provide signals to PORV 1-PCV-68-340A. A signal from each transmitter is required to open the PORV. Similarly, PT 68-322 and PT 68-334 provide signals to PORV 1-PCV-68-334. A continued increase in pressure actuates the high pressure reactor trip at 2,385 psig on a two of four pressure transmitter signal basis.

If the PORVs fail to open on demand or pressurizer pressure continues to rise despite the performance of their designed function, opening of one or more of the safety valves is assumed to be demanded.

3.2.1.14.2.2.2 RCS Primary Relief Water Challenge. The initiating events that will be evaluated by this top event are: inadvertent safety injection (ISI), steam line breaks (SLBOC and SLBIC), steam line PORV/safety valve fails open (MSVO), and inadvertent main steam isolation valve closure (IMSIV). Initiating events resulting in excessive cooldown are considered because they result in a safety injection signal (SIS). IMSIV is considered on the basis that an inadvertent MSIV closure is caused or associated with an SIS. Combinations of losses of instrument buses that cause an inadvertent safety injection signal are also considered. The occurrence of any of the above events is considered for its potential to result in a water challenge to the RCS relief valves in Top Event PR. Evaluation of this top event is handled in the general transient event tree.

3.2.1.14.2.2.3 Pressure Relief Isolation. This action describes the reseating function of the PORVs and the ability to isolate a PORV that fails to close or leaks; i.e., a crack on valve seat. Isolation of a PORV is possible through operator action on the associated block valve (1-FCV-68-332 or 1-FCV-68-333).

Leakage of PORVs and safety valves can be detected through several different ways. Positive indication of PORV position is obtained by an electromagnetic switch (single channel for each PORV). The position of these valves is indicated in the main control room. Temperature sensors downstream of valves (a sensor in each safety valve line and a sensor downstream of both PORVs) also can provide an indication of valve leakage. An increase in a discharge line temperature is an indication of leakage through the associated valves. Acoustic monitors located downstream of the valves are able to detect the flow through the lines.

A temperature alarm in the main control room will indicate when any valve is leaking. If one or both PORVs fails to close, the operator must manually close the proper block valve(s) to isolate the suspected PORV(s).

3.2.1.14.2.2.4 ATWS Pressure Relief. The combined capacity of three safety valves (1.26×10^6 pounds per hour) may not be enough for pressure relief during an ATWS. As presented in Section 1.2 of the system notebook, a conditional probability is considered to include different situations. For Top Event SR, three of three safety valves and either one, two, or no PORVs are required to open. The number of PORVs required to open is dependent on the status of manual rod insertion, the amount (50% or 100%) of auxiliary feedwater (AFW) available for the secondary-side heat removal, and reactivity feedback.

3.2.1.14.2.2.5 Feed and Bleed Following Loss of Secondary Heat Sink. The pressurizer PORVs are used to control rising RCS pressure through "feed and bleed" operations following loss of secondary heat sink. This condition is evidenced by (1) all steam generator wide range levels less than 25% for normal or 35% for adverse containment conditions and (2) total AFW flow less than 470 gpm.

The operator must first establish a feed path to the RCS from either the safety injection pumps or the centrifugal charging pumps (CCP). After feed flow is established, both PORVs and their associated block valves are to be fully opened. The equipment modeled in this top event includes the PORVs (1-PCV-68-340A and 1-PCV-68-334) and the PORV block valves (1-FCV-68-332 and 1-FCV 68-333).

The PORVs and block valves remain open while the operator attempts to restore a secondary heat sink from the AFW, main feedwater, condensate, or high pressure fire protection (HPFP) pumps.

RCS feed and bleed operations can be terminated when all of the following conditions are satisfied:

- Narrow range level in at least one steam generator is greater than 10% (25% for adverse containment condition).
- Core exit thermocouple reading is decreasing.
- T_{hot} is decreasing.

When these conditions are met, the operator terminates feed and bleed by first closing one PORV. While maintaining RCS subcooling above 40°F and pressurizer level greater than 20% (50% for adverse containment), the operator isolates the Boron Injection tank and then closes the second PORV prior to establishing letdown and stopping both safety injection pumps. The operator then stops one CCP and maintains RCS subcooling and pressurizer level.

3.2.1.15 Steam Generator Isolation

3.2.1.15.1 System Function

The function of the steam generator isolation (SGI) that has been modeled is the isolation of any steam generator experiencing a tube rupture and also a faulted steam generator from such causes as a steam line break.

A steam generator that experiences a tube rupture is isolated from the secondary side of the plant to help prevent escape of radioactivity into the environment following this initiating or induced event. Because the primary side of the steam generator is at a far higher pressure than the secondary side, an unisolated tube rupture releases contaminated reactor coolant into the uncontaminated secondary side of the steam generator.

Isolation is also initiated for any steam generator that becomes faulted due to a main steam line break (MSLB) inboard of the associated main steam line isolation valve (MSIV), or upon a failed-open steam generator power-operated relief valve (PORV). This latter isolation is required to mitigate overcooling (i.e., limit positive reactivity insertion and maintain shutdown margin) of the reactor coolant system (RCS) to prevent pressurized thermal shock and to prevent exceeding pressure differential limits between the secondary side of the steam generator and the RCS. (Main steam isolation is modeled via Top Event MS in the main steam system notebook.) A high steam generator pressure differential increases the chances of a tube rupture.

Following successful steam generator isolation or given a small loss of coolant accident (LOCA), the primary coolant system (Top Event DP, modeled in the pressurizer PORV and safety valves notebook) must be cooled down and depressurized. This must be performed in a timely manner so that (1) any primary coolant outflow can be reduced and stopped, and (2) the plant can be placed in a long-term safe shutdown cooling configuration.

3.2.1.15.2 System Operation

The SGI equipment included within the analysis boundary consists of the steam generator PORV, the steam generator safety relief valves, and the isolation valves required to isolate the lines specified in Section 1.2 of the system notebook. This equipment is associated with performing the steam generator isolation function (Top Event SL). In addition, the steam generator PORV and its associated block valve are involved with the RCS cooldown function (Top Event DS in the auxiliary feedwater system notebook). A simplified flow diagram of the steam generator system depicting the system analysis boundary is shown in Section 2 of the system notebook. The components associated with each of the four steam generators that are within the analysis boundary of the SGI are identified in Table 1 of the system notebook.

The following is an account of the configuration of the system during both normal and transient conditions. This is necessary to determine the component manipulations required during transient conditions (either automatic or manual) to accomplish the system functions as specified in Section 1.1 of the system notebook. These component manipulations are then reflected in the system fault trees as component (hardware) failure modes that contribute to failure of the system function. Any operator errors in following the associated procedures to perform manual component manipulations will be included as basic events (where appropriate) in the system fault trees.

3.2.1.15.2.1 Normal Operation. The steam generator isolation system alignment during normal plant operation is in standby mode. During normal operation, the relief valves are normally closed, and the auxiliary feedwater pumps are on standby; i.e., in accordance with the references in Section 1.7.1.1 of the system notebook. This configuration is as follows:

- AFW inlet line isolation valves (LCV-3-174, LCV-3-164, and LCV-3-164A) closed.
- Turbine-driven AFW pump steam supply line isolation valve from steam generator No. 1 (FCV-1-15) open.

- Turbine-driven AFW pump steam supply line isolation valve from steam generator No. 4 (FCV-1-16) closed.
- Steam generator blowdown system line isolation valves (FCV-1-181 and FCV-1-7) open.
- Steam generator PORV (PCV-1-5) closed.
- Steam generator PORV relief line block valve 1-619 open.
- Steam generator safety valves (1-522, 1-523, 1-524, 1-525, and 1-526) closed.
- MSIV (FCV-1-4) open.

During normal plant operation, the steam generator blowdown system is continuously operating for chemistry control.

3.2.1.15.2.2 Accident/Transient Operation. The following is an analysis of the components within the steam generator isolation system analysis boundary for determining the possible failure modes that may prevent steam generator isolation.

The steam generator isolation function is required following a steam generator tube rupture (SGTR) that also includes feedwater isolation.

An SGTR results in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines by tripping all feedwater pumps and closing the main and bypass feedwater isolation valves. When an SGTR occurs, some of the reactor coolant blows down into the affected steam generator, causing the shell-side level and pressure to rise. The primary system pressure is reduced below the secondary safety valve setting. With the exception of the AFW inlet lines, subsequent recovery procedures call for isolation once the ruptured steam generator has been identified. The AFW inlet lines will not be manually isolated until the ruptured steam generator narrow range level is greater than 10%. Abnormal and emergency operating procedures and instructions provide the operator with guidance following detection of an SGTR.

A faulted steam generator results in uncontrolled blowdown from the secondary side of the affected steam generator. This uncontrolled blowdown causes excessive heat transfer from the primary to the secondary side of the steam generator. With the RCS remaining at high pressure with decreasing temperature, pressurized thermal shock of the RCS pressure boundary becomes a major concern. The large pressure differential between the primary and secondary sides could also induce an SGTR. Subsequent recovery procedures for a faulted steam generator call for isolation of all lines leading to and from the affected steam generator once the faulted steam generator has been identified. Note that, unlike the SGTR event sequence, manual isolation of the AFW inlet lines will be performed concurrent with all other lines leading to or from the faulted steam generator. This action is necessary to help mitigate the overcooling effect on the RCS, whereas overcooling was not a concern with the SGTR. An emergency operating procedure is provided for guidance following detection of a faulted steam generator.

Emergency procedures that provide guidance during detection of abnormal conditions in the steam generators are listed in Section 1.7.2.2 of the system notebook.

All steam generator valves within the analysis boundary (identified in Table 1 of the system notebook) are associated with the steam generator isolation function. The initial configuration of the steam generator prior to actuation of the steam generator isolation function is assumed to differ (for some valves) from the configuration at the time that isolation is asked following steam generator pressure relief due to plant trip; i.e., AFW turbine-driven pump has started. This new configuration is assumed as follows (see modeling assumption number 1 in Section 3.1 of the system notebook):

- AFW inlet line isolation valves closed.
- Turbine-driven AFW pump steam supply line isolation valve from steam generator No. 1 (FCV-1-15) open.
- Turbine-driven AFW pump steam supply line isolation valve from steam generator No. 4 (FCV-1-16) closed.
- Steam generator blowdown system suction line isolation valves open.
- Steam generator PORV open.
- Steam generator PORV relief line block valve open.
- Steam generator safety valves open.
- MSIV (FCV-1-4) open.

Thus, considering that all of the above valves must be in the closed position for steam generator isolation, the following is a determination of the possible failure modes for each valve that could prohibit this function from succeeding.

- **AFW Inlet Line Isolation Valves.** There are three AFW control valves and inlet lines associated with each steam generator. They are the turbine-driven AFW pump inlet line, the motor-driven feedwater pump inlet line, and the motor-driven AFW pump inlet line bypass. Each of these contains an air-to-open, level control valve (LCV) that can be used to isolate the steam generator from the AFW. Each valve is normally closed and is demanded closed. Thus, for the turbine-driven AFW pump inlet line LCV, and the motor-driven AFW pump inlet line and bypass LCV, the failure modes that prevent success in performing the steam generator isolation function are for the valves to transfer open, and, except for the bypass valve, fail to close on demand.

  Note: The three AFW inlet lines merge to one line prior to containment penetration.

- **Turbine-Driven AFW Pump Steam Supply Line Isolation Valves.** There is a turbine-driven AFW pump steam supply line associated with both steam generator No. 1 and steam generator No. 4. Steam generator No. 1 is aligned to provide this steam supply to the AFW during normal plant operation.

Since steam generator No. 1 and steam generator No. 4 contain the maximum number of paths to and from a steam generator (because of the steam supply lines to the AFW), and the steam supply to the AFW from steam generator No. 1 is normally open and the steam supply to the AFW from steam generator No. 4 is normally closed, steam generator No. 1 is assumed to experience the SGTR or fault in order to model the maximum number of failure modes that may prohibit success of the steam generator isolation function (see assumption number 2 in Section 3.1 of the system notebook). Should steam generator No. 1 become ruptured or faulted, the steam supply to the turbine-driven AFW pump will be automatically or remote manually switched from steam generator No. 1 to steam generator No. 4. However, the automatic swapover will not occur for all possible scenarios requiring the steam generator isolation function and therefore is not modeled (see modeling assumption number 3 in Section 3.1 of the notebook).

The turbine-driven AFW pump steam supply line isolation valves are motor-operated, fail as-is. Thus, considering that the initial state of the valve associated with steam generator No. 1 is open prior to initiation of the steam generator isolation function, the failure modes that prohibit this function from being successful are as follows:

- Fail to close on demand.
- Transfer open.

- **Steam Generator Blowdown System Suction Line Isolation Valves.** There are two fail-closed solenoid-operated valves (FCV-1-181 and FCV-1-7) in each steam generator blowdown system suction line. These valves are open during normal plant operation (see assumption number 4 in Section 3.1 of the system notebook).

With the steam generator blowdown system in operation, the suction line isolation valves are energized to open. The valves are automatically closed on receipt of a containment isolation system (CIS) signal or an AFW pump start signal or may be remote manually closed. However, because the automatic isolation does not occur for all scenarios requiring steam generator isolation, only remote manual isolation will be modeled (see assumption number 16 in Section 3.1 of the system notebook).

The failure modes for each solenoid-operated valve that prohibit success of the steam generator isolation function when in the steam generator blowdown alignment are as follows:

- Fail to close on demand.
- Transfer open.

Failure to close on demand will be the failure mode for each solenoid-operated valve when the steam generator blowdown system is in operation. (These valves are in the open position initially.) Transfer open will be the failure mode for each solenoid-operated isolation valve when the steam generator blowdown system valves have closed.

- **Safety Valves.** There are five safety valves (SFV) associated with each steam generator. These valves are assumed to have opened in performing the RCS overpressure relief function prior to the requirement to isolate the subject steam generator. Valves must fail to reseat to present a problem. Therefore, the failure mode associated with each SFV that may prohibit success of the steam generator isolation function is failure to reseat. Note that it is not considered feasible that the SFVs would transfer open.

- **PORV.** The PORV is an air-operated fail-closed pressure control valve (air-to-open and spring-to-close). The possibility exists for the PORV to open for the steam generator overpressure relief function should the steam generator pressure increase to the PORV automatic opening setpoint prior to actuation of the steam generator isolation function. Thus, the failure mode of the PORV that prohibits success of the steam generator isolation function is a mechanical failure to close on demand.

- **PORV Block Valve.** The PORV block valve is a manually operated valve. Because there is no requirement for the PORV block valve to be closed prior to performing the steam generator isolation function, the initial state of this valve is assumed to be in the normally open position. Because this valve will be operated manually, one of the failure modes that may prohibit success of the steam generator isolation function is the human error that results in this valve not being closed. This failure mode is not accounted for since a stuck-open steam generator PORV for an SGTR may result in high personal dosage and is counter to operator ALARA training.

- **Main Steam Isolation Valve.** The MSIV is open during normal plant operation. The MSIV is automatically closed on receipt of a steam line break upstream and downstream of the valve. This function is modeled in Top Event MS in the Main Steam System notebook. Top Event SL models the ability of the MSIVs to close for SGTRs, in order to prevent radioactivity from entering the secondary side of the plant.

3.2.1.16 Containment Spray System

3.2.1.16.1 System Function

The design basis of the containment spray system (CSS) is to ensure that the containment pressure does not exceed the containment shell design pressure or the maximum temperature limit following a loss of coolant accident (LOCA) or a main steam line rupture inside containment. The CSS, normally in standby mode, is designed to operate automatically during any design basis event that results in a high-high containment pressure signal. The CSS sprays subcooled borated water into the upper containment atmosphere to obtain full coverage of the containment volume. Spray is supplied through two spray ring headers. Initially, the CSS and ice condenser function simultaneously to remove heat from the containment atmosphere. After the ice is depleted, the CSS and residual heat removal (RHR) spray provide the only active means of containment cooling.

3.2.1.16.2 System Operation

The CSS for this PRA consists of two independent and fully redundant trains of equipment. Each CSS train includes one centrifugal pump driven by an electric motor, a heat exchanger, a spray ring header with 263 nozzles in the upper containment, and the associated pipes and valves. See the simplified containment spray piping and instrumentation diagram (P&ID) in Section 2 of the system notebook for a graphical representation.

3.2.1.16.2.1 Normal Operation. During normal operation, CSS equipment is in standby, and the associated isolation valves (1-FCV-72-39 and 1-FCV-72-2) are closed. The CSS can be actuated either manually from the control room (M-5 or M-6) or automatically by the instrumentation loops monitoring the lower containment pressure. During normal standby conditions, the injection headers are maintained filled with water. The RWST suction valves (1-FCV-72-21 and 1-FCV-72-22) are normally open, ready to allow suction from the RWST during the injection mode. There is an interlock between the RWST suction valves and the containment sump suction valves (between 1-FCV-72-21 and 1-FCV-72-45 for train B, and 1-FCV-72-22 and 1-FCV-72-44 for train A). This interlock does not allow sump suction valves to open for CSS recirculation unless the RWST suction valves are closed, and vice versa. This feature prevents the RWST water from draining into the containment sump during the injection mode and prevents the contamination from the sump water from getting into the RWST during recirculation. In addition, the operators are instructed in accordance with WBN Emergency Sub-Instruction ES-1.2 to ensure that the RHR suction valves (1-FCV-74-3 and 1-FCV-74-21) have automatically closed before opening valves 1-FCV-72-44 and 1-FCV-72-45. The closing of two RHR suction valves also serves to prevent the radioactive contamination to the RWST. The manual switchover to the containment sump from the RWST is initiated when the RWST reaches the lo-lo level.

The containment spray test line manual valves are normally locked closed. The miniflow recirculation motor-operated valves (MOV) (1-FCV-72-13 and 1-FCV-72-34) are normally closed. CSS pump bearing oil cooling is provided by the component cooling system (CCS).

During operation Modes 1 through 4, system operability is controlled by Technical Specification 3/4.6.2. With one containment spray subsystem inoperable, the inoperable subsystem must be restored within 72 hours or be in at least hot standby within the next 6 hours, or it must be restored within the next 48 hours or be in cold shutdown within the next 30 hours.

3.2.1.16.2.2 Accident/Transient Operation. The system is designed so that both trains are automatically started by high-high containment pressure signal (two out of four logic from the containment differential switches). The spray header valves (1-FCV-72-2 and 1-FCV-72-39) open upon high-high containment pressure signal concurrent with starting of their respective CSS pump. The relays associated with this automatic actuation of the CSS are modeled in the engineered safety features actuation system (ESFAS).

To protect the CSS pump(s) from low flow conditions, a minimum flow recirculation line is provided to allow pump discharge to be directed back to the pump suction. This is achieved by opening an MOV when flow in the discharge side drops below the required value or, if upon starting, flow is not achieved in the discharge header within a specified time period.

Manual actuation can be made from the main control room by (1) manually initiating a containment Phase B isolation signal or (2) manually starting the CSS pumps and opening the spray header valves. In the event of a main control room evacuation, the necessary functions can be transferred to the local control boards to ensure that the CSS can be controlled.

The heat sink for the CSS heat exchangers is supplied by the ERCW system. The ERCW water is isolated from the heat exchangers by motor-operated valves at the inlet and outlet of each heat exchanger. Containment heat is removed through these heat exchangers during the recirculation mode.

3.2.1.17 Containment Systems

In the unlikely event of a loss of coolant accident (LOCA) or a degraded core cooling accident, the structural and leakproof integrity of primary containment is to be maintained. The following systems play an important role in ascertaining this integrity during the aforementioned design basis accident (DBA) conditions:

- Primary Containment Isolation
- Ice Condenser
- Hydrogen Management and Combustible Gas Control
- Air Return Fan System
- Containment Sump

The above systems, in part or as a whole, are being modeled in this containment systems analysis. To comply with 10CFR50 (regarding structural integrity of containment), 10CFR100 (regarding the limits of radionuclide that can be released offsite), and 10CFR20 (regarding the limits of radionuclide that can be released onsite), the containment structural and leakproof integrity must be kept intact. The systems being analyzed in the model are important to achieving this objective.

3.2.1.17.1 System Function

The main functions of each system, as applicable to the present model, are defined in this section.

The containment isolation system isolates those fluid systems penetrating the containment that are not required for accident mitigation following a DBA. The isolation minimizes the release of radioactive nuclide/material outside the containment building. The engineered safety features actuation system (ESFAS) conveys signals to actuate the closure of the appropriate valves of the system. The containment isolation function is modeled with two top events in this analysis. Top Event CI models the isolation of all penetrations other than those related to the reactor building purge ventilation system (RBPVS). A simplified diagram of the penetrations modeled in Top Event CI is presented in Section 2 of the system notebook as Figure 1.

Top Event CP models isolation of the RBPVS penetrations. RBPVS lines are modeled separately from the other containment penetrations due to the relatively large size of the RBPVS lines. The RBPVS contains supply and exhaust lines of 8-, 12-, and 24-inch diameter sizes. Each supply and exhaust penetration line is provided with an inboard and an outboard air-operated valve. When the line is in operation (i.e., the valves are open), the valves will fail closed on a loss of power or plant air. According to the plant Technical Specifications, only one pair of RBPVS lines (i.e., one supply and one exhaust line) is allowed to operate at a time during reactor operation. All valves in the remaining purge system lines must be in a closed position. A simplified diagram of the penetrations modeled in Top Event CP is presented in Section 2 of the system notebook as Figure 2.

During a LOCA, the steam in the containment lower compartment makes its way to the containment upper compartment via the ice condenser containment. The ice condenses steam quickly, helping to reduce the sharp increase in the containment pressure during a DBA. The ice condenser function is modeled by Top Event IC in this analysis.

During a DBA, the hydrogen mitigation system (HMS) and combustible gas control system (CGCS) provide a means to control hydrogen concentration inside the containment. A large amount of combustible gas, particularly hydrogen, may be formed as a consequence of the DBA. The primary objective of the CGCS is to bring about the formation of water vapor by spontaneous recombination of hydrogen and oxygen at an elevated temperature. The HMS ignites the excessive quantities of hydrogen. The ignition is carried out in a controlled sequence, so that the integrity of containment is not challenged. The function of the HMS is modeled in Top Event HH. A simplified diagram of the components modeled in Top Event HH is presented in Section 2 of the system notebook as Figure 3. Because the CGCS does not have the capacity to control the hydrogen buildup for the beyond design basis accident conditions modeled in this study, it is not modeled in this analysis.

The primary purpose of the air return fan system is to enhance the ice condenser and the containment spray heat removal operation by circulating air from the upper containment to the lower containment, through the ice condenser, and then back to the upper containment. The system also limits hydrogen concentration in potentially stagnant regions by maintaining circulation. The air return fan system is modeled by Top Event AR in this analysis. A simplified diagram of the components modeled in Top Event AR is presented in Section 2 of the system notebook as Figure 4.

As a result of a LOCA, the coolant leaking from the reactor coolant system (RCS), melted ice from the ice condenser, and water injected into the containment via the containment spray system will collect in the containment sump. The availability of the sump to serve as a reservoir for recirculation cooling is modeled in Top Event SU in this analysis.

3.2.1.17.2 System Operation

The main objective of the containment systems analysis presented here is to model those functions that: (1) mitigate the potential for a degraded core accident, (2) prevent leakage through the containment penetrations during transient conditions, when a LOCA or degraded core accident is in progress, or (3) prevent containment failure. Keeping these objectives in view, functional details of the relevant portions of each system are being given below.

To isolate the containment atmosphere from the environment, all of the pipe lines that penetrate the containment that are not required for accident mitigation are isolated. The following criteria were used to screen containment penetrations from the analysis of containment isolation Top Events CI and CP. Penetrations containing lines that satisfy any of the following criteria are not included in the analysis:

1. The line penetration is required for accident mitigation.

2. The line penetration does not communicate with the reactor coolant system, the containment atmosphere, or the outside environment.

3. The line penetration is isolated during normal power operation by a normally closed, fail-closed valve; a normally closed manual valve; or at least three check valves in series.

4. The system is designed to withstand pressures at least equal to the containment design pressure.

Table 1 in the system notebook lists all of the containment penetrations reviewed for this analysis. Also indicated on Table 1 are the screening criteria listed above, indicating which penetrations were excluded from the analysis. Those penetrations with lines that do not satisfy any of the above criteria and are modeled in Top Event CI or CP are listed in Table 2. All lines for which isolation is required are provided with two barriers so that no single failure will prevent isolation.

The isolation of containment penetrations was modeled in two top events. The larger penetrations provided for containment purge lines are modeled in Top Event CP. The remaining, smaller penetrations are included in the analysis for Top Event CI. The distinction between large and small is important with respect to the size and timing of the radionuclide release.

The ice condenser helps in reducing the increase in containment pressure. This mitigating function may be jeopardized if the ice is not available or if the ice condenser doors fail to open.

The hydrogen ignitors burn pockets of excess hydrogen that may accumulate following a DBA event. The air return fans help in heat distribution by circulating and mixing the upper and lower containment atmospheres. The mixing of the containment atmospheres also helps to deter the formation of high hydrogen concentration pockets in the containment.

The containment sump stores the water spilled from a LOCA line break or containment spray return. The water collected in the sump can be recycled to the residual heat removal and the containment spray system for use on a long-range basis. The availability of sump is subject to flow path from the upper containment to the lower containment being available, and the sump screens remaining unplugged.

3.2.1.17.2.1 Normal Operation. During normal plant operation, the containment systems, except HMS and CGCS, are in a standby status. The containment purge ventilation system is usually in standby mode. It is limited to a maximum of 1,000 hours of operation per year by Technical Specifications. The lineup of each system during normal plant operation is described below.

The position during normal operation for each valve considered in the containment isolation function is listed in Table 2 of the system notebook. Penetrations of the following systems are included in this model:

- Containment Ventilation System
- Containment Connections to the Waste Disposal System
- Radiation Monitoring System
- Reactor Coolant Pump Seal Return

The CGCS is not in service during normal power operations. The function of the CGCS is to limit the buildup of hydrogen in containment during a degraded core accident. The CGCS, on its own, does not have enough capacity to control buildup during beyond design basis accidents analyzed in this study. Therefore, the CGCS is not being modeled as part of the containment systems analysis.

The HMS does help to control the hydrogen buildup during a degraded core accident. This system is manually initiated during an accident after verifying that containment hydrogen concentration is less than 6%. As this is an electrical control system, there is no mechanical system lineup.

During normal power operation, the air return fans are in standby, aligned for automatic startup when a Phase B containment isolation signal is received. The fans will start 10 minutes after the Phase B signal is received. The air ducts for the return fans are aligned to take suction from the upper containment and to discharge to an accumulator room in the lower containment.

The ice condenser system plays no role in the normal operation of the plant but serves only to mitigate the consequences of a LOCA or high energy line break (HELB). All ice bed maintenance is normally done during plant outages. These maintenance activities will not impact the condenser availability. Therefore, no analysis will be carried out for these activities.

Under normal operating plant conditions, the containment sump is dry. It is ready to receive water from a LOCA line break or water from containment spray. The water flows to the sump by gravity. For water injected into the containment via the containment spray system to reach the sump, the upper containment refueling canal drains must be unobstructed.

3.2.1.17.2.2 Accident/Transient Operation. The containment isolation valves modeled in this PRA are listed in Table 2 of the system notebook, including their normal position and the isolation signal required for automatic isolation. All of the isolation valves modeled based on the previous assumptions close on either a containment ventilation isolation signal or a Phase A isolation signal.

The containment ventilation isolation signal is generated by any of the following conditions:

1. Manual or automatic safety injection signal.
2. High radiation in the containment lower compartments.
3. High radiation in the containment upper compartment.
4. High radiation in the containment purge air exhaust.
5. Manual use of Phase A or B containment isolation handswitches.

The Phase A signal can be generated manually or is generated by the manual or automatic actuation of the safety injection signal. The safety injection signal is generated by one or more of the following:

1. High main steam flow coincident with low steam line pressure or low-low primary coolant average temperature in two of four loops.

2. High differential pressure between one main steam line and two of the other three lines.

3. Low pressurizer pressure.

4. Two out of three high containment pressure signals.

5. Manual actuation.

In accordance with design criteria, on a loss of air or electrical power, all of the air-operated valves will fail closed. This ensures that the containment can be isolated under these degraded support system scenarios.

During transient and accident conditions, thermal energy is directed through the ice condenser. As the steam comes in contact with the ice, heat energy is removed, and the steam condenses. This cooling of the containment atmosphere helps to minimize the sudden increase in the containment pressure. The ice condenser also removes some of the iodine nuclide found in the containment atmosphere during accident conditions. The affluent from the ice will collect in the containment sump. Top Event IC models the availability of the ice condenser at the time of the initiating event.

During degraded core accidents (i.e. accidents involving reactor vessel melt-through and release of molten core material), the HMS is used to increase the containment capability to accommodate hydrogen releases. The system consists of two redundant trains of hydrogen ignitors and associated control circuitry. Each train contains 34 ignitors located in various locations throughout the primary containment. The system is manually initiated from the control room during an accident after verifying that containment hydrogen concentration is less the 6%.

The containment air return fan system is composed of two 100%-capacity fans, each capable of removing approximately 40,000 cfm from the upper containment to the lower containment. Either fan will circulate sufficient air throughout the containment. Both fans will start automatically 10 minutes after a Phase B containment isolation signal is actuated. The fans can also be manually started from the control room. The flow path for the air circulation is from the upper containment through a main duct to the lower containment, then to the lower inlet doors of the ice condenser. The air then flows up through the ice condenser and back into the upper containment. Each main duct contains a nonreturn (backdraft) damper that prevents the flow of air from the containment lower compartment to the containment upper compartment during the initial stages of an accident. During accident conditions, the containment air return fan system is capable of operating continuously with temperatures ranging up to 350°F for the first hour, and at 250°F and 100% relative humidity for a year.

When an event occurs that results in a rise in containment pressure or temperature (e.g., a LOCA or steamline break inside containment), water will find its way from a variety of sources into the containment sump. The affluent may be from the reactor coolant system, the containment spray system, or from the ice condenser. In general, the water will collect in the containment sump where it can be returned to the reactor coolant system via the RHR system or to the containment via the containment spray system. Since the containment spray system can pump all of the water in the sump into the upper level of the containment in a short time (approximately 1 hour), the availability of water in the sump for recirculation is subject to drain plugs having been removed from the upper containment compartment after the last extended shutdown.

3.2.1.18 Simplified Drawings

The simplified drawings that describe each system model are provided at the end of this section. The following table lists the number of simplified drawing figures for each system; they are assembled in the sam order as the systems are described.

Section Number	System	Number of Drawing Figures	Total Pages of Figures
3.2.1.1	Electric Power System	21	21
3.2.1.2	Essential Raw Cooling Water System	2	3
3.2.1.3	Component Cooling System	3	3
3.2.1.4	Plant Compressed Air System	4	5
3.2.1.5	Chemical and Volume Control System	4	4
3.2.1.6	Condensate and Feedwater System	1	1
3.2.1.7	Engineered Safety Features Actuation System	2	2
3.2.1.8	Reactor Protection System	2	2
3.2.1.9	Auxiliary Feedwater System	2	2
3.2.1.10	Main Steam System	1	1
3.2.1.11	Residual Heat Removal System	5	5
3.2.1.12	Safety Injection System	1	1
3.2.1.13	RCP Seal Injection and Thermal Barrier Cooling	5	5

Section Number	System	Number of Drawing Figures	Total Pages of Figures
3.2.1.14	Pressurizer Power-Operated Relief Valves and Safety Valves	1	1
3.2.1.15	Steam Generator Isolation	2	2
3.2.1.16	Containment Spray System	2	2
3.2.1.17	Containment Systems	4	6

3.2.2 SYSTEMS ANALYSIS — NOTEBOOK DESCRIPTIONS

The top events or functions required by the plant Level 1 analysis are grouped into functional systems for analysis purposes. Notebooks have been compiled to document the analysis of these top event groups and functional systems. The systems analysis notebooks include the following information:

1. The functional definition of each top event.
2. The required success criteria for each top event.
3. A general description of the system operation during normal power operation and transient/accident conditions.
4. A description of the effect of component testing and maintenance.
5. A table showing the support system dependencies for the components modeled in the top events.
6. The references used in the systems analysis.
7. The assumptions used in constructing the top event models.
8. The fault trees developed for the top event models.
9. The RISKMAN software quantification files.
10. Simplified system drawings.

The RISKMAN software files listed in the notebooks include the quantitative results for each top event split fraction as well as a complete listing of all of the data required to recreate the system models. The results of the top event split fraction quantification for all systems are summarized in Section 3.3.5.

Table 3.2-1 lists the top event names and descriptions corresponding to each of the system groups. The names of the event trees in which each top event appears are also listed in Table 3.2-1. Table 3.2-2 is a list of the system notebooks prepared in support of this document.

3.2.3 SYSTEM DEPENDENCIES

One of the most important and difficult tasks in developing an integrated model for plant response is to explicitly identify all physical and functional intersystem dependencies among the plant systems. Two tables are used to display these intersystem dependencies. Table 3.2-3 shows how failure of each support system (e.g., major electric power board) affects equipment in other support systems. Table 3.2-4 shows how a failure of support system equipment affects frontline system trains of equipment.

To develop these interdependency tables, support system components are first grouped into common functional elements such as system trains, subsystems, and complete plant systems. This grouping is performed in a manner that permits the support-to-support and the support-to-frontline system dependencies to be readily defined. In general, train dependencies are tracked in these tables. This is especially necessary for the construction of event trees that model equipment groups of different trains into separate top events. Only direct system-to-system dependencies are included in the dependency tables. Secondary, or cascaded, system-to-system dependencies are developed through the logic of the support tree models.

Failure of an item along the far left column in each table has an effect on each item along the top row of the table as indicated by an "X" or a number. For example, in Table 3.2-4 failure of 6.9-kV shutdown board 1B-B power supply causes failure of motor-driven auxiliary feedwater pump 1B-B, charging pump 1B-B, safety injection pump 1B-B, RHR pump 1B-B, and containment spray pump 1B-B. These failures are noted by a number in the box that intersects each of the above system pumps and 6.9-kV shutdown board 1B-B. The number is identified in the notes at the bottom of the page. The notes will refer the reader back to Table 1 of the system notebooks for further detail.

Table 3.2-1 (Page 1 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event: Name	Top Event: Description	Event Trees Containing Top Event*: ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Electric Power	A1	480V Shutdown Board 1A1-A	X								
	A1U2	Unit 2 480V Shutdown Board 2A1-A		X							
	A2	480V Shutdown Board 1A2-A	X								
	A2U2	Unit 2 480V Shutdown Board 2A2-A		X							
	A3	6.9-kV and 480V Common Board A	X								
	AA	6.9-kV Shutdown Board 1A-A	X								
	AB	6.9-kV Shutdown Board 2A-A		X							
	B1	480V Shutdown Board 1B1-B	X								
	B1U2	Unit 2 480V Shutdown Board 2B1-B		X							
	B2	480V Shutdown Board 1B2-B	X								
	B2U2	Unit 2 480V Shutdown Board 2B2-B		X							
	B3	6.9-kV and 480V Common Board B	X								
	BA	6.9-kV Shutdown Board 1B-B	X								
	BB	6.9-kV Shutdown Board 2B-B		X							

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large Loss of Coolant Accident (LOCA) Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 2 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event Name	Top Event Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Electric Power (continued)	D1	250V DC Board 1	X								
	D2	250V DC Board 2	X								
	DA	125V DC Battery Board I	X								
	DB	125V DC Battery Board II	X								
	DAAC	120V AC Vital Board 1-I	X								
	DBAC	120V AC Vital Board 1-II	X								
	DC	125V DC Battery Board III		X							
	DCAC	120V AC Vital Board 1-III		X							
	DD	125V DC Battery Board IV		X							
	DDAC	120V AC Vital Board 1-IV		X							
	DG	Unit 1 120V AC Instrument Power Board 1A	X								
	DH	Unit 2 120V AC Instrument Power Board		X							
	FA	Unit 1 Fuel Oil for Diesel 1A-A	X								
	FB	Unit 1 Fuel Oil for Diesel 1B-B	X								

(Event Trees Containing Top Event*)

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 3 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event Name	Top Event Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Electric Power (continued)	FC	Unit 2 Fuel Oil for Diesel 2A-A		X							
	FD	Unit 2 Fuel Oil for Diesel 2B-B		X							
	GA	Unit 1 Diesel 1A-A	X								
	GB	Unit 1 Diesel 1B-B	X								
	GC	Unit 2 Diesel 2A-A		X							
	GD	Unit 2 Diesel 2B-B		X							
	OG	161-kV Offsite Power	X								
	UB1A	6.9-kV Unit Board 1A	X								
	UB1B	6.9-kV Unit Board 1B	X								
	UB1C	6.9-kV Unit Board 1C	X								
	UB1D	6.9-kV Unit Board 1D	X								
	V1	Unit 1 Shutdown Board Room Ventilation	X								
	V2	Unit 2 Shutdown Board Room Ventilation		X							
	VINV1	480V Shutdown Board Room B Ventilation	X								

(Columns ELECT1 through RECL: Event Trees Containing Top Event*)

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 4 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event: Name	Top Event: Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Electric Power (continued)	VINV2	Unit 2 480V Shutdown Board Room B Ventilation		X							
	VT1A	480V Shutdown Transformer Room 1A Ventilation	X								
	VT1B	Shutdown Transformer Room 1B Ventilation	X								
	VT2A	480V Shutdown Transformer Room 2A Ventilation		X							
	VT2B	480V Shutdown Transformer Room 2B Ventilation		X							
Essential Raw Cooling Water (ERCW)	AE	ERCW Train A Pumps			X						
	BE	ERCW Train B Pumps			X						
	CE	ERCW Header 1A-A			X						
	DE	ERCW Header 1B-B			X						
	EE	ERCW Header 2A-A			X						
	FE	ERCW Header 2B			X						
	GE	ERCW Discharge Header A			X						
	HE	ERCW Discharge Header B			X						
	MDE	Maintenance on ERCW Header 1B - Supply to CCS Heat Exchanger A Replaced with ECW Header 2A			X						

Event Trees Containing Top Event*

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 5 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event		Event Trees Containing Top Event*								
	Name	Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Component Cooling Water	AC	Train 1A Component Cooling Water System			X						
	BC	Train 1B Component Cooling Water System			X						
Plant Air System	PE	ERCW Cooling to CAS Compressors			X						
	PA	Train A Auxiliary Control Air			X						
	PB	Train B Auxiliary Control Air			X						
	PD	Nonessential Control Air			X						
Chemical and Volume Control (CVCS)	EB	Emergency Boration, Operator Actions, and Equipment				X					
	VA	Centrifugal Charging Pump 1A-A				X		X			
	VB	Centrifugal Charging Pump 1B-B				X		X			
	VC	One Out of Four Cold Leg Injection Path from CCP				X					
	VF	Two Out of Three CVCS Cold Leg Injection Paths for Medium or Large LOCA						X			
	VS	Supply to Centrifugal Charging Pumps				X		X			
	MU	Makeup to RWST			X						
Condensate and Feedwater	FW	Main Feedwater Continues during Anticipated Transient without Scram (ATWS) Event				X					

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 6 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event		Event Trees Containing Top Event*								
	Name	Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Condensate and Feedwater (continued)	MF	Equipment Needed To Recover Main Feedwater				X					
	OF	Operator Actions To Recover Main Feedwater				X					
Emergency Safeguard Features Actuation (ESFAS)	OS	Manual Operator Backup of ESFAS Alignments			X						
	ZA	Train A ESFAS			X						
	ZB	Train B ESFAS			X						
Reactor Protection	AM	ATWS Mitigating Systems Actuation Circuitry Trips Turbine and Starts Auxiliary Feedwater Independent of SSPS				X					
	MR	Manual Rod Insertion, ATWS Only, Operator Action during First Minute				X					
	PL	Power Level is Less Than 40%, Used In ATWS Only				X					
	RT	Reactor Trips, Control Rods Insert				X					
Auxiliary Feedwater (AFW)	CT	Condensate Storage Tank (CST)			X						
	CTMU	Makeup to the CST			X						
	DS	Operator Depressurizes the Reactor Coolant System (RCS) using the Steam Generator Power-Operated Relief Valves (PORV)				X					

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 7 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event: Name	Top Event: Description	Event Trees Containing Top Event*: ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Auxiliary Feedwater (AFW) (continued)	AF	Discharge Path from the AFW Pumps to the Steam Generators				X					
	MA	Motor-Driven AFW Pump 1A-A				X					
	MB	Motor-Driven AFW Pump 1B-B				X					
	TP	Turbine-Driven AFW Pump				X					
	V3	Component Cooling Water (CCW) and Motor-Driven AFW Ventilation			X						
Main Steam	CD	Condenser, TBVs, and Flow Path for Controlled Cooldown				X					
	MS	Main Steam Isolation Valves (MSIV) Close, Three Out of Four				X					
	TT	Turbine Trip				X					
Residual Heat Removal (RHR)	RA	RHR Pump 1A-A				X	X	X			
	RB	RHR Pump 1B-B				X	X	X			
	RD	RHR Normal Cooldown and Charging				X					
	RF	Two Out of Three RHR Cold Leg Injection Paths for Medium or Large LOCA					X	X			
	RH	RHR and Safety Injection Hot Leg Recirculation					X	X			
	RI	One Out of Four Cold Leg Injection Path from RHR Pumps				X					

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 8 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event		Event Trees Containing Top Event*								
	Name	Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Residual Heat Removal (RHR) (continued)	RL	Recirculation Level Instrumentation for RHR Swapover					X	X			
	RQ	Sump Recirculation is Not Required									
	RW	Refueling Water Storage Tank (RWST)			X						
	RR	Automatic/Manual Swapover from the RWST to the Containment Sump					X	X	X		
	RS	RHR Spray					X	X	X		
	RVA	Train A Sump Swapover Valve, 1-FCV-63-72					X	X	X		
	RVB	Train B Sump Swapover Valve, 1-FCV-63-73					X	X	X		
	CM	Core Melt Occurs during Injection Phase							X		
	EX	Excessive LOCA					X				
Safety Injection (SIS)	CL	Two Out of Three Cold Leg Accumulators						X			
	LCL	Three Out of Three Cold Leg Accumulators					X				
	IP	Two Out of Three SIS Cold Leg Injection Paths for Medium or Large LOCA					X	X			

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 9 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event		Event Trees Containing Top Event*								
	Name	Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Safety Injection System (continued)	S1	Safety Injection Pump 1A-A				X	X	X			
	S2	Safety Injection Pump 1B-B				X	X	X			
	SI	Suction and One Out of Four Cold Leg Injection Paths for Safety Injection Pumps				X					
Seal/Thermal Barrier Cooling	SE	Reactor Coolant Pump (RCP) Seal Injection and RCP Bearing Oil Cooling				X					
	TB	Thermal Barrier Cooling to the Reactor Coolant Pumps				X					
PORVs and Safety Valves	DP	Operator Depressurization of the RCS Using the Pressurizer Spray Valves and PORVs				X					
	OB	Operator Action and PORV Operation To Perform Feed and Bleed RCS				X					
	PI	PORVs Reclosed if Opened in Top Event DP				X					
	PR	Pressurizer PORVs Open To Control RCS Pressure and Reclose				X					
	SR	Steam Relief, ATWS Only, Reactor Pressure is Less Than 3,200 psig				X					
	WC	RCS Primary Relief — Water Challenge				X					
Steam Generator	SL	Isolation of Ruptured Steam Generator				X					
Containment Spray	CH	Containment Spray in Recirculation Mode					X	X	X		
	CSA	Train A Containment Spray Pump and Valves					X	X	X		

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 10 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event		Event Trees Containing Top Event*								
	Name	Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Containment Spray (continued)	CSB	Train B Containment Spray Pump and Valves					X	X	X		
	OT	Operator Terminates Containment Spray							X		
Containment Isolation	AR	Containment Air Return Fans					X	X	X		
	CI	Containment Isolation					X	X	X		
	CP	Containment Purge Isolation					X	X	X		
	HH	Hydrogen Ignitors					X	X	X		
	IC	Ice Condenser					X	X	X		
	SU	Containment Sump					X	X	X		
Recovery	OGR1	Recovery of 161-kV Offsite Power in 1 Hour	X								
	V1R	Recovery of Unit 1 Shutdown Board Room Ventilation in 12 Hours	X								
	V2R	Recovery of Unit 2 Shutdown Board Room Ventilation in 12 Hours		X							
	VNV1R	Recovery of 480V Board Room B Ventilation in 6 Hours	X								
	VNV2R	Recovery of Unit 2 480V Shutdown Board Room B Ventilation in 6 Hours		X							
	VT1AR	Recovery of 480V Shutdown Transformer Room 1A Ventilation in 10 Hours	X								
	VT1BR	Recovery of Shutdown Transformer Room 1B Ventilation in 5 Hours	X								

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 11 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event: Name	Top Event: Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Recovery (continued)	VT2AR	Recovery of 480V Shutdown Transformer Room 2A Ventilation in 10 Hours		X							
	VT2BR	Recovery of 480V Shutdown Transformer Room 2B Ventilation in 5 Hours		X							
	TPR	Restart Turbine-Driven AFW Pump				X					
	REC	Electric Power and Other Recovery Action Split Fraction								X	X
	CCSR	Recovery by Cross-Training			X						
	DSLR	Recovery of ERCW to Diesel from Opposite Side			X						
	CCPR	Align ERCW Cooling to CCP 1A-1			X						
Miscellaneous	A1L	ERCW/Diesel 1A/480V Shutdown Board 1A1-A Dependency			X						
	A1U2L	ERCW/Diesel 2A/480V Shutdown Board 2A1-A Dependency			X						
	A2L	ERCW/Diesel 1A/480V Shutdown Board 1A2-A Dependency			X						
	A2U2L	ERCW/Diesel 2A/480V Shutdown Board 2A2-A Dependency			X						
	AAL	ERCW/Diesel 1A/6.9-kV Shutdown Board 1A-A Dependency			X						
	ABL	ERCW/Diesel 2A/6.9-kV Shutdown Board 2A-A Dependency			X						
	B1U2L	ERCW/Diesel 2B/480V Shutdown Board 2B1-B Dependency			X						
	B2L	ERCW/Diesel 1B/480V Shutdown Board 1B2-B Dependency			X						

Column group header: Event Trees Containing Top Event*

*Legend:

ELECT1 Unit 1 Electric Power Event Tree
ELECT2 Unit 2 Electric Power Event Tree
MECH Mechanical Support Tree
GTRAN General Transient, etc., Tree
LARLOC Large LOCA Tree
MEDLOC Medium LOCA Tree
RECIR Recirculation Event Tree
RECOVERY Recovery Event Tree
RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-1 (Page 12 of 12). Cross-Reference Table for Systems, Top Events, and Event Trees

System	Top Event: Name	Top Event: Description	ELECT1	ELECT2	MECH	GTRAN	LARLOC	MEDLOC	RECIR	RECOVERY	RECL
Miscellaneous (continued)	B2L	ERCW/Diesel 1B/480V Shutdown Board 1B2-B Dependency			X						
	B2U2L	ERCW/Diesel 2B/480V Shutdown Board 2B2-B Dependency			X						
	BAL	ERCW/Diesel 1B/6.9-kV Shutdown Board 1B-B Dependency			X						
	BBL	ERCW/Diesel 2B/6.9-kV Shutdown Board 2B-B Dependency			X						

Legend: Event Trees Containing Top Event

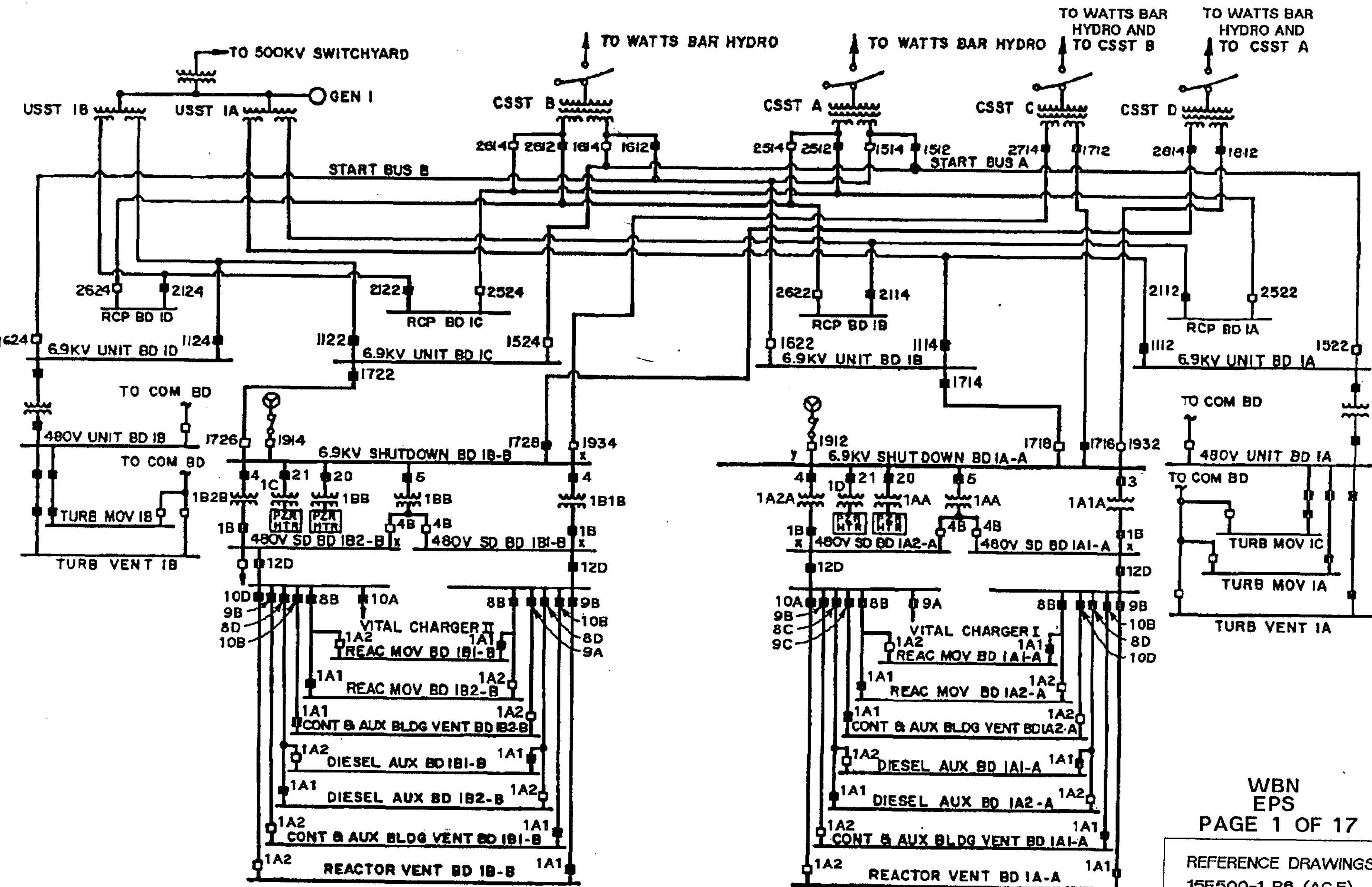
- ELECT1 Unit 1 Electric Power Event Tree
- ELECT2 Unit 2 Electric Power Event Tree
- MECH Mechanical Support Tree
- GTRAN General Transient, etc., Tree
- LARLOC Large LOCA Tree
- MEDLOC Medium LOCA Tree
- RECIR Recirculation Event Tree
- RECOVERY Recovery Event Tree
- RECL Recovery Event Tree for Large or Medium LOCAs

Table 3.2-2. TVA System Notebooks Prepared in Support of the Watts Bar PRA

Electric Power System, Revision 0
Essential Raw Cooling Water System, Revision 0
Component Cooling System, Revision 0
Plant Compressed Air System, Revision 0
Chemical and Volume Control System, Revision 0
Condensate and Feedwater System, Revision 0
Engineered Safety Features Actuation System, Revision 0
Reactor Protection System, Revision 0
Auxiliary Feedwater System, Revision 0
Main Steam System, Revision 0
Residual Heat Removal System, Revision 0
Safety Injection System, Revision 0
RCP Seal Injection and Thermal Barrier Cooling, Revision 0
Pressurizer Power-Operated Relief Valves and Safety Valves, Revision 0
Steam Generator Isolation, Revision 0
Containment Spray System, Revision 0
Containment Systems, Revision 0

SECTION 3.2.1.1 — ELECTRIC POWER SYSTEM FIGURES

OG-MIS000187/4

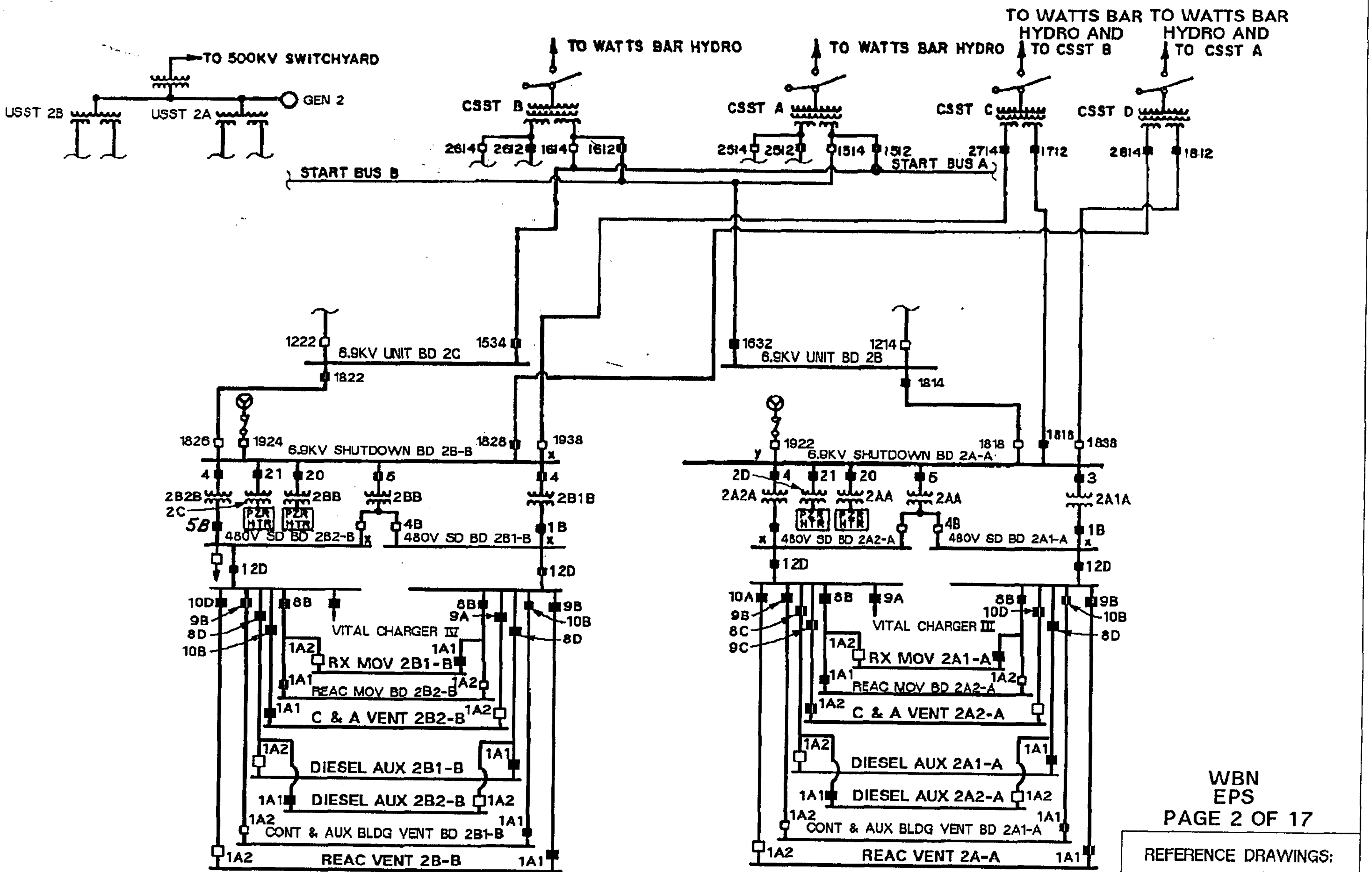


WBN
EPS
PAGE 1 OF 17

REFERENCE DRAWINGS:
15E500-1 R6 (AC.F)
15E500-2 R3 (AC.E)
15E500-1 R7
15E500-2 R6

WBN UNIT I AUX POWER
FIGURE 1

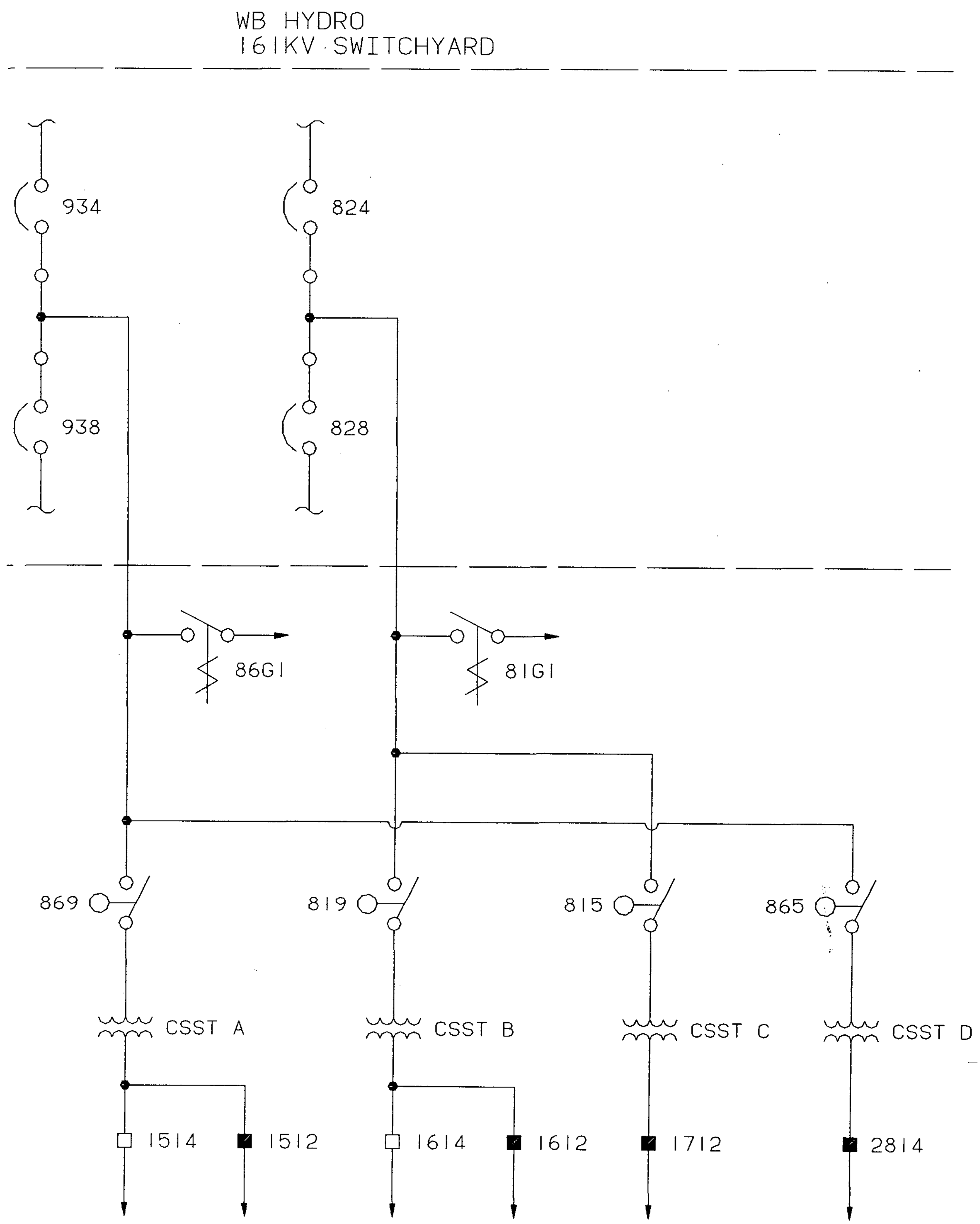
OG-MIS000168



WBN UNIT 2 AUX POWER
FIGURE 2

WBN
EPS
PAGE 2 OF 17

REFERENCE DRAWINGS:
15E500-1 R6 (AC.F)
15E500-2 R3 (AC.E)
15E500-1 R7
15E500-2 R6



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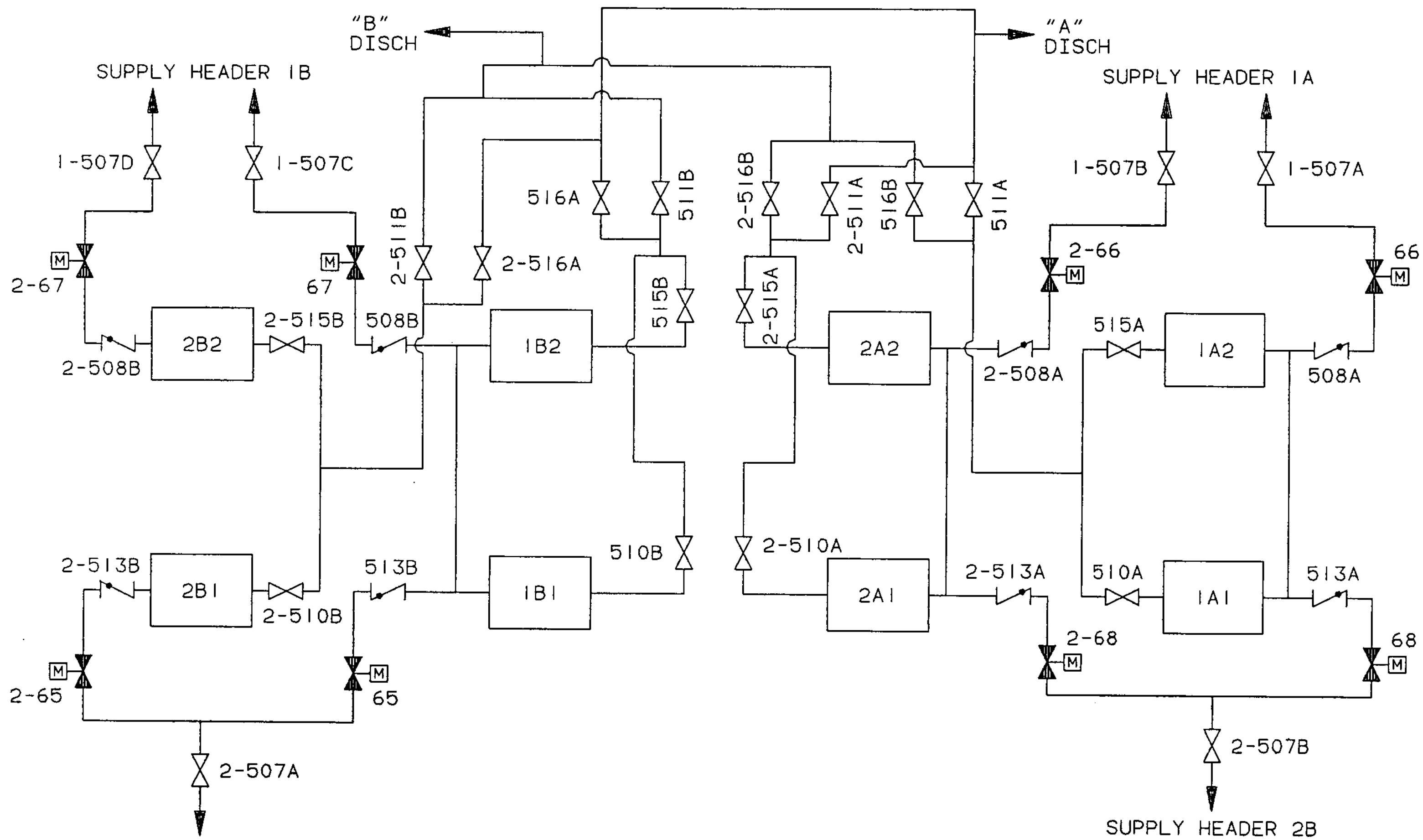
EPS
PAGE 3 OF 17

REFERENCE DRAWINGS:
N/A

FIGURE 3
161KV OFFSITE FEEDS

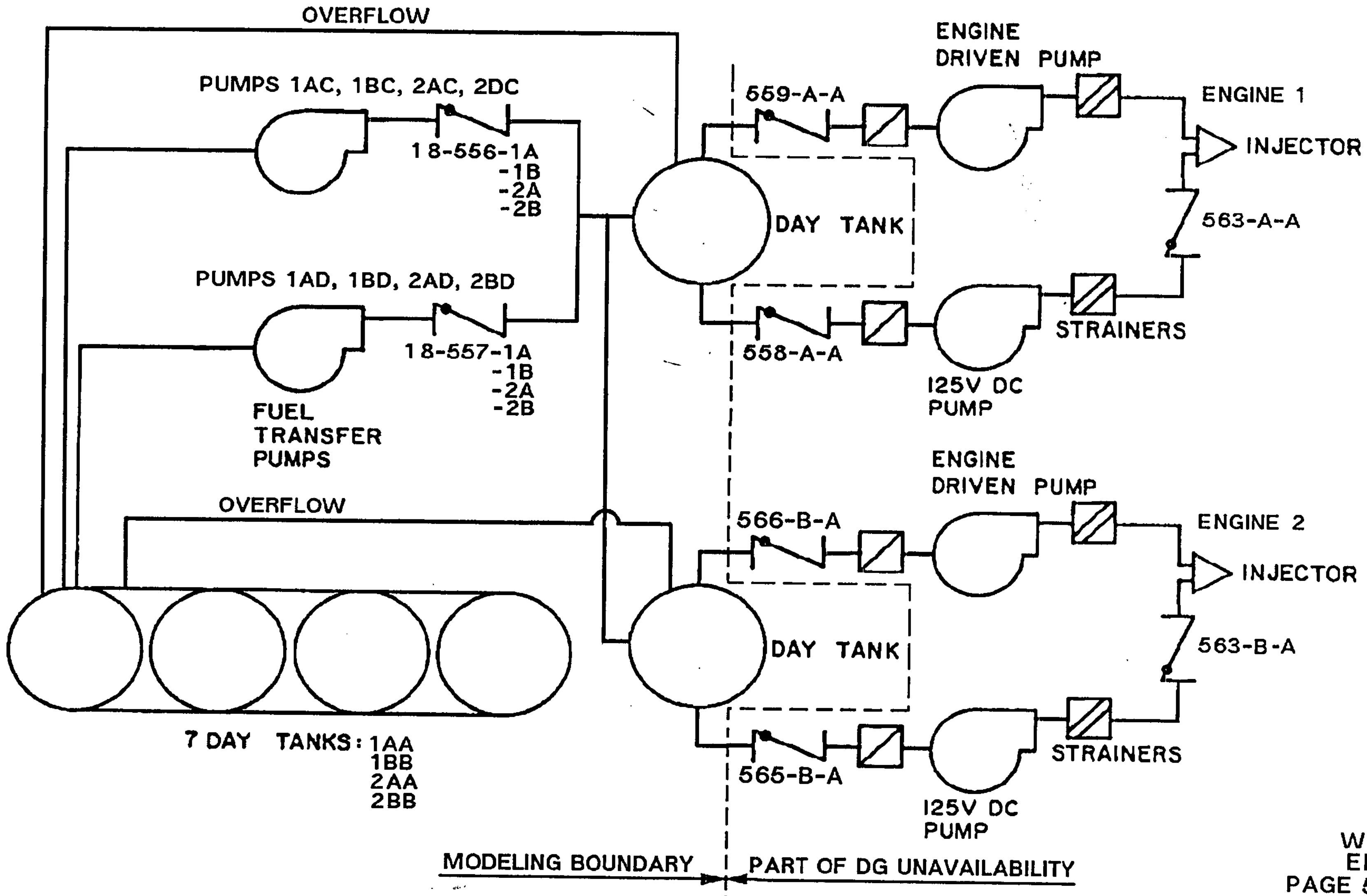


NOTE:
1. "2-" COMPONENT DENOTES A UNIT 2 CONPONENT.
2. SYSTEM PREFIX IS "67".

WBN
EPS

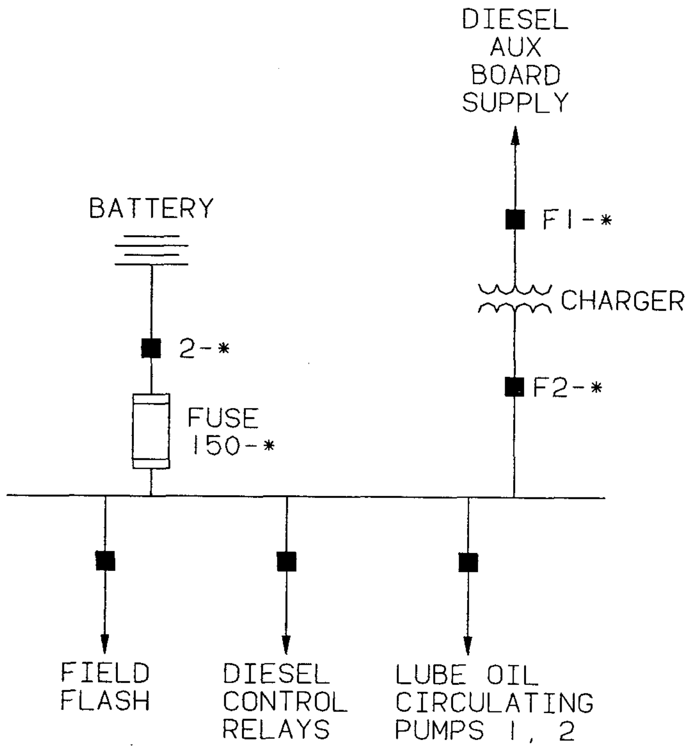
REFERENCE DRAWINGS:
1-47W845-1 R10

FIGURE 4
ESSENTIAL RAW COOLING WATER TO DIESEL GENERATORS



DIESEL GENERATOR FUEL OIL SUPPLY
FIGURE 5

REFERENCE DRAWINGS:
47W840-1 R26 (AC.BB)
47W840-1 R27
47W611-18-1 R5
47W610-18-1 R15
47W610-18-1 R14 (AC.U)



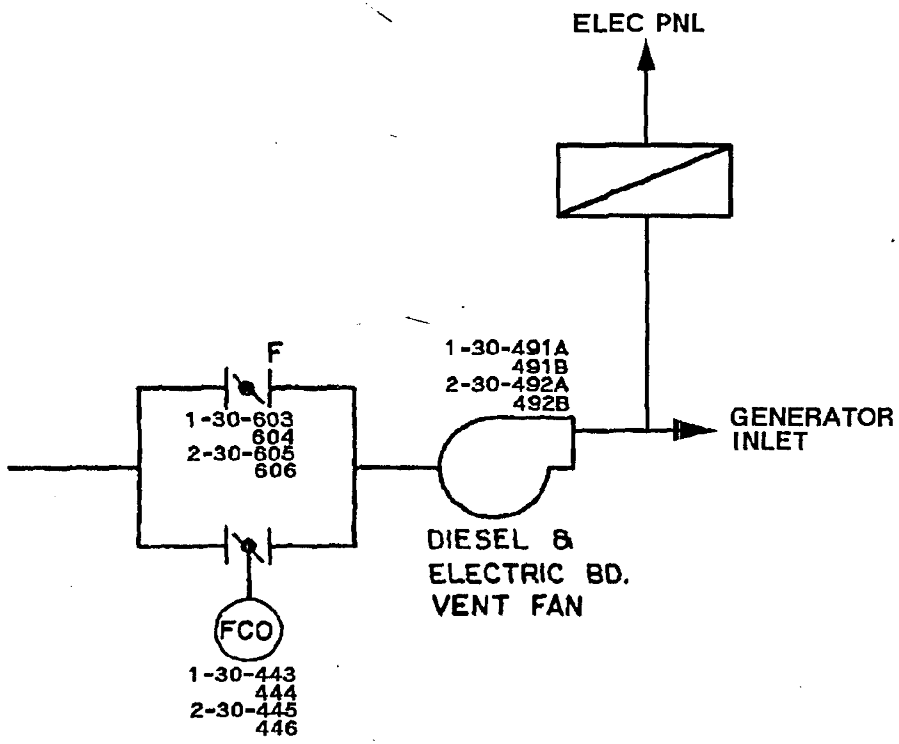
DIESEL	BATTERY	CHARGER	SUPPLY	*SUFFIX
1AA	DG1A	CH1A	1A2-A	1A
1BB	DG1B	CH1B	1B2-B	1B
2AA	DG2A	CH2A	2A2-A	2A
2BB	DG2B	CH2B	2B2-B	2B

FIGURE 6
DIESEL 125V DISTRIBUTION PANEL

REFERENCE DRAWINGS:

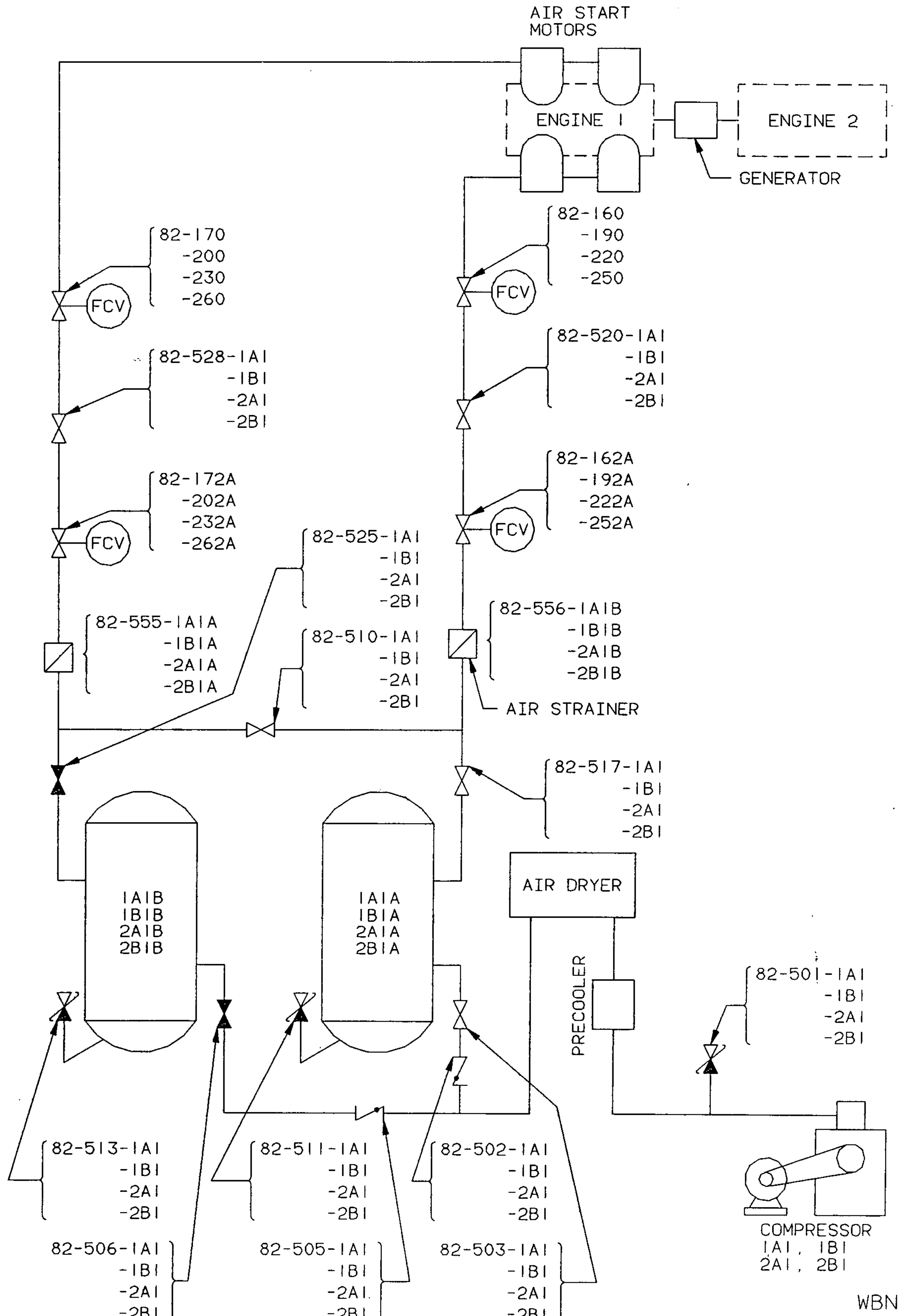
45N727 R2

OG-MS000167/3



DIESEL GENERATOR ROOM COOLING
FIGURE 7

REFERENCE DRAWINGS:
47W866-9 R9
47W866-9 R6 (AC.K)



WBN
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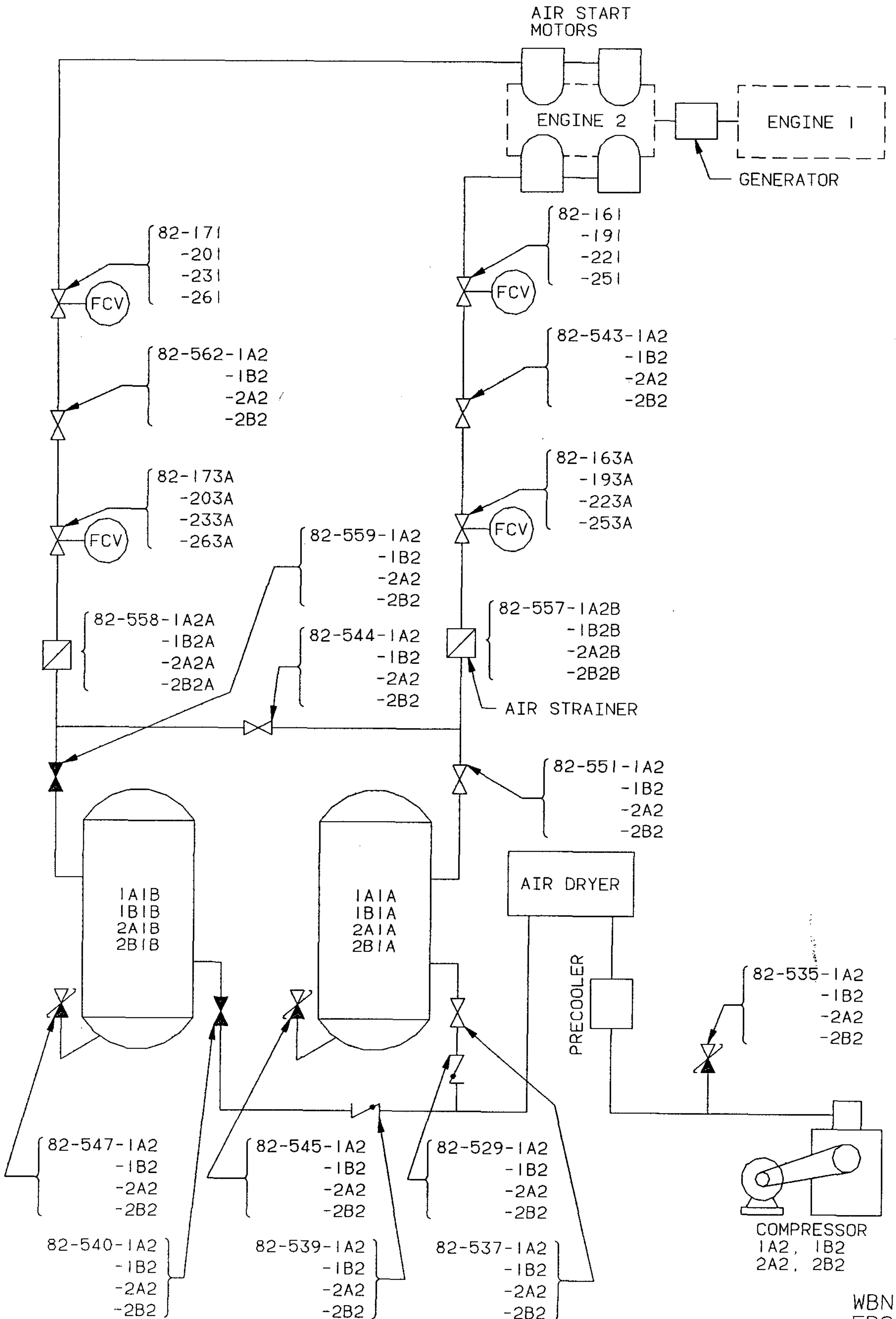
NOTE:
FOUR NUMBERS ARE GIVEN FOR EACH COMPONENT CORRESPONDING TO DIESEL SETS 1A-A, 1B-B, 2A-A, 2B-B.

FIGURE 8
DIESEL AIR START SYSTEM
ENGINE 1A1 (1B1, 2A1, 2B1)

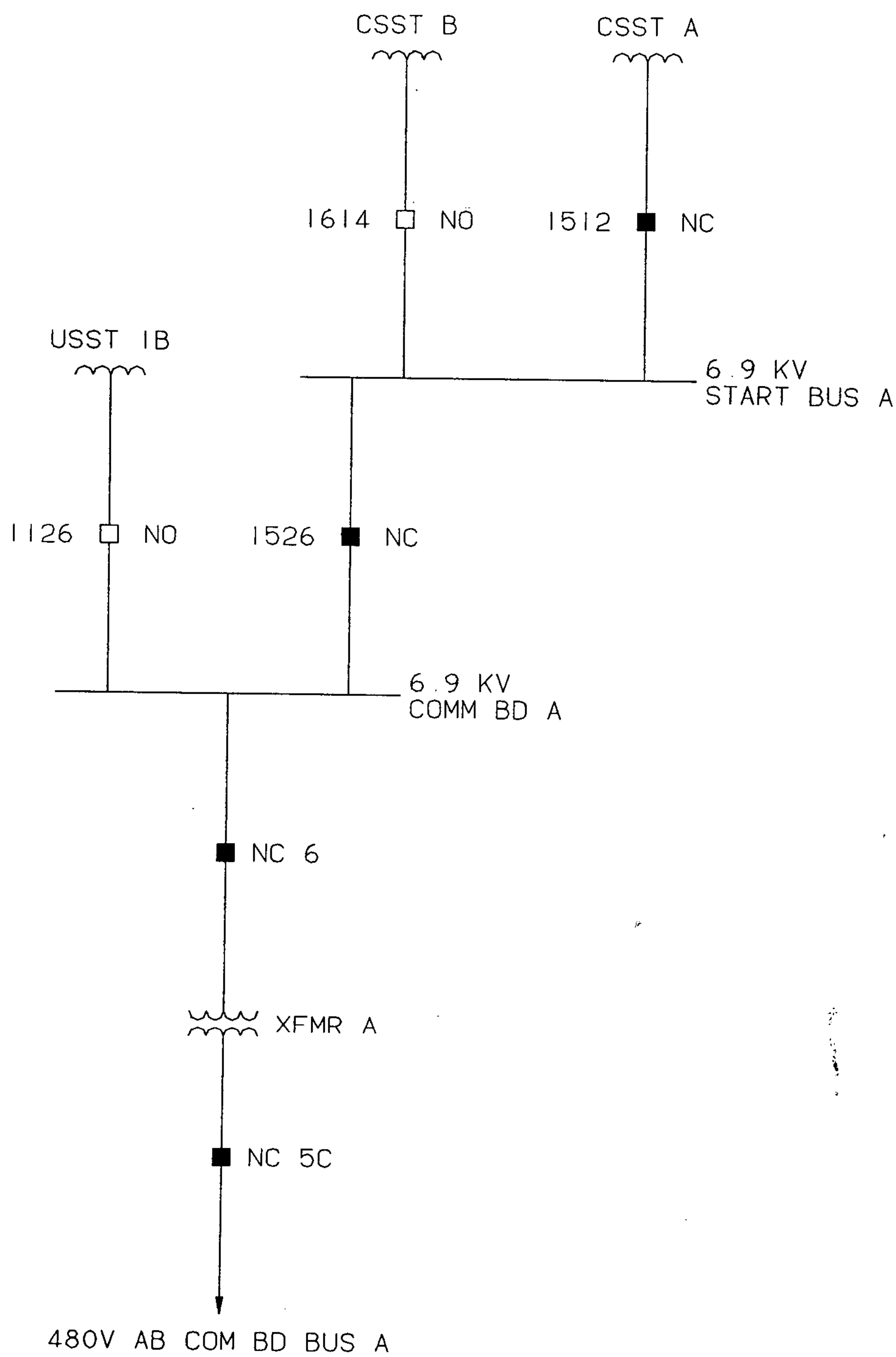
REFERENCE DRAWINGS:
1-47W839 R0



NOTE:
FOUR NUMBERS ARE GIVEN FOR EACH COMPONENT CORRESPONDING TO DIESEL SETS 1A-A, 1B-B, 2A-A, 2B-B.

REFERENCE DRAWINGS:
1-47W839 R0

FIGURE 9
DIESEL AIR START SYSTEM
ENGINE 1A2 (1B2, 2A2, 2B2)



WBN
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PAGE 10 OF 17

REFERENCE DRAWINGS:
1-45W731 R1

FIGURE 10
480V AUX BLDG COMMON BD, BUS A

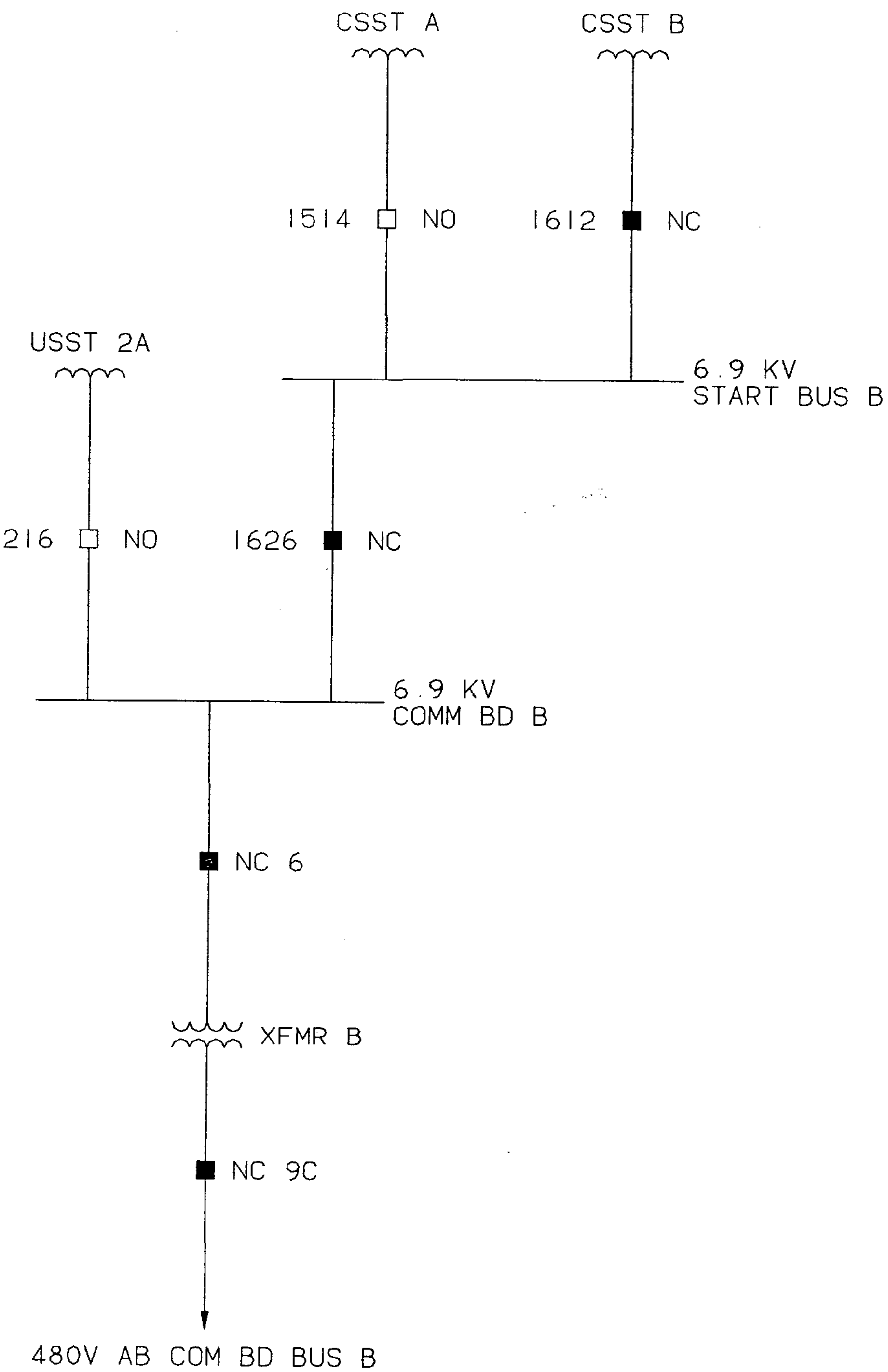
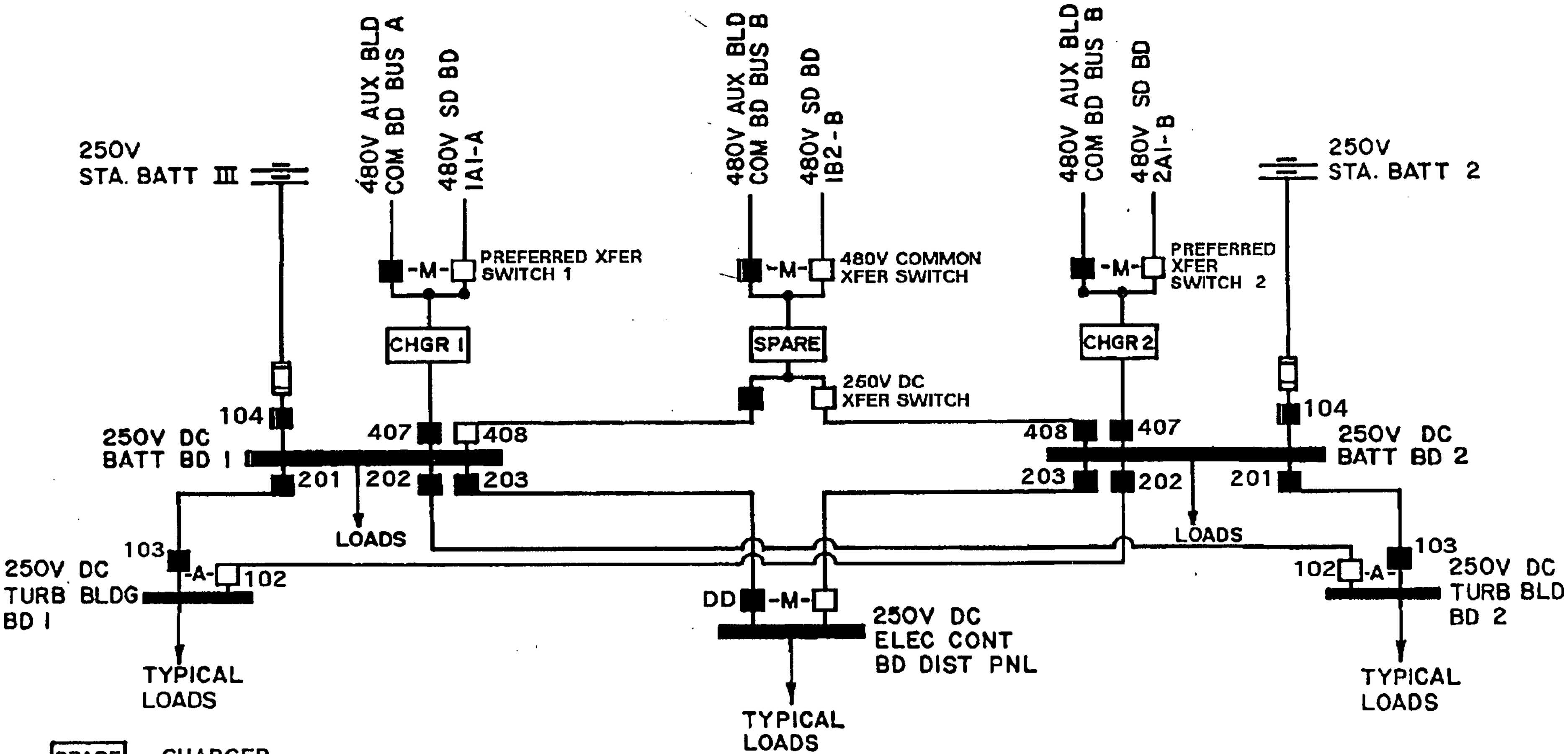


FIGURE 11
480V AUX BLDG COMMON BD, BUS B

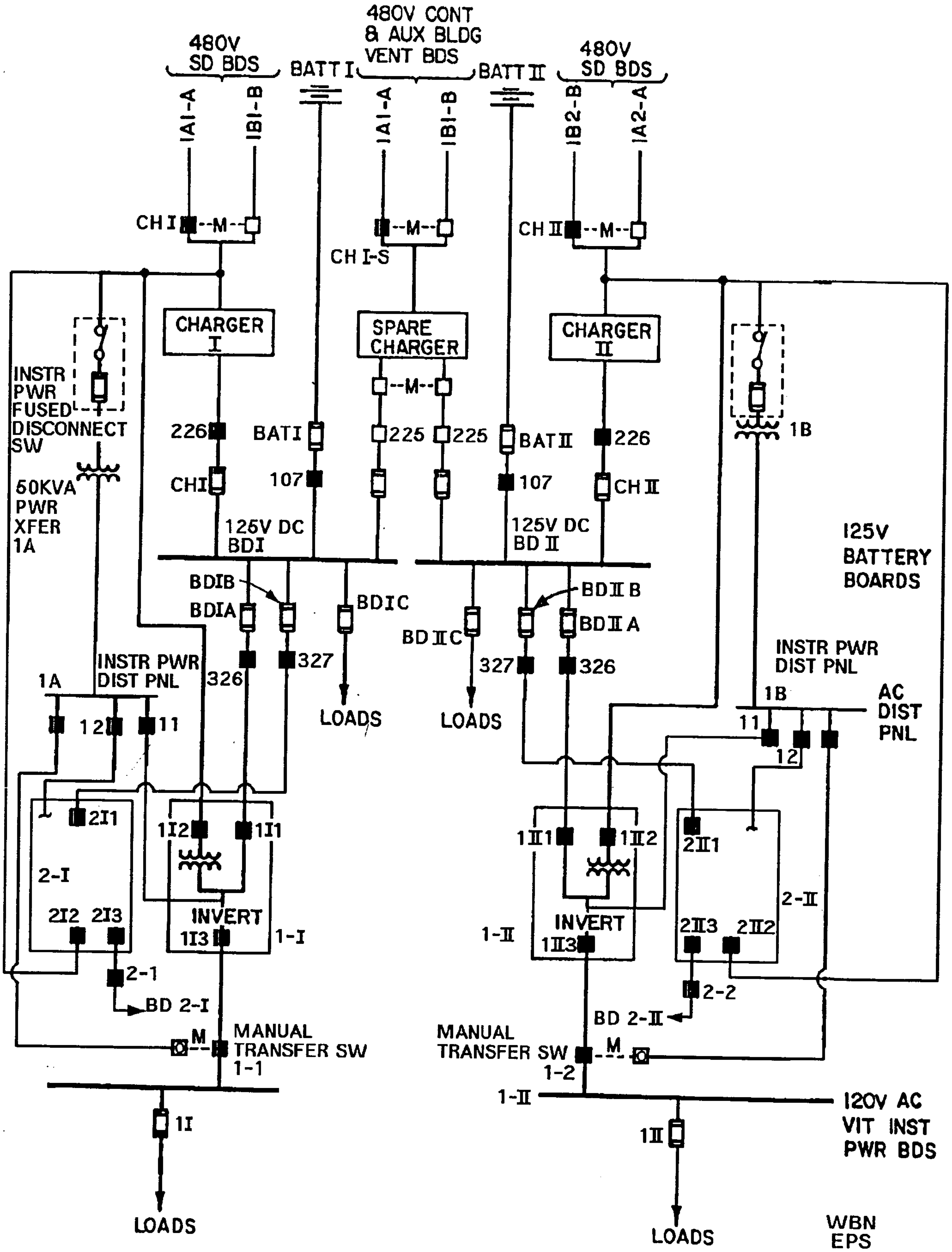
REFERENCE DRAWINGS:
1-45W731 R1



SPARE - CHARGER
- FUSE
- BREAKER NO
- BREAKER NC
-M- - MANUAL
-A- - AUTOMATIC

250V DC POWER SYSTEM
FIGURE 12

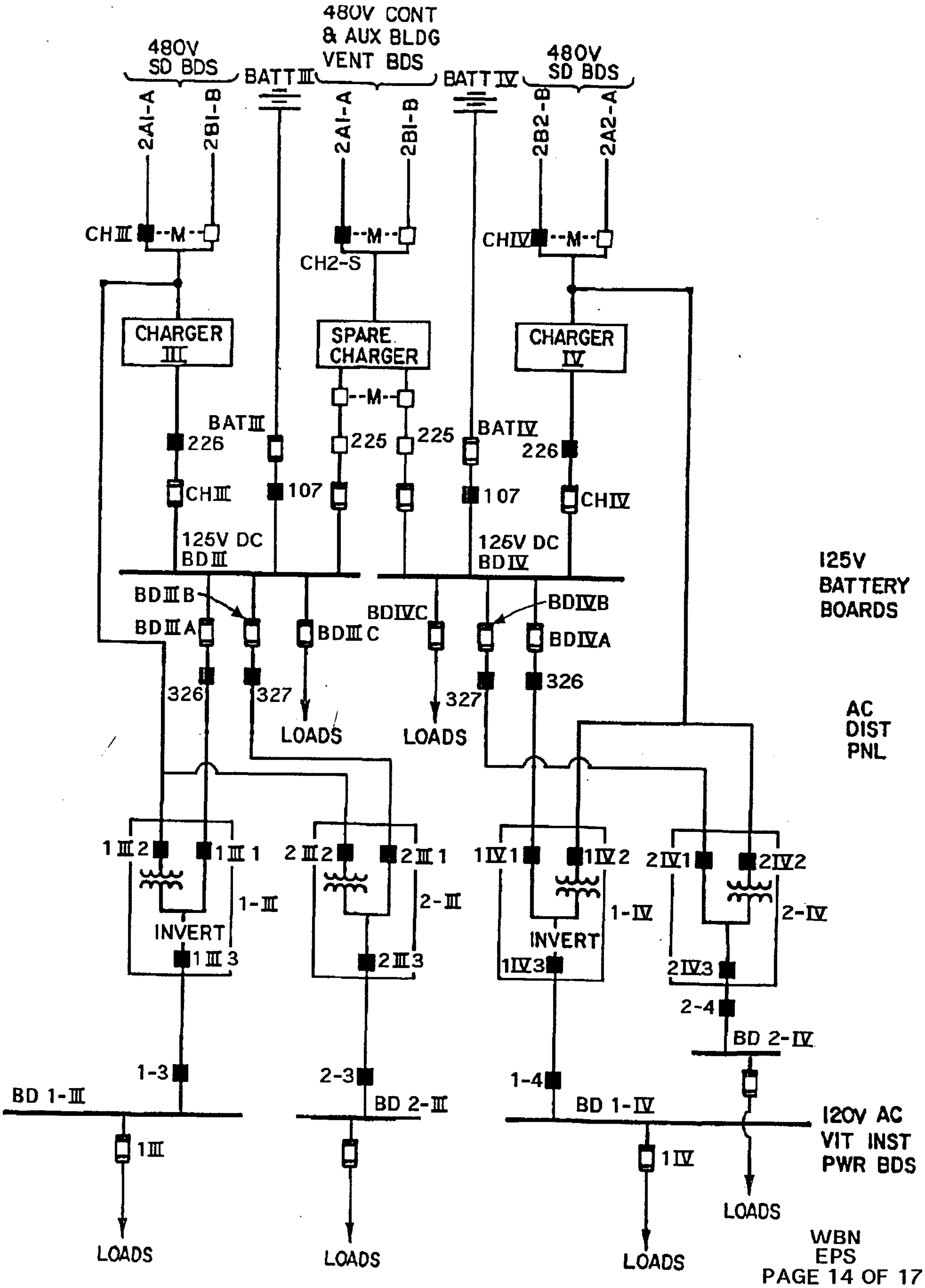
REFERENCE DRAWINGS:
45N700-2 R7 (AC.G)



REFERENCE DRAWINGS:
45N700-1 R4 (AC.H)
45N700-1 R5

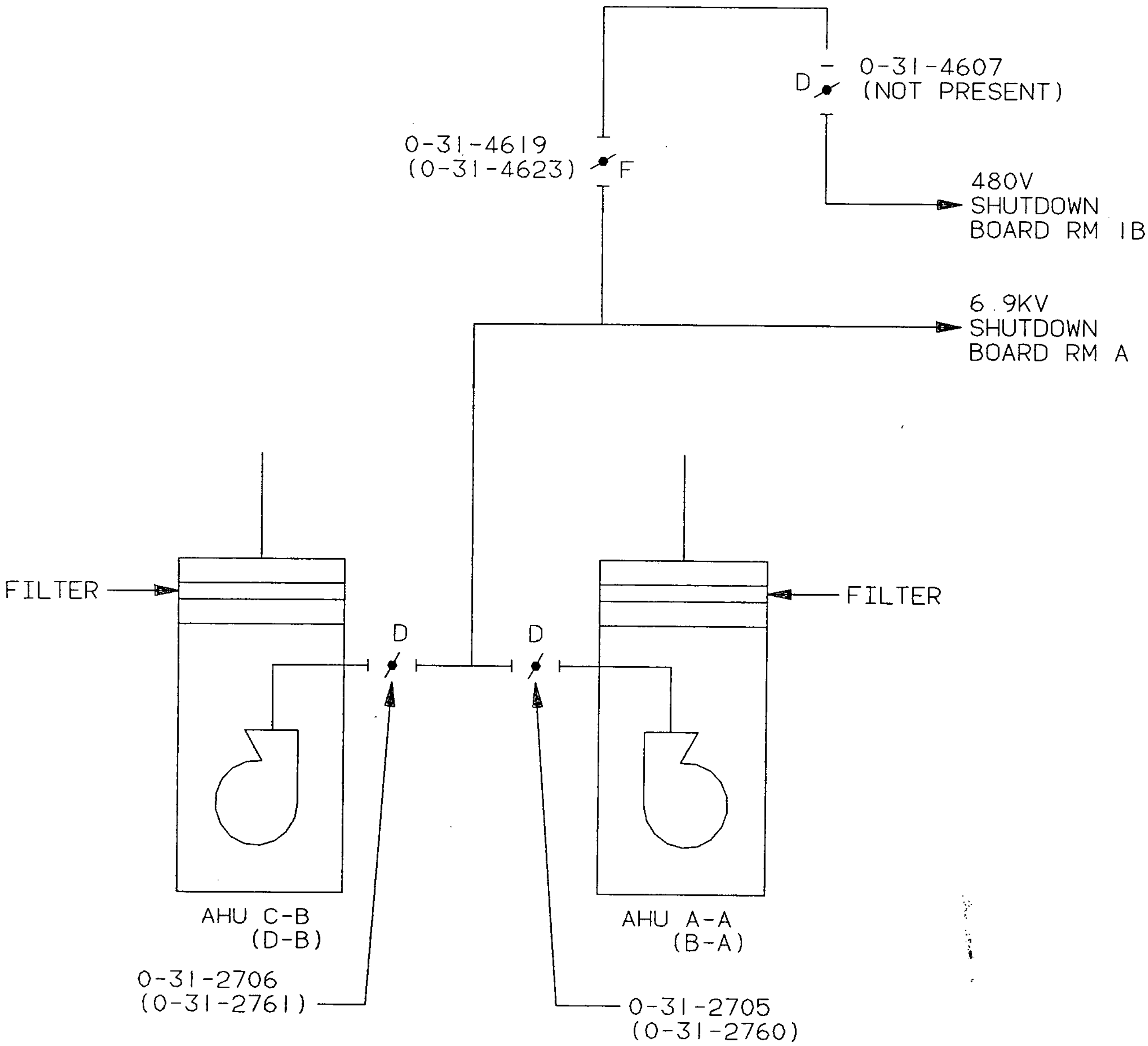
WBN 125V DC AND 120V AC POWER
FIGURE 13

OG-MIS000167/5



WBN 125V DC AND 120V AC POWER
FIGURE 14

REFERENCE DRAWINGS:
45N700-1 R4



NOTE:
UNIT 2 COMPONENT NUMBERS IN PARAENTHESIS.

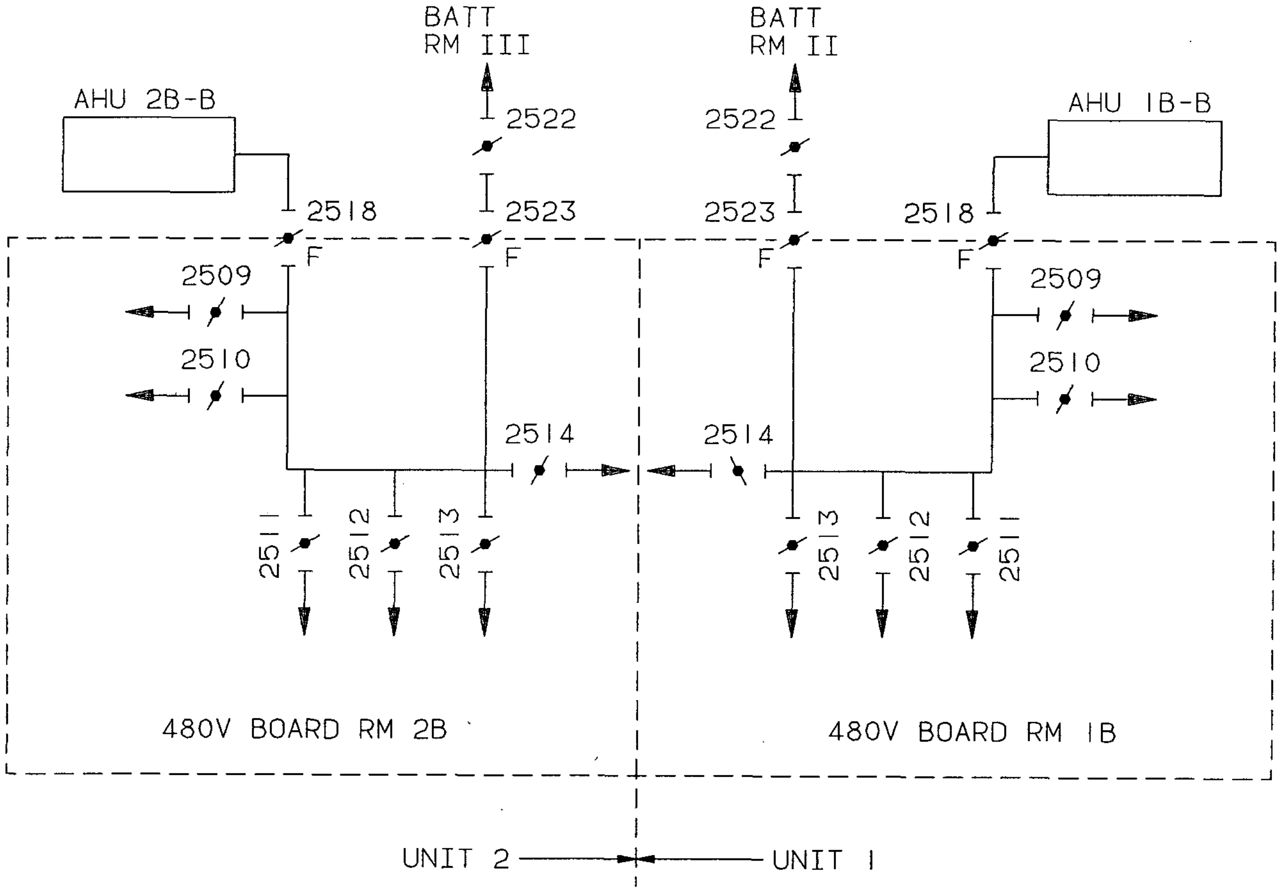
REFERENCE DRAWINGS:

1-47W866-2 R9
1-47W866-3 R2

FIGURE 15
6.9KV SHUTDOWN BOARD ROOM
480V SHUTDOWN BOARD ROOM VENTILATION

EA00.ENG.CS251211

WBN/EPS/16/17.R00



NOTE:

ALL COMPONENTS HAVE PREFIX "1-31-" FOR UNIT 1, OR "2-31-" FOR UNIT 2

ALL DAMPERS ARE BALANCING EXCEPT THOSE WITH AN "F" ADJACENT TO THEM.

WBN
EPS

REFERENCE DRAWINGS
1-47W866-2 R2 1-47W866-3 R3

FIGURE 16
480V BOARD ROOM VENTILATION

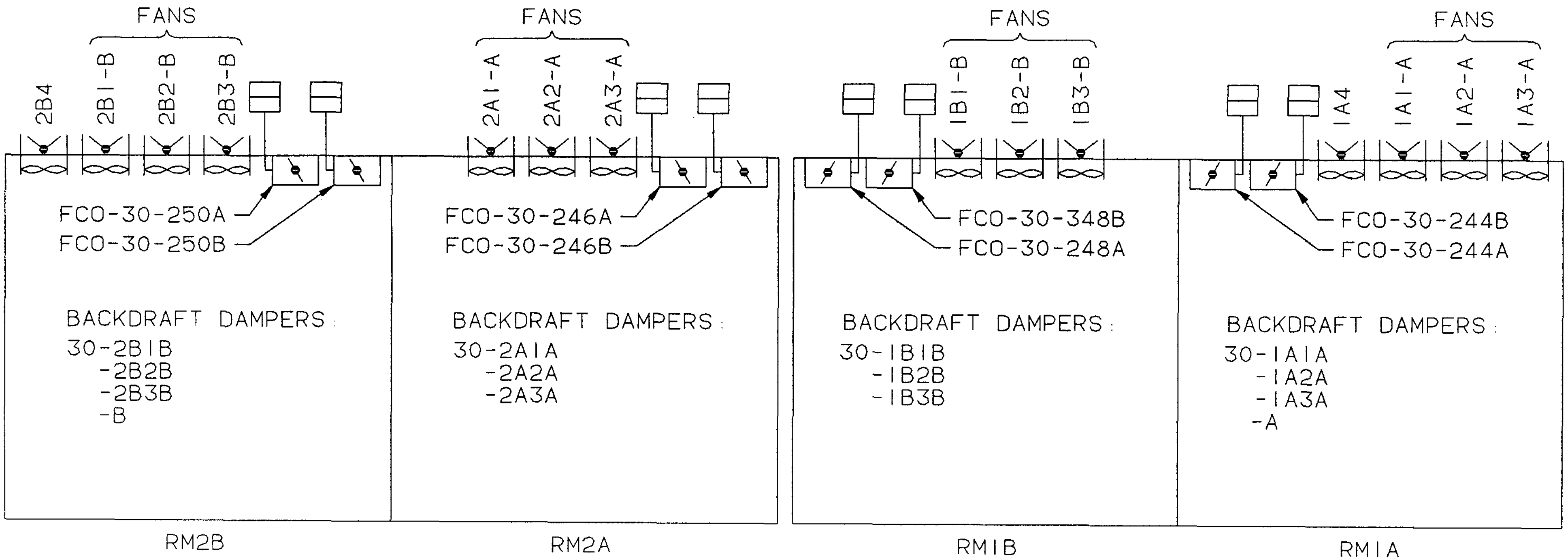
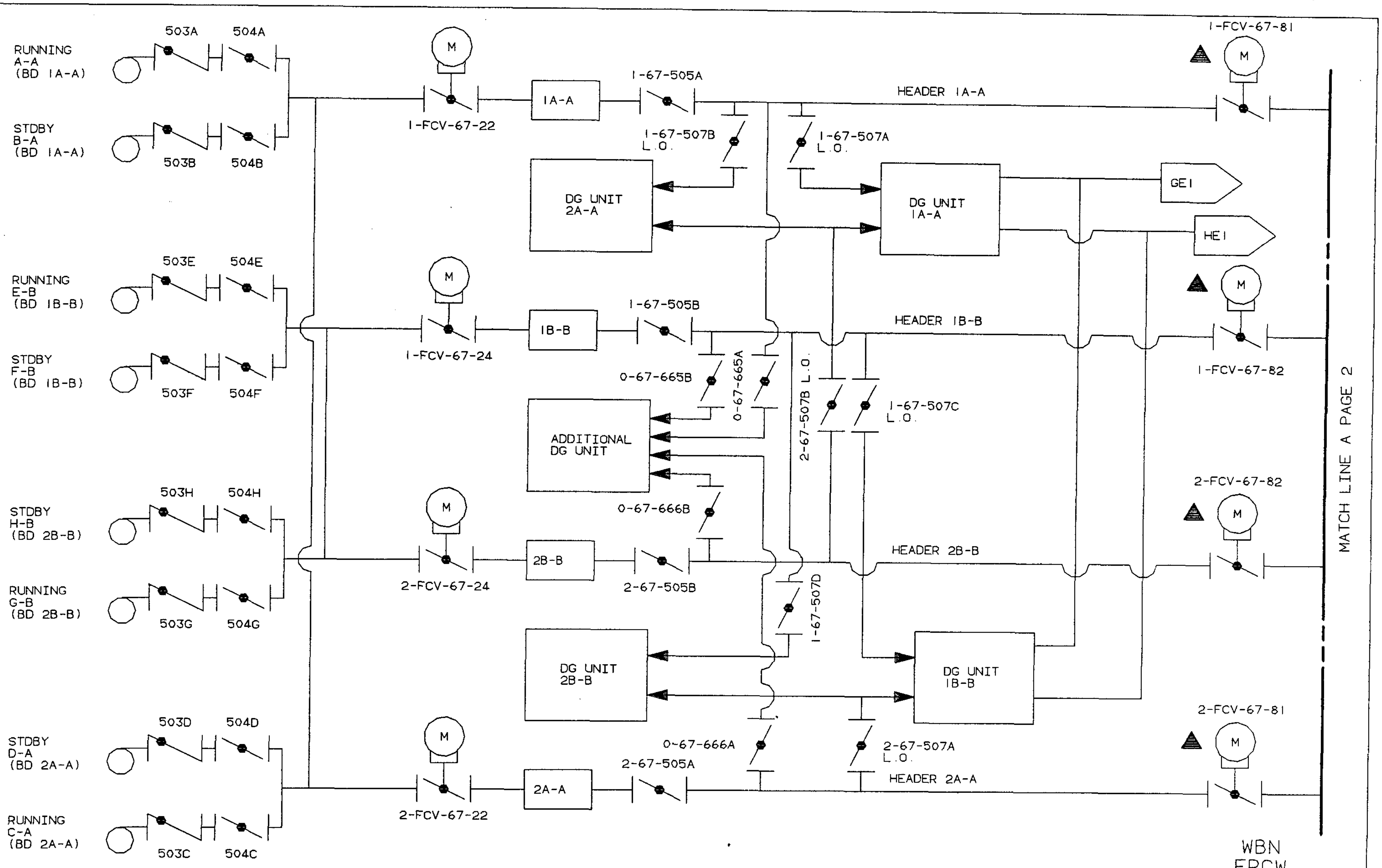


FIGURE 17
480V TRANSFORMER ROOM VENTILATION

REFERENCE DRAWINGS:
1,2-47W866-3 R2

SECTION 3.2.1.2 — ESSENTIAL RAW COOLING WATER SYSTEM FIGURES

EA00,ENG,CS251211



WBN
ERCW
PAGE 1 OF 3

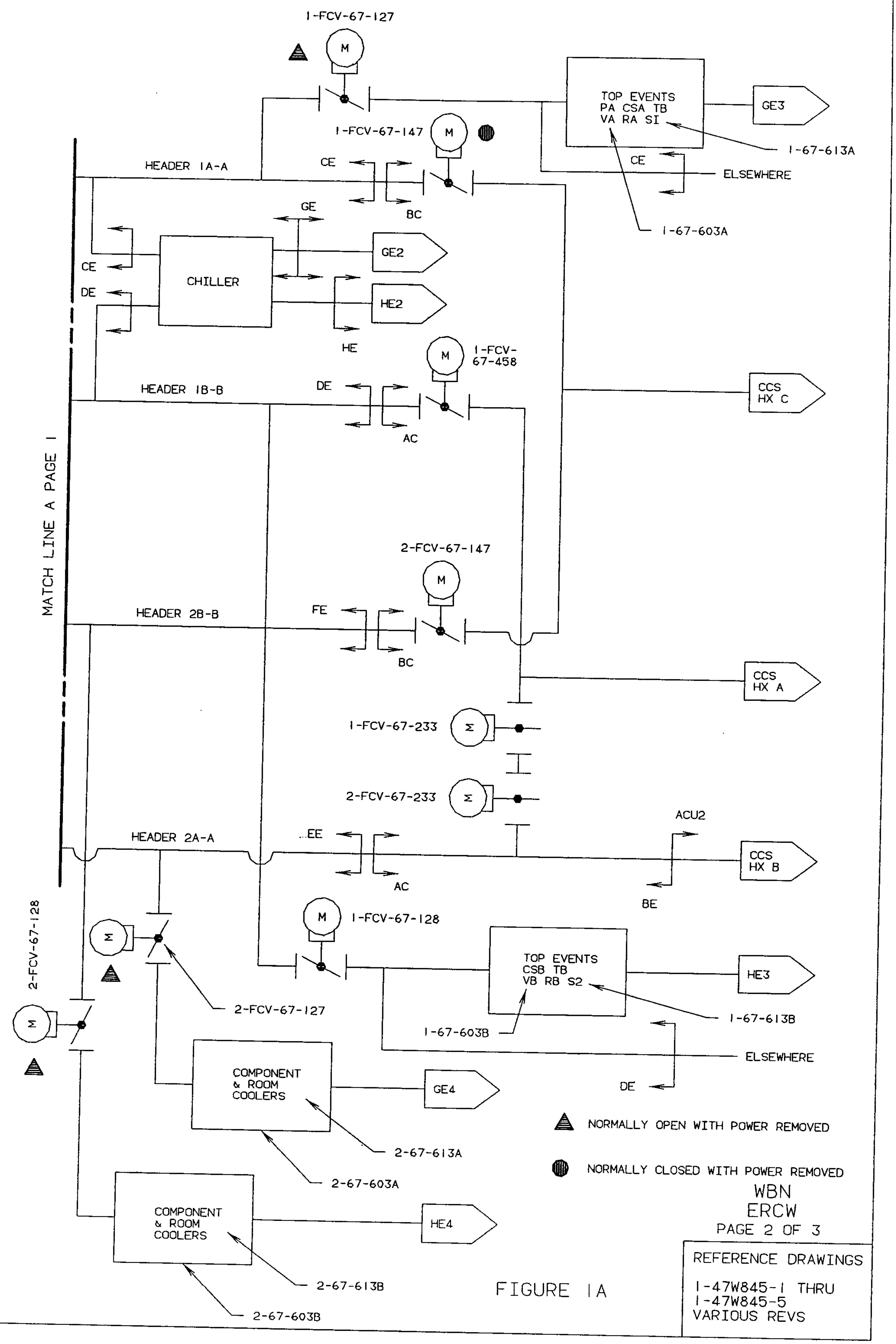
FIGURE 1A

▲ NORMALLY OPEN WITH POWER REMOVED

● NORMALLY CLOSED WITH POWER REMOVED

REFERENCE DRAWINGS

1-47W845-1 THRU
1-47W845-5
VARIOUS REVS



WBN
ERCW
PAGE 2 OF 3

REFERENCE DRAWINGS
1-47W845-1 THRU
1-47W845-5
VARIOUS REVS

FIGURE 1A

WBN/ERCW/03/03,R00

EA00,ENG,CS251211

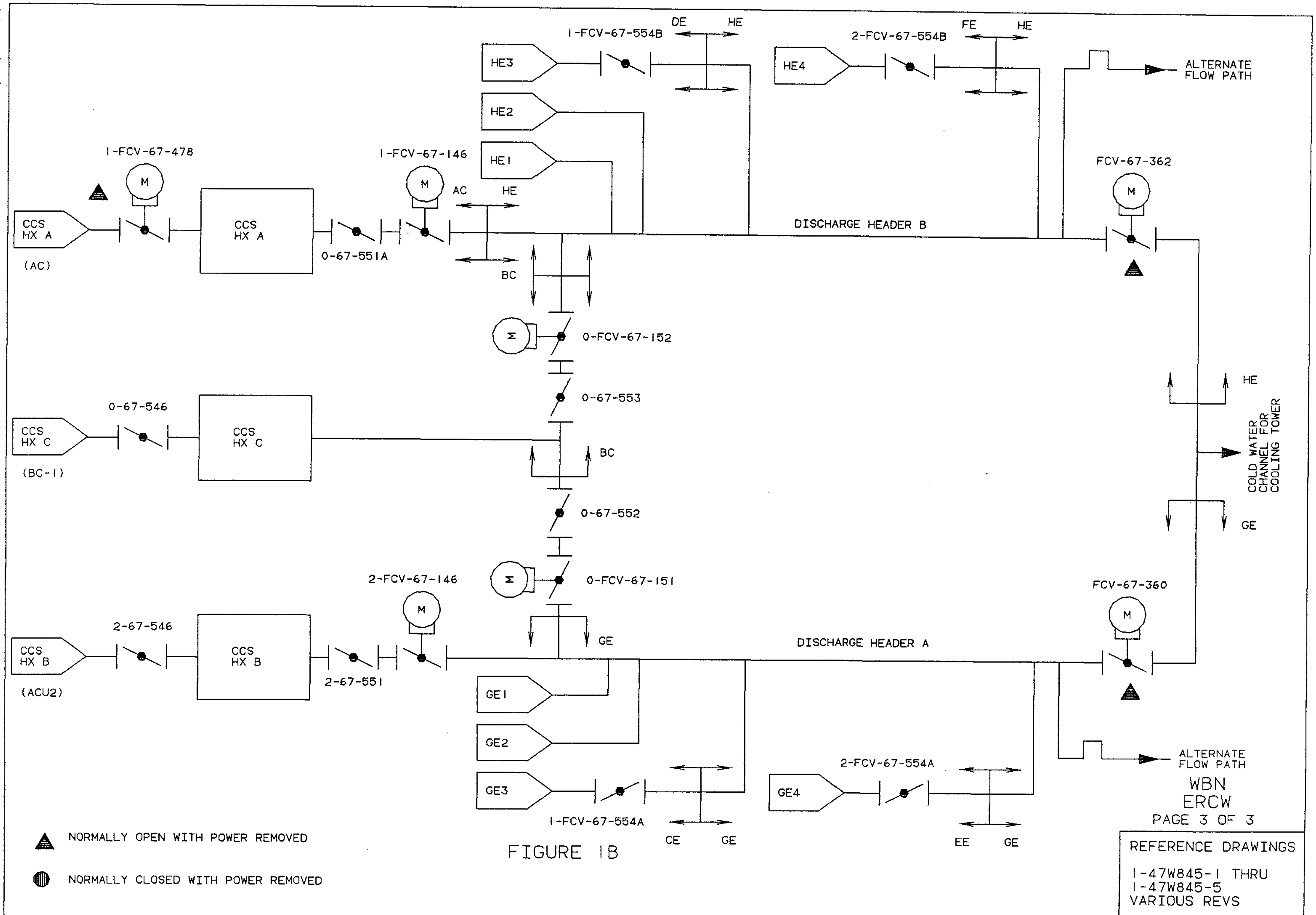


FIGURE 1B

WBN
ERCW
PAGE 3 OF 3

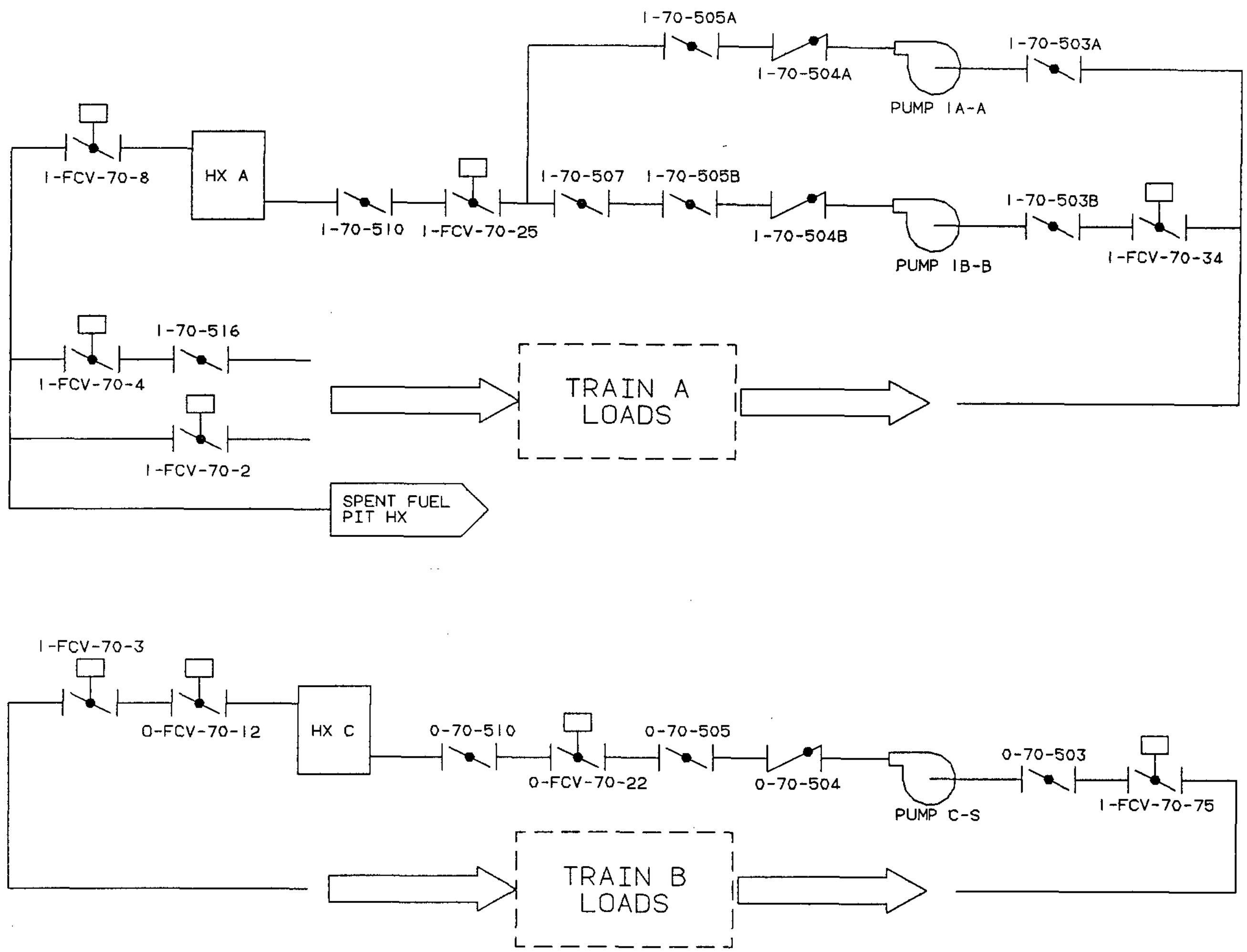
REFERENCE DRAWINGS

1-47W845-1 THRU
1-47W845-5
VARIOUS REVS

SECTION 3.2.1.3 – COMPONENT COOLING SYSTEM FIGURES

WBN/CCS/01/03.R00

EA00.ENG.CS251211



WBN
CCS
PAGE 1 OF 3

REFERENCE DRAWINGS
1-47W859-1 R14

FIGURE 1
COMPONENT COOLING SYSTEM

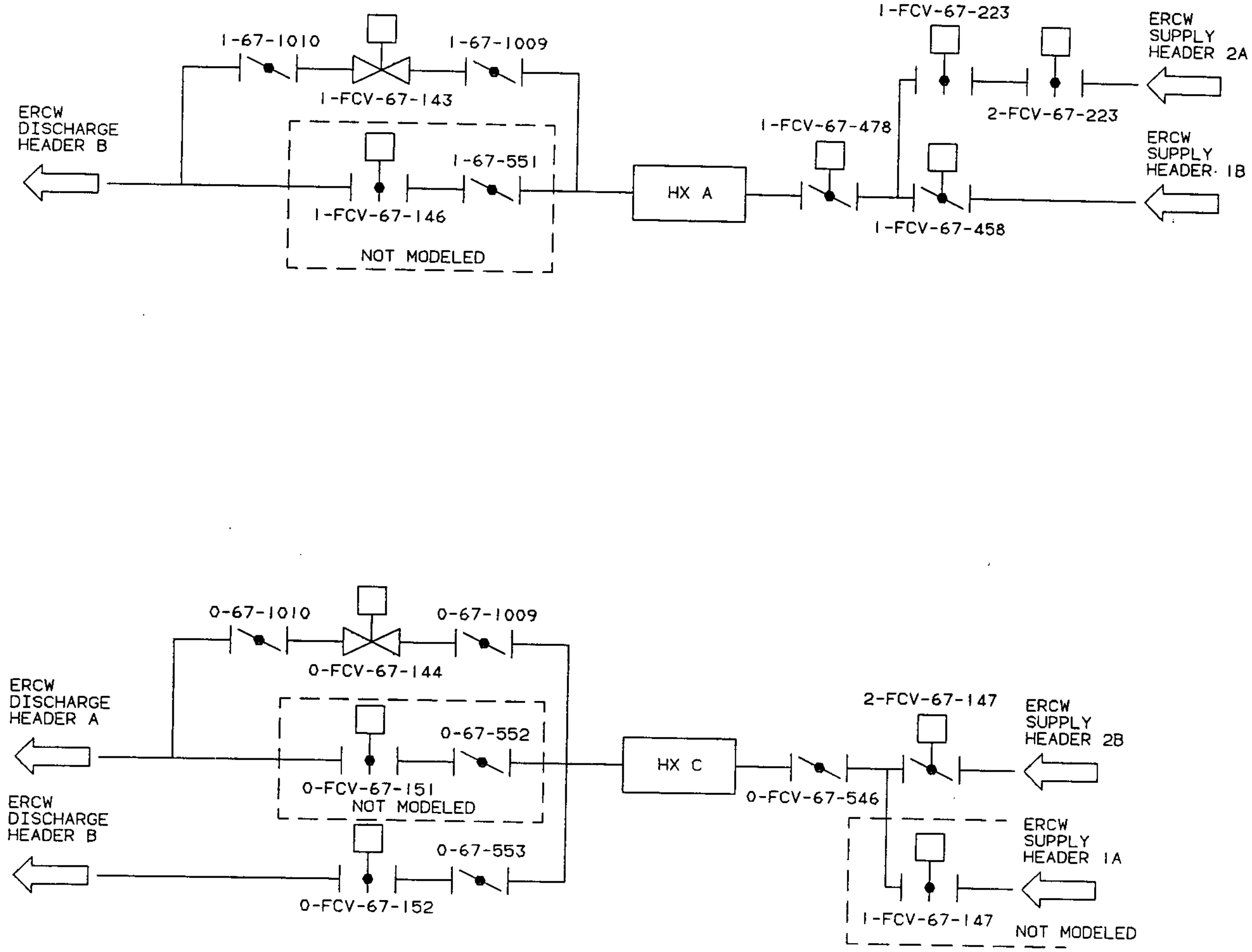


FIGURE 2
COMPONENT COOLING SYSTEM

WBN
CCS
PAGE 2 OF 3

REFERENCE DRAWINGS

1-47W845-2 R8
1-47W610-67-4 R1

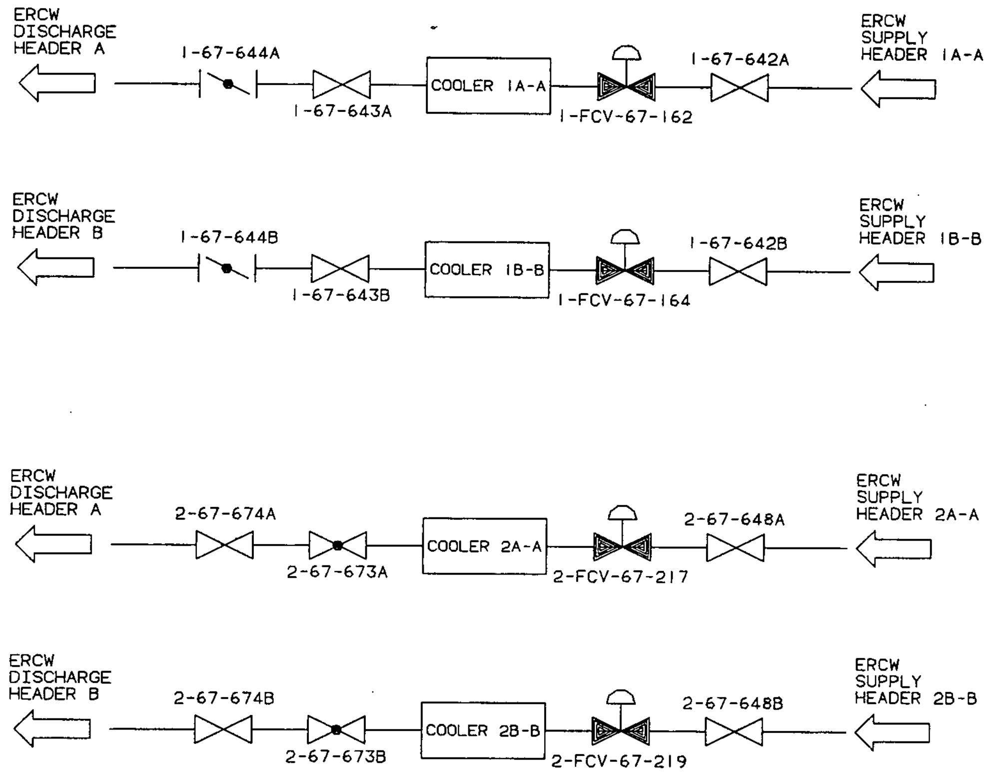


FIGURE 3
COMPONENT COOLING SYSTEM

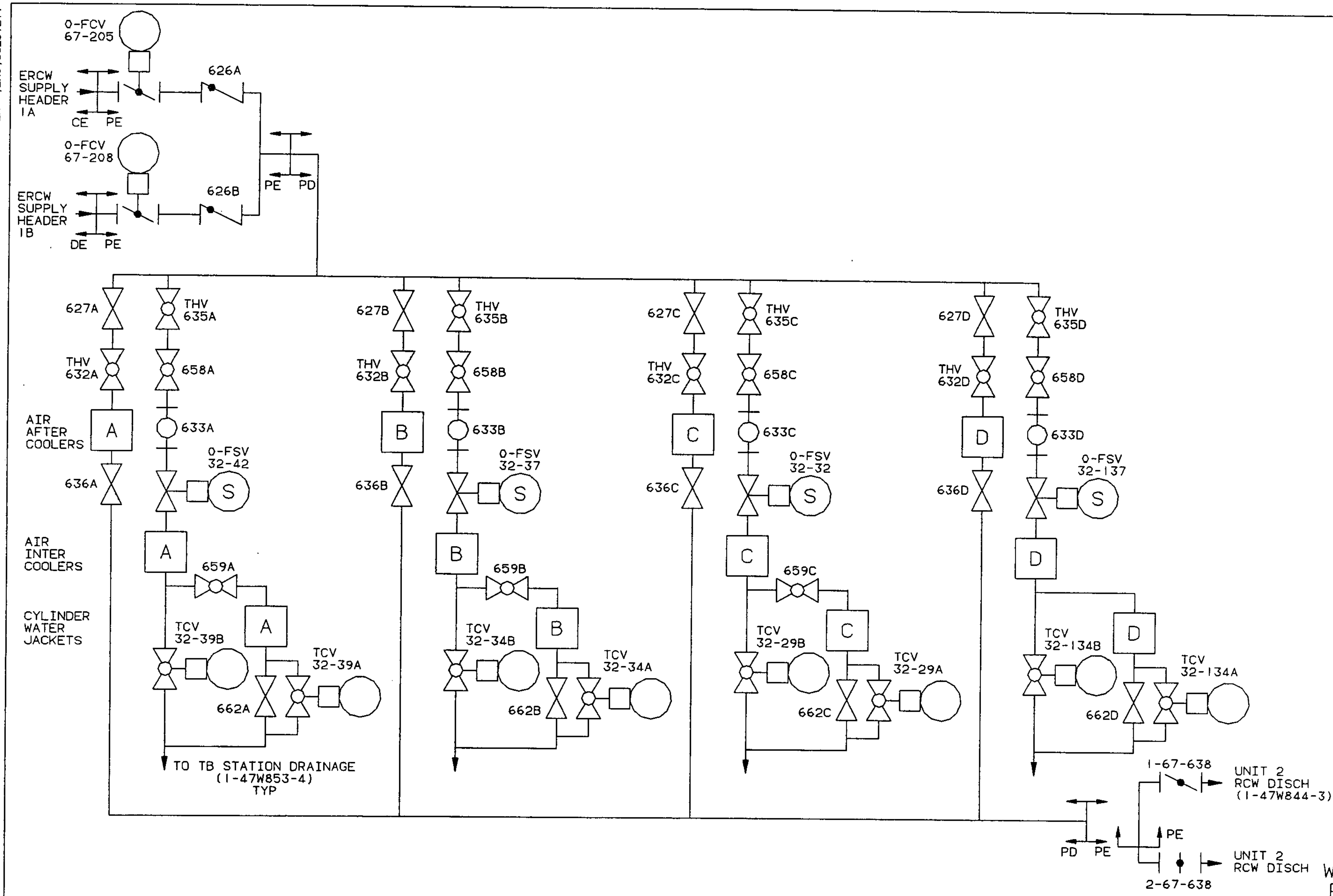
REFERENCE DRAWINGS

1-47W845-4 R5
1-47W845-7 R5

SECTION 3.2.1.4 — PLANT COMPRESSED AIR SYSTEM FIGURES

WBN/PA/01/05,R00

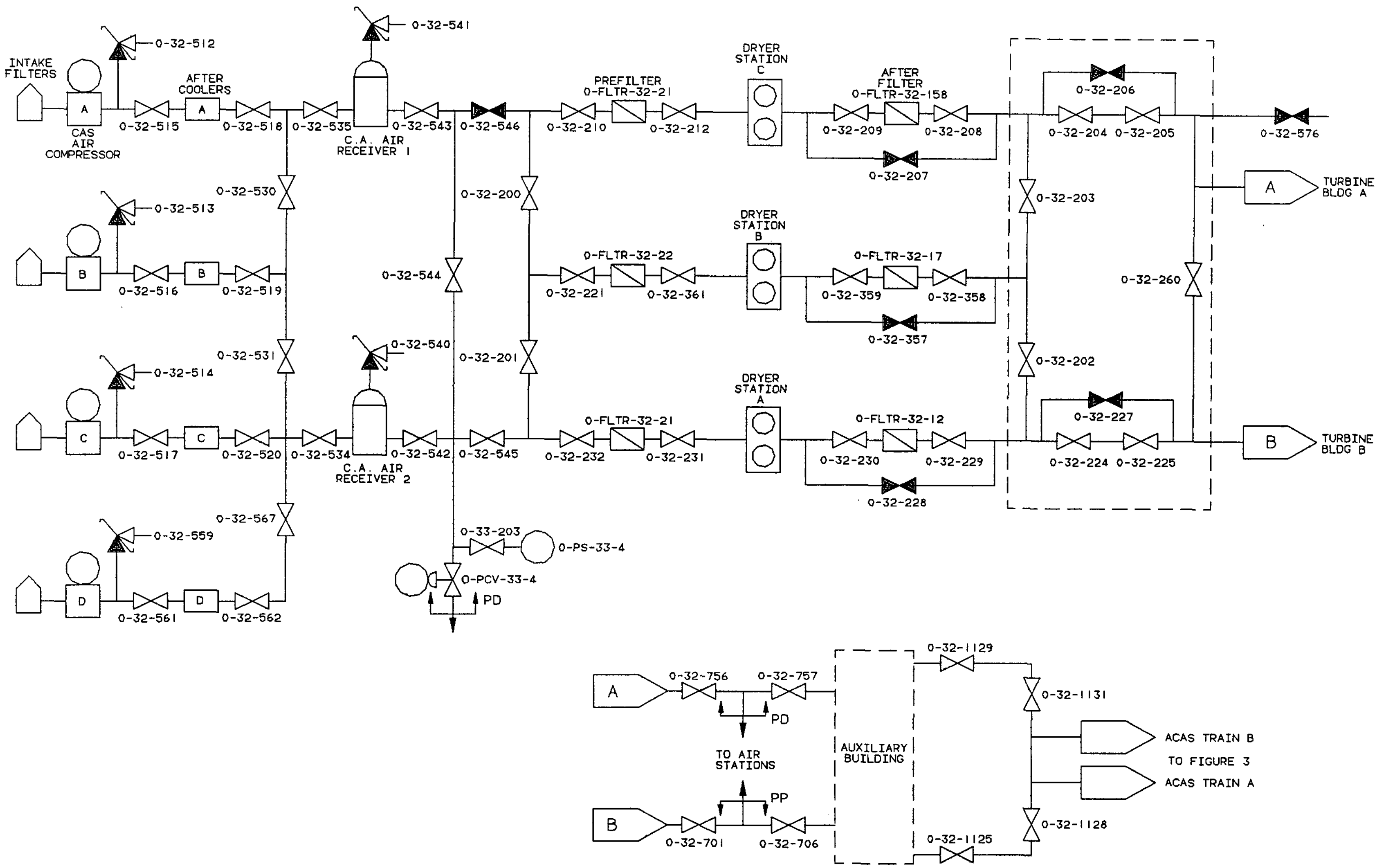
EA00,ENG,CS251211



WBN
PA
PAGE 1 OF 5

FIGURE 1
CAS COMPRESSOR ERCW COOLING SUPPLY

REFERENCE DRAWINGS:
1-47W845-5 R7



WBN
PA
PAGE 2 OF 5

FIGURE 2
CONTROL AND SERVICE AIR SYSTEM

REFERENCE DRAWINGS:
1-47W846-1 R7
1-47W848-1 TO -11
VARIOUS REVS

EA00,ENG,CS251211

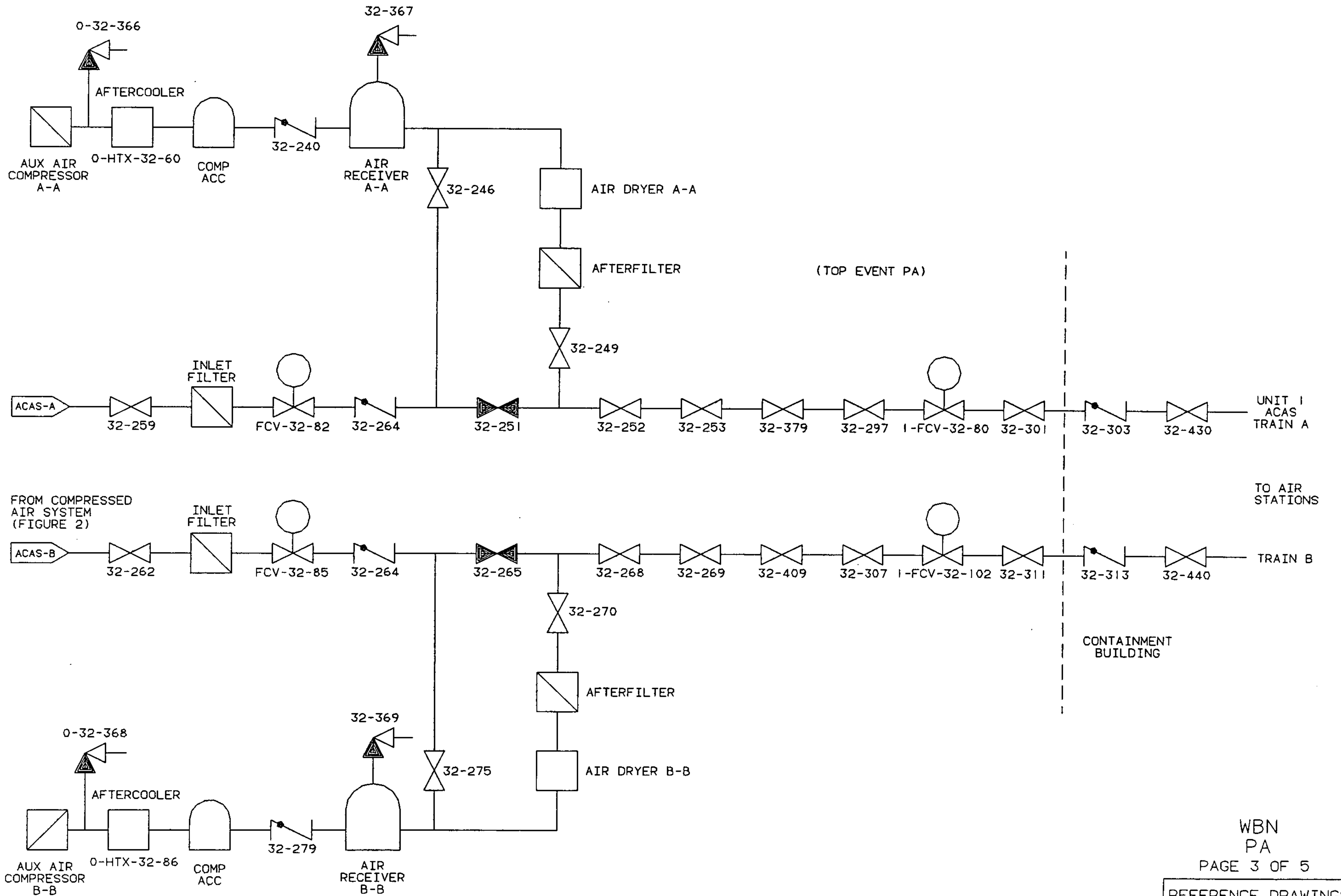


FIGURE 3
AUXILIARY CONTROL AIR SYSTEM

WBN
PA
PAGE 3 OF 5

REFERENCE DRAWINGS:

1,2-47W846-1 THRU
1,2-47W846-13
VARIOUS REVS

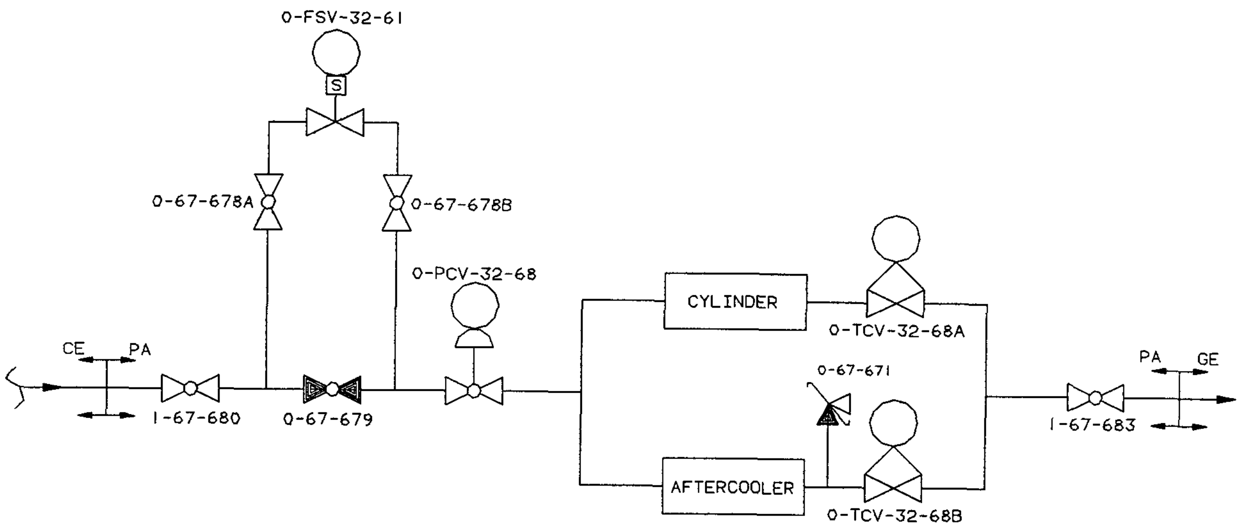


FIGURE 4A
ACAS COMPRESSOR A-A ERCW COOLING

WBN
PA

REFERENCE DRAWINGS:
1-47W845-4 R5

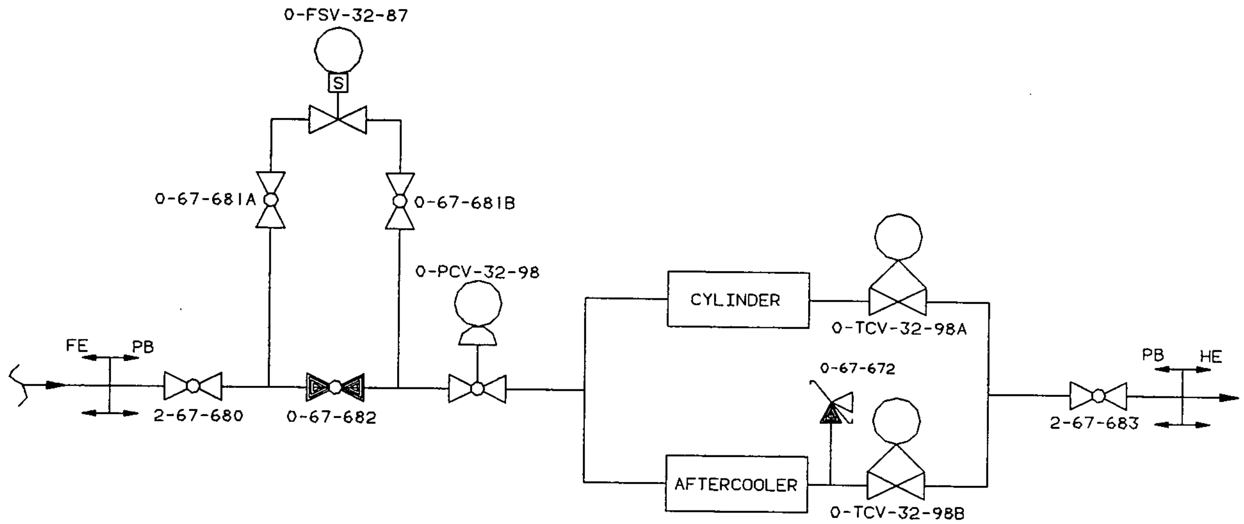
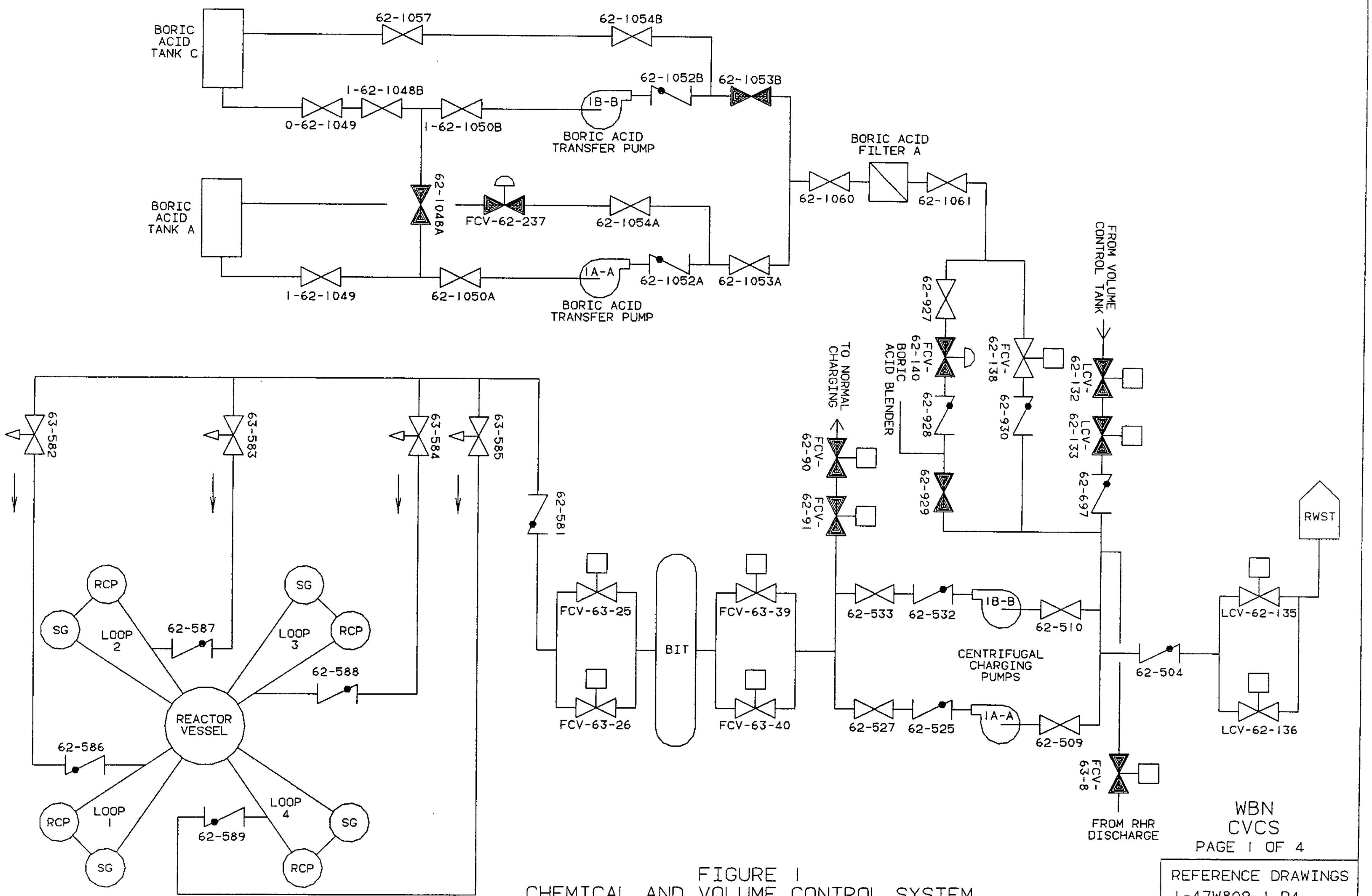


FIGURE 4B
ACAS COMPRESSOR B-B ERCW COOLING

REFERENCE DRAWINGS:

1-47W845-7 R5

SECTION 3.2.1.5 — CHEMICAL AND VOLUME CONTROL SYSTEM FIGURES



WBN
CVCS
PAGE 1 OF 4

FIGURE 1
CHEMICAL AND VOLUME CONTROL SYSTEM
(EMERGENCY BORATION VALVE ALIGNMENT)

REFERENCE DRAWINGS
1-47W809-1 R4
1-47W809-5 R2
1-47W811-1 R3
1-47W809-2 R2

EAOO.ENG.CS251211

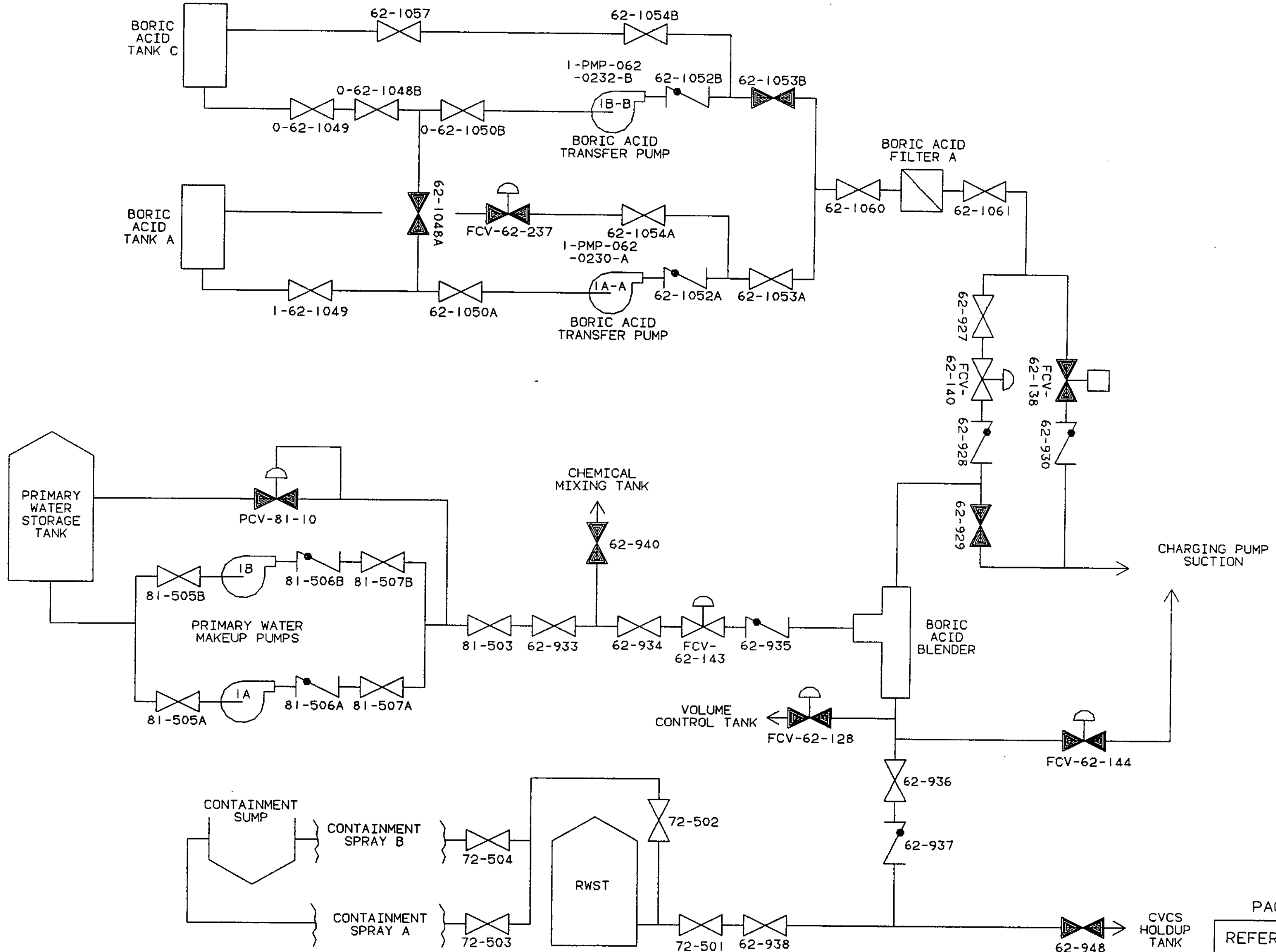


FIGURE 2
CHEMICAL AND VOLUME CONTROL SYSTEM
(MAKEUP TO RWST ALIGNMENT)

WBN
CVCS
PAGE 2 OF 4

REFERENCE DRAWINGS
1,2-47W809-5 R2
1,2-47W809-2 R2
1,2-47W819-1 R2
1,2-47W812-1 R1

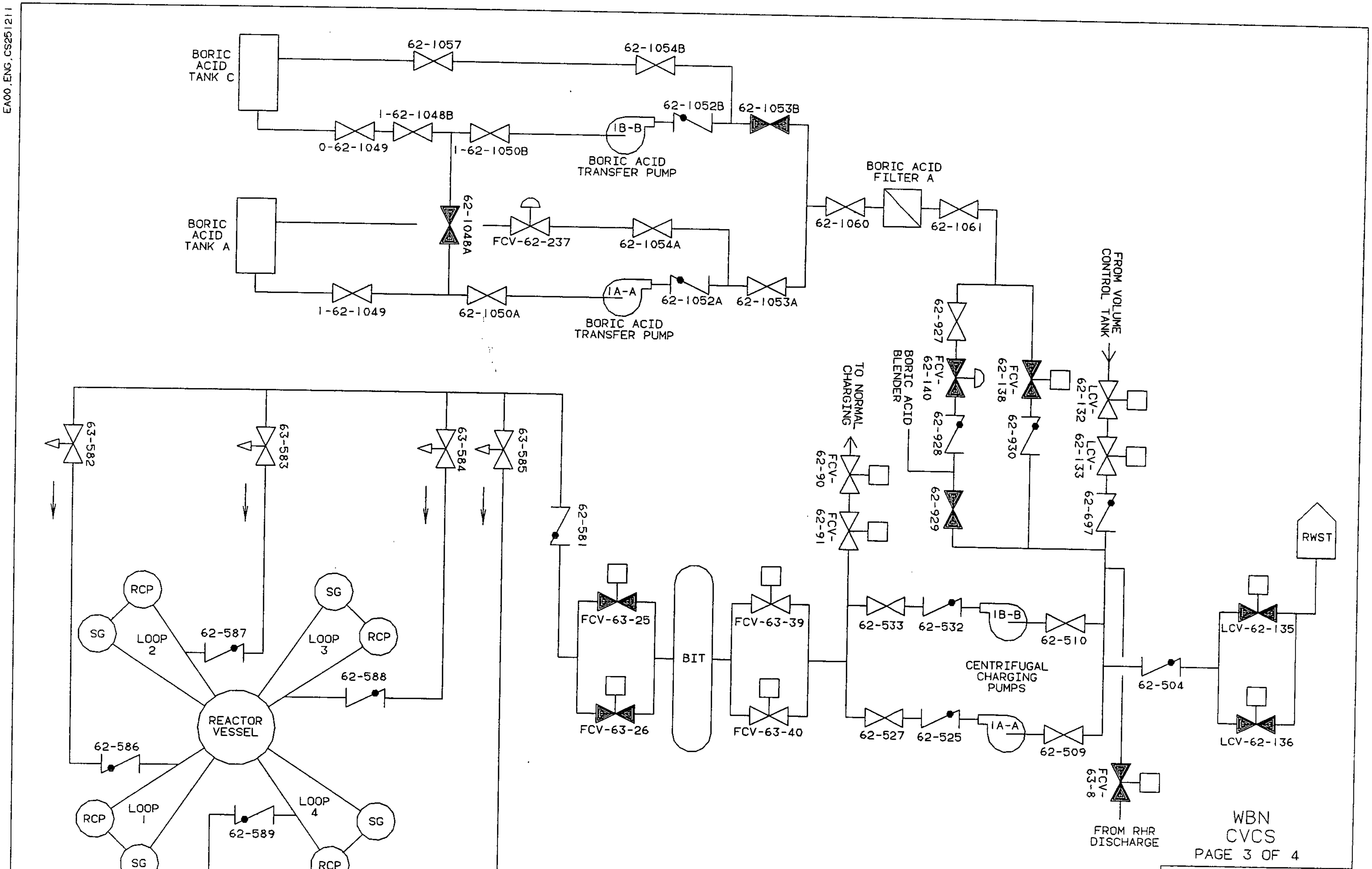
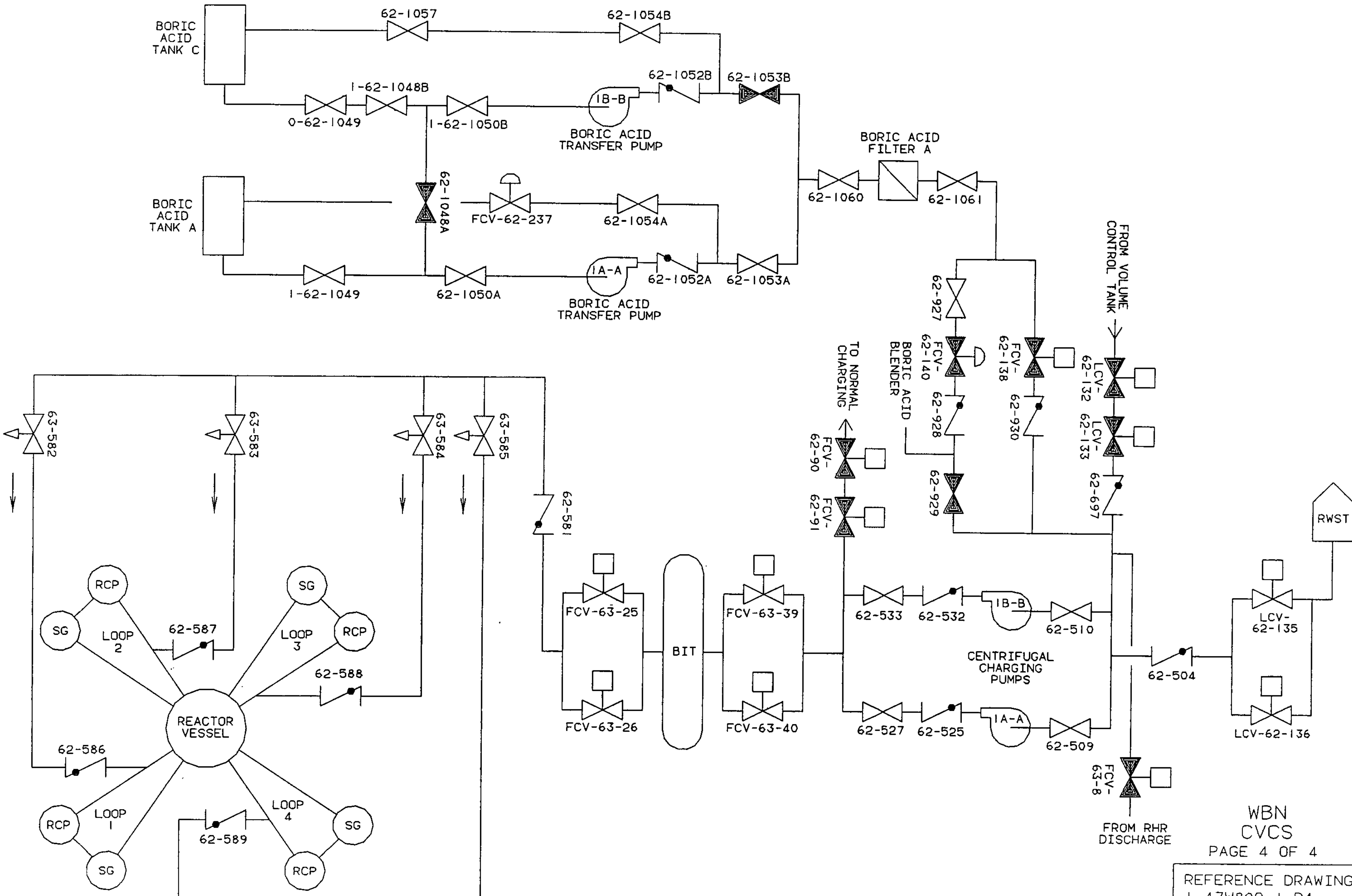


FIGURE 3
CHEMICAL AND VOLUME CONTROL SYSTEM
(NORMAL ALIGNMENT)

REFERENCE DRAWINGS
1-47W809-1 R4
1-47W809-5 R2
1-47W811-1 R3
1-47W809-2 R2

WBN/CVCS/04/04.R00

EA00.ENG.CS251211



WBN
CVCS
PAGE 4 OF 4

REFERENCE DRAWINGS
1-47W809-1 R4
1-47W809-5 R2
1-47W811-1 R3
1-47W809-2 R2

FIGURE 4
CHEMICAL AND VOLUME CONTROL SYSTEM
(COLD LEG INJECTION ALIGNMENT)

SECTION 3.2.1.6 — CONDENSATE AND FEEDWATER SYSTEM FIGURE

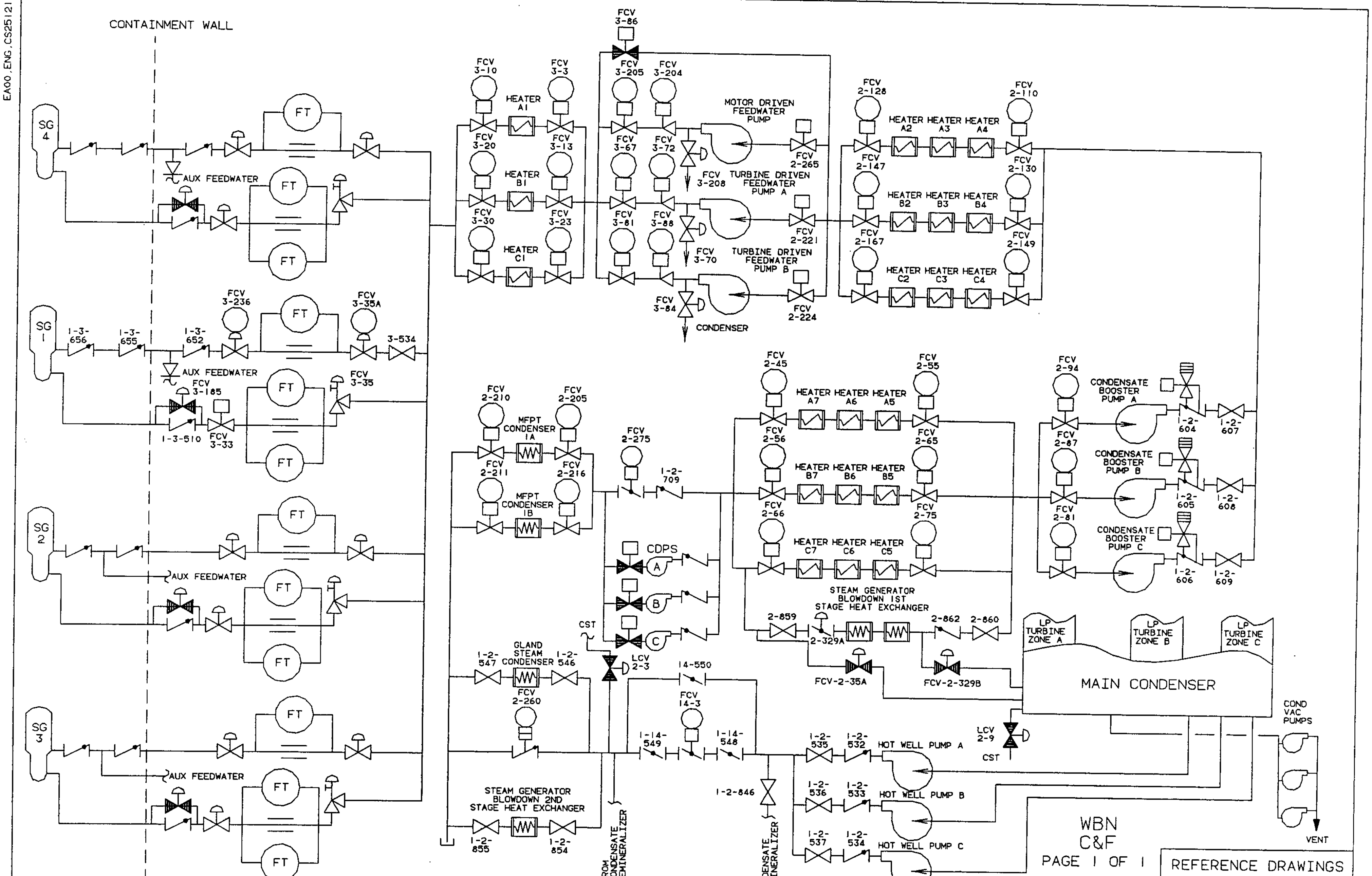


FIGURE 1
CONDENSATE AND FEEDWATER SYSTEMS

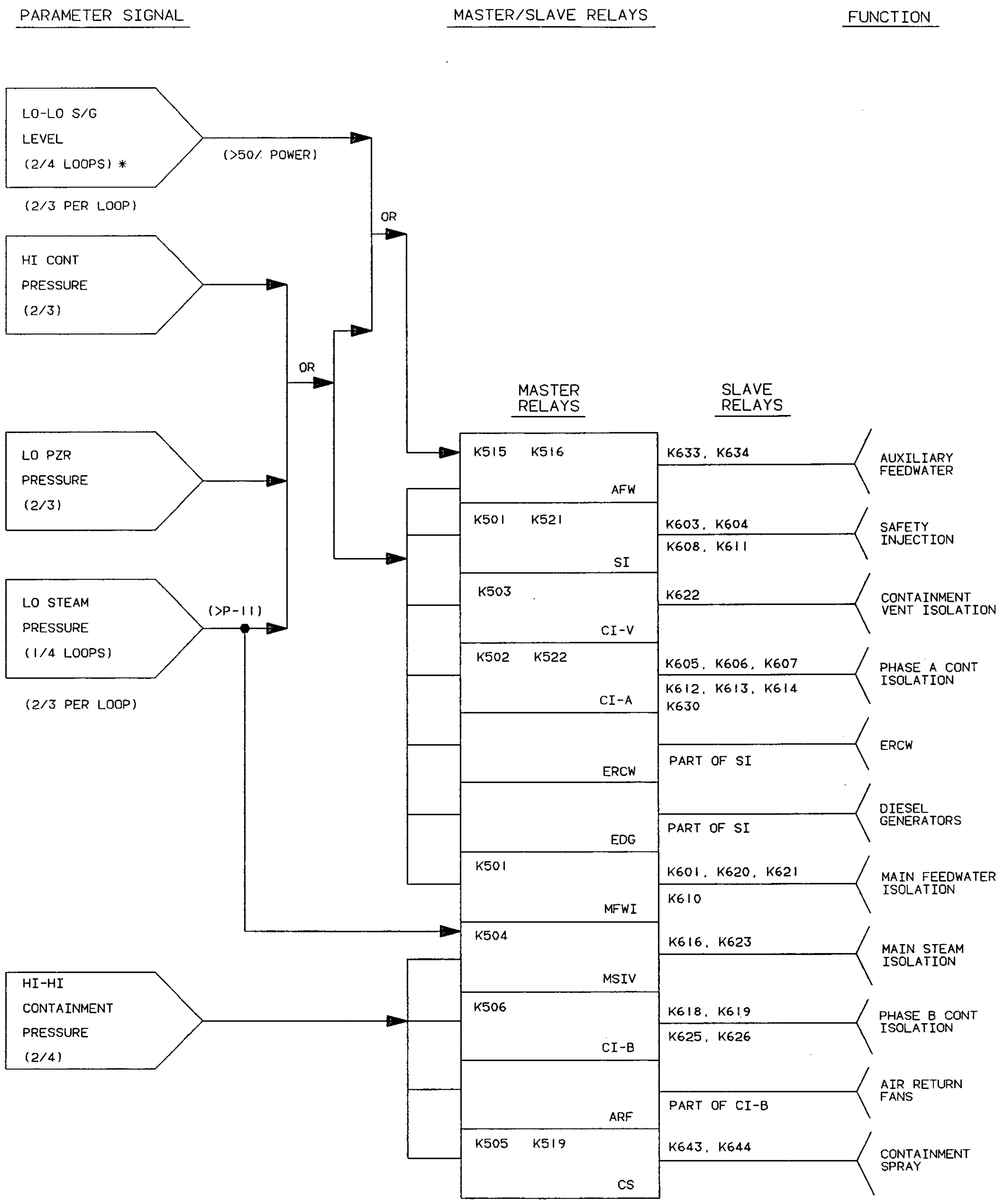
NOTES:
ONLY THE EQUIPMENT AND VALVES LYING WITHIN THE BOUNDARY LINES OF CONDENSATE AND FEEDWATER TOP EVENTS SHOWN IN THIS DIAGRAM.

REFERENCE DRAWINGS
1-47W803-1 R4
1-47W804-1 R0
1-47W804-2 R2

SECTION 3.2.1.7 — ENGINEERED SAFETY FEATURES ACTUATION SYSTEM FIGURES

EA00,ENG,CS251211

WBN/ESFAS/01/02.R00



* MOTOR DRIVEN AFW PUMPS ARE ONE OF FOUR FOR SUCCESS.

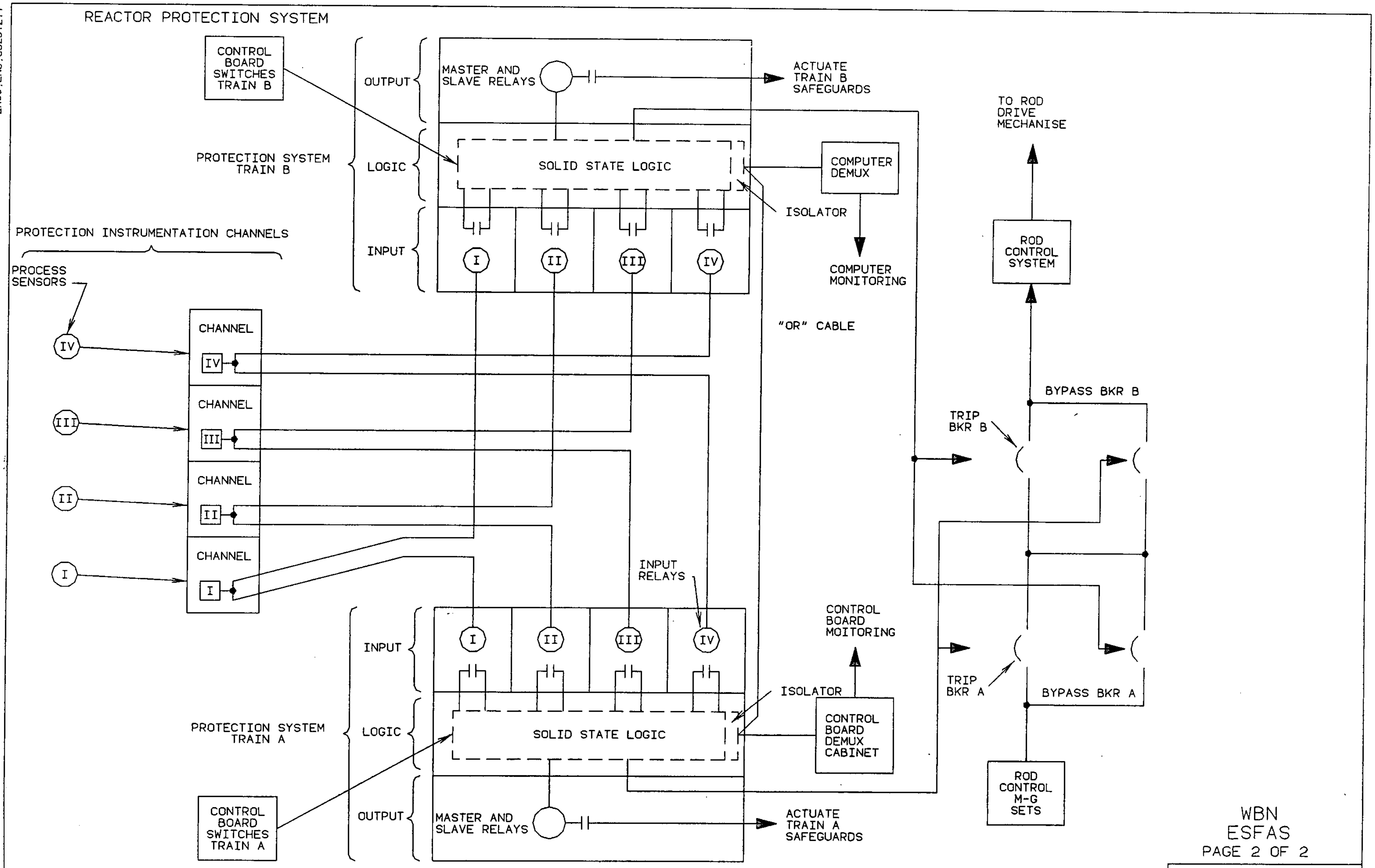
WBN
ESFAS
PAGE 1 OF 2

FIGURE 1
ESFAS SIGNAL SUCCESS DIAGRAM

REFERENCE DRAWINGS
1,2-47W611-99 SERIES VARIOUS REVS

WBN/ESFAS/02/02.R00

EA00.ENG.CS251211



WBN
ESFAS
PAGE 2 OF 2

FIGURE 2
ESFAS SIMPLIFIED DIAGRAM
FOR THE REACTOR PROTECTION SYSTEM

REFERENCE DRAWINGS

FIGURE 3.2-1 OF
TVA N3-99-4003 R2

SECTION 3.2.1.8 — REACTOR PROTECTION SYSTEM FIGURES

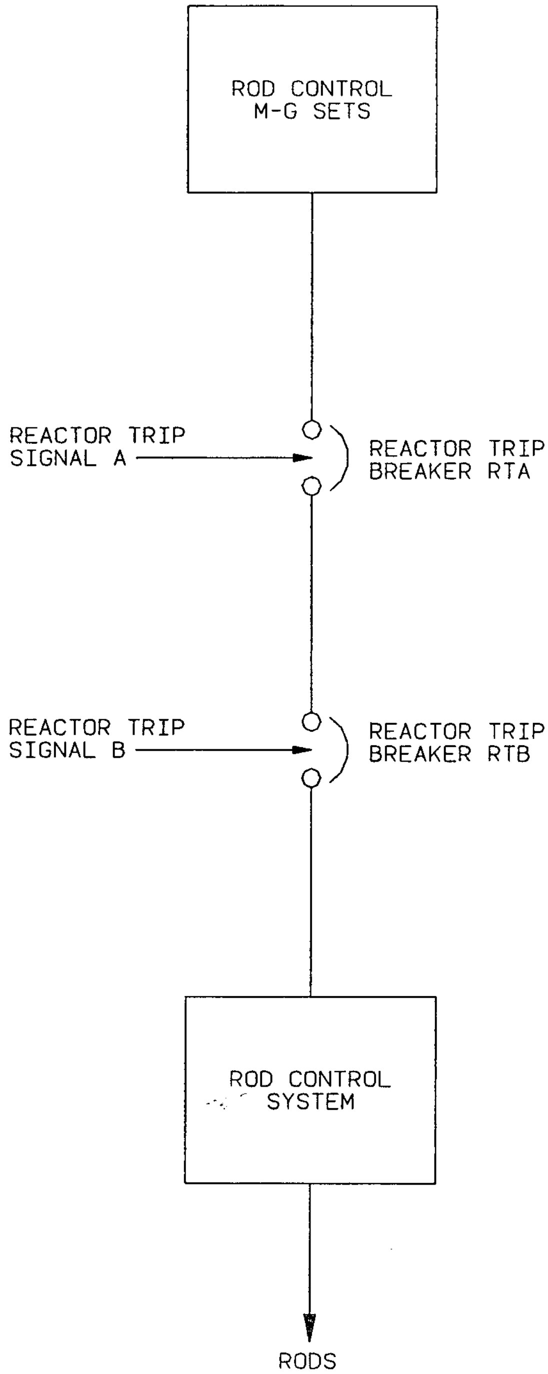


FIGURE 2-1
SIMPLIFIED DRAWING REACTOR TRIP TOP EVENT RT

WBN
RPS
PAGE 1 OF 2

REFERENCE DRAWINGS:
1-45W600-99-1 R0

WBN/RPS/02/02,R00

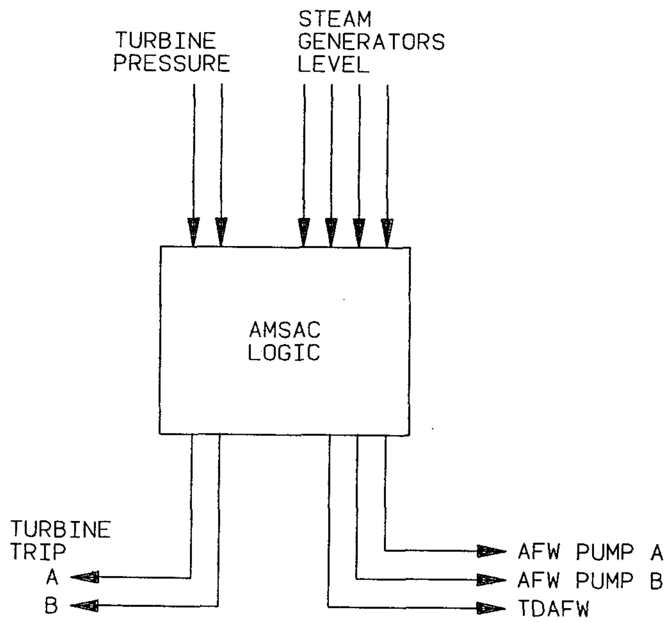


FIGURE 2-2
SIMPLIFIED DRAWING AMSAC TOP EVENT AM

WBN
RPS

REFERENCE DRAWINGS:
ECN E110007

SECTION 3.2.1.9 — AUXILIARY FEEDWATER SYSTEM FIGURES

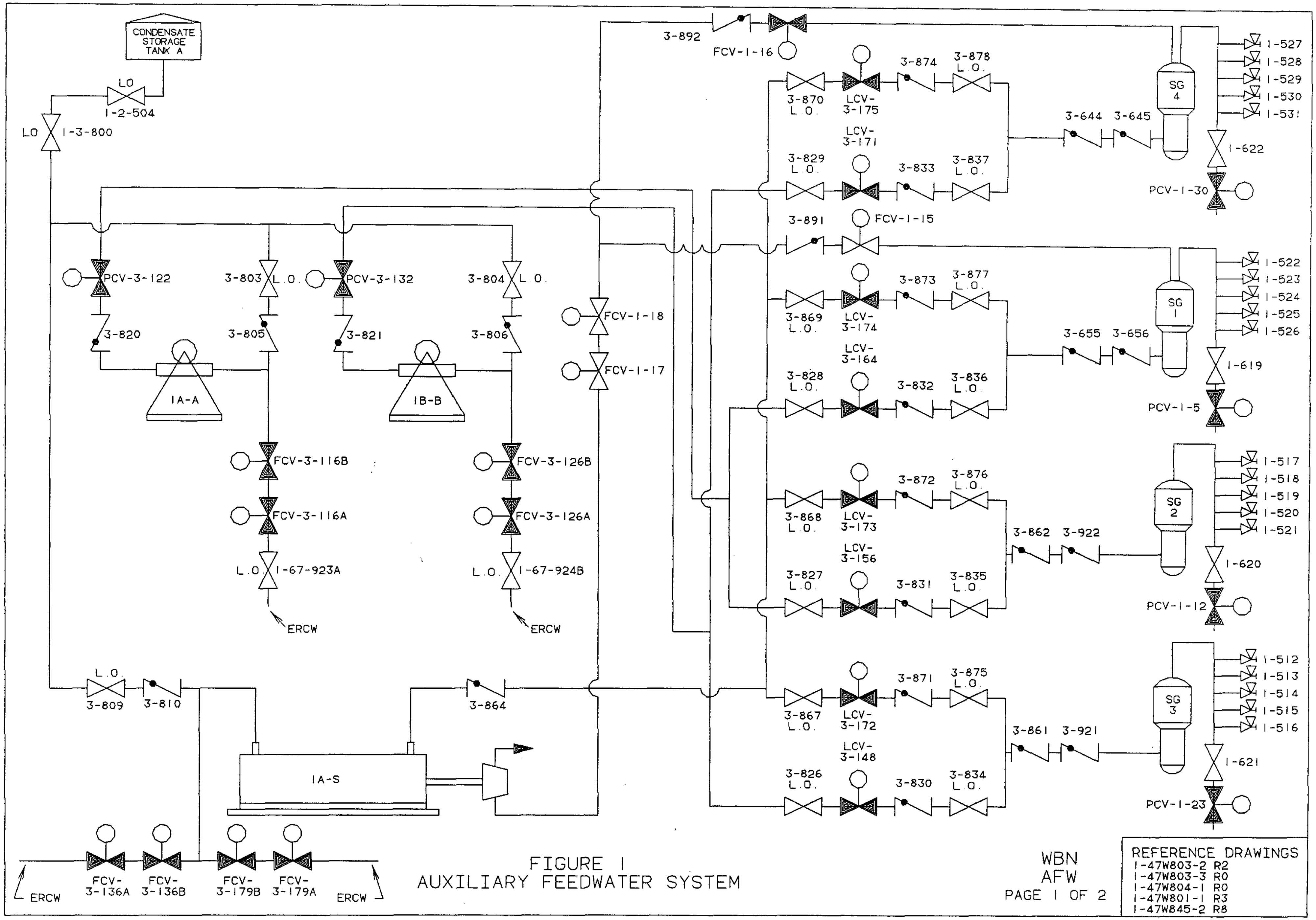
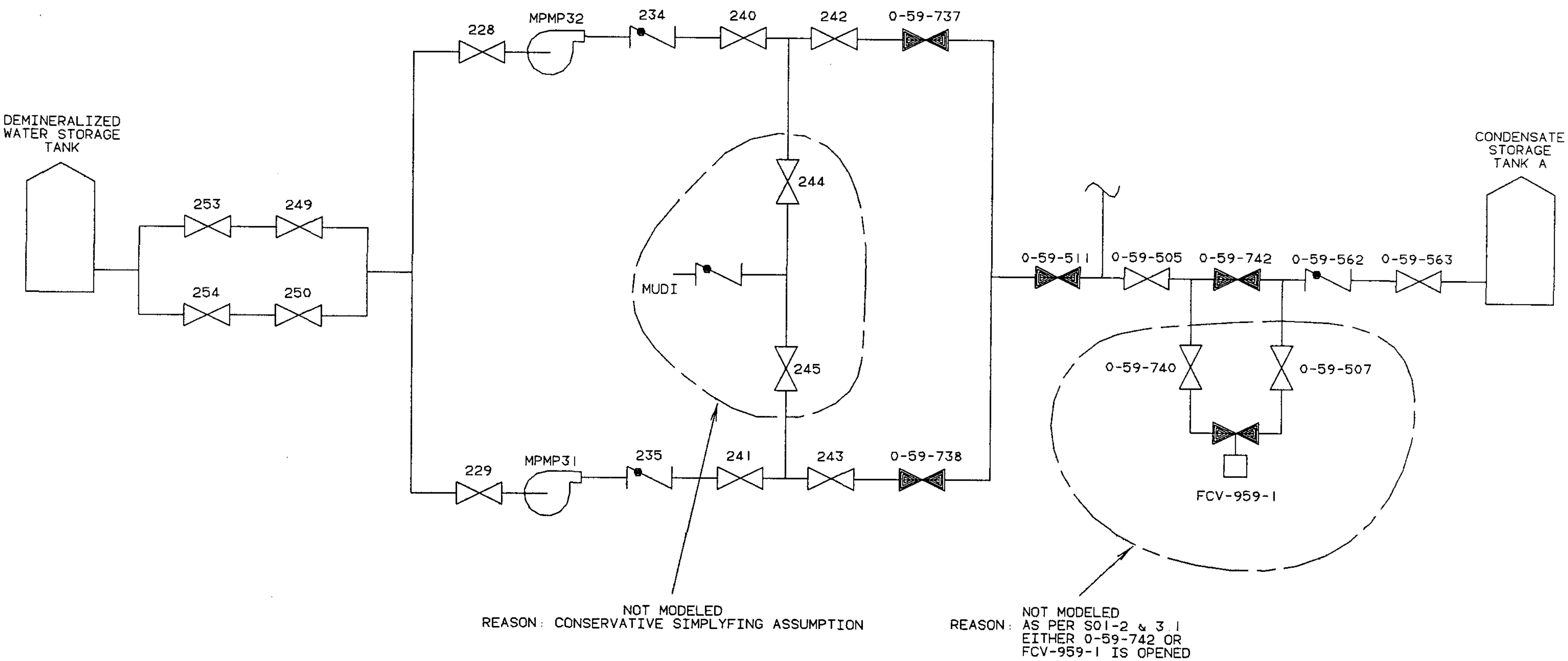


FIGURE 1
AUXILIARY FEEDWATER SYSTEM



WBN
AFW
PAGE 2 OF 2

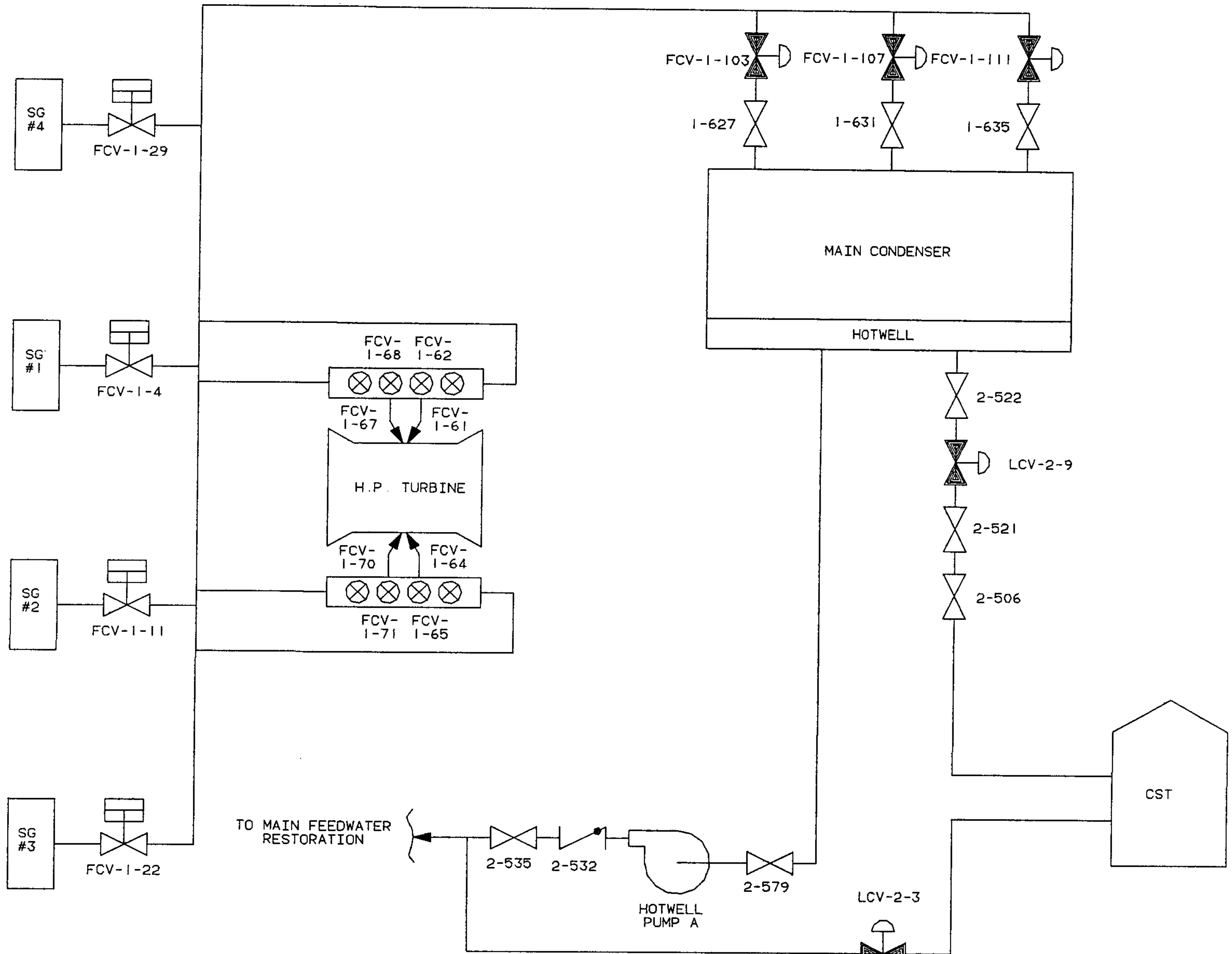
FIGURE 2
AUXILIARY FEEDWATER SYSTEM

REFERENCE DRAWINGS
1-47W804-1 R0
1-47W856-1 R1
1-47W856-2 R1
1-47E872-1 R6

SECTION 3.2.1.10 — MAIN STEAM SYSTEM FIGURES

WBN/MS/01/01,R00

EA00,ENG,CS251211



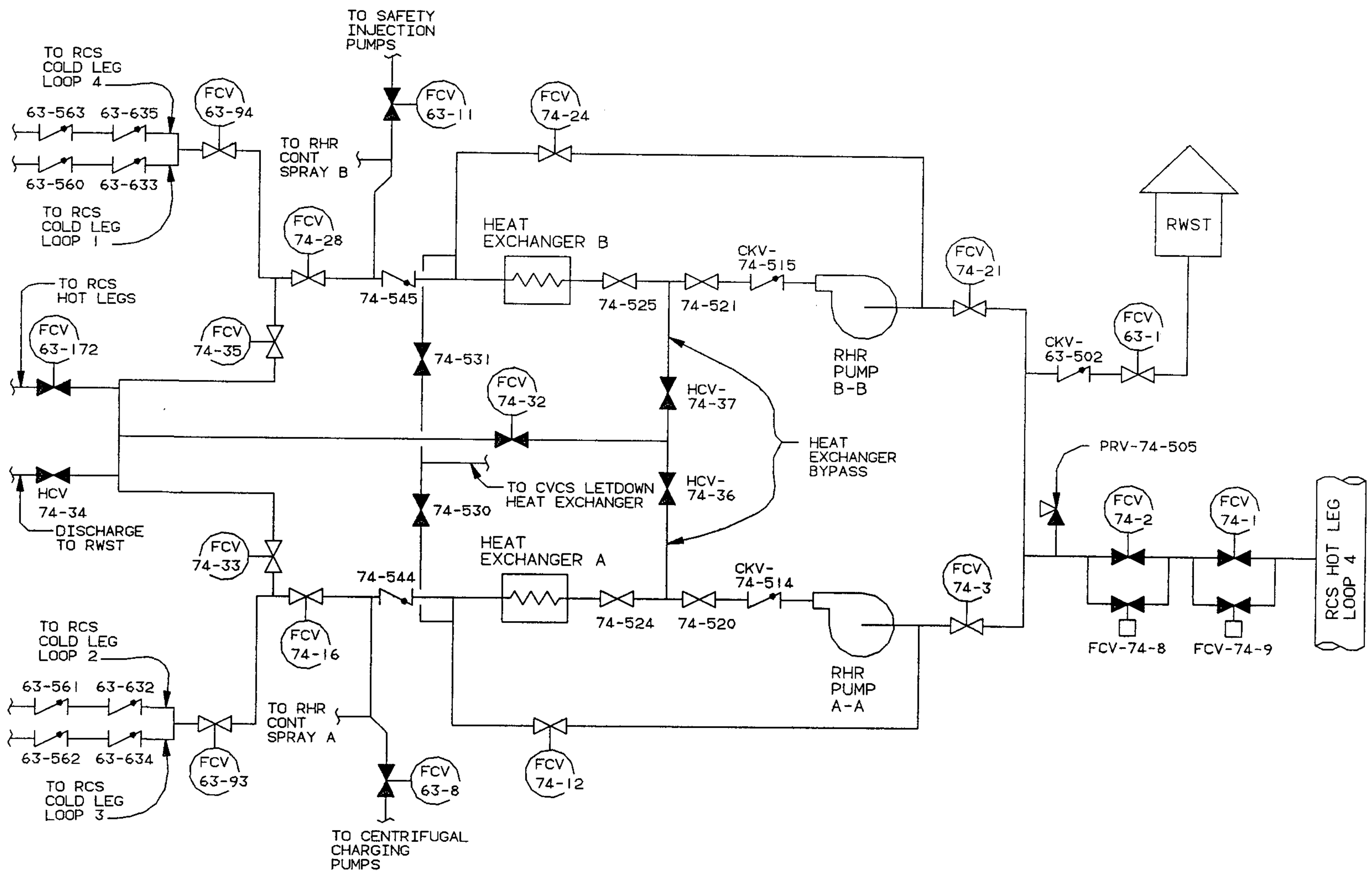
WBN
MS
PAGE 1 OF 1

FIGURE 1
MAIN STEAM SYSTEM
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS
1,2-47W801-1 R3
1,2-47W804-1 R0

SECTION 3.2.1.11 — RESIDUAL HEAT REMOVAL SYSTEM FIGURES

EA00,ENG,CS251211



WBN
RHR
PAGE 1 OF 5

FIGURE 1
RHR STANDBY ALIGNMENT

REFERENCE DRAWINGS:

1-47W810-1 R3
1-47W811-1 R3

EA00.ENG.CS251211

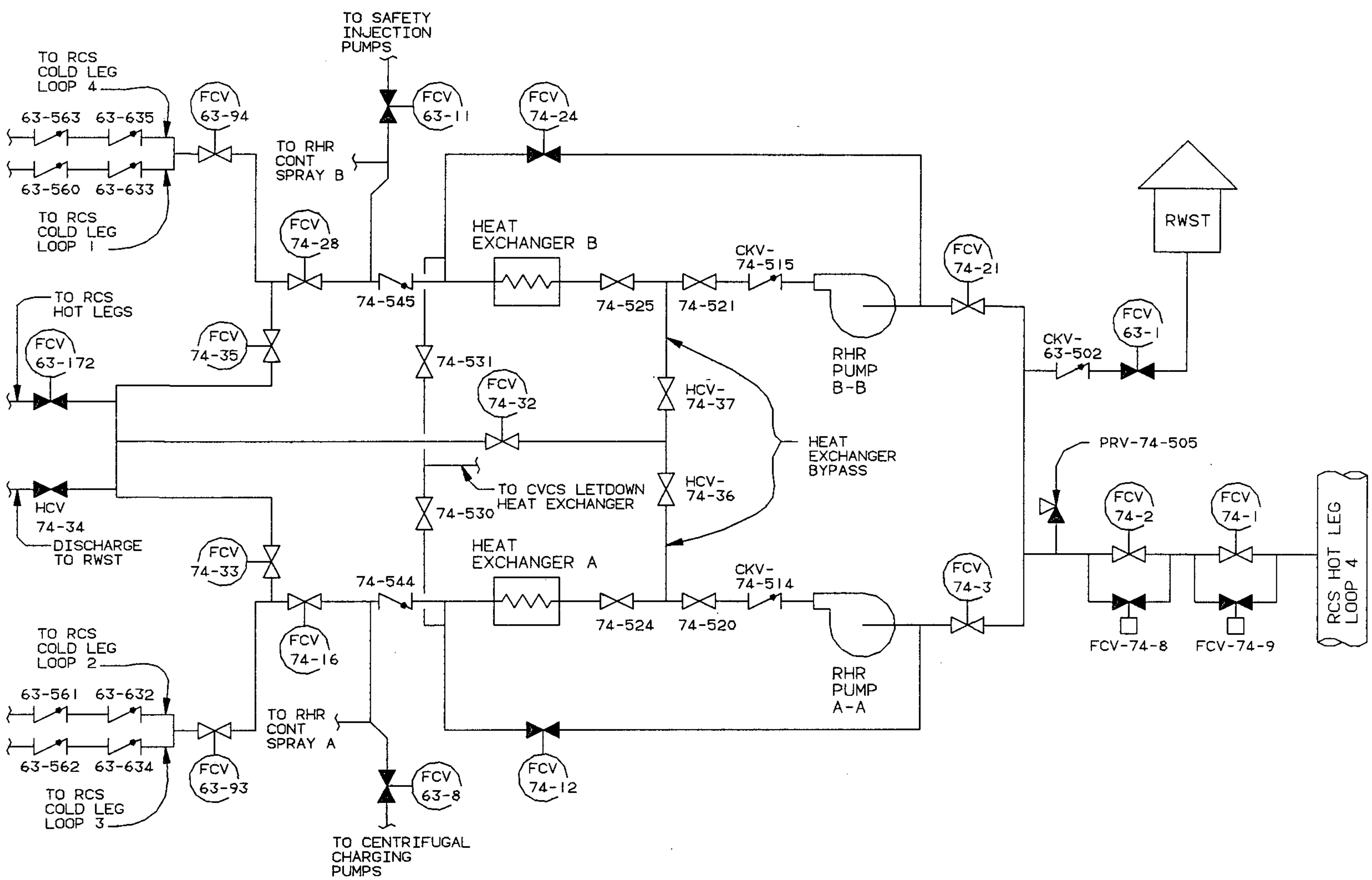


FIGURE 2
RHR ALIGNMENT FOR NORMAL COOLDOWN

WBN
RHR

REFERENCE DRAWINGS:

1-47W810-1 R3
1-47W811-1 R3

WBN/RHR/03/05,R00

EA00.ENG.CS251211

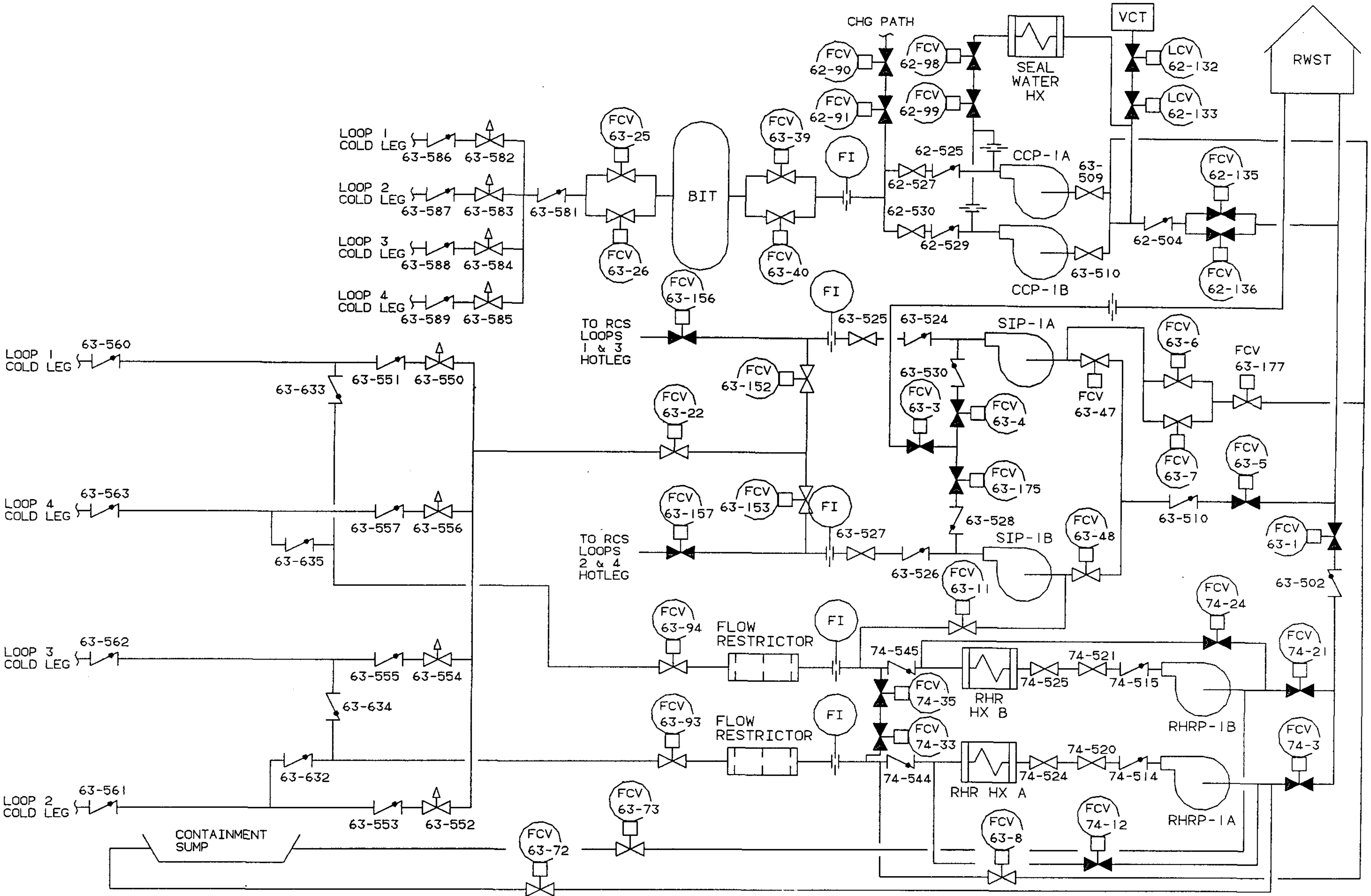


FIGURE 3
COLD LEG RECIRCULATION MODE

WBN
RHR
PAGE 3 OF 5

REFERENCE DRAWINGS:

1-47W810-1 R3
1-47W811-1 R3

WBN/RHR/04/05.R00

EA00.ENG.CS251211

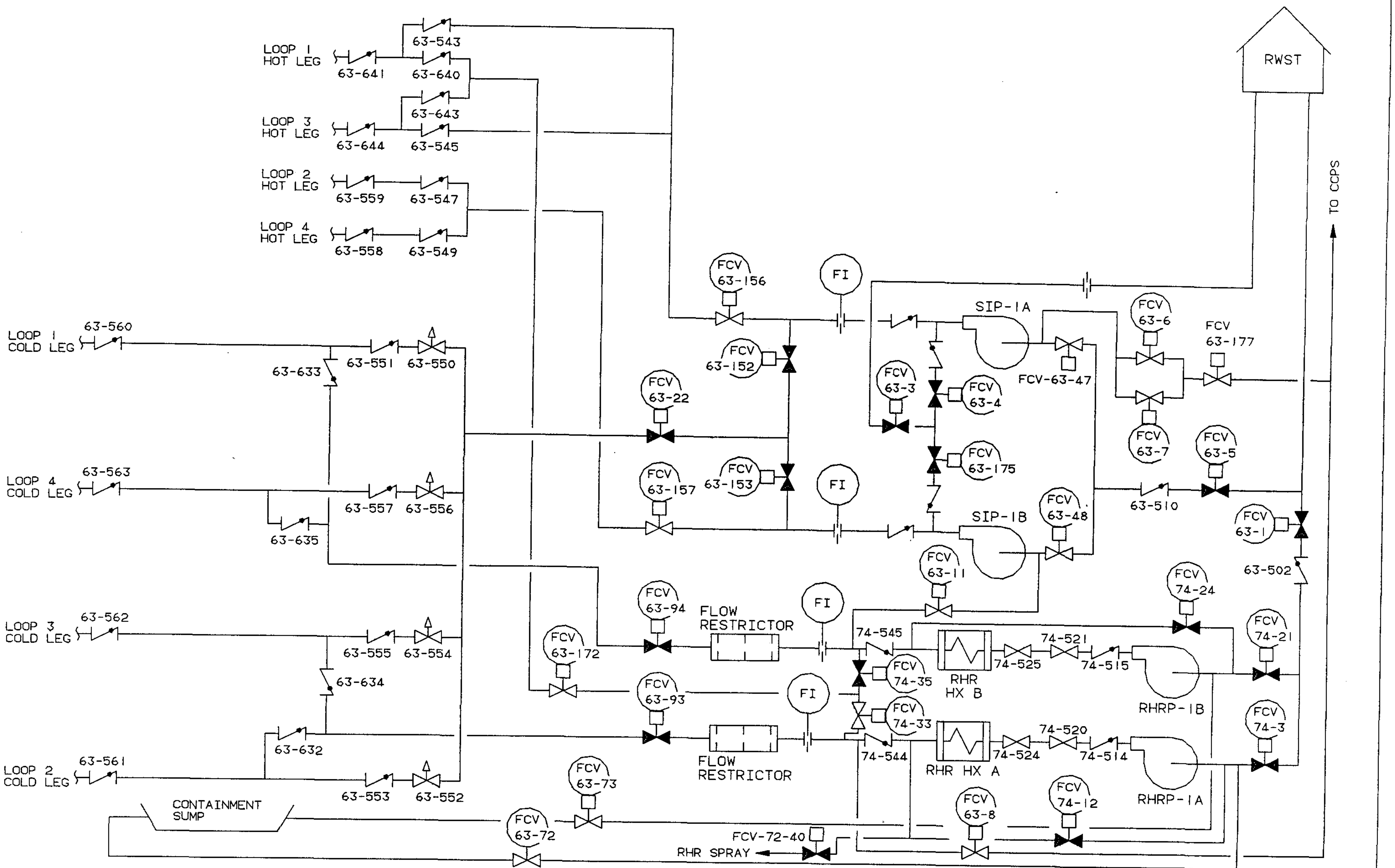


FIGURE 4
HOT LEG RECIRCULATION

WBN
RHR
PAGE 4 OF 5

REFERENCE DRAWINGS:
1-47W810-1 R3
1-47W811-1 R3

EA00,ENG,CS251211

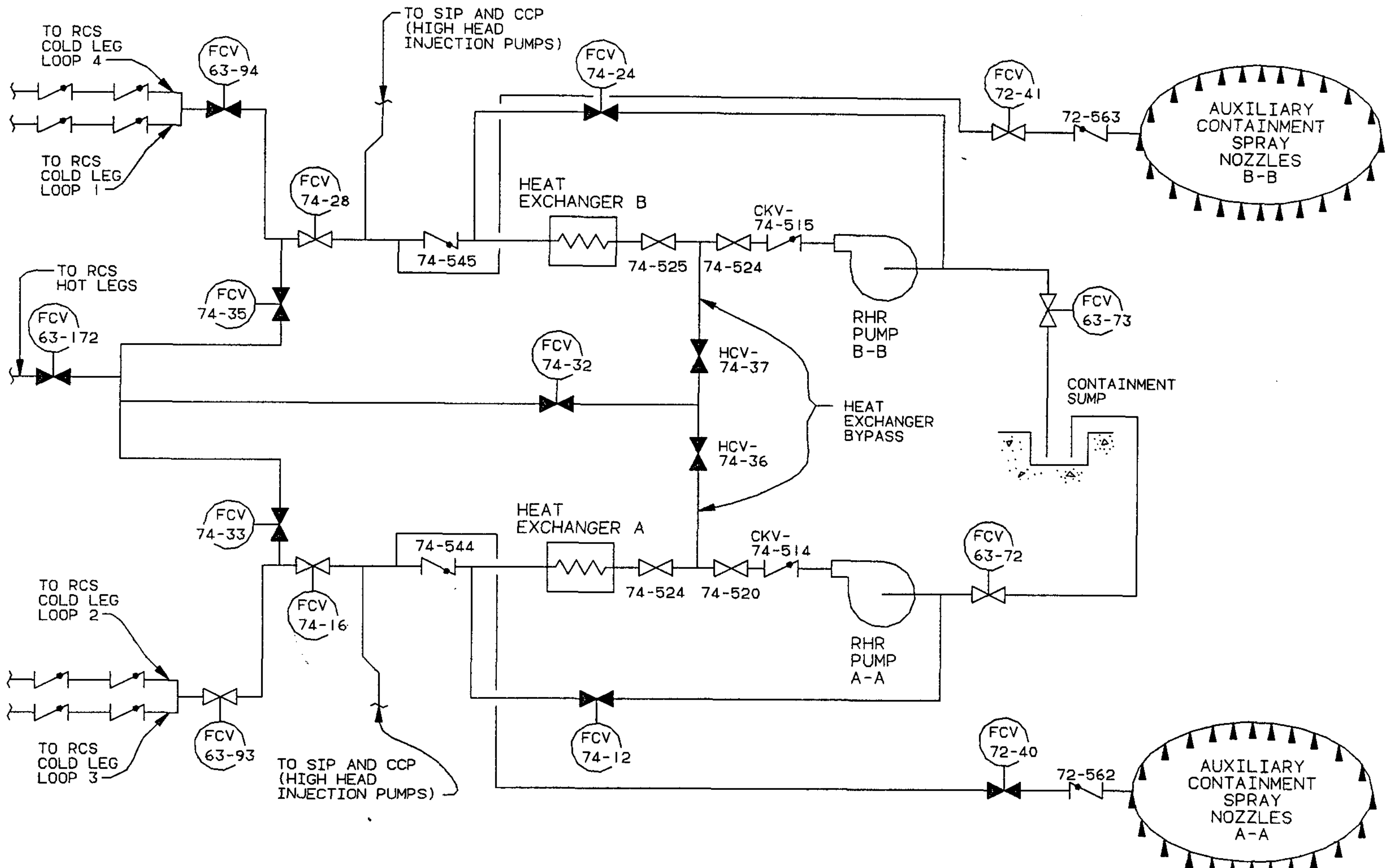


FIGURE 5
RHR CONTAINMENT SPRAY

WBN
RHR
PAGE 5 OF 5

REFERENCE DRAWINGS:

1-47W810-1 R3
1-47W811-1 R3

SECTION 3.2.1.12 — SAFETY INJECTION SYSTEM FIGURE

WBN/SI/01/01.R00

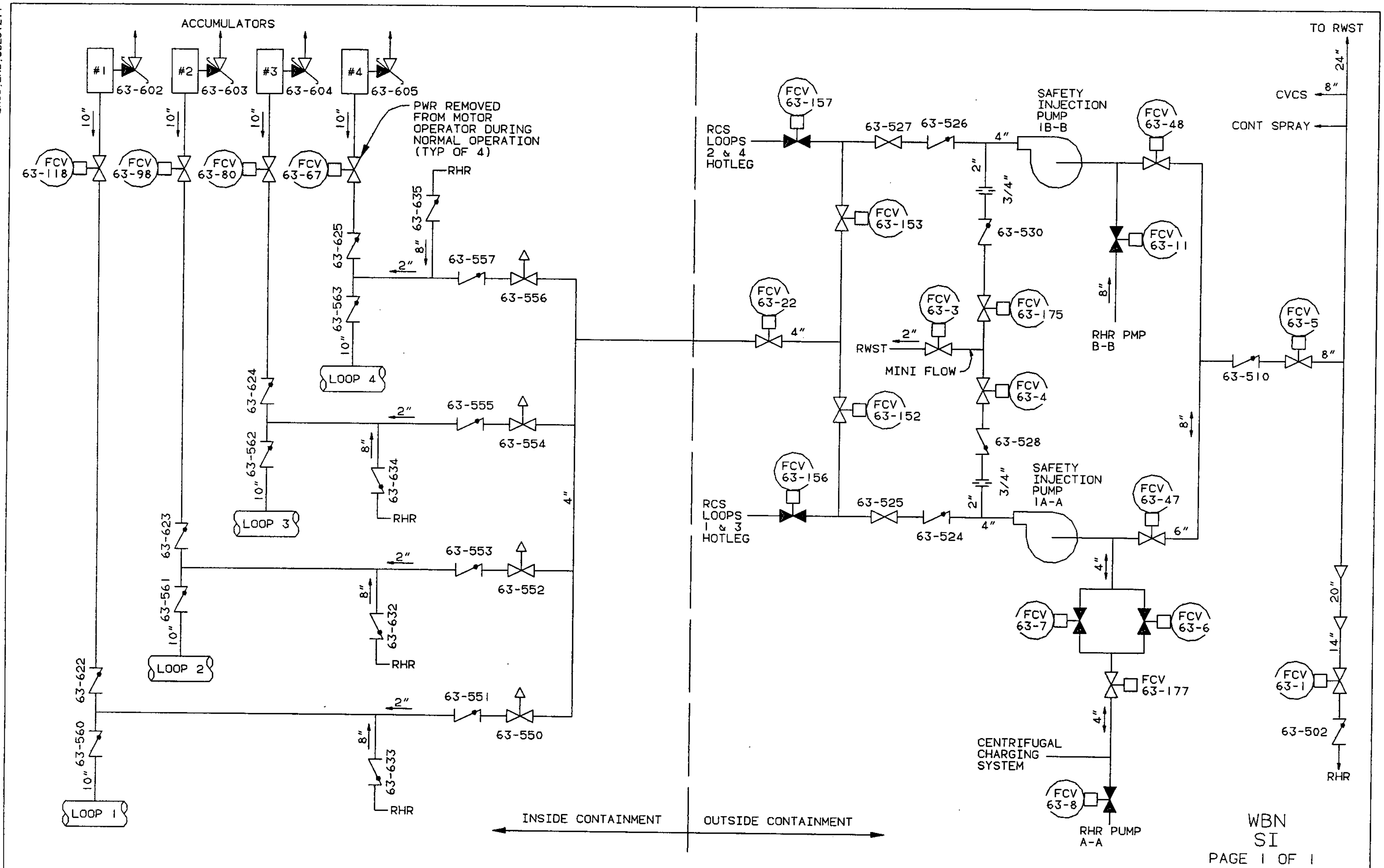
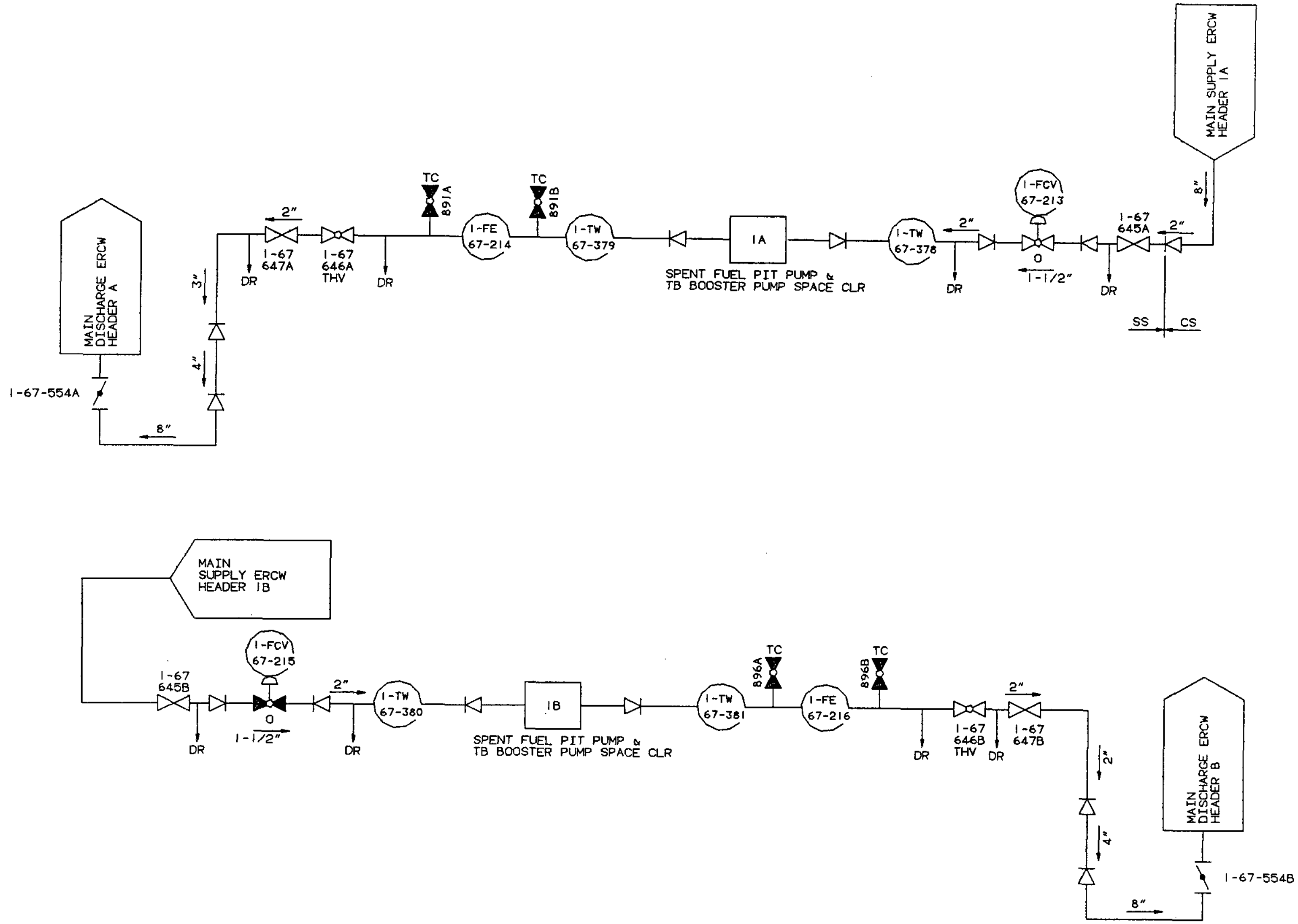


FIGURE 1
SAFETY INJECTION SYSTEM
"COLD LEG" INJECTION MODE

SECTION 3.2.1.13 — RCP SEAL INJECTION AND THERMAL BARRIER COOLING FIGURES

EA00.ENG.CS251211



WBN
RCP
PAGE 5 OF 5

REFERENCE DWGS:
1-47W845-4 R5

FIGURE 5
THERMAL BARRIER BOOSTER PUMP SPACE COOLER SIMPLIFIED P&ID

WBN/RCP/01/05,R00

EA00,ENG,CS251211

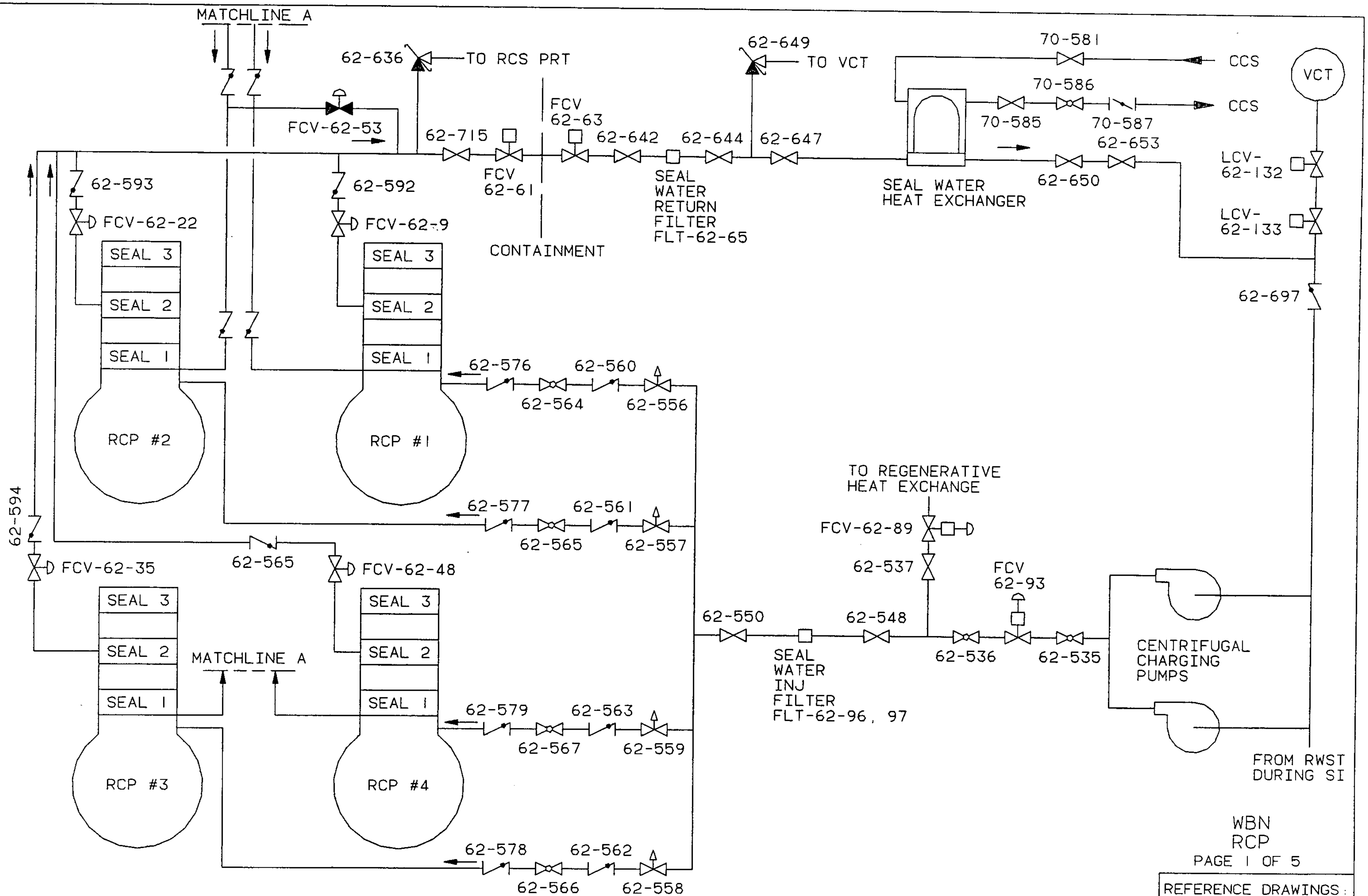
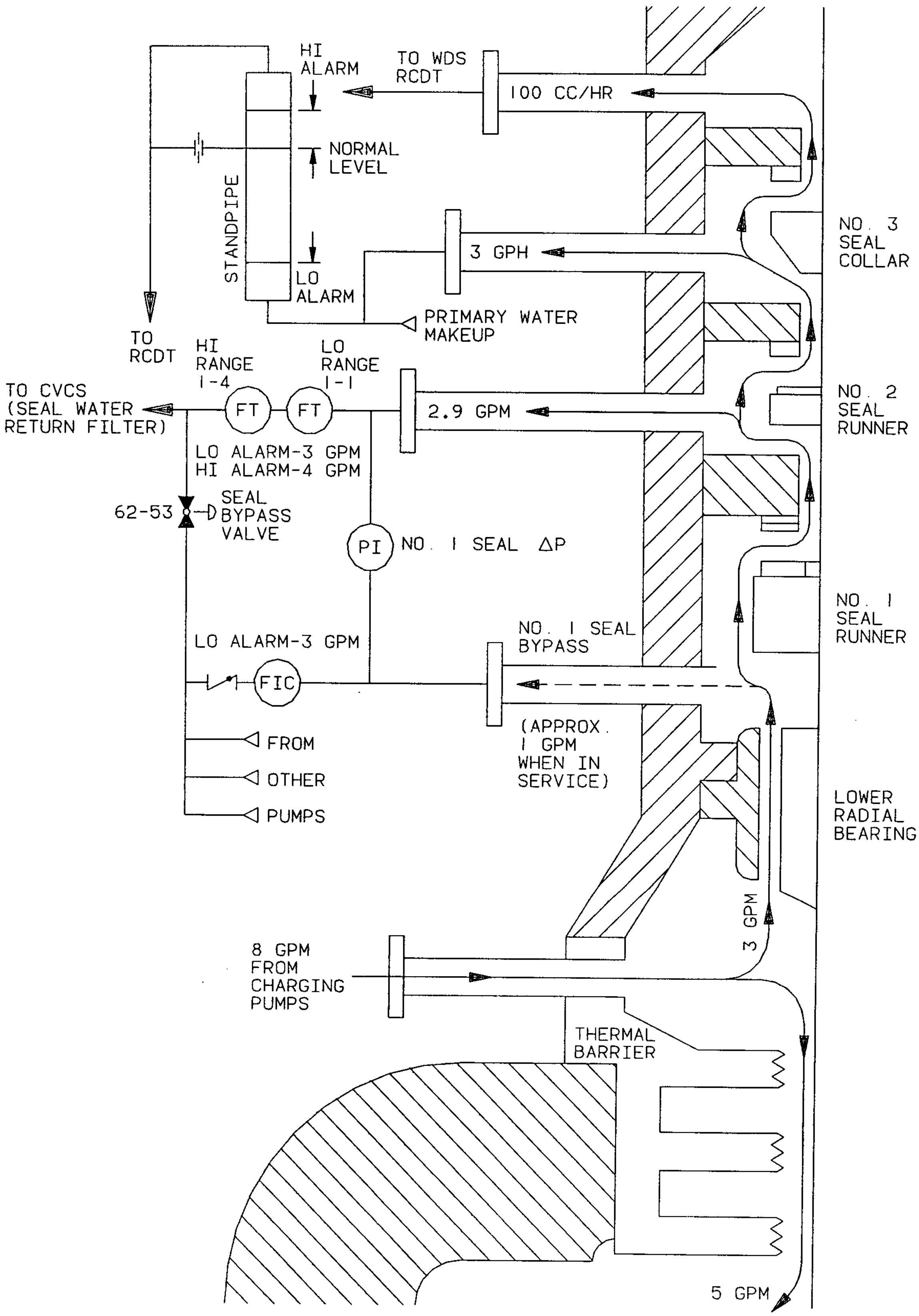


FIGURE 1
RCP SEAL COOLING SIMPLIFIED P&ID

WBN
RCP
PAGE 1 OF 5

REFERENCE DRAWINGS:
1-47W809-1 R4
1-47W859-2 R7

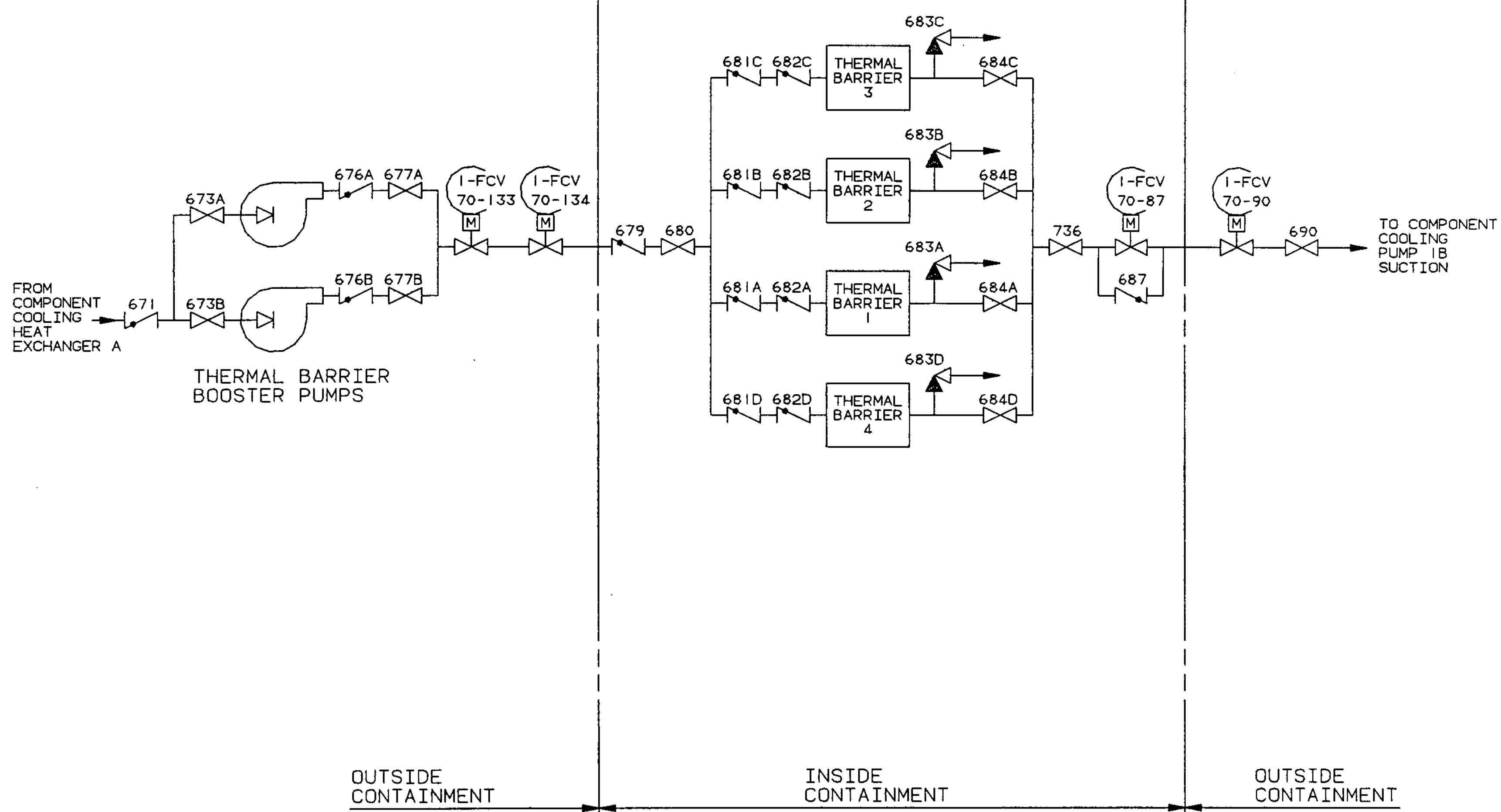


REFERENCE DRAWINGS:
MI-68.04 R9
APPENDIX A

FIGURE 2
RCP SHAFT SEAL INJECTION PATH

NOTES:

1. ONLY ONE BOOSTER PUMP IS NORMALLY RUNNING.
2. NORMAL VALVE ALLIGNMENT SHOWN.



WBN
RCP
PAGE 3 OF 5

REFERENCE DRAWINGS:
1-47W859-2 R7

FIGURE 3
THERMAL BARRIER COOLING SIMPLIFIED P&ID

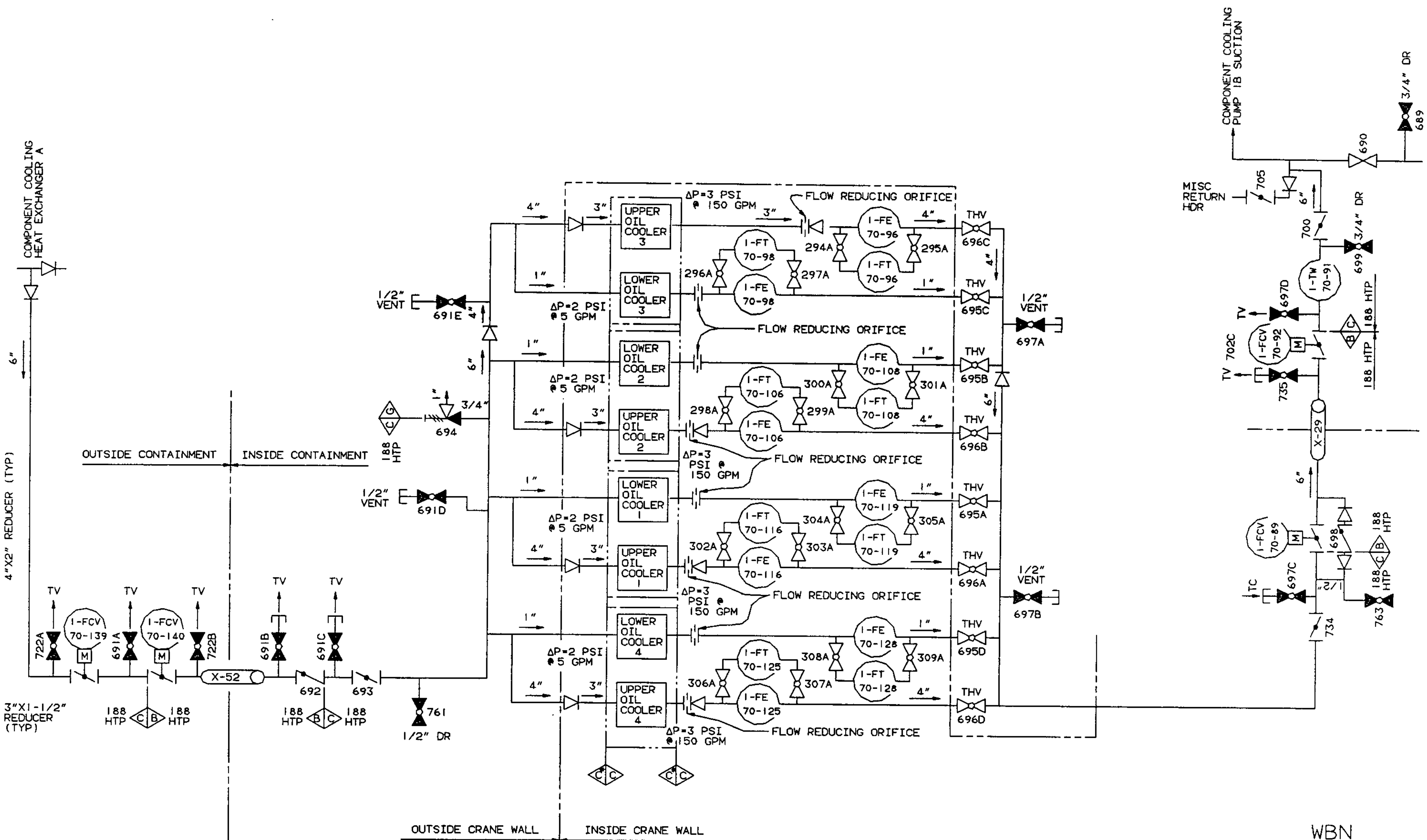
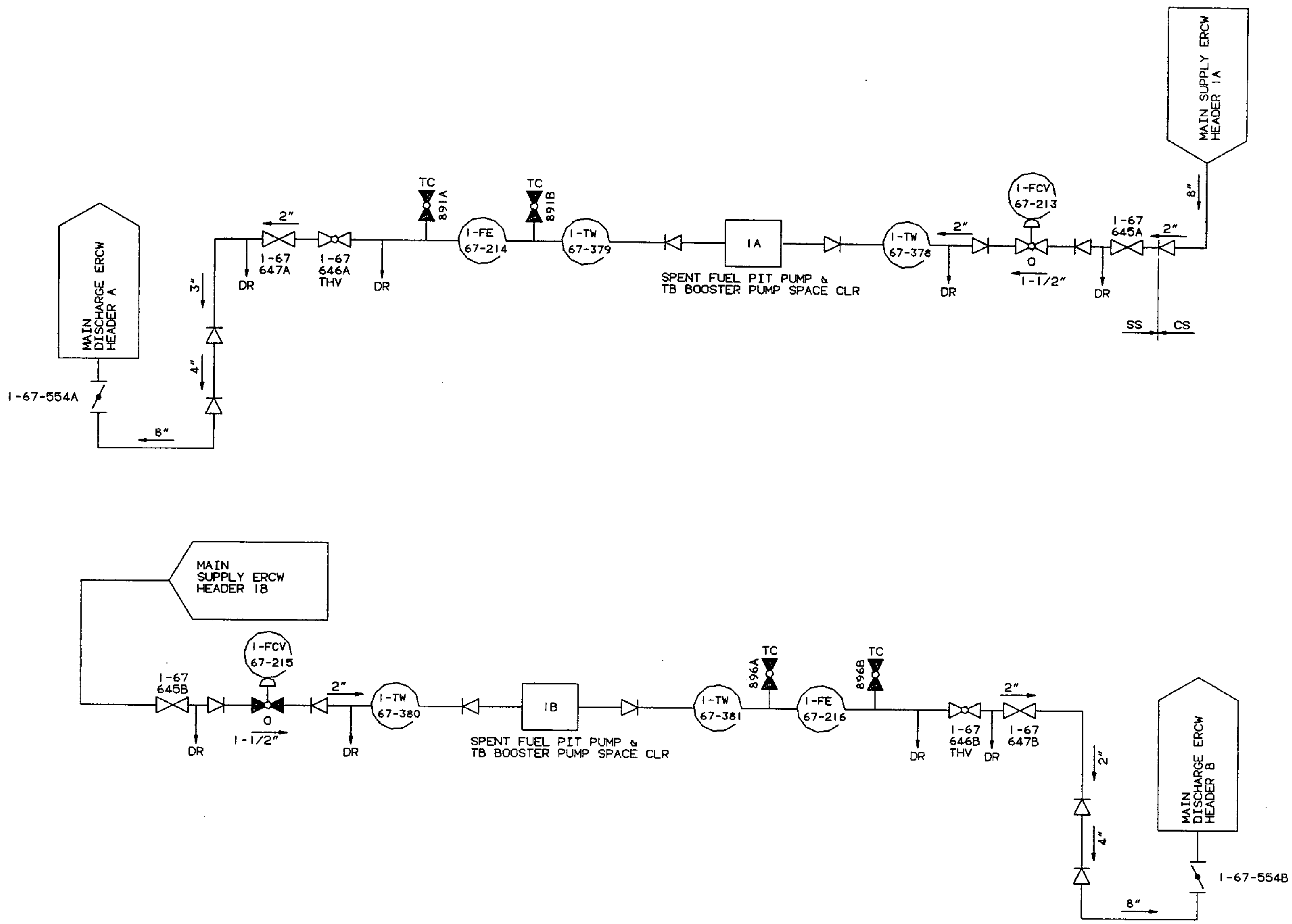


FIGURE 4
REACTOR COOLANT PUMP
BEARING OIL COOLERS

REFERENCE DWGS:

1-47W859-2 R7

EA00,ENG,CS251211



WBN
RCP
PAGE 5 OF 5

REFERENCE DWGS:
1-47W845-4 R5

FIGURE 5
THERMAL BARRIER BOOSTER PUMP SPACE COOLER SIMPLIFIED P&ID

SECTION 3.2.1.14 — PRESSURIZER POWER-OPERATED RELIEF VALVES AND SAFETY VALVES FIGURE

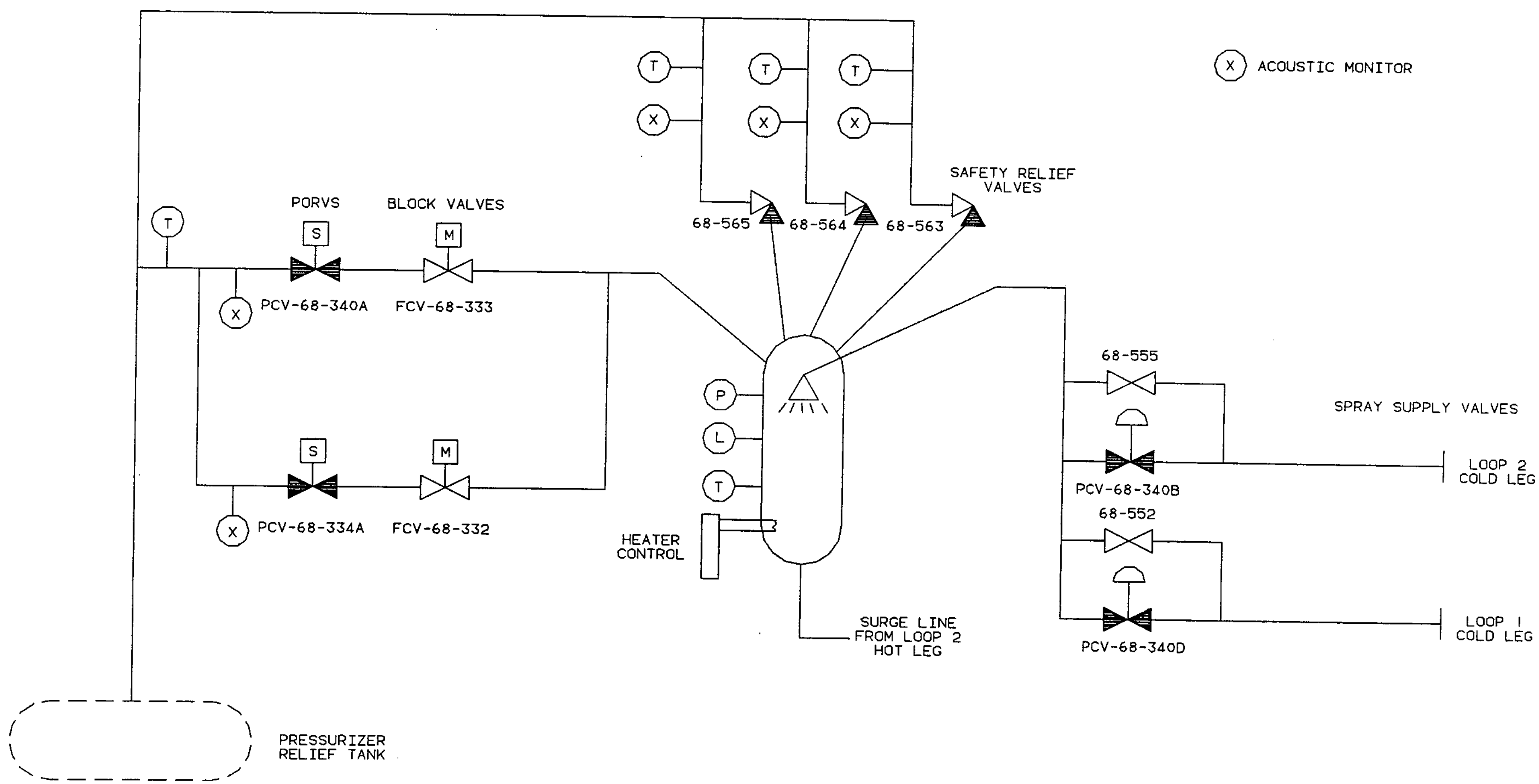


FIGURE 1
PRESSURIZER PORV'S, SV'S
SIMPLIFIED FLOW DIAGRAM

WBN
PORV
PAGE 1 OF 1

REFERENCE DRAWINGS

47W813-1 R35

SECTION 3.2.1.15 — STEAM GENERATOR ISOLATION FIGURE

WBN/SGI/01/01.R00

EA00.ENG.CS251211

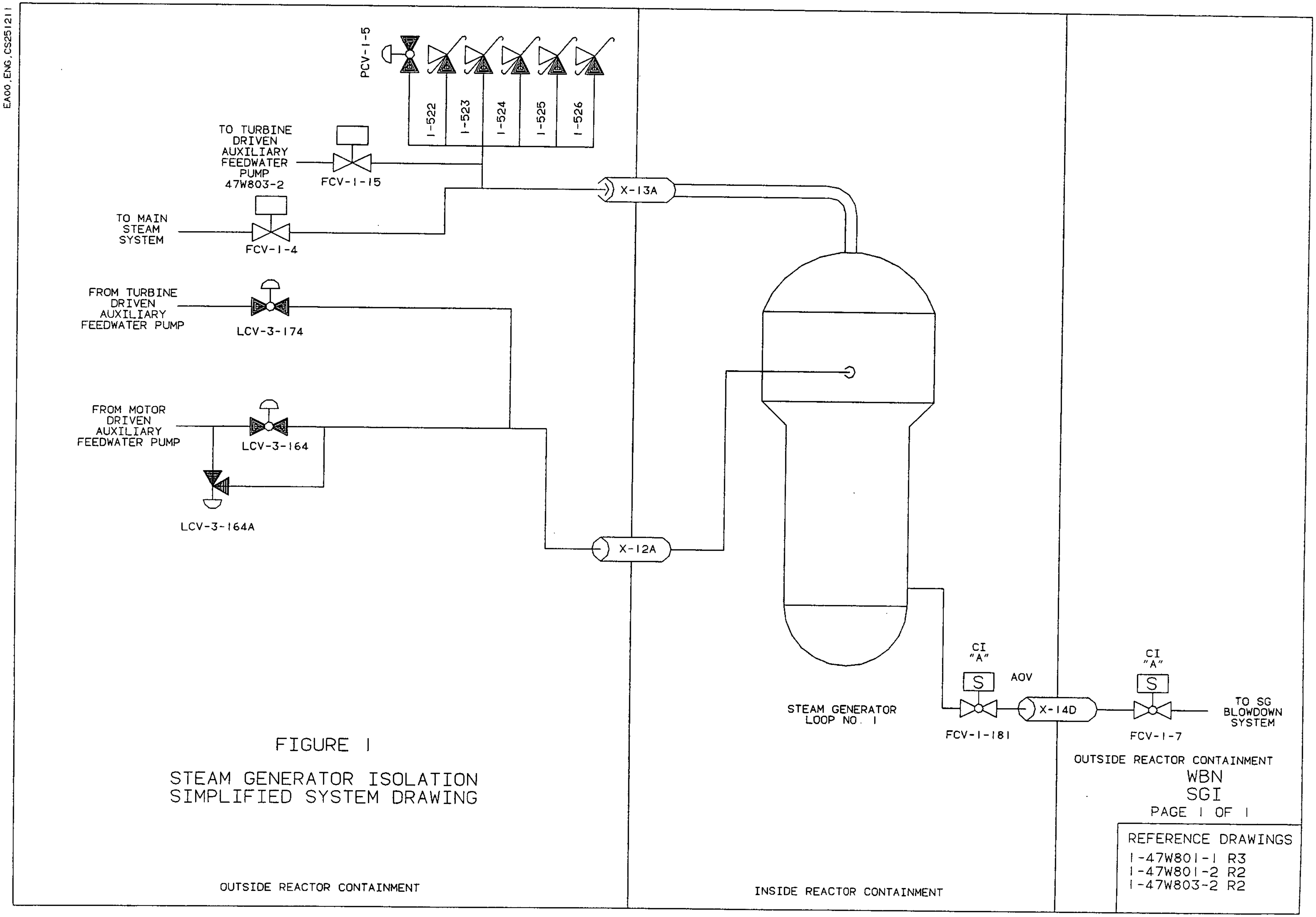


FIGURE 1

STEAM GENERATOR ISOLATION SIMPLIFIED SYSTEM DRAWING

SECTION 3.2.1.16 — CONTAINMENT SPRAY SYSTEM FIGURES

WBN/CTSP/01/02.R00

EA00.ENG.CS251211

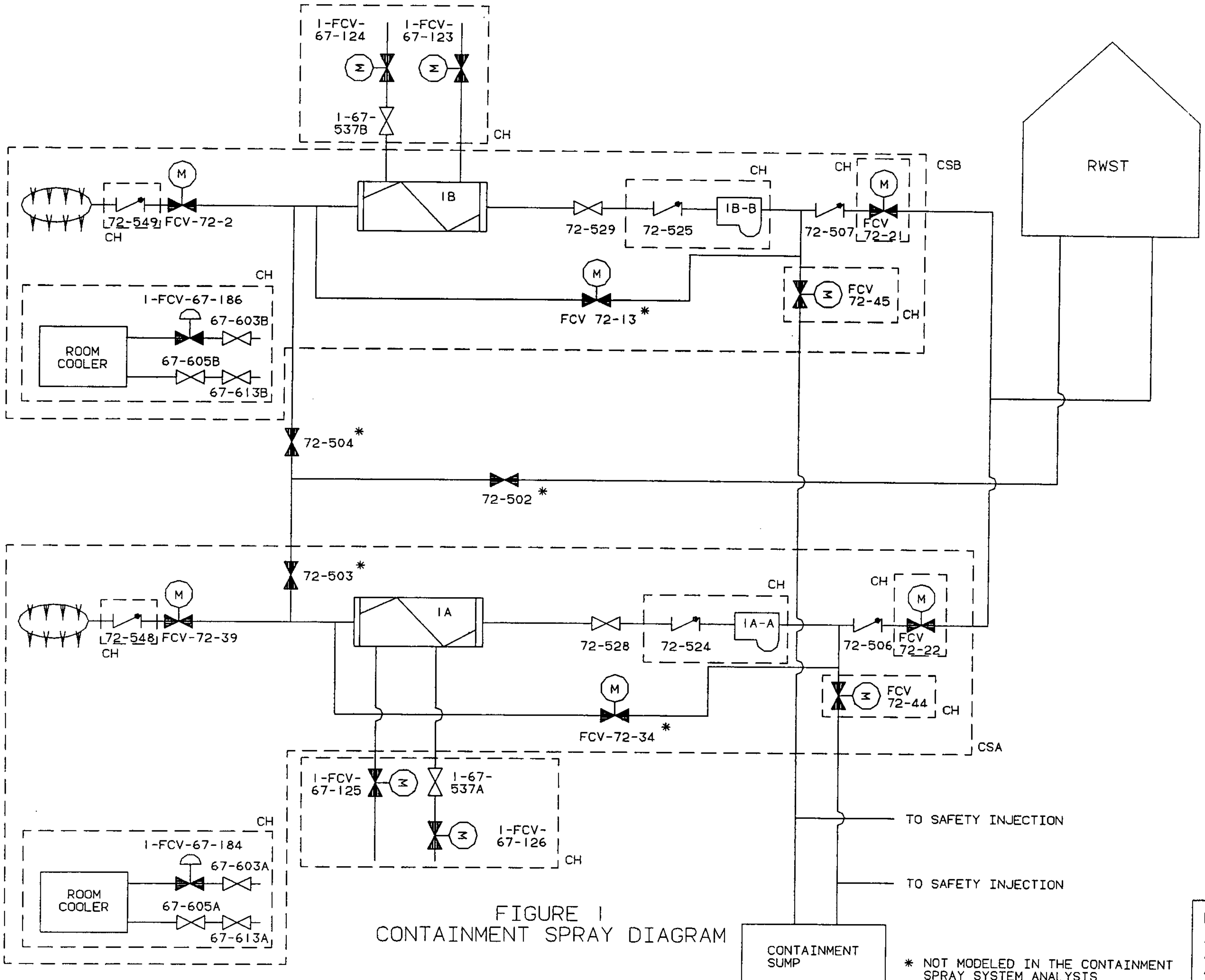


FIGURE 1
CONTAINMENT SPRAY DIAGRAM

* NOT MODELED IN THE CONTAINMENT SPRAY SYSTEM ANALYSIS

WBN
CTSP
PAGE 1 OF 2

REFERENCE DRAWINGS
47W845-2 R8
47W845-4 R5
47W812-1 R1

EA00,ENG,CS251211

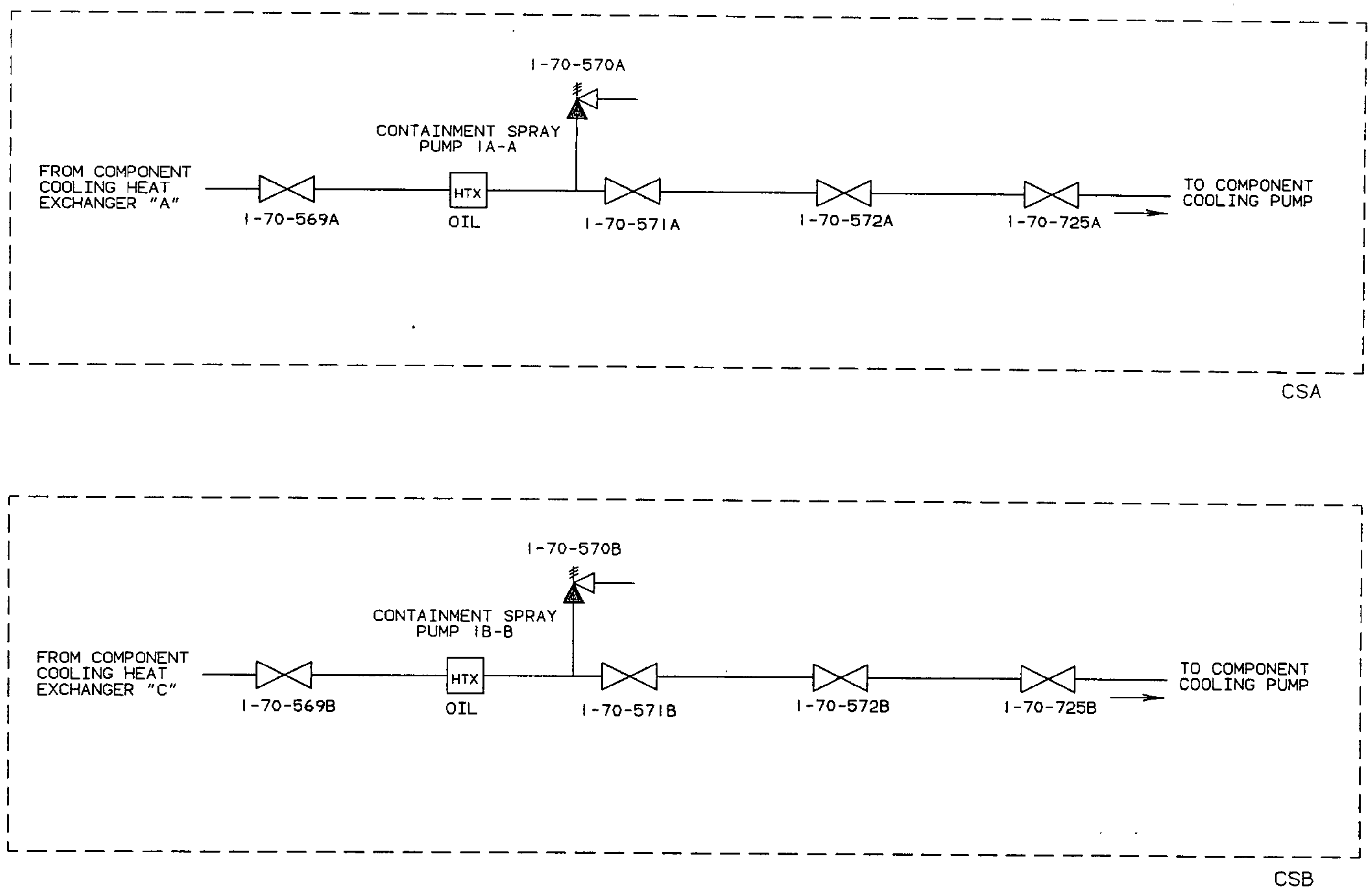


FIGURE 2
CONTAINMENT SPRAY DIAGRAM

WBN
CTSP
PAGE 2 OF 2

REFERENCE DRAWINGS
1-47W859-4 R2

SECTION 3.2.1.17 — CONTAINMENT SYSTEMS FIGURES

WBN/CTSY/01/06,R00

EA00,ENG,CS251211

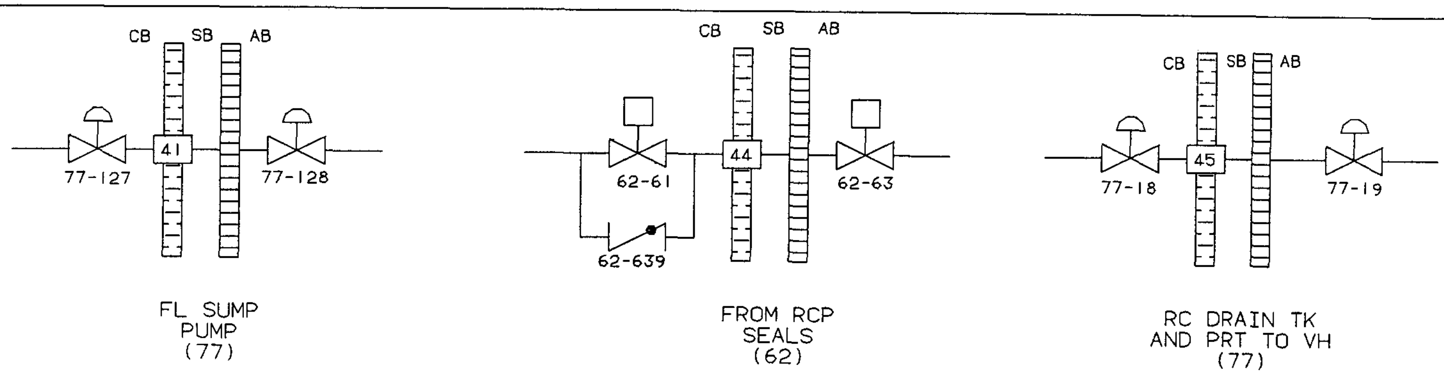


FIGURE 1 (SHEET 1 OF 3)
CONTAINMENT SYSTEMS TOP EVENT CI
SIMPLIFIED DIAGRAM

WBN
CTSY
PAGE 1 OF 6

REFERENCE
TABLE 6.2.4-1
FSAR
AMENDMENT 68

WBN/CTSY/02/06.R00

EA00.ENG.CS251211

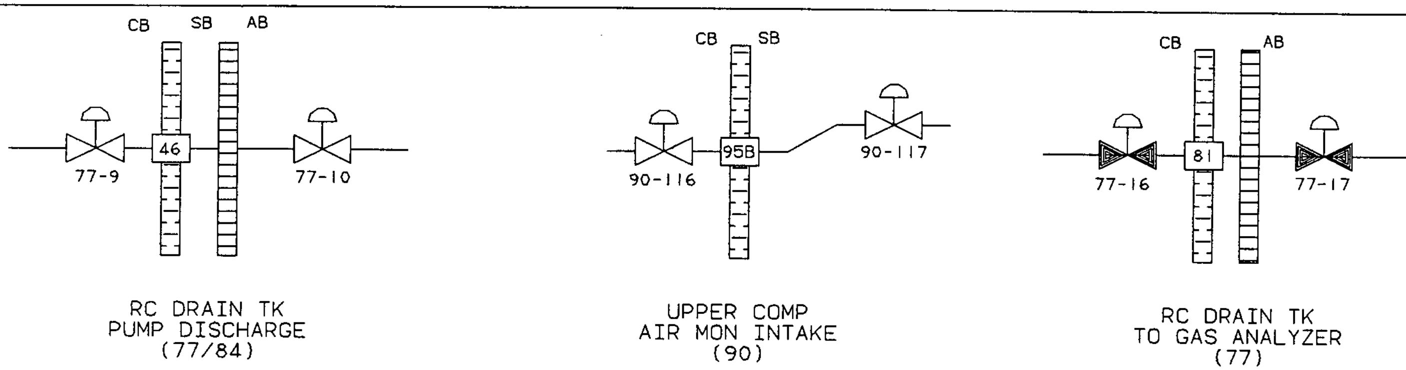


FIGURE 1 (SHEET 2 OF 3)
CONTAINMENT SYSTEMS TOP EVENT CI
SIMPLIFIED DIAGRAM

WBN
CTSY
PAGE 2 OF 6

REFERENCE
TABLE 6.2.4-1
FSAR
AMENDMENT 68

WBN/CTSY/03/06.R00

EA00.ENG.CS251211

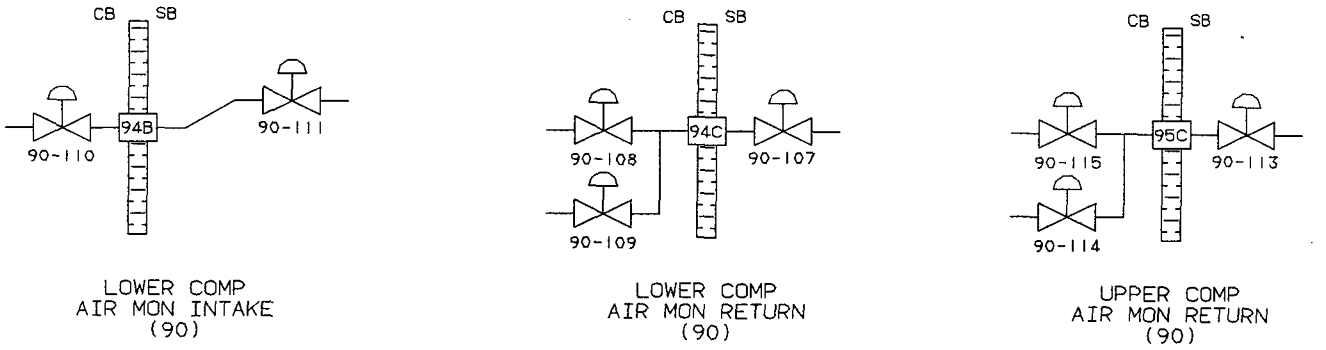
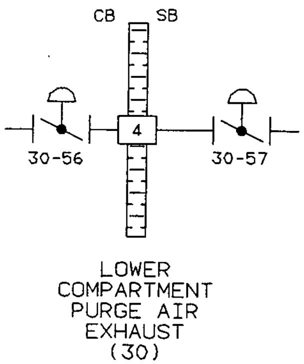


FIGURE 1 (SHEET 3 OF 3)
CONTAINMENT SYSTEMS TOP EVENT CI
SIMPLIFIED DIAGRAM

WBN
CTSY
PAGE 3 OF 6

REFERENCE

TABLE 6.2.4-1
FSAR
AMENDMENT 68



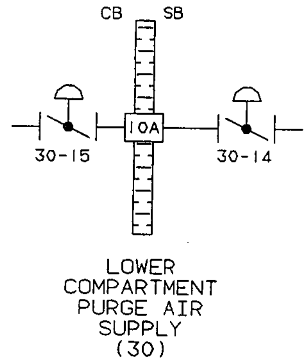


FIGURE 2
CONTAINMENT SYSTEMS TOP EVENT CP
SIMPLIFIED DIAGRAM

WBN
CTSY

REFERENCE

TABLE 6.2.4-1
FSAR
AMENDMENT 68

EA00.ENG.CS251211

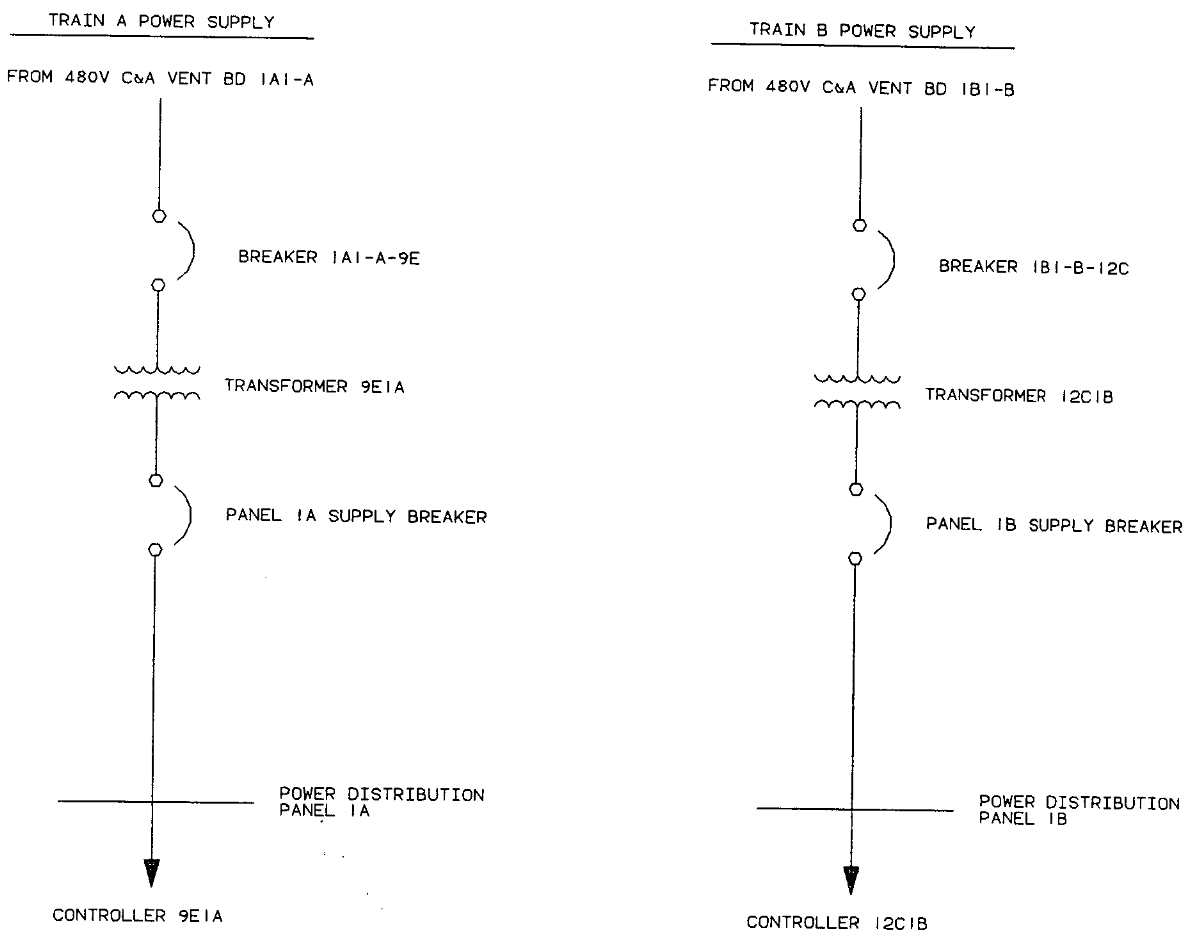


FIGURE 3
CONTAINMENT SYSTEMS
TOP EVENT HH - HYDROGEN IGNITERS
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS
1,2-45W756-2 R1
1,2-45W756-6 R1
1,2-45W760-268-1 R0
1,2-45W1770-8 R1

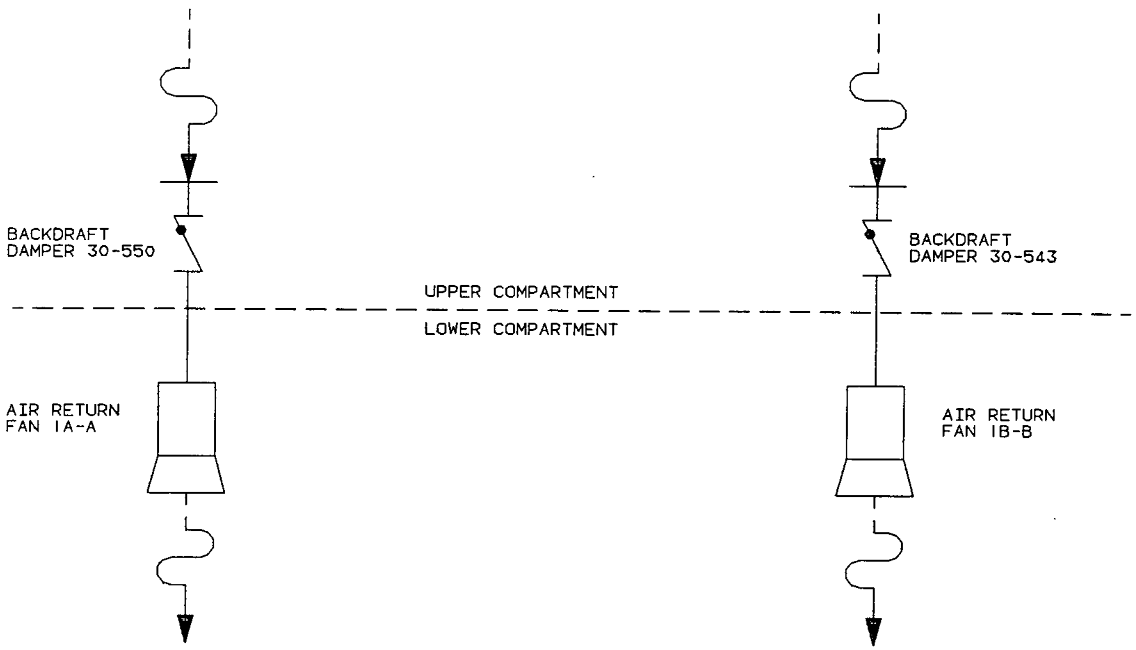


FIGURE 4
CONTAINMENT SYSTEMS TOP EVENT AR(1)
SIMPLIFIED DIAGRAM

REFERENCE DRAWINGS
1-47W866-1 R4

3.3 SEQUENCE QUANTIFICATION

3.3.1 LIST OF GENERIC DATA

3.3.1.1 Introduction

This section presents the database developed for the Watts Bar Nuclear Plant Unit 1 probabilistic risk assessment (PRA) and provides a discussion of the techniques used and steps taken in developing the database.

The following three general areas define the scope of the data analysis presented in this section:

- Component Failure Rates
- Component Maintenance Frequency and Duration
- Internally Caused Initiating Event Frequencies

Common cause failure parameters are presented in Section 3.3.4. Other types of data, such as flood frequencies used in the flood analysis, and human actions, are developed and presented in Section 3.3.3.

The PRA database is primarily based on generic data developed from the cumulative experience of a large population of nuclear power plants documented in the PLG proprietary database. Nevertheless, many plant-specific features were considered in selecting the appropriate generic distributions. For example, some common cause failure parameters were developed based on detailed screening and reclassification of data to ensure applicability of the generic information used for the Watts Bar PRA.

The proprietary PLG generic database has evolved from all of the PRAs that PLG has performed to date. It is based on data collected from U.S. reliability data sources and from operating data of U.S. light water reactors evaluated in past PLG PRAs.

The current database can be updated using the plant-specific data. Such updating can be achieved by means of Bayes' theorem as described later in this section. The following subsections discuss the methods used in developing the data for each of the three general areas; common cause failure parameters are discussed in Section 3.3.4.

3.3.1.2 Basic Concepts

The methodology used to develop the database for this study is based on the Bayesian interpretation of probability and the concept of "probability of frequency." In this context, component failure rates are treated as measurable quantities whose uncertainty is dependent on the state of knowledge of the investigation. The "state of knowledge" is presented in the form of a probability distribution over the range of possible values of that quantity. The probability associated with a particular numerical value of an uncertain but measurable quantity indicates the likelihood that the numerical value is the correct one.

A key issue in developing state of knowledge for the parameters of the PRA models is to ensure that the information regarding each parameter, its relevance, and its value as viewed by the analyst are presented correctly, and that various pieces of information are integrated coherently. "Coherence" is preserved if the final outcome of the process is consistent with every piece of information used and all assumptions made. This is done by using the fundamental tool of probabilistic inference; i.e., Bayes' theorem. Mathematically, Bayes' theorem is written as

$$P(x|E, E_0) = F^{-1} L(E|x, E_0) P(x|E_0) \tag{3.3.1.1}$$

where

$P(x|E, E_0)$ ≡ probability of x being the true value of an unknown quantity in light of new evidence, E, and prior body of knowledge, E_0.

$L(E|x, E_0)$ ≡ likelihood of the new evidence, E, given that the true value is x.

$P(x|E_0)$ ≡ probability of x being the true value of the unknown quantity based on the state of knowledge, E_0, prior to receiving E.

Finally, F is a normalizing factor defined as

$$F \equiv \int_{\text{all } x} L(E|x, E_0) P(x|E_0)\, dx \tag{3.3.1.2}$$

In the context of a plant-specific PRA, three types of information are available for the frequency of elemental events:

Type 1 = the historical information from other similar plants.

Type 2 = general engineering knowledge such as that of the design and manufacture of equipment, sometimes expressed in terms of expert estimate of parameter values or their uncertainty distributions.

Type 3 = the past experience in the specific plant being studied.

The information of types 1 and 2 together constitute the "generic" information, and type 3 is the "plant-specific" or "item-specific" information. The synthesis of information types 1 and 2 to develop generic distributions is explained in Section 3.3.1.3.

Since Watts Bar has had no operating experience yet, the data developed are based entirely on generic information. Any additional plant-specific information collected in the course of operating the Watts Bar unit in the future can be incorporated into the existing data by applying Bayes' theorem. This process is explained in more detail in Section 3.3.2.

It is very important to note that the information type 1 brings an element of plant specificity into the generic data developed for a plant-specific PRA. In general, decisions regarding the relevance and applicability of different pieces of information in developing each generic distribution are made based on type 1 information. Therefore, a piece of information may be judged as being relevant in developing the generic data in one PRA and not relevant in another. As a result, generic distributions for different plant-specific studies could be significantly different. The following sections describe how the general framework described above can be applied for different types of data.

3.3.1.3 Synthesis of Generic Distributions

To discuss the way in which the failure rate distributions were developed based on different types of information, we consider the following information types:

- **Type 1.** Failure data from operating experience at various nuclear power plants.
- **Type 2.** Failure rate estimates or distributions contained in various industry compendia, such as WASH-1400 (Reference 3.3.1-1) and IEEE-500 (Reference 3.3.1-2).

By type 1 information, we mean failure and success data collected from the performance of similar equipment in various power plants. Type 2 information, which could be called processed data, is estimates ranging from the opinion of experts with engineering knowledge about the design and manufacturing of the equipment to estimates based on observed performance of the same class of equipment in various applications.

Normally, type 2 data are either a point estimate, usually referred to as the "best estimate," or a range of values centered about a "best estimate." In some cases, a distribution is provided covering a range of values for the failure rate with the mean or median representing the "best estimate" of the source. For instance, IEEE-500 provides a "low," "high," or "recommended" value for the failure rates under normal conditions and a "maximum" value under extreme environments. WASH-1400, on the other hand, assesses a probability distribution for each failure rate to represent the variability of the available data from source to source. Such distributions are normally centered around a median value judged to be most representative of the equipment in question for nuclear applications.

The methodology used to develop the generic failure rate data uses both types of information to generate generic probability distribution for the failure rates. Such distributions represent variability of the failure rates, from source to source (for type 2 information) and/or from plant to plant (for type 1 information). Obviously, these distributions are in fact, our state of knowledge curves for the failure rate of components. The following discussion helps to clarify the distinction and serves as a prelude to the discussion of the methodology.

Suppose that we have 100 plants and that for each plant the exact value of the failure rate of a particular type of pump is known. Let λ_i be the failure rate of the pump at the ith plant. Suppose further that the λ_i's can be grouped into a limited number of discrete values, say λ_1^*, through λ_5^*, with 20 of the λ_i's being equal , 35 equal to λ_2^*, 25 equal to λ_3^*, 15 equal to λ_4^*, and finally, 5 equal to λ_5^*. The frequency distribution of the λ_i's is then given by the histogram shown in Figure 3.3.1-1.

This histogram represents the "population variability" of the λ_i's because it shows how the failure rate of the particular type of pumps under consideration varies from plant to plant. It is an exact and true representation of the variability of the failure rate at the 100 plants in the population without any uncertainty or ambiguity because the distribution is based on presumed perfectly known failure rates at each and every plant.

Consider now, the case where only estimates, and not the exact values of the failure rates, are available for some, but not all, of the 100 plants in the population. With this state of knowledge, obviously we are not able to know the exact population variability distribution. The question is how one can use this more limited information to estimate the population variability curve and how close the estimate will be to the true distribution, as given in Figure 3.3.1-1.

To answer this question, first note that the desired distribution is a member of the set of all histograms. Because of our limited information, we are uncertain as to which member of that set is, in fact, the true distribution. This situation can be represented by a probability distribution over the set of all possible histograms expressing our state of knowledge about the nature of the true histogram.

For instance, if the entire space, H, of all possible histograms is composed of only n histograms; i.e., if

$$H \equiv \{h_1, h_2, \ldots, h_n\}$$

where h_i represents the ith histogram, the evidence regarding the pump failure rates at different power plants can be used to assess a probability distribution over H as follows:

$$P(H) = \{p_1, p_2, \ldots, p_n\} \quad \text{with} \sum_{i=1}^{n} p_i = 1 \qquad (3.3.1.3)$$

where p_i is the chance that h_i is the true histogram.

Figure 3.3.1-2 depicts the situation in which the variable λ is considered to be continuous, and the desired distribution is a density function.

For a perfect state of knowledge, we would be able to say which h_i is the true distribution; consequently, the corresponding p_i would be equal to 1, and all others equal to 0. However, based on the state of knowledge expressed by Equation (3.3.1.3), our estimate of the true histogram is

$$\bar{h} = \sum_{i=1}^{n} p_i\, h_i \qquad (3.3.1.4)$$

which is called the "expected distribution." Another histogram of interest is one which is assigned the highest chance of being the true histogram. We call it the "most likely distribution," h_m, and we have

$$p_m = \max\{p_i \; i = 1, \ldots, n\} \qquad (3.3.1.5)$$

The problem of obtaining P, as defined by Equation (3.3.1.1), is formulated in the Bayesian context as follows:

$$P(h_i|E) = F^{-1}\, L(E|h_i)P_0(h_i) \qquad (3.3.1.6)$$

where $P_0(h)$ is the prior state of knowledge regarding the set H as defined by Equation (3.3.1.3), and $P(h_i|E)$ is the posterior state of knowledge in light of the evidence E. The evidence is incorporated via the likelihood term $L(E|h_i)$, which is the probability of observing the evidence, given that the true histogram is h. Finally, F is a normalizing factor defined as [see Equation (3.3.1.2)]:

$$F = \sum_{i=1}^{n} L(E|h_i)\, P_0(h_i) \qquad (3.3.1.7)$$

The expected distribution, Equation (3.3.1.4), is our estimate of the true population variability of the failure rate. It shows how the failure rates of similar pumps are distributed among plants in the population. Now, if all we know about a specific pump before we have any experience with it is that it is one member of the population, the population variability curve also becomes our state of knowledge distribution for the failure rate of that specific pump. In other words, generic distributions representing the population variability can also be used to predict the expected behavior of any member of the population, if no other information is available.

For this reason, the generic frequency distributions developed based on type 1 and type 2 information are used as the state of knowledge distributions for the components at Watts Bar.

The following sections describe how types 1 and 2 information can be used to develop generic distribution.

3.3.1.3.1 Generic Distributions Based on Actual Performance Records (Type 1)

The following discussion is based on the method presented in Reference 3.3.1-3. Consider the case where the following set of information is available about the performance of a generic component in N plants:

$$I_1 = \{< k_i, T_i >; i = 1, \ldots, N\} \qquad (3.3.1.8)$$

where k_i is the number of failures of the component in the ith plant during a specific period of time, T_i.

The desired information is $\phi(\lambda)$, the distribution of the failure rate of the component, λ, in light of evidence I_1. This distribution represents the variation of λ from one plant to another, and is analogous to Figure 3.3.1-1.

Following our discussion at the beginning of Section 3.3.1.3, we would like to express a posterior state of knowledge about the true nature of the function $\phi(\lambda)$. To make matters practical, it is assumed that $\phi(\lambda)$ belongs to a particular parametric family of distributions. Let θ be the set of m parameters of $\phi(\lambda)$:

$$\theta = \{\theta_1, \ldots, \theta_m\} \tag{3.3.1.9}$$

For each value of θ, there exists a distribution $\phi(\lambda|\theta)$ and vice versa. Therefore, the state of knowledge distribution over the space of all possible $\phi(\lambda|\theta)$s is the state of knowledge over all possible values of θ and vice versa.

Bayes' theorem, in this case, is written as [see Equation (3.3.1.6)]

$$P(\theta|I_0 I_1) = F^{-1} L(I_1|\theta, I_0)\, P_0(\theta|I_0) \tag{3.3.1.10}$$

where

$P(\theta|I_0 I_1)$ = posterior state of knowledge about θ in light of evidence I_1 and prior information I_0.

$L(I_1|\theta, I_0)$ = the likelihood of evidence I_1 given that the actual set of parameters of $\phi(\lambda)$ is θ.

$P_0(\theta|I_0)$ = prior state of knowledge about θ based on general engineering knowledge I_0.

and F is a normalizing factor

$$F = \int_\theta (I_0|\theta, I_0) P_0(\theta|I_0)\, d\theta$$

The likelihood term is the (conditional) probability of observing the evidence, I_1, given that the data are based on an underlying population variability curve $\phi(\lambda|\theta)$ with θ as the value of its parameters

$$L = P(<k_i, T_i>;\ i = 1, \ldots, N|\theta, I_0) \tag{3.3.1.11}$$

Note that L is also conditional on the prior state of knowledge I_0.

If we assume that the length of operating hours, T_i's, at different plants is independent of one another and that the observed failures, k_i's, also have no dependence (according to our model, each k_i is based on a different underlying failure rate), the joint probability distribution given by Equation (3.3.1.11) can be reduced to the product of the marginal distributions as follows:

$$L(I_1|\theta, I_0) = \prod_{i=1}^{N} P_i(k_i, T_i|\theta, I_0) \tag{3.3.1.12}$$

where $P_i(k_i, T_i|\theta, I_0)$ is the probability of observing k_i failures of the equipment in question during the period T_i in the ith plant assuming that the set of parameters of the underlying population variability curve is θ.

If the failure rate, λ_i, at the ith plant is known exactly, using a Poisson model, the likelihood of observing k_i in T_i can be calculated from

$$P_i(k_i, T_i|\lambda_i) = \frac{(\lambda_i T_i)^{k_i}}{k_i!} \exp(-\lambda_i T_i) \tag{3.3.1.13}$$

However, λ_i is not known. All we know is that λ_i is one of possibly many values of variable λ that represents the variation of the failure rate from plant to plant. In addition, according to our model, λ is distributed according to $\phi(\lambda|\theta)$, with θ being unknown. For this reason, we calculate the probability of observing the evidence, $<k_i, T_i>$, by allowing the failure rate to assume all possible values. This is achieved through averaging Equation (3.3.1.13) over the distribution of λ

$$\begin{aligned} P_i(k_i, T_i|\theta, I_0) &= \int_0^\infty P_i(k_i, T_i|\lambda)\, \phi(\lambda|\theta) d\lambda \\ &= \int_0^\infty \frac{(\lambda T_i)^{k_i} e^{-\lambda T_i}}{k_i!}\, \phi(\lambda|\theta) d\lambda \end{aligned} \tag{3.3.1.14}$$

Depending on the parametric family chosen to represent $\phi(\lambda|\theta)$, the integration in Equation (3.3.1.14) can be carried out analytically or by numerical techniques. For example, if $\phi(\lambda_i|\theta)$ is assumed to be a gamma distribution that has the following form:

$$\phi(\lambda|\alpha, \beta) = \frac{\beta^\alpha}{\Gamma(\alpha)} \lambda^{\alpha-1} e^{-\beta\lambda} \tag{3.3.1.15}$$

with α and β, both nonnegative, as its parameters, the integral can be done analytically resulting in (Reference 3.3.1-4)

$$P_i(k_i, T_i|\alpha, \beta) = \frac{T_i^{k_i}}{k_i!} \frac{\beta^\alpha}{\Gamma(\alpha)} \frac{\Gamma(\alpha + k_i)}{(\beta + T_i)^{\alpha + k_i}} \tag{3.3.1.16}$$

In developing failure rate distributions, $\phi(\lambda|\theta)$ is assumed to be lognormally distributed with μ as the median and σ as the standard deviation of the underlying normal. Then,

$$\phi(\lambda|\mu, \sigma) = \frac{1}{\sqrt{2\pi}\, \sigma\lambda} \exp\left\{-\frac{1}{2}\left(\frac{\ell n\, \lambda - \mu}{\sigma}\right)^2\right\} \tag{3.3.1.17}$$

In this case, Equation (3.3.1.14) is calculated numerically.

The total likelihood for all N plants can now be found by using Equation (3.3.1.14) in Equation (3.3.1.12):

$$L(I_1|\theta, I_0) = \prod_{i=1}^{N} \left\{ \int_0^{\infty} \phi(\lambda|\theta) \frac{(\lambda T_i)^{k_i}}{k_i!} \exp(-\lambda T_i) d\lambda \right\} \quad (3.3.1.18)$$

The posterior distribution resulting from using the likelihood of Equation (3.3.1.18) in Bayes' theorem, Equation (3.3.1.10), is a probability distribution over the m-dimensional space of θ. Any point, θ, in this space has a one-to-one correspondence with a distribution, $\phi(\lambda_i|\theta)$, in the space of $\phi(\lambda|\theta)$. Figure 3.3.1-3 is an example of $P(\theta|I_0, I_1)$ constructed for $\theta = \{\alpha,\beta\}$, the two parameters of gamma distribution based on the pump data from all U.S. nuclear power plants (Reference 3.3.1-4).

The "expected distribution" is obtained from [see Equation (3.3.1.4)]

$$\bar{\phi}(\lambda) = \int_{\theta} \theta(\lambda|\theta) P(\theta|I_0, I_1) d\theta \quad (3.3.1.19)$$

The quantity $\phi(\lambda)$ "summarizes" the information about λ and is used in this study as the model for generic failure distributions.

Sometimes it is also useful to obtain the "most likely distribution" [see Equation (3.3.1.4)]. According to the definition, the most probable distribution of λ is the one whose parameters maximize $P(\theta|I_0 I_1)$. These parameters are therefore the solution of the following system of m equations:

$$\frac{\partial P(\theta|I_0 I_1)}{\partial \theta_i} \Big| \theta_{i,\max} = 0; \qquad i = 1, \ldots, m \quad (3.3.1.20)$$

The methodology discussed above also applies to failure on demand-type data where the evidence is of the form

$$I_1 = \{< k_i, D_i >, i = 1, \ldots, N\} \quad (3.3.1.21)$$

where k_i and D_i are the number of failures and demands in the ith plant, respectively. This can be done if the Poisson distribution used in Equation (3.3.1.14) is replaced by the binomial distribution

$$P(k_i, D_i|\lambda) = \frac{D_i!}{k_i!(D_i - k_i)!} \lambda^{k_i} (1-\lambda)^{D_i - k_i} \quad (3.3.1.22)$$

Example

For motor-operated valve failure on demand, the following data from six plants were available:

Plant	Number of Failures (k)	Number of Demands (D)
1	10	$1.65 \times 10^{+3}$
2	14	$1.13 \times 10^{+4}$
3	7	$1.73 \times 10^{+3}$
4	42	$6.72 \times 10^{+3}$
5	3	$1.26 \times 10^{+3}$
6	31	$9.72 \times 10^{+3}$

These data, which form a set of type 1 information, I_1, were used in mode 1 of the Data Analysis module of RISKMAN (Reference 3.3.1-5), which calculates Equations (3.3.1.14) and (3.3.1.18) and generates $\phi(\lambda)$ based on Equation (3.3.1.19). The result was a 20-bin discrete probability distribution with the following characteristics:

Parameter	Value
5th Percentile	6.82×10^{-4}
50th Percentile	3.06×10^{-3}
95th Percentile	1.42×10^{-2}
Mean	5.09×10^{-3}

3.3.1.3.2 Generic Distributions Using Estimates of Available Sources of Generic Data (Type 2)

As mentioned earlier, generic data frequently are not in the fundamental form given by Equations (3.3.1.8) and (3.3.1.21). Rather, most sources report point or interval estimates, or even distributions for failure rates (type 2 information). These estimates are either judgmental (expert opinion), or based on standard estimation techniques used by the analysts to translate raw data into point or interval estimates, and sometimes into a full distribution.

An example of such estimation techniques is the well-known maximum likelihood estimator given by

$$\lambda_M = \frac{k}{T} \qquad (3.3.1.23)$$

where k is the total number of failures in T units of operating time. Most data sources report λ_M, and not k and T.

To develop a model for constructing generic distributions using this type of data, the following cases are considered.

3.3.1.3.2.1 <u>Estimating an Unknown Quantity Having a Single True Value</u>. The following method is adopted from Reference 3.3.1-6. Suppose that there are M sources, each providing its own estimate of λ, which has a single true, but unknown, value, λ_t. An example is the failure rate of a particular component at a given plant. The true value of that failure rate, λ_t, will be known at the end of the life of the component. Before then, however, the failure rate may be estimated by one or more experts who are familiar with the performance of the component. Let

$$I_2^* = \left\{\lambda_i^*;\, i = 1, \ldots, M\right\} \qquad (3.3.1.24)$$

be the set of such estimates where λ_i^* is the estimate of the ith expert for λ_t.

The objectives are to use information I_2^* and to obtain a state of knowledge distribution for λ_t. Obviously, when everything is known about λ_t, such a state of knowledge distribution is a delta function centered at λ_t.

$$P(\lambda|\text{Perfect Knowledge}) = \delta(\lambda - \lambda_t) \qquad (3.3.1.25)$$

Note that, in Equation (3.3.1.25), λ is used as a variable representing the unknown failure rate.

Assuming a prior state of knowledge, $P_0(\lambda)$, about the quantity λ, Bayes' theorem can be used to incorporate information I_2^* into the prior and to obtain an "updated" state of knowledge about λ.

$$P\left(\lambda|\lambda_1^*, \ldots, \lambda_N^*\right) = k^{-1}\, L(\lambda_1^*, \ldots, \lambda_N^*|\lambda)P_0(\lambda) \qquad (3.3.1.26)$$

For N independent sources of information, the likelihood term, $L(\lambda_1^*, \ldots, \lambda_N^*|\lambda)$ can be written as

$$L(\lambda_1^*, \ldots, \lambda_N^*|\lambda) = \prod_{i=1}^{N} P_i\left(\lambda_i^*|\lambda\right) \qquad (3.3.1.27)$$

where $P_i(\lambda_i^*|\lambda)$ is the probability that the estimate of the ith source is λ_i^*, when the true value of the unknown quantity is λ.

The case of dependent sources of information is discussed in Reference 3.3.1-6. Obviously, if the ith source is a perfect one,

$$P_i\left(\lambda_i^*|\lambda\right) = \delta\left(\lambda_i^* - \lambda\right) \qquad (3.3.1.28)$$

which means that the estimate, λ_i^* , is the true value. The posterior, $P(\lambda|\lambda_i^*, \ldots, \lambda_N^*)$, in this case, will be entirely determined by the estimate of this source

$$P\left(\lambda|\lambda_1^*, \ldots, \lambda_N^*\right) = \delta\left(\lambda - \lambda_i^*\right) \qquad (3.3.1.29)$$

In another extreme, when it is believed that the source is totally unreliable,

$$P_i\left(\lambda_i^* | \lambda\right) = C \qquad (3.3.1.30)$$

where C is a constant. This means that if the true value is λ, the estimate of the ith source can be anything. Using a likelihood of this form in Equation (3.3.1.27) will show that the estimate of this source, as expected, has no effect on shaping the posterior state of knowledge.

The likelihood term in this approach is the most crucial element. It reflects the analyst's degree of confidence in the sources of information, their accuracy, and the degree of applicability of their estimates to the particular case of interest.

As can be seen, the subjective nature of evaluating and "weighting" of the evidence from different sources fits very well in the above formulation. This becomes clearer in discussing the following models for the likelihood functions in Equation (3.3.1.27).

Suppose that in estimating the true value of λ_t, the ith source makes an error of magnitude E. Two simple models relating λ_t, E, and λ_i^* are

$$\lambda_i^* = \lambda_t + E \qquad (3.3.1.31)$$

$$\lambda_i^* = \lambda_t \times E \qquad (3.3.1.32)$$

In the model of Equation (3.3.1.31), if a normal distribution is assumed for the error term of the estimate of each source, the likelihood function will be a normal distribution with mean equal to $\lambda_t + b_i$, where b_i is the expected error, or, in other words, a "bias" term about which the error of the ith source is propagated.

Formally, we have

$$P\left(\lambda_i^* | \lambda_t\right) = \frac{1}{\sqrt{2\pi}\,\sigma_i} \exp\left\{-\frac{1}{2}\left(\frac{\lambda_i^* - (\lambda_t + b_i)}{\sigma_i}\right)^2\right\} \qquad (3.3.1.33)$$

The variance of the likelihood, σ_i^2, is the variance of the error distribution. Values of b_i and σ_i are subjectively assessed by the data analyst, and reflect the credibility and accuracy of the source as viewed by the data analyst. Sometimes, certain information provided by the source, such as the uncertainty bound for the estimate, can be used to assess σ_i.

If, in addition to a normal likelihood function, a normal prior distribution representing the state of knowledge of the data analyst is assumed for λ_t with mean λ_0 and variance σ_0^2, the posterior distribution in Equation (3.3.1.26) will also be normal with mean, λ_p, given by

$$\lambda_p = \sum_{i=0}^{N} w_i\left(\lambda_i^* - b_i\right) \tag{3.3.1.34}$$

and variance

$$\sigma_p^2 = \left(\sum_{i=0}^{N} \frac{1}{\sigma_i^2}\right)^{-1} \tag{3.3.1.35}$$

where w_i, defined as

$$w_i = \left(\frac{\sigma_p}{\sigma_i}\right)^2 \tag{3.3.1.36}$$

is the weight given to the ith source.

Note that

$$\sum_{i=0}^{N} w_i = 1 \tag{3.3.1.37}$$

The mean therefore is a weighted average of the individual estimates after correcting for their expected biases. Also, as can be seen from Equation (3.3.1.36), smaller values of σ_i result in higher weights, implying that the source that is believed to make errors of smaller magnitudes (σ_i is the variance of E) is assigned a higher weight, which is intuitively expected. Extreme cases are when $\sigma_i = 0$ (highest degree of confidence in the ith estimate), for which $w_i = 1$, and when $\sigma_i = \infty$ (no confidence at all) for which $w_i = 0$.

If, instead of the model of Equation (3.3.1.31), the model of Equation (3.3.1.32) is applied and the logarithm of the error is assumed to be normally distributed, the likelihood function for the ith source becomes a lognormal distribution

$$P_i\left(\lambda_i^* | \lambda_t\right) = \frac{1}{\sqrt{2\pi}\,\sigma_i \lambda_i^*} \exp\left\{-\frac{1}{2}\left(\frac{\ell n\lambda_i^* - (\ell n\lambda_t + \ell n b_i)}{\sigma_i}\right)^2\right\} \tag{3.3.1.38}$$

where $\ell n b_i$ is the logarithmic mean error about the logarithm of the true value, $\ell n\lambda_t$, and σ_i is the multiplicative standard deviation. Again, $P_i(\lambda_i^* | \lambda_t)$ is the probability that the estimate of the ith source is λ_i^* when the true value of the failure rate is λ_t. Some evidence in support of the lognormality of $P_i(\lambda_i^* | \lambda_t)$ is provided in References 3.3.1-6 and 3.3.1-7.

By using the model of Equation (3.3.1.38) for individual likelihoods in Bayes' theorem, Equation (3.3.1.26), and assuming a lognormal prior distribution for λ_t, the posterior state of knowledge will also be a lognormal with the following median value:

$$\lambda_{50,p} = \prod_{i=0}^{N} \left(\frac{\lambda_i^*}{b_i} \right)^{w_i} \tag{3.3.1.39}$$

where w_i is defined, as in Equation (3.3.1.36).

The median, then, is a weighted geometric average of the individual estimates after correcting for the multiplicative biases. Note that the usual arithmetic and geometric average methods frequently used in the literature are special cases of these Bayesian normal and lognormal models. For instance, Reference 3.3.1-2 uses the following geometric average of the estimates provided by several experts:

$$\bar{\lambda} = \left(\prod_{i=1}^{N} \lambda_i \right)^{1/N} \tag{3.3.1.40}$$

which assumes equal weights ($w_i = 1/N$), no bias ($b_i = 1$), no prior information, and does not show any uncertainty about the resulting value.

Example

Reference 3.3.1-8 provides a point estimate of 5.60×10^{-3} for the demand failure rate of motor-operated valves. We would like to use this estimate and to obtain a state of knowledge distribution for the MOV failure rates. We use the lognormal model of Equation (3.3.1.38) to express our confidence in the estimated value

$$P\left(\lambda_1^* | \lambda_t\right) = \frac{1}{\sqrt{2\pi}\, \sigma_1 \lambda_1^*} \exp\left\{ -\frac{1}{2} \left(\frac{\ell n \lambda_1^* - (\ell n \lambda_t + \ell n b_1)}{\sigma_1} \right)^2 \right\} \tag{3.3.1.41}$$

where λ_1^* is the estimate (5.60×10^{-3}), and λ_t is the assumed true value of the failure rate that remains an unknown variable at this point. Our subjective judgment about the magnitude of error of the data source is expressed by assigning numerical values to the "bias" term b_1 and the logarithmic standard deviation σ_1.

We assume that there is no systematic bias ($b_1 = 1$). We estimate σ_1 with the aid of range factor, which is a more understandable quantity. Unless otherwise indicated, the range factor here is defined as the ratio of the 95th to the 50th percentiles of the lognormal distribution. Therefore, given the range factor, the value of σ_1 is obtained from the following equation:

$$\sigma_1 = \frac{\ell n \, RF}{1.645} \tag{3.3.1.42}$$

For our example, we assume a range factor of 3. Normally, such a range factor represents a relatively high degree of confidence and means that the source's estimate could be a factor of 3 higher or lower than the true failure rate and that such a statement is made with 90% confidence. Using this range factor in Equation (3.3.1.42) results in a value of 0.67 for σ_1.

If we now use the likelihood of Equation (3.3.1.41) in Bayes' theorem, Equation (3.3.1.26), and assume a flat prior distribution, $P_0(\lambda_t)$, the posterior distribution will be

$$P(\lambda|\lambda_1^* = 5.6\times10^{-3}) = 106.65\ \exp\left\{-\frac{1}{2}\left(\frac{\ell n\,\lambda - \ell n\,5.6\times10^{-3}}{0.67}\right)^2\right\} \qquad (3.3.1.43)$$

which has the following characteristics:

Parameter	Value
5th Percentile	1.87×10^{-3}
50th Percentile	5.60×10^{-3}
95th Percentile	1.68×10^{-2}
Mean	7.01×10^{-3}

3.3.1.3.2.2 Estimating Distributions Using Point Estimates of Various Sources. We now go back to our original problem, which was estimating the generic failure rate distribution $\phi(\lambda|\theta)$. This time, however, we assume that instead of having the set of $<k_i, T_i>$ defined in Equation (3.3.1.8) from various plants, we are given one estimate, λ_i^*, for each plant. That is, the evidence is of the form

$$I_2 = \left\{\lambda_i^*\ i = 1, \ldots, N\right\} \qquad (3.3.1.44)$$

The model to be used is a combination of the methods presented previously and is fully discussed in References 3.3.1-4 and 3.3.1-9. A particular family of parametric distributions, $\phi(\lambda|\theta)$, is assumed for λ, and the information I_2 is used in Bayes' theorem to obtain a posterior distribution over the entire set of possible values of θ and consequently over all possible distributions $\phi(\lambda|\theta)$. Formally,

$$P(\theta|I_2, I_0) = F^{-1}\, L(I_2|\theta, I_0)\, P_0(\theta|I_0) \qquad (3.3.1.45)$$

See the set of definitions immediately following Equation (3.3.1.10) for interpretation of the terms in Equation (3.3.1.45).

The total likelihood function in the present case when λ_i's are independently estimated can be written as [see Equation (3.3.1.12)]

$$L(I_2|\theta, I_0) = \prod_{i=1}^{N} P_i(\lambda_i^*|\theta, I_0) \tag{3.3.1.46}$$

where

$P_i(\lambda_i^*|\theta, I_0)$ ≡ probability that the estimate provided for the ith plant is λ_i^* if the parameter of the population variability distribution of the failure rates is θ. (3.3.1.47)

To make matters clearer, note that we are assuming that the ith source of data is providing an estimate for the failure rate at a particular plant, and all we know is that failure rates vary from plant to plant according to the variability curve $\phi(\lambda|\theta)$. Each λ_i therefore is an estimate of one point in that distribution. As a result, there are two sources of variability in the estimates. First, estimates of individual sources are not necessarily perfect; i.e., they could involve errors and biases, as discussed in the previous section. Second, even if all the sources were perfect, the estimates would still be different due to the actual variation of the failure rate from plant to plant.

Based on our discussion in the previous section, the confidence that we have in the accuracy of the estimate λ_i^* for the failure rate at the ith plant can be modeled by a lognormal distribution [see Equation (3.3.1.38)]. Assuming no bias, we have

$$P_i(\lambda_i^*|\lambda_i) = \frac{1}{\sqrt{2\pi}\,\sigma_i\lambda_i^*} \exp\left\{-\frac{1}{2}\left(\frac{\ell n\lambda_i^* - \ell n\lambda_i}{\sigma_i}\right)^2\right\} \tag{3.3.1.48}$$

where λ_i is the true value of the failure rate at the ith plant. Again, we really do not know λ_i, but we assume that it belongs to $\phi(\lambda|\theta)$, the distribution representing the variability of λ_i's from plant to plant. The relationship between $P_i(\lambda_i^*|\theta, I_0)$ and $\phi(\lambda|\theta)$ is shown in Figure 3.3.1-4.

Therefore, as we did in the case of Equation (3.3.1.14), we can write

$$P_i(\lambda_i^*|\theta, I_0) = \int_0^\infty P_i(\lambda_i^*|\lambda)\,\phi(\lambda|\theta)d\lambda \tag{3.3.1.49}$$

As mentioned earlier, in developing the failure rate distributions, $\phi(\lambda|\theta)$ is assumed to be lognormally defined by Equation (3.3.1.17). With this assumption, the integration in Equation (3.3.1.49) can be done analytically, and the result is

$$P_i(\lambda_i^*|\theta, I_0) = \frac{1}{2\pi\sqrt{\sigma_i^2 + \sigma^2}\ \lambda_i^*} \exp\left\{-\frac{1}{2}\frac{(\ell n\lambda_i^* - \mu)^2}{\sigma_i^2 + \sigma^2}\right\} \tag{3.3.1.50}$$

Equation (3.3.1.45), Bayes' theorem, is now written as:

$$P\left(\theta|\lambda_1^*, \ldots, \lambda_N^*\right) = F^{-1}\prod_{i=1}^{N} P_i\left(\lambda_i^*|\theta, I_0\right) P_0\left(\theta|I_0\right) \tag{3.3.1.51}$$

The most probable and expected distributions of λ can be found in the same way as discussed in Section 3.3.1.3.2. The expected distribution is calculated by using the result of Equation (3.3.1.48) in Equation (3.3.1.19). The parameters of the most likely distribution are shown to be solutions of the following system of equations:

$$\mu = \sum_{i=0}^{N} \frac{\left(\sigma_i^2 + \sigma^2\right)^{-1}}{\sum_{i=0}^{N}\left(\sigma_i^2 + \sigma^2\right)^{-1}} \ell n\lambda_i^* \tag{3.3.1.52}$$

$$\sum_{i=1}^{N}\left[\frac{1}{\sigma_i^2 + \sigma^2} - \left(\frac{\left(\ell n\lambda_i^* - \mu\right)^2}{\sigma_i^2 + \sigma^2}\right)\right] = 0 \tag{3.3.1.53}$$

For perfect sources of information (i.e., $\sigma_i = 0$), the above equations simplify and result in the following solution:

$$\mu = \ell n\left(\prod_{i=1}^{N} \lambda_i^*\right)^{1/N} \tag{3.3.1.54}$$

$$\sigma^2 = \frac{1}{N}\sum_{i=0}^{N}\left(\ell n\, \lambda_i^* - \mu\right)^2 \tag{3.3.1.55}$$

Note that Equations (3.3.1.54) and (3.3.1.55) are similar to the conventional results for fitting a lognormal distribution to a set of estimates. It should also be mentioned that the results of this section apply to any set of failure rate estimates from various sources where a true variability is suspected to exist among the actual values being estimated by each source. For instance, if several generic sources of data provide estimates for a particular type of equipment and it is known or suspected that each source's estimate is based on a different subset of the population, the methods of this section can be applied to obtain a generic distribution representing the "source to source" variability of the failure rate.

Example

The following set of estimates is available for the demand failure rate of MOVs:

Source	Estimate
WASH-1400 (Reference 3.3.1-1)	1.00×10^{-3}
NUREG/CR-1363 (Reference 3.3.1-8)	5.60×10^{-3}
GCR (Reference 3.3.1-10)	1.00×10^{-3}

To use the model of this section, we need to assign range factors to each source as a measure of our confidence in the estimate provided by that source. In this way, we will be able to determine $P_i(\lambda_i^* | \lambda_i)$, Equation (3.3.1.48), for each source.

Following our discussion in the previous example, we assign a range factor of 3 to the estimate of NUREG/CR-1363. For the estimate of WASH-1400, we assign a range factor of 5, which results in a broader likelihood, $P_i(\lambda_i^* | \lambda_i)$, for that source and represents a lesser degree of confidence as compared to NUREG/CR-1363. This is due to the fact that the estimate of NUREG/CR-1363 appears to be based on a larger sample of MOV failures in nuclear applications than does the estimate of WASH-1400. The latter provides a range factor of 3 for the lognormal distribution whose median (1.00×10^{-3}) we have taken as the estimate. Assigning a larger range factor of 5 also means that we believe that WASH-1400 has overstated its confidence in the estimated median value.

The idea of broadening some WASH-1400 distributions when used as generic curves was introduced in an early site-specific PRA study (References 3.3.1-11 and 3.3.1-12) where the WASH-1400 curves (as given) were used as generic prior distributions. It was then found that several posterior distributions, reflecting the evidence of the specific plant, lay in the tail region of the prior distributions on the high side. These results led us to the conclusion that the generic curves had to be broadened to reflect greater uncertainty.

References 3.3.1-13 and 3.3.1-14 provide further support to our decision. In Reference 3.3.1-13, the authors reviewed experimental results that test the adequacy of probability assessments, and they concluded that "the overwhelming evidence from research on uncertain quantities is that people's probability distributions tend to be too tight. The assessment of extreme fractiles is particularly prone to bias." Referring to the Reactor Safety Study, they state, "The research reviewed here suggests that distributions built from assessments of the 0.05 and 0.95 fractiles may be grossly biased."

Commenting on judgmental biases in risk perception, Reference 3.3.1-14 states:

> *A typical task in estimating uncertain quantities like failure rates is to set upper and lower bounds such that there is a 98% chance that the true value lies between them. Experiments with diverse groups of people making many different kinds of judgments have shown that, rather than 2% of true values falling outside the 98% confidence bounds, 20 to 50% do so* [Reference 3.3.1-13]. *Thus, people think that they can estimate such values with much greater precision than is actually the case.*

The numerical effect of using a larger range factor is illustrated in the following table:

Distribution	5th Percentile	Median	Mean	95th Percentile	Range Factor
WASH-1400	3.3×10^{-4}	1.0×10^{-3}	1.2×10^{-3}	3.0×10^{-3}	3
Broadened Distribution	2.0×10^{-4}	1.0×10^{-3}	1.6×10^{-3}	5.0×10^{-3}	5

We see here that the medians are the same and that the mean value increases slightly reflecting the extension of the high side tail of the curve.

For the cases where WASH-1400 was the only source used for a failure rate, the above methodology was used to generate a broader generic curve from the distribution of WASH-1400. The applied range factor, however, was not necessarily the same for each case. For the estimates from the three sources listed previously, the range factors are assigned as follows:

Source	Range Factor
WASH-1400	5
NUREG/CR-1363	3
GCR	10

The above values and the estimates from the three sources were used as input to mode 2 of the Data Analysis module of RISKMAN, which evaluates Equations (3.3.1.48) through (3.3.1.51) and obtains an expected distribution based on an integration similar to Equation (3.3.1.19).

The resulting histogram has the following characteristics:

Parameter	Value
5th Percentile	1.72×10^{-4}
50th Percentile	2.15×10^{-3}
95th Percentile	1.22×10^{-2}
Mean	4.55×10^{-3}

3.3.1.3.3 Generic Distributions Based on a Mixture of Type 1 and Type 2 Data

An obvious extension of the situations discussed in the previous sections is the case where a mixture of types 1 and 2 information is available. In this case, the equivalent of Equations (3.3.1.10) and (3.3.1.45) is

$$P(\theta|I_2, I_1, I_0) = F^{-1}\, L(I_2, I_1|\theta, I_0)P_0(\theta|I_0) \qquad (3.3.1.56)$$

If I_1 and I_2 are independent pieces of information,

$$L(I_2, I_1|\theta, I_0) = L(I_2|\theta, I_0)L(I_1|\theta, I_0) \qquad (3.3.1.57)$$

where the terms in the right side of the equation are defined by Equations (3.3.1.10) and (3.3.1.46).

The expected distribution of λ can now be found from

$$\bar{\phi}(\lambda) = \int_0^\infty \phi(\lambda|\theta)\, P(\theta|I_2\, I_1, I_0)d\theta \qquad (3.3.1.58)$$

Example

As an example, we use the combination of the data given in the examples in the previous sections. This information was used as the main input to the Data Analysis module of RISKMAN, which calculates Equations (3.3.1.56) through (3.3.1.58). The resulting discretized distribution has the following characteristics:

Parameter	Value
5th Percentile	7.28×10^{-4}
50th Percentile	2.96×10^{-3}
95th Percentile	1.01×10^{-2}
Mean	4.27×10^{-3}

A summary of the types 1 and 2 evidence and the results of this example are presented in Figure 3.3.1-5.

3.3.1.3.4 Failure Rate Distributions

Developing a generic database requires a thorough review, analysis, and tabulation of the available generic data for each identified component failure mode. The PLG generic database is proprietary, and is documented in Reference 3.3.1-15. This generic database was used as the generic data basis for Watts Bar. In addition to generic data sources such as WASH-1400 (Reference 3.3.1-1) and IEEE-500 (Reference 3.3.1-2), several well-documented site-specific failure rate data from power plants examined in previous or ongoing risk studies were used in the development of the generic database. This ensures that the final failure rate distributions accurately reflect all of the information that is currently available.

A practical difficulty in using the available generic estimates in the process of developing generic distributions is the lack of standardization in the generic literature. This dictates that using generic sources involves much more than a simple catalog of published failure rate estimates. Each source presents its own unique set of advantages and drawbacks, and these factors must be carefully evaluated before a meaningful comparative analysis may be performed. Typical problems that are encountered include incompatibility between failure and test data, inclusion of failures due to other than hardware-related causes, exclusion of failures due to licensing-based reporting criteria, and a general lack of specific documentation of assumptions, boundary conditions, and methodologies. Often, it is simply not possible to discern the reasons for significant differences among several sources publishing data for the same component failure mode.

Because of the inherent difficulty in ascertaining the direct comparability among these various estimates, the only practical approach to the problem is the assignment of subjective "weighting factors" to each piece of data, based on the perceived compatibility of the source with the desired failure rate information. These weights are assigned by assessing either a range factor or σ parameter for the likelihood functions for each source according to the models discussed in Section 3.3.1.3.2. This process is computerized in RISKMAN, which takes as input various point estimates and corresponding subjective range factors as well as actual plant operating experience of the component in question at various plants. The code then performs Bayesian calculations based on the models and generates an average distribution for the failure rate representing source-to-source and/or plant-to-plant variability of the data. This process involves several iterations in running the code and reviewing the results to ensure that the range of discrete probability distribution is a reasonable representation of the input information and that the binning of the distribution (20 bins or less) was done properly.

In other cases, where only one source of data is available for the component, failure rate distributions are represented as lognormal. In general, these failure rate distributions are derived by defining the median value and range factor as the two most physically meaningful parameters of the lognormal distribution. (The range factor is defined here as the ratio of the 95th percentile to the median, or the square root of the ratio of the 95th and 5th percentiles.) To provide traceable documentation of the data sources used in this analysis, the median value of such distributions is based on published data. The range factor is subjectively assigned so that the resulting 5th and 95th percentiles of the distribution represent realistic bounds for expected or observed component failure rates.

The relative magnitudes of the range factors developed for the various distributions are influenced by a set of consistent evaluation criteria. In general, range factors significantly greater than 10 (i.e., a span of more than 100 in failure frequency between the 5th and 95th percentiles) are considered to produce distributions so broad as to convey a nearly uninformed state of knowledge and therefore would be of marginal utility in any quantification process. The mean value of such a broad distribution, while defined mathematically, is virtually meaningless as a representation of expected component performance because, in truth, very little is known about how the entire population behaves. Some distributions are assigned range factors on the order of 10. Typically, these distributions are characterized by sparse generic data not closely correlated to the desired component failure mode and a relatively low degree of confidence in the available source. It is felt that a distribution this broad conveys only marginal knowledge as to the behavior of a population and is generally indicative of the application of good engineering judgment to minimal prior information. Some distributions are assigned range factors on the order of 3 to 5; i.e., spans of approximately 10 to 25 between the 5th and 95th probability percentiles. While these distributions are still relatively broad, they represent a higher degree of confidence in the failure rate estimate used as the median value.

Treatment of the generic distributions from IEEE-500 (Reference 3.3.1-2) is discussed. This reference contains data for electronic, electrical, and sensing components. The reported values are mainly synthesized from the opinions of some 200 experts (a form of the Delphi procedure is used). Each expert reports a "low," "recommended," and "high" value of the failure rate under normal conditions and a "maximum" value that would be applicable under all conditions (including abnormal ones). The pooling of the estimates is done using a geometric averaging technique; e.g.,

$$\lambda_{max} = \left(\prod_{i=1}^{N} \lambda_{max,\, i} \right)^{1/N} \tag{3.3.1.59}$$

This method of averaging is considered a better representation of the expert estimates, which are often given in terms of negative powers of 10. In effect, the usual arithmetic averages of the exponents are used, which, as discussed in Section 3.3.1.3.2, is a special case of the Bayesian model presented in this report.

Reference 3.3.1-2 does not recommend a distribution. The method of averaging, however, suggests that the authors have in mind a lognormal distribution. Our task now is to determine this distribution from the given information.

The recommended value is suggested to be used as a "best" estimate. The word "best" is, of course, subject to different interpretations. We have decided to use it as the median value mainly for two reasons. First, for skewed, lognormal type distributions, the median is a more representative measure of central tendency than the mean, which is very sensitive to the tails of the distribution. Thus, we suspect that the experts who submitted their "recommended" estimates actually had median values in mind. Experimental evidence (Reference 3.3.1-16) also indicates that assessors tend to bias their estimates of mean values toward the medians. The second reason is that this choice is conservative since the mean value of our resulting distribution is then larger than the "recommended" value. The "maximum" value is taken to be the 95th percentile of the lognormal distribution.

For the majority of the components for Watts Bar, generic component failure rates were taken from PLG Generic Database (Reference 3.3.1-15). In a few cases, additional generic distributions had to be developed for some specific types of equipment.
Reference 3.3.1-15 provides a detailed documentation of the generic distributions used in this study. The main characteristic values of the generic failure rate distributions used for Watts Bar are presented in Table 3.3.1-1.

3.3.1.4 Component Maintenance Data

3.3.1.4.1 Introduction

Maintenance activities that remove components from service and alter the normal configurations of mechanical or electrical systems can provide a significant contribution to the overall unavailability of those systems. This section describes how generic maintenance data were used to develop distributions for generic component maintenance unavailability.

These distributions apply to maintenance performed during normal operation or, in some cases, at hot shutdown (but not during cold shutdown). These include both regularly scheduled preventive maintenance activities and unplanned maintenance events. The specific causes leading to these maintenance activities can include repairs of component failures experienced during operation, repairs of failures discovered during periodic testing, removal of components from service for unplanned testing or inspection, minor adjustments, and hardware modifications.

To quantify maintenance unavailabilities, both the frequency and the mean duration of maintenance are necessary. The frequency defines the rate at which components are removed from service, while the mean duration is the average amount of time that the component will be out of service. The unavailability due to maintenance is calculated according to

$$Q_M \cong f \bullet \tau \qquad (3.3.1.60)$$

where f is the maintenance frequency and τ is the mean maintenance duration or, equivalently, the mean time to repair.

To obtain a state of knowledge distribution for the maintenance-related unavailability Q_M, state of knowledge distributions for both f and τ are needed. Such distributions are developed as described in the following section.

3.3.1.4.2 Frequency of Maintenance

The generic maintenance frequency distributions used for the Watts Bar PRA were selected from generic maintenance frequency distributions developed for 17 different categories of component types and normal service duty; i.e., operating or standby. The basis for these distributions is described in the PLG Generic Database (Reference 3.3.1-15), and the component categories are presented in Figure 3.3.1-6. The corresponding distributions were developed based on observed maintenance data from 14 light water reactor (LWR) operating units covering approximately 150 reactor-years of experience. The statistical method used to develop these distributions was the same as the two-stage method applied in the case of component failure rates. The distributions, consequently, represent the probable range of variation of component maintenance data within the generic population. In the absence of plant-specific data, such population variability distributions are the best estimate of the maintenance frequency of various components. The main characteristics of these distributions are presented in Table 3.3.1-2.

3.3.1.4.3 Duration of Maintenance

As defined in this database, the duration of a maintenance event includes the entire time period during which the affected component is unavailable for operation. This is defined to be the period starting when the component is originally isolated or otherwise removed from service, and ending when the component is returned to service in an operable state. In many cases, this duration may be only weakly dependent on the actual time required for maintenance personnel to effect the needed repairs.

Generic distributions for mean maintenance durations were developed for 12 categories of components based on the component type and the inoperability limitations imposed by plant technical specifications. The basis for these distributions is described in the PLG proprietary database (Reference 3.3.1-15), and the component categories are presented in Figure 3.3.1-7. These distributions were developed based on over 150 reactor-years of experience with 14 LWR units, as collected and analyzed in various PRAs performed by PLG on those reactors. The two-stage methodology described in Section 3.3.1.2 was used to develop the maintenance duration uncertainty distributions. These distributions represent the plant-to-plant variability of mean maintenance duration among the plants in the generic population. The main characteristics of these distributions are presented in Table 3.3.1-3.

3.3.1.5 Internally Caused Initiating Events Frequencies

3.3.1.5.1 Introduction

The initiating events considered for this PRA are divided into two groups according to the method used for quantifying their frequencies. The first group comprises those events for which data available from other nuclear plants are judged to be relevant. Data from other plants are then used, as described in Section 3.3.1.3.2, to create generic distributions for the event frequencies.

The second group consists of events that are caused by loss of support systems. These systems have designs that are unique to plants, and data for similar events from other plants are not relevant. The frequencies of these events are evaluated using system-specific analysis.

3.3.1.5.2 Group 1 Initiating Events

The methodology used to develop the distributions for the frequencies of these initiating events is similar to the two-stage approach used for component failure rates. The details of the development of the generic frequencies and the compiled raw data are described in Reference 3.3.1-15. The initiating events defined for a generic PWR in Reference 3.3.1-15 are directly relevant to the Watts Bar PRA. Table 3.3.1-4 provides the main characteristics of the initiating events frequency distributions.

3.3.1.5.3 Group 2 Initiating Events

The initiating events analysis of Section 3.1.1 identifies support systems that will cause a plant trip when the system fails. Failure of these systems will also impact other support and frontline systems. These are:

- LASD Loss of 6.9-kV Shutdown Board 1A-A (not a reactor trip; evaluated as a precursor)
- LBSD Loss of 6.9-kV Shutdown Board 1B-B (not a reactor trip; evaluated as a precursor)
- LDAAC Loss of 120V Vital AC Board 1-I
- LDBAC Loss of 120V Vital AC Board 1-II

- LDCAC Loss of 120V Vital AC Board 1-III
- LDDAC Loss of 120V Vital AC Board 1-IV
- LVBB1 Loss of 125V Vital Battery Board I
- LVBB2 Loss of 125V Vital Battery Board II
- CCSA Loss of CCS Train A
- CCSTL Total Loss of CCS
- ERCWA Loss of ERCW Train A
- ERCWB Loss of ERCW Train B
- ERCWTL Total loss of ERCW System

The frequencies of each of these events are estimated through system analysis and are presented in the appendices of their respective system notebooks. Table 3.3.1-4 provides the main characteristics of these distributions.

In addition to these system-specific initiating events, loss of control air is also an initiating event for Watts Bar. Loss of control air results in the MSIVs and the main feedwater regulating valves failing in the closed position. On examining the data for "Inadvertent Closure of All MSIVs" in Reference 3.3.1-15, it was noted that, in addition to events caused by human error during MSIV testing, this category included events caused by loss of air. Therefore, loss of control air is not developed as a separate initiating event. The Watts Bar model for the initiating event "Inadvertent Closure of All MSIVs" accounts for the unavailability of the MSIVs for steam dump to the condenser and for the unavailability of the main feedwater system due to closure of the regulating valves.

3.3.1.6 References

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Table 3.3.1-1 (Page 1 of 8). Component Failure Data for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WFBAPR	Boric Acid Transfer Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFBAPS	Boric Acid Transfer Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFBATR	Batteries Fail during Operation	7.53-07	6.36-08	3.79-07	1.64-06
WFBCHR	Battery Chargers Fail during Operation	1.86-05	9.80-07	8.25-06	5.38-05
WFBR1C	Circuit Breaker (480V AC and Above) Fail To Close on Demand	1.61-03	2.68-04	1.07-03	3.40-03
WFBR1O	Circuit Breaker (480V AC and Above) Fail To Open on Demand	6.49-04	6.48-05	3.65-04	1.40-03
WFBR1T	Circuit Breakers (480V AC and Above) Transfer Open	8.28-07	5.48-08	3.79-07	2.28-06
WFBR2C	Circuit Breakers (Less Than 480V) Fail To Close on Demand	2.27-04	8.54-06	8.54-05	8.54-04
WFBR2O	Circuit Breakers (Less Than 480V) Fail To Open on Demand	8.39-04	3.15-05	3.15-04	3.15-03
WFBR2T	Circuit Breakers (Less Than 480V) Transfer Open	2.68-07	2.99-08	1.28-07	8.69-07
WFBUSR	Buses/MCC Fail during Operation	4.98-07	8.74-08	3.40-07	1.13-06
WFCACR	Air Compressors Fail during Operation	9.81-05	9.84-06	4.98-05	2.40-04
WFCACS	Air Compressors Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFCADR	Compressed Air Dryer Fails during Operation	9.11-05	1.41-06	2.19-05	2.76-04
WFCBPR	Condensate Booster Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFCBPS	Condensate Booster Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFCCPR	Component Cooling Water Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFCCPS	Component Cooling Water Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFCDPR	Condensate Transfer Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFCDPS	Condensate Transfer Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

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Table 3.3.1-1 (Page 2 of 8). Component Failure Data for Watts Bar Components					
Designator	**Description**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
WFCSPR	Containment Spray Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFCSPS	Containment Spray Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFCTPR	Centifugal Charging Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFCTPS	Centrifugal Charging Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFDGG1	Diesel Generators Fail during First Hour of Operation	1.70-02	1.07-03	8.24-03	5.36-02
WFDGG2	Diesel Generators Fail After First Hour of Operation	2.51-03	2.97-04	1.49-03	7.44-03
WFDGGS	Diesel Generators Fail To Start	2.14-02	2.50-02	1.35-02	6.44-02
WFDIPR	De-Ionized Water Pumps Fail during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFDIPS	De-Ionized Water Pumps Fail To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFEWPR	ERCW Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFEWPS	ERCW Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFFN1R	Large Fans Fail during Operation	7.88-06	1.55-06	6.23-06	1.58-05
WFFN1S	Large Fans Fail To Start	2.93-03	3.27-04	1.66-03	8.35-03
WFFN2R	Room Ventilation Fans Fail during Operation	7.89-06	1.55-06	6.23-06	1.58-05
WFFN2S	Ventilation Fans Fail To Start	4.84-04	6.00-05	3.00-04	1.50-03
WFFTPR	Fuel Oil Transfer Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFFTPS	Fuel Oil Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFHWPR	Hotwell Pumps Fail during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFHWPS	Hotwell Pumps Fail To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFHXD1	De-Ionized Water Cooled Heat Exchangers (Large) Rupture/Leak during Operation	1.95-06	2.21-07	1.32-06	5.18-06

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

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Table 3.3.1-1 (Page 3 of 8). Component Failure Data for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WFHXD2	De-Ionized Water Cooled Heat Exchangers (Small) Rupture/Leak during Operation	1.95-06	2.21-07	1.32-06	5.18-06
WFHXRL	ERCW Heat Exchangers Rupture/Leak during Operation	1.95-06	2.21-07	1.32-06	5.18-06
WFINVR	Inverters Fail during Operation	1.83-05	1.60-06	1.13-05	4.37-05
WFMAPR	Motor-Driven AFW Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFMAPS	Motor-Driven AFW Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFMFPR	Turbine-Driven MFW Pump Fails during Operation	1.03-03	6.53-05	4.21-04	3.01-03
WFMFPS	Turbine-Driven MFW Pump Fails To Start	3.31-02	6.05-03	2.45-02	8.25-02
WFPWPR	Primary Water Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFPWPS	Primary Water Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFRHPR	RHR Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFRHPS	RHR Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFRTBD	Reactor Trip Breakers Fail To Open On Demand	1.77-03	4.14-04	1.33-03	3.72-03
WFSFPR	Standby MFW Pump Fails during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFSFPS	Standby MFW Pump Fails To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFSIPR	Safety Injection Pumps Fail during Operation	3.42-05	2.68-06	1.77-05	9.32-05
WFSIPS	Safety Injection Pumps Fail To Start	3.29-03	2.01-04	1.63-03	1.12-02
WFTBPR	Thermal Barrier Booster Pump Fails during Operation	3.36-05	2.03-06	1.59-05	9.83-05
WFTBPS	Thermal Barrier Booster Pump Fails To Start	2.35-03	2.47-04	1.45-03	7.38-03
WFTPPR	Turbine-Driven AFW Pump Fails during Operation	1.03-03	6.53-05	4.21-04	3.01-03
WFTPPS	Turbine-Driven AFW Pump Fails To Start	3.31-02	6.05-03	2.45-02	8.25-02
WFTSTD	Turbine Stop/Control Valves Fail To Close On Demand	1.25-04	3.33-05	1.00-04	3.00-04

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

Table 3.3.1-1 (Page 4 of 8). Component Failure Data for Watts Bar Components					
Designator	**Description**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
WFTSTR	Turbine Stop/Control Valves Transfer Closed	2.88-05	1.08-06	1.08-05	1.08-04
WFXR1R	Transformers (4.16-kV and Above) Fail during Operation	1.56-06	2.66-07	1.05-06	3.57-06
WFXR2R	Transformers (4.16-kV to 480V) Fail during Operation	6.87-07	1.05-07	4.47-07	1.37-06
WFXR3R	Transformers, Instrument (480V to 120V) Fail during Operation	1.55-06	7.94-08	7.00-07	4.87-06
ZEICEA	Ice Condenser Available after Initiating Event	1.00-06	3.51-08	3.65-07	3.64-06
ZTBATD	Batteries Fail To Operate on Demand	5.19-04	6.26-05	3.46-04	1.16-03
ZTCISL	Containment Buildings - Large Preexisting Leak	1.44-03	1.76-05	4.80-04	4.24-03
ZTCISS	Containment Buildings - Small Preexisting Leak	3.80-03	1.26-04	1.65-03	1.08-02
ZTCRAD	Single Scram Rod (PWR) - Fail on Demand	3.20-05	1.23-06	9.81-06	9.18-05
ZTDAOD	Dampers, Pneumatic - Fail on Demand	1.52-03	2.37-04	1.08-03	3.32-03
ZTDAOT	Dampers, Pneumatic - Transfer Open or Closed	2.67-07	1.50-08	1.10-07	8.06-07
ZTDBDD	Backdraft Damper - Fail To Open on Demand	2.69-04	5.33-05	1.44-04	6.27-04
ZTDBDP	Backdraft Damper - Transfer Closed or Plugged	1.04-08	2.78-09	8.33-09	2.50-08
ZTDHOT	Dampers, Manual - Transfer Open or Closed	4.20-08	1.57-09	1.30-08	1.19-07
ZTDMOD	Dampers, Motor-Operated - Fail on Demand	4.30-03	7.49-04	2.84-03	1.05-02
ZTDMOT	Dampers, Motor-Operated - Transfer Open or Closed	9.27-08	9.65-09	5.05-08	2.33-07
ZTDRYP	Air Dryer Fails during Operation	9.11-05	1.41-06	2.19-05	2.76-04
ZTEXJL	Expansion Joint Leaks/Ruptures during Operation	2.66-06	9.33-08	9.70-07	9.68-06
ZTFA1P	Filters, Air - Plug during Operation	5.83-06	2.04-07	2.12-06	2.12-05
ZTFA2P	Filters, Oil Removal - Plug during Operation	1.76-05	6.16-07	6.40-06	6.39-05
ZTFA3P	Filters, Compressed Air - Plug during Operation	3.54-05	1.15-06	1.13-05	1.08-04

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

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Table 3.3.1-1 (Page 5 of 8). Component Failure Data for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
ZTFL1P	Filters, Ventilation - Plug during Operation	1.07-06	4.00-08	4.00-07	4.00-06
ZTFU1R	Fuses - Fail Open	9.20-07	2.67-08	2.91-07	2.56-06
ZTHTRR	Heaters/Heat Tracing Lines - Fail during Operation	8.80-06	2.24-07	2.76-06	3.26-05
ZTHXRB	Heat Exchangers - Rupture/Leak during Operation	1.95-06	2.21-07	1.32-06	5.18-06
ZTLC1D	Trip Logic Modules - Fail on Demand	8.52-05	3.20-06	3.20-05	3.20-04
ZTLC1R	Trip Logic Modules - Fail during Operation	2.70-06	1.10-07	1.10-06	1.10-05
ZTLOCV	Loss Of Condenser after Initiating Event on Demand	1.26-01	4.87-02	1.11-01	2.46-01
ZTMGSR	Motor Generators - Fail during Operation	3.59-05	9.60-07	1.10-05	1.20-04
ZTOPFW	Feedwater Unavailability after ATWS; Power > 40% on Demand	5.00-03	1.22-04	3.24-03	1.32-02
ZTPBSD	Switches, Pushbutton - Fail on Demand	2.40-05	8.92-07	8.98-06	8.68-05
ZTPP1B	Pipes (Greater Than 3-Inch Diameter) Rupture/Plug during Operation	8.60-10	3.00-12	9.49-11	3.00-09
ZTPP2B	Pipes (Less Than 3-Inch Diameter) Rupture/Plug during Operation	8.60-09	3.00-11	9.49-10	3.00-08
ZTPS1R	Power Supplies - Fail during Operation	1.71-05	1.03-06	7.60-06	4.90-05
ZTPSHR	Power Supplies (+120V DC ESFAS) - Fail during Operation	1.33-04	5.00-06	5.00-05	5.00-04
ZTPSLR	Power Supplies (+5V or +25V DC ESFAS) - Fail during Operation	5.33-05	2.00-06	2.00-05	2.00-04
ZTRL1D	Relays - Fail on Demand	2.41-04	1.39-05	1.10-04	7.47-04
ZTRL1R	Relays - Fail during Operation	4.20-07	2.39-08	1.98-07	1.31-06
ZTSC1P	Strainers, Service Water - Plug during Operation	6.22-06	8.08-07	3.90-06	1.58-05
ZTSC3P	Traveling Screens, ERCW - Plug during Operation	6.22-06	8.08-07	3.90-06	1.58-05
ZTSEQD	ECCAS/LOP Sequencer - Fail on Demand	2.94-06	4.74-07	1.90-06	6.76-06

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

Table 3.3.1-1 (Page 6 of 8). Component Failure Data for Watts Bar Components					
Designator	**Description**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
ZTSMDR	Signal Modifiers - Fail during Operation	2.94-06	4.74-07	1.90-06	6.76-06
ZTSMP1	Containment Sump Plugged/LOCA-Type Events per Demand	8.26-05	2.89-06	3.01-05	3.00-04
ZTSPNP	Nozzles, Containment Building Spray (One Train) - Plug during Operation	7.06-08	3.49-09	3.00-08	2.58-07
ZTSTCD	Reactor Trip Breaker Shunt Trip Coil Fail To Open on Demand	1.40-04	3.27-05	1.05-04	2.94-04
ZTSWBD	Bistables - Fail on Demand	3.89-07	7.08-08	2.76-07	1.08-06
ZTSWBI	Bistables - Spurious Operation	2.21-06	4.00-09	1.68-07	7.00-06
ZTSWLD	Switches, Level - Fail on Demand	2.69-04	1.15-05	1.09-04	9.37-04
ZTSWPD	Switches, Pressure - Fail on Demand	2.69-04	1.15-05	1.09-04	9.37-04
ZTSWTD	Switches, Temperature - Fail on Demand	2.69-04	1.15-05	1.09-04	9.37-04
ZTTK1B	Tanks, Storage - Rupture/Leak during Operation	2.66-08	1.00-09	1.00-08	1.00-07
ZTTM1X	Temperature Monitor Loops - No Output	3.41-06	3.39-08	6.68-07	1.26-05
ZTTRFR	Transmitters, Flow - Fail during Operation	6.25-06	6.04-07	4.39-06	1.40-05
ZTTRLR	Transmitters, Level - Fail during Operation	1.57-05	3.96-06	1.26-05	3.34-05
ZTTRPR	Transmitters, Pressure - Fail during Operation	7.60-06	8.90-07	4.70-06	1.96-05
ZTTRTR	Transmitters, Temperature - Fail during Operation	1.57-05	3.96-06	1.26-05	3.34-05
ZTUVCD	Reactor Trip Breaker Undervoltage Coil Fail To Open on Demand	2.75-03	6.43-04	2.06-03	5.77-03
ZTV3WD	Valves, Pressure Control (Three-Way) - Fail on Demand	1.52-03	2.37-04	1.08-03	3.32-03
ZTVAOD	Valves, Air-Operated - Fail on Demand	1.52-03	2.37-04	1.08-03	3.32-03
ZTVAOF	Valves, Air-Operated - Fail To Transfer To Failed Position on Demand	2.66-04	1.00-05	1.00-04	1.00-03
ZTVAOT	Valves, Air-Operated - Transfer Open or Closed	2.67-07	1.50-08	1.10-07	8.06-07
ZTVCOD	Valves, Check (Other Than Stop Valves) - Fail on Demand	2.69-04	5.33-05	1.44-04	6.27-04

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

Table 3.3.1-1 (Page 7 of 8). Component Failure Data for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
ZTVCOL	Valves, Check (Other Than Stop) - Gross Reverse Leakage during Operation	5.36-07	9.23-08	3.17-07	1.26-06
ZTVCOP	Valves, Check (Other Than Stop) - Transfer Closed or Plugged	1.04-08	2.78-09	8.33-09	2.50-08
ZTVCSD	Valves, Stop Check Valves - Fail on Demand	9.13-04	7.07-05	4.14-04	2.61-03
ZTVCSL	Valves, Stop Check Valves - Gross Reverse Leakage	5.36-07	9.23-08	3.17-07	1.26-06
ZTVCSP	Valves, Stop Check Valves - Transfer Closed or Plugged	1.04-08	2.78-09	8.33-09	2.50-08
ZTVE1D	Valves, Electrohydraulic (Except TSV, TCV) - Fail on Demand	1.52-03	2.37-04	1.08-03	3.32-03
ZTVE1T	Valves, Electrohydraulic (Except TSV, TCV) - Transfer Open or Closed	2.67-07	1.50-08	1.10-07	8.06-07
ZTVHOT	Valves, Manual - Transfer Open or Closed	4.20-08	1.57-09	1.30-08	1.19-07
ZTVMCX	Disc - Check Valve or Motor-Operated Valve - Rupture during Operation	1.55-08	1.40-10	2.87-09	5.87-08
ZTVMOD	Motor-Operated Valves Fail on Demand	4.30-03	7.49-04	2.84-03	1.05-02
ZTVMOE	Motor-Operated Valves Fail To Close While Showing Closed	1.07-04	2.10-05	7.47-05	3.07-04
ZTVMOT	Motor-Operated Valves Transfer Open or Closed	9.27-08	9.65-09	5.05-08	2.33-07
ZTVR1O	Valves, Safety - Fail To Open on Demand	3.28-04	1.21-05	1.19-04	1.06-03
ZTVR1S	Valves, Safety - Fail To Reseat after Steam Relief on Demand	2.87-03	7.66-04	2.30-03	6.90-03
ZTVR1W	Valves, Safety - Fail To Reseat after Water Relief on Demand	1.00-01	3.45-03	8.37-02	3.33-01
ZTVR2O	Valves, Relief (Except PORVs or Safety) - Fail To Open on Demand	2.42-05	9.95-07	9.49-06	9.04-05
ZTVR2T	Valves, Relief (Except PORVs or Safety) - Premature Open during Operation	6.06-06	9.76-07	4.01-06	1.44-05
ZTVR3C	Valves, Relief (Power-Operated) - Fail To Reseat on Demand	2.50-02	6.66-03	2.00-02	6.00-02
ZTVR3O	Valves, Relief (Power-Operated) - Fail To Open on Demand	4.27-03	1.14-03	3.42-03	1.03-02
ZTVSOD	Valves, Solenoid - Fail on Demand	2.43-03	9.95-05	9.49-04	9.04-03
ZTVSOT	Valves, Solenoid - Transfer Open or Closed	1.27-06	5.19-08	4.91-07	4.07-06

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

Table 3.3.1-1 (Page 8 of 8). Component Failure Data for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
ZTVSRD	Self-Regulated PCV Fails To Operate on Demand	2.66-04	1.00-05	1.00-04	1.00-03
ZTVSRT	Self-Regulated PCV Transfers Open or Closed	3.36-06	1.18-07	1.22-06	1.22-05
ZTVTCD	Valves, Temperature Control (Butterfly) - Fail on Demand	1.52-03	2.37-04	1.08-03	3.32-03
ZTVTCF	Valves, Temperature Control (Butterfly) Fail To Transfer To Failed Position	2.66-04	1.00-05	1.00-04	1.00-03
ZTVTCT	Valves, Temperature Control (Butterfly) - Transfer Open or Closed	4.20-08	1.57-09	1.30-08	1.19-07

Note: 1. Exponential notation is indicated in abbreviated form; e.g., 3.36-05 = 3.36×10^{-05}.
2. Failures during operation and transfer open or close are failure frequency per hour; all others are failure frequency per demand.

Table 3.3.1-2 (Page 1 of 2). Component Maintenance Frequency Data for Watts Bar Components (events per hour)

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WMBAPF	Boric Acid Transfer Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMBATF	Batteries	2.49-05	3.87-06	1.41-05	4.14-05
WMBTCF	Battery Chargers	2.49-05	3.87-06	1.41-05	4.14-05
WMBUSF	Buses	2.66-06	1.29-07	9.86-07	7.04-06
WMCACF	Air Compressors	2.93-04	1.22-05	1.06-04	7.85-04
WMCBPF	Condensate Booster Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMCCPF	Component Cooling Water Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMCDPF	Condensate Transfer Pumps	1.17-04	7.96-06	4.52-05	3.27-04
WMCSPF	Containment Spray Pumps	1.17-04	7.96-06	4.52-05	3.27-04
WMCTPF	Centrifugal Charging Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMDGGF	Diesel Generators	1.03-03	1.65-04	5.99-04	2.13-03
WMDIPF	De-Ionized Water Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMEWPF	ERCW Pumps	3.35-04	2.64-05	1.39-04	8.46-04
WMFN1F	Large Fans	1.47-04	3.85-06	4.03-05	4.05-04
WMFN2F	Small Fans	2.09-04	8.85-06	7.13-05	5.74-04
WMFTPF	Fuel Oil Transfer Pumps	1.17-04	7.96-06	4.52-05	3.27-04
WMHWPF	Hotwell Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMHXRF	Heat Exchangers	4.15-05	2.38-06	1.62-05	1.12-04
WMINVF	Inverters	2.49-05	3.87-06	1.41-05	4.14-05
WMMAPF	Motor-Driven AFW Pumps	1.17-04	7.96-06	4.52-05	3.27-04
WMMFPF	Main Feedwater Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMPWPF	Primary Water Pumps	1.58-04	1.29-05	7.35-05	3.87-04
WMRHPF	RHR Pumps	1.17-04	7.96-06	4.52-05	3.27-04
WMSC1F	ERCW Strainers	9.27-05	5.33-06	3.69-05	2.27-04
WMSFPF	Standby MFW Pump	1.17-04	7.96-06	4.52-05	3.27-04
WMSIPF	Safety Injection Pumps	1.17-04	7.96-06	4.52-05	3.27-04

Note: Exponential notation is indicated in abbreviated form; e.g., 1.58-04 = 1.58×10^{-04}.

Table 3.3.1-2 (Page 2 of 2). Component Maintenance Frequency Data for Watts Bar Components (events per hour)

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WMTPPF	Turbine-Driven AFW Pump	4.19-04	5.99-05	2.41-04	8.89-04
WMVLVF	Valves	2.74-05	3.94-06	1.41-05	5.72-05
WMXFRF	Transformers	4.40-06	1.21-07	1.26-06	1.25-05
ZMSC1F	River Water Strainers	9.27-05	5.33-06	3.69-05	2.27-04

Note: Exponential notation is indicated in abbreviated form; e.g., 1.58-04 = 1.58×10^{-04}.

Table 3.3.1-3 (Page 1 of 3). Component Maintenance Duration Data for Watts Bar Components (hours)

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WMBATD	Batteries	3.85+01	1.37+00	1.37+01	1.17+02
WMBS1D	Buses, Boards (No Technical Specifications)	3.85+01	1.37+00	1.37+01	1.17+02
WMBS2D	Buses, Boards (8-Hour Technical Specifications)	5.56+00	3.20+00	5.68+00	7.50+00
WMBTCD	Battery Chargers	3.85+01	1.37+00	1.37+01	1.17+02
WMCACD	Air Compressor (72-Hour Technical Specifications)	1.31+01	7.84-01	6.01+00	4.04+01
WMCCPD	Component Cooling Water Pumps	1.11+01	1.16+00	6.20+00	3.08+01
WMCSPD	Containment Spray Pumps	1.11+01	1.16+00	6.20+00	3.08+01
WMCDPD	Condensate Transfer Pumps	2.66+02	1.99+00	4.72+01	8.15+02
WMCTPD	Centrifugal Charging Pumps	1.11+01	1.16+00	6.20+00	3.08+01
WMDGGD	Diesel Generator	1.31+01	7.84-01	6.01+00	4.04+01
WMEWPD	ERCW Pumps	1.11+01	1.16+00	6.20+00	3.08+01
WMFN1D	Large Fans (72-Hour Technical Specifications)	1.31+01	7.84-01	6.01+00	4.04+01
WMFN2D	Small Ventilation Fans (72-Hour Technical Specifications)	1.31+01	7.84-01	6.01+00	4.04+01
WMFNND	Small Ventilation Fans (No Technical Specifications)	3.85+01	1.37+00	1.37+01	1.17+02
WMFTPD	Fuel Oil Transfer Pumps	1.11+01	1.16+00	6.20+00	3.08+01
WMHXRD	Heat Exchanger (72-Hour Technical Specifications)	1.31+01	7.84-01	6.01+00	4.04+01
WMMAPD	Motor-Driven AFW Pump	1.11+01	1.16+00	6.20+00	3.08+01
WMPWPD	Primary Water Pump	2.66+02	1.99+00	4.72+01	8.15+02
WMRHPD	RHR Pump	1.11+01	1.16+00	6.20+00	3.08+01
WMSC1D	ERCW Strainer	1.31+01	7.84-01	6.01+00	4.04+01

Note: Exponential notation is indicated in abbreviated form; e.g., 3.85+01 = 3.85 × 10^{01}.

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Table 3.3.1-3 (Page 2 of 3). Component Maintenance Duration Data for Watts Bar Components (hours)

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WMSIPD	Safety Injection Pump	1.11+01	1.16+00	6.20+00	3.08+01
WMTBPD	Thermal Barrier Booster Pumps	2.66+02	1.99+00	4.72+01	8.15+02
WMTPPD	Turbine-Drivern AFW Pump	1.11+01	1.16+00	6.20+00	3.08+01
WMVLSD	Valves (168-Hour Technical Specifications)	1.89+01	1.54+00	1.01+01	5.13+01
WMVNSD	Valves (No Technical Specifications)	1.32+02	7.23-01	1.69+01	4.10+02
WMVSSD	Valves (24- and 72-Hour Technical Specifications)	4.05+00	6.83-01	2.70+00	9.52+00
WMXR1D	Transformers (No Technical Specifications)	3.85+01	1.37+00	1.37+01	1.17+02
WMXR2D	Transformers (8-Hour Technical Specifications)	5.56+00	3.20+00	5.68+00	7.50+00
ZMGNAD	Type A (Nonroutine Maintenance)	1.08+01	6.97+00	9.91+00	1.60+01
ZMGNBD	Type B (Nonroutine Maintenance)	2.09+01	1.31+01	2.02+01	2.88+01
ZMGNCD	Type C (Nonroutine Maintenance)	4.04+01	2.12+01	3.71+01	6.47+01
ZMGNDD	Type D (Nonroutine Maintenance)	1.16+02	7.46+00	9.52+01	2.91+02
ZMGNED	Type E	5.56+00	3.20+00	5.68+00	7.50+00
ZMGNFD	Type F	1.16+00	7.13-01	1.14+00	1.48+00
ZMGNGD	Type G	3.54+00	2.58+00	3.49+00	4.13+00
ZMGNHD	Type H	3.17+03	1.46+03	3.16+03	4.88+03
ZMHXND	Heat Exchangers (No Technical Specifications)	5.83+02	6.34+01	3.68+02	1.53+03
ZMOLSD	Other Equipment (Long Techical Specifications)	3.72+01	8.20+00	2.75+01	7.41+01
ZMOMSD	Other Equipment (48- and 72-Hour Technical Specifications)	1.31+01	7.84-01	6.01+00	4.04+01

Note: Exponential notation is indicated in abbreviated form; e.g., 3.85+01 = 3.85 × 10^{01}.

Table 3.3.1-3 (Page 3 of 3). Component Maintenance Duration Data for Watts Bar Components (hours)

Designator	Description	Mean	5th Percentile	Median	95th Percentile
ZMONSD	Other Equipment (No Technical Specifications)	3.85+01	1.37+00	1.37+01	1.17+02
ZMOSSD	Other Equipment (24-Hour Technical Specifications)	6.26+00	5.46-01	3.42+00	2.02+01
ZMPLSD	Pumps (168-Hour Technical Specifications)	2.87+01	2.58+00	1.57+01	7.27+01
ZMPMSD	Pumps (72 Hour Technical Specifications)	1.11+01	1.16+00	6.20+00	3.08+01
ZMPNSD	Pumps (No Technical Specifications)	2.66+02	1.99+00	4.72+01	8.15+02
ZMPSSD	Pumps (Short Technical Specifications)	7.47+00	1.24+00	5.43+00	1.82+01
ZMVLSD	Valves (Long Technical Specifications)	1.89+01	1.54+00	1.01+01	5.13+01
ZMVNSD	Valves (No Technical Specifications)	1.32+02	7.23-01	1.69+01	4.10+02
ZMVSSD	Valves (Short Technical Specifications)	4.05+00	6.83+01	2.70+00	9.52+00

Note: Exponential notation is indicated in abbreviated form; e.g., 3.85+01 = 3.85×10^{01}.

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Table 3.3.1-4. Summary Of Initiating Events And Precursors To Initiating Events For Watts Bar PRA

Designator	Description	Frequency (events per calendar year) Mean	5th Percentile	Median	95th Percentile
ZEEXL	Excessive LOCA	2.66-07	7.10-09	8.75-08	8.07-07
ZELLOC	Large LOCA	2.03-04	6.73-06	8.11-05	5.75-04
ZEMLOC	Medium LOCA	4.65-04	1.86-05	2.00-04	1.11-03
ZESLOC	Small LOCA, Nonisolable	5.83-03	1.14-04	1.80-03	1.65-02
ZESLI	Small LOCA, Isolable	2.30-02	4.12-04	8.73-03	4.84-02
ZESLBI	Steamline Break Inside Containment	4.65-04	1.86-05	2.00-04	1.11-03
ZESLBO	Steam Line Break Outside Containment	6.04-03	1.84-04	2.18-03	1.74-02
IMSRV	Inadvertent Opening of Main Steam Relief Valves	4.19-03	7.64-05	1.12-03	1.14-02
SGTR	Steam Generator Tube Rupture	2.84-02	2.06-04	5.91-03	8.66-02
OMSIV	Closure of One MSIV	8.66-02	6.31-03	4.46-02	2.22-01
AMSIV	Inadvertent Closure of All MSIVs	1.93-02	5.97-04	1.13-02	5.64-02
EXFW	Excessive Feedwater Flow	1.68-01	8.40-03	7.10-02	5.27-01
CPEX	Core Power Excursion	2.68-02	9.05-04	1.50-02	5.16-02
RT	Reactor Trip	1.35+00	4.09-01	1.07+00	2.79+00
PLMFW	Partial Loss of Main Feedwater	1.13+00	2.02-01	8.04-01	2.57+00
TLMFW	Total Loss of Main Feedwater	1.62-01	1.80-02	9.72-02	4.06-01
LOCV	Loss of Condenser Vacuum	1.18-01	1.90-02	8.36-02	2.64-01
LPF	Loss of Primary Flow	1.76-01	8.41-03	7.54-02	5.12-01
TT	Turbine Trip	1.07+00	3.80-01	9.21-01	1.85+00
ISI	Inadvertent Safety Injection Signal	2.99-02	2.98-04	7.81-03	8.77-02
WELOSP	Loss of Offsite Power Frequency*	8.56-02	6.07-03	4.19-02	2.46-01
LASD†	Loss Of 6.9-kV Shutdown Board 1A-A	3.03-03	4.68-04	2.07-03	7.09-03
LBSD†	Loss Of 6.9-kV Shutdown Board 1B-B	3.04-03	4.70-04	2.07-03	7.06-03
LDAAC†	Loss of 120V Vital AC Board 1-I	1.19-01	1.51-02	7.59-02	2.62-01
LDBAC†	Loss of 120V Vital AC Board 1-II	1.19-01	1.53-02	7.53-02	2.60-01
LDCAC†	Loss of 120V Vital AC Board 1-III**	1.16-01	1.64-02	7.48-02	2.50-01
LDDAC†	Loss of 120V Vital AC Board 1-IV**	1.14-01	1.58-02	7.51-02	2.55-01
LVBB1†	Loss of 125V Vital Battery Board I	5.98-03	1.03-03	3.30-03	1.35-02
LVBB2†	Loss of 125V Vital Battery Board II	5.79-03	1.00-03	3.23-03	1.31-02
CCSA†	Loss of CCS Train A	2.78-02	4.50-03	1.45-02	7.08-02
CCSTL†	Total Loss of CCS	1.11-03	2.47-05	2.84-04	3.09-03
ERCWA†	Loss of ERCW Train A	7.10-04	2.28-05	2.01-04	2.03-03
ERCWB†	Loss of ERCW Train B	7.10-04	2.28-05	2.01-04	2.03-03
ERCWTL†	Total Loss of ERCW System	1.51-05	1.46-07	2.39-06	3.95-05

*The frequency for loss of offsite power is developed as events per site year. The frequency includes events that may happen when the Watts Bar unit is shut down. The above value must be multipled by the availability factor for Watts Bar, = 0.70. The data sources are References 3.3.1-17 and 3.3.1-18.

**Reactor trip will occur on loss of board if that particular board was selected for steam generator level control.

†System-specific analysis. See system notebook appendices.

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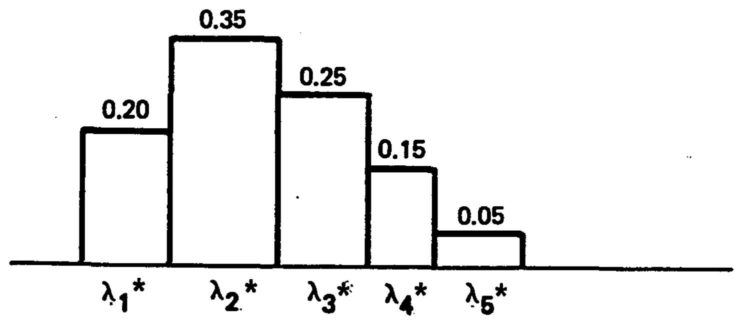


Figure 3.3.1-1. Population Variability of the Failure Rate

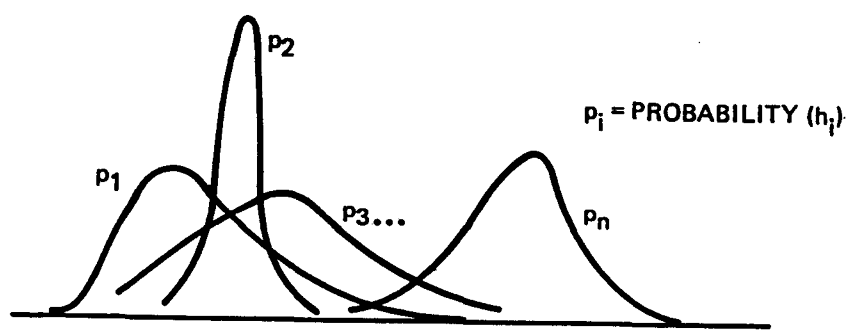


Figure 3.3.1-2. State-Of-Knowledge Distribution Over the Set of Frequency Distributions

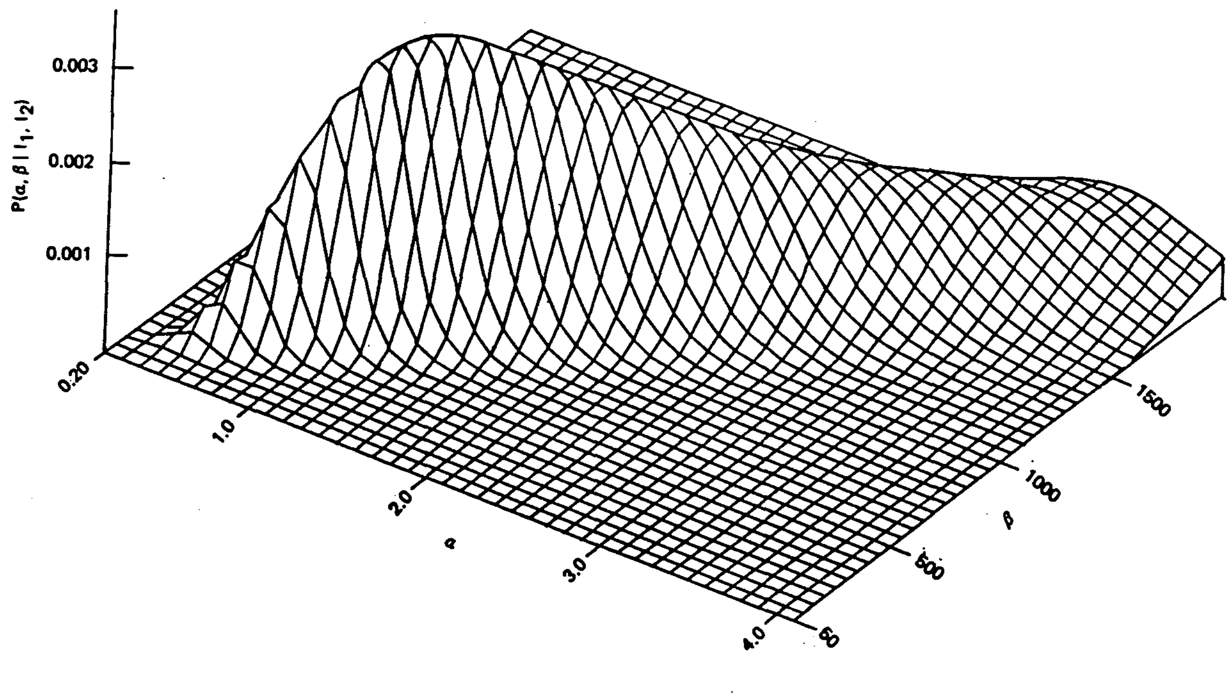


Figure 3.3.1-3. Posterior Distribution for the Parameters of the Distribution of Pump Failure To Start on Demand Rates

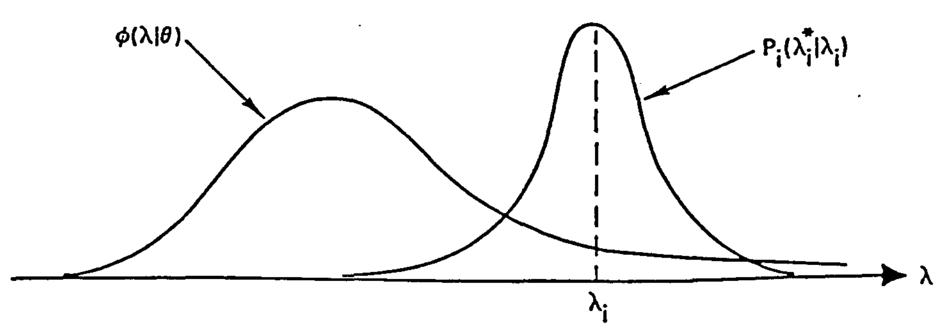


Figure 3.3.1-4. The Relation Between the Variability Curve and Uncertainty about Individual Estimates

EVIDENCE COLLECTED FOR FAILURE RATE				
SOURCE	DATA			
	NUMBER OF FAILURES	NUMBER OF DEMANDS	ESTIMATE	ASSIGNED RANGE FACTOR
TYPE 1				
PLANT A	10	1.65+3		
PLANT B	14	1.13+4		
PLANT C	7	1.73+3		
PLANT D	42	6.72+3		
PLANT E	3	1.26+3		
PLANT F	31	9.72+3		
TYPE 2				
EXPERT 1			1.00-3	5
EXPERT 2			5.60-3	3
EXPERT 3			1.00-3	10

NOTE: EXPONENTIAL NOTATION IS INDICATED IN ABBREVIATED FORM; e.g., 1.65+3 = 1.65 x 10^{+3}

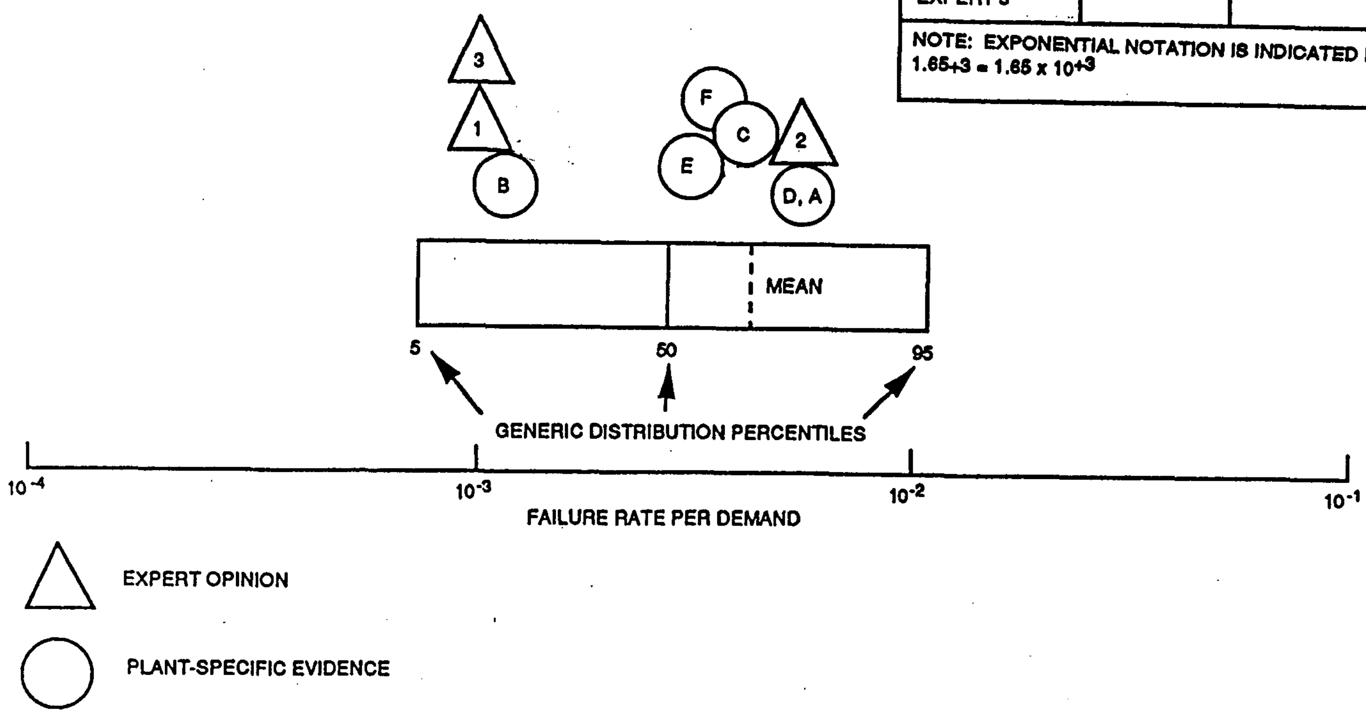


Figure 3.3.1-5. Application of RISKMAN To Develop Generic Distribution for MOV Failure Rates

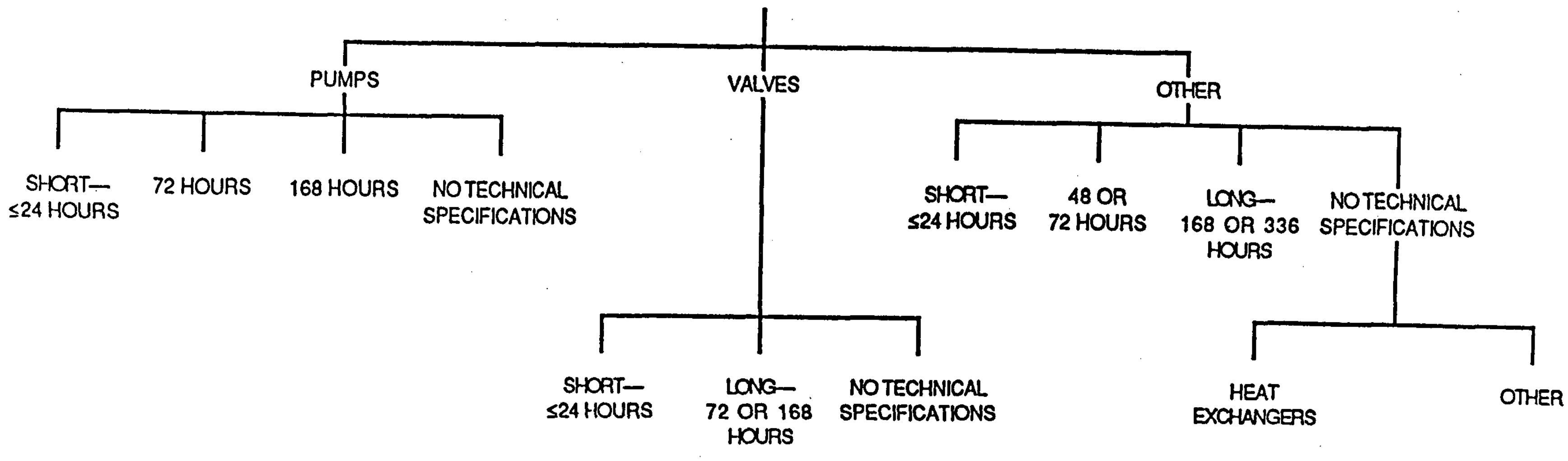


Figure 3.3.1-6. Categorization of Component Types for Generic Maintenance Frequency Distributions

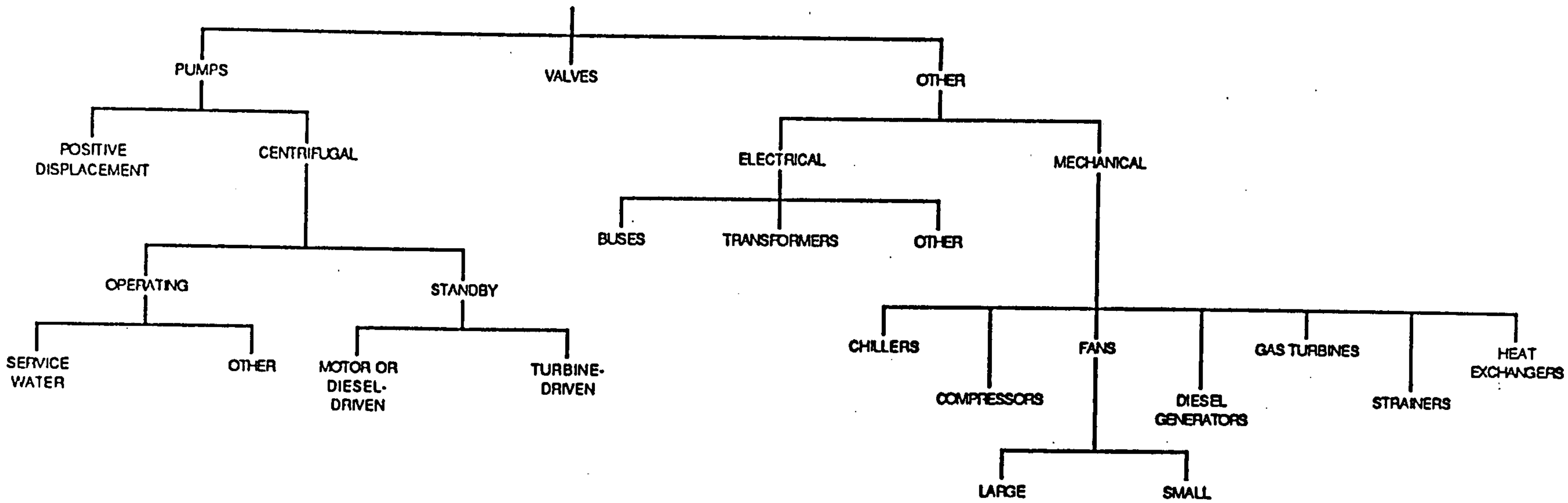


Figure 3.3.1-7. Categorization of Component by Technical Specification for Generic Mean Maintenance Duration Distributions

3.3.2 INCORPORATION OF PLANT-SPECIFIC EVIDENCE

3.3.2.1 Failure and Maintenance Frequency Update

When plant-specific data become available from the accumulation of Watts Bar operating experience, data specialization, or the development of plant-specific failure rate distributions (and maintenance and duration frequency distributions), is achieved by applying Bayes' theorem, as follows:

$$P(\lambda|E_2) = F^{-1} L(E_2|\lambda) P_0(\lambda) \tag{3.3.2.1}$$

where $P(\lambda|E_2)$ is the plant-specific failure rate distribution reflecting the plant-specific experience E_2, and the generic distribution $P_0(\lambda)$ is the prior state of knowledge about the failure rate of the component in question. The likelihood term, $L(E_2|\lambda)$, takes the form of a Poisson distribution when λ is the rate of failure per unit time and the evidence E_2 is k failures in T time units

$$P(k_1T|\lambda) = \frac{(\lambda T)^K}{k!} e^{-\lambda T} \tag{3.3.2.2}$$

If λ is a demand failure frequency and E_2 is k failures in D demands, then $L(E_2|\lambda)$ is a binomial distribution

$$P(k_1D|\lambda) = \frac{D!}{(D-k)!\,k!} (1-\lambda)^{D-k} \lambda^k \tag{3.3.2.3}$$

The magnitude of the effect of adding plant-specific data depends on the relative strength of the data compared with the prior level of confidence expressed in the form of the spread of the prior distribution. Typically, both the location and the spread of the posterior or updated distribution are affected by the plant-specific evidence. The mean value of the updated distribution could be higher or lower than the mean of the generic prior, but adding the plant-specific data normally reduces the spread of the distribution, as shown in the following example. The generic distribution for the MOV demand failure frequency presented in the example of the previous section was updated with 15 failures in 5,315 demands. Calculations were performed using RISKMAN. The following table compares some basic characteristics for the generic prior and updated distributions:

Distribution	Mean (per demand)	5th Percentile	Median	95th Percentile
Generic	4.27×10^{-3}	7.28×10^{-4}	2.96×10^{-3}	1.01×10^{-2}
Updated	2.88×10^{-3}	1.74×10^{-3}	2.70×10^{-3}	3.82×10^{-3}

Another example of how the two-stage Bayesian procedure employed by RISKMAN is used to incorporate plant-specific data is illustrated in Figure 3.3.2-1. In this example, for motor-operated valves, suppose that the plant-specific evidence revealed that there was 1 failure in 1,000 demands at the specific plant being analyzed. The RISKMAN analyst would call up the generic distribution for the failure mode that had been developed previously in the first-stage Bayesian procedure in Figure 3.3.1-5, and input the plant-specific evidence to produce the updated distribution, denoted in Figure 3.3.2-1 as "Posterior 1." As can be seen in this figure, the weight of this evidence pulls down the mean of the posterior distribution toward 1.0×10^{-3}, the point estimate of the plant-specific evidence.

One useful property of Bayes' theorem is that it automatically weights the respective roles of the prior distribution and the evidence according to the amount of evidence applied. So, for example, if five times as much data that happen to be consistent with a point estimate of 1.0×10^{-3} (i.e., 5 failures in 5,000 demands) were collected from the specific plant being analyzed, the updated distribution (Posterior 2) would become very peaked about the point estimate of the evidence such that the role of the prior distribution becomes unimportant. The use of this approach eliminates the need to make and to document difficult and arbitrary decisions about when to use generic and when to use plant-specific data. Even for a plant with much experience, there are insufficient data for some of the rare events that are important (e.g., small loss of coolant accident frequency) to eliminate the need for both sources of data.

Another useful property of Bayes' theorem is that it provides a consistent treatment of any type of evidence, even when that evidence is made up from experience data in which no failures were observed. Suppose that we are using Bayes' theorem to evaluate the failure rate of a pump, λ, at a specific plant that tests the pump N times and observes no failures. Using Bayes' theorem, the probability that the failure rate of the pump is equal to any particular value, say, $\lambda = \lambda^*$, is given by

$$p(\lambda^*|E) = F^{-1} L(E|\lambda^*)p_0(\lambda^*) \qquad (3.3.2.4)$$

where

$$F = \int_0^\infty L(E|\lambda)p_0(\lambda)d\lambda$$

$L(E|\lambda^*)$ = likelihood of observing evidence E, given that the failure rate is λ^*.

If we are quantifying a demand-based failure rate, the appropriate likelihood function is the binomial distribution. If the failure rate on demand is λ, the likelihood of observing exactly K failures in N demands is

$$L(k \text{ failures in N demands}) = \binom{N}{k} \lambda^k (1-\lambda)^{N-k} \qquad (3.3.2.5)$$

So, for zero failures in N demands,

$$L(0 \text{ failures in N demands}) = (1-\lambda)^N \qquad (3.3.2.6)$$

This likelihood function is plotted in Figure 3.3.2-2 for different values of N and λ. To see how Bayes' theorem works for this kind of evidence, assume that λ can take on only one of five discrete values: {1, .03, .01, .003, or .001} and that the prior distribution is uniform over these values; i.e., a "flat distribution." Application of Bayes' theorem for zero failures in N demands is illustrated in Table 3.3.2-1. As can be seen in this table, the posterior distribution is heavily influenced by the prior distribution for N = 10 demands, indicating rather weak evidence. However, for N = 1,000 demands, the posterior essentially vanishes for values of λ in excess of 3×10^{-3} because of the influence of the likelihood function. Thus, zero failures does not pose any problems for the Bayesian approach, and the results are a strong function of the quantity of evidence; i.e., the number of successful demands.

3.3.2.2 Maintenance Duration Update

To use the RISKMAN software to update maintenance duration distributions, the raw plant-specific data have to be processed to a form that is compatible with event frequency updating formulae of Section 3.3.1.2. The plant data must be converted to equivalent values of a number of events, k, and a time period, T. To create these pseudo-data, we make use of the fact that, according to the Poisson model for component failures, the mean value of a failure rate, λ, can be estimated by

$$\hat{\lambda} = \frac{k}{T} \tag{3.3.2.7}$$

where k is the number of failures observed in T time units. The variance of this estimator is given by

$$\text{Var}\left(\hat{\lambda}\right) = \text{Var}\left(\frac{k}{T}\right) = \frac{1}{T^2} \bullet \text{Var}(k) \tag{3.3.2.8}$$

If k has a Poisson distribution, then

$$\text{Var}(k) = \lambda T \approx \hat{\lambda} T = k \tag{3.3.2.9}$$

so Equation (3.3.2.8) can be rewritten as

$$\text{Var}(\hat{\lambda}) = \frac{1}{T^2} \bullet \text{Var}(k) \approx \frac{k}{T^2} \tag{3.3.2.10}$$

By substituting $\overline{m}$ (the mean observed maintenance duration at Watts Bar for $\hat{\lambda}$ in Equations (3.3.2.7) and (3.3.2.10), we then compute appropriate values of k and T by solving the following equations:

$$\frac{k}{T} = \hat{\lambda} = \overline{m} \tag{3.3.2.11}$$

and

$$\frac{k}{T^2} = \text{Var}(\hat{\lambda}) = \text{Var}(\overline{m}) \tag{3.3.2.12}$$

In these equations, the mean observed maintenance duration, $\overline{m}$, is simply the average of the durations of all maintenance events for a particular type of component at Watts Bar:

$$\overline{m} = \frac{\left(\sum_{i=1}^{n} m_i\right)}{n} \tag{3.3.2.13}$$

where m_i is the actual duration of the ith maintenance event for the particular component type being considered and n is the number of such events observed at Watts Bar. The variance of $\overline{m}$ is given by

$$\begin{aligned} Var(\overline{m}) &= Var\left[\frac{\left(\sum_{i=1}^{n} m_i\right)}{n}\right] = \frac{1}{n^2} \bullet Var\left(\sum_{i=1}^{n} m_i\right) \\ &= \frac{1}{n^2} \bullet \sum_{i=1}^{n} Var(m_i) = \frac{1}{n} \bullet \sigma_m^2 \end{aligned} \tag{3.3.2.14}$$

where σ_m^2 is the observed variance of the maintenance duration $m_1, m_2, \dots m_n$.

Now, solving for k and T in Equations (3.3.2.9) and (3.3.2.10), we get:

$$T = \frac{\overline{m}}{Var(\overline{m})} = \frac{\overline{m}}{1/n \bullet \sigma_m^2} = \frac{n \bullet \overline{m}}{\sigma_m^2} \tag{3.3.2.15}$$

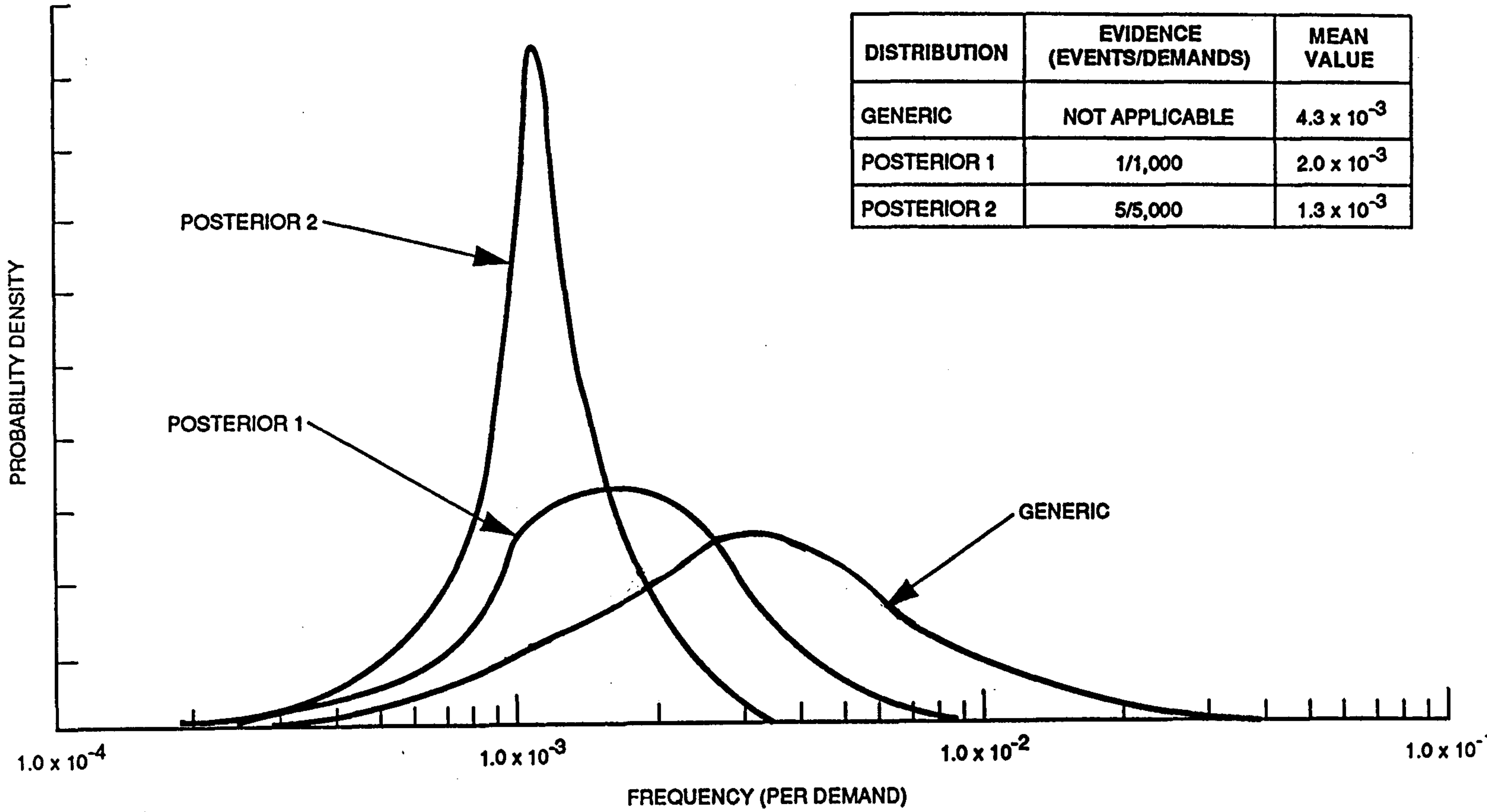
and

$$k = T \bullet \overline{m} = \frac{n \bullet (\overline{m})^2}{\sigma_m^2} \tag{3.3.2.16}$$

The values given by Equations (3.3.2.15) and (3.3.2.16) can be used in a Poisson likelihood function to update generic maintenance durations.

Table 3.3.2-1. Application of Bayes' Theorem for Case of Zero Failures

λ	Prior Distribution $p_0(\lambda)$	Binomial Likelihood Function for Zero Failures $(1 - \lambda)^N$			Posterior Distribution $p(\lambda \mid 0$ failure in N demands)		
		N = 10	N = 100	N = 1,000	N = 10	N = 100	N = 1,000
.1	.2	.35	2.6×10^{-5}	1.8×10^{-46}	.088	1.3×10^{-5}	4.2×10^{-46}
.03	.2	.74	0.47	5.9×10^{-14}	.187	.023	1.4×10^{-13}
.01	.2	.90	.37	4.3×10^{-5}	.229	.178	1.0×10^{-4}
.003	.2	.97	.74	.049	.246	.36	.12
.001	.2	.99	.90	.37	.251	.44	.88



DISTRIBUTION	EVIDENCE (EVENTS/DEMANDS)	MEAN VALUE
GENERIC	NOT APPLICABLE	4.3×10^{-3}
POSTERIOR 1	1/1,000	2.0×10^{-3}
POSTERIOR 2	5/5,000	1.3×10^{-3}

Figure 3.3.2-1. Updating Generic Distributions with Plant-Specific Evidence Using RISKMAN

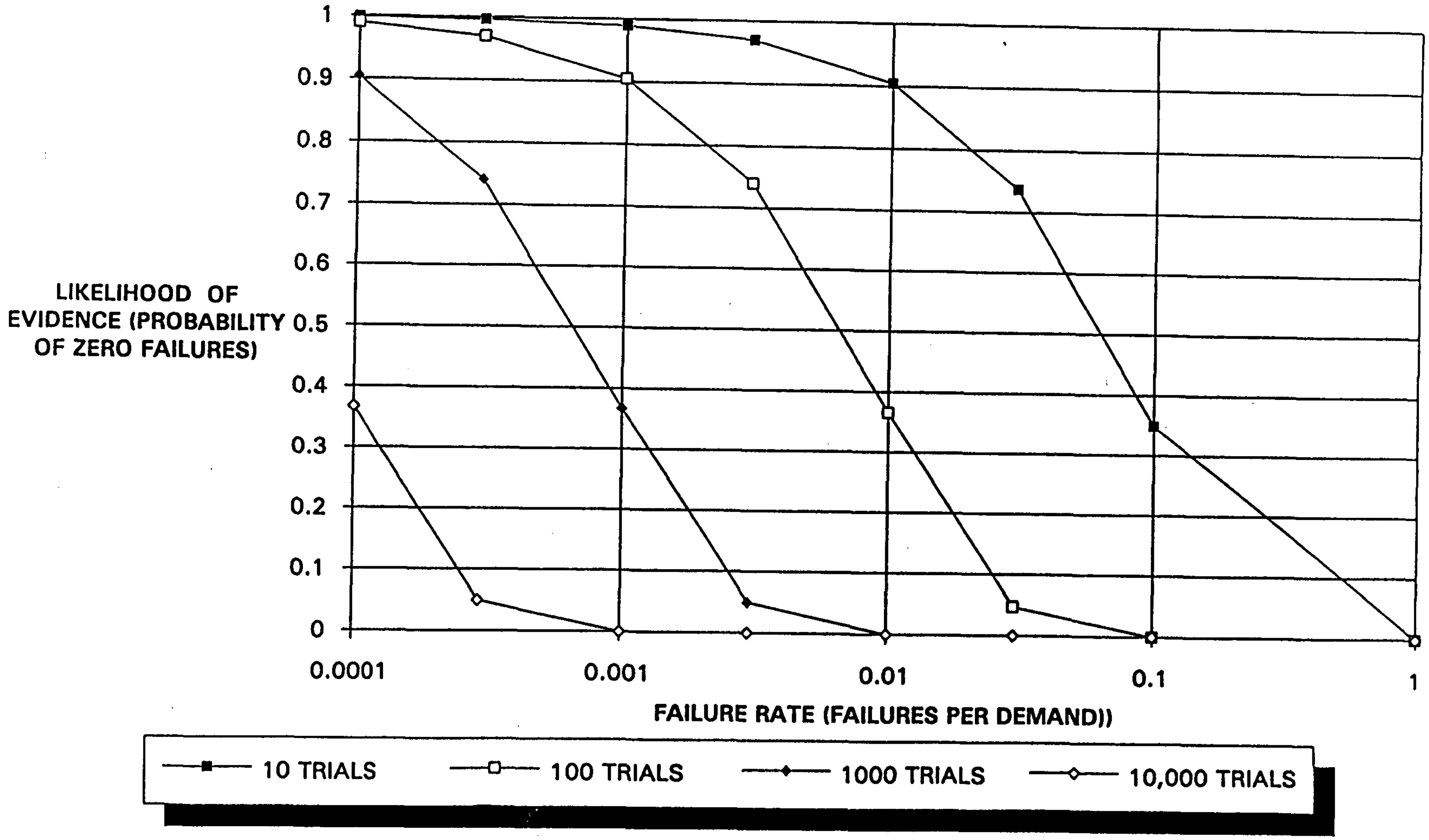


Figure 3.3.2-2. Treatment of Zero Failures Using Binomial Likelihood Function

3.3.3 HUMAN ACTION DATA

This section summarizes the results of the human actions evaluations performed in the Watts Bar Nuclear Plant Probabilistic Risk Assessment (PRA). Section 3.3.3.1 addresses errors made during normal maintenance and testing that leave systems unavailable to perform their safety functions if an initiating event should occur. Section 3.3.3.2 presents the evaluation of errors by the operating crew as they dynamically respond to the plant conditions during the sequence of events following an initiating event. Finally, Sections 3.3.3.3 and 3.3.3.4 address their actions to recover functions through alternate alignments or restoration of failed systems to service. Section 3.3.3.3 addresses those actions identified by a review of the quantification of the plant model, while Section 3.3.3.4 addresses recovery of electric power. The human actions that can initiate plant transients are implicitly included in the initiating event frequencies.

A summary of the methodology used to accomplish the assessment of individual actions is presented in Section 2.3.5.2. A detailed description of the methodology, summaries of the actions evaluated, and calculation packages leading to the results are presented in Appendix B. Section 2.3.5.3 summarizes the electric power recovery methodology, while Section 3.3.3.4 presents the electric power recovery model in its entirety.

3.3.3.1 Results of Routine Action Evaluation

3.3.3.1.1 Summary

The routine action evaluation addresses errors made during normal maintenance and testing that leave systems in an undetected unavailable state that prevents them from performing their safety functions should an initiating event occur. As a surveillance test is performed to verify operability after maintenance, the screening process addresses the surveillance tests accomplished on each system. Each system analyst follows the guidance contained in Appendix B, Section B.2, to identify tests that have a potential to leave components in an undetected unavailable state. The analyst quantifies the unavailability created by those tests that do not meet the screening criteria listed in Section B.2 using the screening error rates summarized in Table 3.3.3-1 for evaluation of risk significance in the plant model. These error rates are derived for an average restoration within a group of typical testing situations following the methodology developed in NUREG/CR-1278 (References 3.3.3.1 and 3.3.3-2).

Based on the results of the systems analyses, the surveillance tests summarized in Table 3.3.3-2 have sufficient enough potential to leave a component in an undetected unavailable state to warrant quantification for inclusion in the plant model. This table identifies the test, describes the error that could lead to the undetected unavailability, lists the impacted top events, and provides a reference to the system model where the test is discussed.

Of the errors listed in Table 3.3.3-2, the following were found to be potentially significant contributors to accident sequences important to risk:

- **Failure To Remove Refueling Cavity Drain Plugs Following Refueling.** When a screening error rate ZHERLL was used to model this error, the loss of inventory to cool the core in the recirculation mode was a dominant contributor to risk. A more detailed evaluation of the procedures addressing this task indicates that the actual failure rate, which is assigned a basic event variable name of WHESUM, is several orders of magnitude lower than the screening value. Consequently, this error is not a contributor to risk (see Section 3.3.3.1.2).

- No other routine human errors made during surveillance tests were found to be a significant contributor to risk.

3.3.3.1.2 Removal of Refueling Cavity Drain Plugs

During refueling, two drain plugs are placed in the refueling cavity to permit flooding for refueling operations. If both plugs remain installed when the plant is placed back in service, the ECCS is guaranteed to fail in the recirculation mode during accident sequences requiring containment pressure control. The water pumped from the containment sump by the containment spray system will be retained in the refueling cavity in the upper containment compartment, thus depleting the inventory available in the containment sump for further core and containment cooling. This would result in a failure of the emergency core cooling system and containment spray functions in the recirculation mode. As the screening value for human error to fail to restore the system to normal configuration following this temporary plant alteration leads to its appearance as a significant contributor to core melt, it is subjected to the more detailed analysis described in Appendix B, Section B.3.2, as summarized in Table 3.3.3-3.

The drain plugs are replaced by vortex eliminators following drainage of the refueling cavity and verified by at least three separate and distinct procedures. Considering the initial error rate and recovery factors outlined in Table 3.3.3-3, the estimated mean likelihood of the plugs remaining in place following a refueling outage is approximately 2×10^{-7} per refueling outage.

If the plugs are inadvertently left in the refueling cavity, this condition is judged to last for only a maximum of 92 days, at which time there is a high likelihood that the separate verification of another test per Surveillance Instruction SI-6.28 will detect the condition. Consequently, the mean unavailability of the drain, based on an 18-month refueling outage, is estimated to be approximately 4×10^{-8} per demand. The distribution associated with this error rate is assigned the variable name WHESUM in the basic event database.

3.3.3.2 Results of Dynamic Human Actions Analysis

The event sequence and systems evaluations identified the operator actions listed in Table 3.3.3-4 as being a potentially important influence for the mitigation of severe core damage sequences. The reasoning for their explicit inclusion in the event sequence models is discussed in the description of the event sequence diagrams and the definition of the event tree top events in Sections 3.1.2 through 3.1.4. This section presents the

- Qualitative description of the tasks required to accomplish the actions successfully, and the conditions under which they must be accomplished.

- Quantitative evaluation of performance-shaping factors reflecting the operators' judgments regarding the degree of difficulty for successfully accomplishing the actions.
- Distributions of the human error rates derived from the quantification evaluation using the adaption of the success likelihood index methodology (SLIM), as summarized in Section 2.3.5.2 and presented in Appendix B, Section B.4.

Insights gained from the evaluation process, to include a comparison of group evaluation perspectives and a trend analysis of seven performance-shaping factor (PSF) ratings, are presented in Appendix B, Sections B.5.4 and B.5.5.

3.3.3.2.1 Qualitative Description of the Dynamic Human Actions

A short description of each action evaluated for the Watts Bar PRA is given in Table 3.3.3-4. Appendix B, Table B-10, presents the Operator Response Forms for each evaluated dynamic human action. The forms are written in accordance with the guidelines contained in Appendix B, Section B.4.1, as summarized in Table 3.3.3-5. The descriptions on the forms were developed by the human action analyst and licensed operators serving on the PRA team, with information provided by the event sequence analysts regarding the conditions under which each action is demanded.

An example of a completed Operator Response Form is given in Table 3.3.3-6. Sufficient detail is provided to permit the operator groups evaluating the actions to recognize the context of the action. However, detailed evaluation of the performance-shaping factors is purposely omitted so that the operators can form their own judgments. The justifications of the time windows for the actions are presented in the top event definitions and Appendix C.

The dynamic human actions were qualitatively evaluated by the three groups of licensed plant operators who performed the quantitative evaluation. These groups discussed the context of each action among themselves before quantitatively evaluating it. In some cases, the groups provided practical comments that assisted the event sequence analyst to improve the plant model. Those found to be useful for clarifying the evaluations were either incorporated into the Operator Response Forms or included in the group comparison and trend analysis presented in Appendix B, Sections B.5.4 and B.5.5.

3.3.3.2.2 Quantitative Evaluations

The quantitative evaluations of the licensed plant operators are elicited and converted to human error rates using an adaptation of the SLIM methodology (References 3.3.3-3 through 3.3.3-5). Three operator groups quantitatively assessed the weight and degree of difficulty score of the seven PSFs in accordance with the guidelines in Appendix B, Section B.4.2, as summarized in Table 3.3.3-7. These evaluations are summarized in Tables 3.3.3-8 and 3.3.3.9.

The failure likelihood index (FLI) evaluations of each group are converted into human error rate estimates independently of the other two groups in accordance with the procedures outlined in Appendix B, Section B.4.3. After the failure rates for the individual groups are obtained, they are merged together, giving equal weight to each evaluation group. The individual actions are grouped by similarity of weights into groups for quantitative evaluation against calibration actions. Calibration actions obtained from evaluations in other PRAs are used to benchmark the failure rates of each group. The identification of each calibration action, the basis for its failure frequency, and source of the calibration PSF weights and scores are documented in a calibration action database (Reference 3.3.3-6). To keep the differences in judgments explicit, no adjustment is made to the normalized weights or individual PSF rating of either the rated actions or the calibration actions during this process. The resulting evaluations are given by individual rating group in Appendix B, Tables B-13 through B-15. An example of the quantification of a group of similarly weighted actions is given in Table 3.3.3-10.

The human error rates used in the PRA are obtained from merging the individual groups of operator evaluations into composite quantitative estimates by assigning equal weights to the evaluations of each operator group. This is done using the MERGE function of the BARP software program (Reference 3.3.3-7), as outlined in Appendix B, Section B.4.4. These composite error rates are given in Table 3.3.3-11.

Some estimates have large range factors. This is due to both the assignment of uncertainty to the derived error rate of each group and the variability of ratings among the groups. Based on the recommendations of Swain and Guttman (Reference 3.3.3-2), the range factor for any individual failure rate must be at least 10 if any of the estimates derived from the group evaluations have a median value of less than 10^{-3} per demand, and 5 otherwise. Therefore, the composite estimates must have at least those range factors. When the estimates derived from the group evaluations diverge, the merging process produces broad distributions whose mean values tend to reflect the most conservative of the group evaluations. However, the entire distribution is retained so that the uncertainty can be accounted for explicitly if the human action appears in risk-dominant sequences that are subjected to uncertainty analysis

3.3.3.2.3 Discussion of Results

The average and the range of the FLIs assessed by the three operator groups are as follows:

Group	Average FLI	Highest FLI	Lowest FLI
1	4.77*	8.56*	1.57
2	5.32**	9.86**	2.20
3	5.14	8.20	2.11

*Excluding two actions assessed as guaranteed to fail due to inadequacy of time to accomplish.
**Excluding three actions assessed as guaranteed to fail due to inadequacy of time to accomplish.

Although not subjected to statistical tests, the average value and distribution of the ratings are considered to be in reasonably close agreement and indicate that the three evaluation groups interpreted the scaling guidance consistently.

Each operator group brought its own perspective to the evaluation process. For some actions, this perspective produced a wide divergence among the error rates derived from the group evaluations. Appendix B addresses those actions that revealed a divergence of opinion among the groups sufficient enough to produce a range factor of greater than 30 in the composite distribution.

In addition to the discussion of significant differences in the evaluations, Appendix B, Section B.5.5, discusses trends observed in the evaluations.

Differences in evaluations, trends, and comments found to be applicable to event sequences that are found to be important during the quantification of the plant model are discussed in Section 6.1.

3.3.3.3 Results of Recovery Analysis

The quantification of the plant model and subsequent analysis of the dominant sequences produced a number of opportunities for operator dynamic and recovery actions, as summarized in Table 3.3.3-12. The reasoning for their explicit inclusion in the event sequence models is discussed in the description of the support system and frontline event trees in Sections 3.1.2 through 3.1.4. Appendix B, Section B.7, presents the recovery action evaluation in detail.

The actions listed in Table 3.3.3-12 explicitly exclude those actions accomplished explicitly to recover offsite power. Those actions are accounted for directly in the model used to quantify the recovery, which is presented in Section 3.3.3.4.

Table B-19 of Appendix B presents the Operator Response Forms for each evaluated recovery action. They follow the same pattern as those shown in Tables 3.3.3-5 and 3.3.3-6. The descriptions on the forms were developed by the human action analyst and licensed operators serving on the PRA team, with information provided by the event sequence analysts regarding the conditions under which each action is demanded.

The recovery actions were qualitatively evaluated by the group of licensed plant operators who performed the quantitative evaluation. This group discussed the context of each action among themselves before quantitatively evaluating it. In some cases, the group provided practical comments that assisted the event sequence analyst to improve the plant model. This included both eliminating some potential recovery actions as being impractical or impossible and offering alternate methods of recovery, where suitable. Wherever appropriate to clarify the evaluations, the operator comments are discussed in the Operator Response Forms.

Only one group of licensed operators quantitatively assessed the recovery actions. This group consisted of a composite of the licensed operator PRA team members from both the Watts Bar and Sequoyah Nuclear Plants, plus one additional operator from the Watts Bar Plant. This mix of operators took advantage of the fact that the recovery actions for the two plants are very similar.

The group assessed the actions using the same procedure followed for the dynamic human actions. These evaluations are summarized in Table 3.3.3-13. The group's FLI evaluations are converted to human error rate estimates in accordance with the procedures outlined in Appendix B, Section B.4.3, which follows that of the dynamic actions discussed in Section 3.3.3.2. The resulting distributions are given in Table 3.3.3-14. Appendix B, Section B.7.3, discusses these results and also provides reasoning for the elimination of some of the recovery actions proposed by the PRA group.

3.3.3.4 Electric Power Recovery

Accident sequences involving station blackout are frequently found to make important contributions to risk in PRAs. Because of the importance of these sequences, realistic treatment of the possibilities for recovery of both offsite and onsite power is needed to obtain realistic frequencies for such sequences leading to severe core damage. To treat such sequences, time-dependent models are needed to account for important time-dependent interactions.

This section describes the calculational model developed for the STADIC computer code to analyze the electric power system hardware and the operators' actions to restore AC power following a loss of all offsite power initiating event. Recognition of the power failure condition would be almost immediate. The nature of the AC power loss condition and the numerous indications available in the control room (dark panels, loss of lighting, failure of all AC equipment, etc.) would make its identification relatively simple. There is therefore a negligible chance that control room personnel could misinterpret a loss of all AC power as being another condition and then take action inappropriate for the situation. The event, however, will cause some degree of stress and confusion among the operators because it is perceived as a severe transient. The principal concerns will be to:

- Restore AC power.
- Maintain and control auxiliary feedwater flow from the turbine-driven auxiliary feedwater pump.
- Monitor core subcooling and reactor coolant inventory.
- Monitor DC power availability and take action to extend battery life.

In this section, the frequency of electric power failure and recovery is analyzed for a number of conditions, depending on the systems available for recovery (e.g., one of two diesel generators, only one diesel generator, and offsite power) and the availability of auxiliary feedwater; i.e., heat removal via the steam generators. Factors that influence the time available to restore AC power include the availability of 125V DC power (i.e., battery lifetime) and the length of time to core damage due to pump seal leakage or power-operated relief valve (PORV) discharge following a loss of all onsite AC power. Coolant inventory loss out the PORV would also occur during station blackout after the time of steam generator dryout with sequences in which the steam-driven auxiliary feedwater pump is postulated to fail.

The result of this analysis is a recovery factor that is the ratio of the following two conditional frequencies, given a loss of offsite power initiating event: the conditional frequency of loss of onsite power (i.e., diesel generators) in a mission time of 24 hours and failure to restore onsite or offsite power before core damage occurs, and the conditional frequency of onsite power failure in a period of 24 hours without considering power recovery. The analysis is applicable to several different initiating events other than loss of offsite power that also entail a loss of offsite power condition; e.g. transient with subsequent loss of offsite power. This analysis is not applicable to seismic-induced loss of offsite power.

3.3.3.4.1 Integrated Model for Electric Power Failure and Recovery

The electric power system analysis presented in Section 3.2.1 evaluates the unavailability of power at the vital buses. The automatic startup and operation of the emergency diesel generators are analyzed after loss of offsite power. An operating mission time of 24 hours is used to quantify the unavailability of onsite power before the application of a detailed recovery.

Recovery analyses often require careful evaluation of human response during abnormal and unfamiliar circumstances. The actions of operations, maintenance, supervisory, and support personnel must be coordinated into a team effort to restore normal plant conditions. Depending on the event scenario, the failed equipment, and the causes for failure, several competing recovery options may be available. These options must be evaluated considering the available personnel, the difficulty of each action, actual and perceived urgency, procedural guidance, training, and experience. Very often, these decisions must be made under conditions of high stress with little time for detailed planning. For these reasons, it is important to apply recovery models for specific event scenarios or for groups of event sequences exhibiting similar system and plant performance characteristics, and requiring similar human responses.

In general, specification of the recovery event scenarios provides two important pieces of information necessary for the evaluation of human actions and equipment response. The initiating event and subsequent system failures define the status of the plant when the operators are required to act. Control room alarms, emergency procedural guidance, and the status of critical plant parameters provide basic input to focus the initial actions. The event tree model defines the nature of the actions that must be taken to restore normal plant response. For each scenario, there is also a fairly well-defined time window for successful system recovery. Core damage will be prevented if the identified recovery actions are completed within this time window. The amount of time available depends on the type of initiating event, and the nature and timing of subsequent component failures.

A realistic model for the recovery of electric power during a specific event scenario must account for the causes and timing of the power failure events, the sequencing of failures and recovery actions, and the available time window for success before the onset of core damage. Equipment failures and recovery can occur at any time during the 24-hour study period after event initiation. Thus, a time-integrated model for failures and recovery actions is necessary to assess the effect of diverse failure causes and to model corresponding responses started at different times after event initiation that require different amounts of time to complete.

In general, the unavailability of power for periods of time longer than T hours after onsite power loss during the 24-hour period after an initiating event can be calculated from the following expression:

$$F(EP,T) = \int_0^{24} \phi_f(t)[1-G(t+T)][1-H(T)]dt \qquad (3.3.3.1)$$

where

F(EP,T) = probability that there is no onsite or offsite power T hours after the occurrence of station blackout; i.e., T hours after the loss of onsite power when offsite power has not been removed.

$\Phi_f(t)dt$ = probability density function for failure of the onsite power system in the time interval between t and t+dt after the loss of offsite power at time 0.

G(t+T) = probability that offsite power is restored within t+T hours after loss of offsite power.

H(T) = probability that onsite power is restored within T hours after the beginning of the station blackout.

T = time interval between the beginning of the station blackout and the point of no return for the return of electric power to prevent core damage.

The electric power recovery factor is expressed as

$$RE = \frac{\int_0^{24} \phi_f(t)[1-G(t+T)][1-H(T)]dt}{\int_0^{24} \phi_f(t)dt} \qquad (3.3.3.2)$$

Note that $\Phi(t)dt$ accounts for all combinations of diesel generator failures to start and to continue running. The recovery functions are also a function of battery life. The above equation is applied differently to different sequences to account for the following types of dependencies:

- Depending on the initiating event, offsite power or one or more diesel generators may not be recoverable. For example, a flood may damage switchgear or a loss of cooling water may lead to an overheated diesel generator.

- The time available to restore electric power during a station blackout is dependent not only on the timing of the blackout after loss of offsite power but also on the sequence. For example, if the steam-driven feedpump is working, the pump seal

LOCA may dictate the time available, whereas, if there is no steam generator heat removal, the steam generator dryout and the loss of coolant out the steam generator PORVs will dictate the time (T). The model includes the variable time window.

- Depending on the event sequence and the application of the Emergency Operating Procedures, operator actions to depressurize the reactor coolant system may occur to lengthen the time available for recovery.

The calculations for the integrated electric power failure and recovery model [Equation (3.3.3.2)] are performed by the Monte Carlo computer simulation program, STADIC. The uncertainty distributions for component failure data and other parameters used in the model calculations are input into STADIC, and the electric power recovery factors are computed in its subroutine SAMPLE for each random sample of parameter values selected from the uncertainty distributions. These calculations produce the ratio of integrated power failure and recovery results over 24 hours to the integrated power failure for 24 hours without recovery.

3.3.3.4.2 Time-Dependent Power Failure Analysis

The evaluation of a detailed power recovery model requires careful treatment of the sequencing of power failures and recovery actions after the initiating event has occurred. Failures may occur at different times during the analysis period. For example, offsite power supply failure could be the cause of a plant transient initiating event, and the onsite diesel generators could fail during operation at some later time. It is possible for some recovery actions to proceed in parallel, while the time sequencing of other recovery actions must be carefully modeled. The actions required to restore power from the offsite grid and to repair a failed diesel generator can be performed at the same time if there are enough available personnel to support both tasks. However, diesel generator repairs and offsite power recovery actions may be started at significantly different times during the event scenario. In general, offsite grid recovery efforts will begin soon after the initial power failure. Diesel generator repairs will begin only after the diesel generators have failed, which may not occur for several hours after the initiating event. Careful treatment of these time dependencies ensures that the model does not incorrectly include the effect of quantified recovery actions before a failure has occurred, such as assigning a frequency for diesel generator recovery within 1 hour after the initiating event when, in fact, the diesel generator has not failed until 2 hours after the event. This treatment also eliminates the quantitative contribution from failures of one power supply that occur after power has been recovered from another source, such as diesel generator failures that occur after offsite power has been restored. After normal power has been restored, the diesel generators will be shut down. The quantitative effect from an analysis of diesel generators operation for time periods after offsite power recovery should not contribute to the power unavailability model results.

The integrated power failure and recovery model characterized by Equation (3.3.3.1) accounts for the time sequencing of failures and recovery actions. The development of the nonrecovery term in this expression, $[1-G(t+T)][1-H(T)]$, allows the evaluation of parallel and time-sequenced recovery models. The specific power recovery options included are described in Section 3.3.3.4.3.

The evaluation of Equation (3.3.3.1) or (3.3.3.2) also requires a quantitative model for the time behavior of electric power failures characterized by the function $\Phi_f(t)$. The failure rate equations of the diesel generator systems analysis are programmed into subroutine QDG to provide this failure function after an initiating event and to consider the sequencing and details of each recovery option.

The electric power recovery scenarios evaluated in this analysis are all initiated by a loss of offsite power at the time of the plant transient. Power failure could be the direct cause of the initiating event, or it could result from power grid instabilities when Watts Bar trips offline from another cause. However, each sequence for this model has offsite power lost at time t = 0. Subroutine QDG models a variety of failure causes of the diesel generators. To develop a time-dependent model for the failure function $\Phi_f(t)$, these subroutines were evaluated for numerous discrete points in time over a 24-hour period after the loss of offsite power. The model programmed in the subroutine provides the cumulative unavailability of onsite power over a specified analysis period. By evaluating this model many times, a cumulative unavailability distribution was derived for the failure of power as a function of time. Demand failures, such as circuit breaker operations and diesel generator starting, are evaluated at time t = 0. Time-dependent failures, such as short circuits and diesel generator operating problems, contribute to the cumulative unavailability function based on the mission time of the analysis. After the cumulative failure function has been quantified in this manner, it can be differentiated to provide the time-dependent power failure function $\Phi_f(t)$ used in the recovery analysis equations. Numerically, this differentiation process is performed by taking the difference between the cumulative unavailabilities at successive discrete time intervals.

3.3.3.4.3 Power Recovery Options

Two types of power recovery actions are evaluated for this analysis. Depending on the cause for its failure, offsite power may be restored at any time during the 24 hours after the initiating event. When offsite power is recovered, the diesel generators can be stopped. The second type of recovery is the restoration of the failed diesel generators. This is considered only for the period after the onsite diesel generators fail.

3.3.3.4.3.1 Offsite Power Recovery. Watts Bar is connected to the offsite grid via five 500-kV transmission circuits. Two 161-kV circuits are connected to Watts Bar Hydro. Part of the output from each generator feeds the offsite grid via the main generator while the remainder feeds station loads via two unit station service transformers (USST) per unit. The two 161-kV circuits feed CSSTs A&D and B&C.

Presently, the 6.9-kV unit boards are normally fed from the USSTs (500-kV switchyard). CSSTs C&D feed the 6.9-kV Class 1E shutdown boards. On a unit trip, the unit boards automatically realign to the 161-kV switchyard through a bus and stepdown transformer. Should the 6.9-kV Class 1E shutdown boards sense a loss of power due to the unavailability of the 161-kV power source, the bus will automatically realign to receive power from the standby emergency diesel generator. In this study, a loss of offsite power is the failure of all power supply from the 500-kV and 161-kV.

Industry data representing the time to recovery of loss of offsite power at nuclear power plants for 63 actual incidents caused by plant-centered losses, grid losses, or severe weather losses have been documented in Reference 3.3.3-8.

The plant-centered causes of offsite power failure were also categorized (Reference 3.3.3-9) into three plant groups (I1, I2, and I3), depending on the plant design factors: (1) independence of the offsite power sources and (2) automatic and manual transfer schemes for the Class 1E buses. Watts Bar was found to be represented closest by the group I2 data. These data were further analyzed in Reference 3.3.3-8 to show various ways to represent analytically the data and to combine the data, depending on the plant to be modeled. The report combined the data to represent a worst case and best case model. It also developed a model based on the group I2 plant-centered data which would represent Watts Bar. These cases have been plotted on Figure 3.3.3-1. Thus, for Watts Bar offsite power recovery base model, the NUREG model for group I2 plants (Watts Bar representative) was selected as the 50th percentile, with the upper and lower bounds selected as the 5th and 95th percentiles, respectively. These figures were given a probability distribution of 0.1, 0.8, and 0.1 for the 5th, 50th, and 95th percentile curves, respectively.

3.3.3.4.3.2 Diesel Generator Power Recovery

3.3.3.4.3.2.1 Diesel Generator Hardware Recovery Model. The emergency diesel generators and their support systems are described in the electric power system analysis presented in Section 3.2. Although each diesel has two independent air-starting systems, each diesel generator requires a supply of 125V DC from its respective DC bus for generator field flashing and generator start and control. The effect of the unavailability of DC power on diesel generator recovery is accounted for in the integrated recovery model. Watts Bar has a spare, fifth diesel generator that could be aligned as a recovery action but has not been considered in the recovery analysis. Alignment of this diesel generator is not proceduralized for loss of all AC conditions.

The most important causes for diesel generator unavailability are diesel generator hardware-related failures either during the startup sequence or during subsequent operation, and diesel generator unavailability due to maintenance at the time of the initiating event. The spare diesel generator can also be used to replace another diesel generator that requires maintenance. No credit for the reduced unavailability from maintenance due to this feature was assumed in this study.

The diesel generator failures include all malfunctions that prevent the unit from delivering stable power to its output bus. These include failures of the engine, generator, mechanical controls, electrical controls, starting systems, and trip systems. The time to return a diesel generator to operation after a hardware failure depends on many factors, such as the cause of failure, repair personnel availability, alternate power supply status, reactor operating conditions, etc.

The causes of diesel generator hardware failures can range from the spurious operation of a trip solenoid to major physical damage of mechanical or electrical components. Recovery from these failures may involve the simple resetting of a local trip interlock and restarting of the diesel generator, or it may require disassembly and repair of the engine, generator, or its control system. If the time available for electric power recovery is relatively short (e.g., less than approximately 2 hours), review of generic diesel generator failure and maintenance data indicates that only the diesel generator startup failures present a significant potential for rapid recovery. Diesel generator failure during operation generally involves more severe problems that require detailed troubleshooting, repairs, or replacement of parts, which are difficult to complete in less than 2 hours.

Finally, because most maintenance events require at least partial reassembly of the diesel generator before it can be started, it is assumed for this analysis that the maintenance contribution to unavailability is also irrecoverable within 2 hours after the initiating event. The following table indicates some of the key actions that can be accomplished within given recovery time periods:

Time Following Operator Response	Action
0 to 5 Minutes	Reset Trip Relay and Attempt Local Manual Restart
5 to 15 Minutes	Troubleshoot Simple Problems; Check Electrical and Mechanical Indications
15 to 30 Minutes	Perform Step-by-Step Problem Diagnosis; Notify Cognizant Engineering and Maintenance Personnel
30 to 60 Minutes	Refer to Technical Manuals and Drawings for Diagnosis of More Complex Failures; Response Time for First Offsite Personnel
1 to 2 Hours	Offsite Personnel Troubleshoot Problems That Do Not Require Component Repair; Make Complex Adjustments to Control Systems
2 to 4 Hours	Replace Simple Failed Components (includes maintenance crew response time)
4 to 8 Hours	Repair Failed Components Requiring Minor Disassembly
8 to 24 Hours	Perform More Complex Repairs
24 to 72 Hours	Make Repairs Requiring Disassembly
> 72 Hours	Overhaul Diesel Engine

It is emphasized that these key actions apply to the recovery for a given failed diesel generator following operator response to that unit and only to the recovery from hardware-related failures. They do not include the time for operators or maintenance personnel to reach the diesel room after the diesel generator fails. It is not the length of time required to recover any one of the failed units. These key actions are used as one piece of information in developing a distribution for the length of time required to recover a failed diesel generator.

The recovery time distribution summarized below applies to situations involving a high urgency for diesel generator repairs. It is not derived directly from actual maintenance event duration data because most diesel generator repairs are not completed under the extremely urgent conditions that would prevail after loss of all offsite and onsite AC power. It is based on a review of diesel generator failure and maintenance records collected from several plants, with an assessment of the severity of the observed failures, and the experience of operations and maintenance experts.

Time To Recover a Failed Diesel Generator	
Time Following Operator Response (hours)	**Probability**
0.0 to 0.5	0.20
0.5 to 1.0	0.10
1 to 2	0.15
2 to 4	0.15
4 to 8	0.20
8 to 24	0.10
> 24	0.10

This distribution is used to model the time needed to restore a single diesel generator to operation after the diesel generator has experienced a hardware failure. It is assumed that repair efforts are continuous from initial troubleshooting until the diesel generator is returned to service, accounting for factors such as the need to call out additional maintenance personnel for major repairs. Recovery cannot begin until someone goes to the diesel generator room to investigate the failure, and this distribution does not include scenario-specific delays for operating or maintenance personnel reaching the room. These personnel response times are evaluated in the next section and are integrated with the hardware repair time distribution to model fully diesel generator recovery for specific failure scenarios.

3.3.3.4.3.2.2 Diesel Generator Recovery Personnel Response Time Model. Station auxiliary operators are responsible for operating the diesel generators and for initial problem troubleshooting. During the normal work day, additional personnel are also available. An auxiliary operator's normal responsibilities include monitoring plant equipment, changing valve positions and system configurations under direction of the control room operators, and performing walk-through inspections of plant areas. During normal shift working conditions, the operators will usually be roving around various locations or will be at a designated watch area. Other possible, but less likely, locations include the main control room or the administration building.

When offsite power is lost, all diesel generators will receive signals to start. If diesel generator failures occur after a loss of all offsite power, trouble alarms for the diesel generators and other safety systems will be annunciated in the main control room. These alarms, electrical equipment inoperability, and initial verification steps in the plant procedures will provide almost immediate notification to the control room operators who may attempt to manually restart the affected engine from its control room switch. However, experience has shown that many failures require local troubleshooting to correct the problem and to reset engine trip relays. The control room operators may also be reluctant to restart quickly a diesel generator that tripped during operation before they determine a cause for the failure. For this analysis, it is assumed that an auxiliary operator must locally investigate all failures before any engine restarts are attempted.

Therefore, local plant operators must be dispatched to the diesel generator building for all diesel generator recovery actions. The control room operators or the control room foreman will contact an auxiliary operator by telephone, radio, or page soon after the diesel generator fails. After he has been notified of the failure, the auxiliary operator will proceed to the diesel generator rooms to investigate the cause, reset engine trip relays, and begin local recovery efforts including manual restart attempts. It is estimated that the operator's response time for going to the diesel generator building from any of his normal duty locations is approximately 5 to 10 minutes after notification. It is assumed that the auxiliary operators will carry key cards to manually unlock controlled access doors, if required.

The following distribution is used to model the response time for an auxiliary operator. It applies to the elapsed time from failure of the diesel generator until the operator begins local troubleshooting activities in the diesel generator room. This time includes initial failure detection time, delays for the control room to contact the operator and describe the problem, the operator transit time to the diesel engine location, and possible additional delays due to communications problems or other considerations that could impede the operator's response.

Time for First Operator Response to Failed Diesel Generator (includes detection time, notification time, and transit time)	
Response Time (minutes)	**Probability**
0 to 5	.01
5 to 10	.25
10 to 15	.50
15 to 20	.20
20 to 30	.03
30 to 60	.01

It is also expected that the in-plant foreman and the onsite maintenance technicians will respond to diesel generator failures that are not corrected quickly by the auxiliary operator. Depending on the status of equipment in other parts of the plant, additional qualified auxiliary operators may also be available to help with the recovery efforts. The participation of this normal complement of shift personnel has been considered in this recovery time distribution.

If two or three diesel generators for Watts Bar fail, the operating and maintenance personnel would concentrate their initial recovery efforts on one of the diesel generators. A preliminary evaluation would be made to determine whether one of the diesel generators could be repaired more quickly than the others, and that diesel generator would receive the most concentrated attention. For example, efforts would be made to restart a diesel generator that tripped spuriously before repairs were started on a diesel engine that sustained extensive mechanical damage. In this recovery model, it is assumed that the initial response team will concentrate its efforts almost exclusively on one diesel generator for approximately 20 minutes after the auxiliary operator reaches the building.

If the first diesel generator is not restored to operation after 20 minutes, it is expected that the response team will begin parallel efforts to recover the other failed units. For example, the maintenance technicians could remain with the first diesel generator to begin component repairs or replacement, while the operators turned their attention to troubleshooting and restart attempts on the other units. As more support personnel respond to the site, repairs of diesel generators can proceed in parallel, and the recovery models can be substantially decoupled.

For this analysis, if three Watts Bar diesel generators have failed, recovery of only one is allowed during the first 30 minutes after initial operator response. Power can be restored from the diesel generators to one vital bus in this interval; then, two vital buses can be crosstied following recovery of one diesel generator so that two power trains are available. For time periods longer than 30 minutes, the model permits recovery to be started on the second diesel generator, and work is assumed to proceed on two diesel generators in parallel until power is restored. Similarly, for time periods longer than 1 hour, the model permits recovery to be started on the third diesel generator, and work is assumed to proceed on three diesel generators in parallel until power is restored. Thus, the minimum amount of time required to begin power recovery to three of the four vital buses by repairing failed diesel generators is more than 30 minutes after the diesel generators fail.

3.3.3.4.3.2.3 Integrated Diesel Generator Power Recovery Model

- **Single Diesel Generator Recovery.** After a loss of offsite power, all vital buses will be deenergized if all four diesel generators fail due to failure of DC power, being out of service for maintenance, or experiencing failure during the starting sequence or the subsequent operation. It is assumed for this analysis that DC power is required for diesel generator recovery. Thus, recovery of only one diesel generator is assumed to be possible if DC power is available to only one diesel generator. The model for a single diesel generator recovery is

$$\Phi_1(T) = \int [\phi_{OR}(t)][\phi_{DH}(T-t)]dt \qquad (3.3.3.3)$$

where

$\Phi_1(T)$ = cumulative frequency of power recovery from a single diesel generator when only one diesel generator is available for recovery.

$\Phi_{OR}(t)dt$ = frequency of auxiliary operator response to diesel generator room between t and t+dt after the failure of diesel generator power.

$\Phi_{DH}(T\text{-}t)$ = cumulative frequency of diesel generator hardware recovery within time (T-t) after operator response.

T = time interval between the beginning of the station blackout and the point of no return for the return of electric power to prevent core damage.

This analysis is performed for conditions when only one diesel generator is available for recovery; i.e., the other diesel generators have failed at t=0 and cannot be recovered within 24 hours. For this analysis, approximately 20% of the single diesel generator unavailability is assumed to be attributed to preexisting maintenance scenarios. The 5th percentile model for the single diesel generator recovery reduces the cumulative frequency of recovery for one diesel generator by 20%. For the 95th percentile, however, a more optimistic view is taken, and it is assumed that this fraction of unavailability is recoverable. This would include, for example, restoring the diesel generator to service after minor maintenance or testing. For the 50th percentile, it is assumed that the fraction of unavailability due to maintenance is recoverable after 2 hours.

The 5th percentile of the single diesel generator recovery model represents a pessimistic model for operator response and delays the auxiliary operator's arrival time by 30 minutes. The 50th percentile of the model represents a delay of the operator's arrival by 10 minutes. The 95th percentile bound represents a more optimistic model for operator response, and no delay in the auxiliary operator's arrival is included.

Figure 3.3.3-2 presents the complementary cumulative distribution for the diesel generator nonrecovery that is derived for these bounding models.

- **One-Out-Of-Two Diesel Generator Recovery.** If power can only be recovered from two diesel generators, successful recovery has been defined for this analysis as the restoration of power from at least one of the two diesel generators and two vital buses being crosstied to receive power from the recovered diesel generator. This recovery model is characterized by the expression

$$\Phi_{1/2}(t) = \Phi_1(t)+[1-\Phi_1(t)][\Phi_1(t-0.5)] \qquad (3.3.3.4)$$

This model allows recovery of the first of two diesel generators to begin when an auxiliary operator arrives at the diesel generator room. Recovery of the second diesel generator begins 30 minutes after the auxiliary operator arrives, and the repairs of both diesel generators are modeled as continuing in parallel thereafter.

Two bounding scenarios are applied as the 5th and 95th percentiles for the diesel generator recovery model.

For the 5th percentile bound, the single diesel generator recovery model [Equation (3.3.3.3)] is used. This model represents a pessimistic model for operator response, and it allows recovery of power from only one diesel generator. Parallel repairs of the second diesel generator are not considered. This bound accounts for possible unidentified dependencies in the recovery efforts for both diesel generators, which could couple the repair time distributions; e.g., limited spare parts availability, limited support personnel availability, etc.

For the 95th percentile bound, the dual diesel generator recovery model [Equation (3.3.3.4)] is used. The recovery of the second diesel generator begins 30 minutes after the operator arrives, and the repairs of both diesel generators are modeled as continuing in parallel thereafter. The 95th percentile bound thus represents a more optimistic assessment of operator response, and it includes a more realistic model for single and parallel diesel generator repairs.

Figure 3.3.3-3 presents the complementary cumulative distribution for the diesel generator nonrecovery derived from these bounding models.

- **One-Out-Of-Three Diesel Generator Recovery.** In this analysis, successful recovery from three failed but recoverable diesel generators is defined as the restoration of power from at least one-of-out-three diesel generators and two vital buses being crosstied to receive power from the recovered diesel generator. This recovery model is characterized by the expression

$$\Phi_{1/3}(t) = \Phi_1(t)+[1-\Phi_1(t)]\{\Phi_1(t-0.5)+[1-\Phi_1(t-0.5)][\Phi_1(t-1.0)]\} \qquad (3.3.3.5)$$

The 5th percentile and 95th percentile bounds of this recovery model are developed in a manner similar to those in the one-out-of-two diesel generator recovery scenarios. Figure 3.3.3-4 presents the complementary cumulative distribution for diesel generator nonrecovery derived from these bounding models.

3.3.3.4.4 Electric Power Recovery Scenarios

Recall from Section 3.3.3.4.1 that a variable time window (T) is used for each recovery option. The available time for recovery is a function of both support system availability and RCS thermal hydraulics. Each diesel generator requires a supply of 125V DC power to start and operate. If, for example, a battery can last for 4 hours after the loss of all onsite power (diesel generators), the auxiliary operators would have a time window of only 4 hours to recover the diesel generators, as long as the thermal-hydraulic window is longer than or equal to 4 hours. The thermal-hydraulic time window is a function of the availability of auxiliary feedwater and the leak rate from the reactor coolant pump (RCP) seals. If, in this case, for example, auxiliary feedwater (the auxiliary turbine-driven feed pump) is available when and after onsite power is lost, the time window for onsite power recovery and for the operator's response time to locally manually operate onsite breakers after DC power is lost is dependent on the leak rate from the RCP seals; i.e., the time to core uncovery from this leak.

3.3.3.4.4.1 Time Window Based on Plant Thermal Hydraulics. Within the framework of the complete loss of AC power, the statuses of the turbine-driven auxiliary feedwater pump and the RCP seals are very important for determining the available time window before the onset of core damage. The time window is the time available for the operators to take action (e.g., restore electric power and restart the auxiliary feedwater pump or the charging pumps) before core damage occurs. This time available for action, in general, increases after the initial reactor trip (at t=0) because the reactor decay heat generation rate decreases with time.

Auxiliary feedwater is assumed to be available as long as AC onsite power is available. When onsite power is lost; i.e., auxiliary feedwater is available only if the turbine-driven feed pump did not fail. The electric power recovery scenarios that can be evaluated by this model include both states (i.e., available or not available) of auxiliary feedwater. In the case in which auxiliary feedwater is available, the turbine-driven pump and its support systems (e.g., condensate storage tank capacity) are assumed to be available for 24 hours after the loss of all AC power. It is assumed in this analysis that a primary system leak from the RCP seals will occur after the systems (i.e., pumps) supplying cooling water to these seals have stopped because of the loss of all AC power and because no seal cooling is supplied from the charging pump. The model was also used to analyze scenarios in which the charging pump continues to provide cooling to RCP seals and thus, no pump seal loss of coolant accident (LOCA) occurs, or a specific time to core uncovery was specified based on other PRA model conditions. For those scenarios in which severe seal degradation occurs (no pump seal cooling), the leak rate from these seals is assumed to be within the range of 84 gpm to 1,920 gpm for all four pumps. A constant leak rate (initiated immediately when all onsite AC power is lost and seal cooling does not exist) of 84 gpm (four pumps) was used in this analysis as the leak rate for the first hour prior to the severe seal damage. The base model for the pump seal leak rates was based on the RCP seal LOCA study of Reference 3.3.3-10 for Westinghouse RCPs with the old style O-rings that exist in the Watts Bar reactor coolant pumps. These data were used to develop the probability leak rate model for this analysis shown in Table 3.3.3-15 and programmed into the model to calculate the time of core uncovery due to a pump seal LOCA.

If auxiliary feedwater is available, along with the steam generators, it can provide sufficient cooling to maintain primary system pressure below all primary system relief valve lift points. Because of this capability, the length of time to core damage will be extended because core decay heat continues to be removed and core coolant inventory is only lost via the degraded reactor coolant pump seals. Without auxiliary feedwater pumps working to remove core decay heat from the reactor coolant system, the primary relief valves open to relieve pressure. The coolant inventory is lost via the RCP seal leakage and through the primary system relief valves. This decreases the primary system water inventory at a much faster rate than through the loss of pump seals alone.

Furthermore, with the auxiliary feedwater system operating, if the operator depressurizes the steam generators, as instructed in the station blackout procedures, primary system pressure will decrease. A decrease in primary system pressure will lower the differential pressure across the RCP seals and therefore decrease the pump seal leak rate and increase the time to core uncovery. However, the seal LOCA data provided in Reference 3.3.3-10 showed little effect of primary system depressurization and, thus, were conservatively neglected in this model.

The time to core uncovery following the loss of onsite power with auxiliary feedwater available is calculated in the model based on the leak rate probability distribution for each time increment following station blackout, as provided in Table 3.3.3-15, and the reactor coolant inventory loss required for core uncovery of 8,345 cubic feet (Reference 3.3.3-11).

However, as the reactor coolant liquid inventory decreases to a level below that of the pump seals, the leakage out the seals becomes steam, and the leakage mass flow rate decreases considerably. The model developed accounts for the two-phase (liquid/steam) flow through the seals. The liquid leakage is considered subcooled at full-power pressure having a density of 45 pounds per cubic foot per information obtained per telecon with Westinghouse. Because the primary coolant is being cooled by the auxiliary feedwater supplying cooling to the steam generators and discharging from the steam generators via the pressure relief valves until primary coolant material circulation stops, the primary coolant temperature will approach the saturation temperature at the steam generator relief valve setting pressure (1,300 psia). Thus, the steam leakage through the pump seals was considered to be saturated steam at 1,300 psia, continuing to leak at the volumetric time-dependent leak rate that occurs at the time the seals are uncovered by liquid (Table 3.3.3-15). The model also considers the amount of steam produced by the decay heat at the time of seal uncovery, and uses the lesser of the volumetric leak rate (Table 3.3.3-15) or that being produced by the decay heat. Even though the model accounts for two-phase flow through the pump seals that lengthens the time of core uncovery from an all-liquid leakage model and accounts for steam generator cooling to reduce the primary pressure to 1,300 psia, it is still conservative as it does not account for continued reflux cooling by the steam generators to further reduce the primary pressure and to continue removing decay heat. A more detailed MAAP analysis would be required to refine the reflux cooling and was not considered to be cost effective at this time.

For those sequences in which auxiliary feedwater is not available following loss of onsite power, the time window was calculated based on the time for the decay heat to provide sufficient energy to (1) boil the steam generators dry, (2) cause the primary pressure to increase to 2,350 psia and open the PORVs, (3) heat the primary inventory so that it expands to cause the pressurizer to become solid with water, (4) form a steam bubble in the reactor vessel top head and displace the top head liquid inventory out of the PORVs, and (5) boil off the remaining liquid inventory of the primary coolant that is above the top of the core while there is simultaneously a loss of primary inventory due to the pump seal leakage. This total energy was calculated to be 31.0×10^7 Btu. Because the pump seal leakage can occur simultaneously with the loss through the PORVs, the inventory loss due to the pump seal leakage was conservatively assumed to be the nominal leakage that might occur over 2 hours (approximately 7,700 cubic feet). At this rate, the RCS volume below the top of the RCP nozzle and above the top of the core is depleted in less than 2 hours. The major contributor to the time of core uncovery is the energy required to boil dry the steam generator. The additional energy required to boil off the primary inventory if no pump seal leakage was accounted for would be approximately 33%. Equating the energy required for Steps 1 through 5 above to the integral decay heat power curve, the time to core uncovery following loss of onsite power with the charging pumps not available was determined to be

$$TCU = (1.450 + TDG^{.7037})^{1.421} \qquad (3.3.3.6)$$

where

TCU = the time of core uncovery in hours following the loss of offsite power.

TDG = the time of onsite power failure (loss of diesel generators) in hours following the loss of offsite power.

If the steam generator shell side inventory is not available to provide the initial cooling of the primary inventory following the loss of onsite power, the time of core uncovery is

$$TCU = (0.041 + TDG^{.7037})^{1.421} \qquad (3.3.3.7)$$

Equations (3.3.3.6) and (3.3.3.7) assume that the primary inventory is 100% at the time that onsite power fails; i.e., charging pumps have kept the RCS inventory full, and the decay heat has been removed by feedwater up to the time of onsite power failure. These equations are believed to be conservative because they do not account for the thermal capacity of the metallic component heatup and the fact that severe fuel clad oxidation actually occurs after core uncovery begins (after approximately 1/3 to 1/2 of core uncovery). Thus, the times calculated from these equations are at the low end of the probability distribution, and no uncertainty was applied to the calculated conservative estimates.

3.3.3.4.4.2 Time Window Based on the Availability of 125V DC Power. During the variable time window established by thermal-hydraulic considerations for diesel generator failure scenarios, the operators will be restoring power to vital buses by recovering either one diesel generator or power from the offsite grid previously described. The availability of 125V DC power is included in this restoration model since DC power must be available to start the diesel. DC power must also be available for the operation of onsite switchgear breakers from the control room. If onsite 125V DC power is not available, these breakers may be operated manually. The breakers in the switchyard have their own independent 250V DC power supply and control system.

The availability of 125V DC power is dependent on the discharge rate, battery temperature, specific gravity, and the minimum useful or final battery voltage. The analysis of the battery availability is highly dependent on the scenario under consideration. For a scenario with no operator action to shed DC loads and with a 100% load on each DC bus, for example, the batteries will last at least 2 hours. This is the licensing design basis discharge used to size the batteries. However, after evaluating the actual plant operating design, Watts Bar engineering personnel have estimated an extended battery availability and developed a procedure to cope with station blackout lasting up to 4 hours (Reference 3.3.3-12). This lengthened battery lifetime is therefore used in this analysis. Because this capacity is still believed to be conservative, no uncertainty is assumed in the analysis; i.e., the 4 hours is at the low end of the probability distribution for battery lifetime.

The diesel generators are assumed in this analysis to be unrecoverable after depletion of the DC batteries. The diesel generators have their own batteries for control power. However, the 125V DC vital boards are required for breaker control on key individual loads. Therefore, the model conservatively requires both DC power sources be available for successful recovery of a diesel generator. The diesel generator's own battery would last longer than the 125V DC vital battery. Therefore, the time available for recovery is conservatively assumed limited by the capacity of vital batteries. The restoration of offsite power and the operation of the turbine-driven AFW pump, however, are assumed to continue after loss of DC power.

3.3.3.4.5 Electric Power Recovery Model Assembly

The time-dependent calculations for the integrated electric power failure and recovery model [Equations (3.3.3.1) and (3.3.3.2)] are performed by the STADIC computer simulation program. The model was developed to compute the ratio of the conditional frequency of onsite power system failure in a mission time of 24 hours with failure to recover diesel generators or offsite electric power before core damage to the conditional frequency of onsite power system failure in a 24-hour period without including recovery. This model considers diesel generator failures after a loss of offsite power initiating event.

The recovery of offsite power without onsite power available was assumed to be delayed for 2 hours, once the batteries have failed, to allow time for making a decision about which breakers to close, for the control room supervisor to brief auxiliary operators, and for the auxiliary operators to close the breakers manually and to correct breaker malfunctions (or to choose alternate paths or a set of breakers). Since DC power is also required to recover the diesel generators, the cumulative nonrecovery frequency for the diesel generators was assumed to remain constant at the value calculated at the time that the plant batteries fail when no offsite electric power is available. In other words, the recovery time available for the restoration of AC power from the diesel generators is limited by either the plant thermal-hydraulic time window (i.e., AFWS, etc.) or the availability of the batteries.

To evaluate some of the event sequences in which the pump seal return lines remain open after loss of onsite electric power, the pump seal leak rates of Table 3.3.3-15 were all increased by 32 gpm to account for the additional inventory loss for these cases.

The specific scenarios evaluated to impact the dominant sequences of the Watts Bar event tree models are shown in Table 3.3.3-16. This table also provides the analysis results for the electric power recovery factors in these scenarios/event tree sequences applied to the Watts Bar event tree models as the base case quantification of the frequency of core melts.

3.3.3.5 References

3.3.3-1. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," PLG-0500, Volume 5, Revision 1 (to be published).

3.3.3-2. Swain, A. D., and H. E. Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-1278, August 1983.

3.3.3-3. Embrey, D. E., "The Use of Performance Shaping Factors and Quantified Expert Judgement in the Evaluation of Human Reliability: An Initial Appraisal," Brookhaven National Laboratory, prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-2986, May 1983.

3.3.3-4. Embrey, D. E., et al., "SLIM-MAUD: An Approach To Assessing Human Error Probabilities Using Structured Expert Judgment," Brookhaven National Laboratory, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3518 (two volumes), March 1984.

3.3.3-5. Rosa, E. A., et al., "Application of SLIM-MAUD: A Test of an Interactive Computer-Based Method for Organizing Expert Assessment of Human Performance and Reliability," Brookhaven National Laboratory, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-4016, September 1985.

3.3.3-6. Dykes, A. A., W. G. He, J. W. Stetkar, and G. A. Tinsley, "Database of Dynamic Human Actions for Calibration of the Conversion of Failure Likelihood Indexes to Human Error Rates," PLG document to be published.

3.3.3-7. PLG, Inc., "BARP (Bayesian Reliability Program)," PLG-0625, Release 2, dated May 1991.

3.3.3-8. U.S. Nuclear Regulatory Commission, "Modeling Time to Recovery of Loss of Off-Site Power at Nuclear Power Plants," NUREG/CR-5032, December 1987.

3.3.3-9. U.S. Nuclear Regulatory Commission, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," NUREG-1032.

3.3.3-10. NUREG/CR-5116-Vol. 1, "Results of Expert Opinion Elicitation on Internal Event Front-End Issues For NUREG-1150: Expert Panel," Sandia 88-0642, April 1988.

3.3.3-11. Watts Bar One-Loop RETRAN-02 Model Control Volume Flow Junction Geometric Calculations TI-ANL-89, Rev. 1, May 23, 1991.

3.3.3-12. Station Blackout Coping and Recovery, SQN-SBO-001, March 30, 1989.

Table 3.3.3-1 (Page 1 of 2). Generic Database Variables Used for System Analysis Screening of Preinitiating Event, Routine Human Error Caused, Undetected Unavailability following Maintenance and Testing

Location of Surveillance	Complexity (See Note on Page 2)	Verification (See Note on Page 2)	Type of Action					
			Realignment Using Manual Controls and Switches Provided by the Design		Realignment from Jumpered Circuits or Other Temporary Plant Modification		Calibrations Left Misaligned or at Unresponsive Setpoints	
			Variable (Note 3)	Mean	Variable	Mean	Variable	Mean
Control Room Area (Includes backs of panels and/or associated equipment)	Low	Yes	ZHERCL	2.0-3	ZHEJCL	1.8-3	ZHECCL	4.9-3
		No	*		*		*	
	Medium	Yes	ZHERCM	5.9-3	ZHEJCM	4.9-3	*	
		No	*		*		*	
	High	Yes	*		*		*	
		No	*		*		*	
Local (single location exterior to the control room area)	Low	Yes	ZHERLL	3.4-3	ZHEJLL	3.2-3	ZHECLL	6.2-3
		No	*		*		*	
	Medium	Yes	ZHERLM	1.5-2	ZHEJLM	1.2-2	*	
		No	*		*		*	
	High	Yes	*		*		*	
		No	*		*		*	
Multiple Locations (excluding the control room area)	Low	Yes	ZHERML	1.0-2	ZHEJML	9.6-3	ZHECML	1.6-2
		No	*		*		*	
	Medium	Yes	ZHERMM	3.2-2	ZHEJMM	2.7-2	*	
		No	*		*		*	
	High	Yes	*		*		*	
		No	*		*		*	

*Refer assessments not having a generic variable associated with it to the human action analyst for a system-specific evaluation. The bases and derivation of the distribution of each generic database variable is contained in Reference 3.3.3-1.

Note: Exponential notation is indicated in abbreviated form; 2.0-3 = 2.0×10^{-3}.

Table 3.3.3-1 (Page 2 of 2). Generic Database Variables Used for System Analysis Screening of Preinitiating Event, Routine Human Error Caused, Undetected Unavailability following Maintenance and Testing

Notes:

1. Complexity Guidance:

Select low complexity only if it is clear that all criteria are satisfied.
Select medium complexity only if no more than two low complexity criteria are out of tolerance.

Low:
- Single objective.
- Very clear procedures (one action/step with individual checkoff, outline or columnar form, easy to interpret).
- Less than 10 closely associated calibrations and/or restorations.
- Items clearly marked and separated.
- Small team working directly with each other.

Medium:
- Repetitive or coordinated objectives.
- Clear procedures (one action/step, "critical steps" having checkoff, narrative form, easy to interpret).
- Less than 10 restorations of varying types.
- Items clearly marked in same general area.
- Team in more than one location with dedicated communication.

High:*
- Diverse objectives.
- More than 10 restorations.
- Items ambiguously marked or in close proximity.
- Team in multiple locations with intermittent communication.
- Any consideration that make assignment of either low or medium complexity uncertain.

2. Verification Guidance:

Yes: Second person verifies and signs off in a separate space provided for that purpose (low dependency between checker and testers).
No: Two people working together verify realignment, or less.* (Moderate or high dependency between checker and testers.)

3. Legend:

ZHERLL
- Complexity (low)
- Location (single local)
- Type of Action (realignment with controls)

*Refer assessments not having a generic variable associated with it to the human action analyst for a system-specific evaluation.

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Table 3.3.3-2. Summary of Routine Human Errors Included in the Systems Analyses To Account for Preinitiating Event, Routine Human Error Caused, Undetected Unavailability following Maintenance and Testing

System Notebook	Top Event	Test Number	Test Name	Database Variable	Description of Error
AFW	AF	SI-7.45	AFW Valve Operability Test	ZHEJLM	Failure to remove jumpers after test.
CONT	SU	SI-6.28	Containment Refueling Canal Drains, Rev 8	WHESUM	Operators fail to remove refueling canal drain plugs following refueling. (Plant specific analysis, see App B, Section B.3.2)
CS	CSA(B)	SI-4.0.5.72.P.1(2).A(B).0	Quarterly Performance Test	ZHERLL	Following completion of text, the potential that the operator fails to properly realign the CCS train for normal configuration exits.
EPS	GAIV	SI-8.1 Rev. 19	Diesel Generator Start and Load Test	ZHEJLL	Pair of fuses are removed and may not be reinstalled at conclusion of test.
EPS	GAIV	SI-8.1	Diesel Generator Start and Load Test	ZHEJLL	Failure to reinstall fuses after test
RHR	RA, RABX	SI-4.0.5.74.V.1.A.0	Valve Full Stroke Exercising During Plant Operation - Residual Heat Removal Train A (or B)	ZHERCL	Handswitches of the RHR suction isolation valve and pump miniflow valve have potential to be left in closed position not AUTO position.
RHR	RABX	SI-4.0.5.74.V.2.B.0	Valve Full Stroke Exercising During Plant Operation - Residual Heat Removal Train B (or A)	ZHERCL	Handswitches of the RHR suction isolation valve and pump miniflow valve have potential to be left in closed position not Auto position.
SIS	S1, SP	SI-4.0.5.63.V.1.A.0	Valve Full Stroke Exercision During Plant Operation - Safety Injection System (Train A)	ZHERCL	The pump control switch of the SIS pump has the potential of being left in the PULL-TO-LOCK position instead of AUTO position.
SIS	S1, SP	SI-4.0.5.63.V.1.A.0	Valve Full Stroke Exercising During Plant Operation - Safety Injection System (Train A)	ZHERCL	The pump control switch of the SIS pump has the potential of being left in the PULL-TO-LOCK position instead of AUTO position.
SIS	SP	SI-4.0.5.63.V.2.B.0	Valve Full Stroke Exercising During Plant Operation - Safety Injection System (Train B)	ZHERCL	The pump control switch of the SIS pump has the potential of being left in the PULL-TO-LOCK position instead of the AUTO position.

Table 3.3.3-3. Assessment of Likelihood that Drain Plugs are Left in Refueling Canal Following Refueling

Procedure/Step	Critical Tasks	Median HEP	Range Factor	NUREG/CR-1278 Table (item)	Reasoning
MI-068.1					
7.8	Remove both drain plugs from refueling canal.	0.003	3	Table 20-7 (2)	Checkoff step in a procedure > 10 steps.
	Total Unrecovered HEP	0.003	3		
Procedure/Step	**Recovery Factors**	**Recovery Factor**	**Range Factor**	**NUREG/CR-1278 Table (item)**	**Reasoning**
FHI-8					
Step 10	Verify drain plug removed and vortex eliminator installed.	0.025	5	Table 20-22 (3 and 11)	Special one-of-a-kind check by operator of maintenance activity. No credit taken for difference in time and procedrue.
SI-6.28					
Para 4.1	Verfiy drain plug removed and vortex eliminator installed and drain free of debris. (This verification is the only objective of the test.)			Zero dependence with previous check.	Surveillance by different people at separate time.
	Accomplish surveillance.	.001	5	Table 20-6 (4) (abnormal operating condition)	Administrative control that test is done during recovery from refueling with mode 4 checklist.
	Product of Recovery Factors	2.5×10^{-5}	10		

Assessed HEP = Total HEP*Recovery Factors
- Assessed HER (Median) 7.5E-8 Range Factor = 13
- Assessed HER (Mean) 2.4E-7
- Assessed HER (95th Percentile) 9.4E-7

Average unavailability given SI-6.28 is performed quarterly with an 18-month refueling cycle (divide by 6)
- Assessed Median Unavailability = 1.3E-08 per demand
- Assessed Mean Unavailability = 4.1E-08 per demand
- Assessed 95th Unavailability = 1.6E-07 per demand

Table 3.3.3-4 (Page 1 of 4). Summary Descriptions of Dynamic Human Actions Evaluated for the Watts Bar PRA

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
AC	HAAC1	Start Standby CCS Pump, Given Running Pump Fails during Non-SI Events.	Two minutes to avoid overheating the Reactor Coolant Pumps.	0.025100
AC	HAAC2	Isolate CCS Train A from Spent Fuel Pool Heat Exchanger	Not time sensitive - assume 45 minutes for heatup of cooled equipment.	0.014100
AE	HAAE1	Start Standby ERCW Pump, Given Running Pump Fails During Non-Safety Injection Events	Not time sensitive - assume 45 minutes for heatup of cooled equipment lower ERCW flow.	0.001500
AE	HAAEIE	Start Standby ERCW Pump To Avert Plant Trip, Given Running Pump Fails during Normal Opertions	Not time sensitive - assume 45 minutes for heatup of cooled equipment at lower ERCW flow.	0.000387
AF	HAAF1	Locally Operate TDAFW Valve, Given Loss of All AC power	Approximately 1 hour to steam generator dryout.	0.156000
CH	HACH1	Transfer Spray From RWST to Sump, Given RHR Swapover was Successful	Four minutes from 8% RWST level until loss of suction to containment spray pumps.	0.000055
CH	HACH2	Transfer Spray from RWST to Sump, Given RHR Swapover Failed	Four minutes from 8% RWST level until loss of suction to containment spray pumps.	0.006930
CI	HACI1	Backup Containment Isolation	Not time sensitive - assume approx. 3 hrs to avoid containment conditions that could result in radioactive releases to the environment	0.113000
CSA CSB	HACS1	Backup CS Pump Initiation, Given Containment Pressure > 2.81 psig	Approx. 20 min. after signal to avoid containment conditions that could result in release of radioactive materials in the environment.	0.002160
CTMU	HACT1	Refill CST during Non-LOCA Events	CST can provide up to 16 hours of makeup if initially at 190,000 gallons.	0.003630
DS	HADS1	Cool Down and Depressurize RCS, Normal Cooldown	Approximately 16 hours before makeup to CST required.	0.002680
DS	HADS2	Cool Down and Depressurize RCS, Given an SGTR with Successful Isolation	Approximately 16 hours before makeup to CST required.	0.001930
DS	HADS3	Cool Down and Depressurize RCS, Given SGTR with Successful Isolation, but High Head Safety Injection Failed	Approximately 16 hours before makeup to CST required.	0.015000

Table 3.3.3-4 (Page 2 of 4). Summary Descriptions of Dynamic Human Actions Evaluated for the Watts Bar PRA

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
DS	HADS4	Cool Down and Depressurize RCS, Given SGTR when Unable To Isolate Ruptured Steam Generator	Approximately 16 hours before makeup to CST required.	0.021500
DS	HADS5	Cool Down and Depressurize RCS, Given SGTR when Unable To Isolate Ruptured S/G and High Head SI Failed	Gain control of leakage within 30 minutes to avoid core uncovery, 16 hours of inventory in CST.	0.061500
DS	HADS6	Cool Down and Depressurize RCS, Given Loss of All AC Power	Initiate as soon as possible to avoid seal LOCA. Seal damage in approx. 1 hour if RCS temperature not lowered by then.	0.101000
DS	HADS7	Cool Down and Depressurize RCS, Given SLOCA with Loss of High Head Safety Injection	Initiate within 30 minutes. Reach RHR setpoints within 2 hours, for a total of 3 hours to avoid sump swapover.	0.034700
EB	HAEB1	Trip CRD MG Power and Initiate Boration, Given ATWS	Approximately 10 minutes.	0.034800
HH	HAHH1	Place Hydrogen Ignitors in Service	Not time sensitive.	0.002720
MR	HAMR1	Manually Insert Control Rods, Given ATWS	One minute or less.	0.008930
MU	HAMU1	Make Up RWST Inventory Following a SGTR Event	Approximately 8 hours of inventory available when RWST = 70%.	0.024600
MU	HAMU2	Make Up RWST Inventory, Given LOCA with Loss of Recirculation	Approximately 10 minutes to empty RWST. Approximately 20 minutes to core uncovery.	0.441000
MU	HAMU3	Make Up RWST, Given LOCA with Loss of Recirculation and Containment Spray	Approximately 10 minutes to empty RWST. Approximately 20 minutes to core uncovery.	0.721000
OB	HAOB1	Establish RCS Feed and Bleed, Given Insufficient Secondary Heat Sink	Approximately 50 minutes to S/G dryout and RCS pressure increase to above PORV setpoint, given RCPs continue.	0.025200
OF	HAOF1	Restore MFW, Given AFW Failed During General Tranisient Not Requiring SI	Approximately 33 minutes to S/G WR < 25%, which requires feed and bleed.	0.038800
OF	HAOF2	Restore Main Feedwater, Given AFW Failed During Transient Requiring SI - SLOCA	Approximately 30 minutes to S/G WR < 25%, which requires feed & bleed.	0.048500
OS	HAOS1	Align ECCS for Core Cooling, Given ESFAS Fails Following a MLOCA or LLOCA	Approximately one minute to mitigate cladding failure.	0.036200

Table 3.3.3-4 (Page 3 of 4). Summary Descriptions of Dynamic Human Actions Evaluated for the Watts Bar PRA

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
OS	HAOS2	Align ECCS for Core Cooling, Given OSFAS Fails, Given a MSLBOC, SGTR, or SLOCA	Bounded by 30 minutes to cladding damage during SLOCA.	0.031700
OS	HAOS3	Manually Start AFW, Given Reactor Trip with No SI Required	Approximately 50 minutes before steam generator dryout.	0.002090
OS	HAOS4	Start AFW, Given ATWS with AMSAC Failure	One minute to steam generator dryout.	0.012200
OS	HAOS5	Backup Restart Timers, Given LOSP and D/G Startup	Approximately 1 hour before steam generator dryout.	0.128000
OT	HAOT1	Place Containment Spray in Standby and Reset Signal	Within 5 minutes of pressure reduction to permissible limits.	0.001650
PR PI	HAPI1	Isolate Open PORV by Closing Block Valve after Depressurizing Using the PORV.	Approximately 5 minutes to PRT disk rupture	0.001430
PR PI	HAPR1	Isolate Open PORV by Closing Block Valve after SI Actuated	Approximately 5 minutes to PRT disk rupture.	0.001430
PR	HAPR2	Isolate Open PORV by Closing Block Valve Prior to Depressurizing to 1,870 psig SI Initiation Signal	Within 30 seconds to avoid SI actuation.	0.005390
RD	HARD1	Place RHR Cooling in Service Following SGTR	Approximately 2 hours available after RHR entry conditions reached until RWST makeup required.	0.006150
NOT USED	HARE1	Shed DC Bus Loads, Given Loss of All AC Power	At least 2 hours [without recharging under normal bus loads] before batteries are depleted.	0.169000
NOT USED	HARE2	Locally Restore RHR Sump Recirculation, When Unable to Transfer From Control Room	Core damage assumed to begin 45 minutes after switchover fails.	0.715000
RH	HARH1	Transfer To Hotleg Recirculation, Given LOCA > 2-Inch Diameter	Not time sensitive - action is required before the time blockage occurs.	0.001780
RR	HARR1	Align High Pressure Recirculation, Given Auto Swapover Succeeds	Approximately 20 minutes from swapover to core uncovery if swapover not completed.	0.001860
RR	HARR2	Align High Pressure Recirculation, Given Auto Swapover Fails	Approximately 20 minutes from swapover to core uncovery if swapover not completed.	0.001860

Table 3.3.3-4 (Page 4 of 4). Summary Descriptions of Dynamic Human Actions Evaluated for the Watts Bar PRA

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
RS	HARS1	Place RHR Spray in Service, Given more than 1 Hour into Event and Containment Pressure High	Tens of minutes available before pressure alters containment conditions.	0.004670
RT	HART1	Manually Trip Reactor, Given SSPS Fails	Within one minute.	0.001510
SE	HASE1	Stop RCPs on Phase B Isolation, Given a Non-LOCA Initiator	Operators evaluated the condition that RCPs will run for 2 minutes before initiation of seal damage. Actually at least 10 min. avail.	0.030600
SE	HASE2	Stop RCPs on Loss of Train A CCS or RCP Cooling Path	At least 10 minutes before RCP damage begins.	0.024100
SL	HASL1	Identify and Isolate Ruptured Steam Generator	40 minutes available before steam generator overfill, given offset rupture of one tube.	0.001750
WC	HAWC1	Control SI To Prevent Water Challange of PORVs	Dependent on fill rate - assume no more than 5 minutes to recognize and react.	0.003720

Table 3.3.3-5. Guidance Regarding Information To Include in Operator Response Forms

TASK IDENTIFIER with the summary reproduced from operation action summary table.

PRECEDING EVENTS
- List initiating events after which action may be required.
- Briefly summarize sequence of events leading to action.
  - Base the sequences on the event sequence diagrams (ESD) and event tree descriptions.
  - Bound the range of possibilities (identify if influenced by initiating event).
- Identify any abnormal plant responses that may complicate the situation.

INDICATIONS OF PLANT CONDITION
- List what the operating crew sees that permits diagnosis that the action is required.
- Estimate how long the condition could exist before indications sufficient for diagnosis are available to the operators.
- Describe parallel indications that can mask the action requirement.

PROCEDURAL GUIDANCE/REQUIRED ACTIONS
- Reference the procedure and steps that will be followed.
- State whether the task is an immediate memorized action.
- Briefly summarize the aspects of the action that could influence the operators' ability to diagnose and accomplish it.
- Identify considerations in addition to procedures that could influence likelihood of success.

CONCURRENT ACTIONS/COMPETING FACTORS
- Identify concurrent actions that could compete for attention.
- Briefly describe alarms, environmental conditions, and other distractions that could impact the operating shift's concentration and produce stress.
- Discuss important aspects of the operator team interactions.

INDICATION OF SUCCESSFUL COMPLETION/SUCCESS IMPACT
- Characterize plant state upon completion based on ESD and event tree success criteria.
- Describe how the operators can determine they have been successful.

FAILURE IMPACT
- Characterize the plant condition following failure to accomplish based on ESD and event tree success criteria.
- Identify later actions the operators have available to respond with once the plant has made a transition to the failed condition.

TIME CONSTRAINTS
- List thermal/hydraulic and physical/equipment response considerations that influence time available before transition to failed condition.
- Summarize what is known about time required to both diagnose and accomplish the tasks.

Table 3.3.3-6. Qualitative Descriptions of Dynamic Human Actions

HAOF1: Restore Main Feedwater, Given AFW Failed during General Transient Not Requiring Safety Injection

PRECEDING EVENTS
- Reactor/turbine trip from full power.
- Station service transfer successful.
- MFW isolation complete.
- Steam generator level declined to < 8% narrow range.

INDICATIONS OF PLANT CONDITION
- AFW total flow < 470 gpm, status light RED.
- Steam generator = 60% WR.
- Shutdown boards energized by offsite power.

PROCEDURAL GUIDANCE/REQUIRED ACTIONS
- Status tree FR-H.1, step 1,5-13.
- Verify AFW flow < 470 gpm while steam generator NR level < 10% (E-0).
- Check steam generator WR levels to determine if 1 steam generator ≥ 25% WR.
  - Yes: Establish secondary heat sink.
  - No: Establish bleed and feed.
- Verify safety injection cleared or blocked, reset MFW isolation.
- Start MFW, condensate pumps, and align valves.

CONCURRENT ACTIONS/COMPETING FACTORS
- ES-0.1 actions.
- Monitoring status trees.

INDICATION OF SUCCESSFUL COMPLETION/SUCCESS IMPACT
- Steam generator level remains above 25% WR and stabilizes in acceptable range.
- Secondary heat sink reestablished.

FAILURE IMPACT
- Transition to feed and bleed (Actions OB1, OB2).

TIME CONSTRAINTS
- Approximately 45 minutes to steam generator WR level <25%, which requires transition to feed and bleed.

Table 3.3.3-7. Summary of the Relationship between the Scoring and Weighting Processes

Score: With respect to the things addressed by this PSF, are the conditions under which the action must be accomplished helping or hindering us to successfully complete it? In other words, we are rating the impact of the conditions on our ability to succeed in accomplishing the action. Interpretation of the range of scores

0-3 Helps

4-6 Is Neutral

7-10 Hinders

Weight: Does a variation between helping and hindering have more influence on the probability that we will successfully complete it than other PSFs? In other words, is this PSF a focus of the action? Do we key in on the things addressed by this PSF?

0 Insignificant compared to other PSFs.

1 Low: unimportant compared to other PSFs.

2 Normal: about the same as other PSFs.

4 High: much more important than other PSFs.

Weighting Thought Process

1. Initially set the weights of every PSF equal to 2.

2. Adjust weights of the PSFs only if you believe that their importance for judging the ease or difficulty of accomplishing the action is significantly (a factor of 2) greater or less than the other PSFs. The weights will be normalized so that the maximum overall failure likelihood index will be a 10, so the effect of increasing all of the weights is the same as increasing none.

3. Generally, actions requiring similar types of skills have the same PSF weights. Some examples of groups of actions where differences in the focus may require different PSF weights are as follows:

   - Immediate recognition and reaction.
   - Actions where diagnosis of need would dominate success.
   - Actions requiring a long sequence of manipulations.
   - Local actions involving coordination of activities.
   - Adjusting or controlling against indications.

Impact of Weight on How the Failure Likelihood Index Changes

Weight	Rating Change Producing the Same Change in the FLI
1	1 → 9
2	3 → 7
4	4 → 6

Table 3.3.3-8 (Page 1 of 3). Raw Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 1 Human Action Evaluations - Raw Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	Tot Wgt
AC1	4	6	2	2	2	7	2	1	2	2	2	1	2	3	16
AC2	3	3	2	4	2	3	2	3	2	4	2	9	2	4	15
AE1	2	3	2	0	2	1	0	0	4	1	2	1	2	4	14
AE2	3	8	2	7	2	6	2	6	3	7	2	6	2	7	16
AF1	4	7	2	9	3	6	2	1	4	10	2	4	4	9	21
CH1	2	3	2	2	2	3	2	0	2	2	2	2	2	2	14
CH2	4	7	2	5	4	7	2	1	2	2	2	1	2	4	18
CI1	4	8	2	5	2	1	2	10	4	8	2	9	2	6	18
CS1	4	5	2	2	2	4	2	1	2	2	2	5	2	5	16
CT1	2	7	2	5	2	1	2	1	2	5	2	1	2	0	14
DS1	2	2	2	2	2	2	2	2	2	6	2	4	2	4	14
DS2	2	5	2	2	2	1	2	0	2	5	2	4	2	2	14
DS3	3	5	2	6	2	2	2	2	2	6	2	6	2	4	15
DS4	4	5	2	8	4	9	2	1	4	9	2	2	2	10	20
DS5	4	7	2	8	4	10	2	1	4	10	2	2	2	10	20
DS6	4	7	3	9	3	10	2	2	4	10	2	2	3	10	21
DS7	3	5	2	2	2	5	2	1	2	6	2	2	2	5	15
EB1	2	0	2	8	4	7	2	0	4	9	3	2	2	5	19
HH1	2	5	2	7	2	2	2	0	2	2	2	5	2	4	14
MR1	2	0	2	2	4	10	2	1	2	2	2	1	2	4	16
MU1	3	8	2	8	2	1	2	4	2	6	2	6	2	2	15
MU2	2	4	2	8	4	10	1	8	2	9	4	6	2	9	17
MU3	2	7	2	8	4	F	1	9	1	9	4	9	4	10	18
OB1	4	5	2	3	2	5	2	1	2	5	2	5	3	8	17
OF1	2	5	2	7	3	5	1	7	4	9	3	7	2	5	17
OF2	2	7	2	7	3	6	1	2	4	9	3	7	2	5	17
OS1	2	8	2	8	4	6	4	5	2	8	4	7	3	8	21
OS2	2	8	2	8	4	6	4	5	2	8	4	7	3	8	21
OS3	2	3	2	2	2	2	2	2	2	2	2	2	2	2	14
OS4	4	7	2	3	4	9	2	3	2	2	2	3	2	5	18
OS5	4	8	4	8	4	10	1	10	4	8	4	9	4	8	25
OT1	3	5	2	2	1	1	2	0	2	2	2	1	2	4	14
PR1	4	8	2	3	4	4	3	0	2	2	2	1	2	5	19
PR2	3	8	2	3	4	9	3	0	2	2	2	1	2	6	18
RD1	3	5	2	6	2	1	2	4	4	7	2	8	2	5	17
RE1	4	8	2	8	2	5	1	8	3	8	2	7	3	6	17
RE2	2	8	2	7	4	F	2	10	4	10	2	10	4	10	20
RH1	2	3	2	2	2	2	2	0	2	2	2	6	2	4	14
RR1	4	3	2	2	2	5	2	0	2	2	2	1	2	4	16
RR2	2	6	2	2	3	5	2	3	2	4	2	8	2	5	15
RS1	2	4	2	2	2	5	2	0	2	2	2	5	2	4	14
RT1	1	4	2	2	3	7	0	0	4	2	4	0	2	0	16
SE1	4	7	2	9	4	7	2	0	2	2	4	1	2	4	20
SE2	4	6	2	9	4	10	2	5	4	8	2	8	2	6	20
SL1	2	2	2	1	3	5	2	1	2	5	3	1	2	4	16
WC1	2	5	2	3	4	6	2	2	2	3	2	1	2	4	16

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-8 (Page 2 of 3). Raw Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 2 Human Action Evaluations - Raw Weights

ID	Actions		Interface		Time		Procedure		Complex		Training		Stress		Tot
Code	W	S	W	S	W	S	W	S	W	S	W	S	W	S	Wgt
AC1	4	7	4	0	2	4	2	0	2	0	2	5	4	8	20
AC2	4	8	4	0	4	7	2	8	4	7	4	9	4	8	26
AE1	4	8	2	0	2	3	2	0	2	0	2	5	2	4	16
AE2	4	10	4	9	4	10	4	10	4	10	4	10	4	10	28
AF1	2	10	2	8	4	10	2	6	2	9	4	7	4	10	20
CH1	2	5	2	2	2	3	2	0	2	2	2	4	2	2	14
CH2	4	9	2	2	2	3	2	0	2	0	2	4	4	8	18
CI1	0	9	4	8	4	7	4	5	4	9	4	9	4	10	24
CS1	2	8	4	2	2	3	2	0	2	0	4	5	2	5	18
CT1	2	8	2	2	1	2	2	0	1	0	2	2	2	5	12
DS1	2	3	4	2	1	0	4	3	2	8	4	7	2	4	19
DS2	2	3	4	2	1	0	4	2	2	7	4	7	2	4	19
DS3	2	3	4	1	2	5	1	10	4	7	4	9	4	8	21
DS4	2	5	4	2	2	5	4	5	2	8	4	7	4	8	22
DS5	2	8	4	2	2	7	1	10	4	7	4	9	4	8	21
DS6	4	8	4	8	4	10	2	2	4	9	4	7	4	10	26
DS7	4	9	4	2	2	3	1	10	4	7	4	9	4	8	23
EB1	1	8	4	2	2	5	2	7	2	8	4	4	4	8	19
HH1	4	5	2	5	2	3	2	0	2	0	2	5	2	5	16
MR1	1	8	2	8	2	5	2	7	2	0	4	7	4	8	17
MU1	4	9	2	8	2	0	4	1	4	8	4	9	2	5	22
MU2	4	9	2	2	4	F	4	0	4	8	2	6	4	10	24
MU3	4	9	2	2	4	F	4	8	4	9	4	10	4	10	26
OB1	4	9	4	8	4	8	2	5	2	2	4	8	4	8	24
OF1	2	8	4	2	2	5	2	7	4	8	4	4	2	6	20
OF2	4	8	4	2	2	7	2	6	4	9	4	9	4	8	24
OS1	4	10	4	2	4	7	2	0	4	7	2	9	4	8	24
OS2	4	10	4	2	4	5	2	0	4	7	2	9	4	6	24
OS3	4	8	2	2	2	3	2	0	2	0	2	7	2	5	16
OS4	4	9	4	2	4	7	4	0	2	0	4	5	4	8	26
OS5	4	10	4	9	2	5	4	0	2	4	4	7	4	10	24
OT1	1	3	2	2	2	1	4	0	2	2	2	5	2	5	15
PR1	2	8	4	0	4	3	2	0	1	0	1	5	2	8	16
PR2	2	8	4	5	4	7	2	0	1	0	1	5	4	8	18
RD1	2	4	4	2	2	3	2	0	2	0	4	2	2	5	18
RE1	4	9	4	10	4	10	4	8	4	7	4	9	4	10	28
RE2	4	10	4	9	4	F	4	0	4	9	4	8	4	10	28
RH1	2	5	2	2	2	3	2	0	2	2	2	7	2	5	14
RR1	2	5	4	2	4	7	4	0	2	0	4	4	2	5	22
RR2	4	10	4	8	4	10	4	0	4	8	4	7	4	8	28
RS1	2	5	2	2	2	3	2	0	2	2	2	7	4	8	16
RT1	1	3	4	0	2	5	2	7	0	5	4	2	4	5	17
SE1	2	7	2	5	4	7	0	7	0	2	2	5	2	8	12
SE2	2	3	4	8	4	5	4	2	1	7	4	3	2	6	21
SL1	2	3	4	2	2	3	4	6	2	2	4	1	2	4	20
WC1	4	5	4	2	4	7	4	9	2	5	2	7	4	8	24

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-8 (Page 3 of 3). Raw Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 3 Human Action Evaluations - Raw Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	Tot Wgt
AC1	4	7	4	8	2	6	1	7	0	1	1	1	2	6	14
AC2	2	6	2	8	2	5	2	3	4	8	2	5	2	5	16
AE1	2	8	2	2	2	3	1	0	1	1	1	4	2	5	11
AE2	4	10	4	8	2	7	2	3	4	9	2	10	2	8	20
AF1	2	8	2	8	1	3	2	5	4	9	2	3	4	9	17
CH1	2	3	2	2	1	3	2	2	1	2	2	1	2	4	12
CH2	2	3	2	2	2	5	2	4	1	2	2	1	2	4	13
CI1	2	9	1	8	2	5	4	3	1	9	2	7	4	9	16
CS1	2	7	2	5	2	5	1	2	1	4	1	3	2	5	11
CT1	0	1	0	2	2	5	1	5	0	1	2	4	2	5	7
DS1	2	5	2	7	0	1	2	3	2	8	2	4	2	4	12
DS2	2	5	2	7	1	4	2	3	2	8	2	3	2	4	13
DS3	2	5	2	7	2	5	2	3	2	8	2	3	2	5	14
DS4	2	5	2	8	2	5	2	3	2	8	2	3	2	6	14
DS5	2	5	2	8	2	5	2	3	2	8	2	3	2	6	14
DS6	2	8	4	8	1	3	2	3	4	9	2	3	4	9	19
DS7	2	8	4	8	2	7	2	3	4	8	2	3	2	6	18
EB1	2	5	4	8	4	8	2	7	4	9	2	4	2	6	20
HH1	2	7	2	5	1	2	2	0	0	2	1	4	2	4	10
MR1	2	5	2	2	4	5	0	0	1	2	0	4	2	5	11
MU1	2	7	2	8	1	3	2	5	4	9	2	8	2	5	15
MU2	2	8	2	8	4	8	2	5	4	9	2	5	4	8	20
MU3	2	8	2	9	4	7	1	5	4	9	2	6	4	8	19
OB1	1	5	2	2	2	5	4	2	2	5	2	2	2	7	15
OF1	1	5	2	8	4	7	2	5	2	8	2	4	2	6	15
OF2	1	5	2	8	4	7	2	5	2	9	2	4	2	6	15
OS1	4	8	2	5	4	7	2	1	2	2	2	3	2	2	18
OS2	4	8	2	5	2	4	2	1	2	2	2	3	2	2	16
OS3	2	8	2	5	2	3	2	1	2	2	2	3	2	2	14
OS4	2	8	2	5	4	8	2	5	2	2	2	3	2	6	16
OS5	2	8	2	8	4	7	2	6	2	8	2	3	2	6	16
OT1	2	3	1	2	1	4	2	1	1	2	1	4	1	2	9
PR1	0	0	2	2	2	5	2	0	0	2	1	1	2	2	9
PR2	1	3	2	0	4	5	2	0	0	2	1	4	2	6	12
RD1	2	5	2	8	1	5	4	3	4	8	2	7	2	4	17
RE1	2	6	1	8	2	7	1	5	2	8	4	5	2	5	14
RE2	4	8	2	8	2	6	1	5	4	8	1	6	4	9	18
RH1	2	4	0	2	0	5	4	0	2	2	2	8	2	5	12
RR1	2	5	2	2	2	5	2	2	2	2	2	3	2	3	14
RR2	2	7	2	5	2	5	2	3	2	3	2	3	2	4	14
RS1	2	3	2	3	2	4	2	0	2	3	1	8	2	5	13
RT1	1	3	2	2	4	5	0	0	0	2	2	4	1	4	10
SE1	4	7	2	6	4	7	0	0	2	4	1	4	2	5	15
SE2	2	8	4	8	4	7	1	5	1	2	1	2	2	4	15
SL1	2	3	4	4	2	4	2	3	2	5	2	1	2	5	16
WC1	2	5	2	2	2	4	2	1	1	5	2	3	2	5	13

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-9 (Page 1 of 3). Normalized Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 1 Human Action Evaluations - Normalized Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedures W	Procedures S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	FLI
AC1	0.25	6	0.13	2	0.13	7	0.13	1	0.13	2	0.13	1	0.13	3	3.50
AC2	0.20	3	0.13	4	0.13	3	0.13	3	0.13	4	0.13	9	0.13	4	4.20
AE1	0.14	3	0.14	0	0.14	1	0.00	0	0.29	1	0.14	1	0.14	4	1.57
AE2	0.19	8	0.13	7	0.13	6	0.13	6	0.19	7	0.13	6	0.13	7	6.81
AF1	0.19	7	0.10	9	0.14	6	0.10	1	0.19	10	0.10	4	0.19	9	7.14
CH1	0.14	3	0.14	2	0.14	3	0.14	0	0.14	2	0.14	2	0.14	2	2.00
CH2	0.22	7	0.11	5	0.22	7	0.11	1	0.11	2	0.11	1	0.11	4	4.56
CI1	0.22	8	0.11	5	0.11	1	0.11	10	0.22	8	0.11	9	0.11	6	7.00
CS1	0.25	5	0.13	2	0.13	4	0.13	1	0.13	2	0.13	5	0.13	5	3.63
CT1	0.14	7	0.14	5	0.14	1	0.14	1	0.14	5	0.14	1	0.14	0	2.86
DS1	0.14	2	0.14	2	0.14	2	0.14	2	0.14	6	0.14	4	0.14	4	3.14
DS2	0.14	5	0.14	2	0.14	1	0.14	0	0.14	5	0.14	4	0.14	2	2.71
DS3	0.20	5	0.13	6	0.13	2	0.13	2	0.13	6	0.13	6	0.13	4	4.47
DS4	0.20	5	0.10	8	0.20	9	0.10	1	0.20	9	0.10	2	0.10	10	6.70
DS5	0.20	7	0.10	8	0.20	10	0.10	1	0.20	10	0.10	2	0.10	10	7.50
DS6	0.19	7	0.14	9	0.14	10	0.10	2	0.19	10	0.10	2	0.14	10	7.76
DS7	0.20	5	0.13	2	0.13	5	0.13	1	0.13	6	0.13	2	0.13	5	3.80
EB1	0.11	0	0.11	8	0.21	7	0.11	0	0.21	9	0.16	2	0.11	5	5.05
HH1	0.14	5	0.14	7	0.14	2	0.14	0	0.14	2	0.14	5	0.14	4	3.57
MR1	0.13	0	0.13	2	0.25	10	0.13	1	0.13	2	0.13	1	0.13	4	3.75
MU1	0.20	8	0.13	8	0.13	1	0.13	4	0.13	6	0.13	6	0.13	2	5.20
MU2	0.12	4	0.12	8	0.24	10	0.06	8	0.12	9	0.24	6	0.12	9	7.76
MU3	0.11	7	0.11	8	0.22 F		0.06	9	0.06	9	0.22	9	0.22	10	FAIL
OB1	0.24	5	0.12	3	0.12	5	0.12	1	0.12	5	0.12	5	0.18	8	4.82
OF1	0.12	5	0.12	7	0.18	5	0.06	7	0.24	9	0.18	7	0.12	5	6.65
OF2	0.12	7	0.12	7	0.18	6	0.06	2	0.24	9	0.18	7	0.12	5	6.76
OS1	0.10	8	0.10	8	0.19	6	0.19	5	0.10	8	0.19	7	0.14	8	6.86
OS2	0.10	8	0.10	8	0.19	6	0.19	5	0.10	8	0.19	7	0.14	8	6.86
OS3	0.14	3	0.14	2	0.14	2	0.14	2	0.14	2	0.14	2	0.14	2	2.14
OS4	0.22	7	0.11	3	0.22	9	0.11	3	0.11	2	0.11	3	0.11	5	5.33
OS5	0.16	8	0.16	8	0.16	10	0.04	10	0.16	8	0.16	9	0.16	8	8.56
OT1	0.21	5	0.14	2	0.07	1	0.14	0	0.14	2	0.14	1	0.14	4	2.43
PR1	0.21	8	0.11	3	0.21	4	0.16	0	0.11	2	0.11	1	0.11	5	3.68
PR2	0.17	8	0.11	3	0.22	9	0.17	0	0.11	2	0.11	1	0.11	6	4.67
RD1	0.18	5	0.12	6	0.12	1	0.12	4	0.24	7	0.12	8	0.12	5	5.35
RE1	0.24	8	0.12	8	0.12	5	0.06	8	0.18	8	0.12	7	0.18	6	7.18
RE2	0.10	8	0.10	7	0.20 F		0.10	10	0.20	10	0.10	10	0.20	10	FAIL
RH1	0.14	3	0.14	2	0.14	2	0.14	0	0.14	2	0.14	6	0.14	4	2.71
RR1	0.25	3	0.13	2	0.13	5	0.13	0	0.13	2	0.13	1	0.13	4	2.50
RR2	0.13	6	0.13	2	0.20	5	0.13	3	0.13	4	0.13	8	0.13	5	4.73
RS1	0.14	4	0.14	2	0.14	5	0.14	0	0.14	2	0.14	5	0.14	4	3.14
RT1	0.06	4	0.13	2	0.19	7	0.00	0	0.25	2	0.25	0	0.13	0	2.31
SE1	0.20	7	0.10	9	0.20	7	0.10	0	0.10	2	0.20	1	0.10	4	4.50
SE2	0.20	6	0.10	9	0.20	10	0.10	5	0.20	8	0.10	8	0.10	6	7.60
SL1	0.13	2	0.13	1	0.19	5	0.13	1	0.13	5	0.19	1	0.13	4	2.75
WC1	0.13	5	0.13	3	0.25	6	0.13	2	0.13	3	0.13	1	0.13	4	3.75

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-9 (Page 2 of 3). Normalized Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 2 Human Action Evaluations - Normalized Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	FLI
AC1	0.20	7	0.20	0	0.10	4	0.10	0	0.10	0	0.10	5	0.20	8	3.90
AC2	0.15	8	0.15	0	0.15	7	0.08	8	0.15	7	0.15	9	0.15	8	6.62
AE1	0.25	8	0.13	0	0.13	3	0.13	0	0.13	0	0.13	5	0.13	4	3.50
AE2	0.14	10	0.14	9	0.14	10	0.14	10	0.14	10	0.14	10	0.14	10	9.86
AF1	0.10	10	0.10	8	0.20	10	0.10	6	0.10	9	0.20	7	0.20	10	8.70
CH1	0.14	5	0.14	2	0.14	3	0.14	0	0.14	2	0.14	4	0.14	2	2.57
CH2	0.22	9	0.11	2	0.11	3	0.11	0	0.11	0	0.11	4	0.22	8	4.78
CI1	0.00	9	0.17	8	0.17	7	0.17	5	0.17	9	0.17	9	0.17	10	8.00
CS1	0.11	8	0.22	2	0.11	3	0.11	0	0.11	0	0.22	5	0.11	5	3.33
CT1	0.17	8	0.17	2	0.08	2	0.17	0	0.08	0	0.17	2	0.17	5	3.00
DS1	0.11	3	0.21	2	0.05	0	0.21	3	0.11	8	0.21	7	0.11	4	4.11
DS2	0.11	3	0.21	2	0.05	0	0.21	2	0.11	7	0.21	7	0.11	4	3.79
DS3	0.10	3	0.19	1	0.10	5	0.05	10	0.19	7	0.19	9	0.19	8	6.00
DS4	0.09	5	0.18	2	0.09	5	0.18	5	0.09	8	0.18	7	0.18	8	5.64
DS5	0.10	8	0.19	2	0.10	7	0.05	10	0.19	7	0.19	9	0.19	8	6.86
DS6	0.15	8	0.15	8	0.15	10	0.08	2	0.15	9	0.15	7	0.15	10	8.15
DS7	0.17	9	0.17	2	0.09	3	0.04	10	0.17	7	0.17	9	0.17	8	6.78
EB1	0.05	8	0.21	2	0.11	5	0.11	7	0.11	8	0.21	4	0.21	8	5.47
HH1	0.25	5	0.13	5	0.13	3	0.13	0	0.13	0	0.13	5	0.13	5	3.50
MR1	0.06	8	0.12	8	0.12	5	0.12	7	0.12	0	0.24	7	0.24	8	6.35
MU1	0.18	9	0.09	8	0.09	0	0.18	1	0.18	8	0.18	9	0.09	5	6.09
MU2	0.17	9	0.08	2	0.17	F	0.17	0	0.17	8	0.08	6	0.17	10	FAIL
MU3	0.15	9	0.08	2	0.15	F	0.15	8	0.15	9	0.15	10	0.15	10	FAIL
OB1	0.17	9	0.17	8	0.17	8	0.08	5	0.08	2	0.17	8	0.17	8	7.42
OF1	0.10	8	0.20	2	0.10	5	0.10	7	0.20	8	0.20	4	0.10	6	5.40
OF2	0.17	8	0.17	2	0.08	7	0.08	6	0.17	9	0.17	9	0.17	8	7.08
OS1	0.17	10	0.17	2	0.17	7	0.08	0	0.17	7	0.08	9	0.17	8	6.42
OS2	0.17	10	0.17	2	0.17	5	0.08	0	0.17	7	0.08	9	0.17	6	5.75
OS3	0.25	8	0.13	2	0.13	3	0.13	0	0.13	0	0.13	7	0.13	5	4.13
OS4	0.15	9	0.15	2	0.15	7	0.15	0	0.08	0	0.15	5	0.15	8	4.77
OS5	0.17	10	0.17	9	0.08	5	0.17	0	0.08	4	0.17	7	0.17	10	6.75
OT1	0.07	3	0.13	2	0.13	1	0.27	0	0.13	2	0.13	5	0.13	5	2.20
PR1	0.13	8	0.25	0	0.25	3	0.13	0	0.06	0	0.06	5	0.13	8	3.06
PR2	0.11	8	0.22	5	0.22	7	0.11	0	0.06	0	0.06	5	0.22	8	5.61
RD1	0.11	4	0.22	2	0.11	3	0.11	0	0.11	0	0.22	2	0.11	5	2.22
RE1	0.14	9	0.14	10	0.14	10	0.14	8	0.14	7	0.14	9	0.14	10	9.00
RE2	0.14	10	0.14	9	0.14	F	0.14	0	0.14	9	0.14	8	0.14	10	FAIL
RH1	0.14	5	0.14	2	0.14	3	0.14	0	0.14	2	0.14	7	0.14	5	3.43
RR1	0.09	5	0.18	2	0.18	7	0.18	0	0.09	0	0.18	4	0.09	5	3.27
RR2	0.14	10	0.14	8	0.14	10	0.14	0	0.14	8	0.14	7	0.14	8	7.29
RS1	0.13	5	0.13	2	0.13	3	0.13	0	0.13	2	0.13	7	0.25	8	4.38
RT1	0.06	3	0.24	0	0.12	5	0.12	7	0.00	5	0.24	2	0.24	5	3.24
SE1	0.17	7	0.17	5	0.33	7	0.00	7	0.00	2	0.17	5	0.17	8	6.50
SE2	0.10	3	0.19	8	0.19	5	0.19	2	0.05	7	0.19	3	0.10	6	4.62
SL1	0.10	3	0.20	2	0.10	3	0.20	6	0.10	2	0.20	1	0.10	4	3.00
WC1	0.17	5	0.17	2	0.17	7	0.17	9	0.08	5	0.08	7	0.17	8	6.17

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-9 (Page 3 of 3). Normalized Weights and Scores of Actions Evaluated by the Three Groups of Licensed Watts Bar Operators

WBN Group 3 Human Action Evaluations - Normalized Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	FLI
AC1	0.29	7	0.29	8	0.14	6	0.07	7	0.00	1	0.07	1	0.14	6	6.57
AC2	0.13	6	0.13	8	0.13	5	0.13	3	0.25	8	0.13	5	0.13	5	6.00
AE1	0.18	8	0.18	2	0.18	3	0.09	0	0.09	1	0.09	4	0.18	5	3.73
AE2	0.20	10	0.20	8	0.10	7	0.10	3	0.20	9	0.10	10	0.10	8	8.20
AF1	0.12	8	0.12	8	0.06	3	0.12	5	0.24	9	0.12	3	0.24	9	7.24
CH1	0.17	3	0.17	2	0.08	3	0.17	2	0.08	2	0.17	1	0.17	4	2.42
CH2	0.15	3	0.15	2	0.15	5	0.15	4	0.08	2	0.15	1	0.15	4	3.08
CI1	0.13	9	0.06	8	0.13	5	0.25	3	0.06	9	0.13	7	0.25	9	6.69
CS1	0.18	7	0.18	5	0.18	5	0.09	2	0.09	4	0.09	3	0.18	5	4.82
CT1	0.00	1	0.00	2	0.29	5	0.14	5	0.00	1	0.29	4	0.29	5	4.71
DS1	0.17	5	0.17	7	0.00	1	0.17	3	0.17	8	0.17	4	0.17	4	5.17
DS2	0.15	5	0.15	7	0.08	4	0.15	3	0.15	8	0.15	3	0.15	4	4.92
DS3	0.14	5	0.14	7	0.14	5	0.14	3	0.14	8	0.14	3	0.14	5	5.14
DS4	0.14	5	0.14	8	0.14	5	0.14	3	0.14	8	0.14	3	0.14	6	5.43
DS5	0.14	5	0.14	8	0.14	5	0.14	3	0.14	8	0.14	3	0.14	6	5.43
DS6	0.11	8	0.21	8	0.05	3	0.11	3	0.21	9	0.11	3	0.21	9	7.11
DS7	0.11	8	0.22	8	0.11	7	0.11	3	0.22	8	0.11	3	0.11	6	6.56
EB1	0.10	5	0.20	8	0.20	8	0.10	7	0.20	9	0.10	4	0.10	6	7.20
HH1	0.20	7	0.20	5	0.10	2	0.20	0	0.00	2	0.10	4	0.20	4	3.80
MR1	0.18	5	0.18	2	0.36	5	0.00	0	0.09	2	0.00	4	0.18	5	4.18
MU1	0.13	7	0.13	8	0.07	3	0.13	5	0.27	9	0.13	8	0.13	5	7.00
MU2	0.10	8	0.10	8	0.20	8	0.10	5	0.20	9	0.10	5	0.20	8	7.60
MU3	0.11	8	0.11	9	0.21	7	0.05	5	0.21	9	0.11	6	0.21	8	7.74
OB1	0.07	5	0.13	2	0.13	5	0.27	2	0.13	5	0.13	2	0.13	7	3.67
OF1	0.07	5	0.13	8	0.27	7	0.13	5	0.13	8	0.13	4	0.13	6	6.33
OF2	0.07	5	0.13	8	0.27	7	0.13	5	0.13	9	0.13	4	0.13	6	6.47
OS1	0.22	8	0.11	5	0.22	7	0.11	1	0.11	2	0.11	3	0.11	2	4.78
OS2	0.25	8	0.13	5	0.13	4	0.13	1	0.13	2	0.13	3	0.13	2	4.13
OS3	0.14	8	0.14	5	0.14	3	0.14	1	0.14	2	0.14	3	0.14	2	3.43
OS4	0.13	8	0.13	5	0.25	8	0.13	5	0.13	2	0.13	3	0.13	6	5.63
OS5	0.13	8	0.13	8	0.25	7	0.13	6	0.13	8	0.13	3	0.13	6	6.63
OT1	0.22	3	0.11	2	0.11	4	0.22	1	0.11	2	0.11	4	0.11	2	2.44
PR1	0.00	0	0.22	2	0.22	5	0.22	0	0.00	2	0.11	1	0.22	2	2.11
PR2	0.08	3	0.17	0	0.33	5	0.17	0	0.00	2	0.08	4	0.17	6	3.25
RD1	0.12	5	0.12	8	0.06	5	0.24	3	0.24	8	0.12	7	0.12	4	5.71
RE1	0.14	6	0.07	8	0.14	7	0.07	5	0.14	8	0.29	5	0.14	5	6.07
RE2	0.22	8	0.11	8	0.11	6	0.06	5	0.22	8	0.06	6	0.22	9	7.72
RH1	0.17	4	0.00	2	0.00	5	0.33	0	0.17	2	0.17	8	0.17	5	3.17
RR1	0.14	5	0.14	2	0.14	5	0.14	2	0.14	2	0.14	3	0.14	3	3.14
RR2	0.14	7	0.14	5	0.14	5	0.14	3	0.14	3	0.14	3	0.14	4	4.29
RS1	0.15	3	0.15	3	0.15	4	0.15	0	0.15	3	0.08	8	0.15	5	3.38
RT1	0.10	3	0.20	2	0.40	5	0.00	0	0.00	2	0.20	4	0.10	4	3.90
SE1	0.27	7	0.13	6	0.27	7	0.00	0	0.13	4	0.07	4	0.13	5	6.00
SE2	0.13	8	0.27	8	0.27	7	0.07	5	0.07	2	0.07	2	0.13	4	6.20
SL1	0.13	3	0.25	4	0.13	4	0.13	3	0.13	5	0.13	1	0.13	5	3.63
WC1	0.15	5	0.15	2	0.15	4	0.15	1	0.08	5	0.15	3	0.15	5	3.46

* Each action identifier code is preceded by an "HA" in the basic event database.

Table 3.3.3-10. Quantification of Operator Group 1 Evaluations into Human Error Rates*

Dynamic Human Action Evaluation for: Watts Bar Nuclear Plant, Unit 1
Evaluation Team: 1
Action Grouping Logic: A - All Performance Shaping Factors Equally Important

Action Code	Preceding & Conc. Actions Weight	Score	Plant Interfaces Weight	Score	Time Adequacy Weight	Score	Procedures Weight	Score	Complexity Weight	Score	Training & Experience Weight	Score	Stress Weight	Score	FLI	P(fail)	LOG(P(fail))
Rated Actions																	
MAX															9.72	1.0E+00	0.00
OS3	0.14	3	0.14	2	0.14	2	0.14	2	0.14	2	0.14	2	0.14	2	2.14	1.4E-05	-4.84
CT1	0.14	7	0.14	5	0.14	1	0.14	1	0.14	5	0.14	1	0.14	0	2.86	4.1E-05	-4.38
RS1	0.14	4	0.14	2	0.14	5	0.14	0	0.14	2	0.14	5	0.14	4	3.14	6.3E-05	-4.20
HH1	0.14	5	0.14	7	0.14	2	0.14	0	0.14	2	0.14	5	0.14	4	3.57	1.2E-04	-3.93
DS1	0.14	2	0.14	2	0.14	2	0.14	2	0.14	6	0.14	4	0.14	4	3.14	6.3E-05	-4.20
DS2	0.14	5	0.14	2	0.14	1	0.14	0	0.14	5	0.14	4	0.14	2	2.71	3.4E-05	-4.47
AF1	0.19	7	0.10	9	0.14	6	0.10	1	0.19	10	0.10	4	0.19	9	7.14	2.3E-02	-1.65
RH1	0.14	3	0.14	2	0.14	2	0.14	0	0.14	2	0.14	6	0.14	4	2.71	3.4E-05	-4.47
DS6	0.19	7	0.14	9	0.14	10	0.10	2	0.19	10	0.10	2	0.14	10	7.76	5.6E-02	-1.25
CH1	0.14	3	0.14	2	0.14	3	0.14	0	0.14	2	0.14	2	0.14	2	2.00	1.2E-05	-4.93
MIN															0.00	6.2E-07	-6.21
Calibration Actions																	
Seabrook ON	0.14	0	0.14	0	0.14	2	0.16	0	0.17	1	0.17	0	0.08	0	0.45	1.0E-06	-6.00
Plant B OR(1A)	0.14	5	0.14	5	0.14	4	0.16	5	0.17	5	0.17	5	0.08	5	4.86	1.0E-04	-4.00
Oyster Crk ZHEMU1	0.17	7	0.13	5	0.15	2	0.18	5	0.12	5	0.15	4	0.10	6	4.84	4.0E-03	-2.40
STP HE0003	0.14	6	0.14	6	0.15	8	0.14	5	0.15	6	0.14	6	0.14	9	6.58	3.0E-02	-1.52
EST_MAX															10.00	9.0E-01	-0.05

Regression Output:

Constant	-6.20
Std Err of Y Est	0.732
R Squared	0.923
No. of Observations	5
Degrees of Freedom	3
X Coefficient(s)	0.6389
Std Err of Coef.	0.1061

*Each action identifier code is preceded by an "HA" in the basic event database.

TAB33310.WBN.08/14/92

Table 3.3.3-11 (Page 1 of 2). Composite Human Error Rates Used for Quantification of the Watts Bar PRA

Action Code	WBN1 Median HER	WBN2 Median HER	WBN3 Median HER	Composite Human Error Rate				
				Mean	5th Percentile	50th Percentile	95th Percentile	Range Factor
HAAC1	2.0-03	2.9-03	4.2-02	2.51-02	5.34-04	3.87-03	8.90-02	14
HAAC2	3.9-03	1.0-02	1.2-02	1.41-02	1.00-03	6.31-03	3.96-02	6
HAAE1	2.9-04	2.0-03	1.5-04	1.50-03	2.33-05	3.76-04	4.82-03	16
HAAE2	3.7-02	1.0+00	1.7-01	4.39-01	1.12-02	1.45-01	1.00+00	9
HAAF1	2.3-02	2.8-01	5.0-02	1.56-01	6.31-03	5.39-02	5.41-01	10
HACH1	1.2-05	2.7-05	2.2-05	5.54-05	1.81-06	1.58-05	1.61-04	10
HACH2	5.6-03	7.0-03	5.7-05	6.93-03	1.12-05	2.53-03	2.15-02	45
HACI1	4.7-02	1.1-01	5.8-02	1.13-01	1.00-02	6.31-02	3.03-01	6
HACS1	2.2-03	3.5-04	7.4-04	2.16-03	5.49-05	7.57-04	6.34-03	11
HACT1	4.1-05	5.1-05	6.6-03	3.63-03	5.46-06	8.64-05	1.40-02	55
HADS1	6.3-05	2.6-03	2.2-03	2.68-03	1.26-05	1.00-03	8.12-03	26
HADS2	3.4-05	2.0-03	8.6-04	1.90-03	6.60-06	4.84-04	6.01-03	31
HADS3	5.1-03	2.1-02	1.2-03	1.50-02	3.98-04	4.14-03	4.76-02	12
HADS4	2.7-02	1.1-02	1.8-03	2.15-02	6.10-04	7.45-03	6.45-02	11
HADS5	6.5-02	4.8-02	1.8-03	6.15-02	6.13-04	2.25-02	1.88-01	19
HADS6	5.6-02	1.0-01	3.5-02	1.01-01	9.89-03	5.27-02	2.68-01	6
HADS7	2.6-03	4.5-02	1.6-02	3.47-02	8.60-04	1.13-02	1.06-01	12
HAEB1	4.6-03	4.8-03	5.6-02	3.48-02	1.00-03	7.07-03	1.20-01	12
HAHH1	1.2-04	2.0-03	2.8-03	2.72-03	2.38-05	1.05-03	8.13-03	19
HAMR1	1.2-03	1.4-02	6.0-04	8.93-03	1.08-04	1.55-03	3.09-02	18
HAMU1	1.1-02	4.8-03	3.0-02	2.46-02	1.48-03	1.00-02	7.09-02	8
HAMU2	1.1-01	1.0+00	8.8-02	4.41-01	2.29-02	1.58-01	1.00+00	7
HAMU3	4.0-02	1.0+00	1.0-01	7.21-01	3.27-02	1.00+00	1.00+00	6
HAOB1	7.3-03	3.4-02	4.7-03	2.52-02	1.25-03	8.82-03	7.68-02	9
HAOF1	3.9-02	1.2-02	2.1-02	3.88-02	3.26-03	1.80-02	1.02-01	6
HAOF2	4.4-02	2.1-02	2.5-02	4.85-02	4.53-03	2.51-02	1.28-01	6
HAOS1	5.3-02	7.8-03	7.0-03	3.62-02	1.58-03	1.02-02	1.16-01	9
HAOS2	5.3-02	2.9-03	3.6-03	3.17-02	6.31-04	5.07-03	1.14-01	14

Note: Exponential notation is indicated in abbreviated form; e.g., 2.0-03 = 2.0 × 10^{-03}.

TAB33311.WBN.07/13/92

Table 3.3.3-11 (Page 2 of 2). Composite Human Error Rates Used for Quantification of the Watts Bar PRA								
Action Code	WBN1 Median HER	WBN2 Median HER	WBN3 Median HER	Composite Human Error Rate				
				Mean	5th Percentile	50th Percentile	95th Percentile	Range Factor
HAOS3	1.4-05	3.6-03	9.6-05	2.09-03	2.95-06	8.49-05	7.89-03	55
HAOS4	1.2-02	6.9-04	9.6-03	1.22-02	1.35-04	4.95-03	3.72-02	17
HAOS5	2.5-01	1.3-02	3.0-02	1.28-01	3.98-03	3.15-02	4.74-01	12
HAOT1	6.7-04	4.3-04	7.5-04	1.65-03	5.08-05	5.14-04	4.78-03	11
HAPR1	2.3-03	1.6-04	5.0-05	1.43-03	8.88-06	1.86-04	5.04-03	26
HAPR2	6.3-03	3.3-03	2.0-04	5.39-03	3.91-05	1.99-03	1.60-02	21
HARD1	6.5-03	9.1-05	4.8-03	6.15-03	1.71-05	2.20-03	1.87-02	34
HARE1	5.8-02	3.5-01	9.8-03	1.69-01	3.15-03	4.71-02	6.31-01	15
HARE2	5.6-01	1.0+00	8.7-02	7.15-01	2.53-02	1.00+00	1.00+00	6
HARH1	3.4-05	9.6-05	3.1-03	1.78-03	5.96-06	1.19-04	6.73-03	36
HARR1	7.2-04	2.1-03	6.3-05	1.86-03	1.25-05	4.85-04	5.90-03	23
HARR2	3.5-03	2.8-02	3.4-04	1.86-03	1.25-05	4.85-04	5.90-03	23
HARS1	6.3-05	8.2-03	9.0-05	4.67-03	8.85-06	1.46-04	1.80-02	49
HART1	5.9-04	1.1-03	4.3-04	1.51-03	5.88-05	6.29-04	4.19-03	9
HASE1	5.3-03	2.7-02	2.4-02	3.06-02	1.58-03	1.43-02	8.77-02	8
HASE2	3.1-02	7.1-03	6.7-03	2.41-02	1.58-03	9.99-03	7.02-02	7
HASL1	1.3-03	9.2-04	2.4-04	1.75-03	3.99-05	6.31-04	4.96-03	12
HAWC1	1.2-03	5.4-03	1.0-04	3.72-03	2.00-05	9.01-04	1.22-02	26

Note: Exponential notation is indicated in abbreviated form; e.g., 2.0-03 = 2.0×10^{-03}.

Table 3.3.3-12 (Page 1 of 2). Summary of Watts Bar Recovery Actions Incorporated into the Plant Model

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
IE	FLAB1R	Identify and Isolate ERCW Header Flood in Auxiliary Building	Hypothesized break large enough to flood RHR and CS pumps within 30 minutes.	0.001010
IE	FLAB3C	Identify and Isolate a Break in the Condensate Storage Tank Discharge Piping, Given Break Can be Isolated	Hypothesized break large enough to release 200,000 gallons within 20 minutes.	0.008860
IE	FLPH1R	Identify and Isolate a Line Break in Intake Piping A1A-A (or one of other three systems) in the Strainer Room	Assume up to 45 minutes to isolate leak before component cooling problems result in the loss of cooled equipment.	0.001550
IE	FLPH2R	Identify and Isolate an ERCW Intake Line Break in the Pump Room of ERCW Intake	Assume up to 45 minutes to isolate leak before component cooling problems require reactor trip.	0.001550
AF	HAFR1	Restore Auxiliary Feedwater Flow, Given Loss of Control Air.	Approximately 50 minutes available after air accumulators lost to steam generator dryout.	0.005030
RL	HARL1	Recover from an Automatic Sump Swapover Failure	Approximately 5 minutes before RWST reaches a point at which pumps lose suction.	0.004030
CCSR	HCCSR1	Align the C-S Pump to the A CCS Heat Exchanger	Estimate 5 to 10 minutes available before charging pump failure.	0.194000
CCPR	HCCSR2	Align and Initiate Alternate Component Cooling to the Charging Pump	Estimate 5 to 10 minutes available before charging pump failure.	0.016100
CCSR	HCCSR3	Align ERCW Header 2A to CCS Heat Exchanger A, Given Loss of B Train ERCW.	Assume 45 minutes available before vital equipment begins to fail due to overheating.	0.001550
	HCRL1	Inadvertently Reset SI Signal, Failure of Automatic Sump Swapover	Not applicable - error of commission, Swain and Guttman estimate	0.003750
DA DB DC DD	HDAR1	Switch to Spare Battery Charger, Given Operating Charger Fails.	At least 2 hours (without recharging under normal bus loads) before batteries are depleted.	0.017800
DS	HDSR1	Cooldown and Depressurize by Cycling S/G PORV's Full Open/Full Closed.	Continuous control requirement until RCS below secondary system pressure. Respond to anomalies within 5 minutes.	0.001250

Table 3.3.3-12 (Page 2 of 2). Summary of Watts Bar Recovery Actions Incorporated into the Plant Model

Top Event	Database Variable	Definition of Action	Time Constraints	Mean HER /Demand
DSLR	HERCW1	Recover ERCW Cooling To Operating Diesel Generator, Given Loss of ERCW to the Diesel Generator	Trip the diesel generators within 5 minutes to avoid overheating failure. Complete alignment and restart within 30 minutes.	0.035500
No Credit Taken	HERCW2	Cool Down with Auxiliary Feedwater and Steam Generator PORV's, Given Total Loss of ERCW, RSW Available To Cool Air Compressor	Initiate cool down within 10 minutes (as soon as possible) to avoid seal damage at 1 hour. Control over 2 hours.	0.012500
DSLR	HERCW3	Cooldown with Auxiliary Feedwater and S/G PORV's, Given Total Loss of ERCW, RSW Not Available To Cool Air Compressors	Initiate cooldown within 10 minutes to avoid seal damage at 1 hour. Control over 2 hours.	0.050000
TPR	HSLR1	Locally Transfer Steam Supply to TDAFP, Given Station Blackout and Loss of Steam Generator #1	Approximately 1 hour to steam generator dryout, given RCPs not running.	0.017800
TPR	HTPR1	Start Turbine Driven Pump, Given it Failed To Start due To Control or Signal Failures	Steam Generator dryout at approximately 50 minutes if RCPs running.	0.008250
V1R V2R	HV1R1	Restore Ventilation to 6.9-k V Switchgear Room	At least 12 hours to accomplish before temperature approaches design limits.	0.000373
VNV1R VNV2R	HVNVR1	Restore Ventilation to the 480V Board Room 1BB (2BB), Given Loss of Room Supply Fan	Six hours to heat up past allowable limits.	0.002080
VT1AR VT1BR VT2AR VT2BR	HVT1AR	Establish Portable Ventilation to the Shutdown Board Transformer Room	Five hours to heat up past allowable limits, given four fully loaded tranformers.	0.002080

Table 3.3.3-13. Raw and Normalized Weights and Scores of Actions Evaluated by the Team of Licensed Operators Evaluating Recovery Actions

WBN Recovery Action Evaluations - Raw Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	Tot Wgt
AFR1	1	3	2	1	2	6	1	0	4	7	4	1	2	4	16
CCSR1	2	9	2	7	4	8	0	10	2	7	2	10	1	6	13
CCSR2	2	7	2	7	4	6	1	3	2	1	2	9	1	5	14
DHARL1	2	2	4	2	3	7	4	1	1	5	2	1	2	2	18
DSR1	1	1	2	4	2	4	1	1	4	1	4	1	2	1	16
ERCWR1	2	8	2	7	4	10	0	0	2	3	2	1	1	6	13
ERCWR2	1	8	2	2	2	7	1	1	4	5	4	4	2	5	16
ERCWR3	1	8	2	8	2	7	1	1	4	9	4	4	2	5	16
FLAB1R	2	1	2	4	3	2	2	0	2	3	4	4	1	5	16
FLAB3C	2	8	2	6	3	4	2	8	2	0	4	4	1	6	16
FLAB3R	GUARRANTEED FAILED - CAN NOT ISOLATE THE LONGEST PIPE														
FLPH1R	2	1	2	4	3	4	2	0	2	3	4	4	1	5	16
FLPH2R	2	1	2	4	3	4	2	0	2	3	4	4	1	5	16
SLR1	2	8	2	9	4	6	0	8	2	1	2	4	1	8	13
TPR1	4	0	1	0	2	3	2	0	2	8	2	1	3	6	16
V1R	2	8	0	4	1	0	1	2	4	1	1	2	1	1	10
VNVR1	2	8	0	4	1	0	1	10	4	1	1	4	1	5	10
VT1AR1	2	8	0	4	1	0	1	10	4	1	1	4	1	5	10

WBN Recovery Action Evaluations - Normalized Weights

ID Code	Actions W	Actions S	Interface W	Interface S	Time W	Time S	Procedure W	Procedure S	Complex W	Complex S	Training W	Training S	Stress W	Stress S	FLI
AFR1	0.06	3	0.13	1	0.13	6	0.06	0	0.25	7	0.25	1	0.13	4	3.56
CCSR1	0.15	9	0.15	7	0.31	8	0.00	10	0.15	7	0.15	10	0.08	6	8.00
CCSR2	0.14	7	0.14	7	0.29	6	0.07	3	0.14	1	0.14	9	0.07	5	5.71
DHARL1	0.11	2	0.22	2	0.17	7	0.22	1	0.06	5	0.11	1	0.11	2	2.67
DSR1	0.06	1	0.13	4	0.13	4	0.06	1	0.25	1	0.25	1	0.13	1	1.75
ERCWR1	0.15	8	0.15	7	0.31	10	0.00	0	0.15	3	0.15	1	0.08	6	6.46
ERCWR2	0.06	8	0.13	2	0.13	7	0.06	1	0.25	5	0.25	4	0.13	5	4.56
ERCWR3	0.06	8	0.13	8	0.13	7	0.06	1	0.25	9	0.25	4	0.13	5	6.31
FLAB1R	0.13	1	0.13	4	0.19	2	0.13	0	0.13	3	0.25	4	0.06	5	2.69
FLAB3CR	0.13	8	0.13	6	0.19	4	0.13	8	0.13	0	0.25	4	0.06	6	4.88
FLAB3RR	GUARRANTEED FAILED - CAN NOT ISOLATE THE LONGEST PIPE														FAIL
FLPH1R	0.13	1	0.13	4	0.19	4	0.13	0	0.13	3	0.25	4	0.06	5	3.06
FLPH2R	0.13	1	0.13	4	0.19	4	0.13	0	0.13	3	0.25	4	0.06	5	3.06
SLR1	0.15	8	0.15	9	0.31	6	0.00	8	0.15	1	0.15	4	0.08	8	5.85
TPR1	0.25	0	0.06	0	0.13	3	0.13	0	0.13	8	0.13	1	0.19	6	2.63
V1R	0.20	8	0.00	4	0.10	0	0.10	2	0.40	1	0.10	2	0.10	1	2.50
VNVR1	0.20	8	0.00	4	0.10	0	0.10	10	0.40	1	0.10	4	0.10	5	3.90
VT1AR1	0.20	8	0.00	4	0.10	0	0.10	10	0.40	1	0.10	4	0.10	5	3.90

Table 3.3.3-14 (Page 1 of 2). Recovery Human Error Rate Distributions Used for Quantification of the Watts Bar PRA

Database Variable	Brief Description	Top Event	Mean HER/ Demand	5th	Median HER/ Demand	95th
FLAB1R	Identify and Isolate ERCW Header Flood in Auxiliary Building	IE	0.001010	0.000036	0.000369	0.003680
FLAB3C	Identify and Isolate a Break in the Condensate Storage Tank Discharge Piping, Given Break Can be Isolated	IE	0.008860	0.000592	0.004290	0.029900
FLPH1R	Identify and Isolate a Line Break in Intake Piping A1A-A (or one of other three systems) in the Strainer Room	IE	0.001550	0.000054	0.000562	0.005620
FLPH2R	Identify and Isolate an ERCW Intake Line Break in the Pump Room of ERCW Intake	IE	0.001550	0.000054	0.000562	0.005620
HAFR1	Restore Auxiliary Feedwater Flow, Given Loss of Control Air.	AF	0.005030	0.000337	0.002440	0.017000
HARL1	Recover from an Automatic Sump Swapover Failure	RL	0.004030	0.000269	0.001950	0.013600
HCCSR1	Align the C-S Pump to the A CCS Heat Exchanger	CCSR	0.194000	0.022800	0.117000	0.584000
HCCSR2	Align and Initiate Alternate Component Cooling to the Charging Pump	CCPR	0.016100	0.001900	0.009780	0.048700
HCCSR3	Align ERCW Header 2A to CCS Heat Exchanger A, Given Loss of B Train ERCW.	CCSR	0.001550	0.000054	0.000562	0.005620
HCRL1	Inadvertently Reset SI Signal, Failure of Automatic Sump Swapover	RL	0.003750	0.000966	0.002960	0.008820
HDAR1	Switch to Spare Battery Charger, Given Operating Charger Fails.	DA DB DC DD	0.017800	0.002090	0.010800	0.053500
HDSR1	Cooldown and Depressurize by Cycling S/G PORV's Full Open/Full Closed.	DS	0.001250	0.000044	0.000456	0.004550
HERCW1	Recover ERCW Cooling To Operating Diesel Generator, Given Loss of ERCW to the Diesel Generator	DSLR	0.035500	0.004190	0.021500	0.107000
HERCW2	Cool Down with Auxiliary Feedwater and Steam Generator PORV's, Given Total Loss of ERCW, RSW Available To Cool Air Compressor	No Credit Taken	0.012500	0.008350	0.006040	0.042100
HERCW3	Cooldown with Auxiliary Feedwater and S/G PORV's, Given Total Loss of ERCW, RSW Not Available To Cool Air Compressors	DSLR	0.050000	0.005900	0.030300	0.151000

Table 3.3.3-14 (Page 2 of 2). Recovery Human Error Rate Distributions Used for Quantification of the Watts Bar PRA

Database Variable	Brief Description	Top Event	Mean HER/ Demand	5th	Median HER/ Demand	95th
HSLR1	Locally Transfer Steam Supply to TDAFP, Given Station Blackout and Loss of Steam Generator #1	TPR	0.017800	0.002090	0.010800	0.053500
HTPR1	Start Turbine Driven Pump, Given it Failed To Start due To Control or Signal Failures	TPR	0.008250	0.000552	0.003990	0.027900
HV1R1	Restore Ventilation to 6.9-k V Switchgear Room	V1R V2R	0.000373	0.000013	0.000136	0.001360
HVNVR1	Restore Ventilation to the 480V Board Room 1BB (2BB), Given Loss of Room Supply Fan	VNV1R VNV2R	0.002080	0.000073	0.000756	0.007550
HVT1AR	Establish Portable Ventilation to the Shutdown Board Transformer Room	VT1AR VT1BR VT2AR VT2BR	0.002080	0.000073	0.000756	0.007550

Table 3.3.3-15. Seal LOCA Flow Rates (GPM) with and without Primary Depressurization

Probability	Cumulative Probability	Flow Rate (in GPM) versus Time after Station Blackout					
		0-1.0 (hours)	1.0-1.5 (hours)	1.5-2.5 (hours)	2.5-3.5 (hours)	4.5-5.5 (hours)	5.5+ (hours)
0.2712	.2712	84	84	84	84	84	84
0.0151	.2863	84	84	84	244	244	244
0.0161	.3024	84	84	244	244	244	244
0.0181	.3205	84	244	244	244	244	244
0.0120	.3325	84	244	433	433	433	433
0.0059	.3384	84	244	433	433	480	698
0.1120	.4504	84	244	1,000	1,000	1,000	1,000
0.0136	.4640	84	480	1,000	1,000	1,000	1,000
0.5302	.9942	84	1,000	1,000	1,000	1,000	1,000
0.0016	.9958	84	1,230	1,230	1,230	1,230	1,230
0.0042	1.0000	84	1,920	1,920	1,920	1,920	1,920

Table 3.3.3-16. Electric Power Recovery Scenario Results

Case*	Number of Diesel Generators Failed	Number of Diesel Generators Available for Recovery	Offsite Power Recoverable	Auxiliary Feedwater Available	Pump Seal LOCA	Pump Seal Return Line Closed	Probability of Onsite Power Failure and Offsite Nonrecovery	Diesel Generator Unavailability	Sequence Recovery Factor**
REC1	2	1	Yes	Yes	Yes	No	1.464-05	1.527-03	3.760-02
REC2	2	1	Yes	Yes	Yes	No	1.464-05	1.527-03	3.760-02
REC3	2	1	Yes	No	Yes	No	3.517-05	1.527-03	9.032-02
REC4	2	1	Yes	Yes	Yes	No	1.464-05	1.527-03	3.760-02
REC5	1	1	Yes	Yes	Yes	No	4.971-04	3.012-02	6.472-02
REC6	1	1	Yes†	Yes	Yes	No	3.581-04	2.994-02	4.690-02

*Legend:
REC2 = a third diesel generator has failed, but it is at the other unit.
REC4 = sequences in which one diesel generator of two at each unit is failed.
REC6 = ventilation failure causes vital buses to fail at 12 hours.
**Includes compensation for the 1-hour recovery factor.
†Available 12 hours.

Note: Exponential notation is indicated in abbreviated form; e.g., 1.464-05 = 1.464×10^{-05}.

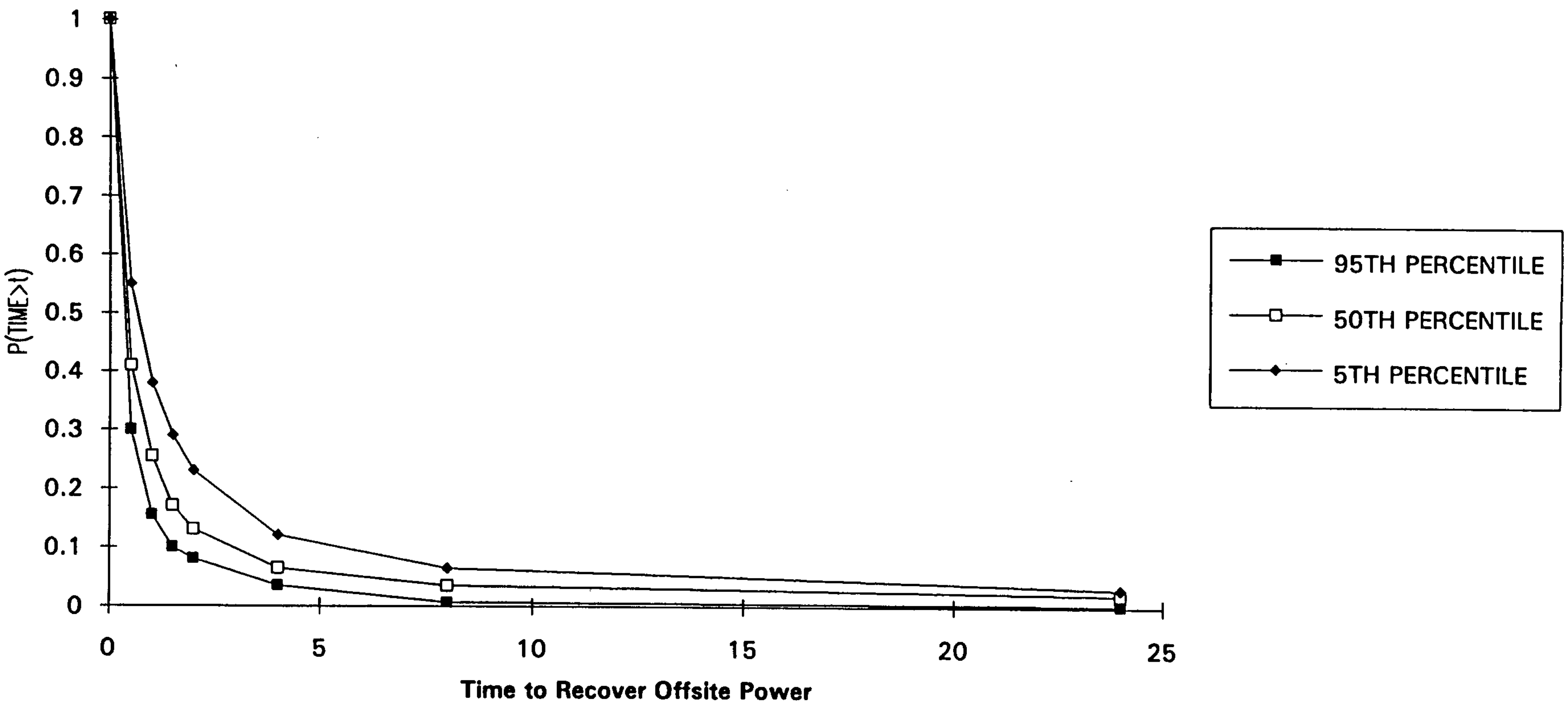

Figure 3.3.3-1. Watts Bar Electric Power Recovery

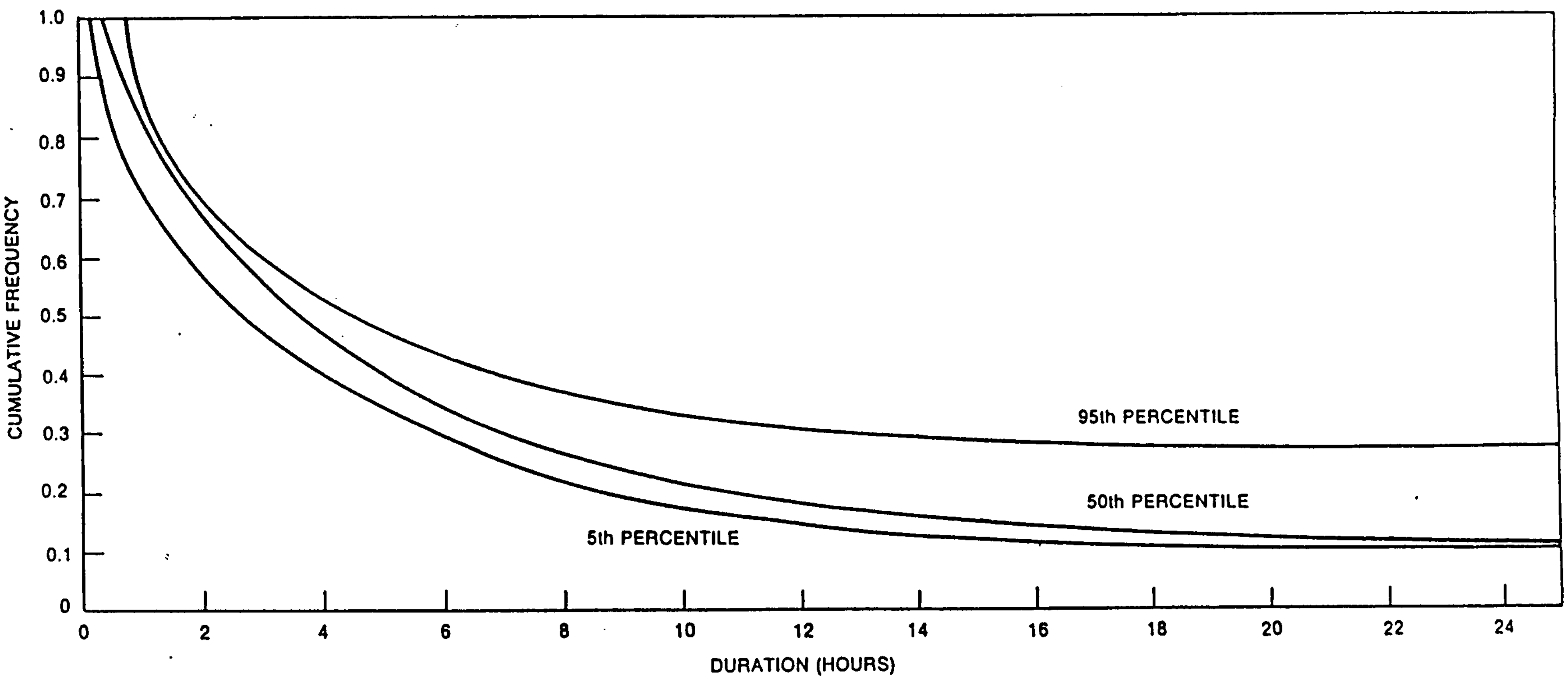

Figure 3.3.3-2. Nonrecovery of Single Diesel Generator

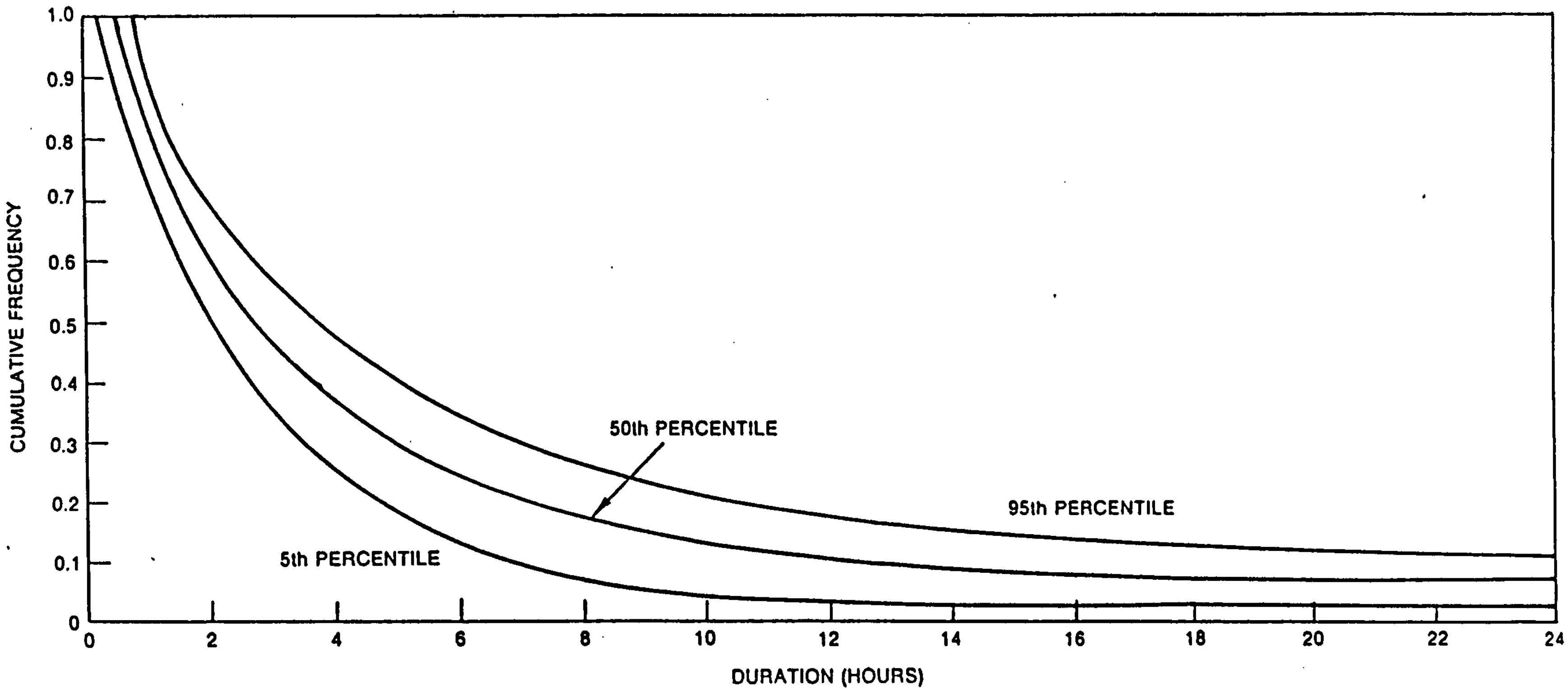

Figure 3.3.3-3. Nonrecovery of One-Out-of-Two Diesel Generators

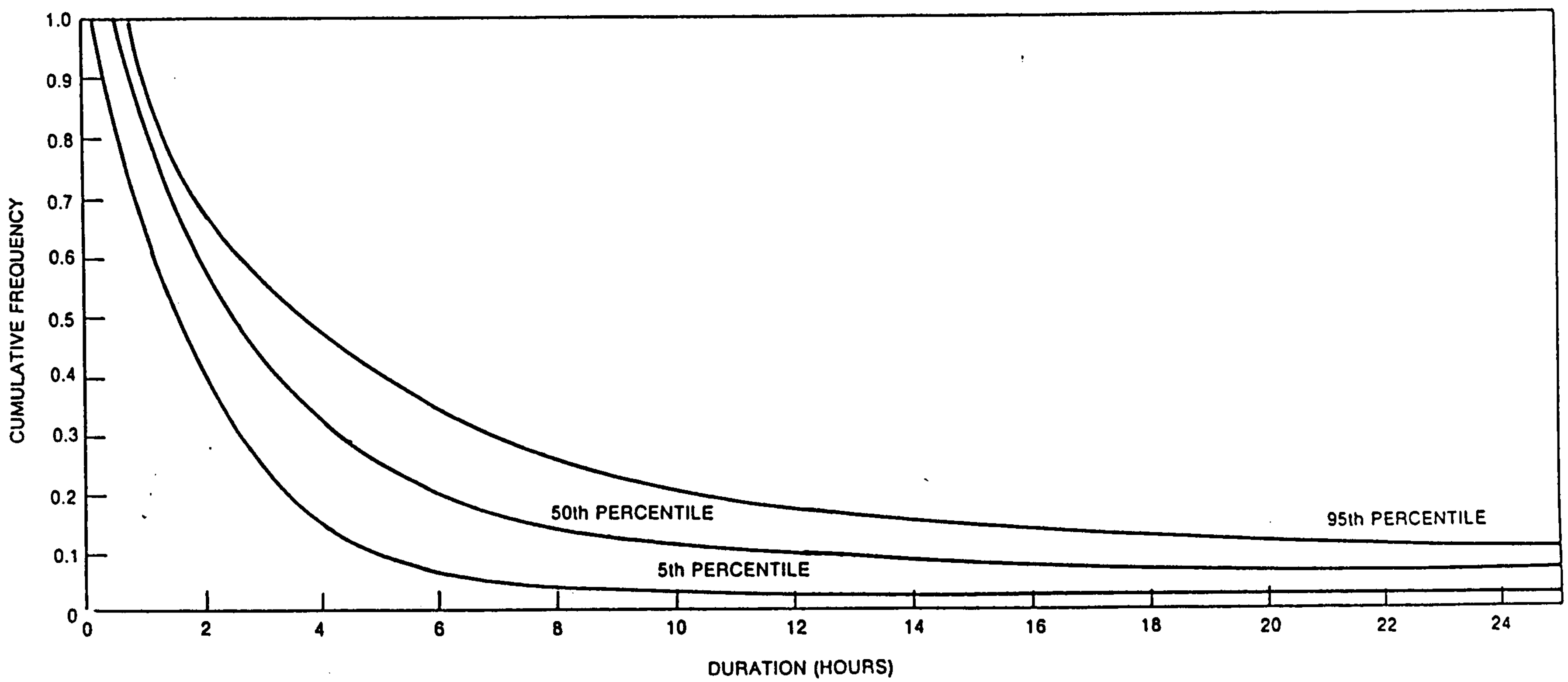

Figure 3.3.3-4. Nonrecovery of One-Out-Of-Three Diesel Generators

3.3.4 COMMON CAUSE FAILURE PARAMETERS

3.3.4.1 Introduction

In the Watts Bar PRA, dependent failures such as common cause failures at the systems level are treated either explicitly by means of identifying causes of dependent failure and incorporating them into the systems or event sequence models, or implicitly by using certain parameters to account for their contribution to the unavailability of the systems. Examples of the first category are the sharing of common components, fires, floods, and certain types of human error during test and maintenance. This section deals with the second category, addressing common cause failures that are not covered in the first category, such as design errors, construction errors, procedural deficiencies, and unforeseen environmental variations.

The parametric model used in this study to quantify the effect of the second category of dependent failures is known as the multiple Greek letter (MGL) method (Reference 3.3.4-1). The following is an overview of the method and the Bayesian technique used in developing state of knowledge distributions reflecting various sources of uncertainty in estimating its parameters. Due to the relatively low frequency of these events, there are insufficient data available to justify the use of the two-stage Bayesian procedure described in Section 3.3.1.2 for failure rates; thus, a modified technique is needed and described below.

3.3.4.2 Multiple Greek Letter Model

The MGL parameters consist of the total component failure probability, Q_t, which includes the effects of all independent and common cause contributions to that component failure, and failure fractions, which are used to quantify the conditional probabilities of all the possible ways that a common cause failure of a component can be shared with other components in the same group, given component failure has occurred. For a group of m redundant components and for each given failure mode, m different parameters are defined. For example, the first four parameters of the MGL model are

Q_t = total failure probability of each component due to all independent and common cause events.

plus

β = conditional probability that the cause of a 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 that is shared by one or more components will be shared by two or more additional components, given that two specific components have failed.

δ = conditional probability that the cause of a component failure that is shared by two or more components will be shared by three or more additional components, given that three specific components have failed.

The general equation that expresses the probability of k specific component failures due to common cause, Q_k, in terms of the MGL parameters, is consistent with the above definitions.

The MGL parameters are defined in terms of the basic parameter model parameters for a group of three similar components as:

$$Q_t = Q_1^{(3)} + 2\,Q_2^{(3)} + Q_3^{(3)} \tag{3.3.4.1}$$

$$\beta^{(3)} = \frac{2Q_2^{(3)} + Q_3^{(3)}}{Q_1^{(3)} + 2Q_2^{(3)} + Q_3^{(3)}}$$

$$\gamma^{(3)} = \frac{Q_3^{(3)}}{2Q_2^{(3)} + Q_3^{(3)}} \tag{3.3.4.2}$$

δ and higher order terms are identically zero.

For a group of four similar components, the MGL parameters are as follows:

$$Q_t = Q_1^{(4)} + 3Q_2^{(4)} + 3Q_3^{(4)} + Q_4^{(4)} \tag{3.3.4.3}$$

$$\beta^{(4)} = \frac{3Q_2^{(4)} + 3Q_3^{(4)} + Q_4^{(4)}}{Q_1^{(4)} + 3Q_2^{(4)} + 3Q_3^{(4)} + Q_4^{(4)}}$$

$$\gamma^{(4)} = \frac{3Q_3^{(4)} + Q_4^{(4)}}{3Q_2^{(4)} + 3Q_3^{(4)} + Q_4^{(4)}} \tag{3.3.4.4}$$

$$\delta^{(4)} = \frac{Q_4^{(4)}}{3Q_3^{(4)} + Q_4^{(4)}}$$

It is important to note that the integer coefficients in the above definitions are a function of m, the number of components in the common cause group. Therefore, it is generally inappropriate to use MGL parameters that were quantified for an m unit group in an ℓ unit group, when $m \neq \ell$. The same comment applies to the other similar multiparameter methods.

The following equations express the probability of multiple component failures due to common cause, Q_k, in terms of the MGL parameters for a three-component common cause group:

$$Q_1 = (1-\beta)\,Q_t$$

$$Q_2 = \frac{1}{2}\,\beta\,(1-\gamma)\,Q_t \tag{3.3.4.5}$$

$$Q_3 = \gamma\,\beta\,Q_t$$

For a four-component group, the equations are

$$Q_1 = (1-\beta)\, Q_t$$

$$Q_2 = \frac{1}{3}\beta\,(1-\gamma)\, Q_t$$

$$Q_3 = \frac{1}{3}\beta\gamma\,(1-\delta)\, Q_t$$

$$Q_4 = \beta\gamma\delta\, Q_t \qquad (3.3.4.6)$$

The generalization of this is given by

$$Q_k = \frac{1}{\binom{m-1}{k-1}}\left(\prod_{i=1}^{k} \rho_i\right)(1-\rho_{k+1})\, Q_t \;(k = 1, \ldots, m) \qquad (3.3.4.7)$$

where $\rho_1 = 1$, $\rho_2 = \beta$, $\rho_3 = \gamma$, ..., $\rho_{m+1} = 0$.

3.3.4.2.1 Point Estimators for the MGL Parameters

The following are simple point estimators for the first three of the MGL parameters:

$$\hat{\beta} = \frac{\sum_{k=2}^{m} kn_k}{\sum_{k=1}^{m} kn_k} \qquad (3.3.4.8)$$

$$\hat{\gamma} = \frac{\sum_{k=3}^{m} kn_k}{\sum_{k=2}^{m} kn_k} \qquad (3.3.4.9)$$

$$\hat{\delta} = \frac{\sum_{k=4}^{m} kn_k}{\sum_{k=3}^{m} kn_k} \qquad (3.3.4.10)$$

where n_k is defined as the number of events involving k components in failed state.

For instance, for a three-unit system (m = 3), we have

$$\hat{\beta} = \frac{2n_2 + 3n_3}{n_1 + 2n_2 + 3n_3} \qquad (3.3.4.11)$$

Similarly,

$$\hat{\gamma} = \frac{3n_3}{2n_2 + 3n_3} \qquad (3.3.4.12)$$

As can be seen from the above estimators, the MGL parameters are essentially the ratios of the number of component failures in various basic events. For instance, in Equation (3.3.4.9), the numerator ($3n_3$) is the total number of components failed in common cause basic events that fail three components (n_3).

3.3.4.2.2 Uncertainty Distribution of the MGL Parameters

The uncertainty distribution of the MGL parameters can be approximated with simple parametric distributions if the observed events are assumed to be independent component failures within different categories of common cause events. In other words, the set $\{n_k\ k = 1, \ldots, m\}$ where n_k is the number of events involving failure of k components due to common cause will be interpreted as $\{kn_k;\ k = 1, \ldots, m\}$ where kn_k is the number of components failed in common cause events involving k component failures, and kn_k events will be assumed to have occurred independently.

With the above assumption, let us define the following conditional probabilities (for a system of these components):

$Z_1 \equiv 1 - \beta$ = conditional probability of component failure being a single failure.

$Z_2 \equiv \beta(1 - \gamma)$ = conditional probability of a component being involved in a double failure.

$Z_3 \equiv \beta\gamma$ = conditional probability of a component being involved in a triple failure.

Note that

$$Z_1 + Z_2 + Z_3 = 1$$

The likelihood of observing n_1 single failures, $2n_2$ component failures due to double failures, and $3n_3$ component failures due to triple failures can be modeled by a multinomial distribution for Z_i's.

$$P(n_1, 2n_2, 3n_3 | Z_1, Z_2. Z_3) = \frac{(n_1 + 2n_2 + 3n_3)!}{(n_1)!(2n_2)!(3n_3)!} Z_1^{n_1} Z_2^{2n_2} Z_3^{3n_3} \qquad (3.3.4.13)$$

Rewriting Equation (3.3.4.13) in terms of β and γ gives

$$P(n_1, 2n_2, 3n_3|\beta, \gamma) = M\,\beta^{2n_2+3n_3}\,(1-\beta)^{n_1}\,\gamma^{3n_3}\,(1-\gamma)^{2n_2} \tag{3.3.4.14}$$

where M is the multinomial multiplier, as in Equation (3.3.4.13).

We can now write Bayes' theorem as

$$\pi(\beta, \gamma|n_1, 2n_2, 3n_3) = \frac{1}{C}\,P(n_1, 2n_2, 3n_3|\beta, \gamma)\pi_0(\beta, \gamma) \tag{3.3.4.15}$$

where π_0 and π are the prior and posterior distribution of β and γ and C is a normalizing factor defined as

$$C = \int_0^1 \int_0^1 P(n_1, 2n_2, 3n_3|\beta, \gamma)\pi_0(\beta, \gamma)d\beta\, d\gamma \tag{3.3.4.16}$$

As the prior, a multinomial distribution can be used

$$\pi_0(\beta, \lambda) = h\,\beta^{A_0-1}\,(1-\beta)^{B_0-1}\,\gamma^{C_0-1}\,(1-\gamma)^{D_0-1} \tag{3.3.4.17}$$

where h is given by

$$h = \frac{\Gamma(A_0 + B_0 + C_0 + D_0)}{\Gamma(A_0)\Gamma(B_0)\Gamma(C_0)\Gamma(D_0)} \tag{3.3.4.18}$$

A flat prior distribution is obtained by setting $A_0 = B_0 = C_0 = D_0 = 1$.

Using Equation (3.3.4.17) in Equation (3.3.4.15) results in a posterior distribution for β and γ that is also multinomial, with parameters

$$\begin{aligned} A &= A_0 + 2n_2 + 3n_3 \\ B &= B_0 + n_1 \\ C &= C_0 + 3n_3 \\ D &= D_0 + 2n_2 \end{aligned} \tag{3.3.4.19}$$

The mode of the posterior distribution occurs at

$$\beta = \frac{A-1}{A+B-2} \tag{3.3.4.20}$$

$$\gamma = \frac{C-1}{C+D-2} \tag{3.3.4.21}$$

The mean values are calculated from

$$\bar{\beta} = \frac{A}{A+B} \tag{3.3.4.22}$$

$$\bar{\gamma} = \frac{C}{C+D} \tag{3.3.4.23}$$

Note that, for the flat prior, the mode of the posterior distribution is

$$\beta = \frac{2n_2 + 3n_3}{n_1 + 2n_2 + 3n_3} \tag{3.3.4.24}$$

$$\gamma = \frac{3n_3}{2n_2 + 3n_3} \tag{3.3.4.25}$$

which corresponds to the point estimates presented earlier for a component common cause group of size $m=3$. As can be seen, the approximate method results in estimators that are similar to the commonly used estimators for the MGL parameters. The commonly used estimators therefore are not exact and should be used only if the magnitude of error introduced is judged to be insignificant compared with other sources of error and uncertainty. More accurate estimators can be found in Reference 3.3.4-2.

3.3.4.3 Data Classification and Screening

Based on the above discussion, the MGL parameters and the associated uncertainty distributions can be assessed if the values of n_k's are known. Ideally, the numerical value of the parameters of the common cause failure models should be estimated in a manner that makes the maximum possible use of event data; i.e., reports of operating experience. This requires review, evaluation, and classification of the available information to obtain specialized failure data. Because common cause failures can dominate the results of reliability and safety analysis, it is extremely important that this analysis of data be performed within a context that represents the engineering and operational aspects of the plant and system being modeled.

Due to the rarity of common cause events and the limited experience of individual plants, the amount of plant-specific data for common cause analysis is very limited. Therefore, in almost all cases, we need to use data from the industry experience and a variety of sources to make statistical inferences about the frequencies of the common cause events. However, due to the fact that there is a significant variability in plants, especially with regard to the coupling mechanisms and defenses against common cause events, the industry experience is not, in most cases, directly applicable to the specific plant being analyzed although much of it may be indirectly applicable. Also, and perhaps equally important, the analysis boundary conditions that dictate what category of components and causes should be analyzed, requires careful review and screening of events to ensure consistency of the database with the assumptions of the system model, its boundary conditions, and other qualitative aspects delineated in the analysis.

The significance of this step has also been emphasized by Reference 3.3.4-3 since an important conclusion of the Common Cause Failure Reliability Benchmark Exercise (CCF-RBE) was that the most important source of uncertainty and variation in the numerical results is data interpretation.

Given the raw data (event reports), a major step is the review and classification of the events to identify where each event fits in a set of predefined categories that describes the type of the event, its cause(s), and its impact; e.g., number of components failed.

The classification of event reports is a rather subjective exercise, particularly in light of the quality of many of the event reports. In an attempt to reduce subjectivity in the screening of event data to identify common cause failures, the CCF-RBE (Reference 3.3.4-3) identified the following rules, which have been somewhat modified:

1. Component-caused functional unavailabilities are screened out since this kind of dependency is normally modeled explicitly.

2. If a specific defense exists that clearly precludes a class of events, all specific events belonging to that class can be screened out.

3. If the cause of the reported event is a train interconnection that, in the plant under consideration, does not exist, the event is considered as an independent failure of one train.

4. Events related to inapplicable plant conditions (e.g., preoperational testing, etc.) are screened out unless they reveal general causal mechanisms capable of occurring during power operation.

5. If the event occurred during shutdown and would be restored before resuming power operation because of preservice testing or if it cannot occur during power operation, the event is screened out.

6. If a second failure in an event happened after the restoration of the first, both failures are considered as independent failures.

7. Events regarding incipient failure modes (e.g., packing leak, etc.) that clearly do not violate component success criteria are screened out.

8. Only the events regarding the failure modes of interest are taken into consideration; events regarding failure modes that are irrelevant to the system logic model are screened out.

3.3.4.4 Event Impact Assessment

A useful tool in developing statistical data from event descriptions is to summarize the outcome of the event classification process up to this point in a form similar to the example given in Table 3.3.4-1.

To complete the description of the event impact at the original plant, the analyst needs to identify the following:

1. **Component Group Size.** The number (m) of (typically similar) components that are believed to have been exposed to the root cause and coupling mechanism of the event.

2. **Number of Components Affected.** The number of components within the component group that were affected (e.g., failed) in the event.

3. **Shock Type.** Whether the cause(s) and coupling mechanism(s) involved were of the type that typically results in the failure of all components within the component group (lethal shock) or not (nonlethal shock).

4. **Failure Mode.** The particular component function affected; e.g., failure to open on demand.

Table 3.3.4-2 summarizes the information about the event for the example event described in Table 3.3.4-1 and introduces the representation called the impact vector (Reference 3.3.4-2).

The binary impact vector of an event that has occurred in a common cause component group of size m has m + 1 elements.*

Each element represents the number of components that can fail in an event. If, in an event, k components are failed, then a 1 is placed in the F_k Position of the binary impact vector, with 0 in other positions. In the example of Table 3.3.4-1, the component group size is 2; therefore, the binary impact vector has three elements: $\{F_0, F_1, F_2\}$. Since two components were failed, we have $F_0 = F_1 = 0$ and $F_2 = 1$. A condensed representation is

$$I = \{0, 0, 1\} \qquad (3.3.4.26)$$

Most of the time, however, the event descriptions are not clear, the exact states of components are not always known, and root causes are seldom identified. Therefore, the interpretation of the event (i.e., the translation of the event descriptions into a form similar to the example in Tables 3.3.4-1 and 3.3.4-2) may require establishing several hypotheses, each representing a different interpretation of the event.

As an example, consider the event classified in Table 3.3.4-3. Since it is not clear whether the third diesel was also actually failed, the binary impact vector is assessed under two different hypotheses (Table 3.3.4-4). Under the first hypothesis, only two diesels are considered failed, while, according to the second hypothesis, all three diesels were failed. The analyst, at this point, needs to assess his or her degree of confidence in each of the two hypotheses. In the example of Table 3.3.4-4, a weight of 0.9 is given to

*Common cause component group is defined as a group of (usually similar) components that are considered to have a high potential of failing due to the same cause (Reference 3.3.4-2).

the first hypothesis, reflecting a very high degree of confidence that only two diesels were actually failed. The weight for the second hypothesis is obviously 0.1 since the weight should add up to 1. This property of the weighting factors assumes that all reasonable hypotheses are accounted for. Note that the data analyst must be in a position to defend and document this assessment.

The expectation values for the impact vectors, taken over the two hypotheses, are

$$\bar{I} = (P_0, P_1, P_2) = (0.9)I_1 + (0.1)I_2 = \{0, 0.9, 0.1\} \quad (3.3.4.27)$$

which is also shown in Table 3.3.4-4. Note that F_i refers to a single binary impact vector, and P_i refers to an average impact vector. The appendix contains the event description and the impacts for each of the common cause events in the PLG generic database.

3.3.4.5 Reinterpretation of Common Cause Events — Creation of a "Plant-Specific Generic" Database

As explained in Section 3.3.4.3, the common cause events in the PLG database (Reference 3.3.4-4) have been analyzed for the original plant. The first step in creating a "plant-specific generic" database is to determine what that event implies for Watts Bar; i.e., what would have happened at Watts Bar if a similar event had occurred. As was mentioned earlier, the same event may not be directly applicable to the plant and system of interest due to several reasons, such as differences in design, operation, common cause defenses, etc. It is therefore essential to reinterpret the event in light of the specific characteristics of the system under consideration.

In general, the differences between the system in which the data originated and the system being analyzed arise in two ways: first, even for systems of the same size, there are physical differences in system design, component type, operating conditions, environment, etc.; second, there can be a difference in system size (degree of redundancy).

In the following discussion, a framework is described with which these two types of differences can be taken into account explicitly in reinterpretation of the event and the assessment of the impact vector for the system of interest.

3.3.4.6 Systems of the Same Size

First, we consider the differences, given the assumption that the system size is the same. The question to be answered is the following: Given all the qualitative differences between the two systems, could the same root cause(s) and coupling mechanism(s) occur in the system being analyzed?

In reality, this step involves a considerable amount of judgment. There are a number of sources of uncertainty. These include the lack of detailed information about the event, its circumstances, the nature of its causes, the nature of defenses in the original system, and the effectiveness of defenses in the system being analyzed. Yet, because of the scarcity of data, there is strong motivation to avoid tossing out the data and extracting from them that evidence that is applicable. Due to the uncertainties involved and the important implications of screening events from the database by declaring them inapplicable, the analyst must have a concrete reason for his judgment. In the cases in which the analyst is uncertain about whether an event is applicable or not, the impact vector of the original system may be modified by a weight reflecting the degree of applicability of the event, as viewed by the analyst. This is similar to the multiple hypothesis situation discussed earlier. Thus, the alternative hypotheses are (1) applicable with probability p, and (2) not applicable with probability (1 - p).

3.3.4.7 Adjustments for Size Difference

The next step is to consider the system size differences. The objective is to estimate or infer what the database of applicable events would look like if it all was generated by systems of the same size (i.e., the number of components in each common cause group) as the system being analyzed. This is done by simulating, in a thought experiment, the occurrence of causes of failures (both independent and dependent) in the system of interest and observing how the impact of these causes changes due to difference in system size. Reference 3.3.4-2 provides a detailed discussion of the background and justification of the need for adjustment in an impact assessment based on system size differences. Reference 3.3.4-2 also develops a set of rules and equations for changing the event impact vectors of the original system to a corresponding set for the system being analyzed.

The rules are presented for the following cases:

1. **Mapping Down.** The case in which the component group size in the original system is larger than in the system being analyzed.

2. **Mapping Up.** The case in which the component group size in the original system is smaller than in the system being analyzed.

3.3.4.8 Mapping Down Impact Vectors

Formulas for mapping down data from systems having four, three, or two components to any identical system having fewer components are presented in Table 3.3.4-5. In this table, $P_k^{(m)}$ represents the kth element of the average impact vector in a system (or component group) of size m. The formulas show how to obtain the elements of the impact vector for smaller size systems when the elements of the impact vector of a larger system are known.

3.3.4.9 Mapping Up Impact Vectors

It can be seen from the results presented above that downward mapping is deterministic; i.e., given an impact vector for an identical system having more components than the system being analyzed, the impact vector for the same size system can be calculated without introducing any new uncertainties. Mapping up, however, as shown in Reference 3.3.4-2, is not deterministic.

To reduce the uncertainty inherent in upward mapping of impact vectors, use is made of a powerful concept that is the basis of the binomial failure rate (BFR) common cause model (Reference 3.3.4-5). This concept is that all events can be classified into one of three categories:

1. **Independent Events.** Causal events that act on components singly and independently.

2. **Lethal Shocks.** Causal events that always fail all the components in the system.

3. **Nonlethal Shocks.** Causal events that act on the system as a whole with some chance that any number of components within the system can fail. Alternatively, nonlethal shocks can occur when a causal event acts on a subset of the components in the system.

When enough is known about the cause (i.e., root cause and coupling mechanism) of a given event, it can usually be classified without difficulty into one of the above categories. If an event is identified as being either an independent event or a lethal shock, the impact vectors can be mapped upward deterministically, as shown below. It is only in the case of nonlethal shocks that an added element of uncertainty is introduced on mapping upward. How each event is handled is separately summarized in the following sections.

3.3.4.9.1 Mapping Up Independent Events

In this case, since the number of independent events in the database is simply proportional to the number of components in the system, it can be shown that $P_I^{(\ell)}$ and $P_I^{(k)}$, the number of independent events in systems with sizes ℓ and k, respectively, are related by the following equation:

$$P_I^{(\ell)} = \frac{\ell}{k} P_I^{(k)} \tag{3.3.4.28}$$

3.3.4.9.2 Mapping Up Lethal Shocks

By definition, a lethal shock wipes out all of the redundant components present within a common cause group. From it follows the simple relationship

$$P_\ell^{(\ell)} = P_j^{(j)} \tag{3.3.4.29}$$

Thus, for lethal shocks, the impact vector is mapped directly.

3.3.4.9.3 Mapping Up Nonlethal Shocks

Nonlethal shock failures are viewed as the result of a nonlethal shock that acts on the system at a rate that is independent of the system size. For each shock, there is a constant probability, ρ, that each component fails. The quantity ρp is the conditional probability of each component failure, given a shock.

Table 3.3.4-6 includes formulas to cover all of the upward mapping possibilities with system sizes up to four. In the limiting cases of $\rho = 0$ and $\rho = 1$, the formulas in Table 3.3.4-6 became identical to Equation (3.3.4.28) (mapping up independent events) and Equation (3.3.4.29) (mapping up lethal shocks), respectively.

While it is the analyst's responsibility to assess, document, and defend his assessment of the parameter ρ, some simple guidelines should help in its quantification.

- If an event is classified as a nonlethal shock and it fails only one component of a group of three or more components, it is reasonable to expect that ρ is small ($\rho < .5$).
- If a nonlethal shock fails a number of components intermediate to the number present, it is unreasonable to expect that ρ is either very small ($\rho \rightarrow 0$) or very large ($\rho \rightarrow 1$).
- If a nonlethal shock fails all the components present in a system, it is reasonable to expect that p is large ($\rho > .5$).

3.3.4.10 Development of Event Statistics from Impact Vectors

Once the impact vectors of all the events in the database are assessed for the system being analyzed, the number of events in each impact category can be calculated by adding the impact vectors; that is,

$$n_k = \sum_{i=1}^{m} P_k^{(i)} \tag{3.3.4.30}$$

where

n_k = total number of basic events* involving failure of k similar components.

$P_k^{(i)}$ = the kth element of the ith impact vector.

The n_k's are used to develop estimates of model parameters.

3.3.4.11 Estimation of the MGL Parameters

The procedure described in the preceding sections was used to develop a Watts Bar-specific generic database for estimating the MGL parameters. The source of data for generic event descriptions and classification was the PLG generic common cause database (Reference 3.3.4-6). The generic events database covers several hundreds of PWR and boiling water reactor (BWR) operating experience for the components of interest in Watts Bar PRA. The generic screening was performed by a team of PLG PRA experts having a

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*A common cause basic event is defined as an event involving common cause failure of a specific subset of components within a common cause component group (Reference 3.3.4-2).

broad range of expertise and background, including former nuclear power plant operators, systems analysts, data analysts, and common cause failure experts. The screening of events for applicability to Watts Bar and the assessment of Watts Bar-specific impact vectors were performed by PLG analysts familiar with the specific systems. The details of this screening are documented in Table 3.3.4-7.

The impact vector for each applicable event was mapped, if needed, to adjust for system size differences between Watts Bar and the plants in the generic database. The number of components in Watts Bar was based on determination of the combination of components assumed to be susceptible to common cause failures. Obviously, the first criteria to apply were to identify components that were modeled in the systems and to determine if those components could be further divided into subgroups of similar components with high susceptibility to common cause failures. As the result of this process, a number of common cause component groups were identified and used as the basis for system size adjustment or mapping of impact vectors.

The result of impact vector assessment and mapping is summarized in Table 3.3.4-8, where for each category of components or set of n_k (k = 1, 2, 3, 4), values are listed. Also provided in the table is the number of independent events adjusted for system size difference between Watts Bar and the plants in the generic population according to Equation (3.3.4.28). This was done by developing an average number of components for the generic plant, P_{GN}, and using it to scale up or down the total number of independent events in the generic population:

$$N_I = N_{WBN} = \frac{P_{WBN}}{P_{GN}} \times N_{GN} \tag{3.3.4.31}$$

where

P_{WBN} = population of the component in Watts Bar.

P_{GN} = average population of the component in the nuclear power plants from which the data are collected.

N_{GN} = number of independent events for the component failure modes in the generic database.

The average population of a component in the generic population was obtained by first tabulating the numbering of the component in each power plant from which the failure data are collected. These are power plants with commercial operating experience. The average population of the component is simply the total number of the components in the plants divided by the number of the plants. For components that are common to both PWRs and BWRs, the average population would be based on the total number of the components in both types of plants. For example, diesel generators and component cooling water pumps are equipment that are common to both PWRs and BWRs, whereas auxiliary feedwater pumps are unique to PWRs.

Table 3.3.4-9 shows the average number of components per plant for each component type of interest and the type(s) of nuclear power plants considered in the calculation. The component population and the corresponding common cause component group for each component type in Watts Bar are provided in Table 3.3.4-8. Given the number of independent events for each component failure mode in the generic database, the average number of components per plant, and the number of components in Watts Bar, the number of independent events for a component failure mode appropriate for the calculation of Watts Bar-specific component common cause failure parameters can be obtained.

The element of plant-specific update can be introduced into the process by incorporating the actual common cause experience of Watts Bar into the data when it becomes available.

Table 3.3.4-8 summarizes the common cause event statistics developed for Watts Bar. It also provides the parameters of the prior distributions used in the Bayesian updating of the data. These parameters, together with the event statistics, provide the parameters of the corresponding posterior distributions according to a generalization of Equation (3.3.4.15) for a four-component system:

- For β-factors,

$$A = A_0 + 2n_2 + 3n_3 + 4n_4$$

$$B = B_0 + n_1$$

- For γ-factors,

$$C = C_0 + 3n_3 + 4n_4$$

$$D = D_0 + 2n_2$$

- For δ-factors,

$$E = E_0 + 4n_4$$

$$F = F_0 + 3n_3$$

The values of n_i include not only the mapped impacts but also the plant-specific experience. Values of A_0, B_0, C_0, D_0, E_0, and F_0 were derived based on characteristics of prior distributions assessed for the category for each component by PLG experts. These distributions reflect the experts' estimate of the likely range of variation of MGL parameters and are provided to supplement the incompleteness of the generic event database with respect to failure modes and causes potentially applicable to Watts Bar but not yet observed in the generic population. The experts panel for the assessment of prior distributions was composed of former senior reactor operators and leading systems analysts, data analysts, and common cause failure experts having a total of over 40 person-years of PRA experience.

The posterior distributions for all MGL parameters are listed in Table 3.3.4-10.

3.3.4.12 References

3.3.4-1. Fleming, K. N., and A. M. Kalinowski, "An Extension of the Beta Factor Method to Systems with High Levels of Redundancy," Pickard, Lowe and Garrick, Inc., PLG-0289, June 1983.

3.3.4-2. Mosleh, A., et al., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," prepared for Electric Power Research Institute and U.S. Nuclear Regulatory Commission, NUREG/CR-4780, Vols. I and II, 1988.

3.3.4-3. Poucet, A., A. Amendola, and P. C. Cacciabue, "Summary of the Common Cause Failure Reliability Benchmark Exercise," Joint Research Centre Report, PER 1133/86, Ispra, Italy, April 1986.

3.3.4-4. Pickard, Lowe and Garrick, Inc., "Database for Probabilistic Risk Assessment for Light Water Nuclear Power Plants," PLG-0500, proprietary, July 1989.

3.3.4-5. Atwood, C. L., "Common Cause Fault Rates for Pumps," EG&G Idaho, Inc., prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-2098, EGG-EA-5289.

3.3.4-6. PLG, Inc., "A Database of Common Cause Events for Risk and Reliability Evaluations," prepared for the Electric Power Research Institute, PLG-0866, March 1992.

Table 3.3.4-1. Example of Event Classification and Impact Assessment — Event Classification

Plant (date)	Status	Event Description
Pilgrim (September 1976)	95% Power	Two RHR torus cooling valves failed to operate. It was found that the failure was due to excessive pressure differential across the valves, which exceeded the capacity of the valve motors.

Table 3.3.4-2. Example of Event Classification and Impact Assessment — Event Impact Assessment

Component Group Size	Impact Vector			Shock Type	Fault Mode
	F_0	F_1	F_2		
2	0	0	01	Nonlethal (L)	Fail to Open on Demand

Table 3.3.4-3. Example of the Assessment of Impact Vectors Involving Multiple Interpretation of Event — Event Classification		
Cause-Effect Diagram		
Plant (date)	Status	Event Description
Maine Yankee (August 1977)	Power	Two diesel generators failed to run due to plugged radiator. The third unit radiator was also plugged.

Table 3.3.4-4. Example of the Assessment of Impact Vectors Involving Multiple Interpretation of Events — Multiple Hypothesis Impact Vector Assessment								
Component Group Size	Hypothesis	Probability	F_0	F_1	F_2	F_3	Shock Type	Fault Mode
3	I_1	0.9	0	0	1	0	Nonlethal (N)	Failure during Operation
	I_2	0.1	0	0	0	1		
	Average Impact Vector (I)		P_0	P_1	P_2	P_3		
			0	0	0.9	0.1		

Table 3.3.4-5. Formulas for Mapping Down Impact Vectors

SIZE OF SYSTEM MAPPING FROM	SIZE OF SYSTEM MAPPING TO (NUMBER OF IDENTICAL TRAINS): 3	2	1
4	$P_0^{(3)} = \frac{1}{4} P_1^{(4)} + P_0^{(4)*}$ $P_1^{(3)} = \frac{3}{4} P_1^{(4)} + \frac{1}{2} P_2^{(4)}$ $P_2^{(3)} = \frac{1}{2} P_2^{(4)} + \frac{3}{4} P_3^{(4)}$ $P_3^{(3)} = \frac{1}{4} P_3^{(4)} + P_4^{(4)}$	$P_0^{(2)} = \frac{1}{2} P_1^{(4)} + \frac{1}{6} P_2^{(4)}$ $P_1^{(2)} = \frac{1}{2} P_1^{(4)} + \frac{2}{3} P_2^{(4)} + \frac{1}{2} P_3^{(4)}$ $P_2^{(2)} = \frac{1}{6} P_2^{(4)} + \frac{1}{2} P_3^{(4)} + P_4^{(4)}$	$P_0^{(1)} = \frac{3}{4} P_1^{(4)} + \frac{1}{2} P_2^{(4)} + \frac{1}{4} P_3^{(4)}$ $P_1^{(1)} = \frac{1}{4} P_1^{(4)} + \frac{1}{2} P_2^{(4)} + \frac{3}{4} P_3^{(4)} + P_4^{(4)}$
3		$P_0^{(2)} = P_0^{(3)} + \frac{1}{3} P_1^{(3)}$ $P_1^{(2)} = \frac{2}{3} P_1^{(3)} + \frac{2}{3} P_2^{(3)}$ $P_2^{(2)} = \frac{1}{3} P_2^{(3)} + P_3^{(3)}$	$P_0^{(1)} = P_0^{(3)} + \frac{2}{3} P_1^{(3)} + \frac{1}{3} P_2^{(3)}$ $P_1^{(1)} = \frac{1}{3} P_1^{(3)} + \frac{2}{3} P_2^{(3)} + P_3^{(3)}$
2			$P_0^{(1)} = P_0^{(2)} + \frac{1}{2} P_1^{(2)}$ $P_1^{(1)} = \frac{1}{2} P_1^{(2)} + P_2^{(2)}$

*THE TERM $P_0^{(4)}$ IS INCLUDED FOR COMPLETENESS, BUT IN PRACTICE, ANY EVIDENCE THAT MIGHT EXIST ABOUT CAUSES THAT IMPACT NO COMPONENTS IN A FOUR-TRAIN SYSTEM WOULD BE "UNOBSERVABLE."

Table 3.3.4-6. Formulas for Upward Mapping of Events Classified as Nonlethal Shocks

SIZE OF SYSTEM MAPPING FROM	SIZE OF SYSTEM MAPPING TO: 2	3	4
1	$P_1^{(2)} = 2(1-\rho)P_1^{(1)}$ $P_2^{(2)} = \rho P_1^{(1)}$	$P_1^{(3)} = 3(1-\rho)^2 P_1^{(1)}$ $P_2^{(3)} = 3\rho(1-\rho)P_1^{(1)}$ $P_3^{(3)} = \rho^2 P_1^{(1)}$	$P_1^{(4)} = 4(1-\rho)^3 P_1^{(1)}$ $P_2^{(4)} = 6\rho(1-\rho)^2 P_1^{(1)}$ $P_3^{(4)} = 4\rho^2(1-\rho)P_1^{(1)}$ $P_4^{(4)} = \rho^3 P_1^{(1)}$
2		$P_1^{(3)} = (3/2)(1-\rho)P_1^{(2)}$ $P_2^{(3)} = \rho P_1^{(2)} + (1-\rho)P_2^{(2)}$ $P_3^{(3)} = \rho P_2^{(2)}$	$P_1^{(4)} = 2(1-\rho)^2 P_1^{(2)}$ $P_2^{(4)} = (5/2)\rho(1-\rho)P_1^{(2)} + (1-\rho)^2 P_2(2)$ $P_3^{(4)} = \rho^2 P_1^{(2)} + 2\rho(1-\rho)P_2^{2}$ $P_4^{(4)} = \rho^2 P_2^{(2)}$
3			$P_1^{(4)} = (4/3)(1-\rho)P_1^{(3)}$ $P_2^{(4)} = \rho P_1^{(3)} + (1-\rho)P_2^{(3)}$ $P_3^{(4)} = \rho P_2^{(3)} + (1-\rho)P_3^{(3)}$ $P_4^{(4)} = \rho P_3^{(3)}$

Table 3.3.4-7 (Page 1 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
DIESEL/GENERATORS	POPULATION 4																	
1	BWR XI.A.208	4	1.000	0.000	0.000	0.000	RUN	N					1.000	0.000	0.000	0.000	APPICABLE-OIL STORED IN COVERED SHED WITH OPEN SIDES	
2	BWR XI.A.289	8	0.930	0.070	0.000	0.000	RUN	N					0.930	0.070	0.000	0.000	APPLICABLE	
3	PWR X1.A.57	3	1.970	0.017	0.000		RUN	N					0.657	1.482	0.013	0.000	APPLICABLE	0.75
4	PWR XI.C.9	3	0.000	0.000	1.000		RUN	L					0.000	0.000	0.000	0.000	DIESELS ARE NOT ALIGNED IN PARALLEL	
5	PWR XI.A.217	2	0.010	0.070			RUN	N					0.000	0.000	0.000	0.000	DIESELS ARE WELL PROTECTED FROM THE WEATHER	
6	PWR X1.A.339	3	0.000	0.900	0.100		RUN	N					0.000	0.225	0.700	0.075	APPLICABLE-TUBES IN LUBE OIL COOLER COULD PLUG	0.75
7	PWR XI.A.337	2	0.000	1.000			RUN	N					0.000	0.063	0.375	0.563	APPLICABLE	0.75
8	BWR XVI.C.32	5	0.000	1.000	0.000	0.000	START	N	0.000	0.000	0.000	0.000					COOLING WATER REALIGNED TO ALT SUCTION PATH FOR VLV MNT	
9	NUREG/CR-1362	3	0.000	0.890	0.110		START	N	0.000	0.223	0.695	0.083					APPLICABLE	0.75
10	NUREG/CR-1362	2	0.000	1.000			START	N	0.000	0.000	0.000	0.000					NOT APPLICABLE-VALVES NOT CLOSED FOR FUEL FILLING	
11	PWR XI.A.230	2	0.000	1.000			START	N	0.000	0.000	0.000	0.000					TRIP BYPASSED ON AUTO START	
12	BWR XI.A.29	3	0.983	0.000	0.017		START	N	0.328	0.737	0.004	0.013					APPLICABLE	0.75
13	BWR X1.A.159	4	0.000	0.900	0.100	0.000	START	N	0.000	0.000	0.000	0.000					TRIP BYPASSED ON AUTO START	
14	PWR XI.A.25	3	0.000	1.000	0.000		RESTART	N	0.000	0.000	0.000	0.000					RESTART OF DIESELS NOT MODELED	
							TOTAL IMPACT AT WBN		0.328	0.960	0.699	0.095	2.587	1.839	1.088	0.638		
									W_DGGS				W_DGGR					
STANDBY PUMPS (AFW)	POPULATION 2																	
1	PWR VII.A.233	4	0.770	0.200	0.030	0.000	START	N	0.000	0.000							PUMPS NOT IN PARALLEL	
2	PWR VIII.A.316	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS ARE NOT P.D. PUMPS	
3	PWR XVI.C.1624	3	0.000	1.000	0.000		START	N	0.000	1.000							APPLICABLE	
4	PWR XVI.C.778	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS DO NOT HAVE THIS TYPE OF LOGIC	
5	PWR XVI.C.801	3	0.000	0.000	1.000		START	L	0.000	0.000							NA-AUX FW IS NOT LIKELY TO BE TAGGED OUT	
6	PWR XVI.C..58	3	0.000	1.000	0.000		START	N	0.667	0.333							APPLICABLE	
7	PWR V11.A.176	4	0.830	0.000	0.000	0.000	RUN	N					0.000	0.000			PUMPS ARE NOT VERTICAL PUMPS	
8	BWR VII.D.15	4	0.997	0.003	0.000	0.000	RUN	N					0.000	1.000			APPLICABLE	
9	BWR VII.D.124	4	0.235	0.765	0.000	0.000	RUN	N					1.000	0.000			PUMPS ARE SEPARATED IN A LARGE AREA	
10	PWR VII.A.289	2	0.000	1.000			START	L	0.000	0.000							NO RECIRC LINES IN EXTREMELY COLD AREAS	
11	BWR XVI.C.411	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
12	BWR XVI.C.488	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
13	BWR XI.B.207	4	0.000	1.000	0.000	0.000	START	N	0.667	0.167							APPLICABLE - FUSE CLIPS USED	
14	PWR VII.B.51	2	0.000	0.970			START	L	0.000	0.000							PUMPS ARE NOT PIGGY-BACK WITH ANY OTHER SYSTEM	
15	PWR XVI.C.188	2	0.000	1.000			START	L	0.000	0.000							PUMPS NOT ROCKED OUT UPON ENTERING COLD SHUTDOWN	
16	PWR XVI.C.559	2	0.000	1.000			START	L	0.000	0.000							NA TO AFW-PUMPS NOT RACKED OUT UPON ENTERING COLD SHUTDOWN	
17	PWR XVI.C.623	2	0.000	1.000			START	L	0.000	0.000							NA TO AFW-PUMPS NOT RACKED OUT DURING COLD SHUTDOWN	
18	PWR XVI.C.669	2	0.000	1.000			START	L	0.000	0.000							OPERATORS NOT LIKELY TO TAG OUT PUMPS DURING POWER OPERATION	
19	PWR XVI.C.801	2	0.000	1.000			START	L	0.000	0.000							NA TO AFW-PUMPS NOT ROCKED OUT DURING COLD SHUTDOWN	
20	PWR VII.B.91	2	0.000	1.000			START	N	0.000	1.000							APPLICABLE	
21	PWR VI.E.80	2+1	0.000	0.000	0.440		RUN	N					0.000	0.000			STRAINER FAILURES MODELED SEPARATELY	
22	PWR VI.E.36	2+1	0.000	1.000			START	N	0.000	0.000							AFW PUMPS TAKE SUCTION FROM BOTTOM OF COND STORAGE TANK	
23	PWR VI.E.49	2+1	0.000	1.000			START	N	0.000	0.000							NA-PUMP TESTED AFTER PACKING IS ADJUSTED	

Table 3.3.4-7 (Page 2 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	ORIGINAL IMPACTS P2	ORIGINAL IMPACTS P3	ORIGINAL IMPACTS P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
24	PWR VI.E.46	2+2	0.000	0.500	0.000	0.000	START	N	0.000	0.000							STRAINERS MODELED SEPARATELY	
25	PWR VI.E.374	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
26	PWR IX.D.205	4+2	0.000	0.000	0.000	1.000	START	N	0.000	0.000							NO TRIP ON LOW SUCTION PRESSURE, SWAP TO ERCW	
27	PWR IX.D.205	4+2	0.000	1.000			START	N	0.000	0.000							NO TRIP ON LOW SUCTION PRESSURE, SWAP TO ERCW	
28	PWR XVI.C.984	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
29	PWR XVI.C.300	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE TO MOTOR DRIVEN PUMPS. CC BETWEEN TD AND MD MODELED AS HUMAN ERROR IN SYSTEM ANALYSIS	
							TOTAL IMPACT AT WBN		1.333	5.50			1.000	1.000				
									WBMAPS				WBMAPR					

STANDBY PUMPS (SI) POPULATION 2

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	ORIGINAL IMPACTS P2	ORIGINAL IMPACTS P3	ORIGINAL IMPACTS P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
1	PWR VII.A.233	4	0.770	0.200	0.030	0.000	START	N	0.533	0.048							APPLICABLE	
2	PWR VIII.A.316	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS ARE NOT P.D. PUMPS	
3	PWR XVI.C.1624	3	0.000	1.000	0.000		START	N	0.000	1.000							APPLICABLE	
4	PWR XVI.C.778	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS DO NOT HAVE THIS TYPE OF LOGIC	
5	PWR XVI.C.801	3	0.000	0.000	1.000		START	L	0.000	1.000							APPLICABLE	
6	PWR XVI.C..58	3	0.000	1.000	0.000		START	N	0.667	0.333							APPLICABLE	
7	PWR V11.A.176	4	0.830	0.000	0.000	0.000	RUN	N					0.000	0.000			PUMPS ARE NOT VERTICAL PUMPS	
8	BWR VII.D.15	4	0.997	0.003	0.000	0.000	RUN	N					0.000	1.000			APPLICABLE	
9	BWR VII.D.124	4	0.235	0.765	0.000	0.000	RUN	N					1.000	0.000			PUMPS ARE SEPARATED IN LARGE AREA	
10	PWR VII.A.289	2	0.000	1.000			START	L	0.000	0.000							NO RECIRC LINES IN EXTREMELY COLD AREAS	
11	BWR XVI.C.411	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
12	BWR XVI.C.488	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
13	BWR XI.B.207	4	0.000	1.000	0.000	0.000	START	N	0.667	0.167							APPLICABLE - FUSE CLIPS	
14	PWR VII.B.51	2	0.000	0.970			START	L	0.000	0.000							PUMPS ARE NOT PIGGY-BACK WITH ANY OTHER SYSTEM	
15	PWR XVI.C.188	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
16	PWR XVI.C.559	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
17	PWR XVI.C.623	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
18	PWR XVI.C.669	2	0.000	1.000			START	L	0.000	0.000							OPERATORS NOT LIKELY TO TAG OUT PUMPS DURING POWER OPERATION	
19	PWR XVI.C.801	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
20	PWR VII.B.91	2	0.000	1.000			START	N	0.000	1.000							APPLICABLE	
21	PWR VI.E.80	2+1	0.000	0.000	0.440		RUN	N					0.000	0.000			STRAINERS MODELED SEPARATELY	
22	PWR VI.E.36	2+1	0.000	1.000			START	N	0.000	0.000							SI PUMPS TAKE SUCTION FROM BOTTOM OF RWST	
23	PWR VI.E.49	2+1	0.000	1.000			START	N	0.000	0.000							NA-PUMPS HAVE BEARINGS NOT PACKING	
24	PWR VI.E.46	2+2	0.000	0.500	0.000	0.000	START	N	0.000	0.000							STRAINERS MODELED SEPARATELY	
25	PWR VI.E.374	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
26	PWR IX.D.205	4+2	0.000	0.000	0.000	1.000	START	N	0.000	0.000							NA - NO LOW SUCTION PRESSURE TRIP	
27	PWR IX.D.205	4+2	0.000	1.000			START	N	0.000	0.000							NA - NO LOW SUCTION PRESSURE TRIP	
28	PWR XVI.C.984	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
29	PWR XVI.C.300	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
							TOTAL IMPACT AT WBN		1.867	10.55			1.000	1.000				
									WBSIPS				WBSIPR					

Table 3.3.4-7 (Page 3 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
STANDBY PUMPS (CS)	POPULATION 2																	
1	PWR VII.A.233	4	0.770	0.200	0.030	0.000	START	N	0.533	0.048							APPLICABLE	
2	PWR VIII.A.316	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS ARE NOT P.D. PUMPS	
3	PWR XVI.C.1624	3	0.000	1.000	0.000		START	N	0.000	1.000							APPLICABLE	
4	PWR XVI.C.778	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS DO NOT HAVE THIS TYPE OF LOGIC	
5	PWR XVI.C.801	3	0.000	0.000	1.000		START	L	0.000	1.000							APPLICABLE	
6	PWR XVI.C..58	3	0.000	1.000	0.000		START	N	0.667	0.333							APPLICABLE	
7	PWR V11.A.176	4	0.830	0.000	0.000	0.000	RUN	N					0.000	0.000			PUMPS ARE NOT VERTICAL PUMPS	
8	BWR VII.D.15	4	0.997	0.003	0.000	0.000	RUN	N					0.000	1.000			APPLICABLE	
9	BWR VII.D.124	4	0.235	0.765	0.000	0.000	RUN	N					1.000	0.000			PUMPS ARE SEPARATED	
10	PWR VII.A.289	2	0.000	1.000			START	L	0.000	0.000							NO RECIRC LINES IN EXTREMELY COLD AREAS	
11	BWR XVI.C.411	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
12	BWR XVI.C.488	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
13	BWR XI.B.207	4	0.000	1.000	0.000	0.000	START	N	0.667	0.167							APPLICABLE - FUSE CLIPS USED	
14	PWR VII.B.51	2	0.000	0.970			START	L	0.000	0.000							PUMPS ARE NOT PIGGY-BACK WITH ANY OTHER SYSTEM	
15	PWR XVI.C.188	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
16	PWR XVI.C.559	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
17	PWR XVI.C.623	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
18	PWR XVI.C.669	2	0.000	1.000			START	L	0.000	0.000							OPERATORS NOT LIKELY TO TAG OUT PUMPS DURING POWER OPERATION	
19	PWR XVI.C.801	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
20	PWR VII.B.91	2	0.000	1.000			START	N	0.000	1.000							APPLICABLE	
21	PWR VI.E.80	2+1	0.000	0.000	0.440		RUN	N					0.000	0.000			NA-STRAINERS MODELED SEPARATELY	
22	PWR VI.E.36	2+1	0.000	1.000			START	N	0.000	0.000							CS PUMPS TAKE SUCTION FROM BOTTOM OF RWST	
23	PWR VI.E.49	2+1	0.000	1.000			START	N	0.000	0.000							NA-PUMPS HAVE NO PACKING	
24	PWR VI.E.46	2+2	0.000	0.500	0.000	0.000	START	N	0.000	0.000							NA-STRAINERS MODELED SEPARATELY	
25	PWR VI.E.374	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
26	PWR IX.D.205	4+2	0.000	0.000	0.000	1.000	START	N	0.000	0.000							NO LOW SUCTION PRESSURE TRIP	
27	PWR IX.D.205	4+2	0.000	1.000			START	N	0.000	0.000							NO LOW SUCTION PRESSURE TRIP	
28	PWR XVI.C.984	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
29	PWR XVI.C.300	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
							TOTAL IMPACT AT WBN		1.867	10.55			1.000	1.000				
									WBCSPS				WBCSPR					
STANDBY PUMPS (RHR)	POPULATION 2																	
1	PWR VII.A.233	4	0.770	0.200	0.030	0.000	START	N	0.533	0.048							APPLICABLE	
2	PWR VIII.A.316	3	0.000	0.000	1.000		START	N	0.000	0.000							PUMPS ARE NOT P.D. PUMPS	
3	PWR XVI.C.1624	3	0.000	1.000	0.000		START	N	0.000	1.000							APPLICABLE	
4	PWR XVI.C.778	3	0.000	0.000	1.000		START	N	0.000	0.000							NA - PUMPS DO NOT HAVE THIS TYPE OF LOGIC	
5	PWR XVI.C.801	3	0.000	0.000	1.000		START	L	0.000	0.000							NA - OPERATORS NOT LIKELY TO TAG OUT OPERATING PUMPS	
6	PWR XVI.C..58	3	0.000	1.000	0.000		START	N	0.667	0.333							APPLICABLE	
7	PWR V11.A.176	4	0.830	0.000	0.000	0.000	RUN	N					0.000	0.000			PUMPS ARE NOT VERTICAL PUMPS	
8	BWR VII.D.15	4	0.997	0.003	0.000	0.000	RUN	N					0.000	1.000			APPLICABLE	

Table 3.3.4-7 (Page 4 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
9	BWR VII.D.124	4	0.235	0.765	0.000	0.000	RUN	N					1.000	0.000			PUMPS ARE SEPARATED	
10	PWR VII.A.289	2	0.000	1.000			START	L	0.000	0.000							NO RECIRC LINES IN EXTREMELY COLD AREAS	
11	BWR XVI.C.411	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
12	BWR XVI.C.488	4	0.000	1.000	0.000	0.000	START	N	0.000	0.000							NA - SINGLE UNIT SITE	
13	BWR XI.B.207	4	0.000	1.000	0.000	0.000	START	N	0.667	0.167							APPLICABLE - FUSE CLIPS USED	
14	PWR VII.B.51	2	0.000	0.970			START	L	0.000	0.000							PUMPS ARE NOT PIGGY-BACK WITH ANY OTHER SYSTEM	
15	PWR XVI.C.188	2	0.000	1.000			START	L	0.000	0.000							NA TO RHR-PUMPS ARE NOT RACKED OUT DURING OPERATION NORMALLY	
16	PWR XVI.C.559	2	0.000	1.000			START	L	0.000	0.000							NA TO RHR-PUMPS ARE NOT RACKED OUT DURING OPERATION NORMALLY	
17	PWR XVI.C.623	2	0.000	1.000			START	L	0.000	0.000							NA TO RHR-PUMPS ARE NOT NORMALLY RACKED OUT DURING OPERATION	
18	PWR XVI.C.669	2	0.000	1.000			START	L	0.000	0.000							OPERATORS NOT LIKELY TO TAG OUT PUMPS DURING POWER OPERATION	
19	PWR XVI.C.801	2	0.000	1.000			START	L	0.000	0.000							NA TO RHR-PUMPS NOT NORMALLY TAGGED OUT DURING OPERATION	
20	PWR VII.B.91	2	0.000	1.000			START	N	0.000	1.000							APPLICABLE	
21	PWR VI.E.80	2+1	0.000	0.000	0.440		RUN	N					0.000	0.000			NA-STRAINERS MODELED SEPARATELY	
22	PWR VI.E.36	2+1	0.000	1.000			START	N	0.000	0.000							RHR PUMPS TAKE SUCTION FROM BOTTOM OF RWST	
23	PWR VI.E.49	2+1	0.000	1.000			START	N	0.000	0.000							NA-PUMPS HAVE BEARINGS	
24	PWR VI.E.46	2+2	0.000	0.500	0.000	0.000	START	N	0.000	0.000							NA-STRAINERS MODELED SEPARATELY	
25	PWR VI.E.374	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
26	PWR IX.D.205	4+2	0.000	0.000	0.000	1.000	START	N	0.000	0.000							NO LOW SUCTION PRESSURE TRIP	
27	PWR IX.D.205	4+2	0.000	1.000			START	N	0.000	0.000							NO LOW SUCTION PRESSURE TRIP	
28	PWR XVI.C.984	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
29	PWR XVI.C.300	2	0.000	1.000			START	L	0.000	1.000							APPLICABLE	
							TOTAL IMPACT AT WBN		1.867	5.548			1.000	1.000				
									WBRHPS				WBRHPR					
OPERATING PUMPS (CHARGING, TB BOOSTER) POPULATION 2																		
1	PWR VIII.B.218	10	0.180	0.000	0.000	0.000	RUN	N					0.000	0.000			PUMPS DON'T USE RIVER WATER FOR SEAL\BEARING COOLING, TB BOOSTER PUMPS SELF-COOLED	
2	PWR VIII.B.159	3	0.000	1.000	0.000		RUN	N					0.667	0.333			APPLICABLE	
3	PWR VI.F.85	2	1.170	0.000			RUN	N					1.170	0.000			APPLICABLE	
4	BWR VIII.C.78	5	1.840	0.076	0.000	0.000	RUN	N					0.000	0.000			PUMPS NOT OPERATING BEYOND CAPACITY	
5	BWR VIII.C.45	5	0.000	0.170	0.000	0.000	RUN	N					0.000	0.000			PUMPS NOT OPERATING BEYOND CAPACITY	
6	BWR VIII.C.33	5	1.170	0.440	0.080	0.000	RUN	N					0.000	0.000			PUMPS NOT OPERATING BEYOND CAPACITY	
7	PWR XVI.C.642	2	0.000	1.000			START	N	0.000	0.000							NA - PUMPS AUTO SELECTED	
8	BWR XI.B.196	2	0.000	1.000			TRIP	N										
9	PWR VIII.B.516	3	2.000	0.000	0.000		RUN	N					0.000	0.000			NA-OIL RESERVOIR\SLINGER TYPE	
							TOTAL IMPACT AT WBN		0.000	0.000			1.837	0.333				
									WBCTPS,WBTBPS				WBCTPR, WBTBPR					
OPERATING PUMPS (ERCW) POPULATION 8																		
1	PWR VIII.B.218	10	0.180	0.000	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	TOP BEARING FED FROM RIVER WATER	
2	PWR VIII.B.159	3	0.000	1.000	0.000		RUN	N					0.000	0.250	0.750	0.000	APPLICABLE	0.75
3	PWR VI.F.85	2	1.170	0.000			RUN	N					0.146	0.548	0.658	0.000	APPLICABLE	0.75

Table 3.3.4-7 (Page 5 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
4	BWR VIII.C.78	5	1.840	0.076	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
5	BWR VIII.C.45	5	0.000	0.170	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
6	BWR VIII.C.33	5	1.170	0.440	0.080	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
7	PWR XVI.C.642	2	0.000	1.000			START	N	0.000	0.063	0.375	0.563					APPLICABLE	0.75
8	BWR XI.B.196	2	0.000	1.000			TRIP	N									FAILURE MODE NOT MODELED	
9	PWR VIII.B.516	3	2.000	0.000	0.000		RUN	N					0.000	0.000	0.000	0.000	NA - OIL BATH	
	TOTAL IMPACT AT WBN								0.000	0.063	0.375	0.563	0.146	0.798	1.408	0.000		
									W_EWPS				W_EWPR					
OPERATING PUMPS (CCS) POPULATION 5																		
1	PWR VIII.B.218	10	0.180	0.000	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS ARE SUPPLIED CLEAN WATER FOR SEAL/BEARING COOLING	
2	PWR VIII.B.159	3	0.000	1.000	0.000		RUN	N					0.000	0.250	0.750	0.000	APPLICABLE	0.75
3	PWR VI.F.85	2	1.170	0.000			RUN	N					0.146	0.548	0.658	0.000	APPLICABLE	0.75
4	BWR VIII.C.78	5	1.840	0.076	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
5	BWR VIII.C.45	5	0.000	0.170	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
6	BWR VIII.C.33	5	1.170	0.440	0.080	0.000	RUN	N					0.000	0.000	0.000	0.000	PUMPS NOT OPERATING BEYOND CAPACITY	
7	PWR XVI.C.642	2	0.000	1.000			START	N	0.000	0.063	0.375	0.563						0.75
8	BWR XI.B.196	2	0.000	1.000			TRIP	N									FAILURE MODE NOT MODELED	
9	PWR VIII.B.516	3	2.000	0.000	0.000		RUN	N					0.000	0.000	0.000	0.000	NOT-APPLICABLE-OIL RESERVOIR	
	TOTAL IMPACT AT WBN								0.000	0.063	0.375	0.563	0.146	0.798	1.408	0.000		
									W_CCPS				W_CCPR					
STANDBY VENTILATION FANS POPULATION 4																		
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					2.000	0.000	0.000	0.000		
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.000	1.000	0.000	0.000		
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000	0.000	0.000	NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000	0.000	0.000	FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					4.000	0.000	0.000	0.000		
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	1.000		
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	0.000	0.000	1.000		
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					0.155	0.605	0.840	0.214		0.75
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	NO LOW FLOW TRIPS	
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					2.667	0.000	0.000	0.000		
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	0.000	0.000	0.000					FANS NOT INTERLOCKED WITH ANY OTHER SYSTEM	
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	1.000	0.000	0.000					EACH RELAY ASSOCIATED WITH ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	1.000	0.000	0.000	0.000					ONE FAN PER TRAIN	
	TOTAL IMPACT AT WBN								5.00	1.00	0.00	0.00	8.82	1.61	0.84	2.21		

Table 3.3.4-7 (Page 6 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
									W_F4SS				W_F4SR					
OPERATING VENTILATION FANS POPULATION 4																		
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					2.000	0.000	0.000	0.000		
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000	NOT APPLICABLE TO NORMALLY OPERATING FANS	
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000	0.000	0.000	NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000	0.000	0.000	FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					2.000	0.000	0.000	0.000		
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	ELEC BLDG FANS ARE NORMALLY OPERATING	
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	0.000	0.000	1.000		
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					0.155	0.605	0.840	0.214		0.75
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	NO LOW FLOW TRIPS	
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					2.667	0.000	0.000	0.000		
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	0.000	0.000	0.000					FANS NOT INTERLOCKED WITH ANY OTHER SYSTEM	
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						0.75
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	1.000	0.000	0.000					EACH RELAY STARTS ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	1.000	0.000	0.000	0.000					ONE FAN PER TRAIN	
							TOTAL IMPACT AT WBN		5.00	1.00	0.00	0.00	6.82	0.61	0.84	1.21		
									W_F4OS				W_F4OR					
STANDBY VENTILATION FANS POPULATION 2																		
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					1.000	0.000				
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.667	0.167				
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000			NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					1.000	0.000				
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	1.000				
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	1.000				
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					1.240	0.380				
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NO LOW FLOW TRIPS	
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					1.333	0.000				
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	0.000							FANS NOT INTERLOCKED WITH ANY OTHER SYSTEM	
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	1.000							EACH COMPONENT STARTS ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	1.000	0.000							ONE FAN PER TRAIN	
							TOTAL IMPACT AT WBN		3.00	1.00			5.24	2.55				

Table 3.3.4-7 (Page 7 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
									W_F2SS				W_F2SR					
OPERATING VENTILATION FANS POPULATION 2																		
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					1.000	0.000				
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.000	0.000			NOT APPLICABLE TO NORMALLY OPERATING FANS	
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000			NOT APPLICABLE TO INDIVIDUAL TRAIN FANS	
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					1.000	0.000				
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			ELEC BUILDING FANS ARE NORMALLY OPERATING	
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	1.000				
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					1.240	0.380				
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NO LOW FLOW TRIPS	
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					1.333	0.000				
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NOT APPLICABLE TO INDIVIDUAL ROOM COOLING FANS	
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	0.000							FANS NOT INTERLOCKED WITH ANY OTHER SYSTEM	
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	1.000							EACH COMPONENT STARTS ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	1.000	0.000							ONE FAN PER TRAIN	
							TOTAL IMPACT AT WBN		3.00	1.00			4.57	1.38				
									W_F2OS				W_F2OR					
MOTOR-OPERATED VALVES POPULATION 4																		
1	PWR VIII.A.548	6	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
2	PWR VI.E.210	3	0.000	0.500	0.000		OPEN/CLOSE	N	0.000	0.125	0.375	0.000						0.75
3	PWR VI.E.276	10	0.930	0.460	0.000	0.000	OPEN/CLOSE	N	0.930	0.460	0.000	0.000						
4	PWR VII.A.99	9	0.711	0.186	0.070	0.003	OPEN/CLOSE	N	0.711	0.186	0.070	0.003						
5	PWR VII.A.137	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	1.000	0.000	0.000	0.000						
6	PWR VII.C.16	3	0.045	0.123	0.188		OPEN/CLOSE	N	0.015	0.065	0.139	0.141						0.75
7	PWR VIII.A.26	2	0.820	0.180			OPEN/CLOSE	N	0.000	0.000	0.000	0.000					NO VALVES AT WBN EXPOSED TO THE WEATHER	
8	BWR VII.D.86	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.063	0.375	0.563						0.75
9	BWR VII.D.49	4	0.627	0.272	0.100	0.000	OPEN/CLOSE	N	0.627	0.272	0.100	0.000						
10	BWR VII.D.46	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.063	0.375	0.563						0.75
11	PWR XVI.C.539	8	0.000	0.000	1.000	0.000	OPEN/CLOSE	N	0.000	0.000	1.000	0.000						
12	PWR VII.C.12	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
13	PWR VII.A.130	30	0.000	0.010	0.010	0.510	OPEN/CLOSE	N	0.000	0.010	0.010	0.510						
14	PWR VII.A.2	4	0.998	0.000	0.000	0.002	OPEN/CLOSE	N	0.998	0.000	0.000	0.002						
15	PWR VII.A.180	45	0.330	0.610	0.000	0.000	OPEN/CLOSE	N	0.330	0.610	0.000	0.000						
16	PWR VIII.B.100	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
17	PWR VI.E.194	12	0.100	0.235	0.665	0.000	OPEN/CLOSE	N	0.100	0.235	0.665	0.000						
18	BWR V.D.4	25	0.536	0.222	0.114	0.128	OPEN/CLOSE	N	0.536	0.222	0.114	0.128						
19	BWR V.D.40	150	0.950	0.050	0.000	0.000	OPEN/CLOSE	N	0.950	0.050	0.000	0.000						

Table 3.3.4-7 (Page 8 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS P1 (FAIL TO OPERATE ON DEMAND)	P2	P3	P4	MAPPED IMPACTS P1 (FAIL DURING MISSION TIME)	P2	P3	P4	REMARKS	RHO
20	PWR XVI.C.1253	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	1.000	0.000	0.000	0.000						
21	PWR XVI.C.1428	N	0.000	1.000	0.00	0.00	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
22	BWR VIII.D.23	28	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000					APPLICABLE - NOT MAPPED UP	
23	BWR VII.D.97, & BWR VII.C.24, 50	76	0.170	0.218	0.243	0.369	OPEN/CLOSE	N	0.170	0.218	0.243	0.369						
24	BWR VII.D.104	2	1.000	0.000			OPEN/CLOSE	N	2.000	0.000	0.000	0.000						
25	BWR VII.D.165	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
26	BWR VII.D.193	4	4.000	0.000	0.000	0.000	OPEN/CLOSE	N	4.000	0.000	0.000	0.000						
27	BWR VII.D.219	81	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
28	BWR VII.D.220	3	0.000	1.000	0.000		OPEN/CLOSE	N	0.000	0.250	0.750	0.000						0.75
29	BWR VIII.C.53	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.063	0.375	0.563						0.75
30	BWR IX.E.236	2	0.015	0.002			OPEN/CLOSE	N	0.000	0.000	0.000	0.000					VALVES MODELED WITH SINGLE SIGNAL	
31	BWR XI.B.35	13	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
32	BWR XVI.C.35	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000	0.000						
							TOTAL IMPACT AT WBN		13.37	11.89	4.59	2.84						
									W_MV4D									
MOTOR-OPERATED VALVES		POPULATION 3																
1	PWR VIII.A.548	6	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							
2	PWR VI.E.210	3	0.000	0.500	0.000		OPEN/CLOSE	N	0.000	0.500	0.000							
3	PWR VI.E.276	10	0.930	0.460	0.000	0.000	OPEN/CLOSE	N	0.928	0.230	0.000							0.75
4	PWR VII.A.99	9	0.711	0.186	0.070	0.003	OPEN/CLOSE	N	0.626	0.146	0.021							
5	PWR VII.A.137	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	0.750	0.000	0.000							
6	PWR VII.C.16	3	0.045	0.123	0.188		OPEN/CLOSE	N	0.045	0.123	0.188							
7	PWR VIII.A.26	2	0.820	0.180			OPEN/CLOSE	N	0.000	0.000	0.000							0.75
8	BWR VII.D.86	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.250	0.750						NO VALVES AT WBN EXPOSED TO THE WEATHER	
9	BWR VII.D.49	4	0.627	0.272	0.100	0.000	OPEN/CLOSE	N	0.606	0.211	0.025							0.75
10	BWR VII.D.46	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.250	0.750							
11	PWR XVI.C.539	8	0.000	0.000	1.000	0.000	OPEN/CLOSE	N	0.000	0.750	0.250							0.75
12	PWR VII.C.12	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							
13	PWR VII.A.130	30	0.000	0.010	0.010	0.510	OPEN/CLOSE	N	0.005	0.013	0.513							
14	PWR VII.A.2	4	0.998	0.000	0.000	0.002	OPEN/CLOSE	N	0.749	0.000	0.002							
15	PWR VII.A.180	45	0.330	0.610	0.000	0.000	OPEN/CLOSE	N	0.553	0.305	0.000							
16	PWR VIII.B.100	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							
17	PWR VI.E.194	12	0.100	0.235	0.665	0.000	OPEN/CLOSE	N	0.193	0.616	0.166							
18	BWR V.D.4	25	0.536	0.222	0.114	0.128	OPEN/CLOSE	N	0.513	0.197	0.157							
19	BWR V.D.40	150	0.950	0.050	0.000	0.000	OPEN/CLOSE	N	0.737	0.025	0.000							
20	PWR XVI.C.1253	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	0.750	0.000	0.000							
21	PWR XVI.C.1428	N	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000						APPLICABLE - NOT MAPPED DOWN	
22	BWR VIII.D.23	28	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							
23	BWR VII.D.97, & BWR VII.C.24, 50	76	0.170	0.218	0.243	0.369	OPEN/CLOSE	N	0.236	0.291	0.430							
24	BWR VII.D.104	2	1.000	0.000			OPEN/CLOSE	N	1.500	0.000	0.000							0
25	BWR VII.D.165	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							

Table 3.3.4-7 (Page 9 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
26	BWR VII.D.193	4	4.000	0.000	0.000	0.000	OPEN/CLOSE	N	3.000	0.000	0.000							
27	BWR VII.D.219	81	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							0.75
28	BWR VII.D.220	3	0.000	1.000	0.000		OPEN/CLOSE	N	0.000	1.000	0.000							0.75
29	BWR VIII.C.53	2	0.000	1.000			OPEN/CLOSE	N	0.000	0.250	0.750							
30	BWR IX.E.236	2	0.015	0.002			OPEN/CLOSE	N	0.000	0.000	0.000						ALL VALVES MODELED WITH SINGLE SIGNAL	
31	BWR XI.B.35	13	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.500	0.000							
32	BWR XVI.C.35	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000	0.000							
							TOTAL IMPACT AT WBN		14.69	10.66	4.00							
									W_MV3D									

MOTOR-OPERATED VALVES POPULATION 2

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS FAIL TO OPERATE ON DEMAND P1	P2	P3	P4	MAPPED IMPACTS FAIL DURING MISSION TIME P1	P2	P3	P4	REMARKS	RHO
1	PWR VIII.A.548	6	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
2	PWR VI.E.210	3	0.000	0.500	0.000		OPEN/CLOSE	N	0.333	0.167								
3	PWR VI.E.276	10	0.930	0.460	0.000	0.000	OPEN/CLOSE	N	0.772	0.077								
4	PWR VII.A.99	9	0.711	0.186	0.070	0.003	OPEN/CLOSE	N	0.515	0.069								
5	PWR VII.A.137	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.000								
6	PWR VII.C.16	3	0.045	0.123	0.188		OPEN/CLOSE	N	0.112	0.229								
7	PWR VIII.A.26	2	0.820	0.180			OPEN/CLOSE	N	0.000	0.000							NO VALVES AT WBN EXPOSED TO THE WEATHER	
8	BWR VII.D.86	2	0.000	1.000			OPEN/CLOSE	N	0.000	1.000								
9	BWR VII.D.49	4	0.627	0.272	0.100	0.000	OPEN/CLOSE	N	0.545	0.095								
10	BWR VII.D.46	2	0.000	1.000			OPEN/CLOSE	N	0.000	1.000								
11	PWR XVI.C.539	8	0.000	0.000	1.000	0.000	OPEN/CLOSE	N	0.500	0.500								
12	PWR VII.C.12	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
13	PWR VII.A.130	30	0.000	0.010	0.010	0.510	OPEN/CLOSE	N	0.012	0.517								
14	PWR VII.A.2	4	0.998	0.000	0.000	0.002	OPEN/CLOSE	N	0.499	0.002								
15	PWR VII.A.180	45	0.330	0.610	0.000	0.000	OPEN/CLOSE	N	0.572	0.102								
16	PWR VIII.B.100	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
17	PWR VI.E.194	12	0.100	0.235	0.665	0.000	OPEN/CLOSE	N	0.539	0.372								
18	BWR V.D.4	25	0.536	0.222	0.114	0.128	OPEN/CLOSE	N	0.473	0.222								
19	BWR V.D.40	150	0.950	0.050	0.000	0.000	OPEN/CLOSE	N	0.508	0.008								
20	PWR XVI.C.1253	4	1.000	0.000	0.000	0.000	OPEN/CLOSE	N	0.500	0.000								
21	PWR XVI.C.1428	N	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000							APPLICABLE - NOT MAPPED DOWN	
22	BWR VIII.D.23	28	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
23	BWR VII.D.97, &	76	0.170	0.218	0.243	0.369	OPEN/CLOSE	N	0.352	0.527								
	BWR VII.C.24,	50																
24	BWR VII.D.104	2	1.000	0.000			OPEN/CLOSE	N	1.000	0.000								
25	BWR VII.D.165	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
26	BWR VII.D.193	4	4.000	0.000	0.000	0.000	OPEN/CLOSE	N	2.000	0.000								
27	BWR VII.D.219	81	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
28	BWR VII.D.220	3	0.000	1.000	0.000		OPEN/CLOSE	N	0.667	0.333								
29	BWR VIII.C.53	2	0.000	1.000			OPEN/CLOSE	N	0.000	1.000								
30	BWR IX.E.236	2	0.015	0.002			OPEN/CLOSE	N	0.000	0.000							ALL VALVES MODELED WITH SINGLE SIGNAL	
31	BWR XI.B.35	13	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.667	0.167								
32	BWR XVI.C.35	4	0.000	1.000	0.000	0.000	OPEN/CLOSE	N	0.000	1.000								

Table 3.3.4-7 (Page 10 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS P1	P2	P3	P4	MAPPED IMPACTS P1	P2	P3	P4	REMARKS	RHO
									FAIL TO OPERATE ON DEMAND				FAIL DURING MISSION TIME					
							TOTAL IMPACT AT WBN		15.06	9.39								
									WBMV2D									
CIRCUIT BREAKERS POPULATION 4																		
1	PWR XI.B.747	4	3.000	0.000	0.000	0.000	DEMAND	N	3.000	0.000	0.000	0.000					APPLICABLE	
2	PWR XI.B.664	2	0.000	1.000			DEMAND	N	0.000	0.063	0.375	0.563					APPLICABLE	0.75
3	PWR XI.B.325	4	0.000	1.000	0.000	0.000	DEMAND	N	0.000	1.000	0.000	0.000					APPLICABLE	
4	PWR XI.A.119	4	0.000	0.000	1.000	0.000	DEMAND	N	0.000	0.000	0.000	0.000					SEQUENCER IS MODELED SEPARATELY	
5	PWR XVI.C.2792	4	0.000	1.000	0.000	0.000	DEMAND	N	0.000	1.000	0.000	0.000					APPLICABLE	
6	BWR V.B.64	2	0.000	1.000			DEMAND	N	0.000	0.063	0.375	0.563					APPLICABLE	0.75
7	BWR XI.A.38	4	0.000	1.000	0.000	0.000	DEMAND	N	0.000	1.000	0.000	0.000					APPLICABLE	
8	BWR XI.A.205	2	0.000	1.000			DEMAND	N	0.000	0.063	0.375	0.563					APPLICABLE	0.75
9	BWR XI.A.217	3	0.000	0.000	1.000		DEMAND	L	0.000	0.000	0.000	1.000					APPLICABLE	
10	BWR XI.A.474	2	1.000	0.000			DEMAND	N	2.000	0.000	0.000	0.000					APPLICABLE	
11	BWR XI.B.7	2	0.000	1.000			DEMAND	N	0.000	0.063	0.375	0.563					APPLICABLE	0.75
12	BWR XI.B.463	4	0.000	1.000	0.000	0.000	DEMAND	N	0.000	1.000	0.000	0.000					APPLICABLE	
13	BWR IX.B.477	4	0.000	0.000	0.000	1.000	DEMAND	N	0.000	0.000	0.000	1.000					APPLICABLE	
14	BWR XI.A.504	3	0.000	0.000	1.000		DEMAND	L	0.000	0.000	0.000	0.000					WBN DOES NOT HAVE SIMILAR LOGIC	
							TOTAL IMPACT AT WBN		5.000	4.250	1.500	4.250						
									W_CB1D									
REACTOR TRIP BREAKERS, UV AND SHUNT TRIP COILS POPULATION 2																		
									UV				BRK					
1	PWR IX.B.13	2	0.000	1.000			DEMAND	UV N	0.000	1.000								
2	PWR IX.B.361	13	0.000	1.000	0.000	0.000	DEMAND	BRK N					0.667	0.167				
3	PWR IX.B.524 & PWR IX.B.585	8	1.000	0.000	0.000	0.000	DEMAND	UV N	0.500	0.000								
4	PWR IX.B.757	2	0.000	1.000			DEMAND	UV N	0.000	1.000								
5	PWR IX.B.761 & PWR IX.B.771	9	0.000	0.000	1.000	0.000	DEMAND	UV N	0.500	0.500								
6	PWR IX.B.777	4	1.000	0.500	0.000	0.000	DEMAND	UV N	0.833	0.083								
							TOTAL IMPACT AT WBN		1.833	2.583			0.667	0.833				
									WBUVCD				WBRTBD					
CHECK VALVES POPULATION 8																		
									RESEAT				OPEN					
1	PWR VII.C.126	5	0.000	0.000	0.000	1.000	OPEN	L									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
2	BWR VII.C.42	2	0.000	1.000	0.000	0.000	OPEN	N									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
3	PWR VII.B.53	4	0.000	1.000	0.000	0.000	OPEN	L									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
4	PWR VI.E.278	4	0.000	1.000	0.000	0.000	RESEAT	N	0.000	1.000	0.000	0.000						
5	PWR VI.D.290	4	0.000	0.000	0.000	0.540	RESEAT	N	0.000	0.000	0.000	0.000					NA	
6	PWR VI.D.319	4	0.075	0.290	0.056	0.000	RESEAT	N	0.000	0.000	0.000	0.000					NA	

Table 3.3.4-7 (Page 11 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS P1 (FAIL TO OPERATE ON DEMAND)	P2	P3	P4	MAPPED IMPACTS P1 (FAIL DURING MISSION TIME)	P2	P3	P4	REMARKS	RHO
7	PWR VI.E.630	12	0.000	0.000	0.000	1.000	RESEAT	N	0.000	0.000	0.000	0.000					THIS FAILURE MODE TYPICAL OF STEAM LINE CHECK VALVES	
8	PWR VII.A.315	4	0.000	1.000	0.000	0.000	RESEAT	N	0.000	1.000	0.000	0.000						
	TOTAL IMPACT AT WBN								0.000	2.000	0.000	0.000						
									WBVC4D									
	CHECK VALVES POPULATION 2																	
									RESEAT				OPEN					
1	PWR VII.C.126	5	0.000	0.000	0.000	1.000	OPEN	L									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
2	BWR VII.C.42	2	0.000	1.000	0.000	0.000	OPEN	N									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
3	PWR VII.B.53	4	0.000	1.000	0.000	0.000	OPEN	L									FAILURE MODE NOT MODELED FOR COMMON CAUSE	
4	PWR VI.E.278	4	0.000	1.000	0.000	0.000	RESEAT	N	0.667	0.167								
5	PWR VI.D.290	4	0.000	0.000	0.000	0.540	RESEAT	N	0.000	0.000							NA	
6	PWR VI.D.319	4	0.075	0.290	0.056	0.000	RESEAT	N	0.000	0.000							NA	
7	PWR VI.E.630	12	0.000	0.000	0.000	1.000	RESEAT	N	0.000	0.000							THIS FAILURE MODE TYPICAL OF STEAM LINE CHECK VALVES	
8	PWR VII.A.315	4	0.000	1.000	0.000	0.000	RESEAT	N	0.667	0.167								
	TOTAL IMPACT AT WBN								1.333	0.333								
									WBVC2D									
	CONTAINMENT AIR RETURN FANS POPULATION 2																	
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					1.000	0.000				
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.667	0.167				
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000			NOT APPLICABLE TO CONTAINMENT AIR RETURN FANS	
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					1.000	0.000				
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	1.000				
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	1.000				
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			FANS HAVE NO OTHER SYSTEM INTERLOCK SIGNALS	
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					1.240	0.380				
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NO LOW FLOW TRIPS	
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					1.333	0.000				
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000			NOT APPLICABLE TO CONTAINMENT AIR RETURN FANS	
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	0.000							FANS NOT INTERLOCKED WITH ANY OTHER SYSTEM	
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	1.000	0.000								
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	1.000							EACH COMPONENT STARTS ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	1.000	0.000							ONE FAN PER TRAIN	
	TOTAL IMPACT AT WBN								3.00	1.00			5.24	2.55				
									WBFN1S				WBFN1R					
	AUXILIARY BUILDING GENERAL VENTILATION POPULATION 4																	
1	PWR XIV.B.115	4	2.000	0.000	0.000	0.000	RUN	N					2.000	0.000	0.000	0.000		

Table 3.3.4-7 (Page 12 of 12). Common Cause Events Impact Vectors for Watts Bar Components

NO.	EVENT ID	POPULATION IN EVENT	ORIGINAL IMPACTS P1	P2	P3	P4	FAILURE MODE	SHOCK TYPE	MAPPED IMPACTS P1 FAIL TO OPERATE ON DEMAND	P2	P3	P4	MAPPED IMPACTS P1 FAIL DURING MISSION TIME	P2	P3	P4	REMARKS	RHO
2	PWR XVI.B.330	6	0.000	1.000	0.000	0.000	RUN	N					0.000	0.000	0.000	0.000		
3	BWR XVI.B.77	3	0.000	0.000	1.000		RUN	L					0.000	0.000	0.000	1.000		1
4	PWR IX.E.615	2	0.000	1.000			RUN	L					0.000	0.000	0.000	1.000		1
5	BWR XVI.B.1104	2	1.000	0.000			RUN	N					2.000	0.000	0.000	0.000		
6	PWR XVI.B.314	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	0.000	AUX BUILDING FANS ARE NORMALLY OPERATING	
7	BWR XVI.B.193	N	0.000	0.000	0.000	1.000	RUN	N					0.000	0.000	0.000	1.000		
8	PWR XVI.C.1112	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	1.000		
9	PWR XVI.B.186	2	1.240	0.380			RUN	N					0.155	0.605	0.840	0.214		0.75
10	PWR IX.F.84	4	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	1.000		1
11	PWR VII.C.34	3	2.000	0.000	0.000		RUN	N					2.667	0.000	0.000	0.000		
12	BWR XVI.B.510	N	0.000	0.000	0.000	1.000	RUN	L					0.000	0.000	0.000	1.000		
13	PWR IX.E.675	4	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						
14	PWR XVI.C.269	2	0.000	1.000			START	N	0.000	1.000	0.000	0.000						
15	BWR XVI.B.501	6	2.000	0.000	0.000	0.000	START	N	2.000	0.000	0.000	0.000						
16	BWR XI.B.185	8	0.000	0.000	0.000	1.000	START	N	0.000	0.000	0.000	1.000					EACH COMPONENT STARTS ONE FAN	
17	PWR XVI.C.2267	2	0.000	1.000			START	N	0.000	1.000	0.000	0.000					ONE FAN PER TRAIN	
							TOTAL IMPACT AT WBN		4.00	2.00	0.00	1.00	6.82	0.61	0.84	6.21		
									W_FNAS				W_FNAR					

* WBDGGS BETA FACTOR
WGDGGS GAMMA FACTOR
WDDGGS DELTA FACTOR

Table 3.3.4-8 (Page 1 of 2). Summary of Common Cause Failure Parameters for Watts Bar Components

NO.	DATA DESIG.	IMPACT VECTORS				NUMBER OF EVENTS AT WBN				NO OF INDEPENDENT EVENTS IN DATA BASE	POPULATION IN GENERIC PLANT	POPULATION AT WBN	PRIOR							POSTERIOR					
		N1	N2	N3	N4	N1WBN	N2WBN	N3WBN	N4WBN	NGN	PGN	PWBN	NI	AO	BO	CO	DO	EO	FO	A	B	C	D	E	F
1	W_DGGS*	0.33	0.96	0.70	0.10					392	2.06	4	761	0.42	8.37	0.30	0.87	0.63	0.88	4.81	769.	2.78	2.79	1.01	2.98
2	W_DGGR	2.59	1.84	1.09	0.64					174	2.06	4	338	0.48	22.9	2.25	5.31	0.79	0.79	9.97	363.	8.06	8.98	3.34	4.06
3	WBMAPS	1.33	5.5							86	**	2	71	1.58	21					12.5	93.1				
4	WBMAPR	1.00	1.00							100	**	2	80	1.58	156.					3.58	237.				
5	WBSIPS	1.87	10.5							86	**	2	71	1.58	21					22.6	93.7				
6	WBSIPR	1.00	1.00							100	**	2	80	1.58	156.					3.58	237.				
7	WBCSPS	1.87	10.5							86	**	2	71	1.58	21					22.6	93.7				
8	WBCSPR	1.00	1.00							100	**	2	80	1.58	156.					3.58	237.				
9	WBRHPS	1.87	5.55							86	**	2	71	1.58	21					12.6	93.7				
10	WBRHPR	1.00	1.00							100	**	2	80	1.58	156.					3.58	237.				
11	WBCTPS	0.00	0.00							46	**	2	23	0.25	15.2					0.25	37.7				
12	WBCTPR	1.84	0.33							81	**	2	41	0.24	30.1					0.90	73.0				
13	WBTBPS	0.00	0.00							46	**	2	23	0.25	15.2					0.25	37.7				
14	WBTBPR	1.84	0.33							81	**	2	41	0.24	30.1					0.90	73.0				
15	W_EWPS	0.00	0.06	0.38	0.56					46	**	4	45	0.42	14.4	1.17	3.8	0.54	0.79	3.92	59.4	4.54	3.92	2.79	1.92
16	W_EWPR	0.15	0.80	1.41	0.00					81	**	4	82	0.37	26.3	1.29	3.91	0.2	0.29	6.19	108.	5.51	5.50	0.2	4.51
17	W_CCPS	0.00	0.06	0.38	0.56					46	**	4	45	0.42	14.4	1.17	3.8	0.54	0.79	3.92	59.4	4.54	3.92	2.79	1.92
18	W_CCPR	0.15	0.80	1.41	0.00					81	**	4	82	0.37	26.3	1.29	3.91	0.2	0.29	6.19	108.	5.51	5.50	0.2	4.51
19	W_F4SS	5.00	1.00	0.00	0.00					24	4	4	24	1.58	21	3.8	23.3	3.8	9.58	3.58	50	3.8	25.3	3.8	9.58
20	W_F4SR	8.82	1.61	0.84	2.21					42	4	4	42	1.58	156.	3.8	23.3	3.8	9.58	16.1	207.	15.1	26.5	12.6	12.1
21	W_F4OS	5.00	1.00	0.00	0.00					24	4	4	24	1.58	156.	3.8	23.3	1.8	8.31	3.58	185.	3.8	25.3	1.8	8.31
22	W_F4OR	6.82	0.61	0.84	1.21					42	4	4	42	0.84	839	1.58	21	1.8	8.31	9.42	887.	8.95	22.2	6.65	10.8
23	WBF2SS	3.00	1.00							24	4	2	12	1.58	21					3.58	36				
24	WBF2SR	5.24	2.55							42	4	2	21	1.58	156.					6.67	182.				
25	WBF2OS	3.00	1.00							24	4	2	12	1.58	156.					3.58	171.				
26	WBF2OR	4.57	1.38							42	4	2	21	0.84	839					3.6	864.				
27	WBFN1S	3.00	1.00							24	4	2	12	1.58	21					3.58	36				
28	WBFN1R	5.24	2.55							42	4	2	21	1.58	156.					6.67	182.				
29	W_FNAS	4.00	2.00	0.00	1.00					24	4	4	24	1.58	156.	3.8	23.3	1.8	8.31	9.58	184.	7.8	27.3	5.8	8.31
30	W_FNAR	6.82	0.61	0.84	6.21					42	4	4	42	0.84	839	1.58	21	1.8	8.31	29.4	887.	28.9	22.2	26.6	10.8
31	W_MV4D	13.4	11.9	4.6	2.8					784	4	4	784	1.58	21	3.8	23.3	1.8	8.31	50.4	818.	28.9	47.0	13.1	22.0
32	W_MV3D	14.7	10.7	4.0						784	4	3	588	1.58	21	3.8	23.3			34.8	623.	15.8	44.6		
33	WBMV2D	15.1	9.4							784	4	2	392	1.58	21					20.3	428.				
34	W_CB1D	5.00	4.25	1.50	4.25					99	4	4	99	1.58	21	3.8	23.3	1.8	8.31	31.5	125	25.3	31.8	18.8	12.8
35	WBRTBD	0.66	0.83							24	3.49	2	13.7	1.58	21					3.24	35.4				
36	WBUVCD	1.83	2.58							23	3.52	2	13.0	1.58	21					6.74	35.9				
37	WBSTCD	0.00	0.00							0	4	2	0	1.58	21					1.58	21				
38	WBVC4D	0.00	2.00	0.00	0.00					21	4	4	21	1.58	156.	3.8	23.3	3.8	9.58	5.58	177.	3.8	27.3	3.8	9.58
40	WBVC2D	1.33	0.33							21	4	4	21	1.58	156.					2.24	178.				
41	WBCDPS	***												1.58	21										
42	WBCDPR	***												1.58	156.										
43	WBPWPS	***												1.58	21										
44	WBPWPR	***												1.58	156.										
45	Z_PXRS	***												1.58	21	3.8	23.3	3.8	9.58						
46	Z_PXRR	***												1.58	156.	3.8	23.3	3.8	9.58						

Table 3.3.4-8 (Page 2 of 2). Summary of Common Cause Failure Parameters for Watts Bar Components

NO.	DATA DESIG.	IMPACT VECTORS				NUMBER OF EVENTS AT WBN				NO OF INDEPENDENT EVENTS IN DATA BASE	POPULATION IN GENERIC PLANT	POPULATION AT WBN		PRIOR						POSTERIOR					
		N1	N2	N3	N4	N1WBN	N2WBN	N3WBN	N4WBN	NGN	PGN	PWBN	NI	A0	B0	C0	D0	E0	F0	A	B	C	D	E	F
47	Z_CMPS	***												0.42	8.37	0.30	0.87	0.63	0.88						
48	Z_CMPR	***												0.48	22.9	2.25	5.31	0.79	0.79						
49	Z_LC1D	***												0.84	839	1.58	21	3.8	9.58						
50	Z_VAOD	***												1.58	21	3.8	23.3	1.8	8.31						
51	Z_VE1D	***												1.58	21	3.8	23.3	1.8	8.31						
52	Z_RL1D	***												1.58	21	1.58	21	3.8	9.58						
53	Z_SWBD	***												1.58	21	1.58	21	3.8	9.58						
54	Z_VSOD	***												1.58	21	3.8	23.3	1.58	21						
55	Z_DMAD	***												1.58	21	3.8	23.3	1.8	8.31						

* WBDGGS BETA FACTOR
WGDGGS GAMMA FACTOR
WDDGGS DELTA FACTOR
** POPULATION FOR THESE COMPONENTS IS PRESENTED IN TABLE WBN
*** GENERIC DATA

Table 3.3.4-9. Average Number of Components per Plant for Each Component Type of Interest

Component	Plant Type	Number of Components	Number of Plants	Average Number per Plant
Diesel Generator	PWR, BWR	138	67	2.06
AFW Motor-Driven Pump	PWR	55	35	1.57
AFW Turbine-Driven Pump	PWR	50	45	1.11
HHSI Pump	PWR	121	45	2.69
HPCI and RCIC Pumps	BWR	44	22	2.0
LHSI, LPCI, RHR Pump	PWR, BWR	193	69	2.80
Containment Spray Pump	PWR	99	42	2.36
Component Cooling Water Pump	PWR, BWR	187	68	2.75
Service Water	PWR, BWR	300	68	4.41
SBLC Pump	BWR	40	20	2.00
Reactor Trip Breaker	PWR	150	43	3.49
MOV*	PWR, BWR	926	67	13.82*
Check Valve*	PWR, BWR	730.8	67	10.91*
Reactor Trip Breaker Undervoltage Trip Attachment	PWR	148	42	3.52
Shunt Trip Attachment	PWR	50	7	7.14

*The MOV and check valve populations are the average number of the respective valves per system for each unit. The systems considered in the population data are core spray, HPCI, LPCI, containment spray, HHSI, LHSI, and AFW systems.

Table 3.3.4-10 (Page 1 of 5). Summary of Common Cause Failure Parameters for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WBCB1D	Beta Factor - Circuit Breakers > 480V Fail on Demand	2.01-01	1.39-01	1.98-01	2.49-01
WBCCPR	Beta Factor - Component Cooling Water Pumps Fail To Run	5.42-02	1.88-02	5.03-02	8.76-02
WBCCPS	Beta Factor - Component Cooling Water Pumps Fail To Run	6.19-02	1.44-02	5.55-02	1.10-01
WBCDPR	Beta Factor - Condensate Transfer Pumps Fail To Run	1.00-02	8.83-04	7.60-03	2.30-02
WBCDPS	Beta Factor - Condensate Transfer Pumps Fail To Start	7.00-02	5.86-04	5.46-02	1.57-01
WBCSPR	Beta Factor - Containment Spray Pumps Fail during Operation	1.49-02	2.93-03	1.31-02	2.75-02
WBCSPS	Beta Factor - Containment Spray Pumps Fail To Start	1.94-01	1.24-01	1.90-01	2.50-01
WBCTPR	Beta Factor - Centrifugal Charging Pumps Fail To Run	1.22-02	1.90-04	7.55-03	3.32-02
WBCTPS	Beta Factor - Centrifugal Charging Pumps Fail To Start	6.59-03	5.65-08	9.77-04	2.59-02
WBDGGR	Beta Factor - Diesel Generator Fails during Operation	2.67-02	1.21-02	2.54-02	3.98-02
WBDGGS	Beta Factor - Diesel Generator Fails To Start	6.22-03	1.69-03	5.63-03	1.07-02
WBEWPR	Beta Factor - ERCW Pumps Fail during Operation	5.42-02	1.88-02	5.03-02	8.76-02
WBEWPS	Beta Factor - ERCW Pumps Fail To Start	6.19-02	1.44-02	5.55-02	1.10-01
WBF2OR	Beta Factor - Operating Fans (Population 2) Fail To Run	4.14-03	8.10-04	3.64-03	7.66-03
WBF2OS	Beta Factor - Operating Fans (Population 2) Fail To Start	2.06-02	4.08-03	1.82-02	3.80-02
WBF2SR	Beta Factor - Standby Fans (Population 2) Fail To Run	3.54-02	1.27-02	3.29-02	5.66-02
WBF2SS	Beta Factor - Standby Fans (Population 2) Fail To Start	9.04-02	1.95-02	8.09-02	1.63-01
WBF4OR	Beta Factor - Operating Fans (Population 4) Fail To Run	1.05-02	4.59-03	9.94-03	1.58-02
WBF4OS	Beta Factor - Operating Fans (Population 4) Fail To Start	1.90-02	3.75-03	1.67-02	3.50-02
WBF4SR	Beta Factor - Standby Fans (Population 4) Fail To Run	7.22-02	4.03-02	6.98-02	9.88-02
WBF4SS	Beta Factor - Standby Fans (Population 4) Fail To Start	6.68-02	1.40-02	5.95-02	1.21-01
WBFN1R	Beta Factor - Containment Air Return Fans Fail To Run	3.54-02	1.27-02	3.29-02	5.66-02

Note: Exponential notation is indicated in abbreviated form; e.g., 2.01-01 = 2.01 × 10^{-01}.

Table 3.3.4-10 (Page 2 of 5). Summary of Common Cause Failure Parameters for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WBFN1S	Beta Factor - Containment Air Return Fans Fail To Start	9.04-02	1.95-02	8.09-02	1.63-01
WBFNAR	Beta Factor - Auxillary Building Ventilation Fans Fail To Run	3.21-02	2.11-02	3.14-02	4.10-02
WBFNAS	Beta Factor - Auxillary Building Ventilation Fans Fail To Start	4.95-02	2.22-02	4.70-02	7.38-02
WBMAPR	Beta Factor - Motor-Driven AFW Pumps Fail To Run	1.49-02	2.93-03	1.31-02	2.75-02
WBMAPS	Beta Factor - Motor-Driven AFW Pumps Fail To Start	1.18-01	6.12-02	1.14-01	1.67-01
WBMV2D	Beta Factor - MOV (Population 2) Fails To Operate on Demand	4.53-02	2.70-02	4.40-02	6.04-02
WBMV3D	Beta Factor - MOV (Population 3) Fails To Operate on Demand	5.29-02	3.62-02	5.19-02	6.62-02
WBMV4D	Beta Factor - MOV (Population 4) Fails To Operate on Demand	5.80-02	4.26-02	5.72-02	7.00-02
WBPWPR	Beta Factor - Primary Water Pumps Fail To Run	1.00-02	8.83-04	7.60-03	2.30-02
WBPWPS	Beta Factor - Primary Water Pumps Fail To Start	7.00-02	5.86-04	5.46-02	1.57-01
WBRHPR	Beta Factor - RHR Pumps Fail during Operation	1.49-02	2.93-03	1.31-02	2.75-02
WBRHPS	Beta Factor - RHR Pumps Fail To Start	1.19-01	6.15-02	1.14-01	1.67-01
WBRTBD	Beta Factor - Reactor Trip Breakers Fail To Open	8.39-02	1.58-02	7.41-02	1.55-01
WBSIPR	Beta Factor - Safety Injection Pumps Fail during Operation	1.49-02	2.93-03	1.31-02	2.75-02
WBSIPS	Beta Factor - Safety Injection Pumps Fail To Start	1.94-01	1.24-01	1.90-01	2.50-01
WBSTCD	Beta Factor - Shunt Trip Coils Fail To Actuate	7.00-02	5.86-04	5.46-02	1.57-01
WBTBPR	Beta Factor - Thermal Barrier Booster Pumps Fail To Run	1.22-02	1.90-04	7.55-03	3.32-02
WBTBPS	Beta Factor - Thermal Barrier Booster Pumps Fail To Start	6.59-03	5.65-08	9.77-04	2.59-02
WBUVCD	Beta Factor - Undervoltage Coils Fail To Actuate on Demand	1.58-01	6.14-02	1.49-01	2.45-01
WBVC2D	Beta Factor - Check Valve (Population 2) Fails To Reseat	1.24-02	8.92-04	1.02-02	2.59-02
WBVC4D	Beta Factor - ERCW Check Valves Fail To Reseat	3.06-02	9.63-03	2.81-02	5.08-02
WDCB1D	Delta Factor - Circuit Breakers > 480V Fail on Demand	5.95-01	4.12-01	5.91-01	7.19-01

Note: Exponential notation is indicated in abbreviated form; e.g., 2.01-01 = 2.01×10^{-01}.

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Table 3.3.4-10 (Page 3 of 5). Summary of Common Cause Failure Parameters for Watts Bar Components

Designator	Description	Mean	5th Percentile	Median	95th Percentile
WDCCPR	Delta Factor - Component Cooling Water Pumps Fail To Run	4.25-02	2.47-08	4.25-03	1.79-01
WDCCPS	Delta Factor - Component Cooling Water Pumps Fail To Start	5.92-01	1.69-01	5.91-01	8.81-01
WDDGGR	Delta Factor - Diesel Generator Fails during Operation	4.51-01	1.30-01	4.35-01	7.13-01
WDDGGS	Delta Factor - Diesel Generator Fails To Start	2.53-01	5.79-04	1.97-01	5.88-01
WDEWPR	Delta Factor - ERCW Pumps Fail during Operation	4.25-02	2.47-08	4.25-03	1.79-01
WDEWPS	Delta Factor - ERCW Pumps Fail To Start	5.92-01	1.69-01	5.91-01	8.81-01
WDF4OR	Delta Factor - Operating Fans (Population 4) Fail To Run	3.81-01	1.66-01	3.69-01	5.53-01
WDF4OS	Delta Factor - Operating Fans (Population 4) Fail To Start	1.78-01	1.32-02	1.49-01	3.69-01
WDF4SR	Delta Factor - Standby Fans (Population 4) Fail To Run	5.10-01	3.08-01	5.04-01	6.55-01
WDF4SS	Delta Factor - Standby Fans (Population 4) Fail To Start	2.84-01	7.88-02	2.65-01	4.72-01
WGCB1D	Gamma Factor - Circuit Breakers > 480V Fail on Demand	4.43-01	3.11-01	4.38-01	5.40-01
WGCCPR	Gamma Factor - Component Cooling Water Pumps Fail To Run	5.00-01	2.13-01	4.91-01	7.14-01
WGCCPS	Gamma Factor - Component Cooling Water Pumps Fail To Start	5.37-01	2.09-01	5.29-01	7.74-01
WGDGGR	Gamma Factor - Diesel Generator Fails during Operation	4.73-01	2.38-01	4.64-01	6.48-01
WGDGGS	Gamma Factor - Diesel Generator Fails To Start	4.99-01	1.28-01	4.85-01	7.91-01
WGEWPR	Gamma Factor - ERCW Pumps Fail during Operation	5.00-01	2.13-01	4.91-01	7.14-01
WGEWPS	Gamma Factor - ERCW Pumps Fail To Start	5.37-01	2.09-01	5.29-01	7.74-01
WGF4OR	Gamma Factor - Operating Fans (Population 4) Fail To Run	2.87-01	1.38-01	2.78-01	4.10-01
WGF4OS	Gamma Factor - Operating Fans (Population 4) Fail To Start	1.31-01	3.14-02	1.18-01	2.29-01
WGF4SR	Gamma Factor - Standby Fans (Population 4) Fail To Run	3.63-01	2.18-01	3.56-01	4.74-01
WGF4SS	Gamma Factor - Standby Fans (Population 4) Fail To Start	1.31-01	3.14-02	1.18-01	2.29-01
WGFNAR	Gamma Factor - Auxillary Building Ventilation Fans Fail To Run	5.66-01	4.21-01	5.62-01	6.65-01

Note: Exponential notation is indicated in abbreviated form; e.g., 2.01-01 = 2.01×10^{-01}.

Table 3.3.4-10 (Page 4 of 5). Summary of Common Cause Failure Parameters for Watts Bar Components					
Designator	**Description**	**Mean**	**5th Percentile**	**Median**	**95th Percentile**
WGFNAS	Gamma Factor - Auxillary Building Vent Fans Fail To Start	2.22-01	9.69-02	2.13-01	3.30-01
WGMV3D	Gamma Factor - MOV (Population 3) Fails To Operate on Demand	2.62-01	1.54-01	2.55-01	3.47-01
WGMV4D	Gamma Factor - MOV (Population 4) Fails To Operate on Demand	3.81-01	2.69-01	3.76-01	4.63-01
WGVC4D	Gamma Factor - ERCW Check Valves Fail To Reseat	1.22-01	2.91-02	1.11-01	2.15-01
WDFNAR	Delta Factor - Auxillary Building Ventilation Fans Fail To Run	7.11-01	5.49-01	7.10-01	8.14-01
WDFNAS	Delta Factor - Auxillary Building Ventilation Fans Fail To Start	4.11-01	1.70-01	3.98-01	6.03-01
WDMV4D	Delta Factor - MOV (Population 4) Fails To Operate on Demand	3.73-01	2.16-01	3.66-01	4.94-01
WDVC4D	Delta Factor - ERCW Check Valves Fail To Reseat	2.84-01	7.88-02	2.65-01	4.72-01
ZBCMPR	Beta Factor - Air Compressor Fails during Operation	2.07-02	3.58-05	8.40-03	6.82-02
ZBCMPS	Beta Factor - Air Compressor Fails To Start on Demand	4.78-02	3.71-05	1.74-02	1.63-01
ZBCRAD	Beta Factor - CRDs Fail To Insert	7.00-02	5.86-04	5.46-02	1.57-01
ZBDMAD	Beta Factor - Motor-/Air-Operated Dampers Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZBLC1D	Beta Factor - Logic Trip Modules Fail on Demand	1.00-03	1.07-05	5.94-04	2.79-03
ZBPXRR	Beta Factor - Fuel Oil Transfer Pumps Fail To Operate	1.00-02	8.85-04	7.62-03	2.31-02
ZBPXRS	Beta Factor - Fuel Oil Transfer Pumps Fail To Start	7.00-02	5.86-04	5.46-02	1.57-01
ZBRL1D	Beta Factor - Mechanical Relays Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZBSWBD	Beta Factor - Bistables, Switches Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZBVAOD	Beta Factor - Air-Operated Valves Fail To Open/Close	7.00-02	5.86-04	5.46-02	1.57-01
ZBVE1D	Beta Factor - Electrohydraulic Valves Fail To Open/Close	7.00-02	5.86-04	5.46-02	1.57-01
ZBVSOD	Beta Factor - Solenoid Valves Fail To Operate on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZDCMPR	Delta Factor - Air Compressor Fails during Operation	4.99-01	9.72-03	4.70-01	9.49-01
ZDCMPS	Delta Factor - Air Compressor Fails To Start	4.17-01	1.56-04	3.47-01	9.14-01
Note: Exponential notation is indicated in abbreviated form; e.g., 2.01-01 = 2.01 × 10^{-01}.					

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Table 3.3.4-10 (Page 5 of 5). Summary of Common Cause Failure Parameters for Watts Bar Components					
Designator	Description	Mean	5th Percentile	Median	95th Percentile
ZDCRAD	Delta Factor - CRDs Fail To Insert	2.84-01	7.88-02	2.65-01	4.72-01
ZDDMAD	Delta Factor - Motor-/Air-Operated Dampers Fail on Demand	1.78-01	1.32-02	1.49-01	3.69-01
ZDLC1D	Delta Factor - Logic Modules Fail on Demand	2.84-01	7.88-02	2.65-01	4.72-01
ZDPXRR	Delta Factor - Fuel Oil Transfer Pumps Fail To Operate	2.84-01	7.88-02	2.65-01	4.72-01
ZDPXRS	Delta Factor - Fuel Oil Transfer Pumps Fail To Start	2.84-01	7.88-02	2.65-01	4.72-01
ZDRL1D	Delta Factor - Mechanical Relays Fail on Demand	2.84-01	7.88-02	2.65-01	4.72-01
ZDSWBD	Delta Factor - Switches, Bistables Fail on Demand	2.84-01	7.88-02	2.65-01	4.72-01
ZDVAOD	Delta Factor - Air-Operated Valves Fail To Open/Close	1.78-01	1.32-02	1.49-01	3.69-01
ZDVE1D	Delta Factor - Electrohydraulic Valves Fail To Open/Close	1.78-01	1.32-02	1.49-01	3.69-01
ZDVSOD	Delta Factor - Solenoid Valves Fail on Demand	2.84-01	7.88-02	2.65-01	4.72-01
ZGCMPR	Gamma Factor - Air Compressor Fails during Operation	2.98-01	4.60-02	2.69-01	5.50-01
ZGCMPS	Gamma Factor - Air Compressor Fails To Start	2.61-01	3.70-05	1.08-01	8.30-01
ZGCRAD	Gamma Factor - CRDs Fail To Insert	2.52-01	7.77-02	2.36-01	4.11-01
ZGDMAD	Gamma Factor - Motor-/Air-Operated Dampers Fail To Open/Close	1.40-01	3.40-02	1.27-01	2.45-01
ZGLC1D	Gamma Factor - Logic Modules Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZGPXRR	Gamma Factor - Fuel Oil Transfer Pumps Fail To Operate	1.40-01	3.40-02	1.27-01	2.45-01
ZGPXRS	Gamma Factor - Fuel Oil Transfer Pumps Fail To Start	1.40-01	3.40-02	1.27-01	2.45-01
ZGRL1D	Gamma Factor - Mechanical Relays Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZGSWBD	Gamma Factor - Switches, Bistables Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01
ZGVAOD	Gamma Factor - Air-Operated Valves Fail To Open/Close	1.40-01	3.40-02	1.27-01	2.45-01
ZGVE1D	Gamma Factor - Electrohydraulic Valves Fail To Open/Close	1.40-01	3.40-02	1.27-01	2.45-01
ZGVSOD	Gamma Factor - Solenoid Valves Fail on Demand	7.00-02	5.86-04	5.46-02	1.57-01

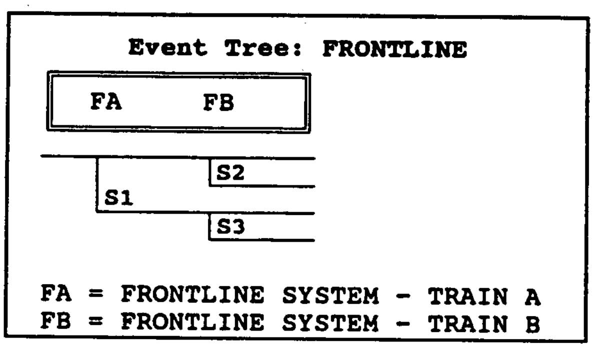
Note: Exponential notation is indicated in abbreviated form; e.g., 2.01-01 = 2.01 × 10^{-01}.

3.3.5 QUANTIFICATION OF UNAVAILABILITY OF SYSTEMS

The Watts Bar individual plant examination (IPE) system models are quantified using the RISKMAN® PRA Workstation Software and data in the Watts Bar database (Section 3.3.1). The Monte Carlo option is used and produces uncertainty distributions for the split fraction totals. In addition to histograms representing each uncertainty distribution, the main parameters of each distribution represented are the mean, 5th percentile, median, and 95th percentile. Table 3.3.5-1 displays the mean parameter for each split fraction. This mean value is used as input for the event tree quantification.

Table 3.3.5-1 also displays the top event associated with the split fraction, and a description of the conditions for which the split fraction is applicable. Often the description makes reference to the status of another top event that precedes the subject top event. This is indicative of a dependency between top events, such as common cause failures or shared components. Split fractions of this type are referred to as conditional split fractions. This method of calculation is used for any system that has been separated into individual trains for the IPE.

A simple example is used to illustrate the development of the equations for the conditional split fractions. Consider two top events in a frontline tree, FA and FB, that are defined such that they are dependent. This could apply to two trains of a system that share common cause and maintenance or to two systems sharing a group of components. Also assume that any necessary support required for Top Events FA and FB is available. The event tree in which these top events appear is shown below.



Let

P($\overline{FA}$) = the unavailability of train FA.

P($\overline{FB}$) = the unavailability of train FB.

P($\overline{FA}$ $\overline{FB}$) = the unavailability of train FA and train FB.

The objective is to define the split fractions S1, S2, and S3 in terms of the above probabilities. Note that S1 is not conditional and is simply P($\overline{FA}$). Also note that S2 and S3 are conditional split fractions and that S2 = P($\overline{FB}$ | FA) and S3 = P($\overline{FB}$ | $\overline{FA}$).

From basic probability theory,

$$P(\overline{FA}\ \overline{FB}) = P(\overline{FA}) * P(\overline{FB} \mid \overline{FA}) \quad => \quad S3 = P(\overline{FB} \mid \overline{FA}) = \frac{P(\overline{FA}\ \overline{FB})}{P(\overline{FA})}$$

The equation for S2 can be obtained as follows:

$$S2 = P(\overline{FB} \mid FA) = \frac{P(FA\ \overline{FB})}{P(FA)}$$

$$= \frac{1 - P(\overline{FA} + FB)}{P(FA)}$$

$$= \frac{1 - P(\overline{FA}) - P(FB) + P(\overline{FA}\ FB)}{P(FA)}$$

$$= \frac{1 - P(\overline{FA}) - P(FB) + P(\overline{FA}) * P(FB \mid \overline{FA})}{P(FA)}$$

$$= \frac{1 - P(\overline{FA}) - P(FB) + P(\overline{FA}) * (1 - P(\overline{FB} \mid \overline{FA})}{P(FA)}$$

$$= \frac{1 - P(\overline{FA}) - P(FB) + P(\overline{FA}) * (1 - P(\overline{FA}\ \overline{FB})/P(\overline{FA}))}{P(FA)}$$

$$= \frac{1 - P(\overline{FA}) - P(FB) + P(\overline{FA}) - P(\overline{FA}\ \overline{FB})}{P(FA)}$$

$$= \frac{1 - P(FB) - P(\overline{FA}\ \overline{FB})}{P(FA)}$$

$$= \frac{P(\overline{FB}) - P(\overline{FA}\ \overline{FB})}{1 - P(\overline{FA})}$$

In summary,

$$S1 = P(\overline{FA})$$

$$S2 = P(\overline{FB} \mid FA) = \frac{P(\overline{FB}) - P(\overline{FA}\ \overline{FB})}{1 - P(\overline{FA})}$$

$$S3 = P(\overline{FB} \mid \overline{FA}) = \frac{P(\overline{FA}\ \overline{FB})}{P(\overline{FA})}$$

The term $P(\overline{FA}\ \overline{FB})$ is calculated using an intermediate fault tree. For the case in which FA has failed due to a support failure, the split fraction used for FB is just $P(\overline{FB})$. This approach can be used to develop the conditional split fractions for any number of top events. For this particular example, a section of Table 3.3.5-1 might look as follows:

FA1	FA	1.0000E-02	REQUIRED SUPPORT AVAILABLE
FAF	FA	1.0000E-00	REQUIRED SUPPORT NOT AVAILABLE
FAB1	FAB	5.0000E-04	INTERMEDIATE TREE
FB1	FB	9.9596E-03	CONDITIONAL ON FA SUCCESS
FB2	FB	5.0000E-02	CONDITIONAL ON FA FAILURE
FB3	FB	1.0000E-02	FA NOT CHALLENGED
FBF	FB	1.0000E-00	REQUIRED SUPPORT NOT AVAILABLE

Here FA1 corresponds to S1, FB1 corresponds to S2, and FB2 correspond to S3.

A description of the methodology for the analysis of systems modeled in the Watts Bar IPE is provided in Section 2.3.5.

Table 3.3.5-1 (Page 1 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
A11	A1	8.3262E-04	All Support Available
A1F	A1	1.0000E+00	Guaranteed Failure
A1LF	A1L	1.0000E+00	ERCW/Diesel 1A/480V SD BD 1A1-A Dependency - Guaranteed Failure
A1LS	A1L	0.0000E+00	ERCW/Diesel 1A/480V SD BD 1A1-A Dependency - Guaranteed Success
A1U21	A1U2	6.9324E-04	All Support Available
A1U2F	A1U2	1.0000E+00	Guaranteed Failure
A1U2LF	A1U2L	1.0000E+00	ERCW/Diesel 2A/480V SD BD 2A1-A Dependency - Guaranteed Failure
A1U2LS	A1U2L	0.0000E+00	ERCW/Diesel 2A/480V SD BD 2A1-A Dependency - Guaranteed Success
A21	A2	6.0348E-04	All Support Available
A2F	A2	1.0000E+00	Guaranteed Failure
A2LF	A2L	1.0000E+00	ERCW/Diesel 1A/480V SD BD 1A2-A Dependency - Guaranteed Failure
A2LS	A2L	0.0000E+00	ERCW/Diesel 1A/480V SD BD 1A2-A Dependency - Guaranteed Success
A2U21	A2U2	5.4685E-04	All Support Available
A2U2F	A2U2	1.0000E+00	Guaranteed Failure
A2U2LF	A2U2L	1.0000E+00	ERCW/Diesel 2A/480V SD BD 2A2-A Dependency - Guaranteed Failure
A2U2LS	A2U2L	0.0000E+00	ERCW/Diesel 2A/480V SD BD 2A2-A Dependency - Guaranteed Success
A31	A3	4.3770E-04	All Support Available
A3F	A3	1.0000E+00	Guaranteed Failure
AA1	AA	2.5520E-05	All Support Available
AA2	AA	6.6400E-04	LOSP
AAF	AA	1.0000E+00	Guaranteed Failure
AALF	AAL	1.0000E+00	ERCW/Diesel 1A/6.9-KV SD BD 1A-A Dependency - Guaranteed Failure
AALS	AAL	0.0000E+00	ERCW/Diesel 1A/6.9-KV SD BD 1A-A Dependency - Guaranteed Success
AB1	AB	2.5520E-05	AA and AB Successful
AB2	AB	5.5650E-04	AA and AB Successful, LOSP
AB3	AB	2.4650E-05	AA or AB Fail
AB4	AB	5.5550E-02	AA or AB Fail, LOSP
AB5	AB	3.9940E-05	AA and AB Fail
AB6	AB	5.6700E-01	AA and AB Fail, LOSP
AB7	AB	5.8890E-04	One BUS BY Support, Other Success, LOSP
AB8	AB	1.1370E-01	One BUS BY Support, Other Fails, LOSP
AB9	AB	6.6400E-04	BOTH BY Support
ABF	AB	1.0000E+00	Guaranteed Failure
ABLF	ABL	1.0000E+00	ERCW/Diesel 2A/6.9-KV SD BD 2A-A Dependency - Guaranteed Failure
ABLS	ABL	0.0000E+00	ERCW/Diesel 2A/6.9-KV SD BD 2A-A Dependency -Guaranteed Success
AC1	AC	1.3290E-04	2 Pump Trains Available, All Support Available
AC10	AC	1.3440E-02	2 Pump Trains Available, BA and BB Fail, OG=F
AC2	AC	1.7350E-02	1 Pump Train Available, (A1 or B1) Fails, Other Supports Available
AC3	AC	1.9330E-02	1 Pump Train Available, (A1 or B1) Fail, OG=F, Other Support Available
AC4	AC	3.0350E-02	1 Pump Train Available, (A1 or B1) Fails, OG=S, (BA and BB) Fail
AC5	AC	3.3020E-02	1 Pump Train Available, (A1 or B1) Fails, OG=F, (BA and BB) Fail
AC8	AC	3.4030E-04	2 Pump Trains Available, OG=F
AC9	AC	1.2810E-02	2 Pump Trains Available, BA and BB Fail, OG=S

Table 3.3.5-1 (Page 2 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
ACF	AC	1.0000E+00	Guaranteed Failure
AE1	AE	3.0425E-05	All Support Available
AE10	AE	1.3533E-04	LOSP, Later Recovered, ALL Other Support AVAILABLE
AE2	AE	1.2314E-02	Loss Of AA or AB
AE3	AE	8.3513E-05	Loss Of DA or DC
AE4	AE	1.5823E-03	Loss Of DA and DC
AE5	AE	6.8632E-03	LOSP, ALL Other Support AVAILABLE
AE6	AE	3.4107E-03	LOSP, Loss Of AA or AB or DA or DC
AE7	AE	3.1892E-05	SI, All Support Available
AE8	AE	1.1472E-02	SI, Loss Of AA or AB
AE9	AE	8.4961E-05	SI, Loss Of DA or DC
AEF	AE	1.0000E+00	GF
AF1	AF	2.6831E-06	All Support Available
AF2	AF	3.0331E-06	Support For 1 SET Of TDP LCVS Unavailable
AF3	AF	2.8698E-06	One MDP NOT Available
AF4	AF	7.0914E-06	1 MDP and ASSOCIATED TDP LCVS NOT Available
AF5	AF	1.9616E-05	TDP or BOTH MDPS Unavailable
AF6	AF	2.2722E-05	1 MDP and TDP Unavailable
AF7	AF	1.5788E-01	SBO (and Loss Of AIR ASSUMED)
AFA1	AF	2.3390E-03	All Support Available (ATWS)
AFA2	AF	3.0570E-02	Support For 1 SET Of TDP LCVS Unavailable (ATWS)
AFA3	AF	3.0460E-02	One MDP NOT Available (ATWS)
AFA4	AF	1.0000E+00	1 MDP and ASSOCIATED TDP LCVS NOT Available (ATWS)
AFA5	AF	5.8530E-02	TDP or BOTH MDPS Unavailable (ATWS)
AFF	AF	1.0000E+00	Guaranteed Failure
AFR1	AF	5.1230E-03	MANUALLY CONTROL AFW
AFTWS1	AFATWS	2.3385E-03	All Support Available
AFTWS2	AFATWS	3.0559E-02	Support For 1 SET Of TDP LCVS Unavailable
AFTWS3	AFATWS	3.0457E-02	One MDP NOT Available
AFTWS4	AFATWS	9.9994E-01	1 MDP and ASSOCIATED TDP LCVS NOT Available
AFTWS5	AFATWS	5.8543E-02	TDP or BOTH MDPS Unavailable
AFTWSF	AFATWS	1.0000E+00	Guaranteed Failure
AM1	AM	1.0000E-03	All Support Available
AMF	AM	1.0000E+00	Guaranteed Failure
AR1	AR	3.1684E-04	ALL Support For THE AIR RETURN FANS Available
AR2	AR	5.6935E-03	Support For One Train Of AIR RETURN FAN Failed
ARF	AR	1.0000E+00	Guaranteed Failure Of Top Event AR
B11	B1	7.0702E-04	All Support Available
B1F	B1	1.0000E+00	Guaranteed Failure
B1LF	B1L	1.0000E+00	ERCW/Diesel 1B/480V SD BD 1B1-B Dependency - Guaranteed Failure
B1LS	B1L	0.0000E+00	ERCW/Diesel 1B/480V SD BD 1B1-B Dependency - Guaranteed Success
B1U21	B1U2	8.4458E-04	All Support Available
B1U2F	B1U2	1.0000E+00	Guaranteed Failure
B1U2LF	B1U2L	1.0000E+00	ERCW/Diesel 2B/480V SD BD 2B1-B Dependency - Guaranteed Failure

Table 3.3.5-1 (Page 3 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
B1U2LS	B1U2L	0.0000E+00	ERCW/Diesel 2B/480V SD BD 2B1-B Dependency - Guaranteed Success
B21	B2	5.9562E-04	All Support Available
B2F	B2	1.0000E+00	Guaranteed Failure
B2LF	B2L	1.0000E+00	ERCW/Diesel 1B/480V SD BD 1B2-B Dependency - Guaranteed Failure
B2LS	B2L	0.0000E+00	ERCW/Diesel 1B/480V SD BD 1B2-B Dependency - Guaranteed Success
B2U21	B2U2	5.7141E-04	All Support Available
B2U2F	B2U2	1.0000E+00	Guaranteed Failure
B2U2LF	B2U2L	.0000E+00	ERCW/Diesel 2B/480V SD BD 2B2-B Dependency - Guaranteed Failure
B2U2LS	B2U2L	0.0000E+00	ERCW/Diesel 2B/480V SD BD 2B2-B Dependency - Guaranteed Success
B31	B3	4.3780E-04	A3 Successful
B32	B3	2.7450E-04	A3 Fails
B3F	B3	1.0000E+00	Guaranteed Failure
BA1	BA	2.5520E-05	AA Successful
BA2	BA	2.4650E-05	AA Fails
BA3	BA	5.8890E-04	AA Successful, LOSP
BA4	BA	1.1370E-01	AA Fails, LOSP
BA5	BA	6.6400E-04	AA Fails BY Support, LOSP
BAF	BA	1.0000E+00	Guaranteed Failure
BALF	BAL	1.0000E+00	ERCW/Diesel 1B/6.9KV SD BD 1B-B Dependency - Guaranteed Failure
BALS	BAL	0.0000E+00	ERCW/Diesel 1B/6.9KV SD BD 1B-B Dependency -Guaranteed Success
BB1	BB	2.5520E-05	AA, AB and BA Successful
BB10	BB	6.6400E-04	Three Previous Trains Fail BY Support
BB11	BB	5.5550E-02	One Previous Fails BY Support, One INDEPENDANT, LOSP
BB12	BB	1.1370E-01	Two Previous Trains Fail BY Support, One INDEPENDANT, LOSP
BB13	BB	5.6700E-01	One Previous Train Fails BY Support, Two INDEPENDANT, LOSP
BB14	BB	7.9320E-01	Three Previous Trains Fail, LOSP
BB2	BB	2.4650E-05	AA or AB or BA Fails
BB3	BB	3.9940E-05	Two Previous BUSES Fail
BB4	BB	4.6750E-05	AA, AB, and BA Fail
BB5	BB	5.3290E-04	AA, AB and BA Successful, LOSP
BB6	BB	4.2890E-02	AA or AB or BA Fails, LOSP
BB7	BB	2.7070E-01	Two Previous BUSES Fail, LOSP
BB8	BB	5.5650E-04	One Previous Train Fails BY Support
BB9	BB	5.8890E-04	Two Previous Train BY Support, LOSP
BBF	BB	1.0000E+00	Guaranteed Failure
BBLF	BBL	1.0000E+00	ERCW/Diesel 1B/6.9-KV SD BD 2B-B Dependency - Guaranteed Failure
BBLS	BBL	0.0000E+00	ERCW/Diesel 1B/6.9-KV SD BD 2B-B Dependency Guaranteed Success
BC1	BC	3.6050E-03	Train A Available with 2 Pump Trains, ALL Supports Available
BC10	BC	5.9370E-03	Train A Available with 2 Pump Trains, BA and BB Fail, OG=F
BC2	BC	3.6000E-03	Train A Available with 1 Pump Train (A1 or B1) Fails, Other Supports Available
BC21	BC	1.7240E-01	Train A Unavailable with 2 Pump Trains, ALL Supports Available
BC22	BC	5.1780E-03	Train A Unavailable with 1 Pump Train (A1 or B1) Fails, Other Supports Available
BC23	BC	1.3210E-02	Train A Unavailable with 1 Pump Train (A1 or B1) Fails, OG=F, Other

Table 3.3.5-1 (Page 4 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
			Supports Available
BC24	BC	4.6660E-03	Train A Unavailable with 1 Pump Train (A1 or B1) Fails, OG=S, BA and BB Fail
BC25	BC	1.0070E-02	Train A Unavailable with 1 Pump Train (A1 or B1) Fails, OG=F, BA and BB Fail
BC28	BC	3.1270E-01	Train A Unavailable with 2 Pump Trains, OG=F
BC29	BC	5.1110E-03	Train A Unavailable with 2 Pump Trains Available, OG=S, BA and BB Fails
BC3	BC	5.9050E-03	Train A Available with 1 Pump Train (A1 or B1) Fails, OG=F, Other Supports Available
BC30	BC	1.4030E-02	Train A Unavailable with 2 Pump Trains, BA and BB Fail, OG=F
BC33	BC	3.6280E-03	Train A Guaranteed Failure, OG=S
BC34	BC	6.0460E-03	Train A Guaranteed Failure, OG=F
BC4	BC	3.5950E-03	Train A Available with 1 Pump Train (A1 or B1) Fails, OG=S, BA and BB Fail
BC5	BC	5.9090E-03	Train A Available with 1 Pump Train (A1 or B1) Fails, OG=F, BA and BB Fail
BC8	BC	5.9420E-03	Train A Available with 2 Pump Trains, OG=F
BC9	BC	3.6080E-03	Train A Available with 2 Pump Trains Available, OG=S, BA and BB Fails
BCF	BC	1.0000E+00	GF
BE1	BE	2.9380E-05	All Support Available, AE=S
BE10	BE	3.0510E-04	Loss Of AA or AB, AE=F
BE11	BE	1.0680E-02	Loss Of (AA or AB) and (BA or BB), AE=S
BE12	BE	1.4300E-01	Loss Of (AA or AB) and (BA or BB), AE=F
BE13	BE	8.0040E-05	Loss Of (AA or AB) and (DB or DD), AE=S
BE14	BE	3.6250E-04	Loss Of (AA or AB) and (DB or DD), AE=F
BE15	BE	1.5200E-03	Loss Of (AA or AB) and DB and DD), AE=S
BE16	BE	6.5890E-03	Loss Of (AA or AB) and DB and DD), AE=F
BE17	BE	2.9350E-05	Loss Of DA or DC, AE=S
BE18	BE	1.2850E-02	Loss Of DA or DC, AE=F
BE19	BE	1.2310E-02	Loss Of (DA or DC) and (BA or BB), AE=S
BE2	BE	3.4430E-02	All Support Available, AE=F
BE20	BE	5.3440E-02	Loss Of (DA or DC) and (BA or BB), AE=F
BE21	BE	8.2440E-05	Loss Of (DA or DC) and (DB or DD), AE=S
BE22	BE	1.2970E-02	Loss Of (DA or DC) and (DB or DD), AE=F
BE23	BE	1.5810E-03	Loss Of (DA or DC) and DB and DD, AE=S
BE24	BE	1.5420E-02	Loss Of (DA or DC) and DB and DD, AE=F
BE25	BE	2.9380E-05	Loss Of DA and DC, AE=S
BE26	BE	6.8700E-04	Loss Of DA and DC, AE=F
BE27	BE	1.2250E-02	Loss Of DA and DC and (BA or BB), AE=S
BE28	BE	5.1270E-02	Loss Of DA and DC and (BA or BB), AE=F
BE29	BE	8.2360E-05	Loss Of DA and DC and (DB or DD), AE=S
BE3	BE	1.2310E-02	Loss Of BA or BB, AE=S
BE30	BE	8.1390E-04	Loss Of DA and DC and (DB or DD), AE=F
BE31	BE	1.5230E-03	Loss Of DA and DC and DB and DD, AE=S
BE32	BE	3.8960E-02	Loss Of DA and DC and DB and DD, AE=F
BE33	BE	6.8630E-03	LOSP, ALL Other Support Available, AE=S

Table 3.3.5-1 (Page 5 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
BE34	BE	7.0120E-03	LOSP, ALL Other Support Available, AE=F
BE35	BE	3.2830E-03	LOSP, Loss Of BA or BB or DB or DD, AE=S
BE36	BE	2.1930E-02	LOSP, Loss Of BA or BB or DB or DD, AE=F
BE4	BE	1.2350E-01	Loss Of BA or BB, AE=F
BE41	BE	6.7360E-03	LOSP, Loss Of AA or AB or DA or DC, AE=S
BE42	BE	4.4130E-02	LOSP, Loss Of AA or AB or DA or DC, AE=F
BE43	BE	3.3430E-03	LOSP, Loss Of (AA or AB or DA or DC) and (BA or BB or DB or DD), AE=S
BE44	BE	2.3230E-02	LOSP, Loss Of (AA or AB or DA or DC) and (BA or BB or DB or DD), AE=F
BE49	BE	3.0870E-05	SI, All Support Available, AE=S
BE5	BE	8.2450E-05	Loss Of DB or DD, AE=S
BE50	BE	3.2090E-02	SI, All Support Available, AE=F
BE51	BE	1.1470E-02	SI, Loss Of BA or BB, AE=S
BE52	BE	4.4240E-02	SI, Loss Of BA or BB, AE=F
BE53	BE	8.3940E-05	SI, Loss Of DB or DD, AE=S
BE54	BE	3.2090E-02	SI, Loss Of DB or DD, AE=F
BE55	BE	1.5810E-03	SI, Loss Of DB and DD, AE=S
BE56	BE	3.0660E-02	SI, Loss Of DB and DD, AE=F
BE57	BE	3.0840E-05	SI, Loss Of AA or AB, AE=S
BE58	BE	1.2300E-04	SI, Loss Of AA or AB, AE=F
BE59	BE	1.1360E-02	SI, Loss Of (AA or AB) and (BA or BB), AE=S
BE6	BE	3.5280E-02	Loss Of DB or DD, AE=F
BE60	BE	2.1150E-02	SI, Loss Of (AA or AB) and (BA or BB), AE=F
BE61	BE	8.3950E-05	SI, Loss Of (AA or AB) and (DB or DD), AE=S
BE62	BE	1.7200E-04	SI, Loss Of (AA or AB) and (DB or DD), AE=F
BE63	BE	1.5220E-03	SI, Loss Of (AA or AB) and DB and DD, AE=S
BE64	BE	6.7480E-03	SI, Loss Of (AA or AB) and DB and DD, AE=F
BE65	BE	3.0870E-05	SI ,Loss Of DA or DC, AE=S
BE66	BE	1.2050E-02	SI ,Loss Of DA or DC, AE=F
BE67	BE	1.1470E-02	SI, Loss Of (DA or DC) and (BA or BB), AE=S
BE68	BE	2.3230E-02	SI, Loss Of (DA or DC) and (BA or BB), AE=F
BE69	BE	8.4130E-05	SI, Loss Of (DA or DC) and (DB or DD), AE=S
BE7	BE	1.5810E-03	Loss Of DB and DD, AE=S
BE70	BE	9.8190E-03	SI, Loss Of (DA or DC) and (DB or DD), AE=F
BE71	BE	1.5810E-03	SI, Loss Of (DA or DC) and DB and DD, AE=S
BE72	BE	1.4830E-02	SI, Loss Of (DA or DC) and DB and DD, AE=F
BE73	BE	3.0960E-05	SI ,Loss Of DA and DC, AE=S
BE74	BE	6.1800E-04	SI ,Loss Of DA and DC, AE=F
BE75	BE	1.1410E-02	SI, Loss Of DA and DC and (BA or BB), AE=S
BE76	BE	4.8930E-02	SI, Loss Of DA and DC and (BA or BB), AE=F
BE77	BE	8.3830E-05	SI, Loss Of DA and DC and (DB or DD), AE=S
BE78	BE	7.9610E-04	SI, Loss Of DA and DC and (DB or DD), AE=F
BE79	BE	1.5230E-03	SI, Loss Of DA and DC and DB and DD, AE=S
BE8	BE	3.5730E-02	Loss Of DB and DD, AE=F
BE80	BE	3.8960E-02	SI, Loss Of DA and DC and DB and DD, AE=F

Table 3.3.5-1 (Page 6 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
BE81	BE	3.0420E-05	Loss Of AA and AB
BE82	BE	1.2310E-02	Loss Of AA and AB and (BA or BB)
BE83	BE	8.3520E-05	Loss Of AA and AB and (DB or DD)
BE84	BE	1.5820E-03	Loss Of AA and AB and DB and DD
BE85	BE	6.8640E-03	LOSP, Loss Of AA and AB
BE86	BE	3.4110E-03	LOSP, Loss Of AA and AB and (BA or BB or DB or DD)
BE87	BE	3.1890E-05	SI, Loss Of AA and AB
BE88	BE	1.1470E-02	SI, Loss Of AA and AB and (BA or BB)
BE89	BE	8.4960E-05	SI, Loss Of AA and AB and (DB or DD)
BE9	BE	2.7000E-05	Loss Of AA or AB, AE=S
BE90	BE	1.3440E-04	LOSP, Later Recovered, AE=S
BE91	BE	7.2450E-03	LOSP, Later Recovered, AE=F
BEF	BE	1.0000E+00	GF
BNAF	BNA	1.0000E+00	Guaranteed Failure
BNAS	BNA	0.0000E+00	Guaranteed Success
BNBF	BNB	1.0000E+00	Guaranteed Failure
BNBS	BNB	0.0000E+00	Guaranteed Success
BNCF	BNC	1.0000E+00	Guaranteed Failure
BNCS	BNC	0.0000E+00	Guaranteed Success
BNNF	BNN	1.0000E+00	Guaranteed Failure
BNNS	BNN	0.0000E+00	Guaranteed Success
BUS1	KVIV	2.5524E-05	Three BUSES Unavailable
BUS2	KVIV	6.2896E-10	Two BUSES Unavailable
BUS3	KVIV	2.5126E-14	One BUS Unavailable
BUS4	KVIV	1.1745E-18	System Unavailable
BUSA	KVIV	6.6403E-04	LOSP, Three BUSES Unavailable
BUSB	KVIV	7.5495E-05	LOSP, Two BUSES Unavailable
BUSC	KVIV	4.2809E-05	LOSP, One BUS Unavailable
BUSD	KVIV	3.3951E-05	LOSP, System
BYAF	BYA	1.0000E+00	Guaranteed Failure
BYAS	BYA	0.0000E+00	Guaranteed Success
BYBF	BYB	1.0000E+00	Guaranteed Failure
BYBS	BYB	0.0000E+00	Guaranteed Success
BYCF	BYC	1.0000E+00	Guaranteed Failure
BYCS	BYC	0.0000E+00	Guaranteed Success
BYNF	BYN	1.0000E+00	Guaranteed Failure
BYNS	BYN	0.0000E+00	Guaranteed Success
C1	AEBES	1.0410E-06	All Support Available
C10	AEBES	9.3703E-07	Loss Of DB and DD
C11	AEBES	9.3661E-07	Loss Of AA and DB
C12	AEBES	9.9649E-07	Loss Of AA and DB and DD
C13	AEBES	9.9512E-07	Loss Of DA and DB and DD
C14	AEBES	9.9344E-07	Loss Of DA and DC and DB and DD
C15	AEBES	1.2639E-05	LOSP, ALL Other Support Available

Table 3.3.5-1 (Page 7 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
C16	AEBES	1.2349E-05	LOSP, Loss Of AA
C17	AEBES	1.6108E-05	LOSP, Loss Of AA and BA
C18	AEBES	1.3255E-05	LOSP, Loss Of AA and AB
C19	AEBES	1.9161E-05	LOSP, Loss Of AA and AB and BA
C2	AEBES	9.5040E-07	Loss Of AA
C20	AEBES	8.9863E-07	SI, All Support Available
C21	AEBES	1.0830E-06	SI, Loss Of AA
C22	AEBES	8.8283E-07	SI, Loss Of DA
C23	AEBES	9.4242E-07	SI, Loss Of DA and DC
C24	AEBES	4.1735E-06	SI, Loss Of AA and BA
C25	AEBES	8.3780E-07	SI, Loss Of AA and DB
C26	AEBES	9.4551E-07	SI, Loss Of AA and DB and DD
C27	AEBES	8.8673E-07	SI, Loss Of DA and DB
C28	AEBES	9.5503E-07	SI, Loss Of DA and DB and DD
C29	AEBES	9.3913E-07	SI, Loss Of DA and DC and DB and DD
C3	AEBES	1.0420E-06	Loss Of DA
C30	AEBES	1.1324E-06	SI, Loss Of AA and AB
C31	AEBES	7.2385E-06	SI, Loss Of AA and AB and BA
C32	AEBES	9.4843E-07	SI, Loss Of AA and AB and DB
C33	AEBES	8.6423E-07	SI, Loss Of AA and AB and DB and DD
C4	AEBES	4.0620E-06	Loss Of AA and BA
C5	AEBES	9.6853E-07	Loss Of DA and DB
C6	AEBES	9.3389E-07	Loss Of AA and AB
C7	AEBES	7.0887E-06	Loss Of AA and AB and BA
C8	AEBES	9.9821E-07	Loss Of AA and AB and DB
C9	AEBES	1.0260E-06	Loss Of AA and AB and DB and DD
CAVF	CAV	1.0000E+00	WATER IN RX CAVITY - Guaranteed Failure
CAVS	CAV	0.0000E+00	WATER IN RX CAVITY - Guaranteed Success
CCPR1	CCPR	1.6100E-02	Recovery Of CCP A BY ALIGNING ERCW HEADER 1A (CE)
CCPRF	CCPR	1.0000E+00	Recovery Of CCP A Failed
CCPRS	CCPR	0.0000E+00	Recovery Of COOLING TO CCP A NOT Required
CCSR1	CCSR	2.0490E-01	ALIGN THE C-S Pump TO THE A CCS HX
CCSR2	CCSR	3.3830E-02	ALIGN and INITIATE ALTERNATE COOLING TO THE CHARGING Pump
CCSR3	CCSR	1.4450E-02	ALIGN ERCW HEADER 2A TO CCS HTX A
CCSRE	CCSR	5.5220E-02	SWAP TO ALTERNATE POWER SUPPLY Given Loss Of NorMAL B2U2
CCSRF	CCSR	1.0000E+00	GUARANTEED FailRUE
CCSRS	CCSR	0.0000E+00	Guaranteed Success
CD1	CD	8.0366E-03	All Support Available
CDBF	CDB	1.0000E+00	CONTAINMENT Bypass Failed
CDBS	CDB	0.0000E+00	CONTAINMENT Bypass SUCCEED
CDF	CD	1.0000E+00	Guaranteed Failure
CE1	CE	1.7716E-03	All Support Available
CE2	CE	1.6423E-03	Loss Of BE
CEF	CE	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 8 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
CF	AEBES	1.0000E+00	Guaranteed Failure
CH1	CH	2.2297E-03	All Support Available, RR=S
CH2	CH	2.3058E-02	Train A Support Failed, RR=S
CH3	CH	2.3213E-02	Train B Support Failed, RR=S
CH4	CH	9.3375E-03	All Support Available, RR=F
CH5	CH	2.9909E-02	Train A Support Failed, RR=F
CH6	CH	3.0232E-02	Train B Support Failed, RR=F
CHF	CH	1.0000E+00	Guaranteed Failure
CI1	CI	4.9642E-03	ALL Required Support Available
CI2	CI	2.3442E-02	Failure Of SSPS Train A or B (I.E. Top Event ZA or ZB) and Top Event OS
CI3	CI	9.0559E-03	Failure Of Train A (or B) 480V Shutdown Board (I.E. Top Event A1 or B2)
CI4	CI	1.1445E-01	STATION BLACKOUT with Top Event OS or ZA and ZB Successful
CI5	CI	1.2745E-01	STATION BLACKOUT with Top Event OS and ZA or ZB Failed
CIF	CI	1.0000E+00	Guaranteed Failure Of Top Event
CL1	CL	9.0020E-06	Given Medium LOCA with IP Succeeded and All Support Available
CL2	CL	9.6670E-06	Given Medium LOCA with IP Failed Due To SIS Pump Unavailable
CL3	CL	7.4480E-04	Given Medium LOCA with IP Failed
CLF	CL	1.0000E+00	Guaranteed Failure
CLRFX1	ICRYY	6.2249E-07	Given Medium LOCA, IP Failed, and All Support Available
CLRFX2	ICRYY	1.9690E-06	Given Medium LOCA with IP and RB Failed
CLRFX3	ICRYY	4.5161E-06	Given Large LOCA, IP Failed, and All Support Available
CLRFX4	ICRYY	5.4005E-04	Given Large LOCA with IP and RB Failed
CLX1	ICRYY	9.6683E-06	Given Medium LOCA with IP and RF Failed Due To Support Unavailable
CMF	CM	1.0000E+00	Guaranteed Failure
CMS	CM	0.0000E+00	Guaranteed Success
COMA	COMMON	1.2014E-07	480V Common System Unavailable
COMB	COMMON	4.3770E-04	480V Common Single Train
CP1	CP	1.5112E-03	ALL Support For CVI Available
CP2	CP	1.8292E-03	Failure Of One Train Of SSPS and Top Event OS
CP3	CP	1.1620E-01	Failure Of ESFAS A and B, and OS
CPF	CP	1.0000E+00	Guaranteed Failure Of Top Event CP
CSA1	CSA	2.0878E-02	ALL Support
CSAB1	CSAB	1.3609E-03	ALL Support
CSAF	CSA	1.0000E+00	Guaranteed Failure
CSB1	CSB	1.9930E-02	CSA Success, Support Available For CSB
CSB2	CSB	6.5180E-02	CSA Failed, Support Available For CSB
CSB3	CSB	2.0880E-02	Support Unavailable For CSA, Support Available For CSB
CSBF	CSB	1.0000E+00	Guaranteed Failure
CSIF	CSI	1.0000E+00	CNMT. SPRAY INJ. - Guaranteed Failure
CSIS	CSI	0.0000E+00	CNMT. SPRAY INJ. - Guaranteed Success
CSRF	CSR	1.0000E+00	CNMT. SPRAY RECIRC. - Guaranteed Failure
CSRS	CSR	0.0000E+00	CNMT. SPRAY RECIRC. - Guaranteed Success
CT1	CT	9.5251E-05	All Support Available
CTF	CT	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 9 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
CTMU1	CTMU	2.5452E-02	ALL Support
CTMUF	CTMU	1.0000E+00	Guaranteed Failure
D11	D1	3.1798E-03	All Support Available
D12	D1	2.9367E-03	Loss Of AC TO CHARGER
D1F	D1	1.0000E+00	Guaranteed Failure
D21	D2	3.1383E-03	All Support Available
D22	D2	2.6488E-03	Loss Of AC TO CHARGER
D2F	D2	1.0000E+00	Guaranteed Failure
DA1	DA	1.0250E-03	All Support Available
DA2	DA	5.8037E-04	Loss Of 480V TOP A1
DAAC1	DAAC	6.9142E-04	All Support Available
DAAC2	DAAC	7.0144E-04	Loss Of DA
DAAC3	DAAC	7.2218E-04	Loss Of A1
DAACF	DAAC	1.0000E+00	Guaranteed Failure
DAF	DA	1.0000E+00	Guaranteed Failure
DB1	DB	9.9437E-04	All Support Available
DB2	DB	5.9108E-04	Loss Of NorMAL POWER
DBAC1	DBAC	6.9267E-04	All Support Available
DBAC2	DBAC	6.9588E-04	Loss Of DB
DBAC3	DBAC	7.3040E-04	Loss Of B1
DBACF	DBAC	1.0000E+00	Guaranteed Failure
DBF	DB	1.0000E+00	Guaranteed Failure
DC1	DC	1.0232E-03	All Support Available
DC2	DC	5.5948E-04	Loss Of 480V TOP A1U2
DCAC1	DCAC	6.9950E-04	All Support Available
DCAC2	DCAC	6.9063E-04	Loss Of DC
DCAC3	DCAC	7.2772E-04	Loss Of A1U2
DCACF	DCAC	1.0000E+00	Guaranteed Failure
DCF	DC	1.0000E+00	Guaranteed Failure
DD1	DD	1.0079E-03	All Support Available
DD2	DD	5.8880E-04	Loss Of NorMAL POWER SUPPLY
DDAC1	DDAC	7.0382E-04	All Support Available
DDAC2	DDAC	6.9688E-04	Loss Of DD
DDAC3	DDAC	7.2193E-04	Loss Of B1U2
DDACF	DDAC	1.0000E+00	Guaranteed Failure
DDF	DD	1.0000E+00	Guaranteed Failure
DE1	DE	1.7423E-04	All Support Available
DE2	DE	1.7784E-04	Loss Of AE
DEF	DE	1.0000E+00	Guaranteed Failure
DG1	DG	3.4414E-04	All Support Available
DG2	GAIV	2.5295E-02	Two Trains Unavailable
DG3	GAIV	6.3665E-03	Three Trains Unavailable
DGA	GAIV	1.3622E-01	Single Train Unavailable
DGF	DG	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 10 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
DGSYS	GAIV	2.4867E-03	System Unavailable
DHF	DH	1.0000E+00	Guaranteed Failure
DHS	DH	0.0000E+00	GUARANTEED SUCCCESS
DP1	DP	3.1384E-06	All Support Available
DP2	DP	1.4122E-05	One PorV Unavailable Due To Support
DP3	DP	1.5416E-04	BOTH PorVS ARE DISABLED Due To Support
DP4	DP	2.5659E-05	One SPRAY VALVE Support Failed
DP5	DP	1.4557E-04	One SPRAY VALVE and One PorV UnAVAIL Due To Support
DP6	DP	1.5814E-03	BOTH PorVS and One SPRAY VALVE DISABLED Due To Support Failure
DP7	DP	1.5120E-02	BOTH SPRAY VALVES DISABLED Due To Support Failure
DP8	DP	9.1117E-02	BOTH SPRAY VALVES and One PorV DISABLED Due To Support Failure
DPF	DP	1.0000E+00	Guaranteed Failure
DS1	DS	8.1532E-03	ALL Support, COOLDOWN & DEPRESS RCS, NorMAL COOLDOWN
DS2	DS	7.3022E-03	ALL Support, COOLDOWN & DEPRESS RCS, SGTR with Isolation
DS3	DS	2.0877E-02	ALL Support, COOLDOWN & DEPRESS RCS, SGTR with Isolation W/O HIGH HEAD SI
DS4	DS	2.7574E-02	ALL Support, COOLDOWN & DEPRESS RCS, with SGTR, S/G NOT ISOLATED
DS5	DS	6.8065E-02	ALL Support, COOLDOWN & DEPRESS RCS, with SGTR, S/G NOT ISOLATED, W/O HIGH HEAD SI
DS6	DS	1.0343E-01	ALL Support, COOLDOWN & DEPRESS RCS, Loss Of ALL AC
DS7	DS	4.0545E-02	ALL Support, Small LOCA
DS8	DS	1.8170E-03	ALL Support, SGTR with Successful Isolation and INJECTION. STEAM DUMPS Available For COOLDOWN
DS9	DS	1.4270E-02	ALL Support, SGTR with Successful Isolation BUT Failed INJECTION. STEAM DUMPS Available For COOLDOWN
DSF	DS	1.0000E+00	Guaranteed Failure
DSLR1	DSLR	3.0160E-02	Recovery Of ERCW TO Diesel FROM OPPOSITE SIDE
DSLRF	DSLR	1.0000E+00	Recovery Of ERCW TO Diesel FROM OPPOSITE SIDE - Guaranteed Failure
DSLRS	DSLR	0.0000E+00	Recovery Of ERCW TO Diesel FROM OPPOSITE SIDE - Guaranteed Success
DSNN	DS	1.0000E+00	A COOLDOWN NOT NEEDED
EB1	EB	4.2076E-02	Given All Support Available, IE IS NOT A LOSP
EB2	EB	4.3911E-02	Given LOSP with All Support Available
EB3	EB	3.9589E-02	Given OffSITE POWER Available with DAAC (or DBAC) or A1U2 (or B1U2) Failed
EB4	EB	4.4075E-02	Given LOSP with DAAC (or DBAC) or A1U2 (or B1U2) Failed
EB5	EB	4.0834E-02	Given OffSITE POWER Available with DAAC (or DBAC) and A1U2 (or B1U2) Failed
EB6	EB	4.2952E-02	Given LOSP with DAAC (or DBAC) and A1U2 (or B1U2) Failed
EB7	EB	3.5000E-02	Given OffSITE POWER Available with Loss Of Support TO HARDWARE For EB
EBF	EB	1.0000E+00	Guaranteed Failure
EE1	EE	1.6740E-03	All Support Available
EE2	EE	1.7431E-03	Loss Of BE
EEF	EE	1.0000E+00	Guaranteed Failure
EXF	EX	1.0000E+00	Guaranteed Failure
EXS	EX	0.0000E+00	Guaranteed Success
FA1	FA	4.5300E-03	All Support Available
FACC	FACC	8.3388E-05	Common CAUSE GLOBAL TERM SWITCH and Pump Failure

Table 3.3.5-1 (Page 11 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
FAI	FAIV	4.5300E-03	Single Train
FAII	FAIV	1.0320E-04	Two Trains
FAIII	FAIV	8.3480E-05	Three Trains
FAIND	FAIV	4.4471E-03	Single Train NO GLOBAL Common CAUSE TERM
FAIV	FAIV	8.3390E-05	FOUR Trains Available
FAS	FA	0.0000E+00	Guaranteed Success
FB1	FB	4.4470E-03	FA Successful
FB2	FB	2.2770E-02	FA Fails
FBS	FB	0.0000E+00	FUEL OIL NOT NECESSARY
FC1	FC	4.4470E-03	FA and FB Successful
FC2	FC	4.4460E-03	FA or FB Fails
FC3	FC	8.0920E-01	FA and FB Fails
FCS	FC	0.0000E+00	FUEL OIL NOT Required
FD1	FD	4.4470E-03	FA,FB,FC Successful
FD2	FD	4.4460E-03	One Previous Trains Fail
FD3	FD	4.4470E-03	Two Previous Trains Fail
FD4	FD	9.9900E-01	Three Previous Trains Fail
FDS	FD	0.0000E+00	FUEL OIL NOT Required
FE1	FE	1.6812E-03	All Support Available
FE2	FE	1.6751E-03	Loss Of BE
FEF	FE	1.0000E+00	Guaranteed Failure
FW1	FW	4.9150E-03	All Support Available
FWF	FW	1.0000E+00	Guaranteed Failure
GA1	GA	1.3620E-01	All Support Available Diesel 1AA
GAF	GA	1.0000E+00	Guaranteed Failure
GAS	GA	0.0000E+00	Diesel NOT Required
GB1	GB	1.2840E-01	GA Successful
GB2	GB	1.8570E-01	GA Fails
GB3	GB	1.3620E-01	GA Fails BY Support
GBF	GB	1.0000E+00	Guaranteed Failure
GBS	GB	0.0000E+00	NOT Required
GC1	GC	1.2220E-01	GA and GB Successful
GC2	GC	1.7060E-01	GA or GB Fails
GC3	GC	2.5170E-01	GA and GB Fail
GC4	GC	1.2840E-01	GA or GB BY Support, Other Train Success
GC5	GC	1.8570E-01	GA or GB BY Support, Other Train Fails
GC6	GC	1.3620E-01	GA and GB BY Support
GCF	GC	1.0000E+00	Guaranteed Failure
GCS	GC	0.0000E+00	Diesel NOT Required
GD1	GD	1.1640E-01	GA, GB, GC Successful
GD10	GD	1.3620E-01	Three Trains Fail BY Support
GD2	GD	1.6360E-01	GA or GB or GC Fails
GD3	GD	2.0500E-01	Two Of GA, GB, or GC Fail
GD4	GD	3.9060E-01	GA, GB, and GC Fail

Table 3.3.5-1 (Page 12 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
GD5	GD	1.2220E-01	Single Train BY Support ALL Others Success
GD6	GD	1.7060E-01	Single Train BY Support, One Train Fails
GD7	GD	2.5170E-01	Single Train BY Support, Other Trains Fail
GD8	GD	1.2840E-01	Two Trains BY Support Other Success
GD9	GD	1.8570E-01	Two Trains BY Support Other Train Fails
GDF	GD	1.0000E+00	GUARANTEED Failed
GDS	GD	0.0000E+00	Diesel NOT Required
GE1	GE	0.0000E+00	NO Supports Required
GEF	GE	1.0000E+00	Guaranteed Failure
HE1	HE	0.0000E+00	NO Supports Required
HEF	HE	1.0000E+00	Guaranteed Failure
HH1	HH	5.0005E-03	ALL Required Support Available
HH2	HH	5.2706E-03	Support For One Train Of Ignitors Failed
HHF	HH	1.0000E+00	Guaranteed Failure Of Top Event HH
HIPRF	HIPR	1.0000E+00	Guaranteed Failure
HIPRS	HIPR	0.0000E+00	Guaranteed Success
HPIF	HPI	1.0000E+00	Guaranteed Failure
HPIS	HPI	0.0000E+00	Guaranteed Success
HPLF	HPL	1.0000E+00	Guaranteed Failure
HPLS	HPL	0.0000E+00	Guaranteed Success
IC1	IC	9.9040E-07	ICE CONDENSER Unavailable
ICRX1	ICRYY	5.7035E-07	Given Medium LOCA with All Support Available
ICRX2	ICRYY	5.7063E-07	Given Medium LOCA with RB Failed
ICRX3	ICRYY	5.9967E-07	Given Large LOCA with All Support Available
ICRX4	ICRYY	2.1956E-06	Given Large LOCA with RB Failed
INAF	INA	1.0000E+00	Guaranteed Failure
INAS	INA	0.0000E+00	Guaranteed Success
INBF	INB	1.0000E+00	Guaranteed Failure
INBS	INB	0.0000E+00	Guaranteed Success
INCF	INC	1.0000E+00	Guaranteed Failure
INCS	INC	0.0000E+00	Guaranteed Success
INNF	INN	1.0000E+00	Guaranteed Failure
INNS	INN	0.0000E+00	Guaranteed Success
INTPRF	INTPR	1.0000E+00	RCS PRESSURE > 2000 PSIA - Guaranteed Failure
INTPRS	INTPR	0.0000E+00	RCS PRESSURE > 2000 PSIA - Guaranteed Success
IP1	IP	9.0383E-04	All Support Available
IPCLX1	ICRYY	6.7324E-07	Given Medium LOCA with RF Failed Due To Support Unavailable
IPCLX2	ICRYY	1.0390E-05	Given Large LOCA with RF Failed Due To Support Unavailable
IPF	IP	1.0000E+00	Guaranteed Failure
IPRFX1	ICRYY	1.2103E-06	Given CL Failed with All Support Available
IPRFX2	ICRYY	4.1662E-06	Given CL and RB Failed
IYAF	IYA	1.0000E+00	Guaranteed Failure
IYAS	IYA	0.0000E+00	Guaranteed Success
IYBF	IYB	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 13 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
IYBS	IYB	0.0000E+00	Guaranteed Success
IYCF	IYC	1.0000E+00	Guaranteed Failure
IYCS	IYC	0.0000E+00	Guaranteed Success
IYNF	IYN	1.0000E+00	Guaranteed Failure
IYNS	IYN	0.0000E+00	Guaranteed Success
LCL1	LCL	3.9540E-03	Given Large LOCA with IP Succeeded and All Support Available
LCL2	LCL	3.9610E-03	Given Large LOCA with IP Failed Due To SIS Pump Unavailable
LCL3	LCL	1.1500E-02	Given Large LOCA with IP Failed
LCLF	LCL	1.0000E+00	Guaranteed Failure
LCLX1	ICRYY	3.9624E-03	Given Large LOCA with IP and RF Failed Due To Support Unavailable
LNAF	LNA	1.0000E+00	Guaranteed Failure
LNAS	LNA	0.0000E+00	Guaranteed Success
LNBF	LNB	1.0000E+00	Guaranteed Failure
LNBS	LNB	0.0000E+00	Guaranteed Success
LNCF	LNC	1.0000E+00	Guaranteed Failure
LNCS	LNC	0.0000E+00	Guaranteed Success
LNNF	LNN	1.0000E+00	Guaranteed Failure
LNNS	LNN	0.0000E+00	Guaranteed Success
LOWPRF	LOWPR	1.0000E+00	RCS PRESSURE NOT LOW (>200 PSIA) - Guaranteed Failure
LOWPRS	LOWPR	0.0000E+00	RCS PRESSURE NOT LOW (>200 PSIA) - Guaranteed Success
LYAF	LYA	1.0000E+00	Guaranteed Failure
LYAS	LYA	0.0000E+00	Guaranteed Success
LYBF	LYB	1.0000E+00	Guaranteed Failure
LYBS	LYB	0.0000E+00	Guaranteed Success
LYCF	LYC	1.0000E+00	Guaranteed Failure
LYCS	LYC	0.0000E+00	Guaranteed Success
LYNF	LYN	1.0000E+00	Guaranteed Failure
LYNS	LYN	0.0000E+00	Guaranteed Success
MA1	MA	7.5990E-03	TP Failed Due To Support FailureS, CST Available and ALL Support
MA10	MA	8.5790E-02	TP Failed, Degraded HVAC, ALTERNATE SUCTION Required and ALL Support
MA11	MA	1.0350E-01	TP Successful, ALTERNATE SUCTION Required with 1 ERCW Train Available and ALL Support
MA12	MA	8.0440E-02	TP Failed, ALTERNATE SUCTION Required with 1 ERCW Train Available and ALL Support
MA13	MA	1.0460E-01	TP Successful, Degraded HVAC, ALTERNATE SUCTION Required with 1 ERCW Train Available
MA14	MA	7.1280E-02	TP Failed, Degraded HVAC, ALTERNATE SUCTION Required with 1 ERCW Train Available
MA1C	TOT	7.5992E-03	TP Failed Due To Support FailureS, CST Available and ALL Support
MA2	MA	1.0160E-01	TP Failed Due To Support FailureS, CST Failed and All Support Available
MA2C	TOT	1.0159E-01	TP Failed Due To Support FailureS, CST Failed and All Support Available
MA3	MA	7.6220E-03	TP Successful, CST Available and All Support Available
MA4	MA	7.2670E-03	TP Failed, CST Available and ALL Support
MA5	MA	7.6720E-03	TP SUCCESFUL, Degraded HVAC, CST Available and All Support Available

Table 3.3.5-1 (Page 14 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
MA6	MA	6.6650E-03	TP Failed, Degraded HVAC, CST Available and All Support Available
MA7	MA	1.0210E-01	TP Successful, ALTERNATE SUCTION Required and ALL Support
MA8	MA	9.3780E-02	TP Failed, ALTERNATE SUCTION Required and ALL Support
MA9	MA	1.0280E-01	TP Successful, Degraded HVAC, ALTERNATE SUCTION Required and ALL Support
MACROS	MACRO	0.0000E+00	Guaranteed Success
MAF	MA	.0000E+00	Guaranteed Failure
MAMB1	TOT	5.6759E-04	All Support Available
MAMB2	TOT	1.3906E-03	CST Unavailable
MB1	MB	7.5990E-03	TP Bypass, MA Bypass, ALL Support
MB10	MB	7.4250E-02	TP Successful, MA Failed, CST Available
MB11	MB	6.7230E-03	TP Failed, MA Successful, CST Available
MB12	MB	8.1540E-02	TP Failed, MA Failed, CST Available
MB13	MB	7.1530E-03	TP Successful, MA Successful, CST Available, Degraded HVAC
MB14	MB	7.4790E-02	TP Successful, MA Failed, CST Available, Degraded HVAC
MB15	MB	6.2180E-03	TP Failed, MA Successful, CST Available, Degraded HVAC
MB16	MB	7.3290E-02	TP Failed, MA Failed, CST Available, Degraded HVAC
MB17	MB	1.0210E-01	TP SUCESSFUL, MA Bypass, CST Unavailable
MB18	MB	9.3780E-02	TP Failed, MA Bypass, CST Unavailable
MB19	MB	1.0350E-01	TP SUCESSFUL, MA Bypass, CST Unavailable, ONLY 1 ERCW Train Available
MB2	MB	1.0160E-01	TP Bypass, MA Bypass, CST Unavailable
MB20	MB	8.0440E-02	TP Failed, MA Bypass, CST Unavailable, ONLY 1 ERCW Train Available
MB21	MB	1.0280E-01	TP SUCESSFUL, MA Bypass, CST Unavailable, Degraded HVAC
MB22	MB	8.5790E-02	TP Failed, MA Bypass, CST Unavailable, Degraded HVAC
MB23	MB	1.0460E-01	TP SUCESSFUL, MA Bypass, CST Unavailable, ONLY 1 ERCW Train Available, Degraded HVAC
MB24	MB	7.1280E-02	TP Fail, MA Bypass, CST Unavailable, ONLY 1 ERCW Train Available, Degraded HVAC
MB25	MB	1.1150E-01	TP Bypass, MA Successful, CST Unavailable
MB26	MB	1.3690E-02	TP Bypass, MA Failed, CST Unavailable
MB27	MB	1.1230E-01	TP Successful, MA Successful, CST Unavailable
MB28	MB	1.2920E-02	TP Successful, MA Failed, CST Unavailable
MB29	MB	1.0080E-01	TP Failed, MA Successful, CST Unavailable
MB3	MB	7.6220E-03	TP Successful, MA Bypass, ALL Support
MB30	MB	2.6080E-02	TP Failed, MA Failed, CST Unavailable
MB31	MB	1.1310E-01	TP Successful, MA Successful, CST Unavailable, Degraded HVAC
MB32	MB	1.2960E-02	TP Successful, MA Failed, CST Unavailable, Degraded HVAC
MB33	MB	9.1500E-02	TP Failed, MA Successful, CST Unavailable, Degraded HVAC
MB34	MB	2.4950E-02	TP Failed, MA Failed, CST Unavailable, Degraded HVAC
MB4	MB	7.2670E-03	TP Fail, MA Bypass, ALL Support
MB5	MB	7.6720E-03	TP SUCESSFUL,Degraded HVAC, MA Bypass, CST Available
MB6	MB	6.6650E-03	TP Failed,Degraded HVAC, MA Bypass, CST Available
MB7	MB	7.0850E-03	TP Bypass, MA Successful, CST Available
MB8	MB	7.4700E-02	TP Bypass, MA Failed, CST Available

Table 3.3.5-1 (Page 15 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
MB9	MB	7.1100E-03	TP Successful, MA Successful, CST Available
MBF	MB	1.0000E+00	Guaranteed Failure
MDE1	MDE	1.0650E-03	MAINTENANCE ALIGNMENT ON STrainER For DE HEADER
MDEF	MDE	1.0000E+00	Guaranteed Failure
MELTBF	MELTB	1.0000E+00	MELT with CNMT BypassED - Guaranteed Failure
MELTBS	MELTB	0.0000E+00	MELT with CNMT BypassED - Guaranteed Success
MELTF	MELT	1.0000E+00	NO Core MELT For Medium LOCA - Guaranteed Failure
MELTIF	MELTI	1.0000E+00	MELT Without CNMT ISOLATED - Guaranteed Failure
MELTIS	MELTI	0.0000E+00	MELT Without CNMT ISOLATED - Guaranteed Success
MELTLF	MELTL	1.0000E+00	Guaranteed Failure
MELTLS	MELTL	0.0000E+00	Guaranteed Success
MELTS	MELT	0.0000E+00	NO Core MELT For Medium LOCA - Guaranteed Success
MELTSF	MELTS	1.0000E+00	MELT with Large PENETRATION Isolation Failure - Guaranteed Failure
MELTSS	MELTS	0.0000E+00	MELT with Large PENETRATION Isolation Failure - Guaranteed Success
MF1	MF	2.3885E-02	All Support Available
MFF	MF	1.0000E+00	Guaranteed Failure
MR1	MR	8.7710E-03	Operator INITIATES MANUAL ROD INSERTION
MRF	MR	1.0000E+00	Guaranteed Failure Of MR
MS1	MS	2.2688E-04	All Support Available
MSF	MS	1.0000E+00	GUARNTEED Failure
MSS	MS	0.0000E+00	GUARANTEED Success
MU1	MU	4.2160E-01	Given SLOCA with CNMT SPRAY Train A (or B) Failed
MU2	MU	4.3170E-01	Given SLOCA and All Support Available
MU3	MU	3.3861E-02	Given SGTR with All Support Available
MU4	MU	4.6724E-01	Given INDUCED SGTR and LOCA
MU5	MU	3.2572E-02	Given SGTR with DAAC (or DBAC, or B1) or A1U2 (or B1U2) Failed
MU6	MU	4.8403E-01	Given INDUCED SGTR and SLOCA with DAAC (or DBAC, or B1) or A1U2 (or B1U2) Failed
MU7	MU	3.2537E-02	Given SGTR with DAAC (or DBAC, or B1) and A1U2 (or B1U2) Failed
U8	MU	4.8404E-01	Given INDUCED SGTR and SLOCA with DAAC (or DBAC, or B1) and A1U2 (or B1U2) Failed
MUF	MU	1.0000E+00	Guaranteed Failure
OB1	OB	4.4179E-02	All Support Available
OBF	OB	1.0000E+00	Guaranteed Failure
Of1	Of	3.8010E-02	Given General Transient NOT REQUIRING SI with AFW Failed
Of2	Of	4.8920E-02	Given TRANSIENT REQUIRING SI with AFW Failed
OfF	Of	1.0000E+00	Guaranteed Failure
OG1	OG	4.8853E-04	All Support Available
OGF	OG	1.0000E+00	Guaranteed Failure
OGR11	OGR1	2.5500E-01	Failure TO Recovery OfFSITE GRID IN One HOUR
OGR1S	OGR1	0.0000E+00	OfFSITE GRID Successful
OS1	OS	3.5650E-02	Operator Failed TO ALIGN ECCS, Given ESFAS Failed FOLLOWING A MLOCA or LLOCA
OS2	OS	3.2500E-02	Operator Failed TO ALIGN ECCS, Given ESFAS Failed (MSLBOC,SGTR,SLOCA)

Table 3.3.5-1 (Page 16 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
OS3	OS	2.0850E-03	Operator Failed TO START AFW, Given RX TRIP with NO SI Required
OS4	OS	1.2140E-02	Operator Failed TO START AFW, Given ATWS with AMSAC Failure
OS5	OS	1.2700E-01	BACKUP RESTART TIMERS, Given LOSP and D/G STARTUP
OSF	OS	1.0000E+00	Guaranteed Failure
OSS	OS	0.0000E+00	Guaranteed Success
OT1	OT	1.6490E-03	OperatorS TERMINATE CNMT SPRAY
OTF	OT	1.0000E+00	GUARANTEED FailRUE
OTS	OT	0.0000E+00	Guaranteed Success
PA1	PA	3.2562E-03	All Support Available
PA2	PA	2.0688E-02	Loss Of PD (CONTROL AIR)
PA3	PA	4.0285E-03	Loss Of A1U2 or CE or GE
PAB1	PAB	4.4760E-05	All Support Available
PAB2	PAB	7.8314E-04	Loss Of PD
PAB3	PAB	5.5098E-05	Loss Of A1U2
PAB4	PAB	3.2550E-03	Loss Of DAAC
PAB5	PAB	2.1042E-02	Loss Of PD and DAAC
PAB6	PAB	6.0252E-05	Loss Of A1U2 and B1U2
PAB7	PAB	4.0481E-03	Loss Of B1U2 and DAAC
PABF	PAB	1.0000E+00	Guaranteed Failure
PAF	PA	1.0000E+00	Guaranteed Failure
PB1	PB	3.2200E-03	All Support Available, PA = S
PB10	PB	2.1040E-02	Loss Of PD and DAAC
PB11	PB	4.0480E-03	Loss Of (FE or B1U2 or HE) and DAAC
PB12	PB	4.0040E-03	Loss Of (FE or B1U2 or HE) and (CE or A1U2 or GE), PA = S
PB13	PB	1.4960E-02	Loss Of (FE or B1U2 or HE) and (CE or A1U2 or GE), PA = F
PB2	PB	1.3750E-02	All Support Available, PA = F
PB3	PB	2.0690E-02	Loss Of PD, PA = S
PB4	PB	3.7850E-02	Loss Of PD, PA = F
PB5	PB	3.2120E-03	Loss Of A1U2 or CE or GE, PA = S
PB6	PB	1.3680E-02	Loss Of A1U2 or CE or GE, PA = F
PB7	PB	4.0060E-03	Loss Of B1U2 or FE or HE, PA = S
PB8	PB	1.6920E-02	Loss Of B1U2 or FE or HE, PA = F
PB9	PB	3.2550E-03	Loss Of DAAC
PBF	PB	1.0000E+00	Guaranteed Failure
PD1	PD	1.1200E-03	All Support Available
PD2	PD	3.5332E-03	Loss Of A2 or B1
PD3	PD	1.3808E-03	Loss Of A3 or B3
PD4	PD	4.0113E-02	Loss Of A2 and B1
PD5	PD	3.2458E-02	Loss Of (A2 or B1) and (A3 or B3)
PD6	PD	7.6925E-03	Loss Of A3 and B3
PDF	PD	1.0000E+00	Guaranteed Failure
PE1	PE	9.9705E-07	All Support Available
PE2	PE	1.1338E-04	Loss Of CE or DE
PEF	PE	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 17 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
PI1	PI	1.4208E-04	ALL Support
PI2	PI	1.2622E-02	A1 Failed
PI3	PI	2.4500E-02	BOTH BLOCK VALVE Support Failed
PIF	PI	1.0000E+00	Guaranteed Failure
PIS	PI	0.0000E+00	Guaranteed Success
PL1	PL	6.6200E-01	PROBABILITY POWER LEVEL > 40%
PR1	PR	4.1449E-04	STEAM CHALLENGE, ALL Support
PR2	PR	9.2581E-04	STEAM CHALLENGE, One PorV Support Failed
PR3	PR	.1814E-03	STEAM CHALLENGE, BOTH PorV Supports Failed
PR4	PR	2.3152E-02	STEAM CHALLENGE, One BLOCK VALVE Support Failed
PR5	PR	2.3853E-02	STEAM CHALLENGE, One PorV Support Failed, BLOCK VALVE Support IN THE Other Train Failed
PR6	PR	4.6109E-02	STEAM CHALLENGE, BOTH BLOCK VALVE Support Failed
PRA	PR	5.6447E-03	Water Challenge, All Support Available
PRB	PR	2.7350E-02	Water Challenge, One PorV Support Failed
PRC	PR	2.9913E-01	Water Challenge, BOTH PorV Support Failed
PRD	PR	2.8169E-02	Water Challenge, One BLOCK VALVE Support Failed
PRE	PR	4.9787E-02	Water Challenge, One PorV Support Failed, One BLOCK VALVE Support IN THE Other Train Failed
PRF	PR	1.0000E+00	Guaranteed Failure
PRG	PR	5.1983E-02	Water Challenge, BOTH BLOCK VALVE Support Failed
PRH	PR	9.7320E-03	OperatorS ISOLATE STUCK OPEN PorV
PRS	PR	0.0000E+00	Guaranteed Success
RA1	RA	1.9068E-02	Given Medium or Large LOCA with All Support Available
RA2	RA	2.2860E-02	Given IE IS NOT A Medium or Large LOCA with All Support Available
RABX1	RABX	1.2461E-03	Given Medium or Large LOCA with All Support Available
RABX2	RABX	1.7812E-03	Given IE IS NOT A Medium or Large LOCA with All Support Available
RAF	RA	1.0000E+00	Guaranteed Failure
R1	RB	1.8170E-02	Given Medium or Large LOCA with All Support Available and RA Succeeded
RB2	RB	1.9070E-02	Given Medium or Large LOCA with RA Failed Due To Support Unavailable and ALL Support TO RB Available
RB3	RB	6.5340E-02	Given Medium or Large LOCA with RA Failed and ALL Support TO RB Available
RB4	RB	2.1570E-02	Given IE IS NOT A Medium or Large LOCA with RA Succeeded and ALL SUPP. Available TO RB
RB5	RB	2.2860E-02	Given IE IS NOT A Medium or Large LOCA with RA Failed Due To SUPP. Failure & All Support Available TO RB
RB6	RB	7.7910E-02	Given IE IS NOT A Medium or Large LOCA with RA Failed and All Support Available TO RB
RBF	RB	1.0000E+00	Guaranteed Failure
RD1	RD	1.1486E-02	Given All Support Available
RD2	RD	1.5082E-02	Given Loss Of One RHR HEAT EXCHANGER Train
RD3	RD	1.9294E-02	Given Loss Of 480V POWER Train 1B1-B (Top Event B1)
RD4	RD	2.3573E-02	Given Loss Of One RHR HEAT EXCHANGER Train and 480V POWER Train 1B1-B (Top Event B1)

Table 3.3.5-1 (Page 18 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
RDF	RD	1.0000E+00	Guaranteed Failure
REC1	REC	3.7600E-02	Two Diesels Fail, TD Success
REC2	REC	3.7600E-02	Two Diesels Fail, TD Success
REC3	REC	9.0320E-02	Two Diesels Fail, TD Fails
REC4	REC	3.7600E-02	Two Diesels Fail, TD Success
REC5	REC	6.4720E-02	One Diesel Fails, TD Success
REC6	REC	4.6900E-02	One DIESEL Fails, TD Success, 6.9KV SD BD VENT. Fails
RECF	REC	1.0000E+00	Guaranteed Failure
RF1	RF	5.1020E-04	Given Medium LOCA with IP and CL Succeeded and All Support Available
RF2	RF	1.6370E-03	Given Medium LOCA with IP and CL Succeeded and RB Failed
RF3	RF	5.7840E-03	Given Medium LOCA, IP Succeeded, CL Failed, and All Support Available
RF4	RF	1.5550E-01	Given Medium LOCA with IP Succeeded and CL and RB Failed
RF5	RF	5.1030E-04	Given Medium LOCA, IP Failed Due To SI PMP Unavailable, CL Succeeded, and ALL SUPP. Available
RF6	RF	1.6400E-03	Given Medium LOCA, IP Failed Due To SI PMP Unavailable, CL Succeeded, and RB Failed
RF7	RF	6.4390E-02	Given Medium LOCA, IP Failed Due To SI PMP Unavailable, CL Failed, and ALL SUPP. Available
RF8	RF	2.0370E-01	Given Medium LOCA with IP Failed Due To SI PMP Unavailable and CL and RB Failed
RF9	RF	7.0870E-04	Given Medium LOCA, IP Failed, CL Succeeded, and All Support Available
RFA	RF	3.9810E-03	Given Medium LOCA with IP and RB Failed and CL Succeeded
RFB	RF	8.4730E-01	Given Medium LOCA with IP and CL Failed and All Support Available
RFC	RF	8.4780E-01	Given Medium LOCA, IP, CL, and RB Failed
RFD	RF	5.0830E-04	Given Large LOCA with IP and CL Succeeded and All Support Available
RFE	RF	1.1050E-03	Given Large LOCA with IP and CL Succeeded and RB Failed
RFF	RF	1.0000E+00	Guaranteed Failure
RFG	RF	9.9140E-04	Given Large LOCA, IP Succeeded, CL Failed, and All Support Available
RFH	RF	1.3610E-01	Given Large LOCA with IP Succeeded and CL and RB Failed
RFI	RF	5.0850E-04	Given Large LOCA, IP Failed Due To SI PMP Unavailable, CL Succeeded, and ALL SUPP. Available
RFJ	RF	1.1060E-03	Given Large LOCA, IP Failed Due To SI PMP Unavailable, CL Succeeded, and RB Failed
RFK	RF	1.1400E-03	Given Large LOCA, IP Failed Due To SI PMP Unavailable, CL Failed, and ALL SUPP. Available
RFL	RF	1.3630E-01	Given Large LOCA with IP Failed Due To SI PMP Unavailable and CL and RB Failed
RFM	RF	6.8370E-04	Given Large LOCA, IP Failed, CL Succeeded, and All Support Available
RFN	RF	2.2060E-03	Given Large LOCA with IP and RB Failed and CL Succeeded
RFO	RF	5.7700E-02	Given Large LOCA with IP and CL Failed and All Support Available
RFP	RF	2.1130E-01	Given Large LOCA, IP, CL, and RB Failed
RFX1	ICRYY	5.1094E-04	Given IP and CL Failed with All Support Available
RFX2	ICRYY	1.6414E-03	Given IP, CL, and RB Failed
RH1	RH	6.6651E-03	Given All Support Available
RH2	RH	1.5014E-02	Given One RHR Pump Train (RA or RB) or 480V Train A (A1) Failed
RHF	RH	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 19 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
RHRSF	RHRS	1.0000E+00	RHR SPRAY RECIRCULATION - Guaranteed Failure
RHRSS	RHRS	0.0000E+00	RHR SPRAY RECIRCULATION - Guaranteed Success
RI1	RI	3.2090E-04	Given ALL Support AVAILABEL and SI Succeeded
RI2	RI	5.0320E-04	Given SI Succeeded and RA (or RB) Failed
RI3	RI	3.2110E-04	Given SI Failed Due To SI PMP Unavailable and All Support Available
RI4	RI	5.0350E-04	Given SI Failed Due To SI PumpS Unavailable and RA (or RB) Failed
RI5	RI	5.3080E-04	Given SI Failed and All Support Available
RI6	RI	8.2330E-04	Given SI Failed and RA (or RB) Failed
RIF	RI	1.0000E+00	Guaranteed Failure
RIX1	SIRIX	3.2112E-04	Given SI Failed
RIX2	SIRIX	5.0349E-04	Given SI and RA (or RB) Failed
RL1	RL	1.7189E-05	Given All Support Available
RL2	RL	1.7877E-05	Given Top Event DCAC (or DDAC) Failed
RL3	RL	3.3109E-04	Given Top Events DCAC and DDAC Failed
RL4	RL	4.2971E-03	Given Top Event DAAC (or DBAC) Failed
RL5	RL	4.2921E-03	Given Top Events DAAC and DBAC Failed
RL6	RL	4.2819E-03	Given Top Events DAAC (or DBAC) and DCAC (or DDAC) Failed
RLF	RL	1.0000E+00	Guaranteed Failure
RQF	RQ	1.0000E+00	Guaranteed Failure
RQS	RQ	0.0000E+00	Guaranteed Success
RR1	RR	2.9371E-03	Given HIGH HEAD RECIRC IS REQD with All Support Available
RR2	RR	1.0922E-02	Given HIGH HEAD RECIRC IS REQD with BOTH CCP Trains Failed
RR3	RR	1.0914E-02	Given HIGH HEAD RECIRC IS REQD with BOTH SIS Train Failed
RR4	RR	1.1327E-02	Given HIGH HEAD RECIRC IS REQD and Loss Of Support TO Train B EQUIPMENT
RR5	RR	1.0815E-02	Given HIGH HEAD RECIRC IS REQD and Loss Of Support TO Train A EQUIPMENT
RR6	RR	1.9750E-02	Given HIGH HEAD RECIRC IS REQD and Top Event B1 Failed
RR7	RR	2.4236E-02	Given HIGH HEAD RECIRC IS REQD and Top Event A1 Failed
RR8	RR	4.6390E-03	Given HIGH HEAD RECIRC IS NOT REQD and Top Event B1 Failed
RR9	RR	4.7274E-03	Given HIGH HEAD RECIRC IS NOT REQD and Top Event A1 Failed
RRA	RR	8.4906E-03	Given HIGH HEAD RECIRC IS NOT REQD, Top Event B1 Failed and Loss Of Support TO Train A EQUIPMENT
RRB	RR	4.6064E-04	Given HIGH HEAD RECIRC IS NOT REQD and All Support Available
RRF	RR	1.0000E+00	Guaranteed Failure
RS1	RS	5.2243E-03	Given All Support Available
RS2	RS	1.4269E-02	Given One RHR Pump Train (RA or RB) Failed
RSF	RS	1.0000E+00	Guaranteed Failure
RT1	RT	1.6159E-04	ALL Support - NO TRIP or LOSP
RT2	RT	1.7472E-04	1 DC Train Failed - NO RX TRIP or LOSP
RT3	RT	6.1438E-04	BOTH DC Trains Failed - NO RX TRIP or LOSP
RT4	RT	1.6879E-04	Failure Of 1 SSPS Train - NO RX TRIP or LOSP
RT5	RT	1.7351E-04	Failure Of DC POWER & SSPS IN SAME Train - NO RX TRIP or LOSP
RT6	RT	1.8024E-04	Failure Of 1 DC Train & OPPOSITE SSPS Train - NO RX TRIP or LOSP
RT7	RT	6.2895E-04	Failure Of 1 SSPS Train & BOTH DC Trains - NO RX TRIP or LOSP
RT8	RT	1.6708E-03	Failure Of BOTH SSPS Trains - NO RX TRIP or LOSP

Table 3.3.5-1 (Page 20 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
RT9	RT	1.6824E-03	Failure Of BOTH SSPS Trains & 1 DC Train
RTA	RT	2.1393E-03	Failure Of BOTH SSPS Trains & BOTH DC Trains - NO RX TRIP or LOSP
RTB	RT	6.4446E-06	LOSP INITIATING EVENT
RTS	RT	0.0000E+00	RX TRIP INITIATING EVENT
RVA1	RVA	8.9290E-03	Given All Support Available
RVABX1	RVABX	5.2425E-04	Given All Support Available
RVAF	RVA	1.0000E+00	Guaranteed Failure
RVB1	RVB	8.4810E-03	Given All Support Available and RVA Succeeded
RVB2	RVB	8.9300E-03	Given All Support Available and RVA Failed Due To Support UnVAILABLE
RVB3	RVB	5.8710E-02	Given All Support Available and RVA Failed
RVBF	RVB	1.0000E+00	Guaranteed Failure
RW1	RW	6.5900E-07	Failure Of RWST
RWF	RW	1.0000E+00	Guaranteed Failure
S11	S1	1.2333E-02	Given All Support Available
S1F	S1	1.0000E+00	Guaranteed Failure
S21	S2	1.1550E-02	Given All Support Available and TOP S1 Succeeded
S22	S2	1.2330E-02	Given All Support Available TO S2 and TOP S1 Failed Due To Support Unavailable
S23	S2	7.4780E-02	Given All Support Available and S1 Failed
S2F	S2	1.0000E+00	Guaranteed Failure
SE1	SE	5.7463E-07	All Support Available with RCP MOTor BEARING Failure
SE2	SE	2.6847E-02	AC Fails
SE5	SE	3.3397E-02	PHASE B PRESENT
SE9	SE	5.3569E-07	VA and VB Fails with RCP MOTor BEARING Failure
SEA	SE	3.3980E-02	VA and VB Fail and PHASE B PRESENT
SEC	SE	1.8531E-04	TB Fails with RCP MOTor BEARING Failure
SED	SE	2.7368E-02	TB and AC Fail
SEF	SE	1.0000E+00	Guaranteed Failure
SEH	SE	3.3625E-02	TB Fails and PHASE B PRESENT
SEI	SE	2.2730E-03	TB Fails and LOSP
SES	SE	0.0000E+00	Guaranteed Success
SGCLGF	SGCLG	1.0000E+00	S.G. COOLING - Guaranteed Failure
SGCLGS	SGCLG	0.0000E+00	S.G. COOLING Guaranteed Success
SI1	SI	1.0315E-03	Given All Support Available
SIF	SI	1.0000E+00	Guaranteed Failure
SIRIX1	SIRIX	5.4755E-07	Given All Support Available
SIRIX2	SIRIX	8.4929E-07	Given RA (or RB) Failed
SL1	SL	5.1699E-02	All Support Available
SL2	SL	3.4001E-02	MSLBOC Fails
SL3	SL	1.6995E-02	ATWS Fails
SLF	SL	1.0000E+00	Guaranteed Failure
SLS	SL	0.0000E+00	Guaranteed Success
SNAF	SNA	1.0000E+00	Guaranteed Failure
SNAS	SNA	0.0000E+00	Guaranteed Success
SNBF	SNB	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 21 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
SNBS	SNB	0.0000E+00	Guaranteed Success
SNCF	SNC	1.0000E+00	Guaranteed Failure
SNCS	SNC	0.0000E+00	Guaranteed Success
SNNF	SNN	1.0000E+00	Guaranteed Failure
SNNS	SNN	0.0000E+00	Guaranteed Success
SP1	SP	9.2231E-04	Given All Support Available
SR1	SR	3.5428E-02	50% AFW FLOW, MR=S, All Support Available
SR2	SR	2.0191E-01	50% AFW FLOW, MR=S, DA (or DB) Failed
SR3	SR	4.0787E-01	50% AFW FLOW, MR=S, DA and DB Failed
SR4	SR	5.0114E-01	50% AFW FLOW, MR=F, All Support Available
SR5	SR	5.9262E-01	50% AFW FLOW, MR=F, DA (or DB) Failed
SR6	SR	7.0104E-01	50% AFW FLOW, MR=F, DA and DB Failed
SRF	SR	1.0000E+00	Guaranteed Failure
SRS	SR	0.0000E+00	Guaranteed Success
SU1	SU	2.3091E-04	CNMT SUMP Unavailable - Large or Medium LOCA EVENTS
SU2	SU	2.9429E-04	CNMT SUMP Unavailable - NOT Large or Medium LOCA EVENTS
SU3	SU	8.8285E-03	CNMT SUMP Unavailable - Core DAMAGE SCENARIOS
SYAF	SYA	1.0000E+00	Guaranteed Failure
SYAS	SYA	0.0000E+00	Guaranteed Success
SYBF	SYB	1.0000E+00	Guaranteed Failure
SYBS	SYB	0.0000E+00	Guaranteed Success
SYCF	SYC	1.0000E+00	Guaranteed Failure
SYCS	SYC	0.0000E+00	Guaranteed Success
SYNF	SYN	1.0000E+00	Guaranteed Failure
SYNS	SYN	0.0000E+00	Guaranteed Success
TB1	TB	3.1053E-04	All Support Available
TB2	TB	4.8679E-02	Loss Of Support TO One BOOSTER Pump
TB3	TB	1.2852E-03	All Support Available with LOSP
TB4	TB	6.0956E-02	Loss Of Support TO One BOOSTER Pump with LOSP
TBF	TB	1.0000E+00	Guaranteed Failure
TD1	TD	1.3288E-04	Train A, 2 Pump Trains Available, ALL Supports Available
TD10	TD	1.3436E-02	Train A, 2 Pump Trains, BA=F, BB=F, OG=F, Other Supports Available
TD2	TD	1.7352E-02	Train A, 1 Pump Train Fails, ALL Supports Available
TD21	TD	3.6281E-03	Train B, ALL Supports Available
TD22	TD	6.0469E-03	Train B, OG=F
TD3	TD	1.9326E-02	Train A, 1 Pump Train, OG Fails, Other Supports Available
TD31	TD	2.2909E-05	Trains A and B, 2 Pump Trains Available, ALL Supports Available
TD32	TD	8.9839E-05	Trains A and B, 1 Pump Train Fails, ALL Supports Available
TD33	TD	2.5535E-04	Trains A and B, 1 Pump Train, OG Fails, Other Supports Available
TD34	TD	1.4162E-04	Trains A and B, 1 Pump Train, 6.9KV BDS BA and BB Fail, OG=S, Other Support Available
TD35	TD	3.3262E-04	Trains A and B, 1 Pump Train, 6.9KV BDS BA and BB Fail, DE=F, OG=F, Other Support Available
TD38	TD	1.0642E-04	Trains A and B, 2 Pump Trains, OG=F, Other Supports Available

Table 3.3.5-1 (Page 22 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
TD39	TD	6.5500E-05	Trains A and B, 2 Pump Trains, BA=F, BB=F, OG=S, Other Supports Available
TD4	TD	3.0348E-02	Train A, 1 Pump Train, 6.9KV BDS BA and BB Fail, OG=S, Other Support Available
TD40	TD	1.8849E-04	Trains A and B, 2 Pump Trains, BA=F, BB=F, OG=F, Other Supports Available
TD5	TD	3.3013E-02	Train A, 1 Pump Train, 6.9KV BDS BA and BB Fail, DE=F, OG=F, Other Support Available
TD8	TD	3.4031E-04	Train A, 2 Pump Trains, OG=F, Other Supports Available
TD9	TD	1.2814E-02	Train A, 2 Pump Trains, BA=F, BB=F, OG=S, Other Supports Available
TOT1	TOT	.7638E-05	ALL Support
TOT2	TOT	1.5434E-04	CST Unavailable
TOT3	TOT	3.5464E-05	Degraded Ventilation
TOT4	TOT	1.5449E-04	Degraded Ventilation and CST Unavailable
TOT5	TOT	2.1250E-03	CST Unavailable and ONLY 1 ERCW Train Available
TOT6	TOT	2.1303E-03	Degraded Ventilation, CST Unavailable and ONLY 1 ERCW Train Available
TP1	TP	6.3520E-02	ALL Support
TP1C	TOT	6.3529E-02	ALL Support
TP2	TP	6.3100E-02	CST Unavailable
TP2C	TOT	6.3102E-02	CST Unavailable
TP3	TP	7.2600E-02	Degraded Ventilation
TP3C	TOT	7.2597E-02	Degraded Ventilation
TP4	TP	7.2160E-02	Degraded Ventilation and CST Unavailable
TP4C	TOT	7.2157E-02	Degraded Ventilation and CST Unavailable
TP5	TP	8.1870E-02	CST Unavailable and ONLY 1 ERCW Train Available
TP5C	TOT	8.1866E-02	CST Unavailable and ONLY 1 ERCW Train Available
TP6	TP	9.0260E-02	Degraded Ventilation, CST Unavailable and ONLY 1 ERCW Train Available
TP6C	TOT	9.0263E-02	Degraded Ventilation, CST Unavailable and ONLY 1 ERCW Train Available
TP7	TP	1.4650E-04	STGR Initiating Event with Support For Swapover Available
TP7C	TOT	5.8602E-04	SGTR Initiating Event, Support Available For Swapover
TP8	TP	2.5000E-01	STGR Initiating Event with No Support Available For Swapever
TPF	TP	1.0000E+00	Guaranteed Failure
TPMA1	TOT	4.6157E-04	ALL Support
TPMA2	TOT	5.9182E-03	CST Unavailable
TPMA3	TOT	4.8388E-04	Degraded Ventilation
TPMA4	TOT	6.1915E-03	Degraded Ventilation and CST Unavailable
TPMA5	TOT	6.5847E-03	CST Unavailable and ONLY 1 ERCW Train Unavailable
TPMA6	TOT	6.4329E-03	Degraded Ventilation, CST Unavailable and ONLY 1 ERCW Train Available
TPR1	TPR	8.0860E-01	Manual Local Start TDAFW Pump AFTER LOSP
TPRF	TPR	1.0000E+00	Manual Local Start TDAFW Pump AFTER LOSP - Guaranteed Failure
TPRS	TPR	0.0000E+00	Manual Local Start TDAFW Pump AFTER LOSP - Guaranteed Success
TT1	TT	1.0797E-05	All Support Available
TT2	TT	9.4482E-05	Given One Train Of Support Failed
TTF	TT	1.0000E+00	Guaranteed Failure

Table 3.3.5-1 (Page 23 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
TTS	TT	0.0000E+00	Guaranteed Success
UB1	UB	2.7547E-03	Single Train
UB1A1	UB1A	2.7550E-03	All Support Available
UB1AF	UB1A	1.0000E+00	Guaranteed Failure
UB1B1	UB1B	2.4510E-03	Unit BOARD 1A Successful
UB1B2	UB1B	1.1290E-01	Unit BOARD 1A Fails
UB1B3	UB1B	2.7550E-03	Unit BOARD 1A Fails BY Support
UB1BF	UB1B	1.0000E+00	Guaranteed Failure
UB1C1	UB1C	2.3290E-03	Unit BOARDS 1A and 1B Successful
UB1C2	UB1C	5.2060E-02	One Previous Unit BOARD Fails
UB1C3	UB1C	5.9090E-01	ALL Previous Trains Fail
UB1C4	UB1C	2.4510E-03	One Unit BOARD Fails BY Support, Other Successful
UB1C5	UB1C	1.1290E-01	One Unit BOARD Fails BY Support, Other Fails
UB1CF	UB1C	1.0000E+00	Guaranteed Failure
UB1D1	UB1D	2.2330E-03	ALL Support
UB1D2	UB1D	4.3340E-02	One Previous Train Fails
UB1D3	UB1D	2.1080E-01	Two Previous Trains Fail
UB1D4	UB1D	8.5410E-01	ALL Previous Trains Fail
UB1D5	UB1D	2.4510E-03	Two Previous Trains BY Support
UB1D6	UB1D	1.1290E-01	Two Previous Trains BY Support, Third Fails INDEPENDENTLY
UB1DF	UB1D	1.0000E+00	Guaranteed Failure
UB2	UB	3.1102E-04	Two Trains
UB3	UB	1.8379E-04	Three Trains
UBSYS	UB	1.5697E-04	System Unavailable
V11	V1	7.9970E-06	ALL Support
V12	V1	3.7900E-03	A2 Fails
V13	V1	3.2660E-03	B2 Fails
V14	V1	1.4330E-05	LOSP
V15	V1	3.7870E-03	LOSP and A2 or B2 Fails
V1A	VT	3.4724E-04	Single Train ALL Support
V1C	VT	3.3920E-04	Single Train 3 FAN Room
V1F	V1	1.0000E+00	Guaranteed Failure
V1R1	V1R	3.8210E-04	Restore Ventilation - BOTH Shutdown BoardS Available
V1RF	V1R	1.0000E+00	Recovery Of Unit 1 Shutdown Board Room Ventilation - Guaranteed Failure
V1RS	V1R	0.0000E+00	Recovery Of Unit 1 Shutdown Board Room Ventilation - Guaranteed Success
V21	V2	7.7860E-06	V1 Success
V210	V2	9.6240E-05	V1 Fails, A2 Fails
V211	V2	1.0820E-04	V1 Fails, B2 Fails
V212	V2	4.5610E-02	V1 Fails, A2U2 Fails
V213	V2	4.4170E-02	V1 Fails, B2U2 Fails
V214	V2	3.1460E-03	V1 Fails, A2 and A2U2 Fail
V215	V2	1.5770E-03	V1 Fails, A2 and B2U2 Fail
V216	V2	7.9970E-06	V1 BY Support
V217	V2	3.7900E-03	V1 BY Support, A2U2 Fails

Table 3.3.5-1 (Page 24 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
V218	V2	3.2660E-03	V1 BY Support, B2U2 Fails
V219	V2	1.3880E-05	V1 Success, LOSP
V22	V2	2.6420E-02	V1 Fails
V220	V2	1.3430E-05	V1 Success, LOSP, A2 or B2 Fails
V221	V2	3.7860E-03	V1 Success, LOSP, A2U2 or B2U2 Fails
V222	V2	3.7890E-03	V1 Success, LOSP, Loss Of 1 POWER SUPPLY TO EACH Train
V223	V2	3.1260E-02	V1 Fails, LOSP
V224	V2	2.5060E-04	V1 Fails, LOSP, A2 or B2 Fails
V225	V2	6.6250E-02	V1 Fails, LOSP, A2U2 or B2U2 Fail
V226	V2	8.2410E-04	V1 Fails, LOSP, Loss Of 1 POWER SUPLY TO EACH Train
V227	V2	1.4330E-05	V1 BY Support, LOSP
V228	V2	3.7870E-03	V1 BY Support, LOSP, A2U2 or B2U2 Fail
V23	V2	7.9970E-06	V1 BY Support
V24	V2	7.6610E-06	V1 Success, A2 Fails
V25	V2	7.6690E-06	V1 Success, B2 Fails
V26	V2	3.7890E-03	V1 Success, A2U2 Fails
V27	V2	3.2650E-03	V1 Success, B2U2 Fails
V28	V2	3.7920E-03	V1 Success, A2 and A2U2 Fail
V29	V2	3.2720E-03	V1 Success, A2 and B2U2 Fail
V2F	V2	1.0000E+00	Guaranteed Failure
V2R1	V2R	3.5340E-04	Restore Ventilation - BOTH Shutdown BoardS Available
V2RF	V2R	1.0000E+00	Recovery Of Unit 2 Shutdown Board Room Ventilation - Guaranteed Failure
V2RS	V2R	0.0000E+00	Recovery Of Unit 2 Shutdown Board Room Ventilation - Guaranteed Success
V31	V3	5.5440E-07	All Support Available
V32	V3	6.0280E-06	Loss Of A1,B1,CE,DE,GE,or HE
V33	V3	2.7003E-04	Loss Of (A1,CE,or GE)and(B1,DE,or HE)
V34	V3	1.4488E-02	Loss Of OG,B3 or PD
V35	V3	4.5854E-05	Loss Of (OG,B3,or PD)&(A1,CE,GE,B1,DE,or HE)
V36	V3	5.1728E-04	Loss of AHUs and One Train of Coolers
V37	V3	3.7074E-02	Loss of AHUs and Loss of Two Coolers
V3F	V3	1.0000E+00	Guaranteed Failure: Failure Of (OG, A1 and B1) or ((A2, B2 or OG) and (CE, GE, or A1) and (HE, DE, or B1))
VA1	VA	7.3280E-03	Given All Support Available
VA2	VA	8.8330E-03	Given LOSP and All Support Available
VAF	VA	1.0000E+00	Guaranteed Failure
VB1	VB	7.3530E-03	Given All Support Available with VA Succeeded
VB2	VB	7.3280E-03	Given Loss Of Support TO VA and Support TO VB Available
VB3	VB	3.8680E-03	Given VA Failed
VB4	VB	8.8110E-03	Given LOSP and VA Succeeded with All Support Available
VB5	VB	8.8330E-03	Given LOSP and VA Failed Due To Support Unavailable
VB6	VB	1.1270E-02	Given LOSP and VA Failed
VBF	VB	1.0000E+00	Guaranteed Failure
VC1	VC	7.0947E-04	Given All Support Available
VC2	VC	8.9203E-03	Given Loss Of Support Train A or B

Table 3.3.5-1 (Page 25 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
VCF	VC	1.0000E+00	Guaranteed Failure
VF1	VF	7.3823E-04	Given All Support Available
VF2	VF	9.0872E-03	Given Loss Of One Support Train (A or B)
VFF	VF	1.0000E+00	Guaranteed Failure
VI1	IEVI	4.0074E-06	RHR INJECTION INITIATING EVENT
VI11	VI1	2.7803E-03	RHR PumpS Fail TO BE ISOLATED
VI21	VI2	5.6100E-03	Manual Valve 74-34 Not Closed
VI31	VI3	8.0690E-04	Check Valve 63-502 Not Closed
VI41	VI4	1.6040E-01	Conditional Frequency of Leak > 800 GPM
VI4D1	VI4D	6.4450E-07	LEAK > RWST Vent Capacity (800 GPM)
VI5F	VI5	1.0000E+00	Operator Fails To Closes MOV 63-1
VI61	VI6	4.9560E-05	1 Of 2 Relief Valves Fail TO OPEN
VI62	VI6	7.4640E-05	1 Of 3 Relief Valves Fail TO OPEN
VI71	VI7	1.6040E-01	Conditional Frequency of Leak > 800 GPM
VI72	VI7	7.4990E-02	Conditional Frequency of Leak > 1700 GPM
VI7D1	VI7D	5.0140E-07	LEAK > 800 GPM
VI7D2	VI7D	3.0276E-07	LEAK > 1700 GPM
VI81	VI8	5.0000E-02	PIPING DOWNSTREAM Fail W/ 800 GPM RELIEF
VI82	VI8	1.0000E-01	PIPING DOWNSTREAM Fail with NO RELIEF
VI83	VI8	7.0000E-01	RHR Failure with 1700 GPM RELIEF
VI91	VI9	1.0000E-01	Operator Failure - 900 GPM TO PRT & Large LOCA Outside
VI92	VI9	1.0000E-01	Operator Failure - No Flow To PRT and Small LOCA Outside
VI93	VI9	1.0000E-01	Operator Failure - No Flow To PRT and Large LOCA Outside
VI94	VI9	1.0000E-01	Operator Failure - 1700 GPM TO PRT and Small LOCA through RHR Pump SEAL
VI95	VI9	1.0000E-01	Operator Failure - 1700 GPM TO PRT and Large LOCA Outside
VI96	VI9	1.0000E-01	Operator Failure - No Flow To PRT and Large LOCA Outside
VI97	VI9	1.0000E-01	Operator Failure - < 900 GPM TO RWST, Isolation, VI13 = F
VI98	VI9	1.0000E-01	Operator Failure - < 900 GPM TO RWST, Isolation VI12 = F
VI99	VI9	1.0000E-01	Operator Failure - > 900 RWST RUPTURE, Core DAMAGE Without Bypass
VI9F	VI9	1.0000E+00	Guaranteed Failure
VIN1	VINV	1.9252E-04	Single Train
VIN1A	VINV	6.6039E-04	Single Train LOSP
VIN2	VINV	2.0593E-06	System
VIN2A	VINV	1.2377E-05	System LOSP
VINV11	VINV1	1.9250E-04	All Support Available
VINV12	VINV1	6.6040E-04	LOSP
VINV1F	VINV1	1.0000E+00	Guaranteed Failure
VINV21	VINV2	1.9050E-04	1B Room Success; ALL Support
VINV22	VINV2	1.0700E-02	1B Room Fails
VINV23	VINV2	6.4840E-04	1B Room Success; LOSP
VINV24	VINV2	1.8740E-02	1B Room Fails; LOSP
VINV2F	VINV2	1.0000E+00	Guaranteed Failure
VIV1	VIV	7.9972E-06	Single Train

Table 3.3.5-1 (Page 26 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
VIV1A	VIV	3.7900E-03	Single Train, A2 Fails
VIV1B	VIV	3.2659E-03	Single Train, B2 Fails
VIV1C	VIV	1.4328E-05	Single Train, LOSP
VIV1D	VIV	3.7864E-03	Single Train, LOSP, A2 or B2 Fails
VIV2	VIV	2.1125E-07	System, ALL Support
VIV2A	VIV	3.6471E-07	System, A2 Fails
VIV2B	VIV	1.1923E-05	System, A2 and A2U2 Fails
VIV2C	VIV	5.9763E-06	System, A2 and B2U2 Fails
VIV2D	VIV	3.5324E-07	System, B2 Fails
VIV2E	VIV	5.3337E-06	System, B2 and A2U2 Fail
VIV2F	VIV	3.1207E-06	System, B2 and B2U2 Fail
VIV2G	VIV	4.4786E-07	System, LOSP
VIV2H	VIV	9.4916E-07	System, LOSP, A2 Fails
VIV2I	VIV	1.2546E-05	System, LOSP, A2 and A2U2 Fail
VNV1R1	VNV1R	2.0100E-03	Restore Ventilation
VNV1RF	VNV1R	1.0000E+00	Recovery Of Unit 1 480V BOARD Room B Ventilation - Guaranteed Failure
VNV1RS	VNV1R	0.0000E+00	Recovery Of Unit 1 480V BOARD Room B Ventilation - Guaranteed Success
VNV2R1	VNV2R	2.0630E-03	Restore Ventilation
VNV2RF	VNV2R	1.0000E+00	Recovery Of Unit 2 480V BOARD Room B Ventilation - Guaranteed Failure
VNV2RS	VNV2R	0.0000E+00	Recovery Of Unit 2 480V BOARD Room B Ventilation - Guaranteed Success
VP1	VP	2.8343E-05	Given All Support Available
VP2	VP	9.9517E-05	Given LOSP and All Support Available
VPA1	VP	7.3272E-03	Failure Of CCP Train A Given All Support Available
VPA2	VP	8.8339E-03	Failure Of CCP Train A Given LOSP
VS1	VS	7.3465E-04	Given SI Required IE and All Support Available
VS11	VS1	8.6510E-04	Check Valve 63-502 Not Closed
VS2	VS	9.1389E-03	Given SI Required IE and Loss Of One Support Train (A or B)
VS21	VS2	7.5940E-06	Manual Valve 74-34 Not Closed
VS3	VS	4.5700E-06	Given IE IS Non-SI Transient
VS31	VS3	3.5490E-02	Conditional Frequency of Leak > RWST Vent Capacity
VS3D1	VS3D	2.5664E-07	Leak > RWST Vent Capacity (900 GPM)
VS4F	VS4	1.0000E+00	Operator Fails To Close MOV 63-1
VS51	VS5	2.2860E-05	Relief Valve 74-505 OpenS
VS61	VS6	5.3400E-05	1 Of 2 Relief Valves Fail To Open
VS71	VS7	3.5490E-02	Conditional Frequency of Leak > 900 GPM
VS7D1	VS7D	2.4924E-07	LEAK > 900 GPM
VS81	VS8	4.3320E-02	Conditional Frequency of Leak > 1700 GPM
VS8D1	VS8D	3.1272E-07	LEAK > 1700 GPM
VS91	VS9	7.0000E-01	RCS Depressurizing at 1700 GPM
VS92	VS9	8.0000E-01	RCS Depressurizing at 900 GPM
VSA1	VSA	1.0000E-01	Operator Failure - 1700 GPM TO PRT & Small LOCA through RHR Pump SEAL
VSA2	VSA	1.0000E-01	Operator Failure - 900 GPM TO PRT and Small LOCA through RHR Pump SEAL

Table 3.3.5-1 (Page 27 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
VSA3	VSA	1.0000E-01	Operator Failure - 1700 GPM TO PRT and Large LOCA Outside
VSA4	VSA	1.0000E-01	Operator Failure - 900 GPM TO PRT and Large LOCA Outside
VSA5	VSA	1.0000E-01	Operator Failure - No Flow To PRT and Large LOCA Outside
VSAF	VSA	1.0000E+00	Guaranteed Failure
VSF	VS	1.0000E+00	Guaranteed Failure
VSIE	IEVS	7.2120E-06	RHR Suction Initiating Event
VT1A1	VT1A	3.4730E-04	All Support Available
VT1A2	VT1A	3.3920E-04	Loss Of Common Board Power Train A
VT1AF	VT1A	1.0000E+00	Guaranteed Failure
VT1AR1	VT1AR	2.0940E-03	Establish Portable Ventilation
VT1ARF	VT1AR	1.0000E+00	Recovery Of Transformer Rooms 1A Ventilation - Guaranteed Failure
VT1ARS	VT1AR	0.0000E+00	Recovery Of Transformer Rooms 1A Ventilation - Guaranteed Success
VT1B1	VT1B	3.3920E-04	All Support Available
VT1BF	VT1B	1.0000E+00	Guaranteed Failure
VT1BR1	VT1BR	2.1820E-03	Establish Portable Ventilation
VT1BRF	VT1BR	1.0000E+00	Recovery Of Transformer Room 1B Ventilation - Guaranteed Failure
VT1BRS	VT1BR	0.0000E+00	Recovery Of Transformer Room 1B Ventilation - Guaranteed Success
VT2A1	VT2A	3.3920E-04	All Support Available
VT2AF	VT2A	1.0000E+00	Guaranteed Failure
VT2AR1	VT2AR	2.1050E-03	Establish Portable Ventilation
VT2ARF	VT2AR	1.0000E+00	Recovery Of Transformer Rooms 2A Ventilation - Guaranteed Failure
VT2ARS	VT2AR	0.0000E+00	Recovery Of Transformer Rooms 2A Ventilation - Guaranteed Success
VT2B1	VT2B	3.4730E-04	All Support Available
VT2B2	VT2B	3.3920E-04	Loss Of Common Board Power Train A
VT2BF	VT2B	1.0000E+00	Guaranteed Failure
VT2BR1	VT2BR	2.0810E-03	Establish Portable Ventilation
VT2BRF	VT2BR	1.0000E+00	Recovery Of Transformer Rooms 2B Ventilation - Guaranteed Failure
VT2BRS	VT2BR	0.0000E+00	Recovery Of Transformer Rooms 2B Ventilation - Guaranteed Success
VTCC	VTCC	7.7840E-07	Common Cause Contribution To 480V Transformer Room Ventilation
WC1	WC	3.6640E-03	Control SI To Prevent Water Challenge Of PorVS
WCF	WC	1.0000E+00	No Water Challenge
WCS	WC	0.0000E+00	Water Challenge
ZA1	ZA	8.4627E-03	General Transient
ZA2	ZA	9.2023E-03	Large LOCA - All Support Available
ZA3	ZA	1.0807E-02	Steam Line Break Inside CNMT (SLBIC) - All Support Available
ZA4	ZA	8.5896E-03	Steam Line Break Outside CNMT (SLBOC) - All Support Available
ZA5	ZA	7.8440E-03	Small LOCA
ZA6	ZA	9.0765E-03	Large LOCA - Loss Of 120V AC II or III or IV
ZA7	ZA	9.9787E-03	Large LOCA - Loss Of 120V AC (II and III) or (II and IV) or (III and IV)
ZA8	ZA	1.0770E-02	SLBIC - Loss Of 120V AC II or III or IV
ZA9	ZA	1.1743E-02	SLBIC - Loss Of 120V AC (II and III) or (II and IV) or (III and IV)
ZAB1	ZAB	5.8715E-04	General Transient
ZAB2	ZAB	6.3607E-04	Large LOCA - All Support Available
ZAB3	ZAB	8.7566E-04	Steam Line Break Inside CNMT (SLBIC) - All Support Available

Table 3.3.5-1 (Page 28 of 28). Watts Bar Quantification Results

Split Fraction Name	Top Event	Split Fraction Value	Split Fraction Description
ZAB4	ZAB	6.0518E-04	Steam Line Break Outside CNMT - All Support Available
ZAB5	ZAB	5.1714E-04	Small LOCA
ZAB6	ZAB	6.3540E-04	Large LOCA - Loss Of 120V AC II or III or IV
ZAB7	ZAB	1.5467E-03	Large LOCA - Loss Of 120V AC (II and III) or (II and IV) or (III and IV)
ZAB8	ZAB	8.4879E-04	SLBIC - Loss Of 120V AC II or III or IV
ZAB9	ZAB	1.6849E-03	SLBIC - Loss Of 120V AC (II and III) or (II and IV) or (III and IV)
ZABF	ZAB	1.0000E+00	Guaranteed Failure
ZAF	ZA	1.0000E+00	Guaranteed Failure
ZB1	ZB	7.9430E-03	ESFAS Train B - General Transient, ZA=S
ZB10	ZB	6.5930E-02	ESFAS Train B - Small LOCA, ZA=F
ZB11	ZB	1.0030E-02	ESFAS Train B - SLBIC, Loss Of 120V AC III or IV, ZA=S
ZB12	ZB	7.8820E-02	ESFAS Train B - SLBIC, Loss Of 120V AC III or IV, ZA=F
ZB13	ZB	8.5180E-03	ESFAS Train B - LLOCA, Loss Of 120V AC III or IV, ZA=S
ZB14	ZB	7.0000E-02	ESFAS Train B - LLOCA, Loss Of 120V AC III or IV, ZA=F
ZB15	ZB	1.0180E-02	ESFAS Train B - SLBIC, Loss Of 120V AC III and IV, ZA=S
ZB16	ZB	1.4350E-01	ESFAS Train B - SLBIC, Loss Of 120V AC III and IV, ZA=F
ZB17	ZB	8.5170E-03	ESFAS Train B - LLOCA, Loss Of 120V AC III and IV, ZA=S
ZB18	ZB	1.5500E-01	ESFAS Train B - LLOCA, Loss Of 120V AC III and IV, ZA=F
ZB19	ZB	1.0770E-02	ESFAS Train B - SLBIC, Loss Of 120V AC I
ZB2	ZB	8.6460E-03	ESFAS Train B - Large LOCA, All Support Available, ZA=S
ZB20	ZB	9.0770E-03	ESFAS Train B - LLOCA, Loss Of 120V AC I
ZB21	ZB	1.1740E-02	ESFAS Train B - SLBIC, Loss Of 120V AC I and (III or IV)
ZB22	ZB	9.9780E-03	ESFAS Train B - LLOCA, Loss Of 120V AC I and (III or IV)
ZB3	ZB	1.0040E-02	ESFAS Train B - SLBIC, All Support Available, ZA=S
ZB4	ZB	8.0530E-03	ESFAS Train B - Steam Line Break Outside CNMT, ZA=S
ZB5	ZB	7.3840E-03	ESFAS Train B - Small LOCA, ZA=S
ZB6	ZB	6.9380E-02	ESFAS Train B - General Transient, ZA=F
ZB7	ZB	6.9120E-02	ESFAS Train B - Large LOCA, All Support Available, ZA=F
ZB8	ZB	8.1020E-02	ESFAS Train B - SLBIC, All Support Available, ZA=F
ZB9	ZB	7.0460E-02	ESFAS Train B - Steam Line Break Outside, ZA=F
ZBF	ZB	1.0000E+00	Guaranteed Failure

3.3.6 GENERATION OF SUPPORT SYSTEM STATES

The probability of success or failure of a top event in a frontline tree may depend on the status of the support systems modeled in the electrical support event tree and in the mechanical support event tree. Advances in software now allow the direct linking of the support trees to the frontline trees, thus eliminating the previous practice of routing data using "support states." This direct linking is part of the general methodology discussed in Section 2.3.

3.3.7 QUANTIFICATION OF SEQUENCE FREQUENCIES

The process for sequence quantification is described in Section 2.3.5.7. That discussion provides the concepts of initiating events, linked event trees, conditional split fractions and associated logic, and plant damage states. This section describes the sequence quantification used in the Watts Bar PRA by providing the relationship between the various trees and the impacts of the initiating events on those trees.

The various initiating events are discussed in Section 3.1.1. Flood initiators are discussed in Section 3.3.8. The discussions of the various event trees are provided in Sections 3.1.2 through 3.1.4. The split fractions used in the quantification process are described in Section 3.3.5. The logic rules are provided in Appendix D.

Figure 3.3.7-1 illustrates the way in which the event trees are linked to delineate the accident sequences from the initiating event categories all the way to the plant damage states. Table 3.3.7-1 maps the event tree descriptions used in Figure 3.3.7-1 to the actual tree names used in the quantification. Note that all initiating events use the macro tree and that all initiating events except the interfacing systems loss of coolant accident (LOCA) initiating events VS and VI proceed through the electrical and mechanical support trees. Of the remaining initiating events, all except medium, large, and excessive LOCAs use the GTRAN tree. Separate event trees are used for these initiators due to the differences in success criteria relative to that assumed in the GTRAN tree.

The different impacts imposed on the GTRAN tree by the initiating event categories that use it for sequence quantification are identified in Table 3.3.7-2.

Table 3.3.7-1. RISKMAN Event Trees and Description	
Event Tree	Description
MACRO	Maps Initiating Events into Macro Names
ELECT1	Unit 1 Electrical Support Equipment
ELECT2	Unit 2 Electrical Support Equipment
MECH	Mechanical Support Equipment
GTRAN	Frontline Tree for General Transients, Small LOCAs, SGTRs, ATWS
RECIRC	Recirculation Tree for GTRAN
RECOVERY	Recovery Tree for GTRAN/RECIRC
GTBINS	Plant Damage State Binning Tree for GTRAN/LOCA
MEDLOCA	Frontline Tree for Medium LOCAs
LARLOCA	Frontline Tree for Large LOCAs
RECL	Recovery Tree Common to Large and Medium LOCAs
MLBINS	Plant Damage State Binning Tree for Medium LOCAs
LLBINS	Binning Tree for Large LOCAs
VILOCA	Frontline Tree for ISLOCAs Originating in the Cold Leg Discharge
VIBIN	Plant Damage State Binning Tree for the VILOCA Tree
VSLOCA	Frontline Tree for ISLOCAs Originating in the Hot Leg Suction
VSBIN	Plant Damage State Tree for the VSLOCA Tree

Table 3.3.7-2 (Page 1 of 3). Sequence Modeling Impacts for Each Initiating Event Category

Initiating Event Category	Impacts of Initiators on Plant System Top Events
CCSA Loss of Component Cooling Water Train A	Top Event AC Guaranteed Failed in the Mechanical Support Tree
CCSTL Total Loss of Component Water	Top Events AC and BC Guaranteed Failed in the Mechanical Support Tree
CPEX Core Power Excursion	Guaranteed Success of Top Event RT[1]
ELOCA Excessive LOCA	Top Event EL = F in Large LOCA Tree
ERCWA Loss of ERCW Train A	Top Event AE Guaranteed Failed, Guaranteed Success of Top Event RT[2]
ERCWB Loss of ERCW Train B	Top Event BE Guaranteed Failed, Guaranteed Success of Top Event RT
ERCWTL Total Loss of ERCW	Top Events AE and BE Guaranteed Failed, Guaranteed Success of Top Event RT
EXMFW Excessive Main Feedwater	Guaranteed Failure of Top Events FW, MF[3]
FLAB2 Flooding, ERCW to Auxiliary Building > 30 Minutes	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Events CE, RA, and RB
FLAB3C Flooding, CST to Auxiliary Building	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Events TP, RA, and RB
FLAB3R Flooding, RWST to Auxiliary Building	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Events RW, RA, and RB
FLPH1A Flood ERCW Strainer Room, Train A	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Event AE
FLPH1B Flood ERCW Strainer Room, Train B	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Event BE
FLTB Internal Flooding in Turbine Building	Guaranteed Success of Top Event RT, Guaranteed Failure of Top Events PD, FW, MF
IMSIV Inadvertent All MSIVs Close	SI/CIA Condition Only[4]. Guaranteed Failure of Top Event PD,Top Event MS Guaranteed Successful, Top Event CD Guaranteed Failed, Top Event WC Questioned[5,6]
ISI Inadvertent Safety Injection	SI/CIA Condition Only, Top Event WC Questioned
LASD Loss of 6.9-kV Shutdown 1A-A	Guaranteed Failure of Top Event AA in the ELECT1 Electrical Support Tree

Table 3.3.7-2 (Page 2 of 3). Sequence Modeling Impacts for Each Initiating Event Category

Initiating Event Category	Impacts of Initiators on Plant System Top Events
LBSD Loss of 6.9-kV Shutdown 1B-B	Guaranteed Failure of Top Event BA in the ELECT1 Electrical Support Tree
LDAAC Loss of 120V AC Vital Board 1-I	Guaranteed Failure of Top Event DAAC in the ELECT1 Electrical Support Tree
LDBAC Loss of 120V AC Vital Board 1-II	Guaranteed Failure of Top Event DBAC in the ELECT1 Electrical Support Tree
LDCAC Loss of 120V AC Vital Board 1-III	Guaranteed Failure of Top Event DCAC in the ELECT1 Electrical Support Tree
LDDAC Loss of 120V AC Vital Board 1-III	Guaranteed Failure of Top Event DDAC in the ELECT1 Electrical Support Tree
LLOCA	Propagated through the Large LOCA Frontline Tree
LOCV Loss of Condenser Vacuum	Guaranteed Failure of Top Events CD, FW, and MF
LOSP Loss of Offsite Power	Guaranteed Failure of Top Event OG in ELECT1 Support Tree
LRCP Loss of One or More RCPs/Primary Flow	Pressurizer PORVs Challenged in Top Event PR
LVBB1 Loss of Battery Board I	Guaranteed Failure of Top Event DA in the ELECT1 Support Tree
LVBB2 Loss Of Battery Board II	Guaranteed Failure of Top Event DB in the ELECT1 Support Tree
MLOCA Medium Break LOCA	Propagated through the Medium LOCA Event Tree
MSIV Inadvertent Closure of One MSIV	SI/CIA Condition, Top Event MS Guaranteed Successful, Top Event WC Questioned
MSVO Steam Generator PORV/Safety Fails Open	SI/CIA Condition Only, Main Steam Isolation Condition, Top Event WC Questioned.
PLMFW Partial Loss of Main Feedwater	Top Event FW Guaranteed Failed after Reactor Trip; Restoration of MFW Possible
RTIE Reactor Trip	Top Event RT Guaranteed Successful
SGTR Steam Generator Tube Rupture	SI/CIA Condition, Top Event RD Guaranteed Failed; Potential for Bypass Sequence
SLBIC Steam Line Break Inside Containment	SI/CIA and CIB Condition[7], Main Steam Line Isolation Condition, Top Event WC Questioned

Table 3.3.7-2 (Page 3 of 3). Sequence Modeling Impacts for Each Initiating Event Category

Initiating Event Category	Impacts of Initiators on Plant System Top Events
SLOCAN Small LOCA Non-Isolable	SI/CIA and CIB Condition, Main Steam Line Isolation Condition, Guaranteed Failure of Top Event SE
TLMFW Total Loss of Main Feedwater	Top Events MF and FW Guaranteed Failed
TTIE Turbine Trip	Top Event TT Guaranteed Success, Turbine Trip Success[8]
VI RHR Discharge Path ISLOCA	Propagated through the VILOCA Tree
VS RHR Suction Path ISLOCA	Propagated through the VSLOCA Tree

Additional Notes/Comments:

1. For initiating event category CPFX (core power excursion), successful reactor trip was assumed since a review of actual events revealed only reactivity decreases.

2. Reactor trip was assumed for all flooding initiators and losses of ERCW. This was a simplifying assumption made because either (1) the initiator did not proceed to plant trip but potentially only resulted in a degraded mode of operation, or (2) the time allowed for manual trip is much longer than that analyzed for ATWS events.

3. Excessive feedwater is assumed to result in a steam generator high-high level trip that closes all feedwater control and isolation valves and trips the main feedwater pumps.

4. Phase A containment isolation (CIA) condition was assumed for any initiating event in which SI would occur.

5. For initiating events that have an SI signal but the RCS is intact, Top Event WC (i.e., challenge for water relief through the pressurizer PORVs) needs to be questioned. It was assumed that the cause of inadvertent closure of MSIVs (IMSIV) could also cause an inadvertent SI, and thus was included as questioning Top Event WC.

6. Loss of instrument air is one cause of closure of all MSIVs. The frequency of loss of control air system (Top Event PD) is included with the closure of all MSIVs initiating event category.

7. Phase B containment isolation (CIB) condition was conservatively assumed for those initiating events in which there is a significant mass/energy release inside containment.

8. Top Event TT does not include the trip signals, so IE TT guarantees success of Top Event TT and a signal to trip the turbine.

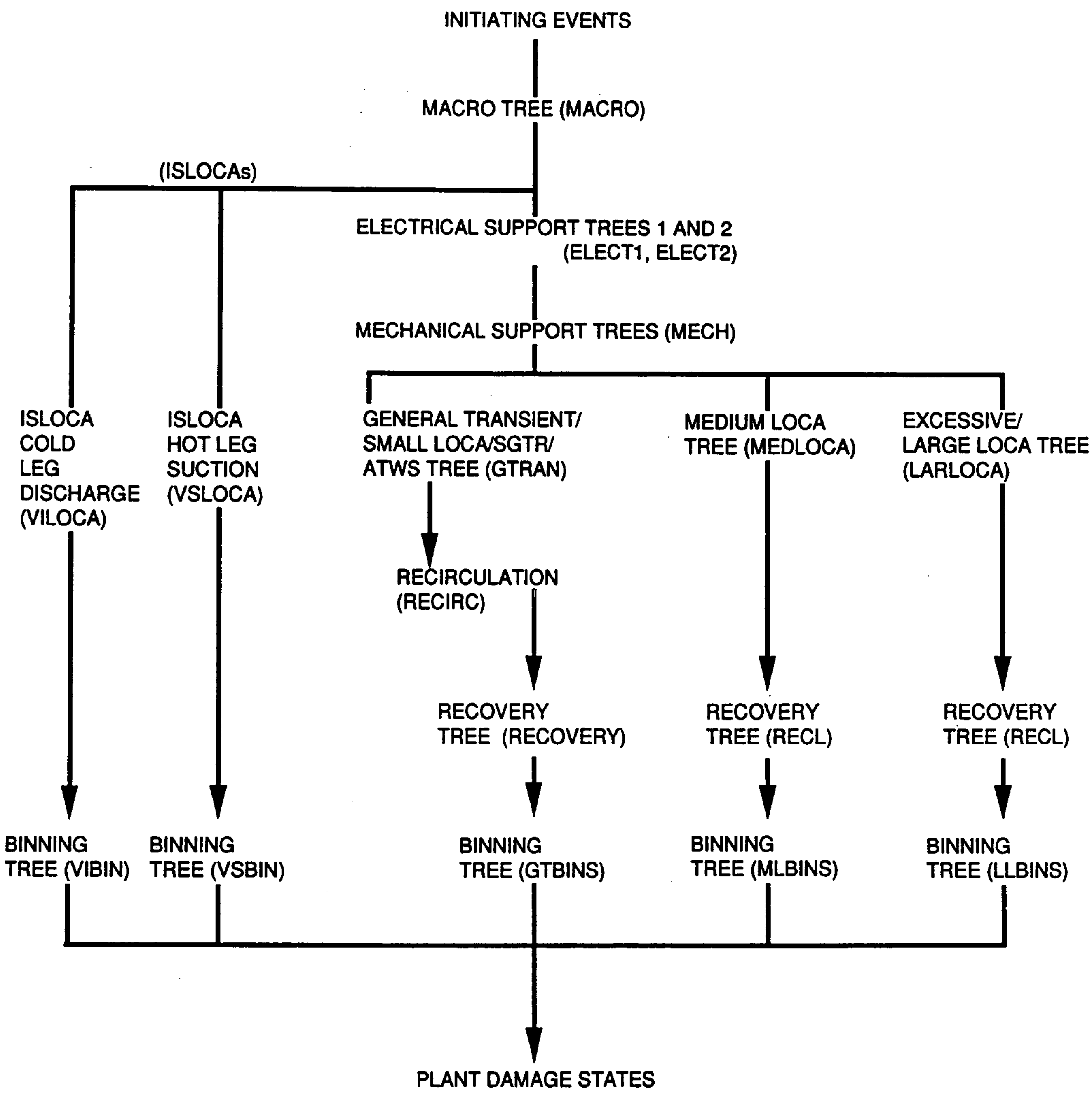


Figure 3.3.7-1. Event Tree Linking

3.3.8 INTERNAL FLOODING ANALYSIS

3.3.8.1 Introduction

An analysis has been completed to identify accident sequences involving internal floods at Watts Bar Nuclear Plant Units 1 and 2. Probabilistic risk assessments (PRA) have shown that spatial hazards such as floods can contribute to core damage frequency since more than one component or system can be affected by the same common cause event. Floods that cause an initiating event and a common mode failure of critical systems (usually support systems that cause additional intersystem dependent failures) are important. This section summarizes a more detailed report that documents this analysis (Appendix E). The analysis identifies internal flooding initiating events and their associated frequencies and impacts on plant equipment. The flood scenarios are treated as initiating events to the transient response model as identified in Sections 3.1.1 and 3.1.2. The quantitative results and contributions to risk from internal floods are summarized in Section 3.4.

3.3.8.2 Methodology and Approach

The basic approach is a screening analysis that first establishes potential major flood sources and safety equipment locations. Flood scenarios are postulated in terms of the flooding source, the extent of propagation to adjacent locations, and the equipment impacted. The frequencies of these scenarios are then quantified as initiating events to the transient event tree model with impacts on event tree top events based on the flood impact to equipment. A more detailed analysis of the flood initiator may be performed, as required, when the risk results are important. The methodology is summarized further below:

- **Plant Familiarization.** Key plant design information that provides details of the plant systems and layout is reviewed. The PRA models are reviewed to ensure familiarity with important intersystem dependencies, success criteria, and plant response models.

- **Flood Experience Review.** Flood data collected from *Nuclear Power Experience* (Reference 3.3.8-1) are reviewed to ensure familiarity with actual flood events, their locations within the plant, and causes. These data are used in the quantification of internal flood scenario initiating event frequencies. Plant-specific screening of flood events in Reference 3.3.8-1 was performed and used in the analysis.

- **Evaluation of Flood Sources.** Using the plant design information and a general knowledge of plant layout, major flood sources and their locations are identified. For example, the Tennessee River supplies the essential raw cooling water (ERCW) system, which supplies cooling to several plant locations. ERCW is identified as a major flood source, and its locations within buildings are identified from the ERCW system flow diagrams.

- **Evaluation of Plant Locations.** Using plant design information such as arrangement drawings, internal flood studies, and information from the evaluation of flood sources, the most important buildings to risk are identified. Then, each building is evaluated further for equipment housed at each elevation/room, flood sources, propagation paths, and means of flood detection and isolation. Flood scenarios are identified for further evaluation when a potential flood is identified that can impact more than one important system.

- **Plant Walk-Through.** A walk-through is conducted to collect additional information and to confirm previous documentation and judgments on flood sources, their potential impact, propagation paths, and detection.

- **Scenario Quantification.** Based on the above, scenarios are postulated, evaluated, and quantified as initiating events with their impact on other plant systems defined. To support quantification, the flood events from Reference 3.3.8-2 were partitioned and screened based on the plant-specific design and arrangement.

- **Risk Model.** The flood scenarios are included as initiating events to the transient event tree model as described in Section 3.1.2.

3.3.8.3 Conclusions

No scenarios that lead directly to core damage were identified. Those scenarios postulated are summarized in Table 3.3.8-1. Figure 3.3.8-1 summarizes the plant-specific screening and partitioning of industry events and the applicable flood scenarios in Table 3.3.8-1. As shown, the scenarios postulated are based on potentially large floods from significant flood sources. Smaller flood sources and leaks were judged to be insignificant due to plant design features, such as the auxiliary building passive sump design, floor openings, and the location of vital electrical and mechanical equipment at higher elevations. The scenarios that were postulated are from the following significant flood sources:

- ERCW from the Tennessee River
- Refueling Water Storage Tank (RWST)
- Condensate Storage Tank (CST)
- Condenser Circulating Water from the Cooling Tower

The following conclusions provide additional insights gained from the analysis:

- Turbine building floods have a relatively high frequency based on industry experience. In addition, significant flood sources such as condenser circulation water, fire water, and raw water are in the turbine building. At Watts Bar, there is no safety equipment in the building, and normal offsite AC power supplies to emergency power are not affected by turbine building floods. Doors from the control bay open into the turbine building, and they are pressure doors, designed to prevent floods from entering. In addition, vital equipment and propagation paths to the auxiliary building are above the outside grade elevation. A large flood scenario in the turbine building is postulated that fails instrument air, feedwater, and the main condenser systems. This scenario is not expected to contribute significantly to risk.

- The auxiliary building at Watts Bar, which houses most of the vital electrical and safeguards equipment, is unique in that a passive sump is provided at the lowest elevation. The vital electrical equipment is located on the two highest elevations where severe floods are unlikely to reach and flood sources in these areas are limited. Outside the electrical areas, the building has several large openings such that floods propagate to the passive sump, which holds approximately 200,000 gallons. Safeguards pump rooms have curbs and flood alarms, and there are sump alarms at the passive sump elevation. The next elevation above the passive sump contains all residual heat removal and containment spray pumps. Scenarios that flood this elevation and fail these pumps, although unlikely, are postulated to occur but are not expected to be important contributors to risk, as auxiliary feedwater, normal feedwater, and safety injection functions are not affected.

- The control building is also an important area as it houses process racks, relays, and controls for the plant. There are fire water sprinklers in rooms and a stand pipe supply to hose reels in the stairwells. In addition, ERCW supplies air conditioning. However, the frequency is low and the size of floods in these areas are relatively small. The impact is on equipment control if not isolated. Personnel are usually present, and the likelihood of operators not maintaining safety functions is judged to be minimal. Therefore, no scenarios were postulated in this building.

- ERCW-related floods have occurred in the industry, and its failure would impact many other systems due to functional dependencies. Even the successful isolation of a major flood can result in the loss of a pump train or supply header. Several ERCW flood scenarios are postulated in the intake pumping station and auxiliary building, as these locations contain most of the ERCW system. These scenarios are not expected to be significant contributors but could be visible in the overall results.

- The RWST and CST are also major flood sources that can empty into the auxiliary building. Loss of this water also results in a common mode failure of other systems that depend on these tanks. Two scenarios are postulated (one associated with each tank), but they are not expected to be important because of the low frequency initiator and the availability of the other tank.

- Although the condenser circulating water is a very large flood source, such floods are limited to the turbine building and CCW pumping station. Floods associated with the turbine building are included in the scenario described for the turbine building.

- Fire water floods were evaluated, but no specific scenarios were postulated. The frequency and impact of fire water floods are assumed to be contained in or enveloped by the turbine building and auxiliary building scenarios. The preaction fire water system used throughout vital areas appears to be reliable with regard to the frequency of initiators and alarms, and the flow capacity is low.

3.3.8.4 References

3.3.8-1. S. M. Stoller Corporation, *Nuclear Power Experience*, Updated Monthly.

3.3.8-2. PLG, Inc., "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants," Vol. 9, Revision 0, Flood Data, PLG-0500, March 1990.

Table 3.3.8-1. Internal Flood Results				
Flood	**Annual Frequency**	**Description**	**Cause of Plant Trip**	**Plant Model Impact**
FLTB	2.0-2	Turbine Building	Loss of Feedwater	Loss of Feedwater, Condenser, and Station Air
FLPH1A	2.3-3	ERCW Strainer Room A	Loss of ERCW Header	Loss of All Four ERCW "A" Pumps and Header
FLPH1B	2.3-3	ERCW Strainer Room B	Loss of ERCW Header	Loss of All Four ERCW "B" Pumps and Header
FLAB2	4.2-6	ERCW in Auxiliary Building for 30 Minutes	Loss of ERCW Header	RHR and Containment Spray Unavailable
FLAB3C	2.8-5	CST Drained to Auxiliary Building	Reactor Trip	CST, RHR, Containment Spray; and One EAFW Pump Unavailable
FLAB3R	3.2-3	RWST Drained to Auxiliary Building	Reactor Trip	RWST, RHR, and Containment Spray Unavailable
Note: Exponential notation is indicated in abbreviated form; e.g., 2.0-2 = 2.0 × 10^{-2}.				

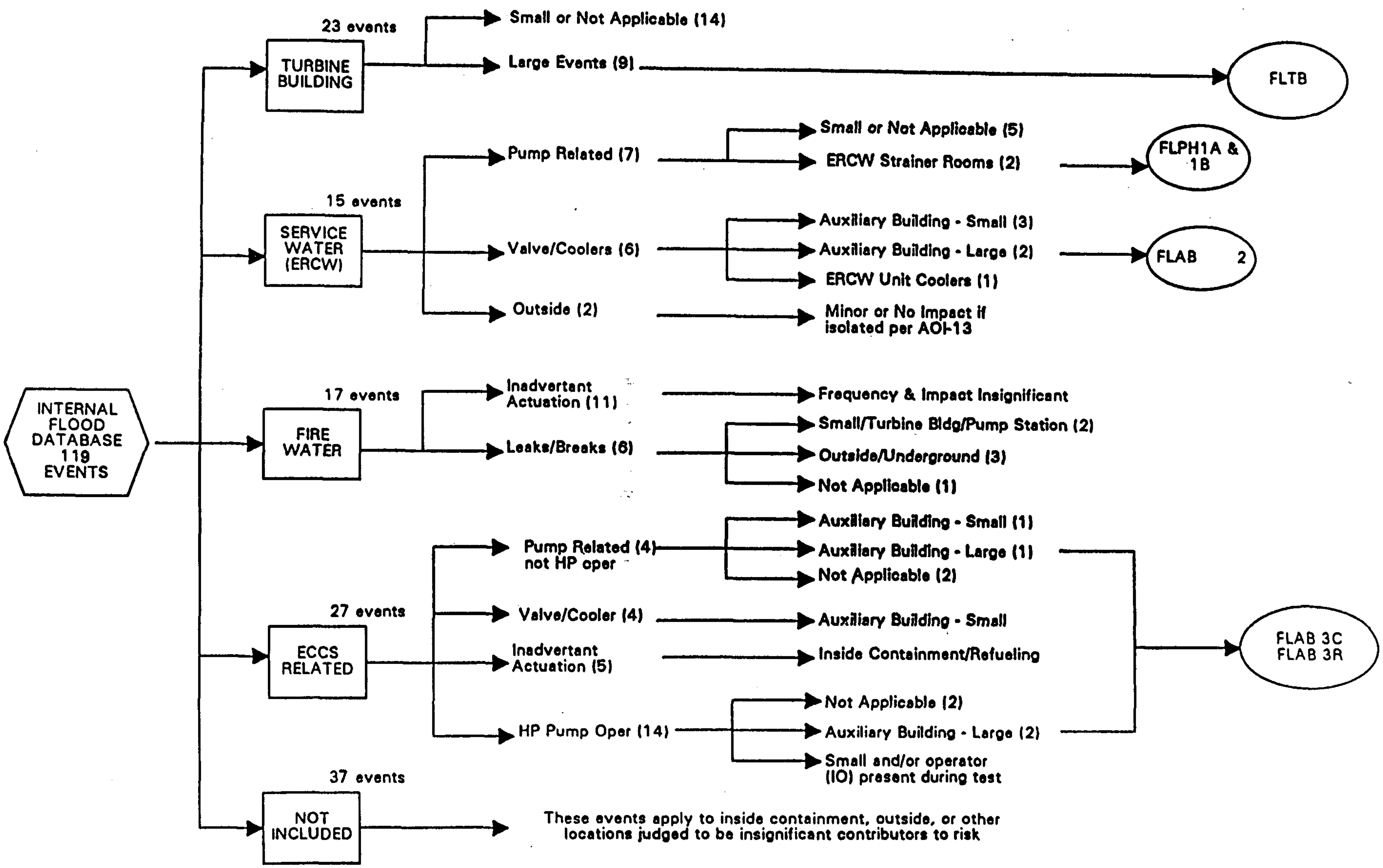


Figure 3.3.8-1. Screening and Partitioning of Flood Events at Watts Bar

3.3.9 INTERFACING SYSTEMS LOCA EVALUATION

An interfacing systems loss of coolant accident (LOCA) is initiated by failures of valves that isolate the reactor coolant system (RCS) from low pressure systems outside containment. These interfaces between the RCS and low pressure systems can be important to risk because the low pressure system can rupture, leading to a LOCA outside containment and unavailability of the same emergency core cooling systems (ECCS) that are used to mitigate the LOCA. In addition, reactor coolant can escape outside containment permitting releases to bypass the containment. This section summarizes a more detailed report that documents this evaluation (Appendix E).

The evaluation includes the identification and quantification of interfacing systems LOCA initiating events, an assessment of low pressure system failure modes and their probabilities, and an accident sequence analysis that considers operator and equipment response to these failures. The following subsections further summarize results, methodology, approach, and evaluation.

3.3.9.1 Summary and Conclusions

The residual heat removal (RHR) system was identified as the most important system outside containment that interfaces with the RCS. Besides the potential for a LOCA outside containment, overpressurizing RHR can also lead to failure of containment sump recirculation. If this occurs, it is important that the operators diagnose the event and isolate the LOCA per Emergency Instructions ECA-1.2, "LOCA Outside Containment," and/or enter ECA-1.1, "Loss of RHR Sump Recirculation," early enough to prevent core damage.

The mean frequency calculated for an initiating LOCA (> 150 gpm) through an RHR cold leg injection path (VI) and RHR suction path (VS) is as follows (see Section 3.3.9.3):

Cold Leg Injection Paths (VI) = 4.0×10^{-6} per reactor-year.

RHR Suction Path (VS) = 7.2×10^{-6} per reactor-year.

The mean annual frequencies of accident sequence end states from quantification of the event trees are summarized in the following table:

End State	Initiating Event (per reactor-year)	
	VI	VS
LOCA	4.0×10^{-6}	6.9×10^{-6}
SUCCESS	3.0×10^{-8}	2.8×10^{-7}
CD	1.0×10^{-11}	N/A
CDB	3.3×10^{-9}	3.7×10^{-8}

The end state definitions are summarized in Section 3.3.9.5. As shown in the above table, initiator VS dominates core damage with bypass results (CD and CDB). This is because the probability of pipe rupture is smaller for initiator VI. New check valves (74-544 and 74-545), added to protect each pump from opposite train operation, isolate most of the RHR system (pumps, heat exchangers, and larger piping) from the four cold leg injection paths. The probability of pipe rupture, given that the discharge relief valves successfully open [800 gpm to pressurized relief tank (PRT)], is approximately 0.05.

Other factors that contribute to the low risk calculated for core damage with bypass (CDB) includes the following:

- The RHR system is designed to 600 psig, and the materials and schedule of piping provide margins over design. For example, the probability of RHR rupture, given that the whole system is pressurized (i.e., initiator VS) and that all three relief valves are successful (1,700 gpm to PRT), is approximately 0.7.
- The total relief capacity of 1,700 gpm, which applies when the whole system is pressurized, reduces the frequency of overpressurizing RHR. If RHR is not overpressurized, there is no opportunity for system rupture and failure. Given successful valve opening, the conditional frequency of an initiating leak through two series valves that exceeds 1,700 gpm reduces the frequency of challenging RHR by approximately an order of magnitude. Similarly, the conditional frequency of exceeding 800 to 900 gpm is included in the model for cases when either a valve fails to open or only the 8-inch piping is being pressurized.

Thus, it always takes additional failures to challenge the RHR system, given an initiating event. Relief valves have to fail or the leak has to exceed the relief valve capacity, and then the probability of rupture or severe leakage is considered.

Initially, concerns were identified with the present Emergency Operating Procedures because the response of the plant and its operators may be that for a small LOCA inside containment (RHR relief to PRT). The concern is that Procedure E-O can take the operators to E-1 (this step is before the interfacing LOCA diagnosis step), thinking that there is a LOCA inside containment. Procedure E-1 is not as clear about entering ECA-1.2 for LOCA outside containment. In addition, the operators were only given a 90% chance of success. As described above, the results are low even with a 0.1 frequency of operator failure. A sensitivity case was run with no credit for the operators. The frequency of CDB for this case is 3.5×10^{-7} per reactor-year. Thus, the results are low even without the operator because the frequency of overpressurizing and rupturing the RHR is small.

3.3.9.2 Interfacing Systems LOCA Paths

Containment penetrations that connect to the RCS were screened to identify the important interfaces with systems outside containment. The screening criteria considered design pressure, pipe diameter, number of isolation valves, and the potential consequences of a LOCA outside containment. Table 3.3.9-1 documents the screening. The RHR system was identified as the most important system outside containment that interfaces with the RCS. Four RHR low pressure cold leg injection paths (each path has two check valves in series) and the shutdown cooling suction path (two sets of parallel motor-operated valves in series) were identified as the interfacing systems LOCA paths to be evaluated further. Figures 3.3.9-1 through 3.3.9-3 provide simplified drawings of the RHR system.

3.3.9.3 Initiating Event Model

Failure models were developed for the four RHR cold leg injection paths (initiator VI) and the RHR shutdown cooling suction path (initiator VS). The final expressions for failure of two valves in series are as follows:

$$VI = 4*\lambda(V1)[\lambda(V1)*T_s/2 + 2*F_d]$$

$$VS = 4*\lambda(V1)[\lambda(V1)*T_s + 2*F_d]$$

where

$\lambda(V1)$ = the valve failure frequency of exceeding 150 gpm, which is beyond charging makeup. This frequency is derived from Figure 3.3.9-4, which was developed from events in Reference 3.3.9-1 and approximately $1.0 \times 10^{+8}$ check valve hours in Reference 3.3.9-2.

T_s = time between tests (18 months).

F_d = the rupture-on-demand frequency.

3.3.9.4 RHR Overpressure Analysis

An overpressure analysis of the RHR system was performed (Reference 3.3.9-3). The pressure capacity (fragility) was analyzed for RHR piping, flanged connections, valve bonnets, heat exchangers, and pumps. Both gross rupture failure modes and leakage failure modes were evaluated with results presented as median capacities and their uncertainties, which included both modeling and material uncertainties.

The RHR system is designed to 600 psig and 400°F. Piping is Schedule 40 Type 304 stainless steel in that portion of RHR designed to 600 psig. The median rupture capacity of piping depends on pipe diameter, material, temperature, and corrosion allowance. The results ranged from 1,600 psig for 18-inch piping at 800°F and a high corrosion allowance to 7,800 psig for 3-inch piping at room temperature with no corrosion allowance.

The 18-inch piping dominates the rupture failure mode when this portion of the system is pressurized. The 14 and 12-inch piping and the RHR heat exchanger tube side cylinder failure contribute to the rupture failure mode.

The probability of the rupture failure mode ranged from 0.8 to 0.05, depending on whether the whole RHR system was pressurized or just the 8-inch discharge piping downstream of check valves 74-544 and 74-545. If the initiating event is one of the injection paths (initiator VI) and these check valves are closed, only 8-inch piping is pressurized. Thus, the larger piping, pumps, and heat exchangers are protected from the overpressure event. In addition, the success of RHR relief valves impacts whether the pressure is sustaining. Failure of relief valves to open is assumed to allow higher material temperatures to occur with high pressures. Note that, in the case of the suction path initiating event (initiator VS), the whole RHR system is pressurized.

The RHR system is equipped with the following relief valves that discharge to the PRT inside containment:

Relief Valve	Location	Capacity
63-626	8-Inch Line to Cold Leg	400 gpm at 600 psig
63-627	8-Inch Line to Cold Leg	400 gpm at 600 psig
74-505	14-Inch Suction Line	900 gpm at 450 psig

If only the 8-inch line is pressurized from the injection path initiator (VI), the leak is assumed to have to be greater than 800 gpm to overpressurize the piping unless a relief valve fails to open. If the whole system is affected by the initiator (i.e., initiator VS), the leak is assumed to have to be greater than 1,700 gpm to overpressurize the piping unless a relief valve fails to open.

The overpressure analysis provided detailed information on leakage at gasketed flange connections including valve bonnets, pump casings, and heat exchanger tube sheet flanges. Since bolt yield stresses are very high, complete failure of the bolts leading to a large leak is unlikely and neglected. However, there is a high likelihood of leakage when the RHR system is overpressurized. Gross leakage pressure (GLP) is reached at approximately reactor operating pressure for valves and pumps. GLP for the heat exchangers is about 1,000 psig. GLP is used to define the onset of gross leakage, or gross leakage pressure, as the point at which the gasket stress is equal to the pressure being retained. Leakage at this point is very small and, as the pressure increases above GLP, the leakage area increases. The RHR heat exchangers have the largest leakage approaching a small LOCA at twice GLP.

The probability and impact of leakage were treated in a simple way. The following summarizes how leakage was treated:

- Only 8-inch piping is pressurized, and both relief valves successfully open (800 gpm). The leakage is limited to only a few valves, and the RCS is depressurizing to the PRT (not sustaining). This is assumed to be insignificant leakage and has no impact on the systems outside containment or operator response. In fact, if the system does not rupture, it is treated as a LOCA inside containment.

- Only 8-inch piping is pressurized, and a relief valve fails to open (0 to 400-gpm relief). Again, the leakage is limited to only a few valves, but it is sustaining, and a small LOCA outside containment is assumed, requiring operator response.

- The whole RHR system is pressurized, and all three relief valves successfully open (1,700 gpm) or two small relief valves open (800 gpm) or the larger relief valve opens (900 gpm). Leakage now includes several valves, pumps, and heat exchangers, but the RCS is depressurizing. A small LOCA outside containment is assumed, requiring operator response.

- The whole RHR system is pressurized, and two relief valves fail to open: the larger suction side valve, and one of the two discharge reliefs (0 to 400-gpm relief).

Leakage includes several valves, pumps, and heat exchangers, but the pressure is sustaining. A large LOCA outside containment is assumed, requiring operator response.

3.3.9.5 Accident Sequence Analysis

Event sequence diagrams (ESD) were developed to document the accident sequence analysis. These ESDs were converted to event trees to quantify accident sequences. The following provides a brief summary of the event tree models for both initiators VI (injection path) and VS (suction path):

- **Secondary Isolation.** Several event tree top events model the status of normally closed secondary isolation valves. This determines whether the RHR pumps and the majority of the system are isolated from the 8-inch discharge piping and/or whether the LOCA outside containment is into the refueling water storage tank (RWST). If the whole RHR system is pressurized, this impacts the probability of rupture and leakage outside containment. If the LOCA is into the RWST and exceeds the vent capacity, the RWST will rupture.

- **Relief Valves Open.** These top events model the relief valves opening. The number of valves that can be successful depends on whether the whole RHR system is pressurized. Success means that the initiating LOCA must exceed the applicable relief capacity to pressurize the system.

- **Leak Exceeds Relief Capacity.** Based on the relief capacity from previous top events, these top events account for the conditional frequency that the initiating event leak exceeds the relief capacity. The initiating events are based on a leak through series valves exceeding 150 gpm. Success means that the leak is within the relief capacity, the system is not overpressurized, and the sequence is binned to LOCA inside containment. (Relief is to the PRT.) Operator and system responses to LOCA inside containment are assumed to be adequate. Failure means that the RHR system is overpressurized to normal RCS operating pressure.

- **RHR System Intact.** Given that the initiating leak exceeds relief valve capacity, these top events model the probability that the system does not rupture outside containment. Failure means there is a large LOCA outside containment requiring operator response. Except for one condition, success is assumed to result in a small LOCA outside containment requiring operator response. The exception is described above in the previous section: only 8-inch piping is pressurized, and the relief valves successfully open (800-gpm relief). It is assumed for this case that leakage will be small and that the sequence is binned to LOCA inside containment. Operator and system responses to LOCA inside containment are assumed to be adequate.

- **Operator Isolates LOCA.** Given a LOCA outside containment (small or large), the operators must diagnose the event and isolate the LOCA as described in ECA-1.2 before the RWST is depleted and switched over to RHR sump recirculation. (RHR pumps are assumed to be unavailable.) There is another opportunity for success if the operators depressurize and try to establish sump recirculation early enough to recognize its failure and enter ECA-1.1 to prevent core damage. ECA-1.1 instructs the operators to makeup to the RWST, preserve the RWST, and to cool down and depressurize to cold shutdown.

Flooding and environmental impacts on auxiliary feedwater and safety injection systems were judged to be minor due to the auxiliary building design. Therefore, the model neglected the failure of the reactor protection function (ATWS) and support systems because they are low frequency contributors. It is assumed that the operator response to LOCA outside containment dominates.

The following summarizes the accident sequence end states:

End State	Description
LOCA	LOCA inside containment and no LOCA outside. No operator response to LOCA outside required.
SUCCESS	LOCA outside containment and operators successful.
CD	Core damage with no containment bypass. The model allows for isolation of LOCA after RWST rupture and core damage. A second chance to isolate after core damage for other sequences is neglected.
CDB	Core damage with containment bypass. Operators fail to isolate LOCA and prevent core damage.

For the core damage end states (CD and CDB), it is assumed that the RHR and containment spray systems, located at the lowest elevation of the auxiliary building, are unavailable. In addition, it is unlikely that a LOCA outside containment (in the auxiliary building) would be covered with water; i.e., reduced source terms from scrubbing.

3.3.9.6 References

3.3.9-1. S. M. Stoller Corporation, *Nuclear Power Experience,* updated monthly.

3.3.9-2. EG&G Idaho, Inc., "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-1363, Vol. 1, June 1980.

3.3.9-3. Wesley, D. A., and H. Hadidi-Tamjed, "Pressure - Dependent Fragilities for the Watts Bar Nuclear Plant RHR System," EQE Engineering Consultants, March 1992.

Table 3.3.9-1 (Page 1 of 2). Containment Penetrations that Connect to RCS — Screening			
Penetration No.	**Description**	**Line Diameter (inches)**	**Screening**
X-15	CVCS Letdown	2	The normal open path (valves 62-73 and 62-77) is small, and valves fail closed on loss of power. Flow and impact outside containment is not significant.
X-16	CVCS Charging	3	There are two or three check valves in series inside containment, two isolation valves outside containment, and the design pressure outside is high. The line is also relatively small. This path is considered insignificant.
X-17	RHR Hot Leg Injection	12	There is a normally closed MOV and two series check valves to each of the two hot legs. These paths were neglected because this failure frequency will be much smaller than that of penetrations X-20A and X-20B which contain only two check valves in series and will dominate the total failure frequency.
X-20A	RHR Injection B	8	Inside containment are two check valves in series to two cold legs. A normally open MOV is outside containment with a low pressure design. Retained as interfacing LOCA path.
X-20B	RHR Injection A	8	Inside containment are two check valves in series to two cold legs. A normally open MOV is outside containment with a low pressure design. Retained as interfacing LOCA path.
X-21	Safety Injection Hot Leg Injection B	4	Normally closed MOV outside containment and two series check valves to two hot legs inside containment. These paths were neglected because penetrations X-20A and X-20B that contain only two check valves in series will dominate the total failure frequency, and the safety injection design pressure is higher with smaller lines.
X-22	Charging Injection	3	Two check valves inside containment. Normally closed MOVs outside and normally operating high pressure system. This path is considered insignificant.

Table 3.3.9-1 (Page 2 of 2). Containment Penetrations that Connect to RCS — Screening

Penetration No.	Description	Line Diameter (inches)	Screening
X-32	Safety Injection Hot Leg Injection A	4	Normally closed MOV outside containment and two series check valves to two hot legs inside containment. These paths were neglected because penetrations X-20A and X-20B that contain only two check valves in series will dominate the total failure frequency, and the safety injection design pressure is higher with smaller lines.
X-33	Safety Injection	4	Two check valves in series to four cold legs, and lines are small. A normally open MOV outside containment and high pressure design. These paths were neglected because the failure frequency of penetrations X-20A and X-20B will dominate.
X-43A	CVCS/RCP Seal Injection	2	There are two check valves in series inside containment, and the lines are small. There are manual valves outside containment, and the design pressure is high. This path is considered insignificant.
X-43B	CVCS/RCP Seal Injection	2	There are two check valves in series inside containment, and the lines are small. There are manual valves outside containment, and the design pressure is high. This path is considered insignificant.
X-43C	CVCS/RCP Seal Injection	2	There are two check valves in series inside containment, and the lines are small. There are manual valves outside containment, and the design pressure is high. This path is considered insignificant.
X-43D	CVCS/RCP Seal Injection	2	There are two check valves in series inside containment, and the lines are small. There are manual valves outside containment, and the design pressure is high. This path is considered insignificant.
X-44	CVCS/Seal Water Return	4	Seal leakoff lines are small (~ 1 inch) and the MOV inside containment can be closed. The design pressure outside containment is low (200 psig) after another MOV. An RCP seal LOCA is required and is modeled in the Level 1 analysis.
X-107	RHR Supply	14	Four combinations of two normally closed MOVs in series inside containment from hot leg 4. Retained as an interfacing LOCA path.

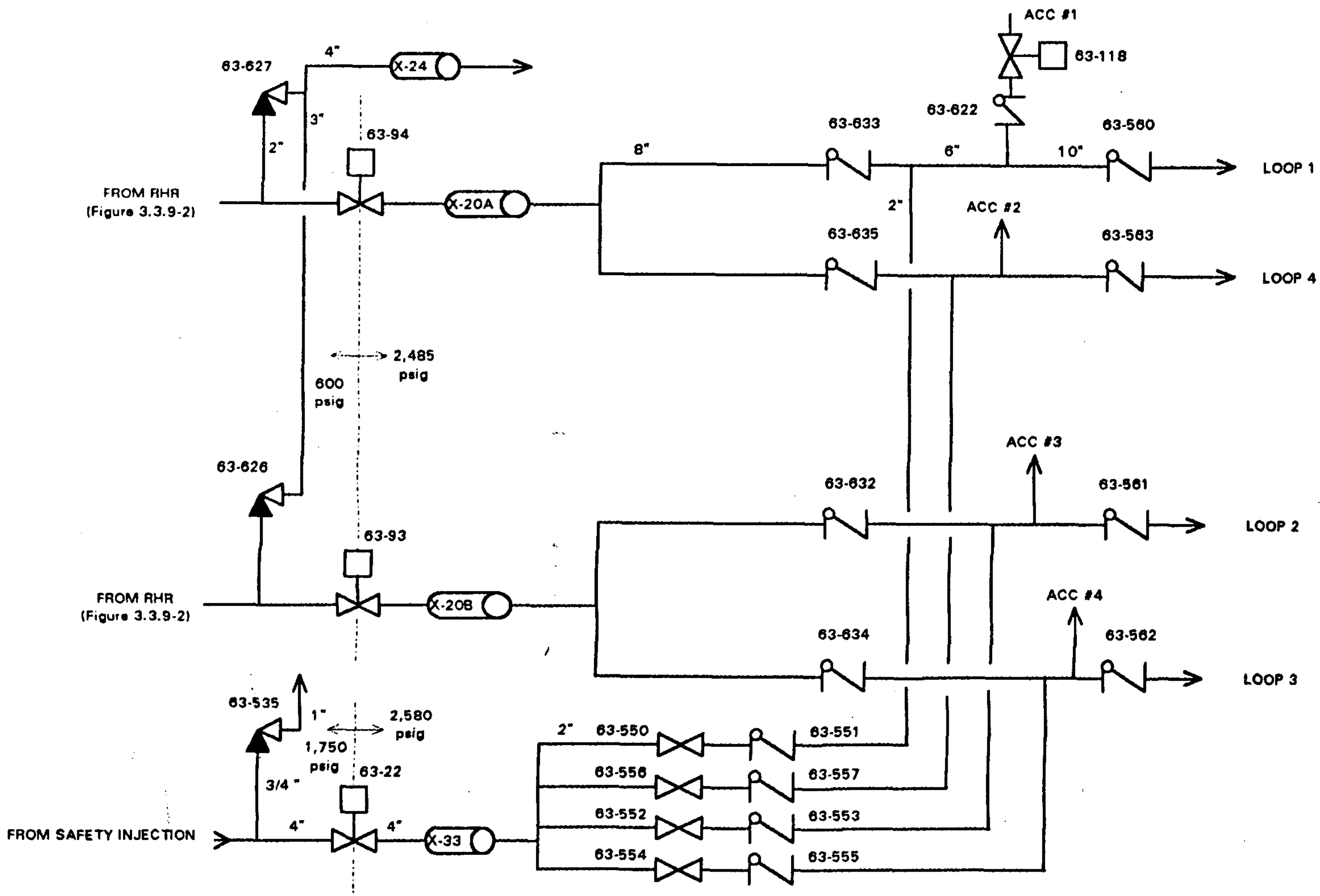


Figure 3.3.9-1. Cold Leg Injection from RHR and Safety Injection

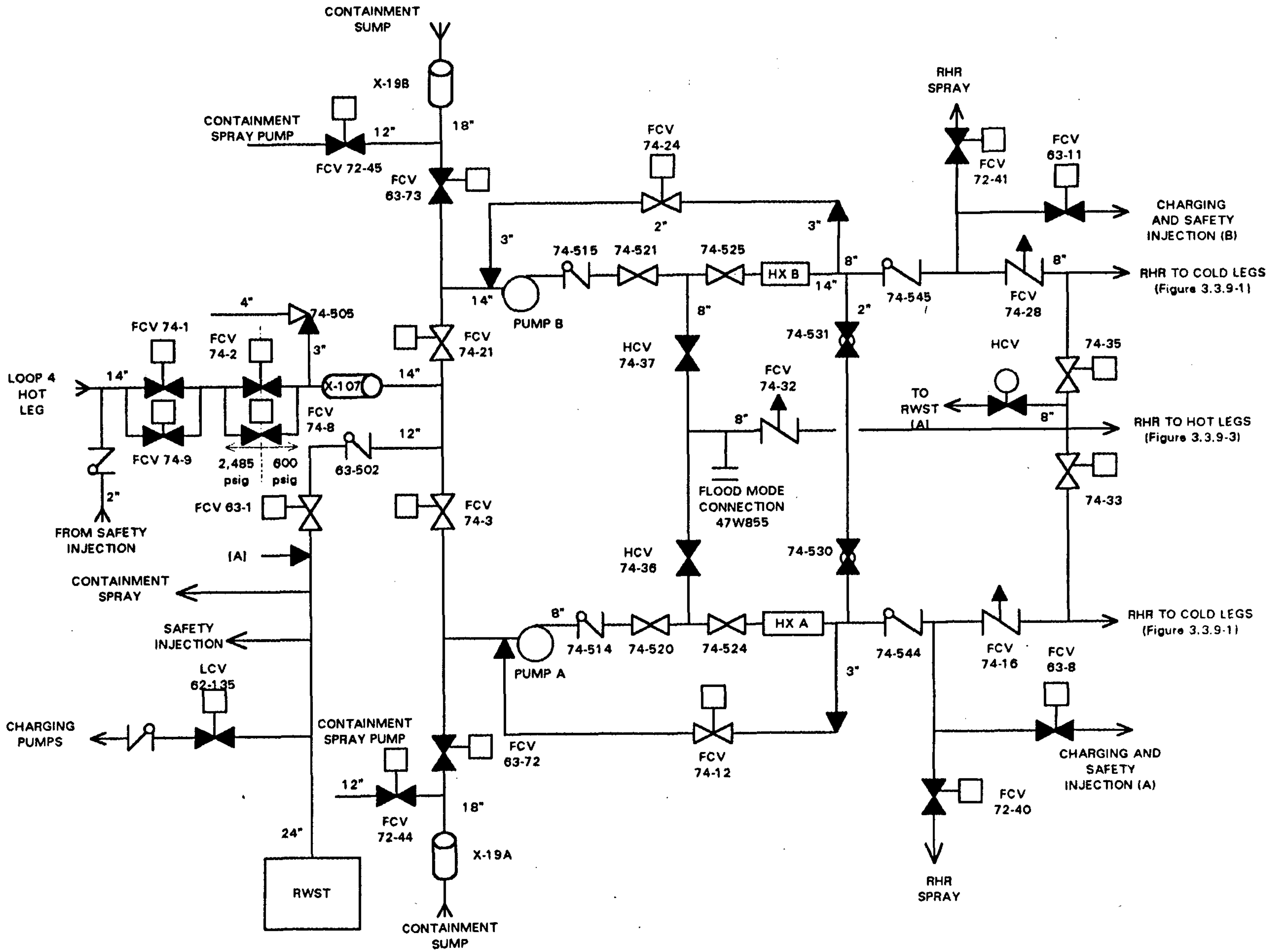


Figure 3.3.9-2. RHR Suction Path and Pumps

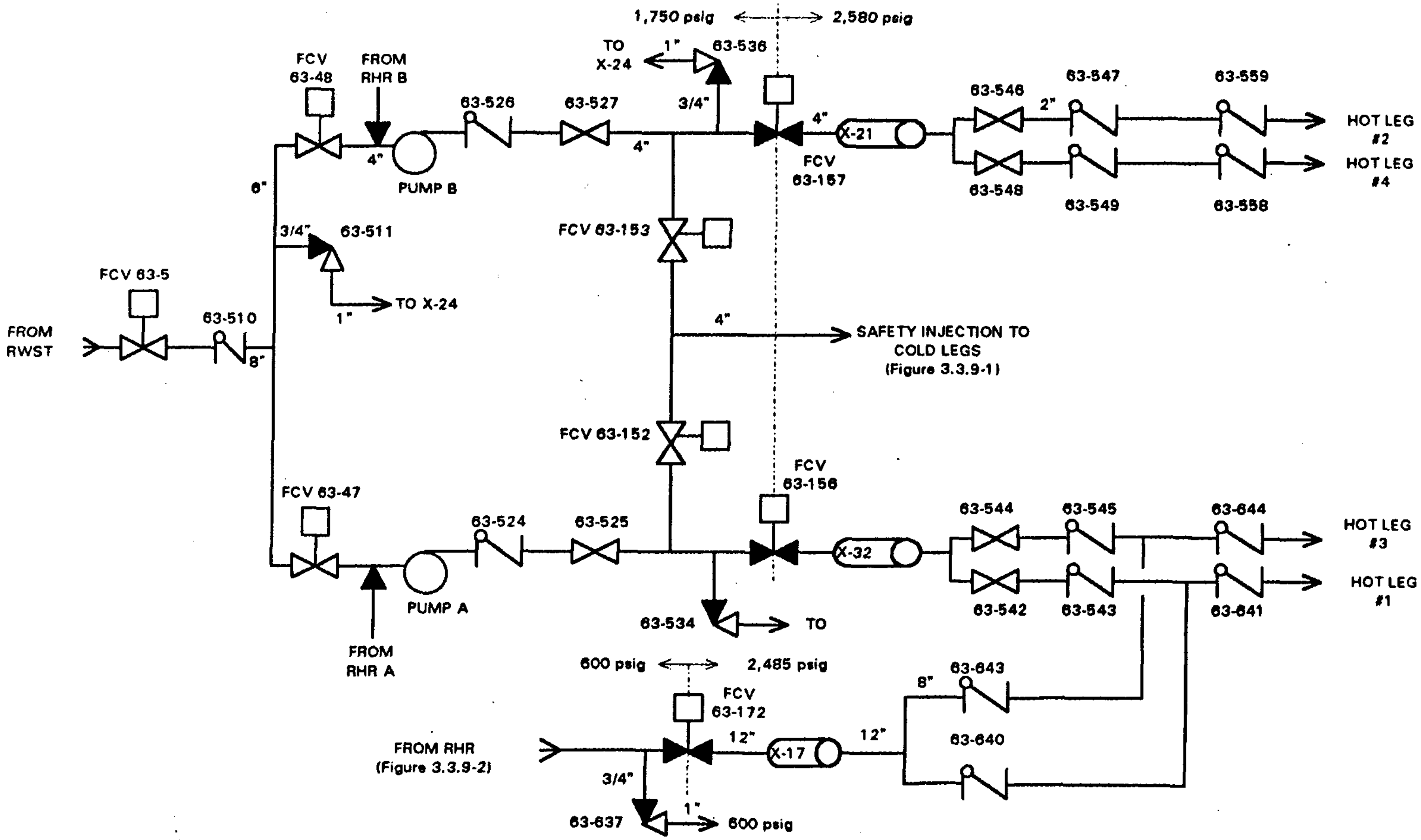


Figure 3.3.9-3. Hot Leg Injection from Safety Injection and RHR

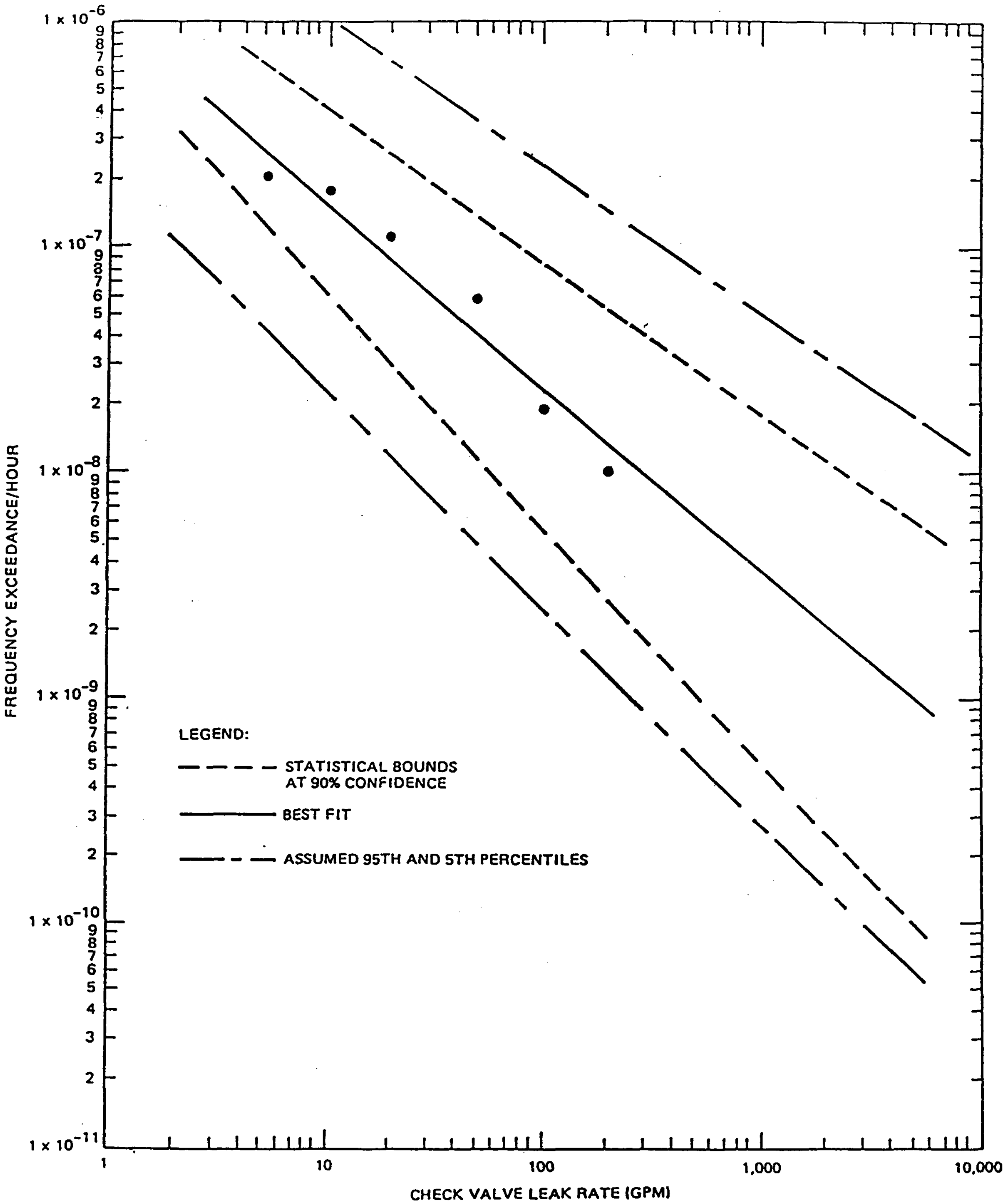


Figure 3.3.9-4. Frequency of Check Valve Leakage Events

3.4 RESULTS AND SCREENING PROCESS

3.4.1 OVERVIEW OF RESULTS AND CONTRIBUTORS

This section presents the results of the Watts Bar Nuclear Plant probabilistic risk assessment (PRA). The plant sequence model includes the responses of all support and frontline systems that are important for determining the core damage frequency and the frequency of all plant damage states, as defined in Section 4.3. The Level 2 analysis to determine the containment response to severe core damage sequences is reported in Section 4.4. The plant model results include contributions from internal initiating events and internal floods.

The total core damage frequency computed for Watts Bar Nuclear Plant is 3.3×10^{-4} per reactor-year. This value is the mean of the uncertainty distribution for Watts Bar Nuclear Plant Unit 1, which is presented in Figure 3.4-1.

The results from the current plant model quantification may be examined in numerous ways. One way to examine the results is at the plant damage state level. Table 3.4-1 presents the frequency of plant damage states that define different categories of core damage scenarios. Sixty-five plant damage states make up the total core damage frequency, where only 24 plant damage states have appreciable frequency. The definitions of the plant damage states are provided in Section 4.3. The logic that is used to assign each sequence to a particular plant damage state is also discussed in Section 4.3.

The plant model quantification results may also be provided by initiating event category. Figure 3.4-2 shows the frequency of core damage that is attributable to sequences grouped by initiating events. The most important initiator is one of the support system faults, the total loss of the component cooling system (CCSTL). This single group accounts for 17.4% of the core damage frequency. The loss of CCS Train A only initiator adds another 10%. Accident sequences that are initiated by loss of offsite power (LOSP) are the second largest group, with 16.6% of the core damage frequency. The top 5 initiators comprise nearly 60% of the total core damage frequency. The results of the plant model quantification will be interpreted further in Section 3.4.3 on vulnerability screening.

Perhaps the most informative way to look at the results is to examine the individual sequences that lead to core damage. A detailed discussion of the top 10 sequences is provided below.

- **Sequence 1 — Loss of CCS with Failure to Trip RCPs.** The highest frequency core damage sequence begins with a total loss of both trains of the component cooling system (CCS). This system failure results in a loss of cooling to the reactor coolant pump (RCP) thermal barrier heat exchangers, and to the upper and lower motor bearings. The loss of motor bearing cooling means that after a period of time with continued pump operation, loss of effective lubrication leads to pump vibration and eventual failure of the bearings. The vibration of the motor and shaft would be transmitted to the RCP seals, which are assumed to fail after a period of time, estimated to be at least 2 minutes, and realistically, 10 minutes is assumed.

The operators are directed by procedures to trip the reactor and the RCPs in the event of no CCS flow to the RCP oil coolers; however, there could be some hesitation in this sequence since their action will result in tripping the unit, and the operators would first try to recover from the loss of all CCS cooling.

Failure to trip the RCPs prior to seal damage is assumed to lead to a small loss of coolant accident (LOCA). The loss of all CCS cooling means that oil cooling for the centrifugal charging pumps and the safety injection pumps is also lost. However, the operators successfully stop the initially running train A charging pump, align essential raw cooling water (ERCW) cooling to its oil coolers, and restart the pump. Charging pump A then provides high pressure injection to make up for the loss of inventory through the damaged RCP seals.

The residual heat removal (RHR) heat exchangers, however, are unavailable for recirculation from the sump due to the loss of CCS cooling. The RHR pumps would also be stopped once recirculation from the sump is required due to the loss of pump seal cooling. Continued operation of the RHR pumps taking suction from the hot water in the sump, without seal cooling, could lead to a LOCA outside containment as a result of an eventual failure of the RHR pump seals.

Therefore, recirculation from the containment sump is unavailable. The loss of sump recirculation leads to eventual core damage due to the loss of inventory control.

The containment spray pumps are also unavailable in this first sequence for recirculation from the containment sump due to the loss of lube oil cooling. The spray pumps may run for a period of time in the injection mode, while taking suction from the cooler refueling water storage tank (RWST) inventory. Operation of the spray pumps in the injection mode would reduce containment pressure but would also shorten the time to empty the RWST.

- **Sequence 2 — Loss of ERCW.** This sequence is initiated by a total loss of all ERCW cooling; i.e., inadequate ERCW flow to both trains A and B for both units. The frequency for loss of all ERCW is derived from combinations of failures involving pumps failing to run, failing to start, and being in maintenance, and of check valves failing to reseat on demand, which leads to a diversion of flow from other, operable pumps.

  Loss of all ERCW, which provides the safety grade ultimate heat sink at Watts Bar, means that CCS cooling is also unavailable. This means that all emergency core cooling system (ECCS) pumps and all RCP seal cooling would be lost. Consequently, core damage is assumed to result eventually due to an RCP seal LOCA without any injection from the RWST. Auxiliary feedwater (AFW) is still available because the turbine-driven AFW pump ventilation is not dependent on ERCW or CCS.

- **Sequence 3 — Loss of CCS with Failure To Trip RCPs or Makeup to RWST.** This sequence is similar to Sequence 1. The only difference is that in this sequence, the operators also fail to attempt to provide make up to RWST for continued high pressure injection. Since containment spray is unavailable to provide make up to the RWST from the containment sump anyway, the added operator failure has no additional impact. No credit for makeup to the RWST via the VCT is assumed as an alternative to recirculation from the sump.

- **Sequence 4 — Loss of All CCS with Failure To Align CCP to ERCW.** This sequence is a variation on the top sequence. In this case, there is also a total loss of all CCS cooling. Unlike the top-ranked sequence, in this sequence, the operators successfully trip the RCPs in time to prevent RCP seal damage due to motor bearing failure. However, the operators do not align ERCW cooling to the centrifugal charging pump for continued RCP seal injection. Therefore, longer term damage to the RCP seals occurs due to loss of all seal cooling. Since none of the high pressure injection pumps have lube oil cooling, core damage results from the seal LOCA without high pressure injection. Again for the sequence, both the RHR and containment spray pumps are also unavailable.

- **Sequence 5 — Loss of Offsite Power Resulting in Unit Blackout.** This sequence is initiated by a loss of offsite power. The Unit 1 onsite diesel generators 1A-A and 1B-B both fail to start. The turbine-driven AFW pump operates successfully so that secondary heat removal is successful. However, the loss of all shutdown power on Unit 1 and a consequential loss of all RCP seal cooling lead to an RCP seal LOCA. Electric power from offsite or from onsite is not recovered before the loss of RCS inventory out the failed RCP seals leads to core uncovery. Core damage then occurs. In the current evaluation of this sequence, no credit was given for aligning the fifth, or C-S, diesel generator to Unit 1.

- **Sequence 6 — Small LOCA with Failure of High Pressure Recirculation.** This sequence is initiated by a small LOCA that is assumed to occur at the RCP seals of one pump. The plant trips, and a safety injection signal is generated. Both the charging and the safety injection pumps actuate to provide RCS inventory control at high pressure. Containment spray pumps come on in response to a high-high containment pressure condition. Automatic swapover of RHR suction to the containment sump is successful, but the operators are postulated in this sequence to fail to align for high pressure recirculation; i.e., the discharge of the RHR pumps is not aligned to the suction of the charging or safety injection pumps. Core damage results because of the loss of inventory control while in recirculation.

  In the Level 2 analysis, this sequence was evaluated using the Modular Accident Analysis Program (MAAP) (Reference 3.4-1) thermal-hydraulic analysis program. Given a successful cooldown by the operators (i.e., as directed by their post-LOCA cooldown procedures), MAAP shows that core damage would not occur due to failure of high pressure recirculation, provided that the RHR pumps operate for low pressure recirculation from the sump. The accumulator inventory keeps the core covered while the RCS cool downs and depressurizes sufficiently for low pressure recirculation. This sequence is assumed to be arrested within the vessel for the Level 2 analysis.

- **Sequence 7 — Loss of All CCS with Failure to Align CCP to ERCW or to Provide Makeup to the RWST.** This sequence is similar to Sequence 4. The only difference is that in this sequence, the operators also fail to attempt to provide makeup to the RWST for continued high pressure injection. Since containment spray is unavailable to provide makeup to the RWST from the containment sump anyway, this added operator failure has no additional impact. No credit for makeup to the RWST via the volume control tank (VCT) is assumed as an alternative to recirculation from the sump.

- **Sequence 8 — Loss of CCS Train A with Failure to Trip RCP and Failure in Recirculation.** This sequence is initiated by a loss of train A of component cooling water with continued operation of train B. Train A of CCS provides the cooling for the RCP thermal barrier heat exchangers and for the RCP motor bearing coolers. Similar to the top sequence, the operators fail to trip the RCPs in time to prevent seal damage due to pump vibration. A small LOCA is assumed to develop. Since train B of CCS is still available, in this sequence, train B of the charging and the safety injection pumps operate for RCS inventory control until the RWST empties. Containment spray pump train B actuates when containment pressure reaches the high-high containment pressure setpoint, and then continues to provide containment heat removal in the recirculation mode. The RHR pump train A is unavailable due to the loss of CCS train A. RHR pump B fails independently. Loss of both RHR pump trains results in a failure of core cooling during recirculation.

- **Sequence 9 — Small LOCA with Failure of Both High Pressure Recirculation and Makeup to the RWST.** This sequence is similar to Sequence 6. The only difference is that in this sequence, the operators also fail to attempt to provide makeup to the RWST for continued high pressure injection. Since containment spray is unavailable to provide makeup to the RWST from the containment sump anyway, this added operator failure has no additional impact. No credit for makeup to the RWST via the VCT is assumed as an alternative to recirculation from the sump.

- **Sequence 10 — Loss of CCS with Failure of Centrifugal Charging Pump 1A-A.** This sequence is initiated by a total loss of all (i.e., both trains of) component cooling water. The operators successfully trip the RCPs in time to prevent early RCP seal failure due to pump vibration. Charging pump A (i.e., the only pump that can currently be aligned to ERCW for backup lube oil cooling independent of CCS) fails independently. Therefore, due to loss of CCS, none of the charging or safety injection pumps are available. The loss of both RCP thermal barrier cooling and of seal injection leads to eventual seal failure. Seal failure results in a small LOCA. The failure of all high pressure injection pumps then leads to core damage with inventory loss through the RCP seals. Neither the RHR pumps nor the containment spray pumps are available for long-term containment heat removal due to the loss of CCS.

3.4.2 APPLICATION OF GENERIC LETTER SCREENING CRITERIA

The U.S. Nuclear Regulatory Commission (NRC) sequence-reporting requirements for the purpose of fulfilling the individual plant examination requirements are discussed in Reference 3.4-2. The Watts Bar PRA plant model provides the results in terms of systemic sequences as opposed to functional sequences. The reporting guidelines for systemic sequences are as follows:

1. Any systemic sequence that contributes 1×10^{-7} or more per reactor-year to core damage.

2. All systemic sequences within the upper 95% of the total core damage frequency.

3. All systemic sequences within the upper 95% of the total containment failure frequency.

4. Systemic sequences that contribute to a containment bypass frequency in excess of 1×10^{-8} per reactor-year.

5. Any other systemic sequence that the utility determines to be important to core damage frequency or to poor containment performance.

The NRC sequence-reporting guidance states that the total number of most significant sequences to be reported should not exceed 100. The accident analysis is also to be limited to sequences initiated from power operation and from hot standby; events that are initiated from cold shutdown or during refueling are specifically excluded. Events that are both initiated from power operation and from hot standby are included in the model and therefore are considered for inclusion in the list of key sequences reported. The NRC reporting guidelines specify that the mean frequency be reported for each sequence. Use of both the mean initiating event category frequencies and the mean values from the system unavailabilities when quantifying each sequence provides a very close approximation to the mean sequence frequencies. In fact, for most of the sequences, these approximate sequence frequencies are equal to the mean. These frequencies are judged to be suitable for reporting here. Monte Carlo error propagation is used to report the complete uncertainty distribution for the total core melt frequency.

The approach that is used to quantify sequences, as described in Section 3.3.7, enables the PRA team to examine any number of the highest frequency sequences down to any frequency cutoff.

Table 3.4-2 presents the 100 highest frequency sequences contributing to the total core damage frequency. This list accounts for sequences whose individual frequency is greater than about 4.3×10^{-7} per reactor-year. The sequences in Table 3.4-2 are presented in terms of the initiating event category, the event failures that occur with frequency less than 1.0, the guaranteed event failures that occur with frequency 1.0 because they are dependent on other events that have failed, and the Level 1 end state or plant damage state to which the sequence belongs. The individual sequence frequency and its percentage contribution to the total core damage frequency are also provided. The top 100 sequences account for more than 60% of the total core damage frequency. The remaining sequences contribute less than 40% of the total core damage frequency.

The front-end analysis for Watts Bar Nuclear Plant includes consideration of containment bypass events from steam generator tube ruptures and interfacing loss of coolant accident (LOCA) initiators. The highest frequency core damage sequences from these initiators are also listed in Table 3.4-2. A trailing "B" or "V" in the plant damage state identifier indicates that the sequence leads to small or large containment bypass paths, respectively. An unisolated containment sequence is identified by a trailing "S" or "L" for small and large containment openings.

Reporting guideline 3 above requests that key sequences contributing to the total containment failure frequency be presented. The back-end analysis is documented in Section 4, with the back-end results provided in Section 4.10.

3.4.3 VULNERABILITY SCREENING

Section 3.4.1 provided a look at the plant model results by examining the key sequences to the core damage frequency. This section interprets the results by examining the contributors that are found in many sequences from several vantage points.

TVA has adopted two sets of criteria for identifying vulnerabilities: one set is based on the results of core damage frequency that are used to evaluate potential vulnerabilities in the systems that protect the reactor core integrity. The second set is based on the results for large early release frequency that are used to evaluate vulnerabilities from the point of view of containment integrity. Each set includes criteria for the numerical results, how the results are distributed across the underlying contributors and the availability of cost effective ways to reduce core damage or large early release frequency.

A vulnerability may exist if the mean core damage frequency exceeds 5×10^{-4} per reactor-year or the mean large early release frequency exceeds 5×10^{-5} per reactor-year. Several PWR plants evaluated using similar PRA data and methods have been reported to the NRC total core damage frequencies in the range of 5×10^{-5} to 5×10^{-4} per reactor-year. These results seem to be typical for modern nuclear power plants in the U.S. For the large early release criteria, some additional margin below total core damage frequency is believed appropriate. TVA has chosen a factor of 10 benefit for the containment as a suitable basis for identifying a vulnerability. Therefore, the criteria for large early release is a factor of 10 below the core damage criteria, or 5×10^{-5} per reactor-year.

Given an exceedance of either of these criteria, a vulnerability is identified, only if a common function, system, operator action, or other common element can be identified which contributes substantially to the total frequency. More than one vulnerability may then be identified. Alternatively, none may be identified if the frequency is well balanced and made up of many different and individually small contributions. Identified vulnerabilities are then to be evaluated for availability of cost effective enhancements.

The occurrence of a vulnerability is therefore based on the total core damage frequency or the early release frequency. If a vulnerability exists, then the specific plant design or operating feature defined as the vulnerability is that which contributes in a substantial way to the frequency criteria being exceeded. To be unique to Watts Bar, the vulnerability must be either a contributor not seen in PRAs for other plants or one that makes a disproportionately high frequency contribution.

3.4.3.1 Event or System Importance

Another perspective of the underlying contributors to risk can be gained by evaluating various importance measures of the individual event tree branch point probabilities, or split fractions, that are evaluated in this study. One importance measure often used is computed by determining the percentage contribution to the total core damage frequency made by all sequences grouped by common failed split fractions. This is in contrast to the look at individual sequences in the previous section.

The accident sequence model contains two types of split fractions: guaranteed failure (GF) split fractions, whose failure frequency is set to equal 1.0 because of functional dependencies on other equipment or operator actions that has already failed in the same accident sequence, and nonguaranteed failure (NGF) split fractions; i.e., those whose split fraction values are other than 1.0.

All of the split fractions for a particular top event can be grouped into one of these two categories. The importance rankings for these groups of split fractions are evaluated separately because the evaluation of each group has different risk management implications. The importance of the highest ranked top events for each group of split fractions is described below.

The risk contribution from guaranteed failed split fractions results from the dependencies between systems and between multiple operator actions; i.e., if the first event fails, the second is then guaranteed to occur. The risk contribution of guaranteed failed split fractions is not associated with the reliability characteristics of the associated system. To reduce or eliminate the importance of these split fractions, it is necessary to attack the dependencies of the important system on the other systems whose failure triggered the guaranteed value. The most important guaranteed failed split fractions are summarized in Table 3.4-3.

The first six split fractions in Table 3.4-3 represent switches in the event trees. They do not involve system failures.

The highest ranked system top event for the importance of guaranteed failed split fractions is Top Event TB. Top Event TB tracks the status of reactor coolant pump (RCP) thermal barrier cooling. Cooling to the RCP thermal barriers is isolated, given a Phase B signal, which is assumed to occur on every LOCA modeled. Thermal barrier cooling is also lost, given failure of train A of component cooling water (CCS). We have already seen that failure of CCS cooling is an important contributor to the core damage frequency. Most of the high-ranking guaranteed failure split fractions reflect a dependence on CCS cooling.

The importance evaluation of the nonguaranteed failure split fractions is summarized in Table 3.4-4. For these split fractions, it is possible to change the core damage frequency by changing the reliability characteristics of the associated system. For this group of split fractions, four different importance measures are used: the percentage contribution of the sequences with that split fraction failed, the factor increase in the core damage frequency when the split fraction is arbitrarily reassigned a value of 1.0, the factor decrease in the core damage frequency when the split fraction is arbitrarily reassigned to a value of 0.0, and the change in core damage frequency per unit change of the split fraction value. These four importance measures are termed importance, risk achievement worth, risk reduction worth, and the core damage frequency derivative. Each of the measures is presented in Table 3.4-4, along with the split fraction value used in the event tree quantification and the frequency of all core damage sequences that involve failure of the split fraction.

The highest ranking nonguaranteed failure split fraction to importance (i.e., by percentage contribution to the total core damage frequency) is split fraction MU4 at 24%. This split fraction models the alignment of makeup to the refueling water storage tank (RWST) for continued high pressure injection in the event of a small LOCA with failure of recirculation from the containment sump. In the current model, this action is only credited if one or both of the containment spray pumps are available to provide the makeup, although it is asked for all sequences; i.e., sometimes when it failed, it would not have been credited anyway due to the failure of containment spray recirculation. Therefore, for this split fraction, a better measure of the significance of this split fraction is provided by the risk achievement worth, which is small. This split fraction has only a marginal impact on core damage frequency, as determined by the risk achievement worth.

The second-ranked nonguaranteed failed split fraction to importance is SED at 21%. This split fraction models the operator action to trip the reactor coolant pumps (RCP) in response to a loss of CCS train A cooling to the RCP upper and lower motor bearing coolers. Failure of a timely trip of the RCPs under these conditions is modeled as resulting in a small LOCA. As we saw from the presentation of sequences in the previous section, this sequence is an important contributor to the total core damage frequency.

The third-ranked nonguaranteed failed split fraction to importance is OGR11 at 15%. This split fraction evaluates the fraction of offsite power losses that are not recoverable within 1 hour. Failure to recover offsite power can result in core damage during unit blackout sequences. Longer term recovery of electric power, but still before core damage, is considered separately in Top Event REC.

The fourth-ranked nonguaranteed failed split fraction to importance is TPR1 at 10%. This split fraction contains the operator action to restart the turbine-driven auxiliary feedwater (AFW) pump in the event of an initial failure to start the pump. It only appears in sequences in which Top Event TP has failed; i.e., where the turbine-driven AFW pump failed to start initially. Failure of Top Event TP, along with failure of TPR1, means that the turbine-driven AFW pump is unavailable.

The fifth-ranked nonguaranteed failed split fraction to importance is GA1, with an importance of 10%. This split fraction models the start and run of the Unit 1 diesel generator 1A-A under loss of offsite power conditions. The failure of this split fraction, in combination with failure of the 1B-B diesel generator, results in a unit blackout.

The sixth-ranked nonguaranteed failed split fraction to importance is RT1 at 10%. This split fraction models reactor trip under the condition when both trip signals from solid state protection system (SSPS) are available. It reflects the contribution from all anticipated transient without scram (ATWS) events to the total core damage frequency. It has an especially high risk achievement worth due to the relatively high reliability of the system.

The seventh-ranked split fraction models the failure to initially start the turbine-driven AFW pump. As might be expected, there is a greater fraction of core damage sequences that involve failure of both the turbine-driven AFW pump to start and not be recovered (i.e. TPR1 fails), than to initially fail to start and then be upgraded; i.e., TP1 fails and TPR1 is successful.

The eighth-ranked nonguaranteed failed split fraction to importance is PL1 at 8%. This split fraction determines the fraction of plant trips occurring from an initial reactor power level above 40%. Its importance is nearly equal to that for RT1, indicating that most of the ATWS sequences resulting in core damage are initiated from power levels above 40%.

The ninth-ranked nonguaranteed failed split fraction to importance is CCPR1 at 8%. This split fraction models the operator action to align cooling water to charging pump A from essential raw cooling water (ERCW), given a loss of train A cooling water from CCS. This action allows continued RCP seal injection, provided that the other support systems to run the charging pump are available. In the current design, only the train A charging pump on Unit 1 can be aligned to ERCW for lube oil cooling.

The tenth-ranked nonguaranteed failed split fraction to importance is GB2 at 8%. This split fraction models the start and run of Unit 1 diesel generator 1B-B, given a loss of offsite power condition with diesel generator 1A-A also failed. GB2 is applicable only for those sequences in which train A diesel generator has also failed. The importance of all of the split fractions for train B (i.e., GB2 and GB1) sum to nearly the same importance as that for GA1; i.e., to 9%. The difference between this sum and the importance for GA1 reflects the small added importance of train A relative to train B under loss of offsite power conditions.

Finally, the eleventh-ranked nonguaranteed failed split fraction to importance is RR1 at 6%. This action models the operator action of selected valves needed to align residual heat removal (RHR) pump discharge to charging and safety injection pump suction for high pressure recirculation from the containment sump, given a LOCA in which the RHR pumps have successfully swapped over to the containment sump for suction. Failure to align for high pressure recirculation in the event of a small or medium LOCA is currently assumed to result in core damage in the Level 1 model.

Table 3.4-5, Most Important Nonguaranteed Failed Top Events, provides an importance ranking of key event tree top events in the Level 1 plant model event trees. All top events with an importance of 1% or greater are included in the table. Because each top event contains a number of different split fractions, this approach is a more general way to examine groups of sequences. The top events are ranked according to their percentage to the core damage frequency involving sequences that include failures of these top events. Sometimes the equipment represented by these events may be unavailable to perform their function only because of failures of the equipment in systems that support them. Sequences that involve such guaranteed failures of the top events are not counted in the ranking in Table 3.4-5. Table 3.4-6 provides an importance ranking of these same event tree top events, but ranked by total importance, i.e., probabilistic failure frequency plus guaranteed event failures.

The highest ranked top events are related to the highest ranked nonguaranteed failed split fractions presented earlier in Table 3.4-4. The reader will note that the key split fractions presented in Table 3.4-4 account for large proportions of the total top event importance.

In addition to the system and event importance just discussed, an importance ranking of individual operator action events is provided in Table 3.4-7. Operator actions are modeled via the top events in the event trees. They may appear as single failure modes of a top event, or in combination with the plant hardware manipulated to achieve the top event's function. The importance to the core damage frequency of each top event split fraction containing an operator action can be determined from Table 3.4-4. Only a portion of the split fraction importance may be attributed to the operator action failure mode. The importance of an operator action to core damage frequency is obtained by multiplying the split fraction importance by the fraction of the split fraction frequency caused by failure of the operator action. When a single operator action is represented by multiple split fractions, the total operator action importance is obtained by summing the importance of each split fraction in which it is included. The important operator actions are generally found in the same split fractions that were ranked as having the highest importance in Table 3.4-4.

As noted in the discussion of split fraction MU4, the importance evaluated for the operator action to provide makeup to the RWST inventory, given a LOCA with loss of recirculation (i.e., HAMU2), is somewhat misleading. The current model takes credit only for makeup to the RWST during a LOCA, if one or both of the containment spray pump trains is available during recirculation; i.e., the spray pumps then refill the RWST from the containment sump via a test line. However, the action to provide makeup to the RWST is asked in the support trees for every sequence. Therefore, MU4 appears in sequences as failed even though it did not contribute to the sequence progression. Its importance, as reported in Table 3.4-7, has been adjusted to reflect the importance only in sequences in which it contributes to core damage. The other actions in Table 3.4-7 do not have this problem, and therefore the importance listed for them are realistic.

3.4.3.2 Sensitivity Cases

Another way of evaluating the contributors to risk is by examining the sensitivity of results to general classes of events. For the Level 1 models, the sensitivity or importance of various groups of events can be determined by reviewing individually the sequences that contribute the most to core damage in a manner similar to the calculation used to compute the importance measures for individual events and systems as presented in the previous section. Alternatively, the sensitivity of various changes to the base models may be computed directly by requantifying all of the plant model event trees and comparing the results to the base case results. For most applications, reviewing the contribution of various event or system groups to the important sequences in the base case is sufficient.

3.4.3.2.1 Core Damage Sensitivity to Systems

Table 3.4-8 presents the contribution of various groups of events to the total core damage frequency. RCP seal failure sequences contribute a very large part of the total core damage frequency; i.e., about 70% of the total. A breakdown of the RCP seal failure sequences is also given in the table. Unit blackouts resulting in core damage with an accompanying seal failure due to loss of all seal cooling make up 9% of the core damage frequency. Use of the C-S diesel generator could reduce the core damage frequency by about 9%. No credit for the C-S diesel generator was assumed in the current analysis.

The other 61% of the RCP seal failures occur with at least one train of shutdown power available. Sequences in this group involve failures of either CCS or ERCW, which lead to failure of CCS, and either a loss of seal injection or of the operators failing to trip the RCPs.

Other groups of contributors can also be seen in Table 3.4-8. Total losses of CCS (i.e., in which both trains of Unit 1 CCS fail make up 38% of the total core damage frequency. The seal LOCAs with power available to at least one shutdown board, makes up 19% of the total core damage frequency. ATWS sequences make up less than 10% of the total core damage frequency. LOCAs involving leakage through the pressurizer power-operated relief valves (PORV), including leakage as an initiating event or failures in response to another plant transient resulting in pressure relief, also make up about 10% of the total.

The current model takes credit for the proceduralized action to align the train A centrifugal charging pump to ERCW in the event that CCS has failed. Lack of credit for this action would raise the core damage frequency by an increment of 1.5×10^{-3} per reactor year. A further reduction in the core damage frequency could be achieved if the capability to provide backup cooling to the train B charging pump was provided. An estimated additional core damage frequency reduction of 2.2×10^{-5} per reactor-year or 7% (i.e., from the current 3.3×10^{-4} per reactor year) would result.

By comparison with the total frequency of seal LOCA core damage sequences, it can be seen that complete losses of CCS make up most of the total but not all. An appreciable fraction of the seal LOCA core damage frequency involves failure of only train A of CCS. Train A of CCS provides cooling for the RCP thermal barriers, the RCP upper and lower bearing coolers, and for the lube oil of the centrifugal charging pump.

Several different ventilation systems were included in the analysis. However, the only ventilation systems that contribute significantly are those that provide cooling for the general area in which the CCS and motor-driven AFW pumps are located. Multiple systems can provide ventilation to this large area. No credit for recovery actions, such as aligning portable ventilation or opening adjacent doors, was considered in the analysis for this location. The current model assumed that any operating motor-driven pumps would fail after 5 hours if the ventilation systems serving this area failed.

Another way of looking at the Level 1 results is to break down the core damage frequency by similar containment end state. Table 3.4-9 presents the breakdown of the core damage frequency by class of end state. The reader is cautioned that these results are only based on the events in the Level 1 plant model. Containment phenomena after core damage may result in failure of containment due to increased pressures, but these failure modes are not addressed here.

Nearly all (i.e., 96%) of the core damage frequency involves sequences in which the containment isolation system has succeeded, and there is no bypass of the containment to the secondary side. Only a tiny fraction of the total core damage frequency (i.e., 2%) involves sequences with an unisolated containment at the time of core damage. The majority of the unisolated sequences involve blackouts in which the seal return line was not locally isolated, as is directed by procedure for losses of all AC power. Containment bypass sequences contribute only 2% of the total core damage frequency. These are largely due to steam generator tube rupture initiating events. There is also a contribution from ATWS sequences in which the increased RCS pressure induces a steam generator tube to rupture and there is a failure to isolate the ruptured steam generator on the secondary side. Interfacing systems LOCAs also involve containment bypass, but the frequency of these events at Watts Bar is less than .01% of the total core damage frequency.

3.4.3.2.2 Core Damage Sensitivity To Plant Specific Failure Data

The important sequences to core damage were requantified with the plant-specific failure data that were collected and reported for the Sequoyah plant and assumed to apply for Watts Bar. Sequoyah also consists of two similar ice condenser units that are owned by TVA and that began commercial operation in the early 1980s. The failure data for the ERCW, CCS, AFW, and onsite electric power systems (i.e., the diesel generators) collected and interpreted from the Sequoyah operating experience were used in a requantification of the Watts Bar system split fractions. The revised split fraction values were then propagated through the important sequences to estimate the impact on the total core damage frequency. As a result, the core damage frequency decreased by about 10%. This indicates that the use of generic data for quantification of the Watts Bar plant model is not significantly different than if plant-specific data from Sequoyah are used.

3.4.3.2.3 Core Damage Sensitivity To Dynamic Operator Action

This sensitivity case is requested by the IPE reporting guidelines in NUREG-1335. The guidance requires that any sequence that drops below the core damage frequency criteria because of a reduction by more than 1 order of magnitude by credit taken for operator actions be discussed. No more than 50 of the most significant sequences are then to be reported.

This requirement was addressed by modifying the failure rate database by raising the dynamic operator action error rates to at least 0.1. Actions whose error rates were already greater than 0.1 were not changed. Electric power recovery factors, which depend more on the types of failures involved rather than on the response of the control room crew, were also left unchanged. Then, all of the split fractions were requantified using the revised database. The resulting split fractions were then used to requantify the Level 1 plant model event trees.

The sequences which were the highest contributors to the core damage frequency were then identified. Some sequences already had frequencies greater than 1×10^{-7} per reactor-year, and now that they are evaluated with higher human error rates, their frequencies are even higher. A brief discussion of the new sequences that appeared above 1×10^{-7} in the sensitivity case is provided below. A complete list of the top 100 ranked sequences from this sensitivity case (i.e., all those with individual frequencies down to 3.8×10^{-6} per reactor-year), including those already identified in the base case, is provided in Appendix D.

Sequences initiated by small LOCAs contribute the highest frequency to core damage for this sensitivity case. The top sequences involve failure of the operators to align for high pressure injection by crosstieing the discharge of the RHR pumps to the suction of the charging and safety injection pumps once the RWST reaches low level; i.e., failure of Top Event RR. These sequences were also found to be important in the base case quantification. Other sequences initiated by a small LOCA, but at a lower frequency, involve the above sequences with an additional failure of the operators to turn on the hydrogen ignitors.

Failures of train A of component cooling water contribute the next most to the core damage frequency for this sensitivity case. These sequences involve failure of the operators to trip the RCPs in response to the loss of RCP motor bearing cooling, resulting in a small LOCA. The sequences then go to core damage because of a failure to align for high pressure recirculation from the containment sump once the RWST inventory is depleted, and a failure to provide makeup to the RWST. Alternatively, recirculation from the containment sump may be failed as a result of an independent failure of the remaining train B of RHR. These sequences also show up in the top list of sequences for the base case.

Steam generator tube ruptures are the third-ranked initiating event group by contribution to core damage frequency for this sensitivity case. The top core damage sequences involve failure of the operators to cool down and depressurize the RCS in order to limit leakage flow to the environment via the secondary side of the ruptured steam generator, and in addition, failure to provide long-term makeup to the RWST. Other sequences involve failure to initiate closed-loop RHR cooling for tube rupture sequences involving an earlier failure to isolate the ruptured steam generator but with a successful action to cooldown and depressurize the RCS. These sequences also show up in the top list of sequences for the base case.

The loss of 120V vital board 1-I initiating event group contributes the fourth most to the frequency of core damage for this sensitivity case. The initiator causes failure of one train of the engineered safety features actuation system (ESFAS). The other train then fails independently in the top sequences. Due to the higher error rates assumed, the top sequences then also involve failure of the operators to back up the automatic actuation signals. The highest frequency sequences involve failure of secondary heat removal or failure to trip the reactor. These sequences were not important in the base case because the operators are likely to perform the necessary equipment actuation.

The isolable small LOCA initiating event group contributes next most to the core damage frequency for this sensitivity case. The isolable small LOCA is assumed to result in a Phase B signal prior to successful isolation. The Phase B condition isolates cooling water to the RCP thermal barrier coolers and to the RCP motor bearing coolers. From this point on, the sequences look similar to those initiated by a loss of CCS train A. The operators fail to trip the RCPs in time to prevent seal damage, resulting in a small LOCA. Core damage then results from a second error, to align for high pressure recirculation. These sequences were not prominent in the base case because of the much lower error rate assigned for tripping the RCPs under Phase B conditions.

Total losses of CCS cooling are the next largest contributors to the core damage frequency for this sensitivity case. These sequences involve failure of the operators to trip the RCPs in response to the loss of RCP motor bearing cooling, resulting in a small LOCA. The sequences then go to core damage because of a failure of recirculation from the containment sump once the RWST inventory is depleted; i.e., CCS cooling for the RHR heat exchangers is unavailable. These sequences also show up in the top list of sequences for the base case.

The seventh-ranked initiator group to the core damage frequency for this sensitivity case involves loss of the 6.9-kV train B shutdown board. The operators then fail to start the standby ERCW pumps on train B, which leads to a loss of ERCW flow to header 1B, and the unavailability of CCS train A. The operators then further fail to align the train A centrifugal charging pump for cooling via ERCW header 1A. The loss of RCP seal cooling then leads to seal damage, and all high pressure injection is unavailable. In an alternate sequence with the same initiator, the operators fail to start the standby CCS pump on train A. This failure then leads to the same sequence of events as the first one; i.e., RCP seal cooling failure resulting in a small LOCA, with failure of all high pressure injection. Another variation on this sequence is, if the operators fail to trip the RCPs in time to prevent seal damage, charging pump A is successfully aligned for high pressure injection, but both RHR trains are unavailable for recirculation from the sump. Makeup to the RWST is also unsuccessful. Each of these sequences is similar to those identified in the base case quantification. The one key difference is the frequency assigned to the likelihood of failing to start the standby pumps. In the base case, the error rate for this action is low.

The eighth-ranked initiator group to core damage frequency for this sensitivity case is loss of RCS flow. For plant trips with RCP flow unavailable to provide pressurizer spray, a challenge for RCS pressure relief was modeled. In the key sequences for this case, the pressurizer relief valves fail to reclose, and the PORV block valves are not isolated. Core damage results when the operators also fail to align for high pressure recirculation.

The top medium LOCA core damage sequence for this sensitivity case involves failure of the operators to align for high pressure recirculation. This sequence also contributed in the base case.

The top large LOCA core damage sequence involved failure of the operators to align for hot leg recirculation 15 hours after switchover to the containment sump.

Sequences initiated by a flood in the train B ERCW strainer room make an appreciable contribution to the core damage frequency for this sensitivity case. The flood causes the loss of the train B ERCW pumps for both units. The operators then fail to align ERCW header 2A as a backup for cooling to CCS train A, and fail to align ERCW backup cooling to the train A charging pump. The result is a loss of all seal cooling with failure of injection. In a related sequence, the operators successfully align backup cooling for the train charging pump but fail to trip the RCPs in time to prevent seal damage. The result is a small LOCA with successful injection but failure of recirculation due to a loss of RHR cooling.

Finally, a family of sequences may be initiated by a turbine trip, reactor trip, or partial loss of main feedwater; i.e., any initiator that has a relatively high frequency of causing a plant trip. In these sequences, both trains of ERCW pumps fail. The frequency of this happening is relatively high because little credit is given in the sensitivity case for the operators manually starting the standby ERCW pumps given failure of the normally running pumps.

The key lesson learned from this sensitivity case is that the new sequences that result primarily involve the failure of two or more operator actions. Those that involve the failure of only one action are already visible in the base case; e.g., failure to align for high pressure recirculation.

3.4.4 DECAY HEAT REMOVAL EVALUATION

Resolution of Unresolved Safety Issue (USI) A-45 has been subsumed into the IPE requirements that allow plant-specific evaluation of the safety adequacy of decay heat removal systems. According to NUREG-1335, the evaluation is restricted to events initiated from power operation and hot standby. A discussion of the decay heat removal capability at Watts Bar for preventing severe accident situations is provided below.

The results for Watts Bar provide indications of the importance of systems that directly perform the decay heat removal function. Table 3.4-10 indicates the importance measures for systems that perform the decay heat removal function at Watts Bar. Four classes of systems were considered: main feedwater, auxiliary feedwater, feed and bleed cooling, and closed-loop residual heat removal. Importance is measured by the percentage of core damage frequency attributable to sequences that involve failure of the indicated split fraction. These measures are not strictly additive because more than one of the ranked split fractions may, and often do, fail in the same sequence.

Three event tree top events are used to represent the main feedwater system. Top Event MF represents the hardware failure modes for successful reactor trip sequences, and Top Event FW represents the hardware failure modes under ATWS conditions. Top Event OF represents the operator response to realign main feedwater after plant trip, given that auxiliary feedwater has failed. Successful realignment, of course, also requires the success of the equipment modeled in Top Event MF.

The most important main feedwater (MFW) system failures occur in sequences for which main feedwater is lost due to the same cause as the initiating event or due to the loss of one or more support systems; i.e., the condition represented by failure of split fraction MFF for successful reactor trip sequences, and split fraction FWF for ATWS sequences. Failures of the operators to realign main feedwater in response to a loss of all AFW, or of independent failures of MFW that prevent it from being realigned, contribute only a small amount to the core damage frequency. Losses of MFW during ATWS account for a large fraction of the total core damage frequency contribution from all ATWS sequences. This is as expected. The relatively large fraction of core damage frequency involving sequences with MFW unavailable requires further interpretation. Many of the main feedwater failures do not directly contribute to the sequence resulting in core damage; rather, MFW is also lost due to many of the same support system failures that lead to core damage

Top Events TP, TPR, MA, MB, and AF model the auxiliary feedwater system. Top Events TP and TPR model the turbine-driven AFW pump train hardware, and the operator action to locally restart the pump if it initially trips off in response to the automatic start attempt. Top Events MA and MB model the motor-driven AFW pump trains. Top Event AF models the suction and discharge valves to and from the AFW pumps to the steam generators. Additional descriptions of these top events can be found in Section 3.1.2, where the frontline event trees are presented.

Combinations of failures of the AFW valves directing flow to the steam generators contribute only a small fraction of the total core damage frequency, with or without reactor trip. On the other hand, sequences involving failure of all three AFW pumps contribute almost 10% of the total core damage frequency. It should not be assumed that all such sequences lead to core damage due to a subsequent failure of feed and bleed cooling. Instead, much of the sequence frequency involves successful initiation of feed and bleed cooling, but core damage then results due to the unavailability of recirculation from the containment sump. Some of the sequences involving failure of all AFW can be recovered by realigning main feedwater.

Feed and bleed cooling is modeled by several top events. Top Events VA, VB, and VC model pump trains A and B of centrifugal charging pumps and the suction and cold leg injection paths, respectively. Top Events S1, S2, and SI model pump trains A and B of safety injection and the suction and cold leg injection paths. Top Event OB models the operator action to initiate feed and bleed cooling and the necessary pressurizer PORV that must open to provide the bleed paths.

Most of the core damage sequences involving failure of feed and bleed cooling occur due to a loss of support systems, which preclude operation of at least one pressurizer PORV train; i.e., split fraction OBF fails in 5% of the core damage frequency. Successful feed and bleed cooling currently requires that both pressurizer PORVs be opened. Failure of the operators to initiate feed and bleed cooling contributes only a small amount; i.e., split fraction OB1 fails in only 0.5% of the core damage sequences.

Top Events RA, RB, RI, and RD model the equipment needed for closed-loop RHR cooling. Top Events RA and RB model the RHR pump trains A and B. Top Event RI models the RHR injection path from the RHR pumps to the cold legs of the RCS. Top Event RD models the RHR hot leg suction path, the CCS cooling water to the RHR heat exchangers, and the equipment required for letdown and normal charging. It also contains the operator action to align for closed-loop RHR cooling, once the RCS has been cooled down and depressurized to the necessary entry conditions. The actions and equipment for RCS cooldown and depressurization are modeled via Top Events DS and DP, respectively.

The current plant models only take credit for closed-loop RHR cooling in the event of a steam generator tube rupture. If the ruptured steam generator is unisolated, by cooling down and depressurizing the RCS for closed-loop RHR cooling, the flow out the leak may be reduced to insignificant levels, thereby preventing core damage. Only about 1.5% of the core damage frequency involves steam generator tube rupture with failure to cool down and depressurize the RCS. The failure of both pump and heat exchanger trains of RHR cooling does contribute significantly to the total core damage frequency. If at least one RHR pump train is fully supported, the contribution to the core damage frequency from all initiators is only about 6%. Loss of support to both trains occurs in a much greater fraction of the core damage frequency; both split fraction RAF and RBF fail in 56% of the sequences that lead to core damage. Much of the time, it is the loss of component cooling water that precludes operation of the RHR trains.

In summary, no particular vulnerabilities of the Watts Bar decay heat removal systems have been identified. The majority of the core damage frequency comes from loss of core cooling scenarios that are caused by failures of the component cooling water system, rather than by failures within the decay heat removal systems.

3.4.5 USI AND GSI SCREENING

NUREG-1335 (Reference 3.4.2) requires that, for those USIs and generic safety issues (GSI) the utility claims are resolved in whole or in part by the IPE submittal, additional information be submitted. In particular, the technical basis for resolving the issue must be provided along with the contribution of the USI or GSI to the core damage frequency or to containment performance.

The technical basis for resolving USI A-45, the evaluation of the decay heat removal function, is provided in Section 3.4.4. No particular vulnerabilities of the systems that are used to perform decay heat removal have been identified.

TVA makes no claim on the resolution of any of the other USIs or GSIs by this submittal. Therefore, no other USIs or GSIs are discussed at the present time.

3.4.6 REFERENCES

3.4-1. Henry, R. E., and M. G. Plys, "MAAP-3.0B — Modular Accident Analysis Program for LWR Power Plants," Electric Power Research Institute, EPRI NP-7071-CCML, November 1990.

3.4-2. U.S. Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

Table 3.4-1. Watts Bar Plant Damage States

Rank	PDS Name	RCS Pressure	Containment Heat Removal	Containment Integrity	Annual Frequency	Percent of Total CDF
1	ENI	High	No	Isolated, Not Bypassed	1.382-04	41.5
2	FCI	High	Yes	Isolated, Not Bypassed	9.033-05	27.1
3	FNI	High	No	Isolated, Not Bypassed	4.348-05	13.1
4	LCI	System	Yes	Isolated, Not Bypassed	1.506-05	4.52
5	GNI	High	No	Isolated, Not Bypassed	8.344-06	2.50
6	BCI	Low	Yes	Isolated, Not Bypassed	5.705-06	1.71
7	ENS	High	No	Not Isolated — Small	4.787-06	1.44
8	EIB	High	Yes	Small Bypass	3.659-06	1.10
9	HGI	High	Yes	Isolated, Not Bypassed	3.094-06	0.93
10	ENB	High	No	Small Bypass	2.665-06	0.80
11	EGI	High	Yes	Isolated, Not Bypassed	2.325-06	0.70
12	KNS	System	No	Not Isolated — Small	2.225-06	0.67
13	KNI	System	No	Isolated, Not Bypassed	2.174-06	0.65
14	FGI	High	Yes	Isolated, Not Bypassed	2.070-06	0.62
15	HNI	High	No	Isolated, Not Bypassed	2.034-06	0.61
16	HCI	High	Yes	Isolated, Not Bypassed	1.572-06	0.47
17	LNI	System	No	Isolated, Not Bypassed	1.433-06	0.43
18	DCI	Medium	Yes	Isolated, Not Bypassed	9.948-07	0.30
19	GNS	High	No	Not Isolated — Small	5.063-07	0.15
20	FCB	High	Yes	Small Bypass	4.336-07	0.13
21	FCS	High	Yes	Not Isolated — Small	2.929-07	0.09
22	KTL	System	No	Not Isolated — Large	2.483-07	0.07
23	FNS	High	No	Not Isolated — Small	1.851-07	0.06
24	ETL	High	No	Not Isolated — Large	1.661-07	0.05
				Total	**3.320-04**	**99.65**

Note: Exponential notation is indicated in abbreviated form; e.g., 1.382-04 = 1.382 × 10^{-04}.

Table 3.4-2 (Page 1 of 30). Top 100 Sequences Contributing to Core Damage

MODEL Name: WBNNEW — Top-Ranking Sequences Contributing to Group : MELT Frequency — 19:05:24 25 JUL 1992

MELT = ALL CORE DAMAGE SEQUENCES

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
1	TOTAL LOSS OF COMPONENT COOLING WATER - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FNI	1.28E-05	4.27
2	TOTAL LOSS OF ERCW	- LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.26E-05	4.17
3	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY	FNI	1.13E-05	3.74

Table 3.4-2 (Page 2 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF RHR SPRAY RECIRCULATION			
4	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	7.68E-06	2.55
5	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION	ENI	7.20E-06	2.39
6	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	6.93E-06	2.30

Table 3.4-2 (Page 3 of 30). Top 100 Sequences Contributing to Core Damage

7	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - FAILURE OF MAKEUP TO RWST	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	6.73E-06	2.24
8	LOSS OF COMPONENT COOLING WATER TRAIN A - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - LOSS OF RHR PUMP 1B-B	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FCI	5.97E-06	1.98
9	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - FAILURE OF MAKEUP TO RWST - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	5.26E-06	1.75
10	TOTAL LOSS OF COMPONENT COOLING WATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	3.44E-06	1.14
11	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - FAILURE OF MAKEUP TO RWST - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	3.34E-06	1.11
12	LOSS OF COMPONENT COOLING WATER TRAIN A - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF SAFETY INJECTION PUMP 1B-B	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FCI	3.19E-06	1.06
13	LOSS OF COMPONENT COOLING WATER TRAIN A - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A	FCI	3.14E-06	1.05

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Table 3.4-2 (Page 4 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION			
14	FLOODING, ERCW STRAINER ROOM - LOSS OF ERCW HEADER 1A	- LOSS OF ERCW TRAIN B PUMPS - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	3.09E-06	1.03
15	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE OF MAKEUP TO RWST - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	3.02E-06	1.00
16	LOSS OF COMPONENT COOLING WATER TRAIN A - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FCI	2.76E-06	.92
17	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 2 DIESEL GENERATOR 2A-A - LOSS OF UNIT 2 DIESEL GENERATOR 2B-B - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D	ENI	2.42E-06	.80

Table 3.4-2 (Page 5 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF 6.9KV SD BD TRAIN A UNIT 2 - LOSS OF 480V SD TRANSFORMER ROOM 2A VENTILATION - LOSS OF UNIT 2 120V AC INSTRUMENT BOARD 2A - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - LOSS OF ERCW TRAIN A PUMPS - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF RHR SPRAY RECIRCULATION			
18	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF UNIT 2 DIESEL GENERATOR 2A-A - LOSS OF UNIT 2 DIESEL GENERATOR 2B-B - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV SD BD TRAIN A UNIT 2 - LOSS OF 480V SD TRANSFORMER ROOM 2A VENTILATION - LOSS OF UNIT 2 120V AC INSTRUMENT BOARD 2A - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS	ENI	2.30E-06	.77

Table 3.4-2 (Page 6 of 30). Top 100 Sequences Contributing to Core Damage

		- FAILURE TO RECOVER ERCW TO DIESEL FROM OPPOSITE TRAIN - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
19	LOSS OF COMPONENT COOLING WATER TRAIN A - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FCI	2.18E-06	.72
20	LOSS OF 6.9 SHUTDOWN BOARD 1B-B - LOSS OF ERCW HEADER 1A	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY	ENI	2.09E-06	.69

Table 3.4-2 (Page 7 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION			
21	FLOOD ERCW STAINER ROOM, TRAIN A - MAINTENANCE ON ERCW HEADER 1B	- LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	2.02E-06	.67
22	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF UNIT 2 DIESEL GENERATOR 2A-A - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV SD BD TRAIN A UNIT 2 - LOSS OF 480V SD TRANSFORMER ROOM 2A VENTILATION - LOSS OF UNIT 2 120V AC INSTRUMENT BOARD 2A - LOSS OF ERCW TRAIN A PUMPS - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B	ENI	1.98E-06	.66

Table 3.4-2 (Page 8 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
23	LOSS OF 6.9 SHUTDOWN BOARD 1B-B - LOSS OF ERCW HEADER 1A - FAILURE OF MAKEUP TO RWST	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.96E-06	.65
24	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF UNIT 2 DIESEL GENERATOR 2B-B - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - LOSS OF ERCW TRAIN B PUMPS - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR)	ENI	1.95E-06	.65

Table 3.4-2 (Page 9 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
25	TURBINE TRIP INITIATING EVENT - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	1.93E-06	.64
26	PARTIAL LOSS OF MAIN FEEDWATER - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	1.78E-06	.59
27	LOSS OF BATTERY BOARD I - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B	- LOSS OF 125V DC BATTERY BD I - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	LCI	1.76E-06	.59
28	TURBINE TRIP INITIATING EVENT - FAILURE OF MAKEUP TO RWST - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	1.69E-06	.56
29	PARTIAL LOSS OF MAIN FEEDWATER	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT	FCI	1.56E-06	.52

Table 3.4-2 (Page 10 of 30). Top 100 Sequences Contributing to Core Damage

	- FAILURE OF MAKEUP TO RWST - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS			
30	LOSS OF 6.9 SHUTDOWN BOARD 1B-B - LOSS OF 480V SHUTDOWN BOARD 1A1-A	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.48E-06	.49
31	LOSS OF OFFSITE POWER - LOSS OF 6.9KV SD BD UNIT 1 TRAIN A - LOSS OF 6.9KV SD BD UNIT 1 TRAIN B - LOSS OF 6.9KV SD BD TRAIN A UNIT 2 - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 480V SD TRANSFORMER ROOM 2A VENTILATION - LOSS OF UNIT 2 120V AC INSTRUMENT BOARD 2A - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM	ENI	1.43E-06	.48

Table 3.4-2 (Page 11 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
32	PARTIAL LOSS OF MAIN FEEDWATER - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF STEAM RELIEF, ATWS ONLY, REACTOR PRESSURE IS LESS THAN 32	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	1.42E-06	.47
33	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF ERCW TRAIN A PUMPS - LOSS OF CCS AND MD AFW PUMPS VENTILIATION	- LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - FAILURE TO COOLDOWN USING STEAM DUMPS, CONDENSER, & HOTWEL - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.39E-06	.46
34	FLOODING, ERCW STRAINER ROOM - LOSS OF 480V SHUTDOWN BOARD 1A1-A	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF ERCW TRAIN B PUMPS - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A	ENI	1.38E-06	.46

Table 3.4-2 (Page 12 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF RHR SPRAY RECIRCULATION			
35	STEAM GENERATOR TUBE RUPTURE - FAILURE OF MAKEUP TO RWST - OPERATOR FAILS TO DEPRESSURIZE THE RCS USING SG PORVS	- LOSS OF RHR NORMAL DECAY HEAT REMOVAL - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - MELT WITH SGTR BYPASS	EIB	1.30E-06	.43
36	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 2 DIESEL GENERATOR 2B-B - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	KNI	1.29E-06	.43
37	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION	GNI	1.27E-06	.42

Table 3.4-2 (Page 13 of 30). Top 100 Sequences Contributing to Core Damage

	- FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION			
38	LOSS OF 6.9 SHUTDOWN BOARD 1A-A - LOSS OF 480V SHUTDOWN BOARD 1B1-B	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1A-A - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - FAILURE TO COOLDOWN USING STEAM DUMPS, CONDENSER, & HOTWEL - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.27E-06	.42

Table 3.4-2 (Page 14 of 30). Top 100 Sequences Contributing to Core Damage

39	PARTIAL LOSS OF MAIN FEEDWATER - FAILURE OF MAKEUP TO RWST - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40% - FAILURE OF STEAM RELIEF, ATWS ONLY, REACTOR PRESSURE IS LESS THAN 32	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	1.25E-06	.41
40	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF ERCW TRAIN A PUMPS - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - FAILURE OF MAKEUP TO RWST	- LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - FAILURE TO COOLDOWN USING STEAM DUMPS, CONDENSER, & HOTWEL - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.21E-06	.40
41	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF 125V DC BATTERY BD III	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTE - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF STEAM GENERATOR COOLING	HGI	1.18E-06	.39

Table 3.4-2 (Page 15 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF RHR SPRAY RECIRCULATION			
42	REACTOR TRIP INITIATING EVENT - LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS	- LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	1.18E-06	.39
43	LOSS OF COMPONENT COOLING WATER TRAIN A - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FNI	1.17E-06	.39
44	STEAM GENERATOR TUBE RUPTURE - FAILURE OF MAKEUP TO RWST - OPERATOR FAILS TO IDENTIFY & ISOLATE RUPTURED STEAM GENERATOR - OPERATOR FAILS TO DEPRESSURIZE THE RCS USING SG PORVS	- LOSS OF RHR NORMAL DECAY HEAT REMOVAL - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - MELT WITH SGTR BYPASS	EIB	1.07E-06	.36
45	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF UNIT 2 DIESEL GENERATOR 2A-A - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV SD BD TRAIN A UNIT 2 - LOSS OF 480V SD TRANSFORMER ROOM 2A VENTILATION - LOSS OF UNIT 2 120V AC INSTRUMENT BOARD 2A - NO POWER AT 6.9 SHUTDOWN BD 2A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B	HGI	1.04E-06	.35

Table 3.4-2 (Page 16 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION			
46	LOSS OF COMPONENT COOLING WATER TRAIN A - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FNI	1.02E-06	.34
47	PARTIAL LOSS OF MAIN FEEDWATER - LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS	- LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	9.84E-07	.33
48	PARTIAL LOSS OF MAIN FEEDWATER - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	9.63E-07	.32
49	LOSS OF ERCW TRAIN B	- LOSS OF ERCW TRAIN B PUMPS	ENI	9.53E-07	.32

Table 3.4-2 (Page 17 of 30). Top 100 Sequences Contributing to Core Damage

	- LOSS OF ERCW HEADER 1A	- LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION			
50	TURBINE TRIP INITIATING EVENT - LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS	- LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	9.31E-07	.31
51	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - FAILURE OF CONTAINMENT ISOLATION - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A	ENS	9.30E-07	.31

Table 3.4-2 (Page 18 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - MELT WITH CI FAILURE AND CONT. PURGE SUCCESS - LOSS OF RHR SPRAY RECIRCULATION			
52	LOSS OF OFFSITE POWER - LOSS OF 6.9KV SD BD UNIT 1 TRAIN A - LOSS OF 6.9KV SD BD UNIT 1 TRAIN B	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - FAILURE TO COOLDOWN USING STEAM DUMPS, CONDENSER, & HOTWEL - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73	ENI	9.30E-07	.31

Table 3.4-2 (Page 19 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
53	SMALL LOCA ISOLABLE, PZR PORV LEAK - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR & S	- LOSS OF RHR SPRAY RECIRCULATION	FCI	9.19E-07	.31
54	TURBINE TRIP INITIATING EVENT - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	9.07E-07	.30
55	LOSS OF BATTERY BOARD II - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A	- LOSS OF 125V DC BATTERY BD II - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	LCI	8.89E-07	.30
56	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - OPERATOR FAILS TO DEPRESSURIZE THE RCS USING SG PORVS - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF UNIT 1 SHUTDOWN BOARD VENTILATION - FAILURE TO RECOVER UNIT 1 SHUTDOWN BD ROOM VENTILATION IN - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF AIR RETURN FANS - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY	ENI	8.64E-07	.29

Table 3.4-2 (Page 20 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
57	TOTAL LOSS OF COMPONENT COOLING WATER - LOSS OF ERCW HEADER 1A	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	8.46E-07	.28
58	PARTIAL LOSS OF MAIN FEEDWATER - FAILURE OF MAKEUP TO RWST - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- MAIN FEEDWATER FAILS TO CONTINUE DURING ATWS EVENT - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	8.44E-07	.28
59	TURBINE TRIP INITIATING EVENT - FAILURE OF MAKEUP TO RWST - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - FAILURE OF EMERGENCY BORATION (OPERATOR ACTIONS & EQUIPEMENT)	- LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS	FCI	7.95E-07	.26
60	LOSS OF BATTERY BOARD I - LOSS OF TRAIN B ESSENTIAL AIR - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES	- LOSS OF 125V DC BATTERY BD I - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	LCI	7.84E-07	.26
61	LOSS OF BATTERY BOARD II - FAILURE OF MAKEUP TO RWST - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A	- LOSS OF 125V DC BATTERY BD II - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN B CONTAINMENT SPRAY	LCI	7.80E-07	.26

Table 3.4-2 (Page 21 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF RHR SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION			
62	TOTAL LOSS OF COMPONENT COOLING WATER - LOSS OF ERCW HEADER 1A - FAILURE OF MAKEUP TO RWST	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	7.42E-07	.25
63	LOSS OF 120 VAC VITAL BOARD 1-III - LOSS OF 125V DC BATTERY BD I - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B	- LOSS OF 120 V AC VITAL BD 1-III - LOSS OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTE - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	LCI	7.14E-07	.24
64	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - CONTAINMENT SUMP IS UNAVAILABLE	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	EGI	7.13E-07	.24
65	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FNI	7.05E-07	.23
66	SMALL LOCA ISOLABLE, PZR PORV LEAK - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR & S	- LOSS OF RHR SPRAY RECIRCULATION	FCI	6.98E-07	.23

Table 3.4-2 (Page 22 of 30). Top 100 Sequences Contributing to Core Damage

67	LOSS OF COMPONENT COOLING WATER TRAIN A - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	6.98E-07	.23
68	TOTAL LOSS OF ERCW - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES	- LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW TRAIN B PUMPS - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	GNI	6.92E-07	.23
69	TOTAL LOSS OF COMPONENT COOLING WATER - LOSS OF 480V SHUTDOWN BOARD 1A1-A	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B	ENI	6.89E-07	.23

Table 3.4-2 (Page 23 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF RHR SPRAY RECIRCULATION			
70	REACTOR TRIP INITIATING EVENT - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	6.64E-07	.22
71	STEAM GENERATOR TUBE RUPTURE - LOSS OF 125V DC BATTERY BD I - OPERATOR FAILS TO DEPRESSURIZE THE RCS USING SG PORVS	- LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - OPERATOR FAILS TO IDENTIFY & ISOLATE RUPTURED STEAM GENERA - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF RHR NORMAL DECAY HEAT REMOVAL - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY - MELT WITH SGTR BYPASS - LOSS OF RHR SPRAY RECIRCULATION	ENB	6.59E-07	.22
72	FLOOD ERCW STAINER ROOM, TRAIN A - LOSS OF 480V SHUTDOWN BOARD 1B1-B	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF ERCW TRAIN A PUMPS - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	6.31E-07	.21
73	REACTOR TRIP INITIATING EVENT - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A - FAILURE OF MAKEUP TO RWST	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B	ENI	6.23E-07	.21

		- LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION			
74	LOSS OF ERCW TRAIN A - MAINTENANCE ON ERCW HEADER 1B	- LOSS OF ERCW TRAIN A PUMPS - LOSS OF ERCW COOLING TO CAS COMPRESSORS - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	ENI	6.23E-07	.21
75	TOTAL LOSS OF COMPONENT COOLING WATER - FAILURE OF MAKEUP TO RWST - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FNI	6.18E-07	.21
76	LOSS OF COMPONENT COOLING WATER TRAIN A - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - FAILURE OF MAKEUP TO RWST	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY	ENI	6.12E-07	.20

Table 3.4-2 (Page 25 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF RHR SPRAY RECIRCULATION			
77	FLOOD ERCW STAINER ROOM, TRAIN A - LOSS OF 480V SHUTDOWN BOARD 1B1-B - FAILURE OF MAKEUP TO RWST	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF ERCW TRAIN A PUMPS - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	5.92E-07	.20
78	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - LOSS OF RHR PUMP 1B-B - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	5.89E-07	.20
79	LARGE & EXCESSIVE BREAK LOCA - FAILURE OF RHR & SIS HOT LEG RECIRCULATION	- LOSS OF RHR SPRAY IN RECIRCULATION	BCI	5.67E-07	.19
80	MEDIUM BREAK LOCA - FAILURE OF AUTOMATIC/MANUAL SWAPOVER FROM THE RWST TO THE CONTAINMEN	- NO INJECTION FOR MLOCA (200-400 PSI) - LOSS OF RHR SPRAY RECIRCULATION	BCI	5.65E-07	.19
81	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - LOSS OF RHR PUMP 1A-A - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	5.62E-07	.19
82	PARTIAL LOSS OF MAIN FEEDWATER - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	5.56E-07	.18
83	LOSS OF 6.9 SHUTDOWN BOARD 1B-B - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A	FNI	5.54E-07	.18

Table 3.4-2 (Page 26 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION			
84	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF UNIT 2 DIESEL GENERATOR 2B-B - FAILURE OF THERMAL BARRIERS TO THE RCPS - FAILURE TO RECOVER POWER BEFORE CORE DAMAGE	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - LOSS OF 6.9KV UNIT 2 SHUTDOWN BOARD 2B-B - LOSS OF 480V SD TRANSFORMER ROOM 2B VENTILATION - LOSS OF 480V SD BD RM 2B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1A-A - NO POWER AT 6.9 SHUTDOWN BD 2B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF RHR SPRAY RECIRCULATION	ENI	5.45E-07	.18
85	STEAM GENERATOR TUBE RUPTURE - LOSS OF 125V DC BATTERY BD I - LOSS OF RHR PUMP 1B-B	- LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - OPERATOR FAILS TO IDENTIFY & ISOLATE RUPTURED STEAM GENERA - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - MELT WITH SGTR BYPASS - LOSS OF RHR SPRAY RECIRCULATION	ENB	5.44E-07	.18
86	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - FAILURE OF MAKEUP TO RWST - CONTAINMENT SUMP IS UNAVAILABLE	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	EGI	5.42E-07	.18

Table 3.4-2 (Page 27 of 30). Top 100 Sequences Contributing to Core Damage

87	TURBINE TRIP INITIATING EVENT - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	5.27E-07	.17
88	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1A-A - LOSS OF ERCW TRAIN B PUMPS - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1A-A - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN B ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	GNI	5.24E-07	.17
89	PARTIAL LOSS OF MAIN FEEDWATER - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A - FAILURE OF MAKEUP TO RWST	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B	ENI	5.22E-07	.17

Table 3.4-2 (Page 28 of 30). Top 100 Sequences Contributing to Core Damage

		- LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION			
90	LOSS OF 6.9 SHUTDOWN BOARD 1B-B - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	FNI	5.20E-07	.17
91	LOSS OF OFFSITE POWER - FAILURE TO RECOVER OFFSITE POWER IN 1 HOUR - LOSS OF UNIT 1 DIESEL GENERATOR 1B-B - LOSS OF ERCW TRAIN A PUMPS - FAILURE OF TURBINE DRIVEN AFW PUMP - FAILURE TO RECOVER TD AFW PUMP START FAILURES IN 30 MINUTES	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF 480V BOARD ROOM B VENTILATION - LOSS OF 6.9KV UNIT BOARD 1A - LOSS OF 6.9KV UNIT BOARD 1B - LOSS OF 6.9KV UNIT BOARD 1C - LOSS OF 6.9KV UNIT BOARD 1D - NO POWER AT 6.9 SHUTDOWN BD 1B-B - LOSS OF CONTROL AIR (NON-ESSENTIAL AIR) - LOSS OF TRAIN A ESSENTIAL AIR - LOSS OF CCS AND MD AFW PUMPS VENTILIATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM - FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF MOTOR DRIVEN AFW PUMP 1B-B - LOSS OF EQUIPMENT NEEDED TO RECOVER MAIN FEEDWATER - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF STEAM GENERATOR COOLING - LOSS OF RHR SPRAY RECIRCULATION	GNI	5.01E-07	.17
92	STEAM GENERATOR TUBE RUPTURE	- FAILURE TO ALIGN CCP A TO ERCW TRAIN A ON LOSS OF CCS A	ENI	4.98E-07	.17

Table 3.4-2 (Page 29 of 30). Top 100 Sequences Contributing to Core Damage

	- LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF TRAIN B COMPONENT COOLING WATER SYSTEM	- LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF RHR SPRAY RECIRCULATION			
93	LARGE & EXCESSIVE BREAK LOCA - FAILURE OF MAKEUP TO RWST - FAILURE OF RHR & SIS HOT LEG RECIRCULATION	- LOSS OF RHR SPRAY IN RECIRCULATION	BCI	4.98E-07	.17
94	MEDIUM BREAK LOCA - FAILURE OF MAKEUP TO RWST - FAILURE OF AUTOMATIC/MANUAL SWAPOVER FROM THE RWST TO THE CONTAINMEN	- NO INJECTION FOR MLOCA (200-400 PSI) - LOSS OF RHR SPRAY RECIRCULATION	BCI	4.96E-07	.16
95	TURBINE TRIP INITIATING EVENT - LOSS OF 480V SHUTDOWN BOARD 1B1-B - LOSS OF ERCW HEADER 1A - FAILURE OF MAKEUP TO RWST	- LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF RHR SPRAY RECIRCULATION	ENI	4.94E-07	.16
96	LOSS OF BATTERY BOARD I - FAILURE TO TRIP REACTOR AND INSERT CONTROL RODS INSERT - POWER LEVEL IS GREATER THAN 40%	- LOSS OF 125V DC BATTERY BD I - LOSS OF MOTOR DRIVEN AFW PUMP 1A-A - LOSS OF STEAM RELIEF, ATWS ONLY, REACTOR PRESSURE IS LE - LOSS OF PZR PORVS OPEN TO CONTROL RCS PRESSURE & RECLOS - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF OPERATOR ACTION TO FEED & BLEED RCS - LOSS OF RHR PUMP 1A-A - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF RHR SPRAY - LOSS OF RHR SPRAY RECIRCULATION	FCI	4.82E-07	.16
97	REACTOR TRIP INITIATING EVENT - LOSS OF 480V SHUTDOWN BOARD 1A1-A - LOSS OF 480V SHUTDOWN BOARD 1B1-B	- LOSS OF 480V SD TRANSFORMER ROOM 1A VENTILATION - LOSS OF UNIT 1 120V AC INSTRUMENT BOARD 1A - LOSS OF SD TRANSFORMER ROOM 1B VENTILATION - LOSS OF TRAIN A COMPONENT COOLING WATER SYSTEM - LOSS OF CENTRIFUGAL CHARGING PUMP 1A-A - LOSS OF CENTRIFUGAL CHARGING PUMP 1B-B	ENI	4.66E-07	.15

Table 3.4-2 (Page 30 of 30). Top 100 Sequences Contributing to Core Damage

		- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF SAFETY INJECTION PUMP 1A-A - LOSS OF SAFETY INJECTION PUMP 1B-B - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B - LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - LOSS OF TRAIN A SUMP SWAPOVER VALVE, 1-FCV-63-72 - LOSS OF TRAIN B SUMP SWAPOVER VALVE, 1-FCV-63-73 - LOSS OF HYDROGEN IGNITORS - LOSS OF RHR SPRAY RECIRCULATION			
98	SMALL LOCA NON-ISOLABLE, RCP SEAL LOCA - FAILURE OF MAKEUP TO RWST - LOSS OF RHR PUMP 1B-B - FAILURE OF AUTOMATIC/MANUAL SWAPOVER TO CONTAINMENT SUMP FOR RHR	- RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOP - LOSS OF RHR SPRAY RECIRCULATION	FCI	4.47E-07	.15
99	SMALL LOCA ISOLABLE, PZR PORV LEAK - FAILURE OF MAKEUP TO RWST - RCP SEAL COOLING FAILED OR RCPS NOT TRIPPED - LOCA DEVELOPS - LOSS OF RHR PUMP 1A-A - LOSS OF RHR PUMP 1B-B	- LOSS OF RHR SPRAY RECIRCULATION	FCI	4.43E-07	.15
100	STEAM GENERATOR TUBE RUPTURE - FAILURE OF MAKEUP TO RWST - OPERATOR FAILS TO IDENTIFY & ISOLATE RUPTURED STEAM GENERATOR - LOSS OF RHR NORMAL DECAY HEAT REMOVAL	- LOSS OF TRAIN A CONTAINMENT SPRAY - LOSS OF TRAIN B CONTAINMENT SPRAY - MELT WITH SGTR BYPASS	EIB	4.35E-07	.14

Table 3.4-3 (Page 1 of 2). Guaranteed Failed Split Fractions Sorted by Importance (RISKMAN Generated)

Model Name: Watts Bar
Split Fraction Importance
Sorted by Importance

......	SF Name...	Importance.....	Achievement..	Reduction...	Derivative..	SF Value.......	Frequency
1.	MELTF	1.0000E+00	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	3.0092E-04
2.	IYAF	9.5811E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.8831E-04
3.	MELTIF	9.5811E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.8831E-04
4.	INTPRF	9.2109E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.7717E-04
5.	RHRSF	8.9323E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.6879E-04
6.	RECF	8.8620E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.6667E-04
7.	TBF	7.7165E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.3220E-04
8.	CSAF	7.4588E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.2445E-04
9.	DPF	7.3627E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.2156E-04
10.	S1F	7.3380E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.2081E-04
11.	RAF	7.3376E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.2080E-04
12.	RRF	7.1771E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1597E-04
13.	DSNN	7.1349E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.1470E-04
14.	CSBF	6.7911E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	2.0436E-04
15.	CSRF	6.6238E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9932E-04
16.	S2F	6.6129E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9899E-04
17.	RBF	6.6125E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9898E-04
18.	VBF	6.5963E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9849E-04
19.	CHF	6.4908E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9532E-04
20.	CSIF	6.4002E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9259E-04
21.	SIF	6.3507E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.9110E-04
22.	CMF	6.3030E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.8967E-04
23.	MSF	5.7254E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.7229E-04
24.	ACF	5.6933E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.7132E-04
25.	SEF	5.5639E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.6743E-04
26.	CAVF	5.0116E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.5081E-04
27.	VCF	4.8243E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.4517E-04
28.	VAF	4.4129E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.3279E-04

Table 3.4-3 (Page 2 of 2). Guaranteed Failed Split Fractions Sorted by Importance (RISKMAN Generated)

......	SF Name...	Importance.....	Achievement..	Reduction...	Derivative..	SF Value.......	Frequency
29.	BCF	3.8895E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.1704E-04
30.	RQF	3.5379E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.0646E-04
31.	MBF	3.5094E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.0561E-04
32.	MUF	3.4006E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.0233E-04
33.	MAF	3.3291E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	1.0018E-04
34.	PDF	2.3919E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	7.1977E-05
35.	B1LF	2.1141E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.3616E-05
36.	RVBF	2.1101E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.3496E-05
37.	VT1BF	2.0768E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	6.2494E-05
38.	CCPRF	1.8163E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.4657E-05
39.	A1LF	1.7952E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.4019E-05
40.	RVAF	1.7912E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.3900E-05
41.	VT1AF	1.7519E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.2718E-05
42.	DGF	1.7519E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.2718E-05
43.	CTMUF	1.6898E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.0850E-05
44.	OGF	1.6748E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	5.0398E-05
45.	V3F	1.6575E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.9877E-05
46.	PAF	1.6434E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.9453E-05
47.	DEF	1.6099E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.8445E-05
48.	B2LF	1.5929E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.7932E-05
49.	BALF	1.5793E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.7524E-05
50.	OTF	1.5654E-01	1.0000E+00	0.0000E+00	0.0000E+00	1.0000E+00	4.7105E-05

Table 3.4-4 (Page 1 of 2). Non-Guaranteed Failed Split Fractions Sorted by Importance (RISKMAN Generated)

MODEL Name: Watts Bar
Split Fraction Importance for Core Damage
Sorted by Importance

......	SF Name...	Importance.....	Achievement..	Reduction...	Derivative..	SF Value.......	Frequency
1.	MU4	2.4079E-01	1.0539E+00	9.5275E-01	3.0429E-05	4.6724E-01	7.2457E-05
2.	SED	2.1104E-01	8.4988E+00	7.8900E-01	2.3200E-03	2.7368E-02	6.3507E-05
3.	OGR11	1.4983E-01	1.4186E+00	8.5673E-01	1.6907E-04	2.5500E-01	4.5085E-05
4.	TPR1	1.0504E-01	1.0183E+00	9.2273E-01	2.8756E-05	8.0860E-01	3.1610E-05
5.	GA1	1.0479E-01	1.6215E+00	9.0200E-01	2.1652E-04	1.3620E-01	3.1532E-05
6.	RT1	9.4822E-02	5.8720E+02	9.0526E-01	1.7643E-01	1.6159E-04	2.8533E-05
7.	TP1	8.6507E-02	1.6286E+00	9.5736E-01	2.0198E-04	6.3520E-02	2.6031E-05
8.	PL1	8.3795E-02	1.0204E+00	9.6009E-01	1.8142E-05	6.6200E-01	2.5215E-05
9.	CCPR1	8.2440E-02	5.6882E+00	9.2329E-01	1.4338E-03	1.6100E-02	2.4808E-05
10.	GB2	7.5046E-02	1.3007E+00	9.3144E-01	1.1110E-04	1.8570E-01	2.2583E-05
11.	RR1	6.4192E-02	2.2791E+01	9.3581E-01	6.5766E-03	2.9371E-03	1.9317E-05
12.	EB1	6.0558E-02	2.3395E+00	9.4116E-01	4.2079E-04	4.2076E-02	1.8223E-05
13.	B11	5.3476E-02	7.5790E+01	9.4708E-01	2.2521E-02	7.0702E-04	1.6092E-05
14.	MU2	5.1784E-02	1.0247E+00	9.8122E-01	1.3090E-05	4.3170E-01	1.5583E-05
15.	CE1	4.7819E-02	2.7210E+01	9.5348E-01	7.9009E-03	1.7716E-03	1.4390E-05
16.	MU6	4.0614E-02	1.0028E+00	9.9739E-01	1.6242E-06	4.8403E-01	1.2222E-05
17.	VA1	4.0207E-02	5.9553E+00	9.6342E-01	1.5021E-03	7.3280E-03	1.2099E-05
18.	REC1	3.8825E-02	1.9938E+00	9.6117E-01	3.1072E-04	3.7600E-02	1.1683E-05
19.	A11	3.4926E-02	4.2087E+01	9.6576E-01	1.2374E-02	8.3262E-04	1.0510E-05
20.	RR5	3.1465E-02	3.8779E+00	9.6853E-01	8.7549E-04	1.0815E-02	9.4684E-06
21.	RB5	3.1009E-02	2.2422E+00	9.7094E-01	3.8254E-04	2.2860E-02	9.3312E-06
22.	REC2	2.9478E-02	1.7545E+00	9.7052E-01	2.3592E-04	3.7600E-02	8.8705E-06
23.	RA2	2.5371E-02	1.8594E+00	9.7989E-01	2.6467E-04	2.2860E-02	7.6345E-06
24.	TP3	2.5112E-02	1.1718E+00	9.8655E-01	5.5731E-05	7.2600E-02	7.5566E-06
25.	GD3	2.3033E-02	1.0407E+00	9.8951E-01	1.5402E-05	2.0500E-01	6.9311E-06
26.	REC3	2.2400E-02	1.2256E+00	9.7760E-01	7.4631E-05	9.0320E-02	6.7406E-06
27.	AC1	2.2271E-02	1.6826E+02	9.7777E-01	5.0338E-02	1.3290E-04	6.7018E-06
28.	GC3	2.1696E-02	1.0114E+00	9.9618E-01	4.5727E-06	2.5170E-01	6.5286E-06

Table 3.4-4 (Page 2 of 2). Non-Guaranteed Failed Split Fractions Sorted by Importance (RISKMAN Generated)

......	SF Name...	Importance.....	Achievement..	Reduction...	Derivative..	SF Value.......	Frequency
29.	BC21	2.1527E-02	1.1026E+00	9.7863E-01	3.7304E-05	1.7240E-01	6.4778E-06
30.	DA1	2.1377E-02	2.1044E+01	9.7943E-01	6.0379E-03	1.0250E-03	6.4326E-06
31.	AC2	2.0710E-02	2.1099E+00	9.8040E-01	3.3987E-04	1.7350E-02	6.2320E-06
32.	DB1	2.0509E-02	2.0794E+01	9.8030E-01	5.9622E-03	9.9437E-04	6.1715E-06
33.	AE5	1.9589E-02	3.8303E+00	9.8044E-01	8.5756E-04	6.8632E-03	5.8947E-06
34.	GC2	1.8854E-02	1.0644E+00	9.8675E-01	2.3364E-05	1.7060E-01	5.6735E-06
35.	AA2	1.8832E-02	2.9300E+01	9.8120E-01	8.5216E-03	6.6400E-04	5.6670E-06
36.	GB1	1.7924E-02	1.0969E+00	9.8572E-01	3.3462E-05	1.2840E-01	5.3935E-06
37.	BA4	1.7130E-02	1.1318E+00	9.8309E-01	4.4758E-05	1.1370E-01	5.1546E-06
38.	RB6	1.6717E-02	1.1909E+00	9.8387E-01	6.2314E-05	7.7910E-02	5.0305E-06
39.	CE2	1.6575E-02	1.1030E+01	9.8350E-01	3.0232E-03	1.6423E-03	4.9876E-06
40.	SR1	1.6532E-02	1.4288E+00	9.8425E-01	1.3376E-04	3.5428E-02	4.9748E-06
41.	V37	1.6417E-02	1.3543E+00	9.8636E-01	1.1073E-04	3.7074E-02	4.9402E-06
42.	AE1	1.6059E-02	0.0000E+00	9.8396E-01	0.0000E+00	3.0425E-05	4.8324E-06
43.	RVB1	1.5510E-02	2.1147E+00	9.9047E-01	3.3830E-04	8.4810E-03	4.6671E-06
44.	BC33	1.5448E-02	5.1190E+00	9.8500E-01	1.2440E-03	3.6280E-03	4.6486E-06
45.	CTMU1	1.5269E-02	7.6904E-01	----------	-----------	2.5452E-02	4.5948E-06
46.	S22	1.5186E-02	2.1173E+00	9.8605E-01	3.4041E-04	1.2330E-02	4.5698E-06
47.	BE2	1.5025E-02	1.4208E+00	9.8500E-01	1.3114E-04	3.4430E-02	4.5213E-06
48.	MA4	1.3526E-02	2.8203E+00	9.8667E-01	5.5177E-04	7.2670E-03	4.0703E-06
49.	CI4	1.1893E-02	9.8295E-01	----------	-----------	1.1445E-01	3.5789E-06
50.	MDE1	1.1861E-02	1.1137E+01	9.8919E-01	3.0538E-03	1.0650E-03	3.5692E-06

Table 3.4-5 (Page 1 of 2). Most Important Nonguaranteed Failed Top Events

	Top Event	Probabilistic	Guaranteed Event	Total	Frequency
1.	MU	3.4971E-01	3.4006E-01	6.8978E-01	2.0757E-04
2.	SE	2.2130E-01	5.5639E-01	7.7769E-01	2.3402E-04
3.	OGR1	1.4983E-01	0.0000E+00	1.4983E-01	4.5085E-05
4.	REC	1.1366E-01	8.8620E-01	9.9986E-01	3.0088E-04
5.	TP	1.1162E-01	2.5582E-02	1.3720E-01	4.1286E-05
6.	RT	1.0620E-01	0.0000E+00	1.0620E-01	3.1958E-05
7.	RR	1.0518E-01	7.1771E-01	8.2289E-01	2.4762E-04
8.	TPR	1.0504E-01	2.5582E-02	1.3063E-01	3.9308E-05
9.	GA	1.0479E-01	2.0125E-03	1.0680E-01	3.2138E-05
10.	GB	9.4282E-02	1.9650E-03	9.6247E-02	2.8962E-05
11.	PL	8.3795E-02	0.0000E+00	8.3795E-02	2.5215E-05
12.	CCPR	8.2440E-02	1.8163E-01	2.6407E-01	7.9464E-05
13.	EB	6.5686E-02	0.0000E+00	6.5686E-02	1.9766E-05
14.	CE	6.4394E-02	1.5505E-01	2.1944E-01	6.6034E-05
15.	RB	5.7732E-02	6.6125E-01	7.1898E-01	2.1635E-04
16.	B1	5.3476E-02	1.5420E-01	2.0768E-01	6.2494E-05
17.	GD	4.9785E-02	8.9896E-04	5.0684E-02	1.5252E-05
18.	BC	4.7463E-02	3.8895E-01	4.3641E-01	1.3132E-04
19.	AC	4.4632E-02	5.6933E-01	6.1396E-01	1.8475E-04
20.	GC	4.3541E-02	7.4699E-04	4.4288E-02	1.3327E-05
21.	VA	4.1045E-02	4.4129E-01	4.8233E-01	1.4514E-04
22.	AE	3.6698E-02	1.1835E-01	1.5505E-01	4.6657E-05
23.	A1	3.4926E-02	1.4027E-01	1.7519E-01	5.2718E-05
24.	BE	3.2791E-02	1.1647E-01	1.4926E-01	4.4915E-05
25.	RA	2.8355E-02	7.3376E-01	7.6212E-01	2.2933E-04
26.	MB	2.3698E-02	3.5094E-01	3.7464E-01	1.1274E-04
27.	DS	2.2452E-02	7.1382E-01	7.3627E-01	2.2156E-04
28.	MA	2.2182E-02	3.3291E-01	3.5509E-01	1.0685E-04
29.	DA	2.1377E-02	1.3810E-02	3.5187E-02	1.0588E-05
30.	RVB	2.0734E-02	2.1101E-01	2.3174E-01	6.9735E-05
31.	DB	2.0509E-02	1.3055E-02	3.3564E-02	1.0100E-05
32.	V3	2.0338E-02	1.6575E-01	1.8609E-01	5.5997E-05

Table 3.4-5 (Page 2 of 2). Most Important Nonguaranteed Failed Top Events

	Top Event	Probabilistic	Guaranteed Event	Total	Frequency
33.	AA	1.9447E-02	1.2082E-01	1.4027E-01	4.2209E-05
34.	BA	1.8686E-02	1.3552E-01	1.5420E-01	4.6402E-05
35.	SR	1.8326E-02	1.7782E-02	3.6108E-02	1.0865E-05
36.	S2	1.6680E-02	6.6129E-01	6.7797E-01	2.0401E-04
37.	CTMU	1.5269E-02	1.6898E-01	1.8425E-01	5.5444E-05
38.	TB	1.5049E-02	7.7165E-01	7.8670E-01	2.3673E-04
39.	CI	1.3789E-02	8.1795E-03	2.1968E-02	6.6106E-06
40.	ZB	1.2974E-02	4.4040E-03	1.7378E-02	5.2293E-06
41.	MDE	1.1861E-02	0.0000E+00	1.1861E-02	3.5692E-06
42.	BB	1.1043E-02	5.0684E-02	6.1727E-02	1.8575E-05
43.	DE	1.0959E-02	1.6099E-01	1.7195E-01	5.1743E-05
44.	PA	1.0896E-02	1.6434E-01	1.7524E-01	5.2732E-05
45.	AB	1.0507E-02	4.4288E-02	5.4795E-02	1.6489E-05
46.	PB	1.0359E-02	1.4486E-01	1.5522E-01	4.6710E-05
47.	SU	9.0033E-03	0.0000E+00	9.0033E-03	2.7093E-06
48.	OS	8.3262E-03	0.0000E+00	8.3262E-03	2.5055E-06
49.	ZA	8.1407E-03	9.2556E-03	1.7396E-02	5.2348E-06
50.	CCSR	7.9888E-03	1.3034E-01	1.3833E-01	4.1625E-05

Table 3.4-6 (Page 1 of 2). Guaranteed Failed Top Events Sorted by Total Importance

	Top Event	Probabilistic	Guaranteed Event	Total	Frequency
1.	MELT	1.0000E+00	0.0000E+00	1.0000E+00	3.0092E-04
2.	REC	8.8620E-01	1.1366E-01	9.9986E-01	3.0088E-04
3.	MELTI	9.5811E-01	0.0000E+00	9.5811E-01	2.8831E-04
4.	IYA	9.5811E-01	0.0000E+00	9.5811E-01	2.8831E-04
5.	INTPR	9.2109E-01	0.0000E+00	9.2109E-01	2.7717E-04
6.	RHRS	8.9323E-01	0.0000E+00	8.9323E-01	2.6879E-04
7.	RR	7.1771E-01	1.0518E-01	8.2289E-01	2.4762E-04
8.	TB	7.7165E-01	1.5049E-02	7.8670E-01	2.3673E-04
9.	SE	5.5639E-01	2.2130E-01	7.7769E-01	2.3402E-04
10.	RA	7.3376E-01	2.8355E-02	7.6212E-01	2.2933E-04
11.	CSA	7.4588E-01	2.5183E-03	7.4840E-01	2.2520E-04
12.	DP	7.3627E-01	1.0179E-03	7.3729E-01	2.2186E-04
13.	DS	7.1382E-01	2.2452E-02	7.3627E-01	2.2156E-04
14.	S1	7.3380E-01	1.6237E-03	7.3543E-01	2.2130E-04
15.	RB	6.6125E-01	5.7732E-02	7.1898E-01	2.1635E-04
16.	MU	3.4006E-01	3.4971E-01	6.8978E-01	2.0757E-04
17.	CSB	6.7911E-01	4.7183E-03	6.8383E-01	2.0578E-04
18.	S2	6.6129E-01	1.6680E-02	6.7797E-01	2.0401E-04
19.	CSR	6.6238E-01	0.0000E+00	6.6238E-01	1.9932E-04
20.	VB	6.5963E-01	1.8067E-03	6.6144E-01	1.9904E-04
21.	CH	6.4908E-01	4.2900E-03	6.5337E-01	1.9661E-04
22.	CSI	6.4002E-01	0.0000E+00	6.4002E-01	1.9259E-04
23.	SI	6.3507E-01	0.0000E+00	6.3507E-01	1.9110E-04
24.	CM	6.3030E-01	0.0000E+00	6.3030E-01	1.8967E-04
25.	AC	5.6933E-01	4.4632E-02	6.1396E-01	1.8475E-04
26.	MS	5.7254E-01	0.0000E+00	5.7254E-01	1.7229E-04
27.	CAV	5.0116E-01	0.0000E+00	5.0116E-01	1.5081E-04
28.	VC	4.8243E-01	3.5259E-05	4.8246E-01	1.4518E-04
29.	VA	4.4129E-01	4.1045E-02	4.8233E-01	1.4514E-04
30.	BC	3.8895E-01	4.7463E-02	4.3641E-01	1.3132E-04
31.	MB	3.5094E-01	2.3698E-02	3.7464E-01	1.1274E-04
32.	MA	3.3291E-01	2.2182E-02	3.5509E-01	1.0685E-04

Table 3.4-6 (Page 2 of 2). Guaranteed Failed Top Events Sorted by Total Importance

	Top Event	Probabilistic	Guaranteed Event	Total	Frequency
33.	RQ	3.5379E-01	0.0000E+00	3.5379E-01	1.0646E-04
34.	CCPR	1.8163E-01	8.2440E-02	2.6407E-01	7.9464E-05
35.	PD	2.3919E-01	5.6422E-03	2.4484E-01	7.3675E-05
36.	RVB	2.1101E-01	2.0734E-02	2.3174E-01	6.9735E-05
37.	CE	1.5505E-01	6.4394E-02	2.1944E-01	6.6034E-05
38.	B1L	2.1141E-01	0.0000E+00	2.1141E-01	6.3616E-05
39.	VT1B	2.0768E-01	0.0000E+00	2.0768E-01	6.2494E-05
40.	B1	1.5420E-01	5.3476E-02	2.0768E-01	6.2494E-05
41.	RVA	1.7912E-01	7.7037E-03	1.8682E-01	5.6218E-05
42.	V3	1.6575E-01	2.0338E-02	1.8609E-01	5.5997E-05
43.	CTMU	1.6898E-01	1.5269E-02	1.8425E-01	5.5444E-05
44.	A1L	1.7952E-01	0.0000E+00	1.7952E-01	5.4019E-05
45.	PA	1.6434E-01	1.0896E-02	1.7524E-01	5.2732E-05
46.	VT1A	1.7519E-01	0.0000E+00	1.7519E-01	5.2718E-05
47.	A1	1.4027E-01	3.4926E-02	1.7519E-01	5.2718E-05
48.	DG	1.7519E-01	0.0000E+00	1.7519E-01	5.2718E-05
49.	DE	1.6099E-01	1.0959E-02	1.7195E-01	5.1743E-05
50.	OG	1.6748E-01	1.5004E-03	1.6898E-01	5.0850E-05

Table 3.4-7. Watts Bar Operator Action Importance to Core Damage Frequency

Operator Action Designator	Description	Operator Action Failure Rate Mean Value	Importance to Core Damage Frequency (%)
HASE2	Stop RCPs upon Loss of Train A CCS or RCP Cooling Path	3.71-02	19
CCPR1	Align ERCW to Charging Pump, Given Loss of CCS Train A	1.61-02	8
HAEB1	Trip CRD Motor Generator Power and Initiate Boration, Given ATWS	7.33-03	5
HARR1	Align High Pressure Recirculation, Given Auto Swapover Succeeds	4.19-03	5
HAAC2	Isolate CCS Train A from Spent Fuel Pool Heat Exchanger, CCSA or CCSTL Initiating Event	2.29-02	5
HAMU2	Make Up RWST Inventory, Given LOCA with Loss of Recirculation	2.06-01	4
HAAEIE	Start Standby ERCW Pump To Avert Plant Trip, Given Running Pump Fails during Normal Operation	3.87-04	2
HAMU1	Make Up RWST Inventory Following an SGTR Event	7.28-03	2

Note: Exponential notation is indicated in abbreviated form; e.g., e.g., 3.71-02 = 3.71×10^{-02}.

Table 3.4-8. Watts Bar Contributors to Core Damage

Case	Frequency per Reactor Year	Percent of Total CDF
RCP Scenarios		
All RCP Seal Failures	2.3-4	70
— Unit Blackouts	3.0-5	9
— Seal LOCAs with One or More Shutdown Boards Available	2.0-4	61
Total Losses of Unit 1 CCS	1.3-4	38
Failure To Trip RCPs, Given CCS Train A Lost, Resulting in an Early Seal LOCA	6.4-5	19
Pressurizer PORV LOCAs	3.4-5	10
ATWS	3.2-5	10
CCS/AFW Pump Area Ventilation	6.0-6	2
Total Core Damage Frequency: 3.3-04 per reactor-year		

Notes:

1. Exponential notation is indicated in abbreviated form; e.g., 2.3-4 = 2.3×10^{-4}.
2. Cases may be subsets of other cases.

Table 3.4-9. Watts Bar Level 1 Containment States with Core Damage		
Containment States	**Frequency Per Reactor-Year**	**Percent of Total CDF**
Containment Intact	3.2×10^{-4}	96
Unisolated Containment	6.7×10^{-6}	2
Containment Bypass	5.9×10^{-6}	2
Total Core Damage Frequency: 3.3×10^{-4} per reactor-year		
Note: Failures due to containment phenomena resulting in increased containment pressure are not included in this table.		

Table 3.4-10 (Page 1 of 2). Importance Evaluation for the Decay Heat Removal Function			
System or Function	**Top Event**	**Comment**	**Percentage of CDF in Which Event Is Failed**
1. Main Feedwater	MF	Losses from All Causes MFF - Failure due to Initiator or Loss of Support MF1 - All Support Available OF1 - Failure to Realign MFW	9.9 9.7 0.0 0.2
	OF	FWF - Failure due to Initiator or Loss of Support in an ATWS	0.3
	FW	ATWS	5.7
2. Auxiliary Feedwater	AF	Failure from AFW Valves Only - All Causes Failure from AFW Valves Only - For ATWS Sequences	0.25 0.21
	MA,MB, TP, TPR	Failure of all Three Pumps - All Causes	9.3
3. Feed and Bleed Cooling	OB	Unavailability from All Causes OBF - Loss of all Pumps or Support for Pressurizer Valves OB1 - Failure To Initiate	5.4 4.9 0.5
4. Residual Heat Removal	DS	DS6 - Failure to Depressurize with Steam Generator Tube Rupture Not Isolated	1.0
		DS8 - Failure To Depressurize with Ruptured Steam Generator Initially Isolated	0.5
	DP	DP7 - Failure to Depressurize RCS Given Support For Both Spray Trains, for Steam Generator Tube Rupture Sequences Failure to Align for RHR Cooling - All Causes	0.1
	RD	RB2/RB3/RB5/RB6 - Failure of Both Trains of RHR, One Or More Trains Of Support Systems Available	1.5

Table 3.4-10 (Page 2 of 2). Importance Evaluation for the Decay Heat Removal Function			
System or Function	**Top Event**	**Comment**	**Percentage of CDF in Which Event Is Failed**
4. Residual Heat Removal (continued)	RA,RB	RAF*RBF - Failure of Both Trains of RHR due Only to Loss of Support systems	5.7
	RI	RI1 - RHR Cold Leg Injection Paths and RWST Suction Line	Negligible

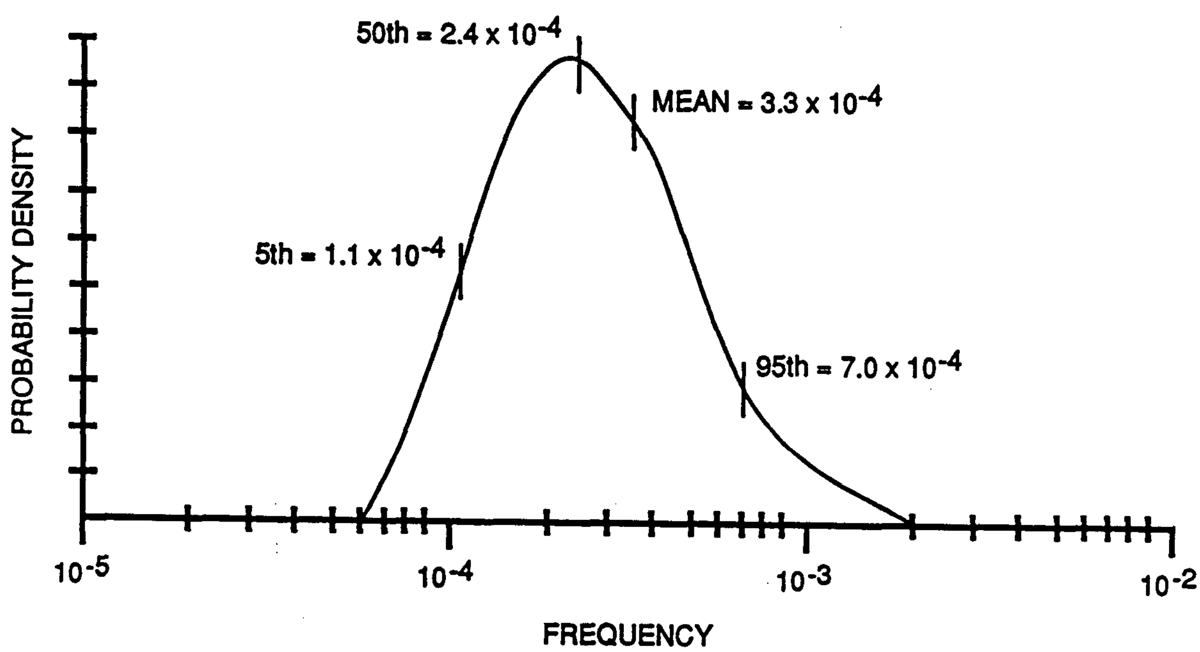

Figure 3.4-1. Probability Distribution for Watts Bar Core Damage Frequency

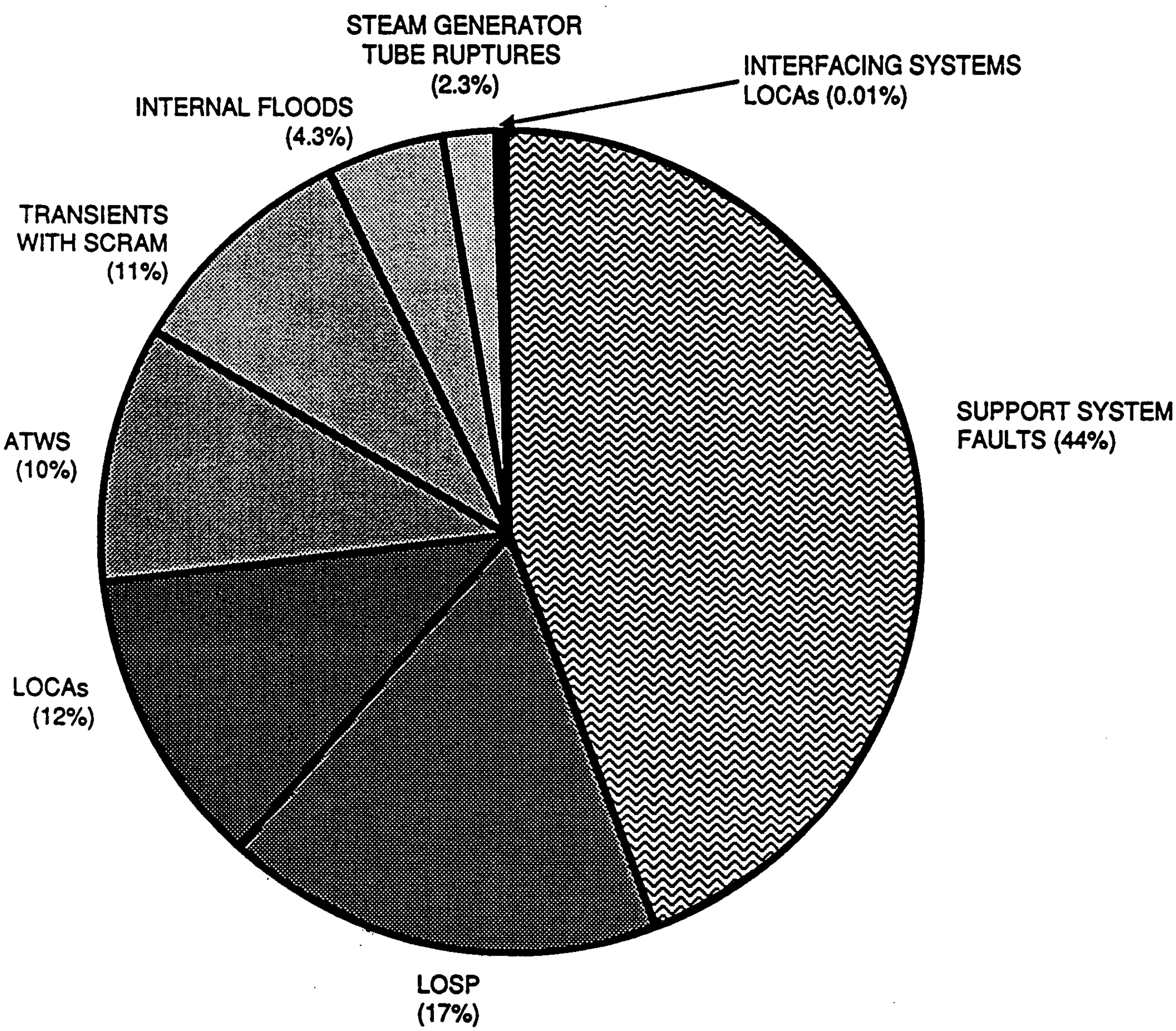

Figure 3.4-2. Category of Watts Bar Initiators Contributing to Core Damage