WOLF CREEK
NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

INDIVIDUAL PLANT EXAMINATION
SUMMARY REPORT

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Submitted By

WOLF CREEK NUCLEAR OPERATING CORPORATION

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1.0 EXECUTIVE SUMMARY

In November 1988, the U.S. Nuclear Regulatory Commission (NRC) staff issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," which established a formal request for utilities to perform an Individual Plant Examination (IPE). Beyond the performance of the IPE, this letter requested utilities to identify potential improvements to address the important contributors to plant risk and implement improvements that they believe are appropriate for their plant.

In August 1989, the NRC issued Supplement 1 to Generic Letter 88-20, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," accompanied by NUREG-1335, "Individual Plant Examination Guidance." These documents provide direction for the performance of the IPE and reporting of summary information to the NRC. The period of performance was three years following utility response regarding the planned methodology and schedule for the IPE performance.

This report provides the requested information for the Wolf Creek Generating Station (WCGS). WCGS is jointly owned by the Kansas City Power & Light Company (47%), Kansas Electric Power Cooperative, Inc. (6%), and Kansas Gas and Electric Company (47%). Wolf Creek Nuclear Operating Corporation (WCNOC) is the Operating Agent for WCGS.

1.1 Background and Objectives

In its Severe Accident Policy Statement (50FR43621) issued in 1985, the NRC concluded that operating nuclear plants pose no undue risk to the public health and safety. However, recognizing that these generic conclusions were derived from a diverse but small sample of the existing plants, the NRC requested that all licensees perform a "limited-scope accident safety analysis" to determine if there might be any unique plant-specific vulnerabilities leading to a core damage accident or to poor containment performance given a core damage event.

WCNOC has responded to Generic Letter 88-20 and its Supplement 1 by performing a Level 1 and Level 2 Probabilistic Risk Assessment for WCGS. WCNOC's goals in performing these analyses include fulfilling the NRC Individual Plant Examination (IPE) requirements and developing a tool which may be used to optimize planning, Plant Modification Request (PMR), and operational decisions and Justifications for Continued Operation (JCOs) on a risk basis. WCNOC intends to use the PRA as a decision optimization tool that can be used with a reasonable effort to aid in the continuation and enhancement of the safe, reliable, and efficient operation of WCGS. Specific functional objectives for the PRA include:

1. The ability to quantitatively assess changes in core damage frequency due to: (a) modifications to component and system design; (b) changes to existing operating and maintenance procedures;

(c) changes in NRC licensing requirements; (d) changes in actual components and systems availability as plant specific operating data becomes available; (e) changes in the Technical Specifications; and (f) changes in safety margins identified by core and plant deterministic analyses.

2. Develop an understanding of containment failure modes, the impact of core damage phenomena and plant features, and the impact of operator actions, and to explore the potential for developing mitigative systems and procedures to minimize the effects of severe accident sequences.

3. Allow qualitative evaluation of failure probabilities and core damage risk through identification of individual risk contributors on a sequence, system, subsystem, module, and component basis.

WCNOC's staff have performed the majority of these analyses, and have been trained in those portions performed by outside consultants. WCNOC intends that these staff members understand and utilize the PRA insights.

This report contains a summary of the methods, results, and conclusions of the IPE, in compliance with the NRC request for information contained in Generic Letter 88-20, Supplement 1 and NUREG-1335. WCNOC has retained supporting analyses, descriptions, and files pertaining to these analyses. Documentation is available at WCNOC offices for NRC review as necessary. To assist NRC reviewers, this submittal is generally structured according to the outline provided in NUREG-1335.

1.2 Plant Familiarization

The WCGS IPE began with familiarization of the staff with the as-built as-operated plant. To accomplish this familiarization, several data collection and documentation activities were undertaken in the first step in the project. System notebooks were prepared for important systems through a combination of analyst review of drawings, system descriptions, and procedures and plant walkdowns to verify the design of the systems, to become familiar with the physical layout of the plant and to visualize restorative actions or alternative systems. Plant records were reviewed to develop a knowledge of plant-specific behaviors such as component failure rates and initiating event frequencies. WCNOC engineers and operators were involved in this familiarization process as well as during various stages of review of the analyses as work progressed.

1.3 Overall Methodology

The WCGS IPE was performed by conducting a Probabilistic Risk Assessment (Level 1 PRA), including an analysis of internal flooding, and a Containment Performance Analysis (Level 2 PRA) of the plant. In performing the IPE, standard PRA systems analysis practices such as those outlined in the PRA Procedures Guide (NUREG/CR-2300) were used.

The WCGS PRA is a full scope investigation of the plant systems and operator responses to transient and accident initiator events. The focus of investigation was on the performance of a realistic assessment of the response of plant systems and the operators to potential accident sequences. The models of plant systems are developed to a level of detail that includes the performance of all key components. The success criteria used to determine whether plant systems achieve their intended safety function were determined for the most important severe accident sequences. The success criteria definition involved consideration of both system capability and the timing of operator responses and system recovery. The Modular Accident Analysis Program (MAAP) code was utilized to determine key success criteria and estimate source terms.

Well known approaches for common cause failure and human error were adopted for the WCGS PRA. In determining the parametric values to be used in the quantification, the available industry databases were reviewed to assure that events and failure modes appropriate for the WCGS and its equipment were utilized. For common cause analyses, the Multiple Greek Letter (MGL) method was used. Human Reliability Analysis (HRA) was performed using THERP (Technique for Human Error Rate Prediction) methodology. Realism was achieved through detailed modeling of operator actions and thorough treatment of operator recovery.

The WCGS containment performance and source term analysis includes plant models and physical processes which reflect the overall plant behavior following core damage. Attention was paid to the interface between the traditional systems analysis and containment analysis portions of the PRA, through the development of a containment safeguards event tree and containment event tree. This process coupled a probabilistic assessment of containment response to postulated initiating events with a physical model to examine plant response. The probabilistic models are embodied in Containment Event Trees (CETs) while the plant physical model is defined in a MAAP parameter file.

1.4 Summary of Major Findings

WCNOC has not identified any vulnerabilites. The results of the PRA indicate that the station blackout event contributes approximately 45% to the total core damage frequency, followed by loss of offsite power contributing 12%. WCNOC has also identified two separate flooding sequences, that combined, contribute approximately 16% to the total core damage frequency. The total core damage frequency of 4.2E-05 is considered acceptably low and typical for the WCGS vintage plant.

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

The WCGS IPE has been performed to identify and resolve severe accident issues. To assure that this purpose was accomplished, WCNOC has performed a Level 1 PRA, including an analysis of internal flooding, and a Level 2 PRA for containment performance analysis.

WCNOC has conducted the PRA to be in compliance with NRC Generic Letter 88-20 and its Supplement. The approach to the PRA has been to perform realistic evaluations of WCGS, with the focus on learning more about the capability of the plant to prevent severe accidents and on the need to effectively respond to accident sequence progression in the event of a severe accident. These evaluations were carried out in a manner that will support management decision-making processes relative to potential enhancement of plant design and operation, aimed at reduction of the risk of core damage and poor containment performance. The PRA process included task planning, methodology development, analysis and documentation. The PRA can be maintained, updated, and used.

The WCGS PRA consisted of major tasks that involved WCNOC and contractor personnel and resources:

1. Project Management
2. Training and Technology Transfer
3. Plant Definition and Information Gathering
4. Initiating Event Analysis
5. Event Tree Analysis
6. Systems Analysis
7. Database Development
8. Human Interaction Assessment, Including Recovery Actions
9. Dependency and Common Failure Analysis
10. Internal Flooding Analysis
11. Final Core Damage Analysis and Scoping Model Development
12. Sensitivity Analysis
13. Containment Performance Analysis
14. Final Report and IPE Report

The analysis was divided into four phases of effort, although considerable overlap occurred during the progression of the above tasks. Phase 1 consisted of plant information collection, data collection, and data analysis. Phase 2 consisted of the event tree and systems analysis to establish system responses to initiating events and the resulting dominant accident sequences. Phase 3 consisted of the performance of tasks related to containment response characterization and the determination of source terms. Phase 4 consisted of the evaluation and documentation of the results.

2.2 Conformance with Generic Letter and Supporting Material

Generic Letter 88-20 requested each utility to perform an Individual Plant Examination for the purpose of: (1) developing an appreciation of severe accident behavior; (2) understanding the most likely severe accident sequences that could occur at its plant; (3) gaining a more quantitative understanding of the overall probabilities of core damage and fission product releases; and, if necessary, (4) reducing the overall probabilities of core damage and fission product releases.

General requirements provided in the Generic Letter for fulfilling the stated purpose are:

(1) The utility staff should be used to the maximum extent possible in the performance of the IPE to insure that they: (a) understand the plant procedures, design, operation, maintenance and surveillance; (b) understand the quantification of the expected sequence frequencies; (c) determine the leading contributors to core damage and unusually poor containment performance; (d) identify proposed plant improvements for prevention and mitigation; (e) examine each of the proposed improvements; and (f) identify which proposed improvements will be implemented and their schedule.

(2) The utility should proceed with the examination of internally initiated events including internal flooding.

(3) The method of examination should either be a PRA that follows the PRA procedures described in NUREG/CR-2300, NUREG/CR-2815 or NUREG/CR-4550, "Analysis of Core Damage Frequency," plus a Containment Performance Analysis that follows the guidance of Appendix 1 to Generic Letter 88-20 or the Industry Degraded Core Rulemaking (IDCOR) front-end method with NRC enhancements, or another systematic method that is acceptable to the staff.

(4) The utility should resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," as part of the IPE.

(5) The utility should carefully examine the results of the IPE to determine if there are worthwhile prevention or mitigation measures that could be taken to reduce the frequency of core damage or improve containment performance.

(6) The utility should report the results of the IPE to the NRC consistent with the criteria provided in the Generic Letter and subsequent guidance provided in NUREG-1335.

(7) The utility should document the examination in a traceable manner and retain it for the duration of the license unless superseded.

(8) The utility should conduct future evaluations for accident management and external events when the guidance for them have been developed.

In response to the Generic Letter, WCNOC issued a letter on October 31, 1989 stating its intent to perform a full scope Level 1 PRA and a Containment Performance Analysis for the WCGS in order to identify, evaluate, and resolve severe accident issues germane to the plant.

WCNOC has invested substantial manpower as well as financial resources into the performance of the PRA. A permanently assigned staff has been involved in all aspects of the PRA. Other WCGS personnel have been involved in various aspects of the evaluation as needed. In addition, a training effort was undertaken to ensure that WCNOC personnel who had a need for understanding the evaluation or parts of the evaluation were informed concerning the risk significance of the results and the plant response, and understood the bases of the PRA.

WCNOC has reviewed the results of the PRA for areas where plant improvements could be effectively made with emphasis on core damage prevention. WCNOC is continuing to evaluate a number of potential improvements even though the over all results of the PRA indicate that the core damage frequency and containment performance are within the expected limits.

2.3 General Methodology

The WCGS IPE program consisted of several discrete major tasks covering the Level 1 and 2 PRA. The IPE was conducted using standard systems analysis practices such as those outlined in NUREG/CR-2300, "PRA Procedures Guide - a Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants" and NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide." A comprehensive task breakdown was developed for the WCGS IPE in order to organize the work to be accomplished. Project management was organized to ensure complete and efficient performance of the effort. Training and technology transfer between contractor and WCNOC personnel included initial orientation to PRA technology and a training session on each major task to ensure the in-house knowledge and capability to utilize the IPE. The technical tasks carried out by WCNOC and its contractors are described in the following paragraphs. Guidelines were developed for many of the key technical tasks.

1. Plant Definition and Information Gathering - This task involved the identification and collection of relevant plant operating and equipment information needed to support the development of plant-specific probabilistic risk models. This information was used to develop system notebooks, to model accident sequences (event trees and fault trees), and to identify critical plant systems, initiating events, and system dependencies. The information used is summarized below in Section 2.4.

2. Initiating Event Analysis - The selection of accident initiating events for the WCGS IPE was made from the collection and analysis of plant trip data. Additionally, plant-specific data evaluation was supplemented by a review of previous analysis of similar plants for their applicability to WCGS. WCGS trip data was collected from Licensee Event Reports (LERs) to identify actual trip events, power level at which the trip occurred, and the failure(s) which caused the trip.

The WCGS accident initiating events included large loss of coolant accident (LOCA), medium LOCA, small LOCA, Steam Generator Tube Rupture (SGTR), interfacing systems LOCA, vessel failure, transient with power conversion, transient without power conversion, steamline/feedline break, loss of offsite power, station blackout, Anticipated Transients Without Scram (ATWS), loss of component cooling water, loss of service water, and loss of a vital DC bus. Power conversion considers the availability of the main feedwater system to mitigate the effects of the transient on the plant.

Several methods were employed to estimate initiating event frequencies. For those events having sufficient WCGS data, each event found in the data was categorized as identified above and the frequency determined by the number of each event in the category. The period of study was from initial plant startup in May 1985 through December 1990. Some events were excluded because of changes in plant equipment and operations. For those events such as loss of offsite power and SGTR, the initiating event frequencies were estimated from generic data for similar plants. The frequency of loss of offsite power was estimated based upon NSAC data and WCNOC/NUMARC station blackout work, and that of SGTR was estimated from the Westinghouse plant population experience data base through 1989. For the small, medium, and large break LOCAs, basic initiating event frequencies were taken from NUREG/CR-4550. Small and medium LOCA frequency estimates also include PORV, safety valve, and random RCP seal failure frequencies which were added to the base frequencies for each category.

3. Event Tree Analysis - Plant-specific event tree models were developed for each initiator. This task included the definition of critical safety functions relevant to the initiating events, development of system level event trees and system success criteria for the various accident sequences, and incorporation of operator actions and consequential failures related to various accident sequences.

4. Systems Analyses - The WCGS plant systems were modeled with fault trees. For each system, the analysis included the development of a detailed system notebook describing the system, its operation, testing, and maintenance, the effect of accident conditions (success criteria, initiator impact, etc.), the system operating experience, the system models and assumptions, quantification, and analyst insights.

   The development of the fault tree models was performed from the top event down. Fault tree development was aided through the development of simplified piping and instrumentation diagrams (P&IDs) and fault tree modules which simplified and standardized the fault tree layouts. The fault trees were developed and quantified using the Westinghouse fault tree GRAFTER Code System.

   The fault tree models incorporate equipment ure, test, maintenance, human reliability modeling, and common cause analysis, where appropriate. Because the analysis utilized the fault tree linking approach to quantification, the appropriate support systems are also included in the fault tree models.

5. Database Development - Plant-specific information was collected from a variety of sources, including work requests, logs, and completed test procedures for selected major component groups for the period from the start of commercial operation, September 3, 1985, through December 31, 1989. For motor operated valves, the time period was from September 3, 1985 through December 31, 1988. This data included information regarding component failures, demands, and run times, and system operating train unavailability due to testing and maintenance. Most of the data used in the WCGS IPE was generic data from IEEE-500, NUREG/CR-4550, and other sources. Plant data were utilized, when statistically meaningful, to estimate failure rates through either classical means or through the use of Bayesian techniques.

6. Human Interaction Assessment - Detailed models were developed to represent the interaction of operators and other plant staff with plant systems and equipment during normal operation and during transient and other accident conditions. The THERP methodology (Technique for Human Error Rate Prediction) was used for the human reliability analysis.

7. Dependency and Common Cause Failure Analysis - This task involved the qualitative and quantitative assessment of system dependencies, interactions, and common susceptibilities which can lead to degradation of the performance of multiple systems or components.

8. Internal Flooding Analysis - A separate analysis was performed to determine whether there are areas in the plant that are susceptible to flooding from internal sources, whether there is sensitive equipment in those areas that could cause a plant shutdown or result in a failed safety system, and if so, the contribution to core damage from the flooding of those areas. The event trees from the other internal event initiators were used to quantify the contribution of flooding to core damage frequency.

9. Core Damage Analysis - The WCGS system fault trees and event tree accident sequences were integrated and quantified to obtain accident sequence cutsets, core damage frequencies for all accident sequences, and identify dominant accident sequences among all event tree results. Sensitivity analysis helps to identify those system failures, operator actions, system interactions, and recovery actions for which operational changes, equipment enhancements, and new procedures may be considered to reduce the probability of core damage. The Westinghouse WLINK Code System was used to perform the accident sequence quantification. The results are the essential data needed as input for the Containment Performance Analysis.

10. Level 2 PRA, Containment Performance Analysis - A containment event tree (CET) was developed with an emphasis on the most probable consequences, including nodes for potential recovery actions. Source terms were developed by analyzing dominant or bounding

accident sequences that led to containment failure using the MAAP code. Source terms were binned into release categories based on the type, timing, and magnitude of the release.

2.4 Information Assembly

The WCGS IPE team reviewed and assembled information from plant-specific sources, similar plant studies, Westinghouse information, and generic sources. Plant walkdowns were an important part of the data collection effort. Information was assembled to familiarize the analysts with the plant, determine the important initiating events and quantify them, determine the component and system failure rates, perform various supporting analyses (e.g., common cause failures), conduct the evaluation of internally initiated flooding events, and develop plant layout insights through the use of plant walkdowns. Walkdowns were specifically used to identify dependencies, identify alternative systems and alignments to address accident management strategies, search for plant characteristics that could impact the transport of radionuclides and the behavior of the containment. Table 2.4-1 provides a list of the important sources of information that were reviewed. The listings of individual references used are documented in the WCGS PRA.

The WCGS IPE team used the current revision of drawings, design documents (such as the USAR), and plant procedures as listed in the PRA documentation. The PRA models reflect the as-built, as-operated condition of the WCGS.

Much of the information was collected at the outset of the project. This information is maintained at the WCNOC offices.

Detailed system notebooks were developed for major systems and several miscellaneous systems that were expected to have an influence on the WCGS PRA results. In addition, notebooks were developed for major analyses of the PRA project (e.g., initiating events, internal flooding, etc.). Plant information sources identified in Table 2.4-1 were used to develop system descriptions and models. Both plant specific and generic sources identified were used to define component availabilities, initiating events and initiating event frequency, important accident sequences, potentially important modeling features, common cause failure rates, and human reliability data. Subsequent sections of this report provide a more detailed discussion of the use of the information collected.

Plant walkdowns were conducted by members of the WCGS PRA team who were responsible for the evaluation of a specific plant system, the containment and/or its systems, and the evaluation of internal flooding. The walkdown teams were led by WCNOC personnel who were knowledgeable about the plant systems and the containment and their detailed arrangement.

Figures 2.4-1 through 2.4-9 show the general areas of the plant evaluated in the walkdowns. The systems and plant environment of most concern are contained primarily in the Auxiliary Building and the Control Building; however, several other buildings or areas were examined because equipment of important systems are located in them. The areas or buildings in which walkdowns were made are:

Reactor Building

Auxiliary Building

Turbine Building

Control Building

Essential Service Water Pumphouse

Circulating Water Screenhouse

Diesel Generator Building

Outside Grounds

Several walkdowns were conducted and are summarized below.

System Walkdowns:

The system fault tree analysts conducted walkdowns of the WCGS systems. WCGS plant personnel were helpful in assisting the team to become familiar with equipment locations, system operations, and test/maintenance practices.

Containment Walkdowns:

The PRA personnel performed a containment walkdown. The team walked down various elevations and compartments of the containment. Potential containment bypass methods were also examined by walkdowns of the Auxiliary Building during this time.

Operator Talkthroughs:

In order to perform the human reliability analysis, the analysts studied appropriate procedures and discussed these actions with plant operators. This talkthrough provided the analysts with insights into the complexity of the tasks, the familiarity of the operators with the required task steps, and the time constraints involved. Analysts involved in the human reliability analysis conducted/participated in the talkthrough.

Internal Flooding Walkdown:

A walkdown was performed primarily to gain an understanding of the special relationships of components and equipment to the various specific hazards. A subsequent walkdown was performed to obtain the relationship for specific hazards identified during analysis. Analysts assigned to this task were the primary participants in this walkdowns.

Checklists were developed to collect information that was needed for the PRA analysis. The scope of these checklists for WCGS included:

Verify components against plant drawings,

Assess room environment (cooling, barriers, open area, etc.),

Assess diligence of maintenance (cleanliness, leaks, equipment condition, stored special equipment, etc.),

Identify local controls and indications available,

Verify dependencies,

Identify flooding information (critical equipment, source of flooding, room drainage, etc.),

Identify potential room hazards.

TABLE 2.4-1
WCGS IPE INFORMATION SOURCES
Page 1 of 3

Plant Specific

WCGS Updated Safety Analysis Report (USAR).

WCGS Technical Specifications, NUREG-1136, June 1985.

WCGS System Descriptions.

Plant System Piping and Instrumentation Diagrams.

Plant Equipment Location Drawings.

Electrical One-Line Diagrams and Schematic Drawings.

Engineered Safety Features Actuation and Reactor Trip Signals Functional Diagrams.

Emergency Procedures (EMGs).

Normal Operating Procedures.

WCGS Off-Normal Procedures (OFNs).

WCGS Alarm Procedures (ALRs)

WCGS Licensee Event Reports (LERs).

Inservice Testing Program for Pumps and Valves, WCOP-02, Rev. 8.

WCGS Maintenance Instruction Set.

WCGS Surveillance Test (STS) Procedures.

Generic Sources

NUREG-0651	"Evaluation of Steam Generator Tube Rupture Events"
NUREG-0909	"January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," April 1982.
NUREG-1335	"Individual Plant Examination: Submittal Guidance," August 1989.
NUREG/CR-0677	"The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes"
NUREG/CR-1174	"Evaluation of Systems Interactions in Nuclear Power Plants," August 1989.

TABLE 2.4-1
WCGS IPE INFORMATION SOURCES
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NUREG/CR-1278	"Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983.
NUREG/CR-2300	"PRA Procedures Guide," January 1983.
NUREG/CR-2678	"Flood Risk Analysis Methodology Development Project Final Report," June 1982.
NUREG/CR-2728	"Interim Reliability Evaluation Program Procedures Guide," January 1983.
NUREG/CR-2815	"Probabilistic Safety Analysis Procedure Guide," Rev. 1, August 1985.
NUREG/CR-3268	"Modular Fault Tree Analysis Procedure Guide," Volumes 1-4, August 1983.
NUREG/CR-3862	"Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," EG&G Idaho, Inc., May 1985.
NUREG/CR-4550	"Analysis of Core Damage Frequency from Internal Events," Volumes 1-4, Rev. 1, January 1990.
NUREG/CR-4780	"Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Volume 1, February 1988 and Volume 2, January 1989.
WASH-1400	"Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear power Plants," October 1975.
EPRI NP-3967	"Classification and Analysis of Reactor Operating Experience Involving Dependent Events," June 1985.
EPRI TR-1000743	MAAP PWR Application Guidelines for Westinghouse and Combustion Engineering Plants
IEEE-500	IEEE Guide to the Collection and Presentation of Electrical Electronic Sensing Component and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations.
NSAC-147	"Loss of Off-Site Power at U.S. Nuclear Power Plants Through 1989," March 1990.
NUMARC 87-00	"Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987.

TABLE 2.4-1
WCGS IPE INFORMATION SOURCES

Generic Letter 88-20	"Individual Plant Examination for Severe Accident Vulnerabilities," November 23, 1988. (and Supplements as applicable)
AEOD/E90-07	"Effects of Internal Flooding of Nuclear Power Plants on Safety Equipment," July 1990.

IDCOR Technical reports.

INPO SOER 85-5	"Internal Flooding of Power Plant Buildings," December 1985.

Westinghouse WCAPs

WCAP-8330	"Westinghouse Anticipated Transients Without Trip Analysis," August 1974.
WCAP-9914	"PORV Sensitivity Study for LOFW-LOCA Analyses," July 1981.
WCAP-10019	"Summary Report on Reactor Vessel Integrity for Westinghouse Operation Plants," December 1981.
WCAP-10541	"Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Rev. 2, November 1986.
WCAP-10590	"Probabilistic Safety Study"
WCAP-11206	"Loss of Feed Flow, Steam Generator Tube Rupture, and Steam Line Break Thermalhydraulic Experiments (MB-2 Tests)," NUREG/CR-4751, October 1986.
WCAP-11992	"Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988.
WCAP-12231	"Station Blackout Coping Assessment for Wolf Creek Generating Station," April 1989.
WCAP-12530	"Nuclear Parameters and Operations Package for Wolf Creek Unit 1, Cycle 5," April 1990.

Figure 2.4-1

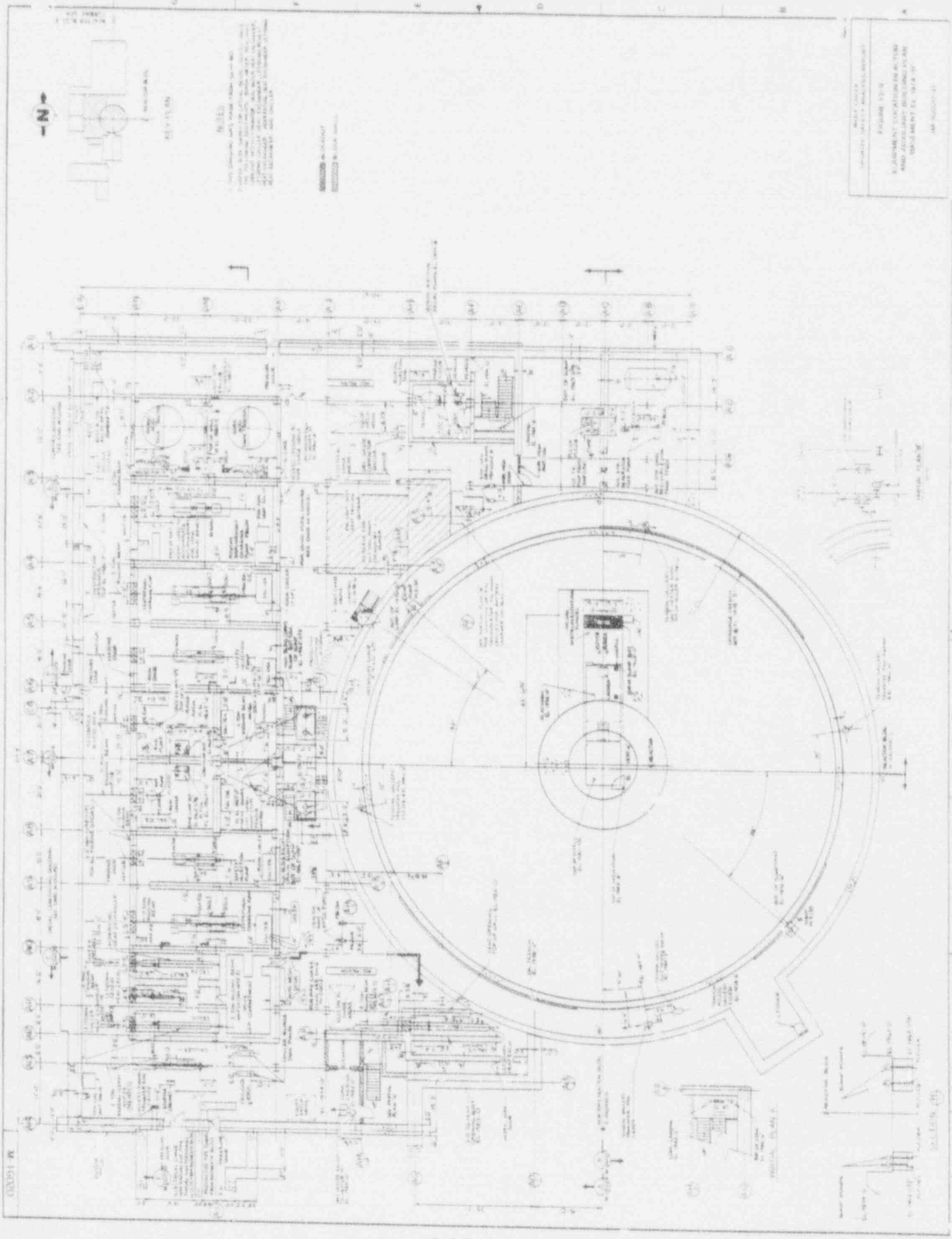


Figure 2.4-2

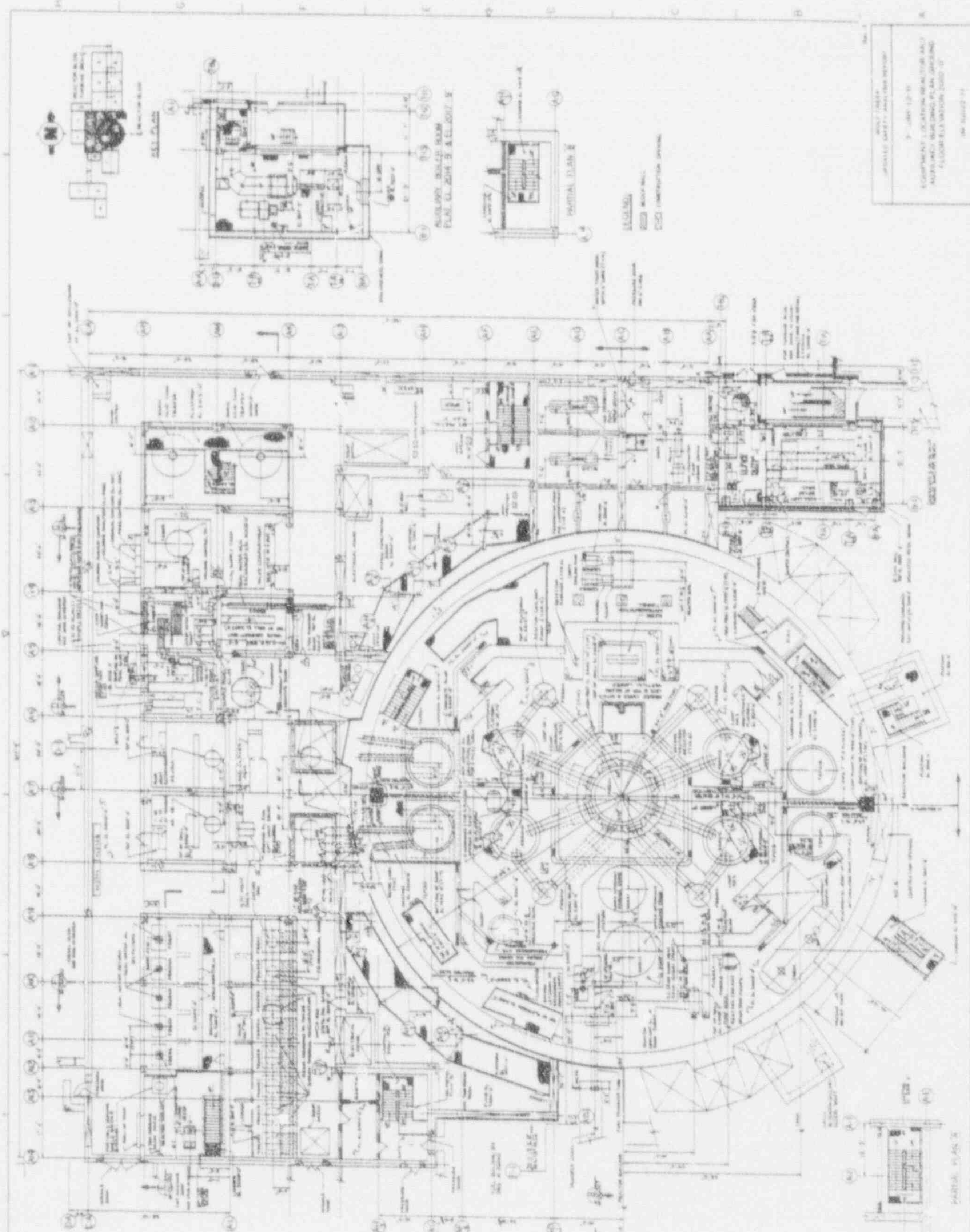


Figure 2.4-3

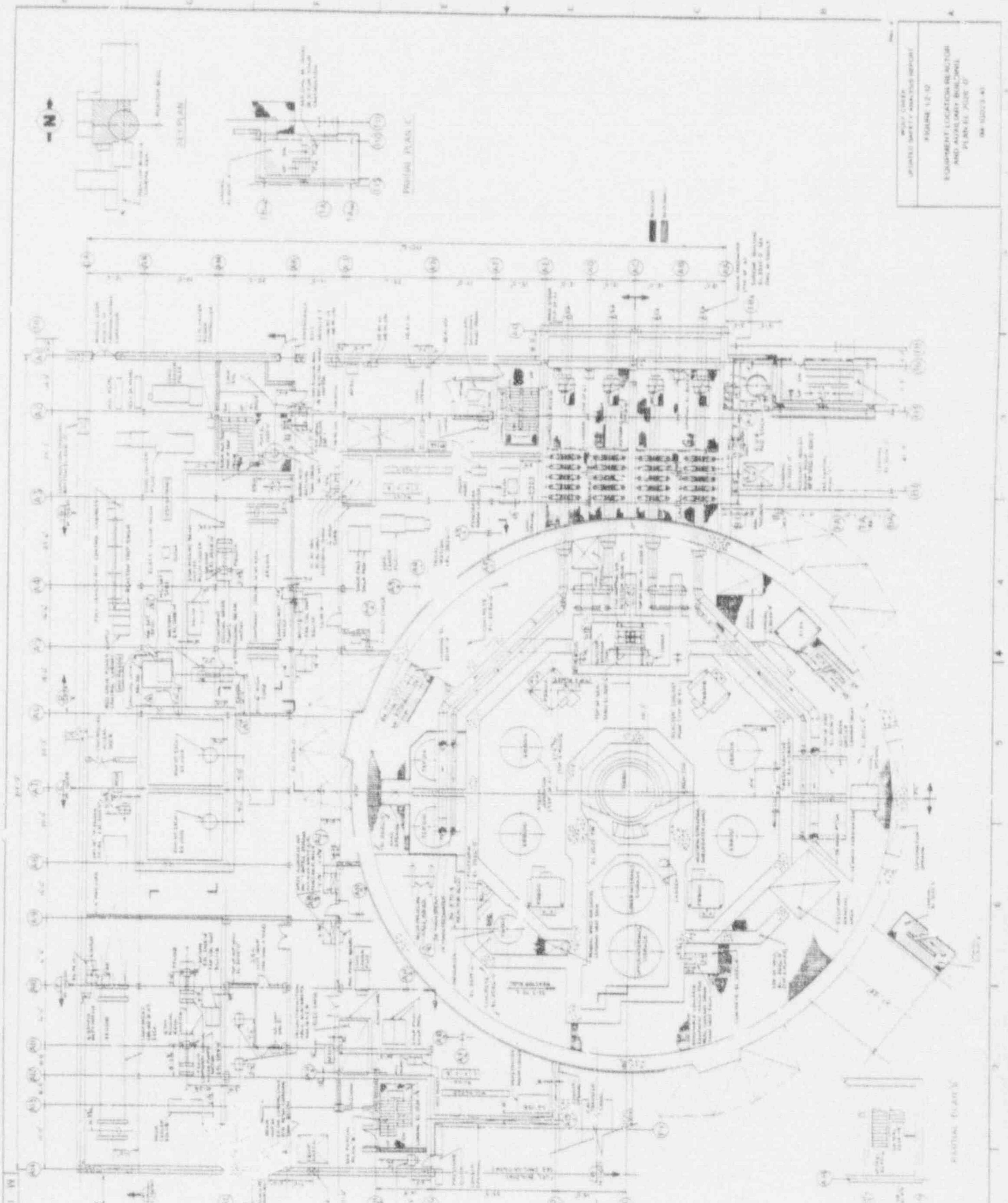
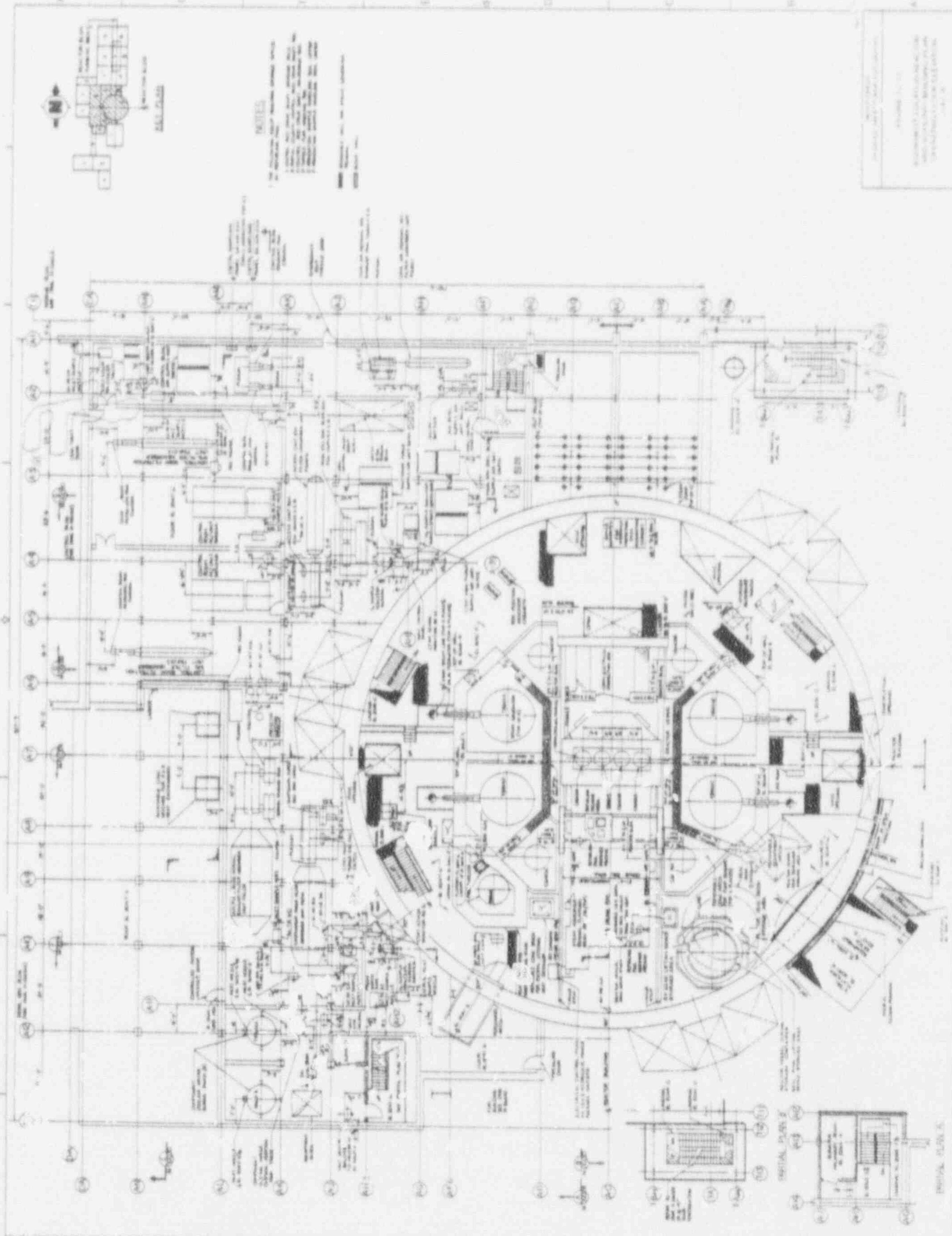


Figure 2.4-4



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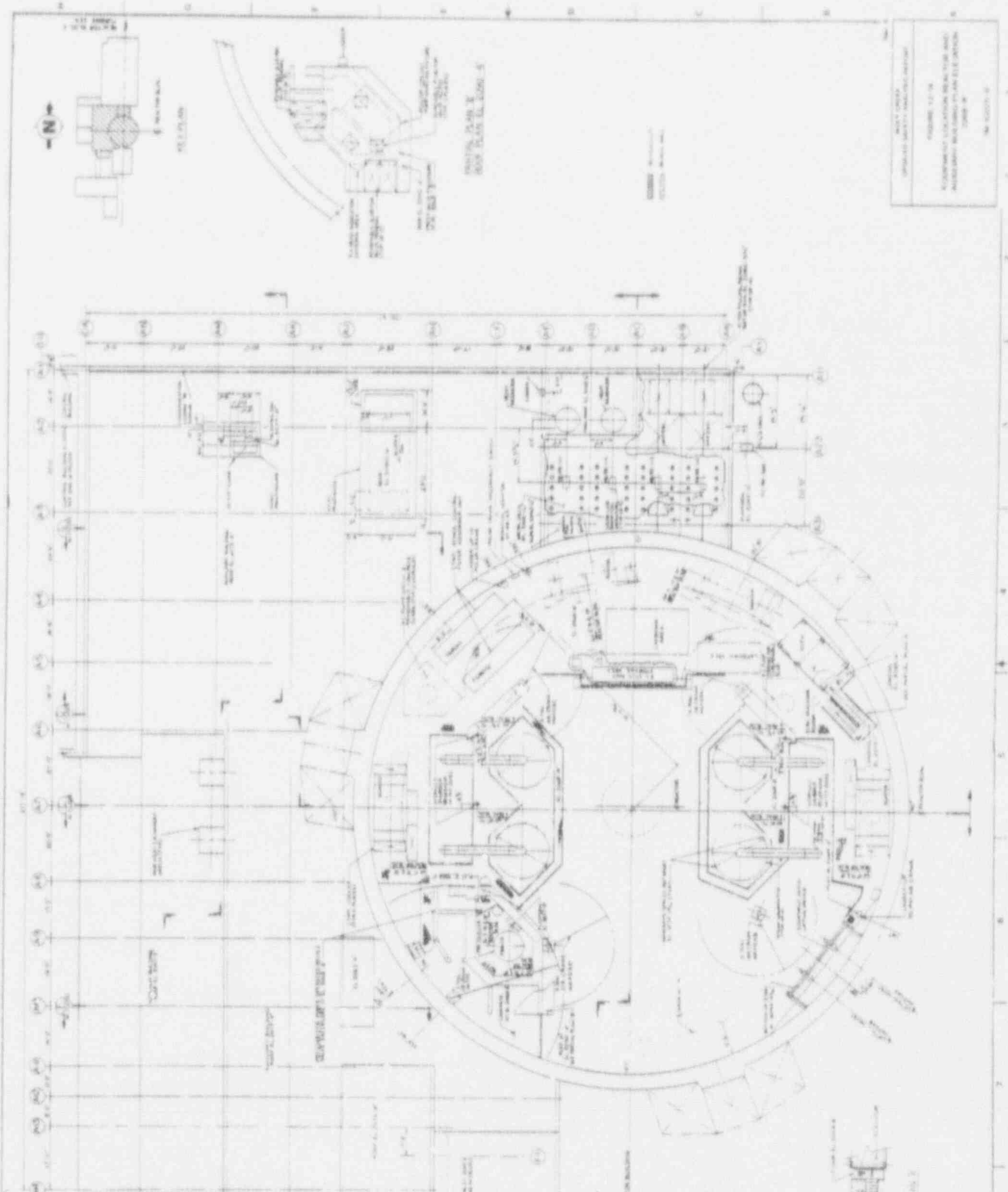
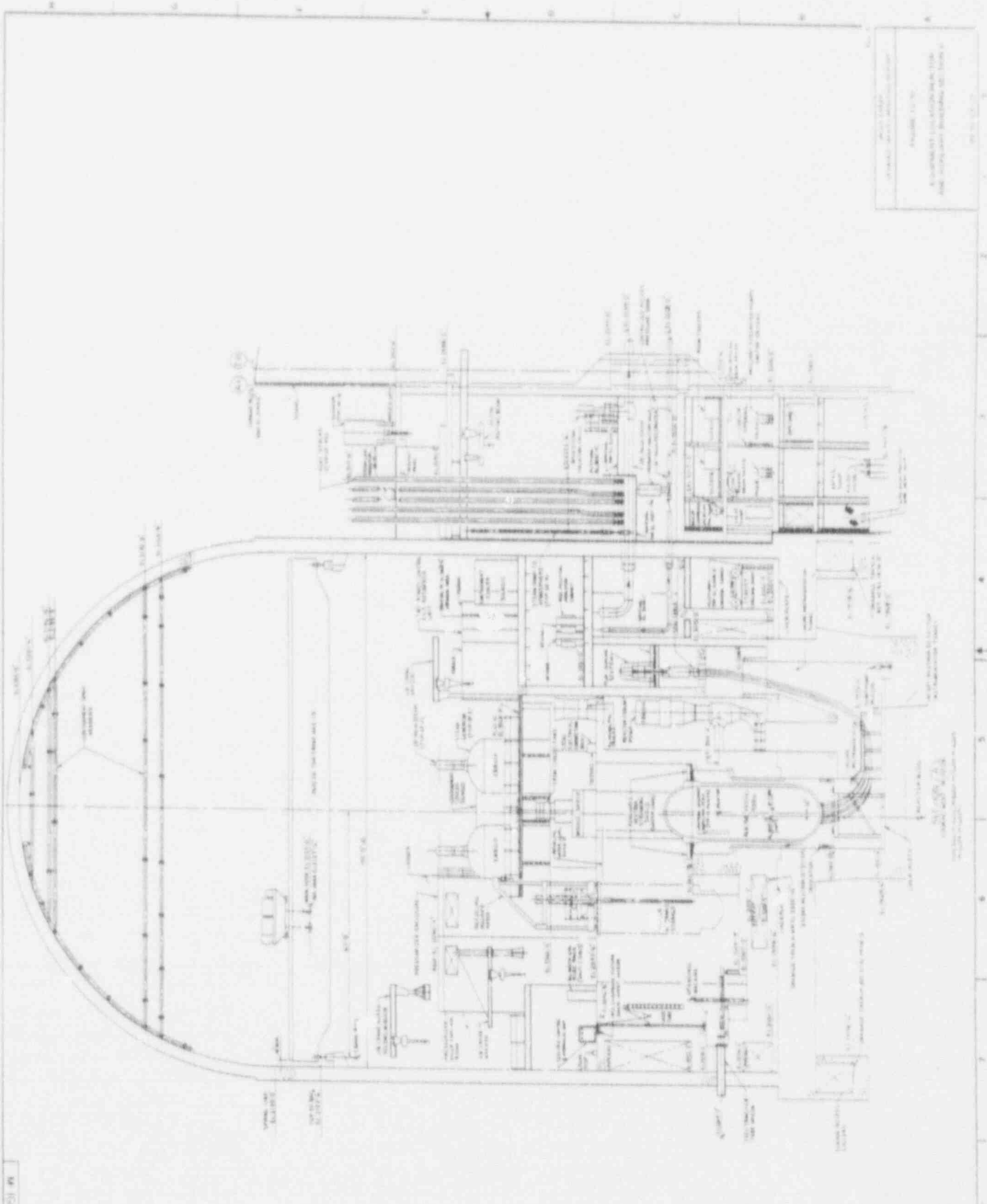


Figure 2.4-6



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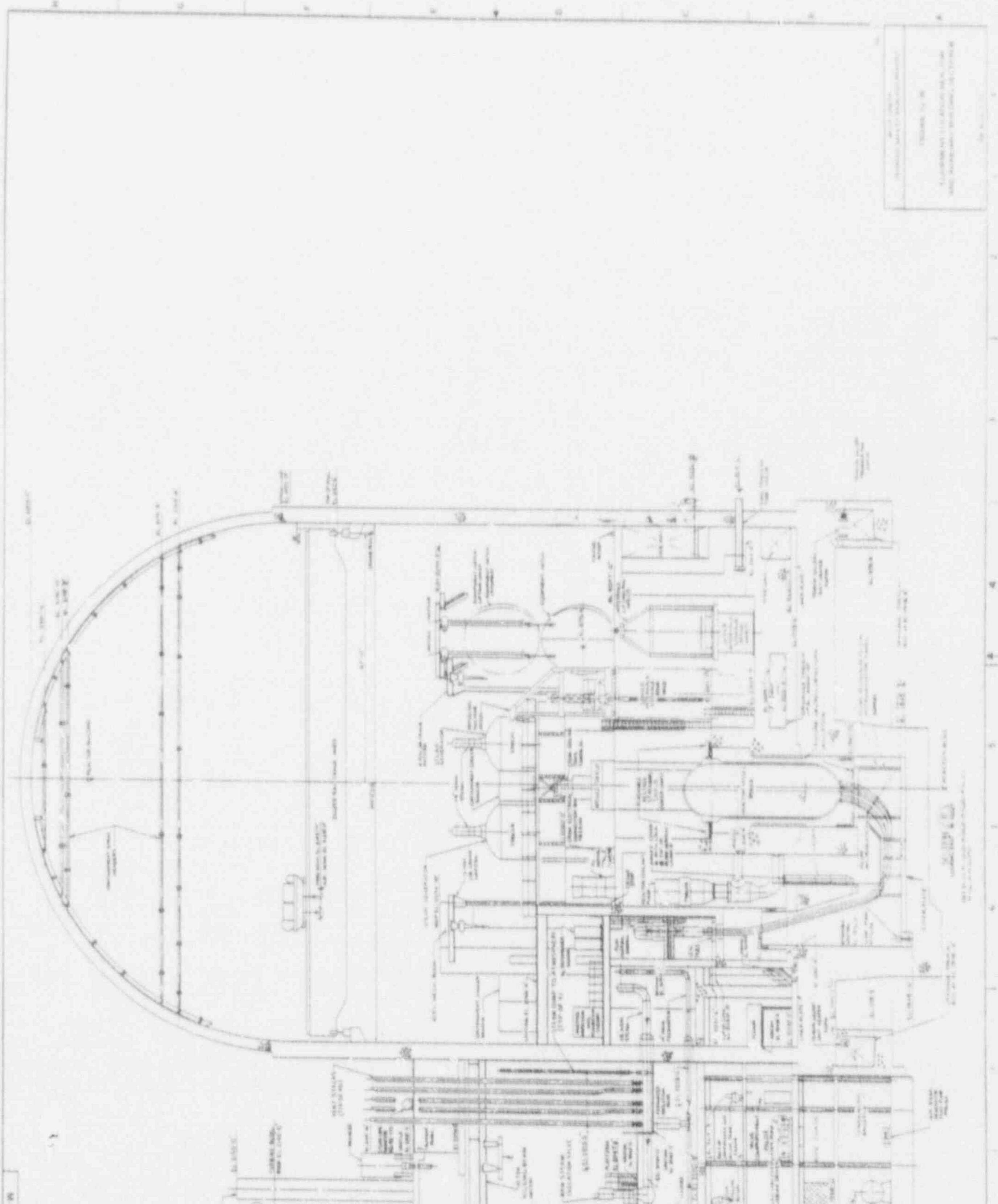


Figure 2.4-8

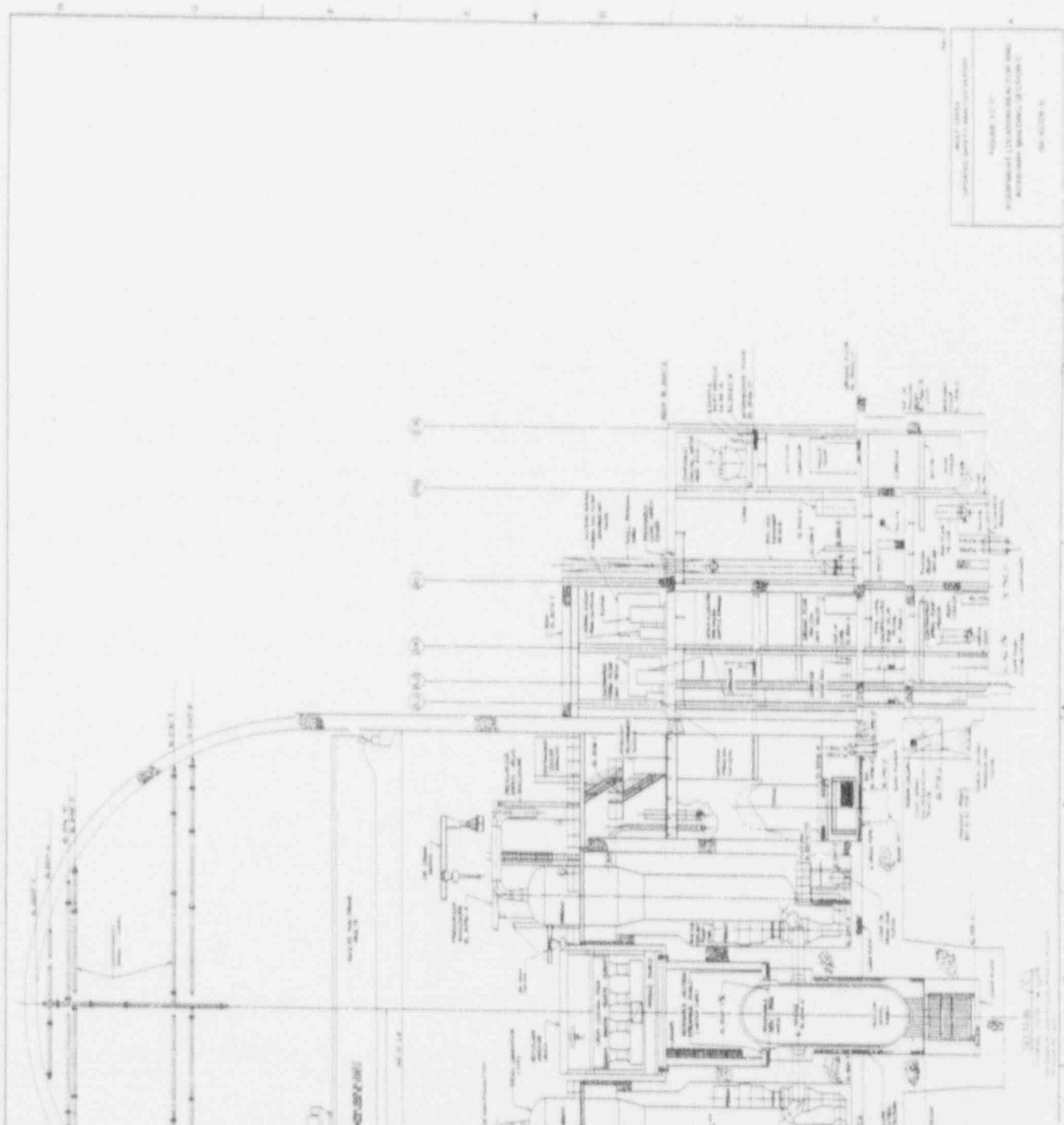
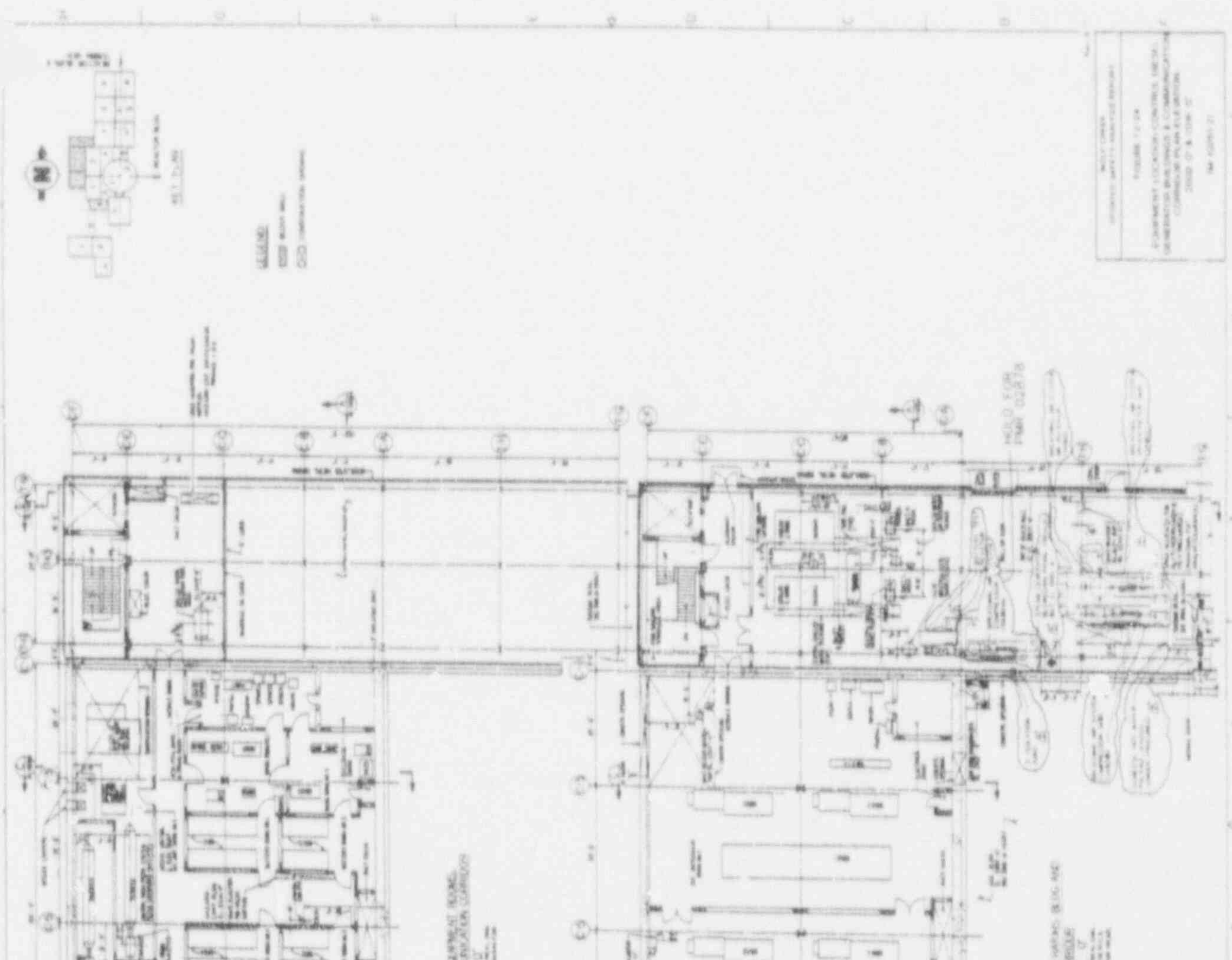


Figure 2.4-9



3.0 FRONT-END ANALYSIS

3.1 Accident Sequence Delineation

This section describes the three key elements in defining the accident sequences. These three elements are the initiating events, the support system modeling, and the event trees.

3.1.1 Initiating Events

A Core Damage Logic Diagram, presented in Figure 3.1-1, was developed to categorize all initiating events for event tree analysis. The rationale for the resulting categorization was based upon the nature and progression of the accident or transient sequences. All postulated events were grouped by the systemic response required to maintain short-term and long-term core cooling. Thus, events were identified that result in a loss of reactor coolant, those that result in insufficient heat removal, those that result in excessive core power, and transients with and without main heat removal systems and with and without vital support systems. This resulted in the identification of 15 internal initiating events for which event tree analysis was performed. The events and their mean annual frequency are:

IDENTIFIER	INITIATING EVENT	FREQUENCY
LLO	Large LOCA	5.00E-04
MLO	Medium LOCA	1.10E-03
SLO	Small LOCA	2.50E-03
SGR	Steam Generator Tube Rupture	1.10E-02
ISL	Nonisolable Interfacing Systems LOCA	6.11E-08
VEF	Reactor Vessel Failure	3.00E-07
TRA	Transient with Power Conversion System	4.30E-00
TRO	Transient without Power Conversion System	1.90E-01
SLB	Steamline/Feedline Break	5.00E-04
LSP	Loss of Offsite Power	5.10E-02
SBO	Station Blackout	2.32E-04
ATW	Anticipated Transient without Scram	3.33E-05
CCW	Loss of Component Cooling Water System	1.62E-04
SWS	Loss of All Service Water	1.76E-05
DCC	Loss of a Vital DC Bus	1.78E-03

An analysis of flooding from internal sources was also performed. This analysis is described in Section 3.3.7.

Frequency estimation for each initiating event was based upon three sources: published industry experience or estimates (LOCA, SGTR, ATWS, Interfacing Systems LOCA, LSP, Vessel Failure, Steamline/Feedline Ruptures); WCGS experience (Transients); and quantitative analysis of WCGS systems (Station Blackout, Losses of CCW, SWS, and vital DC Bus). Several industry frequencies were modified to reflect WCGS systems or experience (e.g., the frequencies of medium and small LOCAs were modified to include the frequency of PORV and Primary Safety Valve failures, and random RCP seal LOCAs; LSP frequency reflects WCGS estimates for severe weather effects).

3.1.2 Front-Line Event Trees

The event tree analysis of the above initiating events identifies those system and operator responses required to maintain primary system inventory and remove reactor decay heat in the period following accident or transient initiation and for the extended period of 24 hours following the event. The success criteria for front-line systems and operator actions are summarized in Table 3.1-1. These criteria are based on USAR analyses and on realistic calculations performed to reflect how the plant systems and operators would perform in the as-built, as-operated plant. As such, these analyses reflect the analysts' best estimates of actual plant response under each modeled accident or transient scenario.

The event trees for front-line initiating events are presented in the following subsections. Each subsection describes the initiating event, the assumptions and modeled dependencies in the analysis, and presents the model event tree diagram. The front-line events are those involving losses of primary and secondary coolant, transients, and Loss of Offsite Power. Special event trees have been developed for ATWS (which includes all transient and accident initiators followed by failure of reactor trip), for transients initiated by losses of station support systems (Component Cooling Water, Service Water, and an essential DC bus), for station blackout, and for Containment Safeguards (which includes fan cooling, spray, and isolation). These latter models are presented in Section 3.1.3. Table 3.1-2 provides a description of each of the nodes included in the front-line and special event trees.

Figure 3.1-1
CORE DAMAGE LOGIC DIAGRAM
PAGE 1 OF 4

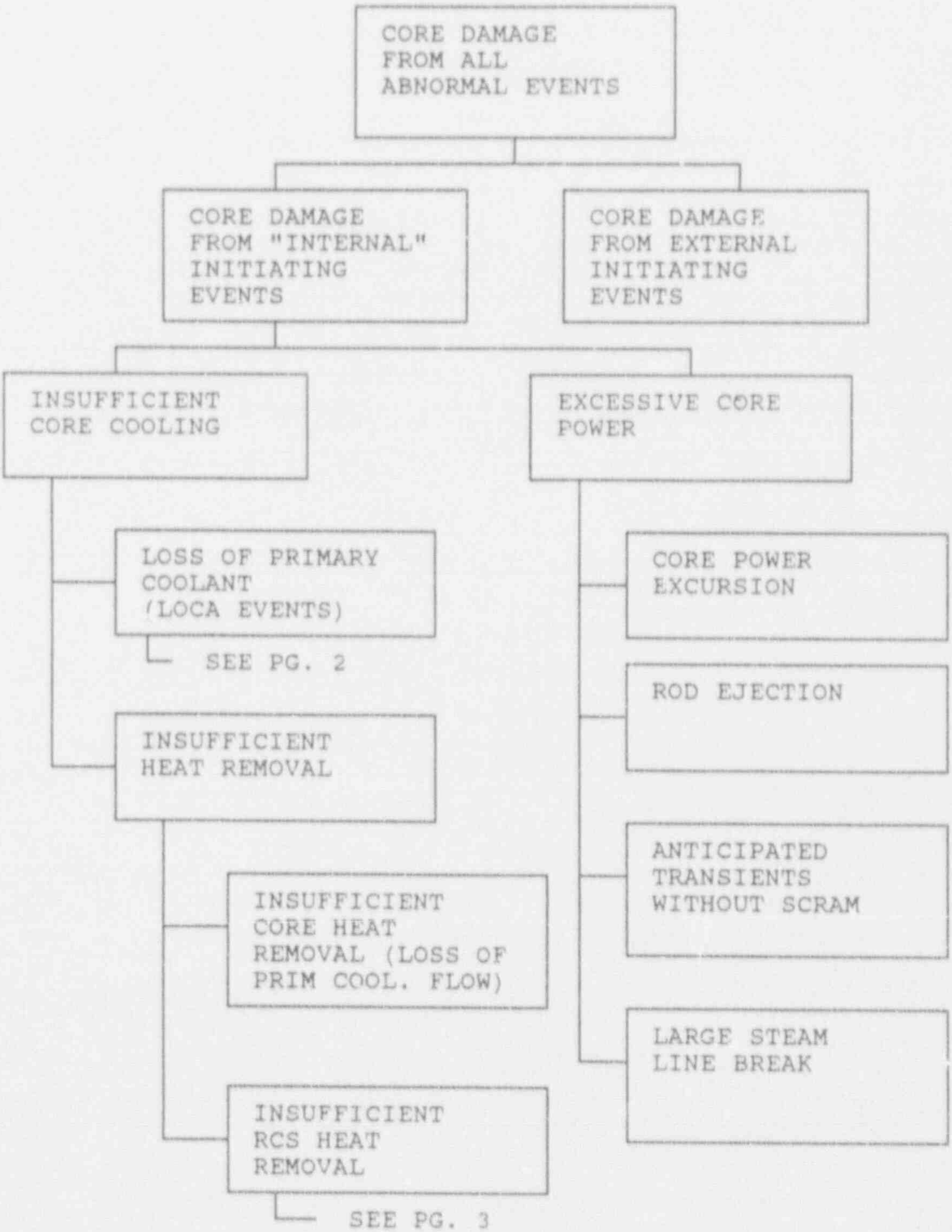


FIGURE 3.1-1
CORE DAMAGE LOGIC DIAGRAM
PAGE 2 OF 4

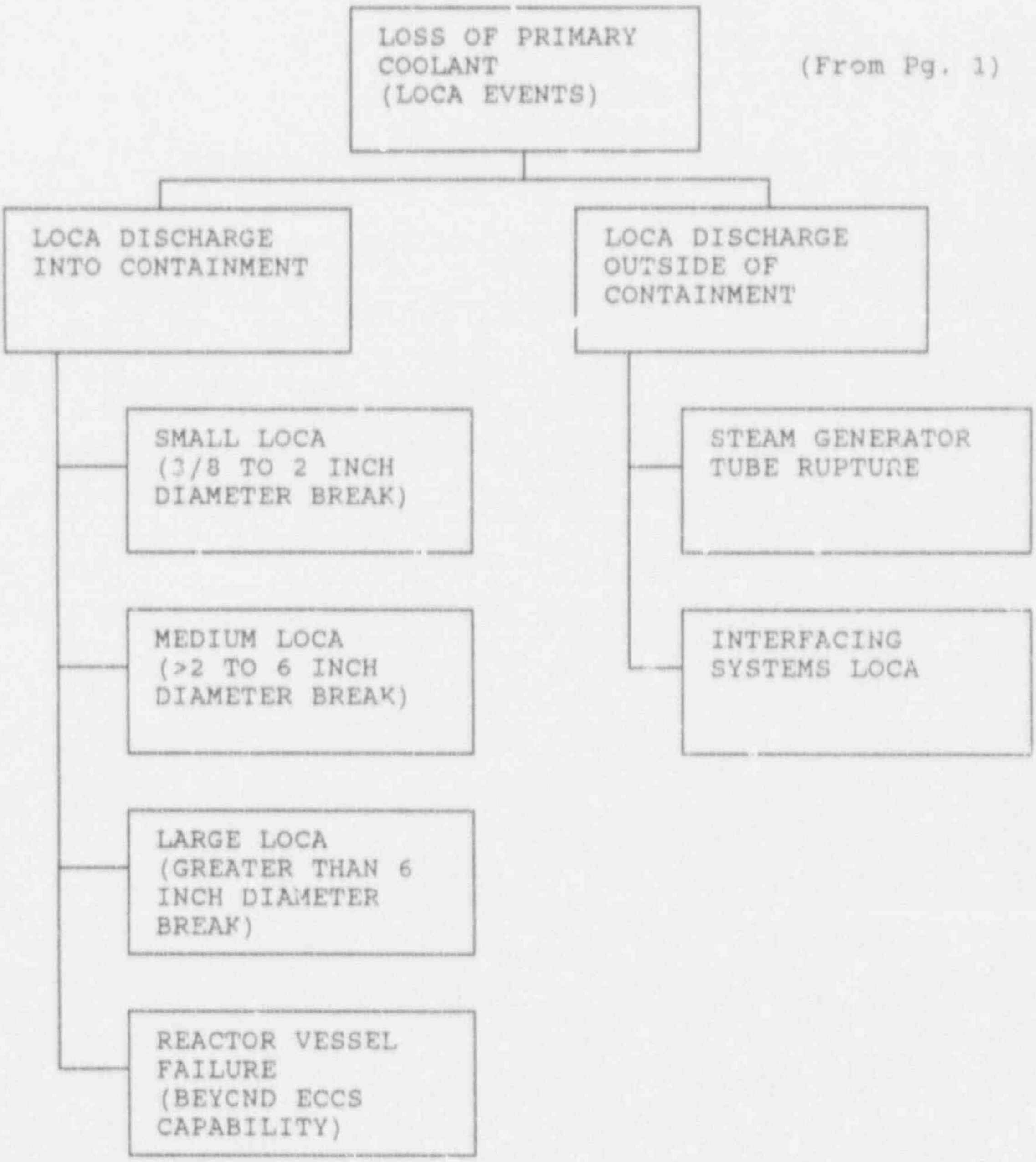


FIGURE 3.1-1
CORE DAMAGE LOGIC DIAGRAM
PAGE 3 OF 4

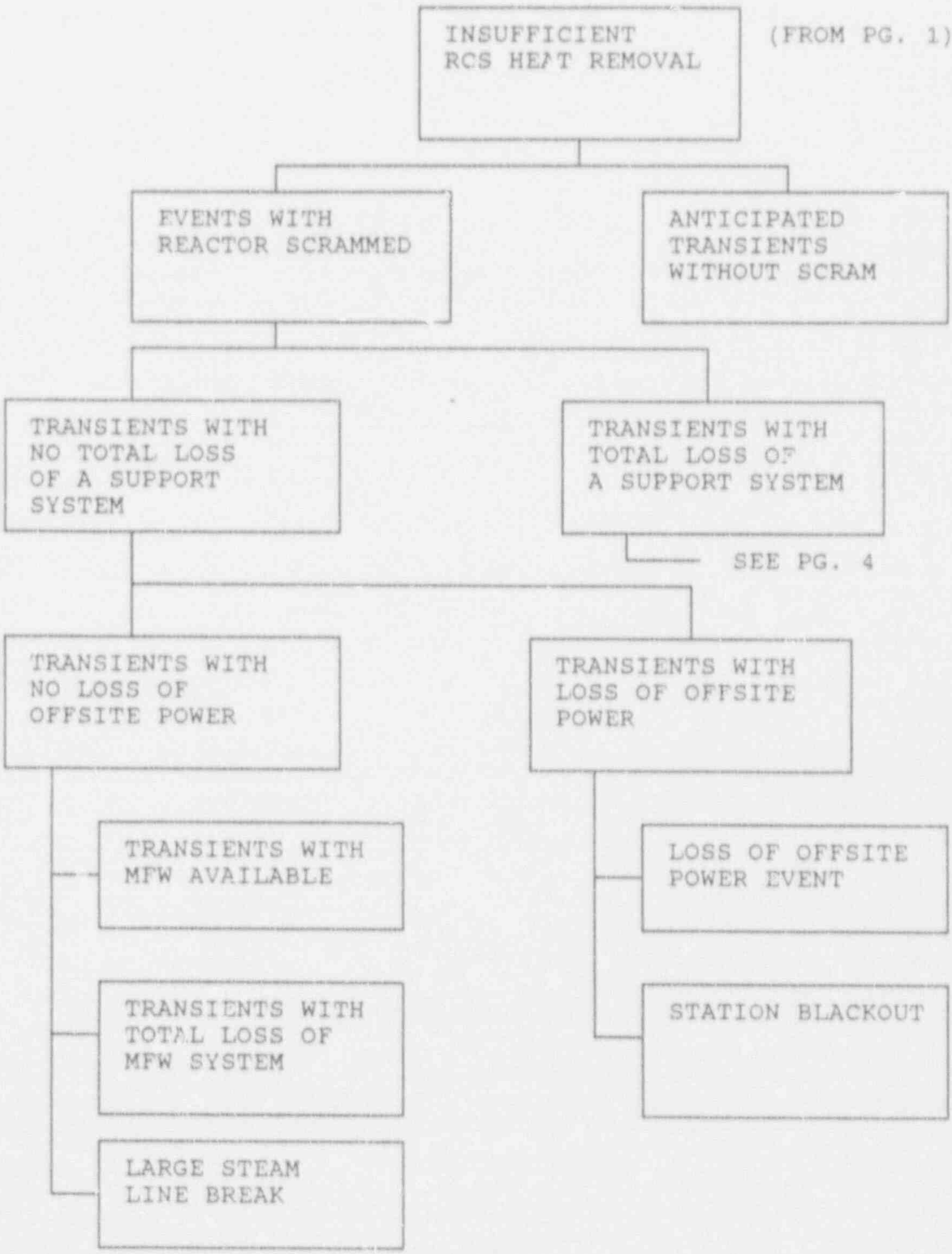


FIGURE 3.1-1
CORE DAMAGE LOGIC DIAGRAM
PAGE 4 OF 4

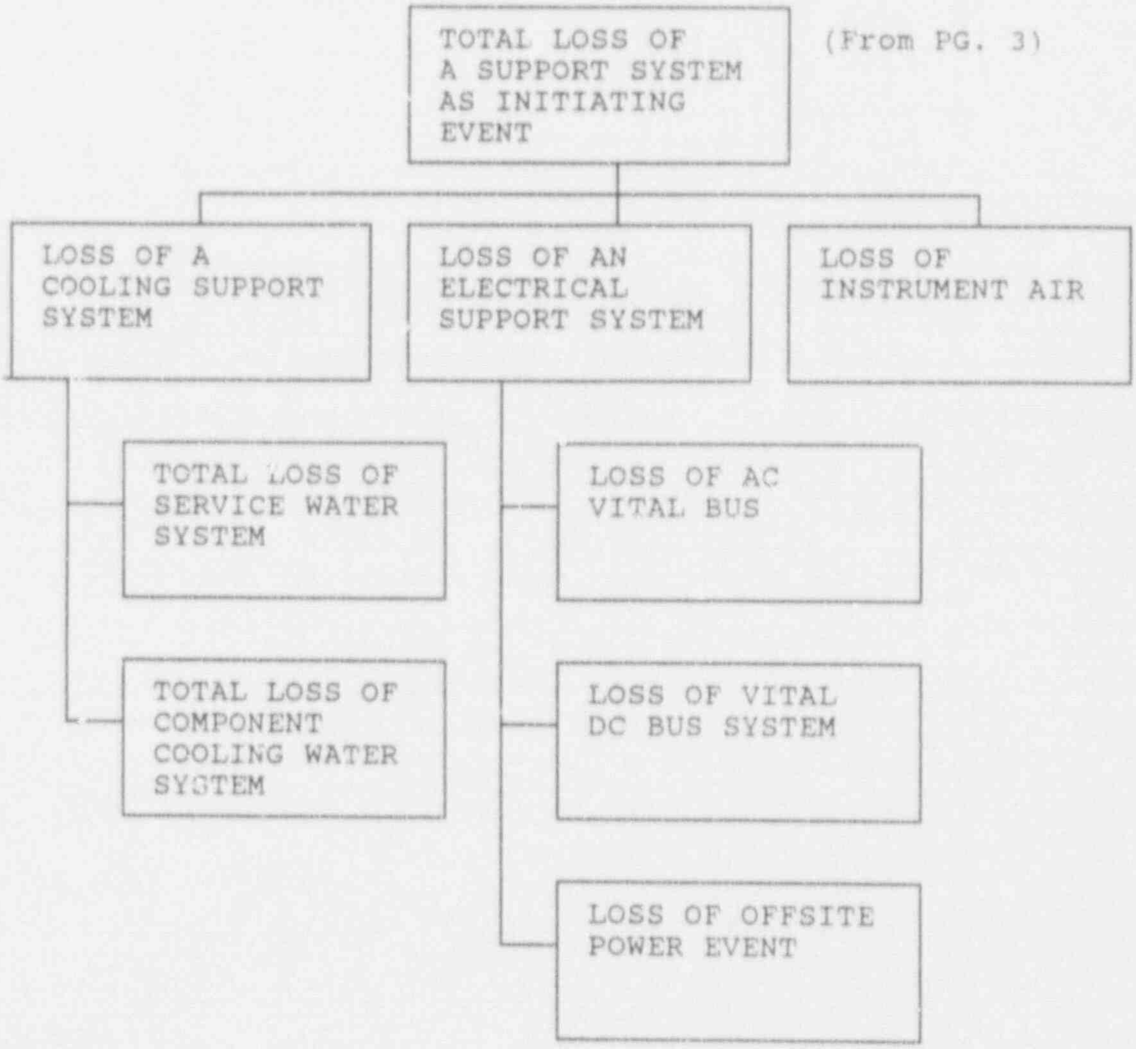


Table 3.1-1
Event Tree Success Criteria Summary for
Front-Line Systems and Operator Actions
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TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Accumulator Injection ACC	1) Large LOCA 2) Medium LOCA 3) Small LOCA	1) and 2) Successful discharge of 3 accumulators into 3 intact loops 3) Successful discharge of 3 accumulators into 3 of 4 loops	Confirm operation of system	None	Not applicable
Inverter Alternate Power Supply ACNN	1) SWS	Operator Action after failure of 120 Volt vital AC inverters due to overheating	Align two of the four 120 VAC instrument buses to alternate power sources	Per OFN-00-021, transformers XNN05 and XNN06 must be available	Variable
Auxiliary Feedwater AF1	1) Medium LOCA	Feed any 1 of 4 steam generators with any AFW pump combination	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours
Auxiliary Feedwater AF2	1) Small LOCA 2) CCW	Feed any 2 of 4 steam generators with any AFW pump combination	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours (30 minute delay acceptable)
Auxiliary Feedwater AF2R	1) TRA	Same as AF2 plus credit is taken for operator action (OFN 00-020)	Start a MDP (AFW pump)* after failure of associated DC bus. * (locally at the switchgear)	Same as AF2	Same as AF2
Auxiliary Feedwater AF2WO	1) LSP	Same as AF2 but no credit is taken for steam dump valves	Same as AF2	Same as AF2	Same as AF2
Auxiliary Feedwater AF2WOR	1) TRO	Same as AF2WO plus credit is taken for operator action (OFN 00-020)	Start a MDP (AFW pump)* after failure of associated DC bus. * (locally at the switchgear)	Same as AF2	Same as AF2

TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Auxiliary Feedwater to Intact SGs AF3	1) SGTR	Feed any 1 of 3 intact steam generators with any AFW pump combination	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours
Auxiliary Feedwater to Ruptured SG AF4	1) SGTR	Feed ruptured steam generator with any AFW pump combination	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours
Auxiliary Feedwater to Intact SGs AF5 AF5A	1) Secondary break	Feed any 2 of 3 intact steam generators with any AFW pump combination. If MSIV fails, assume only MD-AFW pumps are available (AF5A). (*)	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours
Auxiliary Feedwater AF6	1) Loss of SWS	Feed any 2 of 4 steam generators with turbine-driven AFW pump	Confirm operation of system	1) CST and flow path 2) Actuation signal 3) Main steam system and relief valves 4) Electric power	Run for 8 hours (30 minute delay acceptable)
Auxiliary Feedwater AF7	1) Loss of a Vital DC Bus	Feed any 2 of 4 steam generators with any AFW pump combination. (Same as AF2, except 1 MD AFW pump is unavailable due to loss of a DC bus.)	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves	Run for 24 hours (30 minute delay acceptable)

(*) TDAFW pump may not work at low SG pressures with multiple SGs blowing down.

TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Auxiliary Feedwater for ATWS AFA	1) ATWS	Three criteria that depend on other conditions: 1) AFA1 - same as AF2 2) AFA2 - approximately 1000 gpm total flow to 2 of 4 SGs with T-D or any 2 AFW pumps 3) AFA3 - approximately 2000 gpm total flow to 4 of 4 SGs (max flow from 3 AFW pumps)	Confirm operation of system	1) CST and flow path 2) Electric power 3) Actuation signal 4) Main steam system and relief valves (see Section 3.2.13.4 for steam relief requirements)	Run for 24 hours
Auxiliary Feedwater Continues AFC	1) Station blackout	Feedwater from the turbine-driven AFW pump continues for at least 8 hours (failure implies 4 hours)	Possible manual action to control flow and governor valves for turbine-driven AFW pump (see OFN 00-020)	1) DC power continues or operator action to maintain AFW flow if batteries depleted	Run for 8 hours
Auxiliary Feedwater TD AFW Pump AFT	1) Station blackout	Feed any 2 of 4 steam generators with turbine-driven AFW pump	Confirm operation of system	1) CST and flow path 2) Station batteries 3) Actuation signal 4) Main steam system and relief valves	Run for 4 hours (30 minute delay acceptable)
AMSAC AMS	1) ATWS	Signal to start AFW and trip the turbine (if power level is above 40%)	Confirm operation of system	1) Electric power	Not applicable
Restore CCW Cooling CCR	1) Loss of CCW	CCW cooling restored by approximately 8 hours	Actions to restore and align CCW to service	1) Service water 2) Electric power	Not applicable
Failure of CCW A Train CCWA	1) LSP	At least one of the two CCW A train pumps start and remove heat.	Operator action to start the pump	1) AC power 2) SWS 3) DC power	Within 30 minutes
Failure of CCW B Train CCWB	1) LSP	At least one of the two CCW B train pumps start and remove heat.	Operator action to start the pump	1) AC power 2) SWS 3) DC power	Within 30 minutes
Core Not Uncovered CNU	1) Station Blackout 2) Loss of CCW 3) Loss of SWS	Core not uncovered after power, CCW or SWS cooling restored	None	None	Not applicable

TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Cooldown and Depressurize RCS, SGs per EMGs C-31/32 EC3	1) SGTR	Depressurize RCS and ruptured SG to near atmospheric pressure prior to draining RWST (10-20 hour time frame)	Cooldown using intact SGs or possibly ruptured SG, depressurize RCS using spray or one Pzr PORV, reduce SI by stopping high pressure SI pumps, align RHR system for cooldown to cold shutdown	1) CST and flow path 2) RWST and flow path 3) Electric power 4) Steam relief valves and air supply (SG PORVs or steam dump to condenser) 5) Auxiliary feedwater 6) High pressure SI 7) RCS depress. equip. (spray w/RCP or Pzr PORV) 8) RHR System valves 9) CCW cooling	Not applicable
Cooldown and Depressurize RCS, SGs per EMG ES-11 ES1	1) Small LOCA	Cooldown and depressurize RCS to near atmospheric pressure to avoid depleting the RWST (consider for break sizes 0.7" diameter or smaller)	Cooldown RCS using intact SGs, depressurize RCS using spray or one Pzr PORV, reduce SI by stopping high pressure SI pumps, align RHR system for cooldown to cold shutdown	Same as EC3	Not applicable
High Pressure Recirculation HPR	1) Station blackout 2) Loss of CCW 3) Loss of SWS	1 out of 2 RHR pumps provide suction to the high pressure SI pumps: a) HPR1 - 1 of 4 high pressure SI pumps (same as LC2) b) HPR2 - 2 of 2 charging pumps, 1 out of 2 sump valves open	Manual valve changes in SI system, align CCW cooling to RHR Hx, confirm operation of system (see EMG ES-12)	1) Electric power 2) Service water 3) CCW cooling 4) Lo-lo RWST signal	Run for 24 hours
Low-head Recirculation LC1	1) Large LOCA 2) Medium LOCA 3) Small LOCA	1) and 2) 1 out of 2 RHR pumps recirculating flow from the sump to 1 of 3 intact cold legs, 1 of 2 sump valves open 3) 1 out of 2 RHR pumps delivering flow to 1 of 4 cold legs, 1 of 2 sump valves open	Verify auto realignment of SI system, align CCW to RHR Hx, confirm operation of system (see EMG ES-12)	1) Electric power 2) Service water 3) CCW cooling 4) Lo-lo RWST signal	Run for 24 hours

EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
High Pressure Recirculation LC2	1) Medium LOCA 2) Small LOCA 3) SGTR 4) Secondary break 5) Transients (TRA, TRO, LSP)	1 out of 2 RHR pumps recirculating flow from the sump to 1 of 4 high pressure SI pumps delivering flow to 3 of 4 cold legs, 1 out of 2 sump valves open (note: modify to 3 of 3 intact cold legs for medium LOCA)	Manual valve changes in SI system, align CCW coolic to RHR Hx, confirm operation of system (see EMG ES-12)	1) Electric power 2) Service water 3) CCW cooling 4) Lo-lo RWST signal	Run for 24 hours
High Pressure Recirculation LC3	1) Loss of a Vital DC Bus	1 of 1 RHR pump recirculating flow from the sump to 1 of 2 high pressure SI pumps delivering flow to 3 of 4 cold legs. 1 of 1 sump valves open (same as LC2 except one of all frontline safety system trains is not functional due to the event)	Same as LC2	Same as LC2	Run for 24 hours
Low Pressure Injection LPI	1) CCW	Same as SI1 plus no CCW dependency of RHR pumps	Same as SI1 plus operator initially stops RHR pumps and later starts them when needed	Same as SI1 but no CCW dependency	Same as SI1
Low Pressure Recirculation LPR	1) CCW	Same as LC1 plus no CCW dependency of RHR pumps or Heat Exchanger	Verify auto realignment of SI system, confirm operation of system	Same as LC1 but no CCW dependency	Same as LC1
Long Term Shutdown LTS	1) ATWS	Any one of three conditions: 1) Manual insertion of at least half of the control rods within several hours. 2) CVCS emergency boration to 2000 ppm in 6 hours using one charging pump and one BA transfer pump 3) RWST boration to 2000 ppm within 12 hours using one charging pump	Per EMG FR-SI: 1) Manually insert control rods 2) Align BAT and BA transfer pump, operate charging pump in emergency boration mode 3) Align charging pump to RWST, inject thru the BIT to the RCS cold legs	1) Electric power 2) Electric power, CVCS operation 3) Electric power, RWST and flow path	1) Run for 24 hours 2) Run for 6 hours 3) Run for 12 hours

Table 3.1-1

EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Main Feedwater MF1	1) Transient with power conversion system (TRA) 2) Loss of CCW	Feed any 2 of 4 steam generators using a condensate pump and either of the following: 1) 1 of 2 MFW pumps, 2) the startup FW pump, or 3) SG depressurization, condensate pump only	"Jumper" feedwater isolation signal within 30 minutes, start condensate pump, start MFW or startup pump or depressurize SGs to 560 psig (see EMG FR-H1)	1) Electric power 2) Condensate system 3) Air supply for feedwater control valves 4) Main steam system and relief valves	Run for 24 hours (30-60 minute delay acceptable)
Conditional Medium LOCA Coremelt MLOCCW	1) CCW	Conditional probability of coremelt given that a medium LOCA size RCP seal LOCA occurs	NA	NA	24 hours
Manual Trip of RDMGs MRT	1) ATWS	1 of 2 RDMG breakers opened within 2 minutes	Manually trip RDMGS per EMG FR-S1	1) Electrical power	Not applicable
Main Steam Isolation MS1	1) SGTR	Isolation of ruptured SG closure of MSIV on any SG (within about 15 minutes for a design basis SGTR)	Diagnoze ruptured SG, close MSIV on ruptured SG or close MSIVs on any intact SG used for initial cooldown (see EMG E-3)	1) Electric power 2) Hydraulic actuation	Not applicable
Main Steam Isolation MS2	1) Secondary break	Isolation of faulted SG - closure of any 3 of 4 MSIVs	Verify steamline isolation	1) Electric power 2) Hydraulic actuation and signal	Not applicable
Stabilize RCS and Ruptured SG Pressure Before SG Overfill OD1	1) SGTR	Stabilize RCS pressure with ruptured SG pressure before overfill of the ruptured SG (within about 30 minutes for a design basis SGTR)	Initial rapid cooldown using at least one intact SG, depressurize RCS using spray or one Pzr PORV, terminate high pressure SI flow (see EMG E-3)	1) Electric power 2) CST and flow path 3) Steam relief valves and air supply (SG PORVs or steam dump to condenser) 4) Auxiliary feedwater 5) RCS depress. equip. (spray w/RCP or Pzr PORV)	Not applicable
Stabilize RCS and Ruptured SG Pressure After SG Overfill OD2	1) SGTR	Stabilize RCS pressure with ruptured SG pressure after overfill of the ruptured SG (assume approximately 60 minutes)	Initiate rapid cooldown using at least one intact SG, depressurize RCS using spray or one Pzr PORV, terminate high pressure SI flow (see EMG E-3)	Same as OD1	Not applicable

Table 3.1-1
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EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Offsite AC Recovery Fraction O/E	1) SBO	Fraction of AC recovery attributed to offsite power recovery (rather than EDG recovery)	NA	NA	NA
Bleed and Feed OF2	1) Loss of a Vital DC Bus	1 of 1 Pzr PORV opens, 1 of 1 charging pump and 1 of 1 HHSI pump injects to 4 of 4 cold legs. (Bleed and feed initiated prior to secondary dryout - assume at 20 minutes.)	Same as OFC	Same as OFC	Run for 24 hours
Bleed and Feed OFB	1) Small LOCA 2) SGTR 3) Secondary break	2 of 2 Pzr PORVs open (bleed and feed initiated prior to secondary dryout - assume at 30 minutes)	Manually open PORVs and block valves, verify SI pumps running (see EMG FR-H1)	1) Electric power	Run for 24 hours
Bleed and Feed OFC	1) Transients, (TRA, TRO, LSP)	2 of 2 Pzr PORVs open, 1 of 4 high pressure SI pumps inject to 3 of 4 cold legs, (bleed and feed initiated prior to secondary dryout - assume at 30 minutes)	Manually open PORVs and block valves, start SI pumps (see EMG FR-H1)	1) Electric power 2) CCW cooling for SI pumps 3) RWST and flow path	Run for 24 hours
Cooldown and Depressurize the RCS OP1	1) Medium LOCA	Operator initiated cooldown started within 15 minutes using at least 1 SG supplied with feedwater	Perform cooldown using steam dump to condenser or SG PORVs (per EMGs ES-11, FR-C2, or possibly FR-C1)	1) CST and flow path 2) Steam relief valves and air supply (SG PORVs or steam dump to condenser) 3) Auxiliary feedwater	Approximately 1 hour (until break flow and low-head SI flow are able to remove decay heat)

EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Cooldown and Depressurize the RCS OP2	1) Small LOCA	Operator initiated cooldown started within 30 minutes using at least 2 SGs supplied with feedwater	Perform cooldown using steam dump to condenser or SG PORVs (per EMGs ES-11, FR-C2, or possibly FR-C1)	1) CST and flow path 2) Steam relief valves and air valves and air supply (SG PORVs or steam dump to condenser) 3) Auxiliary feedwater	Potentially 24 hours (steam relief needed for decay heat removal)
Terminate High Pressure Safety Injection OST	1) Secondary break	High pressure SI terminated before pressurizer fills - within 10 minutes for large feedline break, 15 minutes for large steamline break, 30 minutes for small secondary break	Secure SI pumps (per EMG E-1 and ES-03)	1) Electric power	Not applicable
Reactor Power Level PLV	1) ATWS	Initial reactor power less than 40% of full power	None	None	Not applicable
Primary Pressure Relief PPR	1) ATWS	3 out of 3 pressurizer safety valves plus X out of 2 pressurizer PORVs open for pressure relief (see PPR writeup in Section 3.2.13.4 for PORV requirements)	None	1) Electric power	Not applicable
RCS Cooldown RCD	1) Station blackout 2) Loss of CCW 3) Loss of SWS	Cooldown using at least 2 of 4 SGs supplied with feedwater (assume cooldown initiated within 60 minutes)	1) Perform cooldown using SG PORVs (per EMG C-0 for station blackout) 2) Perform cooldown using steam dump to condenser or SG PORVs (per normal cooldown or small LOCA procedures)	For both events: 1) CST and flow path 2) Steam relief valves and air supply (SG PORVs or steam dump to condenser) 3) Auxiliary feedwater or main feedwater	Potentially 24 hours (steam relief needed for decay heat removal)

TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
Align RCP Seal Cooling RCPSEAL	1) CCW 2) LSP	Operator Action (after failure of CCW train A)	Align CCW service loop to the operating CCW train B to restore flow to the TBCCs and start a charging pump being cooled by CCW train B and ensure it is aligned to provide RCP seal injection flow	Per OFN 00-004: 1) CCW Train B 2) Charging Pump	Within 30 minutes
Trip RCPs RCPTRIP	1) CCW	Operator Action (after loss of RCP seal cooling)	Stop operating RCP pumps	Per OFN 00-005	Within 10 minutes
Restore RCS Inventory RRI	1) Station blackout 2) Loss of CCW 3) Loss of SWS	Restore safeguards system and initiate high pressure SI flow to RCS cold legs: a) RRI1 - 1 out of 4 high pressure SI pumps (same as SI3) b) RRI2 - 2 of 2 charging pumps (applies if there is no feedwater)	Restore safeguards pumps and cooling, start SI pumps, manually open Pzr PORVs and block valves as required (same EMGs noted above for RCD top event)	1) Electric power 2) CCW cooling for SI pumps 3) RWST and flow path	Run for 24 hours
Manual Reactor Trip or Rod Insertation RT	1) ATW	See operator actions	Operator either manually scrams the rector or performs MRI	Power to controls and instrumentation is available	Within minutes of the reactor trip demand
DC Switchboard Room Cooling SBCOOL	1) SWS	Operator Action	Operator action to locally provide cooling to DC switchboard rooms	Control Room Standing Order #26	Variable
Low Pressure Injection SI1	1) Large LOCA 2) Medium LOCA 3) Small LOCA	1) and 2) 1 out of 2 RHR pumps delivering flow to at least 1 out of 3 intact cold legs 3) 1 out of 2 RHR pumps delivering flow to at least 1 out of 4 cold legs	Confirm operation of system	1) Electric power 2) Actuation signal 3) RWST and flow path 4) CCW cooling for RHR	Run for 24 hours (required for LC1)
High Pressure Safety Injection SI2	1) Medium LOCA	2 out of 4 high pressure SI pumps delivering flow to 3 out of 3 intact cold legs	Confirm operation of system	1) Electric power 2) Actuation signal 3) RWST and flow path 4) CCW cooling for SI pumps	Run initially for up to 4 hours in injection phase; including LC2, 1 high pressure SI pump required for 24 hours

TOP EVENT	EVENT TREES	SYSTEM SUCCESS CRITERIA	NECESSARY OPERATOR ACTIONS	SYSTEM DEPENDENCIES	EXPECTED RUN TIME
High Pressure Safety Injection SI3	1) Small LOCA 2) SGTR 3) Secondary break	1 out of 4 high pressure SI pumps delivering flow to 3 out of 4 cold legs.	Confirm operation of system	1) Electric power 2) Actuation signal 3) RWST and flow path 4) CCW cooling for SI pumps	Run for 24 hours (SI pump required for LC2)
Conditional Small LOCA Coremelt SLOCCW	1) CCW	Conditional probability of coremelt given that a small LOCA size RCP seal LOCA occurs	NA	NA	24 hours
Conditional Small LOCA Coremelt SLOLSP	1) LSP	Conditional probability of coremelt given that a small LOCA size RCP seal LOCA occurs following LSP	NA	NA	24 hours
Secondary Integrity SSV	1) SGTR	All secondary relief valves for the ruptured SG remain closed	None	None	Not applicable
SW restored within 8 hours SWR	1) Loss of SWS	SWS cooling restored by approximately 8 hours	Actions to restore and align SWS to service	1) Electric power	Not applicable
Restore Power by X Hours XHR	1) Station blackout	Power restored to at least one 4.16 kV by X hours (4 or 8 hours, as defined in Section 3.2.12.4)	Actions to restore diesel generator or alternate supply of AC power	None	Not applicable
Restore Power by Y hours YHR	1) Station blackout	Power restored to at least one 4.16 kV bus by Y hours (6, 8, 10 or 12 hours, as defined in Section 3.2.12.4)	Actions to restore diesel generator or alternate supply of AC power	None	Not applicable

TABLE 3.1-2
SUMMARY OF EVENT TREE NODES
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Node	Description
ACC	Accumulator Injection
ACNN	Inverter Alternate Power Supply
AF1	Auxiliary Feedwater - MLO
AF2	Auxiliary Feedwater - SLO
AF2R	Auxiliary Feedwater - TRA
AF2WO	Auxiliary Feedwater - LSP
AF2WOR	Auxiliary Feedwater - TRO
AF3	Auxiliary Feedwater - SGTR
AF4	Auxiliary Feedwater - SGTR
AF5	Auxiliary Feedwater - SLB
AF5A	Auxiliary Feedwater - SLB
AF6	Auxiliary Feedwater - SWS
AF7	Auxiliary Feedwater - DCC
AFA	Auxiliary Feedwater - ATWS
AFC	Auxiliary Feedwater - SBO
AFT	Auxiliary Feedwater - SBO
AMS	AMSAC
CCR	Restore CCW Cooling
CCWA	Failure of CCW A Train
CCWB	Failure of CCW B Train
CNU	Core Not Uncovered
EC3	Cooldown and Depressurize RCS, SGs per EMGs C-31/32
ES1	Cooldown and Depressurize RCS, SGs per EMG ES-11
HPR	High Pressure Recirculation
LC1	Low Pressure Recirculation
LC2	High Pressure Recirculation
LC3	High Pressure Recirculation
LPI	Low Pressure Injection
LPR	Low Pressure Recirculation
LTS	Long Term Shutdown
MF1	Main Feedwater
MLOCCW	Conditional Medium LOCA Coremelt
MRT	Manual Trip of RDMGs
MS1	Main Steam Isolation
MS2	Main Steam Isolation
OD1	Stabilize RCS and Ruptured SG Pressure Before SG Overfill
OD2	Stabilize RCS and Ruptured SG Pressure After SG Overfill
O/E	Offsite AC Recovery Fraction
OF2	Bleed and Feed
OFB	Bleed and Feed
OFC	Bleed and Feed
OP1	Cooldown and Depressurize the RCS
OP2	Cooldown and Depressurize the RCS
OST	Terminate High Pressure Safety Injection
PLV	Reactor Power Level
PPR	Primary Pressure Relief
RCD	RCS Cooldown
RCPSEAL	Align RCP Seal Cooling

TABLE 3.1-2
SUMMARY OF EVENT TREE NODES
Page 2 of 2

RCPTRIP	Trip RCPs
RRI	Restore RCS Inventory
RT	Manual Reactor Trip or Rod Insertion
SBCOOL	DC Switchboard Room Cooling
SI1	Low Pressure Injection
SI2	High Pressure Safety Injection
SI3	High Pressure Safety Injection
SLOCCW	Conditional Small LOCA Coremelt
SLOLSP	Conditional Small LOCA Coremelt
SSV	Secondary Integrity
SWR	Restore Service Water Cooling
XHR	Restore AC Power within X Hours
YHR	Restore AC Power within Y Hours

3.1.2.1 Large LOCA

The large LOCA event tree is presented in Figure 3.1-2. This event tree models plant behavior following a postulated rupture of the RCS ranging in size from a 6" equivalent diameter break up to the equivalent of a double-ended circumferential break of the largest pipe (29") in the RCS. Engineered Safety Features Actuation System (ESFAS) actuation of high and low pressure safety injection is generated on low pressurizer pressure or high containment pressure. Secondary side heat removal is not necessary inasmuch as all decay heat is removed by steam/water exiting the break into the containment volume. Reactor trip is also not required due to core voiding sufficient to stop the nuclear reaction. Long-term shutdown is maintained by the boron concentration of the injected water from the accumulators and Refueling Water Storage Tank (RWST).

The Residual Heat Removal pumps draw on the RWST and inject borated water to the RCS cold legs. Upon a low-low-1 level signal in the RWST, pump suction is automatically switched to the containment sump. The operators verify CCW flow to the RHR heat exchangers as part of the transfer to cold leg recirculation.

This analysis reflects that no credit has been taken for high pressure recirculation using the charging and safety injection pumps drawing water from the discharge of the RHR pumps. Further, no credit for core damage prevention has been taken for the operation of the containment fan coolers and containment spray, which are available to protect the containment from overpressurization and have been modeled in the containment analysis.

Dependencies displayed in the event tree include: (1) if accumulator injection fails, early core damage is assumed to result, and short-term and long-term cooling are not addressed; (2) if low pressure safety injection fails, early core damage is assumed and long-term cooling is not addressed; and (3) if low pressure safety injection succeeds and low pressure recirculation fails, late core damage is assumed.

3.1.2.2 Medium LOCA

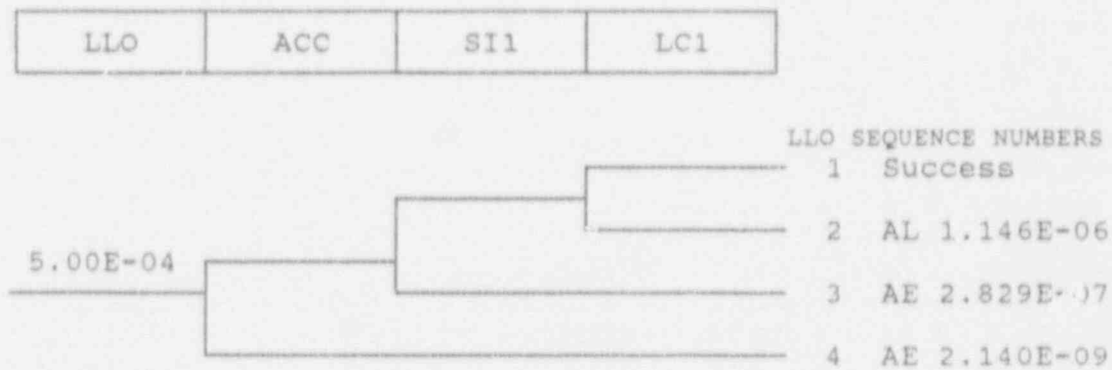
The Medium LOCA event tree is presented in Figure 3.1-3. Medium LOCAs are postulated ruptures of the RCS between about 2" to 6" in equivalent diameter. A failed open Safety Valve has been included in the initiating event frequency estimate. System response varies depending upon the actual break size, with RCS depressurization and the availability of accumulator and low pressure safety injection (which is dependent upon the stabilization pressure of the RCS once safety injection and auxiliary feedwater flows are initiated). If RCS pressure is high, operator action to establish long-term high pressure recirculation will be required. If high pressure injection fails, then the operators have emergency procedures guiding them to depressurize the RCS with the auxiliary feedwater system and establish low-pressure injection and recirculation.

It has been assumed that: (1) accumulator injection is necessary to prevent prompt core damage, despite the fact that RCS pressure will not decrease enough for breaks between about 2"-3" equivalent diameter; and (2) reactor trip is unimportant for plant shutdown due to core voiding and borated water injection by the Emergency Core Cooling Systems (ECCS) for the breaks between 3 to 6 inches.

Dependencies displayed in the event tree include: (1) if accumulator injection fails, then the injection phase is assumed to fail resulting in early core damage; (2) if accumulator injection and high pressure safety injection succeed, then RCS cooldown and depressurization and low pressure safety injection are not addressed; (3) if high pressure safety injection fails, then the availability of auxiliary feedwater and the ability of the operators to cool down and depressurize the RCS and utilize low pressure injection and recirculation is addressed; and (4) if the injection phase fails, recirculation is not addressed because of insufficient water in the containment sump.

Figure 3.1-2

Large LOCA Event Tree



LLO - Large LOCA

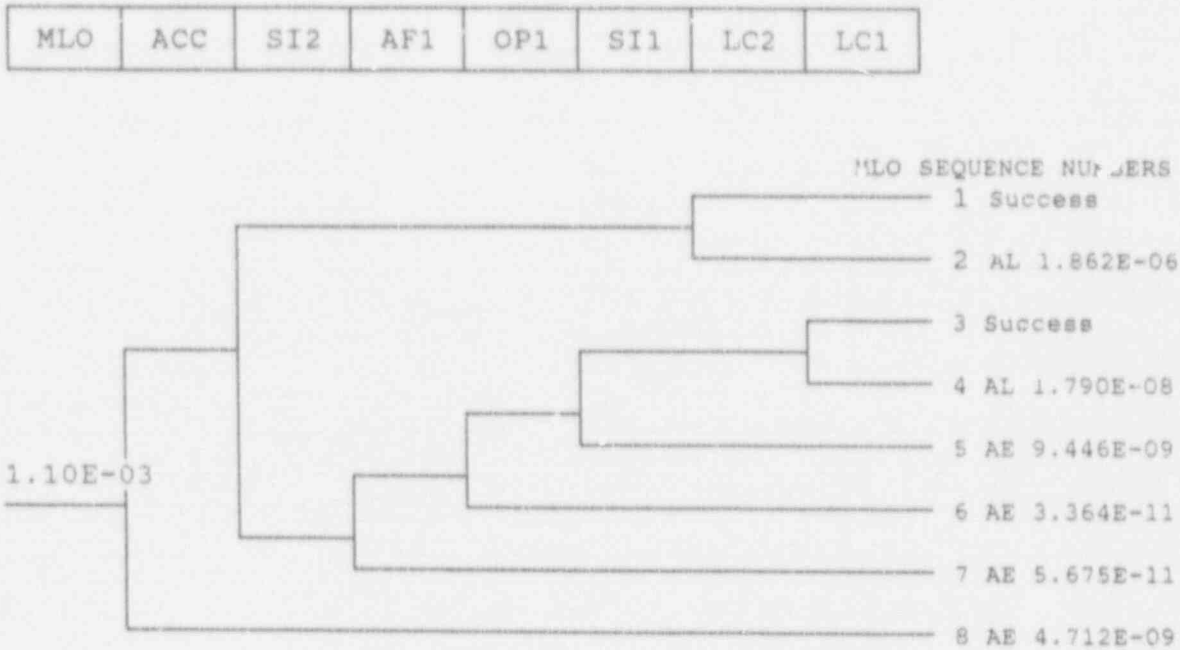
ACC - Accumulator Injection

SI1 - Low Pressure Injection

LC1 - Low Pressure Recirculation

Figure 3.1-3

Medium LOCA Event Tree



MLO - Medium LOCA

ACC - Accumulator Injection

SI2 - High Pressure Safety Injection

AF1 - Auxiliary Feedwater

OP1 - Cooldown and Depressurize the RCS

SI1 - Low Pressure Safety Injection

LC2 - High Pressure Recirculation

LC1 - Low Pressure Recirculation

3.1.2.3 Small LOCA

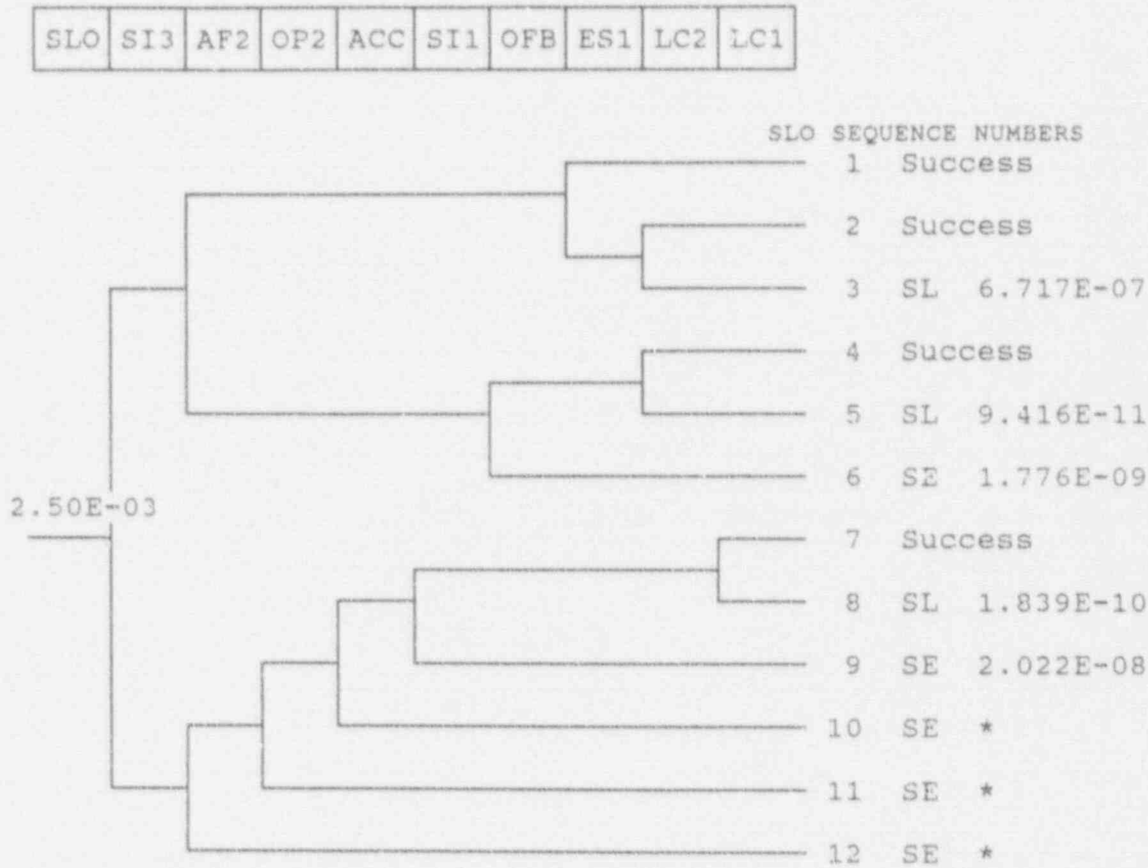
The Small LOCA event tree is presented in Figure 3.1-4. Small LOCAs are ruptures of the RCS in which the break flow exceeds the capacity of the normal charging system but is insufficient to depressurize the system or to allow core decay heat to be removed via ejection of coolant water to the containment. Generally, breaks from 3/8" to 2" equivalent diameter are included. A failed-open, nonisolated PORV and random failure of a Reactor Coolant Pump (RCP) seal are included in the initiating event frequency estimate.

RCS depressurization following a Small LOCA causes reactor trip, ESFAS actuation of the ECCS, and secondary system isolation. Failure of reactor trip is treated in the ATWS model, Section 3.1.3.1 below. High pressure safety injection maintains primary system inventory, while auxiliary feedwater provides for decay heat removal. If secondary cooling is not established, operator action to establish bleed and feed cooling using the pressurizer PORVs is required. In the long-term, high pressure recirculation can be established, or, more preferably, the operators will follow emergency procedures directing them to cool down and depressurize the RCS and align the RHR system for continued cooldown to cold shutdown. High pressure injection or normal charging may be used to maintain RCS inventory after depressurization, if necessary.

Dependencies displayed in the event tree model include: (1) when high pressure injection is available, then operator depressurization of RCS, accumulators, low pressure injection, and low pressure long-term recirculation are not considered; (2) when secondary cooling is available, bleed and feed operation is not addressed; (3) when high pressure injection and auxiliary feedwater are not available, early core damage is assumed; (4) when high pressure injection fails, then operator depressurization of RCS, accumulators, and low pressure injection must be available to prevent early core damage; (5) given high pressure injection, then either auxiliary feedwater or bleed and feed alignment must succeed, as well as long-term cooling; (6) given successful short-term cooling and safety injection, then either successful depressurization and RHR alignment or successful high pressure recirculation cooling must be established.

Figure 3.1-4

Small LOCA Event Tree



* Less than the 1.280E-13 cutoff

SLO - Small LOCA
SI3 - High Pressure Safety Injection
AF2 - Auxiliary Feedwater - SLO
OP2 - Cooldown and Depressurize the RCS
ACC - Accumulator Injection
SI1 - Low Pressure Safety Injection
OFB - Bleed and Feed
ES1 - Cooldown and Depressurize the RCS, SGs per EMG ES-11
LC2 - High Pressure Recirculation
LC1 - Low Pressure Recirculation

3.1.2.4 Steam Generator Tube Rupture

The Steam Generator Tube Rupture (SGTR) Event Tree is presented in Figure 3.1-5. This model applies to breaches of the primary coolant pressure boundary between the primary and secondary systems within the steam generator that exceed the capacity of the normal charging system. Initially, RCS pressure decreases as flow through the tube rupture depletes primary coolant inventory. Reactor trip occurs on either low pressurizer pressure or over-temperature delta-T. Failure of reactor trip is modeled in the ATWS event tree, Section 3.1.3.1, below. Typically, reactor trip is followed by ESFAS actuation on low pressurizer pressure. For smaller tube leakage, the operators may manually trip the reactor and initiate safety injection prior to the automatic actuation. Normal feedwater is isolated on ESFAS, and auxiliary feedwater is initiated for removal of decay heat. Operator action is required to control AFW flow to the unfaulted generators and isolate flow to the ruptured generator.

The operator is relied upon to perform a number of recovery actions following a SGTR. Specifically, the operators are directed by emergency procedures to: (1) identify and isolate the faulted steam generator by terminating auxiliary feedwater flow to the generator and by closing the MSIV on the generator and any auxiliary steam paths such as the steam supply line to the AFW turbine driven pump and SG blowdown lines; (2) perform an initial cooldown of the RCS using the intact generators to assure subcooling margin in the primary coolant; (3) depressurize the RCS to equilibrate pressures in the faulted generator and the RCS and to refill the pressurizer; (4) terminate safety injection to prevent RCS repressurization and overfill of the pressurizer; and (5) perform a post-SGTR cooldown and depressurization to cold shutdown.

In the event the above actions are not performed in a timely manner or in the event of multiple tube ruptures, the steam generator may fill, which may lead to failure of the secondary pressure boundary, by failure of the secondary safety or relief valves in the open position. In this case, loss of reactor coolant via the faulted generator to the atmosphere may occur. The operators are then directed by emergency procedures to cooldown and depressurize the RCS to stop primary coolant inventory loss. The operators have roughly 10 to 20 hours before depletion of RWST inventory to accomplish this cooldown.

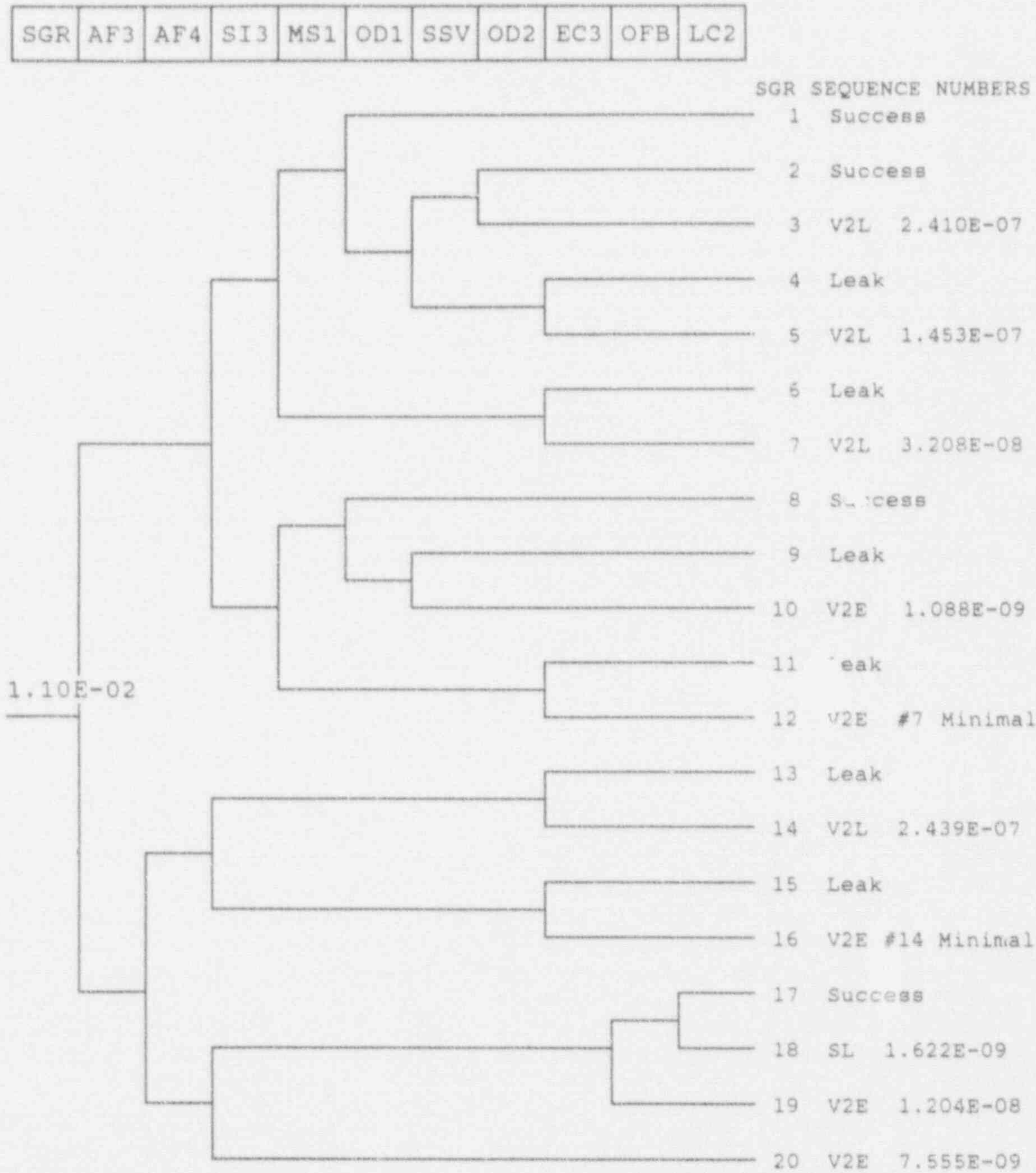
Dependencies displayed in the SGTR event tree model include: (1) if auxiliary feedwater to the intact steam generators is not established, then the faulted generator must be used; (2) failure of auxiliary feedwater to all SG requires bleed and feed operation and long-term high pressure recirculation; (3) given secondary cooling, if safety injection fails, then core damage may be avoided by either successful RCS depressurization and faulted SG isolation sufficient to stop primary fluid loss or rapid cooldown and depressurization to near atmospheric conditions and initiation of RHR cooling; (4) failure of the operators to initially depressurize the RCS to terminate primary to secondary flow requires either successful isolation of the faulted SG and maintenance or restoration of secondary integrity or rapid cooldown and depressurization for RHR cooling; (5) given successful auxiliary feedwater and safety injection, isolation of the faulted generator is required to avoid leakage of reactor coolant to the atmosphere; (6) given secondary cooling with a nonfaulted SG, failure to isolate the faulted generator requires rapid cooldown for RHR cooling; and (7) leakage of reactor coolant to the atmosphere will occur if the faulted SG is used for secondary cooling or if secondary isolation and integrity is not maintained.

3.1.2.5 Interfacing Systems LOCA

No event tree model for the interfacing systems LOCA is presented, inasmuch as this event is assumed to lead to unavoidable late core damage with containment bypass. Hypothesized interfacing systems LOCA may result from failures of the barriers between high pressure RCS piping and low pressure piping in the RHR or High Pressure Safety Injection System (HPSI). If the low pressure piping passes through containment, then ruptures of that piping outside containment result in a LOCA which bypasses containment. Depending upon the location of the break, the line may be nonisolable.

Since the interfacing LOCA occurs outside the containment, core damage will likely take place due to failure of recirculation. However, if the break can be isolated and terminated and safety injection is initiated, then core melt might be delayed or prevented. If long term core cooling can be established, core damage might also be avoided. However, for this analysis, it is simply assumed that if an interfacing LOCA occurs, core damage will follow.

Figure 3.1-5
Steam Generator Tube Rupture Event Tree



SGR - Steam Generator Tube Rupture
AF3 - Auxiliary Feedwater to Intact SGs
AF4 - Auxiliary Feedwater to Ruptured SG
SI3 - High Pressure Safety Injection
MS1 - Main Steam Isolation
OD1 - Stabilize RCS and Ruptured SG Pressure Before SG Overfill
SSV - Secondary Integrity
OD2 - Stabilize RCS and Ruptured SG Pressure After SG Overfill
EC3 - Cooldown and Depressurize RCS, SGs per EMGs C-31/32
OFB - Bleed and Feed
LC2 - High Pressure Recirculation

3.1.2.6 Reactor Vessel Failure

No event tree for the reactor vessel failure is presented because this event models losses of reactor coolant beyond the capability of the ECCS. Postulated events include rupture of more than one large RCS pipe or vessel rupture of sufficient magnitude that the ECCS cannot reflood the core after initial blowdown or thereafter keep the core covered. The event is assumed to result in early core damage regardless of the operation of safety injection. Containment safeguards are addressed separately.

3.1.2.7 Transient with Power Conversion System

The transient with power conversion system (i.e. primarily main feedwater system) event tree is presented in Figure 3.1-6. This model depicts plant functioning following all transient events except those that result in a total loss of the power conversion system (including the main feedwater system). Thus, this model reflects the availability of main feedwater to provide secondary cooling in the event of failure of auxiliary feedwater. Failure of reactor trip is modeled in the ATWS tree, Section 3.1.3.1 below.

Auxiliary feedwater is actuated upon low steam generator level, or manually. Steam relief is provided either by steam dump to the condenser or use of the SG PORVs to the atmosphere. The modeled dependencies include: (1) if secondary cooling is established, the transient is assumed to be successfully terminated; (2) if auxiliary feedwater fails, operator action to establish main feedwater flow or to depressurize the SG and use the condensate pumps is modeled; and (3) if all secondary cooling is lost, operator action to initiate bleed and feed operation is modeled, with long-term cooling provided by high pressure recirculation.

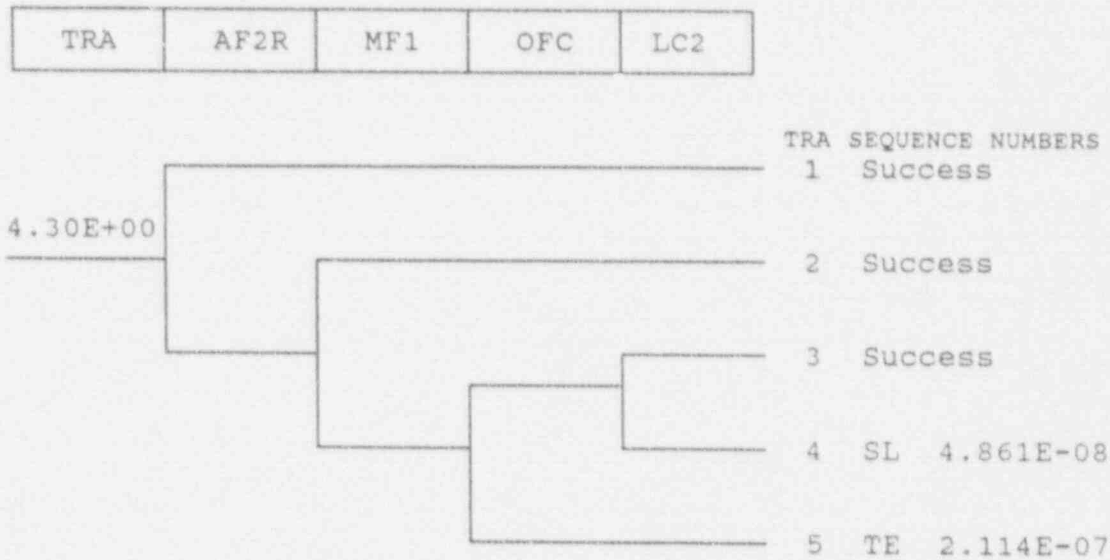
3.1.2.8 Transient without Power Conversion System

The transient without power conversion system event tree is presented in Figure 3.1-7. This model reflects plant behavior following transients in which the power conversion system and main feedwater is unavailable and not recoverable.

Auxiliary feedwater is normally actuated by low SG level or manually, and reactor trip is required. Failure of reactor trip is modeled in the ATWS tree, Section 3.1.3.1 below. Modeled dependencies include: (1) if auxiliary feedwater succeeds, the transient is assumed to be terminated; and (2) if auxiliary feedwater fails, operator action to initiate bleed and feed operation is modeled, with long-term cooling provided by high pressure recirculation.

Figure 3.1-6

Transient With Power Conversion System Event Tree



TRA - Transient with Power Conversion System

AF2R - Auxiliary Feedwater

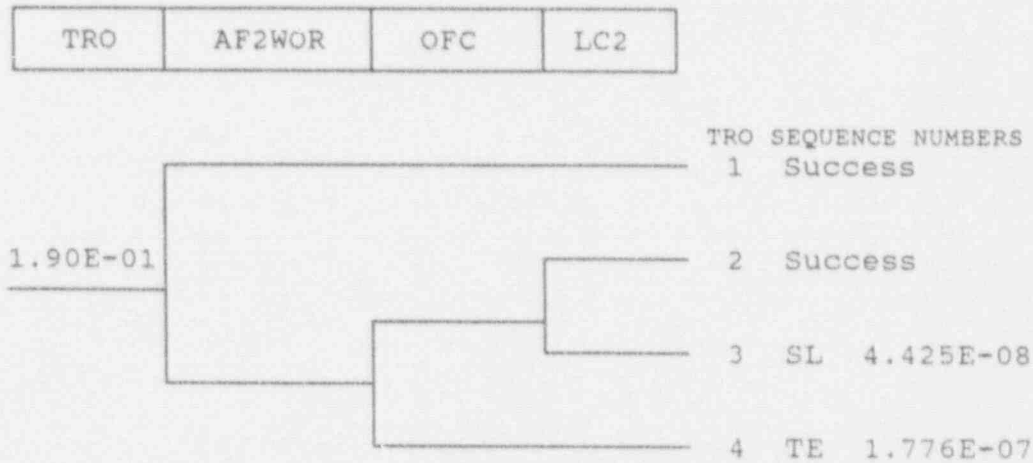
MF1 - Main Feedwater

OFC - Bleed and Feed

LC2 - High Pressure Recirculation

Figure 3.1-7

Transient Without Power Conversion System Event Tree



TRO - Transient without Power Conversion System

AF?WOR - Auxiliary Feedwater

OFC - Bleed and Feed

LC2 - High Head Recirculation

3.1.2.9 Steamline/Feedline Break

The steamline/feedline break event tree is presented in Figure 3.1-8. This event tree applies to those secondary pipe breaks or valve openings that are large enough to cause safety injection actuation and require isolation of the ruptured steam generator. High pressure safety injection is initiated to provide makeup to compensate for reactor coolant volume shrinkage associated with the cooldown and to provide boration for additional shutdown margin. If the break is isolable (downstream of the MSIV or upstream of the MFIV), the initial RCS cooldown will be terminated upon automatic isolation of the rupture and decay heat will cause the RCS to heat up to no-load conditions. If the break is nonisolable, the cooldown will continue and then reverse after the faulted SG completes most of its blowdown. Operator action to terminate high pressure safety injection is required to limit RCS inventory and prevent overfill of the pressurizer. Auxiliary feedwater and secondary cooling using the intact SGs is required for decay heat removal. If secondary cooling is unavailable, the operators are directed by emergency procedures to align bleed and feed.

Reactor trip occurs in these events, but the indications that lead to trip depend upon the size and type of break. A large steamline rupture leads promptly to low steamline pressure MSIV closure and an SI, which also trips the reactor. A medium size blowdown of the SG causes a reduction in Tave, which the reactor control responds to by stepping the control rods out of the core. Trip may occur on either over temperature delta-T or overpower delta-T. SI actuation follows on low pressurizer pressure, low steamline pressure, or high containment pressure. For feedline breaks, the reactor will be tripped by low SG level. If the SG blows down, MSIV closure and SI actuation will occur on low steamline pressure. Smaller breaks progress less quickly than the larger breaks, but the systems will respond similarly.

Displayed dependencies in the steamline/feedline break event tree include: (1) if high pressure safety injection and auxiliary feedwater/secondary cooling succeed, operator action to terminate SI is addressed; (2) operator failure to terminate SI before filling the pressurizer results in a consequential small LOCA; (3) if high pressure SI fails but main steam isolation and auxiliary feedwater are successful, the accident sequence is terminated; (4) if auxiliary feedwater and secondary cooling fails but high pressure SI succeeds, bleed and feed is addressed; (5) bleed and feed requires long-term high pressure recirculation cooling to be aligned to avoid late core damage; and (6) failure of high pressure SI along with either failure to close 3 of 4 MSIVs or failure of the auxiliary feedwater system results in early core damage.

Failure of reactor trip is addressed in the ATWS model, Section 3.1.3.1, below. The frequency of consequential small LOCA caused by the sequence of success of high pressure SI and auxiliary feedwater (regardless of successful main steam isolation) and failure of the operator to properly terminate SI injection flow before filling the pressurizer and causing the failure of a relief valve is included in the initiating event frequency of the Small LOCA event tree.

3.1.2.10 Loss of Offsite Power

The loss of offsite power event tree is presented in Figure 3.1-9. This model applies to those transients that begin with the loss of main plant AC power from the switchyard, in which at least one emergency diesel generator has started and is fully available to mitigate the transient. Failure of both standby generators results in station blackout, which is modeled in Section 3.1.3.5, below.

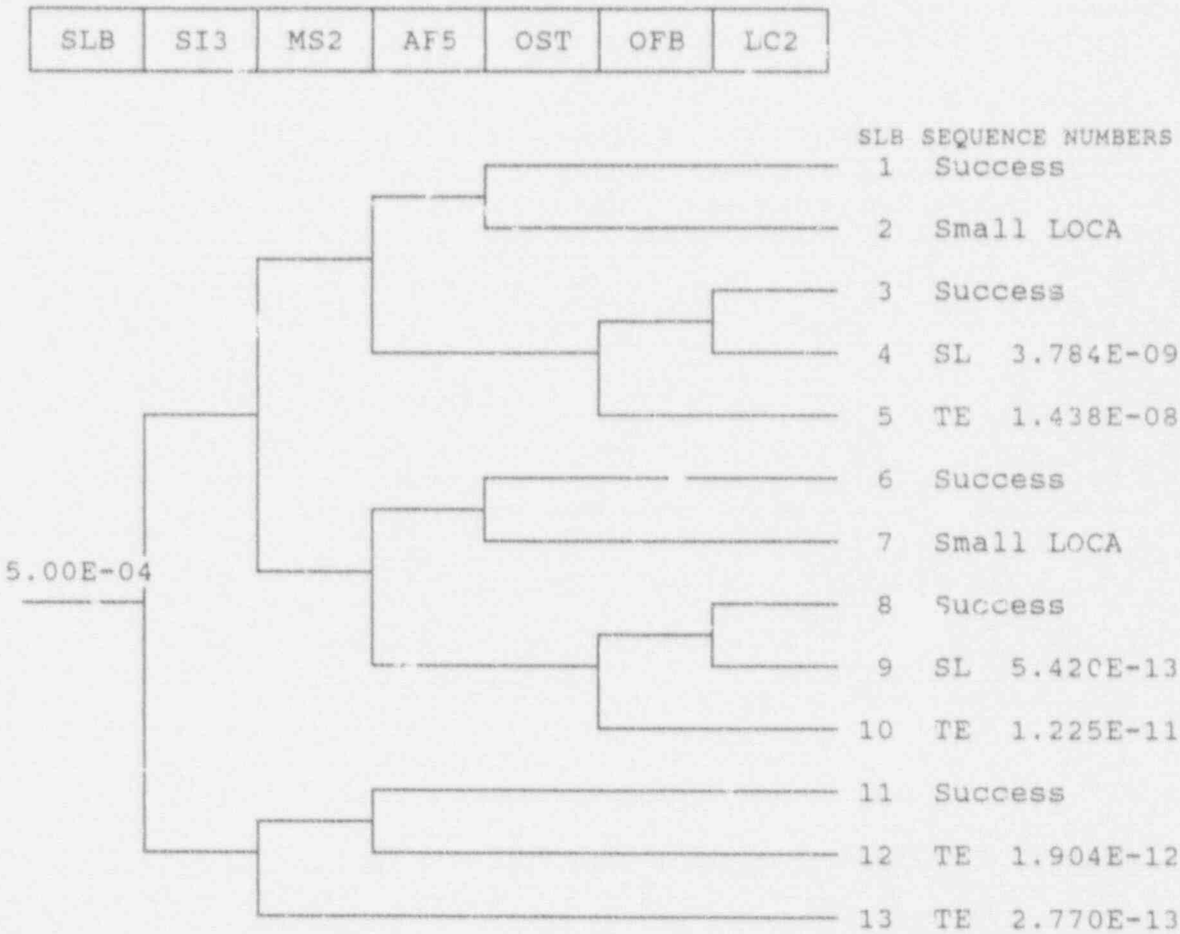
Reactor trip is assumed to occur following deenergization of the trip breakers. Failure of the reactor to trip is addressed in the ATWS model, Section 3.1.3.1, below. Following trip, decay heat is normally removed via the secondary, with auxiliary feedwater flow to the SG and steam venting to the atmosphere through either the PORV or the safety valves. Steam venting using the steam dump valves is not considered due to the unavailability of the circulating water pumps and the condenser. If auxiliary feedwater fails, bleed and feed cooling would be available, requiring the operator to initiate high pressure safety injection and open the pressurizer PORVs to remove heat from the RCS. Long-term cooling requires later alignment of high pressure recirculation, using the RHR system to supply the high pressure ECCS pumps and cool recirculation flow.

Modeled dependencies include: (1) if auxiliary feedwater is successful, the transient is terminated; (2) if auxiliary feedwater fails, operator action to establish bleed and feed and long-term high pressure recirculation is addressed; and (3) if auxiliary feedwater and operator bleed and feed both fail, early core damage results.

The loss of offsite power event tree includes a fifth sequence which models the coremelt probability due to a possible RCP Seal LOCA occurrence. If the Component Cooling Water (CCW) train which was operating at the time of the LSP event (Train A is assumed) fails; RCP seal cooling is lost due to loss of flow through the CCW Service Loop to the RCP Thermal Barrier Cooling Coils (TBCC). Seal injection flow is also assumed to be lost due to loss of lube oil cooling to the charging pump which was aligned for normal seal injection. For this sequence, the "standby" CCW train (Train B by assumption) is considered to be functional and operating. However, the operation of CCW Train B will not, in and of itself, result in re-establishment of RCP seal cooling. Operator action to establish RCP seal cooling is required by either 1) realignment of the CCW Service Loop to CCW Train B, or 2) alignment of a charging pump being supported by CCW Train B cooling (i.e. CCP B) to the alternate or normal seal injection flow path. Failure to re-establish at least one of the above two RCP seal cooling methods within a reasonable period of time (approximately 30 minutes) is assumed to result in an RCP seal LOCA. The RCP seal LOCA for this sequence is assumed to be of a Small LOCA magnitude. Mitigation of this small LOCA is considered using the systems and sequences of the Small LOCA event tree without the support of the offsite AC power source and CCW Train A. Failure to mitigate this small LOCA is assumed to result in early core damage.

Figure 3.1-8

Steamline / Feedline Break Event Tree



SLB - Steamline / Feedline Break

SI3 - High Pressure Safety Injection

MS2 - Main Steam Isolation

AF5 - Auxiliary Feedwater to Intact SGs

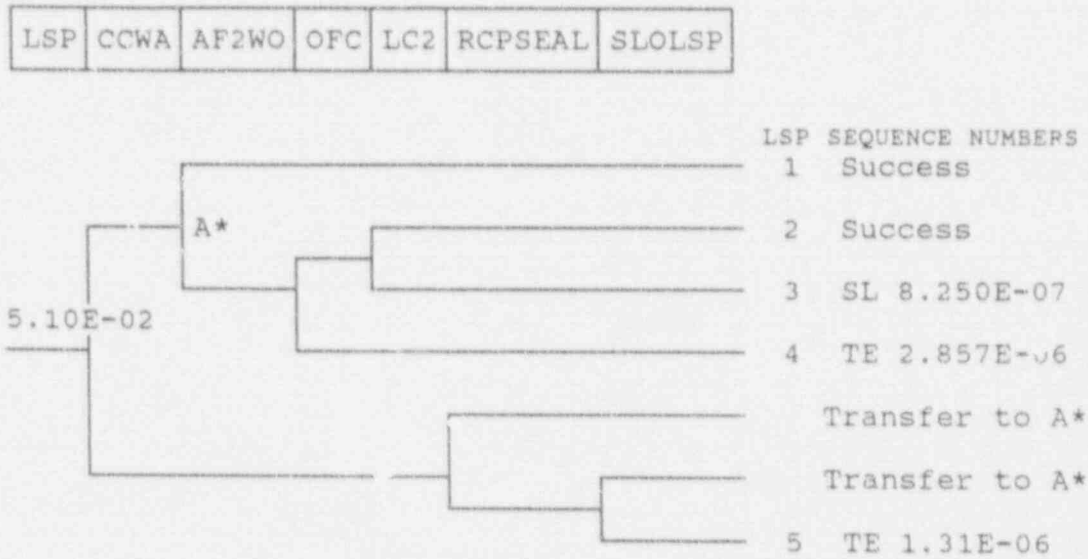
OST - Terminate High Pressure Safety Injection

OFB - Bleed and Feed

LC2 - High Pressure Recirculation

Figure 3.1-9

Loss of Offsite Power Event Tree



LSP - Loss of Offsite Power

CCWA - Failure of CCW A Train

AF2WO - Auxiliary Feedwater

OFC - Bleed and Feed

LC2 - High Pressure Recirculation

RCPSEAL - Align RCP Seal Cooling

SLOLSP - Conditional Small LOCA Coremelt

3.1.3 Special Event Trees

Separate event trees have been used to analyze several special events, including Anticipated Transient Without Scram (ATWS), Loss of Component Cooling Water (CCW), Loss of All Service Water (SWS), Loss of a 125 VDC bus (DCC), Station Blackout (SBO), and containment safeguards. These models are described in the following subsections.

3.1.3.1 Anticipated Transient Without Scram

The anticipated transient without scram (ATWS) event tree is presented in Figure 3.1-10. This model applies to those transient and accident events in which there is a failure of the control rods to drop into the core when needed. Failure of reactor trip coincident with loss of secondary heat sink results in a high pressure transient in the RCS. This model reflects operator and system operations failure to shut down the reactor and prevent overpressurization of the primary system. It is assumed that overpressurization leads to unrecoverable early core damage. The estimate of initiating event frequency includes all transient and accident scenarios in which the reactor trip function fails given a demand. The loss of main feedwater initiating event leads to the most severe RCS pressure transient and is therefore the limiting ATWS precursor. The loss of main feedwater has been analyzed for the ATWS event.

The severity of the RCS pressure transient is a function of initial core conditions determining core power and reactivity feedback conditions. Higher core power at the time of the ATWS and times late in the core cycle will exhibit less negative temperature reactivity feedback, both of which contribute to peak pressures. The peak pressure can be reduced by rapid operator actions to manually trip the reactor or insert the control rods and initiate full auxiliary feedwater flow to the steam generators. WCGS also has installed an ATWS Mitigating System Actuating Circuitry (AMSAC) system, which provides redundant, nonsafety-grade reactor and turbine trips and auxiliary feedwater actuation on low steam generator level. AMSAC is automatically available whenever reactor power is above 40%. For events initiated at power levels below 40%, the peak pressure in the RCS is not predicted to exceed the allowable 3200 psig limit.

Following a transient in which reactor trip fails, the operators are instructed by emergency procedure to: (1) trip the reactor by (in order of preference): initiating a manual trip, manually stepping the control rods into the core, or dispatching an operator to manually open the breakers for the rod drive motor generators (RDMGs); (2) verify or manually trip the turbine; (3) verify auxiliary feedwater actuation; and (4) initiate emergency boration. Emergency boration is too slow to prevent RCS pressurization for limiting ATWS events, and it has not been addressed in this model.

Figure 3.1-10 (Page 1 of 2)

ATWS Event Tree

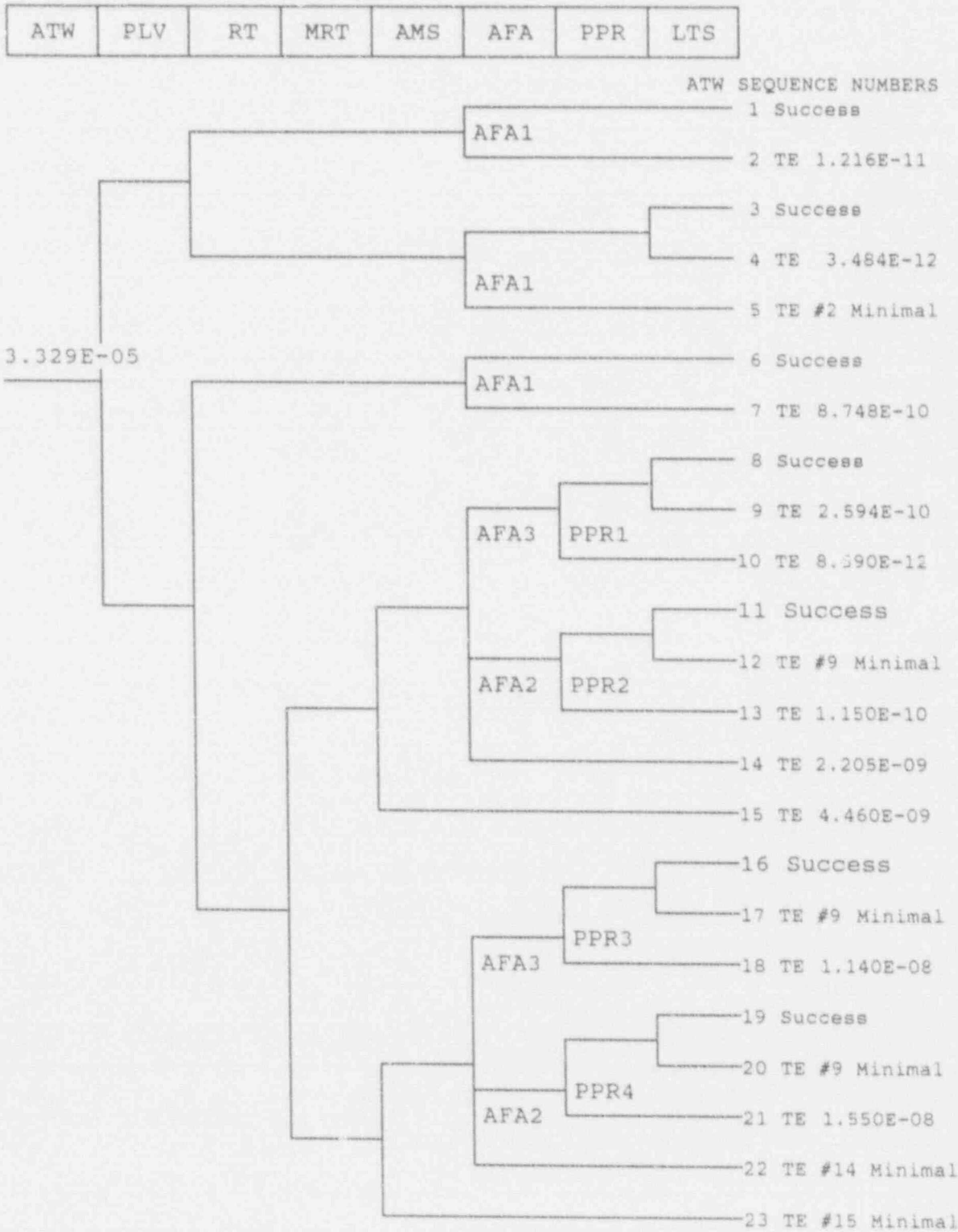


Figure 3.1-10 (Page 2 of 2)

ATWS Event Tree

ATW - ATWS, Anticipated Transient Without Scram

PLV - Reactor Power Level

RT - Manual Reactor Trip or Rod Insertion

MRT - Manual Trip of RDMGs

AMS - AMSAC

AFA - Auxiliary Feedwater for ATWS

AFA1 - Auxiliary Feedwater System (1/3 AFW Pumps to 2/4 Steam Generators)

AFA2 - Auxiliary Feedwater System (2/2 MD or 1/1 TD AFW Pumps to 2/4 Steam Generators)

AFA3 - Auxiliary Feedwater System (2/2 MD and 1/1 TD AFW Pumps to 4/4 Steam Generators)

PPR - Primary Pressure Relief

PPR1 - Pressurizer Pressure Relief (AFA3 and MRT successful)

PPR2 - Pressurizer Pressure Relief (AFA2 and MRT successful)

PPR3 - Pressurizer Pressure Relief (AFA3 successful, MRT fails)

PPR4 - Pressurizer Pressure Relief (AFA2 successful, MRT fails)

LTS - Long Term Shutdown

3.1.3.2 Loss of the Component Cooling Water System

The loss of Component Cooling Water system (CCW) event tree is presented in Figure 3.1-11. This model applies to the transient initiated by failure of the component cooling water system. The CCW supplies cooling water to the seals of the Reactor Coolant Pumps (RCPs), the charging pumps, the high pressure SI pumps, the RHR heat exchangers and the RHR pump seals. Following the loss of CCW flow, the operators are instructed to trip the reactor, turbine, and RCPs. Failure of reactor trip is addressed in the ATWS event tree.

The charging pumps may also be turned off to prevent failure from the loss of lube oil coolant flow. The RCP seals, with neither CCW flow or seal injection, will eventually begin to leak and may fail resulting in significant reactor coolant losses. This loss of reactor coolant will ultimately lead to core damage if CCW flow and safety injection is not recovered. If a safety injection actuation occurs, the operators have to prevent the high pressure SI pumps from starting or turn them off to prevent damage. This allows the pumps to be used later for injection if CCW is recovered.

To reduce the chance of RCP seal damage, the operators are instructed to cool down and depressurize the RCS with the auxiliary feedwater or main feedwater systems and either the SG PORVs or steam dump valves to the condenser. These steps reduce the thermal and pressure challenge to the RCP seal O-rings and reduce the driving head and consequential loss of coolant rates. If auxiliary feedwater, or main feedwater, and RCS cooldown depressurization are successful, RCS inventory replenishment occurs by injection of the accumulator contents. As RCS temperature and pressure are further reduced, injection of the RWST contents may proceed using the low pressure safety injection (RHR) pumps. With continued injection the RWST contents will deplete such that switchover of the RHR pump suction lines to the containment recirculation sumps will be required. Note that this scenario is proceeding without restoration of a CCW cooling supply. This is allowable in that heat removal from the RCS is through the steam generators, and operation of the RHR pumps in either the injection or recirculation mode functions to maintain the RCS inventory and prevent core uncovery. According to the NSSS supplier, RHR pump operation without CCW cooling to the mechanical seal coolers is acceptable for this function. CCW cooling to the RHR heat exchangers is not required so long as heat removal from the RCS is proceeding via the steam generators. With failure of the above described low pressure injection process, continued auxiliary Feedwater or main Feedwater, operation, with RCS cool down and depressurization acts to extend the time available to recover CCW before the core is damaged.

If all sources of feedwater are lost, the steam generators would start to dry out and the RCS would heat up and pressurize due to the loss of secondary heat sink. The pressurizer PORVs and safety valves would release RCS inventory, eventually leading to core uncovery and damage after the reactor coolant saturates. The challenge to the RCP seals is a function of RCS pressure and time. From expected leakage through the RCP seals, the time available before damaging the core is expected to be about 2 hours from the loss of all feedwater flow.

In general, the recovery from this event ultimately requires recovery of CCW flow. If operator cooldown of the RCS is sufficient to minimize seal leakage and the chance of a seal LOCA, it is assumed that CCW flow must be restored in about 8 hours after the transient to prevent core damage. This time period is less than the time for core uncovery given expected RCP seal leakage (non-LOCA).

Dependencies modeled in the event tree include: (1) if all feedwater is lost and CCW is not restored within two hours, an early core damage will result; (2) if CCW has been restored before core damage is postulated (2 hours or 8 hours, depending upon feedwater availability), then high pressure safety injection is required: (a) if feedwater is available, then 1 of 4 SI pumps is sufficient to restore RCS coolant inventory, or (b) if no feedwater is available, then it is assumed that 2 of 2 charging pumps are necessary to establish bleed and feed cooling; (3) if safety injection is successful, then it is assumed that long-term recirculation cooling is necessary to prevent late core damage.

The loss of component cooling water event tree includes sequences 30 and 31 which model the coremelt probability due to possible RCP seal LOCA occurrence. If the operating Component Cooling Water train (Train A assumed) fails on a per year basis, RCP seal cooling is lost due to loss of flow through the CCW service loop to the RCP Thermal Barrier Cooling Coils (TBCC). Seal injection flow is also assumed to be lost due to loss of lube oil cooling to the charging pump which was aligned for normal seal injection. For these sequences, startup and operation of the "standby" CCW train (Train B) is considered to be successful. However, the operation of CCW Train B will not, in and of itself, result in re-establishment of RCP seal cooling. Operator action to establish RCP seal cooling is required by either 1) realignment of the CCW service loop to CCW Train B, or 2) startup and alignment of a charging pump being supported by CCW Train B cooling (i.e. CCP B) to the alternate or normal seal injection flow path. Failure to re-establish at least one of the above two RCP seal cooling methods within a reasonable period of time (approximately 30 minutes) is assumed to result in an RCP seal LOCA. Loss of CCW cooling to the RCPs will not automatically result in a trip of the RCPs. If the operator fails to trip the RCPs within 10 minutes of loss of all RCP seal cooling, an RCP seal LOCA of medium LOCA magnitude is assumed. If the operator successfully trips the RCPs within 10 minutes but fails to re-establish seal cooling within 30 minutes, an RCP seal LOCA of small LOCA magnitude is assumed. Mitigation of the medium or small magnitude RCP seal LOCA is considered using the systems and sequences of the medium and small LOCA event trees, respectively, without the support of CCW Train A. Failure to mitigate the medium or small magnitude RCP seal LOCA is assumed to result in early core damage.

3.1.3.3 Loss of the Service Water System

The loss of Service Water system (SW) event tree is presented in Figure 3.1-12. This model applies to the transient initiated by failure of the service water system and failure of the Essential Service Water system (ESW). The ESW supplies cooling water to the CCW and to several safety-related pumps and room coolers. In addition to losing CCW-cooled loads, the motor driven AFW pumps and the condensate pumps will be lost, as well as the safety-grade backup water supply to the auxiliary feedwater system. For identification of CCW loads, see Section 3.1.3.2 above. The loss of service water causes loss of cooling to the turbine generator auxiliary systems and a turbine trip, which causes a reactor trip. Following the loss of ESW flow, the operators will have to trip the RCPs, CCW pumps, and the charging pumps. If auxiliary feedwater automatically actuates, the motor-driven AFW pumps also will have to be stopped to prevent damage (assumed due to loss of room cooling), and the turbine-driven AFW pump used for reactor cooldown and depressurization. Failure of reactor trip is addressed in the ATWS event tree.

The RCP seals, with neither CCW flow or seal injection, will eventually begin to leak and may fail resulting in significant reactor coolant losses. This loss of reactor coolant will ultimately lead to core damage if ESW, CCW, and safety injection are not recovered. If a safety injection actuation occurs, the operators have to prevent the high pressure SI pumps from starting or turn them off to prevent damage. This allows the pumps to be used later for injection if ESW and CCW are recovered.

To reduce the chance of seal damage, the operators are instructed to cool down and depressurize the RCS with the turbine-driven auxiliary feedwater pump and the SG PORVs or steam dump valves to the condenser. These steps reduce the thermal and pressure challenge to the RCP seals O-rings and reduce the driving head and consequential loss of coolant rates. Depressurization also makes the injection water from the accumulators available to replenish lost RCS inventory. These actions extend the time available to recover ESW and CCW before the core is damaged.

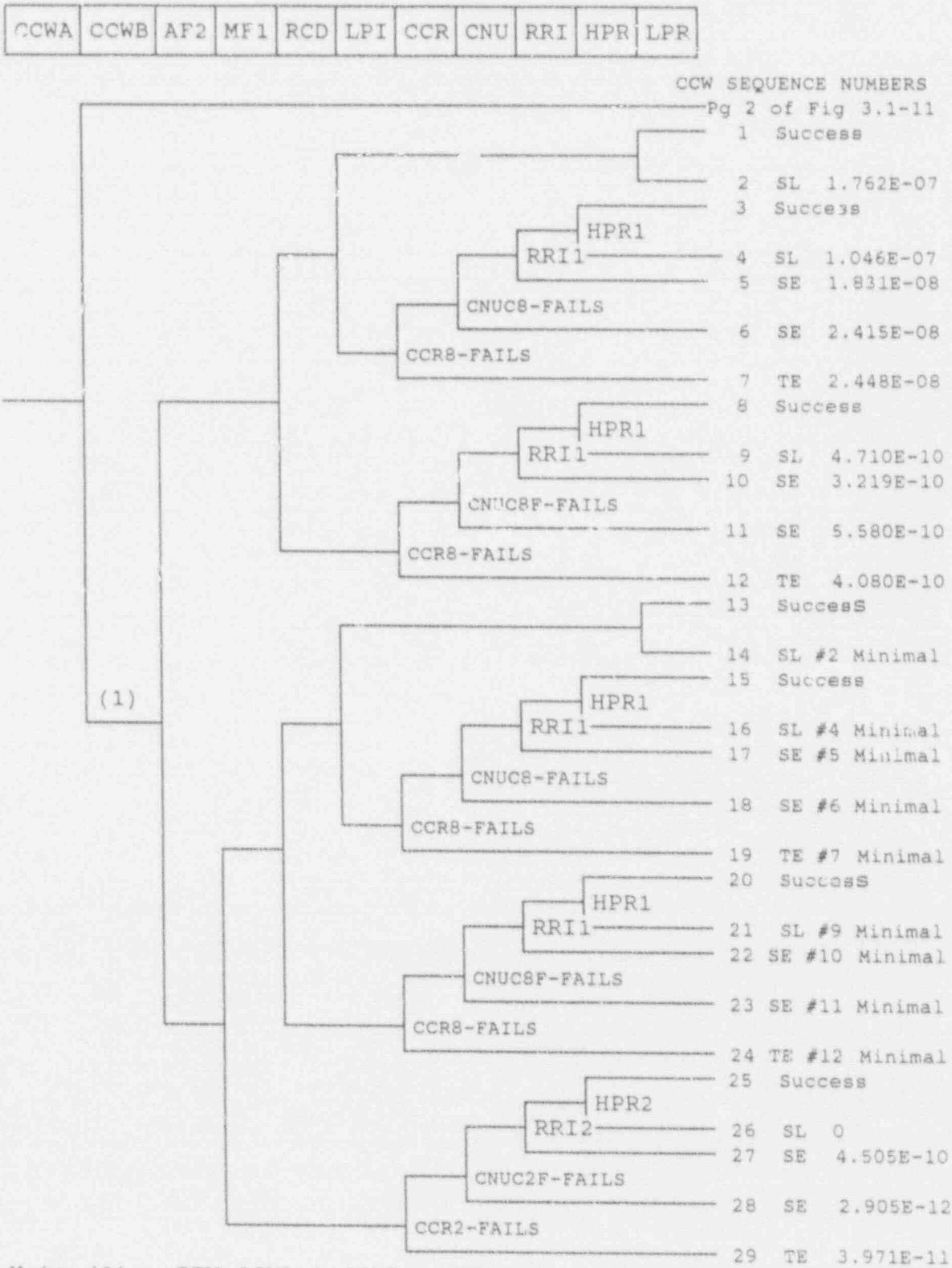
If all sources of feedwater are lost, the steam generators would start to dry out and the RCS would heat up and pressurize due to the loss of secondary heat sink. The pressurizer PORVs and safety valve would release RCS inventory, eventually leading to core uncovery and damage after the reactor coolant saturates. The challenge to the RCP seals is a function of RCS pressure and time. From expected leakage through the RCP seals, the time available before damaging the core is expected to be about 2 hours from the loss of all feedwater flow.

The recovery from this event ultimately requires recovery of ESW and CCW flow. If operator cooldown of the RCS is sufficient to minimize seal leakage and the chance of a seal LOCA, it is assumed that ESW flow must be restored in about 8 hours after the transient to prevent core damage. This time period is less than the time for core uncovery given expected RCP seal leakage (non-LOCA).

Dependencies modeled in the event tree include: (1) if all feedwater is lost and ESW and CCW are not restored within two hours, early core damage will result; (2) if ESW and CCW have been restored before core damage is postulated (2 hours or 8 hours, depending upon auxiliary feedwater availability), then high pressure safety injection is required: (a) if feedwater is available, then 1 of 4 ECCS pumps is sufficient to restore RCS coolant inventory, or (b) if no feedwater is available, then it is assumed that 2 of 2 charging pumps are necessary to establish bleed and feed cooling; (3) if safety injection is successful, then it is assumed that long-term recirculation cooling is necessary to prevent late core damage.

The loss of all service water event tree includes a sequence which addresses the probability of loss of the vital 120 VAC inverters due to loss of room cooling to the DC switchboard rooms. The heat sink for the Class 1E Electrical Equipment Air conditioning Units is provided by the essential service water system. With a loss of all service water, the heat removal function for the DC switchboard rooms will be lost. However, the heat loads for these rooms, with a loss of all service water event, will not decrease. The temperature in these rooms will increase and eventually exceed the high room temperature alarm setpoint, resulting in control room annunciation. Operator action to provide auxiliary cooling by opening doors and possibly installing fans, in accordance with the applicable Control Room Standing Order, will alleviate the high temperature conditions in these rooms. If action to address the high temperature condition in the DC Switchboard rooms is not taken, the vital 120 VAC inverters are assumed to fail. Failure of the inverters will result in loss of motive power to the SG atmospheric relief valves, thereby preventing continuance of the RCS cooldown and depressurization function. However, motive power may be restored to the atmospheric relief valves by local manual action to align the 120 VAC switchboards to their alternate power source. Failure of operator action to locally align the 120 VAC switchboards to their alternate power source upon inverter failure is assumed to fail the RCS cooldown and depressurization function and lead to an early core damage condition.

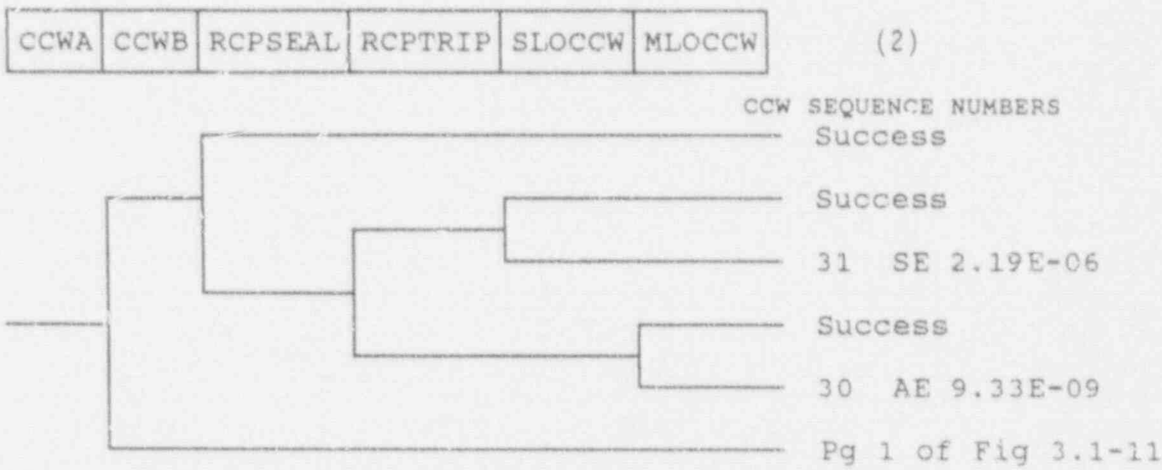
Figure 3.1-11 (Page 1 of 2)
Loss of CCW Event Tree



Note (1): IEV-CCWA * CCWB = IEV-CCW = 1.62E-04/YEAR

Figure 3.1-11 (Page 2 of 2)

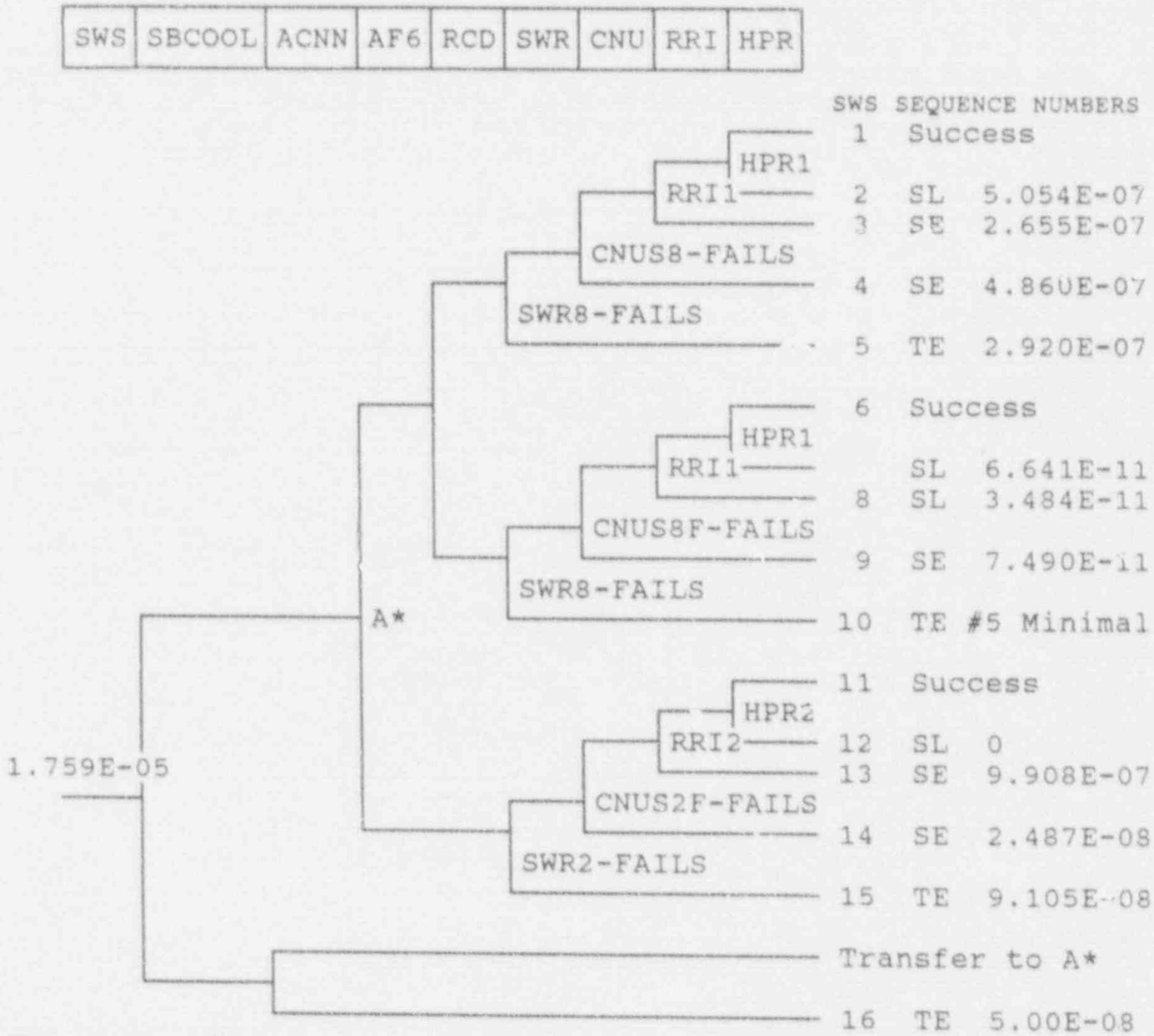
Loss of CCW Event Tree



Note (2): The event sequence numbering is kept the way shown to maintain consistency with the already defined Level 2 sequence numbering scheme.

CCWA - Failure of CCW A Train
CCWB - Failure of CCW B Train
AF2 - Auxiliary Feedwater
MF1 - Main Feedwater
RCD - RCS Cooldown
LPI - Low Pressure Injection
CCR - Restore CCW Cooling
  CCR2 - Restore CCW Cooling within 2 Hours
  CCR8 - Restore CCW Cooling within 8 Hours
CNU - Core Not Uncovered
  CNUC2F - Core Uncovery Occurs within 2 Hours, RCD Fails
  CNUC8 - Core Uncovery Occurs within 8 Hours, RCD Successful
  CNUC3F - Core Uncovery Occurs within 8 Hours, RCD Fails
RRI - Restore RCS Inventory
  RRI1 - RCS Inventory Restoration (1/4 ECCS Pumps Required; 1/2 ESF Trains Available)
  RRI2 - RCS Inventory Restoration (2/2 Charging Pumps Required; 1/2 ESF Trains Available) Function Fails by Definition
HPR - High Pressure Recirculation
  HPR1 - High Pressure Recirculation (1/4 ECCS Pumps Required; 1/2 ESF Trains Available)
  HPR2 - High Pressure Recirculation (2/2 Charging Pumps Required; 1/2 ESF Trains Available) Not Addressed since RRI2 Fails by Definition
LPR - Low Pressure Recirculation
RCPSEAL - Align RCP Seal Cooling
RCPTRIP - Trip RCPs
SLOCCW - Conditional Small LOCA Coremelt
MLOCCW - Conditional Medium LOCA Coremelt

Figure 3.1-12
Loss of SWS Event Tree



SWS - Loss of Service Water System
SBCOOL - DC Switchboard Room Cooling
ACNN - Inverter Alternate Power Supply
AF6 - Auxiliary Feedwater
RCD - RCS Cooldown
SWR - Restore Service Water Cooling
SWR2 - Restore SWS Cooling within 2 Hours
SWR8 - Restore SWS Cooling within 8 Hours
CNU - Core Not Uncovered
CNUS2F - Core Uncovery Occurs within 2 Hours, RCD Fails
CNUS8 - Core Uncovery Occurs within 8 Hours, RCD Successful
CNUS8F - Core Uncovery Occurs within 8 Hours, RCD Fails
RRI - Restore RCS Inventory
RRI1 - RCS Inventory Restoration (1/4 ECCS Pumps Required; 1/2 ESF Trains Available)
RRI2 - RCS Inventory Restoration (2/2 Charging Pumps Required; 1/2 ESF Trains Available) Function Fails by Definition
HPR - High Pressure Recirculation
HPR1 - High Pressure Recirculation (1/4 ECCS Pumps Required; 1/2 ESF Trains Available)
HPR2 - High Pressure Recirculation (2/2 Charging Pumps Required; 1/2 ESF Trains Available) Not Addressed since RRI2 Fails by Definition

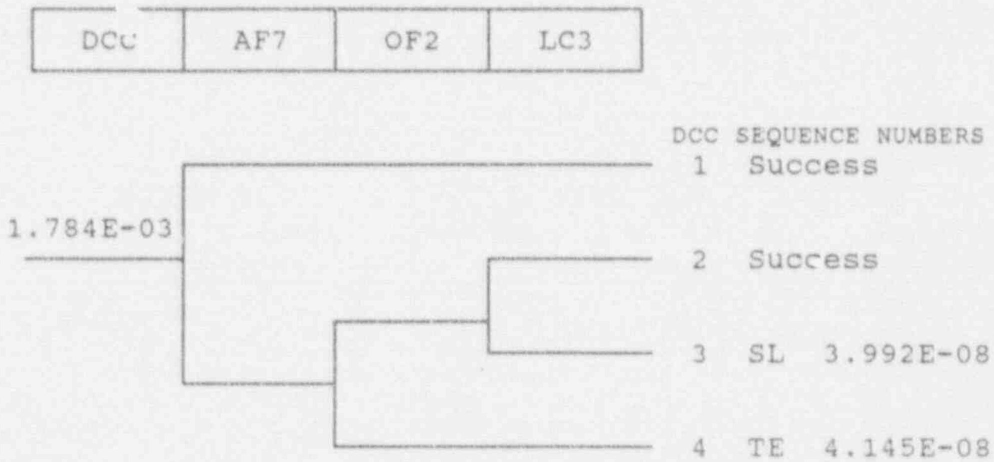
3.1.3.4 Loss of a Vital DC Bus

The loss of a vital DC bus event tree is presented in Figure 3.1-13. This model applies to events resulting in a loss of one vital 125V DC bus. The event will progress as a transient without main feedwater where one train of front line safety systems will be rendered unavailable in response to an automatic actuation signal or manual actuation from the control room. Loss of a DC bus causes a consequential loss of the associated vital AC bus, disabling various flow, temperature, pressure, and level indications as well as causing a loss of control power to various pumps, valves, breakers, and the associated emergency diesel generator. The event tree is identical to that of Transient without Power Conversion System, Section 3.1.2.8 above, except that system success criteria reflect the unavailability, without local manual component actuation, of one train of safety systems.

The electric power loss causes closure of the main feedwater isolation valves, leading to reactor trip on low steam generator level. Failure to scram the reactor is modeled in the ATWS event tree. Modeled dependencies include: (1) if auxiliary feedwater succeeds, the transient is assumed to be terminated; and (2) if auxiliary feedwater fails, operator action to initiate bleed and feed operation is modeled, with long-term cooling provided by high pressure recirculation.

Figure 3.1-13

Loss of a Vital DC Bus Event Tree



DCC - Loss of Vital DC Bus

AF7 - Auxiliary Feedwater

OF2 - Bleed and Feed

LC3 - High Pressure Recirculation

3.1.3.5 Station Blackout

The station blackout event tree is presented in Figure 3.1-14. This model depicts the loss of offsite power transient with coincident loss of all onsite AC power. The reactor will trip upon deenergization of the control rod trip breakers and Control Rod Drive Motor (CRDM) coils, and the turbine driven auxiliary feedwater pump will provide flow to the SG to remove decay heat. Steam dump to the atmosphere will occur through the steam generator PORVs and safety valves. The operators are instructed by emergency procedures to verify auxiliary feedwater flow, to isolate the RCS, perform and verify containment isolation, strip all but essential loads from the DC buses, and initiate RCS cooldown. If no consequential events (e.g., RCP seal LOCA) occur, these actions maintain the plant in a safe, stable condition for at least the time that DC power and the condensate storage tank supply of water to the auxiliary feedwater pump are available.

The station blackout results in a loss of cooling flow from the CCW system and from the charging system to the RCP seals. The seals will leak, and may fail resulting in significant reactor coolant losses which will ultimately lead to core damage if AC power and safety injection are not recovered. To reduce the chance of seal damage, the operators are instructed to cool down and depressurize the RCS with the auxiliary feedwater system and the SG PORVs. These steps reduce the thermal and pressure challenge to the RCP seals O-rings and reduce the driving head and consequential loss of coolant rates. Depressurization also makes the injection water from the accumulators available to replenish lost RCS inventory. These actions extend the time available to recover AC power before the core is damaged.

If auxiliary feedwater is lost, the steam generators would start to dry out and the RCS would heat up and pressurize due to the loss of secondary heat sink. The pressurizer PORVs and safety valves would release RCS inventory, eventually leading to core uncovery and damage after the reactor coolant saturates.

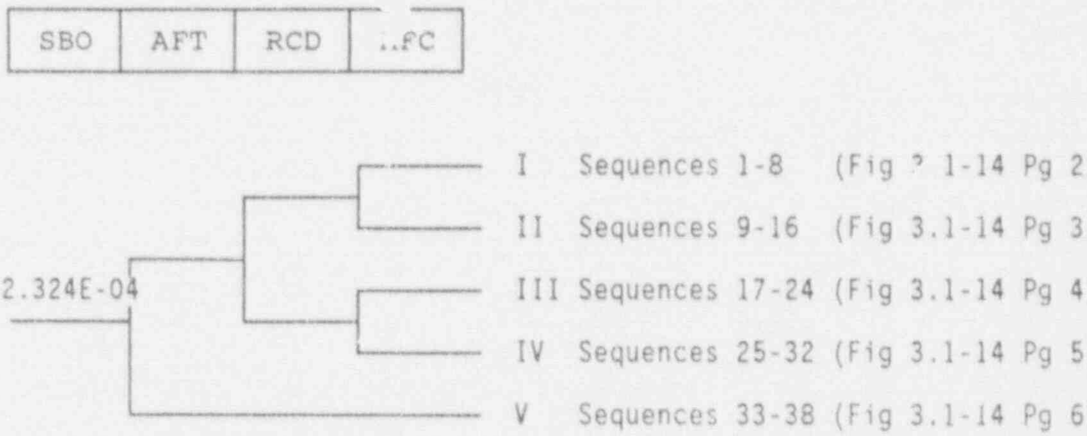
Recovery from station blackout ultimately requires recovery of some AC power source, either from the grid or the diesel generator(s). The timing of events is critical to this analysis, inasmuch as the DC batteries will give the operators indication of parameters such as pressurizer level (which is indicative of RCP seal leakage) and control of PORVs and the turbine-driven auxiliary feedwater pump. The batteries are expected to provide approximately 8 hours of power, with shedding of selected DC loads. After the batteries discharge, auxiliary feedwater flow may be manually maintained by local control and monitoring of local flow indicators (turbine-driven pump failure is nonetheless modeled because while the throttle valve to the turbine fails as is on loss of DC, the speed governing valve fails open, which may result in pump trip on overspeed). In addition to the above, the 8 hour period was chosen since it reflects the availability of the safety-grade backup air supply accumulators to the AFW turbine control valves and the steam generator PORVs (ignoring the possibility of charging the accumulators from nitrogen tanks). This time period also assures the availability of the condensate storage tank supply to the AFW pump and is less than the time for core uncovery given expected RCP seal leakage (non-LOCA).

Further, the challenge to the RCP seals is a function of RCS pressure and time. Once auxiliary feedwater is lost, some time is available to recover AC power before uncovering the core. From expected leakage through the RCP seals, this time is expected to be about 2 hours if RCS cooldown and depressurization was not successfully accomplished. If RCS cooldown and depressurization occurred, then core uncovery is not expected for about 4 hours after the loss of AFW flow. The event tree model reflects recovery of AC power in these time frames. If AC power is recovered before the core is uncovered, the possibility of seal LOCA is addressed. Mitigation of such consequential LOCAs requires high pressure injection in the short-term, and high pressure recirculation in the long-term.

Dependencies modeled in the event tree include: (1) if the turbine-driven auxiliary feedwater pump fails and power cannot be restored within two hours, early core damage will result; (2) if the turbine-driven auxiliary feedwater pump is operating when AC power is recovered (as discussed above), then core uncovery due to RCP seal LOCA is addressed, in which case successful mitigation of an RCP seal LOCA given AC power recovery requires 1 of 4 high pressure SI pumps injecting to the RCS; (3) if the turbine-driven auxiliary feedwater pump fails to continue running (as described above), and power is recovered within about 2 to 4 hours (depending upon the success of the RCS cooldown and depressurization), RCS pressure will be above the shutoff head of the high pressure SI pumps and mitigation of the seal LOCA will require injection from 2 of 2 charging pumps.

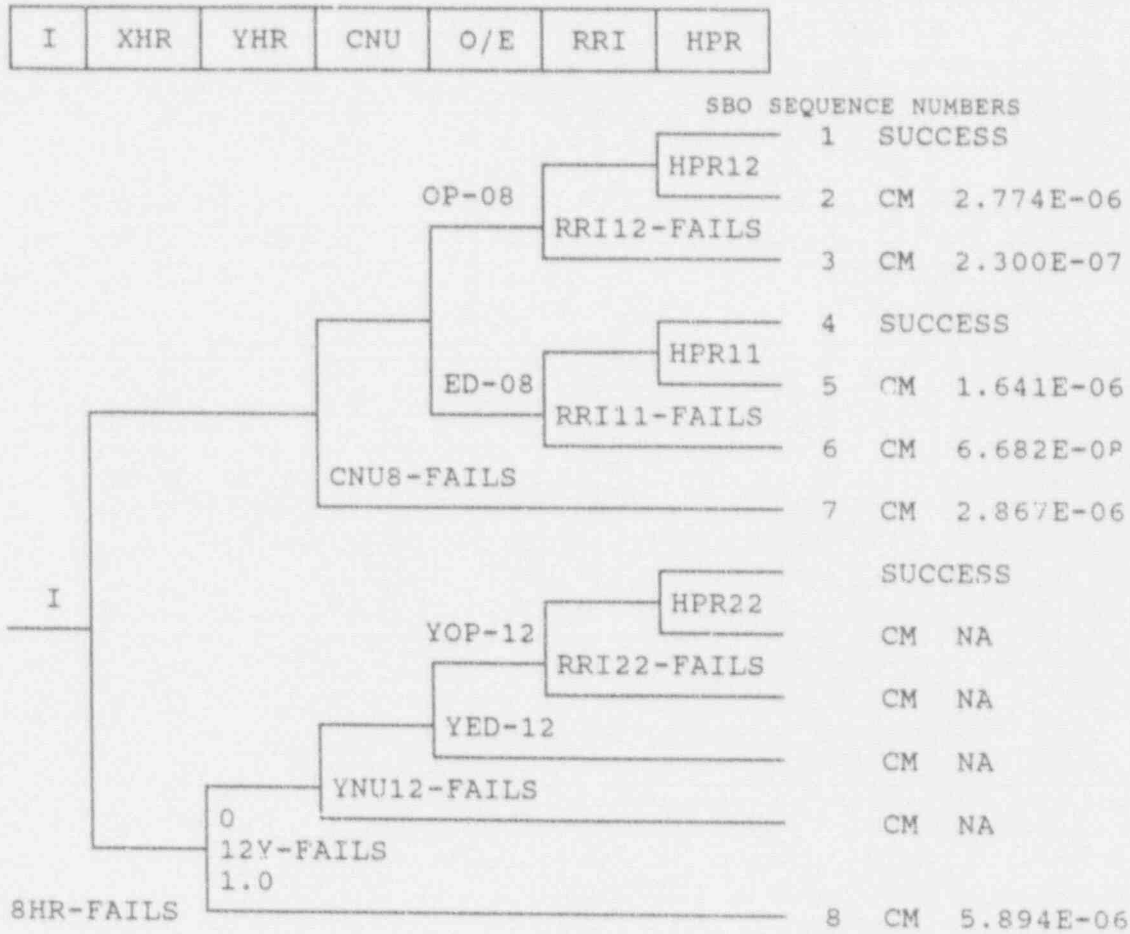
Figure 3.1-14

Station Blackout Event Tree
Page 1 of 8



Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 2 of 8



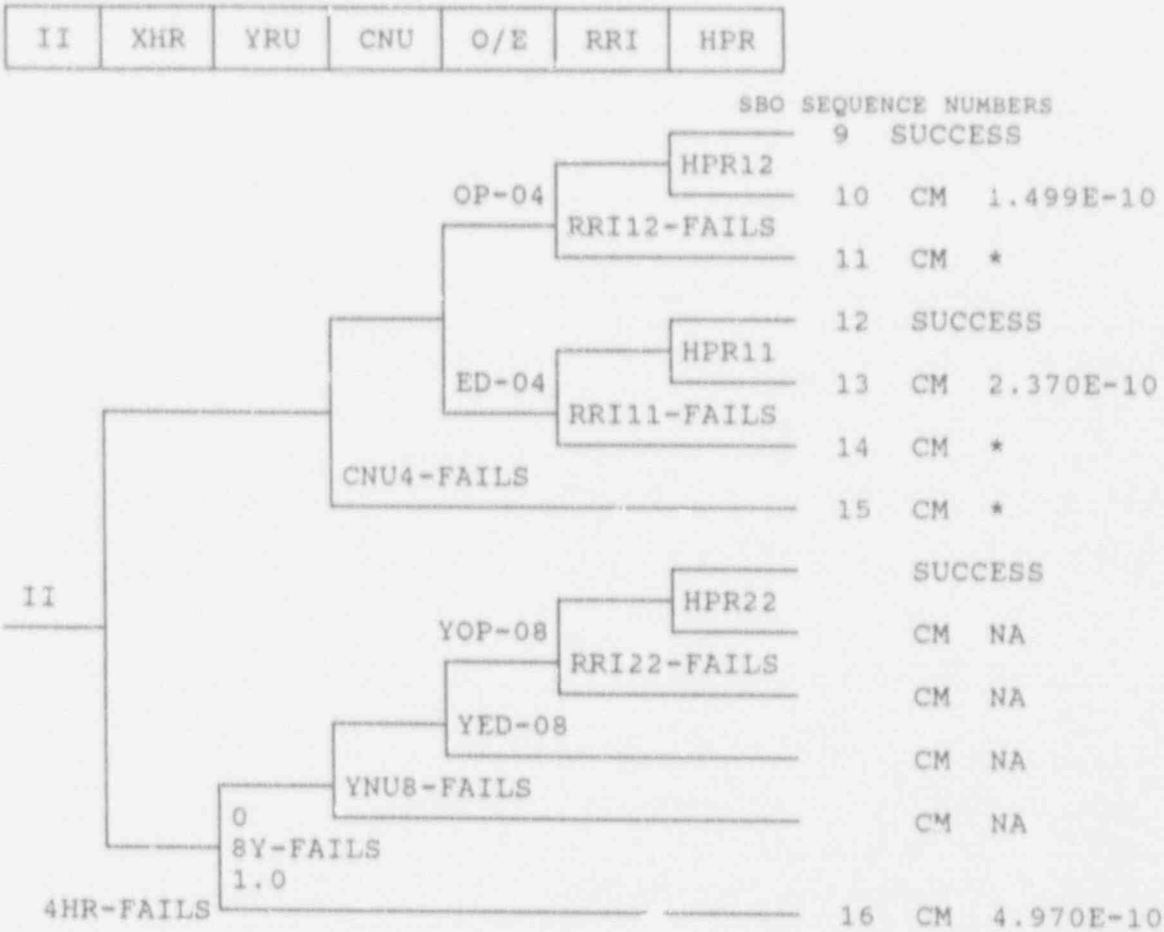
I = RCD SUCCESSFUL

AFC SUCCESSFUL

X = 8 HOURS; Y = 12 HOURS

Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 3 of 8



II = RCD SUCCESSFUL

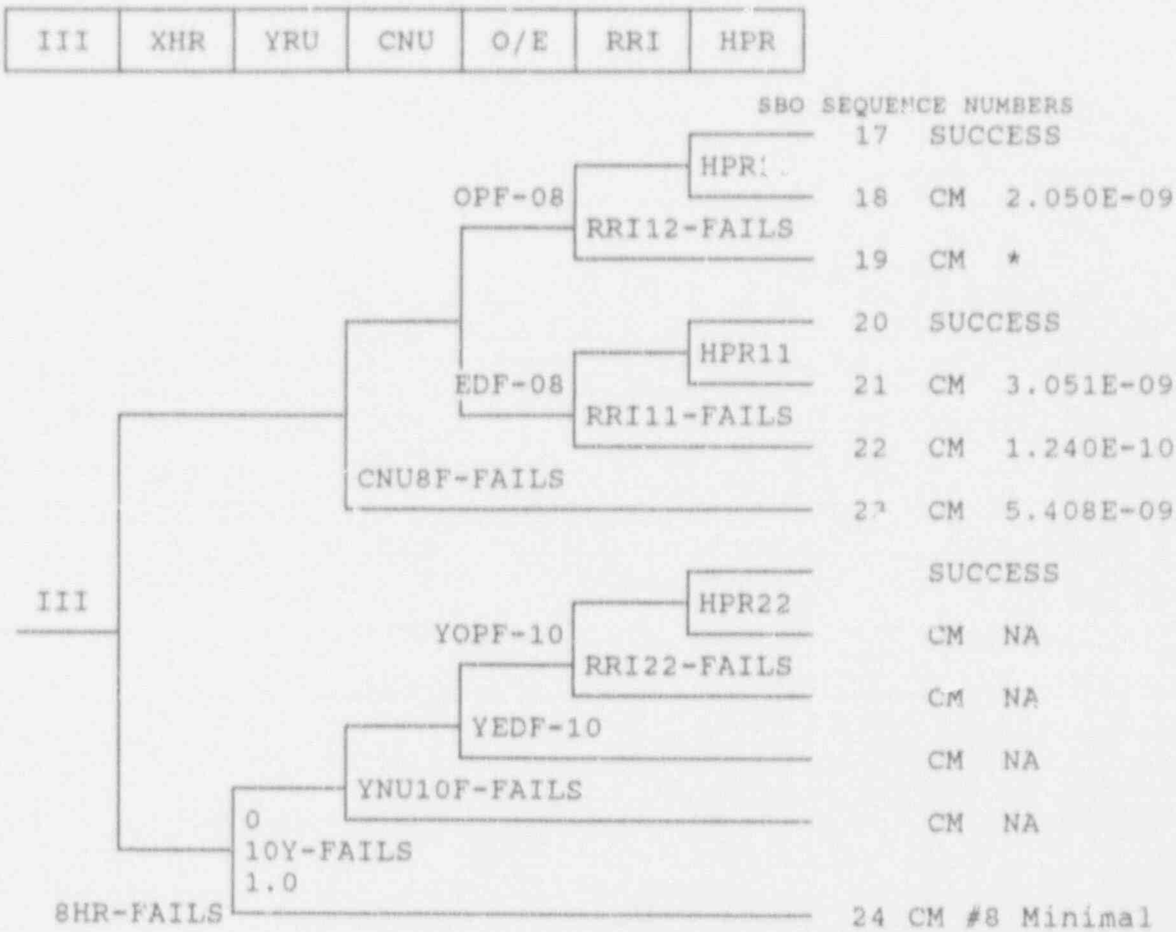
AFC FAILS

X = 4 HOURS; Y = 8 HOURS

* - Less than the 2.25E-11 cutoff

Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 4 of 8



III = RCD FAILS

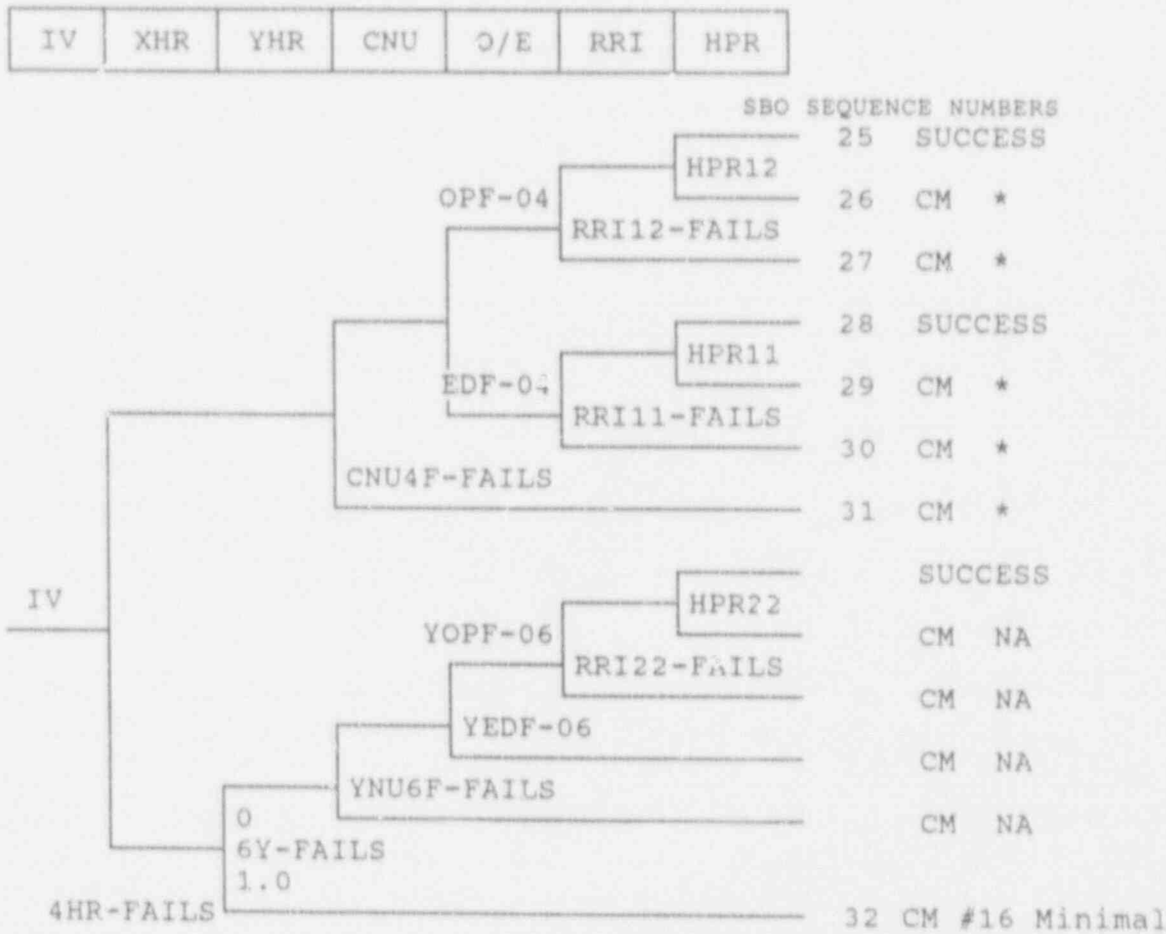
AFC SUCCESSFUL

X = 8 HOURS; Y = 10 HOURS

* - Less than the 2.25E-11 cutoff

Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 5 of 8



IV = RCD FAILS

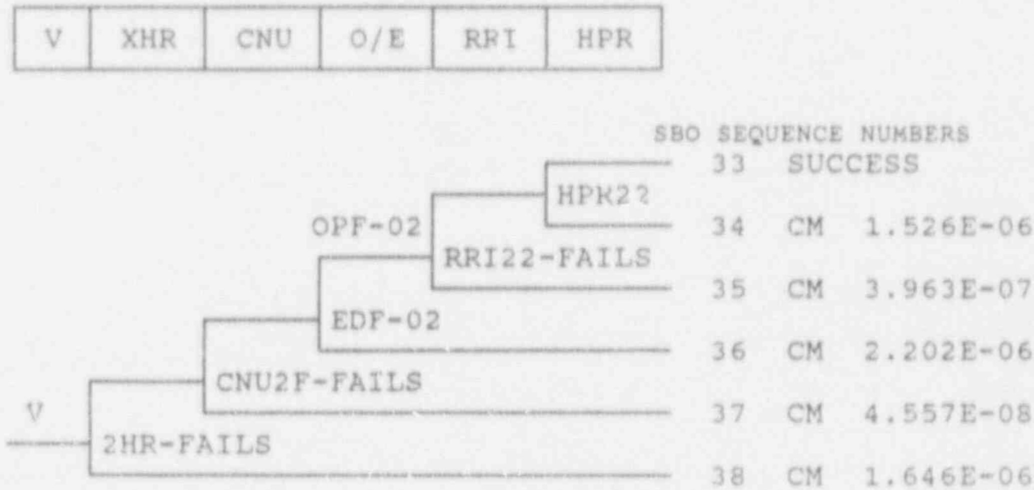
AFC FAILS

X = 4 HOURS; Y = 6 HOURS

* - Less than the 2.25E-11 cutoff

Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 6 of 8



V = RCD FAILS SINCE AFW FAILS.

Event designators are described on Pages 7 and 8 of Figure 3.1-14.

Figure 3.1-14
Page 7 of 8

AFC - Auxiliary Feedwater Continues
AFT - Auxiliary Feedwater TD AFW Pump

CNU - Core Not Uncovered
CNU2F - Core Uncovery Occurs within 2 Hours, RCD Fails
CNU4 - Core Uncovery Occurs within 4 Hours, RCD Successful
CNU4F - Core Uncovery Occurs within 4 Hours, RCD Fails
CNU8 - Core Uncovery Occurs within 8 hours, RCD Successful
CNU8F - Core Uncovery Occurs within 8 Hours, RCD Fails
YNU6F - Core Uncovery Occurs within 6 Hours, RCD Fails
YNU8 - Core Uncovery Occurs within 8 Hours, RCD Successful
YNU10F - Core Uncovery Occurs within 10 Hours, RCD Fails
YNU12 - Core Uncovery Occurs within 12 Hours, RCD Successful

HPR - High Pressure Recirculation
HPR11 - High Pressure Recirculation; 1/4 ECCS Pumps Required, 1/2 ESF Buses Available
HPR12 - High Pressure Recirculation; 1/4 ECCS Pumps Required, 2/2 ESF Buses Available
HPR22 - High Pressure Recirculation; 2/2 Charging Pumps Required, 2/2 ESF Buses Available

O/E - Offsite AC Recovery Fraction
ED-04 - Fraction of 4 Hour AC Power Recovery (4HR) due to Recovery of One Diesel Generator, RCD Successful
ED-08 - Fraction of 8 Hour AC Power Recovery (8HR) due to Recovery of One Diesel Generator, RCD Successful
EDF-02 - Fraction of 2 Hour AC Power Recovery (2HR) due to Recovery of One Diesel Generator, RCD Fails
EDF-04 - Fraction of 4 Hour AC Power Recovery (4HR) due to Recovery of One Diesel Generator, RCD Fails
EDF-08 - Fraction of 8 Hour AC Power Recovery (8HR) due to Recovery of One Diesel Generator, RCD Fails
OP-04 - Fraction of 4 Hour AC Power Recovery (4HR) due to Recovery of Offsite Power, RCD Successful
OP-08 - Fraction of 8 Hour AC Power Recovery (8HR) due to Recovery of Offsite Power, RCD Successful
OPF-02 - Fraction of 2 Hour AC Power Recovery (2HR) due to Recovery of Offsite Power, RCD Fails
OPF-04 - Fraction of 4 Hour AC Power Recovery (4HR) due to Recovery of Offsite Power, RCD Fails
OPF-08 - Fraction of 8 Hour AC Power Recovery (8HR) due to Recovery of Offsite Power, RCD Fails

O/E - Offsite AC Recovery Fraction (continued)

- YED-08 - Fraction of 8 Hour AC Power Recovery (8Y) due to Recovery of One Diesel Generator, RCD Successful
- YED-12 - Fraction of 12 Hour AC Power Recovery (12Y) due to Recovery of One Diesel Generator, RCD Successful
- YEDF-06 - Fraction of 6 Hour AC Power Recovery (6Y) due to Recovery of One Diesel Generator, RCD Fails
- YEDF-10 - Fraction of 10 Hour AC Power Recovery (10Y) due to Recovery of One Diesel Generator, RCD Fails
- YOP-08 - Fraction of 8 Hour AC Power Recovery (8Y) due to Recovery of Offsite Power, RCD Successful
- YOP-12 - Fraction of 12 Hour AC Power Recovery (12Y) due to Recovery of Offsite Power, RCD Successful
- YOPF-06 - Fraction of 6 Hour AC Power Recovery (6Y) due to Recovery of Offsite Power, RCD Fails
- YOPF-10 - Fraction of 10 Hour AC Power Recovery (10Y) due to Recovery of Offsite Power, RCD Fails

RCD - RCS Cooldown and Depressurization

RRI - Restore RCS Inventory
- RRI11 - Restoration of RCS Inventory; 1/4 ECCS Pumps Required, 1/2 ESF Buses Available
- RRI12 - Restoration of RCS Inventory; 1/4 ECCS Pumps Required, 2/2 ESF Buses Available
- RRI22 - Restoration of RCS Inventory; 2/2 Charging Pumps Required, 2/2 ESF Buses Available

SBO - Station Blackout

XHR - Restore AC Power within X Hours
- 2HR - Restore AC Power within 2 Hours
- 4HR - Restore AC Power within 4 Hours
- 8HR - Restore AC Power within 8 Hours

YHR - Restore AC Power within Y Hours
- 6Y - Restore AC Power within 6 Hours given Failure to Restore Power within 4 Hours
- 8Y - Restore AC Power within 8 Hours given Failure to Restore Power within 4 Hours
- 10Y - Restore AC Power within 10 Hours given Failure to Restore Power within 8 Hours
- 12Y - Restore AC Power within 12 Hours given Failure to Restore Power within 8 Hours

3.1.3.5 Containment Safeguards

All of the baseline and special event trees have been linked to a containment safeguards event tree model for calculation of plant damage states. The containment safeguards event tree is presented in Figure 3.1-15. For each core damage sequence, there may be up to 8 plant damage states, reflecting various operating states of the Containment Cooling System (CCS), Containment Spray System (CSS), and Containment Isolation (CIS). Dominant core damage sequences are further analyzed and identified for the Containment Performance Analysis by the functioning of each containment safeguard system.

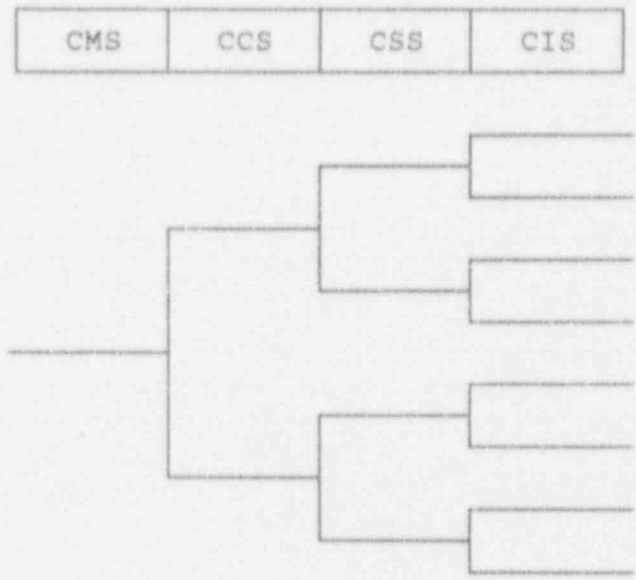
There are no dependencies explicitly modeled in this event tree. Support system dependencies are modeled in the fault tree models for each system.

3.1.4 Support System Modeling

The WCGS IPE uses fault tree linking to quantify the accident sequences, thus no support system event trees were created. The fault tree linking method requires the development of a system fault tree for each of the front-line systems and for each of the support systems modeled. Each front-line system fault tree calls in the appropriate support system fault tree or trees and the linking process quantifies the accident sequences without double-counting support systems.

Figure 3.1-15

Event Tree Defining Status of Containment Safeguards



	Status of CCS	Status of CSS	Status of CIS
1	A	A	A
2	A	A	F
3	A	F	A
4	A	F	F
5	F	A	A
6	F	A	F
7	F	F	A
8	F	F	F

<u>Event</u>	<u>Event Description</u>
CMS	Core Melt Sequence Occurs
CCS	Containment Cooling System
CSS	Containment Spray System
CIS	Containment Isolation
A	Available
F	Failed

3.1.5 Sequence Grouping and Back-End Interface

The event tree analysis identifies the results of events affecting reactor and turbine-generator availability and subsequent failures of safeguards systems. To preserve core integrity, certain functions must be achieved following an event: shutdown of the reactor and removal of reactor decay heat. Multiple systems and methods are available to carry out these functions. These systems are explicitly analyzed by the event tree. When different systems fail, it is possible that core damage will occur. For the PRA, once conditions that might yield core damage were identified by an event tree sequence, core damage was postulated. Neither recovery of systems nor use of non-traditional emergency safeguards methods that hypothetically could be attempted by the operators are addressed beyond this point.

The following different modes of core damage are modeled:

Small LOCA (S) - This category is composed of core damage following a small break of the primary system piping, where a direct release path to the containment atmosphere exists.

Transient (T) - This category is characterized by the release of primary inventory through the pressurizer PORVs or safety valves to the pressurizer relief tank.

Large LOCA (A) - This release category is characterized by a rapid depressurization of the RCS, core uncovery, and core damage, such as could occur following a large pipe or reactor vessel rupture.

V Sequence (V) - This release follows a large interfacing systems LOCA outside containment.

SGTR Release (V2) - This release category is also a bypass of containment with primary coolant being discharged from steam generator PORV or safety valves.

The timing of the core damage is also addressed, reflecting the effects of decay of short-lived radioisotopes inside the core. The two times modeled are:

Early Damage (E) - Occurs when short term core cooling is not available.

Late Damage (L) - Occurs when short term core cooling is available but the operator fails to establish long term cooling.

The core damage categories developed with the above defined parameters are summarized in Table 3.1-3.

TABLE 3.1-3

CORE DAMAGE CATEGORIES

Symbol	Core Damage Category Identification
AE	Large LOCA - Early Core Damage
AL	Large LOCA - Late Core Damage
SE	Small LOCA - Early Core Damage
SL	Small LOCA - Late Core Damage
TE	Transient - Early Core Damage
VL	Interfacing Systems LOCA - Late Core Damage (Containment Bypass - Failure of Recirculation)
V2E	Steam Generator Tube Rupture - Early Core Damage
V2L	Steam Generator Tube Rupture - Late Core Damage

3.2 Systems Analysis

To develop an understanding of the contribution of system performance to accident sequences and to quantify the event trees, an analysis of all key plant systems (from a risk perspective) was performed. This activity included a plant familiarization activity which is documented in system notebooks, a search for dependencies between plant systems, and fault tree analysis performance for each key system. Technical guidelines were prepared for each of these activities to assure a consistent, thorough approach is employed by all analysts throughout each stage of the work. These guidelines also serve as references that document the methods used in detail. This section provides an overview of the methods used in the plant systems analysis activity.

3.2.1 System Descriptions

To ensure the PRA accurately represents how the plant's systems contribute to the overall risk profile, a thorough understanding of key frontline and support systems is essential. Prior to the development of the fault tree logic models, a comprehensive collection, evaluation, and documentation of information was performed for each system. This information was consolidated into a single reference notebook for each system. A typical outline used for the system notebooks is given in Table 3.2-1. The first six sections of the system notebooks contain the essential plant design and operational information necessary to develop the fault trees. Included in these sections are the important dependencies shown in the matrices presented in Section 3.2.3, instrumentation and control requirements, and the results of a review of equipment maintenance and surveillance practices. The plant walkdown, described in Section 2.4 above, was used to verify that the plant configuration modeled in the PRA is consistent with the manner in which the individual systems are installed and operated. The notebooks also contain logic models and the assumptions used to construct the fault tree models, the results of model quantification, and the insights related to that system that were developed during the course of the PRA.

Table 3.2-2 lists the systems modeled in the WCGS PRA. The remainder of this section provides a brief description of these systems and their important design features from a risk analysis perspective. The symbols used in the simplified piping and instrumentation diagrams (P&ID's) for these systems are shown in Figure 3.2-1.

Note: System P&IDs displayed in Section 3.2 are figures taken from the appropriate PRA system notebooks. IPE Section 3.2 figure numbers do not correspond to the PRA system notebook figure numbers.

TABLE 3.2-1
OUTLINE FOR SYSTEM NOTEBOOKS

1.0 SYSTEM FUNCTION

2.0 SYSTEM DESCRIPTION

2.1 Support Systems
2.2 Instrumentation and Controls
2.3 Technical Specification Limitations
2.4 Test and Maintenance
2.5 Component Location

3.0 SYSTEM OPERATION

4.0 PERFORMANCE DURING ACCIDENT CONDITIONS

4.1 Success Criteria
4.2 Initiator Impact on the System

5.0 OPERATING EXPERIENCE

6.0 INITIATING EVENT REVIEW

7.0 SYSTEM LOGIC MODELS

7.1 Assumptions and Boundary Conditions
7.2 Fault Tree Logic Models

8.0 QUANTIFICATION AND RESULTS

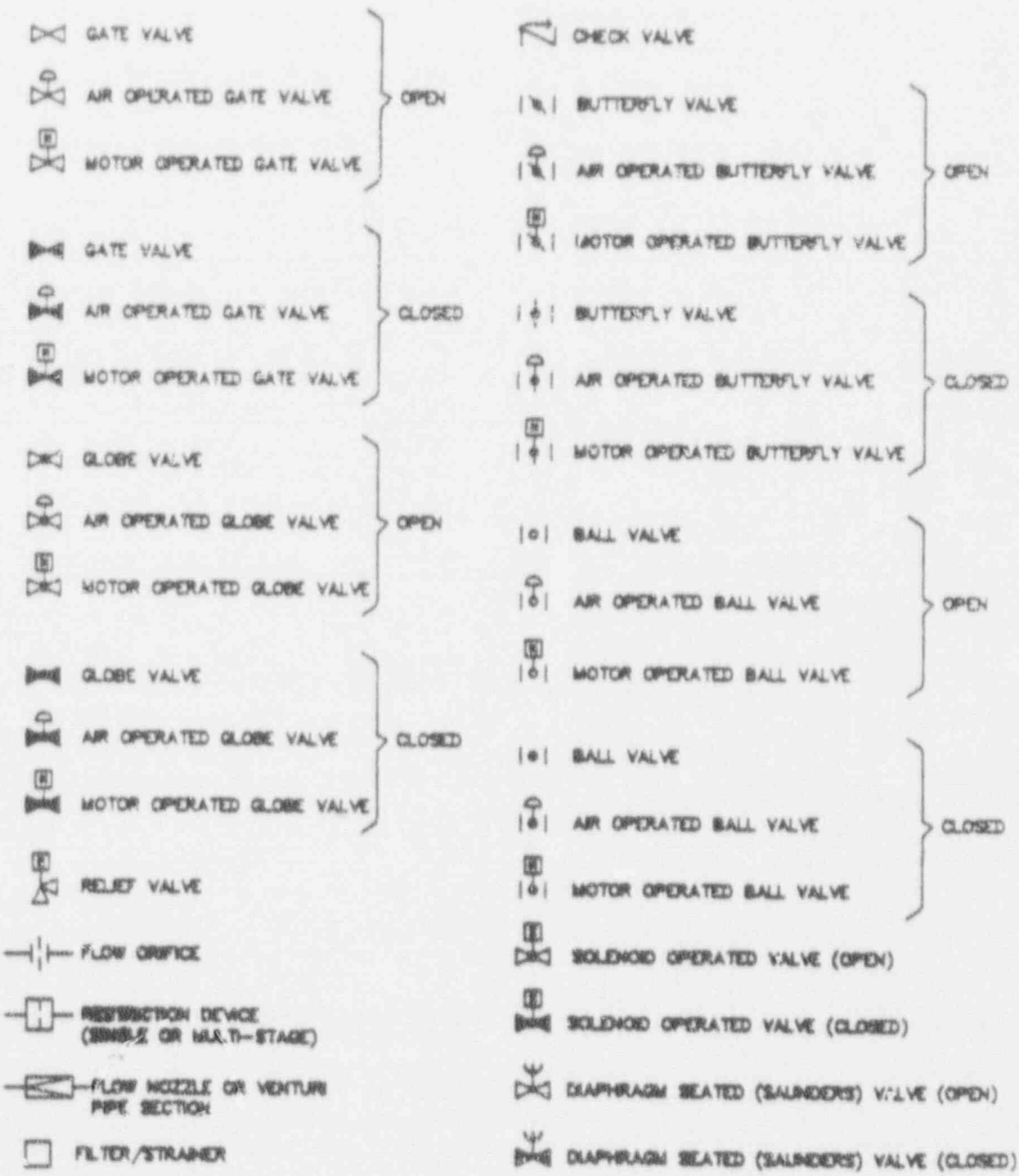
9.0 SYSTEM INSIGHTS

10.0 REFERENCES

TABLE 3.2-2
SYSTEMS MODELED IN THE WCGS IPE

Section	System
3.2.1.1	Auxiliary Feedwater System
3.2.1.2	Chemical and Volume Control System (Portions)
3.2.1.3	Component Cooling Water System
3.2.1.4	Containment Cooling System
3.2.1.5	Containment Isolation
3.2.1.6	Containment Spray System
3.2.1.7	Electrical Power System AC Power System DC Power System Class 1E Electrical Equipment HVAC
3.2.1.8	Emergency Core Cooling System Accumulator Safety Injection Low Pressure Safety Injection/Recirculation Intermediate Pressure Coolant Injection/Recirculation High Pressure Coolant Injection/Recirculation
3.2.1.9	Essential Service Water System
3.2.1.10	Instrument and Control Systems Reactor Trip System Engineered Safety Features Actuation System ATWS Mitigation System Actuating Circuitry
3.2.1.11	Main Feedwater and Condensate Systems
3.2.1.12	Main Steam System
3.2.1.13	Pressurizer Relief System

Figure 3.2-1
Simplified P&ID Component Symbols



3.2.1.1 Auxiliary Feedwater System

The Auxiliary Feedwater System (AFW) provides a safety-grade water supply to the steam generators for removal of reactor decay heat following transient and accident scenarios in which main feedwater is isolated or lost. The system ensures adequate makeup to the steam generators to prevent the reactor coolant system (RCS) pressure from increasing and causing release of coolant through the pressurizer safety or relief valves. For the PRA analysis, the AFW system is the primary source of water to the steam generators for secondary system heat removal. Figure 3.2-2 shows the simplified P&ID of the system.

The AFW consists of two 100 percent capacity motor-driven pumps, one 200 percent capacity steam turbine-driven pump, and associated piping, valves, and instruments. The pumps are normally aligned to take suction from the nonsafety-grade condensate storage tank (CST) with backup suction supply from the safety-grade Essential Service Water System (ESW), which is automatically aligned on low pump suction pressure. The CST has sufficient reserve capacity (minimum of 200,000 gallons) to hold the plant at hot standby for 4 hours and then cooldown the RCS at an average rate of 50 F per hour to a point at which the Residual Heat Removal System (RHR) can be aligned.

The motor-driven AFW pumps each discharge to two steam generators, and the turbine-driven pump discharges to all four steam generators. The motor-driven AFW pumps are supported by the AC and DC power systems and have room coolers supported by essential service water flow. Class 1E motor-operated control valves regulate flow from the motor driven AFW pumps to each associated steam generator. Class 1E air-operated control valves regulate flow from the turbine driven AFW pump to each associated steam generator. The turbine-driven pump receives motive steam from two of the four main steamlines, ensuring a steam supply following a design basis rupture and blowdown of any one steamline.

The success criteria for the AFW system depends upon the initiating event and the availability of steam generators and support systems. These criteria, analyzed by different fault tree models, are described in Table 3.1-1.

Figure 3.2-2
Auxiliary Feedwater System
Simplified P&ID
(Page 1 of 2)

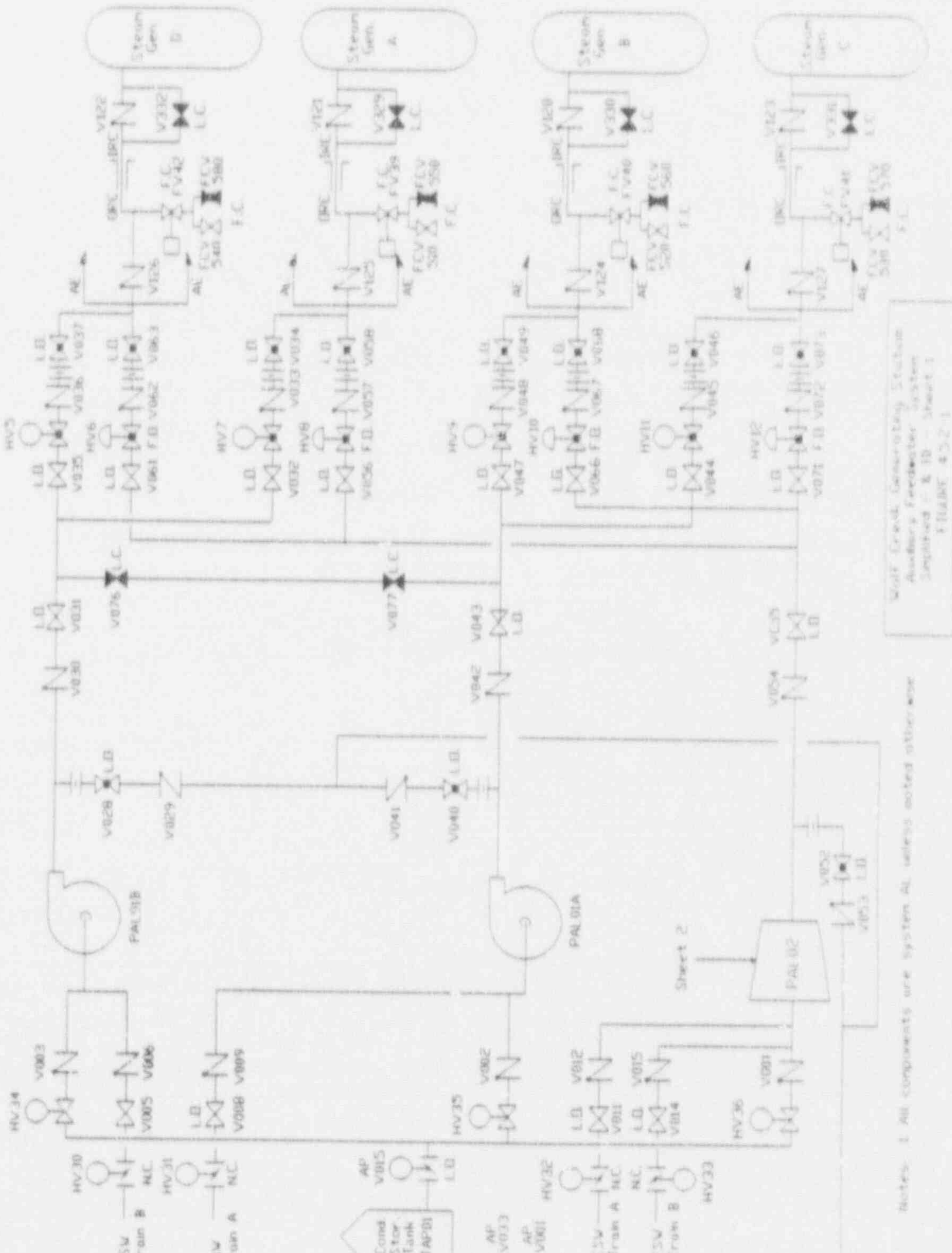
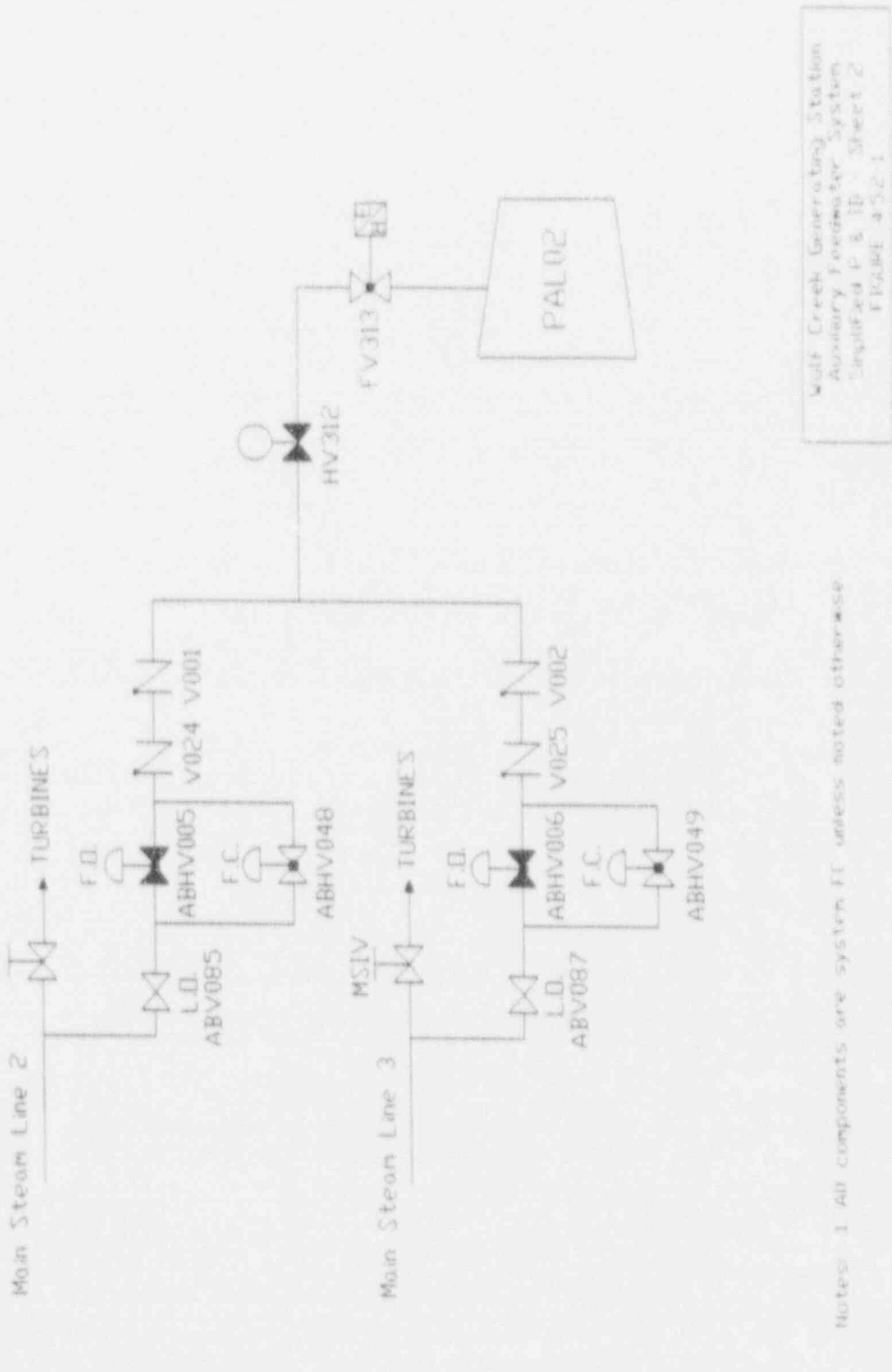


Figure 3.2-2
Auxiliary Feedwater System
Simplified P&ID
(Page 2 of 2)



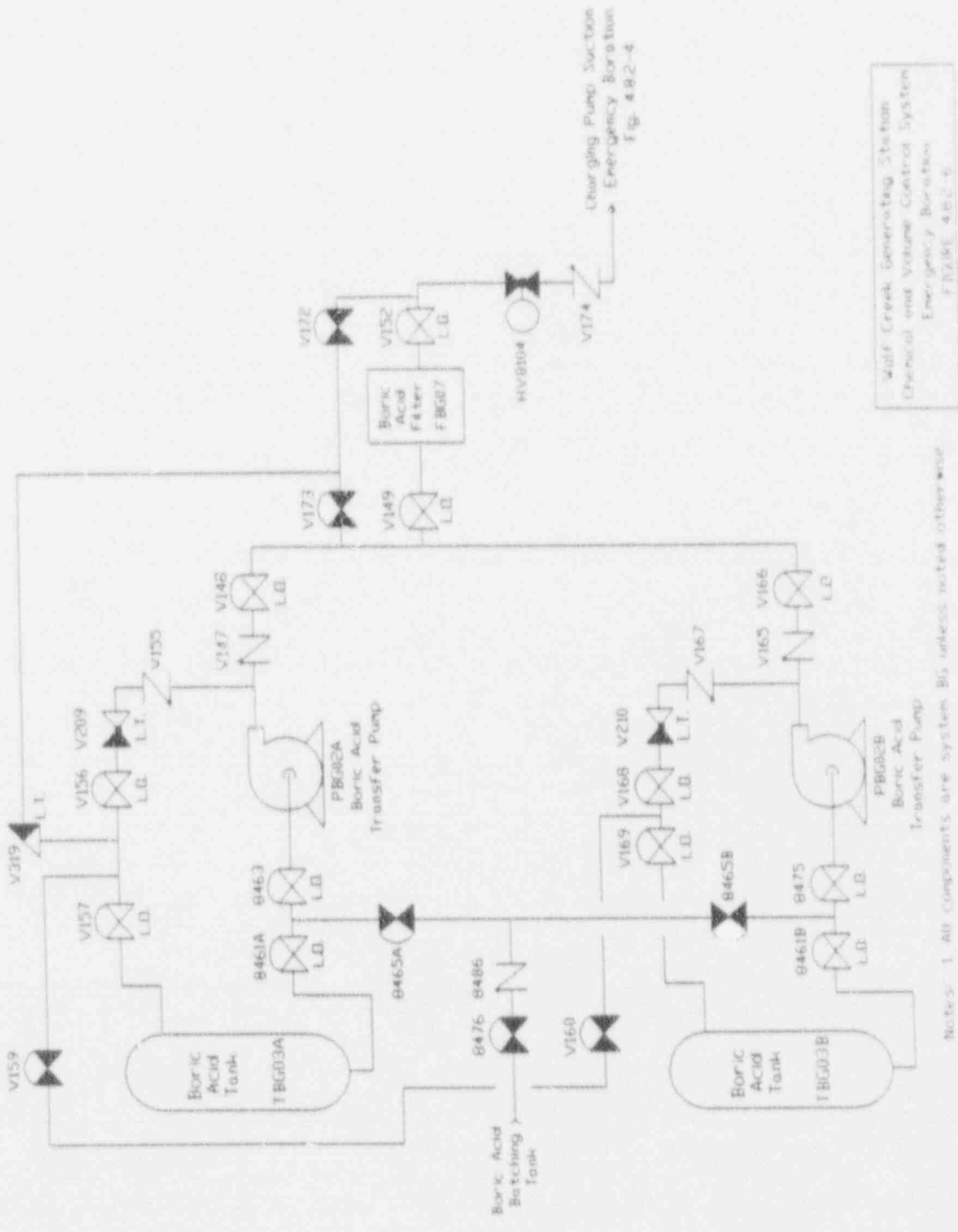
3.2.1.2 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS): (1) maintains required volumetric and chemical composition of the reactor coolant system; (2) provides Reactor Coolant Pump (RCP) seal injection; and (3) provides the standby Emergency Core Cooling System (ECCS) high pressure safety injection function using the charging pumps discharging to the RCS through the Boron Injection Tank. The ECCS function of the system is described in Section 3.2.1.8 below. Figure 3.2-13 shows the simplified P&ID for the charging system aligned in its high pressure safety injection mode and Figure 3.2-3 shows the simplified P&ID for the emergency boration function of the boric acid addition system of the CVCS.

The CVCS provides volume control by a constant feed and bleed process. Makeup to the RCS is provided via makeup injection and via inleakage through the RCP seals. Letdown to the CVCS is through the normal and/or excess letdown lines and the RCP seal water return line. Regenerative and component cooling water heat exchangers cool letdown flow to acceptable levels. The CVCS removes ionic corrosion products and certain fission products, and provides boric acid removal and addition for reactivity control.

The CVCS has been analyzed for its ability to provide emergency boration following an ATWS. The centrifugal charging pumps can take suction from either the Refueling Water Storage Tank (RWST) or from the Boric Acid Tanks (BAT) via the boric acid transfer pumps. The transfer pumps receive power from a Class 1E bus, but are automatically shed on an SI sequence. If previously shed, the pumps can be manually reloaded onto the bus. No credit for the positive displacement charging pump has been taken in this analysis. Successful emergency boration from the BAT requires the successful operation of one of the transfer pumps delivering flow from its associated BAT to one of the centrifugal charging pumps which deliver flow to 3 of 4 RCS cold legs.

Figure 3.2-3
Chemical and Volume Control System - Emergency Boration
Simplified P&ID



3.2.1.3 Component Cooling Water System

The Component Cooling Water System (CCW) provides cooling water to selected nuclear auxiliary components during normal and accident plant operations. One of the two safety-related CCW trains is sufficient to bring the plant to a safe shutdown after a transient or accident. The system is a closed loop system which acts as an intermediate barrier between the Essential Service Water System (ESW) and potentially radioactive systems. The CCW system simplified P&ID is shown in Figure 3.2-4.

Each of two redundant CCW trains includes two 100 percent capacity pumps, a heat exchanger that utilizes the ESW as a heat sink, a surge tank, and associated piping, valves, and instrumentation. Essential equipment that is cooled by the CCW includes the RHR pumps and heat exchangers, the charging pumps, and the intermediate pressure safety injection pumps. The system also provides coolant flow to various nonsafety-related loads, including the RCP thermal barriers and motor bearings, the letdown and excess letdown heat exchangers, and the spent fuel pool cooling heat exchanger. One CCW pump is run during normal operation, and its standby pump will automatically start on low flow or low pressure if the running pump stops. One pump per train will be automatically started following a Safety Injection Signal (SIS) or upon a loss of offsite power. A pump will also be automatically started when a charging pump on the same train is started.

Each CCW surge tank is connected to the suction side of its associated pumps, accommodating any thermal and leakage changes in coolant volume and assuring adequate NPSH to the CCW pumps. Monitoring and alarm of surge tank levels allows detection of any leakage into and out of the system. Automatic makeup to the system is supplied from the demineralized water system. Makeup is also available from the ESW.

Figure 3.2-4
Component Cooling Water System
Simplified P&ID
(Page 1 of 3)

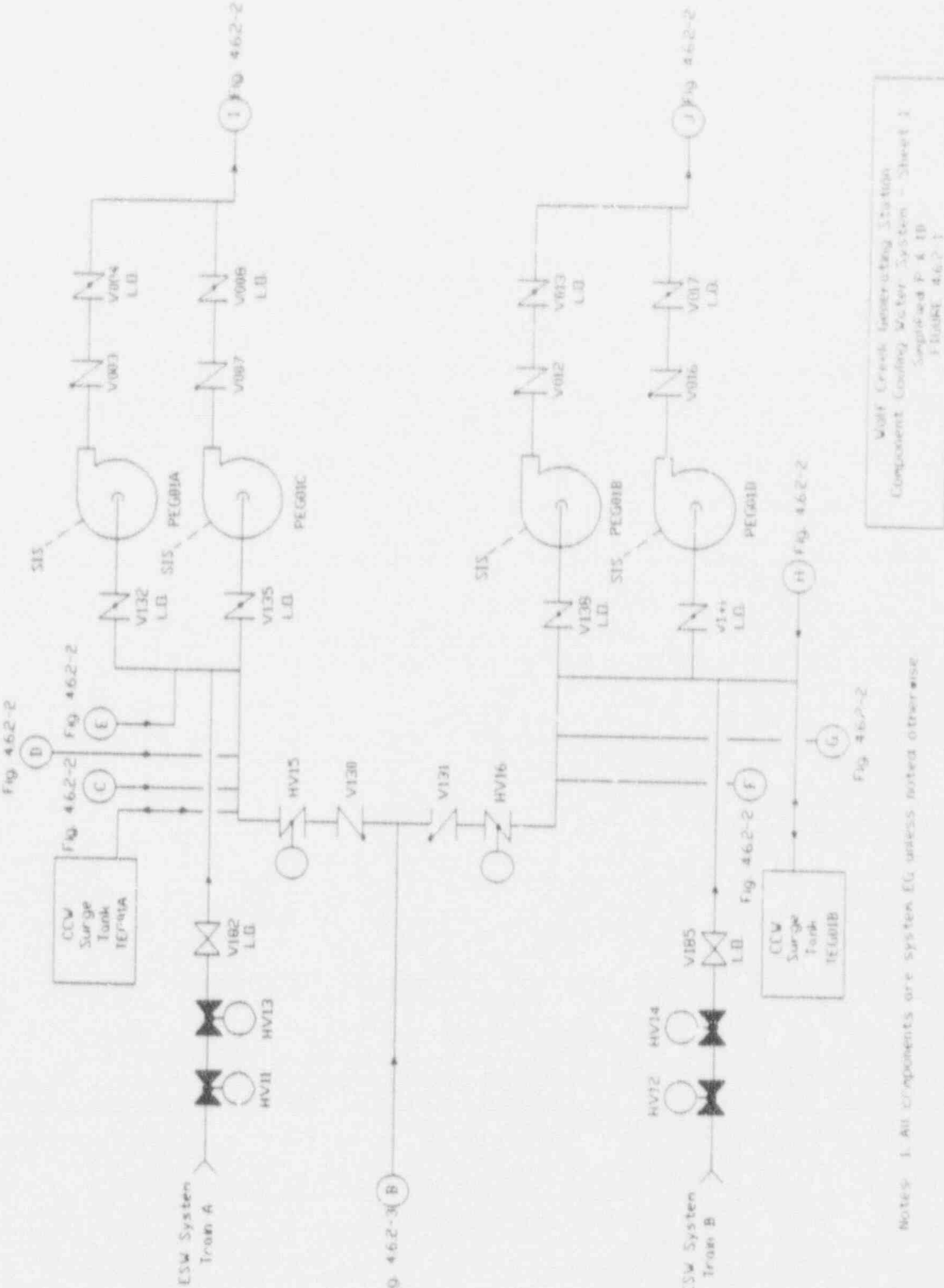


Figure 3.2-4
Component Cooling Water System
Simplified P&ID
(Page 2 of 3)

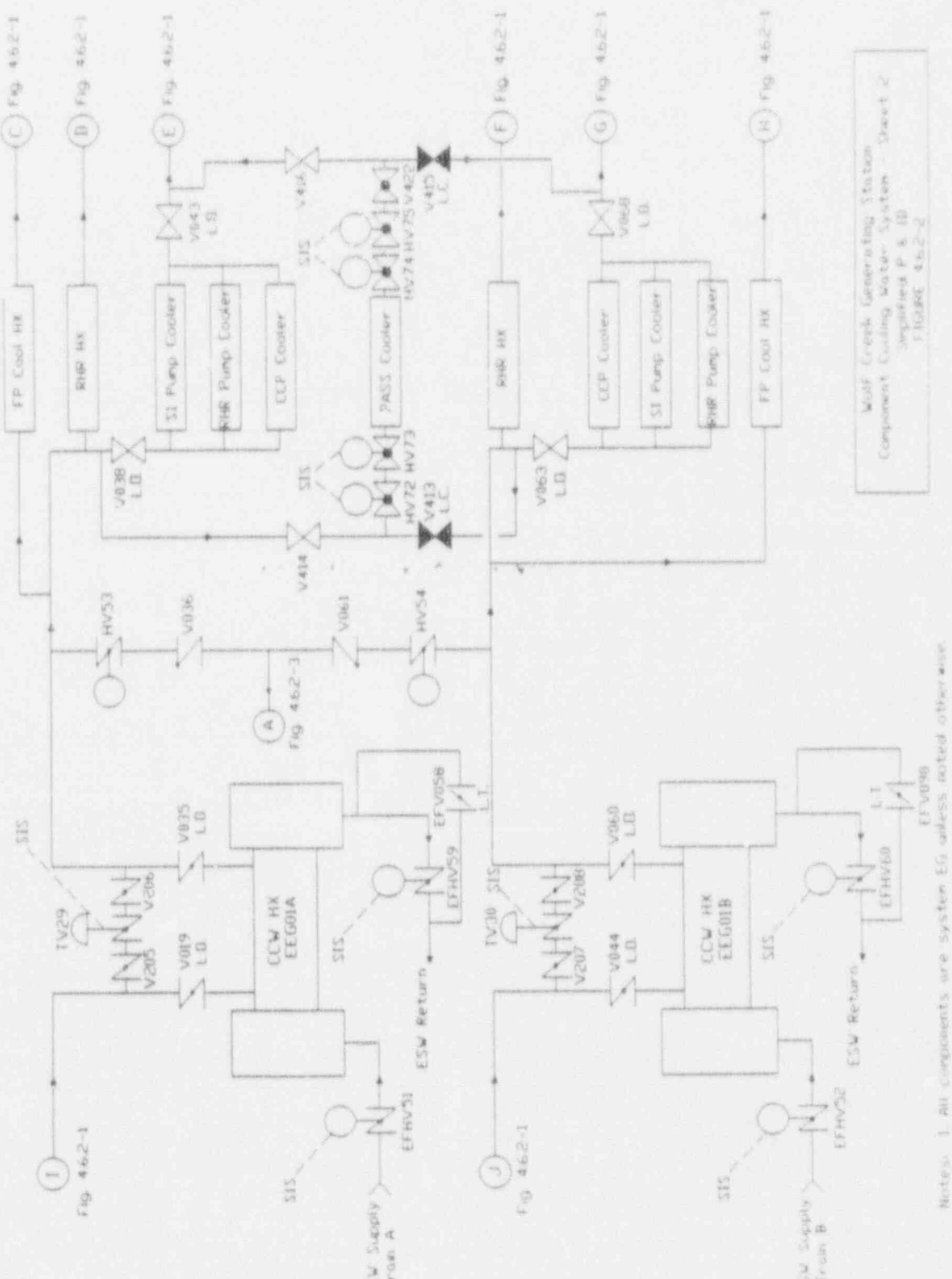
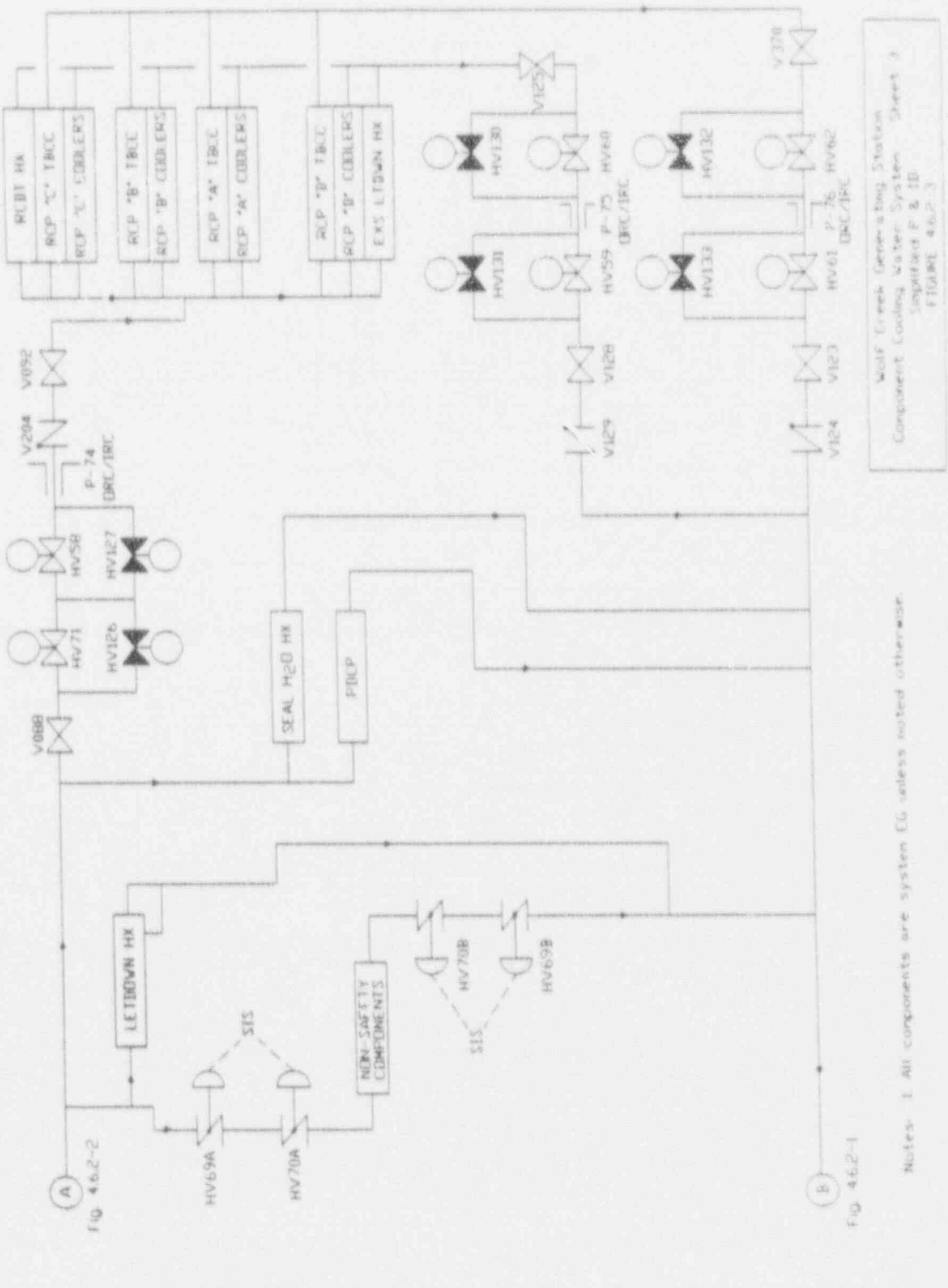


Figure 3.2-4
Component Cooling Water System
Simplified P&ID
(Page 3 of 3)



3.2.1.4 Containment Cooling System

The safety-related containment fan coolers provide a mechanism for removing heat from the containment volume following a LOCA or steamline break inside containment. The reduction of temperature and pressure minimizes the potential leakage of airborne and gaseous radioactivity to the environment. The coolers also function during normal plant operation to maintain a suitable atmosphere for equipment operation. A simplified P&ID for the containment cooling system is shown in Figure 3.2-5.

The coolers recirculate containment air across water-cooled coils. Water supply to the coils is provided by the Essential Service Water System. Two coolers are loaded in parallel on each of the two trains of ESW. All four fans/coolers are normally operated at high speed with low ESW flow to the coils. The fans are automatically switched to low speed and ESW flow is automatically increased to high flow upon SI actuation. As the containment post-accident temperature increases (to approximately 160 F), fusible link plates will separate the cooler from its normal nonsafety-related ductwork flowpath. The success criteria for successful operation of the containment cooling system requires that 2 of 4 fans/coolers switch to slow speed and operate for 24 hours.

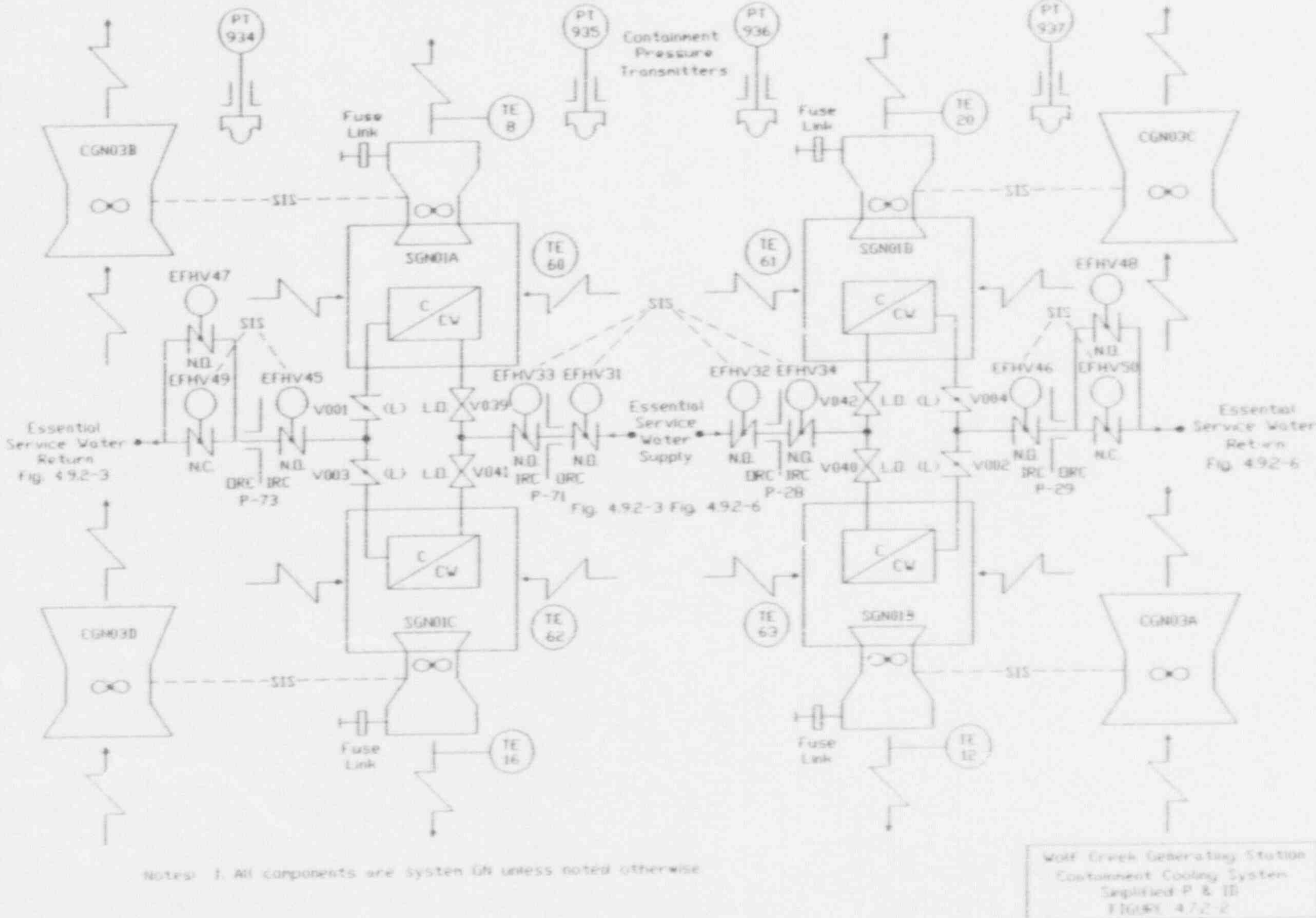


Figure 3.2-5
Containment Cooling System
Simplified P&ID

3.2.1.5 Containment Isolation

Containment isolation refers to the sealing and closure of containment penetrations to limit the potential for releases of radioactive materials following an accident. The scope of containment isolation is the estimation of the probability that isolation cannot be established prior to core damage which leads to a large fission product release. This scope is limited to severe accident sequences which do not bypass the containment and those which lead to containment structural failures.

Three types of containment penetrations are used: (1) electrical penetrations for power, instrumentation, and control cables; (2) mechanical penetrations for passing personnel, equipment, and materials into containment; and (3) fluid system penetrations for all water, steam, nitrogen, and air lines supporting systems inside the containment such as the RCS. Fluid system penetrations are further identified by the type of system supported: (a) fluid systems that are not required to support normal plant operations are administratively controlled (e.g., the refueling cavity drain line); (b) fluid systems that support normal plant operation but are not required under accident conditions (e.g., air supply lines); and (c) fluid systems required following transients and design basis accidents (e.g., ECCS lines).

Electrical penetrations are passive devices providing mechanical seals to prevent leakage. The electrical penetrations have not been modeled using fault trees because they are permanently sealed to prevent the passage of fluids except under conditions of gross failure. The periodic testing of these penetrations is assumed to be adequate to ensure integrity at the time of an accident.

Mechanical penetrations, such as the personnel and equipment hatches and the fuel transfer tube, are generally used only during plant shutdowns. These penetrations provide a mechanical barrier as well as sealant materials on at least one side of the penetration. Mechanical penetrations are normally closed prior to plant operation, and, like the electrical penetrations, are tested to administratively ensure that isolation integrity is established. Failures of the mechanical penetrations have been modeled in the Containment Isolation fault tree.

Each fluid piping system which penetrates the containment is provided with containment isolation features which serve to minimize the release of fission products following a LOCA or fuel handling accident. Isolation of these lines is achieved by closure of two valves in series, with automatic closures initiated by an isolation signal which is tied to the plant safety systems. Double valving ensures isolation of each line given a single failure of one valve. All manual valves which serve as isolation valves are locked or sealed closed, and are subject to Technical Specification surveillance requirements. Provisions are made for the passage of emergency system fluids through the containment boundary. Remote manual operation is available for these valves since automatic closure is not provided.

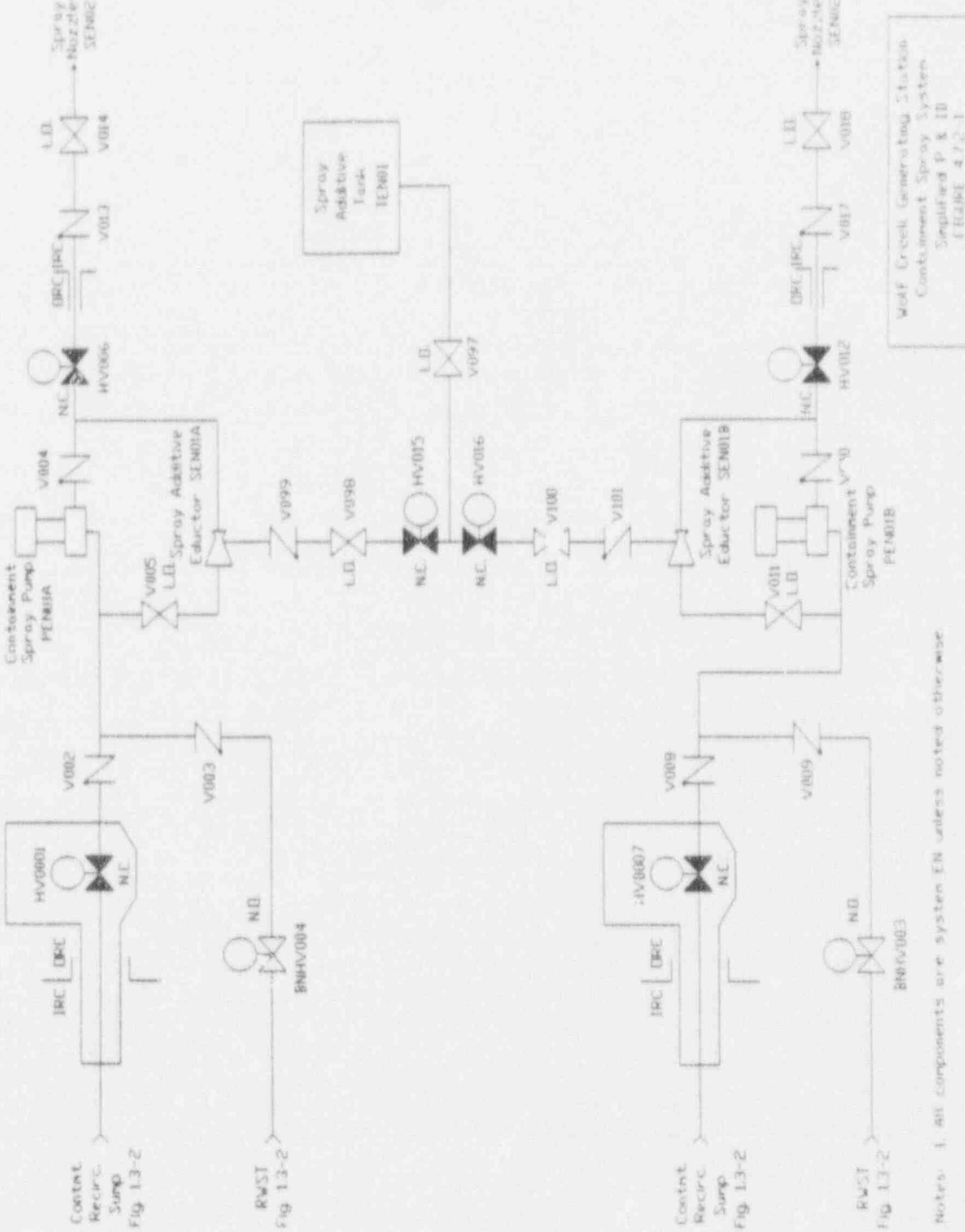
Successful functioning of the electrical, mechanical, and administratively controlled fluid penetrations requires both that there is no pre-existing leakage through each penetration and that the mechanical and sealant materials of each penetration are maintained throughout the accident sequence. Degradation under severe accident conditions has been evaluated within the containment performance analysis.

3.2.1.6 Containment Spray System

The Containment Spray System (CSS) is required in the event of a loss of coolant accident (LOCA) or a main steam line break (MSLB) accident inside containment: (1) to reduce containment temperature and pressure; and (2) to limit potential offsite radiation levels from design basis accidents. The CSS provides two mechanisms to achieve these objectives. First, the CSS delivers cold spray chemical solution water to the containment volume to reduce containment temperature and pressure, thereby diminishing the driving force for leakage of fission products to the environment. Second, the addition of sodium hydroxide to the spray water enhances the removal of airborne fission products from the containment atmosphere, thereby reducing the inventory of fission products available for leakage to the environment. No credit is taken for this mechanism in the WCGS PRA.

A simplified P&ID for the CSS is presented in Figure 3.2-6. The system consists of two separate 100 percent capacity trains, each independently capable of meeting the design bases. Each train consists of a pump, spray header and nozzles, spray additive eductor, containment recirculation sump screens, valves, and necessary piping, instrumentation, and controls. The CSS operates in an injection phase, drawing on the Refueling Water Storage Tank (RWST). Manual or automatic high-3 containment pressure actuation of the system starts the pumps, opens the header isolation valves and the eductor suction valves to the spray additive tank. Manual switchover to the containment recirculation sumps is required after an RWST low-low-2 level alarm.

Figure 3.2-6
Containment Spray System
Simplified P&ID



3.2.1.7 Electrical Power System

The electrical power systems at WCGS are provided as a reliable source of electric power to support the normal and emergency operation of all plant systems. The systems include an AC Power System, which derives power from the station switchyard and has emergency diesel generators for backup, and a DC Power System, which derives power from the AC syste . In addition, the electrical power systems are supported by dedicated air conditioning and ventilation units. Al of these systems are described in the following su sections.

3.2.1.7.1 AC Power System

The AC Power System derives it power from the station switchyard, which is the main outlet for station power and the source of offsite power to WCGS. There are three 345kV lines connecting th witchyard to the area transmission system: (1) the LaCygne line; (2) se Hill line; and (3) the Benton line. Each line is adequa provide the total Engineered Safety Features load requi ed for sa e shutdown of the plant. A one-line diagram of offsite power supply to WCGS is presented in Figure 3.2-7, Page 1.

During normal plant operation, the AC power system is supplied from the unit auxiliary transformer (XMA02). When the main generator is not operating, station power is provided by the start-up transformer (XMR01). Station service distribution buses PA01 and PA02 supply power to the RCPs, to site load 13.8kV buses SL-? through 4, to the makeup water screenhouse, to nonvital 4.16kv buses PB03 and PB04, and to the nonvital 480V system. The 480V system consists of load center unit substations and motor control centers (MCC), and provides nonvital 120/208V AC power to space heaters and various miscellaneous loads. The 120V AC system in turn serves various instrument loads.

The safeguards power system portion of the onsite AC power system supplies power to all ESF loads. The safeguards power system consists of two redundant, physically and electrically separated load groups. The system supplies a few nonvital loads which are considered important for plant operation. Except for the nonvital 120V instrument power, these nonvital loads are shed from the buses on an SIS. Each train includes a ESF transformer, an Emergency Diesel Generator, 4.16kV distrib tion, 480V distribution, and load shedding and sequencing. Simplified one-line diagrams of the two trains of safeguards power are presented in Figure 3.2-7 (Pages 2 and 3). Each bus can be powered f om one of three sources: (1) the ESF transformer for the bus; (2) the Emergency Diese Generator for that bus; or (3) the ESF transformer for the other ESF train, by manual transfer. The preferred power source for ESF transformer XNB01 is the switchyard No. 7 transformer, and that for ESF transformer XNB02 is the startup transformer.

Each Emergency Diesel Generator (DG) is started on underfrequency on its associated bus, an SIS, or manually. Each DG is used exclusively for its own bus, with no provisions for parallel operation. Cooling in support of DG operation i rovided by the Diesel Building HVAC, which also serves as a source of ombustion air for the diesels. Each diesel also receive cooling water from the Essential Service Water System. After coming up to speed and voltage, the DG is loaded onto its bus. Load shedding and sequencing of loads onto the ESF buses is provided by two different sequencers on each bus: a shutdown sequencer and a LOCA sequencer. The sequencing of loads ensures adequate voltage and frequency of the bus as each major load is added, and protects the DG from instabilities. Small loads (e.g., motor operated valves, battery chargers, and certain HVAC components) are not shed and will be energized when bus voltage is present.

The 480V Safeguards Distribution System receives power from the 4.16kV ESF buses, and distributes power to two redundant load groups, either of which is capable of safely shutting down the plant. Each 4.16kV ESF bus supplies two 480V load centers and one MCC, and the four load centers each supply three MCCs, in turn distributing power to all low-voltage safety-related loads and certain nonsafety-related loads, as mentioned above. The 480V buses also provide backup supply via dedicated transformers to the 120V vital AC power system. Each vital instrument AC power supply consists of an inverter, distribution switchboard, and manual transfer device. Each inverter, which is normally in operation, is supplied from an associated 125V DC Class 1E bus. These are discussed further in the following section.

Figure 3.2-7
AC Power System
Simplified One Line Diagram
(Page 1 of 3)

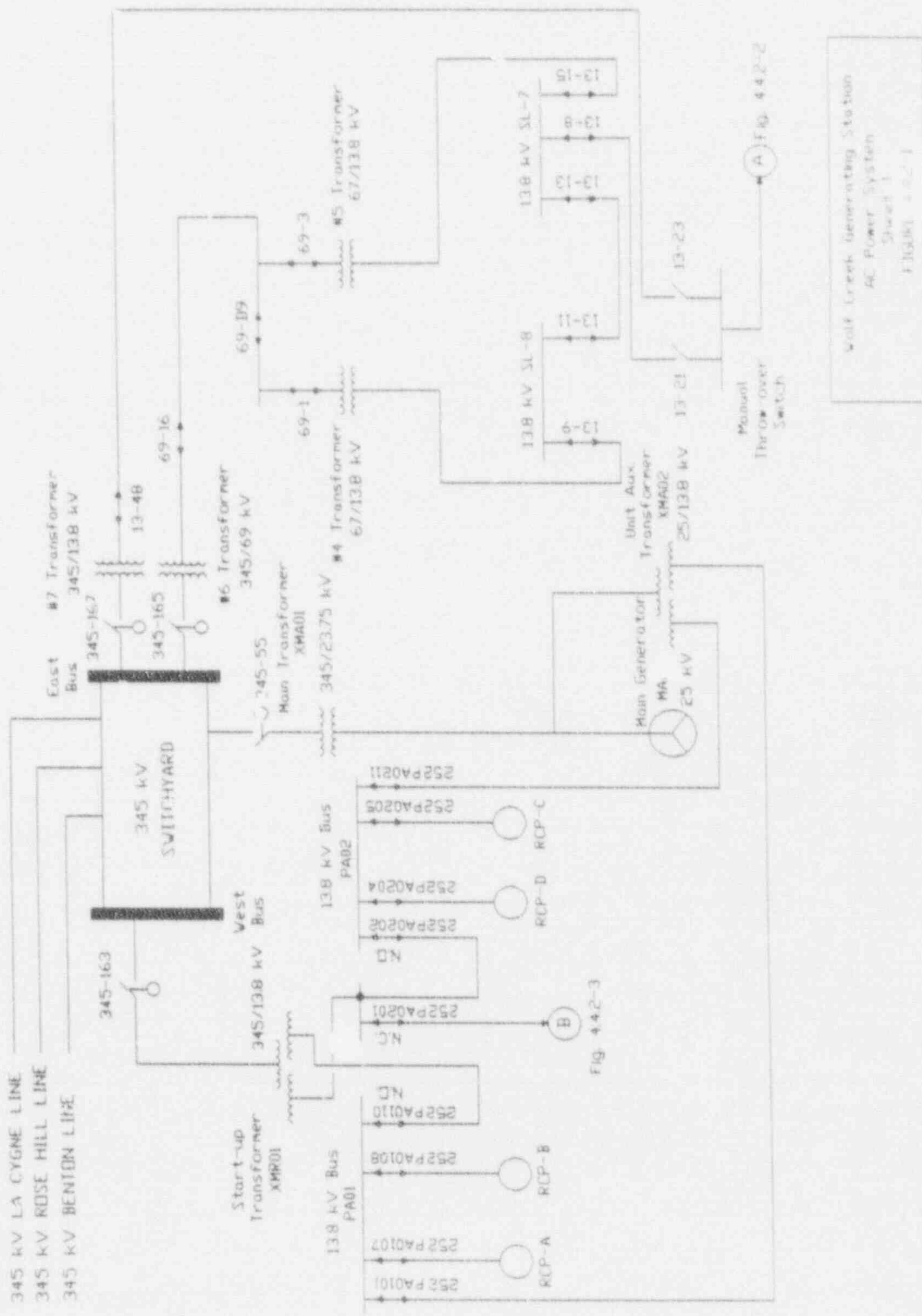


Figure 3.2-7
AC Power System
Simplified One Line Diagram
(Page 2 of 3)

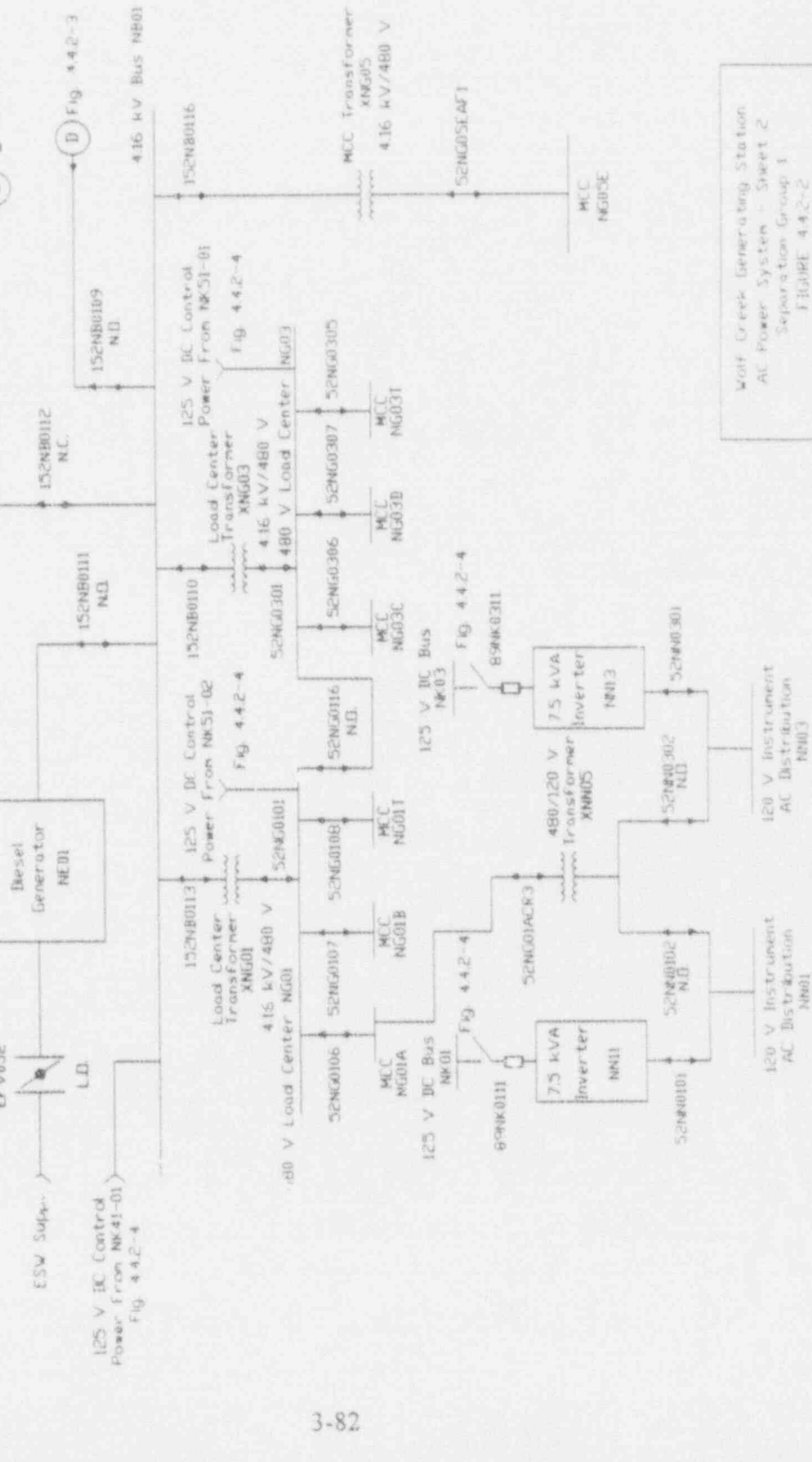
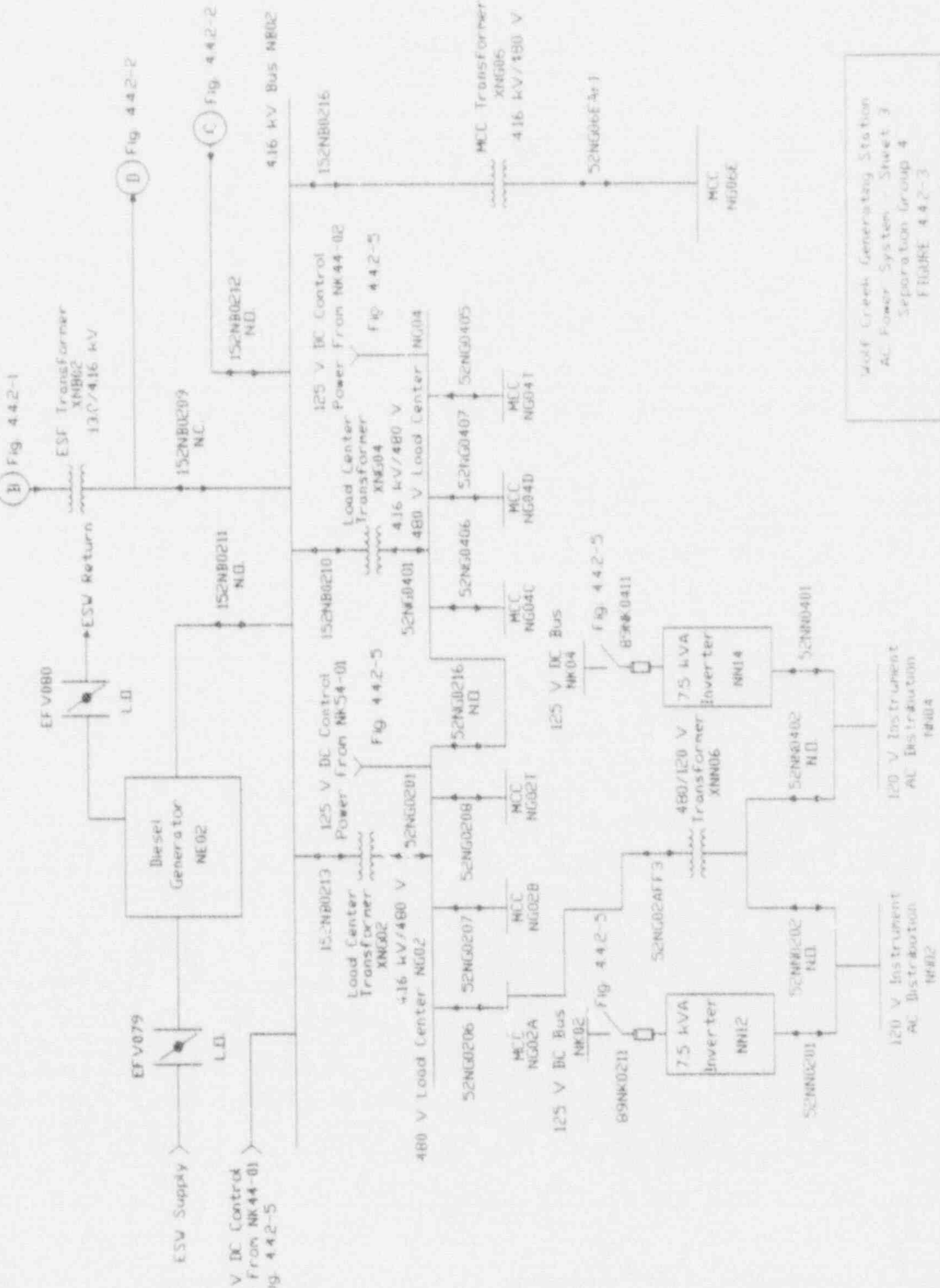


Figure 3.2-7
AC Power System
Simplified One Line Diagram
(Page 3 of 3)



3.2.1.7.2 DC Power System

The plant's DC power system consists of four independent Class 1E 125V DC systems, four nonvital 125V DC systems and one nonvital 250V DC system. The nonvital 125V DC power systems supply nonsafety-related control and instrument loads, emergency lighting, and alternate sources of power for the computer inverters. The nonvital 125V DC battery chargers are fed by Class 1E 480V load centers, but are shed upon loss of offsite power or an SIS. The nonvital 250V DC system provides power to nonvital DC motors, such as emergency lube and seal oil pumps. The system consists of one battery, two chargers, and one bus. Normal power supply is provided by a nonvital 480V AC load center, with backup power available from a Class 1E 480V AC load center.

The Class 1E 125V DC system provides power for Class 1E DC loads and for control and switching of the Class 1E AC systems. A simplified one line diagram of the vital DC system is shown in Figure 3.2-8. The power supply for each DC system consists of one Class 1E battery and one battery charger, the latter powered by the Class 1E 480V AC power system. The system provides vital instrumentation and control power for the Reactor Protection System and the Engineered Safety Features Actuation System.

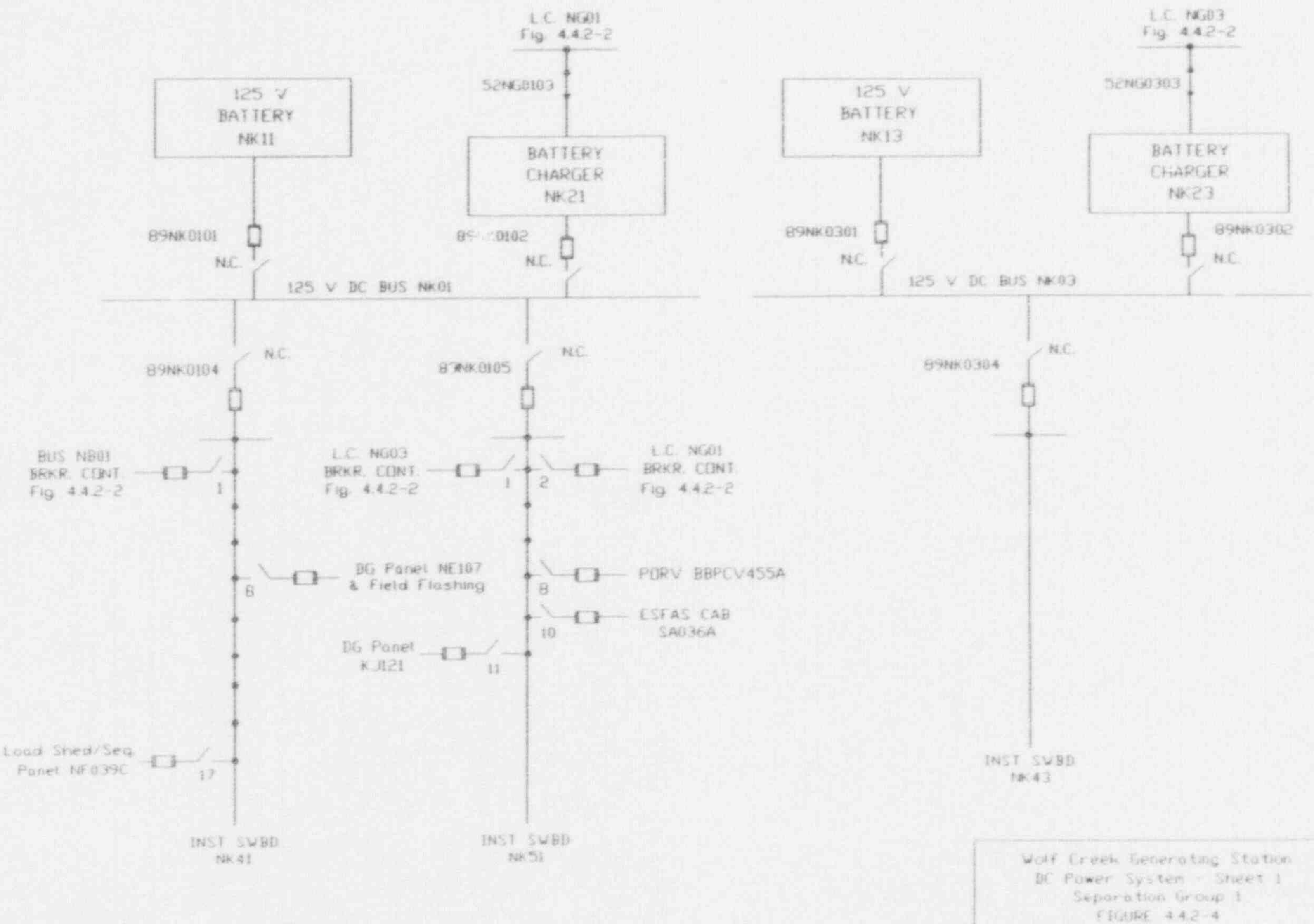


Figure 3.2-8
Vital 125V DC Power System
Simplified One Line Diagram
(Page 1 of 2)

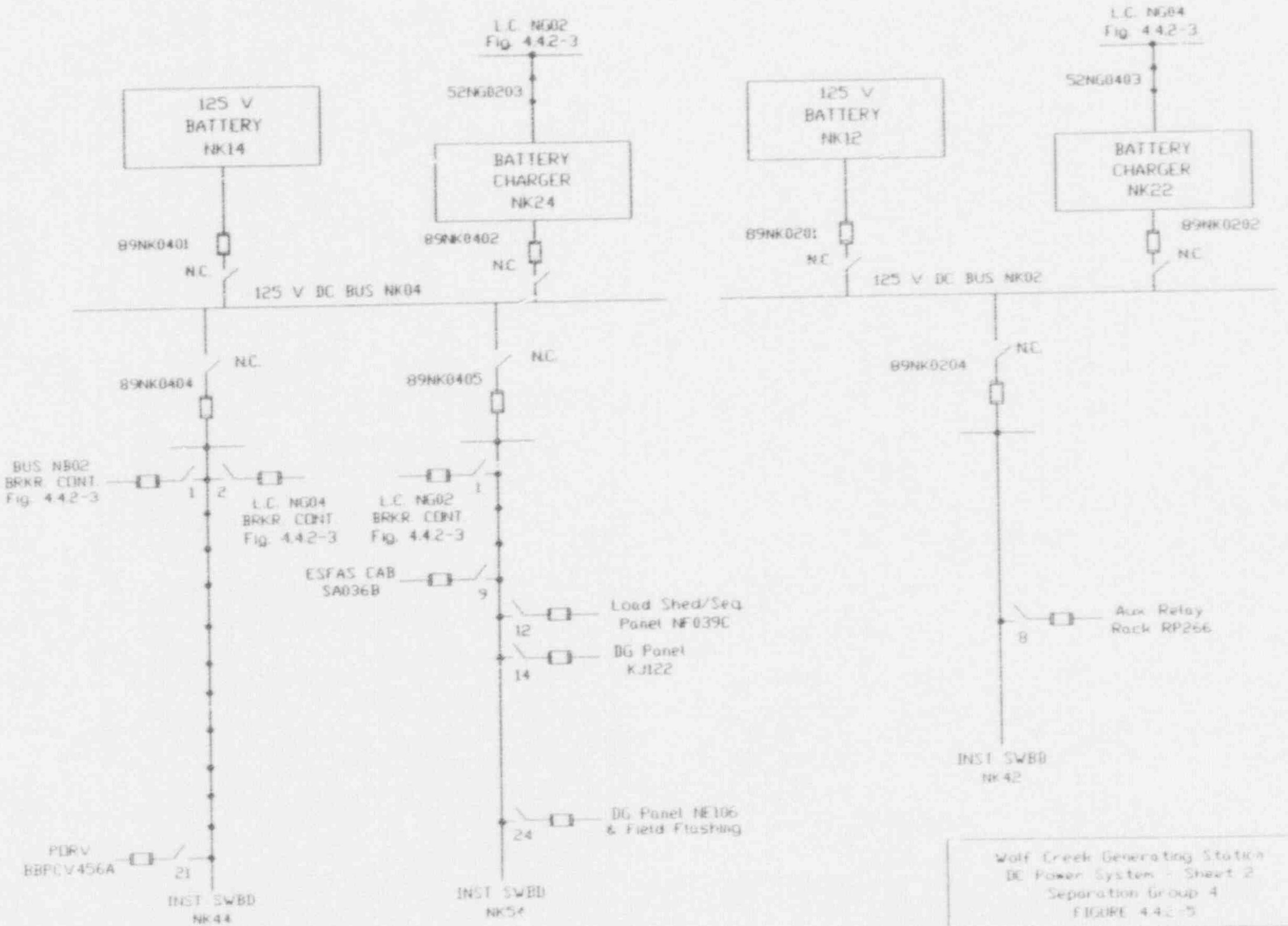


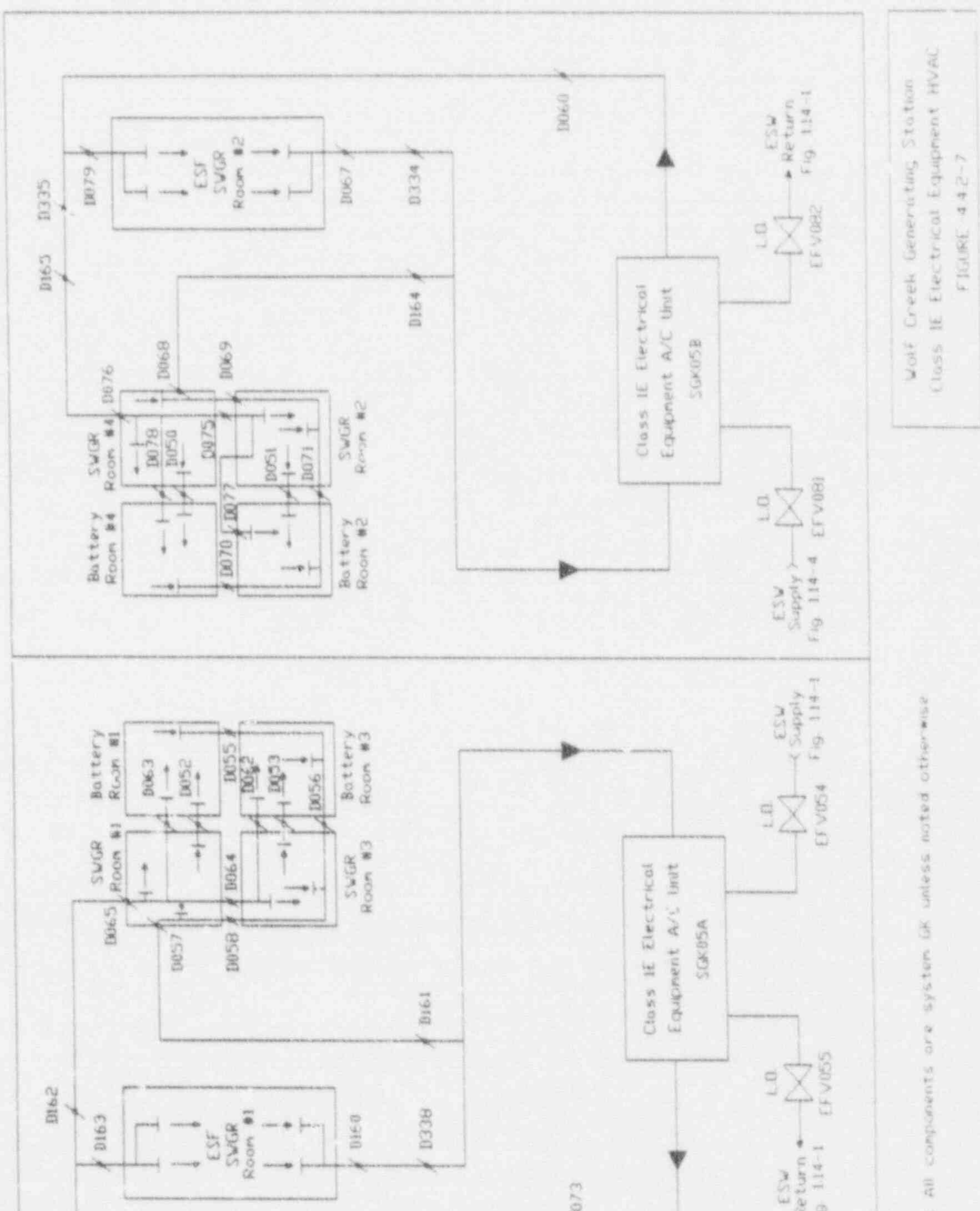
Figure 3.2-8
Vital 125V DC Power System
Simplified One Line Diagram
(Page 2 of 2)

3.2.1.7.3 Class 1E Electrical Equipment HVAC

There are two air conditioning (A/C) units serving the ESF Switchgear, the DC battery, and the DC Switchboard rooms. A simplified P&ID of the system is presented in Figure 3.2-9. Each A/C unit is provided with ESW flow from its associated train of safeguards equipment. The units operate in a continuous recirculation mode with cooling self-regulated by the unit control circuit. The units are normally operating, but will auto-start on loss of preferred AC power, LOCA, or a fuel handling accident.

Failure of the A/C unit is assumed to lead to consequential failure of the associated ESF battery chargers and the 120V AC inverters because of high room temperatures. Thus, these A/C units have been modeled as a support system for the ESF 125V DC power system and for the 120V AC power system (which supports the RPS and ESFAS).

Figure 3.2-9
Class 1E Electrical Equipment HVAC
Simplified P&ID



3.2.1.8 Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) is provided to shutdown and cool the reactor core following design basis accidents. The ECCS is made up of components in the Residual Heat Removal (RHR) system, the Chemical and Volume Control System (CVCS), and the dedicated Accumulators and Safety Injection (SI) systems. These systems provide injection of borated water to the RCS at varying system pressures, ensuring adequate coolant injection capacity at all RCS pressures. The system description provided here is broken down into four functional subsystems: (1) accumulator safety injection; (2) low pressure safety injection and recirculation; (3) high pressure coolant injection and recirculation - safety injection; and (4) high pressure coolant injection and recirculation - boron injection.

Success criteria for the fault tree analyses of the ECCS are presented in Table 3.1-1. All pumps in the ECCS systems are supported by the AC and DC power systems and have room coolers chilled by Essential Service Water flow. Also, Component Cooling Water services the RHR pump seal coolers, RHR heat exchangers, and the lube oil coolers for the charging and SI pumps. Coolant flow to the RHR pumps and heat exchangers is modeled only during the recirculation phase of ECCS operation. Support system dependencies are summarized in Section 3.2.3.

3.2.1.8.1 Accumulator Safety Injection

Each of four accumulators holds over 6000 gallons of borated water, with nitrogen cover gas maintaining pressure between 585 and 665 psig. The accumulators rapidly inject water into the RCS following large and medium LOCAs, limiting peak fuel clad temperatures during the later stages of RCS blowdown and prior to injection by the active low pressure and high pressure safety injection systems. The accumulators may also be used by the operators to help maintain primary coolant inventory during rapid cooldown and depressurization of the RCS following a small LOCA event in which the high pressure safety injection systems fail.

Each independent accumulator is attached to one of the four RCS cold legs by a line having two normally closed check valves and a locked open motor operated valve (MOV). Injection of accumulator inventory occurs if RCS pressure drops below that of the accumulator, opening the check valves to flow. To ensure an open injection pathway, each MOV has its MCC circuit breaker padlocked in the power disconnected position after opening during plant startup. Each MOV also receives automatic signals to open both on a P-11 pressurizer pressure permissive during startup and on an SIS. Figure 3.2-10 presents a simplified P&ID for the accumulator system.

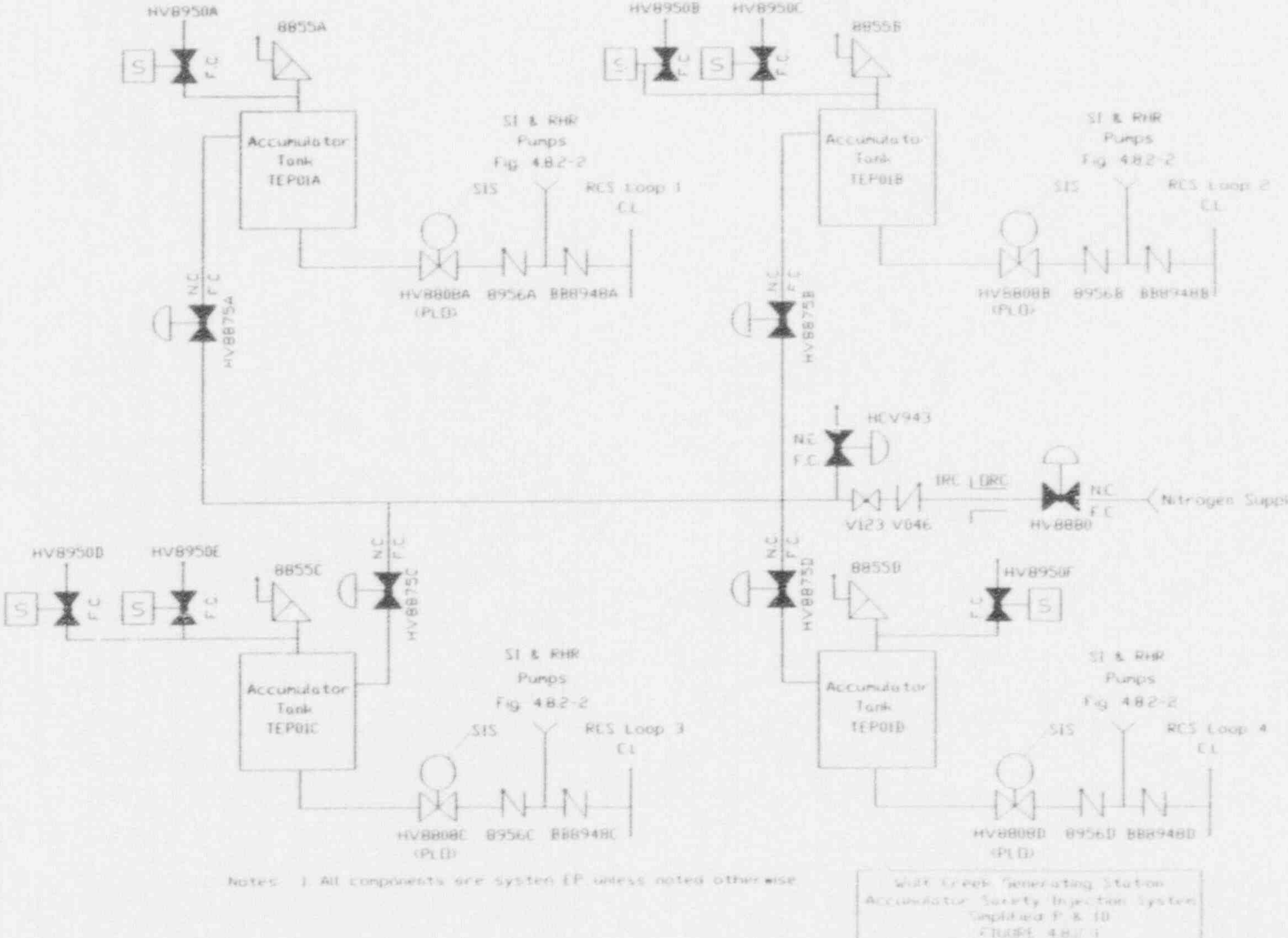


Figure 3.2-10
Accumulator Safety Injection System
Simplified P&ID

3.2.1.8.2 Low Pressure Safety Injection and Recirculation

The Residual Heat Removal System provides the low pressure portion (shutoff head approximately 195 psig) of the ECCS safety injection function along with providing long-term heat removal of core decay heat in recirculation operation. The two RHR pumps are started automatically on a SIS, or they may be manually started. The pumps take suction from the Refueling Water Storage Tank (RWST), and inject water through the RHR heat exchangers to the RCS cold legs. The RHR system is also used for cooling down and maintaining the RCS at cold shutdown conditions.

The RHR system consists of two identical 100 percent capacity trains. Each train consists of a pump, heat exchanger, valves, and associated piping, instrumentation, and controls. A simplified P&ID of the RHR system, aligned in its standby condition for the injection phase of ECCS operation, is shown in Figure 3.2-11. The system is also used for low pressure and high pressure recirculation cooling, where the RHR pump suction is automatically aligned to the containment recirculation sump on low RWST level coincident with an SIS. For high pressure recirculation, RHR discharge from the heat exchangers is manually aligned to the suction of the SI and charging pumps in a piggyback arrangement to cool recirculation flow and to assure adequate NPSH to the high pressure pumps.

The RWST provides storage of borated water to supply all ECCS pumps and the containment spray pumps during the injection mode, following any accident which actuates an SIS. The tank is an outdoor storage, seismic Class 1 tank. It is vented to atmosphere, and has an automatic heating system to prevent freezing. The RWST is periodically used during normal plant operation for testing of the ECCS and containment spray pumps on recirculation flow. RWST inventory is not used for other purposes which would deplete its inventory during power operation. RWST tank levels are indicated and alarmed, and the Technical Specifications require weekly surveillance of volume and boron concentration.

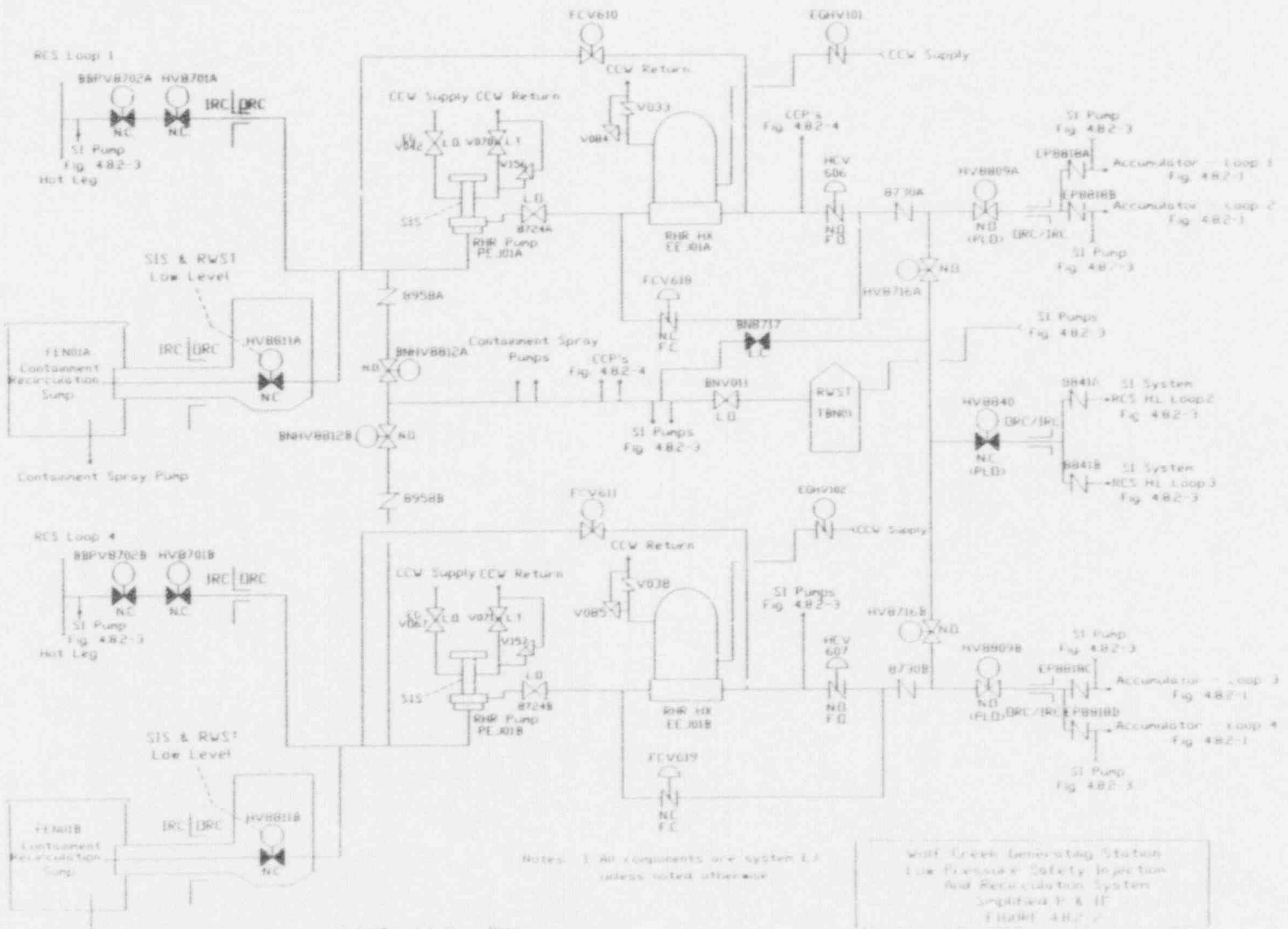


Figure 3.2-11
Low Pressure Safety Injection and Recirculation System
Simplified P&ID

3.2.1.8.3 High Pressure Coolant Injection and Recirculation - Safety Injection

The Safety Injection System (SI) provides injection (shutoff head approximately 1560 psig) for core cooling and negative reactivity (boric acid) addition to the RCS following medium and small LOCAs or steamline/feedline ruptures. Two redundant, full capacity SI pumps take suction from the RWST and deliver injection flow through normally open valves to the RCS cold legs. A simplified P&ID of the system aligned for the injection phase of operation is shown in Figure 3.2-12. The pumps are started automatically on a SIS, or they may be manually started.

In high pressure recirculation operation, the pump suction is aligned manually with the RHR system, drawing flow from the discharge side of the RHR heat exchangers, as discussed in Section 3.2.1.8.2, above.

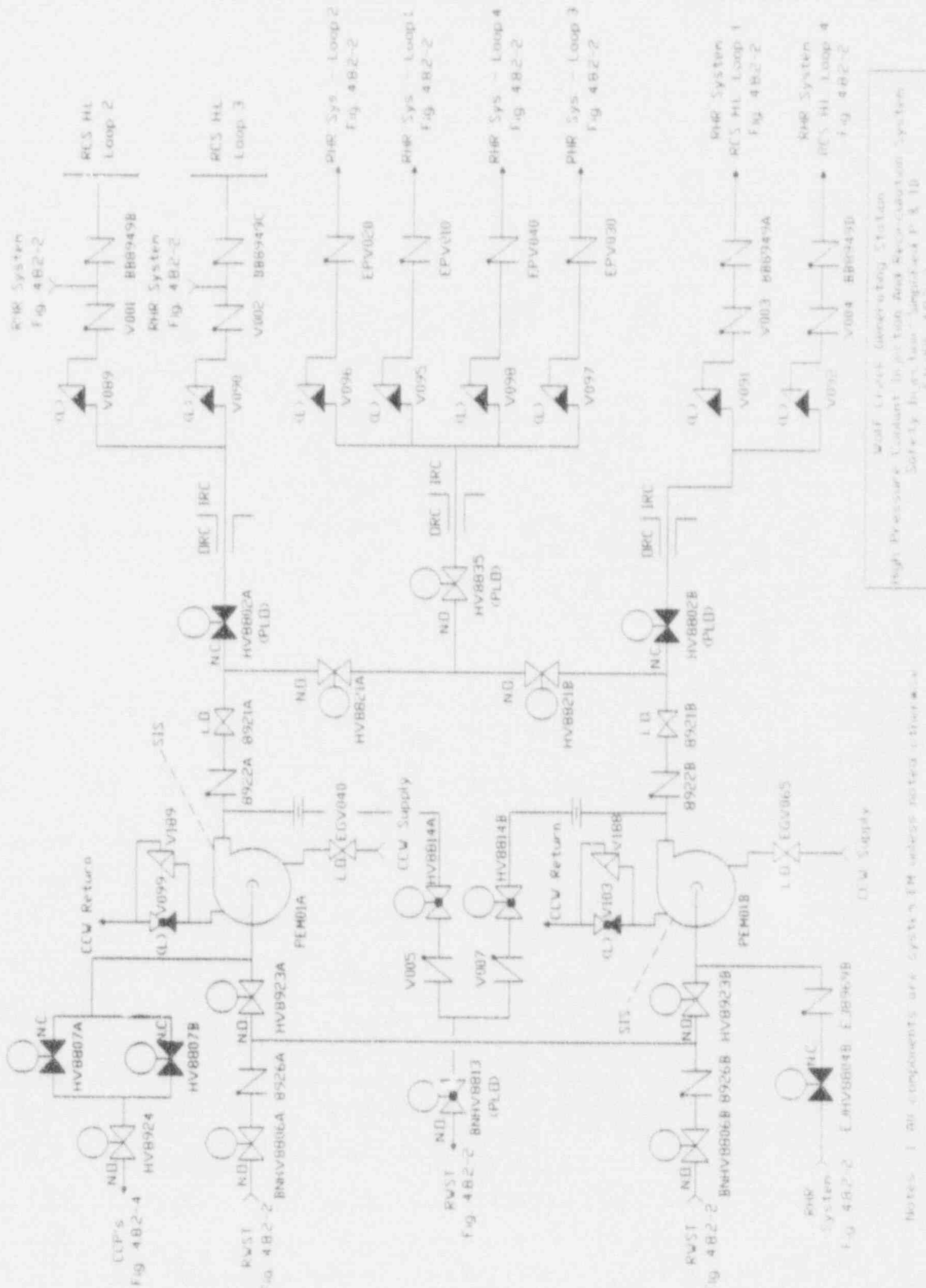
3.2.1.8.4 High Pressure Coolant Injection and Recirculation - Boron Injection

The Boron Injection System provides the high pressure portion (shutoff head approximately 2690 psig) of the ECCS function. The system was designed primarily for medium or small LOCAs (including Steam Generator Tube Ruptures) or secondary side breaks where RCS pressure remains high for an extended period of time. The two centrifugal charging pumps (CCPs) in the CVCS provide the injection function, drawing from the RWST. A simplified P&ID of the system is presented in Figure 3.2-13.

Upon actuation of a SIS, both charging pumps are automatically started, even though one of the pumps may already be operating for normal makeup and RCP seal injection. Pump suction is automatically aligned to the RWST, and the suction valves from the Volume Control Tank are closed. Charging pump discharge is rerouted through the Boron Injection Tank (BIT) by opening parallel inlet and outlet MOVs, and the normal makeup and letdown lines are isolated. Seal injection flow is not isolated by the SIS, and alternative seal injection flow paths from the discharge of each charging pump may also be used by the operators to maintain seal injection flow. The charging pumps may also be aligned to take suction from the Boric Acid Tanks via the Boric Acid Transfer Pumps, as discussed in Section 3.2.1.2, above.

In high pressure recirculation operation, the charging pump suction is aligned manually with the RHR system, drawing flow from the discharge side of the RHR heat exchangers, as discussed in Section 3.2.1.8.2, above.

Figure 3.2-12
High Pressure Coolant Injection and Recirculation - Safety Injection
Simplified P&ID



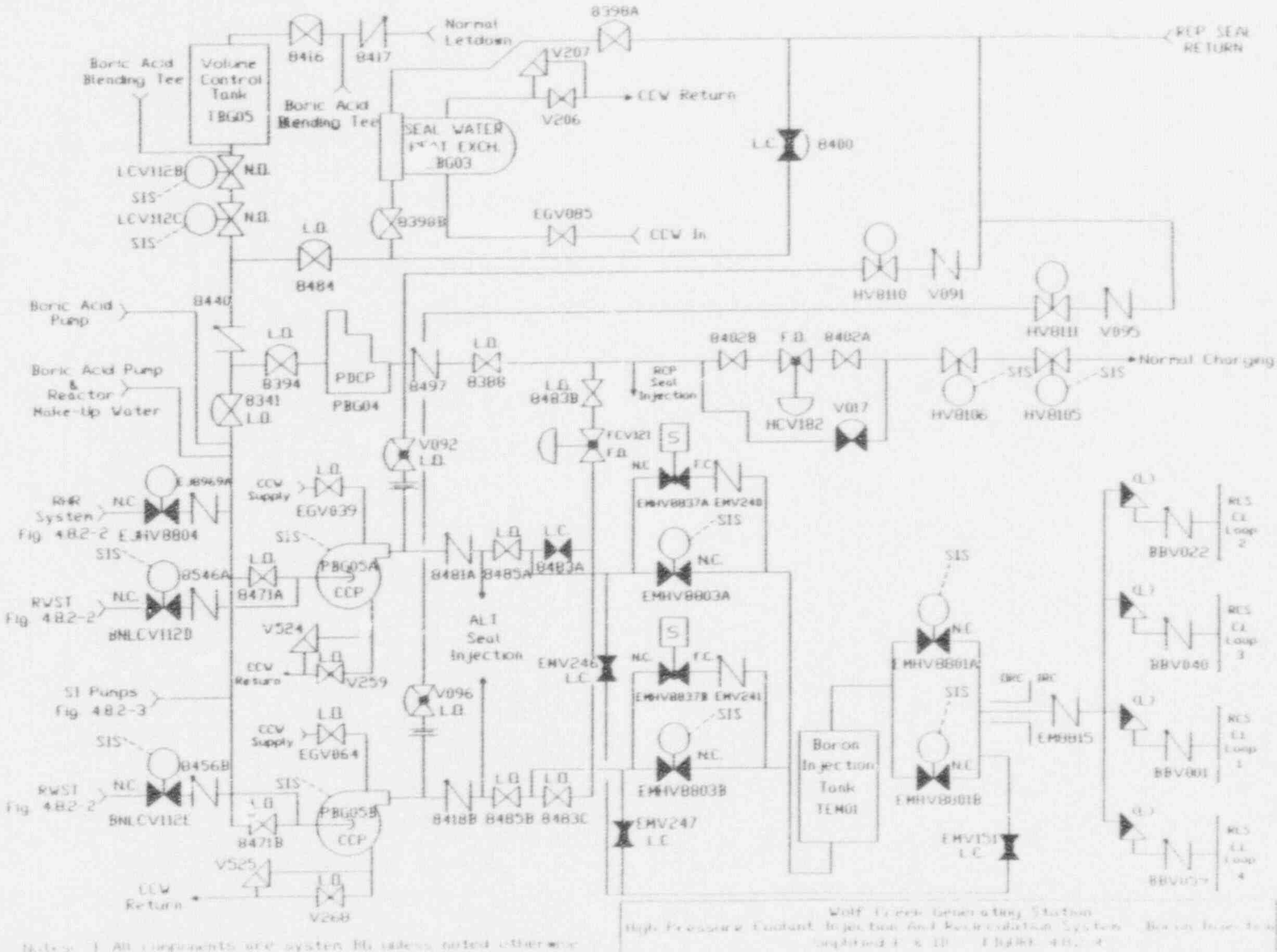


Figure 3.2-13
High Pressure Coolant Injection and Recirculation - Boron Injection
Simplified P&ID

3.2.1.9 Essential Service Water System

The Essential Service Water (ESW) system provides the cooling water requirements for the Component Cooling Water (CCW) heat exchangers, the Diesel Generators, the Containment Coolers, room coolers for the CCW pumps, the charging pumps, the SI pumps, the RHR pumps, the motor driven AFW pumps, the containment spray pumps, the Fuel Pool Cooling pumps, and the Electrical Penetration Rooms, the HVAC condensers for the Control Room and the Class 1E Electrical Equipment air conditioners, and two nonsafety-related instrument/service air compressors. The system also provides an emergency supply of water to the auxiliary feedwater pumps in the event that the Condensate Storage Tank is emptied or otherwise lost as a source of water, and provides emergency makeup water to the Spent Fuel Pool Cooling and Cleanup System and the CCW System.

The ESW system has two redundant flow trains providing flow to safety-related loads needed for safe shutdown of the reactor. Each train contains a pump, pump prelube tank, self-cleaning strainer, traveling water screen, and associated piping, valves, and instrumentation. A simplified P&ID of the ESW system is presented in Figure 3.2-14. The system draws water from and discharges to the Ultimate Heat Sink (UHS). The UHS is a normally submerged cooling pond formed as part of the WCGS main cooling lake behind a seismic Category I dam built across one finger of the lake.

During normal plant operation, the ESW system receives flow from the nonessential Service Water System. A simplified P&ID of the Service Water System is presented in Figure 3.2-15. Upon a SIS actuation, loss of offsite power, or an AFW pump low suction pressure signal, the Class 1E ESW pumps will start, as will their room cooling fans and traveling screens. In addition, the two (2) motor operated isolation valves in each crosstie header joining the ESW to the Service Water System are closed.

The non-essential Service Water System consists of three 50 percent capacity pumps and one 17 percent capacity low flow pump, two 100 percent capacity strainers, four traveling screens, and associated valves, piping, and instrumentation. The system provides cooling flow to various nonsafety-related plant loads, in addition to providing the normal supply to the ESW, as described above. The pumps take suction from the Circulating Water Screenhouse sump, and discharge back to the WCGS cooling lake. A portion of the return flow discharges through the ESW return lines to the UHS.

Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 1 of 6)

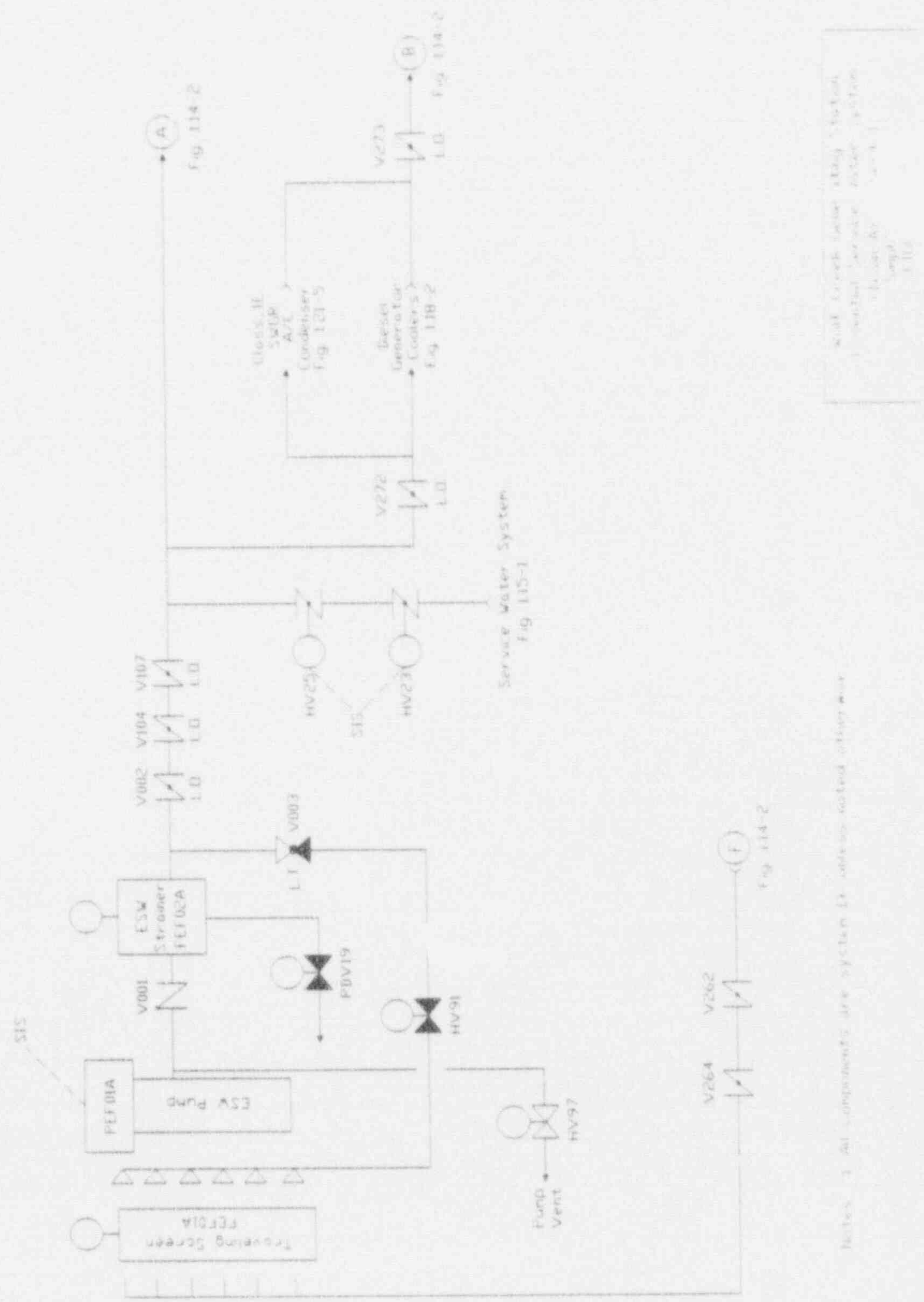


Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 2 of 6)

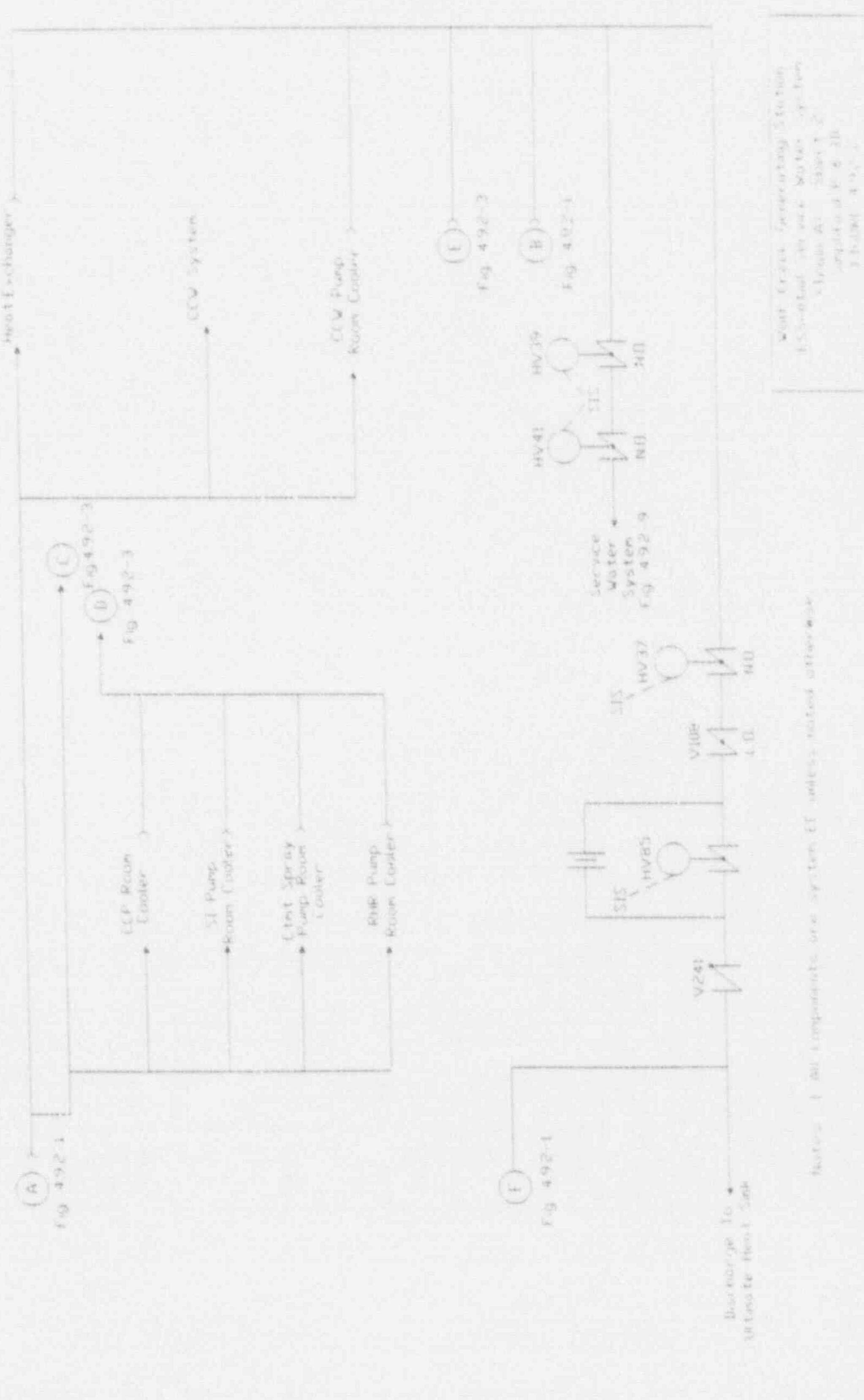


Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 3 of 6)

Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 4 of 6)

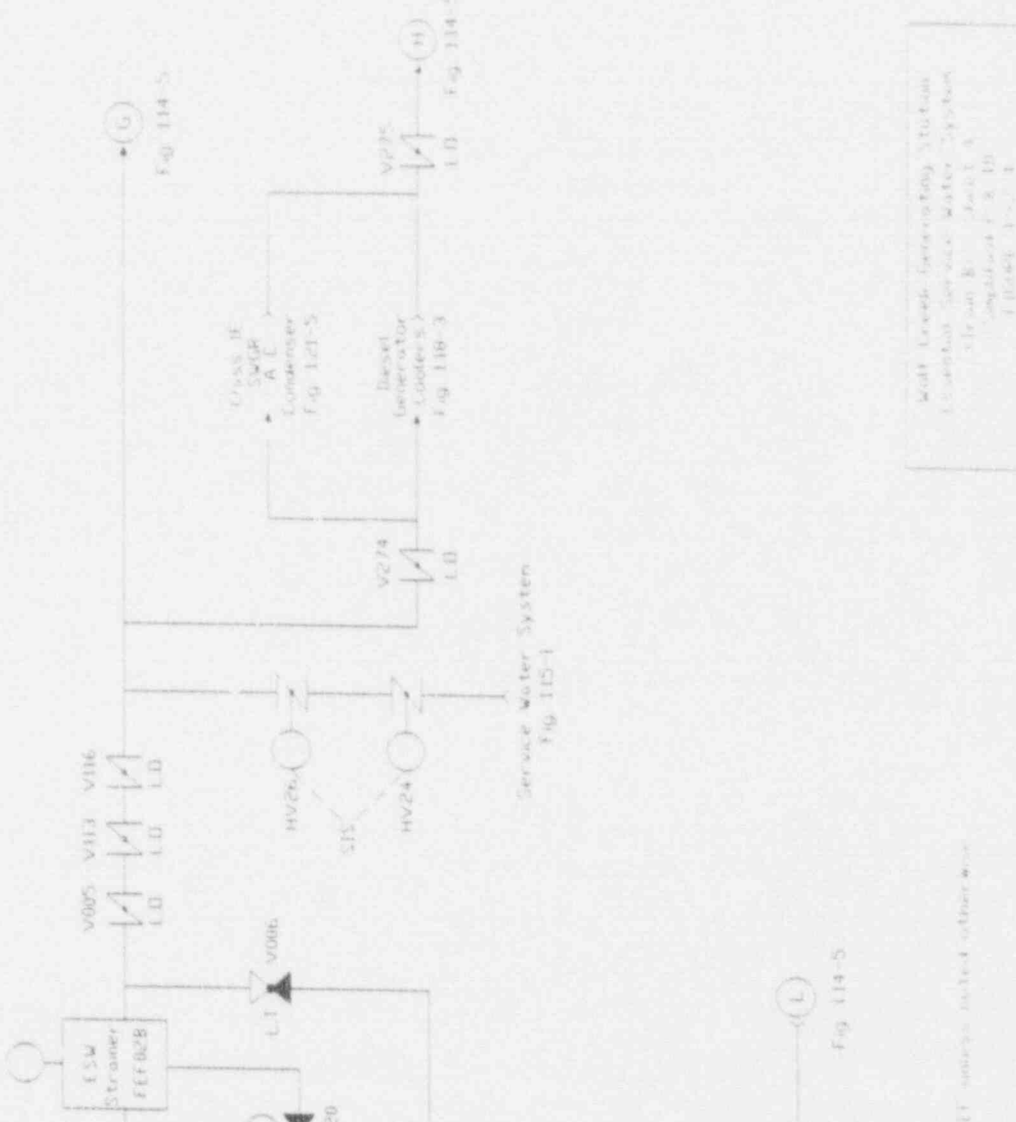


Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 5 of 6)

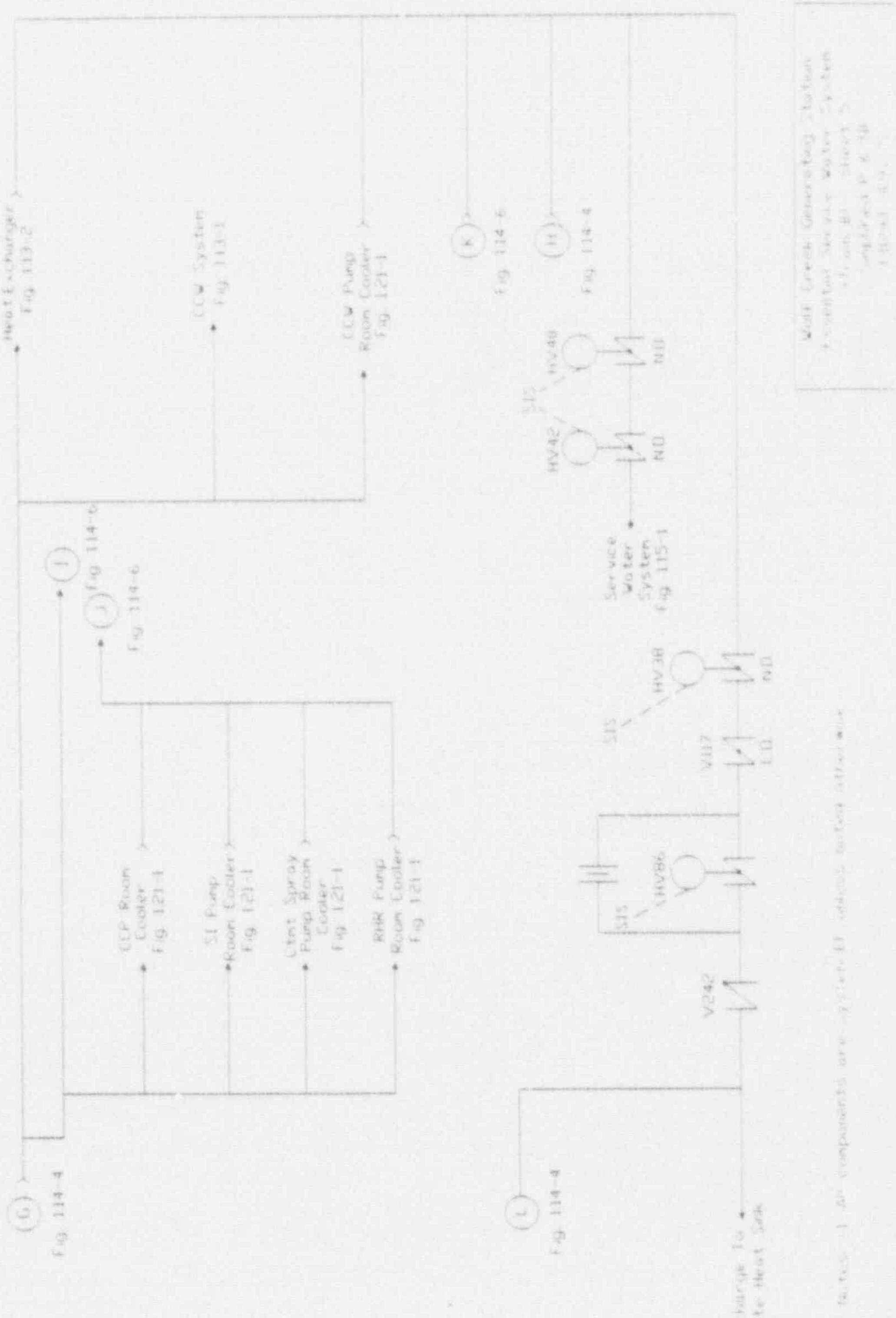


Figure 3.2-14
Essential Service Water System
Simplified P&ID
(Page 6 of 6)

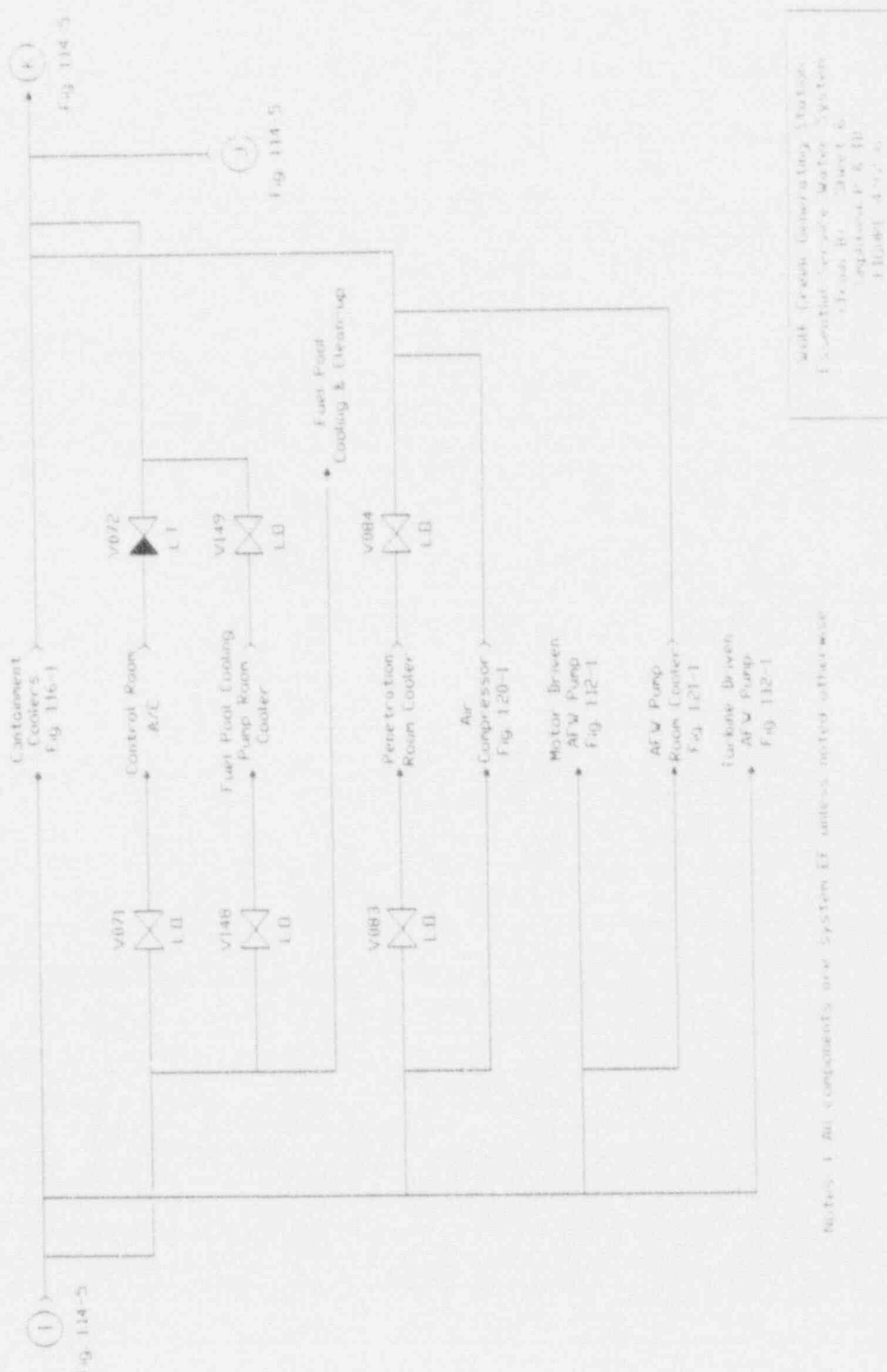
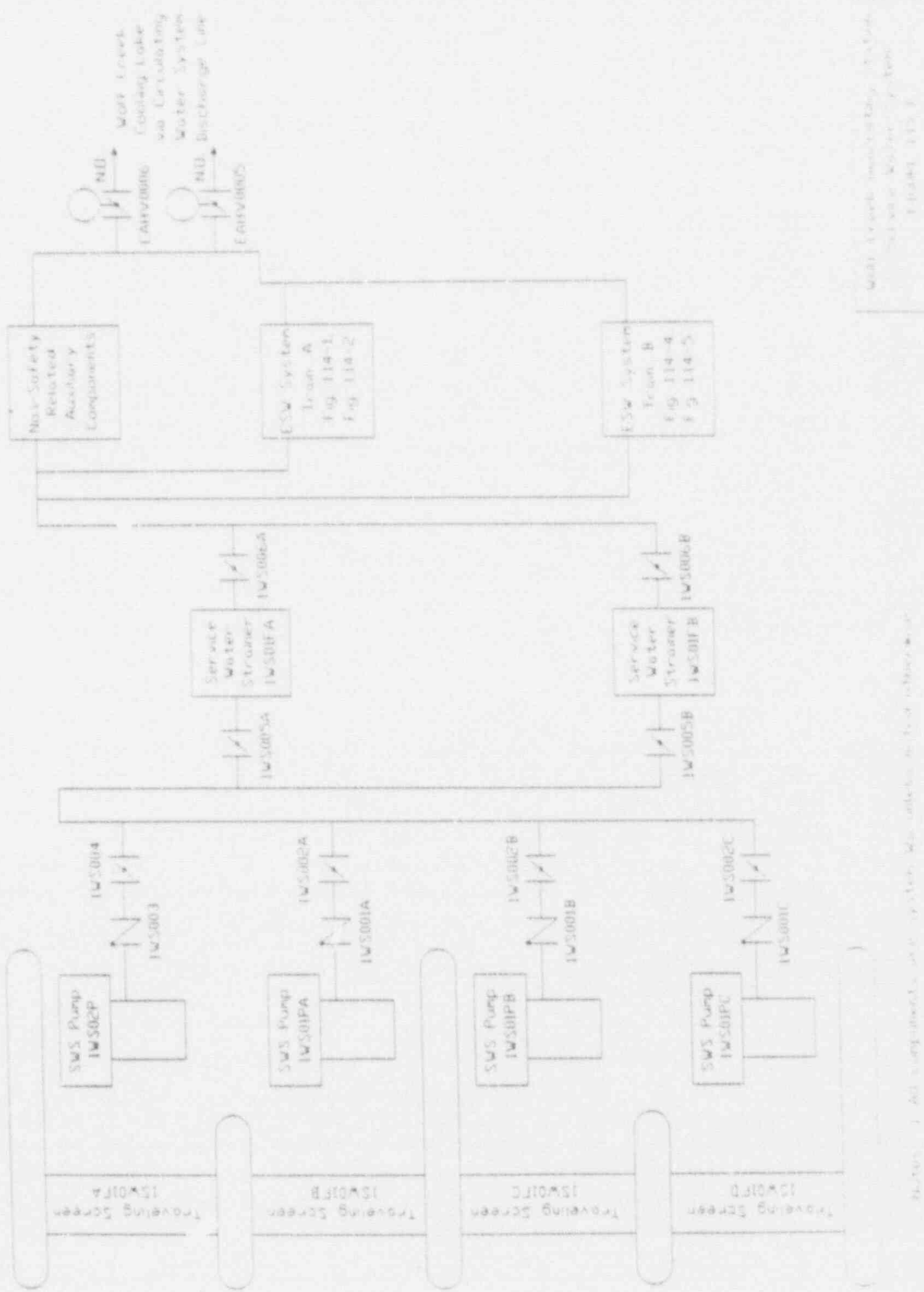


Figure 3.2-15
Nonessential Service Water System
Simplified P&ID



3.2.1.10 Reactor Protection System

The Reactor Protection System (RPS) consists of process and nuclear instrumentation, a solid-state protection system (SSPS), reactor trip switchgear, balance-of-plant engineered safeguards actuation system, and load shedder emergency load sequencers. For reactor trip and engineered safety features actuation, the SSPS contains two redundant logic trains that are physically and electrically independent. The protection portion of the SSPS receives inputs from process instrumentation, nuclear instrumentation, field contacts, and directly from main control board switches.

Process instrumentation includes devices which measure temperature, pressure, fluid flow, and fluid levels. A typical process instrumentation channel includes a sensor, loop power supply, signal conditioning devices, comparators, indicators, recorders, alarm actuating devices, and controllers. Nuclear instrumentation comprises fission chambers for source and intermediate range flux measurement and uncompensated ion chamber assemblies for measuring power range flux. Conditioned signals from the detectors are input to the SSPS.

The RPS has two major categories of protective actions that are addressed within two functionally defined subsystems: the Reactor Trip System (RTS) and the Engineered Safety Features Actuation System (ESFAS). These subsystems are described in the first two of the following subsections. The third subsection discusses the ATWS Mitigation System Actuating Circuitry, which is an alternative system provided at WCGS as a backup to the RTS.

3.2.1.10.1 Reactor Trip System

The Reactor Trip System (RTS) functions to prevent reactor operation outside of prescribed safety limits. The limits of the safe reactor operating envelope are defined by reactor power; RCS temperatures, pressure, and flow; pressurizer level, and secondary system heat removal capabilities. Rapid reactivity shutdown is provided by the insertion of the rod cluster control assemblies (RCCA) by free fall into the core. Duplicate series-connected circuit breakers supply all power to the control rod drive mechanisms (CRDM). The CRDM must be energized for the RCCA to remain withdrawn from the core, and an automatic reactor trip occurs upon loss of power to the CRDM.

If the RTS receives signals indicating an approach to unsafe operating conditions, the system actuates alarms, prevents control rod withdrawal, initiates load runback, and/or opens the reactor trip breakers. The reactor trip breaker shunt trip coils energize on an automatic trip, which also will open the breaker. Certain reactor trip signals are automatically bypassed at low reactor power where they are not required for safety.

The success criteria for the RTS is uniform for all initiating events: one of two reactor trip breakers must open upon demand.

3.2.1.10.2 Engineered Safety Features Actuation System

The Engineered Safety Features Actuation System (ESFAS) senses accident situations and initiates the operation of necessary safety systems to prevent or mitigate damage to the core and ensure containment integrity. For example, generation of a safety injection signal (SIS) results in reactor trip, emergency diesel generator start, phase A containment isolation, control room ventilation isolation, steam generator blowdown isolation, and auxiliary feedwater system actuation. The load shedder emergency load sequencers time the energization of loads to minimize voltage transients on the 4.16kV essential buses.

The ESFAS includes a Balance of Plant ESFAS (BOP ESFAS), which initiates Containment Purge Isolation, Control Room Ventilation Isolation, Fuel Building Isolation, Auxiliary Feedwater Actuation, Auxiliary Feedwater Low Suction Pressure Switchover to ESW, and Steam Generator Blowdown and Sample Isolation actions. The BOP ESFAS works in conjunction with the SSPS, and its components are functionally equivalent to the SSPS instrumentation and components, except that it uses modular bistables and relay drivers rather than solid state cards.

3.2.1.10.3 ATWS Mitigation System Actuating Circuitry

The ATWS Mitigation System Actuating Circuitry (AMSAC) provides a nonsafety grade backup to the Reactor Protection System (RPS) to automatically trip the reactor, trip the turbine-generator, and initiate auxiliary feedwater flow following an Anticipated Transient without Scram (ATWS) event. AMSAC is automatically activated when reactor power is above 40 percent. Analyses have indicated that the peak pressure transient associated with an ATWS for plant power levels below this value will not violate RCS integrity.

3.2.1.11 Condensate and Main Feedwater Systems

The Condensate and Main Feedwater System provides a continuous water supply at required temperature and pressure from the condenser hotwells to the four steam generators. The major system components which are addressed in the WCGS PRA are the Main Feedwater Isolation Valves (FWIV), the main feedwater control valves, the feedwater pumps, and the condensate pumps.

Simplified P&IDs of the Condensate and Main Feedwater System are presented in Figures 3.2-16 and 17. The three 50 percent capacity condensate pumps take suction from the condenser hotwells and discharge through the condensate demineralizers. Flow is then passed through three parallel trains of low pressure feedwater heaters to the suction of the feedwater pumps. There are two main 67 percent capacity turbine-driven feedwater pumps, and a motor driven startup pump having a capacity sufficient to satisfy feedwater requirements up to about 2 percent power. The feedwater pumps discharge through high pressure feedwater heaters to a common header. Each steam generator has an associated feedwater flow control valve and main feedwater isolation valve.

In a plant shutdown, or following a trip, the condensate and main feedwater system may, if plant conditions permit, provide a feedwater supply to the steam generators in place of the auxiliary feedwater system.

Figure 3.2-16
Main Feedwater System
Simplified P&ID
(Page 1 of 2)

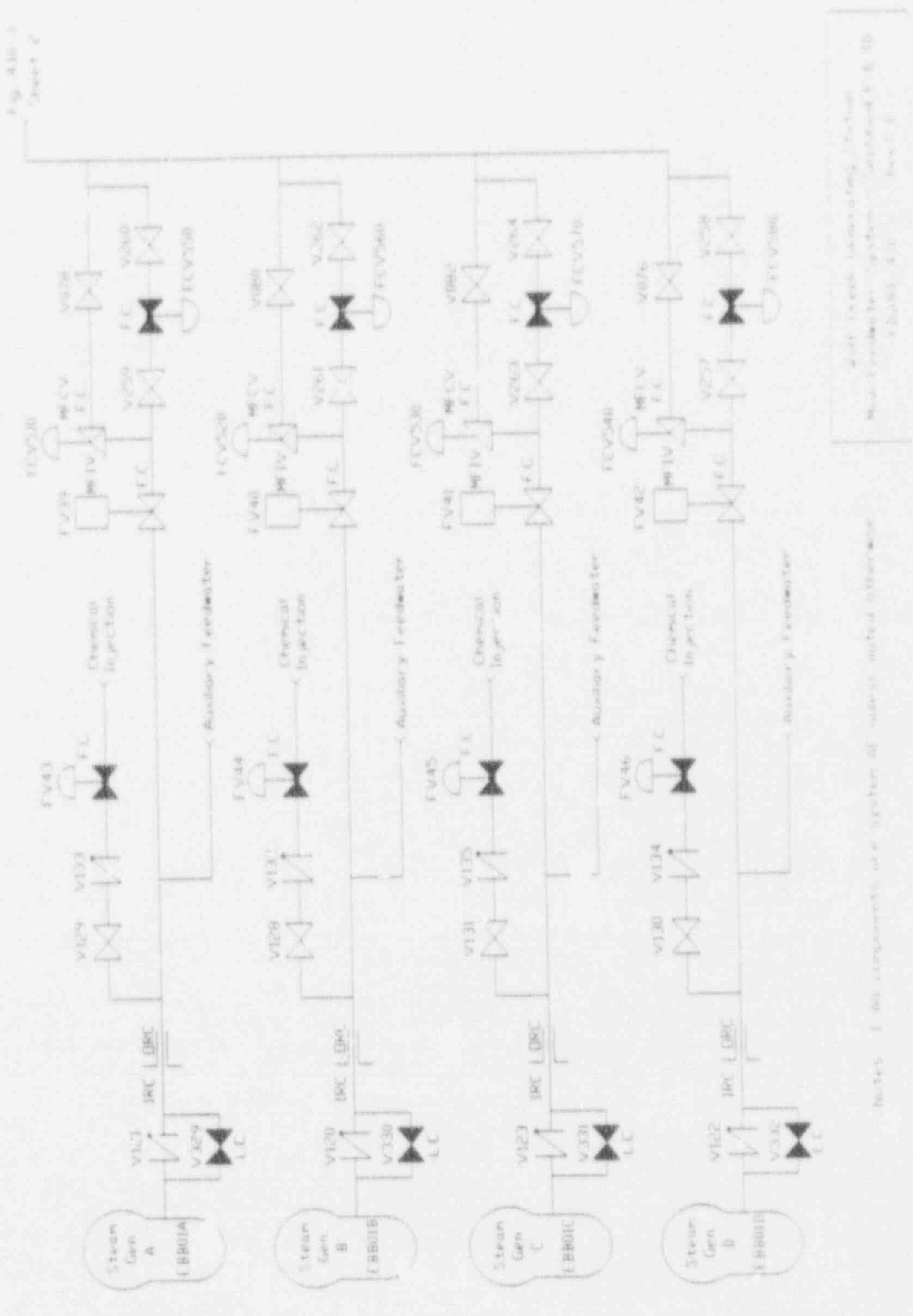
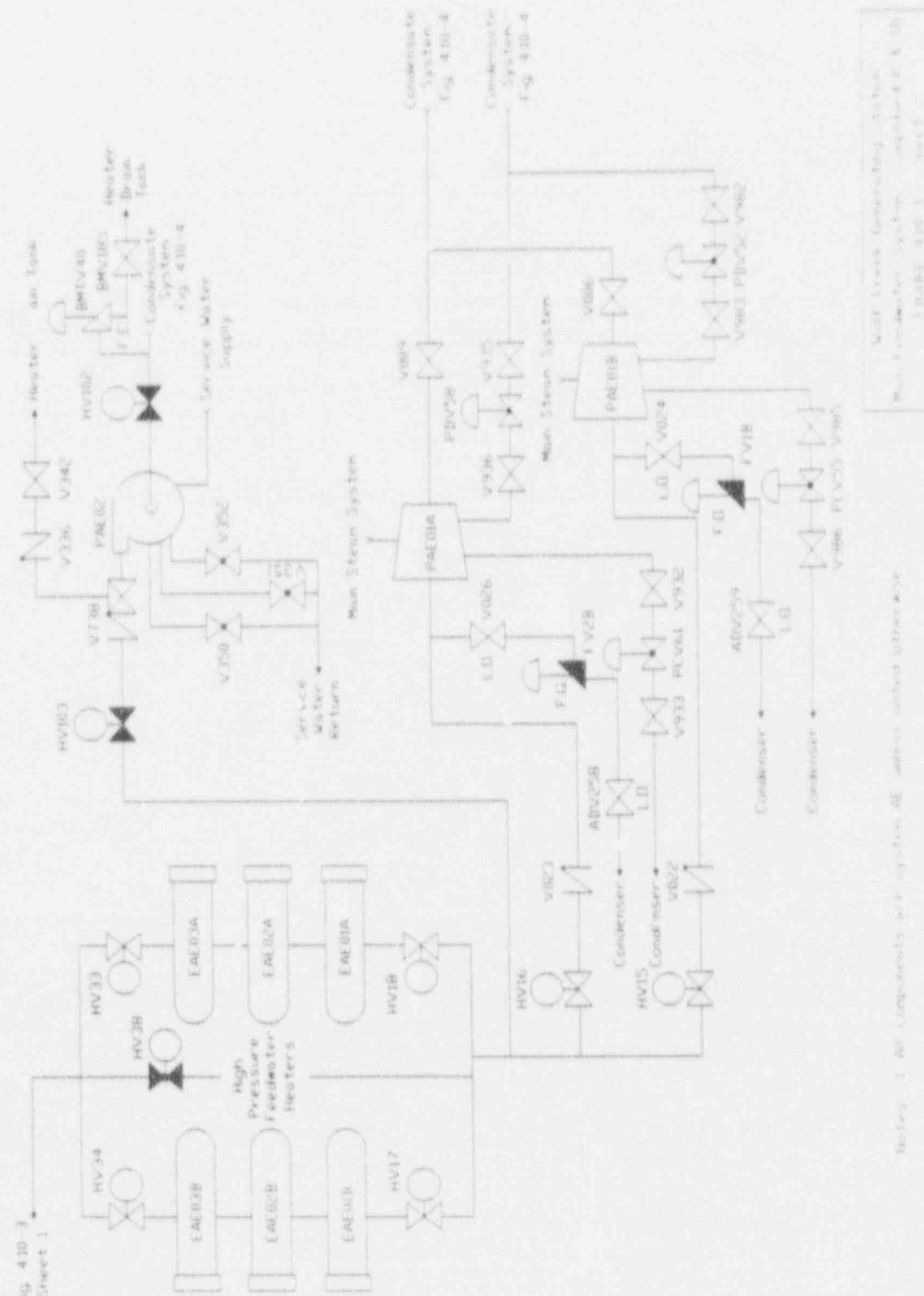


Figure 3.2-16
Main Feedwater System
Simplified P&ID
(Page 2 of 2)



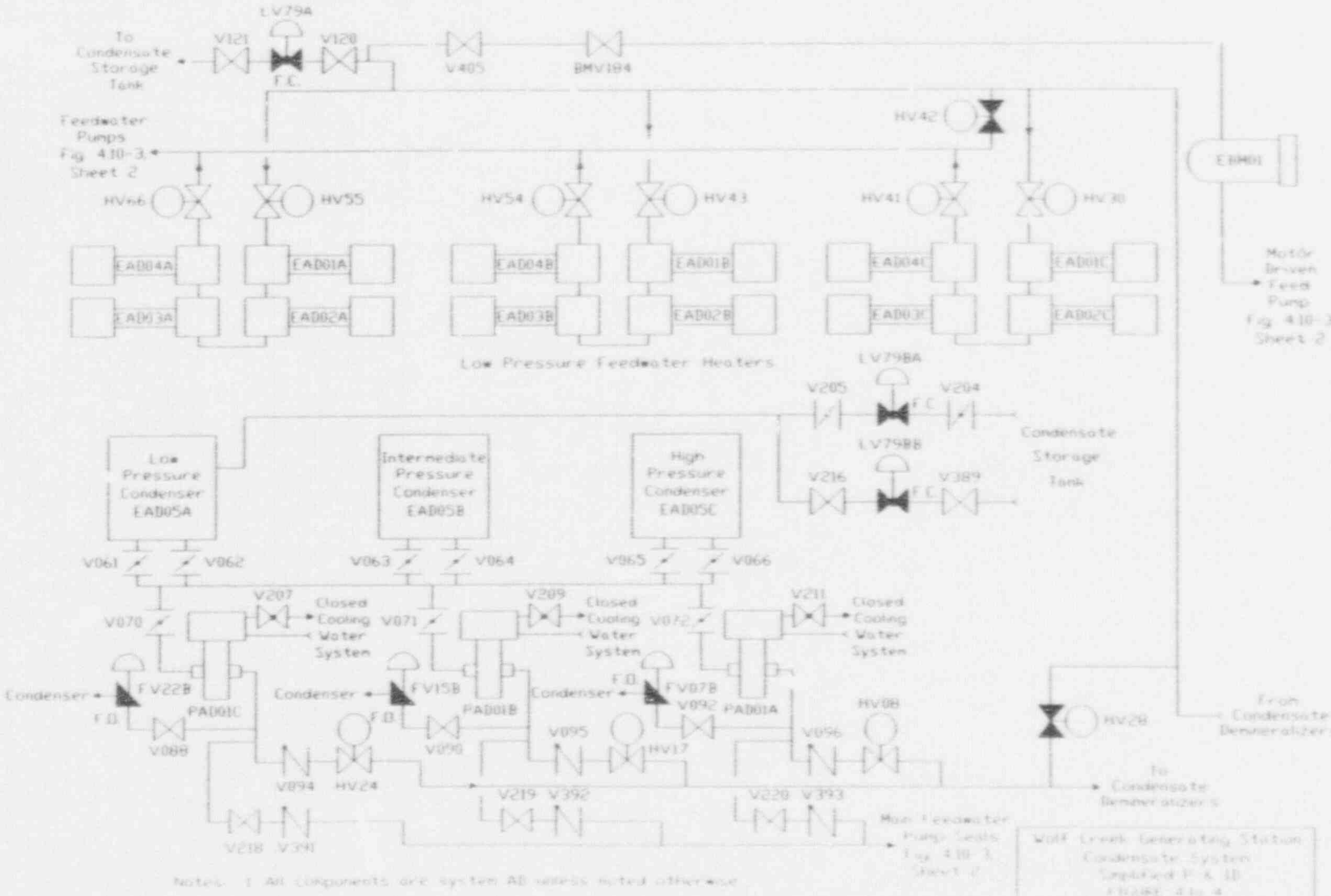


Figure 3.2-17
Condensate System
Simplified P&ID

3.2.1.12 Main Steam System

The Main Steam system supplies saturated steam from the steam generators to the turbine generator for power generation and to auxiliary systems for motive power (e.g., turbine-driven main and auxiliary feedwater pumps). The system also provides means to dissipate heat during plant step load changes and during plant startup via turbine bypass (steam dump) valves or the steamline atmospheric relief valves. The system is protected from overpressurization by banks of safety valves on the steam lines from each steam generator.

The major components of interest for the WCGS PRA are the Main Steam Isolation Valves (MSIV), the atmospheric relief valves, steam dump valves, and safety valves. A simplified P&ID of the Main Steam system is shown in Figure 3.2-18. The MSIVs provide the primary means for isolating secondary system ruptures and establishing a controlled secondary heat removal path. The turbine bypass system dumps steam to the condenser if a turbine load reduction occurs with the plant operating at high power. Following a turbine trip/reactor trip or load reduction greater than 50 percent, or if the turbine bypass system is unavailable, secondary system steam will be released to the atmosphere through the atmospheric relief and/or safety valves.

The Main Steam system has been analyzed for failure to isolate following a Steam Generator Tube Rupture and failure to isolate following a Steamline/Feedline Break event. The system also provides motive steam to the turbine-driven auxiliary feedwater pump. That function and the steam relief function have been analyzed in the fault tree analysis of the AFW system.

Figure 3.2-18
Main Steam System
Simplified P&ID
(Page 1 of 2)

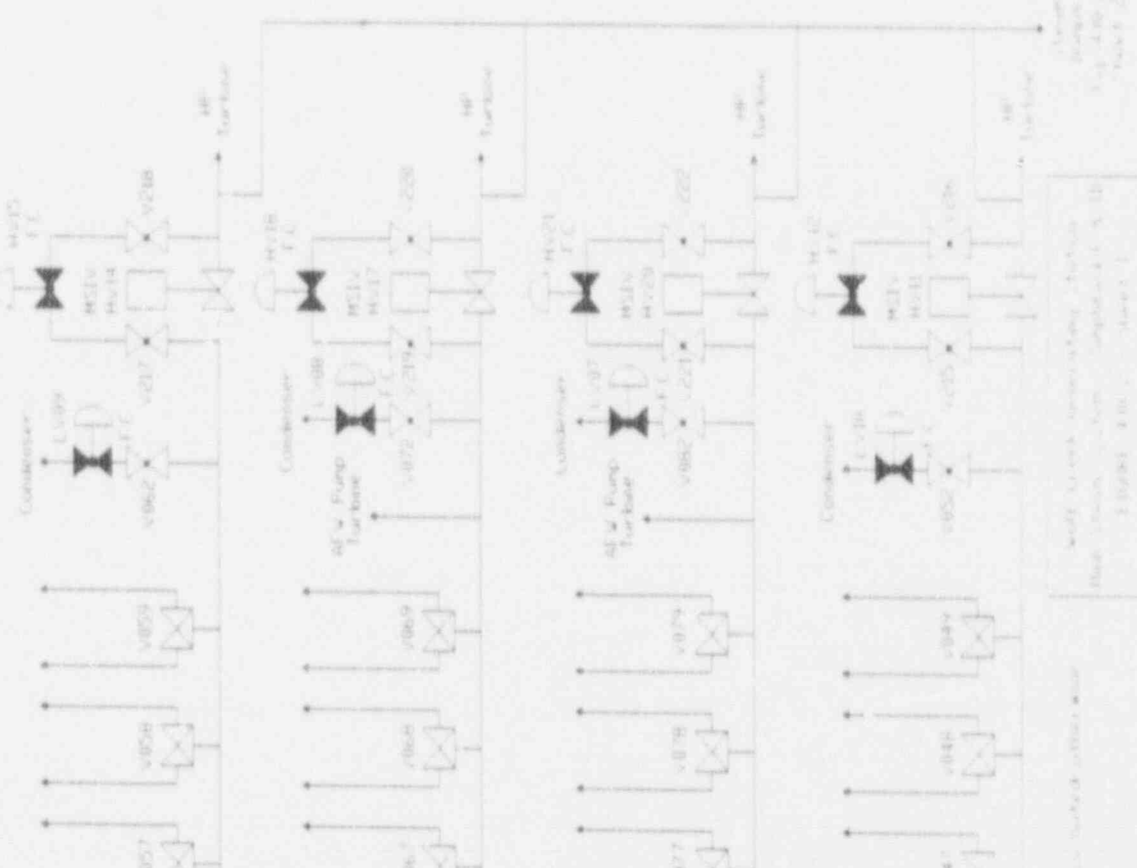
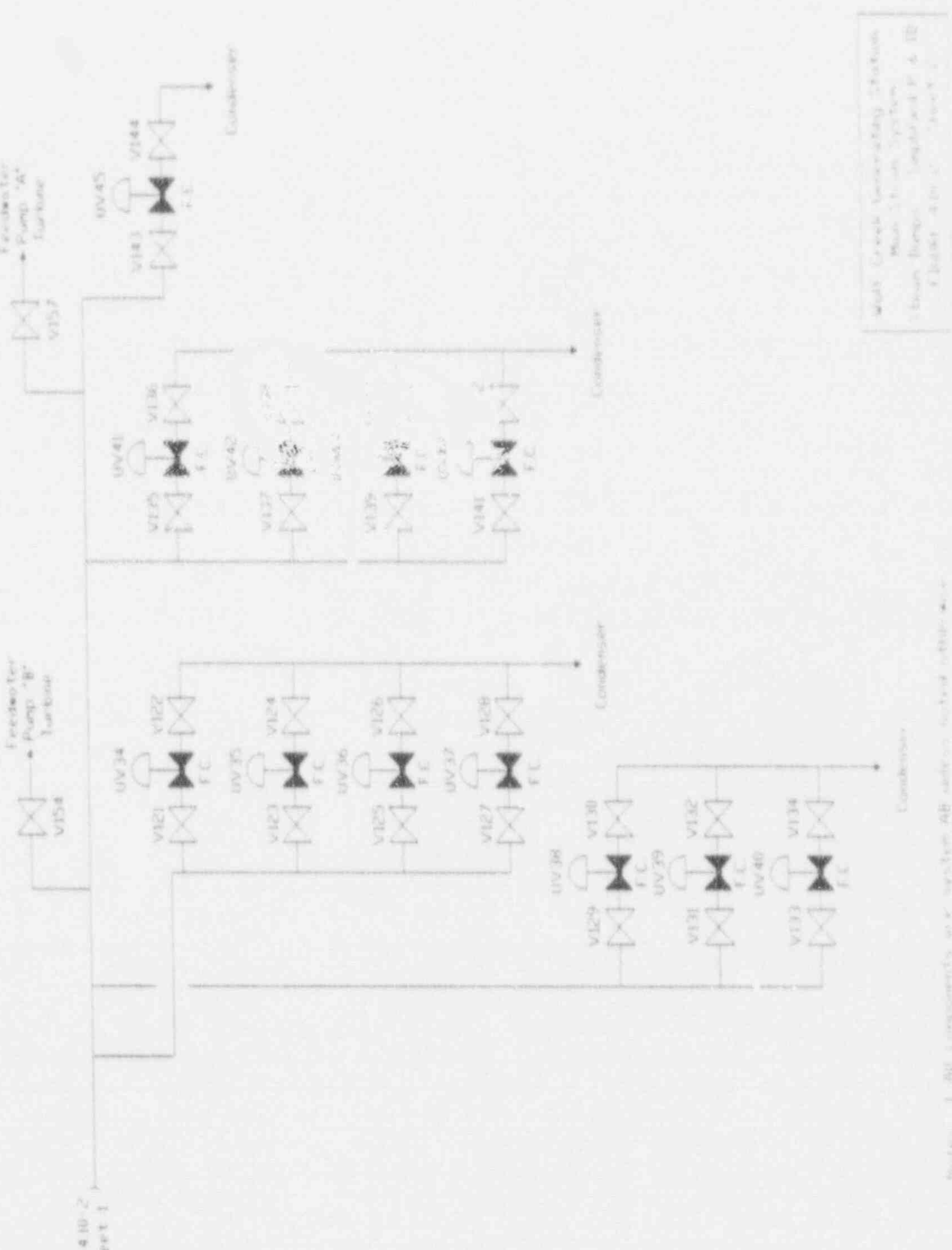


Figure 3.2-18
Main Steam System
Simplified P&ID
(Page 2 of 2)

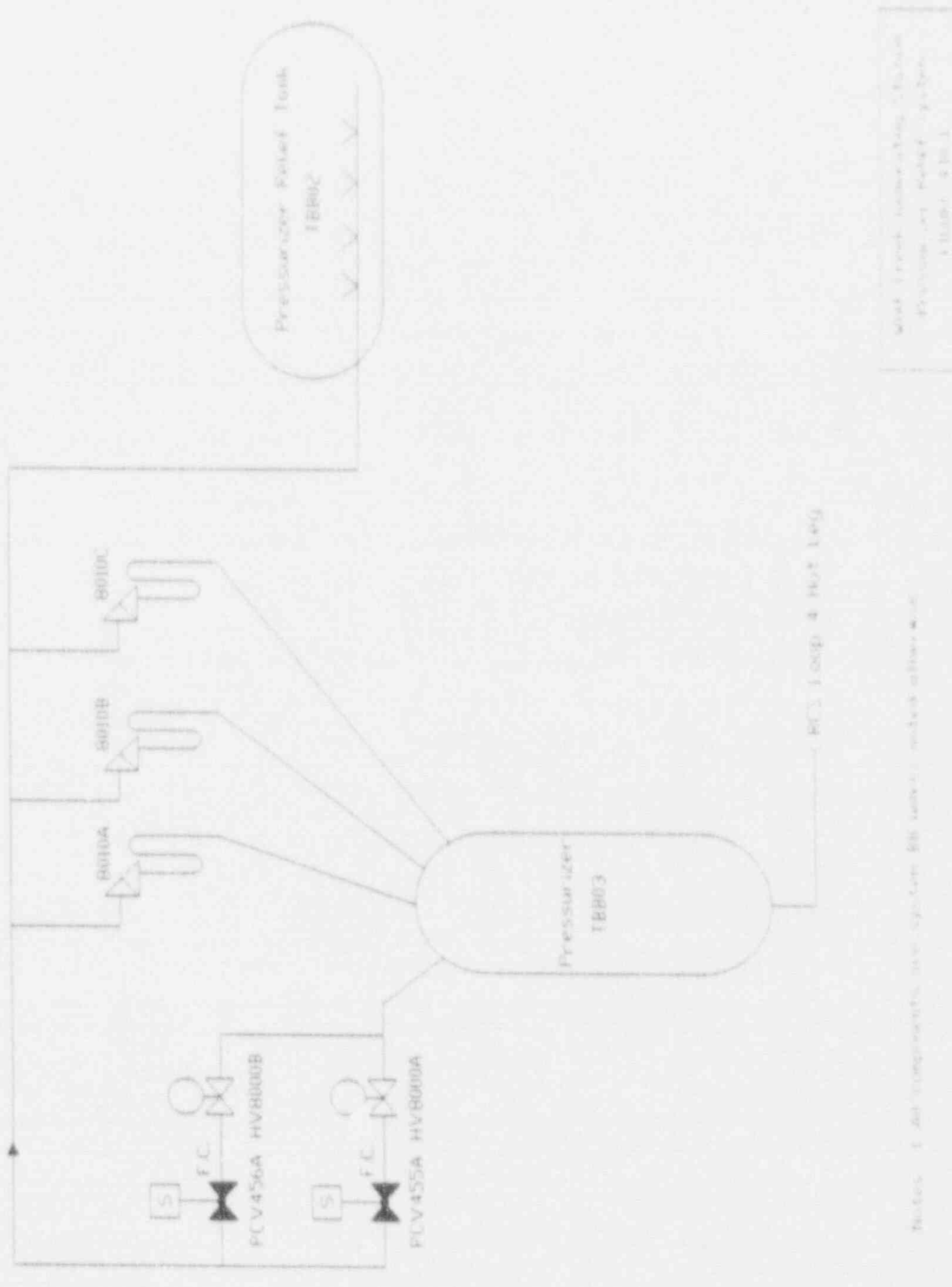


3.2.1.13 Pressurizer Relief System

The RCS pressurizer is designed to maintain system pressure during operation and limit pressure transients by accommodating primary coolant volume changes due to plant load changes. The pressurizer is provided with three safety valves and two Power Operated Relief Valves (PORV). The PORVs are actuated by signals from the pressurizer pressure instrumentation. Actuation setpoints are established to prevent the opening of the pressurizer safety valves. A motor operated block valve is located upstream of each PORV and may be closed in order to isolate an inoperable relief valve or to remove from service a relief valve with excessive leakage. Discharge from the valves is directed to the pressurizer relief tank. A simplified P&ID of the pressurizer relief system is presented in Figure 3.2-19.

The pressurizer relief system has been modeled for bleed and feed cooling of the reactor following failure of secondary heat removal systems. The system has also been modeled for use in depressurization and stabilization of the RCS following a Steam Generator Tube Rupture. Finally, the successful opening of all safety valves and a variable number of PORVs (2 PORVs, 1 PORV, or 2 PORVs instead of one safety valve, depending upon initial reactor power and reactivity feedback conditions, manual trip, and feedwater flow) is required to mitigate the RCS pressure transient associated with the worst-case ATWS event.

Figure 3.2-19
Pressurizer Relief System
Simplified P&ID



3.2.2 System Analysis

Fault tree analysis was used to model the performance of plant systems in the WCGS PRA. These logic models depict the various combinations of hardware faults, human errors, test and maintenance unavailabilities, and other events that can lead to a failure to perform a given safety function. The definition of success for each fault tree is determined by the success criteria established for each event tree heading involving system performance.

Fault trees were developed for both frontline and support systems. Their analysis is conditional on both the initiating event (and its effects), and the availability of support systems that impact system operation. The support system availability is accounted for by linking the support trees into the frontline system fault trees.

The approach used to develop the fault tree models is consistent with the guidance provided in NUREG-2300. A set of fault tree guidelines was developed to ensure a consistent approach was used in establishing modeling assumptions and in structuring the models. They provide guidance in areas such as the selection of random hardware failures to model, treatment of test and maintenance outages, modeling of operator errors, and common cause failure analysis. The following provides a general overview of the fault tree construction process.

STEP 1 Develop Simplified P&ID

A simplified piping and instrumentation diagram (P&ID) was developed from the detailed plant drawings of the systems to provide the level of detail required for the modeling of the system. Support system interfaces, actuation signals, normal component position, etc., are identified on the simplified diagram. The plant drawings were reviewed and then simplified through the elimination of flow paths not directly related with the main process (such as fill and sampling lines). Small diversion flow paths which do not cause failure of the system were also removed. The original WCGS drawings from which the simplified diagrams were derived were documented in the system notebooks.

STEP 2 Develop Fault Tree

Step 2.1 Establish scope of fault tree - The fault tree guidelines were used to establish what modes and basic events should be modeled. They provide guidance to the analyst for selection of faults pertinent to random hardware failures, test outages, maintenance outages, human errors and common cause failures. In addition, they provide guidance on the exclusion of events which should not be included due to their low probability of occurrence relative to other events (e.g., passive failures like pipe ruptures).

Step 2.2 Use fault tree modules to develop fault tree - Fault tree modules served as logic building blocks in the construction of fault trees. In addition, they were used to simplify and standardize fault tree development layout. Modules were defined for the system level, the node level, the segment level, the component level, and the component interface level (actuation, electrical, etc.). The system level module was used to relate the system success criteria to fault logic. The node level modules served as input into the system level module and were applied to totally define the fault logic associated with the segments. Once the node level logic was developed and constructed, the next step was to establish the fault logic associated with each individual segment. This was accomplished using segment level modules which relate components to the segment. Finally, component level modules were used to further define fault contributions related to failure mode elements of each component identified in the segment level module. They relate to hardware failures, test and maintenance outages, operator error, actuation system failure, and support system interfaces (e.g., electrical, cooling).

The Fault Tree Guidelines define the step-by-step process in the development of the fault trees using the fault tree modules. General rules were applied to determine the node level modules to be used based on the system success criteria and flow requirements. The fault tree was developed graphically with the Westinghouse GRAFTER Code System.

STEP 3 Quantify Fault Tree

The fault trees developed in the previous task were quantified using the GRAFTER Code System to determine the system failure probability and to obtain the minimal cutsets. The calculational methods for quantifying the basic event probabilities that were input into the fault tree quantification are presented in the respective technical guideline. General calculational methods were described for hardware failures (both demand and time dependent), maintenance outages, test outages, human errors and common cause failures. A discussion of system mission times was provided and a basic event ID format was provided to maintain consistency within the analyses.

Step 3.1 Calculate basic event probabilities - Utilizing the component failure rates, test and maintenance unavailabilities and other basic event data, the basic event probabilities defined in the fault tree were quantified using the equations provided in the technical guidelines.

Step 3.2 Calculate human error probabilities - The human errors considered in the development of the fault trees and the human error probabilities used in the quantification of the fault trees were developed using the THERP methodology.

Step 3.3 Calculate common cause failure probabilities - Once a fault tree for a system was developed which included, as applicable, random hardware failures, test outages, maintenance outages and human errors, the important common cause component groups were identified for inclusion in the fault trees. The attributes that were used for the identification of common cause failure groups are:

- Component Type
- Component Use/Function (syst isolation, flow modulation, etc.)

- Component initial conditions (i.e., normally closed, initially running, etc.)

- Component failure mode

For each common cause component group identified, common cause events were added to the fault tree. Once all important common cause failures were identified, the Multiple Greek Letter method was used to calculate the common cause failure probability.

With the common cause failure probabilities input into the fault tree, the fault tree was quantified to determine the total system failure probability and to obtain the dominant contributors (cutsets) for the system.

STEP 4 Document Process

The entire process of fault tree development including key assumptions, boundary conditions, and other important information was documented in the associated system notebook. The quantification of the fault tree was also documented in the system notebook in the quantification section (including the computer code used and its input and output). The dominant contributors to system failure were identified and documented in the system notebook. Any key insights identified were also documented in the system notebook.

3.2.3 System Dependencies

The WCGS PRA uses fault tree linking to quantify the accident sequences. The fault tree linking method requires the development of a system fault tree for each of the frontline systems and for each of the support systems modeled. Each frontline system fault tree calls in the appropriate support system fault tree(s) and the linking process quantifies the accident sequences without double-counting support systems.

Table 3.2-3 identifies frontline systems or functions and the support systems on which they rely. Table 3.2-4 identifies support system interdependencies. The support systems modeled include AC power, DC power, component cooling water, essential service water, and RPS/ESFAS signals.

TABLE 3.2-3

FRONTLINE SYSTEM DEPENDENCY ON SUPPPORT SYSTEMS

Front-Line System/ Support System	4160 VAC	480 VAC	125 VDC	120 VAC	RPS/ESFAS	CCW	ESW
Auxiliary Feedwater	X	X	X		X		X
High Pressure Injection	X	X	X		X		X
High Pressure Recirculation	X	X	X		X	X	X
Low Pressure Injection	X	X	X		X		X
Low Pressure Recirculation	X	X			X	X	X
Containment Spray Injection	X	X	X		X		X
Containment Spray Recirculation	X	X	X		X		X
Main Steam			X		X		
Pzr PORVs & Safeties		X	X	X			
Containment Coolers (Fans)		X			X		X
Main Feed & Condensate		X	X				

TABLE 3.2-4

SUPPORT SYSTEM DEPENDENCY ON SUPPORT SYSTEMS

Support System/ Support System	4160 VAC	480 VAC	125 VDC	120 VAC	RPS/ESFAS	CCW	ESW
4160 VAC	-		X		X		X
480 VAC	X	-					
125 VDC			-				
120 VAC		X	X	-			
RPS/ESFAS			X	X	-		
CCW	X	X	X		X	-	X
ESW	X	X	X		X		-

3.3 Sequence Quantification

The following sections describe several analyses that support the quantification of the fault trees and the event trees. These supporting analyses include the generation of plant specific and generic component data, the generation of human error probabilities, the calculation methodology for common cause failure probabilities, the identification of any internal flooding initiating events, and the fault tree and event tree quantification processes.

3.3.1 List of Generic Data

Generic data formed the basis for many of the component failure rates used in the WCGS PRA. If a shortage of plant-specific data existed, generic values were utilized as either the actual failure rates or as the prior distributions for Bayesian updates. Generic failure rates were used for the majority of electrical components modeled in the WCGS PRA. Table 3.3-1 lists all of the generic component failure rates and their respective origins. The following sources were used for generic data:

1) NUREG/CR-4550, Volume 1, Revision 1, "Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines."

2) NUREG/CR-2815, "Probabilistic Safety Analysis Procedure Guide."

3) NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide."

4) IEEE Std., 500-1984, "IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component and Mechanical Equipment Reliability Data for Nuclear-Power Generating Stations."

5) NUREG-75/014, WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants."

NUREG/CR-4550 was the basis for the generic data used in the WCGS PRA. A' other sources were used sparingly to supplement NUREG/CR-4550.

3.3.2 Plant-Specific Data and Analysis

Plant specific data collection and analysis was performed very early in the PRA project. A list of components for which data was to be collected was generated based on the results of industry PRAs and on the opinions of utility personnel. The components for which data was collected include:

- Motor Driven Pumps:

  Auxiliary Feedwater
  Centrifugal Charging
  Essential Service Water
  Component Cooling Water
  Residual Heat Removal
  Safety Injection
  Containment Spray

- Turbine Driven Auxiliary Feedwater Pump

- Emergency Diesel Generators

- Motor Operated Valves

- Essential Service Water Traveling Screens

The boundary for which data was collected on these components includes: (1) pump/valve and motor hardware faults, (2) control circuitry faults, and (3) power circuitry faults (including the circuit breaker). In addition to these boundaries, the following support systems were included within the diesel generator data: lube oil system, intake and exhaust air system, and starting air accumulator. The speed governing valve is also included within the turbine driven AFW pump data.

Data was collected and compiled over the time period from the start of commercial operation, September 3, 1985, through December 31, 1989 for all components except motor-operated valves. The motor-operated valve data was collected from the start of commercial operation through December 31, 1988.

The primary source of operating data for the majority of the components was the control room operator logs, with the diesel generator data checked against the diesel generator start log. Failure data for the plant equipment was collected from work requests, work request logs, LERs, NPRDS, control room operator logs, and equipment out of service logs.

The WCGS PRA utilized the plant-specific component data to calculate failure rates through classical means or through the use of Bayesian techniques. Table 3.3-2 lists all of the plant-specific component failure rates and their respective calculations techniques. Those failure rates calculated using Bayesian techniques used generic prior failure rates from the sources identified in Section 3.3.1.

Determination of unavailabilities due to test, corrective maintenance events, and preventive maintenance procedures was done for the most part on a system train related basis. For the major components within a given system train, performance of any activities which result in unavailability of the component also generally result in unavailability of the associated train. The train unavailabilities, while in Mode 1, for each train in a system, were combined and an average train unavailability for each system was entered in the master data file. The specific system train unavailabilities along with the average train unavailability for each system are indicated in Table 3.3-3.

All of the plant specific component failure rates and system train test/maintenance unavailabilities are listed in Table 3.3-1.

TABLE 3.3-1
WOLF CREEK MASTER DATA TABLE

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MASTER DATA FILE FOR WCGS PRA ANALYSIS

#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
1	XX	ALL	LOGICAL ONE	1.000E+00	0.000E+00	D	NT01
2	XX	ALL	DUMMY PROBABILITY FOR SUB-BASIC EVENTS	1.000E-01	0.000E+00	D	NT02
3	XX	ALL	DUMMY PROBABILITY	1.000E-03	0.000E+00	D	
4	XX	ALL	LOGICAL ZERO	0.000E+00	0.000E+00	D	NT01
5	XX	ALL	DUMMY PROBABILITY FOR SUB-BASIC EVENTS	1.000E-01	0.000E+00	D	NT02
6	XX	ALL	DUMMY PROBABILITY FOR SUB-BASIC EVENTS	1.000E-01	0.000E+00	D	NT02
7	XX	ALL	DUMMY PROBABILITY	1.000E-01	0.000E+00	D	
8	XX	ALL	DUMMY PROBABILITY	1.000E-02	0.000E+00	D	
20	XX	ALL	LARGE LOCA IE FREQUENCY	5.000E-04	1.400E-07	D	NT03
21	XX	ALL	MEDIUM LOCA IE FREQUENCY	1.100E-03	6.800E-07	D	NT03
22	XX	ALL	SMALL LOCA IE FREQUENCY	2.500E-03	3.513E-06	D	NT03
23	XX	ALL	REACTOR VESSEL FAILURE IE FREQUENCY	3.000E-07	5.060E-14	D	NT03
24	XX	ALL	INTERFACING SYSTEMS LOCA IE FREQUENCY	6.110E-08	2.100E-15	D	NT03
25	XX	ALL	STEAM GENERATOR TUBE RUPTURE IE FREQUENCY	1.100E-02	6.800E-05	D	NT03
26	XX	ALL	TRANSIENTS WITH MAIN FEED WATER IE FREQUENCY	4.300E+00	0.000E+00	D	NT03
27	XX	ALL	TRANSIENTS WITHOUT MFW IE FREQUENCY	1.900E-01	2.030E-02	D	NT03
28	XX	ALL	LARGE STEAM/FEED LINE BREAK IE FREQUENCY	5.000E-04	1.400E-07	D	NT03
29	XX	ALL	LOSS OF OFFSITE POWER IE FREQUENCY	5.100E-02	1.462E-03	D	NT03
30	XX	ALL	STATION BLACKOUT IE FREQUENCY	2.324E-04	3.040E-08	D	NT03
31	XX	ALL	ATWS IE FREQUENCY	3.329E-05	6.230E-10	D	NT03
32	XX	ALL	TOTAL LOSS OF CCW SYSTEM IE FREQUENCY	1.620E-04	1.475E-08	D	RF06
33	XX	ALL	TOTAL LOSS OF ESW/SW SYSTEM IE FREQUENCY	1.759E-05	1.740E-10	D	RF09
34	XX	ALL	LOSS OF A VITAL 125 VOLT DC BUS IE FREQUENCY	1.784E-03	1.788E-06	D	RF07
35	XX	ALL	LOSS OF A SINGLE CCW SYSTEM TRAIN - IE FREQUENCY	1.134E-02	7.230E-05	D	RF06
40	MV	ALL	MOTOR-OPERATED VALVE FAILS TO OPEN ON DEMAND - PLANT SPECIFIC	3.709E-03	7.732E-06	D	NT04
41	MV	ALL	MOTOR-OPERATED VALVE FAILS TO CLOSE ON DEMAND - PLANT SPECIFIC	3.709E-03	7.732E-06	D	NT04
42	MV	ALL	MOTOR-OPERATED VALVE TRANSFERS CLOSED	1.000E-07	5.621E-15	HR	RF01
43	MV	ALL	MOTOR-OPERATED VALVE TRANSFERS OPEN	5.000E-07	1.523E-12	HR	RF01
44	MV	ALL	MOTOR-OPERATED VALVE PLUGGED	1.000E-07	5.621E-15	HR	RF01

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
45	MV	ALL	MOV CATASTROPHIC INTERNAL FAILURE	1.000E-07	2.500E-11	HR	RF02
46	MV	ALL	MOV FAILS TO STROKE OPEN OR CLOSED ON DEMAND - GEN ERIC	3.000E-03	5.485E-05	D	RF01
50	AV	ALL	AIR-OPERATED VALVE FAILS TO OPEN ON DEMAND	2.000E-03	2.248E-06	D	RF01
51	AV	ALL	AIR-OPERATED VALVE FAILS TO CLOSE ON DEMAND	2.000E-03	2.248E-06	D	RF01
52	AV	ALL	AIR-OPERATED VALVE TRANSFERS CLOSED	1.000E-07	5.621E-15	HR	RF01
53	AV	ALL	AIR-OPERATED VALVE TRANSFERS OPEN	5.000E-07	1.523E-12	HR	RF01
54	AV	ALL	AIR-OPERATED VALVE PLUGGED	1.000E-07	5.621E-15	HR	RF01
55	AV	ALL	AOV CATASTROPHIC INTERNAL FAILURE	1.000E-07	2.500E-11	HR	RF02
60	SV	ALL	SOLENOID-OPERATED VALVE FAILS TO OPEN ON DEMAND	2.000E-03	2.248E-06	D	RF01
61	SV	ALL	SOLENOID-OPERATED VALVE FAILS TO CLOSE ON DEMAND	2.000E-03	2.248E-06	D	RF01
62	SV	ALL	SOLENOID-OPERATED VALVE TRANSFERS CLOSED	2.000E-07	2.250E-14	HR	RF02
63	SV	ALL	SOLENOID-OPERATED VALVE TRANSFERS OPEN	2.000E-07	2.250E-14	HR	RF02
64	SV	ALL	SOLENOID-OPERATED VALVE PLUGGED	1.000E-07	5.621E-15	HR	RF01
65	SV	ALL	SOV CATASTROPHIC INTERNAL FAILURE	1.000E-07	2.500E-11	HR	RF02
70	HV	ALL	HYDRAULIC-OPERATED VALVE FAILS TO OPEN ON DEMAND	2.000E-03	2.248E-06	D	RF01
71	HV	ALL	HYDRAULIC-OPERATED VALVE FAILS TO CLOSE ON DEMAND	2.000E-03	2.248E-06	D	RF01
72	HV	ALL	HYDRAULIC-OPERATED VALVE TRANSFERS CLOSED	2.000E-07	2.250E-14	HR	RF02
73	HV	ALL	HYDRAULIC-OPERATED VALVE TRANSFERS OPEN	2.000E-07	2.250E-14	HR	RF02
74	HV	ALL	HYDRAULIC-OPERATED VALVE PLUGGED	1.000E-07	5.621E-15	HR	RF01
75	HV	ALL	HOV CATASTROPHIC INTERNAL FAILURE	1.000E-07	2.500E-11	HR	RF02
80	CV	ALL	CHECK VALVE FAILS TO OPEN	1.000E-04	5.621E-09	D	RF01
81	CV	ALL	CHECK VALVE FAILS TO CLOSE	1.000E-03	5.621E-07	D	RF01
82	CV	ALL	CHECK VALVE PLUGGED	4.000E-05	9.000E-10	D	RF01
83	CV	ALL	CHECK VALVE CATASTROPHIC INTERNAL FAILURE	1.000E-07	2.500E-11	HR	RF02
85	XV	ALL	MANUAL VALVE FAILS TO OPEN	1.000E-04	5.621E-09	D	RF01
86	XV	ALL	MANUAL VALVE FAILS TO CLOSE	1.000E-04	5.621E-09	D	RF03
87	XV	ALL	MANUAL VALVE PLUGGED	1.000E-07	5.621E-15	HR	RF01
88	XV	ALL	MANUAL VALVE FAILS TO REMAIN CLOSED	1.000E-04	5.621E-09	D	RF01
90	PV	ALL	PORV/SRV FAILS TO CLOSE	3.000E-02	5.500E-03	D	RF01
91	YV	ALL	SAFETY VALVE FAILS TO OPEN	1.000E-05	5.600E-11	D	RF03

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
92	YV	ALL	SAFETY VALVE FAILS TO CLOSE, ONCE OPEN	1.000E-02	6.100E-04	D	RF03
93	RV	ALL	RELIEF VALVE FAILS TO CLOSE, ONCE OPEN	2.000E-02	2.200E-04	D	RF03
94	RV	ALL	RELIEF VALVE FAILS TO OPEN	3.000E-04	5.500E-07	D	RF03
95	RV	ALL	RELIEF VALVE SPURIOUSLY OPENS	3.900E-06	9.269E-11	HR	RF01
97	MD	ALL	DAMPER FAILS TO OPERATE	3.000E-03	5.485E-05	D	RF01
98	AC	ALL	AIR COOLING UNIT FAILS TO OPERATE	6.000E-06	2.000E-11	HR	RF02
99	MF	ALL	MOTOR DRIVEN FAN FAILS TO START ON DEMAND	3.000E-04	5.058E-08	D	RF01
100	MF	ALL	MOTOR DRIVEN FAN FAILS TO RUN	1.000E-05	5.621E-11	HR	RF01
101	MP	ALL	MOTOR DRIVEN PUMP FAILS TO START ON DEMAND - PLANT SPECIFIC	1.764E-03	1.749E-06	D	NT04
102	MP	ALL	MOTOR DRIVEN PUMP FAILS TO RUN - PLANT SPECIFIC	2.730E-05	4.189E-10	HR	NT04
103	TP	AFW	TURBINE DRIVEN AFW PUMP FAILS TO START ON DEMAND - PLANT SPECIF	4.925E-03	2.176E-05	D	NT04
104	TP	AFW	TURBINE DRIVEN AFW PUMP FAILS TO RUN - PLANT SPECI FIC	4.400E-03	9.873E-06	HR	NT04
105	DP	ALL	DIESEL DRIVEN PUMP FAILS TO START ON DEMAND	3.000E-02	5.059E-04	D	RF01
106	DP	ALL	DIESEL DRIVEN PUMP FAILS TO RUN	8.000E-04	3.900E-06	HR	RF01
107	MP	ALL	MOTOR DRIVEN PUMP FAILS TO START - GENERIC	3.000E-03	5.485E-05	D	RF01
108	MP	ALL	MOTOR DRIVEN PUMP FAILS TO RUN - GENERIC	3.000E-05	5.485E-09	HR	RF01
120	HX	ALL	HEAT EXCHANGER BLOCKAGE	5.700E-06	2.100E-10	HR	RF01
121	HX	ALL	HEAT EXCHANGER FAILURE DUE TO LEAKAGE (TUBE LEAK)	3.000E-06	5.480E-11	HR	RF01
122	HX	ALL	HEAT EXCHANGER SHELL LEAK	3.000E-06	5.480E-11	HR	RF02
123	ST	ALL	STRAINER/FILTER PLUGGED	3.000E-05	5.480E-09	HR	RF02
124	OR	ALL	ORIFICE: FAILURE TO REMAIN OPEN (PLUGGED)	6.000E-07	2.000E-13	HR	RF02
125	OR	ALL	ORIFICE: RUPTURE	3.000E-08	5.480E-[illegible]	HR	RF02
126	AX	INSTAIR	INSTRUMENT AIR COMPRESSOR FAILS TO START ON DEMAND	8.000E-02	3.597E-03	D	RF01
127	AX	INSTAIR	INSTRUMENT AIR COMPRESSOR FAILS TO RUN	2.000E-04	2.430E-07	HR	RF01
128	FL	ESW	ESW SYSTEM TRAVELING SCREEN FAILS TO START ON DEMA ND - PLANT SPECIFIC	1.087E-03	1.664E-06	D	NT04
129	FL	ESW	ESW SYSTEM TRAVELING SCREEN FAILS TO RUN - PLANT S PECIFIC	2.227E-05	1.460E-09	HR	NT04
201	PP	LT3	UP TO 3 INCH DIAM. RUPTURE OR PLUG PER SECTION	8.500E-09	5.100E-15	HR	RF04
202	PP	GT3	MORE THAN 3 INCH DIAM. RUPTURE OR PLUG PER SECTION	8.500E-10	5.100E-17	HR	RF04

TABLE 3.3-1
WOLF CREEK MASTER DATA TABLE
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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
250	MP	ECCS	SAFETY INJECTION PUMPS SURVEILLANCE TEST UNAVAILABILITY - PLANT SPECIFIC	2.170E-04	2.600E-08	D	RF05
255	HX	ALL	HEAT EXCHANGER TEST AND MAINTENANCE UNAVAILABILITY - GENERIC	3.000E-05	5.480E-09	HR	RF01
258	MP	AFW	MOTOR DRIVEN AFW PUMPS SURVEILLANCE TEST UNAVAILABILITY - PLANT SPECIFIC	6.944E-04	2.710E-07	D	RF08
259	TP	AFW	TURBINE DRIVEN AFW PUMP SURVEILLANCE TEST UNAVAILABILITY - PLANT SPECIFIC	6.944E-04	2.710E-07	D	RF08
280	MP	ALL	MOTOR DRIVEN PUMP MAINTENANCE UNAVAILABILITY - GENERIC	2.000E-03	2.438E-05	D	RF01
281	TP	AFW	TURBINE DRIVEN AFW PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.813E-02	1.848E-04	D	NT05
282	DG	ACP	EMERGENCY DIESEL GENERATOR TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.130E-02	7.177E-05	D	NT05
283	MV	ALL	MOTOR-OPERATED VALVE TEST & MAINTENANCE UNAVAILABILITY - GENERIC	2.000E-04	2.248E-08	D	RF01
284	MP	AFW	MOTOR DRIVEN AFW PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.395E-02	1.094E-04	D	NT05
285	MP	ECCS	CENTRIFUGAL CHARGING PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.071E-02	6.448E-05	D	NT05
286	MP	CCW	COMPONENT COOLING WATER PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.027E-02	5.929E-05	D	NT05
287	MP	ESW	ESSENTIAL SERVICE WATER PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	1.206E-02	8.175E-05	D	NT05
288	MP	ECCS	RESIDUAL HEAT REMOVAL PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	6.380E-03	2.288E-05	D	NT05
289	MP	ECCS	SAFETY INJECTION PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	9.000E-03	4.553E-05	D	NT05
290	MP	CTMTSG	CONTAINMENT SPRAY PUMP TRAIN MAINTENANCE UNAVAILABILITY - PLANT SPECIFIC	2.670E-03	4.007E-06	D	NT05
351	XX	EPS	LOSS OF OFFSITE POWER FAILURE FOLLOWING AN INITIATING EVENT	2.000E-04	0.000E+00	D	RF01
352	DG	ACP	DIESEL GENERATOR FAILS TO START ON DEMAND - PLANT SPECIFIC	4.608E-03	1.194E-05	D	NT04
353	DG	ACP	DIESEL GENERATOR FAILS TO RUN - PLANT SPECIFIC	4.347E-03	3.804E-06	HR	NT04
354	BU	ALL	DC BUS FAILURE	1.000E-07	1.604E-14	HR	RF01
355	BT	ALL	BATTERY FAILURE	1.000E-06	5.621E-13	HR	RF01
356	BX	ALL	BATTERY CHARGER FAILURE	1.000E-06	5.621E-13	HR	RF01
357	IN	ALL	INVERTER FAILURE	1.000E-04	5.621E-09	HR	RF01

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
358	DC	ALL	DC MOTOR GENERATOR FAILS TO OPERATE	3.000E-06	5.500E-11	HR	RF02
359	CB	ALL	CIRCUIT BREAKER FAILS TO TRANSFER	3.000E-03	5.485E-05	D	RF01
360	CB	ALL	CIRCUIT BREAKER SPURIOUS TRIP	1.000E-06	5.621E-13	HR	RF01
361	FU	ALL	FUSE PREMATURELY OPENS	3.000E-06	5.500E-11	HR	RF02
362	BU	ALL	AC ELECTRICAL BUS FAILURE	1.000E-07	5.621E-15	HR	RF01
363	TR	ALL	TRANSFORMER FAILURE	2.000E-06	2.438E-11	HR	RF01
366	XX	ELECPWR	FAILURE PROBABILITY OF 480 V MCC	2.640E-05	3.918E-10	D	RF07
367	XX	ELECPWR	FAILURE PROBABILITY OF 480 V LOAD CENTER	9.840E-05	5.442E-09	D	RF07
380	SV	ALL	LIMIT SWITCH FAILS TO OPERATE	1.000E-04	5.600E-09	D	RF03
381	SP	ALL	PRESSURE SWITCH FAILS TO OPERATE	1.000E-04	5.600E-09	D	RF03
382	SR	ALL	MANUAL SWITCH FAILS TO TRANSFER	3.000E-05	5.500E-09	D	RF03
383	FS	ALL	FLOW SWITCH FAILS BY SPURIOUS OPERATION	1.838E-06	1.899E-12	HR	RF10
385	RE	ALL	RELAY FAILS TO TRANSFER OPEN OR CLOSED	3.000E-04	5.500E-07	D	RF03
386	RE	ALL	RELAY COIL FAILS (OPEN OR SHORT)	3.000E-06	5.500E-11	HR	RF02
387	RT	ALL	TIME DELAY RELAY PREMATURELY TRANSFERS	3.000E-04	5.480E-07	D	RF03
388	RT	ALL	TIME DELAY RELAY FAILS TO TRANSFER	3.000E-04	5.480E-07	HR	RF01
389	TT	ALL	TRANSMITTER FAILS TO OPERATE	1.000E-06	5.621E-13	HR	RF01
390	WR	ELECPWR	POWER CABLE FAILURE PER 1000 CIRCUIT FEET	2.830E-06	4.502E-12	HR	RF10
391	WR	ELECPWR	CONTROL CABLE FAILURE PER 1000 CIRCUIT FEE	3.460E-06	6.729E-12	HR	RF10
392	RP	ELECPWR	PROTECTIVE RELAY SPURIOUSLY OPERATES	3.600E-08	7.285E-16	HR	RF10
393	TS	ALL	TEMPERATURE SWITCH FAILS TO TRANSFER	1.000E-04	5.621E-09	D	RF01
395	EB	ALL	BISTABLE FAILS	3.000E-07	5.500E-13	D	RF03
396	ES	ALL	SOLID STATE DEVICE FAILS	3.000E-06	5.500E-11	HR	RF02
397	ET	ALL	TERMINAL BOARD FAILS OPEN OR SHORTS (PER TERMINAL)	3.000E-07	5.500E-13	HR	RF02
450	FU	ALL	FUSE OPENS PREMATURELY	3.000E-06	0.000E+00	HR	RF02
451	RE	ALL	RELAY MECHANICALLY BOUND	4.000E-07	0.000E+00	HR	TOP2
452	RE	ALL	RELAY CONTACTS FAIL TO OPEN/CLOSE	8.500E-06	0.000E+00	D	TOP2
453	RE	ALL	RELAY COIL SHORTED	1.000E-07	0.000E+00	HR	TOP2
454	RE	ALL	RELAY COIL OPEN	1.000E-08	0.000E+00	HR	TOP2
455	RE	ALL	RELAY FAILS DURING OPERATION, COMPOSITE OF 451, 45 3, 454	5.100E-07	0.000E+00	HR	TOP2

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
456	RE	ALL	RELAY CONTACTS SPURIOUSLY OPEN	8.700E-08	0.000E+00	HR	TOP3
457	PX	ALL	LOOP POWER SUPPLY FAILS	5.800E-06	0.000E+00	HR	TOPO
458	SB	ALL	SWITCH (PUSHBUTTON) FAILS	1.700E-07	0.000E+00	HR	TOPO
459	PT	ALL	PRESSURE SENSOR FAILS	2.800E-06	0.000E+00	HR	TOPO
460	EB	ALL	COMPARATOR (BISTABLE) FAILS	2.900E-06	0.000E+00	HR	TOPO
461	LT	ALL	LEVEL SENSOR FAILS	4.900E-0	0.000E+00	HR	TOPO
462	EC	ALL	LEAD/LAG AMPLIFIER FAILS	7.800E-07	0.000E+00	HR	TOP3
463	LG	ALL	UL CARD 3-INPUT CIRCUIT FAILS	4.600E-07	0.000E+00	HR	PORI
464	LG	ALL	UL CARD 4-INPUT CIRCUIT FAILS	9.790E-07	0.000E+00	HR	PORI
465	CA	SS	UV OUTPUT CARD FAILS	7.350E-07	0.000E+00	HR	PORI
466	CA	SF	SAFEGUARDS DRIVER CIRCUIT FAILS	3.960E-07	0.000E+00	HR	PORI
467	ES	SF	SAFEGUARDS DRIVER DIODE FAILS	7.560E-09	0.000E+00	HR	PORI
468	LG	ALL	UL CARD NAND CIRCUIT FAILS	2.0[illegible]E-07	0.000E+00	HR	PORI
469	CB	GEN	REACTOR TRIP BREAKER FAILS TO OPEN	6.890E-05	0.000E+00	D	TOPS
470	CB	GEN	BYPASS BREAKER FAILS TO OPEN	3.490E-04	0.000E+00	D	TOPS
471	CA	SA	NO ESFAS SIGNAL - ONE TRAIN AVAILABLE, NO OPERATOR ACTION (SIA,SIB)	5.490E-03	1.694E-05	D	RF14
472	CA	SA	NO ESFAS SIGNAL - BOTH TRAINS AVAILABLE, NO OPERAT OR ACTION (WESF)	6.698E-05	2.521E-09	D	RF14
473	CA	SA	NO ESFAS SIGNAL - ONE TRAIN AVAILABLE, WITH OPERAT OR ACTION (SIAO,SIBO)	3.904E-03	8.567E-06	D	RF14
474	CA	SA	NO ESFAS SIGNAL - BOTH TRAINS AVAILABLE, WITH OPER ATOR ACTION (WESFO)	3.187E-05	5.709E-10	D	RF14
475	CB	SS	REACTOR TRIP BREAKER MAINTENANCE UNAVAILABILITY	8.335E-07	0.000E+00	D	RF14
476	CB	SS	REACTOR TRIP BREAKER TEST UNAVAILABILITY	1.344E-03	0.000E+00	D	RF14
477	CB	SS	RPS LOGIC TESTING UNAVAILABILITY	1.344E-03	0.000E+00	D	RF14
478	CB	SS	SEQUENCER MAINTENANCE UNAVAILABILITY	8.580E-04	0.000E+00	D	RF14
479	CA	SF	LOCA SEQUENCER LOGIC FAILURE	4.480E-05	0.000E+00	HR	RF14
480	CA	SF	SHUTDOWN SEQUENCER LOGIC FAILURE	4.480E-05	0.000E+00	HR	RF14
481	CA	SF	UV MODULE UNAVAILABLE	2.380E-05	0.000E+00	HR	RF14
482	RE	ALL	COMMON CAUSE; 2 RELAYS FAIL DURING OPERATION: 451, 453, 454	3.570E-09	7.160E-18	HR	RF14
483	IN	ALL	COMMON CAUSE; TWO INVERTERS FAIL TO OPERATE	7.000E-07	2.754E-13	HR	RF14

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
484	CA	SS	COMMON CAUSE; RPS SYSTEM; WRT2 DIVERSE, NO OPERATOR ACTION	4.091E-06	9.410E-12	D	RF14
485	CA	SS	COMMON CAUSE; RPS SYSTEM; WRT20 DIVERSE, WITH OPERATOR ACTION	3.139E-07	5.539E-14	D	RF14
486	CA	SS	COMMON CAUSE; RPS SYSTEM; WRT1 NON-DIVERSE, NO OPERATOR ACTION	2.280E-05	2.922E-10	D	RF14
487	CA	SS	COMMON CAUSE; RPS SYSTEM; WRT10 NON-DIVERSE, WITH OPERATOR ACTION	5.684E-07	1.816E-13	D	RF14
488	CA	SF	COMMON CAUSE; ESFAS SYSTEM; WESF, NO OPERATOR ACTION	2.038E-05	2.335E-10	D	RF14
489	CA	SF	COMMON CAUSE; ESFAS SYSTEM; WESFO, WITH OPERATOR ACTION	2.079E-09	2.430E-18	D	RF14
490	CA	SF	COMMON CAUSE; LOCA SEQUENCERS FAIL	3.136E-07	5.528E-14	HR	RF14
491	CA	SF	COMMON CAUSE; SHUTDOWN SEQUENCERS FAIL	3.136E-07	5.528E-14	HR	RF14
492	HE	SS	FAIL TO MANUALLY TRIP THE REACTOR WITHIN 1 MINUTE (SAME AS 665)	1.360E-02	0.000E+00	D	HRA
493	HE	SF	FAIL TO MANUALLY ACTUATE SAFETY INJECTION WITHIN 20 MINUTES (SAME AS 680)	1.020E-04	0.000E+00	D	HRA
494	CA	NN/EPS	LOSS OF ONE CHANNEL OF 120 VOLT AC VITAL INSTRUMENT POWER	2.498E-03	3.507E-06	D	RF14
501	AV	ECCS/ALL	COMMON CAUSE FAILURE OF 2 OF 2 AIR OPERATED VALVES SPURIOUSLY CLOSE MT = 8.5 HRS	5.950E-09	1.990E-17	D	RF05
502	AV	ECCS/ALL	COMMON CAUSE FAILURE OF 2 OF 2 AIR OPERATED VALVES SPURIOUSLY CLOSE MT = 24.0 HR	1.680E-08	1.590E-16	D	RF05
503	MP	CCW/ESW	COMMON CAUSE FAILURE OF 2/2 CCW,ESW MOTOR-DRIVEN PUMPS TO START & RUN MT = 24HRS	2.904E-05	4.740E-10	D	RF06
504	AV	CCW/ALL	COMMON CAUSE FAILURE OF 4/4 AIR OPERATED VALVES TO CLOSE ON DEMAND,SPUR OPEN 24H	6.432E-06	2.325E-11	D	RF06
505	XX	CCW	COMMON CAUSE FAILURE OF CCW COMPTS BOTH TRAINS NOT MODELED AT COMPT LEVEL - LOCA	8.664E-09	4.220E-17	D	RF06
506	MP	CCW	COMMON CAUSE FAILURE OF 4/4 CCW MOTOR-DRIVEN PUMPS TO START & RUN MT = 24 HRS	5.808E-06	1.896E-11	D	RF06
508	XX	AFW	COMMON CAUSE FAILURES OF AUXILIARY FEEDWATER SYSTEM NOT AT COMPT LVL - AF2 MODEL	1.092E-06	6.703E-13	D	RF08
509	XX	AFW	COMMON CAUSE FAILURES OF AUXILIARY FEEDWATER SYSTEM NOT AT COMPT LVL - AF5 MODEL	1.396E-06	1.095E-10	D	RF08
510	XX	AFW	COMMON CAUSE FAILURES OF AUXILIARY FEEDWATER SYSTEM NOT AT COMPT LVL - AFT MODEL	1.437E-06	1.161E-12	D	RF08
511	CV	ECCS/SI3	COMMON CAUSE FAILURES OF 2/4, 3/4, 4/4 CHECK VALVES TO OPEN ON DEMAND - SI3	1.404E-06	1.110E-12	D	RF05

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
513	XX	ECCS/SI3	COMMON CAUSE FAILURES OF 2/4, 3/4, 4/4 CHECK VALVES TO PREVENT REVERSE FLOW- SI3	1.404E-05	1.110E-10	D	RF05
514	CV	AFW/ALL	COMMON CAUSE FAILURE OF 2/3 CHECK VALVES TO OPEN ON DEMAND	2.100E-07	2.479E-14	D	RF08
515	FL	ESW	COMMON CAUSE FAILURE OF 2/2 ESSENTIAL SERVICE WATER TRAVELING SCREENS TO ST/RUN	1.134E-05	7.230E-11	D	RF09
516	MF	ESW/ALL	COMMON CAUSE FAILURE OF 2/2 MOTOR DRIVEN FANS TO START/RUN - MT = 24HRS	6.480E-06	2.360E-11	D	RF09
518	AC	DCP/RPS	COMMON CAUSE FAILURE OF THE CLASS 1E ELECTRICAL EQUIPMENT AC UNITS TO RUN MT=24	1.728E-06	1.678E-12	D	RF07
519	AC	DCP/RPS	COMMON CAUSE FAILURE OF THE CLASS 1E HVAC UNITS TO START AND RUN MT = 24 HRS	5.328E-06	1.596E-11	D	RF07
520	MP	ECCS	COMMON CAUSE FAILURE OF 2 RUNNING RHR PUMPS - MT=23.5	2.953E-08	4.902E-12	D	RF05
521	MF	ECCS	COMMON CAUSE FAILURE OF 2 RUNNING ECCS PUMP ROOM FANS - MT=23.5	2.820E-06	4.470E-12	D	RF05
522	MV	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 MOVS BY SPURIOUSLY CLOSING MT = 23.5 HRS	8.930E-09	4.483E-17	D	RF05
523	CV	ECCS	COMMON CAUSE FAILURE OF 3 OF 3 CHECK VALVES TO OPEN ON DEMAND	1.600E-07	1.440E-14	D	RF05
524	CV	CCW/ALL	COMMON CAUSE FAILURE OF 2 OF 2 CHECK VALVES TO OPEN ON DEMAND	2.900E-07	4.727E-14	D	RF06
525	MP	ECCS	COMMON CAUSE FAILURE OF 2 RHR PUMPS TO START AND RUN - MT=0.5	8.188E-06	3.770E-11	D	RF05
526	MV	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 MOVS BY SPURIOUSLY CLOSING, MT=372.5 HRS	1.414E-07	1.124E-14	D	RF05
527	MF	ECCS	COMMON CAUSE FAILURE OF 2 STANDBY ECCS PUMP ROOM FANS TO START AND RUN, MT=0.5	3.660E-06	7.530E-12	D	RF05
528	XX	ESW/SWS	COMMON CAUSE FAILURE OF THE NORMAL SERVICE WATER SYSTEM NOT MODELED AT COMPT LVL	1.331E-06	9.958E-13	D	RF09
530	AV	AFW/SGPORV	COMMON CAUSE FAILURE OF STEAM GENERATOR PORVS - AF2, AF2WO MODELS	1.427E-05	1.145E-10	D	RF08
531	AV	AFW/SGPORV	COMMON CAUSE FAILURE OF STEAM GENERATOR PORVS - AF5, AF5A MODELS	2.465E-05	3.420E-10	D	RF08
532	AV	AFW/SGPORV	COMMON CAUSE FAILURE OF STEAM GENERATOR PORVS - AFT MODEL	1.945E-05	2.130E-10	D	RF08
535	XX	CTMTSFG	COMMON CAUSE FAILURE OF CONTAINMENT SPRAY SYSTEM - INJECTION MODE	5.600E-07	1.763E-13	D	RF13
536	XX	CTMTSFG	COMMON CAUSE FAILURE OF CONTAINMENT SPRAY SYSTEM - RECIRCULATION MODE	2.990E-07	5.030E-14	D	RF13

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
537	XX	CTMYSFG	COMMON CAUSE FAILURE OF CONTAINMENT COOLING SYSTEM NOT PLACED AT COMPONENT LEVEL	7.618E-09	3.260E-17	D	RF13
538	MF	CTMTSFG	COMMON CAUSE FAILURE OF HYDROGEN MIXING FANS	1.200E-10	8.094E-21	D	RF13
539	XX	IAIR	COMMON CAUSE FAILURE OF INSTRUMENT AIR SYSTEM EXCL UDING STANDBY COMPRESSORS	3.645E-06	7.47?E-12	D	
540	AX	IAIR	COMMON CAUSE FAILURE OF TWO STANDBY AIR COMPRESSOR S	6.752E-03	2.563E-05	D	
541	SV	PPR	COMMON CAUSE FAILURE OF BOTH PRESSURIZER PORVS	1.400E-05	1.102E-10	D	RF11
542	PV	PPR	COMMON CAUSE FAILURE OF PRESSURIZER SAFETY VALVES IN PPRC - ATWS EVENT	1.260E-07	8.924E-15	D	RF11
544	PV	AFA2/ATWS	COMMON CAUSE FAILURE OF 2 OUT OF 5 SAFETY RELIEF V ALVES ON ONE STEAM LINE - AFA3	1.800E-07	1.820E-14	D	RF08
545	PV	AFA2/ATWS	COMMON CAUSE FAILURE OF 4 OUT OF 5 SAFETY RELIEF V ALVES ON ONE STEAM LINE - AFA3	3.200E-08	5.760E-16	D	RF08
546	XX	AFA2/ATWS	COMMON CAUSE FAILURE OF AUXILIARY FEEDWATER SYSTEM IN AN ATWS EVENT - AFA2 MODEL	1.814E-06	1.850E-12	D	RF08
547	MP	AFW	COMMON CAUSE FAILURE OF 2 MOTOR DRIVEN AFW PUMPS T O START AND RUN 24 HOURS	3.630E-05	7.410E-10	D	RF08
550	AV	CCW/ALL	COMMON CAUSE FAILURE OF 2/4 AIR OPERATED VALVES TO CLOSE ON DEMAND,SPUR OPEN 24H	3.618E-06	7.358E-12	D	RF06
551	CV	CCW/ALL	COMMON CAUSE FAILURE OF 4 OUT OF 4 CHECK VALVES TO OPEN ON DEMAND	8.400E-08	3.966E-15	D	RF06
552	XX	MF1	COMMON CAUSE FAILURE OF MAIN FEEDWATER SYSTEM ON R ESTORATION (FW SYS)- MF1 MODEL	4.404E-05	1.090E-09	D	RF11
553	DG	DGS	COMMON CAUSE FAILURE OF TWO DIESEL GENERATORS TO S TART AND RUN MT = 2.5 HRS	1.078E-04	6.532E-09	D	RF07
554	XX	DGS	COMMON CAUSE FAILURE OF BOTH DG ROOM HVAC SYSTEMS MT = 2.5 HRS	3.190E-05	5.720E-10	D	RF07
555	XX	DGS	COMMON CAUSE FAILURE OF BOTH DG FUEL OIL SYSTEMS MT = 2.5 HRS	4.634E-05	1.207E-09	D	RF07
556	MV	ESW/ALL	COMMON CAUSE FAILURE OF 2/2 MOVS TO CLOSE ON DEMAN D/SPURIOUSLY OPEN MT = 24 HRS	1.414E-05	1.124E-10	D	RF09
557	MV	ESW/ALL	COMMON CAUSE FAILURE OF 2/2 MOVS TO OPEN ON DEMAND /SPURIOUSLY CLOSE MT = 24 HRS	1.410E-05	1.118E-10	D	RF09
558	MV	ESW/ALL	COMMON CAUSE FAILURE OF 4/4 MOVS TO CLOSE ON DEMAN D/SPURIOUSLY OPEN MT = 24 HRS	2.902E-06	4.734E-12	D	RF09
559	HV	ESW/ALL	COMMON CAUSE FAILURE OF 2/2 HOVS BY SPURIOUSLY OPE RATING MT = 24 HRS	3.360E-08	6.346E-16	D	RF09
560	FL	SWS	COMMON CAUSE FAILURE OF 2/4 SERVICE WATER TRAVELIN G SCREENS TO START/RUN MT=12HR	6.048E-06	2.056E-11	D	RF09

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
561	FL	SWS	COMMON CAUSE FAILURE OF 4/4 SERVICE WATER TRAVELIN G SCREENS TO START/RUN MT=12HR	1.075E-05	6.496E-11	D	RF09
562	MP	SWS/ALL	COMMON CAUSE FAILURE OF 2/2 MOTOR DRIVEN WS PUMPS (GENERIC) TO RUN MT=24 HRS	1.080E-05	6.556E-11	D	RF09
563	MV	SWS/ALL	COMMON CAUSE FAILURE OF 2/2 MO VALVES (GENERIC) BY SPURIOUSLY CLOSING MT= 24 HRS	9.120E-09	4.675E-17	D	RF09
564	XX	MSIVS	COMMON CAUSE FAILURE OF MAIN STEAM ISOLATION IN AN SGTR EVENT - MS1 MODEL	2.072E-05	2.413E-10	D	RF11
565	XX	AFW	COMMON CAUSE FAILURE OF AUXILIARY FEEDWATER SYSTEM NOT AT CMPT LEVEL - AF7 MODEL	1.595E-06	1.615E-12	D	RF08
566	AV	AFW/SGPORV	COMMON CAUSE FAILURE OF STEAM GENERATOR PORVS - AF 7 MODEL	2.465E-05	3.420E-10	D	RF08
567	CV	ECCS/LC3	COMMON CAUSE FAILURE OF 2/2 CHECK VALVES TO CLOSE ON DEMAND	2.900E-06	4.727E-12	D	RF05
568	XX	AFW/AF5A	COMMON CAUSE FAILURE OF AUXILIARY FEEDWATER SYSTEM NOT AT COMPONENT LEVEL - AF5A	2.433E-06	3.327E-12	D	RF08
569	AV	ECCS/ALL	COMMON CAUSE FAILURE OF 2 OF 2 AIR OPERATED VALVES SPURIOUSLY OPEN MT = 24 HRS	8.400E-08	3.970E-15	D	RF05
570	MV	MISC/ALL	COMMON CAUSE FAILURE OF 2/4 MOTOR OPERATED VALVES TO OPEN/SPURS CLOSE MT=24 HRS	6.696E-06	2.520E-11	D	RF11
571	XX	ECCS/EC3	COMMON CAUSE FAILURE OF RHR SYSTEM NOT MODELED AT COMP LEVEL IN EC3, ES1 MODELS	3.853E-05	8.345E-10	D	RF11
572	MV	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 MOVS BY SPURIOUSLY CLOSING, MT = 8.5 HRS	3.230E-09	5.860E-18	D	RF05
573	MP	ECCS	COMMON CAUSE FAILURE OF 2/2 RHR PUMPS TO START AND RUN, MT = 23.5 HRS	1.104E-05	6.851E-11	D	RF05
574	MV	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 MOVS BY SPURIOUSLY OPENING MT = 24 HRS	4.560E-08	1.169E-15	D	RF05
575	XX	AFW	COMMON CAUSE FAILURE OF MSIVS TO OPEN TO ACCESS ST EAM DUMPS - AF2 MODEL	4.348E-05	1.062E-09	D	RF08
576	MP	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 HIGH PRESSURE SI/CC P PUMPS TO START/RUN MT=0.5 HR	[illegible].696E-05	1.823E-09	D	RF05
577	MP	ECCS	COMMON CAUSE FAILURE OF 2 OF 2 HIGH PRESSURE SI/CC P PUMPS TO RUN MT=23.5 HRS	2.054E-05	2.371E-10	D	RF05
578	CV	ECCS	COMMON CAUSE FAILURE OF ACCUMULATOR SAFETY INJECTI ON CHECK VALVES	4.164E-06	9.750E-12	D	RF05
579	AV	AFW	COMMON CAUSE FAILURE OF 2/2 AOVS TO OPEN, SPURIOUS LY CLOSE MT = 24 HOURS	1.400E-05	1.102E-10	D	RF08
580	XX	ESW/SWS	COMMON CAUSE FAILURE OF SERVICE WATER SYS AFTER SB O EVENT, NOT AT COMPT LEVEL	8.940E-05	4.492E-09	D	RF09

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
581	AV	CCW/ALL	COMMON CAUSE FAILURE OF 4 OF 4 AIR OPERATED VALVES SPURIOUSLY OPEN MT = 24 HRS	3.840E-08	8.289E-16	D	RF06
582	XX	CCW	COMMON CAUSE FAILURE OF CCW COMPTS BOTH TRAINS NOT MODELED AT COMPT LEVEL - LSP	3.585E-07	7.224E-14	D	RF06
583	MV	AFW/ALL	COMMON CAUSE FAILURE O 2/2 MOVS BY SPURIOUSLY CLOSING MT = 32 HOURS	1.216E-08	8.312E-17	D	RF08
584	CV	AFW	COMMON CAUSE FAILURE OF STEAM SUPPLY (FC) CHECK VALVES	1.124E-06	7.100E-13	D	RF08
585	MV	AFW/ALL	COMMON CAUSE FAILURE OF 2/3 MOVS BY SPURIOUSLY CLOSING MT = 32 HOURS	7.680E-09	3.320E-17	D	RF08
588	XX	ECCS/LTS	COMMON CAUSE FAILURE OF 2 OF 2 BORIC ACID TRANSFER PUMPS - (LTS MODEL)	5.580E-05	1.750E-09	D	RF05
589	XX	MSIVS	COMMON CAUSE FAILURE OF MAIN STEAM ISOLATION IN AN SLB EVENT - MS2 MODEL	4.238E-05	1.009E-09	D	RF11
590	XX	MF1/CONDS	COMMON CAUSE FAILURE OF CONDENSATE SYSTEM ON FW RESTORATION - MF1 MODEL	1.865E-05	1.955E-10	D	RF11
591	XX	AFW/AF2	COMMON CAUSE FAILURF OF N2 GAS ACCUM FOR SG ARVS IN AF2 SERIES MODELS	7.530E-06	3.187E-11	D	RF08
592	XX	AFW/AF5,7	COMMON CAUSE FAILURE OF N2 GAS ACCUM FOR SG ARVS IN AF5 (AND AF7) SERIES MODELS	1.087E-05	6.642E-11	D	RF08
593	CV	ECCS/SI2	COMMON CAUSE FAILURE OF 2/3 & 3/3 CHECK VALVES TO OPEN ON DEMAND - SI2 MODEL	7.900E-07	3.510E-13	D	RF05
594	CV	ECCS/SI2	COMMON CAUSE FAILURE OF 2/3 & 3/3 CHECK VALVES TO CLOSE - SI2 MODEL	7.900E-06	3.510E-11	D	RF05
651	HE	OPA-OP1	HUMAN ERROR FAILURE TO COOLDOWN AND DEPRESSURIZE THE RCS - MLO EVENT, HPSI FAILS	1.730E-04	1.682E-08	D	RF12
652	HE	OPA-OP2	HUMAN ERROR FAILURE TO COOLDOWN AND DEPRESSURIZE THE RCS - SLO EVENT, HPSI FAILS	1.090E-04	6.678E-09	D	RF12
653	HE	OPA-ESWA/B	HUMAN ERROR FAILURE TO REALIGN AN ESW TRAIN TO SERVICE WATER WHEN BOTH ESW FAIL	1.100E-03	6.801E-07	D	RF12
654	HE	OPA-RRI22	HUMAN ERROR FAILURE TO RESTORE RCS INVENTORY IN AN SBO EVENT 2/2 PUMPS - 2 TRNS	7.350E-03	3.037E-05	D	RF12
655	HE	OPA-AFC	HUMAN ERROR FAILURE TO MAINTAIN AFWS COOLING FLOW DURING STATION BLACKOUT	1.360E-04	1.040E-08	D	RF12
656	HE	OPA-RRI12	HUMAN ERROR FAILURE TO RESTORE RCS INVENTORY IN AN SBO EVENT 1/4 PUMPS - 2 TRNS	9.880E-04	5.480E-07	D	RF12
659	HE	OPA-EC3	HUMAN ERROR FAILURE TO COOLDOWN AND DEPRESSURIZE THE RCS IN A SGTR EVENT	2.150E-04	2.598E-08	D	RF12
660	HE	OPA-ES1	HUMAN ERROR FAILURE TO COOLDOWN AND DEPRESSURIZE THE RCS IN A SLO EVENT	1.350E-04	1.024E-08	D	RF12

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
661	HE	OPA-LC1	HUMAN ERROR FAILURE TO SWITCHOVER TO LOW PRESSURE COLD LEG RECIRCULATION	9.170E-04	4.720E-07	D	RF12
662	HE	OPA-LC2,3	HUMAN ERROR FAILURE TO SWITCHOVER TO HIGH PRESSURE COLD LEG RECIRCULATION	4.970E-04	1.380E-07	D	RF12
663	HE	OPA-LTS	HUMAN ERROR FAILURE TO PERFORM LONG TERM SHUTDOWN IN AN ATWS EVENT	4.380E-05	1.070E-09	D	RF12
664	HE	OPA-MF1	HUMAN ERROR FAILURE TO ESTABLISH MAIN FEEDWATER SYSTEM OPERATION	6.020E-04	2.030E-07	D	RF12
665	HE	OPA-RT	HUMAN ERROR FAILURE TO MANUALLY TRIP REACTOR/INSERT CONTROL RODS IN ATWS EVENT	1.360E-02	1.040E-04	D	RF12
666	HE	OPA-MRT	HUMAN ERROR FAILURE TO MANUALLY OPEN THE ROD DRIVE MOTOR GENERATOR BREAKERS	1.550E-01	1.350E-02	D	RF12
667	HE	OPA-SGTR	HUMAN ERROR FAILURE TO DIAGNOSE THAT AN SGTR HAS OCCURED (MS1)	6.360E-06	2.274E-11	D	RF12
668	HE	OPA-RUPSG	HUMAN ERROR FAILURE TO IDENTIFY THE RUPTURED STEAM GENERATOR IN AN SGTR EVENT	6.330E-05	2.252E-09	D	RF12
669	HE	OPA-MS1	HUMAN ERROR FAILURE TO ISOLATE THE RUPTURED STEAM GENERATOR IN AN SGTR EVENT	1.590E-03	1.421E-06	D	RF12
670	HE	OPA-OD1	HUMAN ERROR FAILURE TO ESTABLISH RCS COOLDOWN IN AN SGTR EVENT (OD1-COOL)	5.550E-03	1.731E-05	D	RF12
671	HE	OPA-OD1	HUMAN ERROR FAILURE TO DEPRESSURIZE THE RCS IN AN SGTR EVENT (OD1-DP)	9.550E-03	5.126E-05	D	RF12
672	HE	OPA-OFB/C2	HUMAN ERROR FAILURE TO PERFORM FEED AND BLEED (OFB & OFC)	1.760E-03	1.741E-06	D	RF12
673	HE	OPA-OST	HUMAN ERROR FAILURE TO TERMINATE HIGH PRESSURE SAFETY INJECTION IN A SL/FL BREAK	5.780E-05	1.877E-09	D	RF12
674	HE	OPA-RCD1	HUMAN ERROR FAILURE TO PERFORM RCS COOLDOWN IN A STATION BLACKOUT EVENT	1.660E-03	1.548E-06	D	RF12
675	HE	OPA-RRI11	HUMAN ERROR FAILURE TO RESTORE RCS INVENTORY IN A SBO, CCW OR SWS EVENT - 1 TRN	5.640E-03	1.788E-05	D	RF12
676	HE	OPA-RCD2	HUMAN ERROR FAILURE TO PERFORM RCS COOLDOWN IN A LOSS OF CCW EVENT	1.320E-04	9.794E-09	D	RF12
677	HE	OPA-ALHVS	HUMAN ERROR FAILURE TO REOPEN AO VALVE AFTER STS AL-103 PERFORMANCE	8.290E-04	3.860E-07	D	RF12
678	HE	OPA-CSS	HUMAN ERROR FAILURE TO SWITCH CONTAINMENT SPRAY SUCTION TO RECIRCULATION SUMP	1.700E-03	1.624E-06	D	RF12
679	HE	OPA-RHR	HUMAN ERROR FAILURE TO STOP RHR PUMPS DURING INJECTION MODE OF HIGH PRESSURE EVT	5.010E-04	1.410E-07	D	RF12
681	HE	OPA-OD2	HUMAN ERROR FAILURE TO COOLDOWN AND DEPRESSURIZE THE RCS IN SGTR AFTER OVERFILL	2.900E-03	4.727E-06	D	RF12

TABLE 3.3-1
WOLF CREEK MASTER DATA TABLE

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
682	HE	OPA-MCB	HUMAN ERROR FAILURE TO OPERATE COMPONENT(S) FROM MCB CONTRO'S/AUTO SIGNAL FAILS	1.590E-03	1.421E-06	D	RF12
683	HE	OPA-DCNK01	HUMAN ERROR FAILURE TO LOCALLY OPERATE COMPONENT(S) ON LOSS OF DC BUS/OFN 00-020	4.770E-03	1.279E-05	D	RF12
684	HE	OPA-ABVLS	HUMAN ERROR FAILURE TO REOPEN ISOL VALVES AFTER STS AB-201 PERFORMANCE	3.740E-04	7.862E-08	D	RF12
685	HE	OPA-WS1C2	HUMAN ERROR FAILURE TO START STANDBY SERVICE WATER PUMP(S) ON LOW WS HEADER PRES	1.480E-04	1.231E-08	D	RF12
686	HE	OPA-ESWAB	HUMAN ERROR FAILURE TO START & ALIGN ESW TRAIN(S) ON WS SYSTEM FAILURE (IEV)	2.400E-04	3.237E-08	D	RF12
687	HE	OPA-SWSIEV	HUMAN ERROR FAILURE TO REALIGN FAILED ESW TRAIN TO WS SYSTEM AFTER LINE RUPTURE	2.18[illegible]E-02	2.623E-04	D	RF12
688	HE	OPA-PEG1C	HUMAN ERROR FAILURE TO START CCW PUMP PEG01C AFTER PEG01A & AUTO START FAILS	6.040E-03	2.051E-05	D	RF12
689	HE	OPA-PEG1BD	HUMAN ERROR FAILURE TO START CCW PUMPS PEG01B OR D ON CCW TRAIN A FAILURE	9.420E-03	4.990E-05	D	RF12
690	HE	OPA-RCPSEL	HUMAN ERROR FAILURE TO ALIGN CCW SERVICE LOOP OR START SEAL INJECTION	1.430E-02	1.150E-04	D	RF12
691	HE	OPA-1EHVAC	HUMAN ERROR FAILURE TO PROVIDE VENTILATION TO DC SWITCHBOARD ROOMS	1.000E-01	0.000E+00	D	RF12
692	HE	OPA-RCPTRP	HUMAN ERROR FAILURE TO MANUALLY TRIP THE RCPS ON LOSS OF SEAL COOLING	2.390E-03	3.210E-06	D	RF12
693	HE	OPA-ECCSTP	HUMAN ERROR FAILURE TO MANUALLY TRIP ECCS PUMPS ON LOSS OF COOLING WATER SUPPLY	1.000E-02	0.000E+00	D	RF12
694	HE	OPA-ACNN	HUMAN ERROR FAILURE TO TRANSFER NN SWTCHBOARDS TO ALTERNATE POWER SOURCE	2.840E-02	4.534E-04	D	RF12
701	XX	ECCS	FAILURE OF ONE HIGH PRESSURE ECCS TRAIN DURING INJECTION MODE	1.597E-02	1.434E-04	D	RF05
702	XX	ECCS	FAILURE OF TWO HIGH PRESSURE ECCS TRAINS DURING INJECTION MODE	2.088E-04	2.450E-08	D	RF05
703	XX	ECCS	FAILURE OF THREE HIGH PRESSURE ECCS TRAINS DURING INJECTION MODE	1.254E-06	8.840E-13	D	RF05
704	XX	ECCS	FAILURE OF ONE HIGH PRESSURE ECCS TRAIN DURING RECIRCULATION MODE	9.800E-04	5.390E-07	D	RF05
705	XX	ECCS	FAILURE OF TWO HIGH PRESSURE ECCS TRAINS DURING RECIRCULATION MODE	2.436E-05	3.336E-10	D	RF05
706	XX	[illegible]	FAILURE OF THREE HIGH PRESSURE ECCS TRAINS DURING RECIRCULATION MODE	2.387E-08	3.203E-16	D	RF05
707	XX	ESW	FAILURE OF EITHER TRAIN OF THE ESSENTIAL SERVICE WATER SYSTEM	5.203E-04	1.521E-07	D	RF09

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
708	XX	SWS/ESW	FAILURE OF THE NORMAL PLANT SERVICE WATER SYSTEM	7.061E-03	2.803E-05	D	RF09
709	MV	PRV	FRACTION OF MODE 1 OPERATION TIME WITH ONE PRESSURIZER PORV BLOCK VA[illegible] CLOSED	1.900E-01	0.000E+00	[illegible]	RF11
710	MV	PRV	FRACTION OF MODE 1 [illegible]ERATION TIME WITH BOTH PRESSURIZER PORV BLOCK VALVES OPEN	8.100E-01	0.000E+00	D	RF11
711	AV	AFW	FRACTION OF MODE 1 OPERATION TIME WITH ONE STEAM GENERATOR PORV ISOLATED	2.700E-02	0.000E+00	D	RF11
712	AV	AFW	FRACTION OF MODE 1 OPERATION TIME WITH NO STEAM GENERATOR PORVS ISOLATED	9.730E-01	0.000E+00	D	RF11
713	XX	ECCS/RRI2	FAILURE OF BOTH CCP TRAINS TO DELIVER FLOW - RRI2 W/O PORVS,OPA;SUBS SET TO ZERO	1.392E-03	1.089E-06	D	RF05
714	RV	SGR/SSV	PROBABILITY OF STUCK OPEN SG SAFETY VALVE ONCE CHALLENGED AFTER SG OVERFILL -SGR	5.000E-01	0.000E+00	D	RF11
715	XX	ATW/PLV	FRACTION OF OPERATING TIME (MODES 1 + 2) WITH POWER LEVEL GREATER THAN 40 %	9.860E-01	0.000E+00	D	RF11
716	XX	ATW/SUCPLV	FRACTION OF OPERATING TIME (MODES 1 + 2) WITH POWER LEVEL LESS [illegible] 40 %	1.400E-02	0.000E+00	D	RF11
[illegible]17	RV	ATW/PPR1	PRESSURIZER [illegible] [illegible]ELIEF - ATWS -PPR1 TOP EVENT	1.925E-05	2.083E-10	D	RF11
718	RV	ATW/PPR2	PRESSURIZER PRESSURE RELIEF - ATWS -PPR2 TOP EVENT	2.583E-04	3.750E-08	D	RF11
719	RV	ATW/PPR3	PRESSURIZER PRESSURE RELIEF - ATWS -PPR3 TOP EVENT	1.652[illegible]	[illegible].534E-02	D	RF11
720	RV	ATW/PPR4	PRESSURIZER PRESSURE RELIEF - ATWS -PPR4 TOP EVENT	2.234E-0[illegible]	2.810E-02	D	RF11
721	XX	ATW/AMS	FAILURE OF AMSAC SYSTEM IN ATWS EVENT TREE	1.000E-02	5.621E-05	D	RF11
722	XX	SBO/8HR	FAILURE OF JOINT RECOVERY OF AC POWER WITHIN 8 HOURS AFTER AN SBO EVENT	2.540E-02	0.000E+00	D	RF11
723	XX	SBO/8HR	SUCCESS OF JOINT RECOVERY OF AC POWER WITHIN 8 HOURS AFTER AN SBO EVENT	9.750E-01	0.000E+00	D	RF11
724	XX	SBO/8HR	CORE UNCOVERY OCCURS AFTER SBO - AC POWER IN 8 HOURS - RCS COOLDOWN SUCCESSFUL	1.270E-02	0.000E+00	D	RF11
725	XX	SBO/8HR	CORE DOES NOT UNCOVER AFTER SBO - AC POWER IN 8 HOURS - RCS COOLDOWN SUCCESSFUL	9.870E-01	0.000E+00	D	RF11
726	XX	SBO/8HR	CORE UNCOVERY OCCURS AFTER SBO - AC POWER IN 8 HOURS - RCS COOLDOWN FAILS	1.890E-02	0.000E+00	D	RF11
727	XX	SBO/8HR	CORE DOES NOT UNCOVER AFTER SBO - AC POWER IN 8 HOURS - RCS COOLDOWN FAILS	9.810E-01	0.000E+00	D	RF11
728	XX	SBO/8HR	FRACTION OF JOINT 8 HR AC POWER RECOVERY DUE TO OFFSITE POWER - RCS CD SUCCESS	9.490E-01	0.000E+00	D	RF11
729	XX	SBO/8HR	FRACTION OF JOINT 8 HR AC POWER RECOVERY DUE TO DIESEL GEN - RCS CD SUCCESSFUL	5.060E-02	0.000E+00	D	RF11

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
730	XX	SBO/8HR	FRACTION OF JOINT 8 HR AC POWER RECOVERY DUE TO OF FSITE POWER - RCS CD FAILS	9.405E-01	0.000E+00	D	RF11
731	XX	SBO/8HR	FRACTION OF JOINT 8 HR AC POWER RECOVERY DUE TO DI ESEL GEN - RCS CD FAILS	[illegible]	0.000E+00	D	RF11
732	XX	SBO/4HR	FAILURE OF JOINT RECOVERY OF AC POWER WITHIN 4 HOU RS AFTER AN SBO EVENT	[illegible]-02	0.000E+00	D	RF11
733	XX	SBO/4HR	SUCCESS OF JOINT RECOVERY OF AC POWER WITHIN 4 HOU RS AFTER AN SBO EVENT	9.460E-01	0.000E+00	D	RF11
734	XX	SBO/4HR	CORE UNCOVERY OCCURS AFTER SBO - AC POWER IN 4 HOU RS - RCS COOLDOWN SUCCESSFUL	5.560E-03	0.000E+00	D	RF11
735	XX	SBO/4HR	CORE DOES NOT UNCOVER AFTER SBO - AC POWER IN 4 HO URS - RCS COOLDOWN SUCCESSFUL	9.940E-01	0.000E+00	D	RF11
736	XX	SBO/4HR	CORE UNCOVERY OCCURS AFTER SBO - AC POWER IN 4 HOU RS - RCS COOLDOWN FAILS	6.060E-03	0.000E+00	D	RF11
737	XX	SBO/4HR	CORE DOES NOT UNCOVER AFTER SBO - AC POWER IN 4 HO URS - RCS COOLDOWN FAILS	9.940E-01	0.000E+00	D	RF11
738	XX	SBO/4HR	FRACTION OF JOINT 4 HR AC POWER RECOVERY DUE TO OF FSITE POWER - RCS CD SUCCESS	9.170E-01	0.000E+00	D	RF11
739	XX	SBO/4HR	FRACTION OF JOINT 4 HR AC POWER RECOVERY DUE TO DI ESEL GEN - RCS CD SUCCESSFUL	8.340E-02	0.000E+00	D	RF11
740	XX	SBO/4	FRACTION OF JOINT 4 HR AC POWER RECOVERY DUE TO OF FSITE POWER - RCS CD FAILS	9.160E-01	0.000E+00	D	RF11
741	XX	SBO/4HR	FRACTION OF JOINT 4 HR AC POWER RECOVERY DUE TO DI ESEL GEN - RCS CD FAILS	8.390E-02	0.000E+00	D	RF11
742	XX	SBO/2HR	FAILURE OF JOINT RECOVERY OF AC POWER WITHIN 2 HOU RS AFTER AN SBO EVENT	1.150E-01	0.000E+00	D	RF11
743	XX	SBO/2HR	SUCCESS OF JOINT RECOVERY OF AC POWER WITHIN 2 HOU RS AFTER AN SBO EVENT	8.850E-01	0.000E+00	D	RF11
744	XX	SBO/2HR	CORE UNCOVERY OCCURS AFTER SBO - AC POWER IN 2 HOU RS - RCS COOLDOWN FAILS	3.980E-03	0.000E+00	D	RF11
745	XX	SBO/2HR	CORE DOES NOT UNCOVER AFTER SBO - AC POWER IN 2 HO URS - RCS COOLDOWN FAILS	9.960E-01	0.000E+00	D	RF11
746	XX	SBO/2HR	FRACTION OF JOINT 2 HR AC POWER RECOVERY DUE TO OF FSITE POWER - RCS CD FAILS	8.290E-01	0.000E+00	D	RF11
747	XX	SBO/2HR	FRACTION OF JOINT 2 HR AC POWER RECOVERY DUE TO DI ESEL GEN - RCS CD FAILS	1.710E-01	0.000E+00	D	RF11
748	XX	CCW/8HR	FAILURE OF CCW RECOVERY WITHIN 8 HOURS AFTER A LOS S OF CCW EVENT	1.910E-02	0.000E+00	D	RF11
749	XX	CCW/8HR	SUCCESS OF CCW RECOVERY WITHIN 8 HOURS AFTER A LOS S OF CCW EVENT	9.810E-01	0.000E+00	D	RF11

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#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
750	XX	CSG	FAILURE PROBABILITY OF ALL CLASS A PENETRATIONS TA KEN COLLECTIVELY	1.200E-04	8.094E-09	D	RF13
751	XX	CCW/8HR	CORE UNCOVERY OCCURS AFTER CCW - CCW RESTORED IN 8 HOURS - RCS COOLDOWN SUCCESS	1.920E-02	0.000E+00	D	RF11
752	XX	CCW/8HR	CORE DOES NOT UNCOVER AFTER CCW - CCW RESTORED IN 8 HOURS - RCS COOLDOWN SUCCESS	9.810E-01	0.000E+00	D	RF11
753	XX	CCW/8HR	CORE UNCOVERY OCCURS AFTER CCW - CCW RESTORED IN 8 HOURS - RCS COOLDOWN FAILS	2.660E-02	0.000E+00	D	RF11
754	XX	CCW/8HR	CO[illegible] NOT UNCOVER AFTER CCW - CCW RESTORED IN 8 HOURS - RCS COOLDOWN FAILS	9.730E-01	0.000E+00	D	RF11
755	XX	CCW/2HR	FAILURE OF CCW RECOVERY WITHIN 2 HOURS AFTER A LOS S OF CCW EVENT	9.380E-02	0.000E+00	D	RF11
756	XX	CCW/2HR	SUCCESS OF CCW RECOVERY WITHIN 2 HOURS AFTER A LOS S OF CCW EVENT	9.060E-01	0.000E+00	D	RF11
757	XX	CCW/2HR	CORE UNCOVERY OCCURS AFTER CCW - CCW RESTORED IN 2 HOURS - RCS COOLDOWN FAILS	1.170E-02	0.000E+00	D	RF11
758	XX	CCW/2HR	CORE DOES NOT UNCOVER AFTER CCW - CCW RESTORED IN 2 HOURS - RCS COOLDOWN FAILS	9.880E-01	[illegible]E+00	D	RF11
759	XX	SWS/8HR	FAILURE OF SW RECOVERY WITHIN 8 HOURS AFTER A LOSS OF ALL SERVICE WATER EVENT	1.660E-02	0.000E+00	D	RF11
760	XX	SWS/8HR	SUCCESS OF SW RECOVERY WITHIN 8 HOURS AFTER A LOSS OF ALL SERVICE WATER EVENT	9.830E-01	0.000E+00	D	RF11
761	XX	SWS/8HR	CORE UNCOVERY OCCURS AFTER SWS - SW RESTORED IN 8 HOURS - RCS COOLDOWN SUCCESS	2.810E-02	0.000E+00	D	RF11
762	XX	SWS/8HR	CORE DOES NOT UNCOVER AFTER SWS - SW RESOORED IN 8 HOURS - RCS COOLDOWN SUCCESS	9.720E-01	0.000E+00	D	RF11
763	XX	SWS/8HR	CORE UNCOVERY OCCURS AFTER SWS - SW RESTORED IN 8 HOURS - RCS COOLDOWN FAILS	3.280E-02	0.000E+00	D	RF11
764	XX	SWS/8HR	CORE DOES NOT UNCOVER AFTER SWS - SW RESTORED IN 8 HOURS - RCS COOLDOWN FAILS	9.670E-01	0.000E+00	D	RF11
765	XX	SWS/2HR	FAILURE OF SW RECOVERY WITHIN 2 HOURS AFTER A LOSS OF ALL SERVICE WATER EVENT	8.230E-02	0.000E+00	D	RF11
766	XX	SWS/2HR	SUCCESS OF SW RECOVERY WITHIN 2 HOURS AFTER A LOSS OF ALL SERVICE WATER EVENT	9.180E-01	0.000E+00	D	RF11
767	XX	SWS/2HR	CORE UNCOVERY OCCURS AFTER SWS - SW RESTORED IN 2 HOURS - RCS COOLDOWN FAILS	2.450E-02	0.000E+00	D	RF11
768	XX	SWS/2HR	CORE DOES NOT UNCOVER AFTER SWS - SW RESTORED IN 2 HOURS - RCS COOLDOWN FAILS	9.760E-01	0.000E+00	D	RF11
769	XX	CCW	SUCCESS OF OPERATOR ACTION TO TRIP RCPS ON LOSS OF COOLING WATER	9.976E-01	0.000E+00	D	RF12

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MASTER DATA FILE FOR WCGS PRA ANALYSIS

#	COMP	SYSTEM	FAILURE MODE	FAILRATE	VARIANCE	UNIT	SOURCE
800	XX	SWS/2HR	SUCCESS OF SW RECOVERY WITHIN 2 HOURS GIVEN FLOOD	9.996E-01	0.000E+00	D	FL3
801	XX	SWS/2HR	FAILURE OF SW RECOVERY WITHIN 2 HOURS GIVEN FLOOD	4.300E-04	0.000E+00	D	FL3
802	XX	SWS/8HR	SUCCESS OF SW RECOVERY WITHIN 8 HOURS GIVEN FLOOD	9.996E-01	0.000E+00	D	FL3
803	XX	SWS/8HR	FAILURE OF SW RECOVERY WITHIN 8 HOURS GIVEN FLOOD	4.300E-04	0.000E+00	D	FL3
804	XX	IEV-FL3	IEV-SWS FOR FLOOD SCENARIO FL3 - LARGE RUPTURE	2.680E-06	0.000E+00	D	FL3
805	XX	IEV-FL2	IEV-TRA FOR FLOOD SCENARIO FL2	7.450E-05	0.000E+00	D	FL2
806	XX	IEV-FL1	IEV-TRO FOR FLOOD SCENARIO FL1	2.260E-02	0.000E+00	D	FL1
807	XX	IEV-FL3	IEV-SWS FOR FLOOD SCENARIO FL3 - SMALL RUPTURE	5.360E-06	0.000E+00	D	FL3

NOTES

1. NOTES / REFERENCES:

NT01 = PROBABILITIES OF 1 AND 0 ARE PLACED IN THE DATA BANK TO BE USED AS NEEDED

NT02 = FOR SUPPORT SYSTEM SUB TREES THAT APPEAR AS BASIC EVENTS, THIS TEMPORARY DUMMY PROBABILITY IS ASSIGNED.

NT03 = INITIATING EVENT FREQUENCIES ARE CALCULATED IN SECTION 2.3

NT04 = PLANT SPECIFIC DATA OR BAYESIAN UPDATED

NT05 = PLANT SPECIFIC SYSTEM TRAIN T/M UNAVAILABILITIES

RF01 = NUREG/CR-4550 (SAND86-2084) VOL. 1, REV. 1, JANUARY 1990.

RF02 = NUREG/CR-2815, "PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE," VOL. 1, REV. 1, TABLE C.1, AUGUST 1985, NUMBERS ARE LOGUNIFORM

RF03 = NUREG/CR-2728.

RF04 = WASH 1400.

RF05 = ECCS NOTEBOOK - SECTION 4.8 OF THE WCGS PRA

RF06 = CCW SYSTEM NOTEBOOK - SECTION 4.6 OF THE WCGS PRA

RF07 = ELECTRICAL POWER SYSTEMS NOTEBOOK - SECTION 4.4 OF THE WCGS PRA

RF08 = AFW SYSTEM NOTEBOOK - SECTION 4.5 OF THE WCGS PRA

RF09 = ESW SYSTEM NOTEBOOK - SECTION 4.9 OF THE WCGS PRA

RF10 = IEEE STANDARD 500-1977

RF11 = MISCELLANEOUS NOTEBOOK - SECTION 4.10 OF THE WCGS PRA

RF12 = HUMAN RELIABILITY ANALYSIS - SECTION 4.11 OF THE WCGS PRA

RF13 = CONTAINMENT SAFEGUARDS SYSTEM NOTEBOOK - SECTION 4.7 OF THE WCGS PRA

RF14 = REACTOR PROTECTION SYSTEM NOTEBOOK

TOP0 = WCAP-10271, "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR PROTECTION INSTRUMENTATION SYSTEM, "WESTINGHOUSE PROPRIETARY CLASS 2, TABLE 4.2-1.

TOP2 = WCAP-10271, SUPPLEMENT 2, REVISION 1, "EVALUATIUN OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACIOR PROTECTION INSTRUMENTATION SYSTEM," WESTINGHOUSE PROPRIETARY CLASS 2, TABLE 3.2-1

TOP3 = WCAP-10271, SUPPLEMENT 3, "EVALUATION OF SURVEILLANCE FREQUENCIES AND OUT OF SERVICE TIMES FOR THE REACTOR PROTECTION INSTRUMENTATION SYSTEM," WESTINGHOUSE PROPRIETARY CLASS 2, TABLE 4.2-1.

PORI = FAILURE RATE ESTIMATES FOR UNIVERSAL LOGIC, UNDERVOLTAGE OUTPUT, AND SAFEGUARDS DRIVEN CARDS OF THE SOLID-STATE PROTECTION SYSTEM, CN-PORI-91-021-R0

2. ABBREVIATIONS USED:

XX = ITEM RELATED NOT TO A SPECIFIC COMPONENT

ALL = ITEM APPLIES TO ALL SYSTEMS

IE = INITIATING EVENT

D = DEMAND FAILURE

HR = HOURLY FAILURE RATE (LAMBDA), NEEDS TO BE MULTIPLIED BY A DELTA-T

SUB- = FIRST FOUR CHARACTERS OF THE EVENT ID OF A SUBTREE

OPA- = FIRST FOUR CHARACTERS OF THE EVENT ID OF AN HUMAN ACTION FAILURE

ATWS = ANTICIPATED TRANSIENTS WITHOUT SCRAM

LT3 = LESS THAN OR EQUAL TO 3 INCH DIAMETER PIPE

GT3 = GREATER THAN 3 INCH DIAMETER PIPE

SEE FAULT TREE GUIDELINES TABLE 4.1-1 FOR COMPONENT IDENTIFICATION CODES

3. COMMENTS:

IF A PROBABILITY IS NOT FOUND FOR A SPECIFIC FAILURE TYPE, A SURROGATE VALUE IS USED. THIS SURROGATE VALUE IS OBTAINED BY USING THE FAILURE PROBABILITY OF A SIMILAR COMPONENT.

FOR EXAMPLE, CATASTROPHIC INTERNAL FAILURE PROBABILITIES FOR ALL TYPES OF VALVES ARE TAKEN FROM THE SAME ENTRY IN REFERENCE 02.

IE FREQUENCIES MUST BE USED LIKE DEMAND FAILURES WHEN THEY ARE USED TO QUANTIFY COREMELT ACCIDENT SEQUENCE FREQUENCIES.

COMMON CAUSE FACTORS AND THE EQUATIONS USED TO CALCULATE COMMON CAUSE FAILURE PROBABILITIES ARE FROM W-RMOI GUIDEBOOK 2, "TREATMENT" OF COMMON CAUSE IN FAULT TREE MODELS, APPENDIX C - MULTIPLE GREEK LETTER (MGL) METHOD.
VARIANCES ARE CALCULATED IN ACCORDANCE WITH SECTION 5.2 OF APPENDIX 4.3B.

TABLE 3.3-2

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB.	GEN. VAR.	N	M	CALC. PROB	CALC. VAR	COMMENTS
040	Motor Operated Valve - Fails to Open	/D	3.0E-03	5.50E-05	44	11863	3.71E-03	7.33E-06	Plant
041	Motor operated Valve - Fails to Close	/D	3.0E-03	5.50E-05	44	11863	3.71E-03	7.33E-06	Plant
101	Motor Driven Pump - Fails to Start	/D	3.0E-03	5.50E-05	3	1468	2.04E-03	2.35E-06	Plant
102	Motor Driven Pump - Fails to Run	/HR	3.0E-05	5.50E-09	3	94682	3.17E-05	5.64E-10	Plant
103	Turbine Driven Pump - Fails to Start	/D	3.0E-02	5.50E-03	0	108	5.56E-03	2.94E-05	Bayes
104	Turbine Driven Pump - Fails to Run	/HR	5.0E-03	1.50E-05	0	50	4.46E-03	1.03E-05	Bayes
128	ESW Traveling Screen - Fails to Start	/D	3.0E-03	5.50E-05	0	266	1.13E-03	1.85E-06	Bayes
129	ESW Traveling Screen - Fails to Run	/HR	3.0E-05	5.50E-09	0	3471	2.24E-05	1.49E-09	Bayes
352	Diesel Generator - Fails to Start	/D	3.0E-02	5.10E-04	2	364	5.49E-03	1.70E-05	Plant
353	Diesel Generator - Fails to Run	/HR	2.0E-03	2.25E-06	4	547	4.13E-03	4.13E-06	Bayes

N = Number of failures
M = Number of demands/hours of operation
Bayes = Bayesian Update
Plant = Plant Specific

TABLE 3.3-3

SUMMARY OF WCGS TEST/MAINTENANCE UNAVAILABILITIES

Component Type	Total Duration (Hrs)	Operated in Mode 1 (Hrs)	Unavail. Per Train	Average Unavail.
M.D. AFW Pump Trains				1.40E-02
Train A	438.5	29,893	0.0147	
Train B	394.9	29,893	0.0132	
T.D. AFW Pump Train	542.1	29,893	0.0181	1.81E-02
Charging Pump Trains				1.07E-02
Train A	295.5	29,893	0.0087	
Train B	380.5	29,893	0.0127	
ESW Pump Trains				1.21E-02
Train A	376.5	29,893	0.0126	
Train B	344.3	29,893	0.0115	
CCW Pump Trains				1.03E-02
Train A	391.8	29,893	0.0131	
Train B	427.6	29,893	0.0143	
Train C	299.8	29,893	0.0100	
Train D	108.2	29,893	0.0036	
RHR Pump Trains				6.38E-03
Train A	234.8	29,893	0.0079	
Train B	146.8	29,893	0.0049	
SI Pump Trains				9.00E-03
Train A	301.8	29,893	0.0101	
Train B	235.7	29,893	0.0079	
Cntm Spray Pump Trains				2.67E-03
Train A	80.6	29,893	0.0027	
Train B	78.6	29,893	0.0026	
Diesel Generators				1.13E-02
NE01	281.7	29,893	0.0094	
NE02	393.8	29,893	0.0132	

3.3.3 HUMAN FAILURE DATA

The Human Reliability Analysis (HRA) for the WCGS IPE consists of three phases. The first phase consisted of the HRA analyst, in parallel with the event tree and/or fault tree analysts, identifying all operator actions judged to be critical subtasks for recovery from each initiating event or misalignment following testing activities. The second phase consisted of a control room visit and operator talkthroughs to verify or modify all subtasks modeled in the first phase. Preliminary quantification was then performed as part of the third phase. The results were reviewed by cognizant analysts and revised as necessary. All of the phases are complementary to each other and reinforce the final HRA results.

The Techniques for Human Error Rate Prediction (THERP) method developed at Sandia National Laboratories (SNL) was chosen for the HRA. The THERP method is not a "model" in the usual sense of a hypothetical analogy; rather, it is treated as a form of Boolean modeling. This method represents operator behavior by simple equations dealing with plant equipment parameters, human redundance, training, stress levels, etc. THERP is a relatively simple method used to predict human error probabilities and to evaluate the degradation of a man-machine system likely to be caused by human errors alone, or by operational procedures and plant practices, or by other human characteristics that influence the plant operator's behavior.

The THERP method used is an adequate tool for practical human reliability work; however, the lack of data on human stress factors and the lack of guidance for using appropriate slack time recovery in conjunction with dependency modeling are its major weaknesses. These weaknesses were reduced, although not completely eliminated, by "operator talkthroughs". The THERP method recommends a screening process be applied to HRA. This process involves the assignment of a very high failure probability to certain operator actions. Generally, an operator action with very high HRA which does not have an effect on the system analysis could be eliminated, thus reducing the amount of analysis necessary. However, this initial screening approach was not performed for the WCGS IPE since a realistic model of all operator actions was desired.

Throughout the HRA analysis, extensive efforts were made to ensure consistency in all assumptions.

3.3.3.1 Description of HRA Methodology

The implementation of the THERP method is similar to the application of fault tree methodology; first, it breaks an operator action into subtasks similar to various events in a fault tree. The subtasks are then assembled through the use of ANDed or ORed operations similar to "AND" or "OR" Boolean operations in a fault tree application.

The subtask analysis was the first step in the initial phase of the HRA methodology. This step was performed after the event trees were fully developed. In the subtask analysis, all operator actions identified by the event tree analysts were broken down into specific operator steps which are absolutely necessary for the operator actions to be successful. All operator actions identified by fault tree analysts pertaining to system alignment were also broken down into operator steps in the same fashion.

The WCGS emergency procedures (EMGs), normal and abnormal (OFN) operating procedures, and alarm procedures (ALRs) were used in the HRA for defining all selected operator steps required by the progression of an event or a fault tree.

Talkthroughs with plant operators were performed to verify the subtasks were accurately modeled. The quantification of subtasks was the final step in the methodology. This step was achieved by mathematical presentation and conversion of all independent, conditional, and joint operator steps into human error probabilities (HEPs). In the conversion process, HRA analysts determined five probabilistic parameters: errors of commission, omission, and diagnosis/detection, recovery, and failure to use procedures. Performance Shaping Factors (PSFs) were used concurrently to modify the nominal HEPs (that is, the probability of a given human error when the effects of plant-specific PSFs have not yet been considered). All nominal HEPs were taken from the Swain handbook (NUREG/CR-1278, Section 20, Table A-2).

These three steps (subtask analysis, operator talkthroughs, and quantification) are summarized in the following:

Subtask Analysis

The subtask analysis was initiated by listing only those operator steps that are absolutely necessary to be accomplished within some period of time in accordance with the operator time set in the event trees and associated event success criteria. Each subtask step was selected on the basis that failure of the operator to perform that specific step would significantly impact or result in failure of a safety system. Therefore, most steps which are follow-throughs from or to some operator actions and which do not involve some physical operator activity were screened out. This typically involves steps for checking system parameters while doing other major (physical) steps. The assumption used is that if the plant response is normal or as expected for the accident scenario, then such system parameter responses would be attained. However, some checking steps were taken into account during the quantification activity where they were represented as PSFs to the modeled tasks.

Operator Talkthroughs

In order to verify and validate the initial selected critical steps, a meeting was held between the HRA analyst and WCGS Senior Reactor Operators, trainers, and system analysts. During the meeting, each of the operator actions was reviewed. The review included a discussion of the action steps listed in the emergency operating procedures and a walkthrough of the control room to identify the location of the controls and indications that the operators are using for each of the steps.

Following the meeting, the HRA analysts reviewed the initial work including the time available and actual time for the operator to complete tasks, as estimated in the event trees, and made modifications to the HRA model as necessary.

Quantification

The HRA model is quantified in terms of three phases: (1) cognitive phase (detection, diagnosis, and decision making), (2) action phase, and (3) recovery phase.

The quantification part of the HRA also involves manually developing a probabilistic statement of each operator step defined by the subtask analysis. In other words, the HRA analyst estimates the probabilities of human errors in each step. Although estimates for HEPs are normally based on stored data, expert prediction, or by using extrapolation of existing data from similar operator activities in other nuclear stations, the estimate of HEPs for the WCGS HRA was done by using the published nominal HEPs (Table A-2, Swain handbook, NUREG/CR-1278), and then modifying the nominal HEPs with the PSFs. The HRA analyst also interpreted each modeled step based on the requirement of independent, conditional or joint human activities and then formulated a mathematical expression for each realistic HEP used in the computer code. Finally, the result of each modeled step, in terms of HEPs, was added to obtain an overall HEP value for a particular operator action.

3.3.3.2 HRA Results

Table 3.3-4 lists the HRA results. The results of any HRA are valid only if they are borne out by real-world events. However, based upon engineering judgement, the results are valuable since they provide a reference point for understanding operator performance during the improbable events that would occur during a plant accident. The HRA results are classified as follows:

- if the human error probability (HEP) is greater than or equal to 0.01, then it is likely to occur;

- if the HEP is less than 0.01 but greater than or equal to 1E-04, then the HEP is in the normal range;

- if the HEP is less than 1E-04, then the HEP is unlikely to occur.

Based on the above classification, the HEPs in Table 3.3-4 are grouped as follows:

1. HEP is likely; greater than 0.01	IEHVAC, RT, MRT, ACNN, SYSIEV, ECCSTP, DCNK01, RCPSEL, ESWABF

2.	HEP is in normal range; less than 0.01 but greater than or equal to 1E-04	OP1, OP2, ESWA, ESWB, AFC, LC2/LC3, MF1, MS1, OD1-COOL, RCD1, RCD2, ALHVS, RHR, OD2, MCB, PEG1BD, OD1-DP, OFB & OFC, EC3, FS1, LC1, WS1C2, ESWAB, PEG1C, RCPTRP, RRI11, RRI12, RRI22, CSS
3.	HEP is unlikely; less than 1E-04	LTS, SGTR, RUPSG, OST

NOTE: The HEPs classified as "unlikely" involve either mostly automatic actions, or actions that have a long time period available to be performed.

TABLE 3.3-4
IPE MODEL HRA RESULTS

OPERATOR ACTION DESCRIPTION	ID	HEP
- Provide alternate source to 120 VAC bus	ACNN	2.60E-02
- Maintain AFW flow during station blackout	AFC	1.36E-04
- Reopen air-operated valve	ALHVS	8.29E-04
- Switchover containment spray system	CSS	1.70E-03
- Manually transfer NB01 components on NK01 failure	DCNK01	4.77E-03
- Loss of SGK05A and SGK05B A/C Units	IEHVAC	1.00E-01
- Long term cooldown and depressurization, given SGTR	EC3	2.15E-04
- Manually trip ECCS pumps	ECCSTP	1.00E-02
- Long term cooldown and depressurization, given small LOCA	ES1	1.35E-04
- Realign ESW A to normal service water	ESWA	1.10E-03
- Realign ESW B to normal service water	ESWB	1.10E-03
- Place emergency service water system into operation	ESWAB	2.40E-04
- Switch to ESW due to flood	ESWABF	1.46E-01
- Return ESW B train to normal service water	SYSIEV	2.16E-02
- Perform low-head recirculation	LC1	9.17E-04
- Perform high pressure recirculation	LC2/LC3	4.97E-04
- Long term shutdown	LTS	4.38E-05
- Manually start component on auto start failure	MCB	1.59E-03
- Establish MFW flow	MF1	6.02E-04
- Manually trip RDGMs	MRT	1.55E-01
- Isolate ruptured SG	MS1	1.59E-03
- Initial RCS cooldown, given SG rupture	OD1-COOL	5.55E-03
- Initial RCS Depressurize, given SG tube rupture	OD1-DP	9.55E-03
- Stabilize RCS and ruptured SG after SG overfills	OD2	2.90E-03
- Initiate feed and bleed	OFB & OFC	1.76E-03
- RCS cooldown and depressurization, given medium LOCA with HPSI failure	OP1	1.73E-04
- RCS cooldown and depressurization, given small LOCA with HPSI failure	OP2	1.09E-04
- Terminate HPSI	OST	5.78E-05
- Manually start CCW pump	PEG1C	6.04E-03
- Switch-over CCW Operation to Train B	PEG1BD	9.42E-03
- RCS cooldown during blackout	RCD1	1.66E-03
- RCS cooldown during loss of CCW	RCD2	1.32E-04
- Provide RCP seal injection flow	RCPSEL	1.43E-02
- Manually trip RCPs	RCPTRP	2.39E-03
- Stop RHR pumps	RHR	5.01E-04
- Restore RCS inventory using 1 train of 1	RRI11	5.64E-03
- Restore RCS inventory using 1 train of 2	RRI12	9.88E-04
- Restore RCS inventory using 2 trains of 2	RRI22	7.35E-03
- Manually trip reactor or insert rods	RT	1.36E-02
- Identify ruptured SG	RUPSG	6.33E-05
- Diagnose SGTR event	SGTR	6.36E-06
- Start standby service water pump	WS1C2	1.48E-04

3.3.4 Common Cause Analysis

Common Cause Failure (CCF) is used to describe events that are a subset of dependent events in which two or more components fail due to the same cause at the same time, or in a short interval, and that are a direct result of a shared cause. The common cause failure analysis evaluates and estimates the effects of these dependencies that impact the capability of a system to prevent or mitigate a severe accident.

The WCGS IPE used the Multiple Greek Letter (MGL) method and parametric factors beta, gamma, and delta as defined in NUREG/CR-4780, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events" as follows:

- BETA conditional probability that the common cause of a component failure will be shared by one or more additional components

- GAMMA conditional probability that the common cause of a component failure that is shared by one or more components will be shared by two or more components additional to the first

- DELTA conditional probability that the common cause of a component failure that is shared by two or more components will be shared by three or more components additional to the first

An initial or screening determination of common cause failure probabilites was performed using the generic MGL factors. Subsequently, WCGS performed an analysis to determine values for conditional probabilities for the MGL parametric factors. EPRI event databases were used as the bases for the determination of WCGS specific common cause calculations. The data bases were screened to determine which of the events were likely to happen at WCGS. The EPRI subset of WCGS applicable common cause events served as the base for determination of the MGL parametric factors - beta, gamma, and delta. The WCGS common cause probabilites were recalculated using the new factors and fault trees were requantified. The WCGS specific methodology provides a more realistic model of WCGS common cause failures.

3.3.5 Quantification of Unavailability of Systems and Functions

The WCGS PRA uses event trees, fault trees, and fault tree linking processes to identify fault sequences that would lead to core damage. The fault trees developed for the PRA include front line systems as well as support systems. As needed, system fault trees were combined with fault trees of other systems and/or operator actions to generate unavailabilities of functions such as feed and bleed and secondary depressurization.

The WCGS fault trees were developed and quantified using the Westinghouse GRAFTER Code System software package. The results of the fault tree quantification include system unavailabilities and system cutsets. The fault tree quantification output files are used as input files for the fault tree linking process which is described in Section 3.3.6.

The fault tree quantification process is essentially a two step process: quantify the support system trees and quantify the front line system trees. Each step is discussed below.

Step 1 - Support System Fault Tree Quantification

Support system fault trees appear as subtrees within the frontline system fault trees, thus they should be quantified first. In some instances, a support system fault tree also calls other support system fault trees. For example, a CCW fault tree may call in the support system fault trees for ESW supply, AC power, and DC power. The support system fault trees were quantified to reduce all subtrees in all of the support systems. This replacement of all subtree identifier events in the support systems by their corresponding support system (subtree) cutsets reduces the number of iterations required to perform the core damage frequency calculation. This replacement is called "linking" such that the cutsets for a subtree are linked with the cutsets of the fault tree being quantified in the appropriate places. The fault trees for support systems as used in the core damage quantification are listed in Table 3.3-5. Note that Table 3.3-5 is not comprehensive in that reactor protection signal subtrees are not included.

Step 2 - Front Line System Fault Tree Quantification

Once all the subtrees were linked into the support system fault trees, the front line fault trees were quantified by linking in the support system fault trees. The result for each front line fault tree is the system unavailability (failure probability) and a listing of cutsets which represent all possible unavailabilities of the system given a user-specified truncation value (cutoff). Table 3.3-6 summarizes the frontline system and function unavailabilities. Note that the probabilities listed in the table represent front line system fault trees, with all support systems being available.

3.3.6 Quantification of Sequence Frequencies

For the core damage accident sequence analysis, the Westinghouse WLINK Code System was used. This code uses the fault tree linking method to calculate the plant core damage frequency and identify fault sequences (cutsets) in terms of component failures, operator actions, and other failures. The inputs to the WLINK program include: (1) a listing of the event tree accident sequences to be quantified, (2) the front line fault tree quantification results files, and (3) miscellaneous event tree node probabilities. The scalar event tree node probabilities represent those nodes for which a fault tree was not created (e.g., failure to restore offsite power in 8 hours). Table 3.3-7 lists these scalar event tree node probabilities. The core damage frequency is calculated first for each initiating event category and then for the WCGS total plant core damage frequency.

The results of the accident sequence quantification are presented in Section 3.4.

TABLE 3.3-5

LIST OF SUPPORT SYSTEM FAULT TREES
USED IN FRONT LINE FAULT TREES

Subtree Name	Description
ACNB01	Loss of AC Power at 4160 V ESF Bus NB01
ACNB02	Loss of AC Power at 4160 V ESF Bus NB02
ACNG01	Loss of AC Power at 480 V Load Center NG01
ACNG01A	Loss of AC Power at 480 V MCC NG01A
ACNG01B	Loss of AC Power at 480 V MCC NG01B
ACNG02	Loss of AC Power at 480 V Load Center NG02
ACNG02A	Loss of AC Power at 480 V MCC NG02A
ACNG02B	Loss of AC Power at 480 V MCC NG02B
ACNG03	Loss of AC Power at 480 V Load Center NG03
ACNG03C	Loss of AC Power at 480 V MCC NG03C
ACNG03D	Loss of AC Power at 480 V MCC NG03D
ACNG04	Loss of AC Power at 480 V Load Center NG04
ACNG04C	Loss of AC Power at 480 V MCC NG04C
ACNG04D	Loss of AC Power at 480 V MCC NG04D
CCWA	Loss of Component Cooling Water System - Train A
CCWB	Loss of Component Cooling Water System - Train B
DCNK01	Loss of DC Power at 125 V Bus NK01
DCNK02	Loss of DC Power at 125 V Bus NK02
DCNK03	Loss of DC Power at 125 V Bus NK03
DCNK04	Loss of DC Power at 125 V Bus NK04
DCNK41	Loss of DC Power at 125 V Switchboard NK41
DCNK42	Loss of DC Power at 125 V Switchboard NK42
DCNK43	Loss of DC Power at 125 V Switchboard NK43
DCNK44	Loss of DC Power at 125 V Switchboard NK44
DCNK51	Loss of DC Power at 125 V Switchboard NK51
DCNK54	Loss of DC Power at 125 V Switchboard NK54
ESWA	Loss of Essential Service Water System - Train A
ESWB	Loss of Essential Service Water System - Train B

TABLE 3.3-6

LIST OF FRONT LINE FAULT TREE UNAVAILABILITIES

Fault Tree	Fault Tree Description	Unavailability
ACC	Accumulator Safety Injection	4.28E-06
AF2	AFWS - SLO, MLO, and Loss of CCW Events	2.19E-05
AF2R	AFWS - TRA Event	2.30E-05
AF2WO	AFWS - LSP Event	4.93E-05
AF2WOR	AFWS - TRO Event	4.88E-05
AF4	AFWS - SGR Event	5.23E-03
AF5	AFWS - SLB Event	2.36E-03
AF5A	AFWS - SLB Event	1.88E-02
AF7	AFWS - Loss of 125V DC Event	1.13E-03
AFA2	AFWS - ATWS Event initial power level > 40%	4.77E-03
AFT	AFWS - SBO Event	6.18E-02
EC3	RCS Cooldown and Depressurization - SGR Event	1.24E-03
ES1	RCS Cooldown and Depressurization - SLO Event	1.16E-03
HPR11	High Pressure Recirculation - Subsequent to RRI11	3.42E-02
HPR12	High Pressure Recirculation - Subsequent to RRI12 (Same as LC2)	
HPR22	High Pressure Recirculation - Subsequent to RRI22	3.09E-03
LC1	Low Pressure Recirculation	1.21E-03
LC2	High Pressure Recirculation	1.53E-03
LC3	High Pressure Recirculation - Loss of 125V DC Bus Event	1.67E-03
LPI	Low Pressure Safety Injection - Loss of CCW Event	7.91E-03
LPR	Low Pressure Recirculation - Loss of CCW Event	1.07E-03
LTS	Long Term Shutdown - ATW Event	4.13E-04
MF1	Main Feedwater and Condensate	4.38E-02
MS1	Main Steam Isolation - SGR Event	1.68E-03
MS2	Main Steam Isolation - SLB Event	6.66E-05
OD1	Stabilize RCS and Ruptured SG Pressure Before SG Overfill	1.51E-02
OF2	Operator Bleed and Feed - Loss of 125V DC Bus Event	5.68E-03
OFB	Operator Bleed and Feed - SLO, SGR, and SLB Events	6.60E-03
OFC	Operator Bleed and Feed - TRO, TRA, and LSP Events	6.80E-03
RRI11	RCS Inventory Restoration - Recovery Prior to Core Uncovery - One ESF Bus Available	5.66E-03
RRI12	RCS Inventory Restoration - Recovery Prior to Core Uncovery - Two ESF Buses Available	1.01E-03
RRI22	RCS Inventory Restoration - Recovery Following Start of Core Uncovery - Two ESF Buses Available	4.74E-02
SI1	Low Pressure Safety Injection	6.13E-05
SI2	High Pressure Injection - MLO Event	1.63E-05
SI3	High Pressure Injection - SLO Event	7.23E-08

Note: Unavailabilities represent complete support system availability

TABLE 3.3-7

LIST OF SCALAR EVENT TREE NODES AND PROBABILITIES
Page 1 of 3

	Node	Description	Probability
1	2HR-FAILS	AC POWER NOT RECOVERED WITHIN 2 HOURS	1.1500E-01
2	2HR-SUCCESSFUL	AC POWER RECOVERED WITHIN 2 HOURS	8.8500E-01
3	4HR-FAILS	AC POWER NOT RECOVERED WITHIN 4 HOURS	5.3800E-02
4	4HR-SUCCESSFUL	AC POWER RECOVERED WITHIN 4 HOURS	9.4600E-01
5	8HR-FAILS	AC POWER NOT RECOVERED WITHIN 8 HOURS	2.5400E-02
6	8HR-SUCCESSFUL	AC POWER RECOVERED WITHIN 8 HOURS	9.7500E-01
7	CCR2-FAILS	CCW COOLING NOT RECOVERED WITHIN 2 HOURS	9.3800E-02
8	CCR2-SUCCESSFUL	CCW COOLING RECOVERED WITHIN 2 HOURS	9.0600E-01
9	CCR8-FAILS	CCW COOLING NOT RECOVERED WITHIN 8 HOURS	1.9100E-02
10	CCR8-SUCCESSFUL	CCW COOLING RECOVERED WITHIN 8 HOURS	9.8100E-01
11	CNU2F-FAILS	CORE UNCOVERY OCCURS WITHIN 2 HOURS AFTER SBO	3.9800E-03
12	CNU2F-SUCCESSFUL	CORE REMAINS COVERED 2 HOURS AFTER SBO	9.9600E-01
13	CNU4-FAILS	CORE UNCOVERY OCCURS WITHIN 4 HOURS AFTER SBO, RCS COOLDOWN AND DEPRESSURIZATION (RCD) IS SUCCESSFUL	5.5600E-03
14	CNU4-SUCCESSFUL	CORE REMAINS COVERED 4 HOURS AFTER SBO, RCD SUCCESSFUL	9.9400E-01
15	CNU4F-FAILS	CORE UNCOVERY OCCURS WITHIN 4 HOURS AFTER SBO, RCD FAILS	6.0600E-03
16	CNU4F-SUCCESSFUL	CORE REMAINS COVERED 4 HOURS AFTER SBO, RDC FAILS	9.9400E-01
17	CNU8-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER SBO, RCD SUCCESSFUL	1.2700E-02
18	CNU8-SUCCESSFUL	CORE REMAINS COVERED 8 HOURS AFTER SBO, RCD SUCCESSFUL	9.8700E-01
19	CNU8F-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER SBO, RDC FAILS	1.8900E-02
20	CNU8F-SUCCESSFUL	CORE REMAINS COVERED 8 HOURS AFTER SBO, RCD FAILS	9.8100E-01
21	CNUC2F-FAILS	CORE UNCOVERY OCCURS WITHIN 2 HOURS AFTER CCW EVENT	1.1700E-02
22	CNUC2F-SUCCESS	CORE REMAINS COVERED 2 HOURS AFTER CCW EVENT	9.8800E-01
23	CNUC8-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER CCW EVENT, RCD SUCCESSFUL	1.9200E-02
24	CNUC8-SUCCESSFUL	CORE REMAINS COVERED 8 HOURS AFTER CCW EVENT, RCD SUCCESSFUL	9.8100E-01
25	CNUC8F-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER CCW EVENT, RCD FAILS	2.6600E-02
26	CNUC8F-SUCCESS	CORE REMAINS COVERED 8 HOURS AFTER CCW EVENT, RCD FAILS	9.7300E-01
27	CNUS2F-FAILS	CORE UNCOVERY OCCURS WITHIN 2 HOURS AFTER SWS EVENT	2.4500E-02
28	CNUS2F-SUCCESS	CORE REMAINS COVERED 2 HOURS AFTER SWS EVENT	9.7600E-01
29	CNUS8-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER SWS EVENT, RCD SUCCESSFUL	2.8100E-02
30	CNUS8-SUCCESSFUL	CORE REMAINS COVERED 8 HOURS AFTER SWS EVENT, RCD SUCCESSFUL	9.7200E-01
31	CNUS8F-FAILS	CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER SWS EVENT, RCD FAILS	3.2800E-02

TABLE 3.3-7

LIST OF SCALAR EVENT TREE NODES AND PROBABILITIES
Page 2 of 3

	Node	Description	Probability
32	CNUS8F-SUCCESS	CORE REMAINS COVERED 8 HOURS AFTER SWS EVENT, RCD FAILS	9.6700E-01
33	ED-04	FRACTION OF 4 HOUR AC POWER RECOVERY DUE TO RECOVERY OF ONE DIESEL GENERATOR, RCD SUCCESSFUL	8.3400E-02
34	ED-08	FRACTION OF 8 HOUR AC POWER RECOVERY DUE TO RECOVERY OF ONE DIESEL GENERATOR, RCD SUCCESSFUL	5.0600E-02
35	EDF-02	FRACTION OF 2 HOUR AC POWER RECOVERY DUE TO RECOVERY OF ONE DIESEL GENERATOR, RCD FAILS	1.7100E-01
36	EDF-04	FRACTION OF 4 HOUR AC POWER RECOVERY DUE TO RECOVERY OF ONE DIESEL GENERATOR, RCD FAILS	8.3900E-02
37	EDF-08	FRACTION OF 8 HOUR AC POWER RECOVERY DUE TO RECOVERY OF ONE DIESEL GENERATOR, RCD FAILS	5.9500E-02
38	OP-04	FRACTION OF 4 HOUR AC POWER RECOVERY DUE TO RECOVERY OF OFFSITE POWER, RCD SUCCESSFUL	9.1700E-01
39	OP-08	FRACTION OF 8 HOUR AC POWER RECOVERY DUE TO RECOVERY OF OFFSITE POWER, RCD SUCCESSFUL	9.4900E-01
40	OPA-ACNN	HRA FAILURE TO ALIGN INVERTORS TO ALTERNATE POWER SOURCE (OFN 00-021)	2.8400E-02
41	OPA-AFC	HRA FAILURE TO SHED DC LOADS TO EXTEND BATTERY LIFE TO 8 HOURS	1.3600E-04
42	OPA-MRT	HRA FAILURE TO OPEN DRIVE MECHANISM POWER SUPPLY BREAKERS IN ATWS EVENT	1.5500E-01
43	OPA-OD2	HRA FAILURE IN SGTR TO CONTINUE RCS PRESSURE STABILIZATION AFTER SG OVERFILL OCCURS	2.9000E-03
44	OPA-OP1	HRA FAILURE TO COOLDOWN AND DEPRESSURIZE RCS IN MEDIUM LOCA EVENT	1.7300E-04
45	OPA-OP2	HRA FAILURE TO COOLDOWN AND DEPRESSURIZE RCS IN SMALL LOCA EVENT	1.0900E-04
46	OPA-RCD1	HRA FAILURE IN SBO EVENT TO PERFORM RCS COOLDOWN AND DEPRESSURIZATION	1.6600E-03
47	OPA-RCD2	HRA FAILURE IN CCW, SWS EVENTS TO PERFORM RCS COOLDOWN AND DEPRESSURIZATION	1.3200E-04
48	OPA-RCPSEAL	HRA FAILURE TO REALIGN RCP SEAL COOLING ON FAILURE OF THE OPERATING CCW TRAIN	1.4300E-02
49	OPA-RCPTRIP	HRA FAILURE TO TRIP RUNNING RCPS ON LOSS OF SEAL COOLING	2.3900E-03
50	OPA-RT	HRA FAILURE OF MANUAL REACTOR TRIP/MANUAL ROD INSERTION IN ATWS EVENT	1.3600E-02
51	OPA-SBCOOL	HRA FAILURE TO RESTORE AUXILIARY COOLING TO DC SWITCHBOARD ROOMS ON LOSS OF HVAC	1.0000E-01
52	OPF-02	FRACTION OF 2 HOUR AC POWER RECOVERY DUE TO RECOVERY OF OFFSITE POWER, RCD FAILS	8.2900E-01
53	OPF-04	FRACTION OF 4 HOUR AC POWER RECOVERY DUE TO RECOVERY OF OFFSITE POWER, RCD FAILS	9.1600E-01
54	OPF-08	FRACTION OF 8 HOUR AC POWER RECOVERY DUE TO RECOVERY OF OFFSITE POWER, RCD FAILS	9.4050E-01

TABLE 3.3-7

LIST OF SCALAR EVENT TREE NODES AND PROBABILITIES
Page 3 of 3

	Node	Description	Probability
55	OTH-AMS	FAILURE OF AMSAC SYSTEM	1.0000E-02
56	OTH-CCWS	COMPONENT FAILURE FOLLOWING AN LSP EVENT WHICH RESULT IN AN SBO EVENT	4.5560E-03
57	OTH-HPR22	HIGH PRESSURE RECIRCULATION FAILURE NOT CONSIDERED WITH RRI22 FAILED BY DEFINITION	0.0000E+00
58	OTH-PLV	FRACTION OF REACTOR CRITICAL TIME WITH POWER LEVEL GREATER THAN 40 PERCENT	9.8600E-01
59	OTH-PPR1	FAILURE OF PRESSURIZER PRESSURE RELIEF IN ATWS EVENT (AFA3, MRT SUCCESSFUL)	1.9250E-05
60	OTH-PPR2	FAILURE OF PRESSURIZER PRESSURE RELIEF IN ATWS EVENT (AFA2, MRT SUCCESSFUL)	2.5830E-04
61	OTH-PPR3	FAILURE OF PRESSURIZER PRESSURE RELIEF IN ATWS EVENT (AFA3 SUCCESS, MRT FAILS)	1.6920E-01
62	OTH-PPR4	FAILURE OF PRESSURIZER PRESSURE RELIEF IN ATWS EVENT (AFA2 SUCCESS, MRT FAILS)	2.2340E-01
63	OTH-RRI22	RCS INVENTORY RESTORATION FUNCTION FAILS BY BY DEFINITION (2/2 CCPS NEEDED, 1/2 AVAILABLE)	1.0000E+00
64	OTH-SSV	SECONDARY SIDE RELIEF VALVE FAILS TO RESEAT ONCE CHALLENGED AFTER SG OVERFILL	5.0000E-01
65	OTH-SUCPLV	FRACTION OF REACTOR CRITICAL TIME WITH POWER LEVEL LESS THAN 40 PERCENT	1.4000E-02
66	OTH-SUCSSV	SECONDARY SIDE RELIEF VALVE SUCCESSFULLY RESEATS ONCE CHALLENGED AFTER SG OVERFILL	5.0000E-01
67	RCPTRIP-SUC	HRA ACTION TO TRIP RUNNING RCPS ON LOSS OF ALL SEAL COOLING BEFORE EXTENSIVE SEAL DAMAGE OCCURS IS SUCCESSFUL	9.9760E-01
68	SWR2-FAILS	SWS COOLING NOT RECOVERED WITHIN 2 HOURS	8.2300E-02
69	SWR2-SUCCESSFUL	SWS COOLING IS RECOVERED WITHIN 2 HOURS	9.1800E-01
70	SWR8-FAILS	SWS COOLING NOT RECOVERED WITHIN 8 HOURS	1.6600E-02
71	SWR8-SUCCESSFUL	SWS COOLING IS RECOVERED WITHIN 8 HOURS	9.8300E-01

3.3.7 Internal Flooding Analysis

An internal flooding analysis was performed to determine the vulnerability of the WCGS to flood induced core damage. Equipment may be damaged and fail as a result of flooding or spraying. The impact of these failures was assessed qualitatively, and where necessary, quantitatively using the Level 1 PRA models developed for the event sequence and system analyses.

3.3.7.1 Information Assembly

The following information was assembled and reviewed for the flooding analysis:

1. Architect Engineer Flooding Analyses

   As part of its design basis, WCGS performed flooding calculations to assess vulnerability of vital components to submergence. These calculations used the Standard Review Plan guidance that moderate energy lines which are seismically supported develop cracks, rather than ruptures, resulting in lower outflow rates. The calculations were reviewed to determine where the pipe crack (as opposed to a break) was used, and areas where this assumption was used were re-evaluated to determine the consequences of such a flood.

2. Flood Zones

   The flood zones were chosen to correspond to the existing fire zones developed for compliance with 10CFR50 Appendix R requirements. Barriers separating Appendix R zones were found to be applicable to the internal flooding analysis.

3. Flood Sources

   Flood sources in all flood zones were identified.

4. Inflow and Outflow Paths

   Inflow and outflow paths in each flood zone were identified by reviewing layout and penetration drawings.

3.3.7.2 Major Assumptions

1. When subjected to a flooding or spraying event, unprotected equipment was assumed to fail.

2. Design basis accidents such as containment flooding due to a LOCA, or high energy line breaks, were considered to be outside the scope of this analysis.

3. Flooding calculations performed by the architect engineers were reviewed to ensure accuracy and ar'icability to the PRA. Where inconsistencies with NUREG-1335 were identified, the calculations

were re-evaluated. Assumptions made in these calculations were verified during the walkdown.

4. The vapor and dust seals on selected electrical components were assumed to be capable of resisting water intrusion. Upon walkdown inspection and review of drawings, this was concluded to be a valid assumption.

5. Insulated low and medium energy pipes were assumed to drip if a leak developed. Bare pipes were assumed to be spray sources for a line of sight radius.

6. Lines which were not normally charged (i.e., drain lines and dry fire protection piping) were not considered to be credible flood or spray sources.

7. Environmentally qualified components were assumed to be operable to their safe positions when subjected to adverse conditions.

3.3.7.3 Methodology

Below is a summary of the methodology used in this analysis:

1. A review of possible flood induced initiating events was made. It was concluded that the transient sequence without power conversion available (TRO) could best represent responses required for mitigation of a flooding event occurring in the turbine hall basement. The transient sequence with power conversion available (TRA) best represented responses required for mitigation of events initiated by a spray event which disabled two cabinets unprotected with vapor and dust seals. The loss of service water (SWS) sequence was used to model floods in the control building basement which disabled service water isolation valves.

2. A qualitative screening was performed to identify significant flood events. The capability of a maximum postulated flood to initiate a reactor trip and disable components credited in the PRA was used to select zones warranting further consideration. All other zones were excluded from further analysis.

3. Review of flooding calculations and the walkdown were used to further screen flooding scenarios from further analysis.

4. The internal events accident sequence which was applicable to the flooding scenario in question was used to evaluate flood induced contribution to core damage. The Westinghouse WALT code was used for core damage quantification. Flood induced initiating event frequency, and components which would be affected by the postulated flood were identified, allowing quantification of flood induced core damage frequency.

3.3.7.4 Internal Flooding Results

Four scenarios warranted further analysis and are discussed below.

1) Turbine Hall Floods

Failure of a condenser expansion joint was postulated, resulting in a turbine trip and disabling of heater drain and condensate pumps. The applicable sequence for this scenario is the transient sequence without power conversion (TRO). The initiating event frequency was calculated as the historical frequency of major turbine hall floods, 2.26E-02/yr. The resultant core damage frequency for this scenario was calculated to be 2.64E-08/yr.

2) Room 3302 Spray Scenario

Room 3302, the ESF Switchgear Room (No. 2) contains two unprotected cabinets with ventilation louvers adjacent to a charged fire protection line. The cabinets control two SG atmospheric relief valves, and other auxiliary feedwater components associated with the turbine driven AFW pump. The applicable accident sequence for this scenario is the transient sequence with power conversion available (TRA). The initiating event frequency was estimated as the WASH-1400 pipe break frequency. The cabinets were conservatively assumed to fail upon water intrusion. Core damage frequency arising from this scenario was calculated to be 7.21E-10/yr.

3) Control Building Basement Flood Scenario

Flooding of room 3101, at elevation 1974'-0" in the control building basement due to the rupture of a service water pipe was postulated. Submersion of motor operated valves HV0023, HV0024, HV0025 and HV0026, which are located in this room, would occur, resulting in a loss of service water. Isolation of the flood and the non-essential service water (NESW) system from the essential service water (ESW) system would be impossible until the control building basement is drained. Manual recovery actions are required to successfully isolate NESW from ESW. The applicable accident sequence for this scenario is the loss of service water sequence (SWS). Initiating event frequency was calculated as the WASH-1400 pipe break frequency, and partitioned according to break size. A complete rupture was assumed to go directly to core melt. Recovery actions were credited for the large and small rupture scenarios. Core damage frequencies which arise from this scenario are 8.94E-07/yr (complete rupture) and 2.20E-06/yr (small rupture).

4) ESF Switchgear Room Flooding Scenario

Rooms 3301 and 3302 were found to be susceptible to the rupture of ESW piping in room 3302. In addition, the rupture of ESW piping in either of the diesel generator rooms, 5501 or 5503, could propogate into rooms 3301 and 3302. These rooms contain NB01, NB02, NG01, NG02, NG03 and NG04. Flooding of this equipment would essentially result in a station blackout scenario. No credit has been taken for recovery from this flood via operator action. These 3 rooms contain almost 400 feet of ESW piping both 8" and 4", divided into 12 sections. Using WASH-1400 criteria for pipe sizes greater than 3", the pipe rupture frequency calculates to 4.467E-06/yr.

Total flood-induced core damage is 7.57E-06/yr. This is above the reportable limit for accident sequences which could lead to core damage. No flood induced accident sequences were identified which could lead to containment failure.

3.4 Results and Screening Process

The sequence quantification of the WCGS PRA model resulted in a total core damage frequency of 4.2E-05 per year. A more detailed discussion of the results is presented in this section along with comparison against the NUREG-1335 screening criteria.

Summary of Results

The results of the accident sequence quantification are presented in Table 3.4-1. This table individually lists the top 54 accident sequences for the base PRA model quantification. These 54 sequences account for 99.8% of the total core damage frequency. For each of these sequences the following information is provided:

- Accident sequence frequency
- Percent contribution to total core damage frequency
- Description of the accident sequence
- Sequence event identifiers

The core damage frequency (CDF) and the initiating event frequency are shown in Table 3.4-2 by initiating event. As can be seen from this table, approximately 45% of the CDF comes from the station blackout initiating event. Another five initiating events contribute approximately 40% to the total CDF with no single initiator contributing more than 12%. The remaining 15 initiating events contribute less than 15% of the total core damage frequency.

The top contributor to core damage frequency is station blackout at 45% followed by loss of offsite power at 12%. Two flooding events combine to contribute 16% to core damage. These internal flooding events include flooding of the control building switchgear rooms and recoverable flooding of the control building basement.

3.4.1 Application of Generic Letter Screening Criteria

Following the guidance in NUREG-1335, the following screening criteria were used to determine those important sequences to be reported to the NRC that might lead to core damage or unusually poor containment performance:

a. Any systemic sequence that contributes 1E-7 or more per reactor year to core damage.

b. All systemic sequences within the upper 95 percent of the total core damage frequency.

c. All systemic sequences within the upper 95 percent of the total containment failure probability.

d Systemic sequences that contribute to a containment bypass frequency in excess of 1E-8 per reactor year.

e. Any systemic sequence that the utility determines from previous applicable PRAs or by utility engineering judgment to be an important contributor to core damage frequency or poor containment performance.

The total number of unique sequences reported was determined by the criteria listed above. Table 3.4-3 provides a comparison of the WCGS results to the criteria above. Sequences meeting more than one criterion can be identified by this table.

The most important containment bypass failure sequences for screening criteria "d." are summarized in Table 3.4-4. These containment bypass sequences are primarily steam generator tube rupture events, although MAAP runs indicate that core damage would most likely occur beyond the Level 1 mission time of 24 hours. The interfacing system LOCA core melt sequence also results in a containment bypass with a frequency of 6.11E-08 per year (0.15 percent of the total core damage frequency). Table 3.4-5 identifies the key contributors to core damage including hardware failures, system train unavailability, operator errors, and functional failures such as bleed and feed.

The most important containment failure sequences for screening criteria "c." are summarized in Table 4.3-1 and are comprised of sequences 1 through 27, 38, and 42.

Table 3.4-6 is a concise discussion of the dominant accident sequences' progression, safety issues addressed, and modeling assumptions for those core melt sequences whose frequency is greater than 1E-6.

For WCGS, there were no accident sequences that dropped below the screening criteria because the frequency had been reduced by more than an order of magnitude by taking credit for human recovery actions not defined in the EOPs.

TABLE 3.4-1
DOMINANT SEQUENCES

Reduced Sum of Sequence Probabilities: 4.2815E-05

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	5.89E-06	13.77	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS NOT RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-FAILS
2	4.47E-06	10.44	CONTROL BUILDING SWITCHGEAR ROOMS FLOODING IEV OCCURS	IEV-FL4
3	2.87E-06	6.70	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-FAILS
4	2.86E-06	6.67	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S FROM 1/3 PUMPS	SYS-AF2WO
			OPERATOR BLEED AND FEED COOLING FAILS	SYS-OFC
			BOTH COMPONENT COOLING WATER TRAINS DO NOT FAIL	DEL-CCW
5	2.77E-06	6.48	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 8 HR AFTER SBO FROM OFFSITE W/ RCD	07-08
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-HPR12
6	2.20E-06	5.14	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			COMPONENT FAIL AFTER LSP, SBO EVENT RESULTS	OTH-CCWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			AC POWER IS RECOVERED WITHIN 2 HOURS AFTER AN SBO	2HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 2 HOURS AFTER AN SBO W/O RCD	CNU2F-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 2 HR AFTER SBO FROM EDG W/O RCD	EDF-02
			HIGH PRESSURE SI RESTORATION AFTER SBO, SWS OR CCW FAILS 2/2	OTH-RRI22
7	2.20E-06	5.13	RECOVERABLE CONTROL BUILDING BASEMENT FLOOD IEV OCCURS	IEV-FL3B
			FAILURE TO MITIGATE THE CONTROL BUILDING FLOOD EVENT	SYS-FL3B
8	2.19E-06	5.11	LOSS OF THE OPERATING CCW TRAIN IEV OCCURS	IEV-CCWA
			HRA FAILURE TO PROVIDE RCP SEAL COOL IN TIMELY MANNER	OPA-RCPSEAL
			HRA SUCCESS TO TRIP RUNNING RCP ON LOSS OF SEAL COOL B4 MLO	RCPTRIP-SUC
			RCP SEAL LOCA (SLO) OCCURS AFTER CCWA LOSS (SEAL COOL LOSS)	SYS-SLOCCW
			COMPONENT COOLING WATER SYSTEM TRAIN B DOES NOT FAIL	DEL-CCWBO

TABLE 3.4-1
DOMINANT SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
9	1.86E-06	4.35	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-LC2
10	1.65E-06	3.84	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			AC POWER IS NOT RECOVERED WITHIN 2 HOURS AFTER AN SBO	2HR-FAILS
11	1.64E-06	3.83	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			COMPONENT FAIL AFTER LSP, SBO EVENT RESULTS	OTH-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 8 HR AFTER SBO FROM EDG W/ RCD	ED-08
			HIGH PRESSURE RECIRCULATION FAILS - 1/2 PMPS FROM 1/1 TRAIN	SYS-HPR11
12	1.53E-06	3.56	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			AC POWER IS RECOVERED WITHIN 2 HOURS AFTER AN SBO	2HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 2 HOURS AFTER AN SBO W/O RCD	CNU2F-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 2 HR AFTER SBO FROM OFFSITE W/O RCD	OPF-G2
			HIGH PRESSURE RECIRCULATION FAILS - 2/2 CCPS FROM 2/2 TRAINS	SYS-HPR22
13	1.31E-06	3.06	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			COMPONENT COOLING WATER SYSTEM TRAIN A FAILS	SYS-CCWA
			COMPONENT COOLING WATER SYSTEM TRAIN B DOES NOT FAIL	DEL-CCWB
			HRA FAILURE TO PROVIDE RCP SEAL COOL IN TIMELY MANNER	OPA-RCPSEAL
			RCP SEAL LOCA (SLO) OCCURS - LSP AND CCWA FLS (SEAL COOL LOSS)	SYS-SLOLSP
14	1.15E-06	2.68	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLO
			LOW PRESSURE RECIRCULATION SYSTEM FAILS	SYS-LC1
15	9.91E-07	2.31	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS	IEV-SWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			ONE SWS TRAIN RECOVERED WITHIN 2 HOURS AFTER LOSS OF ALL SWS	SWR2-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 2 HOURS AFTER SWS EVNT W/O RCD	CNUS2F-SUCCESS
			HIGH PRESSURE SI RESTORATION AFTER SBO, SWS OR CCW FAILS 2/2	OTH-RRI22
16	8.94E-07	2.09	NON-RECOVERABLE CONTROL BUILDING BASEMENT FLOOD IEV OCCURS	IEV-FL3A

TABLE 3 4-1
DOMINANT SEQUENCES
Page 3 of 7

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
17	8.25E-07	1.93	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S FROM 1/3 PUMPS HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS BOTH COMPONENT COOLING WATER TRAINS DO NOT FAIL	IEV-LSP SYS-AFZWO SYS-LC2 DEL-CCW
18	6.72E-07	1.57	SMALL LOCA INITIATING EVENT OCCURS RCS COOLDOWN & DEPRESSURIZATION IN SLO PER EMG ES-11 FAILS HIGH PRESSURE RECIRCULATION FAILS - AFTER EC3,ES1 FAILS	IEV-SLO SYS-ES1 SYS-LC2A
19	5.05E-07	1.18	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS ONE SWS TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL SWS CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER SWS EVENT W/ RCD HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	IEV-SWS SWR8-SUCCESSFUL CNUS8-SUCCESSFUL SYS-LC2
20	4.86E-07	1.14	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS ONE SWS TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL SWS CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER SWS EVENT W/ RCD	IEV-SWS SWR8-SUCCESSFUL CNUS8-FAILS
21	3.96E-07	.93	STATION BLACKOUT INITIATING EVENT OCCURS SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP AC POWER IS RECOVERED WITHIN 2 HOURS AFTER AN SBO CORE DOES NOT UNCOVER WITHIN 2 HOURS AFTER AN SBO W/O RCD FRACTION OF AC RECOVERY AT 2 HR AFTER SBO FROM OFFSITE W/O RCD HIGH PRESSURE INJECTION FAILS - 2/2 CCPS FROM 2/2 TRAINS	IEV-SBO SYS-CCWS SYS-AFT 2HR-SUCCESSFUL CNU2F-SUCCESSFUL OPF-02 SYS-RRI22
22	3.00E-07	.70	REACTOR VESSEL FAILURE INITIATING EVENT OCCURS	IEV-VEF
23	2.92E-07	.68	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS SWS COOLING NOT RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL SWS	IEV-SWS SWR8-FAILS
24	2.83E-07	.66	LARGE LOCA INITIATING EVENT OCCURS LOW PRESSURE SAFETY INJECTION SYSTEM FAILS	IEV-LLO SYS-SII
25	2.66E-07	.62	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS ONE SWS TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL SWS CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER SWS EVENT W/ RCD HIGH PRESSURE INJECTION FAILS - 1/2 PUMPS FROM 1/1 TRAIN	IEV-SWS SWR8-SUCCESSFUL CNUS8-SUCCESSFUL SYS-RRI11

TABLE 3.4-1
DOMINANT SEQUENCES
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NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
26	2.44E-07	.57	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/3 SG'S 1/3 PMPS SGR,SLB	SYS-AF5
			RCS COOLDOWN & DEPRESSURIZATION IN SGTR PER EMG C-31/32 FAILS	SYS-EC3
27	2.41E-07	.56	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			RCS AND RUPTURED SG PRESSURE STABILIZATION FAILS B4 OVERFIL	SYS-OD1
			SECONDARY SIDE RELIEF VALVE CLOSES AFTER CHALLENGE - SGTR	OTH-SUCSSV
			HRA FAILURE TO STABILZE RCS AND SG PRESS AFTER OVERFILL IN SGTR	OPA-OD2
28	2.30E-07	.54	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP. SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 8 HR AFTER SBO FROM OFFSITE W/ RCD	OP-08
			HIGH PRESSURE INJECTION FAILS - 1/4 PUMPS FROM 1/2 TRAINS	SYS-ARI12
29	2.11E-07	.49	TRANSIENTS WITH POWER CONVERSION INITIATING EVENT OCCURS	IEV-TRA
			AUXILIARY FEEDWATER SYSTEM FAILS - AF2 WITH OPA-DCNK01	SYS-AF2R
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
			FAILURE OF MAIN FEEDWATER RESTORATION AFTER AFW FAILS	SYS-MF1
			OPERATOR BLEED AND FEED COOLING FAILS	SYS-OFC
30	1.78E-07	.41	TRANSIENTS W/O POWER CONVERSION INITIATING EVENT OCCURS	IEV-TRO
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
			AUXILIARY FEEDWATER SYSTEM FAILS - AF2WO WITH OPA DCNKG1	SYS-AF2WOR
			OPERATOR BLEED AND FEED COOLING FAILS	SYS-OFC
31	1.76E-07	.41	LOSS OF COMPNT COOLING WATER INITIATING EVENT OCCURS	IEV-CCW
			LOW PRESSURE RCS RECIRC W/O RHR HX FAILS IN CCW EVENT	SYS-LPR
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
32	1.45E-07	.34	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			RCS AND RUPTURED SG PRESSURE STABILIZATION FAILS B4 OVERFIL	SYS-OD1
			SECONDARY SIDE RELIEF VALVE STICKS OPEN WHEN CHALLENGE - SGTR	OTH-SSV
			RCS COOLDOWN & DEPRESSURIZATION IN SGTR PER EMG C-31/32 FAILS	SYS-EC3

TABLE 3.4-1
DOMINANT SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
33	1.05E-07	.24	LOSS OF COMPNT COOLING WATER INITIATING EVENT OCCURS	IEV-CCW
			LOW PRESSURE RCS INVERTORY RESTORE FAILS IN CCW EVENT	SYS-LPI
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
			ONE CCW TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL CCW	CCR8-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER CCW EVENT W/ RCD	CNUC8 SUCCESSFUL
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-LC2
34	9.10E-08	.21	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS	IEV-SWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			SWS COOLING NOT RECOVERED WITHIN 2 HOURS AFTER LOSS OF ALL SWS	SWR2-FAILS
35	6.68E-08	.16	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			COMPONENT FAIL AFTER LSP, SBO EVENT RESULTS	OTH-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 8 HR AFTER SBO FROM EDG W/ RCD	ED-08
			HIGH PRESSURE INJECTION FAILS - 1/2 PUMPS FROM 1/1 TRAIN	SYS-RRI11
36	6.11E-08	.14	INTERFACING SYSTEMS LOCA INITIATING EVENT OCCURS	IEV-ISL
37	5.00E-08	.12	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS	IEV-SWS
			HRA FAILURE TO PROVIDE ALTERNTE COOLING ON LOSS OF SGK05A/B	OPA-SBCOOL
			HRA FAILURE TO SWITCH INVERTORS TO ALT POWER PER OFN 00-021	OPA-ACNN
38	4.86E-08	.11	TRANSIENTS WITH POWER CONVERSION INITIATING EVENT OCCURS	IEV-TRA
			AUXILIARY FEEDWATER SYSTEM FAILS - AF2 WITH OPA-DCNK01	SYS-AF2R
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
			FAILURE OF MAIN FEEDWATER RESTORATION AFTER AFW FAILS	SYS-MF1
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-LC2
39	4.56E-08	.11	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			AC POWER IS RECOVERED WITHIN 2 HOURS AFTER AN SBO	2HR-SUCCESSFUL
			CORE UNCOVERY OCCURS WITHIN 2 HOURS AFTER AN SBO W/O RCD	CNU2F-FAILS
40	4.42E-08	.10	TRANSIENTS W/O POWER CONVERSION INITIATING EVENT OCCURS	IEV-TRO
			BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL	DEL-ESW
			AUXILIARY FEEDWATER SYSTEM FAILS - AF2WO WITH OPA-DCNK01	SYS-AF2WOR
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-LC2

TABLE 3.4-1
DOMINANT SEQUENCES
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NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
41	4.14E-08	.10	LOSS OF A VITAL 125 VOLT DC BUS INITIATING EVENT OCCURS AUXILIARY FEEDWATER SYSTEM FAILS - 2/3 SG'S 'N DCC EVENT BLEED AND FEED COOLING FAILS IN DCC EVENT	IEV-DCC SYS-AF7 SYS-OF2
42	3.99E-08	.09	LOSS OF A VITAL 125 VOLT DC BUS INITIATING EVENT OCCURS AUXILIARY FEEDWATER SYSTEM FAILS - 2/3 SG'S IN DC' EVENT HIGH PRESSURE RECIRCULATION FAILS - AFTER DCC EVENT	IEV-DCC SYS-AF7 SYS-LC3
43	3.21E-08	.07	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS MAIN STEAM ISOLATION FAILS IN SGTR EVENT RCS COOLDOWN & DEPRESSURIZATION IN SGTR PER EMG C-31/32 FAILS	IEV-SGR SYS-MSI SYS-EC3
44	2.63E-08	.06	TURBINE BUILDING FLOOD IEV OCCURS FAILURE TO MITIGATE THE TURBINE BUILDING FLOOD EVENT	IEV-FL1 SYS-FL1
45	2.49E-08	.06	LOSS OF ALL SERVICE WATER INITIATING EVENT OCCURS AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP ONE SWS TRAIN RECOVERED WITHIN 2 HOURS AFTER LOSS OF ALL SWS CORE UNCOVERY OCCURS WITHIN 2 HOURS AFTER SWS EVNT W/O RCD	IEV-SWS SYS-AFT SWR2-SUCCESSFUL CNUS2F-FAILS
46	2.45E-08	.06	LOSS OF COMPNT COOLING WATER INITIATING EVENT OCCURS LOW PRESSURE RCS INVERTORY RESTORE FAILS IN CCW EVENT BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL CCW COOLING NOT RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL CCW	IEV-CCW SYS-LPI DEL-ESW CCR8-FAILS
47	2.42E-08	.06	LOSS OF COMPNT CO WATER INITIATING EVENT OCCURS LOW PRESSURE RCS INVERTORY RESTORE FAILS IN CCW EVENT BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL ONE CCW TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL CCW CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER CCW EVENT W/ RCD	IEV-CCW SYS-LPI DEL-ESW CCR8-SUCCESSFUL CNUC8-FAILS
48	2.02E-08	.05	SMALL LOCA INITIATING EVENT OCCURS HIGH PRESSURE SAFETY INJECTION SYSTEM FAILS - 1/4 ECCS PUMPS LOW PRESSURE SAFETY INJECTION SYSTEM FAILS	IEV-SLO SYS-SI3 SYS-SI1
49	1.83E-08	.04	LOSS OF COMPNT COOLING WATER INITIATING EVENT OCCURS LOW PRESSURE RCS INVERTORY RESTORE FAILS IN CCW EVENT BOTH ESSENTIAL SERVICE WATER SYSTEM TRAINS DO NOT FAIL ONE CCW TRAIN RECOVERED WITHIN 8 HOURS AFTER LOSS OF ALL CCW CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER CCW EVENT W/ RCD HIGH PRESSURE INJECTION FAILS - 1/2 PUMPS FROM 1/1 TRAIN	IEV-CCW SYS-LPI DEL-ESW CCR8-SUCCESSFUL CNUC8-SUCCESSFUL SYS-RRI11

TABLE 3.4-1
DOMINANT SEQUENCES
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NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
50	1.79E-08	.04	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE SAFETY INJECTION SYSTEM FAILS - 2/4 ECCS PUMPS	SYS-SI2
			LOW PRESSURE RECIRCULATION SYSTEM FAILS	SYS-LC1
51	1.55E-08	.04	ATWS INITIATING EVENT OCCURS	IEV-ATW
			FRACTION OF OPERATING TIME ABOVE 40 % POWER - ATWS EVENT	OTH-PLV
			HRA FAILURE TO MANUALLY TRIP REACTOR/INSERT RODS IN ATWS	OPA-RT
			HRA FAILURE TO OPEN RDMGS BREAKERS IN AN ATWS EVENT	OPA-MRT
			FAILURE OF PRIMARY PRESS RELIEF - ATWS - 3 SRVS 2 PORVS	OTH-PPR4
52	1.44E-08	.03	STEAM LINE/ FEED LINE BREAK INITIATING EVENT OCCURS	IEV-SLB
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/3 SG'S 1/3 PMPS SGR,SLB	SYS-AF5
			OPERATOR BLEED AND FEED COOLING FAILS - SI PUMPS OPERATING	SYS-OFB
53	1.20E-08	.03	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT OCCURS	IEV-SGR
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/3 SG'S 1/3 PMPS SGR,SLB	SYS-AF5
			AUXILIARY FEEDWATER SYSTEM FAILS - SGTR RUPTURED SG	SYS-AF4
			OPERATOR BLEED AND FEED COOLING FAILS - SI PUMPS OPERATING	SYS-OFB
54	1.14E-08	.03	ATWS INITIATING EVENT OCCURS	IEV-ATW
			FRACTION OF OPERATING TIME ABOVE 40 % POWER - ATWS EVENT	OTH-PLV
			HRA FAILURE TO MANUALLY TRIP REACTOR/INSERT RODS IN ATWS	OPA-RT
			HRA FAILURE TO OPEN RDMGS BREAKERS IN AN ATWS EVENT	OPA-MRT
			FAILURE OF PRIMARY PRESS RELIEF - ATWS - 3 SRVS 2 PORVS	OTH-PPR3

TABLE 3.4-2

CORE DAMAGE FREQUENCY BY INITIATING EVENT

INITIATING EVENT	INITIATING EVENT FREQUENCY (/YR)	CORE DAMAGE FREQUENCY (/YR)	PERCENT CONTRIBUTION
Station Blackout	2.32E-04	1.88E-05	44.87
Loss of Offsite Power	5.10E-02	4.91E-06	11.73
Control Bldg Switchgear Room Flood	4.47E-06	4.47E-06	10.67
Loss of All Service Water	1.76E-05	2.70E-06	6.45
Recoverable Control Bldg Basement Flood	8.04E-06	2.19E-06	5.24
Loss of Operating CCW Train (Leading to Seal LOCA)	1.13E-02	2.19E-06	5.23
Medium LOCA	1.10E-03	1.85E-06	4.42
Large LOCA	5.00E-04	1.37E-06	3.28
Non-Recoverable Control Bldg Basement Flood	8.94E-07	8.94E-07	2.13
Small LOCA	2.50E-03	6.67E-07	1.59
Steam Generator Tube Rupture	1.10E-02	6.26E-07	1.49
Loss of Component Cooling Water	1.62E-04	3.43E-07	0.82
Vessel Failure	3.00E-07	3.00E-07	0.72
Transients			
- With Power Conversion System	4.30	2.25E-07	0.54
- Without Power Conversion System	1.90E-01	1.67E-07	0.40
Interfacing Systems LOCA	6.11E-08	6.11E-08	0.15
Loss of DC Bus	1.78E-03	5.79E-08	0.14
Anticipated Transients without Scram	3.33E-05	3.31E-08	0.08
Turbine Bldg Flood	2.26E-02	1.53E-08	0.04
Steamline/Feedline Break	5.00E-04	1.19E-08	0.03
Room 3302 Spray (Flood)	7.45E-05	3.24E-10	0.00
TOTAL		4.19E-05	100

TABLE 3.4-3

NUREG-1335 SCREENING CRITERIA COMPARISON

CRITERIA		IMPORTANT SEQUENCES
a.	Any systemic sequence that contributes 1E-7 or more per reactor year to core damage.	Through Sequence 33 in Table 3.4-1.
b.	All systemic sequences within the upper 95 percent of the total core damage frequency.	Top 26 sequences in Table 3.4-1 contribute 95 percent.
c.	All systemic sequences within the upper 95 percent of the total containment failure probability.	See Table 4.3-1, Sequences identified.
d.	Systemic sequences that contribute to a containment bypass frequency in excess of 1E-8 per reactor year.	Sequences involving steam generator tube rupture or ISLOCA, specifically sequences 36 and 53. SGTR sequences 26, 27, 32, and 43 were determined to not go to core melt per MAAP analysis and are not included.
e.	Any systemic sequence that the utility determines from previous applicable PRA or by utility engineering judgment to be an important contributor to core damage frequency or poor containment performance.	None were identified with applicability to WCGS.

TABLE 3.4-4

DOMINANT CONTAINMENT BYPASS SEQUENCES

Table 3.4-1 Sequence Number	Frequency	Percent Contribution (to core melt)	Sequence Description
26	2.44E-07	0.57	SGTR event, AFW and cooldown fail
27	2.41E-07	0.56	SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side RV closes
32	1.45E-07	0.34	SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side RV sticks open, cooldown fails
36	6.11E-08	0.14	Interfacing System LOCA
43	3.21E-08	0.07	SGTR event, main steam isolation and cooldown fail
53	1.20E-08	0.03	SGTR event, failure of AFW and bleed and feed cooling

NOTE: MAAP runs indicate that the core will not be damaged within 24 hours for the SGTR event sequences except for sequence 53.

TABLE 3.4-5

MAJOR CONTRIBUTORS TO DOMINANT ACCIDENT SEQUENCES

EVENT DESCRIPTION	PERCENT CONTRIBUTION
AC power is not recovered within 8 hours after SBO	13.91
Turbine driven pump PAL02 fails to start and run (SBO event)	11.10
Diesel generator NE01 fails to start and run	10.62
Turbine driven pump PAL02 fails to start and run	9.59
High pressure SI restoration after SBO, SW or CCW fails	8.62
Operator failure to provide RCP seal cooling in timely manner	8.34
Diesel generator NE02 fails to start and run	7.37
Diesel generator NE01 unavailable due to test or maintenance	6.91
Core uncovery occurs within 8 hrs after SBO, RCD Successful	6.74
MOV EFHV0052 fails to open	6.47
Turbine driven pump PAL02 train unavailable due to test or maintenance	5.22
ESW Train A unavailable due to test or maintenance	5.14
ESW Train B unavailable due to test or maintenance	4.99
Diesel generator NE02 unavailable due to test or maintenance	4.55
RHR pump train B unavailable due to test or maintenance	4.47
AC power is not recovered within 2 hours after SBO	3.82

1. Sequence : SBO-08

Sequence Frequency : 5.89E-06/reactor year
Contribution to Plant Coremelt : 13.77 percent
Initiating Event Frequency : 2.32E-04/reactor year
Conditional Coremelt Frequency : 2.54E-02

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least two of the four steam generators and their associated atmospheric relief valves. Operators follow emergency operation guidelines EMG C-0, and provide rapid cooldown and depressurization to minimize a potential RCP seal LOCA due to lack of seal cooling. The plant staff works on restoring offsite AC power, and also at least one of the onsite emergency AC buses. However, neither AC power source is restored within 8 hours.

Coremelt is postulated due to either:

1. Loss of plant control and potential loss of turbine driven AFW pump operation due to DC battery depletion (loss of air supply in the accumulators for the steam generator atmospheric relief valves is also a consideration); or

2. Potential RCP seal LOCA that may develop.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Station Blackout;
2. Battery depletion after Station Blackout;
3. Potential RCP Seal LOCA after Station Blackout.

Plant Specific Nature of the sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

Credit is taken for:

1. Offsite power recovery;
2. Recovery of AC power on one emergency AC bus.
3. Credit has not been taken for AC power recovery within the time period from the assumed loss of battery power (8 hours) to the expected time of core uncovery (12 hours). Credit for AC power recovery within this time frame may be extended in the future.

2. Sequence : FL-04

Sequence Frequency	:	4.47E-06/reactor year
Contribution to Plant Coremelt	:	10.44 percent
Initiating Event Freruency	:	4.47E-06/reactor year
Conditional Coremelt Frequency	:	1.0

Sequence Description:

Flooding of rooms 3301 and 3302 containing the ESF switchgear occurs, due to a break in an ESW line in room 3301, 5501 or 5502. Due to the postulated loss of all ESW switchgear, a scenario similar to a station blackout occurs.

Coremelt is postulated due to either:

1. Loss of plant control and potential loss of turbine driven AFW pump operation due to DC battery depletion, or

2. Potential RCP seal LOCA that may develop.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Station blackout
2. Battery depletion after station blackout
3. Potential RCP seal LOCA after Station Blackout

Plant Specific Nature of the sequence:

This event sequence, which has consequences similar to those of a Station Blackout, is a low frequency severe accident sequence considered typical for currently operating PWRs. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

No credit is currently taken for recovery from this sequence. This is a recognized conservatism, and credit for operator action may be extended in the future.

3. Sequence : SBO-07

Sequence Frequency : 2.87E-06/reactor year
Contribution to Plant Coremelt : 6.70 percent
Initiating Event Frequency : 2.32E-04/reactor year
Conditional Coremelt Frequency : 1.24E-02

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least two of the four steam generators and their associated atmospheric relief valves. Operators follow emergency operation guidelines EMG C-0, and provide rapid cooldown and depressurization to minimize a potential RCP seal LOCA due to lack of seal cooling. The plant staff works on restoring offsite AC power, and also at least one of the onsite emergency AC buses. At least one of these AC power sources is restored before 8 hours into the event. However, an RCP seal LOCA has already occurred and the core is uncovered before any ECCS pumps can be started for RCS inventory restoration.

Coremelt is postulated due to an RCP seal LOCA and lack of ECCS injection before the core is uncovered.

Safety Issues Addressed:

This event sequence addresses the following safety issues:

1. Station Blackout Event;
2. RCP Seal LOCA after Station Blackout.

Plant Specific Nature of the Sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs requiring RCP seal cooling. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

Credit is taken for:

1. Offsite power recovery;
2. Recovery of AC power on one emergency AC bus;
3. Operator action of cooldown and depressurization, thus reducing the RCP seal LOCA probability.

4. Sequence : LSP-04

Sequence Frequency	:	2.86E-06/reactor year
Contribution to Plant Coremelt	:	6.67 percent
Initiating Event Frequency	:	5.1E-02/reactor year
Conditional Coremelt Frequency	:	5.6E-05

Sequence Description:

A loss of offsite AC power initiating event occurs, leading to reactor trip. Emergency AC power is automatically established on at least one of the two emergency AC power buses. Secondary cooling through at least two of the four steam generators by the auxiliary feedwater system, with steam relief via the associated steam generator atmospheric relief valves, fails. Operators initiate feed and bleed cooling by the emergency procedure EMG FR-H1. This action fails, due to either human reliability or equipment failures. The primary system loses water inventory through the pressurizer PORVs and safety valves.

Coremelt is postulated due to unmitigated LOCA through the pressurizer pressure relief valves.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Loss of Offsite Power (LSP) Event;
2. Loss of auxiliary feedwater cooling following an LSP Event;
3. Failure of the operator bleed and feed function following an LSP Event.

Plant Specific Nature of the Sequence:

An investigation of the failure combinations leading to this sequence shows that:

1. The failure mode "1 steam generator atmospheric relief valve is isolated approximately 3 % of the time in mode 1" appears;

2. The failure mode "1 pressurizer PORV block valve is closed approximately 20% of the time" appears.

These are plant specific failure modes, which, when coupled with the modeling assumptions discussed below, contribute to the sequence frequency.

Modeling Assumptions:

The following modeling assumptions contribute to the sequence frequency:

1. The success criteria for the AFW system requires delivery of flow to at least two out of four steam generators with provision for an associated steam relief mechanism for each "active" steam generator;

5. Sequence : SBO-02

Sequence Frequency	:	2.77E-06/reactor year
Contribution to Plant Coremelt	:	6.48 percent
Initiating Event Frequency	:	2.32E-04/reactor year
Conditional Coremelt Frequency	:	1.19E-02

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least two of the four steam generators and their associated atmospheric relief valves. Operators follow emergency operation guidelines EMG C-0, and provide rapid cooldown and depressurization to minimize a potential RCP seal LOCA due to lack of seal cooling. The plant staff works on restoring offsite AC power, and also at least one of the onsite emergency AC buses. AC power is recovered by restoration of offsite power before 8 hours into the event. In addition, the core has not uncovered at the time of AC power restoration. The RCS inventory restoration function is successful by injection of the RWST inventory using 1 out of 4 high pressure (Charging/SI) ECCS pumps. Recirculation of the containment recirculation sump contents, following depletion of the RWST inventory, fails due to either component failures or human reliability failure of the recirculation switchover function.

Coremelt is postulated due to failure to remove heat from the containment recirculation sump contents and return the cooled inventory to the RCS.

Safety Issues Addressed:

This event sequence addresses the following safety issues:

1. Station Blackout Event;
2. Potential RCP Seal LOCA after Station Blackout;
3. Failure of the recirculation function following Station Blackout.

Plant Specific Nature of the Sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs requiring RCP seal cooling. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

Credit is taken for:

1. Offsite power recovery;
2. Recovery of AC power on one emergency AC bus;
3. Operator action of rapid cooldown and depressurization, thus reducing the RCP seal LOCA probability;
4. Operator action to establish the RCS inventory restoration function once an AC power source is restored.
5. Consideration is included for the fact that component failures which might contribute to the occurrence of the Station Blackout event (i.e. essential service water train failure) are not recovered upon restoration of offsite AC power.

6. Sequence: SBO-36

Sequence Frequency	:	2.20E-06/reactor year
Contribution to Plant Coremelt	:	5.14 percent
Initiating Event Frequency	:	2.32E-04/reactor year
Conditional Coremelt Frequency	:	9.48E-03

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The Auxiliary Feedwater System fails to provide heat removal through at least two of the four steam generators using the turbine driven AFW pump and the associated atmospheric relief valves. An AC powe- source is restored by recovery of one of the emergency diesel generators within 2 hours of occurrence of the initiating event. In addition, the core has not uncovered at the time of AC power restoration. With failure of the AFW System to satisfy its success criteria, rapid cooldown and depressurization of the RCS in accordance with EMG C-0 cannot be performed. Failure of heat removal via the secondary side results in continued high pressure in the RCS. For this high pressure scenario, the success criteria for RCS inventory restoration, upon restoration of an AC power source, requires delivery of flow from both charging pumps and the opening of both pressurizer PORVs. However, with AC power restoration involving recovery of only one emergency diesel generator, AC power will be available to support operation of only one of the required two charging pumps. Therefore, by definition, the conditions of this scenario result in failure to satisfy the success criteria for the RCS inventory restoration function.

Coremelt is postulated due to inadequate "CCS injection with core uncovery by inventory loss through the pressurizer pressure relief valves & ossible RCP seal LOCA.

Safety Issues Addressed:

This event sequence addresses the following safety issues:

1. Station Blackout;
2. Restoration of AC power by recovery of a single emergency diesel generator following Station Blackout;
3. Failure to provide adequate ECCS injection flow to meet the RCS inventory restoration success criteria;
4. Potential RCP Seal LOCA after Station Blackout.

Plant Specific Nature of the Sequence:

This event sequence is a low frequency severe accident sequence considered typical for PWRs requiring decay heat removal and adequate ECCS injection for RCS inventory restoration. The issue of lack of RCP seal cooling and possible subsequent seal LOCA is not dominant for this scenario. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

1. Credit is taken for restoration of an AC power source by recovery of a single emergency diesel generator.

2. For this high RCS pressure scenario, inventory restoration following AC power recovery requires injection by both charging pumps. Injection using the charging pump and SI pump being supported by AC power from the operating diesel generator has not been shown to be adequate for the RCS inventory restoration function. The discharge head capacity of the SI pump is not adequate for flow delivery to the RCS, and the flow delivery to the RCS from a single charging pump is inadequate for the inventory restoration/heat removal function.

7. Sequence : FL-3B

Sequence Frequency	:	2.20E-06/reactor year
Contribution to Plant Coremelt	:	5.13 percent
itiating Event Frequency	:	8.04E-06/reactor year
Conditional Coremelt Frequency	:	2.74E-01

Sequence Description:

As an initial condition to this sequence, the system configuration assumed is operation of at least two normal plant service water pumps providing flow to the normal plant service water system and to both essential service water trains. The initiator is a pipe failure (small to large size range) occurring in the service water/essential service water system piping in Room 3101 of the control building basement, leading to flooding of the room. Depending on the size of pipe break, the water level in the room may submerge the operators for the motor-operated service water/essential service water supply side cross-tie valves (EFHV0023, 0024, 0025 & 0026) and return side cross-tie valves (EFHV0037, 0038, 0039, 0040, 0041 & 0042) before switchover from the normal plant service water supply to the essential service water supply. Subsequent to the pipe break and flood, system failure occurs due to either 1) failure to access the room and manually operate the valves to isolate the break and allow switchover to ESW (resulting in a loss of all service water event), or 2) failure to mitigate the assumed RCP seal LOCA once service water cooling was restored.

Coremelt is postulated due to failure to mitigate the possible RCP seal LOCA.

Safety Issues Addressed:

This event sequence addresses the following safety issues:

1. Internal flooding due to service water/essential service water system pipe break;
2. Recovery from the pipe break and flood (and associated loss of all service water) by local manual action in response to room high sump level alarms;
3. Failure to mitigate the RCP seal LOCA upon recovery of the service water cooling function.

Plant Specific Nature of the Sequence:

A pipe break of the service water line and a potential resultant seal LOCA are typical sequences for PWRs. This sequence is not plant specific in nature.

Modeling Assumptions:

1. Credit is taken for gaining access to room 3101

2. Manually transferring valves EFHV0023, -24, -25 and -26, thereby isolating the ruptured service water lines.

3. Providing service water flow via the ESW pumps.

8. Sequence: CCW-31

Sequence Frequency : 2.19E-06/reactor year
Contribution to Plant Coremelt : 5.11 percent
Initiating Event Frequency : 1.134E-02/reactor year
Conditional Coremelt Frequency : 1.93E-04

Sequence Description:

This sequence is initiated by loss of the operating Component Cooling Water System train (assumed to be CCW Train A), with a frequency of loss determined on a per reactor year basis. With loss of the operating CCW train, cooling is lost to the Reactor Coolant Pump (RCP) Thermal Barrier Cooling Coils (TBCC), due to loss of flow through the CCW service loop. In addition, pump cooling is lost to the operating charging pump which is providing RCP seal cooling via seal injection flow. The standby CCW train (CCW Train B) is successfully started with service water cooling aligned to the associated CCW heat exchanger. However, RCP seal cooling is not restored prior to seal damage due to failure to successfully realign the CCW service loop to the operating CCW Train B; and failure to successfully start the charging pump, which is cooled by CCW Train B, and align it for seal injection cooling. The running RCPs were successfully tripped prior to occurrence of extensive seal damage (i.e.; an RCP seal LOCA of Small LOCA magnitude is assumed due to loss of RCP seal cooling). The RCP seal LOCA (Small LOCA) is not successfully mitigated due to failure of the single ECCS train (Train B) which is supported by the operating CCW train.

Coremelt is postulated due to failure to mitigate the assumed RCP seal LOCA using the ECCS Train B components.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Loss of RCP seal cooling due to loss of the supporting CCW train on a per year basis;
2. Failure to realign RCP seal cooling prior to occurrence of a postulated RCP seal LOCA;
3. Actions to trip any running RCPs on loss of RCP seal cooling and/or CCW cooling (RCP motors);
4. Postulated RCP Seal LOCA on loss of all seal cooling.

Modeling Assumptions:

1. Recovery of RCP seal cooling within 30 minutes by either restoration of seal injection flow, or cooling to the TBCCs by realignment of the CCW service loop to the operating CCW train, will prevent occurrence of an RCP seal LOCA provided any running RCPs are stopped within 10 minutes after a loss of all seal cooling.

2. Failure to stop running RCPs within 10 minutes after a loss of all seal cooling will result in seal damage to such an extent that an RCP Seal LOCA of Medium LOCA magnitude is postulated.

9. Sequence: MLO-02

Sequence Frequency	:	1 [illegible] -06/reactor year
Contribution to Plant Coremelt	:	4.[illegible] percent
Initiating Event Frequency	:	1.10E-03/reactor year
Conditional Coremelt Frequency	:	1.69E-03

Sequence Description:

A Medium LOCA (2 inch to 6 inch diameter break) initiating event occurs, leading to reactor trip. Accumulator safety injection is available from at least two of the three accumulator tanks connected to the intact RCS loops. High pressure safety injection is successful with delivery of RWST inventory by at least two out of four high pressure ECCS (Charging/SI) pumps into the three intact RCS cold legs. High pressure recirculation of the containment recirculation sump contents, following depletion of the RWST inventory, is not successful due to either component failures or human reliability failure of the recirculation switchover function.

Coremelt is postulated due to failure to remove heat from the containment recirculation sump contents and return the cooled inventory to the RCS.

Safety issues Addressed:

1. Medium LOCA Event;
2. Failure of the recirculation function following a Medium LOCA.

Modeling Assumptions:

1. Failure to stop any operating RHR pumps early in the injection phase of the Medium LOCA event is conservatively assumed to result in pump failure due to extended operation without CCW supply to the RHR heat exchanger for cooling of the flow being circulated through the minimum flow loop;

2. The isolation valves in the CCW inlet lines to the RHR heat exchangers (EGHV0101 & 0102) are assumed to be maintained in the closed position. Successful operation of an RHR train for the recirculation function requires opening of the associated CCW inlet isolation valve.

3. No credit is taken for operator action to recover a failed component. For example, operator action to manually open either EGHV0101 or 0102 using the valve operator handwheel may be readily accomplished in a timely manner should one or both of these valves fail to open on demand. However, credit for such recovery actions was not included in the quantification of the Medium LOCA event.

10. Sequence: SBO-38

Sequence Frequency : 1.65E-06/reactor year
Contribution to Plant Coremelt : 3.84 percent
Initiating Event Frequency : 2.32E-04/reactor year
Conditional Coremelt Frequency : 7.11E-03

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The Auxiliary Feedwater System fails to provide heat removal through at least two of the four steam generators using the turbine driven AFW pump and the associated atmospheric relief valves. With failure of the AFW System to satisfy its success criteria, rapid cooldown and depressurization of the RCS in accordance with EMG C-0 cannot be performed. AC power is not restored within 2 hours following the initiating event. For a station blackout event with failure of decay heat removal via the secondary side, steam generator dryout may be expected at approximately one hour with subsequent RCS heatup and inventory loss through the pressurizer relief valves. Core uncovery in this scenario may be expected at approximately two hours following the station blackout event.

Coremelt is postulated due to eith :

1. Core uncovery due to high pressure RCS inventory loss via the pressurizer relief valves; or

2. Potential RCP seal LOCA that may develop.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Station Blackout Event;
2. AFW System failure after Station Blackout;
3. Failure of AC power recovery following Station Blackout;
4. RCS heatup and inventory loss via the pressurizer pressure relief valves after Station Blackout with failure of secondary side decay heat removal;
5. Potential RCP Seal LOCA after Station Blackout.

Plant Specific Nature of the sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs requiring decay heat removal and adequate ECCS injection for RCS inventory restoration. The issue of lack of RCP seal cooling and possible subsequent seal LOCA is not dominant for this scenario. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions

1. Successful secondary side decay heat removal requires AFW flow to at least two out of four steam generators from the turbine driven AFW pump, with steam relief provided by the associated SG atmospheric relief valves.

2. Credit is taken for probabilities of recovering either offsite AC power or one of the emergency AC buses.

11. Sequence: SBO-05

Sequence Frequency : 1.64E-06/reactor year
Contribution to Plant Coremelt : 3.83 percent
Initiating Event Frequency : 2.32E-04/reactor year
Conditional Coremelt Frequency : 7.07E-03

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The turbine driven AFW pump automatically starts and provides heat removal through at least two of the four steam generators and their associated atmospheric relief valves. With shedding of selected DC loads, the plant batteries are expected to last for around 8 hours. Operators follow emergency operation guidelines EMG C-0 and provide rapid cooldown and depressurization of the RCS to minimize a potential RCP seal LOCA due to lack of seal cooling. An AC power source is restored by recovery of one of the emergency diesel generators by 8 hours into the event. In addition, core uncovery has not occurred by the time of AC power recovery. With recovery of an AC power source, RCS inventory restoration proceeds with injection of the RWST contents into the RCS cold legs via at least one of the two ECCS (Charging/SI) pumps powered from the single operating diesel generator. Recirculation of the containment recirculation sump contents, following depletion of the RWST inventory, fails due to either failure of the ECCS components powered from the single operating diesel generator or human reliability failure of the recirculation switchover function.

Coremelt is postulated due to failure to remove heat from the containment recirculation sump contents and return the cooled inventory to the RCS.

Safety issues addressed:

This event sequence addresses the following safety issues:

1. Station Blackout Event;
2. Potential RCP Seal LOCA after Station Blackout;
3. Failure of the recirculation function following Station Blackout.

Plant Specific Nature of the Sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs requiring RCP seal cooling. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

Credit is taken for:
1. Recovery of AC power on one emergency AC bus;
2. Extension of expected battery life up to 8 hours due to shedding of selected DC loads.

12. Sequence : SBO-34

Sequence Frequency	:	1.53E-06/reactor year
Contribution to Plant Coremelt	:	3.56 percent
Initiating Event Frequency	:	2.32E-04/reactor year
Conditional Coremelt Frequency	:	6.59E-03

Sequence Description:

A station blackout initiating event (loss of offsite AC power and loss of onsite emergency AC power) occurs, leading to reactor trip. The Auxiliary Feedwater System fails to provide heat removal through at least two of the four steam generators using the turbine driven AFW pump and the associated atmospheric relief valves. An AC power source is restored by recovery of offsite power within 2 hours of occurrence of the initiating event. In addition, the core has not uncovered at the time of AC power restoration. With failure of the AFW System to satisfy its success criteria, rapid cooldown and depressurization of the RCS in accordance with EMG C-0 cannot be performed. Failure of heat removal via the secondary side results in continued high pressure in the RCS. The RCS inventory restoration function for this high pressure scenario is successful by injection of the RWST inventory using both Charging pumps along with the opening of both pressurizer PORVs. Recirculation of the containment recirculation sump contents, following depletion of the RWST inventory, fails due to either component failures (2 of 2 CCPs required) or human reliability failure of the recirculation switchover function.

Coremelt is postulated due to failure to remove heat from the containment recirculation sump contents and return the cooled inventory to the RCS.

Safety issues Addressed:

This event sequence addresses the following safety issues:

1. Station Blackout Event;
2. Turbine driven AFW failure following Station Blackout;
3. Possible RCP Seal LOCA after Station Blackout;
4. Failure of the recirculation function following Station Blackout.

Plant Specific Nature of the Sequence:

This event sequence is a low frequency severe accident sequence considered typical for currently operating PWRs requiring long term decay heat removal after AC power restoration following a Station Blackout. The issue of lack of RCP seal cooling and possible subsequent seal LOCA is not dominant for this scenario. There is no additional plant specific failure mode observed for this sequence.

Modeling Assumptions:

Credit is taken for:
1. Offsite power recovery;
2. Successful secondary side decay heat removal requires AFW flow to at least two out of four steam generators from the turbine driven AFW pump, with steam relief provided by the associated SG atmospheric relief valves.
3. Operator action to establish the RCS inventory restoration function.

13. Sequence: LSP-05

Sequence Frequency	:	1.31E-06/reactor year
Contribution to Plant Coremelt	:	3.06 percent
Initiating Event Frequency	:	5.10E-02/reactor year
Conditional Coremelt Frequency	:	2.57E-05

Sequence Description:

A loss of offsite AC power initiating event occurs, leading to reactor trip. Emergency AC power is automatically established on at least one of the two emergency AC power buses. The CCW train which was operating at the time offsite power was lost (assumed to be CCW Train A) fails to start and run. With loss of the operating CCW train, cooling is lost to the Reactor Coolant Pump (RCP) Thermal Barrier Cooling Coils (TBCC), due to loss of flow through the CCW service loop. In addition, pump cooling is lost to the operating charging pump which was providing RCP seal cooling via seal injection flow. The standby CCW train at the time offsite power was lost (CCW Train B) is successfully started with service water cooling aligned to the associated CCW heat exchanger. However, RCP seal cooling is not restored prior to seal damage due to failure to successfully realign the CCW service loop to the operating CCW Train B; and failure to start, if necessary, the charging pump which is cooled by CCW Train B and successfully align it for seal injection cooling. Extensive seal damage due to the loss of RCP seal cooling does not occur since the RCPs were stopped by the loss of offsite AC power initiator (i.e. an RCP seal LOCA of Small LOCA magnitude is assumed due to loss of RCP seal cooling). The RCP seal LOCA (Small LOCA) is not successfully mitigated due to failure of the single ECCS train (Train B) which is supported by the operating CCW train.

Coremelt is postulated due to failure to mitigate the assumed RCP seal LOCA using the ECCS Train B components.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Loss of offsite power (LSP) event;
2. Loss of the operating CCW train following an LSP;
3. Failure to realign RCP seal cooling prior to occurrence of a postulated RCP seal LOCA;
4. Postulated RCP seal LOCA on loss of all seal cooling.

Modeling Assumptions:

1. Recovery of RCP seal cooling within 30 minutes by either restoration of seal injection flow, or cooling to the TBCCs by realignment of the CCW service loop to the operating CCW train, will prevent occurrence of an RCP seal LOCA following a loss of offsite power event.

2. No credit is taken for restoration of either offsite AC power, or the one emergency AC bus which might have failed.

14. Sequence: LLO-02

Sequence Frequency	:	1.15E-06/reactor year
Contribution to Plant Coremelt	:	2.68 percent
Initiating Event Frequency	:	5.0E-04/reactor year
Conditional Coremelt Frequency	:	2.30E-03

Sequence Description:

A Large LOCA (greater than 6 inch diameter break) initiating event occurs. Accumulator safety injection is available from at least two of the three accumulator tanks connected to the intact RCS loops. Low pressure safety injection is successful with delivery of RWST inventory by at least one out of two low pressure ECCS (RHR) pumps into at least one of three intact RCS cold legs. Low pressure recirculation of the containment recirculation sump contents, following depletion of the RWST inventory, is not successful due to either component failures or human reliability failure of the recirculation switchover function.

Coremelt is postulated due to failure to remove heat from the containment recirculation sump contents and return the cooled inventory to the RCS.

Safety Issues Addressed:

1. Large LOCA Event;
2. Failure of the recirculation function following a Large LOCA.

Modeling Assumptions:

1. The isolation valves in the CCW inlet lines to the RHR heat exchangers (EGHV0101 & 0102) are assumed to be maintained in the closed position. Successful operation of an RHR train for the recirculation function requires opening of the associated CCW inlet isolation valve.

2. No credit is taken for operator action to recover a failed component. For example, operator action to manually open either EGHV0101 or 0102 using the valve operator handwheel may be readily accomplished in a timely manner should one or both of these valves fail to open on demand. However, credit for such recovery actions was not included in the quantification of the Large LOCA event.

15. Sequence : SWS-13

Sequence Frequency : 9.91E-07/reactor year
Contribution to Plant Coremelt : 2.31 percent
Initiating Event Frequency : 1.76E-05/reactor year
Conditional Coremelt Frequency : 5.63E-02

Sequence Description:

A loss of all service water initiating event occurs, leading to plant trip due to loss of cooling to the turbine auxiliary systems. Loss of all service water is defined as loss of service water flow through at least one safety related service water train from either the associated essential service water system pump train or at least two of the four normal plant service water pumps (three full flow and one low flow). The Auxiliary Feedwater System fails to provide heat removal through at least two of the four steam generators using the turbine driven AFW pump and the associated atmospheric relief valves. With failure of the AFW System to satisfy its success criteria, cooldown and depressurization of the RCS does not occur. Service water cooling to one safety related service water train is restored within 2 hours following the initiating event. In addition, core uncovery due to RCS inventory loss has not occurred by the time of service water restoration. With failure of the cooldown and depressurization function, the RCS is at a high pressure at the time of recovery of cooling water flow through the one safety related service water train. For this high pressure scenario, the success criteria for RCS restoration following service water recovery requires opening both pressurizer PORV's and injection using both Centrifugal Charging Pumps. Since only one safety related service water train is recovered, cooling support is available for operation of only one of the two CCPs. Therefore, by definition, the conditions of this scenario result is failure to satisfy the success criteria for the RCS inventory restoration function.

Coremelt is postulated due to inadequate ECCS injection with core uncovery by inventory loss through the pressurizer pressure relief valves and possible RCP seal LOCA.

Safety Issues Addressed:

This event sequence addresses the following safety related issues:

1. Loss of all service water event;
2. AFW System failure after a loss of all service water;
3. Restoration of service water cooling support by recovery of flow through one safety related service water train;
4. Failure to provide adequate ECCS injection flow to meet the RCS inventory restoration success criteria;
4. Potential RCP seal LOCA after a loss of all service water.

Modeling Assumptions:

1. Credit is taken for actions to recover service water flow to one of the safety related service water trains. Plant specific component recovery probability distributions were developed based on input from WCGS plant staff;
2. Successful secondary side decay heat removal requires AFW flow to at least two out of four steam generators from the turbine driven AFW pump, with steam relief provided by the associated SG atmospheric relief valves. No credit is taken for operation of the motor driven auxiliary feedwater pumps since a loss of all service water would result in a loss of room cooling, with assumed loss of the associated pumps. This modeling assumption will be explored further in the future to possibly credit operator action for providing auxiliary room cooling (i.e. opening doors) to the motor driven AFW pump rooms.
3. For this high RCS pressure scenario, inventory restoration following service water recovery requires injection by both charging pumps. Injection using the charging pump and SI pump being supported by the one train of service water cooling has not been shown to be adequate for the RCS inventory restoration function. The discharge head capacity of the SI pump is not adequate for flow delivery to the RCS, and the flow delivery to the RCS from a single charging pump is inadequate for the inventory rest ration/heat removal function.

3.4.2 Vulnerability Screening

A product of the Wolf Creek PRA is the identification of important plant specific severe accident scenarios. These scenarios take the form of event tree accident sequences which can be used in the assessment of the applicability of various severe accident issues to Wolf Creek. The results of the WCGS PRA have been evaluated against the NUMARC 91-04 Closure Guidelines. The Guidelines were used to identify insights related to severe accidents.

The first step in using the Closure Guidelines was to group the core damage sequences. The groups used were those of Table B-2 of the Closure Guidelines. The grouping was carried out for the top core damage sequences down to a frequency cutoff of 1E-08 for a given sequence. Below that value, sequences were contributing no more than 0.02% to the total core damage frequency and did not significantly influence the evaluation results. In addition, the flooding sequences were not included in the grouping.

One or more sequences were identified in the following Closure Guideline groups:

IA	Accident with loss of primary and secondary heat removal in injection phase
IB	Accident with loss of primary and secondary heat removal in recirculation phase
IIA	Induced LOCA with loss of injection
IIB	Induced LOCA with loss of recirculation
IIIA	Small LOCA with loss of injection
IIIB	Small LOCA with loss of recirculation
IIIC	Large/Medium LOCA with loss of injection
IIID	Large/Medium LOCA with loss of recirculation
IV	Failure of reactivity control
VA	Systems LOCA outside containment
VB	Steam generator tube rupture accidents

The sequence numbers included in each group are listed in Table 3.4-7 with the resulting mean core damage frequency and percent contribution to the total core damage frequency.

The core damage frequency and percent contribution to the total core damage frequency for each group were then evaluated against Tables 1 and 2 of the Closure Guidelines. Table 2 was used for the containment bypass sequences (groups VA and VB only), and Table 1 was used for all other groups.

The comparison shows that only the IA grouping is of interest from the standpoint of the Closure Guidelines. These are the accidents with loss of primary and secondary heat removal in the injection phase. Of those initiators which are part of this grouping, the station blackout event is dominant.

The IA group falls in the category in Table 1 of the Guidelines which suggests that the Licensee: 1) Find a cost effective treatment in EOPs or other plant procedure or minor hardware change with emphasis on prevention of core damage, or 2) If unable to satisfy above response, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure and containment failure.

TABLE 3.4-7

NUMARC CLOSURE GUIDELINES SEQUENCE GROUPING INFORMATION

Sequence Group	Sequence #	Core Damage Frequency	Group % Contribution To Total CDF
IA	SBO - 1,6,10,21,28,35	SBO - 1.04E-05	24.8
	Others - 4,15,23,29, 30,34,37,41,52,53	Others - 4.74E-06	11.3
		TOTAL = 1.51E-05	36.1
IB	5,11,12,17,38,40,42	6.90E-06	16.4
IIA	3,20,25,39,45,46,47, 49	3.76E-06	9.0
IIB	19,31,33	7.86E-07	1.9
IIIA	8,13,48	3.52E-06	8.4
IIIB	18	6.72E-07	1.6
IIIC	22,24	5.83E-07	1.4
IIID	9,14,50	3.03E-06	7.2
IV	51,54	2.69E-08	0.0
VA	36	6.11E-08	0.1
VB	26,27,32,43	6.62E-07	1.6

3.4.3 DECAY HEAT REMOVAL EVALUATION

This section provides a brief evaluation of the decay heat removal functions at the WCGS based upon the results from the IPE as required by the Generic Letter 88-20. The purpose of the evaluation is to identify potential decay heat removal vulnerabilities and to examine whether or not risks attributed to the loss of decay heat removal can be lowered in a cost-effective manner.

The decay heat removal during the first 24 hours following a plant trip is accomplished by the following key functions at WCGS:

During small break LOCA and transient without power conversion system (PCS) events, decay heat removal is via the auxiliary feedwater (AFW) system through the secondary side. If the AFW system fails, bleed and feed operations are needed on the primary side. The bleed and feed operation requires a high pressure injection system, the pressurizer PORVs, and the associated operator actions.

During a transient with PCS event, decay heat is removed via the AFW or main feedwater systems. If both of these systems fail, bleed and feed operations are needed on the primary side.

During a medium LOCA event, decay heat is removed via the AFW system, the steam generator atmospheric relief valves, and directly by the emergency core cooling system (ECCS). This includes the high pressure (charging and SI pumps) and low pressure injection and recirculation systems and the associated operator actions.

During a large LOCA event, decay heat is removed directly by the low pressure (RHR pumps) injection and recirculation system.

Given that successful decay heat removal depends upon the above systems and operations, the following discussions of these functions and their respective features are provided. These discussions start with the AFW system for the small break LOCA and transient events and end with the recirculation systems for the medium and large break LOCA events.

- The AFW system consists of three redundant trains. The two trains of motor-driven AFW pumps feed two steam generators each. The turbine-driven AFW pump is capable of feeding all four steam generators. The normal water supply to the AFW is from the CST and is supplemented by essential service water. The essential service water supply is modeled in the WCGS PRA AFW system fault trees since the CST may not provide enough water to supply the system for a full 24 hours.

  The failure probability of the AFW system was calculated to have a low value of 2.19E-05 with all support systems available. Therefore, the failure of the AFW system was found to be an insignificant contributor to core damage frequency.

- For a transient event with PCS available, if the AFW system is not available, the operators are directed to establish an alternate feedwater supply to the steam generators via the main feedwater system.

  The MFW system includes the following components: two turbine-driven main feedwater pumps, a separate motor-driven feedwater pump, four main feedwater isolation valves, main feedwater control valves, and three motor-driven condensate pumps.

  The MFW system is required to operate in the highly improbable event that the AFW system fails. The failure probability of the MF1 event, which represents the failure of the MFW and condensate system, was calculated to be 4.38E-02, with all support systems available.

- With no feedwater flow to the steam generators, the operators are instructed per EMG FR-H1 to initiate primary side bleed and feed cooling. The cooling path feed function is from the high pressure ECCS (charging and SI) system with the bleed function accomplished by opening the pressurizer PORVs.

  Two of two pressurizer PORVs are required for the bleed function in order to prevent over pressurization of the RCS during various accident conditions. The feed function requires operation of one of four high pressure ECCS pumps to provide cooling to the RCS.

  The bleed and feed operation is used only when the AFW and/or MFW fails, but as described above, this is very improbable. The operator diagnosis and action was calculated to have a failure probability of 1.75E-03. The failure probability of the whole bleed and feed operation is 6.8E-03.

- The high pressure injection portion of the ECCS consists of two SI and two charging pumps. The failure probability of the high pressure injection function was calculated to be relatively low: 1.63E-05 (SI2) for a medium LOCA; 7.23E-08 (SI3) for a small LOCA event; with all support systems available.

- The low pressure injection portion of the ECCS consists of 2 low pressure pumps which can be used when the RCS pressure is below the shutoff head of the RHR pumps. The failure probability of this function (SI1) was calculated as 6.13E-05, with all support systems available.

- When the low level RWST alarm setpoint is reached, the operators transfer from the injection to the recirculation phase based upon EMG ES-12. The operator failure to perform ECCS recirculation is included in the high pressure and low pressure recirculation fault tree models. The failure probability of this operator action for high pressure ECCS recirculation is 4.97E-04 and the probability for the low pressure recirculation is 9.17E-04.

- The high pressure recirculation system (LC2) consists of the two SI and two charging pumps which can be used to maintain the plant in a long-term stable condition for sequences with the RCS pressure above the RHR pump shutoff head. This mode of operation requires alignment of the high pressure ECCS pumps suction to the discharge of the low pressure ECCS pumps. The failure probability of this system, including the operator action, was calculated as 1.53E-03, with all support systems available.

- The low pressure recirculation system (LC1) consists of two RHR pumps which can be used to maintain the plant in long term stable condition for sequences with the RCS pressure less than the RHR pump shutoff head. The failure probability of this system including the operator action, was calculated to be 1.21E-03, with all support systems available.

The dominant contributors to plant core damage are discussed in detail in Section 3.4.1. The dominant contributors (cutsets), relating to decay heat removal, whose frequency is greater than 1E-7/yr are presented below.

Core Damage Frequency	Initiating Event	System Failures(s)
5.47E-07	MLO	Operator fails high pressure switchover for recirculation
4.59E-07	LLO	Operator fails low pressure switchover for recirculation

As presented above, the failure to switch to the ECCS recirculation phase appears to be the dominant contributor relating to failure of the decay heat removal function. The two sequences listed above contribute less than three percent to the total plant core damage frequency.

To summarize the information presented in this section, there are several redundant means for decay heat removal at WCGS. Many of the decay heat removal systems and associated operator actions have relatively low failure probabilities. Several of these systems and operator actions would have to fail in combination to have an impact on the decay heat removal capability at WCGS and that impact is very small. Therefore, cost-effective improvements to the decay heat removal systems such as AFW, MFW, PORVs and ECCS were not explored further.

3.4.4 USI and GSI Screening

This report addresses two unresolved safety issues:

- A-17, "System Interactions in Nuclear Power Plants," was resolved in Generic Letter 89-18 by referring the "internal flooding" issue to the IPE. This issue has been addressed in Section 3.3.7 and it was concluded that no significant vulnerabilities exist in that area.

- A-45, "Decay Heat Removal Evaluation", is addressed in Section 3.4.3. The conclusion from that evaluation is that there are no significant vulnerabilities in that area.

4.0 BACK-END ANALYSIS

This section presents the IPE back-end analysis for WCGS. Back-end analysis is also refered to as Level 2 Containment Analysis.

4.1 PLANT DATA AND PLANT DESCRIPTION

4.1.1 Containment Description

The WCGS containment structure is described below, along with those containment systems which are important to the containment and source term analysis. Detailed plant-specific data are used to model these containment features so as to realistically evaluate the containment response to a core melt accident.

4.1.1.1 Containment Structure

WCGS employs a large, dry containment design. Figures 2.4-7 and 2.4-8 illustrate two vertical sections of the containment. The containment structure is a horizontally and vertically prestressed, post-tensioned concrete cylindrical structure with a hemispherical dome and a conventionally reinforced concrete base slab with a central cavity and instrumentation tunnel to house the reactor vessel. The containment free volume is approximately 2.5E6 ft^3. The inside surface of the structure is lined with a one quarter (1/4) inch thick carbon steel liner plate to ensure a high degree of leak tightness. The containment building foundation is a 10-foot-thick reinforced concrete mat, 154 feet in diameter, founded 11 feet below plant grade. The central reactor cavity and instrumentation tunnel extend below the reactor building foundation, with the bottom of the 5.5-foot-thick foundation slab located 36 feet below grade. The 8-foot-wide tendon access gallery, located beneath the perimeter of the reactor building mat, has a 4.25-foot-thick foundation slab, the bottom of which is 25.25 feet below grade.

Principal nominal dimensions of the reactor building are as follows:

Interior diameter	140 ft
Interior height	205 ft
Height to spring line	135 ft
Base slab thickness	10 ft
Cylinder wall thickness	4 ft
Dome thickness	3 ft
Liner plate thickness	0.25 in
Internal free volume	2.5 x 10^6 cubic ft

The Modular Accident Analysis Program (MAAP) is used in the WCGS PRA to model the plant and containment. In MAAP terminology, the large containment volume above the operating deck (at the 2047' 6" elevation), is referred to as the "Upper Compartment." The "Lower Compartment" is that portion of containment, between the containment floor (2001' 4" elevation) and the operating deck, which is inside the secondary shield wall but outside the reactor shield wall (i.e., the primary shield wall). The "Annular Compartment" is that part of containment below the operating deck but outside the secondary shield wall. The Annular Compartment floor elevation (2000' 0") is 16 inches below the floor of the Lower Compartment.

The open design and significant venting areas for the sub-compartments within the WCGS containment help ensure a well-mixed atmosphere, a feature which inhibits combustible gas pocketing, even for cases in which the hydrogen mixing fans are inoperable. Steel grating around the periphery of the operating deck provides a good flow path between the annular and upper compartments. The lower and upper compartments communicate through large openings around the steam generators and above the reactor coolant pumps. The lower and annular compartments communicate through four manways on the 2001' 4" elevation.

Figures 2.4-2 and 2.4-4 show plan views of the WCGS containment at the 2000' and 2047' 6" elevations, respectively. The four primary system loops and the pressurizer are located inside the secondary shield wall, in the lower compartment. The four accumulator tanks and the pressurizer relief tank are located outside the secondary shield wall, in the annular compartment.

Figures 2.4-1 and 2.4-6 illustrate the reactor cavity and instrument tunnel, along with the containment structure in the vicinity of the seal table. The cavity floor area is about 648 ft^2. The seal table configuration is an important feature of the WCGS containment because it provides an effective structural barrier to debris entrainment from the cavity following a high-pressure vessel blowdown.

The WCGS containment does not facilitate flooding of the reactor cavity. Although water can easily drain from the upper compartment to the annular and lower compartment floors, the water on the lower compartment floor would preferentially drain into the annular compartment since a 6" curb exists around the cavity manway and the annular compartment floor is 16" below that of the lower compartment. Thus, the annular compartment would have to be flooded to a depth of 22" and the lower compartment to a depth of 6", which would require a total liquid volume of ~114,500 gallons , prior to water entry into the cavity via the cavity manway opening. This implies that only limited flooding of the cavity region is possible for sequences in which injection of the RWST inventory occurs and that the cavity would be essentially dry otherwise.

In addition to the description here, containment isolation features are also discussed in section 3.2.1.5. Personnel access to the containment is normally provided through an airlock with interlocked doors. This access hatch and the equipment hatch are located in the upper compartment. An emergency airlock is located in the annular compartment. The equipment, personnel, and emergency hatches all employ non-metallic gaskets.

All containment penetrations are double barrier assemblies consisting of a closed sleeve or a double gasketed closure for the fuel transfer tube. The mechanical systems penetrations are welded to the containment liner. Likewise, the electrical penetration assemblies (EPAs) are constructed to provide a leak tight barrier. The EPAs employ non-metallic seals and potting compounds. There are no penetrations in the immediate vicinity of the seal table enclosure.

4.1.1.2 Containment Systems

The WCGS design includes two containment cooling systems:

1) Four Containment Fan Coolers and
2) Two Containment Spray Trains.

However, only the containment fan cooler system is designed to remove decay heat (long term) from the containment for design basis accident (DBA) events. Containment spray can only be considered as a short-term containment pressure reduction system. The Residual Heat Removal System, although not a containment system, also provides a means of long term containment heat removal. Brief descriptions of the Fan Coolers, Containment Spray, and RHR are provided below. (Also refer to sections 3.2.1.4, 3.2.1.6, and 3.2.1.8.2.)

Containment Fan Coolers

Four main containment cooling units are located at the 2068' 8" elevation in the upper compartment. The cooling units normally take suction from the upper compartment and discharge to the bottom of the containment building through four separate duct systems. Cool air is circulated to the regions around the reactor coolant pumps and the steam generators and then flows upward through large openings in the operating deck floor. When the temperature of the upper compartment gas reaches 160°F, however, a thermal link in the containment fan cooler duct fails and the fan discharge is redirected back to the upper compartment.

Containment Spray System

The Containment Spray System initially takes suction from the Refueling Water Storage Tank (RWST) and injects into the containment via spray headers located on the containment dome. The spray system provides both a potential (short-term) pressure reduction mechanism and a means to remove fission products from the containment atmosphere. It also provides a means for injecting water into the containment.

At the RWST LO-LO-2 alarm, the switchover of containment spray pumps to the recirculation mode is manually initiated. When the containment spray system is in the recirculation mode, the spray pumps take suction from the containment recirculation sumps and discharge to the containment spray headers. There are no heat exchangers in the spray system, therefore, spray recirculation alone cannot remove decay heat from containment.

Residual Heat Removal (RHR) System

During the recirculation mode, the RHR pumps take suction from the two containment recirculation sumps. The containment water is cooled in the RHR heat exchanger and returned to the RCS. Since the RHR heat exchanger cooling water is supplied by the component cooling water (CCW) system which uses the service water system as a heat sink, loss of either of these two systems will negate the containment heat removal capability of the RHR system.

4.1.2 Containment Data

The Modular Accident Analysis Program (MAAP) is used in the WCGS PRA to provide an integrated approach to modeling of plant and containment thermal-hydraulic response and fission product behavior during severe core-damage accidents. MAAP requires plant-specific input data which is compiled into a MAAP Parameter File. The WCGS Parameter File provides a complete, realistic description of WCGS for a MAAP simulation, and the parameter file data is identical for all accident sequences.

4.2 Plant Models and Methods for Physical Processes

The WCGS containment and source term analysis is part of the traditional Level 2 analysis. It includes plant models and physical processes which reflect the overall plant behavior following core damage. This is accomplished by coupling a probabilistic assessment of containment response to postulated initiating events with a physical model to examine plant response. Sequences for initiating events which are dominant contributors to plant risk and other sequences which are judged to be of interest were used in this process. This process also includes the impact of phenomenological uncertainties.

The probabilistic models are embodied in Containment Event Trees (CETs) while the plant physical model is defined in a MAAP parameter file as discussed in Section 4.1.2. This parameter file provides MAAP with information required by the code to perform calculations of plant-specific fission product transport and thermal-hydraulic response to postulated accident sequences. It is also used to study the sensitivity of the source term to phenomenological uncertainties. The MAAP analyses are supplemented with WCGS-specific phenomenological evaluation summaries to provide a complete physical representation of WCGS.

Results obtained with the probabilistic and physical plant models are closely linked. For instance, the CET structure depends on MAAP analyses to 1) define CET nodal success criteria, 2) establish timing of key events for understanding of sequence progression, and 3) determine the accident sequence outcomes. Furthermore, sequences demonstrated by the quantification task to be either dominant contributors to the overall core damage frequency or of structural interest become the basis for MAAP calculations in support of the source term analysis. Finally, MAAP analyses and phenomenological evaluation summaries are used to investigate the effect of phenomenological uncertainties on the source term assessment. The use of MAAP, as suggested, above provides the necessary deterministic complement to the probabilistic analysis. A detailed discussion of the containment event tree models is provided below, followed by a closer examination of the MAAP models and the treatment of key phenomenological issues.

4.2.1 Containment Event Trees (CETs)

The primary function of the containment event tree (CET) is to describe the containment response to a core melt accident accounting for phenomenological issues and system/human behavior. This is accomplished by defining a functional set of top events along with their failure and success states. Each combination of top event success and failure states then leads to a unique CET end state which provides information about ex-vessel sequence progression, containment status, and source term release. Quantification of the CET is performed based on the core damage sequences which come out of the screening process as discussed in Section 4.3.3.1. This quantification results in the assignment of a CET end state to each of the selected Level 1 sequences. Following CET quantification, dominant sequences are selected for source term analysis. These sequences are representative of the entire spectrum of quantified sequences from the CET. Further discussion of the dominant sequence selection is contained in Section 4.3. Thus, in addition to describing the accident sequence beyond core melt, the CET also serves as a directory for binning of sequences in the source term analysis.

The general guidelines used for development of a CET are summarized as follows:

1. CET top events and structure describe the containment response and account for system/human behaviors which most strongly influence the source term assessment.

2. The CET structure provides enough detail such that the severity of the fission product release can be distinguished between CET end states.

3. The CET considers factors which dominate the containment response, thus the top events consider broad categories of systems and phenomena. For example, it is important to know whether water is available in the containment since this will have a major impact on debris coolability and fission product

retention. It is not critical to know exactly which system is supplying the water as long as the water inventory is accounted for. Thus, a CET top event might consider containment wet or dry rather than the availability of individual containment systems.

4. Containment failure modes resulting in early containment failure due to phenomenological uncertainties (i.e., early failure due to ex-vessel steam explosion, early failure due to direct containment heating [DCH], etc.) as described in NRC guidance will not be treated as separate top events, but rather through the phenomenological evaluation summaries and MAAP uncertainty studies. However, they can appear as end states where phenomenological evaluation summaries dictate a specific plant vulnerability.

5. Discussions of the CET top event success criteria consider the impact of success or failure of the node on the source term. This will provide guidance during the binning process.

Based on these guidelines, the structure of the CET has been arranged to first determine the status of the containment and then consider a series of nodes which will describe the accident progression. First, containment status is reflected through a decision having a direct bearing on source term release, as to whether or not the containment is isolated. Accident progression is then addressed through a series of nodes involving phenomenological concerns and/or the availability of system functions pertaining to containment failure and source term analysis. To be consistent with the guidance provided in Appendix A of NUREG-1335, the second node determines if the vessel is at high or low pressure at the time of vessel failure. A high pressure vessel failure would require consideration of some of the phenomenological issues in the WCGS Phenomenological Evaluation Summaries, namely direct containment heating, vessel thrust forces, and steam explosions. The third decision considers the timing of containment failure relative to vessel failure - an important parameter for source term phenomena - because this timing determines if natural fission product removal mechanisms, such as gravity, will have enough time to be effective in reducing source term. Subsequent decision nodes determine the status of those functions (debris coolability, decay heat removal, fission product scrubbing) that either prevent containment failure or mitigate fission product release given that containment failure has occurred. Details of the decision nodes (CET Top Events) and containment success criteria are included in the following sections.

Embodied within the Level I event trees or contained within an intermediate, Containment Safeguards event tree to bridge the Level 1 and Level 2 assessments are the containment systems such as fan coolers and sprays. These systems are maintained within the CET so that the severity of the source term release can be distinguished between CET end states. The CET as described above would remain applicable to those cases where containment failure occurs prior to vessel failure.

As a final bridge between the Level 1 and 2 efforts, the following assumptions have been applied:

- Core damage under Level 1 leads to vessel failure.
- Associated with each plant damage state (PDS) is a set of functional systems. Those systems successful under Level 1 are considered functional under Level 2.
- Failed systems for a plant damage state (PDS) are considered failed throughout Level 2.
- Justification for the basis of the timing of important events (operator actions) is determined in the Level 1 effort. Level 2 analysis takes no credit for operations or recoveries not initiated under Level 1.

4.2.2 CET Top Events and Success Criteria

A number of top events can be considered which produce a CET that describes the ex-vessel sequence progression and that can be used in the source term binning process. Top events which were given consideration were items such as failure of containment isolation for the given accident initiator, availability of water systems to provide both short term and long term coolability of core debris expelled from the reactor vessel, capabilities for decay heat removal from the containment, and operator actions important to sequence progression. Additionally, some severe accident phenomena have been represented by CET top events. Based on a review of the NRC guidance and previous work, a CET for the WCGS has been developed as represented in Figures 4.2-1. The CET top events (nodes) and their success criteria are defined as follows.

Containment Isolation Intact

Containment isolation, as used here, is in agreement with the definition set forth in the WCGS Event Tree Analysis which states that containment isolation "refers to the closure of containment penetrations to limit the release of radioactive fluids following an accident." Containment isolation will be stated as part of the Plant Damage State and can therefore be expressed in the CET with a 0/1 split fraction for a specified sequence.

The impact of success or failure of this CET node is primarily on the timing of source term release. A failure to isolate containment results in an early fission product source term release following the onset of core damage. This source term release will be characterized as receiving no benefit from long term, natural fission product removal mechanisms such as settling.

Low Pressure Vessel Failure

The purpose of this node is to allow quantification of high pressure versus low pressure core damage. For those postulated severe accident scenarios in which a substantial pressure is available within the primary system at the time of vessel failure, high pressure melt ejection could potentially displace core debris into the lower compartment. Entrainment of debris by the steam/hydrogen mixture exiting the vessel and passing through the cavity region is also a possibility. Debris leaving the cavity is deposited in the containment lower compartment as the kinetic energy of the flowing gases decreases and the core debris becomes de-entrained.

The gas velocity required to entrain molten debris can be characterized by the value of the superficial gas velocity required for supporting liquid films.

Following reactor pressure vessel (RPV) failure, the gas velocity and its likelihood of exceeding the "critical" velocity for entrainment increases with increasing RCS pressure. Thus, sequences which result in a high pressure melt ejection following RPV failure exhibit varying degrees of debris displacement and entrainment from the cavity to the containment lower compartment. Typical low pressure sequences, such as a large LOCA, however, result in all of the debris remaining in the cavity region (i.e., no entrainment). In addition to the RCS pressure, the degree of entrainment is influenced by the cavity/instrument tunnel geometry and the amount of molten debris present at the time of RPV failure. The determination of high pressure failure is based on a primary system pressure of 400 psig.

The plant damage state definition for each dominant core melt sequence from the Level 1 analysis includes an indication of the RCS pressure at the time of core damage. Coupled with additional knowledge of the Level 1 sequence progression, this will define *a priori* the likelihood of a high pressure melt ejection (HPME) following RPV failure. Thus success occurs for this node if the RCS is at low pressure prior to RPV failure and HPME is prevented (i.e., all debris remains in the cavity), and failure is defined as when HPME could occur (i.e., high pressure creates the possibility for debris entrainment to the lower compartment).

The occurrence of a high pressure melt ejection can affect containment response by either inducing an early containment failure or by influencing long term sequence progression. It will also impact source term by increasing the airborne fission product concentration. Postulated early containment failure modes resulting from uncertainties surrounding vessel blowdown thrust forces, direct containment heating, and steam explosions are discussed and discounted in the phenomenological evaluation summaries. The impact of HPME on the long term containment response is due to the resulting debris distribution. Debris distribution will affect the requirements for maintaining debris coolability, the degree of molten core-concrete interactions, and the steaming rate of containment water pools.

Late Containment Failure Mode

A late containment failure mode occurs long after vessel failure and allows time for natural fission product removal mechanisms to reduce the mass of airborne fission products in containment. The containment failure mode (early or late) has a large impact on source term. Success of this node for a given sequence, which is defined as a late containment failure mode or containment failure limited to leakage, will be determined based on the results of the phenomenological evaluation summaries and/or MAAP runs.

The NRC, in NUREG/CR-2300, has identified a number of phenomena which could potentially result in early containment failure. Due to the uncertainty surrounding these phenomena, the NRC has recommended that they be considered in the CET. Since the likelihood of early containment failure due to these phenomenological uncertainties is highly dependent on plant specific containment geometries, the present methodology treats these items individually through phenomenological evaluation summaries. These summaries provide a detailed, WCGS specific analysis of the various phenomena and discuss the likelihood and consequences of the phenomena.

Success for this node occurs if none of the pertinent phenomenological uncertainties result in early containment failure as discussed in the summary papers. For the WCGS, it has been found that the occurrence of the phenomena listed in Table 4.2-1 will not threaten the containment structural integrity nor result in an early containment failure. Therefore, this node has a probability for success of 1, for CET quantification.

Although no early containment failure modes are expected, the phenomenological uncertainties could impact the long term, ex-vessel sequence progression. Such effects will be captured in the uncertainty studies. This top event has been included primarily to indicate that phenomenological uncertainties have been considered through the phenomenological evaluation summaries.

Debris Coolable

Debris coolability can be achieved when heat removal from the debris bed exceeds the debris internal decay heat generation rate. Thus, a coolable debris configuration will eventually result in a frozen, or solid, debris bed, while a non-coolable configuration will produce a molten debris pool. Several mechanisms for debris heat removal can be postulated. For instance, conduction heat transfer will occur at the interface of core debris and concrete bodies. If this is the dominant mode of debris heat removal, then concrete attack by the debris will result. For debris pools exposed to the containment atmosphere, radiation heat transfer from the debris to the surrounding gasses and mechanical structures is significant. Finally, for debris beds immersed in water pools, convective heat transfer characterized by nucleate boiling is the dominant heat removal mechanism.

Some uncertainty exists with regard to the ability to cool molten core debris. The NRC has stated (in Generic Letter 88-20) that a debris depth less than 25 cm (9.84 in) may be considered coolable. The WCGS containment geometry is such that if 100% of the core debris is postulated to evenly spread in either the cavity or lower compartment, then the debris depth would be less than 25 cm. Therefore, if an adequate supply of water exists to ensure that the debris is immersed in a water pool throughout the duration of the plant mission time, then debris coolability can be assumed.

Short term debris coolability can be obtained if debris exiting the RPV after vessel failure falls into a water pool and is rapidly quenched. However, since debris coolability occurs via nucleate boiling of the overlying water pool, unless the water pool is replenished it will eventually boil away exposing the core debris and resulting in a reheating of the debris bed. This outcome can be classified as a failure of the Debris Coolable node on the CET.

To replenish the water pool and maintain long term coolability, either an external water source (i.e., RWST) or a recirculation system (i.e., RHR system) must supply a flow rate which exceeds the water pool steaming rate. Thus, the water supply must be sufficient to remove debris decay heat generation. The time available to establish this makeup system is dependent on the debris distribution and the initial water mass available in the containment. Typically, by the time water pool dryout occurs, decay heat levels are within the heat removal capacities of one containment fan cooler unit, or one RHR pump plus heat exchanger. Therefore, operation of one of these systems will ensure success of the CET node for debris coolability. If flow from an external water source, such as from refilling of the RWST, can be established for the duration of the mission time, then this will also result in success of the node. Since systems available for use in the Level 2 analysis are already accounted for in the Level 1 quantification, a review of the plant damage state will reveal the state of the Debris Coolable node on the CET. Thus, for a particular sequence, an appropriate split fraction of 0 or 1 can be assigned to this node.

Success of the Debris Coolable node will prevent core-concrete attack and the accompanying generation of aerosols and airborne fission products. Failure of this node will lead to substantial core-concrete attack and an eventual breach of the containment boundary due to a failure of the cavity basemat. Thus, the Debris Coolable node will have a strong influence on the potential radionuclide release from containment.

Containment Heat Removal

The interaction of core debris with containment water pools, mechanical structures, and atmosphere will result in a heat up and pressurization of the containment. This pressurization will be a function of the containment free volume, the rate of gas production (condensible and non-condensible) and the rate of increase of the containment gas temperature. Since the gas production and temperature rise can be characterized by the levels of decay heat generation within the core debris, it is necessary to establish some form of containment heat removal which meets or exceeds the decay heat generation rates. Failure to do so will result in sustained containment pressurization and an eventual failure at the containment boundary.

The Containment Heat Removal node accounts for the containment response due to the success or failure of containment heat removal systems. Failure of this node will result in a failure of the containment due to overpressure while success will maintain the pressure within the capacity of the containment. During the time period when containment integrity can be challenged due to overpressure, the debris decay heat levels are sufficiently low so that the operation of one containment fan cooler unit or one RHR pump plus heat exchanger can adequately remove containment heat. Thus, operation of one of these systems is required for success of the Containment Heat Removal node. Operation of containment sprays without any heat removal mechanism will affect the timing of containment overpressure but will not prevent its occurrence, therefore containment sprays are not included as part of the Containment Heat Removal success criteria. The split fraction for this node for a particular sequence will be a 0 or 1 depending on the system availability as defined in the plant damage state.

Failure of this node will result in a late containment failure and a release of fission products from the containment. Success of this node will prevent containment failure within the prescribed mission time, however success of the Debris Coolable node is also required to ensure no containment failure will occur. Success or failure of the Containment Heat Removal node will have a significant impact on the ex-vessel sequence progression and the source term quantification.

Fission Product Scrubbing

The quantity and type of radionuclides released following a reactor containment failure during a severe core melt accident is sensitive to the mechanisms available for fission product scrubbing. Fission product scrubbing refers to the removal of radioactive particles from a gas space due to some filtration process. In pressurized water reactors, no filtered containment vents exist therefore fission product scrubbing must be achieved through the operation of containment sprays or the presence of water pools overlying debris beds. Water used in this way can be an extremely effective tool for fission product scrubbing.

Success of the Fission Product Scrubbing node implies that operation of the containment sprays or the presence of water pools have effectively reduced the airborne fission product content. Failure of this node, however, implies that no reduction in the amount of airborne fission products can be credited. A review of the Level 1 sequence definition will be sufficient to determine whether or not fission product scrubbing will occur. Based on this sequence specific assessment, the appropriate split fraction of 0 or 1 can be assigned to the Fission Product Scrubbing node.

The presence of overlying water pools or the operation of containment sprays is closely linked to the state of the Debris Coolable and the Containment Heat Removal nodes. This node differs from the previous ones in that only the fission product removal potential is currently considered rather than the ability to preserve containment integrity as was considered in the previous nodes. Thus, the Fission Product Scrubbing node will not impact the containment failure potential, but it will have a noticeable effect on the fission product source term should containment failure occur.

4.2.3 CET Structure and End States

The CET top events, as described in Section 4.2.2, have been arranged in a manner which takes into account the expected sequence progression and provides insight into the containment response to a postulated severe accident sequence. The combination of top event success and failure states leads to 14 possible CET end states. These end states provide a qualitative description of the ex-vessel sequence progression and source term release and will be useful during the binning process. Among the end states, a number of possible outcomes are shown. These outcomes are defined as follows.

> Leakage - The containment integrity will not be challenged due to either overpressurization or basemat penetration. Minor releases of airborne fission products may occur along normal containment leakage pathways.
>
> Late Containment Failure on Over Pressure - Containment failure due to overpressurization will occur resulting in fission product release. No fission product scrubbing is credited.
>
> Late Containment Failure on Over Pressure - Reduced Fission Product Release - This is essentially the same as the above, except that the source term will be reduced due to fission product scrubbing.
>
> MCCI Induced Containment Failure - Late Containment failure is

expected due to basemat failure resulting from prolonged molten core-concrete interactions (MCCI). This assumes that basemat failure occurs prior to an overpressurization failure and does not credit fission product scrubbing.

Early Containment Failure - Containment failure occurs immediately following RPV failure due to an isolation failure. This results in an early fission product release without the benefit of natural fission product removal mechanisms or fission product scrubbing mechanisms.

Early Containment Failure - Reduced Fission Product Release - This is essentially the same as the above, except that the source term will be reduced due to fission product scrubbing.

4.2.4 CET Quantification

As a precursor to the source term analysis, CET quantification is performed using those core damage sequences which came out of the screening process as discussed in 4.3.3.1. The CET quantification assigns each selected core damage sequence, along with its frequency, to a particular CET end state. These end states and the cumulative frequencies form the basis for binning of like sequences to simplify the source term analysis effort.

The CET quantification process involves following the event tree branching logic for a given sequence to arrive at a particular CET end state. Since the combination of CET top event success and failure states leading to a particular CET end state are largely predetermined by the Level 1 sequence definition, the split fractions for each CET branch can readily be assigned as 0's and 1's. In general, sequences of like initiating event which fall into the same CET end state will have similar radiological consequences and can therefore be binned together for source term analysis purposes. In some cases, different CET end states are binned together. The frequencies for all sequences with the same CET end state are summed up to determine the total frequency for each end state.

To clarify the CET quantification process, an example based on the Large LOCA quantification is provided below. Large LOCA sequences LLO2 and LLO3 passed the screening criteria (See Section 4.3.3.1) and were assessed with the Containment Safeguards Tree. The outcome of the Containment Safeguards Event Tree is summarized in a Plant Damage States (PDS) table which is shown in Table 4.3-1. As an example, consider the plant damage states for sequence LLO3. The portion of the PDS table pertaining to large LOCAs is repeated here as Table 4.2-2 while the large LOCA event tree is shown in Figure 3.1-2. If information provided by the Level 1 definition for sequence 3 in Figure 3.1-2 (Initiator: LLO, ACC success, SI1 failure) is combined with results from the Plant Damage State Table, then the CET end state can be derived as shown in

Table 4.2-3.

Examination of the core damage sequences from all initiating event categories leads to the complete set of quantified CETs shown in Figures 4.2-2 through 4.2-13. Table 4.2-4 summarizes the end state frequencies resulting from the CET quantification. This summary provides insight into the importance of various systems with regards to preventing containment failure.

A review of Table 4.2-4 indicates that the paths leading to containment failures are dominated by late overpressurization failures caused by failures of the containment cooling systems (CET end states 3 and 8), and isolation failures (CET end state 14). Table 4.2-5 provides the total frequency of containment failure for each of the failure modes considered in the CET (excluding containment by-pass sequences).

Late MCCI-induced basemat failures due to failure to inject and recirculate the RWST water inventory (CET end states 6, 10, and 12) do not occur within the 48 hour Level 2 mission time. Thus, this release mode is characterized by normal containment leakage rather than containment failure.

4.2.5 Containment Failure Characterization

Plant-specific phenomenological evaluations have been performed in support of the WCGS PRA to determine the likelihood of all postulated containment failure modes and mechanisms identified in NUREG-1335. These detailed evaluations were performed systematically to address the controlling physical processes or events specific to the WCGS configuration. Modeling and bounding calculations, based upon extensively compiled experimental data, phenomenological uncertainties, and complemented with MAAP calculations in some cases, comprise the general approach taken in these evaluations. Several postulated containment challenges are demonstrated, through the phenomenological evaluations, to be inconsequential for the WCGS containment. These potential failure modes are considered to be very unlikely to occur at WCGS since the predicted pressures resulting from a realistic assessment of these failure mechanisms are far less than the containment ultimate strength.

The failure modes considered unlikely to occur within the 48 hour Level 2 mission time are hydrogen combustion, direct containment heating, steam explosions, molten-core concrete attack, thermal attack of containment penetrations, and vessel thrust forces. More likely to occur are containment over-pressure, containment isolation failure, and containment bypass. Table 4.2-6 summarizes the results of these containment failure mode evaluations.

4.2.5.1 Containment Ultimate Strength

A plant-specific structural analysis of the WCGS containment was conducted to determine its internal pressure capacity and its most likely failure locations. The results of the ultimate pressure analysis determined that the total median (50%) failure pressure is 127.6 psig while the 5% lower bound and the 95% upper bound are 99 psig and 136 psig, respectively.

At containment pressures below 123 psig, the dominant containment failure location is at the large pipe penetrations. Even though the mean capacity of the large pipe penetrations is relatively high (215 psig), this weak link is dominant at lower pressures due to the large uncertainty (logarithmic standard deviation, β = 0.273). At containment pressures above 123 psig, the probability of failure at the containment mid-height region rises sharply and becomes the dominant failure location. This result should be expected since this weak link has the lowest mean capacity (136.4 psig) of any considered and a relatively small uncertainty (β = 0.0497). As the total failure probability approaches unity, failure at the containment mid-height accounts for up to 84% of the total, while failures at the large pipe penetrations contributes 14%. The wall/base slab interface and access openings have a combined contribution up to 2%.

The containment fragility analysis provides a total failure probability curve (i.e., a fragility curve) for the WCGS containment. This curve shows that the median containment failure pressure is 127.6 psig (142.3 psia). Thus there is a 50% probability that the containment will fail at or below this pressure. The overall lower bound ultimate capacity is 99 psig. Thus, there is only a 5% probability of containment failure at or below this pressure.

The best estimate containment failure mode will occur at 127.6 psig due to membrane stresses in the containment mid-height region which exceed the prestress and cause through-concrete cracking and yielding of the liner, reinforcing steel, and prestressing tendons. The conservative, or lower bound, containment failure mode will occur at 114 psig due to tearing of the liner around either the containment purge line or fuel transfer tube penetrations. For applications in MAAP analyses, the lower bound failure mode is used.

4.2.5.2 Unlikely Failure Modes

Hydrogen Combustion

Potential detonability and flammability of the WCGS containment atmosphere are evaluated as part of the PRA. Detonation is evaluated based on both geometric configuration and detonation cell width scaling. Both of these methods conclude that the likelihood of deflagration to detonation transition (DDT) is very low. It is far more likely that combustible gas would be consumed within containment by deflagration rather than detonation. Even for the total station blackout at WCGS, the worst case sequence with respect to hydrogen combustion, it is unlikely that enough hydrogen would accumulate so as to produce a deflagration that could challenge the containment ultimate pressure capacity. Furthermore, the containment would most likely fail due to over-pressurization well before such a large amount of hydrogen could accumulate. None of the sequences addressed in the containment and source term analysis could realistically threaten containment due to hydrogen combustion.

Direct Containment Heating (DCH)

The relevant experiments for DCH have been reviewed and have produced one specific conclusion: given the necessary RCS conditions for high pressure melt ejection, containment structures (geometry) have a first order (dominant) mitigating influence on the potential for DCH. The use of mechanistic models for debris dispersal, which take into account entrainment from the cavity and de-entrainment at the tunnel exit, to evaluate the containment response to a high pressure melt ejection shows the resulting pressurization to be much less than a value that would challenge containment integrity.

Steam Explosions

Separate approaches are used to address in-vessel and ex-vessel steam explosions. The IDCOR work, which is consistent with the recommendation of the NRC sponsored Steam Explosion Review Group, forms the basis for the treatment of in-vessel steam explosions. Results of analyses performed in accordance with significant-scale experiments and expansion characteristics of shock waves form the basis for the treatment of ex-vessel steam explosions.

It is concluded that the slumping of molten debris into the RPV lower plenum could not result in sufficient energy release to threaten the vessel integrity and hence would not lead directly to containment failure. Likewise, evaluations of both the steam generation rate and shock waves induced by ex-vessel explosive interactions show that these would not be of sufficient magnitude to threaten the containment integrity.

Molten Core-Concrete Attack

Molten core-concrete attack within the WCGS containment cavity is evaluated for the most severe accident sequence (i.e., loss of all a.c. power) using a simple, bounding analysis model which assumes that the concrete ablation rate is proportional to the total heat generation rate due to decay heat and chemical reactions. The model uses empirical parameters determined from available experimental data. The evaluation indicates that melt-through of the containment basemat would not occur until well beyond the 24 hour Level 1 mission time. Furthermore, containment failure due to over-pressurization would occur long before basemat melt-through. Therefore, molten core concrete attack is not a l .ely containment failure mode for WCGS.

Thermal Attack of Containment Penetrations

The evaluation of debris dispersal in conjunction with the location of the mechanical and electrical penetrations reveals that it is unlikely for these penetrations to be in direct contact with molten debris dispersed during postulated high pressure melt ejection. The majority of entrained debris would be removed at the seal table enclosure. There are no direct paths by which corium could contact any containment penetrations. The operational limit of the non-metallic materials are shown not to be exceeded by the maximum gas temperatures predicted for containment compartment regions during severe accident sequences. Hence, thermal loading of penetration non-metallic materials would not cause degradation and leakage from the containment under conditions expected at WCGS during a severe accident.

Vessel Thrust Force

The bounding analysis for the magnitude of the thrust force when molten core debris is ejected from the failed reactor at high pressure indicates that this force cannot lift the dead weight of the vessel itself, given a credible break size in the RPV. The likelihood of vessel thrust force causing the reactor to shift its position is then highly unlikely. Even if the vessel could shift, the WCGS containment is configured so that reaction forces cannot be transmitted to the containment wall. Therefore, this postulated failure mode is bounded by the plant design.

4.2.5.3 Failure Modes

Containment Over-pressure

Containment over-pressure, defined as a failure mode caused by steaming and/or generation of non-condensible gases, can be a potential containment failure mode within 48 hours for WCGS. Depending on the specific accident sequence characteristics, over-pressure failures may be observed across a wide range of event times. The potential for containment overpressure failure exists in severe accident scenarios where sufficient containment heat removal is not available. Containment failure is dominated by failure to inject the RWST, failure to align for recirculation, and failure of all containment fan coolers.

Over-pressure failure is expected to be a slow mechanism such that containment pressurization would be approached gradually. The resulting stresses in the containment structures would only induce a relatively small rupture area (i.e., "Leak-before-break" behavior). This failure area is small enough to stabilize the containment pressure at a level below the failure pressure. This is supported by all of the experimental evidence for steel lined concrete containment structures. As discussed in Section 4.2.5.1, the failure location is conservatively assessed at the large pipe penetrations.

Containment Isolation Failure

Containment isolation failure is a possible containment failure mode at WCGS. Containment isolation failure refers to mechanical or operational failure to close containment fluid system penetrations which communicate directly with the containment prior to, or following, the initiation of core damage, in order to limit fission product release to the auxiliary building or to the environment. Containment isolation would fail on one or more of the following conditions:

1) A fluid line or mechanical penetration, which is required to be closed during power operation, has been left unisolated.

2) A fluid line, which has isolation valves which are required to close on an isolation signal, fails to isolate, or

3) A fluid line, which is part of a safety system and is required to remain open following the generation of isolation signals, is not closed by the operators if the system is "failed" or the operation of the system is terminated.

In all of the above conditions for fluid systems, all check valves in fluid lines must also fail to close in order for impaired containment isolation to occur. For example, if a line is protected by two motor operated isolation valves and one check valve, all three must fail to close (possibly different failure modes) to create an unisolated containment condition.

Critical containment penetrations (i.e., those that can lead to significant fission product releases to the environment if they fail to close) are identified based on either of the following screening criteria:

1) the line penetrating containment is a containment sump or reactor cavity sump drain line, or

2) the line penetrating containment is greater than 2 inches in diameter and directly communicates with the containment atmosphere and it is not part of a closed system outside of containment capable of withstanding severe accident conditions.

Failure of containment isolation is addressed in the Containment Event Trees (CETs).

Containment Bypass

Containment bypass is another possible failure mode for WCGS. Containment bypass refers to failure of the pressure boundary between the high pressure RCS and a lower pressure line penetrating containment. This results in a direct pathway from the reactor coolant system to the auxiliary building or the environment, bypassing the containment. Containment bypass is usually considered as an accident initiator that can lead to core damage because the loss of cooling fluid to a location outside containment prohibits the use of ECCS recirculation for long term core cooling. The likely mechanisms for this failure mode, identified for WCGS as being significant in terms of both frequency and potential consequences, are (1) an interfacing systems LOCA and (2) a steam generator tube rupture.

4.2.6 Post Core Damage Accident Progression

Loss of coolant from the primary system, either through a break in the coolant boundary or a loss of heat sink (which in turn promotes over-pressurization of the RCS and subsequent loss of fluid through the safety valves), coupled with failure to inject the RWST, eventually results in uncovering of the reactor core. Core damage occurs once oxidation of the Zircaloy fuel cladding begins. This exothermic chemical reaction between steam and Zircaloy generates heat and produces hydrogen. The reaction is controlled by the availability of steam, which continues to be generated as the primary system inventory boils off. The reaction rate accelerates when the temperature of the Zircaloy exceeds 2871°F (1850 K), and the chemical energy released at this point in the transient exceeds the local decay heat generation. Core melt begins when the fuel temperature reaches the eutectic melt temperature of 4040°F (2500 K).

As the core melts, molten material candles downward until it refreezes on cooler material below. Eventually it re-melts and moves further downward. This downward progression is mainly a function of the temperatures encountered by the melt. Once the melt leaves the core boundaries, it begins attacking the core support structures. Large holes in the lower core support plate allow relocation of the core to the lower plenum of the reactor vessel without melting the entire lower core support plate structure.

In the absence of external cooling of the RPV, relocation of the molten core into the lower head is assumed to lead directly to failure of the reactor vessel; no attempt is made to take credit for potential in-vessel recovery. If the RCS is at high pressure at the time of vessel failure, then high pressure melt ejection (HPME) could possibly displace or entrain a small amount of core debris out of the cavity. However, the majority of this debris is de-entrained by containment structures. If the RCS is at low pressure at the time of vessel failure, then low pressure melt ejection results in no core debris escaping the cavity.

If no water is available to cool the core debris or if the debris boils away the water, then molten core-concrete interaction (MCCI) takes place. Concrete decomposition generates non-condensible gases and also releases a significant amount of water from the concrete, resulting in additional chemical heat generation and hydrogen evolution due to oxidation of metallic constituents within the molten debris. The containment continues to pressurize due to heating of the containment atmosphere and non-condensible gas generation. If no containment heat removal is available, this pressurization induces containment failure.

The time required to fail the containment by over-pressurization depends upon the steaming rate and upon the rate of non-condensible gas generation. The failure mechanism associated with containment overpressure is due to exceeding the ultimate strength of certain key structural components or attachments. This limit is most likely to be approached gradually, so that the energy delivered is only sufficient to induce a relatively small rupture area.

The severity of the source term depends strongly on the containment failure timing. Failure in the immediate time period of vessel failure is clearly the most serious, as the overall airborne fission product mass produced during a severe accident is never greater than it is in the small span of time directly after vessel failure. Whenever containment pressurization lags considerably behind vessel failure, substantial fission product retention through naturally occurring deposition mechanisms (e.g., sedimentation, impaction, etc.) is facilitated.

Finally, failure to isolate the containment results in the direct release of fission products from the containment following core damage. The source term for sequences involving a failure to isolate the containment is to a large degree determined by the area of the isolation failure, the pressure in the containment, and the time at which core damage begins.

Table 4.2-1

POTENTIAL EARLY CONTAINMENT FAILURE MECHANISMS

- Vessel Blowdown Thrust Forces
- Direct Containment Heating
- Steam Explosions
- Hydrogen Combustion
- Thermal Attack of Containment Penetrations

Table 4.2-2

WCGS PLANT DAMAGE STATES FOR LARGE LOCA SEQUENCES

Event Tree Sequence ID	Event Sequence	Fan Coolers	Cont. Sprays	Cont. Isolation	Frequency
LLO3A	SYS-SI1	F	F	A	7.39E-09
LLO3B	SYS-SI1	A	F	A	0.00E+00
LLO3C	SYS-SI1	F	A	A	0.00E+00
LLO3D	SYS-SI1	A	A	A	2.75E-07
LLO2A	SYS-LC1	F	F	A	0.00E+00
LLO2B	SYS-LC1	A	F	A	0.00E+00
LLO2C	SYS-LC1	F	A	A	0.00E+00
LLO2D*	SYS-LC1	A	A	A	1.15E-06
LLO2E	SYS-LC1	A	A	F	1.98E-10

- Frequency of 0.0 means less than 1.0E-15 or excluded due to cutset quantification limit reached
  - A - System Available
  - F - System Failed
- The Sequence I.D.s are derived from the Level I Event Trees (e.g., SBO4C, Station Blackout - Sequence Number 4); and a letter designator for the availability of Containment Cooler, Spray, and Isolation systems as follows:
  - **A** No Coolers, No sprays
  - **B** Coolers; No Sprays
  - C Sprays; No Coolers
  - D Coolers and Sprays
  - E Isolation Failure

* Sequences greater than 1.0E-06 were also treated as containment isolation failures. The containment isolation failure scalar was multiplied by the plant damage states' frequency as follows:

For Station Blackout w/o early power recovery - 2.309E-03
All other sequences - 1.728E-04

Table 4.2-3

Illustration of CET Application to Large LOCA Sequences for WCGS

TOP EVENT	LLO3A QUANTIFICATION	LL03D QUANTIFICATION
Initiator	The sequence initiator is a large LOCA (LLO).	The sequence initiator is a large LOCA (LLO).
Containment Isolation	The PDS indicates containment isolation is available, so this node is a success.	The PDS indicates containment isolation is available, so this node is a success.
Low Pressure Vessel Failure	The Level 1 sequence definition indicates low RCS pressure, therefore, no HPME will occur. This node is a success.	The Level 1 sequence definition indicates low RCS pressure, therefore no HPME will occur. This node is a success.
Late Containment Failure Mode	Since the Low Pressure Vessel Failure node was a success, all containment failures will be late, therefore this node is not considered.	Since the Low Pressure Vessel Failure node was a success, all containment failures will be late, therefore this node is not considered.
Debris Coolable	The PDS indicates that containment heat removal and spray systems are failed. Also, the Level 1 event tree indicates that SI injection fails, thus the RWST contents are never injected. This results in a dry cavity configuration and no debris coolability. Therefore this node is a failure.	Although SI injection is a failure (SI1 failure), the PDS Table indicates that containment spray systems are operable (CSS available), therefore RWST contents will be injected into the containment resulting in a wet cavity configuration. This will ensure debris coolability, thus this node is a success.
Containment Heat Removal	The PDS Table indicates that containment cooling systems are not available (CCS failure), therefore this node is a failure.	The PDS Table indicates that containment cooling systems are available (CCS available), therefore this node is a success.
Fission Product Scrubbing	Since RWST fails to inject (SI1 failure) and containment spray systems fail (CSS failure), no means of fission product scrubbing is available, therefore this node is a failure.	Since containment heat removal is successful, this node is not considered.
End State	The success and failure states of the CET nodes leads to CET end state 6 - Late containment failure.	The success and failure states of the CET nodes leads to CET end state 1 - Normal Leakage.

Table 4.2-4
WCGS CONTAINMENT EVENT TREE END STATE SUMMARY

	CET END STATE													
INITIATOR[1]	1	2	3	4	5	6	7	8	9	10	11	12	13	14
CCW	2D 1.76E-7													
DCC									3A 5.80E-9					
FLD												FL4A 4.47E-6		FL4E 1.03E-8
LLO	2D,3D 1.43E-6					3A 7.39E-9								2E 1.98E-10
LSP							3D,4D,5D 4.67E-6	3C,4C 1.41E-7	3A 5.20E-9					4E,5E 6.88E-10
MLO							2D 1.87E-6		2A 8.79E-9					2E 3.24E-10
SBO	2D,3D,5D 7DED,7DOP 6.18E-6		5A 1.20E-6				34D,35D,36B, 36D 3.98E-06	36C 1.07E-8				8A,36A,38A 7.81E-6		2E,5E,7E,8E 34E,36E, 38E 3.80E-08
SLO							3D 6.72E-7		3A 1.85E-8					
SWS	2B,2D,3B 3D 7.73E-7	2C,3C 4.45E-9	2A 2.79E-9				13D 9.91E-7	13C 4.37E-9		13B 8.92E-9		5A, 13A 2.93E-7		
TRA							5D 2.11E-7							
TRO							4D 1.77E-7					4A 1.10E-9		
VEF	1D 3.0E-7													
TOTAL	8.86E-6	4.45E-9	1.20E-6			7.39E-9	1.26E-05	1.56E-7	3.83E-8	8.92E-9		1.25E-5		4.95E-08

NOTE: 1. MAAF analyses performed for underlined sequences.

2. The sequences in this table are from Table 4.3-1 and Figures 4.2-2 through 4.2-13. Sequences CCW4D, CCW31D, CCW31E, FL3A, FL3B, FL3BE, SGR3D, SGR5D, SGR14A, SGR14D, SWS4B, SWS4C, and SWS4D were excluded from binning and source term analysis based upon specific MAAP case analysis.

Table 4.2-5

Containment Failure Mode Summary

Containment Failure Mode Mode	Release Mode Frequency	CET End States
Normal Leakage	2.15E-05	1,7
Late MCCI Induced Failure *	1.25E-05	4,5,6,10,11,12
Late Overpressurization Failure	1.40E-06	2,3,8,9
Impairment	4.95E-08	14
Early Failure	0.00E+00	13
All Modes Combined	3.63E-05	

* Late MCCI-induced basemat failures due to failure to inject and recirculate the RWST water inventory do not occur within the 48 hour Level 2 mission time. Thus, this release mode is characterized by normal containment leakage rather than containment failure.

Table 4.2-6
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PHENOMENOLOGICAL EVALUATION SUMMARIES ON POSTULATED CONTAINMENT FAILURE MODES

FAILURE MODE	PHENOMENA	ISSUE/FAILURE MECHANISM	MAJOR UNCERTAINTY	IMPACT
1. Hydrogen Combustion	In-vessel H_2 generation Ex-vessel H_2 generation Steam inerting Auto ignition	Breach containment by overpressurization due to H_2 burn or detonation	Amounts of H_2 and CO Flammability of containment atmosphere	No early containment failure Long term containment failure possible if inappropriate recovery action
2. Direct Containment Heating (DCH)	RPV failure Debris dispersion Influence of containment structures Hydrogen combustion/steam inerting Thermal exchange with entire air space	Early breach of containment by rapid overpressurization	Degree of dispersal in containment Hydrogen combustion	Containment pressures for DCH far less than ultimate structure capability
3. Steam Explosions	Missile generation Rapid steam generation Shock waves	Missile impact Early containment overpressurization and breach	Occurrence of multiple conditions required to produce large scale steam explosion	No threat to RPV or containment Promotes debris dispersal and cooling

Table 4.2-6
Page 2 of 3

PHENOMENOLOGICAL EVALUATION SUMMARIES ON POSTULATED CONTAINMENT FAILURE MODES

FAILURE MODE	PHENOMENA	ISSUE/FAILURE MECHANISM	MAJOR UNCERTAINTY	IMPACT
4. Molten Core-Concrete Interactions (MCCI)	Concrete ablation and decomposition Gas evolution (H_2, CO, CO_2) Debris spreading H_2 recombination	Basemat penetration after several days of attack	Presence of water to quench debris Debris coolability	Overpressurization would occur before basemat penetration Basemat penetration yields a "buried" fission product (FP) release path
5. Vessel Blowdown	RPV rupture RPV thrust forces RPV restraints	Failure of containment penetration lines connected to RPV	RPV failure and failure size	No or limited RPV displacement Challenge bounded by design basis
6. Thermal Loading on Penetrations	Degradation of non-metallic components	Containment breach, leakage path	Magnitude and duration of elevated containment gas temperature Behavior of non-metallic materials at high temperature	No loss of containment integrity expected Potential for long term loss of electrical functionality

Table 4.2-6
Page 3 of 3

PHENOMENOLOGICAL EVALUATION SUMMARIES ON POSTULATED CONTAINMENT FAILURE MODES

FAILURE MODE	PHENOMENA	ISSUE/FAILURE MECHANISM	MAJOR UNCERTAINTY	IMPACT
7. Containment Over-pressurization	Noncondensible gas generation Steam generation H_2 burn	Containment breach	Timing, size, and location of containment breach	FP release to environment (air or soil) or other buildings
8. Containment Isolation Failure	Containment piping Operator response Signal dependency	FP release path through unisolated piping	FP plateout/ plugging	Low probability of direct FP path to environment or auxiliary building
9. Containment By-pass	Interfacing Systems LOCA SGTR	FP release path that does not pass through containment air space	FP deposition in building outside containment Number of ruptured SG tubes Size location of break outside containment Water scrubbing at break location FP deposition outside containment	Low probability of direct FP path to environment or auxiliary building

Figure 4.2-1

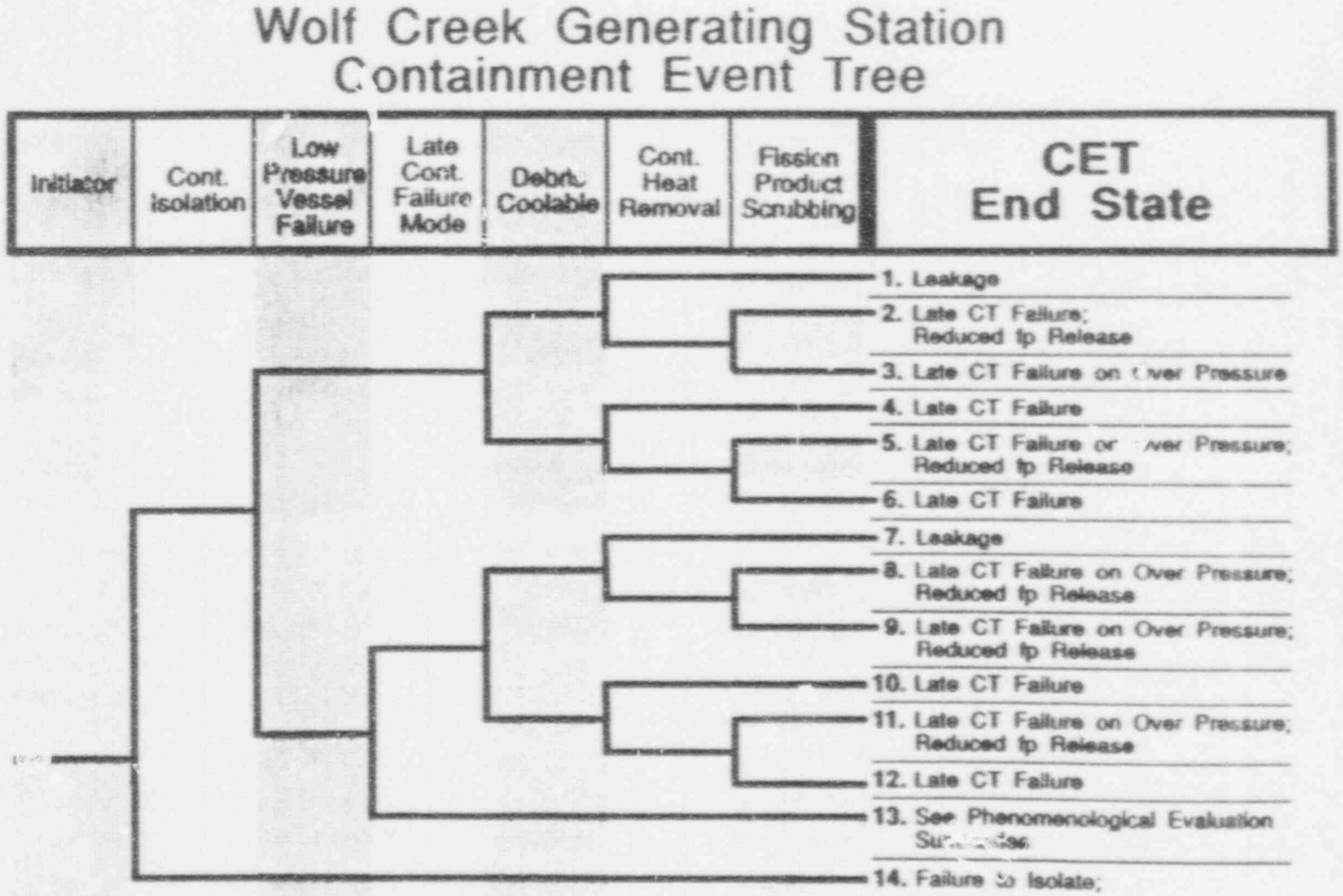
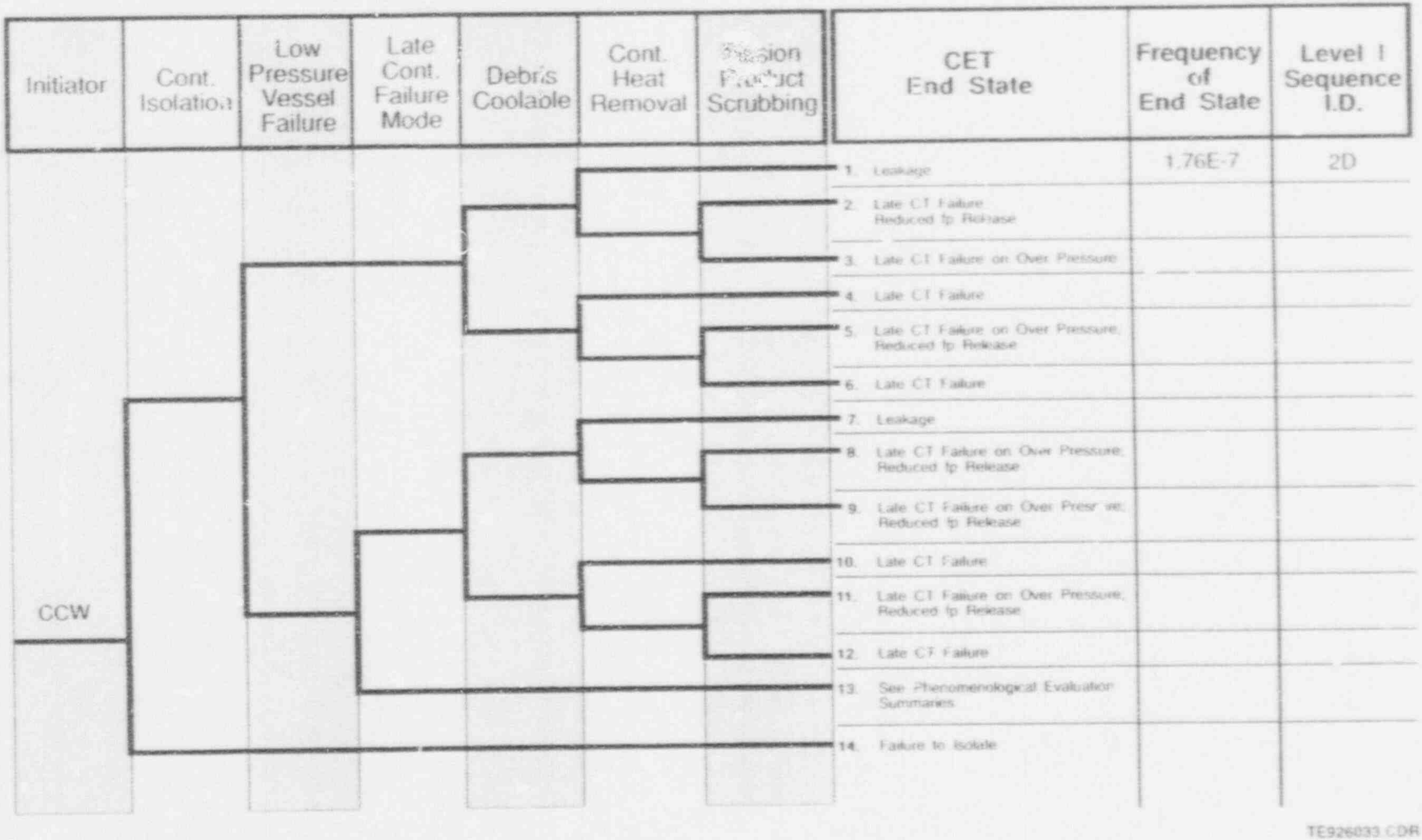

Wolf Creek Generating Station
Containment Event Tree
Initiator
Cont. Isolation
Low Pressure Vessel Failure
Late Cont. Failure Mode
Debris Coolable
Cont. Heat Removal
Fission Product Scrubbing
CET End State
1. Leakage
2. Late CT Failure; Reduced fp Release
3. Late CT Failure on Over Pressure
4. Late CT Failure
5. Late CT Failure on Over Pressure; Reduced fp Release
6. Late CT Failure
7. Leakage
8. Late CT Failure on Over Pressure; Reduced fp Release
9. Late CT Failure on Over Pressure; Reduced fp Release
10. Late CT Failure
11. Late CT Failure on Over Pressure; Reduced fp Release
12. Late CT Failure
13. See Phenomenological Evaluation Summaries
14. Failure to Isolate;

Figure 4.2-2

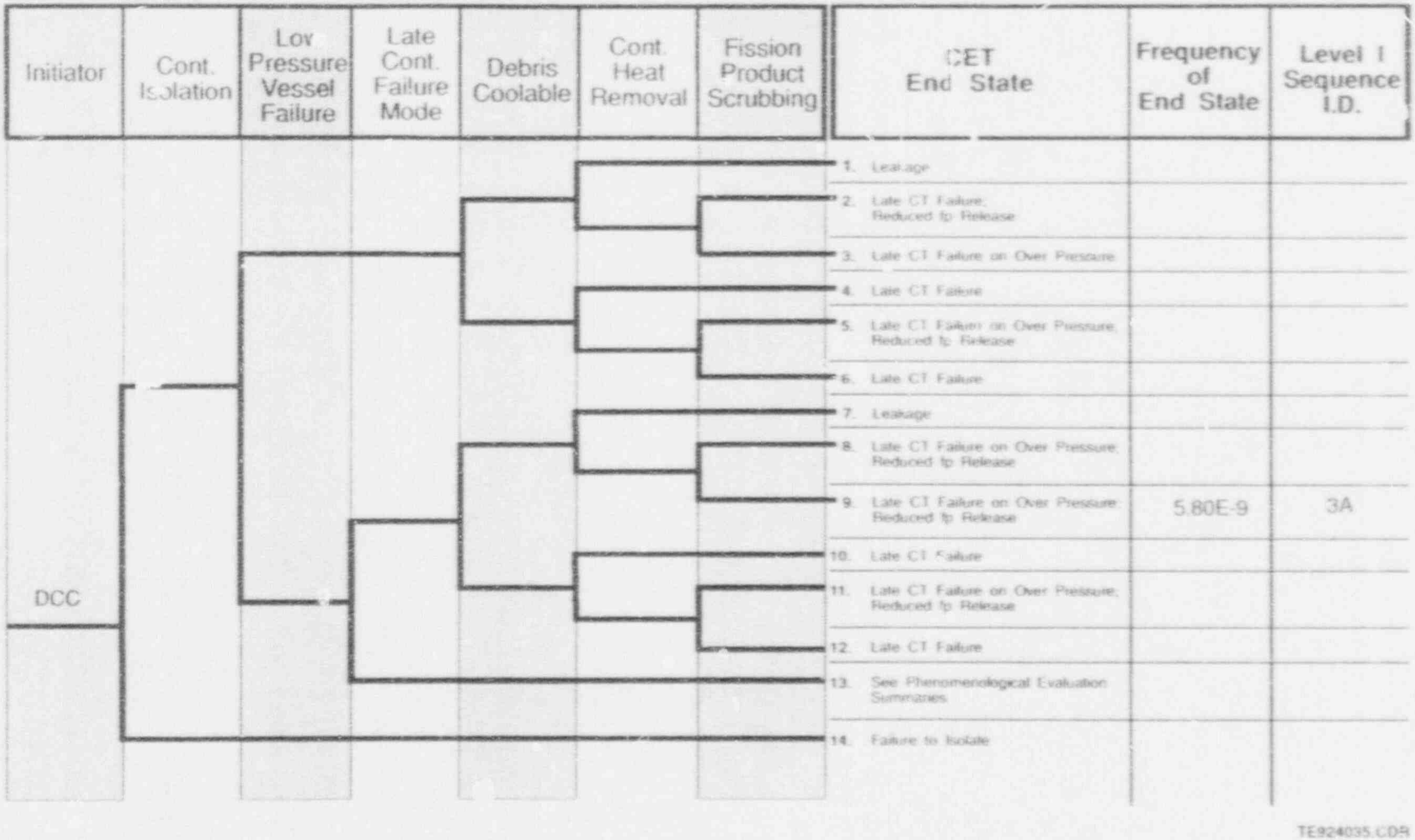
Wolf Creek Generating Station Containment Event Tree



Loss of Component Cooling Water

Figure 4.2-3

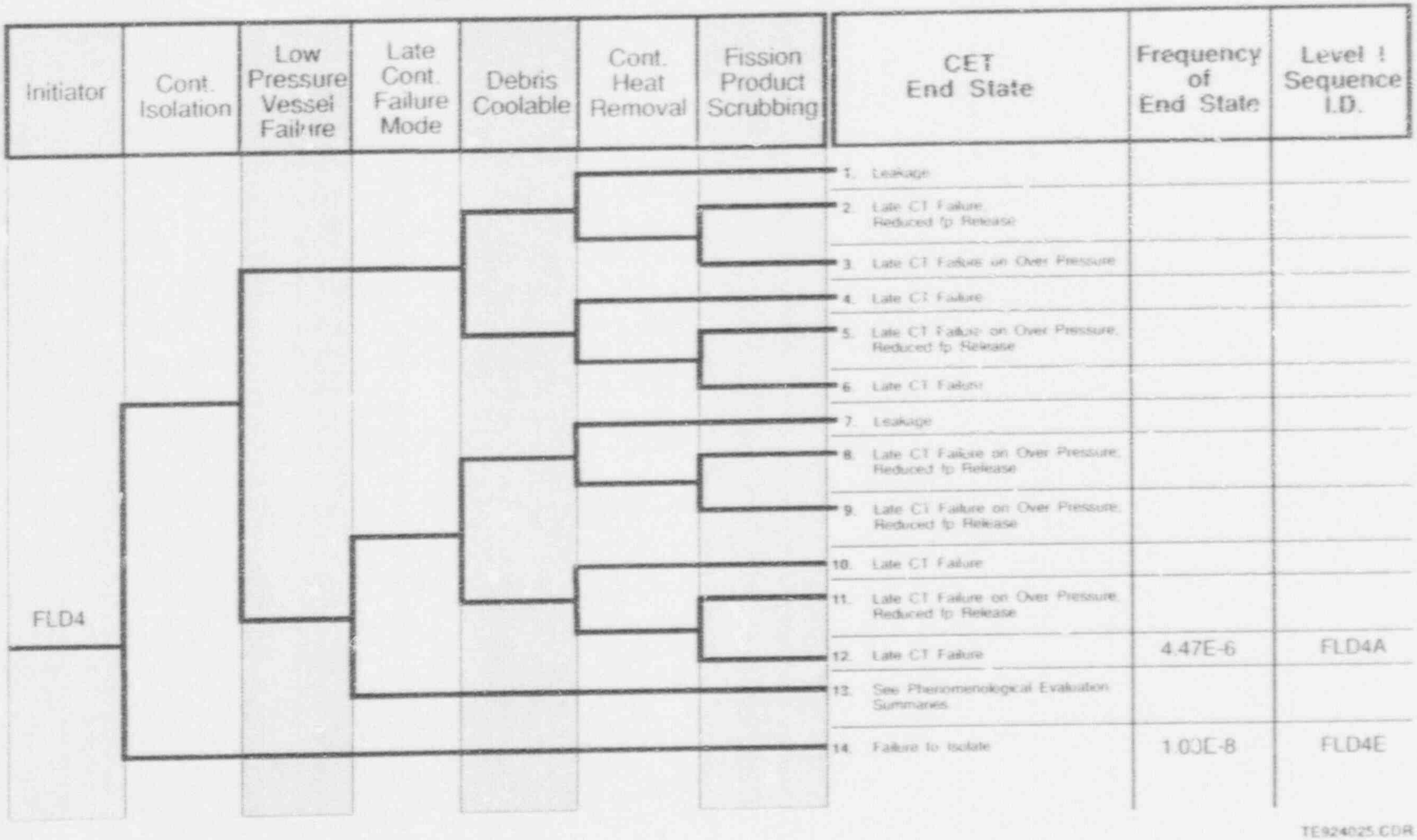
Wolf Creek Generating Station Containment Event Tree



Loss of a Vital DC Bus

Figure 4.2-4

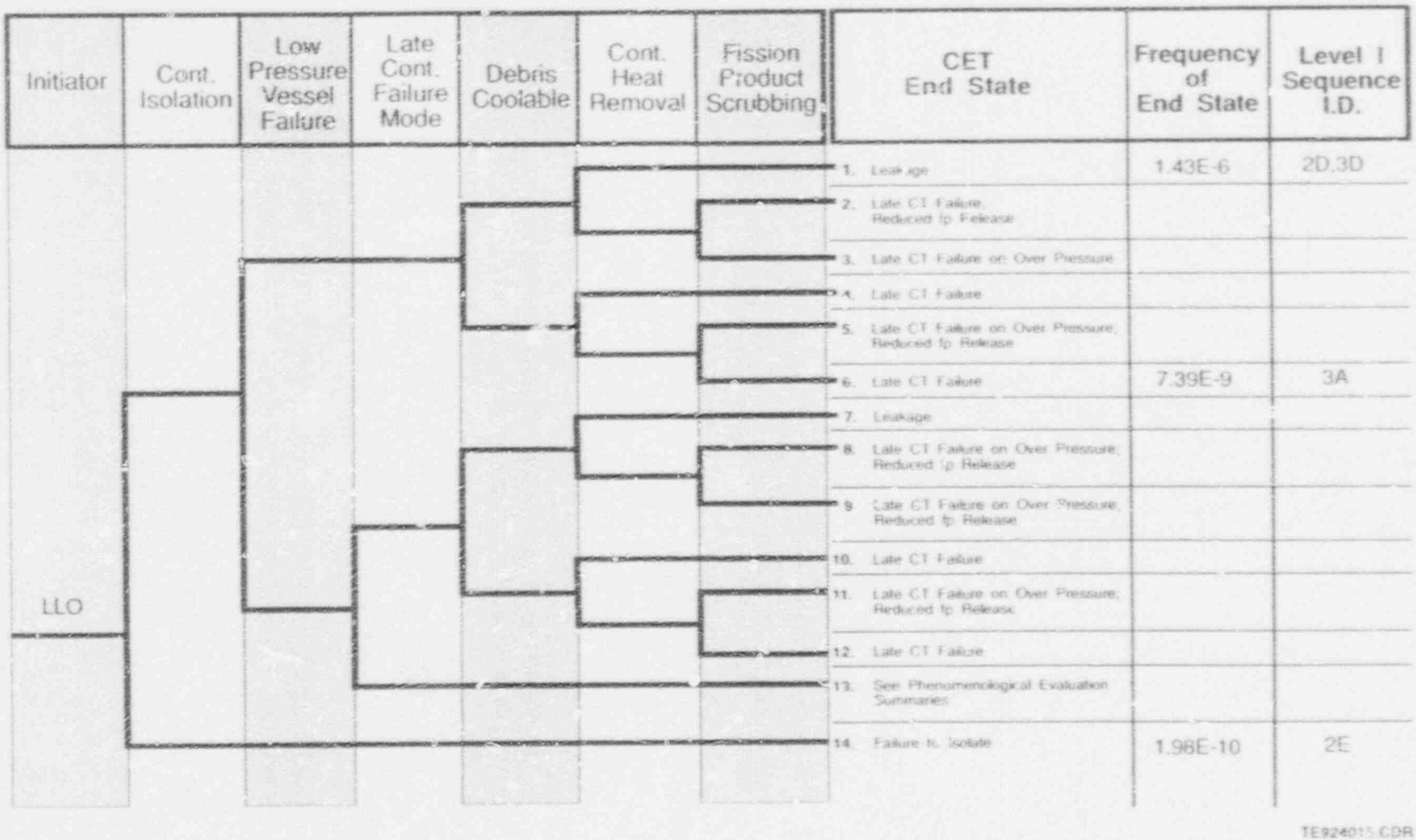
Wolf Creek Generating Station
Containment Event Tree



Internal Flooding

Figure 4.2-5

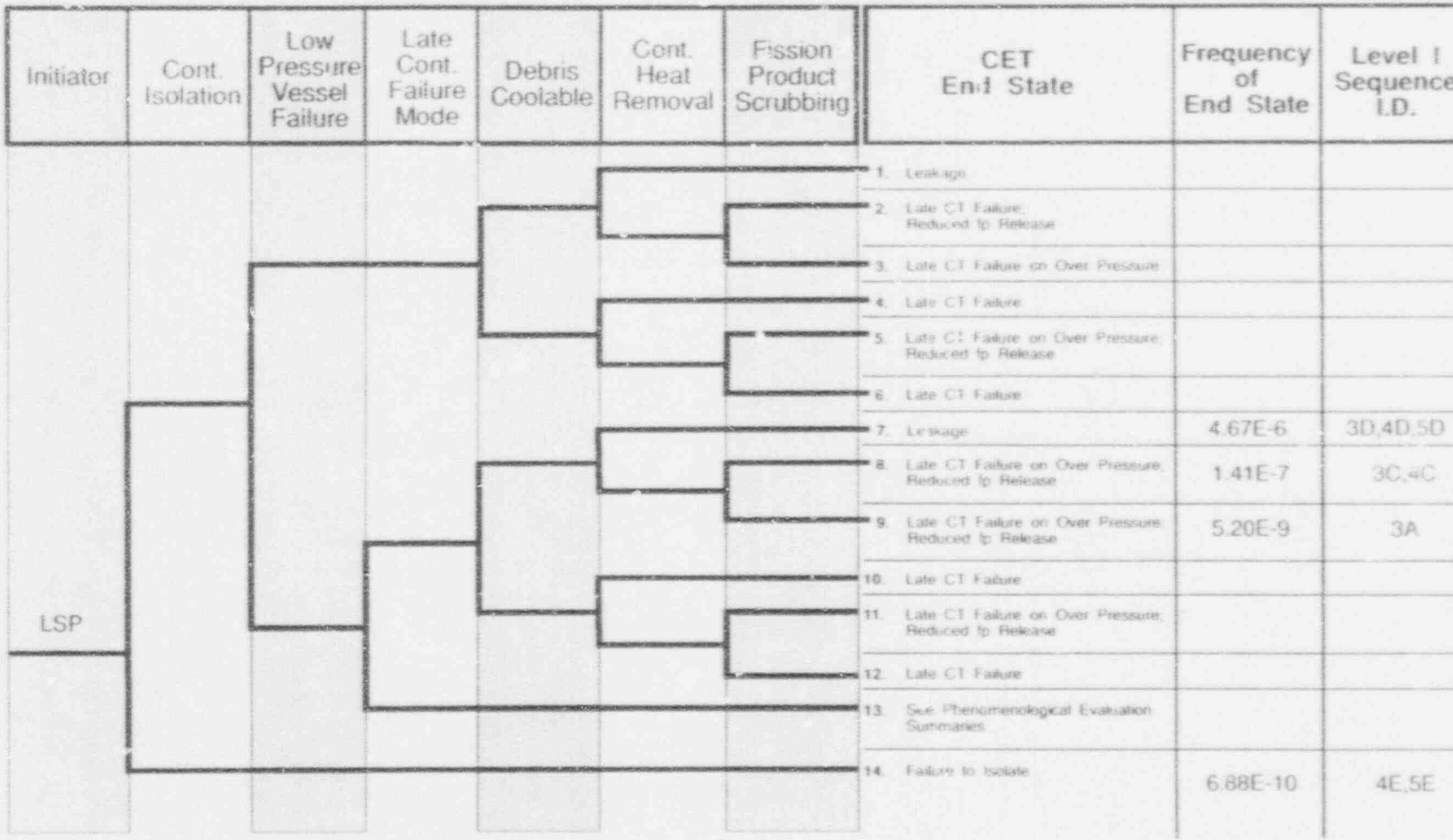
Wolf Creek Generating Station Containment Event Tree



Large LOCA

Figure 4.2-6

Wolf Creek Generating Station Containment Event Tree

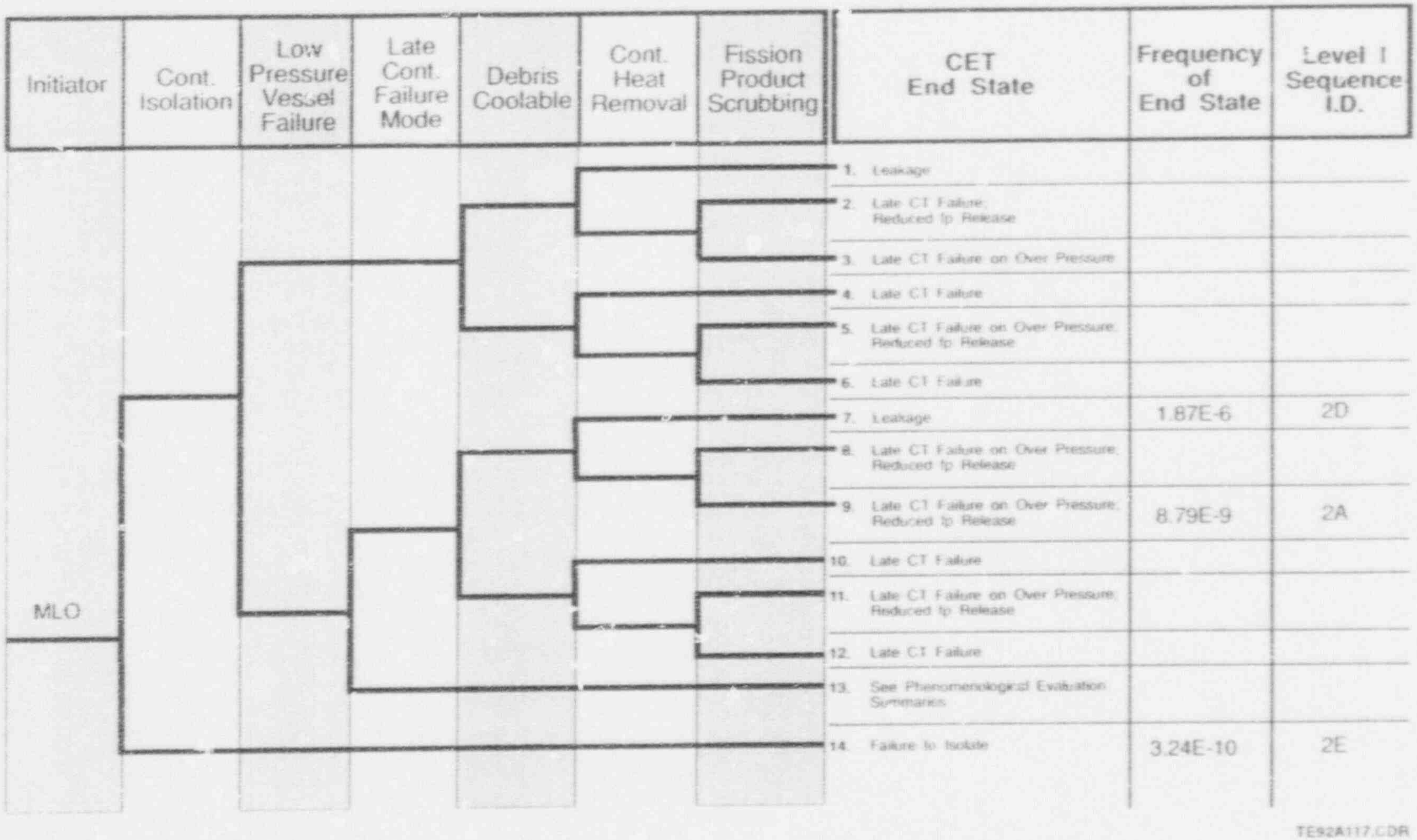


TE924027.CDR

Loss of Offsite Power

Figure 4.2-7

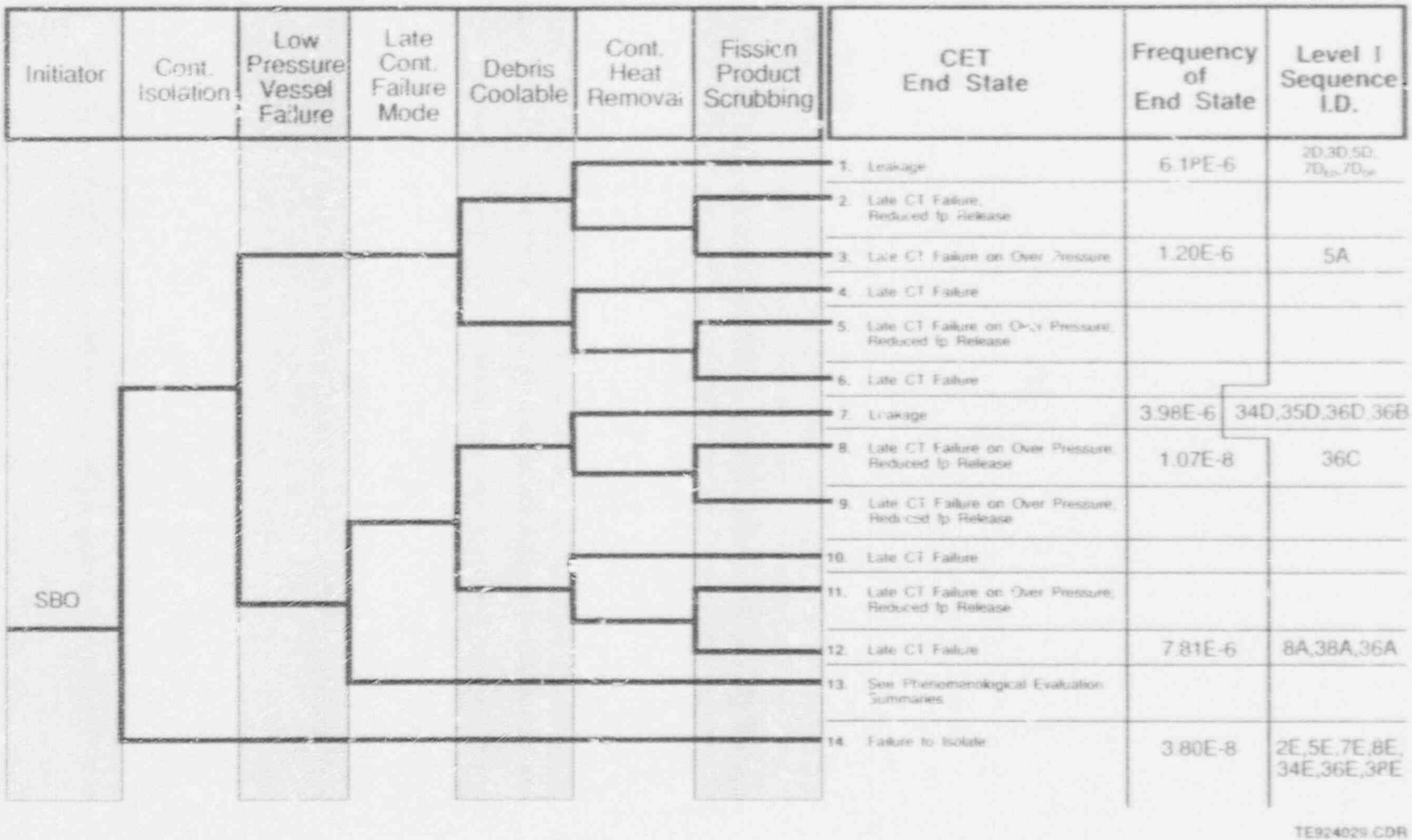
Wolf Creek Generating Station Containment Event Tree



Medium LOCA

Figure 4.2-8

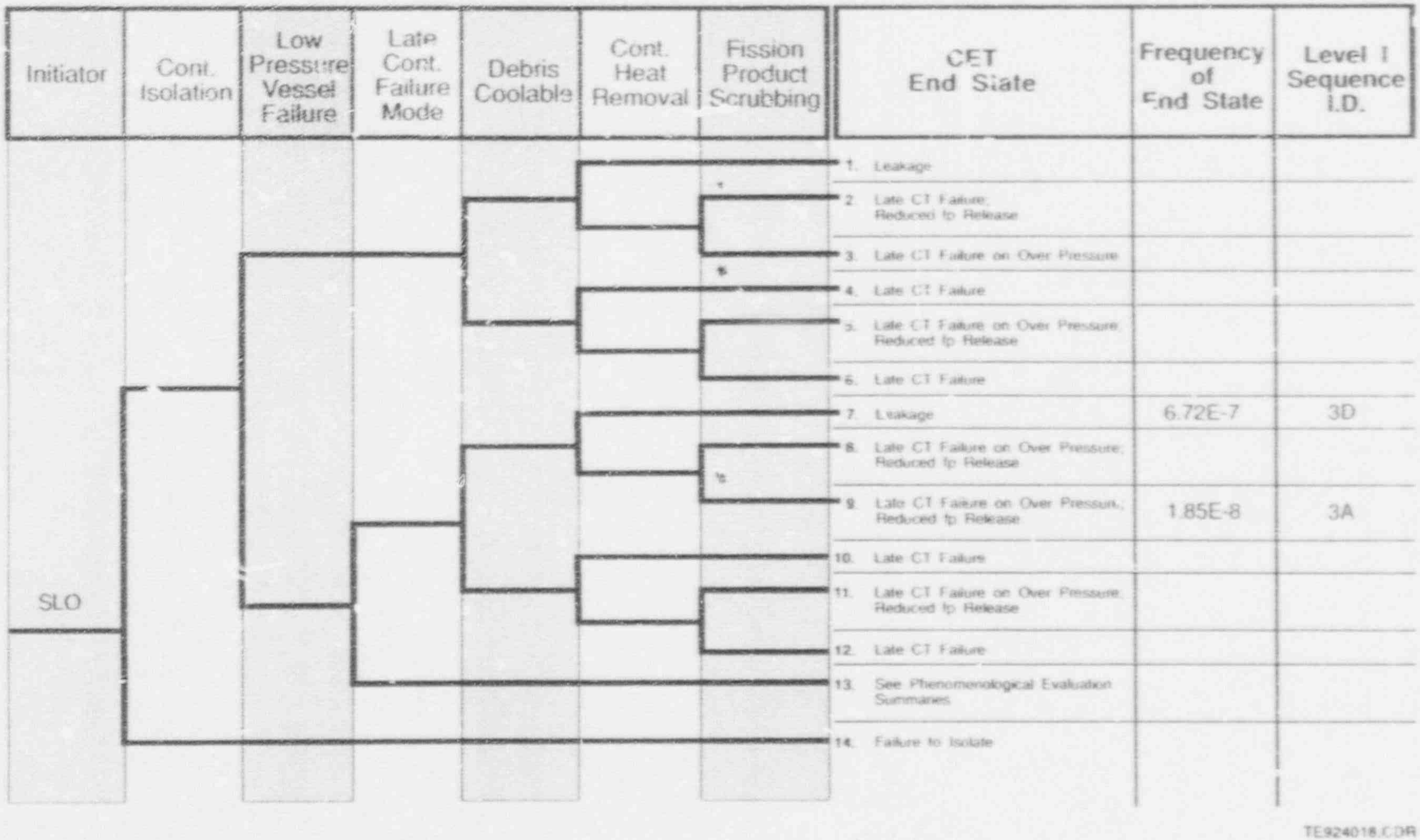
Wolf Creek Generating Station Containment Event Tree



Station Blackout

Figure 4.2-9

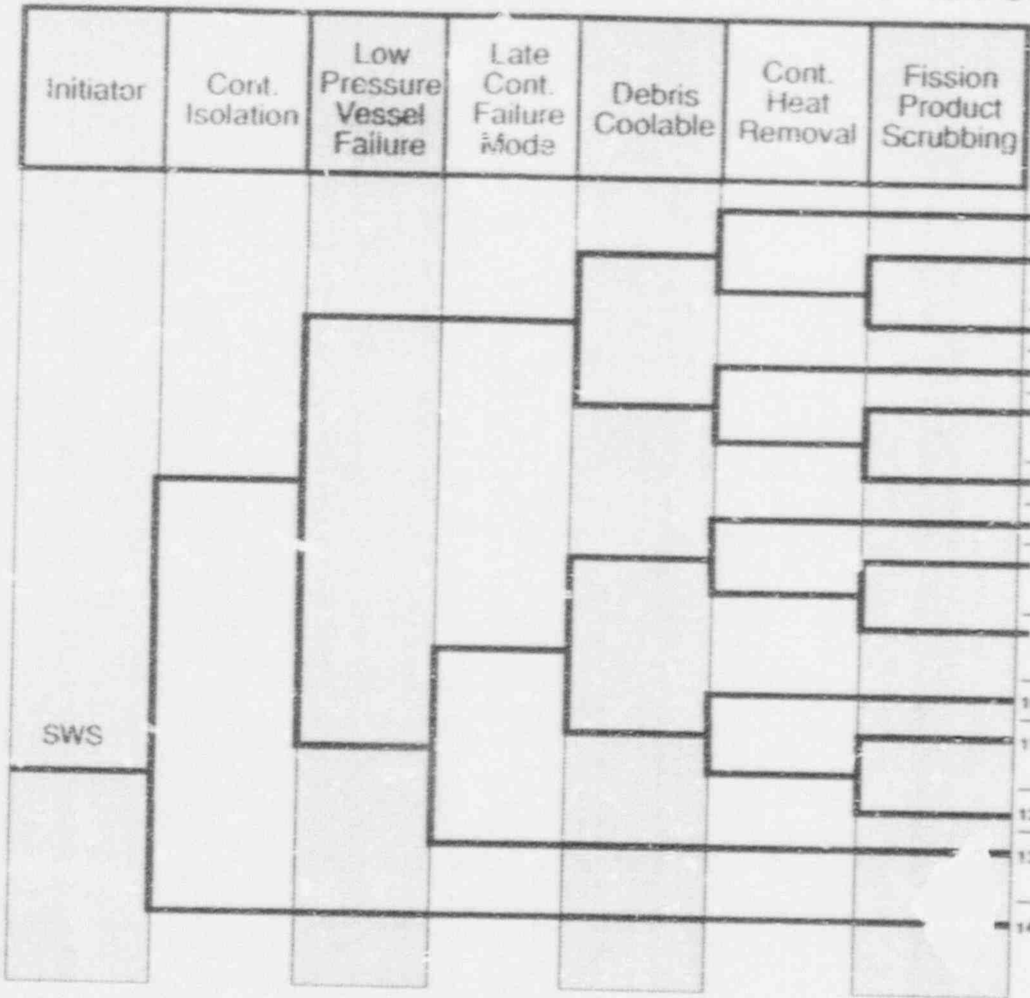
Wolf Creek Generating Station Containment Event Tree



Small LOCA

Figure 4.2-10

Wolf Creek Generating Station Containment Event Tree



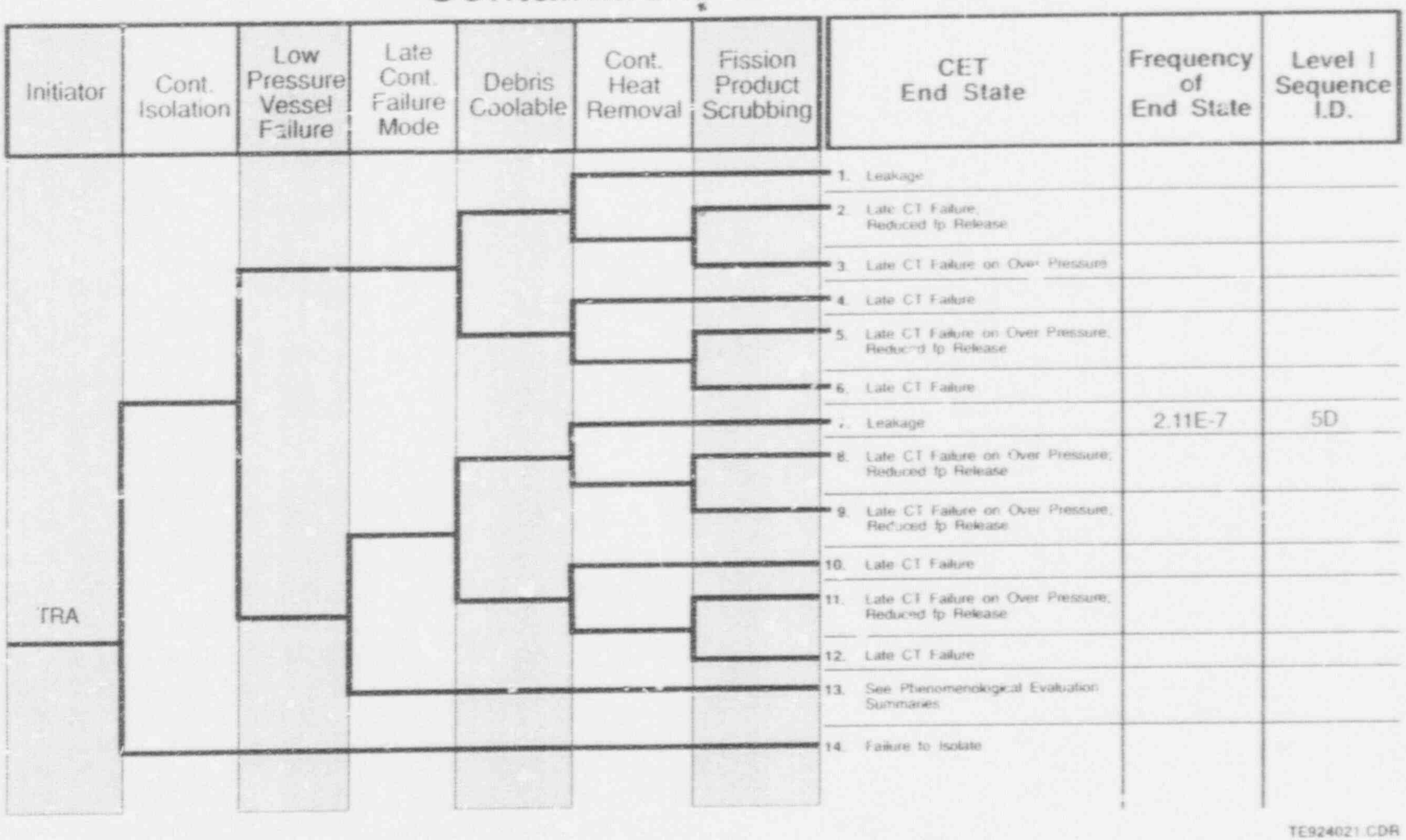
CET End State	Frequency of End State	Level I Sequence I.D.
1. Leakage	7.73E-7	2B,2D,3B,3D
2. Late CT Failure, Reduced fp Release	4.45E-9	2C,3C
3. Late CT Failure on Over Pressure	2.79E-9	2A
4. Late CT Failure		
5. Late CT Failure on Over Pressure, Reduced fp Release		
6. Late CT Failure		
7. Leakage	9.91E-7	13D
8. Late CT Failure on Over Pressure, Reduced fp Release	4.37E-9	13C
9. Late CT Failure on Over Pressure, Reduced fp Release		
10. Late CT Failure	8.92E-9	13B
11. Late CT Failure on Over Pressure, Reduced fp Release		
12. Late CT Failure	2.93E-7	13A, 5A
13. See Phenomenological Evaluation Summaries		
14. Failure to Isolate		

TE924037.*.R

Loss of Service Water

Figure 4.2-11

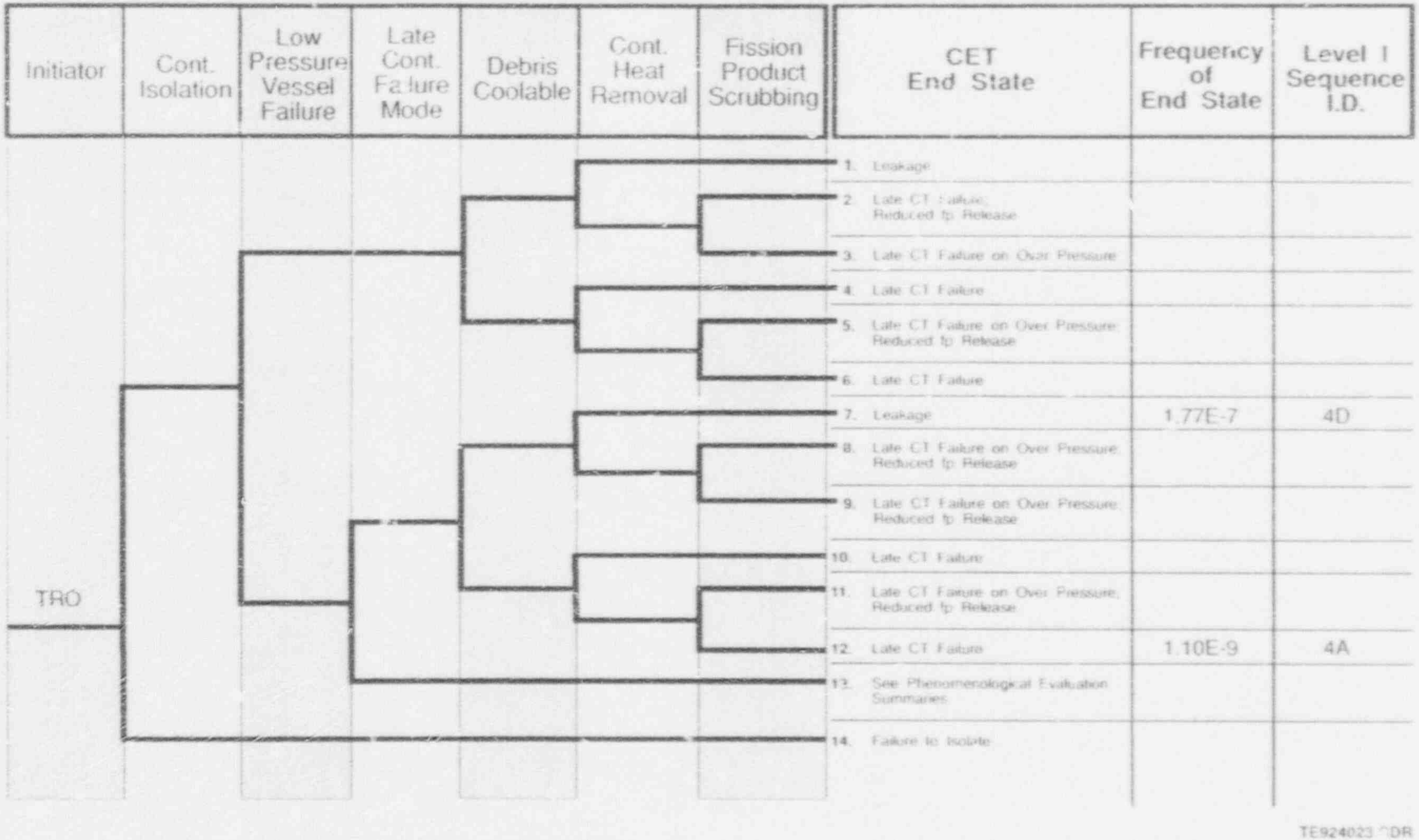
Wolf Creek Generating Station
Containment Event Tree



Transient with Power Conversion

Figure 4.2-12

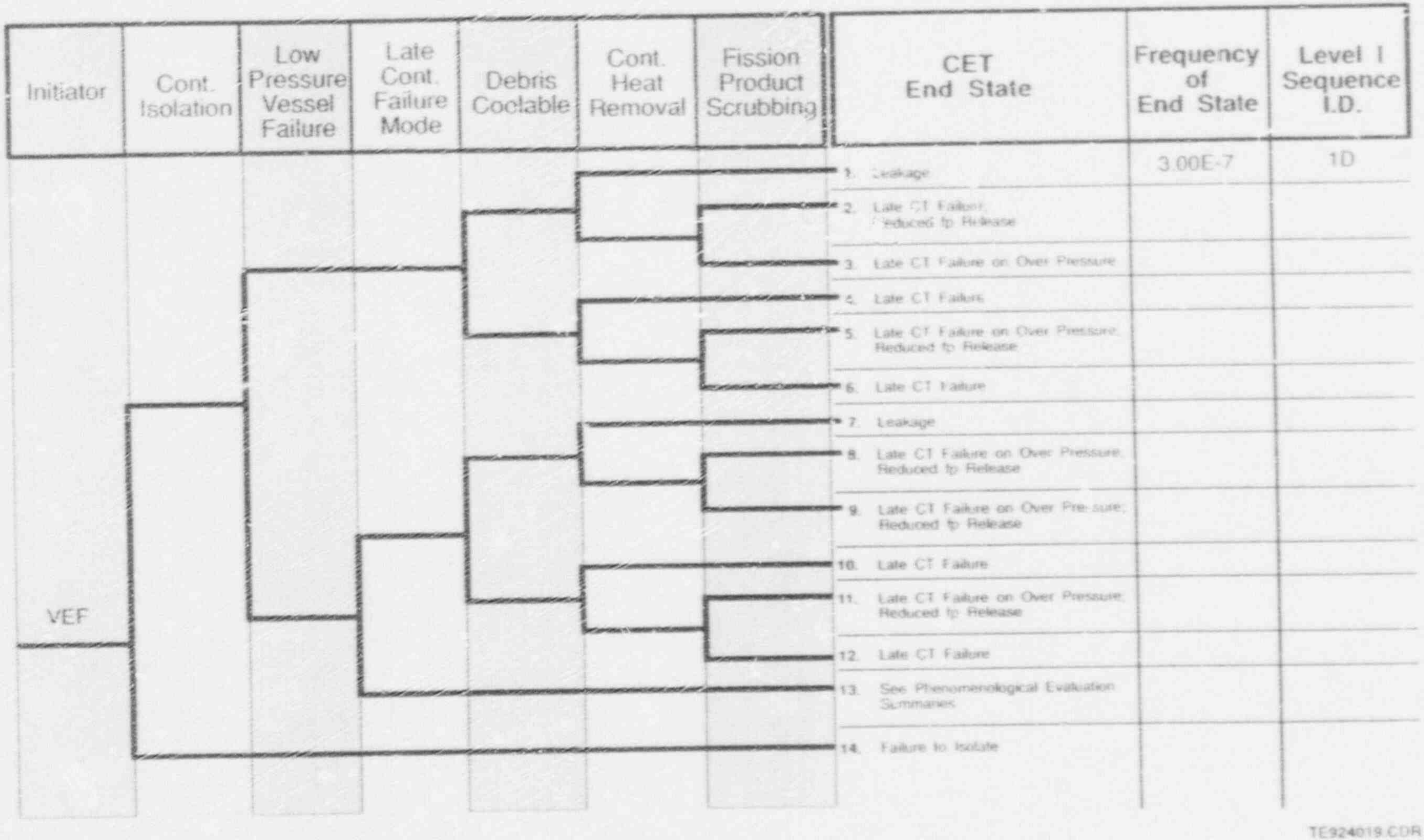
Wolf Creek Generating Station Containment Event Tree



Transient without Power Conversion

Figure 4.2-13

Wolf Creek Generating Station
Containment Event Tree



Vessel Failure

4.3 WCGS PRA - LEVEL 2 SOURCE TERM ANALYSIS

4.3.1 Introduction

Determining nuclear power plant risk, as presented in the Reactor Safety Study (RSS), consists of two major tasks: 1) define the types of severe accidents that occur and their frequency of occurrence, and 2) quantify the consequences of each event. The Wolf Creek Generating Station PRA follows a similar approach. For instance, Task I of the RSS is embodied in the Level 1 PRA analysis which provides accident event trees and core damage frequencies. Next, as input to consequence analysis as described by the RSS Task 2, a source term analysis is performed as part of the Level 2 PRA effort. The purpose of this source term analysis is to quantitatively describe the magnitude and composition of fission product releases to the environment resulting from the severe fuel melt accidents defined in the Level 1 analysis. The remainder of this section is concerned with presentation of the Wolf Creek Generating Station source term analysis and results.

4.3.2 Overview

Although the source term analysis culminates with the quantification of fission product release magnitude, Level 1 results are processed in several steps before source term calculations are actually performed. A majority of the "processing" effort involves reorganizing the Level 1 results into a form suitable for performing the actual source term calculations. This includes Containment Event Tree quantification, as discussed in Section 4.2.4, and grouping of similar sequences into accident sequence "bins" (see Section 4.3.3 below) to reduce the total number of sequences to be analyzed. Source term quantification can then be performed by analyzing a single, representative accident sequence from each bin.

To arrive at fission product releases, a number of phenomena and fission product pathways must be considered through all phases of severe accident sequence progression. For instance, fission products must pass through multiple barriers located along the release pathways, starting with the oxide fuel itself followed by fuel pin cladding, the reactor coolant system and finally, the containment and auxiliary building structures.

Transport of fission products from the initially intact fuel matrix to the environment can best be presented by considering the chronological progression of a core melt accident. During a core melt accident, the transport of fission products, including their transport state and the timing of their release from the intact or molten fuel, varies significantly between volatile and non-volatile fission products and noble gases. Due to the chemical characteristics of volatile fission products, a substantial fraction of these isotopes diffuse through the oxide fuel structure and are released into the fuel pin-cladding gap. Non-volatile fission products, on the other hand, have a much lower affinity for diffusion through the fuel oxide, and thus, are retained within the fuel material. Eventually the fuel cladding will rupture due to the pressure buildup from the volatile gases being released from the fuel material. Concurrent with the cladding failure will be a release into the primary system of the accumulated volatile fission product vapors and the resident noble gases. In the steam environment, most of the volatile fission product vapors would condense and form aerosols. These released fission products may be transported to the containment (or auxiliary building) atmosphere via flow paths through the pressurizer relief and safety valves and pressurizer relief tank rupture disk, or, more directly, via pathways due to breaches in the RCS pressure boundary. In the case of the volatile fission products, significant retention within the primary system could occur as the aerosols deposit on the primary system structures. These "deposited" fission products may re-vaporize late in an accident sequence, however, and follow established pathways out of the RCS.

The onset of core melt accelerates the fission product diffusion process allowing nearly all of the volatile fission products to be released from the fuel material to the primary system. Volatile fission product transport through the primary system and into the containment would then proceed as discussed above. Once again, the non-volatile fission products are retained in the (molten) fuel material. Thus, during the early stages of a core melt accident, most volatile fission products are released to the primary system and containment while the non-volatile fission products flow with the molten core material. This implies that the non-volatile fission products will be transported first to the reactor vessel lower head and then, following lower head failure, to the containment cavity or to both the cavity and the lower compartment regions.

Once in the containment, the molten core debris may begin to attack concrete structures. If this concrete attack occurs, then the ensuing chemical interactions between non-volatile species and the concrete constituents may vaporize some "non-volatile" fission products and release them to the containment gas space in the form of aerosols.

Fission products, volatile and non-volatile alike, which accumulate in the containment gas space are sensitive to a number of fission product removal mechanisms. These mechanisms are important to the fission product retention capability of the containment barrier, especially following breach or impairment of the containment structure. For airborne fission products to be released to the environment they must be transported along with the gas flow through the containment breaches. If active or natural removal mechanisms such as inertial impaction, gravitational settling, or water scrubbing take effect along the pathway from the containment to the outside environment, then a significant reduction in source term release may occur. Fission product pathways encountered in certain severe accident sequences bypass the containment altogether (i.e., steam generator tube rupture, or interfacing systems LOCAs) or lead to early releases from an impaired (i.e., non-isolated) containment and do not benefit from some or all of the aforementioned removal mechanisms. These types of sequences should be expected to have larger source term releases.

The purpose of the Wolf Creek Generating Station source term analysis is to quantitatively describe the magnitude and composition of fission product release to the environment resulting from the severe core damage accidents defined in the Level 1 study. To adequately address the complexities associated with fission product transport and release and to account for the specific Level 1 sequence definitions including operator actions, this analysis relies on the integrated severe accident analysis code, MAAP. This code couples the plant thermal hydraulic responses and fission product behavior to properly model feedback between the two. Furthermore, MAAP can analyze all phases of severe accident progression accounting for the impact of the primary system, containment, engineered safety features, and operator actions. In regard to fission product transport, MAAP begins tracking the fission products as they exist in the normally intact fuel matrix. This initial fission product inventory is organized by chemical properties into 12 groups within MAAP. The initial inventory of each of the 12 fission product groups specific to WCGS as derived from the MAAP parameter file are as follows:

	Fission Product Group	Initial Inventory (lb.)
1)	Noble Gases (Xe, Kr)	1061
2)	CsI (volatile)	91
3)	TeO_2	0
4)	SrO	225
5)	MoO_2	858
6)	CsOH (volatile)	665
7)	BaO	308
8)	La_2O_3 (& Pr_2O_3 + Nd_2O_3, Sm_2O_3 + Y_2O_3)	1646
9)	CeO_2	677
10)	Sb	10
11)	Te_2 (volatile)	88
12)	UO_2 (& NpO_2 + PuO_2)	199,120

The fission product quantities are specific to WCGS. This total inventory of fission products is generally characterized as noble gases (group 1), volatile fission products (groups 2, 6, 11) and non-volatile fission products (groups 3, 4, 5, 7, 8, 9, 10, 12). The source term analysis performed in this section reports the mass fraction released for each of these three categories. Group 2 is used to report the volatile release in the summaries.

4.3.3 Source Term Sequence Selection

Although the source term analysis culminates with the quantification of fission product release magnitude, Level 1 results are processed in several steps before source term calculations are actually performed. A majority of the "processing" effort involves reorganizing the Level 1 results into a form suitable for performing the actual source term calculations. This includes Containment Event Tree quantification and grouping of similar sequences into accident sequence "bins" to reduce the total number of sequences to be analyzed. Source term quantification can then be performed by analyzing a single, representative accident sequence from each bin.

Sequence selection for source term analysis entails the following:

(A) Screening process: From the set of all sequences analyzed during the Level 1 effort, select only a certain number of sequences according to the NUREG-1335 guidelines.

(B) Sequence binning: From the selected sequences identified above, group them together according to their similar end states and containment conditions such that all sequences in the same group would result in similar source terms.

(C) Selection of representative sequences: Select at least one sequence from each group in step B for source term analysis.

(D) Release category: From the source term results in step C, assign a release category to each group identified in Step B.

A screening process based on NRC guidelines contained in NUREG-1335 is applied to the core damage sequences identified in the Level 1 analysis to arrive at a reduced sequence list. For the Wolf Creek PRA, this screening process resulted in a set of 73 sequences of importance for source term analysis. This is the process described above by step (A).

Grouping sequences with expected similar source terms according to their end states and containment conditions is referred to as "binning", as described by step (B) above. Representative sequences are selected in order to make MAAP runs and derive a fission product release. These sequences are those described by step (C) above and referred to as the "analyzed" sequences. The remaining sequences within each group from step (C) are referred to as "bounded" sequences. The accident sequence characteristics of the bounded and analyzed sequences within each grouping are such that approximately the same source term results are expected.

Finally, to complete the analysis, release categories are defined which describe the mode of containment failure and fission product release, and these categories are assigned to each group of sequences based on the MAAP analysis for the analyzed sequences. The source term results for a release category are based on MAAP runs of an analyzed sequence, while the frequency of the release category is the summation of the frequencies for both the analyzed and the bounded sequences assigned to that category. This is the process described above in step (D).

The source term sequence selection process involves a screening process, sequence binning, selection of representative sequences and assignment of source term release categories. Each of these are discussed in greater detail.

4.3.3.1 Screening Process

The purpose of the screening process is to analyze and report those severe accident sequences that are either above a given frequency or, which contribute significantly to the total core damage and containment failure frequencies. The screening process involves several steps in which Level 1 core damage sequences are reviewed and used to generate sequences for Level 2 analysis.

The top Level 1 core damage sequences in order of decreasing frequency are listed in Table 3.4-1. The table includes all core damage sequences with frequencies greater than 1.0E-8 per year. This list of sequences accounts for more than 99.8% of the total core damage frequency.

Each Level 1 sequence in Table 3.4-1 was linked with the containment fan coolers and the containment spray system fault trees as discussed in Section 3.2.3.5. The Containment Safeguards Event Tree includes three separate nodes to address the failure probabilities associated with containment cooling systems (i.e., containment fan coolers and/or RHR heat exchangers), containment spray systems (injection or recirculation mode), and containment isolation. Thus, any dominant accident sequence input to the Containment Safeguards tree could result in eight different end states. All Plant Damage sequences greater than 1E-6 were coupled with containment isolation failure scalars to complete the plant damage state process.

An initial Level 2 assumption (Section 4.2.1) was that all Level 1 sequences would lead to core damage. MAAP results obtained during the source term analysis indicated this was not true. Eleven sequences were subsequently dropped from further Level 2 evaluation as a result of MAAP analysis results. The eleven sequences eliminated from Level 2 were: CCW4D, CCW31D, FL3A, FL3B. SGR3D, SGR5D, SGR14A, SGR14D, SWS4B, SWS4C, and SWS4D; Isolation failure sequences CCW31D and FL3BE were also dropped since their parent sequences (CCW31 and FL3B) were eliminated. The above thirteen sequences totaled 6.52E-6 or 15.2% of the Level 1 total core damage frequency. This left a final total of 58 sequences for non-containment bypass binning and source term analysis with a reduced core damage frequency of 3.63E-5. Containment bypass sequences ISL1 and SGR19 contribute 7.31E-8 (0.20%) to the core damage frequency. The containment failure frequency including interfacing system LOCAs and steam generator tube ruptures is 1.52E-6 (See Table 4.3-4).

The list of WCGS Plant Damage State sequences is shown in Table 4.3-1. Sequences 1 through 27 account for 95% of the core damage frequency. Sequences specifically identified in Table 4.3-1 account for 95% of the containment failure frequency. The comparison to the NUREG-1335 screening criteria are summarized in Section 3.4.

Generic Letter 88-20 Appendix 2, Item 3, requests that 'Any functional sequence that has a core damage frequency greater than 1E-6 per reactor year and that leads to containment failure with a radioactive release greater than the PWR-4 release category of WASH-1400' also be reported. The PWR-4 release category includes those sequences that assume 'the containment is not fully isolated and the containment radioactivity removal systems have failed'. No such sequences meeting these criteria were identified for WCGS.

4.3.3.2 Sequence Binning and Selection of Representative Sequences

The binning of sequences is based on the sequence initiating event category and the CET quantification outcome which results in the grouping of like sequences. The reduced set of 60 sequences which 1) passed the screening criteria and 2) were determined to result in core damage within 24 hours were input to the CET. Results of the CET quantification are presented in Section 4.2.4 and summarized in Table 4.2-4. For the purposes of binning, the sequence bins are referred to by the CET end state.

As an example of the binning process, consider those sequences with CET end state 1 (low pressure vessel failure with normal containment leakage) shown in Table 4.2-4. These sequences fall into the initiating event categories of loss of coolant accidents (i.e., large LOCAs and vessel failure LOCAs), loss of component cooling water, station blackout, and loss of service water. Although the initiating events vary widely between the sequences included in bin 1, the source term resulting from containment normal leakage is sufficiently small that a single value of the fission product release fraction adequately represents the expected release due to any of the initiating events.

One representative sequence from each bin (termed the "analyzed sequence') must then be selected for source term analysis. The analyzed

One representative sequence from each bin (termed the "analyzed sequence") must then be selected for source term analysis. The analyzed sequence is selected either because it has the largest frequency of occurrence of any sequence within the bin or because the analyzed sequence source term is expected to bound that which is expected from the other sequences within the bin. The source term resulting from the analyzed sequence is then assigned to all other sequences in the bin (the "bounded sequences"). Thus, continuing with the above example for CET end state 1, the station blackout, sequence $SB07D_{ED}$, was selected because its source term is expected to bound that of the remaining sequences. Analyzed sequences have been assigned to the other bins in a similar fashion.

The source term for the analyzed sequence in a particular bin is assigned to the bounded sequences in that bin as well. This is accounted for by summing over all sequences within a bin to determine the cumulative frequency associated with the reported source term. For instance, the cumulative frequency for bin 1 is 8.86E-6 occurrences per year. This is then the expected frequency for a release determined from the analyzed sequence. Results of the WCGS Level 2 binning process, as shown in Table 4.3-2, indicate that each sequence selected for Level 2 analysis is assigned, or "binned", with one of twelve sequences selected for MAAP source term analysis.

4.3.3.3 Release Categories

Release categories are defined in Table 4.3-5 in terms of containment failure timing (early or late), containment failure mode (overpressure, impairment, or bypass) and the airborne fractional release of fission products to the environment. Based on the source term results of the analyzed sequences, release categories are assigned to the analyzed sequences and thus to the bin which they represent. The release category designator appears in the fourth column of Table 4.3-2.

4.3.4 MAAP Analyses

Altogether, more than 120 cases for both Level 1 and Level 2 were modeled using MAAP 3.0B Revision 17.02. Of this total, 46 were for Level 1 success criteria verification. Twelve of the sequences were selected in Section 4.3.3.2 for source term analyses; eight other sequences in Table 4.2-4 were also modeled with MAAP.

Several assumptions made for the MAAP calculations are outlined here since they significantly affect the calculated source term results.

(1) The Level 2 analysis assumes a 48 hour mission time, while the Level 1 mission time is 24 hours. Hence, accident progression is studied for a period of time beyond which Level 1 activities would be implemented to alter the course of the accident.

(2) Based on an evaluation of the WCGS containment cavity region , it was concluded that the potential for debris entrainment from the cavity following high pressure melt ejection events is very small. The MAAP analyses, therefore, assume no debris entrainment.

(3) Any equipment assumed failed as part of the Level 1 sequence definition is assumed to remain inoperable for the duration of the accident sequence. This implies that no failed equipment will be recovered during the Level 2 analysis unless specifically defined in the Level 1 event trees.

Several tables were compiled to summarize the source term results. As mentioned previously, Table 4.3-2 provides information regarding the source term binning process. This table also provides a summary of the volatile fission product releases, the release frequency, and the release category for each bin. The release categories are presented in a slightly different form in Table 4.3-3. This table presents the expected frequency of occurrence for the given release categories. Accident progression parameters for all analyzed sequences are contained in Table 4.3-6. This table contains information such as accident timing and conditions, hydrogen burn data, and source term results. Finally, Tables 4.3-7 and 4.3-8 provide an accounting of the airborne fission product releases for all twelve fission product groups at 24 and 48 hours of the twelve Level 2 source term MAAP cases.

4.3.5 Fission Product Release Characteristics

In this section, the plant-specific source term will be characterized according to accident initiators and containment success criteria, based on MAAP analyses of accident sequences selected in Section 4.3.3.2.

There are two phases of fission product release from the damaged core materials, i.e., in-vessel and ex-vessel. The in-vessel phase starts shortly after the core is uncovered and lasts until vessel failure. The release rate during this phase depends on the rate of core uncovery which in turn depends on the RCS coolant depletion rate and the extent of decay heat removal as discussed below.

1) RCS coolant depletion rate: For sequences with a break of the primary system, the RCS coolant depletion rate depends on the LOCA break size. The larger the break size, the earlier the core is uncovered. Hence, for sequences with similar success nodes, vessel failure will occur earlier for a larger break sequence.

2) Actions resulting in decay heat removal: The success for RCS cooldown using safety injection, auxiliary feedwater and steam dump, or primary system depressurization using feed and bleed can substantially remove decay heat, increase fission product retention inside the primary system by reducing vessel wall and internal structure temperatures, and delay vessel failure time for several hours.

3) Primary system pressure during core melt phase: Fission product retention within the primary system is substantially larger for high pressure sequences as opposed to low pressure sequences. As indicated in Table 4.3-6, the majority of volatile fission products (~ 50 - 90%) are retained in the primary system following vessel failure during high pressure sequences. Low pressure sequences, on the other hand, show volatile fission product retention in the primary system of less than 20%. In either case, however, the primary system would continue to be an important barrier to the release of fission products following gross failure of the reactor vessel lower head.

The ex-vessel fission product release characteristics depend on the containment failure mode--fission product retention within the containment will be different for sequences with containment by-pass, containment isolation failures (i.e., impairment), containment overpressurization failures, and containment normal leakage.

For sequences with non-bypassed and non-impaired containments, only a small portion of the volatile fission products will be released to the environment regardless of whether or not containment failure due to overpressurization occurs. The ability of the containment to retain fission products for such sequences depends primarily on the time lapse between volatile fission product release to the containment and containment failure. With a time lapse of about 6 hours, aerosol deposition mechanisms will remove nearly all airborne fission products. Thus, sequences with containment failure less than 6 hours after core melt (i.e., early containment failure, DCC3A as an example) will have significantly larger source term releases than sequences with late containment failure.

Since most volatile fission products are released from the fuel during core melt and prior to vessel failure, the ex-vessel disposition of the debris (i.e., ex-vessel debris distribution and debris interaction with water pools) has only a secondary affect on the source term release. Debris in the containment which is not submerged in water pools may heat up and ablate concrete structures in which case additional fission product volatilization may occur. However, the aerosols generated from the core-concrete interactions would generally be deposited onto the containment structures within a few hours and no significant increase in the source term release would occur.

At Wolf Creek, debris entrainment out of the reactor cavity region following reactor vessel failure is unlikely. Therefore, injection of the RWST inventory is necessary to submerge the core debris through flooding of the cavity region and prevent or terminate the core-concrete attack. In such sequences, interaction between molten core debris and the overlying water pool would maintain debris coolability via boiling heat transfer. Steam which would be generated by this boiling process would cause a gradual pressurization of the containment. In the absence of operable containment heat removal systems, this pressurization would continue until the containment ultimate pressure capacity was surpassed. At this point it is assumed that containment integrity would be lost and a containment breach just large enough to relieve the pressurization would form (i.e., the leak-before-break failure mode). Typical MAAP results for the Wolf Creek dominant accident sequences indicate that overpressurization failure would not occur within the first 40 hours following accident initiation. This late containment failure mode results in a relatively small fission product release since sufficient time has elapsed for natural deposition mechanisms to remove nearly all airborne fission products from the containment atmosphere.

Another mechanism which could potentially improve the fission product retention capability of the containment is scrubbing of airborne fission products through the use of containment sprays. In many sequences, however, operation of containment sprays does not occur. Thus, as indicated by the WCGS PRA results, this extremely effective fission product removal mechanism is seldom of any practical benefit.

For sequences with containment bypass or containment isolation failure, large source term releases should be expected. With a containment bypass sequence, for instance, a relatively large portion of the volatile fission products are released through the bypass path which is established at the beginning of the sequence even if containment failure does not occur. Thus, bypass sequences are characterized by early (i.e., shortly after the onset of core damage) as well as large releases of fission products. Sequences with containment isolation failures, on the other hand, derive some benefit from fission product retention inside the containment. However, since the isolation failures are assumed to occur at the beginning of the sequence, large releases can occur before natural deposition mechanisms can significantly reduce the airborne fission product concentration.

Overall, the Wolf Creek source term quantification indicates that the primary system and containment are capable of retaining nearly all fission products released from the molten core material even if structural integrity of these systems cannot be maintained indefinitely. Thus, sequences which benefit from the fission product retention capability of the primary system and containment (i.e., sequences with late containment failures or normal containment leakage) will have a minimal source term release. Sequences which bypass the containment, or experience an impaired containment, on the other hand, will not receive the full benefit of the containment and will therefore produce larger releases.

Table 4.3-1
WCGS PLANT DAMAGE STATES TABLE
Page 1 of 2

#	SEQUENCE FREQ.	SEQUENCE I.D.	MAAP CASE I. D.	% CDF	SUM OF FREQ.s	TOTAL %
1	5.95E-06	SBO8A	SBO004	14.17	5.95E-06	14.17
2	4.47E-06	FL4A		10.65	1.04E-05	24.82
3	2.74E-06	SBO7DOP		6.51	1.32E-05	31.34
	2.69E-06	LSP4D	LSP003	6.41	1.58E-05	37.74
5	2.64E-06	SBO2D		6.29	1.85E-05	44.04
6	2.20E-06	SBO36D		5.24	2.07E-05	49.28
7	2.20E-06	FL3B		5.24	2.29E-05	54.51
8	2.19E-06	CCW31D		5.21	2.51E-05	59.73
9	1.87E-06	MLO2D	MLO001	4.46	2.69E-05	64.19
10	1.63E-06	SBO38A	SBO005	3.88	2.86E-05	68.07
11	1.45E-06	SBO34D		3.46	3.00E-05	71.52
12	1.29E-06	LSP5D		3.07	3.13E-05	74.59
13	1.20E-06	SBO5A		2.86	3.25E-05	77.46
14	1.15E-06	LLO2D		2.73	3.37E-05	80.19
15	9.91E-07	SWS13D		2.36	3.47E-05	82.55
16	8.94E-07	FL3A		2.13	3.56E-05	84.68
17	6.85E-07	LSP3D		1.63	3.62E-05	86.31
18	6.72E-07	SLO3D	SLO003	1.60	3.69E-05	87.91
19	4.98E-07	SWS2D		1.19	3.74E-05	89.10
20	4.86E-07	SWS4D		1.16	3.79E-05	90.25
21	4.52E-07	SBO5D		1.08	3.83E-05	91.33
22	3.09E-07	SBO35D		0.74	3.87E-05	92.07
23	3.00E-07	VEF1D		0.71	3.90E-05	92.78
24	2.92E-07	SWS5A		0.70	3.92E-05	93.48
25	2.75E-07	LLO3D		0.66	3.95E-05	94.13
26	2.66E-07	SWS3D	SWS006	0.63	3.98E-05	94.76
27	2.44E-07	SGR14D		0.58	4.00E-05	95.35
28	2.41E-07	SGR3D		0.57	4.03E-05	95.92
29	2.31E-07	SBO36A		0.55	4.05E-05	96.47
30	2.11E-07	TRA5D		0.50	4.07E-05	96.97
31	1.98E-07	SBO3D		0.47	4.09E-05	97.44
32	1.77E-07	TRO4D		0.42	4.11E-05	97.87
33	1.76E-07	CCW2D		0.42	4.13E-05	98.28
34	1.46E-07	SGR5D	SGR005	0.35	4.14E-05	98.63
35	1.46E-07	SBO7DED	SBO002	0.35	4.16E-05	98.98
36 *	1.23E-07	LSP4C	LSP002	0.29	4.17E-05	99.27

The Sequence I.D.s are derived from the Level 1 Event Trees (e.g., SBO4C, Station Blackout - Sequence Number 4); and a letter designator for the availability of Containment Cooler, Spray, and Isolation systems as follows:

Sequence ID End Letter derived from status of containment systems:

A No Coolers, No sprays
B Coolers; No Sprays
C Sprays; No Coolers
D Coolers and Sprays
E Isolation Failure

Sequence 36, 38, 41, 42, 43, 44, 45, 47, 49, 53, 54, 57, 58, 59, and 60 contribute >95 % of the containment failure frequency.

Table 4.3-1
WCGS PLANT DAMAGE STATES TABLE

#	SEQUENCE FREQ.	SEQUENCE I.D.	MAAP CASE I. D.	% CDF	SUM OF FREQ.s	TOTAL %
37	1.05E-07	CCW4D	CCW001	0.25	4.18E-05	99.52
38 *	6.11E-08	ISL1	ISL001	0.15	4.18E-05	99.67
39	1.89E-08	SBO36B		0.05	4.18E-05	99.57
40	1.85E-08	SLO3A		0.04	4.18E-05	99.61
41 *	1.78E-08	LSP3C		0.04	4.18E-05	99.65
42 *	1.37E-08	SBO8E		0.03	4.19E-05	99.69
43 *	1.20E-08	SGR19D		0.03	4.19E-05	99.71
44 *	1.07E-08	SBO36C		0.03	4.19E-05	99.74
45 *	1.03E-08	FL4AE		0.02	4.19E-05	99.76
46	8.92E-09	SWS13B	SWS005	0.02	4.19E-05	99.79
47 *	8.79E-09	MLO2A	MLO002	0.02	4.19E-05	99.81
48	7.59E-09	SGR14A		0.02	4.19E-05	99.82
49 *	7.39E-09	LLO3A	LLO003	0.02	4.19E-05	99.84
50	6.32E-09	SWS2B		0.02	4.19E-05	99.86
51	6.32E-09	SBO7E		0.02	4.19E-05	99.87
52	6.10E-09	SBO2E		0.01	4.19E-05	99.89
53 *	5.80E-09	DCC3A	DCC001	0.01	4.19E-05	99.90
54 *	5.20E-09	LSP3A	LSP004	0.01	4.19E-05	99.91
55	5.08E-09	SBO36E		0.01	4.20E-05	99.92
56	4.38E-09	SWS4B	SWS008	0.01	4.20E-05	99.94
57 *	4.37E-09	SWS13C		0.01	4.20E-05	99.95
58 *	3.76E-09	SBO38E		0.01	4.20E-05	99.95
59 *	3.30E-09	SWS2C		0.01	4.20E-05	99.96
60 *	2.79E-09	SWS2A		0.01	4.20E-05	99.97
61	2.78E-09	SBO5E		0.01	4.20E-05	99.98
62	2.35E-09	SWS3B		0.01	4.20E-05	99.98
63	2.14E-09	SWS4C		0.01	4.20E-05	99.99
64	1.[illegible]3E-09	SWS13A	SWS004	0.00	4.20E-05	99.99
65	1.15E-09	SWS3C	SWS007	0.00	4.20E-05	99.99
66	1.10E-09	TRO4A		0.00	4.20E-05	99.99
67	4.65E-10	LSP4E		0.00	4.20E-05	100.00
68	3.80E-10	FL3BE		0.00	4.20E-05	100.00
69	3.78E-10	CCW31E		0.00	4.20E-05	100.00
70	3.24E-10	MLO2E		0.00	4.20E-05	100.00
71	2.51E-10	SBO34E		0.00	4.20E-05	100.00
72	2.23E-10	LSP5E		0.00	4.20E-05	100.00
73	1.98E-10	LLO2E		0.00	4.20E-05	100.00

The Sequence I.D.s are derived from the Level 1 Event Trees (e.g., SBO4C, Station Blackout - Sequence Number 4); and a letter designator for the availability of Containment Cooler, Spray, and Isolation systems as follows:

Sequence ID End Letter derived from status of containment systems:

- A No Coolers, No sprays
- B Coolers; No Sprays
- C Sprays; No Coolers
- D Coolers and Sprays
- E Isolation Failure

Sequence 36, 38, 41, 42, 43, 44, 45, 47, 49, 53, 54, 57, 58, 59, and 60 contribute >95 % of the containment failure frequency.

Table 4.3-2

Level 2 MAAP Source Term Quantification Summary for WCGS

MAAP RUN SUMMARY						
Analyzed Seq. ID	CET End State	Binned Sequences	Source Term Release Category	Cumulative Frequency	% Volatile Release for Analyzed Sequence	Total % Release: (Freq x % Release)
$SBO7D_{ED}$	1	CCW2D, LLO2D, LLO3D, SBO2D, SBO3D, SBO5D, $SBO7D_{ED}$, $SBO7D_{DP}$, SWS2B, SWS2D, SWS3B, SWS3D, VE?1D	S	8.86E-6	2.17E-3	1.92E-8
$SBO7D_{ED}$	2	SWS2C, SWS3C	[illegible]	4.45E-9	2.17E-3	9.66E-12
SBO5A	3	SBO5A, SWS2A	[illegible]	1.20E-6	7.43E-2	8.92E-8
LLO3A	6	LLO3A	K	7.39E-9	6.58E-3	4.86E-11
LSP4D	7	LSP3D, LSP4D, LSP5D, MLO2D, SBO34D, SBO35D, SBO36B, SBO36D, SLO3D, SWS13D, TRA5D, TRO4D	S	1.26E-5	2.91E-3	3.68E-8
LSP4C	8	LSP3C, LSP4C, SBO36C, SWS13C	K	1.56E-7	4.54E-5	7.08E-12
DCC3A	9	DCC3A, LSP3A, MLO2A, SLO3A	J	3.83E-8	8.82	3.38E-7
SWS13B	10	SWS13B	A	8.92E-9	1.56E-2	1.39E-10
SWS13A	12	FL4A, SBO8A, SBO36A, SBO38A, SWS13A, SWS15A, TRO4A	A	1.25E-5	4.99E-4	6.24E-9
SBO38E	14	FL4E, LLO2E, LSP4E, LSP5E, MLO2E, SBO2E, SBO5E, SBO7E, SBO8E, SBO34E, SBO36E, SBO38E	G	4.95E-8	6.92	3.43E-7
ISL1D	Bypass	ISL1D	T	6.11E-[illegible]	85.8	5.24E-6
SGR19D	Bypass	SGR19D	D	1.2[illegible]-8	5.77	6.92E-8
Total Frequency and Total Release of Sequences Considered for Level 2:				3.55E-5 (97.8% of CDF)		6.14E-6 % Volatile Released Per Year

NOTES:

1. Sequences CCW4D, CCW31D, CCW31E, FL3A, FL3B, FL3BE, SGR3D, SGR5D, SGR14A, SGR14D, SWS4B, SWS4C, and SWS4D (6.52E-6 total CDF) are excluded from the Cumulative Frequency since MAAP analysis did not indicate core uncovery/damage within the Level 1 24 hour mission time.
2. Source term from sequence $SBO7D_{ED}$ used for Bin 2, since SWS3C did not go to core damage.
3. Reduced Core Damage Frequency for Level 2 = 3.63E-5 per year.

Table 4.3-3
WOLF CREEK GENERATING STATION
AIRBORNE RELEASE CATEGORY AND PROBABILITY

RELEASE CATEGORY	DEFINITION	FREQUENCY	P(RC|CD)[1,2]
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful)	2.15E-5	0.593
A	No containment failure within mission time	1.25E-5	0.345
K	Late containment failure - <0.1% volatiles released	1.37E-6	3.78E-2
T	Containment bypassed - >10% volatiles released	7.31E-3	2.01E-3
G	Containment failure prior to vessel failure - <10% volatiles released	4.95E-8	1.36E-3
J	Early containment failure - <10% volatiles released	3.83E-8	1.06E-3
D	Containment bypassed with noble gases and up to 10% of the volatiles released	1.20E-8	3.31E-4

NOTE:
1. Conditional probability of release category given core damage.
2. Reduced Core Damage Frequency for Level 2 = 3.63E-5 per year.

Table 4.3-4
CONTAINMENT RELEASE MODE
AND CONTRIBUTION TO SOURCE TERM

Release Mode	Failure Frequency	Contribution to Source Term, %	Total Volatile Release (% per year)
Containment Bypass (ISL1, SGR19D)	7.31E-8	86.5	5.31E-6
Containment Overpressure Failure (Bin 2,3,8,9)	1.40E-6	6.9	4.27E-7
Containment Isolation Failure (Bin 14)	4.95E-8	5.6	3.43E-7
Containment Failure Frequency	1.52E-6	----	----
Containment Normal Leakage (Bin 1,4,5,6,7,10,11,12)	3.40E-5	1.0	6.23E-8
Total	2.67E-5	100.0	6.14E-6

Containment Failure Frequency: 1.52E-6
Conditional Probability of Containment Failure Given Core Damage:
P(CF/CD) = 1.52E-6 / 3.63E-5 = 0.042

TABLE 4.3-5
Page 1 of 2
RELEASE CATEGORY DEFINITION

Release Category	Definition
A	No containment failure within 48 hour mission time but failure could eventually occur without accident management action; noble gases and less than 0.1% volatiles released.
B	Containment bypassed with noble gases plus less than 0.1% of the volatiles released.
C	Containment bypassed with noble gases plus up to 1% of the volatiles released.
D	Containment bypassed with noble gases and up to 10% of the volatiles released.
E	Containment failure prior to vessel failure with noble gases and less than 0.1% of the volatiles released (containment isolation impaired).
F	Containment failure prior to vessel failure with noble gases and up to 1% of the volatiles released (containment isolation impaired).
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired).
H	Early containment failure with the noble gases and less than 0.1% volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful).
I	Early containment failure with noble gases and up to 1% of the volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful).
J	Early containment failure with noble gases and up to 10% of the volatiles released (containment failure within six hours of vessel failure, containment not bypassed; isolation successful).
K	Late containment failure with noble gases and less than 0.1% volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).

TABLE 4.3-5
Page 2 of 2
RELEASE CATEGORY DEFINITION

Release Category	Definition
L	Late containment failure with noble gases and up to 1% of the volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).
M	Late containment failure with noble gases and up to 10% of the volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).
N	Late containment failure with noble gases and up to 1% of the volatiles and up to 0.1% of the non-volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).
P	Not used.
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful).
T	Containment bypassed with noble gases and more than 10% of the volatiles released.
U	Containment failure prior to vessel failure with the noble gases and more than 10% of the volatile fission products released (containment isolation impaired).
V	Early containment failure with noble gases and more than 10% of the volatiles released (containment failure within 6 hours of vessel failure; containment not bypassed; isolation successful).
W	Late containment failure with noble gases and more than 10% of the volatiles released (containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful).

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE								
MAAP Case I.D.	CCW001	CCW002	DCC001	4DCC001		LLO003	1LLO003	2LLO003
Sequence Frequency	1.05E-7	2.19E-6	5.80E-9	—		7.39E-9	—	—
Sequence Designator	CCW4D	CCW31D	DCC3A	DCC3A		LLO3A	LLO3A	LLO3A
Sensitivity Parameter (See 4.4.0)	—	—	—	TTRX=0.5 HR		—	TTRX=0.5 HR	PCF=13[illegible].7 PSIA
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	—	—	12.14	12.40		0.008	0.14	0.01
Onset of Core Melt (hr)	—	—	13.3	13.56		0.40	0.35	0.40
Time of Vessel Failure (hr)	—	—	15.29	16.05		0.89	1.29	0.89
Time of Containment Failure (hr)	—	—	15.45	16.06		16.76	16.4	23.55
Annular Compartment Temperature (max. °F)	147	145	348	345		445	450	446
Annular Compartment Pressure (max. psi)	20	[illegible]	124	123		132	124	139
Fraction of Clad Reacted in Vessel	0.0	0.0	0.459	0.464		0.343	0.354	0.343
H2 Mass Burned (lbm)	0.0	0.0	0.0	0.0		276	253	275
Rx Cavity Condition @ End of Mission Time	WET	WET	DRY	DRY		DRY	DRY	DRY
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	0.0	0.0	0.26	0.01		6.4	6.4	6.4
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	0.0	0.0	95.2	95.1		30.4	30.7	24.8
Volatile FP Released from Containment (%)	0.0	0.0	8.82	9.26		6.58E-3	5.68E-3	4.94E-3
Non-Volatile FP Released from Containment (%)	0.0	0.0	0.471	0.410		1.86E-3	2.13E-3	1.23E-3
Volatile FP Retained in Primary System %	0.0	0.0	44.1	44.4		4.60	4.43	4.60

NOTES:
- A. Containment Isolation Failure
- B. Containment bypass (SGR, ISL).
- 1. Mission time for Level 2 is 48 hours.
- 2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE								
MAAP Case I.D.	LSP002	LSP003	LSP004	MLO001	MLO002	ISL001	ISL002	
Sequence Frequency	1.23E-7	2.69E-6	[illegible].20E-9	1.87E-6	8.79E-9	6.11E-8	6.11E-8	
Sequence Designator	LSP4C	LSP4D	LSP3A	MLO2D	MLO2A	ISL1D	ISL1D	
Sensitivity Parameter (See 4.4.0)	---	---	---	---	---	ABB=3"	ABB=2"	
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	2.23	2.22	2.04	0.97	0.97	3.56	4.48	
Onset of Core Melt (hr)	2.89	2.88	21.59	11.74	11.75	4.09	5.2	
Time of Vessel Failure (hr)	3.27	3.26	23.91	13.83	13.88	4.46	6.73	
Time of Containment Failure (hr)	23.32	---	19.64	---	13.96	B	B	
Annular Compartment Temperature (max. °F)	323	352	346	318	352	175	180	
Annular Compartment Pressure (max. psi)	114	41	114	35	128	25	27	
Fraction of Clad Reacted in Vessel	0.440	0.438	0.483	0.391	0.362	0.440	0.433	
H2 Mass Burned (lbm)	0.0	1573	0	417	0.0	663	683	
Rx Cavity Condition @ End of Mission Time	WET	DRY	WET	WET	DRY	DRY	DRY	
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	0.0	5.6	0.002	0.002	1.63	5.7	5.2	
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	89.8	9.89E-2	91.1	8.49E-2	95.6	98.5	96.5	
Volatile FP Released from Containment (%)	4.54E-5	2.92E-[illegible]	7.76	1.43E-3	6.60	85.8	79.4	
Non-Volatile FP Released from Containment (%)	1.06E-9	1.32E-4	0.717	4.11E-5	0.317	0.25[illegible]	14.3	
Volatile FP Retained in Primary System %	85.2	63.1	59.5	18.6	18.4	12.0	19.0	

NOTES:
A. Containment Isolation Failure
B. Containment bypass (SGR, ISL).
1. Mission time for Level 2 is 48 hours.
2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE
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SEQUENCE TYPE								
MAAP Case I.D.	SBO002	1SBO002	2SBO002	3SBO002	4SBO002	SBO004		
Sequence Frequency	1.46E-7	---	---	---	---	5.90E-6		
Sequence Designator	SBO7DED	SBO7DED	SBO7DED	SBO7DED	SBO7DED	SBO8A		
Sensitivity Parameter (See 4.4.0)	---	FLPHI = 100	FCFH = 0.06	DXHIG = 0.1	FENTR = 1.0	---		
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	2.12	2.12	2.12	2.12	2.12	11.7		
Onset of Core Melt (hr)	3.04	3.04	3.04	3.04	3.04	12.8		
Time of Vessel Failure (hr)	3.66	3.66	3.66	3.66	3.66	13.6		
Time of Containment Failure (hr)	---	---	---	---	---	A		
Annular Compartment Temperature (max. °F)	592 (2)	365	605 (2)	563	528 (2)	303		
Annular Compartment Pressure (max. psi)	70	67	71	86	61	38		
Fraction of Clad Reacted in Vessel	.470	0.470	0.474	0.470	0.470	0.445		
H2 Mass Burned (lbm)	1254	685	1262	1100	1269	897		
Rx Cavity Condition @ End of Mission Time	WET	WET	WET	Wet	WET	DRY		
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	1.1	1.1	1.1	1.1	1.3	4.6		
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	0.110	0.113	0.110	0.111	0.110	82.8		
Volatile FP Released from Containment (%)	2.17E-3	2.20E-3	2.26E-3	2.22E-3	2.21E-3	1.86		
Non-Volatile FP Released from Containment (%)	7.73E-5	7.73E-5	7.38E-5	7.67E-5	9.97E-5	0.250		
Volatile FP Retained in Primary System %	15.1	14.9	15.3	14.4	15.0	82.5		

NOTES:
A. Containment Isolation Failure
B. Containment bypass (SGR, ISL).
1. Mission time for Level 2 is 48 hours.
2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE								
MAAP Case I.D.	SBO005	1SBO005	2SBO005	3SBO005	4SBO005	5SBO005	6SBO005	7SBO005
Sequence Frequency	1.68E-6	---	---	---	---	---	---	---
Sequence Designator	SBO38A	SBO38A	SBO38A	SBO38A	SBO38A	SBO38A	SBO38A	SBO38A
Sensitivity Parameter (See 4.4.0)	---	FLPHI = 100	FCRBLK = 1	FCHF = 0.06	FCRDR = 0.8	FCSIVP = 0.1	FCSIVP = 0.1, 1.0	ALKNOM = 2*
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	2.22	2.22	2.22	2.22	2.22	2.22	2.23	2.22
Onset of Core Melt (hr)	2.87	2.87	2.87	2.87	2.87	2.87	2.88	2.88
Time of Vessel Failure (hr)	3.25	3.25	3.32	3.25	3.25	3.27	3.26	3.25
Time of Containment Failure (hr)	A	A	A	A	A	A	A	A
Annular Compartment Temperature (max. °F)	364	365	364	365	365	364	365	393
Annular Compartment Pressure (max. psi)	62	61	59	62	61	61	61	67
Fraction of Clad Reacted in Vessel	0.451	0.451	0.302	0.443	0.451	0.464	0.440	0.454
H2 Mass Burned (lbm)	917	916	771	956	916	974	945	372
Rx Cavity Condition @ End of Mission Time	DRY	DRY	DRY	DRY	DRY	DRY	DRY	DRY
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	5.6	5.6	5.6	5.6	5.6	5.6	5.6	5.6
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	94.5	94.5	94.0	94.5	94.5	94.2	94.6	74.1
Volatile FP Released from Containment (%)	6.92	6.92	6.49	6.27	6.92	5.14	6.74	0.811
Non-Volatile FP Released from Containment (%)	0.212	0.212	0.234	0.228	0.212	0.206	0.211	8.86E-2
Volatile FP Retained in Primary System %	68.7	68.7	72.8	70.6	68.6	75.2	69.3	87.6

NOTES:
A. Containment Isolation Failure
B. Containment bypass (SGR, ISL).
1. Mission time for Level 2 is 48 hours.
2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE
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SEQUENCE TYPE								
MAAP Case I.D.	8SBO005	9SBO005	10SBO005	11SBO005	12SBO005	SGR005	SLO003	ISLO003
Sequence Frequency	—	—	—	—	—	1.46E-7	6.72E-7	1.85E-8
Sequence Designator	SBO38A	SBO38A	SBO38A	SBO38A	SBO38A	SGR3D	SLO3D	SLO3A
Sensitivity Parameter (See 4.4.0)	ALKNOM = 18"	ALKNOM = 6"	DXHSG = 0.1	ABB = 1.0	FENTR = 1.0	—	—	—
CORE/CONTAINMENT RESPONSE (1)			(See Table 4.4-2)					
Time of Core Uncovery (hr)	2.22	2.21	2.22	0.52	2.22	—	26.47	26.86
Onset of Core Melt (hr)	2.87	2.87	2.87	0.79	2.87	—	28.49	23.87
Time of Vessel Failure (hr)	3.25	3.27	3.25	1.36	3.25	—	29.06	29.46
Time of Containment Failure (hr)	A	A	A	A	A	—	—	29.60
Annular Compartment Temperature (max. °F)	361	342	358	364	365	168	[illegible]5	348
Annular Compartment Pressure (max. psi)	31	49	61	62	61	17	[illegible]	117
Fraction of Clad Reacted in Vessel	0.442	0.461	0.451	0.377	0.451	0.0	0.440	0.434
H2 Mass Burned (lbm)	489	282	0	458	916	0.0	564	0.0
Rx Cavity Condition @ End of Mission Time	DRY	DRY	DRY	DRY	DRY	DRY	WET	WET
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	5.7	5.6	5.6	6.4	3.6	0.0	0.0	0.0
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	99.5	98.4	94.3	96.0	94.5	0.0	4.49E-2	76.3
Volatile FP Released from Containment (%)	14.6	11.1	6.3	5.25	6.92	0.0	7.60E-4	0.243
Non-Volatile FP Released from Containment (%)	0.413	0.969	0.212	0.220	0.210	0.0	1.57E-8	6.43E-5
Volatile FP Retained in Primary System %	53.8	55.0	70.8	37.7	68.7	0.0	50.0	51.4

NOTES:
A. Containment Isolation Failure
B. Containment bypass (SGR, ISL).
1. Mission time for Level 2 is 48 hours.
2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE
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SEQUENCE TYPE								
MAAP Case I.D.	SWS003	SWS004	1SWS004	2SWS004	SWS005	SWS006	SWS007	SWS008
Sequence Frequency	9.91E-7	1.23E-9	—	—	8.92E-9	2.66E-7	1.15E-9	4.38E-9
Sequence Designator	SWS13D	SWS13A	SWS13A	SWS13A	SWS13B	SWS3A	SWS3C	SWS4B
Sensitivity Parameter (See 4.4.0)	—	—	FCRBLK=1	TTRX=0.5 HR	—	—	95 GPM S.L.	125 GPM S.L.
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	2.21	2.20	2.20	2.20	2.22	—	—	—
Onset of Core Melt (hr)	2.86	2.87	2.87	2.87	2.88	—	—	—
Time of Vessel Failure (hr)	3.25	3.26	3.35	3.75	3.27	—	—	—
Time of Containment Failure (hr)	—	—	—	—	—	—	—	—
Annular Compartment Temperature (max. °F)	338	437	426	425	362	162	162	169
Annular Compartment Pressure (max. psi)	42	112	10[illegible]	113	43	20.10	20.10	21.01
Fraction of Clad Reacted in Vessel	0.453	0.454	0.309	0.456	0.460	0.0	0.0	0.0
H2 Mass Burned (lbm)	577	233	209	240	1617	0.0	0.0	0.0
Rx Cavity Condition @ End of Mission Time	WET	DRY	DRY	DRY	DRY	DRY	SRY	DRY
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	0.0	5.6	5.6	5.6	5.6	0.0	0.0	0.0
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	0.105	0.130	0.129	0.131	0.098	0.0	0.0	0.0
Volatile FP Released from Containment (%)	6.74E-5	4.99E-4	2.71E-4	5.84E-4	1.56E-2	0.0	0.0	0.0
Non-Volatile FP Released from Containment (%)	1.03E-8	8.20E-5	3.82E-5	7.70E-5	1.30E-4	0.0	0.0	0.0
Volatile FP Retained in Primary System %	78.0	89.2	96.9	87.9	76.4	100	100	100

NOTES
A. Containment Isolation Failure
B. Containment bypass (SGR, ISL).
1. Mission time for Level 2 is 48 hours.
2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-6
WOLF CREEK GENERATING STATION
MAAP RUN SUMMARY TABLE

SEQUENCE TYPE								
MAAP Case I.D.	SBO010	SGR007	1SGR007					
Sequence Frequency	1.20E-6	1.20E-8	1.20E-8					
Sequence Designator	SBO5A	SGR19D	SGR19D					
Sensitivity Parameter (See 4.4.0)	----	1 SI Pump W/ Level Control	1 CHRG Pump No Level Control					
CORE/CONTAINMENT RESPONSE (1)								
Time of Core Uncovery (hr)	18.80	3.87	24.41					
Onset of Core Melt (hr)	19.50	4.76	25.92					
Time of Vessel Failure (hr)	21.24	5.46	26.44					
Time of Containment Failure (hr)	42.06	B	B					
Annular Compartment Temperature (max. °F)	[illegible]	271	309					
Annular Compartment Pressure (max. psi)	33[illegible]	33	35					
Fraction of Clad Reacted in Vessel	0.530	0.505	0.510					
H2 Mass Burned (lbm)	0.0	0.0	1262					
Rx Cavity Condition @ End of Mission Time	Wet	Wet	Dry					
Cavity Concrete Abaltion Depth at End of Mission Time (ft)	0.0	0.0	2.2					
FISSION PRODUCT DISTRIBUTION AT END OF MISSION TIME								
Noble Gas Released from Containment (%)	33.2	66.7	32.0					
Volatile FP Released from Containment (%)	0.074	5.77	2.60					
Non-Volatile FP Released from Containment (%)	3.66E-5	9.05E-3	3.63E-5					
Volatile FP Retained in Primary System %	52.6	59.[illegible]	59.1					

NOTES:
- A. Containment Isolation Failure
- B. Containment bypass (SGR, ISL).
- 1. Mission time for Level 2 is 48 hours.
- 2. Peak temperature is due to short lived spike following H_2 burn.

Table 4.3-7

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
24 HOURS AFTER ACCIDENT INITIATION

	Airborne Release (%) @ 24 Hours Analyzed Case Identifier					
Bin	1	2	3	6	7	
	$SBO7D_{ED}$	$SBO7D_{ED}$	SBO5A	LLO3A	LSP4D	
1) Noble	5.51E-02	5.51E-02	8.91E-03	7.92	4.62E-02	
2) CsI	2.09E-03	2.09E-03	1.15E-03	3.54E-03	1.74E-03	
3) TeO_2	3.26E-05	3.26E-05	0.0	2.71E-04	2.14E-06	
4) SrO	4.55E-05	4.55E-05	3.87E-06	3.76E-04	8.71E-05	
5) MoO_2	2.41E-04	2.41E-04	1.65E-04	4.18E-04	2.13E-05	
6) CsOH	2.02E-03	2.02E-03	9.12E-04	7.48E-03	1.53E-03	
7) BaO	8.91E-05	8.91E-05	5.08E-05	2.58E-04	4.43E-05	
8) La_2O_3	6.37E-06	6.37E-06	2.61E-07	7.68E-05	1.01E-05	
9) CeO_2	7.72E-05	7.73E-05	1.57E-06	8.18E-04	1.31E-04	
10) Sb	1.88E-03	1.88E-03	1.44E-03	0.145	1.37E-03	
11) Te_2	2.60E-03	2.60E-03	0.0	0.137	1.74E-03	
12) UO_2	2.52E-07	2.52E-07	0.0	3.35E-06	4.46E-07	

Table 4.3-7

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
24 HOURS AFTER ACCIDENT INITIATION
Page 2 of 3

	Airborne Release (%) @ 24 Hours Analyzed Case Identifier					
Bin	8	9	10	12	14	
	LSP4C	DCC3A	SWS13B	SWS13A	SBO38E	
1) Noble	6.39	55.6	4.56E-02	5.92E-02	74.9	
2) CsI	4.54E-05	3.34	8.44E-04	4.51E-04	1.49	
3) TeO_2	0.00E+00	0.0	7.46E-04	3.19E-05	1.07E-01	
4) SrO	1.29E-8	7.87E-02	8.33E-05	4.27E-05	1.20E-01	
5) MoO_2	1.45E-07	1.44	7.80E-06	2.75E-05	4.34E-02	
6) C H	4.30E-05	1 85	8.45E-04	4.92E-04	1.58	
7) BaO	7.25E-08	8.68E-1	4.65E-05	2.96E-05	7.15E-02	
8) La_2O_3	8.48E-10	1.16E-2	9.75E-6	6.35E-06	1.92E-02	
9) CeO_2	1.06E-09	4.57E-1	1.28E-04	7.94E-05	2.10E-01	
10) Sb	1.98E-06	7.00	1.32E-03	1.84E-03	4.14	
11) Te_2	0.00E+00	4.11E-03	9.96E-04	3.14E-03	6.45	
12) UO_2	0.00E+00	8.13E-08	4.31E-07	2.61E-07	6.74E-04	

Table 4.3-7

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
24 HOURS AFTER ACCIDENT INITIATION
Page 3 of 3

Bin	Airborne Release (%) @ 24 Hours Analyzed Case Identifier					
	Cont. Bypass	Cont. Bypass				
	ISL001	SGR19D				
1) Noble	98.2	66.7				
2) CsI	84.4	5.77				
3) TeO_2	1.72E-03	0.0				
4) SrO	3.76E-01	2.69E-02				
5) MoO_2	8.81	1.58				
6) CsOH	85.6	5.50				
7) BaO	2.48	3.60E-01				
8) La_2O_3	2.97E-02	1.79E-03				
9) CeO_2	2.50E-01	9.05E-03				
10) Sb	34.8	8.42				
11) Te_2	3.24	0.0				
12) UO_2	7.74E-04	0.0				

Table 4.3-8

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
48 HOURS AFTER ACCIDENT INITIATION
Page 1 of 3

Bin	Airborne Release (%) @ 48 Hours Analyzed Case Identifier					
	1	2	3	6	7	
	$SBO7D_{ED}$	$SBO7D_{ED}$	SBO5A	LLO3A	LSP4D	
1) Noble	1.10E-01	1.10E-01	33.2	30.4	9.89E-02	
2) CsI	2.17E-03	2.17E-03	7.43E-02	6.58E-03	2.92E-03	
3) TeO_2	3.26E-05	3.26E-05	0.0	5.10E-04	2.21E-06	
4) SrO	4.55E-05	4.55E-05	8.05E-05	8.34E-04	8.78E-05	
5) MoO_2	2.41E-04	2.41E-04	3.44E-03	7.45E-04	2.13E-05	
6) CsOH	2.11E-03	2.11E-03	6.93E-02	2.23E-02	2.75E-03	
7) BaO	8.91E-05	8.91E-05	1.08E-03	5.34E-04	4.47E-05	
8) La_2O_3	6.37E-06	6.37E-06	5.79E-06	1.73E-04	1.02E-05	
9) CeO_2	7.73E-05	7.73E-05	3.66E-05	1.86E-03	1.32E-04	
10) Sb	1.91E-03	1.91E-03	4.75E-02	0.478	1.48E-03	
11) Te_2	2.60E-03	2.60E-03	0.0	0.533	1.75E-03	
12) UO_2	2.52E-07	2.52E-07	0.0	7.64E-06	4.52E-07	

Table 4.3-8

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
48 HOURS AFTER ACCIDENT INITIATION

Page 2 of 3

Bin	Airborne Release (%) @ 48 Hours Analyzed Case Identifier					
	8	9	10	12	14	
	LSP4C	DCC3A	SWS13B	SWS13A	SBO38E	
1) Noble	89.8	95.2	9.79E-02	1.30E-01	94.5	
2) CsI	4.54E-05	8.82	1.56E-02	4.99E-04	6.92	
3) TeO_2	0.00E+00	0.0	7.51E-04	3.37E-05	1.07E-01	
4) SrO	1.29E-08	9.37E-02	8.41E-05	4.37E-05	1.20E-01	
5) MoO_2	1.45E-07	1.47	7.98E-06	2.[illegible]	4.34E-02	
6) CsOH	4.42E-05	5.10	1.34E-03	5.64E-[illegible]	6.[illegible]	
7) BaO	7.25E-08	8.89E-01	4.75E-05	3.00E-05	7.18E-02	
8) La_2O_3	8.48E-10	1.21E-02	9.86E-06	6.54E-06	1.92E-02	
9) CeO_2	1.06E-09	4.72E-01	1.30E-04	8.20E-05	2.12E-01	
10) Sb	1.58E-05	12.0	1.45E-03	2.08E-03	4.68	
11) Te_2	0.00E+00	2.18	1.00E-03	3.15E-03	6.45	
12) UO_2	0.00E+00	1.69E-05	4.37E-07	2.70E-07	6.78E-04	

Table 4.3-8

WOLF CREEK GENERATING STATION
AIRBORNE FISSION PRODUCT RELEASE (%)
48 HOURS AFTER ACCIDENT INITIATION
Page 3 of 3

	Airborne Release (%) @ 48 Hours Analyzed Case Identifier					
Bin	Cont. Bypass	Cont. Bypass				
	ISL001	SGR19D				
1) Noble	98.5	66.7				
2) CsI	85.8	5.77				
3) TeO_2	1.94E-03	0.0				
4) SrO	3.76E-01	2.69E-02				
5) MoO_2	8.81	1.58				
6) CsOH	87.6	5.50				
7) BaO	2.48	3.60E-01				
8) La_2O_3	2.97E-02	1.79E-03				
9) CeO_2	2.51E-01	9.05E-03				
10) Sb	34.8	8.42				
11) Te_2	3.27	0.0				
12) UO_2	7.75E-04	0.0				

4.4 WCGS IPE - LEVEL 2 SENSITIVITY ANALYSIS

4.4.1 Approach

NUREG-1335 has identified in-vessel and ex-vessel phenomena which might have significant effects on containment failure timing and the associated source term release. These phenomena are listed in Table 4.4-1. The purpose of this section is to address the above phenomenological uncertainties.

In order to address the uncertainties associated with in-vessel and ex-vessel phenomena outlined in NUREG-1335, a two step approach was taken. The first step addressed the phenomena in detailed Phenomenological Evaluation Summary papers. These phenomenological evaluations are concerned with the unlikelihood or likelihood of early or late containment failure mechanisms, as well as containment fragility. Since the phenomena were conservatively evaluated as not contributing to containment failure, uncertainties associated with these phenomena no longer considered.

The second step addresses, through the variation of relevant MAAP model parameters, phenomenological uncertainties not predisposed of in the Phenomenological Evaluation Summaries. The ranges of MAAP model parameters for sensitivity analyses are recommended in the EPRI document EPRI TR-100167. With these two steps, all phenomena relevant to Wolf Creek are addressed.

4.4.2 Scope

Table 4.4-1 lists (1) phenomena identified in NUREG-1335 for sensitivity analyses, (2) WCGS analyses performed to address the corresponding phenomena, and (3) related MAAP parameters for sensitivity runs. Table 4.4-2 lists the values of the MAAP parameters used with various base case dominant accident sequences to perform the sensitivity study. Refer to Table 4.4-1 to correlate the MAAP parameter with the uncertainty.

Phenomena that may have impact on source term and will be studied as sensitivity cases are as follows:

- Hydrogen burn completeness
- In-vessel hydrogen production/core relocation
- Hot leg creep rupture failure in a high pressure sequence
- RV failure mode
- Containment failure pressure and area

- Volatile fission product release/retention in the primary system

- Ex-vessel debris coolability

Results of the phenomenological evaluation summaries and MAAP sensitivity analyses concludes that WCGS is not susceptible to the failures and the source term is not affected by the severe accident phenomena. MAAP sensitivity cases of the threshold for hydrogen burn did not perform as expected. Investigation of these cases will continue. Summaries of the MAAP analyses are shown in Table 4.3-5.

Although not specifically a part of the sensitivity analysis discussed here, RCP seal LOCAs created their own sensitivity study. Many of the event trees contain the node CNU, which pose the time and break-flow dependent question, 'Is the core uncovered?' MAAP analysis of numerous sequences with cooldown indicated that a significantly larger break flow was required to satisfy this node. The event tree CNU node generally indicates that a seal LOCA break flow of 21-50 gpm per reactor coolant pump will lead to core uncovery at eight hours. MAAP analysis indicated that a break flow of 100-200 gpm was necessary to force core uncovery in the cases with cooldown. Since the larger break size was used in these sequences, then the correspondingly smaller frequencies for these LOCAs should be used.

Table 4.4-1
Sensitivity Studies

Phenomena	Analyses Performed	Related MAAP Parameter
-Performance of containment heat removal systems	- Conservatively assumed only 1 train of Fan Coolers in MAAP analyses. No further sensitivity analysis is required.	
-In-Vessel Phenomena		
-H_2 production and combustion in containment	- MAAP analysis (SBO7D_{ED}, SBO38A) using a higher value of "flame flux multiplier" which promotes H_2 burn completeness	FLPHI
	- MAAP analysis (SBO38, SWS13A) with and w/o core blockage model which results in different in-vessel hydrogen production	FCRBLK
	- Threshold for H_2 burn during SBO increased to induce larger burns for SBO7D_{ED}, SBO38A	DXHIG
	- Discussed in summary paper on hydrogen combustion; sensitivity analysis is required	
- Core relocation characteristics	- MAAP analysis (SBO38A, SWS13A) with and w/o core blockage model	FCRBLK
- Fuel/coolant interactions	- MAAP analysis (SBO7D_{ED}, SBO38A) using reduced critical heat flux	FCHF
- Mode of RV melt-through	- Discussed in summary paper on thrust forces at RPV failure. Thrust forces would not cause containment failure; sensitivity analysis on thrust forces not required.	
	- Base case analyses assumed a 60 sec. melt-through time; sensitivity with 1800 sec melt-through time performed for DCC3A, LLO3A	TTRX
- Induced failure of RCS pressure boundary at high RCS pressure/temperature	- SBOs assumed range of induced pump seal LOCAs from 21 gpm to 480 gpm per pump. Analysis of SBO38A with hot leg failure was performed	ABB

Table 4.4-1
Sensitivity Studies
Page 2 of 2

Phenomena	Analyses Performed	Related MAAP Parameter
- Ex-Vessel Phenomena		
- Direct containment heating (at high RCS pressure)	- Addressed in summary paper on DCH as not causing containment failure; no further analysis is required.	
- Potential for early containment failure due to pressure load	- Early containment failure modes due to ex-vessel steam explosions and hydrogen detonation addressed in summary papers as not causing containment failure; no further analysis is required. - Early containment failure (defined as less than 6 hrs after vessel failure) were analyzed in base case DCC3A, SLO3A.	
- Early failure via debris attack of containment penetrations	- Uncertainties addressed in summary paper on Penetration Thermal Attack as not causing containment failure, no further analysis is required.	
- Long-term core-concrete interaction	- Discussed in MCCI summary paper as not causing basemat failure prior to containment overpressure. - Concrete erosion with varying degree of severity is a common phenomenon in base case sequences.	
- Water availability	- It was a common situation in several base cases that core debris remained uncooled in the dry cavity.	
- Debris coolability	- MAAP analysis ($SBO7D_{ED}$, SBO38A) assumed reduced debris coolability by reducing FCHF.	FCHF

Table 4.4-2

MAAP SENSITIVITY RUN MATRIX FOR WCGS

Sensitivity	DCC3A	LLO3A	$SBO7D_{ED}$	SBO38A	SWS13A
INDUCED ABB = 1'				11SBO005	
ALKNOM = 2"				7SBO005	
ALKNOM = 6"				9SBO005	
ALKNOM = 18"				8SBO005	
DXHIG = 0.10			3SBO002	10SBO005	
FCHF = .06			2SBO002	3SBO005	
FCRBLK = 1				2SBO005	1SWS004
FCRDR = 0.8				4SBO005	
FCSIVP = 0.1				5SBO005	
FCSIVP = 0.1 @ T_0 = 1.0 @ VF				6SBO005	
FENTR = 1.0				12SBO005	
FLPHI = 100			1SBO002	1SBO005	
TTRX = 1800	4DCC001	1LLO003			2SWS004
PCF = 137.7		2LLO003			

Note: The MAAP case I.D. is listed for each sensitivity study (See Table 4.3-6).

4.5 LEVEL 2 CONCLUSIONS

The Level 2 analysis has reviewed containment failure modes and provided estimates of sequence progression and source term releases for those accident sequences that have significant contribution to the total Wolf Creek core damage frequency. Not only has this analysis provided quantitative information regarding containment failure probabilities and source term releases, but it has yielded useful, quantitative insights into the performance of the Wolf Creek containment as well. These insights are summarized below.

Overall, given core damage, there is a 95.8% probability of containment success. That is, there exists a 95.8% probability that the final barrier to fission product release to the environment will not be breached, impaired, or by-passed.

The contribution of the various fission product release modes and the associated range of volatile airborne source term, given core damage, are summarized as follows:

Release Mode	Contribution to Source Term, %	Total Volatile Release (% per year)
Containment Bypass	86.5	5.31E-6
Containment Overpressure Failure	6.9	4.27E-7
Containment Isolation Failure	5.6	3.43E-7
Containment Normal Leakage	1.0	6.23E-8
Total	100.0	6.14E-6

As shown, the overall release frequency of 6.14E-6% of volatiles released per year is dominated by releases due to containment bypass sequences of which interfacing system LOCAs make up the vast majority. These sequences contribute 86.5 % to the total release frequency. Containment over-pressurization contributes 6.9% and containment isolation failure sequences contribute 5.6%.

Other insights pertaining to the Wolf Creek containment are:

- The single most important feature of the Wolf Creek containment with regards to fission product retention is its ability to be maintain intact for several tens of hours following core melt. This allows natural deposition mechanisms to remove most airborne fission products from the containment atmosphere.

- No vulnerabilities to early containment failure modes are evident at Wolf Creek.

- About 34% of sequences going to core damage resulted in significant concrete ablation in the cavity (5-6 feet) by end of the 48 hour mission time due to MCCI. Sequences with this outcome are contained in source term bins 10 and 12 shown on Table 4.3-2. In general, these sequences will lead to overpressurization failures of the containment prior to the basemat failure. Certain sequences, however, have operable fan coolers which prevent overpressurization, yet concrete attack is significant (see SWS13B, for example). These types of sequences would result in basemat failures, assuming no recovery action, within 50 to 60 hours.

- The primary system proved to be a good barrier to fission product release even after vessel failure and during containment bypass sequences. This factor should not be neglected when considering the fission product retention capability of WCGS.

4.6 Generic Letter 88-20, Supplement 3: Containment Performance Improvements (CPI)

Generic Letter 88-20 Supplement 3 announces the completion of the NRC staff's CPI program, which focuses on the vulnerability of containments to severe accident challenges. As discussed in Section 4.2.5.2 of this report, hydrogen combustion is an unlikely failure mode for WCGS.

Potential detonability and flammability of the WCGS containment atmosphere was evaluated as part of the Level 2 PRA. Severe accident sources of combustible gases were identified and bounding calculations performed. No credit was taken for steam inerting of the containment and complete combustion was assumed. Based on adiabatic isochoric complete combustion of the selected bounding hydrogen inventory, the resulting containment pressure was less than the calculated value for the ultimate containment capacity.

Detonation due to a transition from deflagration to detonation (DDT) was considered. Two methods were used to assess the likelihood of the transition to detonation given the existence of a deflagration. Both of the methods resulted in a very low likelihood of a DDT at WCGS. Due to the small size of ignition sources required to initiate a deflagration it is far more likely that combustible gases would be consumed within containment by deflagration rather than by detonation. Furthermore, the containment would most likely fail due to over-pressurization well before such a large amount of hydrogen could accumulate.

None of the sequences addressed in the containment and source term analysis could realistically threaten containment due to hydrogen combustion. No WCGS containment vulnerabilities were identified as a result of Supplement 3 to Generic Letter 88-20.

5.0 UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

The Risk Assessment group was formed to develop the IPE for WCGS. Risk Assessment is part of WCNOC's Nuclear Analysis Division. Other individuals were identified throughout the company to coordinate IPE/PRA activities. These PRA Coordinators were selected from Operations, Training, Engineering (NPE), Licensing, and Safety Engineering. The PRA Coordinators were responsible for obtaining required input from their respective organizations and providing the information to the Risk Assessment analysts. The PRA Coordinators also made provision for the review, by their respective organization, of the PRA documentation during the development process.

5.2 Composition of Independent Review Team

The Nuclear Safety Engineering (NSE) group performed the independent review team function for the WCGS IPE. The Nuclear Safety Engineering group reports to the Chairman of the WCGS Nuclear Safety Review Committee (NSRC) and is responsible for safety audits/surveillance of the plant and other tasks as assigned by the NSRC.

The NSE engineers performing the review have Bachelor of Science Degrees in Nuclear Engineering, Mechanical Engineering, Electrical Engineering, and Electrical/Electronics Engineering. They have a combined 73 years of engineering experience, with 67 years related to the nuclear field. This includes 41 years of experience with the WCGS plant.

The Union Electric IPE team also performed a review of the WCGS PRA and its results. This provided an additional level of review by individuals knowledgeable in the PRA development process and familiar, by way of their experience with the Callaway plant, the WCGS power block design. WCGS and Callaway are sister plants designed and constructed under the Standardized Nuclear Unit Power Plant System (SNUPPS) concept.

5.3 Areas of Review, Major Comments, and Resolution of Comments

The WCGS PRA was reviewed at various stages of development by WCNOC personnel, Westinghouse personnel, and Union Electric Personnel. Some of the major comments received are listed below.

1) The interfacing system LOCA for WCGS utilized NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes", to determine the frequency of occurrence. An assumption was then made that if the event occurred, core damage would result. Union Electric utilized NSAC-154, "ISLOCA Evaluation Guidelines", which provides more insights into the ISLOCA event. WCNOC did not revise the existing WCGS PRA ISLOCA analysis, but intends to analyze the event using NSAC-154 guidelines for future reference and use.

2) Performance of the WCGS flooding analysis was accomplished using original WCGS flooding hazards analysis calculations for screening

purposes. Peer reviews of the flooding work identified problems with the use of these original calculations for screening. The calculations were based in part on assumptions different than those required for performance of the PRA internal flooding analysis. Therefore, the PRA flooding analysis missed potential flooding scenarios. The flooding analysis was reevaluated with additional flooding scenarios identified. The results from the analysis of the identified flooding scenarios were incorporated into the WCGS PRA.

3) The peer review of the WCGS station blackout (SBO) quantification process identified a potential for missing some important system dependencies. The WCGS quantification of the SBO event originally utilized a scalar value for the failures of both onsite emergency AC power systems following a loss of offsite power. The SBO event was requantified with the failures of both onsite emergency AC power systems imported as a cutset file instead of a scalar value. Quantification in this manner accounts for component or system failures which might impact the availability of other systems but which would not necessarily be restored by recovery of an offsite AC power source.

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

6.1 Unique Safety Features

WCNOC does not believe that WCGS has any safety features that are unique to the nuclear industry for this vintage of plant, although some design features of WCGS do help to reduce the likelihood of a core damage accident.

The design of the Emergency Core Cooling System (ECCS) at WCGS includes four high pressure pumps. The provision of two safety injection (SI) pumps, in addition to the two centrifugal charging pumps, gives WCGS a total of four high pressure ECCS pumps to provide RCS inventory makeup (injection) flow along with makeup flow for the feed and bleed cooling function.

The design of the ECCS at WCGS also includes provisions allowing either of the two low pressure (RHR) pumps to provide suction supply to all four high pressure ECCS pumps. This provision increases the likelihood of successful performance in the recirculation mode of the RCS inventory makeup and feed and bleed functions.

In addition to the two essential service water system pump trains, the design of the nonessential service water system at WCGS provides four pumps (three full flow and one low flow) for cooling water supply. The four nonessential service water pumps provide additional flexibility to meet plant cooling water needs and significantly reduces the likelihood of losing all service water cooling support.

The design of the WCGS containment reduces the frequency and magnitude of potential releases. The large, dry containment provides for approximately 2.5 million cubic feet of free volume. A containment capacity evaluation revealed that the containment can withstand pressures approximately twice the design pressure. The structural strength and volume features allow the containment to withstand a large mass and energy release without failing. Given the probability of containment failure, analysis shows the large dry containment would contain most releases for a period of time, allowing settling and impaction of airborne fission products prior to containment failure, thereby reducing the release.

6.2 Plant Improvements

WCNOC has not identified any vulnerabilities at WCGS and believes that the probability of core damage at WCGS is acceptably low. However, WCNOC is evaluating several areas of plant enhancements which, if implemented, would provide some reduction in the WCGS core damage frequency.

6.2.1 Plant Improvements Under Consideration

HIGH TEMPERATURE QUALIFIED RCP SEAL O-RINGS

The WCGS design presently does not utilize high temperature qualified RCP seal package O-rings which have recently been made available by Westinghouse. WCNOC will evaluate the use of RCP seal packages with the new O-rings as the need for replacements arises. Preliminary evaluations and discussions with Westinghouse have indicated that the new qualified O-rings would likely provide a reduction in the core damage frequency on the order of 2-5 percent.

REPLACEMENT OF THE POSITIVE DISPLACEMENT CHARGING PUMP

WCGS is evaluating replacement of the positive displacement charging pump (PDP) with a third centrifugal charging pump (CCP). The addition of a third CCP would be enhanced if the pump was self-cooled and powered from an independent back-up AC power supply. This design would provide a diverse means of RCP seal injection (cooling), RCS makeup, and another source of flow injection for bleed and feed operations.

PROVIDE A SWITCH TO BYPASS FEEDWATER ISOLATION IN ORDER TO RESTORE MAIN FEEDWATER

Following a reactor trip, main feedwater is isolated and auxiliary feedwater is actuated to provide for secondary heat removal. If auxiliary feedwater fails, procedures direct operators to restore main feedwater. To restore main feedwater, jumpers must be manually installed. There is a relatively short time for the operators to successfully accomplish this task prior to steam generators drying out. The human error probability associated with this task would be reduced by permanent installation of a main control board switch to accomplish this task. Plant modifications to accomplish this will be evaluated.

EQUIPMENT DEPENDENCE ON ROOM COOLING

Many of the key components included in the PRA have modeled dependencies on room cooling. WCNOC will evaluate, for selected key components, the impact on component function resulting from a loss of room cooling. If necessary, procedural guidance for implementation of actions to supply auxiliary cooling to specific rooms may be developed.

EMERGENCY PROCEDURES ASSOCIATED WITH TOTAL LOSS OF CCW & SW

The total loss or failure of the component cooling water (CCW) or service water (SW) supply systems is not directly addressed by the WCGS emergency procedures. The emergency procedures are based on the Westinghouse Emergency Operating Procedure (EOP) guidelines, which also do not directly address these events. WCNOC Operations is evaluating this with the Westinghouse Owners Group and other utilities. Since the Westinghouse Owners Group is currently involved in developing Accident Management guidelines, any additional WCGS event mitigating instructions could be included in the accident management procedures.

INTERNAL FLOODING

Internal flooding initiating events currently provide a significant contribution to the core damage frequency. Any action taken to reduce the initiating event frequency or to improve the mitigation of the event will improve the WCGS core damage frequency. WCNOC will initiate additional evaluation in this area to identify any procedural or hardware modifications to reduce the risk due to internal flooding.

GENERAL

WCNOC is working with the Westinghouse owners group to develop generic Accident Management guidelines Once developed, WCNOC will address plant specific accident management procedures for WCGS. Those procedures will be used to provide additional direction, as identified, for mitigation of WCGS specific events.

7.0 SUMMARY AND CONCLUSIONS

7.1 SUMMARY

WCNOC's goals in performing a PRA were to 1) fulfill the NRC Individual Plant Examination requirements (Generic Letter 88-20), 2) address Unresolved Safety Issues (USI) A-17 and A-45, and 3) develop a decision making tool based on plant risk.

WCNOC has met the first objective by performing a level 1 PRA for internal events including an internal flooding analysis, and a full level 2 PRA for containment analysis. This exceeds the minimum requirements for meeting the IPE and this report documents the results of our efforts as required.

The PRA results for the internal flooding analysis showed no significant vulnerabilities exist for WCGS, although two flooding sequences contribute approximately 16% to the total core damage frequency. WCNOC will evaluate these sequences for further means of mitigation and plant improvements as stated in Section 6.0. However, WCNOC considers this evaluation, with the and results of no significant hazards, sufficient to met the closure requirements of USI A-17.

Similarly, as required by Generic Letter 88-20, the WCGS PRA has addressed USI A-45, "Decay Heat Removal Evaluation", with the determination that no significant vulnerabilities exist for this function. A detailed discussion in Section 3.4.3 of this submittal provides discussion regarding the redundant means of decay heat removal which exists at WCGS. Several highly reliable systems and operator actions would have to fail in combination to have an impact on the decay heat removal capability, and the risk impact would be small. WCNOC believes it has met the closure requirements for USI A-45.

The containment performance improvement (CPI) program of Generic Letter 88-20, Supplement 3 was evaluated as part of the WCGS Level 2 PRA. As discussed in section 4.6 of this report, no containment vulnerabilities were identified from hydrogen detonation. WCNOC believes it has met the closure requirements for Supplement 3 of Generic Letter 88-20.

WCNOC chose to perform a Level 1 and 2 PRA as a means of meeting the IPE requirements and for the versatility and variety of uses the PRA provides after its development. The WCGS PRA models are PC-based; making use of integrated event tree and fault tree analysis codes. WCNOC has performed a majority of the work in-house and now has trained personnel capable of utilizing the PRA models.

The risk analysis group is currently working with several organizations within WCNOC to utilize the PRA and its results. WCNOC is currently working at integrating the PRA results into the WCGS response to the NRC Maintenance Rule while the WCNOC Training Department has integrated part of the event tree analysis information into the Operator Training Program. The PRA information was also utilized in initial development of the 1992 Emergency Plan Exercise. The Plant Trending and Evaluation and Quality Assurance organizations have received a ranking of equipment important to WCGS from a risk perspective, based on the PRA results.

7.2 CONCLUSION

WCNOC has met its goal for performing the IPE and determined that WCGS has no significant vulnerabilities. However, some potential areas of improvement have been identified and are being considered and evaluated for implementation. These changes, if implemented, would in all likelihood reduce an already acceptably low core damage frequency of 4.2E-05.

WCNOC intends to increase its efforts at promoting and integrating the PRA and its results into the day-to-day activities of the plant and WCGS personnel. The WCGS PRA is the culmination of the effort to respond to the Generic Letter 88-20 requirements. But the PRA is also part of WCNOC's efforts to utilize a risk based tool for evaluation of plant design and procedural changes and for guidance in addressing the severe accident management issue in the future.