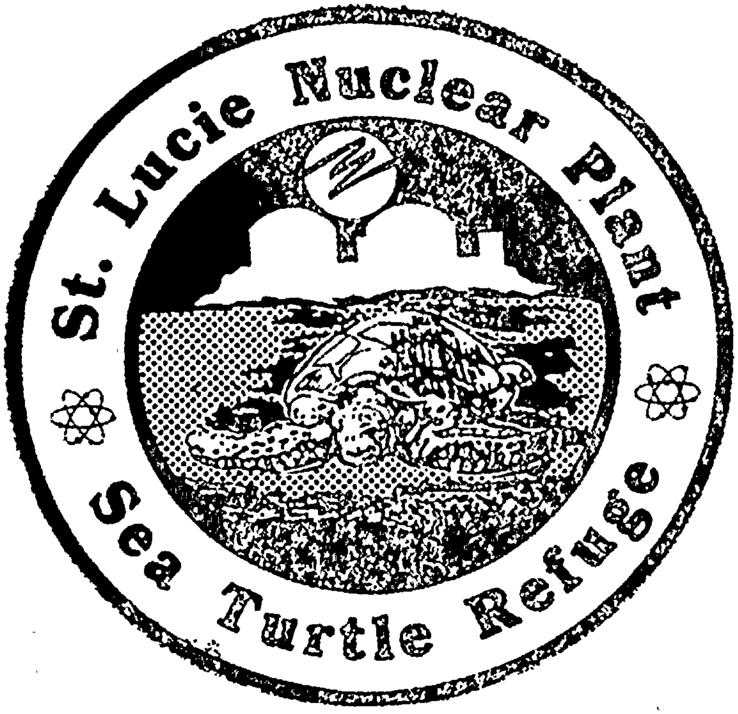
ST. LUCIE UNITS 1 & 2
INDIVIDUAL PLANT EXAMINATION
SUBMITTAL



FPL

DECEMBER, 1993

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ADV	Atmospheric Dump Valve
AFW	Auxiliary Feed Water
AOV	Air Operated Valve
ATWS	Anticipated Transient Without Scram
BOP	Balance Of Plant
CAFTA^{tm}	Computer Aided Fault Tree Analysis
CARP^{tm}	Computerized Analysis of Reliability Parameters
CCF	Common Cause Failure
CCI	Core-Concrete Interaction
CCS	Containment Cooling System
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDS	Core Damage Sequence
CE	ABB Combustion Engineering
CET	Containment Event Tree
CI	Containment Isolation
CIS	Containment Isolation Signal
CHRS	Containment Heat Removal System
CR	Control Room
CRDM	Control Rod Drive Mechanism
CS	Containment Spray
CSAS	Containment Spray Actuation Signal
CSR	Cable Spreading Room
CSS	Containment Spray System
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
CW	Circulating Water
DCH	Direct Containment Heating
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPS	Electrical Power System
EQ	Environmental Qualification
ERIN	Engineering and Research, Incorporated
ESD	Event Sequence Diagram
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FCV	Flow Control Valved
FPL	Florida Power & Light
FSAR	Final Safety Analysis Report
GSI	Generic Safety Issue
HFE	Human Failure Event
HPI	High Pressure Injection

TABLE OF ACRONYMS

HPME	High Pressure Melt Ejection
HPR	High Pressure Recirculation
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
IA	Instrument Air System
ICW	Intake Cooling Water
IE	Initiating Event
INPO	Institute for Nuclear Power Operations
IPE	Individual Plant Examination
IREP	Interim Reliability Examination Program
ISLOCA	Interfacing System Loss of Cooling Accident
ISEG	Independent Safety Evaluation Group
LC	Load Center
LCV	Level Control Valve
LER	Licensee Event Report
LOCA	Loss Of Coolant Accident
LOFW	Loss of Feed Water
LOG	Loss of Grid
LOOP	Loss Of Offsite Power
LPR	Low Pressure Recirculation
LPSI	Low Pressure Safety Injection
LWR	Light Water Reactor
MAAP^{tm}	Modular Accident Analysis Program
MCC	Motor Control Center
MFW	Main Feed Water
MOV	Motor Operated Valve
MSIS	Main Steam Isolation System
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
MTC	Maintenance, Test and Calibration
MW	Megawatts
NJPS	Nuclear Job Planning System
NO	Nuclear Operator
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NREP	Nuclear Reliability Examination Program
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
ONOP	Off-Normal Operating Procedure
OTC	Once-Through-Cooling
P&ID	Piping & Instrumentation Diagram
PC/M	Plant Change or Modification
PCS	Power Conversion System

TABLE OF ACRONYMS

PCV	Pressure Control Valve
PDF	Probability Density Function
PDS	Plant Damage State
PORV	Power Operated Relief Valve
PPC	Primary Pressure Control
PRA	Probabilistic Risk Assessment
PWO	Plant Work Order
PWR	Pressurized Water Reactor
PZR	Pressurizer
QA	Quality Assurance
RAB	Reactor Auxiliary Building
RCB	Reactor Containment Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMIEP	Risk Methodology Integration and Evaluation Program
RMQS[tm]	Risk Management Query System
RO	Reactor Operator
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRAG	Reliability and Risk Assessment Group
RT	Reactor Trip
RV	Reactor Vessel
RWT	Refueling Water Tank
SAIC	Science Applications International Corporation
SDC	Shutdown Cooling
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SHARP	Systematic Human Action Reliability Procedure
SI	Safety Injection
SIT	Safety Injection Tank
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
T&M	Test & Maintenance
TCW	Turbine Cooling Water
TO	Turbine Operator
TT	Turbine trip
USI	Unresolved Safety Issue
VCT	Volume Control Tank

1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

For several years, Florida Power and Light (FPL) has monitored developments in the area of Probabilistic Risk Assessment (PRA). FPL has also performed several probabilistic analyses to support management decisions concerning nuclear plant design, operation and maintenance. These analyses were typically directed at understanding contributors to safety system unavailability or the frequency of individual sequences of events.

After 1988, the Nuclear Energy Department management foresaw the need to develop more fully the technology of PRA within FPL. A team was chartered to develop a set of recommendations for the future of PRA at FPL. Subsequently, a course of action was adopted that would result in the development of full-scale, detailed PRAs for the Turkey Point and St. Lucie Nuclear Plants.

To accomplish these objectives, the Nuclear Engineering Department established a group accountable for the development, application, and maintenance of the PRAs. This PRA group solicited support from critical interfacing departments such as Nuclear Fuels and the plant Operations, Maintenance, Technical and Training Departments.

Because of its age and design, a PRA was performed first for the Turkey Point Plant. Following completion of Turkey Point's analysis, the St. Lucie PRA would then be developed.

As these decisions were being made, the NRC's Individual Plant Examination Program (IPEP) was also being shaped and defined. As a minimum, the scope of the FPL PRAs must encompass that of the IPEP. Based on the Generic Letter 88-20 content [Ref. 1.0-1], FPL determined that the Turkey Point and St. Lucie PRAs should include a Level 1 PRA for internal initiating events, a limited scope Level 2 Containment Performance Analysis, and an assessment of the risk due to internal flooding.

By July of 1989, FPL let a contract to Science Applications International Corporation (SAIC) for assistance in development of the Turkey Point PRA. Since this effort was "new technology" for FPL, SAIC would provide project management service and technology transfer. FPL engineers would perform at least 50% of the work, thus supporting the objective of bringing the PRA technology in-house. In practice, FPL performed well over half the work on the Turkey Point analysis and gained valuable experience in almost all aspects of PRA technology. Based on the experience and technology gained during development of the Turkey Point PRA, FPL developed the St. Lucie PRA with minimal contractor support.

This report documents the work performed to estimate a core damage frequency (CDF) for St. Lucie Units 1 and 2 and to satisfy the provisions of Generic Letter 88-20.

1.2 Plant Familiarization

St. Lucie Plant Units 1 and 2 are located on Hutchinson Island in St. Lucie County about halfway between the cities of Fort Pierce and Stuart on the East Coast of Florida. Each unit is a pressurized water-type reactor (PWR) with a nuclear steam supply system (NSSS) designed by Combustion Engineering, Inc. (CE) and rated for a full power core thermal output of 2700 megawatts. Unit 1 began commercial operation in 12/76 and Unit 2 in 8/83.

The Reactor Coolant System (RCS) of each unit is arranged as two closed loops connected in parallel to the reactor vessel. Each loop has one outlet (hot leg) pipe, one steam generator, two inlet (cold leg) pipes and two reactor coolant pumps. An electrically heated pressurizer is connected to the hot leg of one loop and a safety injection line is connected to each of the four cold legs. The RCS operates at a nominal pressure of 2235 psig.

The reactor buildings are dual containment design comprised of a steel containment vessel surrounded by an annular space and enclosed by a reinforced concrete shield building. The containment vessel steel shell is designed to confine the radioactive material that could be released from a postulated design basis Loss-of-Coolant Accident (LOCA). The shield building is a concrete structure that surrounds the annulus and steel containment vessel. It protects the containment vessel from external missiles and provides biological shielding and a means of collecting radioactive products that may leak from the containment following a major hypothetical accident.

Engineered Safety Features (ESF) systems with the containment ensure that the off-site radiological consequences following any LOCA do not exceed the regulations. The ESF include: (a) independent redundant systems (Containment Cooling System (CCS) and Containment Spray System (CSS)) to remove heat from and reduce the pressure in the containment vessel, (b) a high and low pressure Safety Injection System (SIS), (c) a Shield Building Ventilation System and an Iodine Removal System, (d) a Containment Isolation System, (e) a hydrogen control system, and (f) a control room habitability system.

Feedwater to the steam generators is provided by two motor driven main feedwater pumps per unit. Each unit also has an Auxiliary Feedwater System (AFW) consisting of two motor driven pumps and one pump driven by a steam turbine. This system provides a source of water inventory to the steam generators during plant startup, hot standby, and during plant cooldown, and provides heat removal to bring the Reactor Coolant System to the shutdown cooling system activation window. One condensate storage tank per unit provides a large volume of water to support operation of the AFW system.

Off-site power from the utility grid comes from the switchyard via two startup transformers per unit. During normal operation, each unit receives power from the main generator through two unit auxiliary transformers. When necessary, on-site AC power is provided by two independent emergency diesel generators per unit.

Equipment heat loads are removed by a closed Component Cooling Water (CCW) System, which rejects heat to the Intake Cooling Water (ICW) System.

1.3 Overall Methodology

The St. Lucie PRA was developed to satisfy the provisions of the Individual Plant Examination (IPE) process; that is to perform a "systematic examination to identify any plant-specific vulnerabilities to severe accidents . . . ." The IPE has several goals, including the development of an appreciation for severe accident behavior, to understand the most likely severe accidents for St. Lucie, to gain a "more quantitative" understanding of core damage probabilities and potential fission product releases, and finally to reduce these probabilities by appropriate plant changes where required. The St. Lucie Units 1 and 2 PRA scope and process were designed specifically to meet these goals.

1.3.1 Internal Events Methodology

Standard event tree/fault tree methods were employed to understand the most probable core damage states for the plant. The St. Lucie analysis used the small event tree/large fault tree philosophy. Functional event trees were developed for each class of unique initiating events identified; top logic was then developed to link the statement of functional failure to that of system failure criteria.

Detailed fault trees were developed for both Units 1 and 2 for each front-line system identified in the top logic. Also, these front-line systems' support systems had fault trees developed. To ensure the traceability of the supporting data, detailed system description notebooks were created to document the analytical effort.

The Unit 1 and Unit 2 fault tree basic events were then quantified with a mixture of generic and St. Lucie plant specific data. The scope of the plant specific data analysis included initiating event frequencies and plant specific failure data for component types requested by Generic Letter 88-20. The project established a six-year data window as the basis for quantifying failure rates and maintenance unavailability.

Human failure events were also quantified. Methods compatible with those outlined in the Systematic Human Action Reliability Procedure [Ref. 1.0-2] were employed to develop conservative screening values for human events; more detailed analysis was used for important recovery events.

The SAIC enhanced version of the EPRI-developed CAFTA code was used to integrate the event trees and fault trees into a plant model. Model development, integration and quantification was performed on personal computers.

St. Lucie plant personnel involvement was a key factor in the project. The individual system analysts performed walkdowns, as required, to verify the completeness and correctness of their models. Operations, Maintenance, Technical, ISEG, and Training department personnel were consulted throughout the analysis. Operations and Training Department personnel were particularly instrumental in the identification and quantification of operator recovery events.

St. Lucie is an open plant (i.e., no enclosed turbine building). For the internal flooding events, it was recognized that most sources of "floods" would simply run-off across the plant area to either the intake or discharge canals. Fire zones were chosen as the unit of examination. For each zone, screening questions were employed (Does the water source trip the plant?, is there PRA equipment in the zone?, does the PRA equipment become damaged by the water source? (either immersion or spray)). For the areas that did not pass the screening analysis, the contribution to core damage frequency was determined by "failing" the zone's PRA related equipment and analyzing the CAFTA-based plant model.

1.3.2 Containment Performance Methodology

A simplified, limited-scope approach was taken for this portion of the analysis. To make the transition from core damage states identified by the internal events analysis to plant damage states, a containment systems status "bridge" tree was constructed and appended to the binned core damage sequences. This bridge tree assesses the unavailability of containment isolation systems and containment sprays/emergency containment cooling systems and helps categorize the various core damage states into plant damage states. The containment event tree then provides insights into the phenomenological factors affecting the core melt and subsequent containment failure and release modes. The EPRI-developed MAAP Code was used to gain St. Lucie specific knowledge about the progression of the accident from melt to release. Containment failure modes and release categories are the outcome of this portion of the overall effort.

1.4 Summary of Major Findings

FPL has performed a Level 1 and limited scope Level 2 PRA for St. Lucie Units 1 & 2 in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities". The objectives for this assessment are consistent with the objectives given in the generic letter. FPL personnel have been directly involved in all aspects of the development, quantification, and documentation of the PRA models. The approach included system, procedure, and drawing reviews, discussions with Operations, Training, Technical Staff, and other plant personnel, and independent peer reviews by PRA experts to ensure that the models are consistent with accepted PRA practices.

As a result, the IPE provides a comprehensive and detailed analysis of the severe accident behavior of St. Lucie Units 1 & 2. The overall likelihood of core damage and fission product release from the containment from internally initiated events has been quantified consistent with the guidance provided in Generic Letter 88-20. The relative contribution to core damage frequency from the different accident sequence types has been determined.

The major findings are presented here in two components: findings from the Internal Events analysis, and the Containment Performance analysis.

1.4.1 Internal Events (Level 1) Findings

- The overall core damage frequency due to internally initiated events for St. Lucie 1 is 2.3×10^{-5}/yr and for St. Lucie Unit 2 is 2.6×10^{-5}/yr. This is much less than the NRC safety goal of 1×10^{-4}/yr and illustrates a high level of safety.

- The overall core damage frequency for St. Lucie Units 1 & 2 is within the range of past PRAs performed for PWRs. Thus, the susceptibility to core damage at St. Lucie Units 1 & 2 is not unlike other PWRs.

- A chart of the dominant accident sequences is shown in Figure 1.4-1. It shows that the largest contributor to core damage risk is small-small (1/2" - 3") LOCAs. Total loss of feedwater events are also important accident sequences for core damage risk. Section 3.7 presents the Level 1 results in more detail.

- St. Lucie has several means of providing feedwater to the steam generators for decay heat removal. No vulnerability related to USI A-45, Decay Heat Removal, has been identified.

1.4.2 Containment Performance (Level 2) Findings

- The St. Lucie Units 1 & 2 large dry containment design provides adequate capability to mitigate severe accidents. No unusually poor containment performance has been found. A chart of the containment analysis results is shown in Figure 1.4-2.

- The greatest threat to containment integrity is due to a loss of all containment heat removal during an accident where the RCS is at high pressure. Steam generation without the ability to remove heat and condense steam increases the likelihood that high pressure melt ejection at vessel breach can fail the containment.

- A key feature of the St. Lucie containment design is that for almost all accident sequences, the reactor cavity is flooded with water. This decreases the likelihood of reactor vessel failure due to ex-vessel cooling and results in lower releases (due to retention of fission products in the RCS and scrubbing of ex-vessel fission products by the water) compared to if the vessel were to fail and the core were to fall on a dry cavity floor.

- The open design of the St. Lucie containment means that local hydrogen accumulation (identified in Generic Letter 88-20, Supplement 3, containment performance improvement issues) is not a significant contribution to containment failure.

The St. Lucie Unit 1 and Unit 2 PRA has been performed in a manner consistent with the objectives stated in Generic Letter 88-20 and the results found that there are no plant unique severe accident vulnerabilities.

1.5 Report Organization

Section 2 of NUREG-1335, "Individual Plant Examination: Submittal Guidance" provided a standard Table of Contents for submittals in response to Generic Letter 88-20. This report adheres to the standard format as far as practical. The following provides a brief guide to this report's organization:

SECTION 1.0 - Executive Summary - Overview of the project, its scope and results.

SECTION 2.0 - Examination Description - Details on what methods were applied to perform the various components of the analysis, discussion on how the intent of Generic Letter 88-20 was met by the analysis.

SECTION 3.0 - Core Damage Analysis: "Front-End Analysis" - Details on the Internal Events analysis leading up to the core damage condition (includes the Accident Sequences, Systems Analysis, Internal Flooding Analysis, Reliability data, Human Reliability Analysis, Quantified Core Damage Sequences, results of the "Front-End" work performed, and proposed resolution of any USIs and GSIs addressed by the St. Lucie PRA).

SECTION 4.0 - Containment Performance Analysis: "Back-End Analysis" - Details on the features of the St. Lucie containment structures, core and plant damage state binning, containment systems "bridge" tree, Containment Event Tree, quantification of containment failure modes and radionuclide release characterization.

SECTION 5.0 - Utility Participation and Internal Reviews - Project organization, project reviews, major comments and their resolution.

SECTION 6.0 - Plant Improvements and Unique Safety Features - Discussion of how potential vulnerabilities were analyzed and any countermeasures identified.

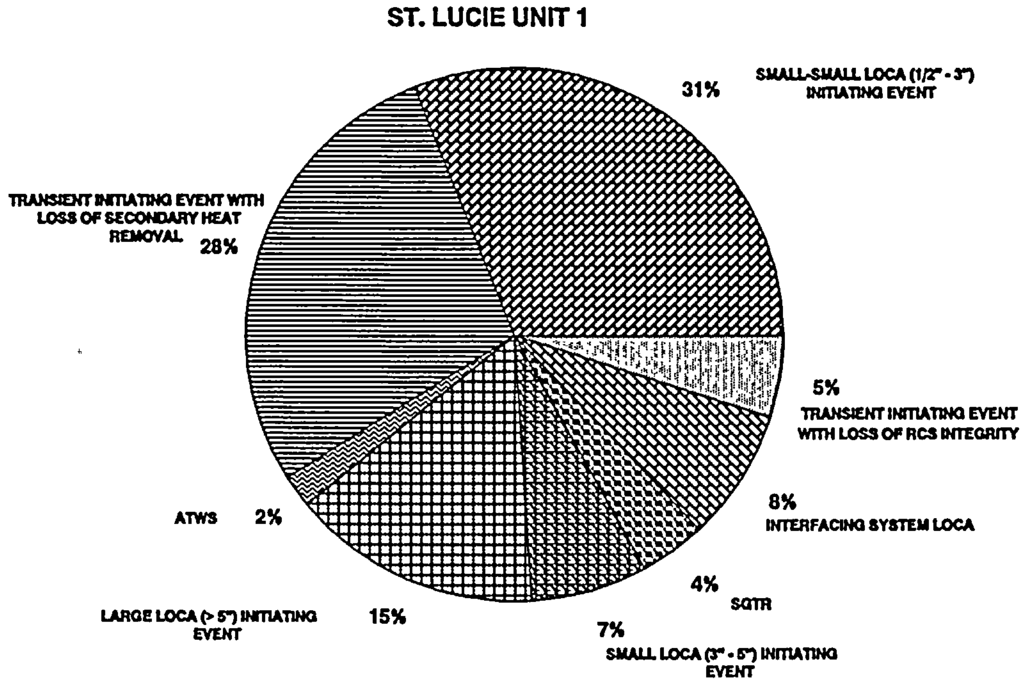
SECTION 7.0 - Summary of Results and Conclusions.

1.6 Section 1.0 References

1.0-1 NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, November 23, 1988.

1.0-2 EPRI-NP-3583, Systematic Human Action Reliability Procedure (SHARP), 1984.

Figure 1.4-1 Summary of St. Lucie Unit 1 and Unit 2 Level 1 Results



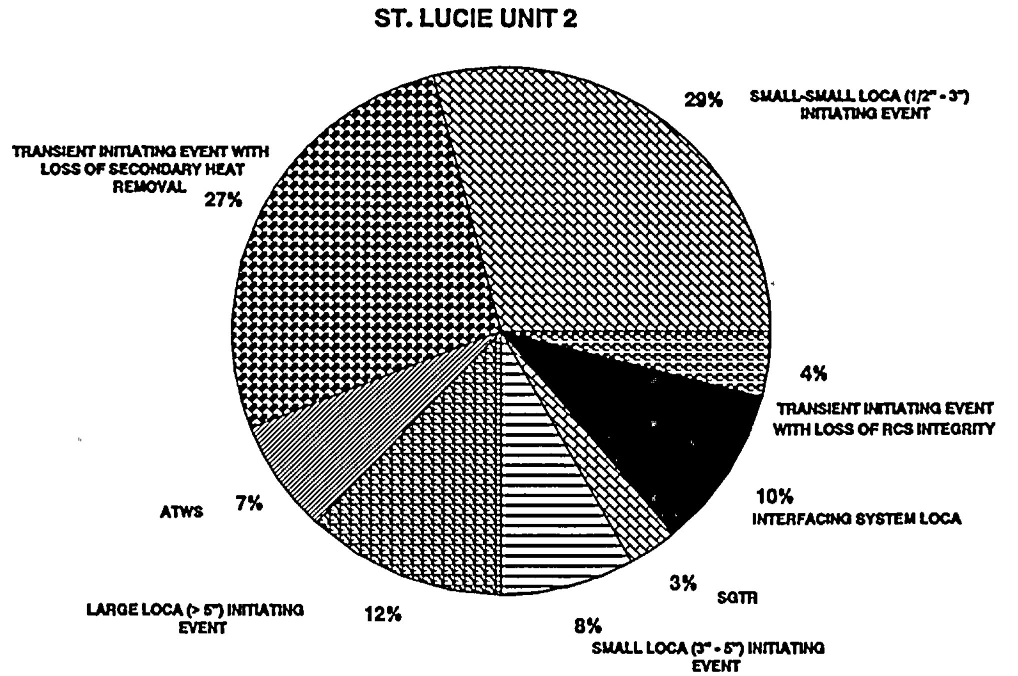
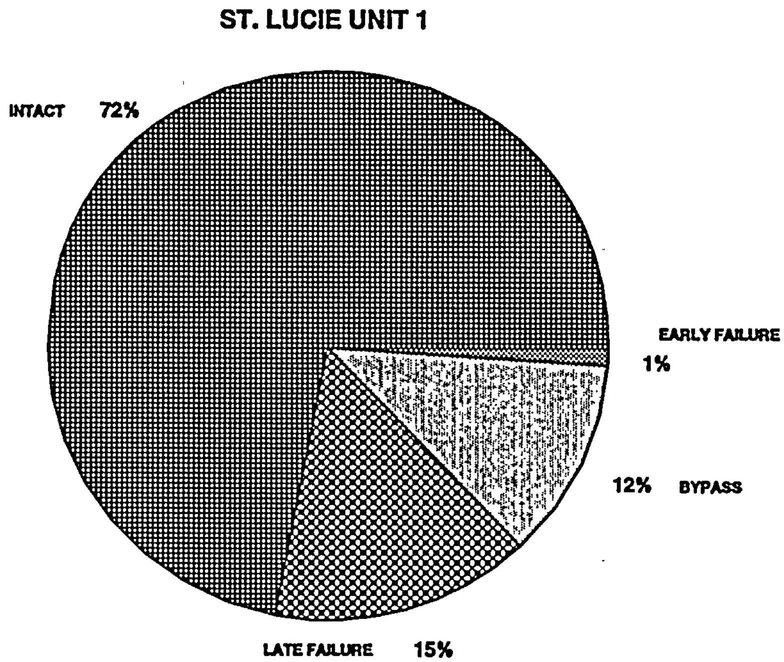
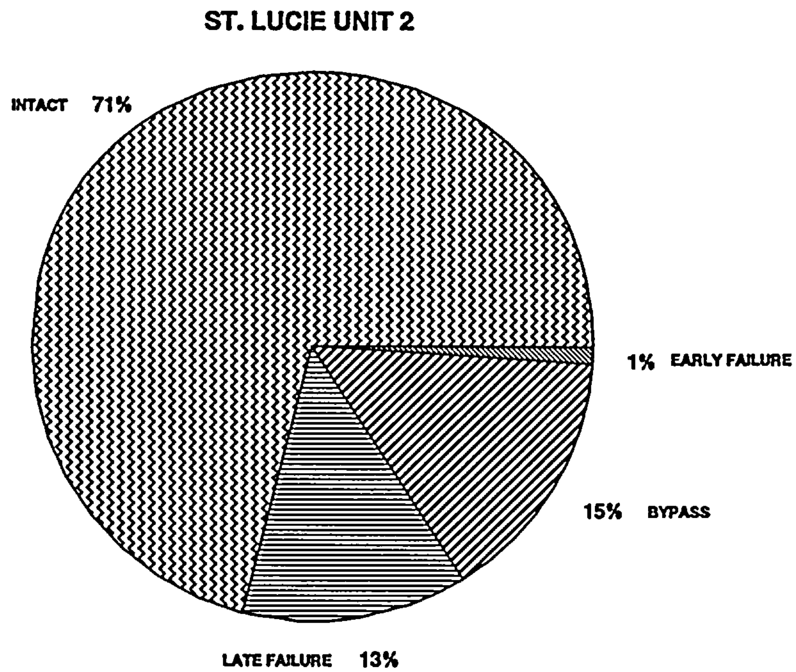


Figure 1.4-2 Summary of St. Lucie Unit 1 and Unit 2 Level 2 Results





2.0 EXAMINATION DESCRIPTION

2.1 Introduction

To satisfactorily comply with Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," FPL chose Probabilistic Risk Assessment (PRA) as the technical approach for the St. Lucie analysis. Integrated with this objective, FPL developed the PRA so that it can be used routinely by trained company employees. To accomplish this, a group of engineers was established in the Nuclear Engineering Department. This group performed more than 50% of the Turkey Point PRA, which was submitted in June of 1991 [Ref. 2.0-1]. The NRC staff evaluation of the Turkey Point IPE [Ref. 2.0-2] concluded that the evaluation was complete with the level of detail consistent with the information requested by NUREG-1335. Approximately 50% of the FPL engineers who worked on development of the St. Lucie PRA were task leaders for various portions of the Turkey Point analysis. This experience was applied to the St. Lucie PRA, where the project was totally managed by FPL personnel with minimal use of outside contractors.

The project scope can be generally described as:

- A Level 1 PRA and Limited Scope Level 2 PRA, including consideration of internal floods. The Level 1 PRA seeks to identify and quantify the combinations of accident initiators and plant response failures that can lead to core damage. The Level 2 PRA seeks to identify and quantify the combinations of additional containment response failures and phenomena that could yield a significant radioactive release. The "limited scope" qualifier refers to the use of simplified event trees and logic trees to model the accident progression. The MAAP code was employed by the project to gain certain specific insights into the behavior of the St. Lucie Plant containment behavior during the accident.

- PRA workstations consisting of Personal Computers and PRA software to contain the PRA models and software tools supporting calculations and decision-making.

2.2 Conformance with Generic Letter and Supporting Material

2.2.1 General Conformance

Three primary documents describe the Nuclear Regulatory Commission's request to perform an Individual Plant Examination for Severe Accident Vulnerabilities: Generic Letter GL-88-20 [Ref. 2.0-3], which describes the examination's purpose and process, Generic Letter GL-88-20, Supplement 1 [Ref. 2.0-4], which initiated the examination process and its accompanying NUREG-1335, "Individual Plant Examination: Submittal Guidance" [Ref. 2.0-5], which delineates the guidance for reporting the results of the plant examination.

On October 31, 1989, FPL issued a letter [Ref. 2.0-6] outlining the proposed FPL IPE Program Plan for Turkey Point Units 3 and 4 and St. Lucie Units 1 and 2. FPL proposed to perform a Level 1 Probabilistic Risk Assessment (PRA), a containment performance analysis following the guidance of Appendix 1 to GL-88-20, and to include internal flooding.

The NRC responded to the FPL proposal in February, 1990 [Ref. 2.0-7]. In the NRC response, they concluded that the proposed approach, methodology and schedule were acceptable.

2.2.2 Specific Conformance

FPL reviewed Generic Letter 88-20 and extracted the important issues relating to specific conformance with the provisions of the letter. The following references the report section detailing this conformance or summarizes FPL's conformance with these important provisions:

1. Licensee Staff Involvement - See Section 5.

2. Approach to Satisfy the Examination - Level 1 PRA, including Internal Flooding, plus a Limited Scope Level 2 Containment Performance Analysis.

3. Resolution of USI A-45, Decay Heat Removal - See Section 3.7.

4. Resolution of Other USI/GSI - See Section 3.7.

5. Reporting of Potentially Important Functional Sequences - See Section 3.7.

6. Correction of any Identified Vulnerabilities - See Section 6.

7. Documentation Requirements - NUREG-1335, Individual Plant Examination: Submittal Guidance has been used to establish both the format and content of this report.

8. Containment System Performance Examination - See Section 4.

2.3 General Methodology

2.3.1 Overview

To clearly organize and specify the work to be accomplished for the St. Lucie PRA, a comprehensive task breakdown was developed. Eight (8) major tasks were defined. An overview of each is provided below. Project specific procedures were developed for each of these key technical tasks.

1) Accident Sequence Analysis - Identification of potential accident initiators and development of related accident sequence models leading to core damage. The output is used both to understand the progression of the accident to the core damage state, and also as input to the containment analysis.

2) Systems Analysis - Development of plant, containment system, and isolation models, including recovery actions, incorporating component failure, maintenance and test unavailability, human reliability actions and system-specific accident initiators.

3) Data Analysis - Development of generic and plant specific component reliability data, maintenance and/or test unavailability data and initiating event frequencies.

4) Human Reliability/Recovery Analysis - Development of screening data for human action events and refined data using appropriate human error modeling techniques for dominant actions and human recovery actions. Identification of accident recovery scenarios/actions and quantification of non-recovery probabilities for incorporation into the integrated plant model.

5) Quantification/Integration - Integration and quantification of system fault trees and accident sequences and containment event trees to obtain risk-related results and measures.

6) Containment Performance Analysis - Development of the containment event tree and logic models to analyze the response of the containment to core damage scenarios and related containment interaction phenomena, identification of potential source terms and potential release categories for combinations of plant and containment states.

7) Internal Flood Analysis - Assessment of risk due to internal flooding by identifying flood vulnerability, flood scenarios and risk quantification.

8) Sensitivity/Uncertainty/Modifications Evaluation - Performance of uncertainty analysis and analysis of sensitivity of core damage and containment failure states to key assumptions, identify and evaluate conceptual plant changes to improve risk.

2.3.2 Technical Approach

The approach for the St. Lucie PRA was developed to address the project objectives in an integrated fashion. The technical steps are essentially the same as those established to analyze Turkey Point and thus comply with the Individual Plant Examination (IPE) provisions of GL-88-20 while also developing St. Lucie plant specific computer models and tools that can be the basis for future risk management and operational decision-making. The general technical approach by which this was accomplished is summarized below.

The technical approach for the Level 1 and limited-scope Level 2 PRA and the origin of the technical task breakdown can be characterized as a "Small Event Tree, Large Fault Tree" approach. To begin the Level 1 analysis, potential accident initiators were identified and grouped according to similar St. Lucie Plant response (e.g. transients, LOCAs). "Small" functional event trees were then developed to identify accident sequence scenarios for each group of initiators. These sequences define the plant responses and the resulting core damage bins. For each plant response function node in the event tree, "Top Logic" fault trees were developed to convert the functional response to identification of specific St. Lucie plant system failures and human interactions. For each system response, "Large" component-level fault trees were developed including front-line and support system relationships plus related human interactions. System-related features that could provide additional accident initiators were fed back into the accident sequences. The system fault trees also include test and/or maintenance activities that can contribute to system unavailability.

Component failure and unavailability data, as well as initiating event frequencies, were constructed from a combination of generic and St. Lucie Plant specific failure information. Common Cause failures were modeled to the extent where published industry data existed. Human reliability data was similarly incorporated with the use of screening values and specific, detailed modeling of important human actions. To develop the Level 1 core damage frequencies, a fault tree linking approach was used to combine the system level fault trees and top logic for the specific accident sequences. This process led to the quantification and identification of accident cutsets, that is, the combinations of initiating event, component failures, component unavailabilities, common cause failures and human errors that can yield the accident sequence and related core-damage end states.

To extend the analysis to a Level 2 PRA; which incorporates the influence of severe accident phenomena and containment performance given a core damage sequence, interfaces with containment systems, containment phenomena and potential for radioactivity release were included. From the systems perspective, this interface is accomplished by fault tree modeling of containment systems response for the containment states related to the core damage condition. Potential containment isolation failures or bypass were also modeled. From the accident sequence perspective, the interface was included by extending the end state of the core damage sequence to a plant damage state through modeling of the containment systems and containment boundary conditions that link the two states (containment systems "bridge tree"). Phenomenological effects (e.g., core-concrete interaction, wet cavity effects) were then incorporated by development of a containment event tree which characterized the sequence of events which could lead to a radioactivity release. For a limited-scope Level 2 analysis, such as was performed here, the containment performance analysis was limited to the most significant accident sequences; only relevant combinations of plant damage states, containment system states, and containment failure modes were addressed. Both reference plant analyses and the MAAP code (including specific St. Lucie containment features) were used for estimating phenomenological effects.

The influence of internal flooding was addressed using the Level 1 plant damage model. In general, a "hazard" analysis was performed identifying the means through which the internal flooding event could cause an accident initiator or fail a system's response to an initiator. This analysis involved assessing the spatial interactions effects and determination of the vulnerability of the identified Level 1 basic events to the flooding hazard. The flood/spray sources were identified, and a "truth-table" screening approach followed to identify flood sources for which core damage sequence frequencies were to be calculated. From this point, damage scenarios were defined and solved in much the same manner as the baseline Level 1 analysis.

In the Level 1 and internal flooding analyses, consideration was also given to recovery actions (i.e., the likelihood that alternative steps may be taken through operator action(s) to circumvent continued progression of an accident). The probability that such actions may fail was also considered. Recovery was addressed considering the existing failures, plant conditions, and time involved, plus the information available to the operator. This was accomplished through review of cutsets, identification of recovery steps and quantification of non-recovery probabilities. Discussions with St. Lucie Plant operations and training personnel and walkdowns of both in- and ex-control room actions provided the basis for this portion of the analysis. The individual cutsets, identified through the initial quantification effort, were thus expanded to include additional events which must also fail to allow the accident sequence to continue. The addition of these events and the related analyses added realism and additional plant-specific details to the models, which were then requantified. Level 2 modeling included not only recovery actions that were carried forward from the Level 1 analysis, but also some considerations of actions beyond the Level 1 effort.

At essentially each step of the technical approach, the FPL analysts utilized PC-based computer workstations with the CAFTA software to build models (i.e. event trees and fault trees) and databases (e.g., component reliability data). These models and databases were linked together for solution and manipulated (i.e., through Sensitivity Task activities) for risk management insights.

2.3.3 Vulnerability Identification and Treatment

Section 8 of GL-88-20 describes the NRC's expectation that the licensee would move "expeditiously to correct any identified vulnerabilities that it determines warrant correction." Further, the NRC states that it will act to require plant change should the IPE identify instances where the NRC regulations are not met by the plant design or should an analysis pursuant to 10CFR50.109 reveal a significant benefit be gained (considering the cost) by a plant change.

Based on these statements of intent, FPL developed the following criteria for vulnerability identification and treatment: 1) If the PRA development effort identified a plant feature that was outside the current St. Lucie design/operating basis, that feature would be reported and FPL would immediately commence efforts to correct the feature, 2) If the PRA development effort identified a plant feature that contributed to a significant fraction of the core damage frequency, strategies to correct the feature would be identified and reported herein along with FPL's schedule for their completion. Section 3.7.2 discusses vulnerability screening.

2.4 Information Assembly

2.4.1 General Documentation Assembly

At the start of the project, a list of specific information needed to begin the PRA was established. This list was developed based on experience gained in development of the Turkey Point PRA. This information was assembled by the various analysts as required. A cutoff date of 11/91 was used for plant changes so that the models could be frozen and quantified.

A list of the information utilized includes:

FSAR System Description - system design and operation

Piping and Instrumentation Drawings - system design and interfaces

Electrical One-line Drawings - electric power requirements and interfaces

Licensee Event Reports - initiating events and unique failure mechanisms

Monthly Operating Reports - initiating events and operating history

Technical Specifications - operating limits and surveillance frequencies

Emergency and Off-Normal Operating Procedures - operator response to accident conditions

Special Studies and Analyses - system response during accident conditions to establish success criteria

The St. Lucie PRA model is accurate and represents the as-built, as-operated plant based on the following:

- complete documentation of the model with adequate control and review of any changes made
- access to and use of controlled plant drawings and Emergency Operating procedures
- reviews by and interactions with Operations, Engineering, and other plant personnel
- review of the system models by plant personnel and outside contractor experts
- plant walkdowns performed to gain an understanding of the spatial relationships of equipment
- containment walkdowns performed by key Level 2 analysts

2.4.2 Other PRA Insights

Past PRA studies reviewed as part of the development effort for the St. Lucie PRA include the following:

1. NUREG/CR-4374, "A Review of the Oconee-3 Probabilistic Risk Assessment - Internal Events, Core Damage Frequency"
2. NUREG/CR-5245, "A Review of the Crystal River Unit 3 Probabilistic Risk Assessment: Internal Events, Core Damage Frequency"
3. NUREG/CR-4552, "A Review of the Seabrook Station Probabilistic Safety Assessment - Containment Failure Modes and Radiological Source Terms"
4. NUREG/CR-4142, "A Review of the Millstone 3 Probabilistic Safety Study."
5. NUREG/CR-4589, "Review of Selected Areas of Yankee Rowe Probabilistic Safety Study."
6. NUREG/CR-2515, "Crystal River - 3 Safety Study"

7. NSAC-60, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3"

8. Northeast Utilities Co. "Millstone Unit 3 Probabilistic Safety Study"

9. PLG-0300, "Seabrook Station Probabilistic Safety Assessment"

10. WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants"

The principal benefit obtained from reviewing the actual PRAs is to gain a perspective on the issues addressed by these studies. Of all the PRA tasks, the Accident Sequence Analysis is the one where the insights from other PRAs are most beneficially applied. Within this task, the identification and grouping of Initiating Events and LOCA sizes/categories were aided the most. Finally, information and insights were gained from review of the Waterford 3, San Onofre Units 2 and 3, and ANO-2 IPE submittals.

2.4.3 PRA Plant Walkdowns

Several different types of walkdowns were conducted as an integral part of the St. Lucie PRA development. Although the analysis was conducted primarily offsite, the engineers performing the work became very familiar with the actual plant and system layouts. Support from the St. Lucie Plant Operations, Maintenance and Technical Departments was readily available to assist the PRA team in their understanding of plant, system and equipment operation, as required.

System Level -	The system fault tree analysts conducted walkdowns of their respective systems. St. Lucie System Engineers assisted the PRA team in understanding equipment locations, and system operations, tests and maintenance.
Containment -	The Unit 2 refueling outage of 1992 provided an excellent opportunity to walkdown one of the St. Lucie containments (they are essentially identical). The PRA team members assigned to the Containment Performance Analysis task took advantage of this opportunity to assess various elevations and compartments of the containment. Potential containment bypass methods were also examined by walkdowns of the auxiliary building during this time.

Recovery Actions - Several accident mitigation and recovery actions identified by the PRA analysis occur via ex-control room equipment manipulations. After studying the plant procedures directing these actions and discussing the actions with plant operators, the PRA analyst walked down many of these actions. Understanding of both the relative simplicity and the timing of these actions was the goal of this type of walkdown. This is an especially important type of walkdown since it is a prime input to the assessment of the human reliability for the specified action.

Internal Flooding Analysis - Walkdowns were performed to gain an understanding of the spatial relationships of equipment to the various flooding hazards.

2.5 Section 2.0 References

2.0-1 W. H. Bohlke to NRC, Turkey Point Units 3 and 4 Individual Plant Examination for Severe Accident Vulnerabilities (GL 88-20), L-91-184, dated June 25, 1991.

2.0-2 L. Raghavan (NRC) to J. H. Goldberg, Individual Plant Examination (IPE) for Severe Accident Vulnerabilities, Generic Letter 88-20 - Turkey Point Units 3 and 4 (TAC Nos. M74482 and M74483), dated October 15, 1992.

2.0-3 NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, November 23, 1988.

2.0-4 NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, Supplement 1, August 29, 1989.

2.0-5 NUREG-1335, Individual Plant Examination: Submittal Guidance, August 1989.

2.0-6 J. H. Goldberg to NRC, Turkey Point Units 3 and 4, St. Lucie Units 1 and 2, Individual Plant Examination for Severe Accident Vulnerabilities, L-89-389, dated October 31, 1989.

2.0-7 G. E. Edison (NRC) to J. H. Goldberg, Turkey Point 3 and 4 - St. Lucie 1 and 2 - Receipt of 60 Day Response to Generic Letter 88-20, Individual Plant Examinations, dated November 22, 1989.

3.0 CORE DAMAGE ANALYSIS: "FRONT-END ANALYSIS"

3.0.1 Background

This section documents the core damage risk assessment conducted for the St. Lucie Units 1 and 2 PRA. The St. Lucie core damage analysis methods are consistent with the PRA Procedures Guide (NUREG/CR-2300), previous PRAs (such as Oconee PRA (NSAC-60), Seabrook PRA, and Turkey Point) and NUREG-1150 core damage analyses.

The scope of the core damage risk assessment includes a full treatment of internal transient and accident initiating events, such as various categories of Reactor Trips, LOCAs, and ATWS. Internal Flooding was also analyzed for impact on potential core damage sequences.

3.0.2 Section Organization

The organization of this report follows closely that requested by NUREG-1335, "Individual Plant Examination: Submittal Guidance." The accident sequences are first defined, with the discussion focused on the initiating event selection and grouping, followed by the selection of accident sequence functions and event tree development. The St. Lucie systems necessary to perform the functions are identified and described, along with their success criteria. The data necessary to quantify the plant model, including initiating event frequency, equipment failure probabilities and human reliability parameters are then summarized. The accident sequence quantification methods for internal events are presented. Prior to the overall results discussion, the Internal Flooding analysis findings are presented. The composite "front-end", results are then discussed; with a comparison made to the previous Sandia analysis of St. Lucie's Decay Heat Removal capabilities.

These and other topics are supported by the discussions in the following sections:

- Accident Sequence Delineation (Section 3.1)
- Systems Analysis (Section 3.2)
- Reliability Data Analysis (Section 3.3)
- Human Reliability/Recovery Analysis (Section 3.4)
- Core Damage Sequence and Plant Damage Sequence Quantification (Section 3.5)
- Internal Flood Analysis (Section 3.6)
- Front-End Results and Screening (Section 3.7)

3.1 Accident Sequence Analysis

3.1.1 Initiating Event Analysis

3.1.1.1 Introduction

PRA is the process of understanding which are the most important accident sequences that could lead to a damaged core and, further, to a release of radioactive material from the containment. Accident sequences begin with an initiating event. An initiating event is a component failure or human error which causes a demand for a reactor trip. In almost every case, the reactor will trip and other systems will perform safety functions to bring the plant to hot or cold shutdown.

If the reactor does not trip or if other safety systems fail to perform their safety functions, the core may heat up and if unmitigated will be damaged. Such a scenario, called a core-damage accident sequence, is unlikely because of the redundancy in nuclear power plant safety systems. Consequently, initiating events which demand a reactor trip and fail or degrade safety systems will be important features of the most likely core damage accident sequences.

The principal output of the Initiating Events Analysis is a list of the most important initiating events. The process of identifying initiating events includes three basic steps:

1. The first step is to identify in as practical a manner as possible a "complete" list of the component or human failures which can cause a reactor trip. Because most reactor trips have little or no impact on safety systems, it is unnecessary to identify every cause in order to identify the risk significant accidents. Therefore, this step generally involves collecting initiating event lists from sources of *risk significant* initiators. These sources include other PRAs as well as their NRC reviews, plant safety analyses of both limiting and likely events (e.g., FSAR), and St. Lucie reactor trips.

2. The second step is to organize the list into groups with equivalent impact on the plant. Equivalent impact implies that the plant response is sufficiently similar in all important aspects (i.e., which plant mitigating systems should respond and if they are degraded). For example, initiators which cause a PORV to open are grouped because they may cause a LOCA.

3. After completion of these two steps, additional initiators are identified and/or modified as part of the fault tree analysis of plant systems and the success criteria analysis for the event trees. The initiators developed from the systems analysis, commonly called system initiators, are included on a case-by-case basis depending on their potential risk impact. During this third and final step, additional analyses are performed to identify risk-significant dual unit initiators.

Section 3.1.1.2 describes St. Lucie design features which impact initiating event grouping. Section 3.1.1.3 describes the categories of St. Lucie initiating event groups and their impact on plant response. Section 3.1.1.4 describes the findings of NRC PRAs and NRC reviews of PRAs and the potential implication of "NRC staff positions" on the St. Lucie PRA. The following discussion is applicable to both St. Lucie Unit 1 and Unit 2 except where otherwise noted.

3.1.1.2 St. Lucie Specific Design Features Affecting Initiating Event Selection

This section discusses some key St. Lucie plant design features which have been identified because of their potential impact on initiating event selection and grouping.

Certain St. Lucie plant design features affect the potential risk significance of initiating events and the corresponding search and evaluation process. The following description of those features follows the process by which they are identified.

Subsection	Focus
3.1.1.2.1	Systems designed to produce thermal and electric power.
3.1.1.2.2	Systems shared by both Units 1 and 2 which have the potential to generate a dual unit trip.
3.1.1.2.3	Supporting systems, such as Component Cooling Water, which have the potential to generate a unit trip and affect the operability of plant safety systems.

3.1.1.2.1 Power Production Systems

Past PRAs have generally divided power production system initiators into three categories:

- reactor trip
- loss of main feedwater
- loss of offsite power

An evaluation of St. Lucie power production systems indicates a more detailed categorization is necessary. This categorization is needed to properly account for both realistic operator recovery actions in St. Lucie Emergency Operating Procedures as well as for plant features which may result in accidents uniquely important at St. Lucie. Two design features are addressed below:

Main Feedwater

Offsite Power

Main Feedwater. Unlike Westinghouse plants, main feedwater (MFW) at St. Lucie will not isolate after a reactor trip (although AFAS would isolate MFW). A general reactor trip will be significantly different in character from a loss of Main Feedwater caused reactor trip because the general reactor trip does not make main feedwater unavailable.

Because Main Feedwater or Auxiliary Feedwater may fail to operate, St. Lucie EOPs identify a number of means to recover heat sink [Ref. 3.1-1, 3.1-2, 3.1-3]. These means include recovering Condensate (if steam generator pressure is less than condensate pump discharge pressure, 600 psig) or Feedwater by restoring or bypassing components which have failed or been actuated (closed by control systems) and which may have initially caused the loss of feedwater.

Two design features affect the selection of St. Lucie initiating events. First, condensate unavailability precludes heat sink recovery using either Main Feedwater or Condensate. Therefore, initiating events which completely fail Condensate are potentially more significant than other loss of main feedwater initiators. Second, both main feedwater pump discharge headers converge into a single line after high pressure feedwater heaters. Therefore, a break in this line may be risk significant because it would depressurize the line and divert flow away from the steam generator for each of the above-mentioned means of feedwater recovery.

Offsite Power. A loss of offsite power is particularly important at St. Lucie because it can cause both units to trip and fail Main Feedwater and Condensate as well. A two-unit trip is believed to be sufficiently different in potential risk such that losses of offsite power are divided into those which cause only a single unit trip and those which cause a dual-unit trip.

3.1.1.2.2 Shared Systems

The following discussion is extracted from FSAR 3.1.5 for general design criterion 5 [Ref. 3.1-4, 3.1-5], Sharing of Structures, Systems or Components.

Startup transformers may be paralleled under administrative control. No other structures, systems or components important to safety are shared between St. Lucie Unit 2 and Unit 1 except for seismic instrumentation and ultimate heat sink.

The following facilities are shared by both nuclear units:

a) ultimate heat sink

b) fire protection system

c) switchyard, telemetering and load dispatch equipment

d) seismic instrumentation

e) site and off-site environmental monitors

f) service building

g) steam generator blowdown process facility

All facilities listed are constructed so that no single failure can in any way preclude safe shutdown of the plant.

An accident in one unit will not affect safe shutdown of the other unit. An accident in any of the shared features may result in reduced load operation of either or both units, but the capability for safe shutdown is unaffected by such an accident. The only safety component common to the two units is the ultimate heat sink.

In the unlikely event of a loss of the preferred shutdown power, both Units 1 and 2 have their own 100 percent capacity redundant diesel generator sets which would be available for safe shutdown.

The ultimate heat sink emergency canal supplies emergency cooling water to both Units 1 and 2 if the ocean water intake pipes become unavailable. The canal has sufficient cross-sectional flow area to mitigate the consequences of a LOCA on one unit while safely shutting down the other unit.

3.1.1.2.3 Support Systems

The Intake Cooling Water System (ICW) provides cooling water to the Component Cooling Water System (CCW), the Turbine Cooling Water System (TCW) and the Steam Generator Open Blowdown Cooling System (SGOBD) during normal operation. During accident conditions, the ICW System only provides cooling water to the CCW System since the flow path to the TCW and SGOBD Systems are automatically isolated.

Two startup transformers are provided for each unit. Each startup transformer steps down the voltage from 240kV to 6.9kV and 4.16kV. There are two non-safety related DC buses and three safety related buses per unit.

For St. Lucie Unit 1, four redundant 120VAC single phase instrument power buses provide power to essential instrumentation and control loads. Each bus is supplied separately from an inverter powered from one of the vital 125VDC buses. To permit maintenance of any inverter without de-energizing the corresponding instrument bus, two redundant maintenance bypass buses provide power through "make before break" transfer switches. Breaker interlocks are provided to prevent simultaneous connection of more than one instrument bus to a maintenance bypass bus. For St. Lucie Unit 2, four pairs of 120VAC single phase instrument buses provide uninterruptible power to Engineered Safety Features Actuation (ESFAS) and Reactor Protective System (RPS) instrumentation. Each pair of instrument buses is supplied from an inverter connected to one of the vital 125VDC buses. To permit maintenance without disabling the corresponding instrument bus, maintenance bypass transformers and voltage regulators are provided for each pair of instrument buses.

The standby AC power supply for each unit consists of two emergency diesel generator (EDG) sets, their associated air starting and fuel supply systems, and automatic control circuitry. Each engine has a self-contained cooling system which consists of a forced circulation cooling water system which cools the engine directly and an air cooled radiator system which removes heat from the cooling water. The cooling system requires no external source of power and does not depend on any plant cooling system. The engines of each EDG have self-contained lube oil systems consisting of a lube oil sump located at the base of the engine and engine driven lube oil pump, piping, and heat exchanger. The lube oil heat exchanger is served by the EDG cooling water system and thus no external source of power or other plant type systems are required. Each EDG has an independent air starting system. Each EDG is provided with air receivers which have sufficient air to start a cold EDG five (5) times. The air starting system does not depend on normal plant electrical power except for 125VDC control power.

The Unit 1 instrument air system consists of four (two full capacity, two half capacity) instrument air compressors located outside containment as well as two instrument air compressors located inside containment. The Unit 2 instrument air system consists of four (two full capacity and two half capacity) instrument air compressors located outside containment (i.e., no compressors inside containment). Unit 1 and Unit 2 instrument air systems may be cross-connected. The cross-connection consists of normally closed pressure regulating valves which are actuated when system pressure in either unit decreases to 85 psig.

HVAC Systems considered important from a plant risk assessment standpoint include: the electrical equipment room subsystem, the turbine switchgear room ventilation subsystem, and the ECCS room ventilation subsystem. For Unit 2, the intake structure is enclosed and ventilation there is assumed to be required to ensure components within the enclosure are cooled and perform their safety function.

3.1.1.3 Identification and Grouping of Initiating Events

In the previous section, potentially important St. Lucie design features were highlighted. This section describes the general process by which those design features and other criteria are used to identify risk significant initiators and group them into categories. Each such group uniquely impacts St. Lucie safety system operation in response to a reactor trip. In the following section, these groups are compared to the results of other studies and to NRC reviews of such studies.

The process of identifying initiating events involves assembling generic sources of reactor trip experience and reviewing that experience in conjunction with St. Lucie reactor trips and St. Lucie design features. Generic sources include EPRI NP-2230 [Ref. 3.1-6] and other PWR PRAs [Refs. 3.1-7 - 3.1-17]. The first step in the review is to categorize the initiating event groups reported in these sources using the following general Pressurized Water Reactor classification scheme:

- Transients which do not affect mitigating systems
- Transients affecting the Power Conversion System
- Transients causing a PORV to open (RCS pressure increasing)
- Transients causing an SI signal or a main steam isolation signal (MSIS)
- Transients initiated by a loss of offsite power
- Transients initiated by support systems
- Transients initiated by front line systems

- LOCAs
- ATWS

Table 3.1-1 presents the initiating event groups from other PRAs. The following systematically describes the initiating event groups corresponding to these categories. The first subsection defines the transient groups from the first five categories. The second subsection defines transients initiated by front-line or support systems. (These initiator groups are identified through a cooperative effort with the systems analysis.) The third subsection defines the LOCA analysis categories.

3.1.1.3.1 Transient Initiators

Experience from past PRAs and a review of the St. Lucie design indicates that a few key effects of initiators are important to the ability of plant systems to prevent core damage. The following discusses those effects.

The Power Conversion System (PCS) is an important means of ensuring that secondary heat removal (if PCS fails, AFW will provide backup capability) is available. Secondary heat removal is evaluated in the event tree development process. If it fails, once-through cooling must be implemented to prevent core damage.

A transient can develop into a LOCA if a PORV opens, or sticks open, and is not subsequently isolated. While most transients which open a PORV are easily isolated, the causes of openings and the possibility of dependent causes of block valve failure or operator ability to diagnose a LOCA must be systematically examined to ensure these types of LOCAs are not risk significant.

Typically, SI signals significantly affect the configuration and initiation of safety and non-safety systems. Since these signals may occur as a result of plant response, it is appropriate to identify the accident sequence conditions, including initiators. For Unit 1, SI would isolate MFW leading to reactor trip; for Unit 2, without further complications and if operator takes appropriate actions within 10 minutes, no MFW isolation would occur.

Because offsite power is used by most mitigating systems, it is important to account for the effect of its loss.

By reviewing generic transient initiating event groups and considering the above criteria and the plant design features described in Section 3.1.1.2, the following list of St. Lucie transient initiating event groups was obtained.

T_1 - Reactor Trip - This initiating event results from a system disturbance that causes the reactor protection system to insert control rods to terminate the nuclear chain reaction. This event does not cause a severe challenge to safety systems. Because this initiator does not degrade or fail any other safety systems, it only rarely is an initiator of a risk-significant sequence. (A spurious reactor trip signal is an example event for this group.)

T_2 **- Reactor Trip With PORV Challenge** - This initiating event represents a class of transients that result in an increase in primary system pressure to the PORV opening setpoint. For St. Lucie Units 1 and 2, the PORV opening setpoint (2400 psia for unit 1, 2370 psia for unit 2) is the same as the RCS high pressure reactor trip setpoint. Transients causing RCS pressurization without another anticipatory reactor trip will cause a PORV to open. RCS pressurization can be caused by a power-cooling mismatch (energy addition) or increasing inventory (mass addition). Primary system pressure will increase to the PORV and reactor trip set points, thereby causing a reactor trip and PORV challenge. Successful reclosure of the PORVs results in a situation similar to a reactor trip; failure of the PORVs to reclose results in a LOCA which can be isolated by the operators if the PORV block valve can be closed.

T_3 **- Loss of Power Conversion System** - These initiating events result from failures in the PCS (i.e., condensate/feedwater, steam, turbine and condenser including ADVs and steam dump system). A loss of the PCS results in pressurization of the primary system immediately prior to a reactor trip. In the event of coincident failure of ADV/steam dump systems or pressurizer sprays, the PORV may open. Auxiliary feedwater must be actuated to remove decay heat from the steam generator, or the heat sink must otherwise be recovered.

T_{3a}. Loss of Main Feedwater, but recoverable. This group of initiating events includes transients where failures in the PCS likely induce an anticipatory reactor trip on a low-low steam generator level signal. However, at least one train of condensate and main feedwater remains operable.

T_{3c}. Loss of Main Feedwater, but not recoverable. This group of initiating events includes total loss of condensate pumps or blockage of the MFW flowpath; therefore, neither MFW nor condensate recovery can occur.

T_{3d}. Loss of MFW due to feedline break. A section of feedline (downstream of FW HTR and upstream of steam generator feed flow control valves) is not isolable and would lead to loss of feedwater, if a pipe or valve rupture were to occur. Only AFW is available, since other heat sink recovery measures would preferentially pump water through the break since the steam generator would be at higher pressure.

T_{3e}. Excessive feedwater would cause a turbine trip due to high-high steam generator level, followed by a reactor trip. The high-high level would isolate feedwater and result in a low level on one steam generator, and AFAS actuation. The operator will be required to restart feed pumps for Unit 1 because high-high steam generator level also trips the main feed pump and closes the main feedwater discharge header block valves.

T_4 **- Loss of Offsite Power** (Non Loss of Grid Events) - Loss of offsite power results in a reactor trip, turbine generator trip, loss of PCS and a demand for emergency diesel generators. Restart of normally operating safety-related component and intake cooling water is necessary. Loss of offsite power can occur due to failure in the grid (Loss of Grid) or at the switchyard (T_4). PORVs are challenged due to delayed reactor trip on high pressurizer pressure.

T_5 **- Steamline Break Upstream of MSIVs** - This event will blowdown a steam generator and rapidly depressurize the primary system, thereby causing a safety injection signal, main steam isolation, and main feedwater isolation. The steam supply to the turbine-driven AFW pumps from that steam generator is also unavailable. This event also includes spurious opening of atmospheric dump valves or steam generator safety relief valves.

T_6 - Steamline Break Downstream of MSIVs - This event will blowdown a steam generator and rapidly depressurize the primary system, causing a safety injection signal, main steam isolation signal, and main feedwater isolation. However, if the MSIS signal fails to close all MSIVs, then a more severe cooldown transient results.

This event also includes unisolated opening of the steam dump to condenser valves. Feedline or condensate line break events, other than those included in T_{3d}, cause a similar plant response as these steamline breaks. These events are included in T_6.

T_7 - Spurious Main Steam Isolation Signal (MSIS) - This transient involves events where MSIS is actuated but not required. The major impact of the spurious MSIS actuation includes reactor trip, main feedwater isolation, and MSIV closure. Operator actions are required to reset MSIS to make feedwater available. This event also includes manual initiation of MSIS. For Unit 1, T_7 also includes spurious SI actuation because FW isolations occur due to spurious SI and the reactor trips due to low-low SG level.

T_8 - PORV Sticking Open - This transient involves failure of pressurizer pressure transmitters. T_{8a} represents failing high of pressurizer pressure transmitters causing PORV 1404 to remain open. T_{8b} represents failing high of pressurizer pressure transmitters causing PORV 1405 to remain open. The T_8 event is similar to T_2 except that for T_2 PORVs have failed to reclose; for T_8 the PORVs will not reclose because pressurizer pressure transmitters send signals to keep the PORVs open.

To model the functional response of the plant to these transients, a single transient event tree was constructed (See Figures 3.1-1 and 3.1-7; note that there is one tree per unit and their construction is identical). This event tree will also prove sufficiently robust to model the effects of front-line and support system initiators discussed in the next subsection.

The frequencies for each of these initiators were developed in part based on generic data (see Section 3.3). The EPRI document NP-2230 initiator classification includes frequencies for each of these transients.

3.1.1.3.2 Front-Line System Initiators

In addition to initiators that have direct effects upon the safety functions described in Section 3.1.1.3.1, other initiators that can impact the safety functions through front-line and support systems must be identified. This set of initiators originates from front-line or support systems failures that can cause a transient as well as degrade the mitigation capability of the plant.

Each St. Lucie front-line system and support system were examined to determine if a failure could occur in that system that would cause or require a reactor trip, coincident with a degraded state of the system.

3.1.1.3.2.1 Chemical and Volume Control System

The CVCS is normally in operation providing normal makeup to the RCS. A system initiating event would be failure of normal charging. It could also lead to low pressurizer level if normal letdown flow also was not isolated.

3.1.1.3.2.2 Emergency Safeguards Actuation System (ESFAS)

ESFAS includes Safety Injection Actuation Signal (SIAS), Recirculation Actuation Signal (RAS), Containment Spray Actuation Signal (CSAS), Containment Isolation Signal (CIS), Main Steam Isolation Signal (MSIS), and Auxiliary Feedwater Actuation Signal (AFAS). For Units 1 and 2, MSIS will cause a plant trip and affect main feedwater availability. The spurious MSIS actuation is included in T_7 (Spurious MSIS). For Unit 2, a spurious SIAS without complications (i.e., other coexisting or subsequent failures) and operator actions will trip the plant but not isolate the main feedwater. For Unit 1, SIAS actuation will trip the plant. Although AFAS will isolate MFW for Unit 2, no other effects are expected and it is thus not treated as a special transient.

3.1.1.3.2.3 Primary System Pressure Control

This class of events represents failure of specific elements of the Primary System Pressure Control System (PPC) that result in the potential for or an actual demand on the PORVs to open, and will cause a reactor trip. If the PORVs inadvertently open and fail to reclose, a reactor trip due to low pressurizer pressure may occur. These faults are treated individually due to the specific actions required to open and close the PORVs and the ability of the pressurizer (PZR) sprays to mitigate some of these events. T_2, T_{8a}, and T_{8b} represent PPC-related system initiators.

3.1.1.3.2.4 Containment Isolation System

No risk significant containment isolation system initiators were identified. Failure of certain portions of the Containment Isolation System may cause a plant trip. This event is included in T_1. Because no degradation of a mitigative safety system occurs, no special initiator is defined. Failure of the isolation paths either would not cause plant trip, would not be likely because they required multiple passive and active failures (i.e. low frequency), or were not air-to-air paths (i.e. low release magnitude).

3.1.1.3.2.5 Auxiliary Feedwater System

Loss of auxiliary feedwater system does not trip the plant. However, rupture of a steam supply line between a Steam Generator and an AFW steam admission valve or a rupture of a water line between the Main Feedwater line and the first check valve would trip the plant and degrade AFW response.

Both the frequency of a steam supply line failure (passive failure) and its impact on the AFW system (one of two steam sources to one of three AFW pumps) are judged to be low; consequently this system initiator was judged to be insignificant.

The water line failure frequency is also low (passive failure); but more importantly the impact is no different from a Main Feedwater Line break, since AFW flow to the corresponding Steam Generator would be isolated by operators anyway. Therefore, this failure is considered implicitly in T_5 (Steam Line Break upstream of MSIVs).

3.1.1.3.2.6 Safety Injection Tank, High Pressure and Low Pressure Safety Injection/Residual Heat Removal System

Loss of any of these systems does not cause a plant trip. Injection line breaks between the RCS and the first check valve in each of the systems will cause a LOCA and prevent the system from delivering flow through the corresponding line. The frequency of such an event is judged to be small, since the relative amount of piping is small compared to the rest of the RCS piping. Regardless, the system models assume that no flow enters the RCS in the loop in which the LOCA occurs, so the specific impact of injection line failure is already accounted for and requires no additional modeling.

3.1.1.3.2.7 Reactor Protection System

Failure/malfunction of RPS may cause a plant trip but, in doing so, performs the safety function of this system. All such events are included in T_1.

3.1.1.3.2.8 Containment Spray System

Inadvertent containment spray system actuation might cause a plant trip, although no specific analysis was performed to confirm this. However, spray actuation's only adverse effect would be to draw down the unit's RWT. It is assumed that operators would terminate spray flow relatively quickly. The resulting loss of RWT inventory was judged to be insignificant.

3.1.1.3.2.9 Power Conversion System

Loss of the Power Conversion System (main feedwater, condensate, spurious Atmospheric Dump or Condenser Dump Valve opening) will cause a plant trip. Of particular importance is steam line break/feedline breaks that will trip the plant and affect AFW steam supply and feedwater supply to steam generators. These events are included in T_5, T_6 and T_{3d}.

3.1.1.3.3 Support Systems Initiators

As was done for the Frontline Systems in Section 3.1.1.3.2, each potential support system initiator was evaluated in terms of the following:

1. Impact on normal operation

   The immediate response of the plant following each initiator is modeled based on applicable off-normal procedures, one line diagrams, or other available plant specific studies. For example, since RPS is energized by DC power, loss of a DC bus will trip the plant.

2. Attendant primary system failures

   The primary system failures considered include those associated with systems that are required to safely shut down the plant after the trip induced by the special initiator. These systems provide the following functions:

   a) Secondary Heat Removal; AFW, MFW, Steam relief, and Once-through cooling
   b) Core Cooling; HPSI, LPSI, Charging, and Safety Injection Tanks
   c) RCS Integrity; PORV and RCP seals
   d) Support systems; AC power, DC power, ICW, CCW, TCW, Instrument Air and HVAC

   The effect of the special initiator on the systems required to accomplish the above functions is included in the system fault trees. However, the major effects are highlighted below to assure that each initiator is properly classified.

3. Resolution

   It is important to classify each special initiator into one of the following for correct modeling of the plant behavior after the trip:

   1. Reactor Trip
   2. Reactor Trip with PORV opening
   3. Loss of power conversion system (recoverable/unrecoverable)
   4. Loss of offsite power
   5. Steamline break upstream of MSIVs
   6. Steamline break downstream of MSIVs
   7. Spurious MSIS actuation
   8. LOCAs
   9. SGTR
   10. ATWS

The following support systems were considered for potential system initiators:

1. Vital 125 VDC A, B,
2. 120 VAC Vital Instrument Panels
3. 4 kV or 6.9kV Bus A, B,
4. Instrument Air
5. ICW
6. CCW
7. HVAC
8. TCW

3.1.1.3.3.1 Loss of Vital 125 VDC

Two battery chargers are operated in parallel on each vital DC bus. Each battery charger is supplied by a vital 480 VAC MCC. An Off-Normal Operating Procedure provides instructions for operator action in the event of loss of an emergency DC bus [Ref 3.1-18].

In addition to a reactor trip (caused by reactor trip breaker opening due to loss of DC power), loss of a DC bus results in:

- Generator Lockout,
- SIAS, CIAS and MSIS actuation,
- PORV actuation.

All other effects of the loss of a 125VDC bus are modeled indirectly through the dependency of control power or motive power associated with components of mitigating systems.

3.1.1.3.3.2 Vital Instrument AC System

An Off-Normal Operating Procedure [Ref. 3.1-19] indicates that the main effect of loss of the 120V instrument AC system (class 1E) is a 1-out-of-3 logic for RPS, ESFAS and AFAS. The loss of a 120V instrument bus does not lead to a reactor trip without other concurrent malfunctions. A condition which could lead to a reactor trip is loss of an instrument bus concurrent with a RPS or ESFAS channel in trip. This results in satisfying the 2-out-of-4 trip logic. It is for this scenario that loss of a 120VAC instrument bus is considered a special initiator.

3.1.1.3.3.3 4kV and 6.9kV Buses

An Off-Normal Operating Procedure [Ref. 3.1-20] indicates that the major vital AC loads list includes mostly ECCS (HPSI, LPSI, Containment Spray, CCW, ICW) and AFW pumps, and load centers. Loss of one vital 4kV bus may not trip the plant. Loss of a non-vital 4kV bus will result in loss of a condensate pump. Loss of a 6.9kV bus will result in loss of a MFW pump and two RCPs. The loss of a non-vital 4kV or 6.9kV bus is therefore treated as a special initiator.

3.1.1.3.3.4 Instrument Air

An Off-Normal Operating Procedure [Ref. 3.1-21] indicates that with instrument air at 85 psig decreasing on Unit 1, the cross-tie from Unit 2 will open. The cross-tie will close if Unit 2 pressure decreases to 85 psig or Unit 1 pressure increases to 95 psig. This procedure further instructs the operator to ensure that the standby compressor has started and service air compressor is running and to open the "service air cross-tie to instrument air isolation". If instrument air pressure decreases to less than 75 psig, the operator evaluates the need to shutdown the unit. It is conservative to assume that the reactor trips, and thus loss of instrument air is treated as a special initiator.

3.1.1.3.3.5 Intake Cooling Water (ICW)

Malfunction of the ICW system is covered by an Off-Normal Operating Procedure [Ref. 3.1-22]. If adequate cooling to CCW cannot be maintained, the operator is instructed to immediately shutdown the reactor and secure ICW to TCW flow. It is conservatively assumed that the reactor trips, and thus loss of ICW is treated as a special initiator.

3.1.1.3.3.6 Component Cooling Water (CCW)

An Off-Normal Operating Procedure [Ref. 3.1-23] provides instructions to re-establish CCW flow or isolate affected components in the event of a malfunction in the CCW system. If CCW flow to the RCPs has been lost for 10 minutes, the reactor and the turbine are required to be tripped [Unit 2 has an automatic trip]. If flow to three containment cooling units cannot be maintained for 45 minutes, the reactor must be shutdown. It is conservatively assumed that the reactor trips, and thus loss of CCW is treated as a special initiator.

3.1.1.3.3.7 HVAC

The containment cooling system consists of three normally operating fan-coolers (four in total) to maintain ambient containment temperature at less than 120°F. Failure of more than two fan coolers will result in a manually initiated plant shutdown. Other HVAC systems include the control room ventilation system, the auxiliary building ventilation system, the fuel handling building ventilation system, the turbine building ventilation system, the diesel generator building ventilation system, and for Unit 2, the intake structure ventilation system. Because of the relatively long term effects of loss of a HVAC system and potential recovery actions available, no special HVAC initiating event is assumed.

3.1.1.3.3.8 Turbine Cooling Water (TCW)

An Off-Normal Operating Procedure provides instructions in the event of a failure in the Turbine Cooling Water (TCW) system. For a complete loss of TCW, plant shutdown (manual trip) may be required if lowering the power level does not terminate various alarms. Since TCW supports main feedwater pumps, condensate pumps, and Instrument Air compressors, loss of TCW is treated as a special initiator.

3.1.1.3.4 Loss of Coolant Accidents (LOCAs)

LOCAs are another category of events that can affect the safety functions. The development of LOCA categories is based upon the capabilities of the Emergency Core Cooling System to make up for coolant loss through a breach in the Reactor Coolant System (RCS) pressure boundary. The ECCS requirements are categorized in Table 3.1-2. These analyses are based upon input from a review of generic license calculations, Emergency Procedure Guidelines background calculations, success criteria from other PRAs, and the FSAR. Final characterization was based on a more realistic thermal hydraulic analysis [Ref. 3.1-30]. Three categories have been identified: Small-Small, Small, and Large LOCAs. These categories are described below.

3.1.1.3.4.1 Small-Small LOCA (S1) (Breaks 1/2" < D < 3")

A Small-Small LOCA initiating event is a break in the RCS pressure boundary in some location other than the steam generator that exceeds normal charging flow. For these break sizes, the normal charging system cannot maintain level in the pressurizer. Break sizes less than 1/2" in diameter are considered leaks rather than small-small LOCAs, since the normal charging system can maintain RCS inventory so that RCS pressure and pressurizer level do not decrease. For those leaks, slight system depressurization may occur, but no immediate automatic trip or safety injection signal would be generated unless charging failed.

For small-small (S1) LOCAs, the normal charging system cannot maintain pressurizer level and pressure. The S1 LOCA will depressurize the RCS and cause a reactor trip and safety injection signal to be generated. Provided that a secondary heat sink exists, the RCS will reach an equilibrium pressure which corresponds to the pressure at which the liquid phase break flow equals the high pressure pumped safety injection flow. Because the primary system pressure remains above the HPSI shutoff, once-through cooling must be established to remove additional decay heat if the secondary heat sink does not exist.

3.1.1.3.4.2 Small LOCA (S2) (Breaks 3" < D < 5")

This break size is large enough that decay heat removal through a steam generator is not required, and small enough so that primary system pressure remains elevated above the safety injection tank pressure and LPSI pump shutoff head.

3.1.1.3.4.3 Large LOCA (A) (Breaks D > 5")

A large break LOCA results in a fast depressurization requiring activation of the safety injection tanks and the Low Pressure Safety Injection system. High Pressure Safety Injection by itself is not adequate for this event.

3.1.1.3.4.4 Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) differs from other initiating events in several important respects, and for convenience it will therefore be evaluated by a separate event tree. Credible tube failures range in severity from leak rates of a few gallons to several hundred gallons per minute for the guillotine rupture of several tubes. The event chosen as representative of this range is the complete severance of a single tube. This choice is made on the basis that less than complete failure will result in much smaller leak rates, generally within the capacity of the normal make-up system, and a fairly normal shutdown can take place.

On the other hand, multi-tube failures are not explicitly addressed because they are much less likely and because the success criteria for systems called upon to respond are substantially the same as those for the failure of a single tube. In fact, multiple-tube failures may aid in depressurizing the RCS, a necessary action in recovering from a tube failure.

3.1.1.3.5 Other Initiators

3.1.1.3.5.1 Reactor Vessel Failure

Failure of the reactor vessel has been addressed in certain other PRAs by a vessel neutron fluence monitoring program. A vessel failure may lead to core damage directly and is normally considered to be unlikely.

3.1.1.3.5.2 Interfacing System LOCAs

Interfacing System LOCAs are events that occur at the pressure boundary of the primary system and the low pressure interfacing systems. Failure of the pressure boundary is postulated to result in a LOCA outside the containment (most likely in the auxiliary building). The environmental effects of such an event may induce failure of systems required for mitigation (e.g., LPSI). Such scenarios can result in containment bypass situations where fission products can directly escape containment. The frequency of this event is dependent upon the testing procedures used for verification of the pressure boundary, as well as the procedures for testing interfacing systems.

3.1.1.4 Comparison with Initiating Events of Other PRAs

Appendix B of NUREG-1335 [Ref 3.1-39] identifies a number of NRC reviews of industry PRAs. Together with the NUREG-1150 studies of the Zion, Surry, and Sequoyah PRAs [Refs. 3.1-15, 3.1-

16 and 3.1-25], this information reflects NRC staff positions which should be considered in the development of IPE models. The following NRC reviews of PWR initiating events analysis were evaluated:

- Oconee PRA [Ref. 3.1-26]
- Crystal River PRA [Ref. 3.1-27]
- Seabrook PRA [Ref. 3.1-28]
- Millstone 3 PRA [Ref. 3.1-29]
- Yankee Rowe PRA [Ref. 3.1-17]

Three general conclusions resulted from these reviews. First, considering initiating events from other PRAs and EPRI NP-2230 [Ref. 3.1-6] is sufficient to ensure a level of completeness in initiating event analysis consistent with NRC desires and other PRAs.

Second, in general the NRC found industry studies of initiating event groups to be sufficiently wide in scope. More often, NRC reviews tended to reduce the number of initiating event groups. Many initiating event groups, such as loss of condenser vacuum and turbine trip, were grouped with other categories, such as loss of main feedwater and reactor trip.

Lastly, the principal disagreements between NRC reviewers and industry analysts tended to center around assigning EPRI NP-2230 initiating event categories to initiating event groups. The following example EPRI PWR categories varied in recommended assignments to initiating event groups both between industry and NRC reviewers and among NRC reviewers:

15 Loss or reduction in feedwater flow (one train)

21 Feedwater flow instability--operator error

22 Feedwater flow instability--mechanical instability

In some applications, these categories were assigned to loss of feedwater categories, significantly increasing their impact on plant response. In other applications, these categories were apportioned about equally between reactor trip and loss of feedwater categories. The frequency of these categories compiled by EPRI was higher than the total loss of feedwater categories. Consequently, if fifty percent of these categories are assigned to total loss of feedwater, the frequency of that event is increased.

NRC's recent PRA studies (NUREG-1150) do not alter the above. EPRI PWR categories 21 and 22 were assigned to a total loss of feedwater initiating event group. Therefore, the loss of feedwater frequencies should be expected to be lower than NUREG-1150 evaluations.

NUREG-1150 also provides conditional probabilities for PORV opening for various initiating event groups. For reactor trip events, a 0.014 probability of PORV opening is assigned based on a review of the data in WCAP-9804 [Ref. 3.1-31]. For loss of offsite power and bus losses, a probability of 0.1 is assigned. No source is provided for this latter estimate. For event scenarios which fail both condenser and atmospheric dumps, a probability of PORV opening of 1.0 is assigned, apparently based on engineering judgement. Consequently, the review of WCAP-9804, St. Lucie thermal hydraulic analysis [Refs. 3.1-30 and 3.1-32], and PORV openings experienced at St. Lucie were important in developing more realistic PORV opening estimates.

In conclusion, NRC staff positions on initiating events appear to cause significant potential conservatism only in determining a loss of feedwater frequency and a PORV opening frequency. Detailed assessments of feedwater recovery removed the first conservatism. A plant specific evaluation of PORV opening including operating experience addressed the second conservatism.

3.1.1.5 Initiating Event Summary List

The initiating events and their treatment in the St. Lucie PRA project is summarized in Table 3.1-3.

3.1.2 Accident Sequence and Functional Event Tree Development

3.1.2.1 Introduction

Section 3.1.1 identified and categorized the plant events that result in a reactor trip or accident such as a LOCA or Main Steam Line Break. In the process of risk assessment, the next step is to develop an understanding of how these events which challenge the plant's safety systems proceed to either a "safe" or "core damage" condition. This section describes the methods used for development of the accident sequences for St. Lucie Units 1 and 2.

Core damage sequences are characterized by the following factors:

1) mission(s) of the various plant systems,

2) equipment performance requirements for each mission,

3) systems actuated or recovered by operators, and

4) system time sequencing and operating periods.

There are many plant systems and procedures and, as a result, many possible sequences of events. Since only certain systems and procedures directly affect whether core damage can occur, the many possible sequences of events can be described more simply in terms of systems on the "front line" and the "critical" safety functions they perform.

Consequently, a straightforward method to accurately identify core damage sequences is to identify the critical safety functions needed, their sequences of success, failure, and recovery, and their corresponding front line system requirements. The following subsections describe this method and the resulting definitions of core damage accident sequences investigated.

3.1.2.1.1 Critical Safety Functions

Initiators begin with a unit at power operation. Because safety systems are designed for shutdown rather than full power loads, the reactor must first be brought to a subcritical condition. Those initiators requiring shutdown may threaten the core by causing either a loss of RCS integrity or a loss of operating heat sink systems. Therefore, the initial defense against core damage is to maintain or recover integrity and heat sink. The EOPs are designed to guide the operators in maintaining these "critical" safety functions (CSFs).

Table 3.1-4 lists the safety functions identified in the St. Lucie plant EOPs.

3.1.2.1.2 Relationship Between Frontline And Support Systems

The accident sequence definition task primarily addresses the operation of "frontline" systems (such as HPSI, AFW, LPSI, etc.); the necessary support systems such as AC power, DC control power, CCW/ICW, HVAC and ECCS actuation are embedded in the frontline system analysis. The interface/interdependency is considered and analyzed in each frontline system . The dependency matrices for frontline/support systems are provided as a part of Section 3.2 (Systems Analysis).

3.1.2.1.3 Success Criteria

The success criteria for any initiating event (see Table 3.1-5 for initiator listing for St. Lucie Units 1 and 2) are the minimal system operations requirements and/or operator actions to maintain a stable core configuration and thus, ultimately, to prevent core damage. A stable core configuration is maintained if all CSFs are satisfied. Operator actions depend on the systems available at any one time; therefore, the success criteria for timing of each action can only be developed in the context of the sequence of events and system operability.

The principal sources of information on the development of success criteria are Combustion Engineering nonproprietary reports in the CEN series and the final safety analysis report (FSAR) for St. Lucie. In these reports, the criterion for acceptable plant/system performance is the conservative peak fuel cladding temperature (2200°F) stipulated in licensing analyses [Ref. 3.1-33] or design basis DNBR requirement not violated [Ref. 3.1-34].

In addition, the analyses in all available documentation use the conservative estimates of decay heat generation rates specified by licensing regulations. Any success criteria that are based on these values are therefore conservative. The development of more realistic core temperature limits, decay heat predictions, and degraded-core parameters for every event would have required considerable analysis beyond the scope of this study.

Perhaps the most significant impact of adopting more realistic criteria is to extend the time available for a system to be started or recovered, thereby increasing the probability that the operator takes the correct action. A set of initial MAAP analyses was performed to gain a better understanding of the sequence of events for transients where Appendix K type calculations result in a more prompt demand for operator actions (assuming that human reliability has a time correlation, this would result in high failure probabilities for operator actions).

As discussed in Section 3.1.1, the initiating events are grouped into two major categories: Transients and LOCAs. For transients, further subdivisions into T_1, T_2, T_3, T_4, T_5, T_6, T_7, T_8 are made. For LOCAs, further subdivisions into Large LOCA (A), Small LOCA (S2), and Small-Small LOCA (S1) are made. In addition, SGTR and ATWS are added for enhanced understanding and better modeling of these events. Special initiators, including Loss of Grid, are also included for completeness.

Accident sequences are inherently dynamic, hence they may move from one group to another. For example, a transient characterized by repeated opening of a PORV becomes a LOCA if that PORV sticks open. Conversely, an accident sequence involving rupture of a steam generator tube (like a small-small LOCA) behaves like a transient if the operator succeeds in isolating the faulted steam generator, cooling down the RCS and reducing its pressure to a point below the pressure of the faulted steam generator.

The events leading to a loss of coolant from the RCS (LOCA) vary, and thus the timing of that accident sequence cannot be precisely determined. However, there are clearly two phases of operation: injection, the timing of which is governed by the rate of depletion of the refueling-water tank (RWT), and recirculation following the depletion of the RWT. The system performance requirements for each of these phases are as follows:

> Injection Phase: By removing decay heat from the core, the systems or groups of systems actuated immediately following the loss of coolant maintain adequate inventory to prevent fuel temperature from exceeding the specified limit.
>
> Recirculation Phase: When RWT level is low, systems are realigned to establish stable, long-term heat removal from the core and the containment.

In the case of transients, success is usually achieved by the simple process of satisfactory secondary heat removal. However, in the event of failure of all secondary heat removal, the transient can effectively be turned into a small LOCA by the opening of both PORVs.

The success criteria for each of the safety function and associated systems depend on the initiating events and the pre-existing conditions (i.e., scenarios) as discussed above. Sections 3.1.2.2 through 3.1.2.7 describes the applicable success criteria for each of the initiators.

3.1.2.1.4 Methodology For Accident Sequence Definition

The method used to define accident sequences includes development of functional event trees and top logic fault trees for each function. Figures 3.1-1 through 3.1-12 provide the St. Lucie specific functional event trees.

The failure events are represented by fault trees. Accident sequences are quantified by linking the fault trees for the failure events together. Success events in a sequence are also considered explicitly in obtaining the sequence solutions.

The sequence types are defined in the sequence class column and fall into three categories: core-damage (CD), non-core-damage (NCD i.e., OK), and transfers to other event trees (LOCA, ATWS). Not every path includes a branch point for each top event in the event tree, since the success or failure of an event is in many cases predetermined by the status of events preceding it or is irrelevant in evaluating the end states of interest. A branch point is included when failure of an event affects the need for, or the likelihood of, an event later in the tree, or if it is relevant in discriminating between core-melt end states for the consequence analysis. The Containment Performance Analysis Task identified the systems and sequences which affect the consequences of core-damage sequences.

Supporting logic for the event tree top events (other than those events corresponding directly to the failure of a single system) are developed to reflect the success criteria. This appears in the form of "top-logic" fault trees (see Appendix A). These "top-logic" fault trees relate the top events in the system fault trees, effects of the initiating events, and any top-level human errors to the functions represented by the event tree top events.

Accident sequence descriptions, including event trees and top logic, are described below for each of the initiators previously described. These initiators belong to a number of classes of accidents and are grouped according to the safety functions required for their mitigation.

In general, the top event gates and event tree headings are prefixed with U1 for Unit 1, and U2 for Unit 2 in the top event logic and event trees. For the sake of brevity, in the following section where discussions are applicable to both Units 1 and 2, the U1 and U2 prefix are omitted. For certain systems, where Unit 1 and Unit 2 gates are not prefixed by U1 or U2, the corresponding gates for each unit are taken directly from the system fault trees and are included in the event tree top logic of Unit 1 and Unit 2. For the following discussions, where Unit 1 and Unit 2 information differs, the Unit 2 data may be shown in brackets ([ ]).

3.1.2.2 Event Tree and Supporting Logic for Transients

Eight types of transient initiators, each of the which has a unique impact on the likelihood of core damage, were considered in the study of St. Lucie Units 1 and 2. These unique impacts are highlighted in the following discussion. Those eight initiators are:

- T_1 - Reactor Trip
- T_2 - Reactor Trip with PORV Challenge
- T_3 - Loss of Power Conversion System
- T_4 - Loss of Offsite Power
- T_5 - Steamline/Feedline Break Upstream of Main Steam Isolation Valves (MSIVs)

T_6 - Steamline Break Downstream of MSIVs

T_7 - Spurious MSIS Actuation

T_8 - PORV Sticking Open

The Loss of Power Conversion System transient (T_3) was further broken down into four separate initiators. These were:

T_{3a} - Loss of Main Feedwater, recoverable

T_{3c} - Loss of Main Feedwater, not recoverable

T_{3d} - Loss of Main Feedwater due to feedline break

T_{3e} - Excessive Feedwater

In addition, the following support system (or special) initiators were considered:

Loss of 125VDC Bus
Loss of 4kV Bus
Loss of 6.9kV Bus
Loss of Instrument Air
Loss of ICW
Loss of CCW
Loss of TCW
Loss of 120VAC Instrument Bus
Loss of Grid

The transient event tree is shown in Figure 3.1-1 for Unit 1 and Figure 3.1-7 for Unit 2.

Transient initiators result in a demand for a number of safety functions and, therefore, a number of potential accident sequences. The first of these is the requirement for the reactor protection system (RPS) to terminate the nuclear chain reaction. A special event tree is developed for those events where subcriticality is not satisfied, (i.e., ATWS). Primary system integrity must be maintained, ensuring enough mass to allow for long-term heat transfer to the steam generators. Secondary heat removal (i.e., steam generator cooling using main or auxiliary feedwater and secondary relief valves or the main condenser) is required to prevent boil off of primary system inventory. If steam generator cooling is not available, then once-through cooling can remove decay heat directly by releasing primary coolant to the containment through the PORVs and replacing it with water from the HPSI system (bleed and feed cooling or once-through cooling). Once the once-through cooling mode has been established, either secondary heat removal must be recovered or high pressure safety recirculation (HPSR) must take place.

Failure of primary system integrity can result from relief valve challenges occurring at about the time of the reactor trip, or through subsequent failure of the RCP seals. (The latter event, if it occurs, is delayed until some time after the trip.)

3.1.2.2.1 Top Event Descriptions

EVENT K: REACTOR SUBCRITICALITY

Transient initiating events place a demand on the RPS to terminate the nuclear chain reaction, reducing core power to decay heat levels. Failure of the RPS results in severe challenges to the ability to maintain RCS heat removal and integrity given that a transient initiating event has occurred. For this reason, TK sequences are treated by transfer to a special event tree for ATWS sequences. The treatment of ATWS sequences is discussed in Section 3.1.2.7. All remaining events in the transient event tree are defined within the context of successful RPS operation. No top event logic is developed for reactor subcriticality. The fault tree for subcriticality developed under gate KW01 can be used to quantify accident sequence.

EVENT Q - RCS INTEGRITY

Loss of RCS integrity can result from either of two causes, a PORV sticking open or a loss of reactor coolant pump seal integrity.

Pressure relief for the primary system at St. Lucie is provided by two solenoid pilot operated PORVs and three pressurizer SRVs. The setpoint of the PORVs is 2400 psia [2370 psia] and the setpoint of the SRVs is 2500 psia. Motor operated block valves 1405 and 1403 [1477 and 1476] are installed upstream of PORV 1404 and 1402 [1475 and 1474], respectively.

Pressure relief occurs if the primary system cannot transfer sufficient heat to the secondary system. Failure to transfer sufficient heat could occur shortly after reactor trip during a core power-feedwater mismatch. Event Q addresses the potential for loss of RCS integrity at this time.

Only a portion of transient initiators will result in a sufficient core power-feedwater mismatch to cause a relief valve to actuate. Only T_2 initiators would result in PORV actuation. Furthermore, even if both PORVs fail to open or are blocked, pressure is unlikely to increase to near the SRV setpoint. Thus, in no transient initiator assessed in this study is an SRV expected to open during the time period considered in event Q.

During this early period of plant response, steam relief from the PORVs will be sufficient to compensate for the temporary mismatch in RCS heat removal. Experience with relief valve failures indicates that they are more likely to reclose under steam relief conditions rather than liquid relief.

The second potential cause of loss of RCS integrity is failure of the RCP seals. The St. Lucie reactor coolant pumps, by design and field experience, are not susceptible to seal failure resulting from loss of seal cooling water. The reactor coolant pumps are equipped with Byron-Jackson (BJ) four series-arranged face seals, all of which are designed for 2500 psig. A seal leakage chamber structurally designed for 2500 psia is provided to collect controlled seal leakage and conduct it to a closed system. The fourth face seal is provided as an integral part of the seal leakage chamber to prevent liquid or gaseous leakage from escaping to the atmosphere. This seal is capable of holding against 2500 psia in the static condition and during coastdown following failure of the three series-arranged main seals. When holding against 2500 psia in the static condition, the seal leakage should not exceed the normal operating seal leakage. The seal assembly is cooled by circulating the controlled leakage through a coiled tube heat exchanger cooled by component cooling water.

Component cooling water to the reactor coolant pumps is not required to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100 [Ref. 3.1-5]. The St. Lucie design, however, has accommodated the potential for loss of CCW to the RCP seals in two ways:

1) The pumps have inherent capability to accommodate interruption of cooling, and

2) Diverse and redundant intelligence has been provided to the operator.

Based on the pump manufacturer recommendations, the operator is directed to trip a RCP if CCW is lost and cannot be restored within 10 minutes. This requirement is reflected in St. Lucie emergency and off-normal operating procedures. Low CCW flow to each pump is indicated and alarmed in the control room. In addition, RCP intelligence such as bearing temperatures, heat exchanger temperatures, and controlled bleedoff temperatures are indicated and/or alarmed in the control room for each pump. On Unit 2, an automatic reactor trip will occur if pump cooling is not restored within 10 minutes.

FPL has conducted a test of RCP seals under simulated loss of AC power conditions at full temperature and pressure. Loss of AC power would result in loss of the component cooling water to the seals. After approximately 50 hours at coolant conditions of 550°F and 2250 psig, the RCP seal cartridge still performed satisfactory with the pump idle. Although some seal damage was observed during the post-test inspection, the maximum seal leakage during the test was only 16 gph.

The high reliability of the RCP seal design used at St. Lucie is also evidenced by the fact that no failures leading to significantly large leakages have occurred at CE plants with Byron Jackson pumps throughout their operating history [3.1-38].

Based on the above discussion, it is concluded that:

1) The potential for RCP seal failure during normal operation is bounded by the small-small and small LOCA frequencies and is therefore not explicitly modeled.

2) RCP seal LOCAs will not occur during station blackout conditions.

3) RCP seal LOCAs will not occur if the RCP is tripped within 10 minutes after a loss of CCW.

It has been conservatively assumed that operation in excess of 10 minutes without CCW cooling will lead to a catastrophic failure of the RCP seals.

EVENT B - SECONDARY HEAT REMOVAL

This event addresses the likelihood of secondary heat removal failure and its recovery before steam generator dryout. St. Lucie has two motor-driven AFW pumps and one turbine-driven pump available for each unit.

The normal response after a reactor trip is for the plant to stabilize at hot shutdown conditions, with heat removal provided by the steam generators.

If main feedwater is lost to the steam generators and insufficient auxiliary feedwater flow is being provided, EOP-06 ("Total Loss of Feedwater") provides instructions for operators to go to Functional Recovery (EOP-15), and initiate once-through cooling.

For main steamline break scenarios, primary makeup is included (via HPSI or CVCS) to consider the initial overcooling and reduced inventory in the RCS.

EVENT F: ONCE-THROUGH (Bleed and Feed) COOLING

If secondary heat removal is not recovered, wide range steam generator level indication will decrease. Heat transfer degradation will occur after approximately 30 minutes when most of the tube bundle is uncovered and the RCS begins to heat up. The RCS heat up will also be indicated by the increasing pressurizer level and pressure caused by RCS fluid volume expansion. The PORVs will open before SG dryout and the end of this phase, steam relief, will then occur. If the PORVs opened early in the event, they will also open again later.

EVENT U: CORE COOLING (SHORT TERM)

For transient-induced small-small LOCAs, early core cooling is accomplished by high pressure safety injection. If HPSI is unavailable, an operator must depressurize and use LPSI to provide core cooling.

EVENT X: LONG TERM ONCE-THROUGH (Bleed and Feed) COOLING

Event Node U1XT02 (U2XT02) represents long term core cooling for sequences where RCS integrity is maintained, secondary heat removal is not successful, but bleed and feed (once-through) cooling is available. Only high pressure recirculation is credited. Shutdown cooling is considered not possible due to failure of secondary heat removal. In order for high pressure recirculation to be successful, containment heat removal by either one containment spray or at least 2-out-of-4 containment fan coolers is required. Event Node U1XT01 was added to address the long term cooling after depletion of the condensate storage tank (only for Unit 1; Unit 2 CST has a larger capacity and does not deplete in 24 hours). Three means of long term cooling were credited: AFW long term cooling, shutdown cooling, and once-through cooling. For shutdown cooling to be successful it is assumed that controlled depressurization by steam generator ADVs or steam dump to condensers is available. In addition, normal charging is required to provide inventory control/makeup. Because a separate loss of normal charging event has not been modeled, M1BORATN01 (for emergency boration) is used as a surrogate. This approximation is conservative, but is expected to be reasonable because the additional failures due to boration suction paths over a short period of time (2 hours) are small compared with the failures of the charging system common to both normal charging and emergency boration (e.g., charging pumps).

Event U1XS101 represents failure of long term core cooling for a transient induced small-small LOCA. Long term AFW cooling is a necessary condition to reach shutdown entry conditions. If shutdown cooling is not available, high pressure recirculation in conjunction with containment heat removal represents a feasible means to provide long term cooling. For Unit 2, event U2XS101 is similar to event U1XS101 except no long term AFW cooling failure is included for the shutdown cooling function. Event U1XS102 (U2XS102) represents long term core cooling failure due to high pressure recirculation failure or failure of containment heat removal.

Table 3.1-6 summarizes the success criteria for the St. Lucie Transient Event Tree.

3.1.2.2.2 Unit 1 and Unit 2 Differences

Unit 1 has one more sequence due to a smaller size CST. The additional sequence accounts for various ways to provide long term cooling.

Unit 2 does not have a corresponding event for XT01 (U1XT01 exists but not U2XT01). Event U1XS101 has one additional failure associated with AFW long term cooling compared with event U2XS101.

Unit 1 requires two of two PORVs to open for successful once-through cooling. Unit 2 requires one of two PORVs.

3.1.2.3 Event Tree and Supporting Logic for a Small-Small LOCA(S1)

This study of St. Lucie 1 and 2 considers two small LOCA initiators, a LOCA whose size is between an equivalent 1/2" and 3" diameter pipe (Small-Small LOCA, S1) and a LOCA whose size is between an equivalent 3" and 5" diameter pipe (Small LOCA, S2). Figure 3.1-2 and 3.1-8 show the functional Small-Small LOCA event tree.

In the case of small-small LOCAs, as opposed to the other LOCAs, a heat sink must be provided. The energy lost through the break is insufficient to remove heat from the RCS. For this reason, the event tree considers the Heat Sink function and the corresponding Secondary Heat Removal "subfunction". If secondary heat removal succeeds, core cooling (both injection and recirculation) must still be provided to prevent core uncovery.

The requirement to achieve recirculation is conservative. In the small-small LOCAs that have occurred in PWR operating experience, recirculation has never been required. One of two cases has occurred: the LOCA may have been isolable by closing a valve, or depressurization and cooldown terminated the leak sufficiently to allow LPSI cooling to be implemented before injection water supplies were depleted. Discussions with plant operations staff revealed that using LPSI for shutdown cooling is a viable option for the Small-Small LOCA and was considered in this analysis.

3.1.2.3.1 Top Event Descriptions

EVENT K: REACTOR SUBCRITICALITY

The S1 LOCA initiating event places a demand on the RPS to terminate the nuclear chain reaction, reducing core power to decay heat levels. Failure of the RPS results in severe challenges to the ability to maintain RCS heat removal and to prevent further loss of integrity given that a Small-Small LOCA initiating event has occurred. For this reason, S1K sequences are treated by transfer to a special event tree for ATWS sequences.

EVENT B_{S1} - SECONDARY HEAT REMOVAL

In the transient event tree description of secondary heat removal a number of systems were identified for providing water to the steam generators. In the case of the S1 LOCA, AFW is the only system considered in the top logic. A LOCA will cause a safety injection signal which, in turn, will close the feedwater isolation valves (Unit 1 only) and provide a close signal to the already closed bypass valves. Use of main feedwater or condensate will each require resetting the safety injection signal and opening bypass valve(s). These actions were input for consideration by the Recovery task of this study.

EVENT F_{S1}: ONCE-THROUGH (Bleed and Feed) COOLING

For S1 LOCAs, it is conservative to use the same requirement for once-through (bleed and feed) cooling as that for transient. The break itself may be able to depressurize as effectively as one PORV. However, the time available for an operator to initiate once-through cooling may be shorter due to faster inventory loss.

EVENT U_{S1}: CORE COOLING (SHORT TERM)

For S1 LOCAs, early core cooling represents inventory makeup to prevent core uncovery and subsequent core heatup which if uncorrected will lead to core damage. High pressure safety injection is the primary source to perform the early core cooling function. Although not credited, if HPSI is unavailable, an operator may depressurize and use LPSI to provide core cooling.

EVENT X_{S1}: CORE COOLING (LONG TERM)

Long Term core cooling for S1 LOCAs is similar to that for those transients in which RCS integrity is not maintained. The reactor can be brought to hot standby, hot shutdown, or cold shutdown conditions. The following options to accomplish long term core cooling are considered:

1. RHR Shutdown Cooling
2. High Pressure Recirculation and containment heat removal

If secondary heat removal is available and the RCS has been depressurized, RHR shutdown cooling will provide long term cooling. If the RCS remains at high pressure with successful operation of HPSI (the most likely case), and portions of the high pressure recirculation path fail, the operators may still depressurize the RCS and use shutdown cooling. In addition to one of two RHR heat exchangers, one of three Charging Pumps is required to maintain cold shutdown. As discussed in transient long term cooling in Section 3.1.2.2.1, normal charging is approximated by emergency boration in the top logic.

Table 3.1-7 summarizes the success criteria for the Small-Small LOCA initiator.

3.1.2.3.2 Unit 1 and Unit 2 Differences

For Unit 1, long term AFW operation requires CST makeup. For Unit 2, the larger size of the CST provides sufficient inventory for the mission time of 24 hours. An SI signal will close feedwater isolation valves for Unit 1, but not for Unit 2.

3.1.2.4 Event Tree and Supporting Logic for a Small LOCA (S2)

This break size (between 3" and 5") is large enough that decay heat removal through the steam generator is not required, and small enough so that primary system pressure remains elevated above the safety injection tank pressure and LPSI shutoff head. Therefore, the S2 LOCA event tree does not contain the primary-secondary heat removal function. Early core cooling must be provided by the High Pressure Safety Injection System. Long Term Cooling is provided by high pressure recirculation and containment heat removal. The recirculation actuation signal (RAS) occurs earlier than in a S1 LOCA. The S2 LOCA event tree appears in Figures 3.1-3 and 3.1-9.

3.1.2.4.1 Top Event Descriptions

EVENT K: REACTOR SUBCRITICALITY

The S2 LOCA initiating event places a demand on the RPS to terminate the nuclear chain reaction, reducing core power to decay heat levels. Failure of the RPS results in severe challenges to the ability to maintain RCS heat removal and to prevent further loss of integrity given that a S2 LOCA has occurred. For this reason, S2K sequences are treated by transfer to a special event tree for ATWS sequences.

EVENT U_{S2}: HIGH HEAD SAFETY INJECTION

This event represents failure of 1-out-of-2 HPSI trains to provide initial core cooling for a S2 LOCA.

EVENT X_{S2}: LONG TERM CORE COOLING

Long term core cooling for a S2 LOCA can be achieved by high pressure recirculation and containment heat removal.

For a S2 LOCA, the plant cannot be placed in cold shutdown before the RWT level is less than the amount required to transfer to recirculation (i.e., level at 4' for Unit 1, 5.67' for Unit 2). An alternate source of suction for the HPSI pumps must be established to maintain core cooling. This mode of cooling involves using the HPSI pumps to draw suction from the reactor coolant and RWT inventory that has collected in the containment sump. This mode of cooling, referred to as "Cold Leg Recirculation," could then be sustained until the plant can be placed in a cold shutdown condition. It is noted that LPSI pumps are isolated upon a recirculation actuation signal (RAS). In the top logic, a backup option of hot leg recirculation is provided. If cold leg recirculation fails, hot leg recirculation can be initiated.

Table 3.1-8 summarizes success criteria for Small LOCA.

3.1.2.4.2 Unit 1 and Unit 2 Differences

Unit 1 uses LPSR for hot leg recirculation, Unit 2 uses HPSR for hot leg recirculation.

3.1.2.5 Event Tree and Supporting Logic for a Large LOCA (A)

The large break LOCA (>5") results in a fast depressurization of the primary system. This depressurization results in significant voiding within the core that terminates the nuclear chain reaction. The reactor trip safety function is therefore automatically satisfied. The injection function (event U) does not require the HPSI system for injection, but does require additional safety injection tank and LPSI flow. The recirculation function (event X_a) requires cold leg recirculation and containment heat removal. Hot leg recirculation is not included as a core damage sequence due to the low likelihood that boron precipitation might occur given the low concentration (2.5 - 3.5 weight percent). A 12% concentration was assumed in previous FSAR boron analyses [Ref. 3.1-36, 3.1-37]. Failure of hot leg recirculation was included as a sensitivity issue (see Section 3.7). Although not included as a core damage sequence, hot leg recirculation is still reflected in the large LOCA top logic.

Normal use of the shutdown cooling system is considered not possible. The large LOCA event trees appear in Figures 3.1-4 and 3.1-10. Descriptions of the events in the tree follow.

3.1.2.5.1 Top Event Descriptions

EVENT U_a: CORE COOLING INJECTION FAILURE

This event represents failure of core cooling during the injection phase and is assumed to occur if both low pressure safety injection trains or 2-out-of-4 safety injection tanks fail to respond.

EVENT X_a: FAILURE OF LONG-TERM CORE COOLING

For a large LOCA, cold leg recirculation and containment heat removal is necessary since entry to shutdown conditions is not likely to be satisfied. Cold leg recirculation (XCA) must take place within about 20 minutes after the accident. Failure of this event can occur if automatic actuation and the operator fails to switchover or high pressure recirculation fails.

Table 3.1-9 summarizes the success criteria for a Large LOCA.

3.1.2.5.2 Unit 1 and Unit 2 Differences

Unit 1 uses LPSR for hot leg recirculation, Unit 2 uses HPSR for hot leg recirculation.

3.1.2.6 Event Tree and Supporting Logic for a Steam Generator Tube Rupture (SGTR)

The methods and/or operator actions required to respond to a SGTR event are outlined in EOP-04. The objectives of the EOP's are to limit the release of radioactive effluents from the ruptured steam generator, stop primary-to-secondary leakage to prevent steam generator overfill, and restore reactor coolant inventory to ensure adequate core cooling and plant pressure control. The SGTR functional event tree is depicted on Figures 3.1-5 and 3.1-11.

3.1.2.6.1 Top Event Descriptions

EVENT K: FAILURE OF THE RPS TO TRIP THE REACTOR

A SGTR initiating event places a demand on the RPS to terminate the nuclear chain reaction, reducing core power to decay heat levels. Failure of the RPS results in severe challenges to the ability to maintain RCS heat removal and integrity given that a SGTR has occurred. For this reason, RK sequences are treated by transfer to a special event tree for ATWS sequences.

EVENT U_R: HIGH PRESSURE SAFETY INJECTION WITH DEPRESSURIZATION

After a SGTR, an SI signal occurs and high pressure injection is actuated to provide inventory makeup. Thermal-hydraulic analyses based on the MAAP program indicate that with secondary heat removal available, the effect of HPSI is minimal; without secondary heat removal, HPSI is included as part of once-through cooling. This event was thus deleted in the SGTR event tree.

EVENT B_r: FAILURE OF SECONDARY HEAT REMOVAL

This event represents a loss of secondary heat removal through the SGs. MFW is isolated by SI for Unit 1, but is recoverable by operator action. While MFW is not isolated by an SI signal for Unit 2, both MFW and AFW are feasible ways of removing secondary heat. This event is the same as that for the transients (BT01).

EVENT D_r: RUPTURED STEAM GENERATOR ISOLATION

Two functions must be achieved in order to isolate a faulted steam generator: mechanical isolation of the faulted steam generator and cooldown and depressurization of the faulted steam generator below the SG SRV setpoint.

The mechanical isolation requires the isolation/reclosure of the ADV's/SRVs, closure of the MSIV and MSIV bypass valve on the faulted SG, isolation of that steam generator's blowdown lines, and isolation of the AFW steam supply from the faulted generator. The failure to isolate any of these paths is assumed to result in steam generator isolation failure D_r. The faulted steam generator must be depressurized below its SRV setpoint or steam leakage would continue from the SRV; to accomplish this the primary system must be cooled and depressurized.

The depressurization function is accomplished by either pressurizer main or auxiliary spray or by secondary side steam dump through either the ADV's or condenser steam dump valves on the intact steam generator. Either pressurizer spray failure or loss of the steam dump function from the intact steam generator will result in steam generator isolation failure. Additionally, the operator actions must be performed correctly and timely in order to assure the success of the mechanical isolation and cooldown and depressurization functions, and to preclude overfill related secondary failure.

EVENT F_r: ONCE-THROUGH (Bleed & Feed) COOLING

This event is similar to that for a S1 LOCA. Compared to transients with loss of secondary heat removal, there is more time available for the operators to initiate the once-through cooling because of the additional inventory from high pressure safety injection.

EVENT X_r: LONG TERM CORE COOLING HEAT REMOVAL

Depending on the availability of secondary heat removal and whether the leakage from the faulted steam generator is isolated, three (two for Unit 2) branches are possible for long term core cooling:

1. XR01: This top logic includes AFW long term failure and shutdown cooling failure. For Unit 2, this event is not applicable; AFW cooling considers both the short term and long term cooling.

2. XR02: This top logic is for the case where faulted SG leakage is not isolated but secondary heat removal is available; contained HPSI or shutdown cooling is needed to assure long term cooling.

3. XR03: This top logic is for the case where once-through cooling was initiated subsequent to failure of secondary heat removal. Only high pressure recirculation and containment heat removal is credited to provide long term cooling. Shutdown cooling is not assumed to be available because of failure of secondary heat removal.

Table 3.1-10 summarizes success criteria for SGTR.

3.1.2.6.2 Unit 1 and Unit 2 Differences

Unit 1 has one more sequence than Unit 2 due to the larger size of its CST. For Unit 1 there are four valves (2 air operated ADVs and 2 manual isolation valves), for Unit 2 there are 8 valves (4 motor operated ADVs and 4 motor operated isolation valves).

3.1.2.7 Event Tree and Supporting Logic for ATWS

In the previous discussion of transients, small LOCAs, small-small LOCAs and SGTRs, the function Subcriticality was indicated first on the associated event trees. Failure of Subcriticality sequences were transferred to the ATWS event tree. In this section, these transfers are considered in more detail. This analysis is based largely on the CEOG ATWS Analysis [Ref. 3.1-35].

An ATWS event results when the Critical Safety Function (CSF) Subcriticality cannot be attained by automatic or immediate (within one minute) manual insertion of control rods into the core. ATWS sequences can be initiated by transients with loss of heat sink (T1, T3, T4, T5, T6, and T7) or integrity (T2, and T8) as well as by SGTR or small (S2) or small-small LOCAs (S1). For large LOCA, success of the Core Cooling CSF results when the core refloods with borated water; the borated water brings the reactor subcritical. If core reflood fails to occur, the core melts and the question of subcriticality becomes moot.

The ATWS event tree models the response of the three remaining CSFs:

Integrity
Heat Sink
Core Cooling (Early and Late time phases)

An ATWS scenario imposes a significant heat load on reactor safety systems because the core remains at a power level higher than the normal decay heat levels; many of the systems which CSFs employ to mitigate transients or LOCAs are ineffective.

The increased heat loads also impose new threats to core cooling systems. Because the heat capacity of the RCS is small when compared to the heat quickly generated by the core during full or partial power, the post-ATWS response is characterized by a rapid pressurization of the RCS. If this pressurization increases to a peak pressure of 3700 psia, a value employed by Ref. 3.1-35, the functionality of core cooling systems may be affected. Also, there is uncertainty over whether reactor coolant system integrity can be maintained above this level, with integrity failures postulated to vary from induced small pipe breaks to rupture of the vessel. Because of these concerns, the CEOG ATWS Analysis assumed core damage will occur if the peak RCS pressure of 3700 psia is exceeded. Safety margins beyond this point have not been quantified.

The initial core operating conditions influence the magnitude of ATWS heat loads and the resulting RCS pressurization. Core power level and Moderator Temperature Coefficient (MTC) both play important roles. If either is low enough, the RCS cannot pressurize to the ATWS stress limit. Other ranges for these values eliminate certain threats to core cooling. The impact of the above conditions on the Critical Safety Functions and the associated event tree top logic are described below. Figures 3.1-6 and 3.1-12 present the event tree for an ATWS for Unit 1 and Unit 2 respectively.

3.1.2.7.1 Top Event Descriptions

KW - Subcriticality

This event represents manual actions to make subcriticality successful.

QK - RCS Integrity

Regardless of initial conditions, RCS pressure will exceed the pressurizer PORV and SRV setpoints. Each valve must reclose to prevent a S1 LOCA. The top logic for this event models three SRVs and two PORVs although closed PORV block valves may prevent or terminate this latter LOCA path.

BK - Secondary Heat Removal

Secondary Heat Removal is the only portion of the Heat Sink CSF capable of removing ATWS heat loads. Once-through cooling cannot because the PORVs are not large enough to reduce RCS pressure for HPSI. Secondary Heat Removal capability itself is compromised due to the short time available. Recovery of auxiliary feedwater (locally), main feedwater, or depressurization and use of condensate is impractical. It is assumed that auxiliary feedwater to both SGs is required for ATWS scenarios.

CK - Core Cooling After ATWS

Core cooling requirements may vary depending on the state of RCS integrity. A stuck open SRV or PORV requires additional core cooling capacity.

CK - RCS Integrity Succeeded

Core cooling can fail after an ATWS event due to one of two causes. First, emergency boration using the CVCS may fail to lower the power level in the core. If this occurs, RCS pressure will remain high and RCS inventory losses out the PORVs and SRVs will exceed the inventory from Charging and HPSI systems. Second, the RCS may pressurize above Stress Level C (ATWS peak pressure of 3700 psia) and fail core cooling for the previously mentioned reasons.

Stress level C can be exceeded due to two causes. First, if both high power and high MTC are the initial conditions, RCS pressurization to greater than Stress Level C cannot be prevented. In the CEOG ATWS Analysis of a 2700 MWh plant, an MTC of greater than -8 pcm/°F was used to develop the probability of these initial conditions. For St. Lucie Unit 1, cycle 11, MTC greater than -8 pcm/°F is expected 25% of the time, while that for St. Lucie Unit 2, cycle 7, is 5%.

Secondly, Stress Level C can be exceeded if pressure relief fails to control RCS pressurization. It is assumed that all three SRVs and both PORVs on Unit 1 (one PORV on Unit 2) must open to maintain the peak pressure below 3700 psia. A low MTC value also will prevent this failure mode.

CKQ - RCS Integrity Failed

Core cooling can fail due to three causes. The first two conditions, Stress Level C exceeded and emergency boration failure are described in the logic for RCS integrity maintained. The third failure mode results from the need to provide inventory makeup from HPSI.

XK - Core Cooling (late time phase)

This logic is similar to the logic for Core Cooling (Late) from the transient event tree. If RCS integrity is maintained, long term AFW is assumed to maintain the core in a stable condition or controlled shutdown cooling (for Unit 1 only). If RCS integrity has failed, high pressure safety recirculation and containment heat removal is assumed to be required.

Table 3.1-11 summarizes ATWS success criteria.

3.1.2.7.2 Unit 1 and Unit 2 Differences

There are two major differences between Unit 1 and Unit 2 ATWS Scenarios:

1. Fraction of time MTC is greater (i.e. less negative) than -8 pcm/°F; for Unit 1, the value is 0.25 while for Unit 2 the value is 0.05.

2. Unit 1 has one more sequence than Unit 2 because long term cooling by AFW necessitates CST makeup.

3.1.3 Accident Sequence Bin Characteristics

The core damage sequences delineated in the functional event trees are classified based on the core damage timing and reactor coolant system pressure at reactor vessel melt-through. The containment performance analysis has defined criteria for classifying accident sequences into unique bins with similar impact on containment performance. Other binning criteria such as containment pressure boundary status, containment safeguards status, and water availability in the reactor cavity are addressed in the containment performance analysis.

Three time periods are defined for purposes of characterizing the time of release and associated potential off-site consequences:

1. Less than 2 hours;

2. Greater than 2 hours but less than 6 hours;

3. Greater than 6 hours.

These time periods are assigned in this task based on the following rules:

1. Large LOCAs (less than 2 hours),

2. Transients/small LOCAs without AFW coupled with failure of once-through cooling (2 to 6 hours), and

3. Failure of long term cooling for LOCAs/transients (greater than 6 hours).

The RCS pressure at which core damage occurs is a key parameter that influences debris dispersal and attendant core debris cooling ex-vessel, and Direct Containment Heating (DCH). Four reactor pressure ranges are selected to distinguish core damage sequences which could affect subsequent events in the accident progression. These are: 1. Greater than 2000 psig; 2. Between 600 and 2000 psig; 3. Between 200 and 600 psig; 4. Less than 200 psig. Additional discussions on accident sequence binning is provided in Section 4.

Each of the core damage functional sequences were characterized based on the binning factors presented above. Table 3.1-12 describes the core damage bins.

3.1.4 Section 3.1 References

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TABLE 3.1-1 GENERIC TRANSIENT INITIATING EVENT GROUPS AND FREQUENCIES

Source	RT/TT	Loss of RCS Flow	Core Power Increase	Loss of PCS	Partial Loss of Feedwater	Closure of 1 MSIV	Loss of condenser Vacuum	Steamline Break Inside containm-ent	Steamline Break Outside containm-ent	Excessive Feedwater	Loss of Offsite Power	Loss of compon-ent cooling water	Loss of service water	Spurious safety injection	Loss of instrument air	Loss of power to necessary bus	Steam genera-tor tube rupture
WASH-1400	7.0		.2	3.0													
NP-2230	2.31/1.76	partial/full .39/.03	.22	.15	1.88	.23	.20			.70	.14		(1) .01	.06		.09	
Crystal River (NREP)	8.48			1.78							.32	.01					
Zion	3.77/3.69	3.58E-1	2.28E-2	5.17		2.52E-1		9.40E-4	9.40E-4		5.76E-2	9.40E-4	(all) 9.40E-4	6.36E-1			2.44E-2
Oconee	4.9			line break/other 6.4E-1/-9.3E-4	6.9E-1	N/A	2.1E-1	3.0E-4	3.0E-3	9.2E-2		included	(all) line break/other 9.3E-4/-2.2E-3	Pzr 1.0E-2/-4.4E-2	1.7E-1	MCC/4kV 2.0E-2/-5.4E-3	8.6E-3
Seiswell	3.77/3.69	3.58E-1	2.28E-2	5.17		2.52E-1		9.40E-4	9.40E-4		3.50E-2	9.40E-4	(all) 9.40E-4	6.36E-1			2.44E-2
ANO-1 (IREP)	7.1			1.0							.32		(all) 2.6E-3			3.5E-2 (AC) 1.8E-2 (DC)	
Millstone-3	3.03/2.33	4.19E-1	7.18E-2	7.29E-1				3.88E-4	3.78E-2		1.1E-1		(1) 1.27E-2	4.99E-2		3.91E-3 (DC) 6.15E-2 (VAC)	3.92E-2
Seabrook	3.13/1.95	5.6E-1	2.73E-2	3.31E-1	2.53	(all) 3.54E-1/-2.44E-3	4.18E-1	4.65E-4	(valves) 6.04E-3/-4.94E-2	1.38	1.35E-1	1.39E-6	(all) 2.52E-6	6.40E-2		3.35E-2 (DC)	1.38E-2
Sequoyah	6.3			7.22E-1							9.05E-2					5.0E-3	1.0E-2
Surry	7.3			9.4E-1							7.7E-2					5.0E-3	1.0E-2
Yankee Rowe	2.35/-8.49E-1			1.98E-1		4.32E-2	1.47E-2	2.73E-3	(PWR-29) 2.73E-3/-2.21E-2	2.74E-3	6.16E-2	1.24E-3	(all) 1.24E-3		1.24E-3	1bus/-2buses 3.0E-3/-1E-5	1.06E-2
Range Low/High	2.3 - 8.48	.34 - .56	2.3E-2 - 2.2E-1	.15 - 5.2	.69 - 2.53	.043 - .354	.015 - .42	3.0E-4 - 2.7E-3	9.4E-4 - 3.8E-2	2.7E-3 - 1.38	.035 - .32	1.39E-6 - .01	2.5E-6 - .01	.050 - .636	.001 - 0.17	5E-3 - 9E-2	8.6E-3 - 3.9E-2

TABLE 3.1-2

LOCA SUCCESS CRITERIA SUMMARY

					SYSTEM REQUIREMENTS					
LOCA Type	SIS Generated	CVCS	AFW	HPSI	HPSR (1/2)	LPSI (1/2)	LPSR (1/2)	SIT	DHR
Small-Small (S1)	Yes	1/3	1/3	1/2	X	N/A	N/A	N/A	X
Small (S2)	Yes	N/A	N/A	1/2	X	N/A	N/A	N/A	N/A
Large (A)	Yes	N/A	N/A	N/A	X	1/2	N/A	3/4	N/A

TABLE 3.1-3

INITIATING EVENT TREATMENT SUMMARY

Initiator	Description	Treatment
T_1	Reactor/Turbine Trip	Transient Event Tree
T_2	Reactor Trip (PORV challenged)	Transient Event Tree
T_3	Loss of PCS	Transient Event Tree
T_4	Loss of Offsite Power	Transient Event Tree
T_5	Steamline Break Upstream of MSIVs	Transient Event Tree
T_6	Steamline Break Downstream of MSIVs	Transient Event Tree
T_7	Spurious MSIS (for Unit 2); Spurious SI and MSIS (for Unit 1)	Transient Event Tree
T_8	PORV Sticking Open	Transient Event Tree
S_1	Small-Small LOCA	S1 LOCA Event Tree
S_2	Small LOCA	S2 LOCA Event Tree
A	Large LOCA	Large LOCA Event Tree
R	Steam Generator Tube Rupture	SGTR Event Tree
Special	Loss of DC Buses.	Transient Event Tree
	Loss of 4kV Buses	Transient Event Tree
	Loss of 6.9kV Buses	Transient Event Tree
	Loss of 120VAC Instrument Bus	Transient Event Tree
	Loss of TCW, ICW, CCW, and IA.	Transient Event Tree
	Loss of Grid	Judged to be insignificant for dual-unit core damage; single-unit core damage includes blackout tie.

TABLE 3.1-4 FUNCTIONAL RECOVERY SUCCESS PATHS

SAFETY FUNCTIONS		SUCCESS PATHS
REACTIVITY CONTROL	1	CEA Insertion
	2	Emerg. Boration CVCS
	3	Nuclear Instrumentation
	4	Emerg. Boration HPSI
MAINT OF VITAL AUX - DC	1	DC Normal Alignment
	2	Alternate DC Alignment
MAINT OF VITAL AUX - AC	1	AC Power from Offsite
	2	AC from Diesel Generators
	3	AC Power from Other Unit
RCS INVENTORY CONTROL	1	Charging and Letdown
	2	SIAS and Charging Pumps
RCS PRESSURE CONTROL	1	Heaters and Spray
	2	Charging and Letdown
	3	Safety Injection
	4	Steam Generator Heat Removal
	5	PORVs
RCS & CORE HEAT REMOVAL	1	Forced Circulation with no SIAS
	2	Natural Circulation with no SIAS
	3	Steam Generator with SIAS
	4	Once-Through-Cooling
CONTAINMENT ISOLATION	1	Normal Containment Conditions
	2	CIAS
CONTAINMENT TEMP &	1	Normal Conditions
PRESS CONTROL	2	Containment Coolers
	3	CSAS
CNTMT COMBUSTIBLE GAS	1	H2 Concentration less than 3.5%
	2	H2 Concentration greater than or equal to 3.5%

Table 3.1-5
ST. LUCIE UNIT 1 AND UNIT 2 INITIATOR LISTING

		INITIATING EVENT	GATE NAME UNIT 1	GATE NAME UNIT 2
	T_1 -	Reactor Trip	ZZT1U1	ZZT1U4
	T_2 -	Reactor Trip with PORV Challenge	ZZT2U1	ZZT2U4
	T_{3a} -	Loss of Main Feedwater, but recoverable	ZZT3AU1	ZZT3AU2
	T_{3c} -	Loss of Main Feedwater, but not recoverable	ZZT3CU1	ZZT3CU2
	T_{3d} -	Loss of MFW due to feedline break	ZZ3DU1	ZZ3DU2
			ZZ3DU1A	ZZ3DU2A
			ZZ3DU1B	ZZ3DU2B
	T_{3e} -	Excessive Feedwater	ZZT3EU1	ZZT3EU2
	T_{4A} -	Loss of Offsite Power to 1A and 2A Startup	ZZT4A	ZZT4A
	T_{4B} -	Loss of Offsite Power to 1B and 2B Startup	ZZT4B	ZZT4B
SI	T_5 -	Steamline Break Upstream of main steam isolation valves (MSIVs)	ZZ5U1A	ZZ5U2A
			ZZ5U1B	ZZ5U2B
SI	T_6 -	Steamline Break Downstream of MSIVs	ZZT6U1	ZZT6U2
SI	T_7 -	Spurious SI Actuation	ZZT7U1	ZZT7U2
SI	T_{8a} -	PORV 1404 [1474] STICKING OPEN	ZZT8AU1	ZZT8AU2
SI	T_{8b} -	PORV 1402 [1475] STICKING OPEN	ZZT8BU1	ZZT8BU2
SI	S_1 -	Small-Small LOCA S1 (Breaks 1/2"<D<3")	ZZS1U1	ZZS1U2
SI	S_2 -	Small LOCA S2 (Breaks ~ 3"-5" in D)	ZZS2U1	ZZS2U2
SI	A -	Large LOCA (Breaks D > 5")	ZZAU1	ZZAU2
SI	R -	Steam Generator Tube Ruptures	ZZRU1A	ZZRU2A
			ZZRU1B	ZZRU2B

Table 3.1-5 (Cont'd)
ST. LUCIE UNIT 1 AND UNIT 2 INITIATOR LISTING

INITIATING EVENT	GATE NAME UNIT 1	UNIT 2
SPECIAL INITIATORS		
LOSS OF 125V DC BUS 1A	ZZDC1A	
LOSS OF 125V DC BUS 1B	ZZDC1B	
LOSS OF 125V DC BUS 2A		ZZDC2A
LOSS OF 125V DC BUS 2B		ZZDC2B
LOSS OF 4KV BUS A	ZZ4KV1A2	ZZ4KV2A2
LOSS OF 4KV BUS B	ZZ4KV1B2	ZZ4KV2B2
LOSS OF 6KV BUS A	ZZ6KV1A1	ZZ6KV2A1
LOSS OF 6KV BUS B	ZZ6KV1B1	ZZ6KV2B1
LOSS OF INSTRUMENT AIR	ZZIAU1	ZZIAU2
LOSS OF ICW	ZZICWU1	ZZICWU2
LOSS OF CCW	ZZCCWU1	ZZCCWU2
LOSS OF TCW	ZZTCWU1	ZZTCWU2
LOSS OF 120VAC INSTRUMENT BUS	ZZMAU1	ZZMAU2
	ZZMBU1	ZZMBU2
	ZZMCU1	ZZMCU2
	ZZMDU1	ZZMDU2
LOSS OF GRID (BOTH UNITS 4KV BUSES A, B, AND 6.9KV BUSES A, B)	ZZLOG	ZZLOG

TABLE 3.1-6
TRANSIENT SUCCESS CRITERIA SUMMARY INFORMATION

INITIATORS: T_1 - REACTOR TRIP, T_2 - REACTOR TRIP (PORV CHALLENGED), T_3 - LOSS OF PCS, T_4 - LOOP, T_5 - SLB UPSTREAM OF MSIV, T_6 - SLB DOWNSTREAM OF MSIV

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	CORE HEAT REMOVAL, LATE	COMMENTS
RPS	1. 1/2 MFW Pumps OR 2. 1/3 AFW Pumps OR 3. 1/2 HPSI Pumps and 2 PORVs Open Unit 1 (1 PORV Open Unit 2) (in Once-Through Cooling)	1. Operator Secures RCP within 10 minutes of loss of CCW to RCP with power to RCP still available and PORV Reclose (if opened)	1. 1/2 HPSR and Containment Heat Removal OR 2. (Unit 1 only) SDC or CST makeup or Once-Through-Cooling	1. Secondary steam relief assumed available. 2. AFW to 1 SG sufficient. 3. No RCS Pressure safety relief required. 4. Failure of RCS integrity goes to S_1 tree. 5. Unit 2 CST is sufficient for the mission time of 24 hours.

NOTE: Systems and Major Components listed are for each unit, except where noted.

TABLE 3.1-7
SMALL-SMALL LOCA SUCCESS CRITERIA SUMMARY INFORMATION

INITIATORS: SMALL-SMALL LOCA, S_1 (between 1/2" and 3")

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	CORE HEAT REMOVAL, LATE	COMMENTS
RPS	1. 1/2 HPSI Pump and 1/3 AFW Pump	Lost as a result of initiator	1. 1/2 HPSR and Containment Heat Removal OR 2. 1/2 RHR Shutdown Cooling and Long Term CST Makeup (Unit 1)	1. Failure of RPS transfers to ATWS Tree. 2. For S_1, if AFW and charging flow are available, the operator can depressurize the RCS and go on closed cycle cooling before the RWT is emptied, thereby, eliminating the requirement for recirculation.

NOTE: Systems and Major Components listed are for each unit, except where noted.

TABLE 3.1-8
SMALL LOCA SUCCESS CRITERIA SUMMARY INFORMATION

INITIATORS: SMALL LOCA, S_2 (between 3" and 5")

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	CORE HEAT REMOVAL, LATE	COMMENTS
RPS	1. 1/2 HPSI Pump OR 2. 1/2 LPSI and Depressurization	See Comments	1. 1/2 HPSR and Containment Heat Removal	1. Failure of RPS transfers to ATWS Tree. 2. RCS integrity is lost as a result of the initiator.

NOTE: Systems and Major Components listed are for each unit, except where noted.

TABLE 3.1-9
LARGE LOCA SUCCESS CRITERIA SUMMARY INFORMATION

INITIATORS: LARGE LOCA, A (> 5")

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	CORE HEAT REMOVAL, LATE	COMMENTS
RPS	1. 1/2 LPSI and 3/4 SITs	See Comments	1/2 HPSR and Containment Heat Removal	1. Injection of LPSI into one RCS loop was considered sufficient. 2. Reactor subcriticality is not explicitly required. If RPS fails, the reactor will be maintained subcritical by injection of RWT inventory. 3. RCS integrity is lost as a result of the initiator.

NOTE: Systems and Major Components listed are for each unit, except where noted.

TABLE 3.1-10
SGTR SUCCESS CRITERIA SUMMARY INFORMATION

INITIATORS: SGTR

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	CORE HEAT REMOVAL, LATE	COMMENTS
RPS	1. 1/3 AFW Pumps OR 2. 1/2 MFW Pumps OR 3. Once-Through (Bleed and Feed) Cooling UNIT 1 2/2 PORV AND 1/2 HPSI UNIT 2 1/2 PORV AND 1/2 HPSI	Depressurize RCS to less than SG-RV setpoint and isolate ruptured SG • MSIV • SG Blowdown Line • Main Steam bypass valve • AFW Steam Supply	1. Long Term AFW and SDC OR 2. (S/G not available) HPSI Injection or SDC OR 3. HPSR and Containment Heat Removal	1. Definition of RCS boundary expanded to include SG; hence, SG integrity must be considered too. SG integrity includes SRV recloses.

NOTE: Systems and Major Components listed are for each unit, except where noted.

TABLE 3.1-11
ATWS SUCCESS CRITERIA SUMMARY INFORMATION

EVENT: ATWS

REACTOR SUBCRITICALITY	CORE HEAT REMOVAL, EARLY	RCS INTEGRITY	RCS PRESSURE RELIEF	COMMENTS
Manual insertion of control rods by operator AND Emergency boration using 1 charging pump, taking suction from Boric Acid Tank, discharging through the charging line and remaining at elevated temperature to maintain subcriticality.	1 AFW Pump; 2 SGs	All SRV, PORV must reclose after initial ATWS pressurization	SHORT TERM CORE COOLING 1. Emergency Boration AND 3 SRVs and 2 PORVs (1 PORV Unit 2) AND (If RCS integrity lost) HPSI LONG TERM CORE COOLING 1. Long Term AFW or SDC 2. (If RCS integrity lost) HPSR and Containment Heat Removal	1. Entry into the ATWS tree assumes the RPS failed. 2. AFW must be supplied to 2 of 2 SG. 3. If MTC >-8 pcm/°F pressure relief not possible. 4. MTC criteria apply to high initial power only. 5. Dual Unit ATWS probability is assumed to be negligible.

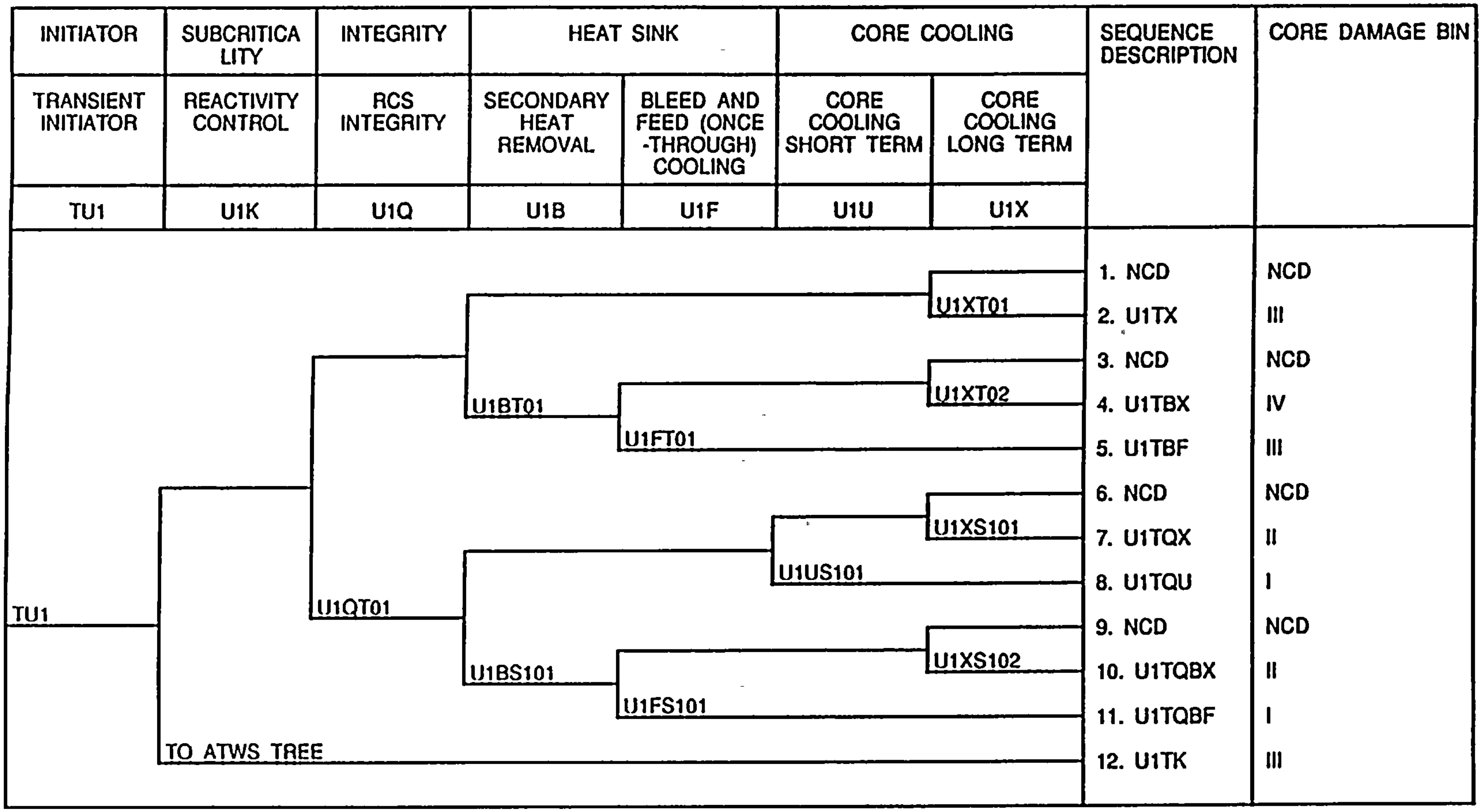
NOTE: Systems and Major Components listed are for each unit, except where noted.

Table 3.1-12

CORE DAMAGE BIN (SEQUENCE CHARACTERISTICS)

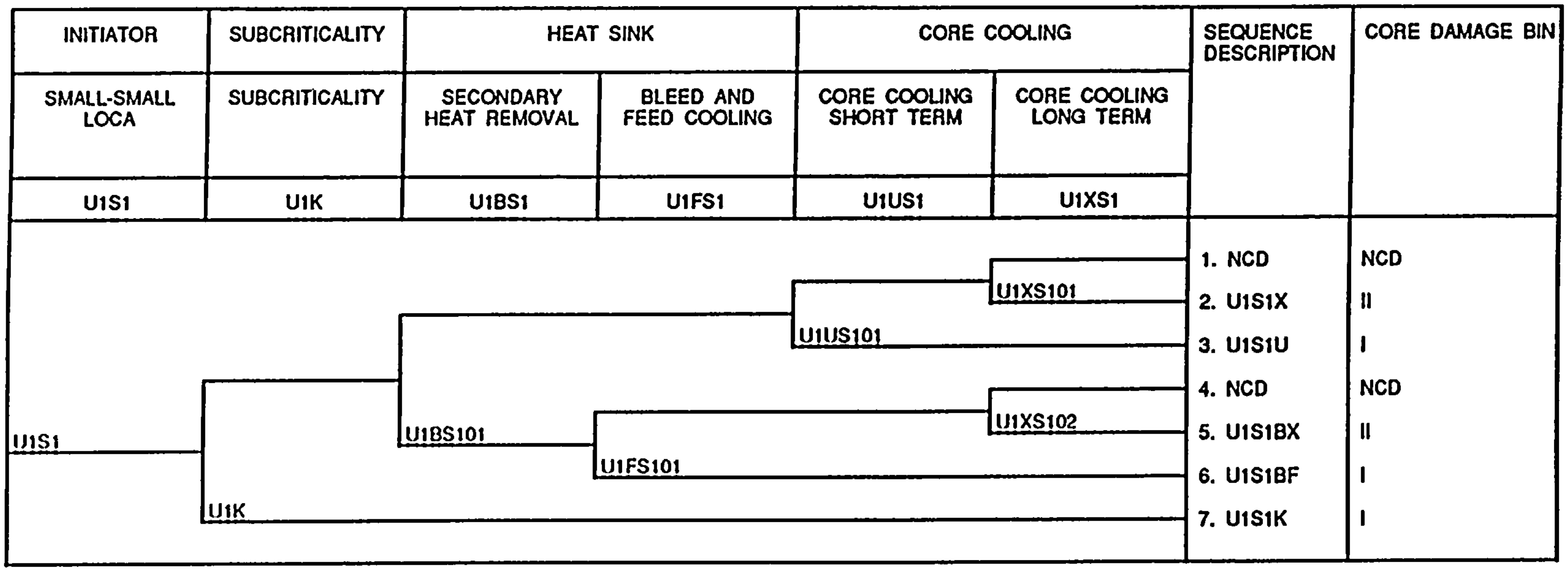
Core Damage Bin[1]	Scenario Description
I	RCS pressure and leakage rates associated with small-break LOCAs, with early melting of the core (e.g., within about 2 hours after the break occurs).
IR[1]	Similar to Bin I except for the radioactivity release through the steam generator.
II	RCS Pressure and leakage rates associated with small-break LOCAs, with late melting of the core (e.g., during recirculation).
IIR[1]	Similar to Bin II except for the radioactivity release path through the steam generator.
III	High RCS pressure and leakage rates associated with boil-off of the reactor coolant through cycling pressurizer relief valves, with early core melting (within about 2 hours).
IV	High RCS pressure and leakage rates associated with boiloff of the reactor through cycling relief valves with late melting of the core.
V	Large rates of leakage from the RCS and low pressures associated with large break LOCAs and failure of coolant injection, resulting in early melting of the core.
VI	Large-break LOCA conditions with failure of coolant recirculation and late melting.
CB (Not used in Level I)	Containment bypass sequences in which the RCS leakage bypasses the containment (e.g., interfacing systems LOCA or steam generator tube rupture).
SBO (Not explicitly delineated in Event Tree)	Station blackout sequences in which the engineered safeguards systems may be recoverable.

[1] Based upon Nuclear Safety Analysis Center, Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3, NSAC-60, June 1984. For the St. Lucie PRA Core Damage Bins, IR and IIR bins are conservatively combined with I and II respectively.



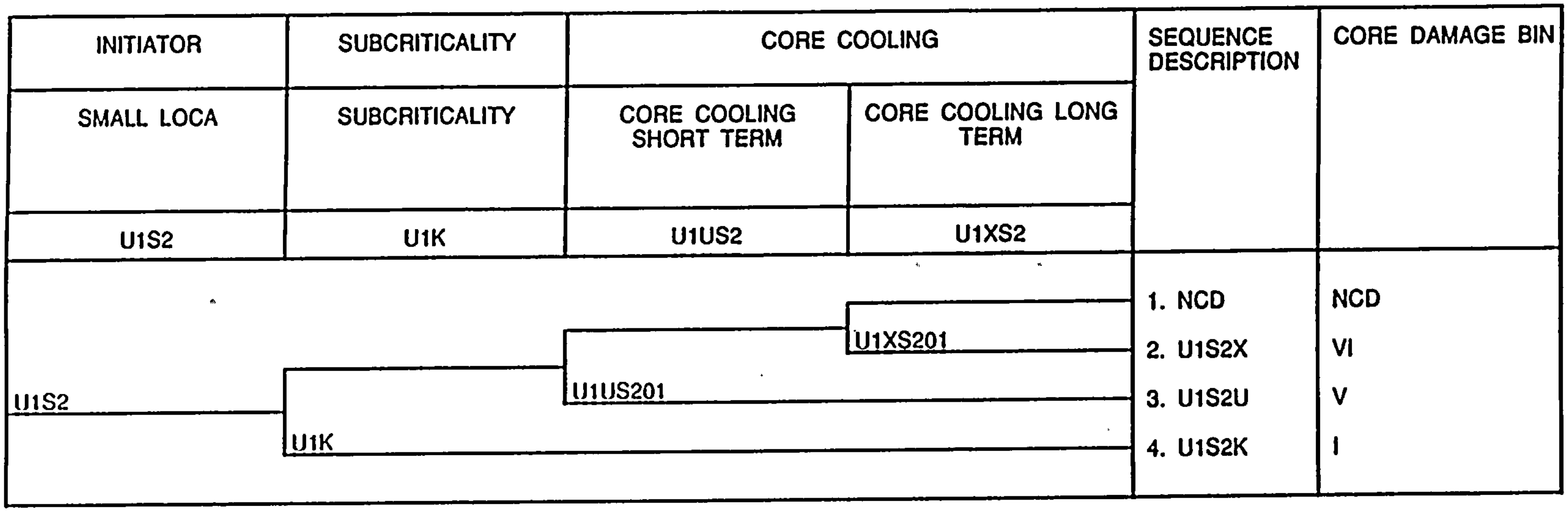
ST LUCIE UNIT 1 TRANSIENT EVENT TREE

Figure 3.1-1 Transient Functional Event Tree for St. Lucie Unit 1



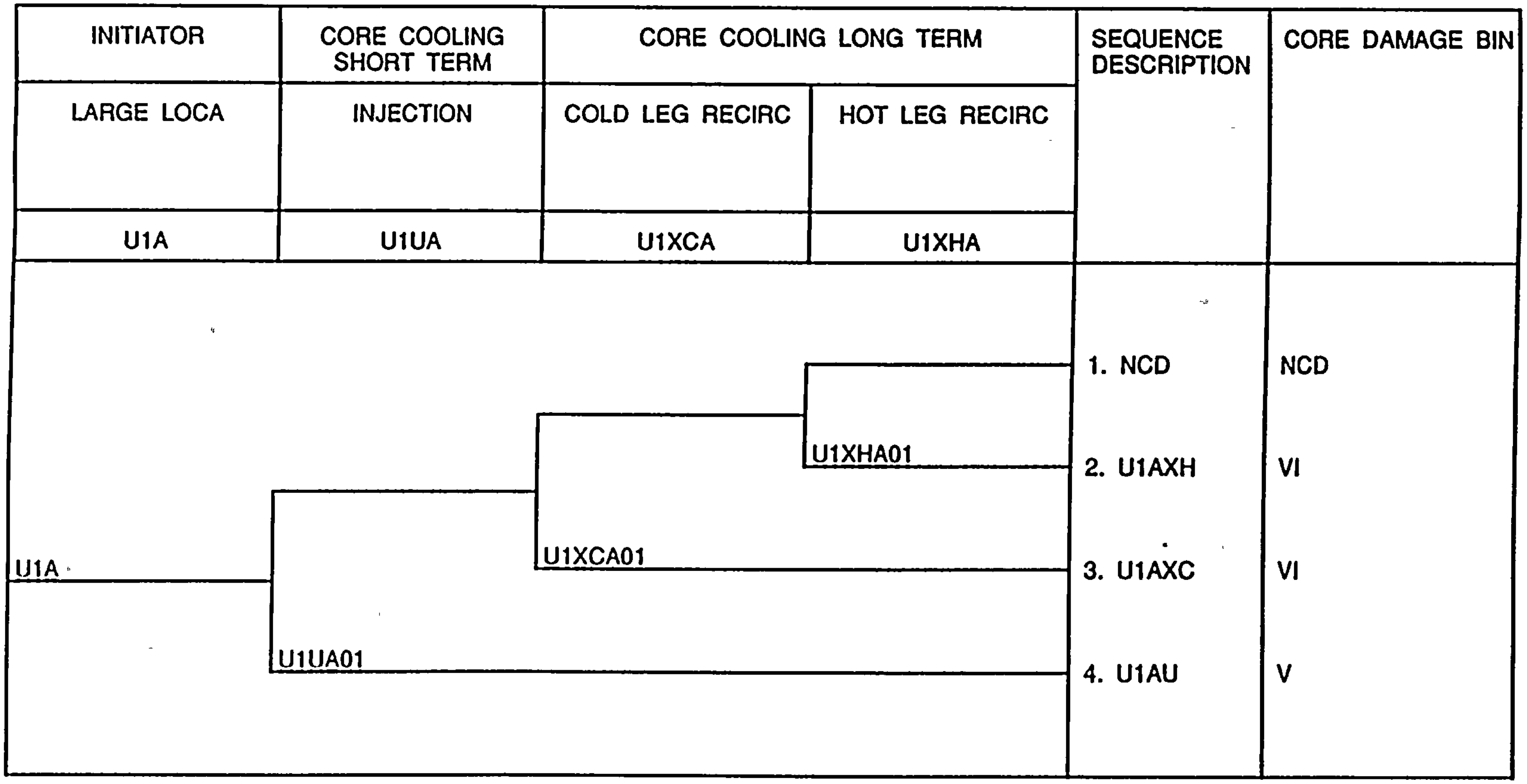
ST LUCIE UNIT 1 S1 LOCA EVENT TREE

Figure 3.1-2 Small-Small LOCA Functional Event Tree for St. Lucie Unit 1



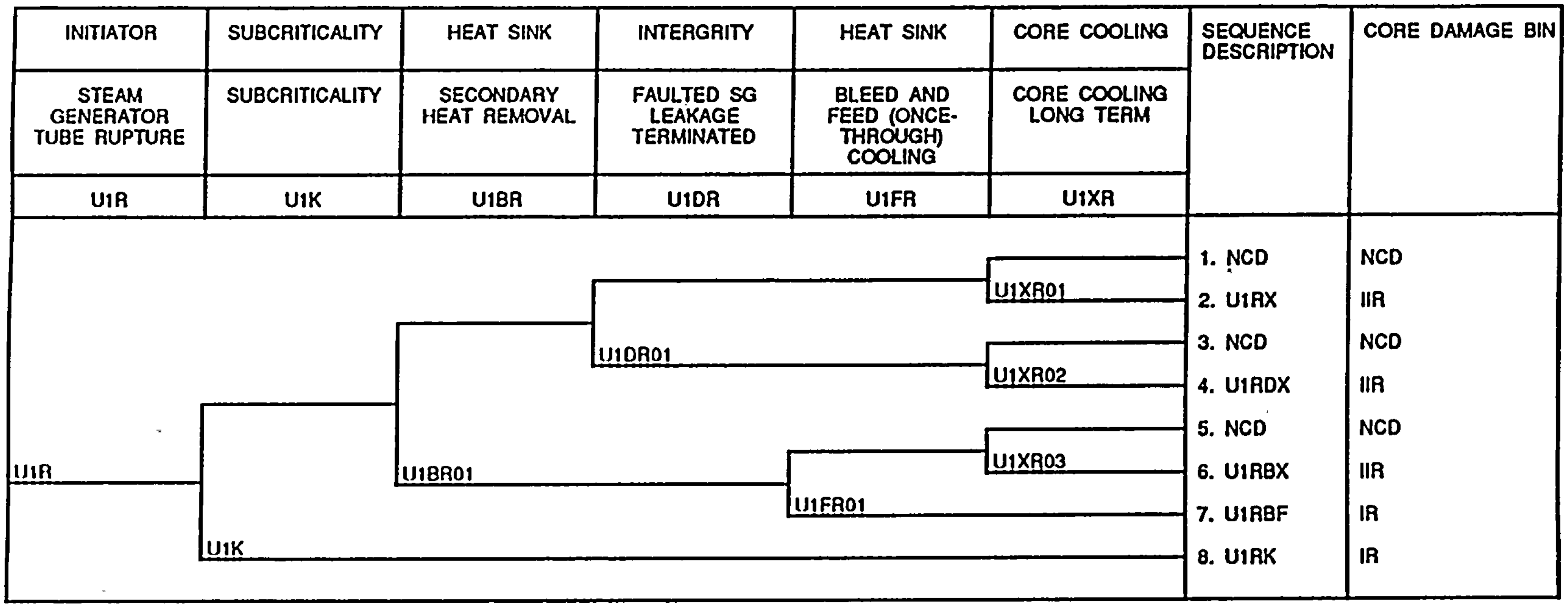
ST LUCIE UNIT 1 S2 LOCA EVENT TREE

Figure 3.1-3 Small LOCA Functional Event Tree for St. Lucie Unit 1



ST LUCIE UNIT 1 LARGE LOCA EVENT TREE

Figure 3.1-4 Large LOCA Functional Event Tree for St. Lucie Unit 1



ST. LUCIE UNIT 1 SGTR EVENT TREE

Figure 3.1-5 SGTR Functional Event Tree for St. Lucie Unit 1

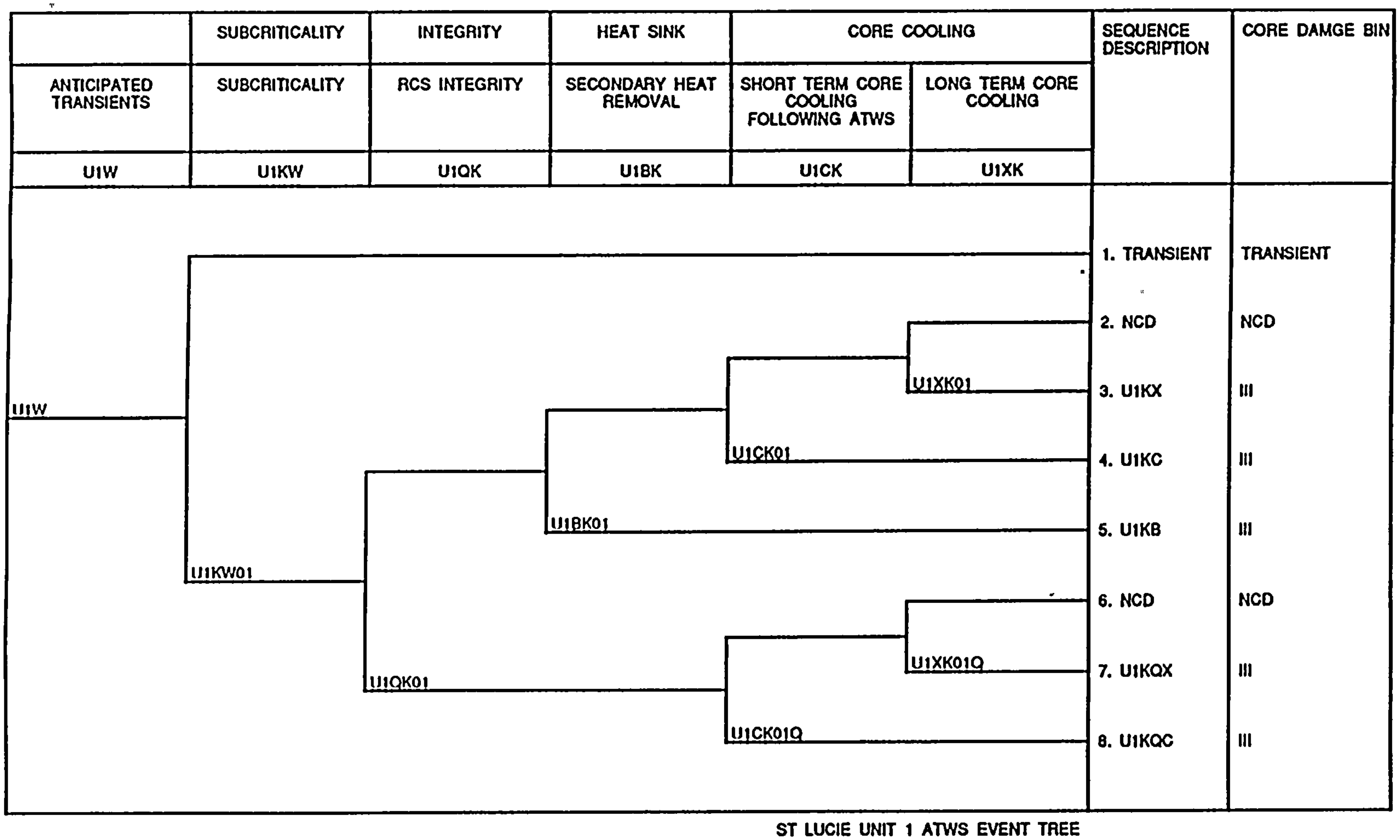


Figure 3.1-6 ATWS Functional Event Tree for St. Lucie Unit 1

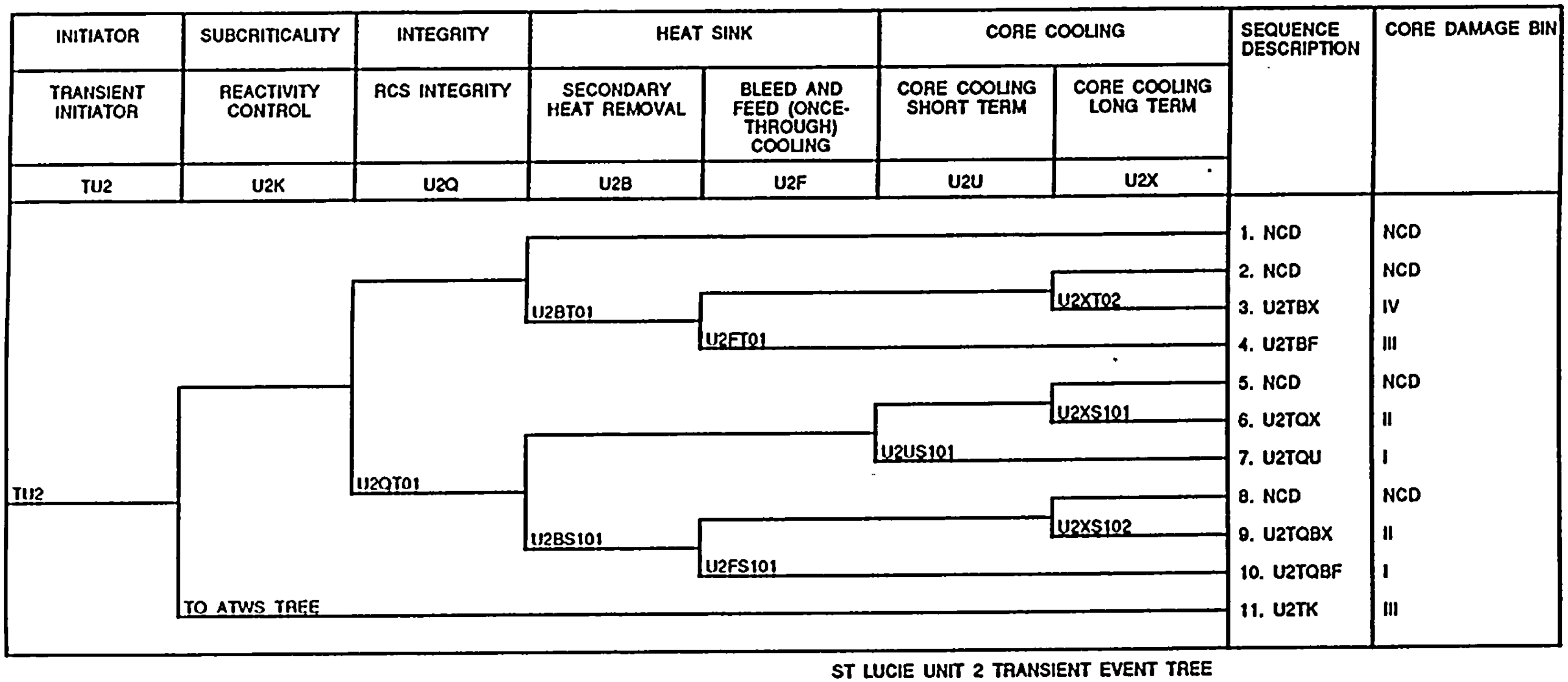
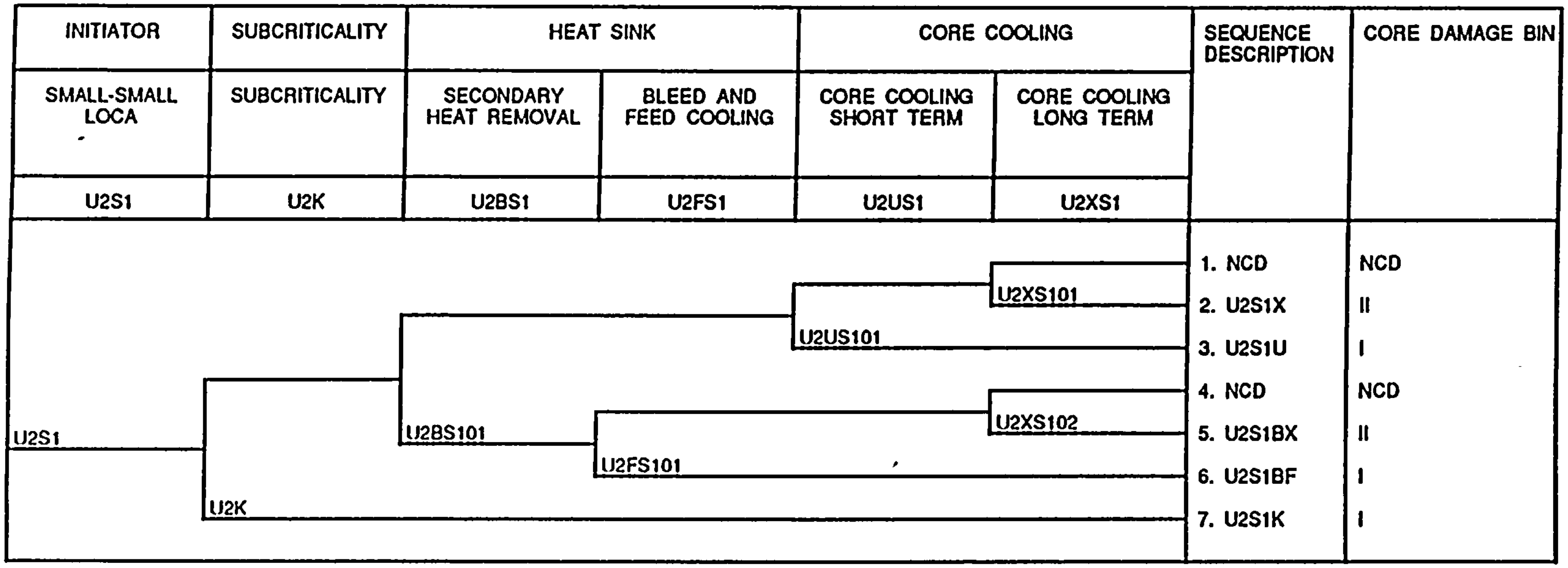
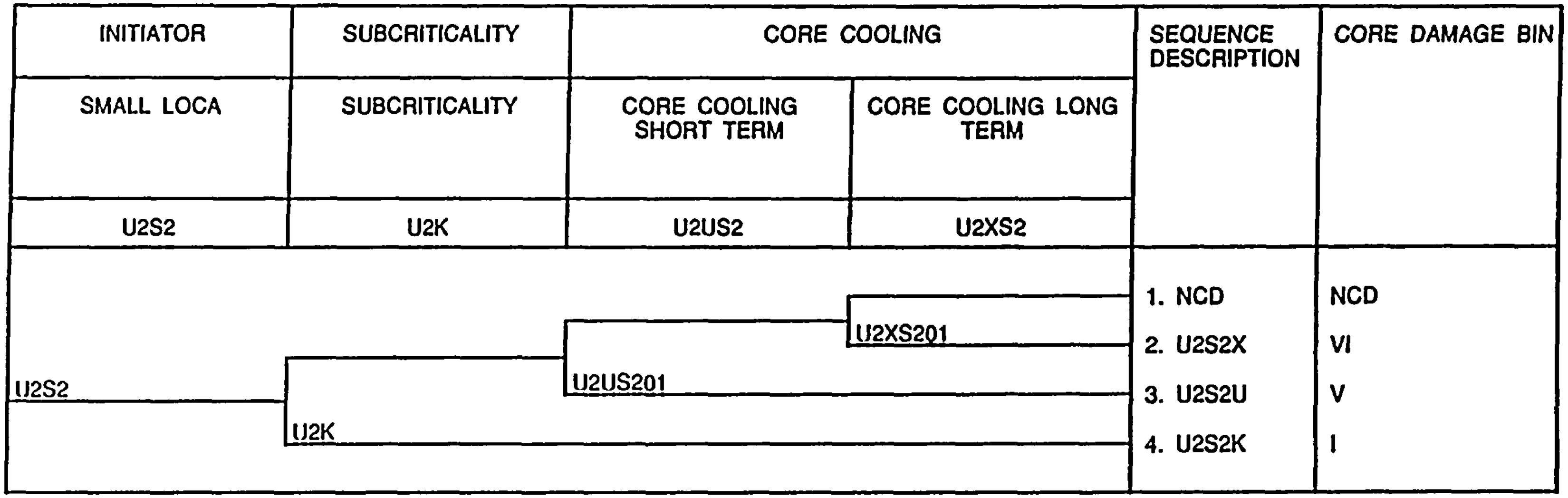


Figure 3.1-7 Transient Functional Event Tree for St. Lucie Unit 2



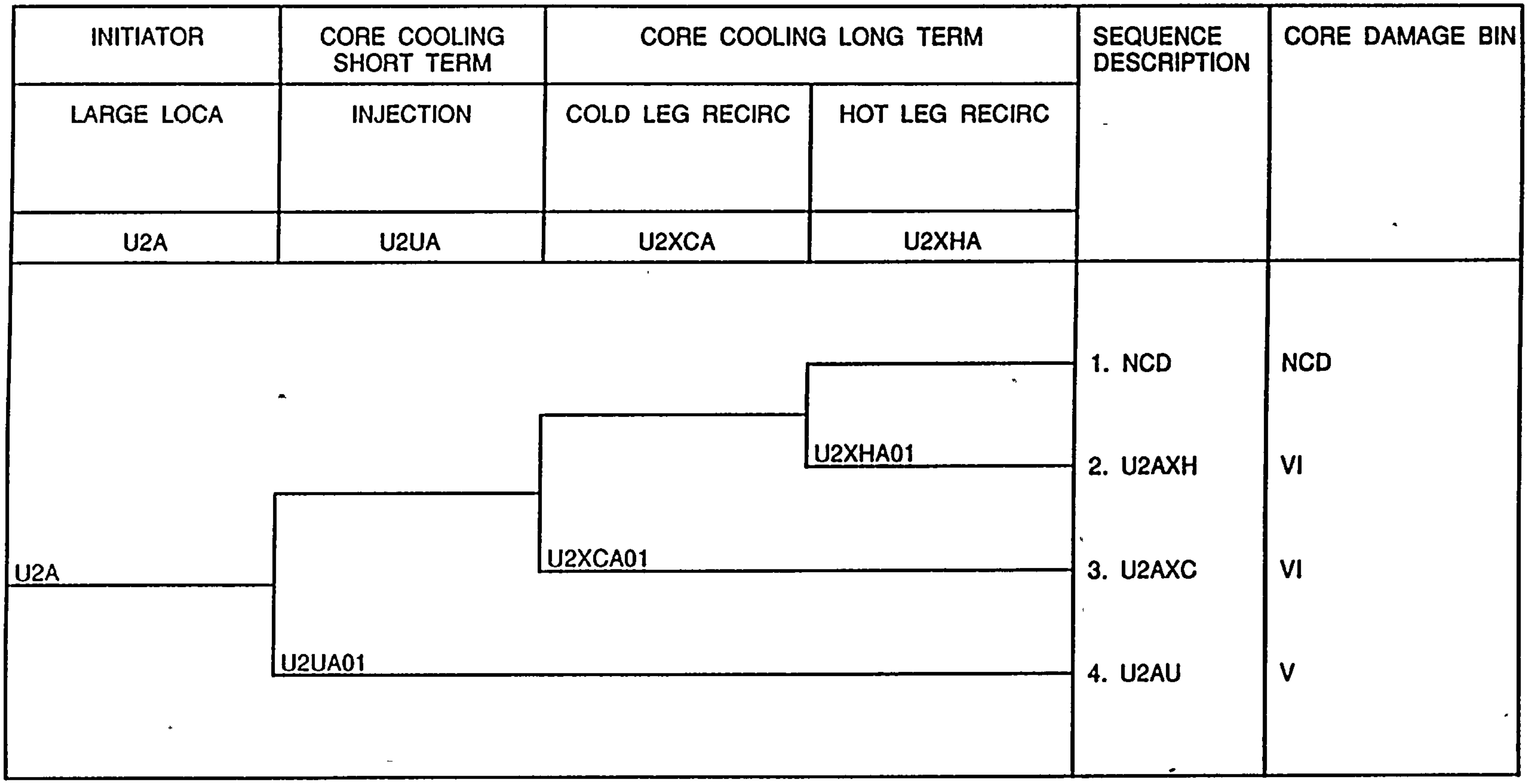
ST LUCIE UNIT 2 S1 LOCA EVENT TREE

Figure 3.1-8 Small-Small LOCA Functional Event Tree for St. Lucie Unit 2



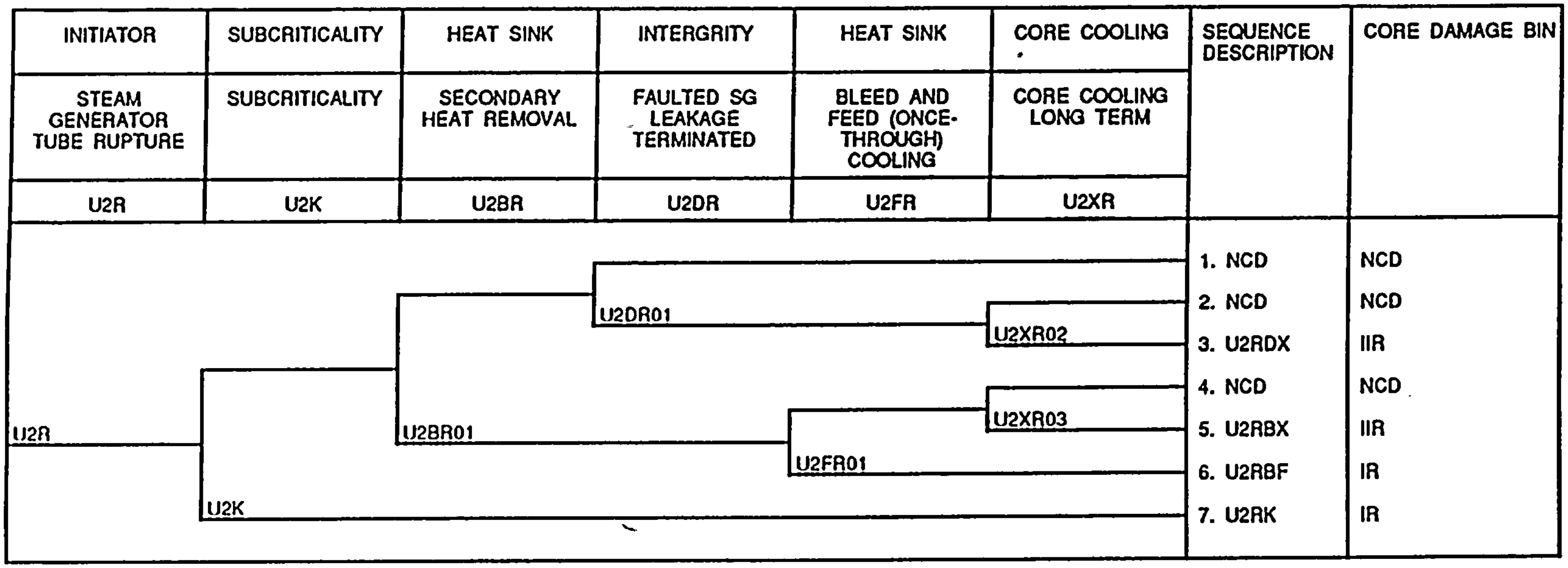
ST LUCIE UNIT 2 S2 LOCA EVENT TREE

Figure 3.1-9 Small LOCA Functional Event Tree for St. Lucie Unit 2



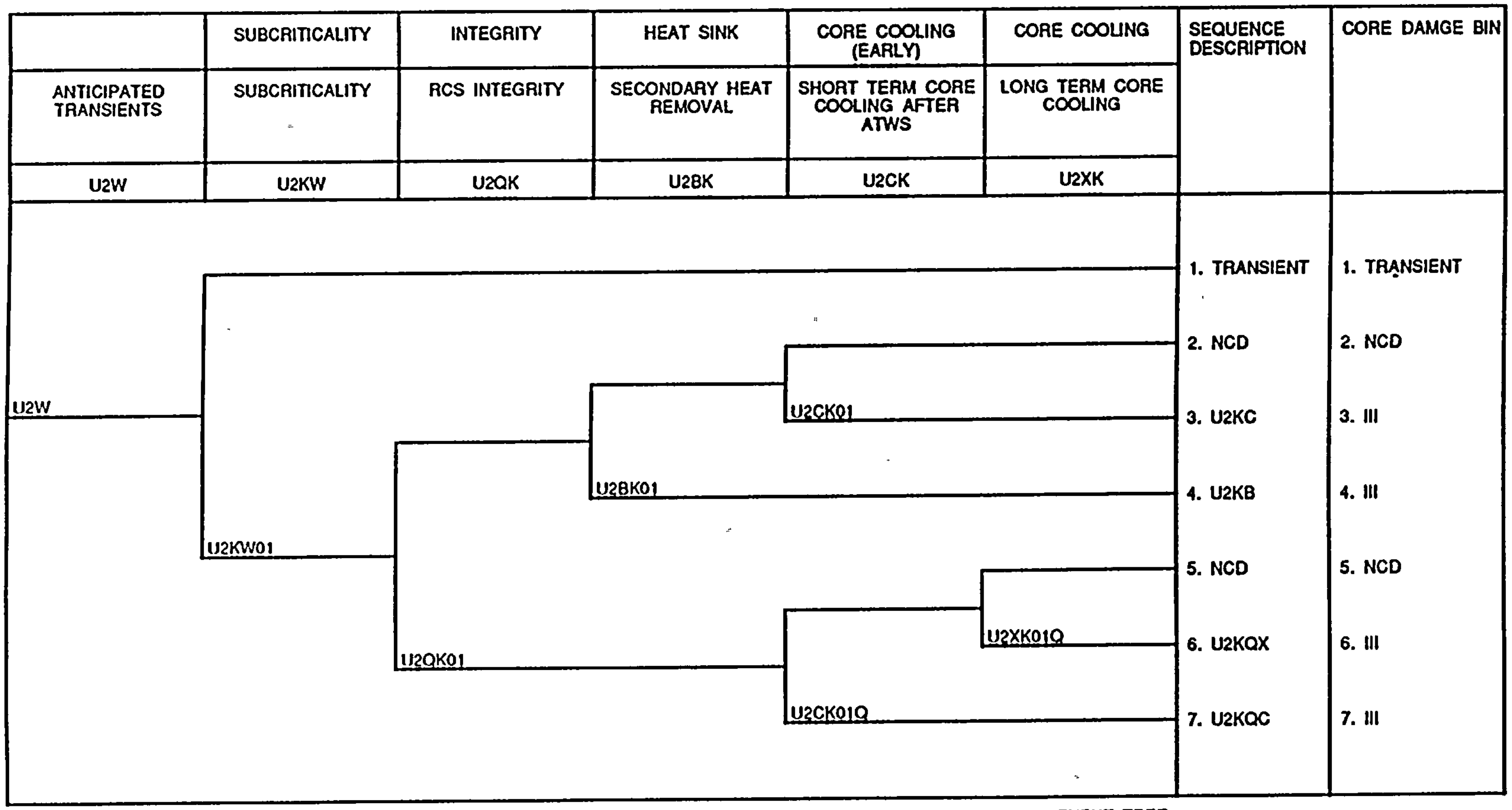
ST LUCIE UNIT 2 LARGE LOCA EVENT TREE

Figure 3.1-10 Large LOCA Functional Event Tree for St. Lucie Unit 2



ST. LUCIE UNIT 2 SGTR EVENT TREE

Figure 3.1-11 SGTR Functional Event Tree for St. Lucie Unit 2



ST LUCIE UNIT 2 ATWS EVENT TREE

Figure 3.1-12 ATWS Functional Event Tree for St. Lucie Unit 2

3.2 Systems Analysis

3.2.1 System Analysis Scope

The St. Lucie systems analysis effort resulted in the development of both Unit 1 and Unit 2 fault tree models that describe the ways in which plant systems can fail and subsequently contribute to accident sequences that lead to core damage and then to a release of radionuclides from the containment. The determination of systems for which modeling was required was based upon the results of the Accident Sequence Analysis task (Section 3.1) and the Containment Performance Analysis task (Section 4). This assessment of the relevant St. Lucie accident sequences culminated with the creation of event trees that identified those front-line systems whose successful operation is required in order to mitigate specific accident scenarios and the criteria for successful operation of each of the front-line systems.

Fault tree top logic was constructed to reflect the logic from the functional event trees and to aid in identifying those systems which provide the necessary safety functions. The support system fault trees were linked to appropriate points in the front-line system models. The top logic fault trees were in turn linked through the event sequence top logic in order to create an integrated model of core damage. Data for all events including equipment failure and human errors was subsequently applied to the fault tree models. The integrated plant model for each Unit was then solved to define the combinations of events that lead to core damage, along with their frequencies. As such, the method used for determining the core damage frequency was the "large fault tree, small event tree" approach. This effectively forced the creation of fault tree models that define in a logical and detailed manner the ways that the applicable system can fail.

System models were constructed for a total of 17 systems per unit. For many systems, alternate "top events" were developed based on the various requirements for successful system operation as defined by the success criteria for different event sequences. In addition, alternate system operating modes occurring during an accident sequence (e.g., injection mode followed by recirculation mode) was addressed in the models. Therefore, each system model may contain several fault trees. Table 3.2-1 lists the system fault tree models and their safety functions, along with the major St. Lucie systems that were addressed in each model.

3.2.2 Systems Analysis Methodology

For the St. Lucie Units 1 and 2 Level 1 PRA, the accident sequence failure models consisted of a set of large fault trees that were linked together based upon the logic specified in the event trees. The fault trees incorporated all significant contributors to the system failure state including front-line component failures, support system component failures, associated test and maintenance failures, and operator errors where appropriate. This approach assured that the system interactions which arise due to functional dependencies between different systems were modeled explicitly.

3.2.2.1 Fault Tree Development

The process involved in the development of the fault tree models was designed to create a consistent and reliable end product. Task procedures were developed for use by the systems analysis team. Procedure content includes basic event naming schemes, style guidelines, system notebook requirements, and the interface between the systems analysis, event sequence, human reliability, data, and quantification efforts. The fault tree analysis began by identifying and defining the system to be modeled. Design drawings and descriptions, system manuals, SAR sections, Technical Specifications, operating procedures, and test and maintenance procedures were collected and reviewed. Each system analyst was then required to become familiar with the design and operation of the assigned system prior to the development of the fault tree.

Fault tree top events for each unit were defined by the Accident Sequence Analysis task and were developed in terms of major system trains or blocks (e.g., combinations of flow paths for the case of a fluid system). This approach simplified any subsequent modification of the fault trees to accommodate modified or added events as they were identified in the Quantification and Containment Performance Analysis tasks. In addition, in order to reduce the effective size of the system models, small portions of fault tree branches which were statistically independent (i.e., the failure events in the branch, including their support system failures, did not appear elsewhere in other branches) were identified. The basic events within each independent branch were then grouped together, or modularized, to effectively form one basic event. This modularization process resulted in a significant reduction in fault tree complexity without loss of structure and permitted solution of the fault trees using a personal computer. This modularization also adds conservatism to the overall quantification, since module probabilities routinely are larger than any of their single component parts.

3.2.2.2 General Modeling Guidelines

Unique system fault trees were developed for both Unit 1 and Unit 2. The top events and success criteria of each fault tree were defined by the Accident Sequence Analysis and the Containment Performance Analysis tasks. Different top events were created based on the system's function, its configuration, and the initiating event. Appropriate references to subtree gates were made rather than redeveloping the same subtree for a number of different top events.

An important aspect of the integrated fault tree approach is the incorporation of support system events into the system models. In order to ensure that this linking was correctly done, a set of system physical and operating boundary conditions were employed and only one analyst modeled the failure modes of any one system. The physical boundary condition was typically defined to be the point on the Piping and Instrument Diagram (P&ID) where valves and other components were no longer identified by the letter designator common to the system being modeled. Interfacing components, however, (such as load circuit breakers, cooling water isolation valves, and solenoid valves controlling the flow of air to air-operated valves) were modeled in the front-line system. The operating boundary condition was typically the configuration of the system at full power operating conditions; that is, the plant was assumed to have been operating at full power prior to occurrence of the initiating event. Connections with other systems were reviewed for their ability to divert sufficient flow from the system such that it would not be able to perform its safety function.

Each unit's fault tree contains those faults which would interrupt, divert, or cause loss of a process flow path, or interrupt required support functions. The level of detail employed in modeling these faults was limited to the extent that appropriate failure data existed. Table 3.2-2 lists the failure modes for each type of component considered during model development. Simplifying assumptions were made when applying the component failure modes to modeling flow path faults. These rules-of-thumb type assumptions were employed because past PRA experience and typical component failure data have shown that certain failure modes (or groups of failure modes) are not significant contributors to flow path failure in the presence of other existing single failures. These assumptions include:

Passive Failures: Multiple pipe breaks were not modeled. Only a single, worst case, pipe rupture event was included.

Flow Diversion: Failures in small lines which divert flow away from a train or component, but which have no significant impact on system function, were excluded from the model. A diversion path of less than 1/3 diameter of the primary flow rate was considered to meet this criterion. Minimum recirculation paths were only modeled if they were required for component operability during the mission time specified for the sequence under consideration.

Redundancy: Events which require numerous component failures in order to occur (i.e., several basic events below an AND gate) were excluded if their combination was of very low probability and there were no dependencies involved.

Other assumptions made that concerned the system operation or success criteria for the system being modeled were included in the System Description Notebook.

Test or maintenance activities which could be performed while at power were included in the fault trees only if they caused the unavailability of a component or set of components. These two events were addressed in terms of groups of components removed for a specific test or maintenance action.

Human errors associated with post-initiator response (front-line human errors) were included as high in the fault tree as practical. Operator actions during recovery were included in the fault trees where possible. Proper inclusion of such actions was verified by the recovery analysis portion of the Quantification Task. Additional post-initiator operator recovery events were also added during the recovery analysis task. Pre-initiator human errors (e.g., failure to restore a train to operation following maintenance) were included in the fault trees as appropriate.

3.2.2.3 Nomenclature

Gate, basic event, module, and house (logic flag) event names were developed using a ten (10) character coding scheme which permitted the models to be easily reduced and quantified using CAF386(FTAP). The application of this scheme is described below for each event type.

1. Gates

The gates labeling scheme is designed for consistency and traceability across the systems. Each gate is labeled as XUaaaaaaaa where X is a one-letter code denoting the system (see Table 3.2-3), U is the one-digit number denoting the unit (except for electrical gates - this number designates the bus type), and aaaaaaaa is an alphanumeric code of up to 8 characters describing the gate or indicating its sequence in the fault tree.

2. Basic Events

Basic events were named as follows: XYYAUddddd, where X is the system designator (see Table 3.2-3), YY is the component type code (see Table 3.2-2), A designates the failure mode (see Table 3.2-2), U is the unit designator, and ddddd is descriptive nomenclature (e.g., component tag number).

In general, basic event system designators correspond to plant systems, even though they may appear in a fault tree which uses a different system designator in the gate names. For example, failure of the MFW flow regulation valves may be affected by the transfer close of a manual valve in the Instrument Air System. This event was designated with an "I" system designator event although it appears in the MFW system fault tree (system designator "F"). An exception to this is the naming of circuit breaker basic events. System designators for these events follow that of the load controlled by the circuit breaker. For example, failure of the MFW pumps would occur with failure of AC breakers with basic event system designator "F".

3. Modular Events

Modular events were designated XMMUdddddd, where X is the system designator, "MM" identifies the basic event as a module, U is the unit designator, and dddddd is a unique numeric designator to describe the module contents.

4. Test and Maintenance Events

Test and maintenance events were designated XTMUdddddd, where X is the system designator, "TM" identifies the basic event as a test and maintenance unavailability event, and dddddd is a unique alphanumeric designator to describe the components involved.

5. House Events

House events (logic flags) were designated ZZnnnn, where "ZZ" identifies the event as a flag, and nnnn is a unique identifier.

6. Initiating Events

The nomenclature used in describing initiating events is described in Section 3.1.

7. Human Failure Events

Human (pre-initiator) failure events were named XHFMUddddd, where X is the system designator, "HF" identifies that the event involves human failures, M is a timing designator ("L" for pre-initiator events and "D" for post-initiator events), U is the unit designator, and ddddd is descriptive nomenclature.

8. Common-Cause Failure Events

Common-cause failure (CCF) events (i.e., beta factors) were named XYYACCFBF$, where X is the system designator, YY is the component type code, A is the failure mode, and "CCFBF$" denotes that the event involves a CCF. CCF events were incorporated into the fault trees using modular events. CCF events were not mixed with other types of failures in the same module nor were several CCF component groups mixed in the same module. Each CCF modular event consists of one two-event cut set, the first event representing the failure of one member of the CCF group and the second event representing the common cause beta factor.

3.2.2.4 Modularization

Detailed fault trees were modularized to aid in the reduction of the size of the linked fault trees (i.e., plant model) for quantification purposes. Modularization is the process of combining individual basic events into one event. It does not eliminate dependencies, high order fault combinations, or low probability events. Only statistically independent basic events were combined into an equivalent modular event. In addition, only "single-level" "OR" gates were typically modularized.

To ensure that the logic model was consistently maintained during the modularization process and to facilitate recovery analysis, the following were NOT included within modules:

1. References to other gates,

2. House events (logic flags),

3. Basic events used more than once in the fault tree,

4. Basic events which could be used in another fault tree (e.g., a valve which could be part of the failure logic in two fault trees). Breakers associated with a single component could be modularized with that component; however, breakers on a DC or 120V AC panel may not be modularized because they often power multiple components through fused distribution panels,

5. Test and maintenance events,

6. Faults which cannot occur at the same time as another similar fault and which must be identifiable during the cutset review process, and

7. Human failure events.

3.2.2.5 System Analysis Results

Results of the system analysis effort yielded:

- a simplified P&ID for each system model;
- detailed modularized fault trees for each top event required to quantify the plant model;
- cutset files listing the basic events contained in each module;
- failure probabilities for the modular and basic events contained in each fault tree;
- list of mutually exclusive system events (events, such as test and maintenance, which cannot occur concurrently); and a
- identification of modeled system dependencies.

Simplified system P&IDs/one-line diagrams and brief discussions for each system are provided in Appendix B. A summary of modeled system dependencies, at the train level, is provided in Table 3.2-5.

The system models are documented in system description notebooks and are maintained for every model listed in Table 3.2-1. Each notebook also contains a comprehensive collection of system information. Table 3.2-4 describes the revision 0 system description notebook contents.

3.2.3 Dependent Failure Treatment

Dependent failure events of the following types were considered in each system fault tree model:

1. Inter-system dependencies,
2. Human failures,
3. Initiating events, and
4. Common-cause failures.

Failures due to inter-system dependencies (e.g., support systems) were modeled explicitly in the fault trees. Human failure events which defeat an entire system or group of systems were modeled at the top levels of the fault tree. Failures due to the effects of an initiating event were modeled by including the initiating event in the fault trees. Failures of common-cause groups were modeled explicitly in the fault trees using modular events which represent failures of common-cause failure components groups within the same system. The definition of a common-cause failure group was based on consideration of factors such as:

1. component type (e.g., motor-operated valve, turbine-driven pump), including any special design or construction features,
2. component use (e.g., isolation, sensing),
3. component manufacturer,
4. component internal conditions (e.g., pressure, temperature),
5. component external environment (e.g., humidity, temperature),
6. component location,
7. component initial condition (e.g., normally open, energized) and operating characteristics (e.g., running, open),
8. component testing procedures (e.g., test interval, lineup), and
9. component maintenance procedures (e.g., preventive maintenance frequency).

Common-cause failure beta factors were determined for the following component groups:

1. Diesel Generators
2. Pumps
3. Motor-Operated Valves
4. Batteries
5. Fans
6. Reactor Trip Breakers
7. Air Operated Valves
8. Check Valves
9. Safety Relief Valves

The credibility of these groups relative to their failure modes was evaluated during the Data Analysis task (Section 3.3) using the results of EPRI research (Reference 3.2-2).

3.2.4 Section 3.2 References

3.2-1 USNRC, "Procedures for Treating Common-Cause Failures in Safety and Reliability Studies", NUREG/CR-4780 (EPRI NP-5613), January 1983.

3.2-2 EPRI, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events", EPRI NP-3697, June 1985.

Table 3.2-1

St. Lucie PRA System Models

System Model Included	Safety Functions	St. Lucie Plant Systems
	LEVEL 1 PRA ANALYSIS	
• AC Electric Power	• Equipment motive power • Control Power	• AC Power • Emergency Diesel Generators
• Component Cooling Water (CCW) System	• RCS equipment cooling	• Component Cooling Water (CCW)
• DC Electric Power	• Equipment control power • Instrumentation power	• DC Power
• Auxiliary Feedwater (AFW)	• Backup steam generator water supply	• Auxiliary Feedwater (AFW)
• Engineered Safety Features Actuation System (ESFAS)	• Signals for systems (HPSI, LPSI/SDC, CSS, CIS, AFW, PCS, RCS, ICW, EDGs) actuated or isolated on an ESFAS, RAS, SIAS, CIAS, AFAS, CSAS, or MSIS condition	• Engineered Safety Features Actuation System (ESFAS)
• High Pressure Safety Injection/Makeup (HPSI)	• RCS makeup for small and large LOCAs, or transients if RCS not intact • RCS makeup after a SGTR • Feed and Bleed cooling	• High Pressure Safety Injection (HPSI)
• Instrument Air (IA)	• Motive power for various systems including main steam, HVAC, CCW and ICW systems	• Instrument Air (IA)
• Low Pressure Safety Injection (LPSI)	• RCS makeup for large LOCAs • Shutdown cooling	• Low Pressure Safety Injection (LPSI)
• Heating, Ventilation, and Air Conditioning (HVAC)	• Maintain ambient temperature control to permit proper functioning of equipment	• Electrical equipment room ventilation, ECCS area ventilation, Turbine SWGR room ventilation, Intake structure ventilation
• Power Conversion System (PCS)	• Steam dump and bypass control after a transient or SGTR • RCS heat removal after a transient, small LOCA, or SGTR	• Feedwater • Condensate • Main Steam • Steam Dump and Bypass Control • Feedwater Control • Circulating Water • Atmospheric Dump

Table 3.2-1

St. Lucie PRA System Models

System Model Included	Safety Functions	St. Lucie Plant Systems
• Primary Pressure Control (PPC)	• RCS pressure control and pressure boundary integrity after a transient, or SGTR • Feed and Bleed cooling	• Pressurizer Code Safety Valves • PORVs • Pressurizer Spray Valves
• Safety Injection Tanks (SIT)	• RCS makeup after a large LOCA	• Safety Injection Tanks (SIT)
• Intake Cooling Water (ICW)	• Provide cooling for CCW and turbine cooling water	• Intake Cooling Water (ICW)
• Containment Spray (CSS)	• Containment heat removal	• Containment Spray (CSS)
• Containment Cooling	• Containment heat removal	• Containment Fan Coolers • Containment Spray (CSS)
• Chemical & Volume Control System (CVCS)	• Provide a means of injecting boric acid into the RCS to effect an emergency shutdown • Charging flow to RCS during LOCA events • Provide Auxiliary Pressurizer for RCS pressure control during shutdown to allow pressurizer cooling	• Charging Pumps • Volume Control Tank • Boric Acid Makeup Tanks • Boric Acid Pumps
	<u>LEVEL 2 PRA ANALYSIS</u>	
• Containment Isolation (CIS)	• Forms barrier between post-trip containment atmosphere and external environment	• Containment Isolation (CIS)
• Containment Spray (CSS)	• Containment heat removal	• Containment Spray (CSS)
• Containment Cooling	• Containment heat removal	• Containment Fan Coolers • Containment Spray (CSS)

Table 3.2-2

COMPONENT TYPE AND FAILURE MODE DESIGNATOR

EVENT TYPE CODE/ FAILURE MODE	DESCRIPTION	UNIT*
AC A	Air cooling unit fails to start	N
AC F	Air cooling unit fails to run	H
AD F	Air dryer fails to deliver flow	H
AF F	Air filter fails to deliver flow	H
AM A	Air compressor fails to start	N
AM F	Air compressor fails to run	H
AR F	Air receiver local faults	H
AV C	Air-operated valve fails to close	N
AV K	Air-operated valve transfers closed	H
AV N	Air-operated valves fail to open	N
AV R	Air-operated valve transfers open	H
B1 F	>4 kV bus fault	H
B2 F	<4 kV bus fault	H
B4 F	120 V bus fault	H
BC F	Batter charger - no output	H
BD F	DC bus fault	H
BI F	Bistable, spurious operation	H
BI N	Bistable fails to operate on demand	N
BT D	Battery - no output (demand)	N
BT F	Battery - no output (hourly)	H
CB D	AC breaker fails to operate	N
CB K	AC breaker transfers closed	H
CB R	AC breaker transfers open	H
CD D	DC breaker fails to trip (over current)	N
CD R	DC breaker transfers open	H
CF R	Fuse fails open	H
CI R	DC interrupter transfers open	H
CS R	DC disconnect switch transfers open	H
CT C	Contact fails to open	N
CT K	Contact transfers closed	H
CT N	Contact fails to close	N
CT R	Contact transfers open	H
CV C	Check valve fails to close	N
CV G	Check valve rupture	H
CV K	Check valve transfers closed	H
CV N	Check valve fails to open	N
CV R	Check valve transfers open	H
DG A	Diesel generator fails to start	N
DG F	Diesel generator fails to run	H
EI D	E/I converter fails to respond	H
EP D	E/P converter fails to respond	H
FC F	Dropout register fails to fall	N
FE F	Flow element fails	H
FE H	Flow element fails high	H
FE L	Flow element fails low	H
FS D	Flow switch fails to respond	N

* H refers to an hourly failure rate; N is a demand failure rate.

Table 3.2-2

COMPONENT TYPE AND FAILURE MODE DESIGNATOR

EVENT TYPE CODE/ FAILURE MODE	DESCRIPTION	UNIT*
FS H	Flow switch fails high	H
FS L	Flow switch fails low	H
FT D	Flow transmitter fails to respond	H
FT H	Flow transmitter fails high	H
FT L	Flow transmitter fails low	H
HR F	Hydrogen recombiner fails to recombine	H
HX F	Heat exchanger cooling capability fails	H
HX J	Heat exchanger tube rupture	H
HX P	Heat exchanger plugs	H
IN F	Inverter - no output	H
IR F	Regulating rectifier - no output	H
IV F	Static voltage regulator - no output	H
LC D	Logic circuit fails to generate signal	H
LC F	Logic circuit - false output	H
LS D	Level switch fails to respond	N
LS H	Level switch fails high	H
LS L	Level switch fails low	H
LT D	Level transmitter fails to respond	N
LT H	Level transmitter fails high	H
LT L	Level transmitter fails low	H
LY D	Signal processor module fails to respond	H
LY L	Signal processor module fails low	H
MC C	Backflow damper fails to close	N
MC K	Damper transfers closed	H
MC N	Backflow damper fails to open	N
MC R	Damper transfers open	H
MF A	Motor-driven fan fails to start	N
MF F	Motor-driven fan fails to run	H
MP A	Motor-driven pump fails to start	N
MP F	Motor-driven pump fails to run	H
MV C	Motor-operated valve fails to close	N
MV G	Motor-operated valve ruptures	H
MV K	Motor-operated valve transfers closed	H
MV N	Motor-operated valve fails to open	N
MV R	Motor-operated valve transfers open	H
NZ P	Spray nozzles plugged	H
PH F	Conditional event, loss of function	N
PP J	Piping rupture	H
PP P	Piping plugged	H
PS D	Pressure switch fails to respond	N
PS H	Pressure switch fails high	H
PS L	Pressure switch fails low	H
PT D	Pressure transmitter fails to respond	H
PT H	Pressure transmitter fails high	H
PT L	Pressure transmitter fails low	H
PV R	Pressure control valve transfers open	H

* H refers to an hourly failure rate; N is a demand failure rate.

Table 3.2-2

COMPONENT TYPE AND FAILURE MODE DESIGNATOR

EVENT TYPE CODE/ FAILURE MODE	DESCRIPTION	UNIT*
PX F	Power supply - no output	H
RA F	Radiation element fails to respond	H
RE B	Relay fails to deenergize	N
RE E	Relay fails to energize	N
RE K	Relay - spurious transfer (to energized)	H
RE R	Relay - spurious transfer (to deenergized)	H
RV C	Relief valve fails to close	N
RV N	Relief valve fails to open	N
RV R	Relief valve - spurious open	H
RY N	PSV or S/G safety valve fails to open	N
RY Q	PSV or S/G safety valve fails to reseat after liquid relief	N
RY T	PSV or S/G safety valve fails to reseat after steam relief	N
RZ N	PORV fails to open	N
RZ Q	PORV fails to reseat after liquid relief	N
RZ T	PORV fails to reseat after steam relief	N
SC D	Stop-check valve fails to respond	N
SC G	Stop-check valve leakage	H
SC N	Stop-check valve fails to open	N
SM P	Containment sump plugged	N
ST A	Motor-driven strainer fails to start	N
ST F	Motor-driven strainer fails to run	H
SV C	Solenoid valve fails to close	N
SV K	Solenoid valve transfers closed	H
SV N	Solenoid valve fails to open	N
SV R	Solenoid valve transfers open	H
SW C	Hand switch fails to close	N
SW K	Hand switch transfers closed	H
SW N	Hand switch fails to open	N
SW R	Hand switch transfers open	H
SX N	Speed switch fails to open	N
SZ C	Valve position switch fails to close	N
SZ K	Valve position switch transfers closed	H
SZ R	Valve position transfers open	H
T1 F	kV transformers fault	H
T6 F	480V-240V transformer fault	H
TK G	Tank leakage	H
TK J	Tank rupture	H
TP A	Turbine-driven pump fails to start	N
TP F	Turbine-driven pump fails to run	H
TS B	Temperature switch fails to open	N
TS E	Temperature switch fails to close	N
TS H	Temperature switch fails high	H
TS L	Temperature switch fails low	H
TT D	Temperature transmitter fails to respond	H
TT H	Temperature transmitter fails high	H
TT L	Temperature transmitter fails low	H

* H refers to an hourly failure rate; N is a demand failure rate.

Table 3.2-2

COMPONENT TYPE AND FAILURE MODE DESIGNATOR

EVENT TYPE CODE/ FAILURE MODE	DESCRIPTION	UNIT*
TV D	Temperature indicating controller fails to respond	H
TV H	Temperature indicating controller fails high	H
TV L	Temperature indicating controller fails low	H
XV C	Manual valve fails to close	N
XV K	Manual valve transfers closed	H
XV N	Manual valve fails to open	N
XV R	Manual valve transfers open	H

* H refers to an hourly failure rate: N is a demand failure rate.

Table 3.2-3

SYSTEM DESIGNATOR

Fault Tree System Code	
A	Auxiliary Feedwater System (AFW)
B	Safety Injection Tanks (SIT)
C	Component Cooling Water System (CCW)
D	Containment Heat Removal System (CHRS)
E	Electric Power System - AC and DC (EPS)
F	Power Conversion System (PCS)
G	High Pressure Safety Injection System (HPSI)
H	Instrument Air System (IA)
I	Heating and Ventilation (HVAC)
J	Low Pressure Safety Injection and Shutdown Cooling System (LPSI/SDC)
K	Containment Isolation System (CIS)
L	Containment Spray System (CS)
M	Chemical and Volume Control System (CVCS)
N	Reactor Protection System/Engineered Safety Features Actuation System (RPS/ESFAS)
O	Primary Pressure Control System (PPC)
P	Interfacing LOCA/Containment Bypass System (V)
Q	Intake Cooling Water System (ICW)

TABLE 3.2-4

SYSTEM DESCRIPTION NOTEBOOK TABLE OF CONTENTS

TABLE 3.2-4

SYSTEM DESCRIPTION NOTEBOOK TABLE OF CONTENTS

TABLE 3.2-5 ST. LUCIE SYSTEM LEVEL DEPENDENCY MATRIX (Sheet 1 of 2)

			AC Power Train				DC Power Train				ESFAS						CCW Train		ICW Train		IA	HVAC
			NSR	A	B	AB	NSR	A	B	AB	SIAS	CSAS	AFAS	CIAS	MSIS	RAS	A	B	A	B		
FRONT LINE SYSTEMS CORE DAMAGE MITIGATION	SIT																					
	AFW	A		X									X									
		B			X								X									
		C								X			X									
	HPSI	A		X				X(2)			X					X	X					X
		B			X				X(2)		X					X		X				X
	LPSI	A		X				X(1)			X					X	X				X	X
		B			X				X(1)		X					X		X			X	X
	SDC	A		X													X					X
		B			X													X				X
	PCS	A	X	X				X			X(1)		X(2)	X	X						X	X
		B	X		X				X		X(1)		X(2)	X	X						X	X
	CHRS Fan	A		X							X						X					
	CHRS Fan	B		X							X						X					
	CHRS Fan	C			X						X							X				
	CHRS Fan	D			X						X							X				
	CSS	A		X				X				X					X				X	X
		B			X				X			X						X			X	X
	CVCS	A		X				X			X										X	
		B			X				X		X										X	
		C(3)				X					X											
	PPC (PORVs)	A		X				X														
		B			X				X													

(1) UNIT 1 ONLY
(2) UNIT 2 ONLY
(3) "C" CHARGING PUMP

TABLE 3.2-5 ST. LUCIE SYSTEM LEVEL DEPENDENCY MATRIX (Sheet 2 of 2)

				AC Power Train NSR	AC Power Train A	AC Power Train B	AC Power Train AB	DC Power Train NSR	DC Power Train A	DC Power Train B	DC Power Train AB	ESFAS SIAS	ESFAS CSAS	ESFAS AFAS	ESFAS CIAS	ESFAS MSIS	ESFAS RAS	CCW Train A	CCW Train B	ICW Train A	ICW Train B	IA	HVAC
CONTAINMENT DAMAGE MITIGATION	CIS		A		X(1)							X			X	X							
			B									X			X	X							
	CHRS	Fan	A		X							X						X					
		Fan	B		X							X						X					
		Fan	C			X						X							X				
		Fan	D			X						X							X				
	CSS		A		X				X				X					X				X	X
			B			X				X			X						X			X	X
SUPPOT SYSTEMS	AC Power		NSR	X	X	X			X	X													X
			A	X	X				X			X											X
			B	X		X				X		X											X
			AB		X	X					X	X											
	DC Power		NSR	X																			
			A		X																		X(2)
			B			X																	X
			AB						X	X													
	IA			X	X	X	X								X			X(1)					
	CCW		A		X							X								X			
			B			X						X									X		
	ICW		A		X							X											X(2)
			B			X						X											X(2)
	HVAC			X	X	X			X	X		X			X								

(1) UNIT 1 ONLY
(2) UNIT 2 ONLY
(3) "C" CHARGING PUMP

3.3 Reliability Data Analysis

Numerical data is required to support many aspects of the St. Lucie PRA. This section documents the generic and plant specific data values used for quantification of the St. Lucie Units 1 and 2 core damage frequency.

For the St. Lucie PRA, the following types of data requirements exist:

a) Component Failure Rates - Demand or hourly failure rates for the components modeled in the System Fault Trees,

b) Test and Maintenance (T/M) Unavailability - Fraction of time a component is not able to perform its function due to test and/or maintenance activities,

c) Common Cause Data - "Beta Factors" are provided to quantify the common cause events modeled in the System Fault Trees,

d) Initiating Event Frequencies - Frequencies (events per reactor year) of transient and accident categories.

This report section only summarizes the information and data that was used to generate the data set and discusses the data analysis techniques. The detailed calculations are recorded in the data analysis task report [Ref. 3.3-1].

3.3.1 Component Reliability Parameters

Most basic events in the St. Lucie risk models represent random failures of system equipment following a reactor trip. Calculation of the probabilities associated with these basic events, therefore, requires an extensive set of component level reliability parameters. The term "reliability parameters" refers to both failure rates and failure-on-demand probabilities. Component reliability parameters are assumed to be constants in this PRA; that is, effects such as infant mortality, wear-out, or other exposure dependencies are not addressed.

There are two sources of component reliability parameter information: (1) generic data and (2) plant specific data. The following sections discuss the collection and analysis of generic and plant specific data, and the process that was used to compile these data sources into the St. Lucie PRA data set.

3.3.1.1 Generic Data

Reliability parameter estimates based on generic data were used for component types and failure modes when plant specific data could not be obtained. The following process was used to develop the generic data set:

1. Basic events which require component reliability parameters for estimation of failure probabilities were identified from the system fault trees.

2. The basic events were then grouped according to the type of component (e.g., motor-driven pump, relay, etc.) and failure mode (e.g., fails to start, spurious operation, etc.).

3. Generic data sources were identified which provide reliability parameter estimates relevant to the component types and failure modes specified.

4. The generic data sources identified were then classified as to their origin (e.g., expert opinion, U.S. commercial nuclear industry failure experience, etc.), scope (e.g., number of components considered in the population, dates of data, relevance, etc.), and quality (e.g., failure counts based on plant maintenance records or LERs; exposure hours based on actual experience or estimates, etc.).

5. The most appropriate generic data sources were then selected by matching the needs of the PRA to the available information.

6. Finally, the generic data sources were aggregated into a composite estimate if more than one source was identified and selected for a particular component type and failure mode.

A "generic data source" provides component reliability parameter estimates based on (1) the failure experience of other utilities or industries, or (2) expert opinion. There are many such sources; hence, the bulk of the generic data effort consisted of identifying, classifying, and selecting generic data that was relevant. As it is difficult to collect meaningful plant specific data on many types of equipment, a significant amount of the component reliability parameter estimates were taken from generic data.

Since generic data plays an important role in risk and reliability analysis studies, SAIC has developed a comprehensive generic data set for use in commercial nuclear power plant PRAs. It is based on the accumulated experience of many SAIC and industry studies and has been subjected to extensive review. With this data set as the basis, a Generic Data Notebook for the Turkey Point and St. Lucie PRA projects was developed [Ref. 3.3-2] using the steps outlined above. The generic data is presented in Table 3.3-1.

3.3.1.2 Plant Specific Component Reliability Data

The calculation of a component failure rate is fundamentally a straightforward task. The number of failures of the component for a specific failure mode is divided by the number of demands or operating hours or, in general, the exposure of the component to the failure mode. PRA does not attempt to understand the underlying frequency distribution of the failure mode, but rather, imposes a constant failure rate on all failure modes identified in the model. Table 3.3-2 provides a listing of the component types for which plant specific data analysis was performed per the requirements of the Data Analysis Procedure [Ref. 3.3-3].

The INPO Nuclear Plant Reliability Data System (NPRDS) database was the primary source of component failure data. This database captures data for the majority of the components modeled in the PRA, and based on the INPO reporting requirements should contain records of failures applicable to the PRA. Where required, data for components not included under the scope of NPRDS was obtained from the FPL Nuclear Job Planning System (NJPS) and/or interviews with system engineers, maintenance, and operations personnel. Only failures judged as catastrophic (i.e. events where the failure mode of concern occurred, valve failed to open, etc.) were considered in the calculation.

The number of demands or operating hours for the component is, in general, not recorded in readily accessible databases. For the failure rate calculations, exposure estimates were determined through a combination of review of operator logs and by understanding how the system is operated, tested, and maintained. Unit operating history (i.e., number of trips, operating hours, etc.) was also considered.

The same project guideline used for the Turkey Point PRA for setting the data window was used for the St. Lucie analysis - a minimum of five years of operating experience as the targeted data window. Plant history was then reviewed to determine the closest unit refueling outage to this target. The actual data window for this analysis includes approximately 5.75 years of operating experience for St. Lucie Unit 1 (1/85 to 10/91) and for St. Lucie Unit 2 (6/86 to 4/92) unless otherwise noted.

Initial plant specific data analysis was performed separately for both Unit 1 and Unit 2. Due to the limited data, either the number of failures or small exposure (demands or operating hours), it was determined that combining the Unit 1 and Unit 2 data would provide a more realistic representation of the site specific history. A bayesian update was then performed (using generic data as the prior distribution) to obtain many of the failure rates used in the St. Lucie PRA for component types with plant specific failure history. Plant specific history was used for diesel generator fails-to-start and a combination of plant and generic data was used for diesel generator fails-to-run.

Table 3.3-3 provides the failure rates used based plant specific experience and the estimation methods.

3.3.2 Test and Maintenance Unavailability

Equipment Out-of-Service Logs were reviewed to determine out-of-service (unavailable) hours for the T/M events included in the fault tree models. Since the PRA assumes that the Unit was operating (on-line) when the initiating event occurs, only T/M unavailability when the Unit is on-line was considered.

Where a system consists of multiple trains (e.g., 2 trains of HPSI), the unavailability data for both trains was combined (averaged) to obtain one T/M probability applicable to both trains. Since both St. Lucie units have similar equipment and maintenance practices, the Unit 1 and Unit 2 data was then combined, except for PORV flow paths, to calculate the final unavailability probabilities. Unit specific probabilities were estimated for the PORV flow paths. This data was not combined due to the differences in once-through-cooling success criteria (2-out-of-2 PORV flow paths for Unit 1 vs. 1-out-of-2 flow paths for Unit 2) and the Technical Specification requirement that one Unit 2 block valve be closed with the unit at power (due to the larger relief capacity of the Unit 2

PORV's). T/M unavailability events were incorporated into the fault tree model at the train level where possible.

Tables 3.3-4 and 3.3-5 list the Test and Maintenance events used in the St. Lucie PRA.

3.3.3 Common Cause Failure Data

For the St. Lucie PRA, the estimation of common-cause failure (CCF) probabilities was based on NUREG/CR-4780 [Ref. 3.3-4]. As noted in the NUREG, the term "common cause failure" refers to an event in which two or more components fail at the same time, or within a short interval, from a shared cause. Common cause failures can thus be considered as a subset of dependent failures (excluding functional dependency such as a common support system) that must be understood, modeled, and quantified in a study such as this.

CCF events are relatively infrequent, so it is unlikely that a single utility would have sufficient operating history to build an adequate CCF data set. Generic sources of CCF are not common either, consequently the publicly available CCF data sources come from NRC- or EPRI-sponsored sources.

The process used for treating CCFs began with a review of the fault tree models in order to identify groups of components that were similar in nature. This determination was based primarily upon whether the grouped components (1) provided the same function and (2) operated in the same system and environment. Other distinguishing characteristics such as component manufacturer, testing policy, and maintenance policy were found to be dependent on the above factors. It should be noted that CCF data for only a limited number of components is available (which further suggests the rarity of such events), so the project specific review was equally limited. Following this review, basic events for the components and applicable failure modes were incorporated into the fault trees and basic event data set. Review of the fault trees was also performed to assure proper placement of the CCF event.

CCF data was applied using a beta factor, as provided in the generic sources, to each group of components based upon the type of components involved (e.g., pumps, MOVs, and EDGs). This beta factor was then multiplied by the independent component failure probability to determine the probability of CCF for the group. The common-cause failure data is presented in Table 3.3-6.

3.3.4 Initiating Event Frequencies

The general practice of Probabilistic Risk Assessment is to identify the many different types of transients and accidents that can affect the nuclear plant and then collapse them into categories based on the similarity of plant response to the event. This analysis is documented in Section 3.1 (Accident Sequence Analysis).

Upon completion of this identification and categorization process, the task of Data Analysis is to develop frequencies (events per reactor year) for the individual events and then sum them into category frequencies. Initiating event frequencies have been developed utilizing two methods: (1)

use of plant specific experience, and (2) use of generic data. For events which have occurred during the data window, plant specific experience was utilized. Plant specific data was obtained for both Unit 1 and Unit 2. Due to the limited number of initiating events during the data window and the similarities in design, operation, and maintenance, the final plant specific initiating event frequencies calculated were based on combined Unit 1 and Unit 2 data. For events which have not occurred during the data window, generic data or Bayesian updating was utilized.

Table 3.3-7 list the initiating event frequencies used in the St. Lucie PRA.

3.3.5 Section 3.3 References

3.3-1 Data Analysis Report, PSL PRA 3.0, Revision 0.

3.3-2 Generic Data Notebook for Commercial Nuclear Power Plant Probabilistic Risk Assessment, SAIC 163-03-00, Revision 2, January, 1991.

3.3-3 Data Analysis Procedure, RRAG-PSL-006, Revision 1.

3.3-4 USNRC and EPRI, *Procedures for Treating Common Cause Failures in Safety and Reliability Studies*, NUREG/CR-4780, Vols 1 and 2, January 1988. (Also issued as EPRI NP-5613, Vols 1 and 2, February 1988.)

Table 3.3-1 Generic Failure Data

Type Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
AC A	AIR COOLING UNIT	FAILS TO START	H	L	2.08-04	9.44-06	8.52-05	7.69-04	9.02+00		G
AC F	AIR COOLING UNIT	FAILS TO RUN	H	L	1.05-05	1.63-06	7.06-06	3.06-05	4.34+00		G
AD F	AIR DRYER	FAILS TO DELIVER FLOW	H	L	5.23-07	3.38-07	5.07-07	7.61-07	1.50+00		G
AF F	AIR FILTER	FAILS TO DELIVER FLOW	H	L	7.23-06	1.16-06	4.91-06	2.08-05	4.24+00		G
AM A	AIR COMPRESSOR	FAILS TO START	H	L	1.27-01	9.65-02	1.25-01	1.63-01	1.30+00		G
AM F	AIR COMPRESSOR	FAILS TO RUN	H	L	2.48-03	2.74-05	5.0[illegible]	9.49-03	1.86+01		G
AR F	AIR RECEIVER	LOCAL FAULTS	H	L	6.00-07	1.04-08	1.56-07	2.32-06	1.49+01		G
AV CM	AIR-OPERATED VALVE	FAILS TO OPERATE	H	L	2.17-03	3.12-04	1.42-03	6.46-03	4.55+00		G
AV KR	AIR-OPERATED VALVE	SPURIOUS OPERATION	H	L	3.74-06	3.57-07	2.10-06	1.23-05	5.88+00		G
B1 F	>4 KV BUS	FAULT	H	L	4.50-08	1.42-09	1.56-08	1.71-07	1.10+01		G
B2 F	<4 KV BUS	FAULT	H	L	1.19-07	2.31-09	3.26-08	4.60-07	1.41+01		G
B4 F	120 V BUS	FAULT	H	L	1.19-07	2.31-09	3.26-08	4.60-07	1.41+01		G
BC F	BATTERY CHARGER	NO OUTPUT	H	L	7.78-06	3.52-07	3.18-06	2.87-05	9.04+00		G
BD F	DC BUS	FAULT	H	L	4.50-08	1.42-09	1.56-08	1.71-07	1.10+01		G
BI F	BISTABLE	SPURIOUS OPERATION	H	L	1.03-06	9.00-08	5.57-07	3.45-06	6.19+00		G

Table 3.3-1 Generic Failure Data

Type	Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
BI	N	BISTABLE	FAILS TO OPERATE ON DEMAND	N	L	2.25-07	7.13-08	1.89-07	5.00-07	2.65+00		G
BS	P	BASKET STRAINER	PLUGS	H	L	2.66-05	1.00-06	1.00-05	1.00-04	****+00		G
BT	D	BATTERY	NO OUTPUT (DEMAND)	N	L	6.61-03	7.12-04	3.88-03	2.12-02	5.45+00		G
BT	F	BATTERY	NO OUTPUT (HOURLY)	H	L	1.93-06	1.75-07	1.06-06	6.40-06	6.05+00		G
CB	D	AC BREAKER	FAILS TO OPERATE	N	L	1.16-03	2.03-04	8.11-04	3.24-03	3.99+00		G
CB	K	AC BREAKER	TRANSFERS CLOSED	H	L	2.02-06	2.04-07	1.16-06	6.55-06	5.66+00		G
CB	R	AC BREAKER	TRANSFERS OPEN	H	L	1.87-06	1.89-07	1.07-06	6.06-06	5.66+00		G
CD	D	DC BREAKER	FAILS TO TRIP (OVER CURRENT)	N	L	8.83-04	9.27-05	5.14-04	2.85-03	5.54+00		G
CD	R	DC BREAKER	TRANSFERS OPEN	H	L	3.80-06	3.48-08	7.09-07	1.44-05	2.03+01		G
CF	R	FUSE	FAILS OPEN	H	L	6.38-07	2.40-08	2.40-07	2.39-06	9.98+00		G
CI	R	DC INTERRUPTER	TRANSFERS OPEN	H	L	2.00-08	3.47-10	5.18-09	7.74-08	1.49+01		G
CS	R	DC DISCONNECT SWITCH	TRANSFERS OPEN	H	L	1.41-06	3.47-07	1.10-06	3.50-06	3.18+00		G
CT	CN	CONTACT	FAILS TO OPEN/FAILS TO CLOSE	N	L	2.27-06	1.22-07	1.00-06	8.21-06	8.21+00		G
CT	KR	CONTACT	SPURIOUS OPERATION	H	L	7.05-08	5.11-09	3.54-08	2.44-07	6.91+00		G
CV	C	CHECK VALVE	FAILS TO CLOSE	N	L	1.63-03	1.14-04	8.06-04	5.69-03	7.06+00		G

Table 3.3-1 Generic Failure Data

Type	Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
CV	K	CHECK VALVE	TRANSFERS CLOSED	H	L	1.69-06	4.33-07	1.34-06	4.13-06	3.09+00		G
CV	N	CHECK VALVE	FAILS TO OPEN	N	L	1.45-04	3.47-05	1.12-04	3.64-04	3.24+00		G
CV	R	CHECK VALVE	TRANSFERS OPEN	H	L	9.46-07	4.97-08	4.13-07	3.43-06	8.32+00		G
DG	A	DIESEL GENERATOR	FAILS TO START	N	L	1.76-02	2.22-03	1.10-02	5.45-02	4.96+00		G
DG	F	DIESEL GENERATOR	FAILS TO RUN	H	L	2.25-03	1.72-04	1.15-03	7.71-03	6.70+00		G
EI	D	E/I CONVERTER	FAILS TO RESPOND	H	L	2.19-07	4.22-09	5.98-08	8.47-07	1.42+01		G
EL	S	ELECT. PENETRATION	SEAL FAILURE	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
EP	D	E/P CONVERTER	FAILS TO RESPOND	H	L	1.00-07	7.55-09	5.10-08	3.44-07	6.75+00		G
FC	F	DROPOUT REGISTER	FAILS TO FALL	N	L	1.00-06	1.74-08	2.59-07	3.87-06	1.49+01		G
FE	F	FLOW ELEMENT	FAILS	H	L	9.16-06	7.85-06	9.12-06	1.06-05	1.16+00		G
FE	P	FLOW ELEMENT	PLUGS	D	L	3.75-04	1.00-04	3.00-04	9.00-04	3.00+00		G
FL	S	FUEL TRANSFER TUBE	SEAL FAILURE	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
FS	D	FLOW SWITCH	FAILS TO RESPOND	N	L	1.00-08	1.74-10	2.59-09	3.87-08	1.49+01		G
FS	HL	FLOW SWITCH	FAILS HIGH/FAILS LOW	H	L	2.80-06	2.13-07	1.43-06	9.61-06	6.72+00		G
FT	D	FLOW TRANSMITTER	FAILS TO RESPOND	H	L	1.81-06	3.64-07	1.33-06	4.83-06	3.64+00		G

Table 3.3-1 Generic Failure Data

Type	Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
FT	H	FLOW TRANSMITTER	FAILS HIGH	H	L	2.04-06	5.57-07	1.64-06	4.83-06	2.94+00		G
FT	L	FLOW TRANSMITTER	FAILS LOW	H	L	1.83-06	3.82-07	1.36-06	4.82-06	3.55+00		G
HR	F	HYDROGEN RECOMBINER	FAILS TO RECOMBINE	H	L	2.68-06	3.15-07	1.63-06	8.41-06	5.17+00		G
HX	F	HEAT EXCHANGER	COOLING CAPABILITY FAILS	H	L	1.95-05	5.86-07	6.60-06	7.44-05	1.13+01		G
HX	J	HEAT EXCHANGER	TUBE RUPTURE	H	L	1.60-07	1.98-08	9.91-08	4.96-07	5.00+00		G
HX	P	HEAT EXCHANGER	PLUGS	H	L	2.20-06	2.15-07	1.24-06	7.18-06	5.78+00		G
IN	F	INVERTER	NO OUTPUT	H	L	2.87-05	9.67-07	1.03-05	1.09-04	1.06+01		G
IR	F	REGULATING RECTIFIER	NO OUTPUT	H	L	1.07-06	3.57-07	9.09-07	2.31-06	2.55+00		G
IV	F	STATIC VOLTAGE REGUL	NO OUTPUT	H	L	7.11-06	1.99-06	5.77-06	1.67-05	2.89+00		G
LA	F	EQUIP HATCH/AIR LOCK	FAILS TO FUNCTION	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
LA	S	EQUIP HATCH/AIR LOCK	SEAL FAILURE	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
LC	D	LOGIC CIRCUIT	FAILS TO GEN SIGNAL/FALSE OUT	H	L	3.89-06	6.84-08	1.01-06	1.50-05	1.48+01		G
LI	D	LEVEL INDICATOR	FAILS TO RESPOND	H	L	2.50-07	4.34-09	6.48-08	9.67-07	1.49+01		G
LI	H	LEVEL INDICATOR	FAILS HIGH	H	L	2.50-07	4.34-09	6.48-08	9.67-07	1.49+01		G
LI	L	LEVEL INDICATOR	FAILS LOW	H	L	2.50-07	4.34-09	6.48-08	9.67-07	1.49+01		G

Table 3.3-1 Generic Failure Data

Type Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
LS D	LEVEL SWITCH	FAILS TO RESPOND	N	L	3.00-08	1.17-08	2.65-08	6.00-08	2.26+00		G
LS H	LEVEL SWITCH	FAILS HIGH	H	L	2.31-06	1.31-07	1.04-06	8.31-06	7.97+00		G
LS L	LEVEL SWITCH	FAILS LOW	H	L	2.31-06	1.31-07	1.04-06	8.31-06	7.97+00		G
LT D	LEVEL TRANSMITTER	FAILS TO RESPOND	H	L	2.14-06	4.62-07	1.60-06	5.57-06	3.47+00		G
LT H	LEVEL TRANSMITTER	FAILS HIGH	H	L	2.02-06	3.72-07	1.44-06	5.57-06	3.87+00		G
LT L	LEVEL TRANSMITTER	FAILS LOW	H	L	2.06-06	4.05-07	1.50-06	5.56-06	3.70+00		G
LY DL	SIGNAL PROCESSOR MOD	FAILS TO RESPOND/FAILS LOW	H	L	6.42-07	1.49-07	4.93-07	1.63-06	3.30+00		G
MC CN	AIR-OPERATED DAMPER	FAILS TO OPERATE	N	L	2.18-03	6.09-04	1.77-03	5.14-03	2.90+00		G
MC KR	AIR-OPERATED DAMPER	SPURIOUS OPERATION	H	L	5.09-06	2.24-07	2.05-06	1.88-05	9.16+00		G
MD CN	MOTOR-OP DAMPER	FAILS TO OPERATE	N	L	2.18-03	6.09-04	1.77-03	5.14-03	2.90+00		G
MD KR	MOTOR-OP DAMPER	SPURIOUS OPERATION	H	L	5.09-06	2.24-07	2.05-06	1.88-05	9.16+00		G
ME S	MECHANICAL PENETRA.	SEAL FAILURE	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
MF A	MOTOR-DRIVEN FAN	FAILS TO START	N	L	2.08-04	9.44-06	8.52-05	7.69-04	9.02+00		G
MF F	MOTOR-DRIVEN FAN	FAILS TO RUN	H	L	1.24-05	4.60-06	1.08-05	2.55-05	2.35+00		G
MP A	MOTOR-DRIVEN PUMP	FAILS TO START	N	L	4.84-03	5.45-04	2.89-03	1.53-02	5.30+00		G

Table 3.3-1 Generic Failure Data

Type Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
MP F	MOTOR-DRIVEN PUMP	FAILS TO RUN	H	L	8.45-05	3.71-06	3.41-05	3.13-04	9.18+00		G
MV C	MOTOR-OPERATED VALVE	FAILS TO CLOSE	N	L	6.01-03	6.72-04	3.58-03	1.91-02	5.33+00		G
MV K	MOTOR-OPERATED VALVE	TRANSFERS CLOSED	H	L	1.52-06	3.62-07	1.18-06	3.82-06	3.25+00		G
MV N	MOTOR-OPERATED VALVE	FAILS TO OPEN	N	L	5.07-03	1.39-03	4.09-03	1.20-02	2.94+00		G
MV R	MOTOR-OPERATED VALVE	TRANSFERS OPEN	H	L	1.36-06	2.39-07	9.53-07	3.80-06	3.99+00		G
PP JP	PIPING	FAILURE	H	L	5.53-07	4.05-08	2.78-07	1.91-06	6.88+00		G
PS D	PRESSURE SWITCH	FAILS TO RESPOND	N	L	4.50-05	4.32-08	2.35-06	1.28-04	5.44+01		G
PS HL	PRESSURE SWITCH	FAILS HIGH/FAILS LOW	H	L	8.45-07	1.21-07	5.51-07	2.52-06	4.57+00		G
PT D	PRESSURE TRANSMITTER	FAILS TO RESPOND	H	L	1.47-06	3.43-07	1.13-06	3.74-06	3.30+00		G
PT H	PRESSURE TRANSMITTER	FAILS HIGH	H	L	1.49-06	3.51-07	1.15-06	3.76-06	3.27+00		G
PT L	PRESSURE TRANSMITTER	FAILS LOW	H	L	1.47-06	3.39-07	1.12-06	3.73-06	3.32+00		G
PV R	PRESSURE CONTROL VLV	TRANSFERS OPEN	H	L	1.06-05	7.84-07	5.37-06	3.67-05	6.84+00		G
PX F	POWER SUPPLY	NO OUTPUT	H	L	1.40-06	9.34-07	1.36-06	1.99-06	1.46+00		G
RA F	RADIATION ELEMENT	FAILS TO RESPOND	H	L	3.42-06	2.12-06	3.30-06	5.13-06	1.56+00		G
RE BE	RELAY	FAILS TO OPERATE ON DEMAND	N	L	7.65-05	5.08-06	3.69-05	2.69-04	7.28+00		G

Table 3.3-1 Generic Failure Data

Type Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
RE KR	RELAY	OPERATIONAL FAILURE	H	L	3.94-07	3.83-08	2.22-07	1.29-06	5.80+00		G
RO P	RESTRICTING ORIFICE	PLUGS	D	L	3.75-04	1.00-04	3.00-04	9.00-04	3.00+00		G
RV C	RELIEF VALVE	FAILS TO CLOSE	N	L	5.18-03	1.12-04	1.49-03	2.00-02	1.34+01		G
RV N	RELIEF VALVE	FAILS TO OPEN	N	L	2.12-04	7.90-06	7.94-05	7.98-04	1.00+01		G
RV R	RELIEF VALVE	SPURIOUS OPEN	H	L	1.69-06	2.84-07	1.17-06	4.80-06	4.11+00		G
RY N	PSV, S/G SAFETY VLV	FAILS TO OPEN	N	L	1.40-04	2.91-06	3.97-05	5.41-04	1.36+01		G
RY Q	PSV, S/G SAFETY VLV	FAILS TO RESEAT AFTER LIQUID	N	L	1.00-01	3.75-03	3.75-02	3.75-01	1.00+01		G
RY T	PSV, S/G SAFETY VLV	FAILS TO RESEAT AFTER STEAM	N	L	7.45-03	1.05-03	4.83-03	2.23-02	4.62+00		G
RZ N	PORV	FAILS TO OPEN	N	L	4.15-03	6.60-06	2.92-04	1.29-02	4.42+01		G
RZ Q	PORV	FAILS TO RESEAT AFTER LIQUID	N	L	5.00-03	1.88-04	1.88-03	1.88-02	1.00+01		G
RZ T	PORV	FAILS TO RESEAT AFTER STEAM	N	L	5.00-03	1.88-04	1.88-03	1.88-02	1.00+01		G
SC DN	STOP-CHECK VALVE	FAILS TO OPERATE	N	L	1.61-03	2.86-05	4.22-04	6.23-03	1.48+01		G
SH P	CONTAINMENT SUMP	PLUGGED	N	L	2.20-05	3.82-07	5.70-06	8.51-05	1.49+01		G
SP S	SPARE PIPING PENETR.	SEAL FAILURE	H	L	1.00-06	2.67-07	8.00-07	2.40-06	3.00+00		G
ST A	MTR-DRIVEN STRAINER	FAILS TO START	N	L	2.08-04	9.44-06	8.52-05	7.69-04	9.02+00		G

Table 3.3-1 Generic Failure Data

Type Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
ST F	MTR-DRIVEN STRAINER	FAILS TO RUN	H	L	7.85-06	7.97-07	4.51-06	2.55-05	5.66+00		G
SV CN	SOLENOID VALVE	FAILS TO OPERATE	N	L	2.83-03	1.78-04	1.34-03	1.00-02	7.51+00		G
SV K	SOLENOID VALVE	TRANSFERS CLOSED	H	L	4.09-07	1.08-07	3.26-07	9.85-07	3.02+00		G
SV R	SOLENOID VALVE	TRANSFERS OPEN	H	L	4.09-07	1.08-07	3.26-07	9.85-07	3.02+00		G
SW C	HAND SWITCH	FAILS TO CLOSE	N	L	2.59-08	4.50-10	6.71-09	1.00-07	1.49+01		G
SW K	HAND SWITCH	TRANSFERS CLOSED	H	L	8.00-08	1.89-09	2.41-08	3.08-07	1.28+01		G
SW N	HAND SWITCH	FAILS TO OPEN	N	L	2.00-08	3.47-10	5.18-09	7.74-08	1.49+01		G
SW R	HAND SWITCH	TRANSFERS OPEN	H	L	8.00-08	1.89-09	2.41-08	3.08-07	1.28+01		G
SX N	SPEED SWITCH	FAILS TO OPEN	N	L	2.46-04	2.05-05	1.31-04	8.32-04	6.37+00		G
SZ C	VLV POSITION SWITCH	FAILS TO CLOSE	N	L	2.46-04	2.93-05	1.50-04	7.70-04	5.12+00		G
SZ KR	VLV POSITION SWITCH	TRANSFERS OPEN/CLOSED	H	L	4.44-06	7.98-07	3.14-06	1.24-05	3.94+00		G
T1 F	KV TRANSFORMERS	FAULT	H	L	2.08-06	1.50-07	1.04-06	7.20-06	6.92+00		G
T6 F	480V-240V TRANS	FAULT	H	L	1.90-06	8.99-08	7.93-07	6.99-06	8.81+00		G
TK GJ	TANK	LEAKAGE/RUPTURE	H	L	5.52-06	2.51-06	5.05-06	1.01-05	2.01+00		G
TN F	TRAVELLING SCREEN	FAILS TO RUN	H	L	6.85-04	1.19-05	1.78-04	2.65-03	1.49+01		G

Table 3.3-1 Generic Failure Data

Type	Code	Component	Failure Mode	Unit	Dist	Mean	Lower	Median	Upper	P1	P2	Basis
TP	A	TURBINE-DRIVEN PUMP	FAILS TO START	H	L	2.62-02	4.18-03	1.78-02	7.58-02	4.26+00		G
TP	F	TURBINE-DRIVEN PUMP	FAILS TO RUN	H	L	8.91-05	1.09-05	5.49-05	2.77-04	5.06+00		G
TS	BE	TEMPERATURE SWITCH	NO FUNCTION WITH SIGNAL	H	L	1.20-07	4.69-08	1.06-07	2.40-07	2.26+00		G
TS	HL	TEMPERATURE SWITCH	FUNCTION WITHOUT SIGNAL	H	L	9.20-07	2.30-08	2.85-07	3.54-06	1.24+01		G
TT	D	TEMP TRANSMITTER	FAILS TO RESPOND	H	L	1.47-06	2.21-07	9.74-07	4.30-06	4.42+00		G
TT	HL	TEMP TRANSMITTER	FAILS HIGH/FAILS LOW	H	L	1.81-06	4.41-07	1.41-06	4.49-06	3.19+00		G
TV	D	TEMP INDICATING CTRL	FAILS TO RESPOND	H	L	2.12-06	8.47-07	1.88-06	4.19-06	2.22+00		G
TV	HL	TEMP INDICATING CTRL	FAILS HIGH/FAILS LOW	H	L	1.36-06	5.39-07	1.21-06	2.70-06	2.24+00		G
XV	CN	MANUAL VALVE	FAILS TO OPERATE	H	L	3.47-04	2.57-05	1.76-04	1.20-03	6.84+00		G
XV	K	MANUAL VALVE	TRANSFERS CLOSED	H	L	1.94-07	2.77-08	1.27-07	5.80-07	4.58+00		G
XV	R	MANUAL VALVE	TRANSFERS OPEN	H	L	1.30-07	1.49-08	7.81-08	4.08-07	5.23+00		G

Table 3.3-2

PLANT-SPECIFIC DATA ANALYSIS SCOPE BY COMPONENT TYPE	
SYSTEM	COMPONENT TYPES
Auxiliary Feedwater	Pumps, Valves
Safety Injection Tanks	Valves
Component Cooling Water	Heat Exchangers, Pumps, Valves
Containment Ventilation and Heat Removal	Heat Exchangers, Dampers, Fans, Valves
Containment Isolation	Valves
Containment Spray	Pumps, Valves
Chemical and Volume Control	Pumps, Valves
Electric Power	Batteries, Battery Chargers, Diesel Generators, Invertors, Load Circuit Breakers, Transformers
High Pressure Safety Injection	Pumps, Valves
Ventilation and Air Conditioning	Dampers, Fans, Heat Exchangers, Valves
Instrument Air	Compressors, Dryers, Valves
Low Pressure Safety Injection/Shutdown Cooling	Heat Exchangers, Pumps, Valves
Power Conversion System	Pumps, Valves
Primary Pressure Control	Valves
Intake Cooling Water	Pumps, Strainers, Valves

Table 3.3-3

Plant Specific Failure Data

	Failure Rate	Error Factor
Diesel Generators		
Fail-to-Start:	8.26E-03/Demand (3)	1.60
Fail-to-Run:	2.54E-03/Hour (1)	10.0
Motor Operated Valves		
Fail-to-Open:	6.16E-03/Demand (2)	1.50
Fail-to-Close:	2.40E-03/Demand (2)	1.95
Motor Driven Pumps		
Fail-to-Start:	1.80E-03/Demand (2)	2.57
Fail-to-Run:	6.88E-05/Hour (2)	1.37
Air Operated Valve		
Fails-to-Operate:	4.95E-03/Demand (2)	1.50
4kV Circuit Breakers		
Fails-to-Operate:	2.77E-03/Demand (3)	1.53
Battery Charger		
No Output	4.53E-05/Hour (3)	1.55

(1) Combination of Plant Specific and Generic Data
(2) Bayesian Estimation
(3) Plant Specific Data

Table 3.3-4 St. Lucie Unit 1 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
1	AFW	ATM1AFWP1A	AFW PUMP 1A IN T/M	4.52E-03
1	AFW	ATM1AFWP1B	AFW PUMP 1B IN T/M	4.52E-03
1	AFW	ATM1AFWP1C	AFW PUMP 1C IN T/M	1.83E-02
1	CCW	CTM114-9	VALVE 14-9 IN T/M	7.44E-05
1	CCW	CTM11410	VALVE 14-10 IN T/M	7.44E-05
1	CCW	CTM1CCWHXA	CCW HX 1A IN T/M	7.22E-03
1	CCW	CTM1CCWHXB	CCW HX 1B IN T/M	7.22E-03
1	CCW	CTM1PCMOVS	MV 14-2 OR 14-4 IN T/M	2.95E-04
1	CHRS	DTM1HVS1A	CONT COOLER 1A IN T/M	4.46E-04
1	CHRS	DTM1HVS1B	CONT COOLER 1B IN T/M	4.46E-04
1	CHRS	DTM1HVS1C	CONT COOLER 1C IN T/M	4.46E-04
1	CHRS	DTM1HVS1D	CONT COOLER 1D IN T/M	4.46E-04
1	CSS	LTM1PUMPA	CS PUMP 1A IN T/M	5.26E-03
1	CSS	LTM1PUMPB	CS PUMP 1B IN T/M	5.26E-03
1	CVCS	MTM1BAMPA	BAM 1A PUMP IN T/M	3.35E-02
1	CVCS	MTM1BAMPB	BAM PUMP 1B IN T/M	3.35E-02
1	CVCS	MTM1CHGPA	CHARGING PUMP 1A IN T/M	7.75E-02
1	CVCS	MTM1CHGPB	CHARGING PUMP 1B IN T/M	7.75E-02
1	CVCS	MTM1CHGPC	CHARGING PUMP 1C IN T/M	7.75E-02
1	EPS	ETM11AACHG	1AA BATTERY CHARGER IN T/M	1.22E-02
1	EPS	ETM11ACHG	1A BATTERY CHARGER IN T/M	1.22E-02
1	EPS	ETM11AEDG	EDG 1A IN T/M	1.90E-02
1	EPS	ETM11AINV	1A INST INV IN T/M	3.56E-04
1	EPS	ETM11BBCHG	1BB BATTERY CHARGER IN T/M	1.22E-02
1	EPS	ETM11BCHG	1B BATTERY CHARGER IN T/M	1.22E-02
1	EPS	ETM11BEDG	EDG 1B IN T/M	1.90E-02
1	EPS	ETM11BINV	1B INST INV IN T/M	3.56E-04
1	EPS	ETM11CINV	1C INST INV IN T/M	3.56E-04
1	EPS	ETM11DINV	1D INST INV IN T/M	3.56E-04
1	EPS	ETM1ASU	1A S/U TRANSFORMER IN T/M	5.31E-03
1	EPS	ETM1BSU	1B S/U TRANSFORMER IN T/M	5.31E-03
1	ESFAS	NTM1BA101	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA102	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA103	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA104	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA105	BISTABLE BYPASSED FOR T/M	2.89E-04

Table 3.3-4 St. Lucie Unit 1 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
1	ESFAS	NTM1BA106	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA108	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA110	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA112	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA201	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA202	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA203	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA204	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA205	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA206	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA208	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA210	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA212	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA301	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA302	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA303	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA304	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA305	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA306	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA308	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA310	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA312	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA401	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA402	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA403	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA404	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA405	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA406	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA408	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA410	BISTABLE BYPASSED FOR T/M	2.89E-04
1	ESFAS	NTM1BA412	BISTABLE BYPASSED FOR T/M	2.89E-04
1	HPSI	GTM107-1A	VALVE 07-1A IN T/M	4.05E-04
1	HPSI	GTM107-1B	VALVE 07-1B IN T/M	4.05E-04
1	HPSI	GTM107-2A	VALVE 07-2A IN T/M	5.49E-04
1	HPSI	GTM107-2B	VALVE 07-2B IN T/M	5.49E-04
1	HPSI	GTM1PUMPA	HPSI PUMP 1A IN T/M	7.80E-03

Table 3.3-4 St. Lucie Unit 1 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
1	HPSI	GTM1PUMPB	HPSI PUMP 2B IN T/M	7.80E-03
1	HPSI	HPSI VLVS	HPSI FLOW VALVES IN T/M (PER VALVE)	5.63E-04
1	HVAC	ITM1ECCEXA	ECCS EXHAUST PATH 'A' IN T/M	7.66E-03
1	HVAC	ITM1ECCEXB	ECCS EXHAUST PATH 'B' IN T/M	7.66E-03
1	HVAC	ITM1HVS4A	HVS 4A IN T/M	7.13E-02
1	HVAC	ITM1HVS4B	HVS 4B IN T/M	7.13E-02
1	IA	HTM1CMP1C	INST AIR COMP 1C IN T/M	8.01E-03
1	IA	HTM1CMP1D	INST AIR COMP 1D IN T/M	8.01E-03
1	LPSI	JTM13615	VALVE 3615 IN T/M	3.31E-04
1	LPSI	JTM13625	VALVE 3625 IN T/M	3.31E-04
1	LPSI	JTM13635	VALVE 3635 IN T/M	3.31E-04
1	LPSI	JTM13645	VALVE 3545 IN T/M	3.31E-04
1	LPSI	JTM1PUMPA	LPSI PUMP 1A IN T/M	5.53E-03
1	LPSI	JTM1PUMPB	LPSI PUMP 1B IN T/M	5.53E-03
1	LPSI	JTM1SDCHXA	1A SDC HX IN T/M	1.78E-03
1	LPSI	JTM1SDCHXB	1B SDC HX IN T/M	1.78E-03
1	PPC	ZZ1ABKSHUT	'A' PORV BLOCK VALVE CLOSED WITH POWER	6.62E-02
1	PPC	ZZ1ABLKRO	'A' PORV BLOCK VALVE CLOSED W/O POWER	7.65E-04
1	PPC	ZZ1BBKSHUT	'B' PORV BLOCK VALVE CLOSED WITH POWER	6.62E-02
1	PPC	ZZ1BBLKRO	'B' PORV BLOCK VALVE CLOSED W/O POWER	7.65E-04

Table 3.3-5 St. Lucie Unit 2 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
2	AFW	ATM2AFWP2A	AFW PUMP 2A IN T/M	4.52E-03
2	AFW	ATM2AFWP2B	AFW PUMP 2B IN T/M	4.52E-03
2	AFW	ATM2AFWP2C	AFW PUMP 2C IN T/M	1.83E-02
2	CCW	CTM2CCWHXA	CCW HX 2A IN T/M	7.22E-03
2	CCW	CTM2CCWHXB	CCW HX 2B IN T/M	7.22E-03
2	CHRS	DTM2HVS1A	CONT COOLER 1A IN T/M	4.46E-04
2	CHRS	DTM2HVS1B	CONT COOLER 1B IN T/M	4.46E-04
2	CHRS	DTM2HVS1C	CONT COOLER 1C IN T/M	4.46E-04
2	CHRS	DTM2HVS1D	CONT COOLER 1D IN T/M	4.46E-04
2	CSS	LTM2PUMPA	CS PUMP 2A IN T/M	5.26E-03
2	CSS	LTM2PUMPB	CS PUMP 2B IN T/M	5.26E-03
2	CVCS	MTM2BAMPA	BAM PUMP 2A IN T/M	3.35E-02
2	CVCS	MTM2BAMPB	BAM PUMP 2B IN T/M	3.35E-02
2	CVCS	MTM2CHGPA	CHARGING PUMP 2A IN T/M	7.75E-02
2	CVCS	MTM2CHGPB	CHARGING PUMP 2B IN T/M	7.75E-02
2	CVCS	MTM2CHGPC	CHARGING PUMP 2C IN T/M	7.75E-02
2	EPS	ETM22AACHG	2AA BATTERY CHARGER IN T/M	1.22E-02
2	EPS	ETM22ACHG	2A BATTERY CHARGER IN T/M	1.22E-02
2	EPS	ETM22AEDG	EDG 2A IN T/M	1.90E-02
2	EPS	ETM22AINV	2A INST INV IN T/M	3.56E-04
2	EPS	ETM22BBCHG	2BB BATTERY CHARGER IN T/M	1.22E-02
2	EPS	ETM22BCHG	2B BATTERY CHARGER IN T/M	1.22E-02
2	EPS	ETM22BEDG	2B EDG IN T/M	1.90E-02
2	EPS	ETM22BINV	2B INST INV IN T/M	3.56E-04
2	EPS	ETM22CINV	2C INST INV IN T/M	3.56E-04
2	EPS	ETM22DINV	2D INST INV IN T/M	3.56E-04
2	EPS	ETM2ASU	2A S/U TRANSFORMER IN T/M	5.31E-03
2	EPS	ETM2BSU	2B S/U TRANSFORMER IN T/M	5.31E-03
2	ESFAS	NTM2BA101	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA102	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA103	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA104	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA105	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA106	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA108	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA110	BISTABLE BYPASSED FOR T/M	2.89E-04

Table 3.3-5 St. Lucie Unit 2 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
2	ESFAS	NTM2BA112	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA201	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA202	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA203	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA204	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA205	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA206	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA208	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA210	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA212	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA301	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA302	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA303	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA304	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA305	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA306	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA308	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA310	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA312	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA401	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA402	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA403	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA404	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA405	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA406	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA408	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA410	BISTABLE BYPASSED FOR T/M	2.89E-04
2	ESFAS	NTM2BA412	BISTABLE BYPASSED FOR T/M	2.89E-04
2	HPSI	GTM207-1A	VALVE 07-1A IN T/M	4.05E-04
2	HPSI	GTM207-1B	VALVE 07-1B IN T/M	4.05E-04
2	HPSI	GTM207-2A	VALVE 07-2A IN T/M	5.49E-04
2	HPSI	GTM207-2B	VALVE 07-2B IN T/M	5.49E-04
2	HPSI	GTM2MINRCA	A' MIN RECIRC PATH IN T/M	3.47E-04
2	HPSI	GTM2MINRCB	B' MIN RECIRC PATH IN T/M	3.47E-04
2	HPSI	GTM2PUMPA	HPSI PUMP 2A IN T/M	7.80E-03
2	HPSI	GTM2PUMPB	HPSI PUMP 2B IN T/M	7.80E-03

Table 3.3-5 St. Lucie Unit 2 Test & Maintenance Probabilities

Unit	System	T/M Basic Event	Description	T/M Probability
2	HPSI	HPSI FLOW VLVS	HPSI FLOW VALVES IN T/M (PER VALVE)	5.63E-04
2	HPSI	JTM23615	VALVE 3615 IN T/M	3.31E-04
2	HPSI	JTM23625	VALVE 3625 IN T/M	3.31E-04
2	HPSI	JTM23635	VALVE 3635 IN T/M	3.31E-04
2	HPSI	JTM23645	VALVE 3545 IN T/M	3.31E-04
2	HVAC	ITM2ECCEXA	ECCS EXHAUST PATH 'A' IN T/M	7.66E-03
2	HVAC	ITM2ECCEXB	ECCS EXHAUST PATH 'B' IN T/M	7.66E-03
2	HVAC	ITM2HVS4A	HVS 4A IN T/M	7.13E-02
2	HVAC	ITM2HVS4B	HVS 4B IN T/M	7.13E-02
2	IA	HTM2CMP2C	INST AIR COMP 2C IN T/M	8.01E-03
2	IA	HTM2CMP2D	INST AIR COMP 2D IN T/M	8.01E-03
2	LPSI	JTM23615	VALVE 3615 IN T/M	3.31E-04
2	LPSI	JTM23625	VALVE 3625 IN T/M	3.31E-04
2	LPSI	JTM23635	VALVE 3635 IN T/M	3.31E-04
2	LPSI	JTM23645	VALVE 3645 IN T/M	3.31E-04
2	LPSI	JTM2PUMPA	LPSI PUMP 2A IN T/M	5.53E-03
2	LPSI	JTM2PUMPB	LPSI PUMP 2B IN T/M	5.53E-03
2	LPSI	JTM2SDCHXA	2A SDC HX IN T/M	1.78E-03
2	LPSI	JTM2SDCHXB	2B SDC HX IN T/M	1.78E-03
2	PPC	ZZ2ABKSHUT	'A' PORV BLOCK VALVE CLOSED WITH POWER	7.10E-01
2	PPC	ZZ2ABLKRO	'A' PORV BLOCK VALVE CLOSED W/O POWER	1.00E-01
2	PPC	ZZ2ABLKROR	A' PORV BLOCK VALVE CLOSED W/O POWER BUT RECOVERABLE	1.90E-01
2	PPC	ZZ2BBKSHUT	'B' PORV BLOCK VALVE CLOSED WITH POWER	3.00E-02
2	PPC	ZZ2BBLKRO	'B' PORV BLOCK VALVE CLOSED W/O POWER	1.00E-01
2	PPC	ZZ2BBLKROR	B' PORV BLOCK VALVE CLOSED W/O POWER BUT RECOVERABLE	1.90E-01

TABLE 3.3-6 GENERIC SCREENING DATA FOR CCF EVENTS

Type Code	CCF Beta Factor Basic Event	Description	Beta Factor
AC	DACACCFBFS	CVHRS air cooling unit fails to start	0.11
	DACFCCFBFS	CVHRS air cooling unit fails to run	0.11
AV	xAVCCCFBFS[1]	air-operated valve fails to open	0.11
	xAVNCCFBFS[1]	air-operated valve fails to close	0.11
BT	EBTFCCFBFS	battery - no output (hourly)	0.08
CB	NCBDCCFBFS	reactor trip breaker fails to operate	0.19
CV	xCVNCCFBFS[1]	check valve fails to open	0.06
DG	EDGACCFBFS	diesel generator fails to start	0.05
	EDGFCCFBFS	diesel generator fails to run	0.05
MF	IMFACCFBFS	HVAC fan fails to start	0.13
	IMFFCCFBFS	HVAC fan fails to run	0.13
MP	CMPACCFBFS	CCW pump fails to start	0.03
	CMPFCCFBFS	CCW pump fails to run	0.03
	GMPACCFBFS	HHSI pump fails to start	0.17
	GMPFCCFBFS	HHSI pump fails to run	0.17
	JMPACCFBFS	RHR pump fails to start	0.11
	JMPFCCFBFS	RHR pump fails to run	0.11
	LMPACCFBFS	CS pump fails to start	0.05
	LMPFCCFBFS	CS pump fails to run	0.05
	QMPACCFBFS	ICW pump fails to start	0.03
	QMPFCCFBFS	ICW pump fails to run	0.03
	yMPACCFBFS[2]	motor-driven pump fails to start	0.10
	yMPFCCFBFS[2]	motor-driven pump fails to run	0.10
MV	xMVCCCFBFS[1]	motor-operated valve fails to close	0.08
	xMVNCCFBFS[1]	motor-operated valve fails to open	0.08
RY	FRYNCCFBFS	S/G safety valve fails to open	0.07
	ORYNCCFBFS	primary safety fails to open	0.07
	ORYQCCFBFS	primary safety valve fails to close after liquid relief	0.07
	ORYTCCFBFS	primary safety valve fails to close after steam relief	0.07
TP	ATPACCFBFS	AFW pump fails to start	0.03
	ATPFCCFBFS	AFW pump fails to run	0.03

[1]The "x" system designator means that this CCF data applies to all systems.

[2]The "y" system designator means that this CCF data applies to all systems not specifically addressed above (e.g., charging pumps, and condensate pumps).

Table 3.3-7 Initiating Event Frequencies

Initiator	Description	Frequency (Per Year)	Estimation Method
T1	Reactor/Turbine Trip	1.58	Plant Specific
T2	Reactor Trip (PORV Chall)	2.97E-01	Plant Specific
T3A	LOFW - Recoverable	3.96E-01	Plant Specific
T3C	LOFW - Not Recoverable	9.89E-02	Plant Specific
T3D	Feedline Break - Upstream Feedline Break - Downstream (A S/G) Feedline Break - Downstream (B S/G)	1.00E-03 1.00E-03 1.00E-03	Generic Generic Generic
T3E	Excessive Feedwater	1.98E-01	Plant Specific
T4	Loss of Offsite Power - A Train Loss of Offsite Power - B Train	1.83E-06 1.83E-06	Fault Tree Fault Tree
T5	Steamline Break - Upstream - A S/G Steamline Break - Upstream - B S/G	5.00E-05 5.00E-05	Generic Generic
T6	Steamline Break - Downstream	4.00E-04	Generic
T7	Spurious MSIS Spurious SI	9.89E-02 5.00E-02	Plant Specific Generic
T8A	PORV Sticking Open (A S/G)	1.03E-03	Generic
T8B	PORV Sticking Open (B S/G)	1.03E-03	Generic
S1	Small-Small LOCA	1.42E-03	Generic
S2	Small LOCA	4.06E-04	Generic
A	Large LOCA	2.66E-04	Generic
R	Steam Generator A Tube Rupture Steam Generator B Tube Rupture	4.89E-03 4.89E-03	Generic Generic
Loss of DC Bus	Loss of DC Bus - A Train Loss of DC Bus - B Train	3.94E-04 3.94E-04	Generic Generic
Loss of 4kV Bus	Loss of 4kV Bus - A Train Loss of 4kV Bus - B Train	3.94E-04 3.94E-04	Generic Generic
Loss of 6.9kV Bus	Loss of 6.9kV Bus - A Train Loss of 6.9kV Bus - B Train	3.94E-04 3.94E-04	Generic Generic
Loss of 120VAC Instrument Bus	Loss of 120VAC Instrument Bus	9.89E-02	Plant Specific
Loss of TCW	Loss of TCW	9.41E-04	Bayesian Update
Loss of ICW	Loss of ICW	2.68E-04	Bayesian Update
Loss of CCW	Loss of CCW	9.41E-04	Bayesian Update
Loss of IA	Loss of Instrument Air	9.20E-02	Generic
Loss of Grid	Loss of Grid	1.50E-01	Generic

3.4 Human Reliability/Recovery Analysis

3.4.1 HRA Scope

Human performance plays an important role in plant safety and in the plant's risk profile. During transient and accident situations, the St. Lucie shift operating crews are expected to terminate the event sequence with a safely shutdown reactor. To succeed, they must correctly diagnose the event and act as needed to deal with complex situations and system interactions. The St. Lucie PRA considers the influence of the operator throughout the analysis but primarily in the Accident Sequence, Systems Analysis, and Recovery Analysis tasks. In these tasks, the human reliability discipline was applied to assure proper incorporation of human failure events into accident sequence models, system fault trees, and integrated sequence cutsets. Human Reliability/Recovery Analysis (HRA) was used to provide guidelines for human failure treatment, model review, assignment of screening values and other quantification assistance. HRA is also instrumental in review of preliminary results, refined quantification of human failure events and incorporation of potential recovery scenarios, most of which involve human actions.

The HRA consists of two major efforts: (1) a model incorporation effort and (2) a detailed analysis. The model incorporation effort is a direct support function to the PRA model development. The accident delineation and system fault tree efforts required the identification of credible human failure events (HFE) for inclusion in their PRA models.

The HRA is scoped according to currently available HRA techniques and the typical requirements of the IPE [Ref. 3.4-1]. The results of the HRA are incorporated into the PRA as human failure events which have both point and range estimates for the occurrence probabilities associated with them.

The primary manner in which the scope of the HRA is defined is by the *types* of human failure events that are incorporated into the PRA. HFE events were classified as being those that occur prior to the initiator (pre-initiators), or HFE that occur after the initiator (post-initiators).

Notice that human-induced initiators were not separately identified since their impact would not significantly affect subsequent performance responses and, hence, would not be distinguishable from an initiator due to a hardware failure of the same functional type. Note also that so-called commission errors, in the sense of being due to misdiagnosis or faulty decision making (i.e., cognitive commission errors), are not modeled. There is no established method for modeling this type of HFE, and in accordance with Reference 3.4-1 p. C-19, no new methods were developed. Finally, time-dependency is generally not considered by HR analysts as relevant to pre-initiator tasks, i.e., maintenance, testing, and calibration (MTC), and, hence, only time-independent pre-initiators were considered.

Pre-initiators are considered time-independent and are due to what HRA calls the *slip* failure mode [Ref. 3.4-2]. Post-initiators may have time-dependent or time-independent characteristics and hence may involve slips or time related failures (i.e., the failure to respond in time which may or may not be due to a *mistake*).

The human failure probabilities presented here are based on published generic information from other analyses, simulator evaluations, and insights from past PRA. Site specific information from St. Lucie is used when appropriate.

The HR analyst reviewed St. Lucie procedures, walked down the plant and control facilities and had discussions with various plant personnel in the Operations and Training Departments. The HR analyst also reviewed the PRA logic models for the appropriateness of included HFE and to search for possible omissions. The logic models had been developed under general HRA guidelines but a continuing review and discussion process assured that the models made sense relative to HRA. All final non-recovery HFE were identified and associated with sequence cutsets only upon approval from the HR analyst.

Since pre-initiator HFE only add another failure mode to systems and components that must already be modeled, and since the data used to quantify component failure rates does not segregate human root causes, it would be conservative to omit pre-initiators entirely. As a result, the HRA focused on those MTC actions that might lead to failures of multiple trains of equipment, thus acting like common-cause failures.

A final scope/incorporation concern involves how much the HRA may be said to be St. Lucie specific. This is essentially a "data" question. First, there are no databanks of HRA data in the sense of being taken from nuclear utilities and being appropriate to severe accident conditions, especially relative to post-initiator actions. NUREG/CR-1278, referred to as THERP [Ref. 3.4-3], contains tables of HRA estimates (Chapter 20), some of which may have originated from the analysis of data, but none of which are from the nuclear utility setting. NPRDS, LER, and other industry data efforts also do not provide sufficient distinction of the human root causes of events thus making them ineffective as HRA data sources.

Hence, all current HRA techniques "generate" their own data, often simply as expert judgment. As such, the HRA techniques used in this PRA were "generic." However, this judgment process, one that is required for any type of HRA, incorporates as much information as possible from St. Lucie and is influenced by plant specifics such as configuration, procedure, practice, and training. The HFE identified in the HRA were specific to St. Lucie by being based on the St. Lucie emergency operating procedures and other procedures, the crew structure of St. Lucie crews, and the fact that the instrumentation and controls of the St. Lucie control rooms include that which the procedures and HFE would require for operation. In the case of actions that seem to be time-dependent, available and expected response times are integral factors. The available times are provided by people who understand the process timing and thus are simply an input of the HRA, not a product of the HRA. Response times could be taken from operator estimates or simulators as surrogate data for in-control actions, and from walkdowns for ex-control actions. As a result, the HRA results for the St. Lucie PRA might be described as being generated by generic models and insights but influenced by specific St. Lucie information and configuration.

As a check on the HRA and PRA integration, the recovery analysis task reviews the cutsets. This allows for the integration of the human and hardware elements of the PRA, including a full timing of the elements that are time-dependent. This also allows for the context of the cutset to be specified in detail so that the HFE included make sense operationally and procedurally and so that an opportunity is afforded to identify additional HFE as necessary.

Hence, although the HFE quantification is straightforward, there is a considerable subjectivity in qualitatively basing the choice of parameters and a commensurate uncertainty in the modeling effort. To compensate somewhat for these uncertainties, as complete as possible qualitative basis is provided for each important HFE. In this way, the HRA may rely more on qualitative analysis than other elements of the PRA.

3.4.2 HRA Methodology

Pre-initiator HFE were screened at 0.003 [a nominal value from Reference 3.4-3], with a beta factor of 0.1 for multiple train events were modeled. All post-initiator HFE were specifically quantified and thus no screening values were utilized.

There were two types of techniques used for quantification of HFE and recovery events: one for "time-independent" HFEs and the other for "time-dependent" HFEs. These techniques were considered by the general PRA community to be adequate for the purposes of IPE and for the HFE types mentioned in Section 3.4.1.

The time-independent technique was applied to slips, whether occurring pre-initiator or post-initiator. The time-independent technique is a variant on the "technique for human error rate prediction" (THERP) [Ref. 3.4-3], and is similar to the ASEP HRA procedure [Ref. 3.4-4].

The time-dependent technique was applied to untimely responses (i.e., the major decisional actions made from the control room or the equipment manipulations made locally, ex-control room). The time-dependent technique is a system of TRCs developed by SAIC [Refs. 3.4-5 and 3.4-6] that is similar to the HCR [Ref. 3.4-7] and RMIEP [Ref. 3.4-8] TRC methods.

Both techniques are parametric in the sense that the analyst must identify certain types of qualitative information, which then induces a choice in the value of one or more model parameters. These parameters then comprise the independent variables of the quantification algorithms, the parameters being quite different for the different techniques. The parameters are:

For the time-independent technique:

1. A basic human failure probability (taken from the Chapter 20 tables in Reference 3.4-3, or a default of "0.003",

2. A multiple component beta factor, or a default of "1",

3. A dependency factor for other personnel, or a default of "1", and

4. Any number (typically less than 2) of performance shaping factors [Ref. 3.4-3 or analyst's judgement].

The net estimate of HFE occurrence probability is the product of the above parameters. The range on this estimate follows from considering this point estimate as lognormally distributed with an error factor, the default of which is "10".

For the time-dependent in-control room technique:

1. A net available time as the difference between total available time and other times as human factors considerations require, such as a cue that occurs at a time later than time zero, or an action that takes more than a minute or so to carry out,

2. A type factor: "0.25" for verifications actions; "0.5" for rule-based actions; "1" for others,

3. A success likelihood factor to reflect various performance shaping factors, or a default to the nominal value of "0.5", which was typically done for this HRA,

4. A burden factor: "1" when no burden is assumed; "2" otherwise, and

5. A model uncertainty factor, which is fixed at "1.68".

The net estimate of HFE occurrence probability is generated by feeding the above parameters into a doubled error factor lognormal distribution. The range on this estimate follows from the second uncertainty factor.

For the time-dependent ex-control room technique:

1. A net available time as the difference between total available time and other times as human factors considerations require,

2. An estimate from operations personnel's judgement or walkdowns of the expected time to locate, access, and manipulate the equipment,

3. Additional time to reflect the potential delaying effects of specific types of ex-control room hazards (e.g., contamination, steam),

4. A hazard factor to reflect the uncertainties due to specific types of ex-control room hazards, and

5. A model uncertainty factor, which is fixed at "1.68".

The net estimate of HFE occurrence probability is generated by feeding the above parameters into a doubled error factor lognormal distribution. The range on this estimate follows from the second uncertainty factor.

Finally, the issue of dependency is critical related to HRA since the actions of people, successful or not, are highly context dependent. Interpersonnel dependencies were modeled explicitly for slips. The crew in the control room is modeled as a unit for untimely responses. Ex-control actions post-initiator are assumed performed by a single person.

3.4.3 Results

The result of the HRA is an HFE and recovery event Database consisting of the HFEs and recovery events, each of which includes the following minimum information:

1. Applicable unit,
2. Its event designator,
3. A descriptor of the event,
4. A type identification, at a minimum: pre-initiator slip, post-initiator untimely response, or post-initiator slip,
5. A mean occurrence probability estimate.

Note that the last estimate is the result of a detailed quantification for post-initiator HFE and recovery events and the screening value for pre-initiator HFE.

Table 3.4-1 is the Database for pre-initiator HFE events and Tables 3.4-2 and 3.4-3 provide the Database for post-initiator and recovery events.

Typical documentation for quantified post-initiator HFE and recovery events include:

1. Recovery Analysis Worksheet - event summary and non-recovery probability
2. HFE Record Basis Sheet - event description with bases for quantification of non-recovery probability (event type, failure mode, timing considerations, etc.)
3. HFE Record Sheet - SAIC ORCA worksheets showing calculation results including mean, 5th, and 95th percentile estimates.

3.4.4 Section 3.4 References

3.4-1 NUREG-1335, "Individual Plant Examination, Submittal Guidance," August 1989.

3.4-2 J. T. Reason, "Absent-Mindedness and Cognitive Control," in J. Harris & P. Morris (eds.), Everyday Memory, Actions and Absent-Mindedness, London: Academic Press, 1983, pp. 113-132.

3.4-3 NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications: Final Report," August 1983.

3.4-4 NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," USNRC, Washington, D.C., February 1987.

3.4-5 Dougherty, E.M. and J.R. Fragola, Human Reliability Analysis, John Wiley and Sons, New York, NY, 1988.

3.4-6 E. M. Dougherty, "An Ex-Control Room Human Reliability Model," Transactions of the 1989 Winter Meeting of the American Nuclear Society, TANSAO 60 1-792, November 28, 1989.

3.4-7 EPRI NP-6560-L "A Human Reliability Analysis Approach Using Measurements for Individual Plant Examination," Electric Power Research Institute, Palo Alto, CA, 1990.

3.4-8 NUREG/CR-4834, "Recovery Actions in PRA for the Risk Methods Integration Program (RMIEP)," Vol. 2, USNRC, Washington, D.C., February 1987.

TABLE 3.4-1
PRE-INITIATOR HFE EVENTS

UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
1	AHFL109108	AFW PUMP 1A MANUAL VALVE V09108 MISPOSITIONED	SLIP	3.00E-03	10
1	AHFL109124	AFW PUMP 1B MANUAL VALVE V09124 MISPOSITIONED	SLIP	3.00E-03	10
1	AHFL109140	AFW PUMP 1C MANUAL VALVE V09140 MISPOSITIONED	SLIP	3.00E-03	10
1	BHFL1LVL	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT LEVEL SENSORS	SLIP	3.00E-04	10
1	BHFL1PRS	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS	SLIP	3.00E-04	10
1	DHFL1HVS1A	OPERATOR FAILS TO RESTORE HVS 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1B	OPERATOR FAILS TO RESTORE HVS 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1C	OPERATOR FAILS TO RESTORE HVS 1C FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1D	OPERATOR FAILS TO RESTORE HVS 1D FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	EHFL1EDG1A	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	EHFL1EDG1B	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	GHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	GHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	HHFL1STBYA	OPERATOR FAILS TO PUT CTMT AIR COMPRESSOR 1A IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYB	OPERATOR FAILS TO PUT CTMT AIR COMPRESSOR 1B IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYC	OPERATOR FAILS TO PUT AIR COMPRESSOR 1C IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYD	OPERATOR FAILS TO PUT AIR COMPRESSOR 1D IN STANDBY	SLIP	3.00E-03	10
1	JHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1SDA	OPERATOR FAILS TO RESTORE SDC HX 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1SDB	OPERATOR FAILS TO RESTORE SDC HX 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	LHFL1PUMPA	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	LHFL1PUMPB	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	MHFL1V2154	VALVE V2154 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10

TABLE 3.4-1
PRE-INITIATOR HFE EVENTS

UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
1	MHFL1V2155	VALVE V2155 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	NHFL1PPCCF	COMMON CAUSE MISCALIBRATION OF PRESSURIZER PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1SGPCF	COMMON CAUSE MISCALIBRATION OF THE STEAM GENERATOR PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1CPCCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1CRMCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT RADIATION MONITORS	SLIP	3.00E-04	10
2	AHFL209108	AFW PUMP 2A MANUAL VALVE V09108 MISPOSITIONED	SLIP	3.00E-03	10
2	AHFL209124	AFW PUMP 2B MANUAL VALVE V09124 MISPOSITIONED	SLIP	3.00E-03	10
2	AHFL209140	AFW PUMP 2C MANUAL VALVE V09140 MISPOSITIONED	SLIP	3.00E-03	10
2	BHFL2LVL	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT LEVEL SENSORS	SLIP	3.00E-04	10
2	BHFL2PRS	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS	SLIP	3.00E-04	10
2	DHFL2HVS1A	OPERATOR FAILS TO RESTORE HVS 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1B	OPERATOR FAILS TO RESTORE HVS 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1C	OPERATOR FAILS TO RESTORE HVS 1C FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1D	OPERATOR FAILS TO RESTORE HVS 1D FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	EHFL2EDG2A	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	EHFL2EDG2B	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	GHFL2PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	GHFL2PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	HHFL2STBYC	OPERATOR FAILS TO PUT AIR COMPRESSOR 1C IN STANDBY	SLIP	3.00E-03	10
2	HHFL2STBYD	OPERATOR FAILS TO PUT AIR COMPRESSOR 1D IN STANDBY	SLIP	3.00E-03	10
2	JHFL2PUMPA	OPERATOR FAILS TO RESTORE PUMP 2A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	JHFL2PUMPB	OPERATOR FAILS TO RESTORE PUMP 2B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10

TABLE 3.4-1 PRE-INITIATOR HFE EVENTS					
UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
2	JHFL2SDCA	OPERATOR FAILS TO RESTORE SDC HX 2A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	JHFL2SDCB	OPERATOR FAILS TO RESTORE SDC HX 2B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	LHFL2PUMPA	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	LHFL2PUMPB	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	MHFL2V2154	VALVE V2154 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	MHFL2V2155	VALVE V2155 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	NHFL2PPCCF	COMMON CAUSE MISCALIBRATION OF PRES-SURIZER PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2SPCCF	COMMON CAUSE MISCALIBRATION OF THE STEAM GENERATOR PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2CPCCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2CRMCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT RADIATION MONITORS	SLIP	3.00E-04	10

TABLE 3.4-2 ST. LUCIE UNIT 1 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
1	R#AFWCMP	OPERATOR FAILS TO ACTUATE AFW COMPONENT(S)	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	R#AFWXVLVS	OPERATOR FAILS TO MANUALLY OPEN AFW X-TIE VLVS AND SG FLOW VLVS	3.68E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
1	R#DCAB	OPERATOR FAILS TO REALIGN POWER SUPPLY TO 125VDC BUS AB	5.57E-03	3	IN CR	MISTAKE	NA	RESPONSE	NO	4	30	1,3	HU
1	R#DGFO	OPERATOR FAILS TO OPEN DG FO FILL VALVE BYPASS	3.68E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
1	R#PCSIAS	OPERATOR FAILS TO RESTORE PCS FOLLOWING ESFAS ACTUATION	3.5E-01	10	IN CR/EX CR	SLIP/ INADEQUATE RESPONSE	1	NA	NA	EX CR - 30	EX CR - 43	1	HU
1	R#PCSMFW	OPERATOR FAILS TO RECOVER MAIN FEED WATER	1.21E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	5	55	1,3	HU
1	RAFW1CST	OPERATOR FAILS TO SWITCH AFW TO UNIT 2 CST	3.15E-04	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	20	520	1,2	HU
1	RAFW1SGTR	OPERATOR FAILS TO REALIGN AFW AND ISOLATE THE FAULTED SG FOLLOWING SGTR	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RCSS1CSAS	OPERATOR FAILS TO ACTUATE CSS COMPONENTS	6.00E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RCST1TWST	OPERATOR FAILS TO PROVIDE LONG TERM MAKEUP TO CST VIA TWST	1.55E-05	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	530	1,2	HU
1	RCVC1RWT	OPERATOR FAILS TO SWITCH CHARGING PUMP SUCTION TO RWT	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2	1.88E-02	6.4	IN CR	MISTAKE	NA	RULE	YES	10	60	1	HU
1	RHPS1RAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING RAS	1.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RHPS1SIAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING SIAS	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RHVA1ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF PWR	5.59E-03	10	IN CR/EX CR	MISTAKE/ INADEQUATE RESPONSE	1	RULE	NO	IN CR - 30/ EX CR - 5	IN CR - 120/ EX CR - 90	1,2,4	HU
1	RIA1AB	OPERATOR FAILS TO ALIGN 1A OR 1B IA COMPRESSOR	5.49E-04	10	IN CR/EX CR	MISTAKE/ INADEQUATE RESPONSE	1	RULE	NA	IN CR - 120/ EX CR - 45	1080	1,2	HU
1	RLPS1HTLEG	OPERATOR FAILS TO INITIATE HOT LEG RECIRC	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RPCS1XFSD	OPERATOR FAILS TO RESTORE PCS FOLLOWING EXCESSIVE FEEDING OF SG'S	1.67E-02	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	11	60	1	HU

TABLE 3.4-2 (Cont'd) ST. LUCIE UNIT 1 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
1	RPPCI-PORV	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH	3.00E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RPPCIBLPWR	OPERATOR FAILS TO RESTORE POWER TO PORV BLOCK VALVE	2.04E-01	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	5	11	1,1	HU
1	RTOPIRLTC	OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RTOPIROTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOPISIOTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SI LOCA	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOPISIRCP	OPERATOR FAILS TO SECURE RCP'S FOLLOWING LOSS OF SEAL COOLING	3.00E-04	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RTOPITERM	OPERATOR FAILS TO TERMINATE LEAKAGE ON FAULTED SG	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RTOPITLTC	OPERATOR FAILS TO IMPLEMENT SDC (TRANSIENT)	4.38E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	180	1440	1	HU
1	RTOPITOTC	OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING FOR TRANSIENT	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOPIWBOR	OPERATOR FAILS TO BORATE DURING ATWS	6.00E-03	10	IN CR	SLIP	4	NA	NA	NA	NA	NA	HU
1	RTOPWLTC	OPERATOR FAILS TO IMPLEMENT SDC (ATWS)	4.38E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	180	1440	1	HU

TIMING SOURCE KEY:

1 - OPERATOR ESTIMATE

2 - WALKDOWN

3 - MAPP

4 - PROCEDURES

EVALUATION TYPE KEY:

HU - HUMAN

TABLE 3.4-3 ST. LUCIE UNIT 2 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
2	R#AFWCMP	OPERATOR FAILS TO ACTUATE AFW COMPONENTS	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	R#AFXVLVS	OPERATOR FAILS TO UTILIZE AFW X-CONNECT VLVS	3.68E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
2	R#2PCSMFW	OPERATOR FAILS TO RECOVER MAIN FEED WATER	1.42E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	5	53	1,3	HU
2	R#DCAB	OPERATOR FAILS TO REALIGN POWER SUPPLY TO 125VDC BUS AB	5.57E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	4	30	1,3	HU
2	R#DGFO	OPERATOR FAILS TO RECOVER EDG BY OPENING DG FILL VALVE BYPASS	3.68E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
2	R#PCSIAS	OPERATOR FAILS TO RESTORE PCS FOLLOWING ESFAS ACTUATION	1.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RAFW2SGTR	OPERATOR FAILS TO REALIGN AFW AND ISOLATE THE FAULTED SG FOLLOWING SGTR	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RCVC2RWT	OPERATOR FAILS TO SWITCHCHARGING PUMP SUCTION TO RWT	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1	1.88E-02	6.4	IN CR	MISTAKE	NA	RULE	YES	10	60	1	HU
2	RHPS2HTLEG	OPERATOR FAILS TO INITIATE HOT LEG RECIRC	5.31E-03	10	IN CR/EX CR	SLIP/ INADEQUATE RESPONSE	1	NA	NA	EX CR 30	EX CR 360	1,4	HU
2	RHPS2RAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING RAS	1.50E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RHPS2SIAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING SIAS	3.0E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF PWR	4.20E-02	3.2	IN CR	MISTAKE	1	RESPONSE	NO	IN CR 30	IN CR 120	1,2	HU
2	RPCS2XFSD	OPERATOR FAILS TO RESTORE PCS FOLLOWING EXCESSIVE FEEDING OF SG'S	1.67E-02	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	11	60	1	HU
2	RPPC2-PORV	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH	3.00E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RPPC2BLPWR	OPERATOR FAILS TO RESTORE POWER TO PORV BLOCK VALVE	2.04E-01	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	5	11	1,2	HU
2	RTOP2RLTC	OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RTOP2ROTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR	7.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU

TABLE 3.4-3 (Cont'd) ST. LUCIE UNIT 2 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
2	RTOP2SIOTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH-COOLING FOR SI LOCA	7.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RTOP2SIRCP	OPERATOR FAILS TO SECURE RCP'S FOLLOWING LOSS OF SEAL COOLING	3.00E-04	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RTOP2TOTC	OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING FOR TRANSIENT	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
2	RTOP2WBOR	OPERATOR FAILS TO BORATE DURING ATWS	6.00E-03	10	IN CR	SLIP	4	NA	NA	NA	NA	NA	HU

TIMING SOURCE KEY:

1 - OPERATOR ESTIMATE
2 - WALKDOWN
3 - MAPP
4 - PROCEDURES

EVALUATION TYPE KEY:

HU - HUMAN

3.5 Core Damage and Plant Damage Sequence Quantification

3.5.1 Quantification Overview

The core damage sequence integration and quantification process is based on the solution of a linked core damage sequence model which includes sequence logic, top logic models, front line system models, support system models, and logic flags. The sequence logic models were developed from the event tree models and evaluated to obtain raw sequence cutsets which are stripped of cutsets containing mutually exclusive events or cutsets which violate the success criteria of the sequences. Core damage sequence cutsets resulting from the initial solution were reviewed and non-recovery events were either "hardwired" or manually added to each appropriate cutset.

The plant damage cutsets were then determined by linking the core damage models with the containment safeguards systems models via a "Bridge" event tree and associated fault trees. Plant damage end states were then defined for input to the Level 2 analysis documented in Section 4.

The following inputs were used to generate the accident sequence cutsets:

1) Sequence Fault Tree Logic Models
2) System Fault Trees
3) Data Analysis Results
4) Human Reliability/Recovery Analysis Results
5) Logic Flag Files
6) Batch Programs
7) Macro Programs

The following codes were used for the quantification of the Level 1 PRA model. All codes were used on a personal computer.

ETA-II - Event Tree Development and Documentation [Ref. 3.5-1]

CAF386 - Fault Tree Editor/Basic Event Quantification [Ref. 3.5-2]

SAIPLOT - Fault Tree Plot Generation [Ref. 3.5-3]

CUT386 - Cut Set Solution [Ref. 3.5-2]

CSED386 - Cut Set Editor [Ref. 3.5-2]

DELEXC - Elimination of Cut Sets Containing Mutually-Exclusive Events [Ref. 3.5-4]

DELTERM - Elimination of Success Cut Sets [Ref. 3.5-4]

IPRA - File Manipulation Program [Ref. 3.5-5]

RMQS - Quantification of Accident Sequences, Importance Calculations [Ref. 3.5-5]

UNCERT - Uncertainty Analysis [Ref. 3.5-6]

A flow chart depicting the quantification tasks and the computer codes used to support these efforts is shown in Figure 3.5-1. The information on this figure is summarized below.

ETA-II is used to store the functional event trees as developed in the Accident Sequence Delineation (Section 3.1). The system and top logic fault tree models that are based upon the requirements of the event trees were created and stored on the CAFTA workstation. The human reliability analysts used SAIC's ORCA spreadsheets to estimate the human error probabilities (Section 3.4). The information developed from both of these data related codes was then used as input in order to complete the plant model. All fault tree models were plotted using SAIC's SAIPLOT program.

The fault tree models for every accident sequence (both failure and success states) were solved (i.e., cutsets were generated) using the CAF386 and CUT386 codes contained within the CAFTA workstation. Mutually-exclusive event pairs were then removed from the generated cutsets using DELEXC. DELTERM was then used to eliminate cutsets that implied the failure of systems which were defined to be successful for a given accident sequence.

Recovery analyses were then performed on the cut sets. The recovery analyses consists of: 1) modeling recovery actions identified during the review of intermediate core-damage cutsets which remained following DELEXC and DELTERM processing and 2) appending the recovery cutsets to the appropriate intermediate cutsets or hardwiring the recovery actions and re-quantifying to produce the final core-damage cutsets. This effort was completed entirely within the CAFTA workstation.

The RMQS program was used to perform sensitivity and importance analyses on the final recovered cutsets using information extracted from the PRA. This is accomplished through the use of the IPRA program which loads the final sequence cutsets, modules, and basic events into RMQS in order to perform these evaluations. Finally the UNCERT program extracts information from the RMQS database using a RMQS generated report, and performs an uncertainty analysis on the RMQS model by Monte Carlo iteration methodology.

3.5.2 Quantification Input Files

Prior to the generation of accident sequence cutsets, several input files must be developed. These input files are needed to correctly model the accident scenario and allow the cutsets to be generated (see Figure 3.5-1).

1. **Sequence Logic Files** - Accident sequences were coded into fault tree logic from the event tree model logic (Section 3.1). Fault tree top logic was developed to connect the event tree functions with the front line system fault tree logic. The top gates which represent failure branches on the event tree were "AND"ed to quantify the accident sequences, and were "OR"ed for the success branches. The success branch logic was used to remove cutsets that implied the failure of systems which were defined to be successful for a given accident sequence.

2. **Flag Files** - Some of the system models contain flags that are set to true or false depending on the plant configuration, initiator, and/or accident sequence that is being quantified. For example, these sets of flags are used to differentiate between injection and recirculation modes of operation. Also contained in the flag files are true/false settings for the initiating events which define the accident sequence represented by the selected flag file.

3. **Plant Fault Tree Model** - The front line system fault tree models are linked with the support system fault tree models and connected via the top logic to the event tree logic. This linking process forms the plant model and is accomplished with the CAF386 software.

4. **Data Bases** - Three files make up the data bases that are required whenever a fault tree model is loaded into CAF386 - a basic event file, a gate name file and a type code file. The basic event file contains the event names, descriptions and probabilities that are reflected in the fault tree model events when they are viewed using CAF386. The gate names and descriptions in the fault tree are obtained from the gate name file. The type code file contains event failure rates for the various category of components (e.g., MOVs, PUMPs ...) and failure modes (e.g., failure to open, failure to start ...). The basic event file interacts with the type code file to calculate the event probability as a function of the failure rate, exposure time and method of calculation. This interaction is driven by the component code which identifies the component type and failure mode.

5. **Mutually-Exclusive Events File** - Two events may appear in an accident sequence cutset which could not occur simultaneously. For example, it was assumed that only one initiator can occur at any one time, as well as maintenance events on separate trains of the same system. Cutsets containing these combinations of events could be removed through the use of NOT logic, but this introduces a significant amount of additional work on the part of the quantification codes. It was easier and less time consuming to remove these events from the cutset results than to add logic to the model to perform the same task. These events were placed in a "mutually-exclusive event" file and processed by DELEXC.

3.5.3 Quantification of Accident Sequences

The cutsets for an accident sequence are generated from the combined logic of two files - the accident sequence flag file and the plant model file. The combined logic models for both the success and failure portions of each accident sequence are submitted to CAF386(FTAP) to generate the input files (.FTP) for CUT386. The following paragraphs describe the computer codes that were used to solve the plant model and produce the accident sequence cutsets.

1. Generating "Raw" Cutsets: CAF386/FTAP/CUT386

CAF386 is CAFTA's Database and Fault Tree Editor which is used to build the databases and create the fault trees. CAF386 options allow the fault trees to be aggregated and modified to represent the desired accident sequence; then a CUT386(FTAP) input file is created. CUT386 is CAFTA's cutset generation program. An automated set of batch files is typically used to execute the programs since they can be set up for an overnight run. Among the reasons for developing an automated approach are accident sequences require several hours execution for cutset generation and the efficiencies associated with having the repetitive set of key strokes captured in executable batch and macro files.

2. Deleting Mutually-Exclusive Events: DELEXC

The DELEXC program was used to delete the cutsets containing mutually-exclusive pairs of events. It reads the CUT386(FTAP) output (raw cutsets) and compares the cutsets with the events in the mutually-exclusive file. If a pair of mutually-exclusive events are matched in a cutset, then the cutset is deleted. DELEXC also adds a line at the end of the FTAP output which states the number of cutsets that were deleted during the post FTAP processing. This procedure was performed on both the success and failure portions of the accident sequences.

3. Generating Final Accident Sequence Cutsets: DELTERM

The final step in obtaining accident sequence cutsets was to delete those cutsets from the failure portion of the sequence which violate the success criteria for that sequence. The significance of this can best be seen through an example. Figure 3.5-2 contains a simple event tree with an initiating event A and the system models I and II. Suppose that component X is in both the system I and system II. Failure of X will fail both system I and system II. The model for sequence number 2 (i.e., the failed portion) would be A*II. System I is assumed to have not failed. However, a cutset could be generated with the initiator A and the component X. This cutset would also fail system I which violates the system success criteria for the sequence. Thus, the DELTERM program reads the failure sequence cutsets and then deletes all cutsets which are also present in the success sequence cutsets. DELTERM would remove any cutsets containing A and X from the A*II accident sequence cutsets.

3.5.4 Recovery Analysis

The next step in the quantification process involves identifying the most important core damage sequences and their cutset contributors. These are then examined for potential recovery actions that may represent either "pure" operator actions or combinations of operator/hardware events. These are developed as "non-recovery" events and their failure probabilities assessed. These events are then applied to the core damage sequence cutsets and the "recovered" core damage sequences obtained.

Nonrecovery events were determined by (1) identifying the recovery options for each dominant cutset and by (2) developing nonrecovery events based on the review of the recovery options. The review was based on plant procedures, walkdowns, and/or the operators' training.

The nonrecovery events were then quantified (see Section 3.4). This quantification effort required the support and interaction of the Recovery, System, and Data Analysis Tasks. The Human Recovery Task developed human failure models for all post-initiator nonrecovery events. The human recovery analyst determined the task subtype (i.e., verification, rule following, response, ex-control room), mode (i.e., slip, mistake, untimely response), and technique (i.e., time-dependent vs. time-independent) classifications for this event. Means, medians, 5th and 95th percentiles for the nonrecovery event were calculated.

Manual addition of non-recovery events, using cutset editor (CSED386), to each cutset as appropriate has an advantage in that only appropriate recovery actions are included and inappropriate and/or multiple recoveries are avoided. The disadvantages are that the manual entry process is very tedious since each individual cutset must be reviewed, the appropriate recovery action must be determined and manually entered, and then, if the sequence is re-quantified later, the new sequence cutsets must be reviewed and recovery actions applied to the appropriate cutsets. For these reasons, the manual entry of non-recovery events was avoided where possible.

Hardwiring of non-recovery events was used where appropriate. The type of human event determined the logic that was applied. That is, if the event would only be required if a hardware failure occurred, the event was modelled as part of an "AND" gate with the hardware failures. For example, the HPSI pump should receive an automatic start signal following a SIAS; therefore, it would take a hardware failure of the ESFAS system to send a start signal to the pump <u>and</u> a failure of the operator to manually start the pump for the pump not to receive a start signal. In contrast, if the success of the mitigating function can only occur if some operator action is taken then the non-recovery event would be modelled as part of an "OR" gate. For example, initiation of Shutdown Cooling requires operator action; therefore, SDC will fail if the hardware (LPSI pumps, valves, heat exchangers, etc.) fails <u>or</u> the operator fails to initiate shutdown cooling.

The primary advantage of hardwiring the non-recovery events is that the events are included in the cutsets during quantification and need not be added manually after quantification. It also ensures that recovery events are applied to all applicable cutsets and not to just the most important ones (typically the manual recovery process is only applied to the most likely cutsets, and when the overall sequence probability is "low enough", the process is ended). There are disadvantages to this approach as well. Since all the non-recovery events are included in the model, the potential exists that non-independent multiple recoveries could be included in a single cutset. If so, this would result in a greatly reduced cutset probability. To compensate for this, non-recovery events are set to a probability of 1.0 during quantification such that cutsets with multiple non-recoveries will not be truncated. Then each sequence is reviewed to determine which non-recovery events are appropriately applied to that sequence. All the remaining non-recoveries are redefined as "TRUE" in the flag file such that they will not erroneously reduce the probability of valid cutsets.

3.5.5 Plant Damage Cutset Quantification

The last phase of the Level 1 quantification includes the status of the Containment Safeguards Systems (Sprays, Coolers and Isolation) and the determination of Plant Damage States. These states represent the output of the Level 1 analysis necessary for input to the Level 2 Containment Performance Analysis.

The relationship of the functional core damage sequences described above (RCS and core cooling status) and the containment safeguards systems' status is developed in the bridge tree. The entry state to the bridge tree is the set of core-damage sequences. The structure of the bridge tree is similar to that of an event tree, for which the top events would include core/containment status and containment safeguard systems-related questions. These consider the dependency of the containment safeguard systems on the entry state conditions of the functional sequence. This structured approach helps cross-check the core-damage sequence mapping into plant damage states.

The spectrum of core and containment conditions following core-damage are portrayed in the bridge tree as a set of end states which are identified with a sequence ID. The bridge tree end states that result in similar containment response and radionuclide release characterization are called Plant Damage States (PDS). The sequence ID's follow a simplified nomenclature to categorize the front-line sequence characteristics of the core-damage sequences (i.e., I, IR, II, IIR, III, IV, V and VI) that is coupled with an added identifier for the containment safeguards state (A, B, C, D, E, F, G, and H).

Mechanistically, the Containment Systems "Bridge Tree" was developed, the associated system fault trees linked and the subsequent plant model analyzed. The plant damage cutset results were then transmitted to the Level 2 containment performance and quantification effort.

3.5.6 Sensitivity/Importance Analysis

3.5.6.1 Risk Management Query System

The primary tool used to perform sensitivity analyses is the SAIC Risk Management Query System (RMQS). The RMQS processor provides the capability to quantify the total core damage probability of all accident sequences and to conduct sensitivity and importance evaluations of the PRA model. By reassembling a database from the fault tree cutsets, event tree sequences and component probability data risk measures for the plant can be readily re-evaluated for safety significance.

As examples of RMQS capabilities, reports can be generated to determine the contributions of individual components or combinations of sequences to core melt. The same type of report can also be developed for system contributors. In addition, changes to individual component probabilities and conditions can be more readily modified in the RMQS database than in the PRA model to allow snapshot "what if" sensitivity calculations.

3.5.6.2 Importance Analysis

Each of the inputs to the PRA model contribute to the frequency of core damage by a certain amount. The magnitude of this contribution is described by an importance measure. In this study, importance is measured by the Fussell-Vesely importance measure for both basic events and for systems as a whole. This importance measure is the weighted proportion of cutsets which have a given basic event or system in them. The Fussell-Vesely importance measure also indicates the maximum degree of improvement in core damage frequency that can be obtained by reducing the basic event or system failure rate to zero. The results of the importance analysis are given in Section 3.7.

3.5.6.3 Sensitivity Analysis

Sensitivity analysis involves the determination of how much the results of the quantification change for a certain change in the failure data. This is useful for evaluating the significance of uncertainty in a particular failure event to the overall results or, conversely, determining which failure events must be known with the greatest certainty to minimize the uncertainty of the results. The results of the sensitivity study are given in Section 3.7.

3.5.7 Section 3.5 References

3.5-1 SAIC ETA-II, Version 2.1c.

3.5-2 SAIC CAFTA, Version 2.2c.

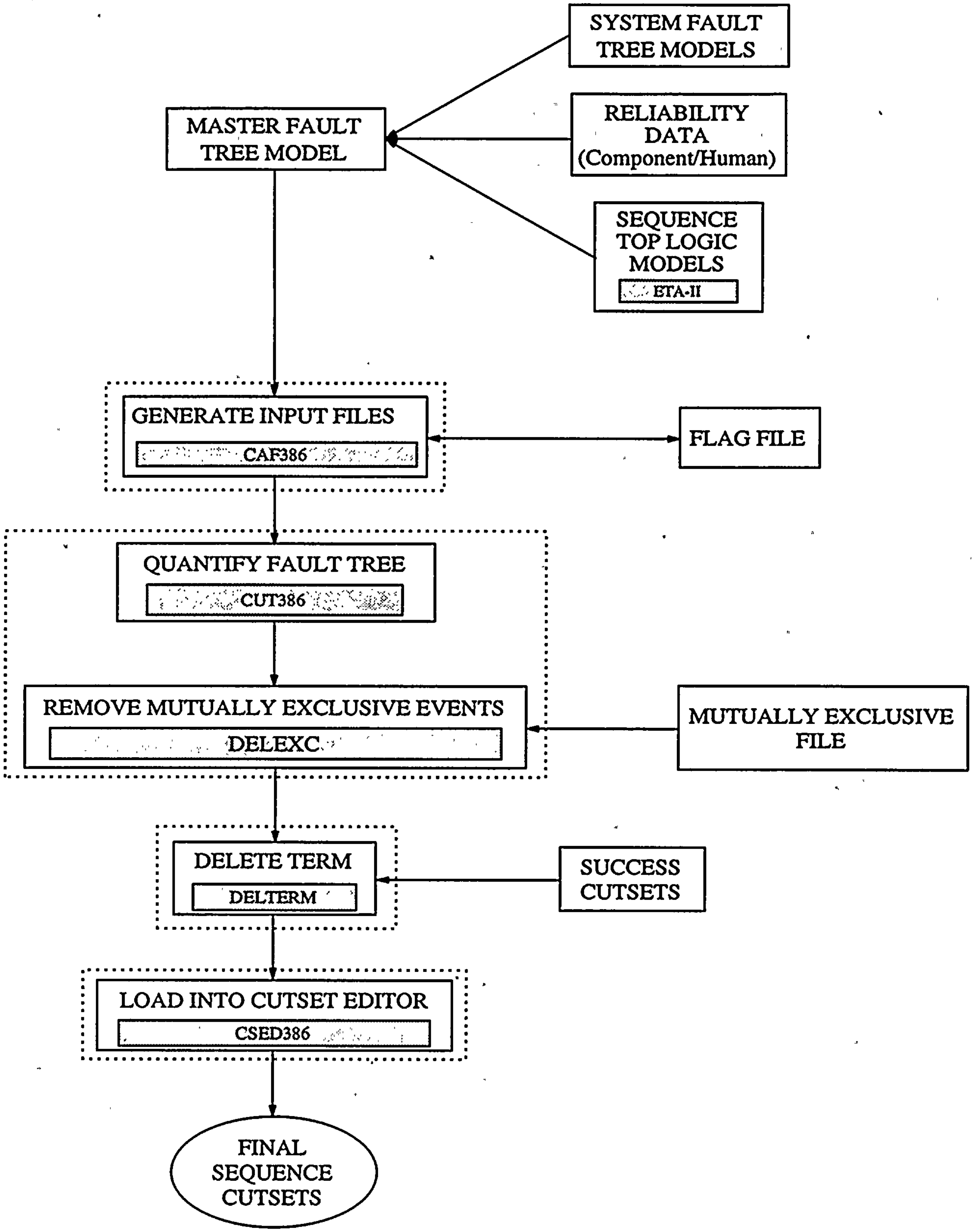
3.5-3 SAIC SAIPLOT, Version 2.2c.

3.5-4 PRAQUANT, Version 2.1d.

3.5-5 SAIC RMQS, Version 2.4f.

3.5-6 SAIC UNCERT, Version 1.1e.

Figure 3.5-1 Accident Sequence Quantification Process for St. Lucie Units 1 & 2



C:\PSLFINAL\F35-1.DRW

Figure 3.5-2 Delete Term Sample

A	I	II	SEQUENCE NUMBER	SUCCESS LOGIC	FAILURE LOGIC
			1	---	---
			2	1	A*II
			3	---	A*I

C:\PSLFINAL\GRAPHICS\FIG35-2.DRW

3.6 Internal Flood Analysis

3.6.1 Analysis Methodology

Internal floods are considered to have a potential for contributing to the overall core damage frequency if a particular flood scenario could cause a transient (i.e. initiating event) while also degrading some PRA system's mitigation capability. The general approach for this task utilizes engineering judgement and screening analysis to identify flood source/spray source/flow path/exposed component scenarios that might contribute significant risk, followed by the mapping of these scenarios onto the internal events models for quantification where required.

Flood zones were defined as identical to the fire zones described in each unit's FSAR. A zone-by-zone screening was performed utilizing a combination of plant drawing reviews, plant walkdowns, and review of previous internal flooding analyses. For screening purposes, a bounding flood and/or spray scenario was postulated within a zone. If it was concluded that the resulting flood level or spray could cause failure of a component(s), the effect of loss of the specific component(s) was then analyzed in terms of whether the affected component is associated with initiating event and/or PRA equipment (defined below). A specific zone was screened from further consideration if (1) no flood/spray source exists within the zone and the zone is not susceptible to flood or spray initiated in another zone, or (2) no postulated flood/spray scenario would cause both an initiating event and damage to PRA equipment. The zone-by-zone screening was documented by screening worksheets (Tables 3.6-1 and 3.6-2).

An initiating event is an event that causes a demand for a reactor trip, as defined in the Accident Sequence Analysis (Section 3.1). Events which result in loss of components such that a controlled, manual shutdown is required (e.g. due to technical specification requirements) are not considered initiating events in the context of this study.

PRA equipment is any equipment included in the St. Lucie Plant internal events fault trees. Only PRA equipment or related items whose failure could lead directly to failure of PRA equipment (e.g. terminal boxes) need to be considered since the fault trees specify the scope of failures that can lead to core damage. To aid in the identification of PRA equipment in various flood zones, the component database tables found in the PRA System Descriptions were used in conjunction with the Essential Equipment Lists [Ref. 3.6-2], fire zone descriptions from Section 9.5 of each unit's FSAR [Ref. 3.6-3 and 3.6-4], and the Appendix R Safe Shutdown Analyses [Ref. 3.6-5] to develop a listing of PRA Equipment by Flood Zone.

During the screening process, the potential effects of flooding and/or spray on junction boxes within a zone were considered as well as the effects on major components (e.g. motors, switchgear). A component was considered to be vulnerable to flooding if the postulated flood level could reach the component's critical height. For motors, the critical height was assumed to be the bottom of the motor casing. It was assumed that switchgear and busses would fail if they are standing in six inches of water since typical contacts of large breakers are found at approximately this level. For other types of electrical cabinets, it was assumed that the critical height was the level of the lowest exposed electrical connection. A component's vulnerability to spray was based on the type of component (e.g. outdoor vs. indoor type), its location relative to the spray source, and engineering judgement as to whether the postulated spray scenario would result in water contact on electrical connections.

For zones where a postulated flood/spray scenario could result in both an initiating event and damage to a PRA component(s), and the scenario could not be screened out by further refinement of the analysis, an initiating event frequency was estimated. A core damage frequency was then determined based on the flood/spray-induced initiating event frequency and assumed failure of the affected PRA components.

The intent of the flood analysis was to identify candidate flood scenarios for which preventive measures were needed. Where required, appropriate corrective action to reduce the risk due to such internal floods would be recommended.

3.6.2 Screening Results

Seven screening categories were identified and utilized to categorize the susceptibility of each flood zone to flood/spray scenarios. These are as follows:

1. Flood Source in Zone
2. Flood Propagation Thru Zone
3. Spray Source in Zone
4. IE (Initiating Event) Equipment Flood Damage
5. IE (Initiating Event) Equipment Spray Damage
6. PRA Equipment Flood Damage
7. PRA Equipment Spray Damage.

Tables 3.6-1 and 3.6-2 give initial screening results for all St. Lucie flood zones. The last column indicates whether, based on the above analysis methodology, the zone was initially screened out. Note that later assessments resulted in screening zones from concern that were not initially screened from concern. Therefore, the last columns of Table 3.6-1 and Table 3.6-2 indicate the results of an intermediate step in the screening process, and not the final screening results. A zone was screened out only if flood or spray would not cause an initiating event and damage PRA equipment at the same time. The flood zones not initially screened out (i.e. those with an "F" in the last column) have the zone number underlined in the first column of Tables 3.6-1 and 3.6-2. The zones not initially screened from concern on Unit 1 are 1, 3, 5, 14, 15, 19, 27, 31, 32, 33, 34, 35, 41, 47, 55E, 56, 57, 60, 62, 70, and 82. The zones not initially screened from concern on Unit 2 are 3, 6, 13, 15, 16, 19, 20, 24, 28, 34, 34*, 37, 40, 42, 44, 47, 49, 50, 51E, and 52.

Zones not initially screened from concern were later screened based on qualitative and/or quantitative assessment. The results of these assessments are documented below. One notable factor in several assessments is the role of plant operators in flood detection. These individuals travel pre-determined circuits at a prescribed frequency to take log readings on numerous plant components. Given their training and responsibility for vigilance, it is likely they would also notice flood waters or evidence of nearby flooding (such as gross leakage beneath doors) during their rounds. Thus several of the qualitative assessments take credit for early detection of flooding by plant operators.

3.6.2.1 Unit 1, Flood Zone 1 - Steam Trestle/Aux Feed Pumps Area
Unit 2, Flood Zone 6 - Steam Trestle

On one side of the steam trestle, the feedwater line is directly over the 1A (2A) and 1B (2B) AFW pumps. On the other side of the trestle, the other feedwater line is directly over the 1C (2C) AFW pump. Rupture of a feedwater line could fail either a) both motor driven AFW pumps or b) the turbine driven AFW pump only. These scenarios could not be screened out qualitatively; calculation of the contribution to core damage frequency is discussed in Section 3.6.3.

3.6.2.2 Unit 1, Flood Zone 3 - Intake Cooling Water Area
Unit 2, Flood Zone 13 - Intake Cooling Water Pumps
Unit 2, Flood Zone 49 - Intake Structure

The concern is if one pump discharge line sprays the adjacent pump motor, and the third pump is out of service, all ICW could be lost. However, several factors make this an incredible event. First, this event could only happen when A or B pump is out of service, since C is the middle pump on both units. This is a small fraction of the time for power operation (A and B are the normally running pumps). Second, only a small class of pipe breaks would have the right size, geometry and location to spray an adjacent pump's motor. Third, due to the motor's construction it is unlikely that spray would contact electrical parts. Therefore, an event of this nature is not deemed credible.

3.6.2.3 Unit 1, Flood Zone 5 - Component Cooling Area
Unit 2, Flood Zone 3 - CCW Building

1) The CCW pit on Unit 1 and CCW room on Unit 2 contain ICW and CCW piping above and below the ground elevation (18 foot elevation. Several barriers to the complete flooding of these CCW areas exist. Low flow breaks, or "cracks", have already been analyzed in the Unit 2 FSAR, Appendix 3.6.F; the analysis is assumed to be essentially equivalent for Unit 1. (Though the reference addresses safety-related equipment only, all PRA equipment in the CCW areas is safety-related.) The FSAR concludes that no safety-related equipment is damaged from the assumed crack. For large breaks of CCW piping the maximum flood depth is 2.0 feet, which is bounded by flooding from ICW. If the entire volume of the CCW system flooded either unit's CCW pit, the resulting depth of water in the pit would be less than 2.0 ft. For large breaks of ICW piping, the control room may receive low pressure alarms. On Unit 2 the same alarms exist from indicating switches. In addition to these alarms, the control room would receive high level alarms from the respective area's sump. Another barrier to complete flooding is response time available. Before the CCW pit or room can start filling, the large pipe trench to the RAB must first fill since it is at a lower elevation. Once the pit or room starts to fill (at a slower rate since it has a larger area), no PRA equipment is vulnerable until the level reaches at least the 17.2 foot elevation on Unit 1 or at least the 18.5 foot elevation on Unit 2. It is unlikely that the level would reach those elevations since plant personnel enter the CCW areas fairly frequently. Since loss of PRA

equipment due to flood is not deemed credible in this scenario, it is screened from further consideration.

2) (Unit 2 only - Unit 1 pumps are shielded from each other) The concern is that if one pump discharge line sprays the adjacent pump motor, and the third pump is out of service, all CCW could be lost. However, several factors make this an incredible event. First, this event could only happen when A or B pump is out of service, since C is the middle pump. This is a small fraction of the time for power operation (A and B are the normally running pumps). Second, only a small class of pipe breaks would have the right size, geometry and location to spray an adjacent pump's motor. Third, due to the motor's construction it is unlikely that the spray would contact electrical parts. Therefore, an event of this nature is not deemed credible.

3.6.2.4 Unit 1, Flood Zone 14,15 - Condensate Pump and Condenser Area/Condensate Pump Pit
Unit 2, Flood Zone 47 - Turbine Building

1) The Sandia analysis of this scenario [Ref. 3.6-6] suggests that a rupture of circulating water piping would cause operators to secure the circulating water pumps rapidly, and only a short gush of water would occur which would spread out over the area. However, if complete flooding of the pit is assumed, all three condensate pumps would be lost. (Note that this assumption is not based on calculated flood volume and therefore may be conservative.) Flooding of all three condensate pumps bounds any spray concerns for this zone. The flood scenario could not be screened out qualitatively; calculation of the contribution to core damage frequency is discussed in Section 3.6.3.

2) The effect of flooding/spray from the eyewash station in the Unit 2 non-1E DC equipment room was addressed. The equipment of concern is the 125V DC Bus 2D. Its role in the PRA model is only as a backup supply to the Unit 2 Vital Inverter Bus. Its loss would not significantly affect the plant since the Unit 2 Vital Inverter Bus would continue to be powered from its normal source. Since there is no initiating event connected with this scenario, it is not addressed further.

3.6.2.5 Unit 1, Flood Zone 19 - Feedwater Pumps 1A and 1B

The concern is damage to certain PTs/FTs, resulting in loss of one feedwater pump (the deluge could not affect PTs/FTs for both feedwater pumps at once). Several factors make this an incredible scenario. First, only a small class of pipe breaks would have the right size, geometry and location to deluge the FTs/PTs. Second, much of the heater drain fluid would flash to steam, reducing the amount of water available to impact the FTs/PTs. Third, the PTs/FTs are designed for exposure to the environment and it is unlikely that sufficient water would impinge upon the components to fail them. Even if this scenario was to occur, the loss of a single feedwater pump is not of great importance since the other feedwater pump, the condensate pumps, and the auxiliary feedwater pumps remain available for use. This scenario is therefore screened from further analysis.

3.6.2.6 Unit 1, Flood Zone 27 - Aerated Waste Storage Tank (AWST) Area
Unit 1, Flood Zone 55E - Main Hallway East (El 19.5)
Unit 2, Flood Zone 19 - RAB East Hallway & Miscellaneous Equipment Room
Unit 2, Flood Zone 20 - RAB East-West Common Hallway
Unit 2, Flood Zone 51E - RAB Hallway, East of Column Line RAH

The lowest initiating event or PRA equipment on the -0.5 foot elevation are the boric acid makeup pumps which are about a foot off the floor, the charging pumps which are about two feet off the floor, and DC Power Panels 254 and 255 (Unit 2 only) which are susceptible at about two feet off the floor. 0.38 foot of water would accumulate at the -0.5 foot elevation from release of the (full) aerated waste storage tank volume at the west end of the building. This agrees with Sandia's analysis [Ref. 3.6-6] which concludes that release of the AWST contents to the -0.5 foot elevation would result in less than one foot of water on the floor. (The study also concludes that release of the caustic storage tank contents is bounded by the AWST scenario.) This level is not enough to damage any of the initiating event or PRA equipment on the -0.5 foot elevation. The release of all 4 (full) hold-up tanks' contents would not result in water escaping into the rest of the RAB. Release of the contents of the RWT via the pipe tunnel pathway is bounding for this scenario (water reaches a depth of 4.3 feet). Refer to Section 3.6.3 for further discussion of this scenario.

3.6.2.7 Unit 1, Flood Zone 31 - Shutdown Heat Exchanger 1B Room
Unit 1, Flood Zone 32 - Shutdown Heat Exchanger 1A Room
Unit 2, Flood Zone 15 - Shutdown Heat Exchanger Rooms

Since the plant is assumed initially at full power, the worst case break is that of a CCW line. CCW would be lost from one train, flooding that train's room. Assumed flooding of either room bounds any spray damage concerns. The only initiating event or PRA equipment in either room is the equipment associated with that train's operation. The two trains' rooms are separated by a wall seven feet high. Each SDHX room has approximately 26,000 gallons of volume available before the wall to the next room would be topped. The total CCW system volume is estimated to be about 78,000 gallons, which is three times one room's volume. However, several factors inhibit flooding of this area and make topping the dividing wall improbable. First, it is unlikely that the full CCW system volume would enter the room. Second, each room has a drain to the ECCS pump rooms which would allow adequate drainage for smaller leaks and reduce flood volume for larger leaks. Third, the control room would be alerted to the problem from numerous alarms; the CCW surge tank has 3 control room alarms for low level and the ECCS pump room sumps each have high and high high level alarms in the control room. Finally, the RAB hallway outside the SDHX rooms is frequently travelled and has a plant operator passing through it periodically. It is unlikely that seepage from below the door to the RAB hallway would go unnoticed for long. In any event, the CCW leak would likely be terminated before 26,000 gallons escaped. Since only one CCW train is assumed affected, there is no initiating event for this scenario and it is screened from further analysis.

3.6.2.8 Unit 1, Flood Zone 33 - Pipe Tunnel
Unit 2, Flood Zone 24 - RAB Pipe Tunnel

Consistent with Sandia's approach [Ref. 3.6-6], it is assumed that doors which only have a latch keeping them shut against the force of flood water immediately open and pass the flood water. This means that a flood of the pipe tunnel on either unit is assumed to immediately flood the respective RAB open areas, since there are doors that open outward from the pipe tunnel on each unit. Several factors make early detection and mitigation of such flooding highly probable. First, the RAB hallway is frequently travelled. Second, drainage to the 1100 gallon ECCS pump room sumps would result in high and high high level alarms in the control room. Third, source-specific alarms and operator actions are likely whether the source is the RWT or CCW. If the RWT (either unit) was filled to the high high alarm level, only about 136,000 gallons could spill before a low level alarm would sound in the control room. This would certainly alert operators to the source of flooding, and flooding is easily terminated by closing the RWT outlet MOVs. If this operator action is not taken, the scenario is still screened out based on the final level after continued flooding of the RAB until the RWT level drops to the recirculation actuation setpoint where its outlet MOVs automatically close. If the operators additionally fail to close the "Drain to Safeguards Room Sumps" valves, as directed by ONOP 1-0210031 or 2-0210031 [Ref. 3.6-7, 3.6-8] any resultant flooding of the ECCS pump rooms is bounded by the analysis in Section 3.6.2.9. If the operators do close these valves, 4.3 feet of water could accumulate on the -0.5 foot elevation and would result in failure of the boric acid makeup pumps, charging pumps, and - on Unit 2 only - DC power panels 254 and 255. However, no initiating event occurs as a result of this scenario and it is screened from further analysis. Flooding from CCW in the pipe tunnel is considered bounded by the RWT analysis, since CCW is a closed system containing only about 78,000 gallons. Also, timely identification and termination of CCW leakage is supported by surge tank low level alarms and off-normal procedures for loss of CCW.

3.6.2.9 Unit 1, Flood Zone 34, 35 - 1A and 1B Emergency Core Cooling System Rooms
Unit 2, Flood Zone 16 - ECCS Pump Rooms

Early detection and mitigation are highly probable, since each train's room has a sump with high and high high level alarms in the control room. (The sumps are only 1100 gallon capacity, so they would fill quickly.) Further, if the RWT (either unit) was filled to the high high alarm level, only about 136,000 gallons could spill before a low level alarm would sound in the control room. This would certainly alert operators to the source of flooding, and flooding is easily terminated by closing the RWT outlet MOVs. If this operator action is not taken, the scenario is still screened out based on the final level following continued flooding of an ECCS pump room until the RWT level drops to the recirculation actuation setpoint where its outlet MOVs automatically close. One of two things could happen. First, the water could be contained in the room due to the watertight doors installed. (The room floor is at the -10 foot elevation; the doors are at the -0.5 foot elevation.) The room would flood to the 12.8 foot elevation. Since HVAC ductwork is at the 15 foot elevation, transport of the water via HVAC openings is not a concern. This scenario would result in the failure of all HPSI, LPSI, and containment spray pumps, but would cause no initiating event. The other possibility is failure of the watertight doors or other penetrations to contain the water, resulting in flood of the -0.5 foot elevation of the RAB. The resulting water depth on the -0.5 foot elevation would be about 2.1 feet. This water level would damage the boric acid makeup pumps, charging pumps, and - on Unit 2 only - DC Power Panels 254 and 255, in addition to all HPSI, LPSI, and containment spray pumps. Again, there would be no initiating event (see Section 3.6.2.8.) Since there is no initiating event in either scenario, both are screened from further analysis.

3.6.2.10 Unit 1, Flood Zone 41 - Hold-up Tank Enclosure
Unit 2, Flood Zone 40 - Hold-up Tank Cubicles

This zone is sealed up to the 19.5 foot elevation where the walls separating the four tanks end and water could communicate between tank cubicles. There is a fire door at the 19.5 foot elevation, so all four cubicles would have to fill before spilling significant water into the RAB. There is a total volume of over 200,000 gallons available up to the 19.5 foot elevation. Since the combined volume of all 4 tanks is 160,000 gallons, no water could leave the hold-up tank rooms. No initiating event or PRA equipment would be damaged inside or outside this zone from a release equal to the combined volume of all 4 hold-up tanks, so the scenario is screened from further analysis.

3.6.2.11 Unit 1, Flood Zone 47 - "AB" Switchgear Room
Unit 2, Flood Zone 28 - "A/B" Switchgear Room

Spray from overhead piping in the AB switchgear room could disable an MG set or the AB switchgear. A conservative assumption is made that a single spray source might affect both. Loss of an MG set would trip the reactor. The AB switchgear powers "C" ICW pump and "C" CCW pump, which are normally in standby. This scenario could not be screened out qualitatively; quantification of the contribution to core damage frequency is discussed in Section 3.6.3.

3.6.2.12 Unit 1, Flood Zone 56 - "B" Switchgear Room
Unit 2, Flood Zone 34 - "B" Switchgear Room

Flooding to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to fire pump auto start alarms in the control room which would prompt investigation, and the vigilance of plant operators (and other personnel) who pass through this room periodically. However, spray could impact components in this room. On Unit 1 spray could affect the 1B5 MCC, but no initiating event would result. Therefore, spray is not considered further for Unit 1. On Unit 2, spray could affect the 2B DC Bus, which would cause an initiating event and loss of PRA equipment. (Note that although spray from different sections of piping could affect other equipment in the room, no initiating event occurs.) This scenario could not be screened out qualitatively; calculation of the contribution to core damage frequency is discussed in Section 3.6.3.

3.6.2.13 Unit 1, Flood Zone 57 - Cable Spread Room
Unit 2, Flood Zone 52 - Cable Spread Room

Flooding to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to fire pump auto start alarms in the control room which would prompt investigation, and the vigilance of plant operators (and other personnel) who pass through this room periodically. However, spray could affect components in this room. On Unit 1, spray could damage the Vital AC Bus; in the plant model loss of this bus affects turbine runback, the feedwater control system, and steam dump to the condenser. It is conservatively assumed that a reactor trip would occur due to loss of feedwater. This scenario could not be screened out qualitatively; calculation of the contribution to core damage frequency is discussed in Section 3.6.3. On Unit 2, spray could damage 120VAC Instrument Bus MB and cause a reactor trip. However, the increase in the probability of reactor trip is negligible. Using the same pipe rupture failure rate calculated for the Unit 1 scenario (see Section 3.6.3) results in an initiating event frequency increase of 3.6E-6/year over the baseline model frequency of 9.89E-2/year. This is an increase of about 0.004%, which is considered negligible. Therefore, the Unit 2 scenario is screened from further analysis.

3.6.2.14 Unit 1, Flood Zone 60 - "A" Switchgear Room
Unit 2, Flood Zone 37 - "A" Switchgear Room

Flooding resulting from the eyewash station piping to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to the small diameter and low pressure of the source, a drain near the source, and the vigilance of plant operators who pass through this room periodically. Therefore this scenario is screened from further analysis.

3.6.2.15 Unit 1, Flood Zone 62 - Resin Addition Tank Area
Unit 2, Flood Zone 50 - Boric Acid Batching Room

Flooding from broken or open piping in this area would find its way to the -0.5 foot elevation via drains and the stairwell. Due to the high visibility of water falling down the stairwell and the vigilance of plant operators, such a leak would be discovered and terminated in a timely manner. Flooding from the BA batch tank is limited to its volume, 636 gallons. 0.03 foot of water would accumulate on the floor if the entire volume of the BA batch tank were released. No initiating event or PRA equipment damage results from this small quantity of water, so the scenario is screened from further analysis.

3.6.2.16 Unit 1, Flood Zone 70 - Control Room
Unit 2, Flood Zone 42 - Control Room Envelope

The flow rate from a failure of the sink/washroom plumbing would be minimal due to the small size of the piping. It is not deemed credible that a quantity of water sufficient to result in component damage would accumulate in the control room; the control room is continuously occupied, so any leakage or flooding would be quickly identified and rectified before damage occurred to PRA or initiating event equipment. This conclusion is in agreement with Sandia's analysis [Ref. 3.6-6]. This scenario is screened from further analysis.

3.6.2.17 Unit 1, Flood Zone 82 - Component Cooling Water Surge Tank Room
Unit 2, Flood Zone 44 - Component Cooling Water (CCW) Surge Tank Room

Timely identification of CCW leakage is supported by CCW surge tank low level alarms. Each unit's surge tank has a volume of only 2000 gallons, so level would drop quickly on a leak too large for floor drains to accommodate. There are 3 control room alarms for CCW surge tank low level. An "N" train alarm occurs at 3 feet from the bottom of the tank. Low level alarms for A and B trains each occur at 2.5 feet from the bottom of the tank. If it is assumed, consistent with Sandia's approach [Ref. 3.6-6], that doors which only have a latch keeping them shut against the force of flood water immediately open and pass the flood water, no substantial flooding could occur since on both units doors open to a roof from the CCW surge tank room. More importantly, since each surge tank is baffled to ensure that a failure of one train will not disable the redundant train, the worst case scenario is loss of a single train of CCW. It is assumed that this scenario would result in a controlled, manual shutdown; thus there is no initiating event and the scenario is screened from further analysis.

3.6.2.18 Unit 2, Flood Zone 34* - "A" Train DC Equipment Room (in Zone 34)

Flooding in this zone is not deemed credible due to the exclusion of Zone 34 as a flood source (see Section 3.6.2.12).

3.6.3 Core Damage Frequency Calculations

The following addresses the specific core damage frequency calculations performed for internal flooding/spray scenarios which were determined during the screening to have the potential to cause both an initiating event and damage to PRA equipment. The software used for the calculations was RMQS.

3.6.3.1 CDF Calculation for Unit 1, Flood Zone 1; Unit 2, Flood Zone 6

To determine the increase in core damage frequency due to deluge of AFW pumps from rupture of MFW piping, four core damage frequency calculations were performed using RMQS. One calculation for each unit modeled failure of the MFW line over the motor driven AFW pumps, and one calculation for each unit modeled failure of the MFW line over the turbine driven AFW pump. Since initiating events already exist in the models for feedline breaks in these locations (ZZT3DU1A, ZZT3DU1B, ZZT3DU2A, and ZZT3DU2B), the only change in the model required was to account for loss of AFW pump(s). This was accomplished by increasing the appropriate failure-to-start event(s) by ten percent. Ten percent was chosen as a bounding value, since there is no way to accurately determine the true failure-to-start contribution of deluge from MFW line rupture. It was estimated that less than ten percent of the MFW piping represented by the feedline break initiating events is located directly over AFW pumps.

The increase in core damage frequency for deluge of the motor driven AFW pumps is 1E-7/year for Unit 1 and less than 1E-7/year for Unit 2. The increase in core damage frequency for deluge of the turbine driven AFW pumps is 2E-7/year for Unit 1 and 1E-7/year for Unit 2. These contributions to core damage frequency are all much less than 1E-6/year.

3.6.3.2 CDF Calculation for Unit 1, Flood Zone 14,15; Unit 2, Flood Zone 47

To determine the increase in core damage frequency due to the loss of all condensate pumps from rupture of circulating water piping, core damage frequency was recalculated for each unit using RMQS. In each case, the Loss of Main Feedwater - Not Recoverable initiator (ZZT3CU1, ZZT3CU2) was increased by 4.04E-2/year. This increase is based on the probability of circulating water pipe ruptures at the condenser - 8 pipes at 4.8E-3/year each (3.84E-2/year subtotal) - and expansion joint ruptures - 8 joints at 2.5E-4/year each (2.0E-3/year subtotal) - for a total of 4.04E-2/year [Ref. 3.6-11, 3.6-12, 3.6-13].

The increase in core damage frequency for loss of all three condensate pumps is 2E-7/year for Unit 1 and 2E-7/year for Unit 2. These contributions to core damage frequency are both much less than 1E-6/year.

3.6.3.3 CDF Quantification for Unit 1, Flood Zone 47; Unit 2, Flood Zone 28

To determine the increase in core damage frequency due to the loss of an MG Set and the 4160V and 480V AB Switchgear from spray, the Unit 1 internal events model was modified and then requantified. The modification was an increase in the basic event probabilities for fault of the 4160V and 480V AB Buses by 3.1E-5. This increase is arrived at using an average leakage rate per pipe section of 3.5E-9/section-hour, from EPRI's "Pipe Failures in U.S. Commercial Nuclear Power Plants" [Ref. 3.6-9]:

$$(3.5\text{E-}9/\text{section-hour})(1\ \text{section})(8760\ \text{hours}) = 3.1\text{E-}5$$

No increase in the initiating event (Reactor Trip - ZZT1U1) frequency was necessary since an increase of 3.1E-5/year would be insignificant compared to the baseline frequency of 1.58/year.

The increase in core damage frequency for loss of an MG Set and both the 4160V and 480V AB Switchgear is 7.8E-15/year. This contribution to core damage frequency is far below 1E-6/year. Since the increase in core damage frequency is so low on Unit 1, no equivalent quantification was performed for Unit 2.

3.6.3.4 CDF Calculation for Unit 2, Flood Zone 34

To determine the increase in core damage frequency due to the loss of DC Bus 2B from spray, core damage frequency was recalculated using RMQS. Before recalculating, the Loss of DC Bus 2B initiating event (ZZDC2B) was increased by 3.6E-6/year. This increase was calculated using EPRI's feedwater and condensate pipe rupture failure rate of 4.1E-10/section-hour for Combustion Engineering plants [Ref. 3.6-9, Table 4.4-2]:

(4.1E-10/section-hour)(1 section)(8760 hours/year) = 3.6E-6/year

The failure rate for feedwater and condensate piping was used because EPRI's analysis does not cover fire protection piping, and feedwater/condensate piping failure rates are assumed to be consistent with those for fire protection piping. A rupture failure rate was used instead of a leakage failure rate because of EPRI's inclusion of small, low pressure piping in the rupture data [Ref. 3.6-9, page 4-2].

The increase in core damage frequency for loss of DC Bus 2B is less than 1E-7/year. This contribution to core damage frequency is much less than 1E-6/year.

3.6.3.5 CDF Quantification for Unit 1, Flood Zone 57

To determine the increase in core damage frequency on Unit 1 due to loss of the Vital AC Bus from fire protection piping spray, the internal events model was modified and then requantified. The first modification was an increase in the Vital Bus 1 Fault basic event probability by 3.6E-6. This increase was calculated using EPRI's feedwater and condensate pipe rupture failure rate of 4.1E-10/section-hour for Combustion Engineering plants [Ref. 3.6-9, Table 4.4-2]:

(4.1E-10/section-hour)(1 section)(8760 hours) = 3.6E-6

The failure rate for feedwater and condensate piping was used because EPRI's analysis does not cover fire protection piping, and feedwater/condensate piping failure rates are assumed to be consistent with those for fire protection piping. A rupture failure rate was used instead of a leakage failure rate because of EPRI's inclusion of small, low pressure piping in the rupture data [Ref. 3.6-9, page 4-2]. The second modification to the internal events model was an increase in the Loss of Main Feedwater - not recoverable initiator (ZZT3CU1) frequency by 3.6E-6/year.

The increase in core damage frequency for loss of the Vital AC Bus on Unit 1 is 2.7E-10/year. This contribution to core damage frequency is far below 1E-6/year.

3.6.4 Conclusions

Based on the above evaluation, there is no credible internal flood/spray scenario which provides a significant contribution to the overall risk of St. Lucie Units 1 and 2. The maximum cumulative increase in core damage frequency from the scenarios discussed above is 5E-7/year for Unit 1 and 5E-7/year for Unit 2. These values are both much less than 1E-6/year, the screening value for importance. This evaluation addresses internal floods initiated by both safety related and non-safety related components, and supports the conclusions of previous St. Lucie internal flood evaluations [Ref. 3.6-3, 3.6-4, 3.6-6].

3.6.5 Section 3.6 References

3.6-1. Accident Sequence Analysis Report for St. Lucie Units 1 and 2 Probabilistic Risk Assessment, PRA Task 1, Rev. 0

3.6-2. Appendix R Essential Equipment Lists, 8770-B-049 Rev. 1 (Unit 1), 2998-B-049 Rev. 1 (Unit 2)

3.6-3. St. Lucie Unit 1 Final Safety Analysis Report (FSAR), Amendment 11.

3.6-4. St. Lucie Unit 2 Final Safety Analysis Report (FSAR), Amendment 7.

3.6-5. Appendix R Safe Shutdown Analysis, 8770-B-048 Rev. 1 (Unit 1), 2998-B-048 Rev. 1 (Unit 2)

3.6-6. NUREG/CR-4710 SAND 86-1797, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop Pressurized Water Reactor".

3.6-7. St. Lucie Unit 1, Off-Normal Operating Procedure 1-0210031 Rev. 2, Flooding of Reactor Auxiliary Building.

3.6-8. St. Lucie Unit 2, Off-Normal Operating Procedure 2-0210031 Rev. 2, Flooding of Reactor Auxiliary Building.

3.6-9. EPRI TR-100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants", July 1992.

3.6-10. Procedure RRAG-PSL-011, Rev. 0, "Reliability and Risk Assessment Group Internal Flood Analysis Procedure", St. Lucie Units 1 and 2 Probabilistic Risk Assessment.

3.6-11. Generic Data Notebook for Commercial Nuclear Power Plant Probabilistic Risk Assessment, Document No. 163-03-00.

3.6-12. Internal Floods Analysis Procedure, SAIC-139-90-080, April 15, 1991.

3.6-13. NSAC-60, Oconee PRA, Volume 1, Page 9-192, Table 9-32.

3.6-14. Component Cooling Pumps Foundations - Masonry, 8770-G-671 Rev. 8, 2998-G-671 Rev. 6

Table 3.6-1: Unit 1 Flood Zones and Initial Screening Results

Flood Zone	Flood Zone Description	Flood Source in Zone	Flood Propagation Thru Zone	Spray Source in Zone	IE Equipment Flood Damage	IE Equipment Spray Damage	PRA Equipment Flood Damage	PRA Equipment Spray Damage	Screened Out
1	Steam Trestle/Aux Feed Pumps Area	T	T	T	F	T	F	T	F
2	Condensate Storage Tank Closure	T	T	T	F	F	F	F	T
3	Intake Cooling Water Area	T	T	T	F	T	F	T	F
4	Diesel Oil Storage Tank Area	T	T	T	F	F	F	F	T
5	Component Cooling Area	T	T	T	T	F	T	F	F
6	Diesel Generator 1A Room	T	T	T	F	F	T	T	T
7	Diesel Generator 1B Room	T	T	T	F	F	T	T	T
8	Refueling Water Tank	T	T	T	F	F	F	F	T
9	Primary Water Storage Tank Area	T	T	T	F	F	F	F	T
10	Gas Storage Building	F	T	F	F	F	F	F	T
11	Miscellaneous Oil Storage Building	T	T	T	F	F	F	F	T
12	Water Plant	T	T	T	F	F	F	F	T
13	Turbine Lube Oil Reservoir Area	T	T	T	F	F	F	F	T
14	Condensate Pump and Condenser Area	T	T	T	T	T	T	T	F
15	Condensate Pump Pit	T	T	T	T	F	T	F	F
15A	Condensate Polisher Area	T	T	T	F	F	F	F	T
16	Main, Auxiliary and Start-up Transformer Area	T	T	T	F	F	F	F	T
17	Turbine Cooling Water Pumps and Heat Exchanger Area	T	T	T	F	F	F	F	T
18	Instrument and Station Compressors Area	T	T	T	F	F	F	F	T
19	Feedwater Pumps 1A and 1B	T	T	T	F	T	F	T	F
20	Heater Drain Pump Area	T	T	T	F	T	F	F	T
21	Turbine Switchgear Room ("A" Switchgear Room)	T	F	T	F	F	F	F	T
22	"B" Turbine Switchgear Room ("B" Switchgear Room)	T	F	T	F	F	F	F	T
23	LP Heater Area	T	T	T	F	F	F	F	T
24	Isolated Phase Bus Area	F	T	F	F	F	F	F	T
25	Turbine Deck	T	F	T	F	F	F	F	T
26	Containment	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
27	Aerated Waste Storage Tank Area	T	T	T	T	F	T	F	F
29	Chemical Drain Tank Area	T	T	T	F	F	F	F	T
31	Shutdown Heat Exchanger 1B Room	T	T	T	T	F	T	T	F
32	Shutdown Heat Exchanger 1A Room	T	T	T	T	F	T	T	F
33	Pipe Tunnel	T	T	T	T	F	T	F	F
34	"1A" Emergency Core Cooling System Room	T	T	T	T	F	T	F	F
35	"1B" Emergency Core Cooling System Room	T	T	T	T	F	T	F	F
36	Main Hallway (El. -0.50')	T	T	T	F	F	F	F	T
36A	Charging Pump Access Hallway	T	T	F	F	F	F	F	T
37	Boric Acid Condensate Room	T	T	T	F	F	F	F	T
38	Charging Pump 1C Cubicle	T	T	T	F	F	T	T	T

Table 3.6-1: Unit 1 Flood Zones and Initial Screening Results

Flood Zone	Flood Zone Description	Flood Source in Zone	Flood Propagation Thru Zone	Spray Source in Zone	IE Equipment Flood Damage	IE Equipment Spray Damage	PRA Equipment Flood Damage	PRA Equipment Spray Damage	Screened Out
39	Gas Decay Tank Area	T	T	T	F	F	F	F	T
40	Boric Acid Make-up Tank Room	T	T	T	F	F	T	F	T
41	Hold-up Tank Enclosure	T	T	T	T	F	T	F	F
43	Control Building Personnel Room	T	T	T	F	F	F	F	T
44	Radio Chemistry Lab	T	T	T	F	F	F	F	T
44A	"A" Cable Loft Enclosure	T	T	T	F	F	F	F	T
45	Piping Penetration Room	T	T	T	F	F	F	T	T
46	Containment Purge Room	F	F	F	F	F	F	F	T
47	"AB" Switchgear Room	T	T	T	F	T	F	T	F
48	Letdown Heat Exchanger Room	T	T	T	F	F	F	F	T
49	Volume Control Tank Room	T	T	T	F	F	T	T	T
50	Demineralizer Area	T	T	T	F	F	T	T	T
51	Drumming Station	T	T	T	F	F	F	F	T
52	Boric Acid Concentrator Cubicles	T	T	T	F	F	F	F	T
54	Laundry and Decontamination Area	T	T	T	F	F	F	F	T
55E	Main Hallway - East (El. 19.5')	T	T	T	T	F	T	F	F
55W	Main Hallway West (El. 19.5'), Cable Loft Area	T	T	T	F	F	F	F	T
56	"B" Switchgear Room	T	F	T	T	T	T	T	F
57	Cable Spread Room	T	F	T	T	T	T	T	F
57B	Static Inverter Room (B Electrical Equipment Cubicle)	F	F	F	F	F	F	F	T
58	1B Battery Room	T	F	T	F	F	T	T	T
59	1A Battery Room	T	F	T	F	F	T	T	T
60	"A" Switchgear Room	T	F	T	T	F	T	F	F
61	HVAC Equipment Area	T	T	T	F	F	T	T	T
62	Resin Addition Tank Area	T	T	T	T	F	T	F	F
62B	HVE-9B Fan Room	T	T	T	F	F	T	T	T
63	Closed Blowdown Heat Exchanger Area	T	T	T	F	F	F	F	T
64	Fuel Pool Pumps and Heat Exchanger Area	T	T	T	F	F	F	F	T
65	New Fuel Container Storage Area	T	T	T	F	F	F	F	T
66	Spent Fuel Pool Area	T	T	T	F	F	F	F	T
67	Cask Washdown Area	T	T	T	F	F	F	F	T
68	New Fuel Storage Area	T	T	T	F	F	F	F	T
69	Spent Fuel Pool HVAC Area	T	T	T	F	F	F	F	T
70	Control Room	F	T	F	T	F	T	F	F
71	Control Room HVAC Area	T	F	T	F	F	F	F	T
72	Auxiliary Building Roof	T	F	T	F	F	F	F	T
73	Technical Support Center and Offices	T	T	T	F	F	F	F	T
74	Spent Resin Tank Room	T	T	T	F	F	F	F	T

Table 3.6-1: Unit 1 Flood Zones and Initial Screening Results

Flood Zone	Flood Zone Description	Flood Source in Zone	Flood Propagation Thru Zone	Spray Source in Zone	IE Equipment Flood Damage	IE Equipment Spray Damage	PRA Equipment Flood Damage	PRA Equipment Spray Damage	Screened Out
75	Charging Pump 1B Cubicle	T	T	T	F	F	T	T	T
76	Charging Pump 1A Cubicle	T	T	T	F	F	T	T	T
77	"A" Electrical Penetration Room (East)	F	F	F	F	F	F	F	T
78	"B" Electrical Penetration Room (West)	F	F	F	F	F	F	F	T
79	Raw Water Storage Tank and Fire Pump Area	T	T	T	F	F	F	F	T
80	Steam Generator Blowdown Tank Room	T	T	T	F	F	F	F	T
81	Area Between Auxiliary Building and Containment	T	T	T	F	F	F	F	T
82	Component Cooling Water Surge Tank Room	T	T	T	T	F	T	F	F
83	RAB West Stairwell	F	T	F	F	F	F	F	T

Table 3.6-2: Unit 2 Flood Zones and Initial Screening Results

Flood Zone	Flood Zone Description	Flood Source in Zone	Flood Propagation Thru Zone	Spray Source in Zone	IE Equipment Flood Damage	IE Equipment Spray Damage	PRA Equipment Flood Damage	PRA Equipment Spray Damage	Screened Out
1	Diesel Oil Storage Tank 2A	T	T	T	F	F	F	F	T
2	Diesel Oil Storage Tank 2B	T	T	T	F	F	F	F	T
3	CCW Building	T	T	T	T	T	T	T	F
4	Refueling Water Storage Tank	T	T	T	F	F	F	F	T
5	Primary Water Tank	T	T	T	F	F	F	F	T
6	Steam Trestle	T	T	T	F	T	F	T	F
7	Yard Area	F	T	T	F	F	F	F	T
8	Diesel Generator Building 2A	T	T	T	F	F	T	T	T
9	Diesel Generator Building 2B	T	T	T	F	F	T	T	T
10	Condensate Storage Tank Enclosure	T	T	T	F	F	F	F	T
11	Turbine Lube Oil Reservoir	T	T	T	F	F	F	F	T
12	Transformer Yard	F	T	T	F	F	F	F	T
13	Intake Cooling Water Pumps	T	T	T	F	T	F	T	F
14	Reactor Containment/Shield Building	N/A.	N/A	N/A	N/A	N/A	N/A	N/A	N/A
15	Shutdown Heat Exchanger Rooms	T	T	T	T	F	T	T	F
16	ECCS Rooms	T	T	T	T	F	T	F	F
17	Boric Acid Tank Room above solid floor at El. 12.08'	T	T	T	F	F	T	F	T
18	Charging Pump Rooms	T	T	T	F	F	T	T	T
19	RAB East Hallway & Miscellaneous Equipment Room	T	T	T	T	F	T	F	F
20	RAB East-West Common Hallway	T	T	T	T	F	T	F	F
21	RAB Personnel Rooms	T	T	T	F	F	F	F	T
22	Train "A" Electrical Penetration Room	F	F	F	F	F	F	F	T
23	Train "B" Electrical Penetration Room	F	F	F	F	F	F	F	T
24	RAB Pipe Tunnel	T	T	T	T	F	T	F	F
25	RAB HVAC Plenum	F	F	F	F	F	F	F	T
26	Volume Control Tank Room	T	T	T	F	F	F	F	T
27	Letdown Heat Exchanger Room	T	F	T	F	F	F	F	T
28	"A/B" Switchgear Room	T	F	T	F	T	F	T	F
29	Drumming Station Room	F	F	F	F	F	F	F	T
30	Ion Exchanger Room	T	T	T	F	F	F	F	T
31	Waste & Boric Acid Concentrators	T	T	T	F	F	F	F	T
32	PASS & Radiation Monitoring Room	T	F	T	F	F	F	F	T
33I	Instrument Repair Shop	T	T	T	F	F	F	F	T
33II	Radiochemistry Lab	T	F	T	F	F	F	F	T
33III	Sample Room	T	F	T	F	F	F	F	T
34	"B" Switchgear Room	T	T	T	T	T	T	T	F
34*	"A" Train DC Equipment Room (in zone 34)	F	T	F	T	F	T	F	F
35	Battery Room "A"	T	F	T	F	F	T	T	T

Table 3.6-2: Unit 2 Flood Zones and Initial Screening Results

Flood Zone	Flood Zone Description	Flood Source in Zone	Flood Propagation Thru Zone	Spray Source in Zone	IE Equipment Flood Damage	IE Equipment Spray Damage	PRA Equipment Flood Damage	PRA Equipment Spray Damage	Screened Out
36	Battery Room "B"	T	F	T	F	F	T	T	T
37	"A" Switchgear Room	T	F	T	T	F	T	F	F
38	ECCS 9B Ventilation Room	F	F	F	F	F	F	F	T
39	RAB HVAC Equipment Room	T	T	T	F	F	T	F	T
40	Hold-up Tank Cubicles	T	T	T	T	F	T	F	F
41	RAB Blowdown Heat Exchanger	T	T	T	F	F	F	F	T
42	Control Room Envelope	T	F	T	T	F	T	F	F
43	RAB Electrical Equipment Exhaust Fan Room	T	F	T	F	F	T	F	T
44	Component Cooling Water (CCW) Surge Tank Room	T	F	T	T	F	T	F	F
45	Fuel Handling Building HVAC Room	T	T	T	F	F	F	F	T
46	Fuel Handling Building	T	T	T	F	F	F	F	T
47	Turbine Building	T	T	T	T	T	T	T	F
48	RAB Electrical Equipment Area Supply Fan Room	F	F	F	F	F	F	F	T
49	Intake Structure	T	T	T	F	T	F	T	F
50	Boric Acid Batching Room	T	T	T	T	F	T	F	F
51E	RAB Corridor, east of column line RAH	T	T	T	T	F	T	F	F
51W	Cable Loft	T	T	T	F	F	F	F	T
51*	"A" Cable Penetration Room Extension	T	F	T	F	F	F	F	T
52	Cable Spread Room	T	F	T	T	T	T	T	F
53	RAB West Stairwell	F	T	F	F	F	F	F	T

3.7 Front-End Results and Screening

This section summarizes the St. Lucie PRA results with respect to the screening guidance for reporting given in Generic Letter 88-20 and NUREG-1335.

3.7.1 Application of Generic Letter Screening Criteria

Generic Letter 88-20 (GL 88-20) and NUREG-1335 provide two alternative criteria for reporting the results of IPEs. The first approach, defined in the generic letter, defines reporting criteria on the basis of functional sequences. The alternative approach described in NUREG-1335 utilizes systemic sequences. The St. Lucie PRA model utilizes a functional event tree approach which results in accident sequences which are based on the functional failures which lead to core damage. For functional sequence groupings, the generic letter provides the following screening criteria for reporting IPE results:

1. Any functional sequence that contributes 1×10^{-6} or more per reactor year to core damage,

2. Any functional sequence that contributes 5% or more to the total core damage frequency,

3. Any functional sequence that has a core damage frequency greater than or equal to 1×10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to PWR-4 release category of WASH-1400,

4. Any functional sequences that contribute to a containment bypass frequency in excess of 1×10^{-7}/yr, or

5. Any functional sequences that the utility determines from previous applicable PRAs or by utility engineering judgement to be important contributors to core damage frequency or poor containment performance.

Section 3.1 describes the St. Lucie PRA functional accident sequences. In addition, other events such as ISLOCA and internal flooding were evaluated separately in scoping analyses. For the purposes of reporting, these events have been considered as separate "functional" groups.

A summary of the total core damage frequency contribution of the dominant St. Lucie Unit 1 and Unit 2 IPE functional groups is provided in Figures 3.7-1 and 3.7-3, respectively. A total of 9 functional groups on Unit 1 and 11 functional groups on Unit 2 have been considered to meet at least one of the screening criteria for reporting.

The following section provides a discussion of each of the functional sequences identified as meeting the screening criteria provided in Appendix 2 of Generic Letter 88-20. Separate subsections are provided for the Unit 1 and Unit 2 results. The discussions include a description of accident progression and specific key assumptions. Additional detail on severe accident progression for all plant damage states is provided in Section 4.

3.7.1.1 St. Lucie Unit 1 Dominant Sequences

The St. Lucie Unit 1 results are summarized in Figures 3.7-1 and 3.7-2, and Tables 3.7-1, 3.7-2, and 3.7-3. The Dominant Cutsets are provided in Table 3.7-4.

3.7.1.1.1 Unit 1 Sequences > 1 x 10^{-6}/yr and/or > 5% of the Total CDF

3.7.1.1.1.1 Unit 1 Sequences TBF and TBFB

The TBF and TBFB sequences represent 27% (6.36 x 10^{-6}/yr) of the total core damage frequency (CDF) of St. Lucie Unit 1. These sequences involve a transient initiating event with failure of secondary heat removal (main and auxiliary feedwater) and failure of once-through-cooling (OTC). The normal response after a reactor trip is for the unit to stabilize at hot shutdown conditions with heat removal provided by the steam generators. This is normally accomplished by the reactor coolant pumps providing forced circulation of the reactor coolant and main or auxiliary feedwater pumps providing makeup to the secondary. If RCPs are not available, reactor coolant flow is provided by natural circulation.

If all feedwater is lost (main and auxiliary), the operator is directed by the emergency operating procedures to initiate once-through-cooling. Successful once-through-cooling requires one HPSI pump and both PORV flow paths. Once-through-cooling must be initiated before steam generator dryout (approx. 20 min.). Successful secondary heat removal can be accomplished, however, if main or auxiliary feedwater is recovered within 1 hour. For steamline breaks, it is also assumed that primary makeup via HPSI is required due to the initial overcooling and reduced inventory in the RCS.

- <u>TBFB: Station Blackout</u>

  Station blackout related (TBFB) cutsets represent 11% (2.64 x 10^{-6}) of the total Unit 1 CDF. The dominant cutsets involve a loss of grid initiating event with failure of all four emergency diesel generators (EDG's) (two on Unit 1 and two on Unit 2). Dominant failures include common cause failures of all four EDG's or failure of both Unit 1 EDG's with failure to align the blackout crosstie (operator fails to align crosstie or crosstie hardware failures).

  The loss of grid initiator results in loss of main feedwater. A subsequent loss of EDG power to both safety related 4kV buses results in unavailability of the two motor driven auxiliary feedwater pumps, both HPSI pumps, and all battery chargers. The turbine driven AFW pump is assumed to fail after battery depletion since all control and indicator power is lost (i.e., manual operation of AFW pump was not credited). Recovery of offsite power was credited for recovery of main feedwater or recovery of the motor driven AFW pumps.

The operator action associated with the Unit 1 TBFB cutsets is failure to implement the Unit 1 to Unit 2 blackout crosstie (REPS1XTIE) following a blackout on Unit 1. This action involves aligning one of the Unit 1 safety related 4 kV busses to one of the Unit 2 EDGs via the 1AB and 2AB 4 kV busses.

- TBF: Non-Station Blackout

Non-station blackout (TBF) cutsets represent 16% (3.72 x 10^{-6}) of the total Unit 1 CDF. The dominant cutsets involve a steamline break initiating event with common cause failure of HPSI injection valves to open or common cause failure of the HPSI pumps to start. As discussed above, HPSI injection is assumed to be required to makeup for the initial overcooling and reduced inventory in the RCS. Other dominant cutsets involve a loss of DC bus initiator with:

a) loss of power to the other train's safety related 4kV bus (results in unavailability of both motor driven AFW pumps, both HPSI pumps, and loss of the turbine driven AFW pump following battery depletion), or

b) failure of the other train's motor driven AFW pump and the turbine driven AFW pump.

Operator actions associated with Unit 1 TBF cutsets include failure to initiate once-through cooling (RTOP1TOTC) and failure to re-align the 1AB DC bus to the 1A DC bus following loss of the 1B DC bus (R#DCAB) (this is required to recover the turbine driven AFW pump).

3.7.1.1.1.2 Unit 1 S1X Sequence

The S1X sequence contributes 19% (4.34 x 10^{-6}/yr) to the total Unit 1 CDF. The S1X sequence involves a small-small LOCA (1/2" - 3") with successful secondary heat removal, successful short term core cooling (HPSI injection) and failure of long term core cooling. The options considered for long term core cooling are shutdown cooling (LPSI pumps and shutdown cooling heat exchangers) or high pressure recirculation and containment heat removal. To implement SDC, the RCS must be cooled and depressurized to the SDC entry conditions. High pressure recirculation is automatically initiated when the RWT level drops to 4 ft. Successful high pressure recirculation requires one HPSI pump and either one containment spray pump or 2-out-of-4 containment coolers for containment heat removal.

The dominant cutsets include:

a) common cause failures of CCW N-header isolation valves to close (fails to remove non-safety related heat load from CCW system),

b) common cause failure of containment sump valves to open,

c) common cause miscalibration of RWT level transmitters (screening value used),

d) common cause failure of ICW motor operated valves to close (fails to isolate TCW heat load from ICW system),

e) common cause failure of HPSI pumps.

The operator actions associated with Unit 1 S1X cutsets involve the operator failing to manually actuate RAS components following failure of the automatic signal (RHPS1RAS), and the operator failing to re-start electrical equipment room fans (RHVA1ELEQ) following a loss of power.

3.7.1.1.1.3 Unit 1 S1U Sequence

The S1U sequence accounts for 12% (2.74 x 10^{-6}/yr) of the total Unit 1 CDF. This sequence involves a small-small LOCA (1/2" - 3") with successful decay heat removal and failure of short term core cooling. Early core cooling represents inventory makeup to prevent core uncovery and subsequent core heatup which if uncorrected will lead to core damage. High pressure safety injection (1-out-of-2 HPSI pumps) is the primary source to perform the early core cooling function. If HPSI is unavailable, the operator must depressurize and use LPSI for core cooling.

Dominant S1U cutsets include:

a) common cause failure of HPSI injection valves to open,

b) HPSI minimum recirculation valve transferring closed,

c) common cause failure of HPSI pumps to start,

c) common cause failure of HPSI pumps to run during injection.

There are no operator actions associated with Unit 1 S1U cutsets.

3.7.1.1.1.4 Unit 1 AU Sequence

The AU sequence accounts for 10% (2.37 x 10^{-6}/yr) or the total Unit 1 CDF. This sequence involves a large LOCA (> 5") with failure of core cooling during the injection phase. Successful core cooling requires 1-out-of-2 LPSI pumps and 3-out-of-4 safety injection tanks.

Dominant AU cutsets include:

a) common cause failure of LPSI injection valves to open,

b) failure of SIT flowpaths,

c) common cause failure of SIT's due to miscalibration of level or pressure transmitters (note that screening value was used),

d) LPSI pump common discharge header flow control valve and flow control valve bypass valve transfers closed during standby,

e) common cause failure of LPSI pumps to start or run during injection.

The operator action associated with the dominant AU cutsets involves the operator failing to manually start ECCS components (RHPS1SIAS) following failure of the automatic signal.

3.7.1.1.1.5 Unit 1 AXC Sequence

The AXC sequence accounts for 5% (1.12 x 10^{-6}/yr) of the total Unit 1 CDF. This sequence involves a large LOCA (> 5") with failure of cold leg recirculation. Failure of cold leg recirculation can occur if automatic actuation and the operator fails to switchover from injection via the RWT to recirculation via the sump, or there are high pressure recirculation component failures.

Dominant cutsets include:

a) common cause failure of the CCW N-header isolation valves to close,

b) common cause failure of HPSI injection valves to open,

c) common cause failure of containment sump valves to open,

d) common cause failure of HPSI pumps to start or run during recirculation.

The operator actions associated with the Unit 1 AXC cutsets include the operator failing to manually actuate RAS (RHPS1RAS) or ECCS (RHPS1SIAS) components following failure of the automatic signal, and the operator failing to re-start electrical equipment room fans (RHVA1ELEQ) following a loss of power.

3.7.1.1.2 Unit 1 Containment Bypass Sequences > 1 x 10^{-7}/yr

3.7.1.1.2.1 Unit 1 Interfacing System LOCA (ISLOCA)

ISLOCAs contribute 8% (1.74 x 10^{-6}/yr) to the total St. Lucie Unit 1 CDF. The potential flow paths are the four safety injection lines to the RCS loops and two shutdown cooling suction lines from the RCS loops. This analysis used expressions developed and presented in NUREG/CR-5102.

For the safety injection lines, ISLOCA could occur if two check valves and a normally closed motor operated valve were to fail in the open state. The ISLOCA frequency through each Unit 1 safety injection line is estimated to be 3.02 x 10^{-8}/yr (1.21 x 10^{-7}/yr total for all four lines).

For the shutdown cooling suction lines, an ISLOCA could occur if two normally closed suction isolation valves were to fail in the open state. The ISLOCA frequency through each shutdown cooling suction line is estimated to be 8.11 x 10^{-7}/yr (1.62 x 10^{-6}/yr total for both suction lines). However, if the CE Owner's Group (CEOG) data [Ref. 3.7-4] was used in lieu of NUREG/CR-5102, the estimated CDF contribution would be less than 1.0 x 10^{-6}/yr.

3.7.1.1.2.2 Steam Generator Tube Rupture (Unit 1)

Steam generator tube ruptures contribute 4% (8.16 x 10^{-7}/yr) to the total Unit 1 CDF. The dominant contributor (4.70 x 10^{-7}/yr) is the RDX sequence. This sequence involves a SGTR with failure to isolate the faulted SG and failure of long term cooling. Two functions must be achieved in order to isolate a faulted SG: mechanical isolation and cooldown and depressurization below the SG SRV setpoint. Long term cooling involves HPSI or shutdown cooling. Dominant RDX cutsets are related to RWT faults (tank or line ruptures), common cause failures of HPSI valves, and common cause failures of HPSI pumps. The RBF sequence contributes 3.15 x 10^{-7}/yr to the total Unit 1 CDF. This sequence involves loss of secondary heat removal (MFW and AFW) and failure of once-through-cooling. Dominant RBF cutsets are related to failure of AFW, and the operator failing to recover MFW or to initiate once-through-cooling. Other SGTR related sequences contributed < 1 x 10^{-7}/yr.

The operator actions associated with the Unit 1 SGTR cutsets include the operator failing to switch charging pump suction from the VCT to the RWT before VCT depletion (RCVC1RWT), the operator failing to bypass a failed EDG fuel oil fill valve (R#DGFO), the operator failing to manually actuate ECCS components following failure of the automatic signal (RHPS1SIAS), the operator failing to initiate long term core cooling (RTOP1RLTC), and the operator failing to align the 1A or 1B instrument air compressor (RIA1AB) after loss of the normal (1C and 1D) compressors.

3.7.1.2 St. Lucie Unit 2 Dominant Sequences

The St. Lucie Unit 2 results are summarized in Figures 3.7-3 and 3.7-4, and in Tables 3.7-5, 3.7-6, and 3.7-7. The Dominant Cutsets are provided in Table 3.7-8.

3.7.1.2.1 Unit 2 Sequences > 1 x 10^{-6}/yr and/or > 5% of Total CDF

3.7.1.2.1.1 Unit 2 Sequences TBF and TBFB

The TBF and TBFB sequences represent 22% (5.67 x 10^{-6}/yr) of the total core damage frequency (CDF) of St. Lucie Unit 2. These sequences involve a transient initiating event with failure of secondary heat removal (main and auxiliary feedwater) and failure of once-through-cooling (OTC). The normal response after a reactor trip is for the unit to stabilize at hot shutdown conditions with heat removal provided by the steam generators. This is normally accomplished by the reactor coolant pumps providing forced circulation of the reactor coolant and main or auxiliary feedwater pumps providing makeup to the secondary. If RCPs are not available, reactor coolant flow is provided by natural circulation.

If all feedwater is lost (main and auxiliary), the operator is directed by the emergency operating procedures to initiate once-through-cooling. Successful once-through-cooling requires one HPSI pump and 1-out-of-2 PORV flow paths. Once-through-cooling must be initiated before steam generator dryout (approx. 20 min.). Successful secondary heat removal can be accomplished, however, if main or auxiliary feedwater is recovered within 1 hour. For steamline breaks, it is also assumed that primary makeup via HPSI is required due to the initial overcooling and reduced inventory in the RCS.

- TBFB: Station Blackout

Station blackout related (TBFB) cutsets represent 10% (2.64×10^{-6}) of the total Unit 2 CDF. The dominant cutsets involve a loss of grid initiating event with failure of all four emergency diesel generators (EDG's) (two on Unit 2 and two on Unit 1). Dominant failures include common cause failures of all four EDG's or failure of both Unit 2 EDG's with failure to align the blackout crosstie (operator fails to align crosstie or crosstie hardware failures).

The loss of grid initiator results in loss of main feedwater. A subsequent loss of EDG power to both safety related 4kV buses results in unavailability of the two motor driven auxiliary feedwater pumps, both HPSI pumps, and all battery chargers. The turbine driven AFW pump is assumed to fail after battery depletion since all control and indicator power is lost (i.e., manual operation was not credited). Recovery of offsite power was credited for recovery of main feedwater or the motor driven AFW pumps.

The operator action associated with the Unit 2 TBFB cutsets is failure to implement the Unit 2 to Unit 1 blackout crosstie (REPS2XTIE) following a blackout on Unit 2. This action involves aligning one of the Unit 2 safety related 4 kV busses to one of the Unit 1 EDGs via the 2AB and 1AB 4 kV busses.

- TBF: Non-Station Blackout

Non-station blackout (TBF) cutsets represent 12% (3.03×10^{-6}) of the total Unit 2 CDF. The dominant cutsets involve a steamline break initiating event with common cause failure of HPSI injection valves to open or common cause failure of the HPSI pumps to start. As discussed above, HPSI injection is assumed to be required to makeup for the initial overcooling and reduced inventory in the RCS. Other dominant cutsets involve a loss of DC bus initiator with:

a) loss of power to the other train's safety related 4kV bus (results in unavailability of both motor driven AFW pumps, both HPSI pumps, and loss of the turbine driven AFW pump following battery depletion), or

b) failure of the other train's motor driven AFW pump and the turbine driven AFW pump.

The operator action associated with the Unit 2 TBF cutsets is failure to initiate once-through cooling (RTOP2TOTC).

3.7.1.2.1.2 Unit 2 S1X Sequence

The S1X sequence contributes 19% (5.0 x 10^{-6}/yr) to the total Unit 2 CDF. The S1X sequence involves a small-small LOCA (1/2" - 3") with successful secondary heat removal, successful short term core cooling (typically HPSI injection) and failure of long term core cooling. The options considered for long term core cooling are shutdown cooling (LPSI pumps and shutdown cooling heat exchangers) or high pressure recirculation and containment heat removal. To implement SDC, the RCS must be cooled and depressurized to the SDC entry conditions. High pressure recirculation is automatically initiated when the RWT level drops to 6 ft. Successful high pressure recirculation requires one HPSI pump and either one containment spray pump or 2-out-of-4 containment coolers for containment heat removal.

The dominant cutsets include:

a) common cause failures of CCW N-header isolation valves to close (fails to remove non-safety related heat load from CCW system),

b) common cause failure of containment sump valves to open,

c) common cause miscalibration of RWT level transmitters (screening value used),

d) common cause failure of ICW valves to isolate TCW heat loads,

e) common cause failure of HPSI pumps to run during recirculation.

The operator actions associated with the Unit 2 S1X cutsets include the operator failing to open the EDG fuel oil fill valve bypass valves following fill valve failure (R#DGFO), and the operator failing to manually actuate ECCS components following failure of the automatic signal (RHPS2SIAS).

3.7.1.2.1.3 Unit 2 S1U Sequence

The S1U sequence accounts for 10% (2.5 x 10^{-6}/yr) of the total Unit 2 CDF. This sequence involves a small-small LOCA (1/2" - 3") with successful decay heat removal and failure of short term core cooling. Early core cooling represents inventory makeup to prevent core uncovery and subsequent core heatup which if uncorrected will lead to core damage. High pressure safety injection (1-out-of-2 HPSI pumps) is the primary source to perform the early core cooling function. If HPSI is unavailable, the operator must depressurize and use LPSI for core cooling.

Dominant S1U cutsets include:

a) common cause failure of HPSI injection valves to open,

b) common cause failure of HPSI pumps to start,

c) common cause failure of HPSI pumps to run during injection.

The operator actions associated with the Unit 2 S1U cutsets include the operator failing to recover an EDG following failure of the fuel oil tank automatic fill valves (R#DGFO), and the operator failing to manually actuate ECCS components following failure of the automatic signal (RHPS2SIAS).

3.7.1.2.1.4 Unit 2 AU Sequence

The AU sequence accounts for 8% (2.19×10^{-6}/yr) or the total Unit 2 CDF. This sequence involves a large LOCA (> 5") with failure of core cooling during the injection phase. Successful core cooling requires 1-out-of-2 LPSI pumps and 3-out-of-4 safety injection tanks.

Dominant AU cutsets include:

a) common cause failure of LPSI injection valves to open,

b) common cause failure of SIT's due to miscalibration of level or pressure transmitters (note that screening value was used),

c) LPSI pump common discharge header flow control valve and flow control valve bypass valve transfers closed during standby,

d) common cause failure of LPSI pumps to start or run.

There are no operator recovery actions associated with Unit 2 AU cutsets.

3.7.1.2.1.5 Unit 2 KC Sequence

The KC sequence accounts for 7% (1.76×10^{-6}) of the total St. Lucie Unit 2 CDF. This sequence involves an ATWS event where RCS integrity and secondary heat removal is successful and short term core cooling fails. Core cooling can fail due to one of two causes. First, emergency boration using the CVCS may fail to lower the power level in the core. This results in RCS pressure remaining high, and RCS inventory losses out of the PORVs and SRVs will exceed the inventory from the charging and HPSI systems. Second, the RCS may pressurize above Stress Level C (ATWS peak pressure of 3700 psia) and fail core cooling for the reasons discussed above. It is assumed that all three SRV's and 1-out-of-2 PORV's must open to maintain the peak pressure below 3700 psia.

Dominant KC cutsets are related to transient initiators with a favorable moderator coefficient, a mechanical fault preventing control rod insertion and unavailability of both PORV flow paths.

The operator recovery actions associated with the Unit 2 KC cutsets involve the operator failing to initiate emergency boration (RTOP2WBOR), and the operator failing to align the charging pump suction to the RWT before VCT depletion (RCVC2RWT).

3.7.1.2.1.6 Unit 2 S2X Sequence

The S2X sequence accounts for 5% (1.39×10^{-6}) of the total St. Lucie Unit 2 CDF. This sequence involves a small LOCA (3" - 5") initiating event with failure of long term core cooling. Long term core cooling is provided by high pressure recirculation and containment heat removal.

Dominant S2X cutsets involve:

a) common cause failure of the CCW N-header isolation valves to close (fails to remove the non-safety related heat loads from the CCW system),

b) common cause failure of containment sump valves to open,

c) common cause miscalibration of RWT level transmitters (screening value used),

d) common cause failure of HPSI pumps to run during recirculation.

The operator action associated with the Unit 2 S2X cutsets involves the operator failing to initiate hot leg recirculation (RHPS2HTLEG). Hot leg recirculation can be used if cold leg recirculation fails.

3.7.1.2.1.7 Unit 2 TBX Sequence

The TBX sequence accounts for 5% (1.28×10^{-6}) of the total St. Lucie Unit 2 CDF. The TBX sequence involves a transient initiating event with failure of secondary heat removal (main and auxiliary feedwater), successful once-through-cooling, and failure of long term core cooling. Long term cooling is assumed to require high pressure recirculation via 1-out-of-2 HPSI pumps. For high pressure recirculation to be successful, containment heat removal by either one containment spray pump or at least 2-out-of-4 containment fan coolers must also be available.

Dominant TBX cutsets involve a loss of grid initiating event with failure of one train's emergency diesel generator, failure of the other train's motor driven AFW pump (or flow path) and failure to recover the electrical equipment room's HVAC. It was conservatively assumed that both safety related 125VDC buses are dependent on room cooling for long term operation. Failure of the DC busses would result in loss of the turbine driven AFW pump and loss of control power for high pressure recirculation components that are required to support long term heat removal.

The operator actions associated with the Unit 2 TBX cutsets include the operator failing to re-establish electrical equipment room cooling following a loss of offsite power (RHVA2ELEQ), the operator failing to bypass a failed fuel oil fill valve to recover an EDG (R#DGFO), the operator failing to align the AFW cross-connect valves to recovery AFW (R#AFXVLVS), and the operator failing to manually actuate RAS components following failure of the automatic signal (RHPS2RAS). Note that the action to recover the electrical equipment room cooling is only required if the fan that was not running prior to the loss of power fails to automatically re-start.

3.7.1.2.1.8 Unit 2 AXC Sequence

The AXC sequence accounts for 4% (1.11 x 10^{-6}/yr) of the Unit 2 CDF. This sequence involves a large LOCA (> 5") with failure of cold leg recirculation. Failure of cold leg recirculation can occur if automatic actuation and the operator fails to switchover from injection via the RWT to recirculation via the sump or there are high pressure recirculation component failures.

Dominant cutsets include:

a) common cause failure of the CCW N-header isolation valves to close,

b) common cause failure of HPSI injection valves to open,

c) common cause failure of containment sump valves to open,

d) common cause failure of HPSI pumps to start or run during recirculation.

The operator action associated with the Unit 2 AXC cutsets involves the operator failing to manually actuate ECCS components following failure of the automatic signal (RHPS2SIAS).

3.7.1.2.2 Unit 2 Containment Bypass Sequences > 1 x 10^{-7}/yr

3.7.1.2.2.1 Unit 2 Interfacing System LOCA (ISLOCA)

ISLOCAs contribute 10% (2.72 x 10^{-6}/yr) to the total St. Lucie Unit 2 CDF. The potential flow paths are the four safety injection lines to the RCS loops and two shutdown cooling suction lines from the RCS loops. This analysis used expressions developed and presented in NUREG/CR-5102.

For the safety injection lines, an ISLOCA could occur if three check valves and a normally closed motor operated valve were to fail in the open state. The ISLOCA frequency through each Unit 2 safety injection line is estimated to be 5.5 x 10^{-11}/yr (2.2 x 10^{-10}/yr total for all four lines).

For the shutdown cooling suction lines, an ISLOCA would occur if two normally closed suction isolation valves were to fail in the open state. The ISLOCA frequency through each shutdown cooling suction line is estimated to be 1.36 x 10^{-6}/yr (2.72 x 10^{-6}/yr total for both suction lines). However, if the CE Owner's Group (CEOG) data [Ref. 3.7-4] was used in lieu of NUREG/CR-5102, the estimated CDF contribution would be less than 1.0 x 10^{-6}/yr.

3.7.1.2.2.2 Steam Generator Tube Rupture (Unit 2)

Steam generator tube ruptures contribute 3% (8.99 x 10^{-7}) to the total Unit 2 CDF. The dominant contributor (8.0 x 10^{-7}/yr) is the RDX sequence. This sequence involves a SGTR with failure to isolate the faulted SG and failure of long term cooling. Two functions must be achieved in order to isolate a faulted SG: mechanical isolation and cooldown and depressurization below the SG SRV setpoint. Long term cooling involves HPSI or shutdown cooling. Dominant RDX cutsets are related to RWT faults (tank or line ruptures), common cause failures of HPSI valves, and common cause failures of HPSI pumps. Other SGTR sequences contribute < 1 x 10^{-7}.

The operator actions associated with the Unit 2 RDX cutsets involve the operator failing to switch charging pump suction from the VCT to the RWT before VCT depletion (RCVC2RWT), and the operator failing to initiate long term core cooling (RTOP2RLTC).

3.7.1.2.3 Radioactive Release Sequences Comparable to WASH-1400 PWR-4 (Unit 1 & Unit 2)

WASH-1400 developed several categories for radioactive releases, among them being the PWR-4 category. PWR-4 includes the following release fractions for major radionuclides: Noble Gases - 0.6, Organic Iodine - 0.003, Iodine - 0.09, Tellurium - 0.03, Barium - 0.005, Ruthenium - 0.003, Lanthanum - 0.0004.

Section 4 describes the containment performance analysis conducted within the scope of this study. FPL's approach to understanding the most important plant damage sequences and their associated radioactive releases relies heavily on a "pinch-point" binning method. The core damage sequences obtained from the Level 1 analysis were first binned into categories, the dominant cutsets associated with these bins were then combined with the cutsets obtained through evaluation of the containment safeguards systems bridge tree. The output of this effort includes the dominant plant damage states' cutsets. These are then input to the Containment Event Tree to determine containment failure modes/sequences and associated radioactive releases.

In this process, the distinct identity of the functional core damage sequences described above is lost. The process does allow the containment failure sequence and associated source term to be decomposed and thereby traced to the dominant plant damage cutsets. To aid in understanding the transition from core damage state to plant damage state to radioactive release category, Table 3.7-14 develops a matrix of the relationship of core damage sequence to plant damage state. Tables 3.7-15 and 3.7-16 then identifies those plant damage states and their associated containment event tree end states which are predicted to exceed the PWR-4 release category as defined by WASH-1400.

For this study, "exceeding" the PWR-4 release category is defined as exceeding the WASH-1400 table values for release of Cesium (0.09) and Iodine (0.04) fractions. Noble gas releases were more severe than WASH-1400 PWR-4 for all PDSs. The remaining fractions are excluded due to their insignificant amounts. Integrated releases up to 24 hours were calculated for V sequences, up to 50 hours for cases where containment does not fail, and up to 10 hours after containment failure for cases where containment fails.

CET quantification includes various degraded events with huge uncertainty or low probability. These uncertain or unlikely events generally lead to source terms of high magnitudes. Thus, all CET end states contain release fractions greater than that associated with PWR-4.

Since the GL-88-20 and NUREG-1335 reporting criteria do not include the conditional probability associated with the release, essentially all plant damage states will have source terms greater than PWR-4. A number of PDSs will have a very low probability of exceeding this criteria, however.

3.7.2 Vulnerability Screening

One of the purposes of the IPE is to identify plant specific severe accident vulnerabilities and, where appropriate, modify hardware and procedures to help prevent or mitigate the severe accident. The NRC, however, has not defined what constitutes a vulnerability. Instead, the NRC has left this definition up to each utility to define in its IPE submittal.

As discussed in Section 3.7.1, GL 88-20 has defined screening criteria for selecting important severe accident sequences which should be reported to the NRC in the IPE submittals. The NRC states, however, that the screening criteria values do not represent a threshold for vulnerability determination.

NUMARC 91-04 [Ref. 3.7-1] provides guidelines intended to be a framework which utilities can use for closure of severe accident issues. NUMARC provides quantitative guidelines for consideration of when procedural or hardware changes should be implemented. The NUMARC guideline CDF values above which the largest emphasis is placed on changes to eliminate or reduce the source of the accident sequence initiator are as follows:

1. A sequence (except containment bypass) with a mean CDF greater than 1×10^{-4}/yr, or greater than 50% of the total CDF, or

2. A containment bypass sequence with a mean frequency of greater than 1×10^{-5}/yr, or greater than 20% of the total CDF.

The NUMARC document, however, does not specifically identify the above values as a threshold for determining when a severe accident sequence should be considered a vulnerability.

For the purposes of this evaluation, a vulnerability is defined as:

(1) A failure which contributes a disportionately large contribution to the total CDF or significant release probabilities and in turn is considered significantly higher than those of PRAs for similar plants, or

(2) A failure which has any unusual and significant impact on the total CDF or release probabilities.

The core damage probability results from Section 3.7.1 were reviewed for any core damage vulnerabilities. Based on the following, there are no vulnerabilities at St. Lucie Unit 1 or Unit 2:

1. No individual functional sequence has a core damage frequency greater than 4.34×10^{-6} for Unit 1 and 5.00×10^{-6} for Unit 2.

2. No individual functional sequence contributes more than 19% to the total core damage probability for Unit 1 or Unit 2. The most likely individual cutset has a probability of 9.98×10^{-7} for Unit 1 and Unit 2, or less than 5% of the total core damage probability.

3. No containment bypass sequence contributes more than 2.72 x 10-6/yr or 10% to the total CDF.

4. No unusual and significant failures were found.

The total estimated core damage probability of 2.3×10^{-5}/yr for Unit 1 and 2.6×10^{-5}/yr for Unit 2 is significantly less than the NRC staff's core damage frequency objective of 1×10^{-4} per year. This core damage probability is within the range of previous, published PWR risk estimates and indicates that St. Lucie does not have an unusual core damage risk.

3.7.3 Importance Analysis

Each of the inputs to the PRA model contribute to the frequency of core damage by a certain amount. The magnitude of this contribution is described by an importance measure. The Fussell-Vesely (F-V) importance measure is used in this study for both basic events and for systems as a whole. This importance measure is the weighted proportion of cutsets which have a given basic event or system in them:

$$F\text{-}V\ (SYS_i) = \left[\sum_{i=1}^{\infty} SEQ_i\ (SYS_i) \right] / \ CDF$$

where CDF = total core damage frequency

Tables 3.7-9 and 3.7-10, for Unit 1 and Unit 2 respectively, show the model elements that have a F-V importance of 0.1% or larger. Figure 3.7-5 shows the estimated system importance for Unit 1 and Unit 2. The high importance of the HPSI system is due to the various functions supported:

- injection and recirculation following a small-small LOCA

- recirculation following a small LOCA

- high pressure recirculation failures following a large LOCA
- once-through-cooling following loss of secondary heat removal

The CCW system supports HPSI operation and provides cooling to the RCP seals.

3.7.4 Sensitivity Analysis

Sensitivity analysis involves the determination of how much the results of the quantification change for a certain change in the data or assumptions used. This is useful for evaluating the significance of uncertainty in a particular failure event to the overall results or, conversely for determining which failure events must be known with greater certainty.

The basic events with a F-V importance of 1% or greater were reviewed to determine the dominant failure modes. The dominant events identified for both Unit 1 and Unit 2 included:

- Common Cause Failures
- Off-Site Power Recovery
- Motor Operated Pump Failures
- MOV Failures
- Operator Recovery Actions
- Pre-Initiator HFE
- PORV Flow Path Unavailability
- Diesel Generator Failures
- Test & Maintenance Unavailability

Based on the above, the following data sensitivity analyses were performed:

1) Increased common causes failure event probabilities by a factor of 10.

2) Increased all offsite power non-recovery probabilities by a factor of 10.

3) Increased all motor driven pump fail-to-start probabilities by a factor of 10.

4) Increased all motor driven pump fail-to-run probabilities by a factor of 10.

5) Increased all MOV fail-to-open and fail-to-close probabilities by a factor of 10.

6) Increased all operator non-recovery probabilities by a factor of 10.

7) Increased all pre-initiator HFE probabilities by a factor of 10.

8) Increased the Unit 1 PORV flow path unavailability to a value similar to that used for Unit 2.

9) Decreased the Unit 2 PORV flow path unavailability to a value similar to that used for Unit 1.

10) Increased the diesel generator fail-to-start and fail-to-run probability by a factor of 10.

11) Increased all test and maintenance unavailability probabilities by a factor of 10.

12) Decreased all test and maintenance unavailability probabilities by a factor of 10.

RMQS was used to calculate the change in CDF for the cases listed above. The results of these sensitivity runs are provided in Table 3.7-11.

The following provides a brief discussion of the results of the sensitivity analyses summarized in Table 3.7-11:

An increase in all MOV failure probabilities by a factor of 10 has a significant effect on the CDF (486% increase on Unit 1 and 503% increase on Unit 2). This is not unexpected since many MOVs in key systems must change position in order to prevent or mitigate an accident. An increase in the independent MOV failure rates also result in higher common cause failure probabilities which contribute to the higher CDF. The data used in this PRA represents a reasonable estimation of the MOV failure rates since it takes into account plant specific experience in conjunction with generic data. Thus, even with an increase in failure probability greater than the error factor, the CDF is still on the order of 1 x 10^{-4}.

A factor of 10 increase in the common cause failure probabilities results in a CDF increase of 291% on Unit 1 and 272% on Unit 2. Since common cause events fail multiple trains of the same system, they would tend to have a significant effect on the CDF. There is limited data available to estimate the beta factors used to calculate the common cause failure probabilities. As such, the beta factors tend to be conservative. The impact of common cause failures on the CDF, therefore, is more likely over-estimated.

The estimated change in CDF due to a factor of 10 increase in the diesel generator failure rates is 261% for Unit 1 and 258% for Unit 2. Since essentially all components require electric power for motive force, it is expected that an increase in the failure rate could have a significant effect on the CDF given the relatively large loss of grid initiating event frequency used. The failure rate used in the PRA was based on St. Lucie experience and is thought to be a reasonable estimate.

Two sensitivity studies were performed regarding test and maintenance unavailability data. First, a factor of 10 increase in the assumed PRA values results in an increase in the CDF of 248% for Unit 1 and 235% for Unit 2. Second, a decrease in the assumed PRA values by a factor of 10 results in a decrease in the CDF of 17% for Unit 1 and 16% for Unit 2. This shows that the CDF is more sensitive to an increase in total unavailability. The test and maintenance unavailability data used in this analysis was based on historical data obtained from review of operator equipment out-of-service logs. The actual contribution to the total CDF, therefore, is an accurate representation of the testing and maintenance practices at St. Lucie and should not significantly influence the uncertainty of the analysis.

The assumed higher flow path unavailability (in this case related to closure of the block valve with the block valve breaker open) results in an approximate 226% increase in the Unit 1 CDF. This is due to the fact that the Unit 1 success criteria for OTC and for pressure relief following an ATWS is 2-out-of-2 PORV flow paths. The unavailability of one flow path, therefore, would fail these functions. The Unit 2 PORV flow path unavailability data used for this Unit 1 sensitivity study is significantly higher than the Unit 1 plant specific data. The Unit 2 sensitivity study shows that with the Unit 1 PORV path unavailability data, there is an approximate 10% decrease in the Unit 2 CDF. This is primarily due to the Unit 2 OTC success criteria (1-out-of-2 PORV flow paths).

The remaining changes show that uncertainty in the motor operated pump, offsite power non-recovery, operator recovery, and pre-initiator HFE probabilities contribute to the overall uncertainty in the analysis. The assumed factor of 10 increase is within the error factors and thus it is concluded that the events do not significantly influence the analysis uncertainty. Even with the higher failure probabilities, the CDF is well below 1×10^{-4}.

As discussed in Section 3.1, it was assumed that initiation of hot leg recirculation is not required to prevent boron precipitation, and thus was not considered a core damage sequence. The Unit 1 and Unit 2 fault trees were re-quantified to assess the potential effect on the CDF if it were assumed that hot leg recirculation is required. The results of this analysis show that there would not be a significant increase in the CDF.

3.7.5 Uncertainty Analysis

Uncertainties can be grouped into basically two types typically associated with PRA model development. These two basic types are parameter value uncertainty and modeling uncertainty. These uncertainties arise in the analysis at every step in the process and can be both qualitative and quantitative. The range of uncertainty is further dependent on the completeness of the PRA analysis.

Parameter value uncertainties are typically quantitative in that they are related to failure rates, frequencies and unavailabilities. To quantify this uncertainty, probabilistic distributions were developed for each cutset parameter value. These parameter value distributions were then propagated to the accident sequence level through the Monte Carlo technique utilized by the UNCERT program. Through multiple iterations on a random seed, a high confidence mean value with a log-normal distribution was generated, using the mean and error factor values of each parameter (and the accident sequences as a whole) found in the accident sequence cutsets being evaluated. UNCERT was used to develop overall model parameter value uncertainty by loading a RMQS text file output and then selecting the number of samples desired. The UNCERT program Monte Carlo propagation of basic event distributions to a total core damage frequency uncertainty distribution results are given by a fifth and ninety-fifth core damage frequency of 4.55 x 10^{-6} and 6.24 x 10^{-5} for Unit 1 and 4.96 x 10^{-6} and 7.16 x 10^{-5} for Unit 2, respectively around a 2.21 x 10^{-5} Unit 1 mean and 2.41 x 10^{-5} Unit 2 mean using cutsets representing 98% of the total internal events CDF and 5000 samples. Tables 3.7-12 and 3.7-13 provide the uncertainty analysis results.

Modeling uncertainties have been handled largely by either modeling conservatisms or qualitative considerations. One important source of uncertainty is believed to be the completeness and the accuracy of the models that define the core damage sequences. Since it is difficult to hypothesize all postulable accident initiators, it is believed that the finite number of identified initiators introduces a slight bias that tends to under predict core damage frequency. This bias tends to be offset by the initiator class grouping philosophy which groups similar initiators together, takes the cumulative frequency, and then assumes the impact of the most limiting initiator in the group.

System modeling inaccuracies are a related source of uncertainty that multiple levels of review and reconsideration during model iteration does not necessarily eliminate. Since during the review and iteration phase of the analysis it was easier to identify and correct those sequences which significantly over predict the core damage frequency than those which could have been over-looked with respect to under prediction, modeling inaccuracy uncertainty is believed to introduce a slight bias that tends to under predict core damage frequency. This is compensated, to a large extent, by conservative assumptions that could not be or were chosen not to be removed in the modeling. An example of conservative model uncertainty involves the "loss of makeup" sequences. These sequences are very slow to evolve, requiring many hours to reduce the RCS inventory to the point where core uncovering begins. No credit was taken for restoring components from maintenance or repairing failed components during this time. Another conservatism arises from the modeling of time related failures. These failures are assumed to occur at t=0 in the accident sequence. Credit for a finite availability of components before failure (DHR systems for example), therefore, may not be accounted for when determining the available time to take operator actions to prevent or mitigate core damage. Also, many HVAC dependencies are conservatively assumed since it is difficult to determine a component's vulnerability to ambient temperatures above design values.

Another important source of uncertainty is in the modeling and quantification of human actions. While attempts were made to identify human actions that would help mitigate important core damage sequences, it is possible that some such actions were overlooked. In addition, most of the important human actions modeled are either procedural actions performed under some degree of stress or non-procedural recovery actions, for which there is only limited experience. Thus, the quantification of these important human actions had to be based on the subjective judgement of the analyst. Several thousand cutsets are generated during the quantification process and many of the recovery actions are manually added to the cutsets. There may be many lower probability cutsets, therefore, which are recoverable but no operator action is included due to the magnitude of the effort and the analyst's judgement that the applicable sequence has been adequately recovered. This could account for a slight bias that may over predict the core damage frequency.

The criteria used to define the success criteria of systems are believed to have a moderate bias that tends to over predict core damage frequency. Such criteria are mostly taken from safety analyses in which conservative assumptions are made. MAAP analyses have been used in some cases to determine more "realistic" success criteria.

Dependencies between systems were explicitly modeled, and dependencies within a system were modeled to the extent that the analyst judged them appropriate in accordance with existing data. However, it is estimated that dependent component failures that may not be apparent from the plant data, and thus were not modeled, could account for a moderate bias that may under predict core damage frequency. Tending to offset this potential under prediction is the general belief that the beta factor approach and values used for common-cause failures is a conservative interpretation of a limited data set and therefore tends to over predict core damage frequency.

In summary, the total core damage frequency uncertainty distribution results given by a fifth and ninety-fifth core damage frequency of 4.55×10^{-6} and 6.24×10^{-5} for Unit 1 and 4.96×10^{-6} and 7.16×10^{-5} for Unit 2 around a mean of 2.21×10^{-5}/yr for Unit 1 and 2.41×10^{-5}/yr for Unit 2. The uncertainty analysis was based on cutsets representing 98% of the estimated internal events CDF (Unit 1 internal events point estimate = 2.14×10^{-5}/yr (98% = 2.10×10^{-5}/yr), Unit 2 internal events point estimate = 2.35×10^{-5}/yr (98% = 2.30×10^{-5}/yr)). This uncertainty distribution is thought to be representative of the base PRA model quantitative results uncertainty. Additional analyses that have been performed, such as ISLOCA and Internal Flooding, introduce additional uncertainties which may tend to over predict the overall core damage and offsite release frequencies due to their conservative scoping (or screening) nature.

Although there are sources of uncertainty in the quantitative St. Lucie Unit 1 and Unit 2 PRA results, the quantitative and qualitative results provide useful information for assessing insights into the plant's capability to respond to accident conditions. Since there are inherent uncertainties, the potential user should realize that PRA results are not adequate to provide the sole basis for decisions. PRA results, however, can be combined with other types of evaluation to make decisions regarding modifications or areas of emphasis in operations, training and plant design.

3.7.6 Sequences Eliminated Because of Human Recoveries

The IPE submittal guidance, NUREG-1335, requests the identification of core damage sequences that drop below the core damage frequency screening criteria because the frequency was reduced by more than an order of magnitude by credit taken for operator actions. To determine which St. Lucie sequences fell into this category, all operator recovery action probabilities that were less than 0.1 were increased to 0.1 in RMQS and a new CDF calculated. With all operator actions at 0.1 or above, the additional sequences above the screening criteria are TX, TQX, TQU, RX, TBX, TQBF, and KC for Unit 1, and TQX, and TQU for Unit 2. A brief description of each sequence is given below:

TX - Transient initiator with failure of long term core cooling (Unit 1 sequence only).

TQX - Transient initiator with loss of RCS integrity and failure of long term core cooling.

TQU - Transient initiator with loss of RCS integrity and failure of short term core cooling (injection).

TQBF - Transient initiator with loss of RCS integrity, failure of secondary heat removal and failure of once-through-cooling.

TBX - Transient initiator with failure of secondary heat removal and failure of long term core cooling.

RX - SGTR with failure of long term core cooling (Unit 1 sequence only).

KC - ATWS with failure of short term core cooling.

The dominant operator actions included in the Unit 1 TX and RX cutsets are related to the operator failing to (1) makeup to the Unit 1 condensate storage tank (CST) to support long term heat removal via AFW or (2) establish shutdown cooling. The applicable operator events include RAFWICST, RCSTITWST, and RTOPITLTC. The Unit 1 CST can support AFW operation for approximately 10 hours before either (1) makeup is required (via treated water storage tank), (2) the AFW pump suction must be re-aligned to the Unit 2 CST, or (3) shutdown cooling must be established.

The operator action related to the dominant Unit 1 and Unit 2 TQX and TQU cutsets involves the failure of the operator to trip the reactor coolant pumps (RCPs) after loss of CCW to the pump seals (RTOP1S1RCP and RTOP2S1RCP). It is assumed that failure to trip the RCPs within 10 minutes following loss of seal cooling will result in a seal LOCA.

The Unit 1 TQBF sequence has two important operator actions. First, the operator must trip the RCPs if cooling is lost to the seals (RTOP1S1RCP). Second, the operator must initiate once-through-cooling if secondary heat removal (main and auxiliary feedwater) fails (RTOP1S1OTC).

The dominant operator action in the Unit 1 TBX sequence involves the operator failing to re-establish electrical equipment room cooling (RHVA1ELEQ) following a loss of grid initiating event. Per emergency operating procedures, the operator is instructed to re-start the fans within two hours. This event primarily affects "B" train components. The dominant TBX cutsets involve the operator action discussed above with failure of "A" train components. Other operator actions related to the TBX sequence include the operator failing to manually start AFW components after failure of the automatic signal (R#AFWCMP) and failure of the operator to re-align the 1AB DC bus from the 1B DC bus to the 1A DC bus (R#DCAB) to recover the turbine driven AFW pump.

The dominant operator actions related to the Unit 1 KC sequence involve failure to emergency borate (RTOP1WBOR) and to re-align the charging pump suction to the RWT (RCVC1RWT) before boric acid makeup tank depletion.

3.7.7 Decay Heat Removal Evaluation

In NUREG-1289 [Ref. 3.7-2], the staff defines the systems related to the decay heat removal function as those components and systems required to maintain primary and secondary coolant inventory control and to transfer heat from the reactor coolant system to an ultimate heat sink following shutdown of the reactor for normal events or abnormal transients such as loss of main feedwater, loss of offsite power, and small-break loss-of-coolant accidents (SBLOCAs). The A-45 program was not concerned with anticipated transients without scram, interfacing system loss-of-coolant accidents, or those emergency core cooling systems that are required only during the reflood phase to maintain coolant inventory and dissipate heat for a short period following a large LOCA. The USI A-45 program also considered supporting systems such as the component cooling water system, essential service water system, and emergency onsite AC and DC power systems that are required for various modes of decay heat removal. The reliability of the reactor protection system was not addressed, and successful shutdown of the reactor is assumed. The transition from reactor trip to hot shutdown (excluding the reflooding phase in a large LOCA), the transition from hot shutdown to cold shutdown, and maintaining cold shutdown conditions were considered as part of the NRC program. However, the latter two phases did not receive the same degree of quantitative analysis as the first. In addition, the USI A-45 program was directed toward prevention of accidents that lead to core damage and not to mitigation of such accidents.

The primary method for removing decay heat in pressurized water reactors (PWRs) is through the steam generators. St. Lucie Units 1 and 2 each have two different steam generator feedwater supply systems: main feedwater and auxiliary feedwater. The main feedwater system of each unit consists of two motor driven main feedwater pumps and three motor driven condensate pumps. The condensate pumps can also be used to supply water directly to the steam generators if the steam generators are depressurized to less than 600 psig. The auxiliary feedwater system for each unit consists of two motor driven pumps and one turbine driven pump. Each auxiliary feedwater pump is capable of maintaining secondary inventory after a reactor trip. Transfer of heat from the reactor coolant system is accomplished by feeding the steam generators and dumping the steam produced into the condenser using the turbine steam bypass system or into the atmosphere via the atmospheric dump valves. The High Pressure Safety Injection system is used for primary inventory control during a small break LOCA event.

St. Lucie Units 1 and 2 also has the capability to provide decay heat removal by once-through-cooling. Once-through-cooling is a feed and bleed operation in which core decay heat is removed by opening the power operated relief valves (PORVs) (2-out-of-2 valves on Unit 1 and 1-out-of-2 valves on Unit 2) on the pressurizer and injecting coolant with a HPSI pump. This operation effectively transfers core heat to the containment. In order to achieve a long-term stable state, heat must be removed from containment. All of these decay heat removal options are modeled in the St. Lucie PRA.

The primary front-line systems supporting decay heat removal at St. Lucie, therefore, are main feedwater (and condensate), auxiliary feedwater and the once-through-cooling systems. The A-45 studies [Ref. 3.7-3] and other PRAs show that the support systems (component cooling water, intake cooling water, and electric power systems) can be of equal or even greater importance in controlling decay heat removal vulnerabilities. The St. Lucie PRA models, therefore, include the consideration of AC and DC power, service water (intake and component cooling), instrument air and HVAC systems necessary to support front-line system operation.

3.7.7.1 Evaluation Objective

The purpose of USI A-45 is to evaluate the adequacy of current designs to ensure that LWRs do not pose unacceptable risk as a result of DHR function failures. The primary objectives of the USI A-45 program are to evaluate the safety adequacy of DHR systems in existing LWR power plants and to assess the value and impact (or benefit-cost) of alternative measures for improving the overall reliability of the DHR function.

At the time the USI A-45 program commenced, the NRC also started to develop a set of qualitative safety goals and quantitative design objectives (QD0). To aid progress in the USI A-45 program, some interim QDOs were defined with the knowledge that these might have to be changed later in the program to conform with those finally decided on by the Commission. The principal quantitative design objective selected for USI A-45 is the frequency of core damage due to failure of the DHR function. An interim value of 1 x 10^{-5} per reactor-year is proposed for this QDO.

3.7.7.2 Evaluation Approach

The experience gained from application of PRA analysis to U.S. LWRs in the USI A-45 program and other programs suggests that, when the examinations for severe accident vulnerabilities (IPEs and IPEEEs) called for as part of the Severe Accident Policy have been completed, the existing plants will fall into three broad categories as far as the quantifiable adequacy of their DHR function is concerned. Pending further guidance from the Commission, the following quantitative values (expressed as frequency means) have been used by the staff as a basis for categorization:

Category	Classification of Level of DHR Vulnerability	Core Damage Probability
1	Frequency of core damage due to failures of DHR function acceptably small or reducible to an acceptable level by simple improvements.	< 3 x 10^{-5}/yr
2	DHR performance characteristics intermediate between Categories 1 and 3.	> 3 x 10^{-5}/yr and < 3 x 10^{-4}/yr
3	Frequency of core damage so large that prompt action to reduce probability of core damage due to failure of DHR to an acceptable level is necessary.	> 3 x 10^{-4}/yr

The choice between the various alternatives for the resolution of USI A-45, therefore, takes into account this variability in the performance characteristics of the DHR function in the existing LWRs.

3.7.7.3 St. Lucie DHR Evaluation

NUREG/CR-4710 [Ref. 3.7-3] documents an NRC investigation of the adequacy of decay heat removal (DHR) at the St. Lucie Plant. The NRC internal events analysis identified ten "potential vulnerabilities". The dominant failures associated with these "vulnerabilities" included:

- common cause failure of batteries (affects capability to start diesel generators)
- common cause failure of component cooling water pumps (results in loss of cooling to ECCS components)
- containment sump recirculation valve failures
- common cause failure of diesel generators with failure of the turbine driven AFW pump
- local diesel generator faults with failure of the turbine driven AFW pump
- local fault on one train's diesel generator with failure of the opposite train's battery
- common cause failure of intake cooling water pumps (results in loss of cooling to component cooling water system)
- common cause testing/maintenance induced failures in the safety injection or recirculation actuation signal logic

The NRC analysis concluded that there were no recommended modifications that would significantly reduce the St. Lucie core damage probability. One of the observations of the NRC study, however, was that emergency electric power availability appears to be a key issue. Since the NRC DHR study was completed, the St. Lucie plant has installed a Unit 1/Unit 2 blackout crosstie. This crosstie provides the capability to power the safe shutdown loads of both units from any one of the four emergency diesel generators (two diesel generators are installed on each unit). The dominant failure modes identified in the NRC evaluation are included in the St. Lucie Unit 1 and Unit 2 PRA models. The St. Lucie evaluation also takes into account the capability to use once-through-cooling as an alternative means of DHR.

3.7.7.4 Evaluation Conclusion

The core damage frequency contribution from sequences which meet the NRC's definition of decay heat removal (transients and small break LOCAs (S1 and S2)) is $< 2 \times 10^{-5}$/yr for Unit 1 and for Unit 2. This falls into category 1 of the NRC's vulnerability classification scheme. Thus, the conclusion of this IPE is that St. Lucie has no unique decay heat removal vulnerabilities.

3.7.8 USI and GSI Screening

The St. Lucie Unit 1 and Unit 2 PRA has not been used to evaluate any USIs or GSIs other then the A-45 evaluation discussed above. At a later date, however, the IPE models may be used in regulatory applications including USIs and GSIs.

3.7.9 Section 3.7 References

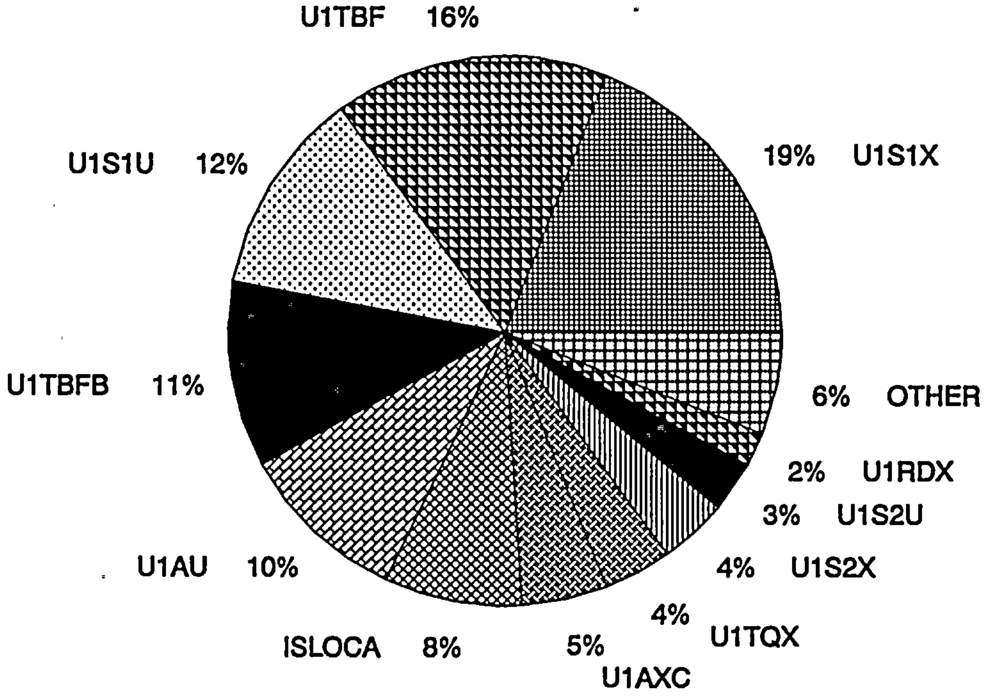
3.7-1 NUMARC 91-04, "Severe Accident Issue Closure Guidelines", January, 1992

3.7-2 NUREG-1289, "Resolution of Generic Issues A-45"

3.7-3 NUREG/CR-4710, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop Pressurized Water Reactor", dated August, 1987.

3.7-4 CEOG Task 582, "Risk Evaluation of Removal of Shutdown Cooling System Auto-Closure Interlock," September 1989.

Figure 3.7-1 St. Lucie Unit 1 Core Damage Frequency by Sequence

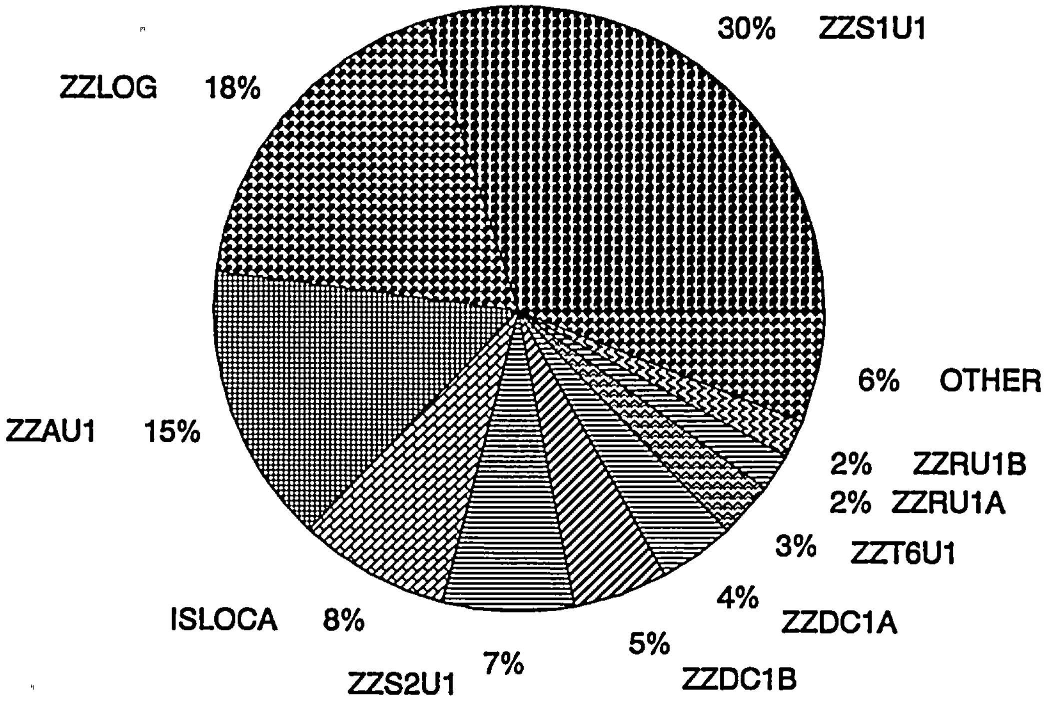


Sequence	Description	Core Damage Frequency Contribution	% of Total
U1S1X	Small-Small LOCA w/Long Term Core Cooling Failure	4.34E-06	19%
U1TBF	Transient w/Secondary Heat Removal and OTC Failure (Non-Blackout)	3.72E-06	16%
U1S1U	Small-Small LOCA w/Injection Failure	2.74E-06	12%
U1TBFB	Transient w/Secondary Heat Removal and OTC Failure (Blackout)	2.64E-06	11%
U1AU	Large LOCA w/Injection Failure	2.37E-06	10%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
U1AXC	Large LOCA w/Cold Leg Recirc Failure	1.12E-06	5%
U1TQX	Transient w/Loss of RCS Integrity and Long Term Core Cooling Failure	9.31E-07	4%
U1S2X	Small LOCA w/Long Term Core Cooling Failure	9.29E-07	4%
U1S2U	Small LOCA w/Injection Failure	6.72E-07	3%
U1RDX	SGTR w/Failure to Terminate Leakage and Long Term Core Cooling Failure	4.70E-07	2%
Other		1.48E-06	6%
Total Freq:		2.3E-05	

Table 3.7-1 St. Lucie Unit 1 Core Damage Frequency by Sequence

Sequence	Description	Core Damage Frequency Contribution	% of Total
U1S1X	Small-Small LOCA w/Long Term Core Cooling Failure	4.34E-06	19%
U1TBF	Transient w/Secondary Heat Removal and OTC Failure (Non-Blackout)	3.72E-06	16%
U1S1U	Small-Small LOCA w/Injection Failure	2.74E-06	12%
U1TBFB	Transient w/Secondary Heat Removal and OTC Failure (Blackout)	2.64E-06	11%
U1AU	Large LOCA w/Injection Failure	2.37E-06	10%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
U1AXC	Large LOCA w/Cold Leg Recirc Failure	1.12E-06	5%
U1TQX	Transient w/Loss of RCS Integrity and Long Term Core Cooling Failure	9.31E-07	4%
U1S2X	Small LOCA w/Long Term Core Cooling Failure	9.29E-07	4%
U1S2U	Small LOCA w/Injection Failure	6.72E-07	3%
U1RDX	SGTR w/Failure to Terminate Leakage and Long Term Core Cooling Failure	4.70E-07	2%
U1KC	ATWS w/ Short Term Core Cooling Failure	3.98E-07	2%
U1TQU	Transient w/Loss of RCS Integrity and Injection Failure	3.23E-07	1%
U1RBF	SGTR w/Secondary Heat Removal Failure and OTC Failure	3.15E-07	1%
U1TBX	Transient w/Secondary Heat Removal Failure and Long Term Core Cooling Failure	2.65E-07	1%
U1TX	Transient w/Failure of Long Term Core Cooling	1.13E-07	<1%
U1RBX	SGTR w/Secondary Heat Removal Failure and Long Term Core Cooling Failure	3.07E-08	<1%
U1KB	ATWS w/Secondary Heat Removal Failure	1.51E-08	<1%
U1TQ1UB	Transient w/Loss of RCS Integrity and Injection Failure (Blackout)	9.02E-09	<1%
U1S1BF	Small-Small LOCA w/Secondary Heat Removal Failure and OTC Failure	5.53E-09	<1%
U1TQBF	Transient w/Loss of RCS Integrity, Secondary Heat Removal Failure, and OTC Failure (Non-Blackout)	5.01E-09	<1%
U1RX	SGTR w/Failure of Long Term Core Cooling	2.03E-10	<1%
U1TQBX	Transient w/Loss of RCS Integrity, Secondary Heat Removal Failure, and Long Term Core Cooling Failure	3.75E-12	<1%
U1KX	ATWS w/Failure of Long Term Core Cooling	1.00E-13	<1%
Total Freq:		2.32E-05	

Figure 3.7-2 St. Lucie Unit 1 Core Damage Frequency by Initiator



Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U1	Small-Small LOCA	7.09E-06	30%
ZZLOG	Loss of Grid	4.11E-06	18%
ZZAU1	Large LOCA	3.48E-06	15%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
ZZS2U1	Small LOCA	1.60E-06	7%
ZZDC1B	Loss of DC Bus 1B	1.08E-06	5%
ZZDC1A	Loss of DC Bus 1A	9.75E-07	4%
ZZT6U1	Steamline Break Downstream of MSIVs	6.60E-07	3%
ZZRU1A	SGTR - S/G 1A	4.29E-07	2%
ZZRU1B	SGTR - S/B 1B	3.86E-07	2%
Other		1.59E-06	6%

Total Freq: 2.3E-05

Table 3.7-2 St. Lucie Unit 1 Core Damage Frequency by Initiator

Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U1	Small-Small LOCA	7.10E-06	31%
ZZLOG	Loss of Grid	4.08E-06	18%
ZZAU1	Large LOCA	3.49E-06	15%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
ZZS2U1	Small LOCA	1.60E-06	7%
ZZDC1B	Loss of DC Bus 1B	1.10E-06	5%
ZZDC1A	Loss of DC Bus 1A	9.90E-07	4%
ZZT6U1	Steamline Break Downstream of MSIVs	6.60E-07	3%
ZZRU1A	SGTR - S/G 1A	4.35E-07	2%
ZZRU1B	SGTR - S/G 1B	3.90E-07	2%
ZZCCWU1	Loss of CCW	2.83E-07	1%
ZZT3CU1	LOFW - Not Recoverable	2.38E-07	1%
ZZT7SIU1	Spurious SIAS	1.91E-07	1%
ZZT1U1	Reactor Trip	1.80E-07	1%
ZZT3AU1	LOFW - Recoverable	9.95E-08	<1%
ZZT2U1	Reactor Trip (PORV Challenge)	9.49E-08	<1%
ZZT5U1A	Upstream Steamline Break - S/G 1A	8.90E-08	<1%
ZZT5U1B	Upstream Steamline Break - S/G 1B	8.55E-08	<1%
ZZICWU1	Loss of ICW	8.06E-08	<1%
ZZT3EU1	Excessive Feedwater	6.82E-08	<1%
ZZIAU1	Loss of Instrument Air	6.08E-08	<1%
ZZT8AU1	PORV Sticking Open - S/G 1A	3.75E-08	<1%
ZZT8BU1	PORV Sticking Open - S/G 1B	3.74E-08	<1%
ZZT7MSU1	Spurious Main Steam Isolation	2.31E-08	<1%
ZZMAU1	Loss of Instrument Bus 1MA	4.39E-09	<1%
ZZMBU1	Loss of Instrument Bus 1MB	4.39E-09	<1%
ZZMCU1	Loss of Instrument Bus 1MC	4.34E-09	<1%
ZZMDU1	Loss of Instrument Bus 1MD	4.34E-09	<1%
ZZ4KV1B2	Loss of 4kV Bus 1B2	8.35E-10	<1%
ZZ4KV1A2	Loss of 4kV Bus 1A2	4.05E-10	<1%
ZZT3DU1A	Feedline Break S/G 1A	2.81E-10	<1%
ZZT3DU1B	Feedline Break S/G 1B	2.72E-10	<1%
ZZTCWU1	Loss of TCW	2.50E-10	<1%
ZZT3DU1	Feedline Break (Common)	2.34E-10	<1%
ZZ6KV1A1	Loss of 6.9kV Bus 1A1	9.54E-11	<1%
ZZ6KV1B1	Loss of 6.9kV Bus 1B1	9.54E-11	<1%
ZZT4B	Loss of Offsite Power "B" Train	6.38E-12	<1%
ZZT4A	Loss of Offsite Power "A" Train	4.38E-12	<1%
Total Freq:		2.32E-05	

Table 3.7-3 St. Lucie Unit 1 Core Damage Contribution by Transient Group

Transient Group	Transient Initiators in Group	Sequence Contribution								
		TQBF	TQU	TQX	TBF	TBFB	TBX	TX	K	Total (Per Year)
1	ZZLOG	1.66E-10	1.72E-07	2.57E-07	6.55E-07	2.64E-06	2.40E-07	1.09E-07	3.45E-08	4.11E-06
2	ZZT5U1A ZZT5U1B ZZT6U1 ZZT8AU1 ZZT8BU1 ZZDC1A ZZDC1B ZZT7SIU1	2.00E-09	1.03E-07	1.57E-07	2.84E-06		2.54E-08	3.48E-09	2.65E-08	3.16E-06
3	ZZT1U1 ZZT2U1 ZZT3AU1 ZZT3CU1 ZZT3EU1	2.80E-09	1.41E-08	1.54E-07	2.18E-07			1.08E-10	2.91E-07	6.80E-07
4	ZZT3DU1 ZZT3DU1A ZZT3DU1B ZZT4AU1 ZZT4BU1 ZZ4KV1A2 ZZ4KV1B2 ZZ6KV1A1 ZZ6KV1B1 ZZCCWU1 ZZICWU1 ZZTCWU1 ZZIAU1 ZZMAU1 ZZMBU1 ZZMCU1 ZZMDU1 ZZT7MSU1		4.27E-08	3.63E-07				1.67E-10	6.11E-08	4.67E-07
Total Transient Contribution		4.97E-09	3.32E-07	9.31E-07	3.71E-06	2.64E-06	2.65E-07	1.13E-07	4.13E-07	8.41E-06

Table 3.7-4 St. Lucie Unit 1 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U1TBFB	III	9.98E-07	4.7	ZZLOG	LOSS OF GRID
				EMM14CCFTR	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO RUN
				REPS1CASE6	OFF-SITE POWER NON-RECOVERY CASE 6:CCF OF DIESELS TO RUN
U1S1X	II	7.73E-07	3.6	ZZS1U1	SMALL-SMALL LOCA
				CMM1AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U1S1U	I	7.00E-07	3.3	ZZS1U1	SMALL-SMALL LOCA
				GMM1HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U1S1X	II	7.00E-07	3.3	ZZS1U1	SMALL-SMALL LOCA
				GMM1SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U1S1U	I	5.95E-07	2.8	ZZS1U1	SMALL-SMALL LOCA
				GMM1MRMOV	MINIMUM RECIRC LINE MOTOR VALVES TRANSFER CLOSED
U1TBFB	III	5.15E-07	2.4	ZZLOG	LOSS OF GRID
				EMM14CCFTS	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO START
				REPS1CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
U1S1U	I	4.35E-07	2.0	ZZS1U1	SMALL-SMALL LOCA
				GMM1MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U1S1X	II	4.26E-07	2.0	ZZS1U1	SMALL-SMALL LOCA
				NHFL1RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS
U1TQX	II	2.82E-07	1.3	ZZCCWU1	LOSS OF CCW
				RTOP1S1RCP	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U1S1X	II	2.73E-07	1.3	ZZS1U1	SMALL-SMALL LOCA
				QMM1MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U1S2X	VI	2.21E-07	1.0	ZZS2U1	SMALL LOCA
				CMM1AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U1S2U	V	2.00E-07	.9	ZZS2U1	SMALL LOCA
				GMM1HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U1S2X	VI	2.00E-07	.9	ZZS2U1	SMALL LOCA
				GMM1SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U1S1U	I	1.99E-07	.9	ZZS1U1	SMALL-SMALL LOCA
				GMM1CFTRIS	CCF OF HPSI PUMPS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
U1S1X	II	1.99E-07	.9	ZZS1U1	SMALL-SMALL LOCA
				GMM1CFTRRS	CCF OF HPSI PUMPS TO RUN DURING REC FOLLOWING SMALL-SMALL LOCA
U1TBF	III	1.97E-07	.9	ZZT6U1	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM1HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U1AXC	VI	1.45E-07	.7	ZZAU1	LARGE LOCA
				CMM1AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U1AU	V	1.31E-07	.6	ZZAU1	LARGE LOCA
				JMM1HCVCCF	COMMON CAUSE FAILURE OF LPSI INJECTION VALVES TO OPEN DURING
U1AXC	VI	1.31E-07	.6	ZZAU1	LARGE LOCA
				GMM1HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U1AXC	VI	1.31E-07	.6	ZZAU1	LARGE LOCA
				GMM1SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U1TBF	III	1.28E-07	.6	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				ETM1ASU	1A STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
U1TBF	III	1.28E-07	.6	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				ETM1BSU	1B STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
U1S2U	V	1.24E-07	.6	ZZS2U1	SMALL LOCA
				GMM1MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U1TBF	III	1.22E-07	.6	ZZT6U1	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM1MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U1S2X	VI	1.22E-07	.6	ZZS2U1	SMALL LOCA
				NHFL1RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS
U1S2U	V	1.14E-07	.5	ZZS2U1	SMALL LOCA
				GMM1FTRCFI	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
U1TBF	III	1.12E-07	.5	ZZT6U1	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM1FTRCFI	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A1	SIT 1A1 INJECTION PATH FAILS
				ZZCLB1A2	LOCA IN COLD LEG 1A2

Table 3.7-4 St. Lucie Unit 1 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A1	SIT 1A1 INJECTION PATH FAILS
				ZZCLB1B1	LOCA IN COLD LEG 1B1
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A1	SIT 1A1 INJECTION PATH FAILS
				ZZCLB1B2	LOCA IN COLD LEG 1B2
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A2	SIT 1A2 INJECTION PATH FAILS
				ZZCLB1B2	LOCA IN COLD LEG 1B2
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A2	SIT 1A2 INJECTION PATH FAILS
				ZZCLB1A1	LOCA IN COLD LEG 1A1
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11A2	SIT 1A2 INJECTION PATH FAILS
				ZZCLB1B1	LOCA IN COLD LEG 1B1
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B1	SIT 1B1 INJECTION PATH FAILS
				ZZCLB1A1	LOCA IN COLD LEG 1A1
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B1	SIT 1B1 INJECTION PATH FAILS
				ZZCLB1B2	LOCA IN COLD LEG 1B2
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B1	SIT 1B1 INJECTION PATH FAILS
				ZZCLB1A2	LOCA IN COLD LEG 1A2
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B2	SIT 1B2 INJECTION PATH FAILS
				ZZCLB1A2	LOCA IN COLD LEG 1A2
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B2	SIT 1B2 INJECTION PATH FAILS
				ZZCLB1B1	LOCA IN COLD LEG 1B1
U1AU	V	9.68E-08	.5	ZZAU1	LARGE LOCA
				BMM11B2	SIT 1B2 INJECTION PATH FAILS
				ZZCLB1A1	LOCA IN COLD LEG 1A1
U1TBFB	III	9.38E-08	.4	ZZLOG	LOSS OF GRID
				EMM1CCFDGR	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO RUN FOR 24 HOURS
				REPS1CASE6	OFF-SITE POWER NON-RECOVERY CASE 6:CCF OF DIESELS TO RUN
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	9.35E-08	.4	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM1CCFDGR	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO RUN FOR 24 HOURS
				REPS1CASE6	OFF-SITE POWER NON-RECOVERY CASE 6:CCF OF DIESELS TO RUN
U1AXC	VI	8.14E-08	.4	ZZAU1	LARGE LOCA
				GMM1MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U1TQX	II	8.04E-08	.4	ZZICWU1	LOSS OF ICW
				RTOP1S1RCP	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U1AU	V	7.98E-08	.4	ZZAU1	LARGE LOCA
				BHFL1LVL	CCF OF SITs DUE TO MISCALIBRATION OF SIT LEVEL SENSORS
U1AU	V	7.98E-08	.4	ZZAU1	LARGE LOCA
				BHFL1PRS	CCF OF SITs DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS
U1S2X	VI	7.80E-08	.4	ZZS2U1	SMALL LOCA
				QMM1MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE
U1AXC	VI	7.47E-08	.3	ZZAU1	LARGE LOCA
				GMM1FTRCFR	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING RECIRCULATION
U1TBFB	III	7.34E-08	.3	ZZLOG	LOSS OF GRID
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				REPS1CASE4	OFF-SITE POWER NON-RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	7.32E-08	.3	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				REPS1CASE4	OFF-SITE POWER NON-RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN

Table 3.7-4 St. Lucie Unit 1 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U1S1X	II	7.07E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				GMM1ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GTM1PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE
U1S1X	II	7.07E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				GMM1BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
				GTM1PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U1S1X	II	7.03E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				CMM1MPFCCF	CCW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
U1S1X	II	7.03E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				QMM1MPFCCF	ICW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
U1TBF	III	6.66E-08	.3	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				ECBD120102	AC BREAKER 20102 FAILS ON DEMAND (A AUX)
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
U1TBF	III	6.66E-08	.3	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				ECBD120302	AC BREAKER 20302 FAILS TO CLOSE (B SU)
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
U1S1X	II	6.54E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GMM1BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
U1S1X	II	6.54E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GMM1ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
U1RDX	IIR	6.48E-08	.3	ZZRU1A	SG 1A TUBE RUPTURE
				GTKJ1RWT	REFUELING WATER TANK RUPTURE
U1RDX	IIR	6.48E-08	.3	ZZRU1B	SG 1B TUBE RUPTURE
				GTKJ1RWT	REFUELING WATER TANK RUPTURE
U1AU	V	6.35E-08	.3	ZZAU1	LARGE LOCA
				JAVK13306S	AIR-OPERATED VALVE FCV-3306 TRANSFERS CLOSE DURING STANDBY
				JMVK103-2S	MOTOR-OPERATED VALVE MV-03-2 TRANSFERS CLOSED DURING STANDBY
U1S1X	II	5.78E-08	.3	ZZS1U1	SMALL-SMALL LOCA
				GMM1ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GMM1BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
U1TBFB	III	5.68E-08	.3	ZZLOG	LOSS OF GRID
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				ETM11BEDG	1B EDG IN TEST OR MAINTENANCE
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	5.68E-08	.3	ZZLOG	LOSS OF GRID
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				ETM11AEDG	EDG 1A IN TEST OR MAINTENANCE
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	5.66E-08	.3	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				ETM11AEDG	EDG 1A IN TEST OR MAINTENANCE
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
U1TBFB	III	5.66E-08	.3	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				ETM11BEDG	EDG IN TEST OR MAINTENANCE
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
U1AU	V	5.27E-08	.2	ZZAU1	LARGE LOCA
				JMM1MPACFI	COMMON CAUSE FAILURE OF LPSI PUMPS TO START DURING INJECTION
U1S1U	I	5.22E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMM1PAFTS	FAILURE OF HPSI PUMP A TO START
				GTM1PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE
U1S1U	I	5.22E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMM1PBFTS	FAILURE OF HPSI PUMP B TO START
				GTM1PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U1AXC	VI	5.11E-08	.2	ZZAU1	LARGE LOCA
				QMM1MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE

Table 3.7-4 St. Lucie Unit 1 Dominant Cutsets

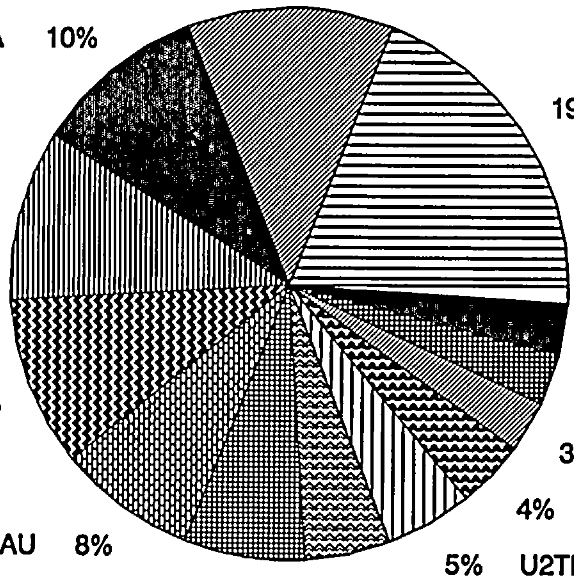
Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U1TBF	III	4.89E-08	.2	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AMM1PAFTS	AFW PUMP 1A FAILS TO START
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1TBF	III	4.89E-08	.2	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				AMM1PBFTS	AFW PUMP 1B FAILS TO START
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1TBFB	III	4.84E-08	.2	ZZLOG	LOSS OF GRID
				EMM1CCFDGS	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO START
				REPS1CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1AU	V	4.83E-08	.2	ZZAU1	LARGE LOCA
				JMM1MPFCFI	COMMON CAUSE FAILURE OF LPSI PUMPS TO RUN DURING INJECTION
U1S1X	II	4.83E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GMM1PBFTS	FAILURE OF HPSI PUMP B TO START
U1S1X	II	4.83E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GMM1PAFTS	FAILURE OF HPSI PUMP A TO START
U1TBFB	III	4.82E-08	.2	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM1CCFDGS	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO START
				REPS1CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
U1TBF	III	4.69E-08	.2	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
				ATM1AFWP1B	AFW PUMP 1B TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
U1TBF	III	4.69E-08	.2	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
				ATM1AFWP1A	AFW PUMP 1A TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
U1TBF	III	4.37E-08	.2	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AMM1SGAP1A	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS
				R#DCAB	OPERATOR FAILS TO REALIGN 'AB' DC BUS
U1S1X	II	4.27E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMM1ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GMM1PBFTS	FAILURE OF HPSI PUMP B TO START
U1S1X	II	4.27E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMM1BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
				GMM1PAFTS	FAILURE OF HPSI PUMP A TO START
U1KC	III	4.13E-08	.2	ZZT3AU1	LOSS OF MAIN FEEDWATER BUT RECOVERABLE
				-ZZMTCUNF	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
				NMM1CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ1ABKSHUT	'A' BLK VLV CLOSE W/POWER
U1KC	III	4.13E-08	.2	ZZT3AU1	LOSS OF MAIN FEEDWATER BUT RECOVERABLE
				-ZZMTCUNF	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
				NMM1CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ1BBKSHUT	'B' BLK VLV CLOSED W/POWER
U1S1U	I	3.64E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMVK13654S	MOTOR-OPERATED VALVE V3654 TRANSFERS CLOSED DURING STANDBY
				GTM1PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U1S1U	I	3.64E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GMVK13656S	MOTOR-OPERATED VALVE V3656 TRANSFERS CLOSED DURING STANDBY
				GTM1PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE
U1TBF	III	3.40E-08	.2	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AMM1PAFTS	AFW PUMP 1A FAILS TO START
				ATM1AFWP1C	AFW PUMP 1C TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
U1TBF	III	3.40E-08	.2	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				AMM1PBFTS	AFW PUMP 1B FAILS TO START
				ATM1AFWP1C	AFW PUMP 1C TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
U1S1X	II	3.37E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GMVK13654S	MOTOR-OPERATED VALVE V3654 TRANSFERS CLOSED DURING STANDBY
U1S1X	II	3.37E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GMVK13656S	MOTOR-OPERATED VALVE V3656 TRANSFERS CLOSED DURING STANDBY

Table 3.7-4 St. Lucie Unit 1 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U1S1U	I	3.32E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE
				GTM1PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE
U1S1U	I	3.32E-08	.2	ZZS1U1	SMALL-SMALL LOCA
				GHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE
				GTM1PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U1TBFB	III	3.29E-08	.2	ZZLOG	LOSS OF GRID
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				EMM1BDGFTS	EDG 1B FAILS TO START
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	3.29E-08	.2	ZZLOG	LOSS OF GRID
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				EMM1ADGFTS	EDG 1A FAILS TO START
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
				REPS1XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
U1TBFB	III	3.28E-08	.2	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM11AEDG	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
				EMM1BDGFTS	EDG 1B FAILS TO START
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
U1TBFB	III	3.28E-08	.2	ZZLOG	LOSS OF GRID
				E2XTIETOU1	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
				EMM11BEDG	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
				EMM1ADGFTS	EDG 1A FAILS TO START
				REPS1CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
U1TBF	III	3.18E-08	.1	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AMM1PAFTR	AFW PUMP 1A FAILS TO RUN
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1TBF	III	3.18E-08	.1	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				AMM1PBFTR	AFW PUMP 1B FAILS TO RUN
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1S1U	I	3.15E-08	.1	ZZS1U1	SMALL-SMALL LOCA
				GMM1PAFTS	FAILURE OF HPSI PUMP A TO START
				GMM1PBFTS	FAILURE OF HPSI PUMP B TO START
U1TBF	III	3.11E-08	.1	ZZDC1B	LOSS OF DC BUS 1B FOR UNIT 1
				AHFL109108	AFW PUMP 1A MANUAL VALVE V09108 MISPOSITIONED
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1TBF	III	3.11E-08	.1	ZZDC1A	LOSS OF DC BUS 1A FOR UNIT 1
				AHFL109124	AFW PUMP 1B MANUAL VALVE V09124 MISPOSITIONED
				AMM1PCFTS	AFW PUMP 1C FAILS TO START
U1KC	III	3.10E-08	.1	ZZT2U1	REACTOR TRIP (PORV ACTUATED)
				-ZZMTCUNF	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
				NMM1CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ1ABKSHUT	'A' BLK VLV CLOSE W/POWER
U1KC	III	3.10E-08	.1	ZZT2U1	REACTOR TRIP (PORV ACTUATED)
				-ZZMTCUNF	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
				NMM1CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ1BBKSHUT	'B' BLK VLV CLOSED W/POWER
U1S1X	II	3.08E-08	.1	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE
U1S1X	II	3.08E-08	.1	ZZS1U1	SMALL-SMALL LOCA
				CTM1CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE
U1AU	V	3.01E-08	.1	ZZAU1	LARGE LOCA
				JMM1PAFTRI	FAILURE OF LPSI PUMP A TO RUN DURING INJECTION
				JMM1PBFTRI	FAILURE OF LPSI PUMP B TO RUN DURING INJECTION

TOTAL: 1.29E-05 60.11% of CM Total Frequency 2.14E-05

Figure 3.7-3 St. Lucie Unit 2 Core Damage Frequency by Sequence

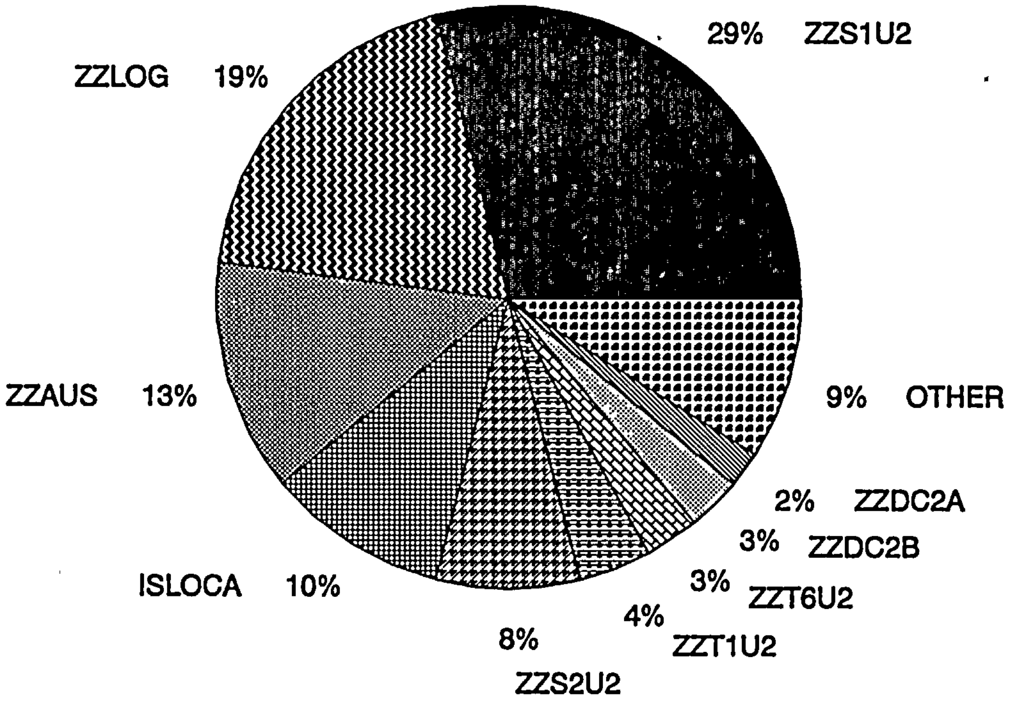


Sequence	Description	Core Damage Frequency Contribution	% of Total
U2S1X	Small-Small LOCA w/Long Term Core Cooling Failure	5.00E-06	19%
U2TBF	Transient w/Secondary Heat Removal and OTC Failure (Non-Blackout)	3.03E-06	12%
ISLOCA	Interfacing System LOCA	2.72E-06	10%
U2TBFB	Transient w/Secondary Heat Removal and OTC Failure (Blackout)	2.64E-06	10%
U2S1U	Small-Small LOCA w/Injection Failure	2.50E-06	10%
U2AU	Large LOCA w/Injection Failure	2.19E-06	8%
U2KC	ATWS w/ Short Term Core Cooling Failure	1.76E-06	7%
U2S2X	Small LOCA w/Long Term Core Cooling Failure	1.39E-06	5%
U2TBX	Transient w/Secondary Heat Removal Failure and Long Term Core Cooling Failure	1.28E-06	5%
U2AXC	Large LOCA w/Cold Leg Recirc Failure	1.11E-06	4%
U2TQX	Transient w/Loss of RCS Integrity and Long Term Core Cooling Failure	8.04E-07	3%
U2RDX	SGTR w/Failure to Terminate Leakage and Long Term Core Cooling Failure	8.00E-07	3%
U2S2U	Small LOCA w/Injection Failure	7.17E-07	3%
Other		3.00E-07	1%
Total Freq:		2.62E-05	

Table 3.7-5 St. Lucie Unit 2 Core Damage Frequency by Sequence

Sequence	Description	Core Damage Frequency Contribution	% of Total
U2S1X	Small-Small LOCA w/Long Term Core Cooling Failure	5.00E-06	19%
U2TBF	Transient w/Secondary Heat Removal and OTC Failure (Non-Blackout)	3.03E-06	12%
ISLOCA	Interfacing System LOCA	2.72E-06	10%
U2TBFB	Transient w/Secondary Heat Removal and OTC Failure (Blackout)	2.64E-06	10%
U2S1U	Small-Small LOCA w/Injection Failure	2.50E-06	10%
U2AU	Large LOCA w/Injection Failure	2.19E-06	8%
U2KC	ATWS w/ Short Term Core Cooling Failure	1.76E-06	7%
U2S2X	Small LOCA w/Long Term Core Cooling Failure	1.39E-06	5%
U2TBX	Transient w/Secondary Heat Removal Failure and Long Term Core Cooling Failure	1.28E-06	5%
U2AXC	Large LOCA w/Cold Leg Recirc Failure	1.11E-06	4%
U2TQX	Transient w/Loss of RCS Integrity and Long Term Core Cooling Failure	8.04E-07	3%
U2RDX	SGTR w/Failure to Terminate Leakage and Long Term Core Cooling Failure	8.00E-07	3%
U2S2U	Small LOCA w/Injection Failure	7.17E-07	3%
U2TQU	Transient w/Loss of RCS Integrity and Injection Failure	1.93E-07	1%
U2RBF	SGTR w/Secondary Heat Removal Failure and OTC Failure	9.09E-08	<1%
U2RBX	SGTR w/Secondary Heat Removal Failure and Long Term Core Cooling Failure	8.17E-09	<1%
U2TQBFB	Transient w/Loss of RCS Integrity, Secondary Heat Removal Failure, and OTC Failure (Blackout)	4.22E-09	<1%
U2TQBF	Transient w/Loss of RCS Integrity, Secondary Heat Removal Failure, and OTC Failure (Non-Blackout)	2.24E-09	<1%
U2S1BF	Small-Small LOCA w/Secondary Heat Removal Failure and OTC Failure	1.12E-09	<1%
U2KB	ATWS w/Secondary Heat Removal Failure	2.10E-10	<1%
U2TQBX	Transient w/Loss of RCS Integrity, Secondary Heat Removal Failure, and Long Term Core Cooling Failure	7.51E-11	<1%
Total Freq:		2.62E-05	

Figure 3.7-4 St. Lucie Unit 2 Core Damage Frequency by Initiator



Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U2	Small-Small LOCA	7.49E-06	29%
ZZLOG	Loss of Grid	4.95E-06	19%
ZZAU2	Large LOCA	3.30E-06	13%
ISLOCA	Interfacing System LOCA	2.73E-06	10%
ZZS2U2	Small LOCA	2.11E-06	8%
ZZT1U2	Reactor Trip	1.17E-06	4%
ZZT6U2	Steamline Break Downstream of MSIVs	6.99E-07	3%
ZZDC2B	Loss of DC Bus 2B	6.67E-07	3%
ZZDC2A	Loss of DC Bus 2A	5.69E-07	2%
Other	N/A	2.56E-06	9%

Total Freq: 2.6E-05

Table 3.7-6 St. Lucie Unit 2 Core Damage Frequency by Initiator

Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U2	Small-Small LOCA	7.49E-06	29%
ZZLOG	Loss of Grid	4.95E-06	19%
ZZAU2	Large LOCA	3.30E-06	13%
ISLOCA	Interfacing System LOCA	2.72E-06	10%
ZZS2U2	Small LOCA	2.11E-06	8%
ZZT1U2	Reactor Trip	1.17E-06	4%
ZZT6U2	Steamline Break Downstream of MSIVs	6.99E-07	3%
ZZDC2B	Loss of DC Bus 2B	6.67E-07	3%
ZZDC2A	Loss of DC Bus 2A	5.69E-07	2%
ZZRU2B	SGTR - S/G 2B	4.58E-07	2%
ZZRU2A	SGTR - S/G 2A	4.41E-07	2%
ZZT3CU2	LOFW - Not Recoverable	3.52E-07	1%
ZZCCWU2	Loss of CCW	2.83E-07	1%
ZZT3AU2	LOFW - Recoverable	2.44E-07	1%
ZZT2U2	Reactor Trip (PORV Challenge)	2.01E-07	1%
ZZT3EU2	Excessive Feedwater	1.32E-07	1%
ZZIAU2	Loss of Instrument Air	1.06E-07	<1%
ZZT5U2A	Upstream Steamline Break - S/G 2A	9.14E-08	<1%
ZZT5U2B	Upstream Steamline Break - S/G 2B	9.14E-08	<1%
ZZICWU2	Loss of ICW	8.05E-08	<1%
ZZT8BU2	PORV Sticking Open - S/G 2B	3.37E-08	<1%
ZZMAU2	Loss of Instrument Bus 2MA	8.91E-09	<1%
ZZMBU2	Loss of Instrument Bus 2MB	8.91E-09	<1%
ZZMCU2	Loss of Instrument Bus 2MC	8.91E-09	<1%
ZZMDU2	Loss of Instrument Bus 2MD	8.91E-09	<1%
ZZT7U2	Spurious SIAS	8.91E-09	<1%
ZZT8AU2	PORV Sticking Open - S/G 2A	1.51E-09	<1%
ZZ4KV2A2	Loss of 4kV Bus 2A2	6.66E-10	<1%
ZZ4KV2B2	Loss of 4kV Bus 2B2	6.66E-10	<1%
ZZTCWU2	Loss of TCW	9.25E-11	<1%
ZZT3DU2	Feedline Break (Common)	9.01E-11	<1%
ZZT3DU2A	Feedline Break S/G 2A	9.01E-11	<1%
ZZT3DU2B	Feedline Break S/G 2B	9.01E-11	<1%
ZZ6KV2A1	Loss of 6.9kV Bus 2A1	4.01E-11	<1%
ZZ6KV2B1	Loss of 6.9kV Bus 2B1	4.01E-11	<1%
ZZT4B	Loss of Offsite Power "B" Train	1.01E-11	<1%
ZZT4A	Loss of Offsite Power "A" Train	3.29E-12	<1%
Total Freq:		2.62E-05	

Table 3.7-7 St. Lucie Unit 2 Core Damage Contribution by Transient Group

Transient Group	Transient Initiators in Group	Sequence Contribution							
		TQBF	TQU	TQX	TBF	TBFB	TBX	K	Total (Per Year)
1	ZZLOG	2.90E-09	6.21E-08	9.31E-08	9.14E-07	2.68E-06	1.11E-06	3.89E-08	4.90E-06
2	ZZ5U2A ZZ5U2B ZZT6U2 ZZT7U2 ZZT8AU2 ZZT8BU2 ZZDC1A ZZDC1B	4.95E-10	3.25E-08	4.16E-08	1.87E-06		1.88E-07	2.06E-08	2.15E-06
3	ZZT1U2 ZZT2U2 ZZT3AU2 ZZT3CU2 ZZT3EU2	1.69E-09	1.24E-08	1.54E-07	2.88E-07			1.64E-08	2.10E-06
4	ZZT3DU2 ZZT3DU2A ZZT3DU2B ZZT4AU2 ZZT4BU2 ZZ4KV2A2 ZZ4KV2B2 ZZ6KV2A1 ZZ6KV2B1 ZZCCWU2 ZZICWU2 ZZTCWU2 ZZIAU2 ZZMAU2 ZZMBU2 ZZMCU2 ZZMDU2		4.59E-08	4.58E-07				1.12E-08	5.15E-07
Total Transient Contribution		5.09E-09	1.53E-07	7.47E-07	3.07E-06	2.68E-06	1.30E-06	1.71E-06	9.66E-06

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2TBFB	III	9.98E-07	4.2	ZZLOG	LOSS OF GRID
				EMM24CCFTR	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO RUN
				REPS2CASE6	OFF-SITE POWER RECOVERY CASE 6: CCF OF DIESELS TO RUN
U2S1X	II	7.73E-07	3.3	ZZS1U2	SMALL-SMALL LOCA
				CMM2AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U2S1U	I	7.00E-07	3.0	ZZS1U2	SMALL-SMALL LOCA
				GMM2HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U2S1X	II	7.00E-07	3.0	ZZS1U2	SMALL-SMALL LOCA
				GMM2SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U2S1X	II	5.68E-07	2.4	ZZS1U2	SMALL-SMALL LOCA
				NHFL2RWLCF	COMMON CAUSE MISCALIBRATION OF RWT LEVEL TRANSMITTERS
U2TBFB	III	5.15E-07	2.2	ZZLOG	LOSS OF GRID
				EMM24CCFTS	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO START
				REPS2CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
U2S1U	I	4.35E-07	1.8	ZZS1U2	SMALL-SMALL LOCA
				GMM2MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U2KC	III	4.25E-07	1.8	ZZT1U2	REACTOR TRIPS
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2TQX	II	2.82E-07	1.2	ZZCCWU2	LOSS OF CCW
				RTOP2S1RCP	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U2S1X	II	2.73E-07	1.2	ZZS1U2	SMALL-SMALL LOCA
				QMM2MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U2KC	III	2.24E-07	1.0	ZZT1U2	REACTOR TRIPS
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKRO	'B' BLK VLV CLOSED W/O POWER
U2S2X	VI	2.21E-07	.9	ZZS2U2	SMALL LOCA
				CMM2AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U2S2U	V	2.00E-07	.9	ZZS2U2	SMALL LOCA
				GMM2HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U2S2X	VI	2.00E-07	.9	ZZS2U2	SMALL LOCA
				GMM2SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U2S1U	I	1.99E-07	.8	ZZS1U2	SMALL-SMALL LOCA
				GMM2CFTRIS	CCF OF HPSI PUMPS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
U2S1X	II	1.99E-07	.8	ZZS1U2	SMALL-SMALL LOCA
				GMM2CFTRRS	CCF OF HPSI PUMPS TO RUN DURING RECIRC FOLLOWING SMALL-SMALL LOCA
U2TBF	III	1.97E-07	.8	ZZT6U2	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM2HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U2TBX	IV	1.79E-07	.8	ZZLOG	LOSS OF GRID
				AMM2SGAP2A	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS TO S/G 2A
				EMM22BEDG	EDG 2B FAILS TO RUN
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5AU2	ELECT EQUIP ROOM SUPPLY FAN 5A IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
U2TBX	IV	1.79E-07	.8	ZZLOG	LOSS OF GRID
				AMM2SGBP2B	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS TO S/G 2B
				EMM22AEDG	EDG 2A FAILS TO RUN
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5BU2	ELECT EQUIP ROOM SUPPLY FAN 5B IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
U2KC	III	1.66E-07	.7	ZZT1U2	REACTOR TRIPS
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				ZZPWRLVL	REACTOR AT HIGH POWER BEFORE TRIP

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2S2X	VI	1.62E-07	.7	ZZS2U2	SMALL LOCA
				NHFL2RWLCF	COMMON CAUSE MISCALIBRATION OF RWT LEVEL TRANSMITTERS
U2AXC	VI	1.45E-07	.6	ZZAU2	LARGE LOCA
				CMM2AVCCCF	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U2AU	V	1.31E-07	.6	ZZAU2	LARGE LOCA
				JMM2HCVCCF	COMMON CAUSE FAILURE OF LPSI INJECTION VALVES TO OPEN DURING
U2AXC	VI	1.31E-07	.6	ZZAU2	LARGE LOCA
				GMM2HCVCCF	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
U2AXC	VI	1.31E-07	.6	ZZAU2	LARGE LOCA
				GMM2SMVCCF	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
U2TBF	III	1.28E-07	.5	ZZDC2B	LOSS OF DC BUS 2B FOR UNIT 2
				EMM22AEDG	EDG 2A FAILS TO RUN
				ETM2ASU	2A STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
U2TBF	III	1.28E-07	.5	ZZDC2A	LOSS OF DC BUS 2A FOR UNIT 2
				EMM22BEDG	EDG 2B FAILS TO RUN
				ETM2BSU	2B STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
U2S2U	V	1.24E-07	.5	ZZS2U2	SMALL LOCA
				GMM2MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U2TBF	III	1.22E-07	.5	ZZT6U2	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM2MPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U2S2U	V	1.14E-07	.5	ZZS2U2	SMALL LOCA
				GMM2FTRCFI	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
U2S2X	VI	1.14E-07	.5	ZZS2U2	SMALL LOCA
				GMM2FTRCFR	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING RECIRCULATION
U2KC	III	1.14E-07	.5	ZZT1U2	REACTOR TRIPS
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABLKROR	'A' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2TBF	III	1.12E-07	.5	ZZT6U2	STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
				GMM2FTRCFI	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
U2KC	III	1.07E-07	.5	ZZT3AU2	LOSS OF MAIN FEEDWATER BUT RECOVERABLE
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A1	SIT 2A1 INJECTION PATH FAILS
				ZZCLB2B1	LOCA IN COLD LEG 2B1
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A1	SIT 2A1 INJECTION PATH FAILS
				ZZCLB2A2	LOCA IN COLD LEG 2A2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A1	SIT 2A1 INJECTION PATH FAILS
				ZZCLB2B2	LOCA IN COLD LEG 2B2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A2	SIT 2A2 INJECTION PATH FAILS
				ZZCLB2A1	LOCA IN COLD LEG 2A1
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A2	SIT 2A2 INJECTION PATH FAILS
				ZZCLB2B1	LOCA IN COLD LEG 2B1
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22A2	SIT 2A2 INJECTION PATH FAILS
				ZZCLB2B2	LOCA IN COLD LEG 2B2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B1	SIT 2B1 INJECTION PATH FAILS
				ZZCLB2B2	LOCA IN COLD LEG 2B2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B1	SIT 2B1 INJECTION PATH FAILS
				ZZCLB2A1	LOCA IN COLD LEG 2A1

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B1	SIT 2B1 INJECTION PATH FAILS
				ZZCLB2A2	LOCA IN COLD LEG 2A2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B2	SIT 2B2 INJECTION PATH FAILS
				ZZCLB2B1	LOCA IN COLD LEG 2B1
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B2	SIT 2B2 INJECTION PATH FAILS
				ZZCLB2A2	LOCA IN COLD LEG 2A2
U2AU	V	9.68E-08	.4	ZZAU2	LARGE LOCA
				BMM22B2	SIT 2B2 INJECTION PATH FAILS
				ZZCLB2A1	LOCA IN COLD LEG 2A1
U2TBFB	III	9.38E-08	.4	ZZLOG	LOSS OF GRID
				EMM2CCFDGR	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO RUN
				REPS2CASE6	OFF-SITE POWER RECOVERY CASE 6: CCF OF DIESELS TO RUN
				REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
U2TBFB	III	9.07E-08	.4	ZZLOG	LOSS OF GRID
				E1XTIETOU2	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
				EMM2CCFDGR	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO RUN
				REPS2CASE6	OFF-SITE POWER RECOVERY CASE 6: CCF OF DIESELS TO RUN
U2AXC	VI	8.14E-08	.3	ZZAU2	LARGE LOCA
				GMM2HPACCF	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
U2TQX	II	8.04E-08	.3	ZZICWU2	LOSS OF ICW
				RTOP2S1RCP	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U2KC	III	7.99E-08	.3	ZZT2U2	REACTOR TRIP (PORV ACTUATED)
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2AU	V	7.98E-08	.3	ZZAU2	LARGE LOCA
				BHFL2LVL	CCF OF SITs DUE TO MISCALIBRATION OF SIT LEVEL SENSORS
U2AU	V	7.98E-08	.3	ZZAU2	LARGE LOCA
				BHFL2PRS	CCF OF SITs DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS
U2S2X	VI	7.80E-08	.3	ZZS2U2	SMALL LOCA
				QMM2MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE
U2AXC	VI	7.47E-08	.3	ZZAU2	LARGE LOCA
				GMM2FTRCFR	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING RECIRCULATION
U2S1U	I	7.35E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				GMVR23523	MOTOR-OPERATED VALVE V3523 TRANSFERS OPEN DURING STANDBY
				GMVR23551	MOTOR-OPERATED VALVE V3551 TRANSFERS OPEN
				ZZHLR-INJ	PROBABILITY THAT CORE COOLING DURING INJECTION WILL BE LOST
U2S1U	I	7.35E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				GMVR23540	MOTOR-OPERATED VALVE 3540 TRANSFERS OPEN DURING STANDBY
				GMVR23550	MOTOR-OPERATED VALVE V3550 TRANSFERS OPEN
				ZZHLR-INJ	PROBABILITY THAT CORE COOLING DURING INJECTION WILL BE LOST
U2TBFB	III	7.34E-08	.3	ZZLOG	LOSS OF GRID
				EMM22AEDG	EDG 2A FAILS TO RUN
				EMM22BEDG	EDG 2B FAILS TO RUN
				REPS2CASE4	OFF-SITE POWER RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN
				REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
U2TBFB	III	7.10E-08	.3	ZZLOG	LOSS OF GRID
				E1XTIETOU2	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
				EMM22AEDG	EDG 2A FAILS TO RUN
				EMM22BEDG	EDG 2B FAILS TO RUN
				REPS2CASE4	OFF-SITE POWER RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN
U2S1X	II	7.07E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				GMM2ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GTM2PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE
U2S1X	II	7.07E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				GMM2BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
				GTM2PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U2S1X	II	7.03E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				CMM2MPFCCF	CCW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2S1X	II	7.03E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				QMM2MPFCCF	ICW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
U2TBX	IV	6.75E-08	.3	ZZLOG	LOSS OF GRID
				AMM2PAFTS	AFW PUMP 2A FAILS TO START
				EMM22BEDG	EDG 2B FAILS TO RUN
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5AU2	ELECT EQUIP ROOM SUPPLY FAN 5A IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
U2TBX	IV	6.75E-08	.3	ZZLOG	LOSS OF GRID
				AMM2PBFTS	AFW PUMP 2B FAILS TO START
				EMM22AEDG	EDG 2A FAILS TO RUN
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5BU2	ELECT EQUIP ROOM SUPPLY FAN 5B IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
U2TBF	III	6.66E-08	.3	ZZDC2B	LOSS OF DC BUS 2B FOR UNIT 2
				ECBD220102	AC BREAKER 20102 FAILS TO CLOSE
				EMM22AEDG	EDG 2A FAILS TO RUN
U2TBF	III	6.66E-08	.3	ZZDC2A	LOSS OF DC BUS 2A FOR UNIT 2
				ECBD220302	AC BREAKER 20302 FAILS TO CLOSE
				EMM22BEDG	EDG 2B FAILS TO RUN
U2S1X	II	6.54E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				CTM2CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GMM2BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
U2S1X	II	6.54E-08	.3	ZZS1U2	SMALL-SMALL LOCA
				CTM2CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GMM2ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
U2RDX	IIR	6.48E-08	.3	ZZRU2A	SG 2A TUBE RUPTURE
				GTKJ2RWT	REFUELING WATER TANK RUPTURE
U2RDX	IIR	6.48E-08	.3	ZZRU2B	SG 2B TUBE RUPTURE
				GTKJ2RWT	REFUELING WATER TANK RUPTURE
U2KC	III	5.99E-08	.3	ZZT1U2	REACTOR TRIPS
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABLKRO	'A' BLK VLV CLOSED W/O POWER
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2S1X	II	5.78E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMM2ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GMM2BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
U2TBFB	III	5.68E-08	.2	ZZLOG	LOSS OF GRID
				EMM22AEDG	EDG 2A FAILS TO RUN
				ETM22BEDG	2B EDG UNAVAILABLE DUE TO MAINTENANCE
				REPS2CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FAILS TO START/1 DIESEL FAILS TO RUN
				REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
U2TBFB	III	5.68E-08	.2	ZZLOG	LOSS OF GRID
				EMM22BEDG	EDG 2B FAILS TO RUN
				ETM22AEDG	EDG 2A UNAVAILABLE DUE TO MAINTENANCE
				REPS2CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FAILS TO START/1 DIESEL FAILS TO RUN
				REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
U2KC	III	5.61E-08	.2	ZZT3AU2	LOSS OF MAIN FEEDWATER BUT RECOVERABLE
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKRO	'B' BLK VLV CLOSED W/O POWER
U2TBX	IV	5.57E-08	.2	ZZLOG	LOSS OF GRID
				AMM2SGAP2A	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS S/G 2A
				ETM22BEDG	2B EDG UNAVAILABLE DUE TO MAINTENANCE
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5AU2	ELECT EQUIP ROOM SUPPLY FAN 5A IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2TBX	IV	5.57E-08	.2	ZZLOG	LOSS OF GRID
				AMM2SGBP2B	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS TO S/G 2B
				ETM22AEDG	EDG 2A UNAVAILABLE DUE TO MAINTENANCE
				RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
				ZZHVS5BU2	ELECT EQUIP ROOM SUPPLY FAN 5B IDLE PRIOR TO INITIATOR
				REPS2CA1A2	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
U2S1U	I	5.49E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				IMM2EERCCF	COMMON CAUSE FAILURE OF ELECT EQUIP ROOM FANS TO RUN
U2TBFB	III	5.49E-08	.2	ZZLOG	LOSS OF GRID
				E1XTIETOU2	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
				EMM22AEDG	EDG 2A FAILS TO RUN
				ETM22BEDG	2B EDG UNAVAILABLE DUE TO MAINTENANCE
				REPS2CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FAILS TO START/1 DIESEL FAILS TO RUN
U2TBFB	III	5.49E-08	.2	ZZLOG	LOSS OF GRID
				E1XTIETOU2	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
				EMM22BEDG	EDG 2B FAILS TO RUN
				ETM22AEDG	EDG 2A UNAVAILABLE DUE TO MAINTENANCE
				REPS2CASE3	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FAILS TO START/1 DIESEL FAILS TO RUN
U2KC	III	5.33E-08	.2	ZZT3EU2	EXCESSIVE FEEDWATER
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2AU	V	5.27E-08	.2	ZZAU2	LARGE LOCA
				JMM2MPACFI	COMMON CAUSE FAILURE OF LPSI PUMPS TO START DURING INJECTION
U2S1U	I	5.22E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMM2PAFTS	FAILURE OF HPSI PUMP A TO START
				GTM2PUMPB	HPSI PUMP B IN TEST OR MAINTENANCE.
U2S1U	I	5.22E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMM2PBFTS	FAILURE OF HPSI PUMP B TO START
				GTM2PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE
U2AXC	VI	5.11E-08	.2	ZZAU2	LARGE LOCA
				QMM2MVCCCF	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
U2TBFB	III	4.84E-08	.2	ZZLOG	LOSS OF GRID
				EMM2CCFDGS	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO START
				REPS2CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
				REPS2XTIE	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
U2AU	V	4.83E-08	.2	ZZAU2	LARGE LOCA
				JMM2MPFCFI	COMMON CAUSE FAILURE OF LPSI PUMPS TO RUN DURING INJECTION
U2S1X	II	4.83E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				CTM2CCWHXA	CCW HX A IN TEST OR MAINTENANCE
				GMM2PBFTS	FAILURE OF HPSI PUMP B TO START
U2S1X	II	4.83E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				CTM2CCWHXB	CCW HX B IN TEST OR MAINTENANCE
				GMM2PAFTS	FAILURE OF HPSI PUMP A TO START
U2TBFB	III	4.68E-08	.2	ZZLOG	LOSS OF GRID
				E1XTIETOU2	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
				EMM2CCFDGS	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO START
				REPS2CASE5	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
U2S1X	II	4.27E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMM2ASUMP	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
				GMM2PBFTS	FAILURE OF HPSI PUMP B TO START
U2S1X	II	4.27E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMM2BSUMP	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
				GMM2PAFTS	FAILURE OF HPSI PUMP A TO START
U2RDX	IIR	4.25E-08	.2	ZZRU2B	SG 2B TUBE RUPTURE
				GMM2RCVCCF	COMMON CAUSE FAILURE OF RWT OUTLET CHECK VALVES TO OPEN
U2RDX	IIR	4.25E-08	.2	ZZRU2A	SG 2A TUBE RUPTURE
				GMM2RCVCCF	COMMON CAUSE FAILURE OF RWT OUTLET CHECK VALVES TO OPEN

Table 3.7-8 St. Lucie Unit 2 Dominant Cutsets

Sequence Name	Plant Damage Class	Freq. Measure	Percent (%)	Accident Sequence Events	
U2KC	III	4.21E-08	.2	ZZT2U2	REACTOR TRIP (PORV ACTUATED)
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED W/POWER AVAILABLE
				ZZ2BBLKRO	'B' BLK VLV CLOSED W/O POWER
U2KC	III	4.16E-08	.2	ZZT3AU2	LOSS OF MAIN FEEDWATER BUT RECOVERABLE
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				ZZPWRLVL	REACTOR AT HIGH POWER BEFORE TRIP
U2KC	III	4.04E-08	.2	ZZLOG	LOSS OF GRID
				-ZZMTCUNFU2	MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE (UNIT 2)
				NMM2CEDM	MECHANICAL FAULT PREVENTING ROD INSERTION
				ZZ2ABKSHUT	'A' BLK VLV CLOSED WLOSS OF MAIN FEEDWATER BUT RECOVERABLE
				ZZ2BBLKROR	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
U2TBF	III	3.85E-08	.2	ZZT3CU2	LOSS OF MAIN FEEDWATER BUT NOT RECOVERABLE
				AMM2HDMOV	COMMON CAUSE FAILURE OF AFW AC REGULATING VALVES
				AMM2PCFTS	AFW PUMP 2C FAILS TO START
				ZZ2ABLKRO	'A' BLK VLV CLOSED W/O POWER
				ZZ2BBLKRO	'B' BLK VLV CLOSED W/O POWER
U2S1U	I	3.64E-08	.2	ZZS1U2	SMALL-SMALL LOCA
				GMVK23654S	MOTOR-OPERATED VALVE V3654 TRANSFERS CLOSED DURING STANDBY
				GTM2PUMPA	HPSI PUMP A IN TEST OR MAINTENANCE

TOTAL: 1.38E-05 58.79% of CM Total Frequency 2.35E-05 (INTERNAL EVENTS)

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
GMM1HCVCCF	6.54E-02	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
REPS1CASE6	5.63E-02	OFF-SITE POWER NON-RECOVERY CASE 6:CCF OF DIESELS TO RUN
CMM1AVCCCF	5.41E-02	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE
GMM1SMVCCF	4.92E-02	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
EMM14CCFTR	4.72E-02	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO RUN
GMM1MPACCF	4.04E-02	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
EMM11BEDG	4.03E-02	EDG 1B FAILS TO RUN (24 HOUR EXPOSURE)
EMM11AEDG	4.02E-02	EDG 1A FAILS TO RUN (24 HOUR EXPOSURE)
AMM1PCFTS	3.28E-02	AFW PUMP 1C FAILS TO START
RTOP1S1RCP	3.10E-02	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
GTM1PUMPA	3.07E-02	HPSI PUMP A IN TEST OR MAINTENANCE
REPS1CASE7	2.92E-02	OFF-SITE POWER RECOVERY CASE 7: 1 DIESEL FAILS TO START
REPS1CASE5	2.90E-02	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
GTM1PUMPB	2.88E-02	HPSI PUMP B IN TEST OR MAINTENANCE
GMM1MRMOV	2.78E-02	MINIMUM RECIRC LINE MOTOR VALVES TRANSFER CLOSED
GMM1ASUMP	2.66E-02	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
GMM1BSUMP	2.63E-02	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
GMM1PAFTS	2.61E-02	FAILURE OF HPSI PUMP A TO START
NHFL1RWLCF	2.60E-02	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS
REPS1XTIE	2.51E-02	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2
E2XTIETOU1	2.44E-02	FAILURE OF THE UNIT 2 ELECTRICAL SYSTEM TO SUPPLY UNIT 1
EMM14CCFTS	2.41E-02	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO START
GMM1PBFTS	2.40E-02	FAILURE OF HPSI PUMP B TO START
CTM1CCWHXA	2.25E-02	CCW HX A IN TEST OR MAINTENANCE
CTM1CCWHXB	2.22E-02	CCW HX B IN TEST OR MAINTENANCE
RTOP1TOTC	2.19E-02	OPERATOR FAILS TO DO BLEED & FEED (ONCE-THROUGH) COOLING
REPS1CASE3	2.00E-02	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FTS (OR T&M) OTHER DIESEL FTR
ETM11AEDG	1.97E-02	EDG 1A IN TEST OR MAINTENANCE
QMM1MVCCCF	1.90E-02	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
ATM1AFWP1C	1.84E-02	AFW PUMP 1C TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
ZZ1ABKSHUT	1.83E-02	'A' BLK VLV CLOSE W/POWER
NMM1CEDM	1.81E-02	MECHANICAL FAULT PREVENTING ROD INSERTION
GMVK13656S	1.80E-02	MOTOR-OPERATED VALVE V3656 TRANSFERS CLOSED DURING STANDBY
AMM1SGBP1B	1.75E-02	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS
GMVK13654S	1.66E-02	MOTOR-OPERATED VALVE V3654 TRANSFERS CLOSED DURING STANDBY
ETM11BEDG	1.64E-02	1B EDG IN TEST OR MAINTENANCE
GHFL1PUMPA	1.63E-02	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE
ZZ1BBKSHUT	1.60E-02	'B' BLK VLV CLOSED W/POWER
AMM1HDMOV	1.55E-02	COMMON CAUSE FAILURE OF AFW AC REGULATING VALVES
ETM1BSU	1.54E-02	1B STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
EMM1ADGFTS	1.51E-02	EDG 1A FAILS TO START
GHFL1PUMPB	1.50E-02	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE
AMM1SGAP1A	1.49E-02	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS
ZZBLKV1403	1.49E-02	PORV 1402 BLOCK VALVE OPEN (MOV 1403 UNIT 1)
EMM1BDGFTS	1.41E-02	EDG 1B FAILS TO START
BMM11A1	1.37E-02	SIT 1A1 INJECTION PATH FAILS
BMM11A2	1.37E-02	SIT 1A2 INJECTION PATH FAILS
BMM11B1	1.37E-02	SIT 1B1 INJECTION PATH FAILS
BMM11B2	1.37E-02	SIT 1B2 INJECTION PATH FAILS
OMM1PORVA	1.34E-02	INDEPENDENT FAILURES OF PORV TRAIN A
ZZBLKV1405	1.33E-02	PORV 1404 BLOCK VALVE OPEN (MOV 1405, UNIT 1)
GMM1FTRCFI	1.21E-02	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
OMM1PORVB	1.19E-02	INDEPENDENT FAILURES OF PORV TRAIN B
ETM1ASU	1.14E-02	1A STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
EMM1CCFDGR	1.12E-02	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO RUN FOR 24 HOURS
GMM1CFTRIS	1.06E-02	CCF OF HPSI PUMPS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
RHVA1ELEQ	1.05E-02	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
ECBD120302	9.98E-03	AC BREAKER 20302 FAILS TO CLOSE (B SU)
AMM1PBFTS	9.79E-03	AFW PUMP 1B FAILS TO START
AMM1PAFTS	9.54E-03	AFW PUMP 1A FAILS TO START
GMM1CFTRRS	9.48E-03	CCF OF HPSI PUMPS TO RUN DURING REC FOLLOWING SMALL-SMALL LOCA
QMM1-21-3	9.11E-03	MV-21-3 FAILS TO CLOSE WITH SI
QMM1-21-2	9.10E-03	MV-21-2 FAILS TO CLOSE WITH SI
REPS1CASE1	8.74E-03	OFF-SITE POWER RECOVERY CASE 1: BOTH DIESELS FAIL TO START
GTKJ1RWT	7.86E-03	REFUELING WATER TANK RUPTURE
RTOP1ROTC	7.47E-03	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR
R#PCSIAS	7.42E-03	OPERATOR FAILS TO RECOVER PCS FOLLOWING SIAS
CMM1PAFTR	7.39E-03	CCW PUMP A FAILS TO RUN
REPS1CASE4	7.38E-03	OFF-SITE POWER NON-RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN
QMM1PAFTR	7.37E-03	ICW PUMP A FAILS TO RUN
AMM1STTVLV	7.36E-03	MODULAR EVENT FOR AFW TURBINE PUMP TRIP AND THROTTLE VALVE
ATM1AFWP1B	7.29E-03	AFW PUMP 1B TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
CMM1PBFTR	7.27E-03	CCW PUMP B FAILS TO RUN
QMM1PBFTR	7.25E-03	ICW PUMP B FAILS TO RUN
ATM1AFWP1A	7.22E-03	AFW PUMP 1A TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
GMM1ARWT	7.04E-03	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM RWT
REPS1CASE9	7.04E-03	OFF-SITE POWER RECOVERY CASE 9:CCF EDG TO START AND AFW PP FTS OR T/M
ECBD120102	6.83E-03	AC BREAKER 20102 FAILS ON DEMAND (A AUX)
AMM1PBFTR	6.64E-03	AFW PUMP 1B FAILS TO RUN
AHFL109124	6.61E-03	AFW PUMP 1B MANUAL VALVE V09124 MISPOSITIONED
AMM1PAFTR	6.54E-03	AFW PUMP 1A FAILS TO RUN
R#DCAB	6.47E-03	OPERATOR FAILS TO REALIGN 'AB' DC BUS
ECBD120411	6.44E-03	AC BREAKER 20411 FAILS TO OPEN (1B3 FROM 1B2)
CMM1MPFCCF	6.42E-03	CCW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
QMM1MPFCCF	6.42E-03	ICW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
AHFL109108	6.34E-03	AFW PUMP 1A MANUAL VALVE V09108 MISPOSITIONED

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
GMM1BRWT	6.31E-03	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM RWT
JMM1PAFTRI	6.23E-03	FAILURE OF LPSI PUMP A TO RUN DURING INJECTION
JMM1PBFTRI	6.23E-03	FAILURE OF LPSI PUMP B TO RUN DURING INJECTION
ECBD120209	6.15E-03	AC BREAKER 20209 FAILS TO OPEN (1A3 FROM 1A2)
JMM1HCVCCF	6.13E-03	COMMON CAUSE FAILURE OF LPSI INJECTION VALVES TO OPEN DURING
JMVK13206S	5.84E-03	MOTOR-OPERATED VALVE V3206 TRANSFERS CLOSED DURING STANDBY
JMVK13207S	5.84E-03	MOTOR-OPERATED VALVE V3207 TRANSFERS CLOSED DURING STANDBY
REPS1CASE8	5.72E-03	OFF-SITE POWER RECOVERY CASE 8: 1 DIESEL FAILS TO RUN
JMVR13-1AS	5.21E-03	MOTOR-OPERATED VALVE MV-03-1A TRANSFERS OPEN DURING STANDBY
JMVR13-1BS	5.21E-03	MOTOR-OPERATED VALVE MV-03-1B TRANSFERS OPEN DURING STANDBY
EMM1CCFDGS	5.18E-03	COMMON CAUSE FAILURE OF EDG'S 1A AND 1B TO START
MMM1RWT	5.11E-03	LOCAL FAULTS IN RWT LINE
OMM1V1404O	4.74E-03	LOCAL FAILURES PREVENTING OPERATION OF PORV TRAIN B
OMM1V1402O	4.73E-03	LOCAL FAILURES PREVENTING OPERATION OF PORV TRAIN A
REPS1CAS11	4.38E-03	OFF-SITE POWER RECOVERY CASE 11
AMM1PCFTR	3.94E-03	AFW PUMP 1C FAILS TO RUN
BHFL1LVL	3.73E-03	CCF OF SITs DUE TO MISCALIBRATION OF SIT LEVEL SENSORS
BHFL1PRS	3.73E-03	CCF OF SITs DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS
RAFW1CST	3.60E-03	OPERATOR FAILS TO SWITCH SUCTION OF AFW PUMPS TO UNIT 2 CST
GMM1FTRCFR	3.59E-03	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING RECIRCULATION
AHFL109140	3.54E-03	AFW PUMP 1C MANUAL VALVE V09140 MISPOSITIONED
EHFL1EDG1A	3.53E-03	OPERATOR FAILS TO PROPERLY ALIGN EDG FOLLOWING MAINTENANCE
EHFL1EDG1B	3.43E-03	FAILURE TO PROPERLY ALIGN SYSTEM FOLLOWING MAINTENANCE
ECBD1XTIE	3.33E-03	UNIT 1 BLACKOUT CROSSTIE BREAKER FAILS ON DEMAND
REPS1CA1A2	3.20E-03	OFFSITE POWER RECOVERY WITHIN 2 HRS
ZZ1BBLKRO	3.06E-03	'B' BLK VLV CLOSED W/O POWER
ZZ1ABLKRO	3.05E-03	'A' BLK VLV CLOSED W/O POWER
RCVC1RWT	3.05E-03	OPERATOR FAILS TO PROPERLY SWITCH SUCTION TO RWT
FMM1SGCVLV	2.99E-03	COMMON CAUSE FAILURE OF SG CHECK VALVES

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
JAVK13306S	2.97E-03	AIR-OPERATED VALVE FCV-3306 TRANSFERS CLOSE DURING STANDBY
JMVK103-2S	2.97E-03	MOTOR-OPERATED VALVE MV-03-2 TRANSFERS CLOSED DURING STANDBY
JTM1PUMPA	2.85E-03	LPSI PUMP A IN TEST OR MAINTENANCE
JTM1PUMPB	2.85E-03	LPSI PUMP B IN TEST OR MAINTENANCE
IMM1EQRM04	2.84E-03	LOCAL FAILURES OF SUPPLY FAN HVS 5B TO START
JMM1PAFTSI	2.83E-03	FAILURE OF LPSI PUMP A TO START DURING INJECTION
JMM1PBFTSI	2.83E-03	FAILURE OF LPSI PUMP B TO START DURING INJECTION
AMM1SGALCV	2.82E-03	MODULAR EVENT FOR NOT CLOSING MV-09-11
AMM1SGBLCV	2.82E-03	MODULAR EVENT FOR NOT CLOSING MV-09-12
ECBR140514	2.65E-03	AC BREAKER 40514 TRANSFERS OPEN
AMM1MPCSTV	2.53E-03	FAILURE OF AFW MOTOR PUMP COMMON SUCTION VALVES
JMM1MPACFI	2.46E-03	COMMON CAUSE FAILURE OF LPSI PUMPS TO START DURING INJECTION
GMM1AFTRRS	2.42E-03	HPSI PUMP 1A FAILS TO RUN DURING REC FOLLOWING SMALL-SMALL LOCA
GMM1BFTRRS	2.42E-03	HPSI PUMP 1B FAILS TO RUN DURING REC FOLLOWING SMALL-SMALL LOCA
ESVN1AFOFL	2.30E-03	EDG FILL VALVE FAILS TO OPEN
RPPC1-PORV	2.28E-03	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH
JMM1MPFCFI	2.26E-03	COMMON CAUSE FAILURE OF LPSI PUMPS TO RUN DURING INJECTION
ESVN1BFOFL	2.10E-03	EDG FILL VALVE FAILS TO OPEN
IMM1ECCS14	2.09E-03	MOTOR DAMPER D-9A UNAVAILABLE
IMM1ECCS15	2.09E-03	MOTOR DAMPER D-9B UNAVAILABLE
AMM1SGBP1C	2.06E-03	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FORM TURBINE PUMP
AMM1MPDEM	2.00E-03	DEMAND COMMON CAUSE FAILURE OF MOTOR DRIVEN PUMPS
APPJ18C56	1.98E-03	RUPTURE OF PUMP SUCTION LINE 8-C-56
ATKJ1CST	1.98E-03	TANK CST RUPTURES
RTOP1WBOR	1.96E-03	OPERATOR FAILS TO BORATE DURING ATWS
GMM1PAFTRI	1.94E-03	FAILURE OF HPSI PUMP A TO RUN DURING INJECTION
CMM1CCWHXA	1.94E-03	NO FLOW THROUGH CCW HX A
AMM1SGAP1C	1.90E-03	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM TURBINE PUMP

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
CMM1CCWHXB	1.89E-03	NO FLOW THROUGH CCW HX B
AMM1MPOPR	1.83E-03	OPERATING COMMON CAUSE FAILURE OF MOTOR DRIVEN PUMPS
R#AFXVLVS	1.83E-03	OPERATOR FAILS TO MANUALLY OPEN AFW X-TIE VLVS AND SG FLOW VALVES
ET1F11B2LC	1.81E-03	1B2 LC TRANSFORMER FAULT
GMM1PBFTRI	1.77E-03	FAILURE OF HPSI PUMP B TO RUN DURING INJECTION
GMM1AFTRIS	1.76E-03	HPSI PUMP 1A FAILS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
JHFL1PUMPA	1.75E-03	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE
JHFL1PUMPB	1.75E-03	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE
AMM1HDDCV	1.74E-03	COMMON CAUSE FAILURE OF AFW DC REGULATING VALVES
OMVN1-1403	1.73E-03	MOTOR-OPERATED VALVE 1403 FAILS TO OPEN
CPPJ1HDRA	1.72E-03	PIPING RUPTURE IN CCW HDR A
CPPJ1HDRB	1.72E-03	PIPING RUPTURE IN CCW HDR B
CPPJ1HDRN	1.72E-03	PIPING RUPTURE IN CCW HDR N
ECBR140503	1.68E-03	AC BREAKER 40503 TRANSFERS OPEN
ECBR120402	1.68E-03	AC BREAKER 20402 TRANSFERS OPEN
GMM1BFTRIS	1.55E-03	HPSI PUMP 1B FAILS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
GPPJ1HPSII	1.55E-03	PIPE RUPTURE OF HPSI COMMON HEADER DURING INJECTION
GTM107-2A	1.48E-03	MV-07-2A TEST AND MAINTENANCE
GTM107-2B	1.44E-03	MV-07-2B TEST AND MAINTENANCE
NLCD1AF1D	1.41E-03	LOGIC CIRCUIT AF1D FAILS TO GENERATE SIGNAL
QMM1PAFTS	1.36E-03	ICW PUMP A FAILS TO START
GCVN13217	1.35E-03	CHECK VALVE V3217 FAILS TO OPEN
GCVN13227	1.35E-03	CHECK VALVE V3227 FAILS TO OPEN
GCVN13237	1.35E-03	CHECK VALVE V3237 FAILS TO OPEN
GCVN13247	1.35E-03	CHECK VALVE V3247 FAILS TO OPEN
NLCD1AF2D	1.34E-03	LOGIC CIRCUIT AF2D FAILS TO GENERATE SIGNAL
AMM1AFWPCV	1.33E-03	COMMON CAUSE FAILURE OF AFW PUMP DISCHARGE CHECK VALVES
AMM1CSTCV	1.33E-03	COMMON CAUSE FAILURE OF CST DISCHARGE CHECK VALVES
AMM1HDCVLV	1.32E-03	COMMON CAUSE FAILURE OF AFW HEADER CHECK VALVES

Table 3.7-9 St. Lucie Unit 1 Basic Event Importance

Basic Event	F-V Importance	Description
AMM1CST-XT	1.30E-03	CST CROSSTIE FAILS TO PERMIT FLOW
NLCD1RPS	1.24E-03	LOGIC CIRCUIT RPS FAILS TO GENERATE SIGNAL
RAFW1SGTR	1.23E-03	OPERATOR FAILS TO REALIGN AFW AND ISOLATE THE FAULTED SG FOLLOWING LOSS OF OFFSITE POWER
EREE186GP	1.22E-03	LOCKOUT RELAY 86GP FAILS TO ENERGIZE
CMM1PAFTS	1.20E-03	CCW PUMP A FAILS TO START
GMM1CVNCCF	1.20E-03	COMMON CAUSE FAILURE OF SIS LINE CHECK VALVES TO OPEN
GMM1RCVCCF	1.20E-03	COMMON CAUSE FAILURE OF RWT OUTLET CHECK VALVES TO OPEN
ET1F11A2LC	1.20E-03	1A2 LC TRANSFORMER FAULT
EMM1DC-AB	1.13E-03	FAILURE OF 125 VDC FEEDER BREAKERS TO OPERATE DURING REALIGNMENT
QMM1PBFTS	1.12E-03	ICW PUMP B FAILS TO START
ECBR140214	1.11E-03	AC BREAKER 40214 TRANSFERS OPEN
ECBR120210	1.10E-03	AC BREAKER 20210 TRANSFERS OPEN
GMM1INJCCF	1.09E-03	COMMON CAUSE FAILURE OF HPSI INJECTION CHECK VALVES TO OPEN
GMM1PCVCCF	1.09E-03	COMMON CAUSE FAILURE OF HPSI PUMP DISCHARGE CV TO OPEN
MMM1V2501	1.09E-03	MOV V2501 FAILS TO CLOSE INDEPENDENT FAILURES
OMVN1-1405	1.08E-03	MOTOR-OPERATED VALVE 1405 FAILS TO OPEN
ECBR140203	1.07E-03	AC BREAKER 40203 TRANSFERS OPEN
GPPJ1HPSIR	1.07E-03	PIPE RUPTURE OF HPSI COMMON HEADER DURING RECIRCULATION
GMM1PAFTRR	1.03E-03	FAILURE OF HPSI PUMP A TO RUN DURING RECIRCULATION

Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
NMM2CEDM	7.39E-02	MECHANICAL FAULT PREVENTING ROD INSERTION
ZZ2ABKSHUT	6.04E-02	'A' BLK VLV CLOSED W/POWER AVAILABLE
GMM2HCVCCF	5.97E-02	COMMON CAUSE FAILURE OF HPSI INJECTION VALVES TO OPEN
ZZ2BBLKROR	5.65E-02	'B' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
REPS2CASE6	5.10E-02	OFF-SITE POWER RECOVERY CASE 6: CCF OF DIESELS TO RUN
CMM2AVCCCF	4.88E-02	N-HEADER AIR OPERATED ISOLATION VALVES FAIL TO CLOSE DUE TO COMMON CAUSE FAILURE
EMM22BEDG	4.86E-02	EDG 2B FAILS TO RUN
EMM22AEDG	4.74E-02	EDG 2A FAILS TO RUN
RHVA2ELEQ	4.67E-02	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF OFFSITE POWER
GMM2SMVCCF	4.50E-02	COMMON CAUSE FAILURE OF SUMP OUTLET MOTOR VALVES TO OPEN
ZZ2BBLKRO	4.38E-02	'B' BLK VLV CLOSED W/O POWER
EMM24CCFTR	4.30E-02	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO RUN
REPS2CA1A2	4.15E-02	OFFSITE POWER RECOVERY: CASE 1 WITH 2 HR RECOVERY TIME
GMM2MPACCF	3.69E-02	COMMON CAUSE FAILURE OF HPSI PUMPS TO START
GTM2PUMPA	3.21E-02	HPSI PUMP A IN TEST OR MAINTENANCE
NHFL2RWLCF	3.20E-02	COMMON CAUSE MISCALIBRATION OF RWT LEVEL TRANSMITTERS
ZZ2ABLKRO	3.11E-02	'A' BLK VLV CLOSED W/O POWER
RTOP2S1RCP	2.92E-02	OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
GMM2ASUMP	2.91E-02	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM SUMP
GTM2PUMPB	2.81E-02	HPSI PUMP B IN TEST OR MAINTENANCE
GMM2PAFTS	2.75E-02	FAILURE OF HPSI PUMP A TO START
REPS2CASE5	2.63E-02	OFF-SITE POWER RECOVERY CASE 5: CCF OF DIESELS TO START
GMM2BSUMP	2.63E-02	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM SUMP
GMM2PBFTS	2.52E-02	FAILURE OF HPSI PUMP B TO START
AMM2PCFTS	2.45E-02	AFW PUMP 2C FAILS TO START
CTM2CCWHXA	2.45E-02	CCW HX A IN TEST OR MAINTENANCE
AMM2SGBP2B	2.42E-02	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS
REPS2XTIE	2.30E-02	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1
AMM2SGAP2A	2.22E-02	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM MTR PUMPS
EMM24CCFTS	2.19E-02	COMMON CAUSE FAILURE OF UNIT 1 AND UNIT 2 DIESELS TO START
E1XTIETOU2	2.17E-02	FAILURE OF THE UNIT 1 ELECTRICAL SYSTEM TO SUPPLY UNIT 2
CTM2CCWHXB	2.13E-02	CCW HX B IN TEST OR MAINTENANCE
REPS2CASE7	2.11E-02	OFF-SITE POWER RECOVERY CASE 7: 1 DIESEL FAILS TO START
ETM22BEDG	2.07E-02	2B EDG UNAVAILABLE DUE TO MAINTENANCE

Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
ETM22AEDG	1.94E-02	EDG 2A UNAVAILABLE DUE TO MAINTENANCE
REPS2CASE3	1.82E-02	OFF-SITE POWER RECOVERY CASE 3: 1 DIESEL FAILS TO START/1 DIESEL FAILS TO RUN
GMVK23656S	1.75E-02	MOTOR-OPERATED VALVE V3656 TRANSFERS CLOSED DURING STANDBY
RTOP2TOTC	1.74E-02	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING (TRANSIENT)
GHFL2PUMPA	1.72E-02	OPERATOR FAILS TO RESTORE PUMP A FOLLOWING MAINTENANCE
QMM2MVCCCF	1.71E-02	ICW MOTOR OPERATED VALVES FAIL TO CLOSE DUE TO COMMON CAUSE
GMVK23654S	1.60E-02	MOTOR-OPERATED VALVE V3654 TRANSFERS CLOSED DURING STANDBY
GHFL2PUMPB	1.58E-02	OPERATOR FAILS TO RESTORE PUMP B FOLLOWING MAINTENANCE
EMM2BDGFTS	1.55E-02	EDG 2B FAILS TO START
EMM2ADGFTS	1.50E-02	EDG 2A FAILS TO START
ATM2AFWP2C	1.38E-02	AFW PUMP 2C TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
ETM2ASU	1.38E-02	2A STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
RPPC2BLPWR	1.35E-02	OPERATOR FAILS TO RESTORE POWER TO BLOCK VALVE
ZZBLKV1477	1.35E-02	BLOCK VALVE OPEN
ETM2BSU	1.34E-02	2B STARTUP TRANSFORMER UNAVAILABLE DUE TO MAINTENANCE
BMM22A1	1.24E-02	SIT 2A1 INJECTION PATH FAILS
BMM22A2	1.24E-02	SIT 2A2 INJECTION PATH FAILS
BMM22B1	1.24E-02	SIT 2B1 INJECTION PATH FAILS
BMM22B2	1.24E-02	SIT 2B2 INJECTION PATH FAILS
AMM2HDMOV	1.24E-02	COMMON CAUSE FAILURE OF AFW AC REGULATING VALVES
OMM2PORVB	1.21E-02	INDEPENDENT FAILURES OF PORV TRAIN B
AMM2PBFTS	1.19E-02	AFW PUMP 2B FAILS TO START
ZZ2ABLKROR	1.19E-02	'A' BLOCK VALVE CLOSED WITH POWER REMOVED BUT RECOVERABLE
ZZHLR-INJ	1.10E-02	PROBABILITY THAT CORE COOLING DURING INJECTION WILL BE LOST
GMM2FTRCFI	1.09E-02	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING INJECTION
QMM2-21-3	1.04E-02	MV-21-3 FAILS TO CLOSE WITH SI
AMM2PAFTS	9.86E-03	AFW PUMP 2A FAILS TO START
QMM2-21-2	9.73E-03	MV-21-2 FAILS TO CLOSE WITH SI
CMVC214-17	9.60E-03	MOTOR-OPERATED VALVE MV-14-17 FAILS TO CLOSE FOLLOWING SI
GMM2CFTRIS	9.55E-03	CCF OF HPSI PUMPS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
EMM2CCFDGR	9.17E-03	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO RUN
ECBD220102	8.78E-03	AC BREAKER 20102 FAILS TO CLOSE
REPS2CASE8	8.75E-03	OFF-SITE POWER RECOVERY CASE 8: 1 DIESEL FAILS TO RUN

Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
GMM2CFTRRS	8.67E-03	CCF OF HPSI PUMPS TO RUN DURING RECIRC FOLLOWING SMALL-SMALL LOCA
ECBD220302	8.58E-03	AC BREAKER 20302 FAILS TO CLOSE
CMM2PAFTR	8.42E-03	CCW PUMP A FAILS TO RUN
QMM2PAFTR	8.34E-03	ICW PUMP A FAILS TO RUN
GMM2ARWT	8.25E-03	LOCAL FAULTS OF ECCS PUMP A SUCTION LINE FROM RWT
REPS2CASE1	8.12E-03	OFF-SITE POWER RECOVERY CASE 1: BOTH DIESELS FAIL TO START
GMM2FTRCFR	8.09E-03	COMMON CAUSE FAILURE OF HPSI PUMPS TO RUN DURING RECIRCULATION
GMM2BRWT	7.89E-03	LOCAL FAULTS OF ECCS PUMP B SUCTION LINE FROM RWT
CMM2PBFTR	7.68E-03	CCW PUMP B FAILS TO RUN
QMM2PBFTR	7.59E-03	ICW PUMP B FAILS TO RUN
GTKJ2RWT	6.99E-03	REFUELING WATER TANK RUPTURE
REPS2CASE9	6.90E-03	OFF-SITE POWER RECOVERY CASE 9 - DIESEL CCF TO START
ECBD220411	6.90E-03	AC BREAKER 20411 FAILS TO OPEN
REPS2CASE4	6.71E-03	OFF-SITE POWER RECOVERY CASE 4: BOTH DIESELS FAIL TO RUN
AMM2SGAP2C	6.58E-03	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM TURBINE PUMP
GMM2MNRCCA	6.51E-03	LOCAL FAILURES OF COMMON ECCS PUMP TRAIN A MINIMUM RECIRC VALVE
ECBD220209	6.21E-03	AC BREAKER 20209 FAILS TO OPEN
GMM2MNRCCB	6.17E-03	LOCAL FAILURES OF COMMON ECCS PUMP TRAIN B MINIMUM RECIRC VALVE
CMM2MPFCCF	6.16E-03	CCW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
QMM2MPFCCF	6.16E-03	ICW PUMP FAILS TO RUN DUE TO COMMON CAUSE FAILURE
JMM2HCVCCF	5.57E-03	COMMON CAUSE FAILURE OF LPSI INJECTION VALVES TO OPEN DURING
GMVR23523	5.52E-03	MOTOR-OPERATED VALVE V3523 TRANSFERS OPEN DURING STANDBY
GMVR23540	5.52E-03	MOTOR-OPERATED VALVE 3540 TRANSFERS OPEN DURING STANDBY
GMVR23550	5.52E-03	MOTOR-OPERATED VALVE V3550 TRANSFERS OPEN
GMVR23551	5.52E-03	MOTOR-OPERATED VALVE V3551 TRANSFERS OPEN
MMM2RWT	4.94E-03	LOCAL FAULTS IN RWT LINE
ATM2AFWP2B	4.81E-03	AFW PUMP 2B TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
JMVK23301S	4.77E-03	MOTOR-OPERATED VALVE FCV-3301 TRANSFERS CLOSED DURING STANDBY
JMVK23306S	4.77E-03	MOTOR-OPERATED VALVE FCV-3306 TRANSFERS CLOSED DURING STANDBY
IMM2EERCCF	4.70E-03	COMMON CAUSE FAILURE OF ELECT EQUIP ROOM FANS TO RUN
GMM2RCVCCF	4.59E-03	COMMON CAUSE FAILURE OF RWT OUTLET CHECK VALVES TO OPEN
EMM2CCFDGS	4.54E-03	COMMON CAUSE FAILURE OF EDG'S 2A AND 2B TO START

Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
JMVR23536S	4.27E-03	MOTOR-OPERATED VALVE V3536 TRANSFERS OPEN DURING STANDBY
JMVR23539S	4.27E-03	MOTOR-OPERATED VALVE V3539 TRANSFERS OPEN DURING STANDBY
AMM2PBFTR	4.15E-03	AFW PUMP 2B FAILS TO RUN
ATM2AFWP2A	4.10E-03	AFW PUMP 2A TRAIN UNAVAILABLE DUE TO TEST/MAINTENANCE
AMM2SGBP2C	4.10E-03	MODULAR EVENT FOR HEADER VALVES IN FLOW-PATH FROM TURBINE PU
AHFL209124	3.96E-03	AFW PUMP 2B MANUAL VALVE V09124 MISPOSITIONED
ECBR240203	3.64E-03	AC BREAKER 40203 TRANSFERS OPEN
BHFL2LVL	3.39E-03	CCF OF SITs DUE TO MISCALIBRATION OF SIT LEVEL SENSORS
BHFL2PRS	3.39E-03	CCF OF SITs DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS
EHFL2EDG2B	3.38E-03	FAILURE TO PROPERLY ALIGN SYSTEM FOLLOWING MAINTENANCE
AHFL209108	3.28E-03	AFW PUMP 2A MANUAL VALVE V09108 MISPOSITIONED
AMM2PAFTR	3.26E-03	AFW PUMP 2A FAILS TO RUN
EHFL2EDG2A	3.14E-03	FAILURE TO PROPERLY ALIGN FOLLOWING MAINTENANCE
ECBR240520	3.09E-03	AC BREAKER 40520 TRANSFERS OPEN
ECBD2XTIE	3.06E-03	UNIT 2 BLACKOUT CROSSTIE BREAKER FAILS ON DEMAND
RCVC2RWT	2.80E-03	OPERATOR FAILS TO PROPERLY SWITCH SUCTION TO RWT
GMM2AFTRRS	2.75E-03	HPSI PUMP 2A FAILS TO RUN DURING RECIRC FOLLOWING SMALL-SMALL LOCA
GMM2BFTRRS	2.57E-03	HPSI PUMP 2B FAILS TO RUN DURING RECIRC FOLLOWING SMALL-SMALL LOCA
AMM2PCFTR	2.50E-03	AFW PUMP 2C FAILS TO RUN
CMM2CCWHXA	2.34E-03	NO FLOW THROUGH CCW HX A
JMM2PAFTSI	2.33E-03	FAILURE OF LPSI PUMP A TO START DURING INJECTION
JMM2PBFTSI	2.33E-03	FAILURE OF LPSI PUMP B TO START DURING INJECTION
JTM2PUMPA	2.28E-03	LPSI PUMP A IN TEST OR MAINTENANCE
JTM2PUMPB	2.28E-03	LPSI PUMP B IN TEST OR MAINTENANCE
JMM2MPACFI	2.24E-03	COMMON CAUSE FAILURE OF LPSI PUMPS TO START DURING INJECTION
CMM2CCWHXB	2.20E-03	NO FLOW THROUGH CCW HX B
GMM2PAFTRR	2.14E-03	FAILURE OF HPSI PUMP A TO RUN DURING RECIRCULATION
FMM2SGCVLV	2.12E-03	COMMON CAUSE FAILURE OF SG CHECK VALVES
JMM2MPFCFI	2.05E-03	COMMON CAUSE FAILURE OF LPSI PUMPS TO RUN DURING INJECTION
GMM2PBFTRR	2.01E-03	FAILURE OF HPSI PUMP B TO RUN DURING RECIRCULATION
ESVN2A1FL	1.98E-03	EDG FUEL FILL VALVE 2A1 FAILS TO OPEN
ESVN2A2FL	1.98E-03	EDG FUEL FILL VALVE 2A2 FAILS TO OPEN
ET1F22A2LC	1.98E-03	2A2 LC TRANSFORMER FAULT
GTM207-2A	1.86E-03	MV-07-2A TEST AND MAINTENANCE
AHFL209140	1.85E-03	AFW PUMP 2B MANUAL VALVE V09140 MISPOSITIONED

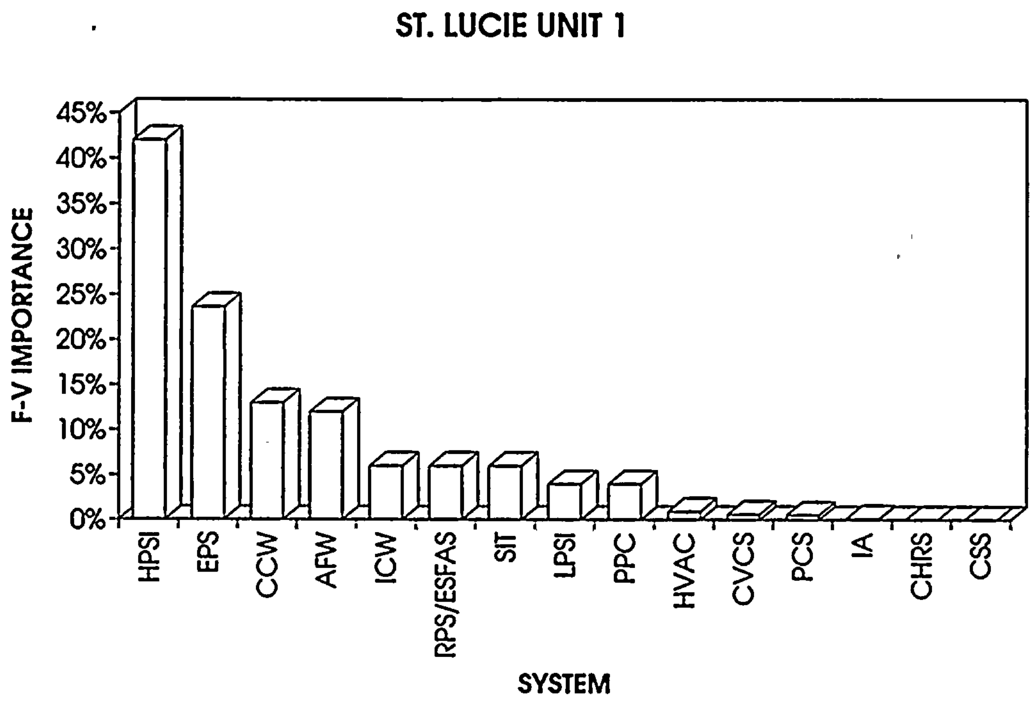
Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
ESVN2B1FL	1.81E-03	EDG FUEL FILL VALVE 2B1 FAILS TO OPEN
ESVN2B2FL	1.81E-03	EDG FUEL FILL VALVE 2B2 FAILS TO OPEN
RTOP2WBOR	1.79E-03	OPERATOR FAILS TO BORATE DURING ATWS
ECBR220213	1.77E-03	AC BREAKER 20213 TRANSFERS OPEN
ECBR240219	1.77E-03	AC BREAKER 40219 TRANSFERS OPEN
GTM207-2B	1.72E-03	MV-07-2B TEST AND MAINTENANCE
GMM2AFTRIS	1.71E-03	HPSI PUMP 2A FAILS TI RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
CXVK214126	1.69E-03	MANUAL VALVE 14126 TRANSFERS CLOSED
GMM2BFTRIS	1.67E-03	HPSI PUMP 2B FAILS TO RUN DURING INJ FOLLOWING SMALL-SMALL LOCA
CXVK214127	1.58E-03	MANUAL VALVE 14127 TRANSFERS CLOSED
R#DGFO	1.58E-03	OPERATOR FAILS TO RECOVER EDG BY OPENING DG FILL VALVE BYPASS
GMM2PAFTRI	1.55E-03	FAILURE OF HPSI PUMP A TO RUN DURING INJECTION
GMM2PBFTRI	1.55E-03	FAILURE OF HPSI PUMP B TO RUN DURING INJECTION
APPJ28C56	1.43E-03	RUPTURE OF PUMP SUCTION LINE 8-C-56
ATKJ2CS	1.43E-03	TANK CST RUPTURES
JHFL2PUMPA	1.43E-03	OPERATOR FAILS TO RESTORE PUMP 2A FOLLOWING MAINTENANCE
JHFL2PUMPB	1.43E-03	OPERATOR FAILS TO RESTORE PUMP 2B FOLLOWING MAINTENANCE
GPPJ2HPSII	1.40E-03	PIPE RUPTURE OF HPSI COMMON HEADER DURING INJECTION
IMM2ECCS14	1.39E-03	MOTOR DAMPER D-9A UNAVAILABLE
IMM2ECCS15	1.39E-03	MOTOR DAMPER D-9B UNAVAILABLE
ET1F22B2LC	1.37E-03	2B2 LC TRANSFORMER FAULT
ECBR220402	1.27E-03	AC BREAKER 20402 TRANSFERS OPEN
AMM2ACSV	1.27E-03	COMMON CAUSE FAILURE OF AC SOLENOID VALVES
GCVN23217	1.23E-03	CHECK VALVE V3217 FAILS TO OPEN
GCVN23227	1.23E-03	CHECK VALVE V3227 FAILS TO OPEN
GCVN23237	1.23E-03	CHECK VALVE V3237 FAILS TO OPEN
GCVN23247	1.23E-03	CHECK VALVE V3247 FAILS TO OPEN
ECBR240503	1.22E-03	AC BREAKER 40503 TRANSFERS OPEN
GPPJ2HPSIR	1.20E-03	PIPE RUPTURE OF HPSI COMMON HEADER DURING RECIRCULATION
GMM2AMINRC	1.19E-03	LOCAL FAULTS OF HPSI PUMP A MIN RECIRC LINE
HHFL2STBYD	1.14E-03	OPERATOR FAILS TO PUT AIR COMPRESSOR 2D IN STANDBY
GMM2BMINRC	1.13E-03	LOCAL FAULTS OF HPSI PUMP B MIN RECIRC LINE
GTM207-1A	1.13E-03	MV-07-1A TEST AND MAINTENANCE
GTM207-1B	1.11E-03	MV-07-1B TEST AND MAINTENANCE
RPPC2-PORV	1.08E-03	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH
GMM2CCWPA	1.06E-03	LOCAL FAILURES OF CCW TO HPSI PUMP A

Table 3.7-10 St. Lucie Unit 2 Basic Event Importance

Basic Event	F-V Importance	Description
MMM2V2501	1.05E-03	MOV V2501 FAILS TO CLOSE INDEPENDENT FAILURES
GMM2CCWPB	1.01E-03	LOCAL FAILURES OF CCW TO HPSI PUMP B

Figure 3.7-5 System Importance



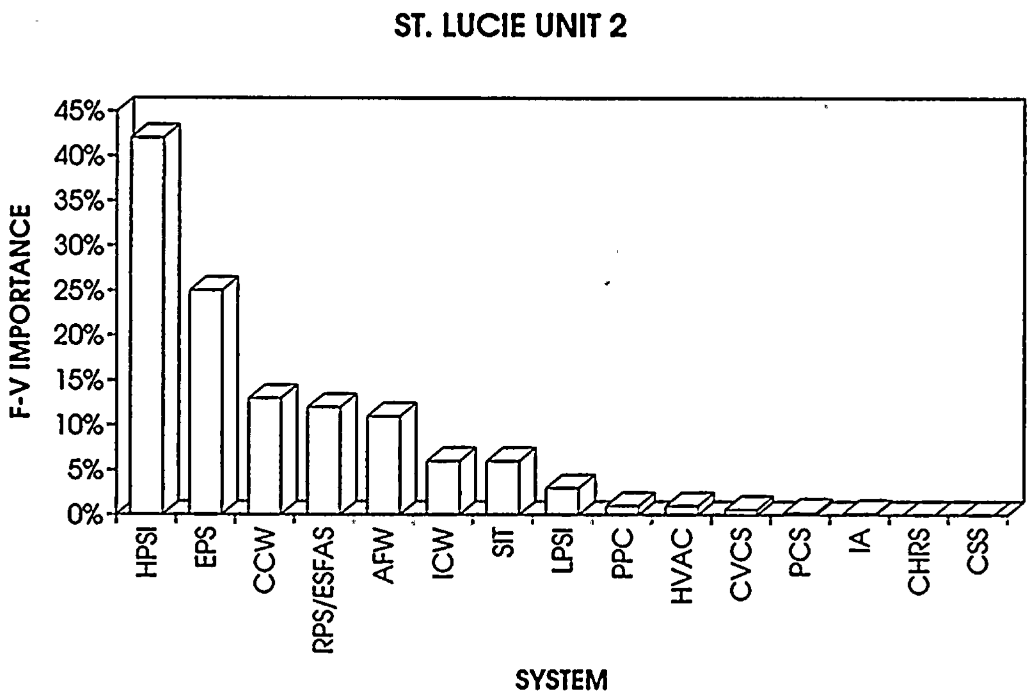


Table 3.7-11 Sensitivity Analysis Results Summary

CHANGE	UNIT 1		UNIT 2	
	CDF	% CHANGE	CDF	% CHANGE
BASELINE	2.32E-05	NA	2.62E-05	NA
INCREASED ALL MOV FAILURE PROBABILITIES (FAIL-TO-OPEN & FAIL-TO-CLOSE) BY A FACTOR OF 10	1.36E-04	486%	1.58E-04	503%
INCREASED ALL COMMON CAUSE FAILURE RATES BY A FACTOR OF 10	9.06E-05	291%	9.75E-05	272%
INCREASED THE DIESEL GENERATOR FAIL-TO-START AND FAIL-TO-RUN PROBABILITY BY A FACTOR OF 10	8.37E-05	261%	9.37E-05	258%
INCREASED ALL TEST AND MAINTENANCE UNAVAILABILITY PROBABILITIES BY A FACTOR OF 10	8.07E-05	248%	8.78E-05	235%
DECREASED ALL TEST AND MAINTENANCE UNAVAILABILITY PROBABILITIES BY A FACTOR OF 10	1.92E-05	-17%	2.20E-05	-16%
INCREASED ALL MOTOR DRIVEN PUMP FAIL-TO-START FAILURE RATE BY A FACTOR OF 10	3.86E-05	66%	4.14E-05	58%
INCREASED ALL MOTOR OPERATED PUMP FAIL-TO-RUN PROBABILITIES BY A FACTOR OF 10	4.68E-05	102%	5.26E-05	101%
INCREASED THE UNIT 1 PORV FLOW PATH UNAVAILABILITY TO A VALUE SIMILAR TO THAT USED FOR UNIT 2	7.57E-05	226%	NA	NA
DECREASED THE UNIT 2 PORV FLOW PATH UNAVAILABILITY TO A VALUE SIMILAR TO THAT USED FOR UNIT 1	NA	NA	2.37E-05	-10%
INCREASED ALL OFFSITE POWER NON-RECOVERY PROBABILITIES BY A FACTOR OF 10	4.98E-05	115%	6.32E-05	141%
INCREASED ALL OPERATOR NON-RECOVERY PROBABILITIES BY A FACTOR OF 10	4.56E-05	97%	5.79E-05	121%
INCREASED ALL PRE-INITIATOR HFE BY A FACTOR OF 10	4.42E-05	91%	4.79E-05	83%
ASSUMED FAILURE TO INITIATE HOT LEG RECIRCULATION FOLLOWING A LARGE LOCA IS A CORE DAMAGE SEQUENCE	2.98E-05	28%	2.82E-05	8%

Table 3.7-12 St. Lucie Unit 1 Uncertainty Analysis Summary

Input Options

Filename	: .\REPORTS
Module Name	: [TOP]
Sample Size	: 5000
Seed	: 9937817
Point Estimate	: 2.10E-05
Number of Modules	: 188
Total Cutsets In All Modules	: 5085
Number of Basic Events	: 979
Number of Type Codes	: 169
Inputs Missing Distribution	: 36

Moments
(With 95% Confidence)

	Low	Estimate	High
Mean	4.55E-06	2.21E-05	6.24E-05
Standard Deviation	3.68E-05	3.60E-05	3.54E-05
Skewness	-	1.05E+01	-
Kurtosis	-	1.73E+02	-

Percentiles
(With 95% Confidence)

	Low	Estimate	High
Minimum	-	1.92E-06	-
2.5	3.56E-06	3.75E-06	3.91E-06
5.0	4.38E-06	4.55E-06	4.70E-06
10.0	5.60E-06	5.77E-06	5.97E-06
20.0	7.34E-06	7.48E-06	7.64E-06
25.0	8.06E-06	8.29E-06	8.47E-06
30.0	8.99E-06	9.27E-06	9.51E-06
40.0	1.09E-05	1.12E-05	1.14E-05
50.0	1.32E-05	1.35E-05	1.39E-05
60.0	1.61E-05	1.64E-05	1.69E-05
70.0	1.97E-05	2.04E-05	2.11E-05
75.0	2.25E-05	2.32E-05	2.40E-05
80.0	2.62E-05	2.71E-05	2.82E-05
90.0	4.07E-05	4.28E-05	4.45E-05
95.0	5.90E-05	6.24E-05	6.75E-05
97.5	8.29E-05	8.75E-05	1.00E-04
Maximum	-	8.85E-04	-

Figure 3.7-6 St. Lucie Unit 1 Uncertainty Analysis Summary

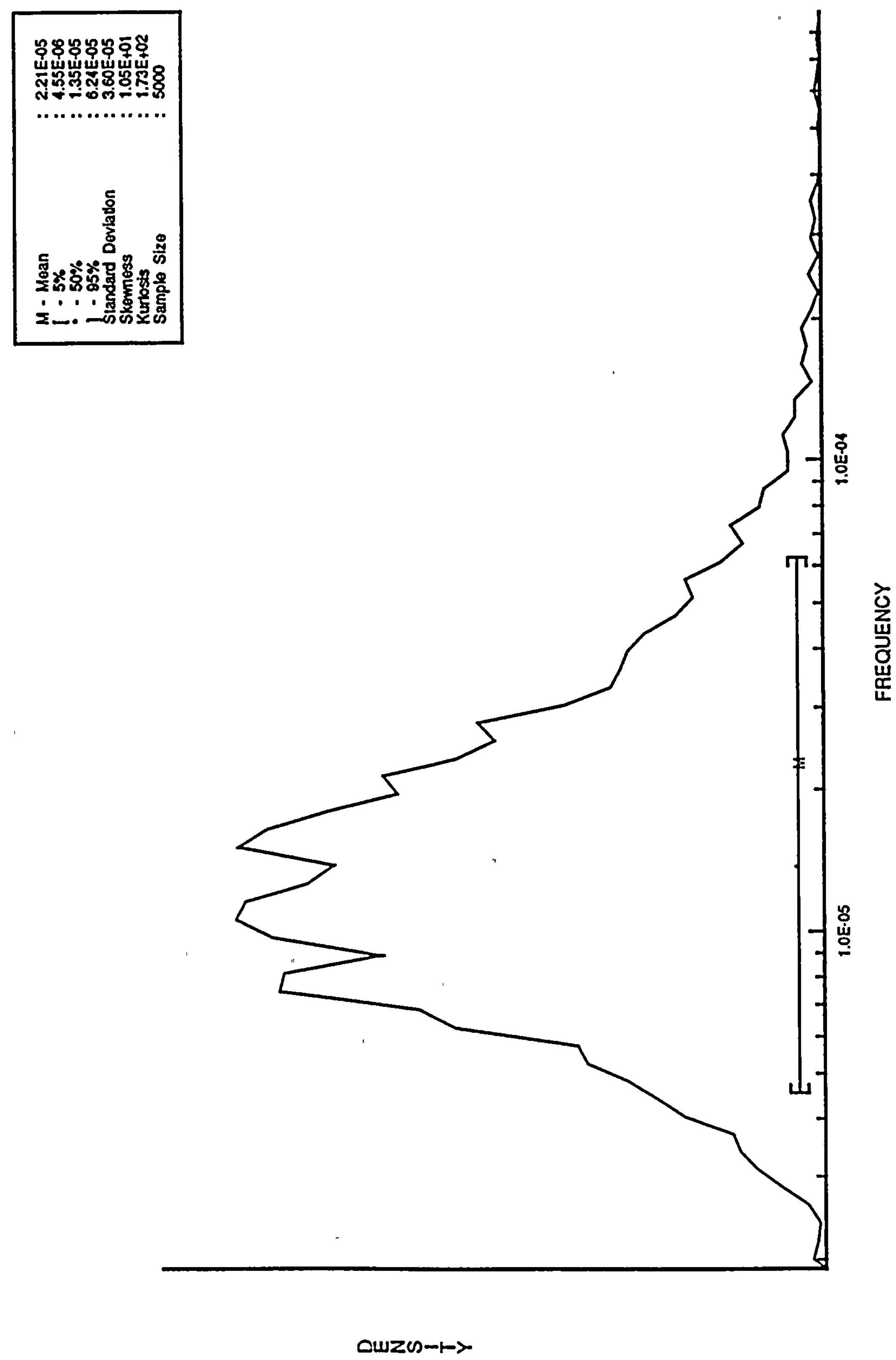


Table 3.7-13 St. Lucie Unit 2 Uncertainty Analysis Summary

Input Options

Filename	: .\REPORTS
Module Name	: [TOP]
Sample Size	: 5000
Seed	: 3960164
Point Estimate	: 2.30E-05
Number of Modules	: 146
Total Cutsets In All Modules	: 4181
Number of Basic Events	: 771
Number of Type Codes	: 148
Inputs Missing Distribution	: 31

Moments
(With 95% Confidence)

	Low	Estimate	High
Mean	4.96E-06	2.41E-05	7.16E-05
Standard Deviation	3.68E-05	3.61E-05	3.54E-05
Skewness	-	1.01E+01	-
Kurtosis	-	1.99E+02	-

Percentiles
(With 95% Confidence)

	Low	Estimate	High
Minimum	-	2.21E-06	-
2.5	4.00E-06	4.13E-06	4.34E-06
5.0	4.79E-06	4.95E-06	5.11E-06
10.0	5.92E-06	6.14E-06	6.29E-06
20.0	7.79E-06	8.00E-06	8.22E-06
25.0	8.77E-06	9.00E-06	9.28E-06
30.0	9.72E-06	9.96E-06	1.02E-05
40.0	1.18E-05	1.21E-05	1.24E-05
50.0	1.43E-05	1.46E-05	1.50E-05
60.0	1.72E-05	1.77E-05	1.82E-05
70.0	2.17E-05	2.25E-05	2.33E-05
75.0	2.51E-05	2.60E-05	2.69E-05
80.0	2.94E-05	3.07E-05	3.20E-05
90.0	4.59E-05	4.74E-05	5.00E-05
95.0	6.74E-05	7.16E-05	7.71E-05
97.5	9.51E-05	1.02E-04	1.15E-04
Maximum	-	1.09E-03	-

Figure 3.7-7 St. Lucie Unit 2 Uncertainty Analysis Summary

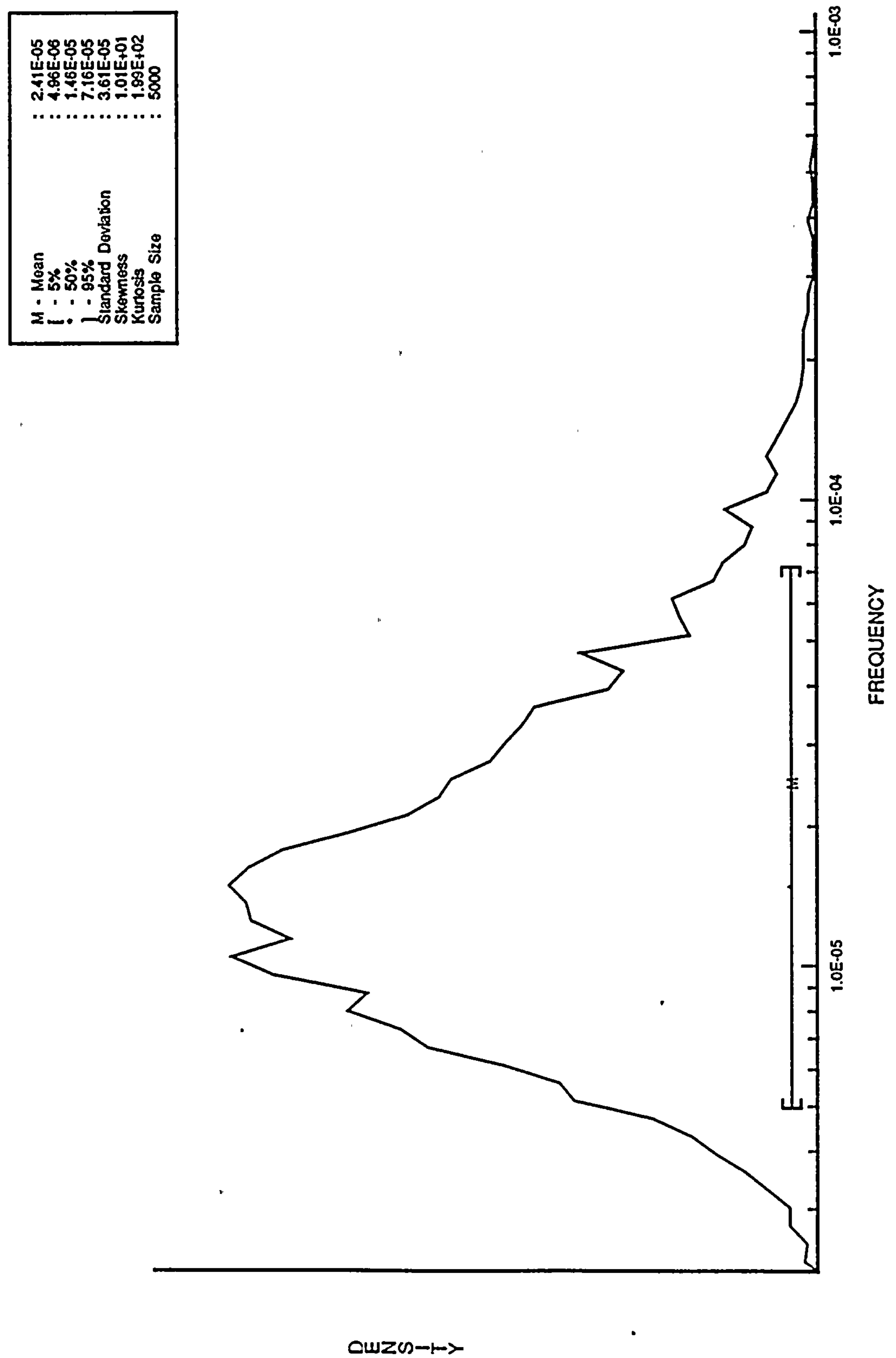


Table 3.7-14 Core Damage Sequence/Plant Damage State Matrix

	PLANT DAMAGE STATE									
	IB	IIA	IIB	IIE	IIF	3B	3H	VB	VIA	VIB
Core Damage Sequence	Small-Small LOCA (Early Core Cooling Failure)	Small-Small LOCA, (Late Core Cooling Failure)	Small-Small LOCA, (Long-term Core Cooling Failure)	Small-Small LOCA (Long-term Core Cooling Failure)	Transient Induced LOCA (Long-term Core Cooling Failure)	Transients (Secondary Heat Removal Failure)	Blackout	Small & Large LOCA (Early Core Cooling Failure)	Small & Large LOCA, (Long-term Core Cooling Failure)	Small & Large LOCA, (Long-term Core Cooling Failure)
Containment Status	ECC OK; CS injection available and activated	ECC OK; CS OK During Injection	ECC OK; CS injection available and activated	No ECC; CS injection only	No ECC; CS OK During Injection and Recirculation	ECC OK; CS OK	No ECC; No CS	ECC OK; CS OK During Injection and Recirculation	ECC OK; CS injection failure	ECC OK; CS OK during injection & recirculation

Table 3.7-15 St. Lucie Unit 1 Conditional Probability of PDS Sequences with PWR-4 Comparable Source Terms

PDS	Mean Freq. per year	Containment Event Tree End States																
		B2-L	B2-R	B4-L	B4-R	B6-L	B6-R	C2-L	C2-R	C4-L	C4-R	C6-L	C6-R	D2-R	D4-R	E2-R	E4-R	E6-R
IB	2.76E-6	4.06E-7	ε	1.45E-6	ε	ε	ε	8.19E-4	2.05E-6	3.19E-3	8.01E-6	1.83E-4	4.6E-7	4.02E-6	1.31E-6	ε	ε	ε
IIA	1.73E-6	1.50E-6	ε	5.35E-6	ε	3.65E-7	ε	9.60E-4	2.41E-6	4.72E-3	1.18E-5	2.68E-4	6.71E-7	4.40E-6	2.56E-6	ε	ε	ε
IIB	1.54E-6	ε	ε	ε	ε	1.69E-3	4.23E-6	ε	ε	ε	ε	8.57E-2	2.15E-4	ε	6.15E-4	ε	ε	ε
IIE	1.49E-6	8.39E-5	2.10E-7	5.26E-4	1.32E-6	2.11E-5	ε	7.25E-4	1.82E-6	5.46E-3	1.37E-5	1.83E-4	4.58E-7	3.91E-6	1.31E-6	ε	ε	ε
IIF	9.31E-7	1.98E-2	4.96E-5	1.36E-2	3.42E-5	4.34E-2	1.08E-4	2.56E-2	6.40E-5	1.78E-2	4.47E-5	4.73E-2	1.18E-4	8.64E-5	4.52E-4	ε	ε	ε
3B	3.93E-6	1.03E-6	ε	3.66E-6	ε	1.07E-7	ε	6.59E-4	1.65E-6	3.23E-3	8.09E-6	7.89E-5	1.97E-7	4.05E-6	5.65E-7	ε	ε	ε
3H	2.72E-6	7.10E-5	1.78E-7	6.54E-5	1.64E-7	ε	ε	4.15E-2	1.04E-4	4.45E-1	1.11E-3	1.02E-7	ε	6.25E-6	1.22E-6	2.65E-3	2.71E-2	5.15E-4
VB	2.99E-6	6.87E-5	1.72E-7	4.31E-4	1.08E-6	1.35E-4	3.37E-7	7.54E-4	1.90E-6	4.81E-3	1.21E-5	1.25E-3	3.14E-6	3.43E-6	8.99E-6	ε	ε	ε
VIA	7.01E-7	5.26E-5	1.32E-7	3.31E-4	8.30E-7	1.04E-4	2.60E-7	2.40E-3	6.02E-6	1.68E-2	4.21E-5	4.37E-3	1.10E-5	3.44E-6	8.97E-6	1.63E-5	1.25E-4	4.09E-5
VIB	6.31E-7	6.86E-5	1.73E-7	4.31E-4	1.08E-6	1.35E-4	3.38E-7	7.56E-4	1.90E-6	4.82E-3	1.21E-5	1.25E-3	3.14E-6	3.44E-6	8.97E-6	ε	ε	ε

* NOTE: IF VALUE IS LESS THAN 1.0E-7, ε WILL BE USED TO REPRESENT IT.

Table 3.7-16 St. Lucie Unit 2 Conditional Probability of PDS Sequences with PWR-4 Comparable Source Terms

PDS	Mean Freq. per year	Containment Event Tree End States																
		B2-L	B2-R	B4-L	B4-R	B6-L	B6-R	C2-L	C2-R	C4-L	C4-R	C6-L	C6-R	D2-R	D4-R	E2-R	E4-R	E6-R
IB	2.39E-6	4.06E-7	ε	1.45E-6	ε	ε	ε	8.19E-4	2.05E-6	3.19E-3	8.01E-6	1.83E-4	4.6E-7	4.02E-6	1.31E-6	ε	ε	ε
IIA	1.86E-6	1.50E-6	ε	5.35E-6	ε	3.65E-7	ε	9.60E-4	2.41E-6	4.72E-3	1.18E-5	2.68E-4	6.71E-7	4.40E-6	2.56E-6	ε	ε	ε
IIB	1.71E-6	ε	ε	ε	ε	1.69E-3	4.23E-6	ε	ε	ε	ε	8.57E-2	2.15E-4	ε	6.15E-4	ε	ε	ε
IIE	1.51E-6	8.39E-5	2.10E-7	5.26E-4	1.32E-6	2.11E-5	ε	7.25E-4	1.82E-6	5.46E-3	1.37E-5	1.83E-4	4.58E-7	3.91E-6	1.31E-6	ε	ε	ε
IIF	5.34E-7	1.98E-2	4.96E-5	1.36E-2	3.42E-5	4.34E-2	1.08E-4	2.56E-2	6.40E-5	1.78E-2	4.47E-5	4.73E-2	1.18E-4	8.64E-5	4.52E-4	ε	ε	ε
3B	4.46E-6	1.03E-6	ε	3.66E-6	ε	1.07E-7	ε	6.59E-4	1.65E-6	3.23E-3	8.09E-6	7.89E-5	1.97E-7	4.05E-6	5.65E-7	ε	ε	ε
3H	2.64E-6	7.10E-5	1.78E-7	6.54E-5	1.64E-7	ε	ε	4.15E-2	1.04E-4	4.45E-1	1.11E-3	1.02E-7	ε	6.25E-6	1.22E-6	2.65E-3	2.71E-2	5.15E-4
VB	2.86E-6	6.87E-5	1.72E-7	4.31E-4	1.08E-6	1.35E-4	3.37E-7	7.54E-4	1.90E-6	4.81E-3	1.21E-5	1.25E-3	3.14E-6	3.43E-6	8.99E-6	ε	ε	ε
VIA	7.18E-7	5.26E-5	1.32E-7	3.31E-4	8.30E-7	1.04E-4	2.60E-7	2.40E-3	6.02E-6	1.68E-2	4.21E-5	4.37E-3	1.10E-5	3.44E-6	8.97E-6	1.63E-5	1.25E-4	4.09E-5
VIB	1.08E-6	6.86E-5	1.73E-7	4.31E-4	1.08E-6	1.35E-4	3.38E-7	7.56E-4	1.90E-6	4.82E-3	1.21E-5	1.25E-3	3.14E-6	3.44E-6	8.97E-6	ε	ε	ε

* NOTE: IF VALUE IS LESS THAN 1.0E-7, ε WILL BE USED TO REPRESENT IT.

4.0 CONTAINMENT PERFORMANCE (LEVEL 2) ANALYSIS

4.1 Background

This report documents the containment event tree (CET) analysis conducted for the St. Lucie PRA. The St. Lucie PRA CET analysis methodology is consistent with the PRA procedures guidelines (NUREG/CR-2300), and previous PRAs (such as the Oconee PRA (NSAC-60), Seabrook PRA), and NUREG-1150 [Refs. 4.0-4 & 4.0-6] accident progression analyses for reference large, dry PWRs. The analysis makes use of the EPRI Generic Framework for IPE Back-End Analysis for CET logic model development and quantification [Ref. 4.0-1]. This is supported by an analysis of the containment performance using plant-specific MAAP 3.0B [Ref. 4.0-2] accident simulation code calculations. In certain cases, the containment's response to the physical processes during an accident is also evaluated comparatively against existing reference plant analysis. The spectrum of severe accident progression is evaluated in a probabilistic framework. Severe accidents that lead to containment failure and environmental release of fission products are grouped into sets of release categories.

4.2 Report Organization

The CET analysis documented in this report consists of an overview of the dominant plant damage states (PDS) and a discussion of the quantification process and presentation of the results. Several MAAP calculations conducted to assess the response of the St. Lucie containment to severe accident phenomena are used in the characterization of the containment's performance as well as in the determination of the likelihoods of the CET branch points. Plant-specific features are considered in the analysis of the potential challenges to containment integrity and determination of the conditional probability of containment failure modes. St. Lucie specific components, systems and structures, including applicable emergency operating procedures were considered in the analysis.

The discussion in this report and the analysis conducted are supported by the following appendices documenting these topics:

- Containment Performance Features (Appendix C):
- Plant Damage States Binning (Appendix D);
- Containment Event Tree Analysis (Appendix E);
- Severe Accident Progression Analysis (Appendix F); and
- Containment Failure Pressure Characterization (Appendix G).

The accident progression analysis models are essentially encoded in the MAAP 3.0B code for PWRs. Details of the analytical and numerical solutions to severe accident progression modeling and phenomenological assumptions are described in the MAAP Users Manual and will not be included in this report.

The CET analysis described in this report is a follow-on effort of Level 1 analysis results [Ref. 4.0-3] which includes an analysis of the recovery of accident sequences that lead to core damage. The recovery measures considered in the Level 1 analysis are generally applied in the Level 2 analysis regime. The objective is to rank the risk importance of the identified accident scenarios that would lead to early containment failure. Radionuclide release characterization is used as a measure of the severity of the accident.

The characteristics of the dominant plant damage state (PDS) contributors and their frequencies will be described first, followed by a discussion of the CET analytical process and results, and finally the radionuclide release characteristics of the dominant PDS.

4.3 Plant Damage States

Plant damage states (PDS) are a combination of core damage state with containment safeguard states. The dominant core damage sequences from the Level 1 analysis were studied further for contributions to the plant damage states. LOCAs and transients with failure of secondary heat removal were determined in level 1 task as the dominant contributors to the core damage frequency. The binning of the PDS is discussed in detail in Appendix D.

Containment isolation failures were quantified separately to facilitate the PDS quantification. This separate treatment of containment isolation is a reasonable approximation because there is no dependency between the containment spray, emergency containment coolers, and the containment isolation. The containment isolation failure probability is used in the CET top event logic for early containment failure.

Based on iterations of several preliminary level 2 PDS quantification results, several factors were considered in simplifying the final PDS cutsets:

1. Any Level 1 functional sequences with core damage frequency less than 1.0E-7/yr are not included for Level 2 PDS quantification.

2. Sequences with significant recovery credited in the core damage quantification and not likely to have all containment safeguards failing are directly transferred to applicable PDS cutsets without further quantification.

3. Other sequences with core damage frequency greater than 1.0E-7/yr and no significant recovery actions included in the cutsets were requantified based on the same approach as Level 1 quantification.

4. ATWS sequences with the core damage frequency greater than 1.0E-7/yr are retained but not requantified. These sequences are used to represent cases where no containment safeguards failures exist. This is reasonable, because ATWS sequences involving failures of containment spray or containment coolers are negligible.

5. SGTR and V sequences are not requantified because they bypass containment. CETs for these sequences are not developed; releases are based on EPRI framework NSAC/159 [Ref. 4.0-1].

The mean frequencies of dominant PDS are summarized in Table 4.0-1A and 4.0-1B for St. Lucie Unit 1 and Unit 2 respectively.

Each of the dominant PDS is described below (in descending order of the St. Lucie Unit 1 PDS frequency):

PDS 3B. This group of PDS consists mainly of sequences initiated by transients such as loss of main feedwater and other simple reactor trips followed by failure of secondary heat removal. Injection Systems (HPSI and LPSI) are available but do not inject due to the high RCS pressure. Injection Systems will deliver water via RCS at vessel breach to cavity. Containment spray and ECCs are available. CS is actuated before vessel breach because the containment pressure does exceed the setpoint of containment spray (10 psig for Unit 1, and 5.4 psig for Unit 2). The frequency of PDS 3B is approximately 3.93E-6/yr for Unit 1, and 4.46E-6/yr for Unit 2.

PDS VB. This group of PDS is dominated by Small LOCA and Large LOCA with no injection leading to early core damage. Emergency containment coolers and containment sprays are available during both injection and recirculation phases. The frequency of PDS VB is 2.97E-6/yr for Unit 1, and 2.86E-6/yr for Unit 2.

PDS IB. This group of PDS is dominated by Small-Small LOCA (S1 LOCA), with no HPSI. ECCs are available and in operation, CS is in operation (containment spray actuation setpoint reached). The frequency of PDS IB is 2.76E-6/yr for Unit 1 and 2.39E-6/yr for Unit 2.

PDS 3H. This group of PDS is dominated by SBO and other transients with no AFW. In addition, no ECC or CS is available throughout the accident scenarios. The frequency of PDS 3H is 2.72E-6/yr for Unit 1 and 2.64E-6/yr for Unit 2, respectively.

PDS IIA. This group of PDS is dominated by Small-Small LOCA with recirculation failure. CS is available only during injection, ECCs are available for both injection and recirculation. The frequency of PDS IIA is 1.73E-6/yr for Unit 1, and 1.86E-6/yr for Unit 2.

PDS IIB. This group of PDS is dominated by small-small LOCA with successful emergency coolant injection but recirculation failure. Both ECCs and spray are in operation. The frequency of PDS IIB is 1.54E-6/yr for Unit 1 and 1.77E-6/yr for Unit 2, respectively.

PDS IIE. This PDS is dominated by sequences involving Small-Small LOCAs with recirculation failures. Emergency Containment Coolers (ECCs) are not available, while containment spray is available during injection but fails during recirculation. The frequency of this PDS is approximately 1.49E-6/yr for Unit 1, and 1.51E-6/yr for Unit 2.

4.4 Accident Progression and CET Quantification

This section describes the quantification of the CET event nodes developed for the St. Lucie Level 2 Analysis. The process used revolves around the quantification of the basic events in the logic trees provided in Appendix E. The logic trees that support the top event nodes are fault tree representations of the relationship of severe accident phenomena, systems operation, and operator actions. The CET event node logic trees are analogous to the system fault trees. These are quantified using fault tree quantification methods. The component failures include physical processes and system failures that affect containment performance and source terms. The information used includes "generic" data on phenomenological issues and plant-specific information regarding systems performance and containment response. For example, containment failure probabilities are evaluated using the St. Lucie plant-specific containment parameters (e.g., containment volume and ultimate capacity) for those failure modes that are generally limited by physical processes (e.g., overpressure failures). The boundary conditions relative to the physical processes (i.e., severe accident phenomena or fission product release mechanisms) that impact containment response, mitigation, or recovery of the severe accident progression can be obtained from the accident sequence PDS definitions and the CET nodes.

Probabilistic data applicable to severe accident phenomenological events and containment failure events are generally not available. IDCOR (Technical Report for Task 4.1) developed detailed containment and phenomenological event trees to characterize the sequence progression, but these, unfortunately, were not quantified. Instead, simplified CETs were developed for the IPEM, for which qualitative likelihoods were assessed. WASH 1400 used a simple representation of the containment failure modes to quantify various release events, and industry PRAs have used similar approaches while incorporating the mitigating effects of containment safeguards on the likelihood of containment failure. More recent CETs developed for NUREG-1150, significantly more complicated than the IPEM and previous PRAs, were quantified in detail. While that level of detail is not necessary for this assessment, the probabilistic information (and basis for the assigned probabilities) is very useful in quantifying the likelihoods of the CET branch points for St. Lucie.

4.4.1 Quantification Rationale

The rationale used in the quantification effort is developed on two levels: general phenomena and accident specific. The estimates of conditional probabilities of general events that appear in several of the CETs are based on generic information related to severe accident phenomena. Accident specific probabilities are developed on the basis of the boundary conditions derived from the unique PDS that is, in turn, defined from identified PDSs and dominant sequence cut sets. In either case, the probability values used are relative values, meant to provide insights on the containment performance under postulated severe accident conditions. The probability values used for the phenomenological issues contained in the logic trees are essentially based on the NUREG-1150 accident progression event tree analysis contained in the NUREG/CR-4551 assessment of the ranges of event likelihoods. PDS specific basic event probability is supported by the MAAP 3.0B calculations for St. Lucie (Appendix F).

Table 4.0-2 summarizes the qualitative terms used in the NUREG-1150 accident progression event tree analysis that are also adopted in this quantification. The values in Table 4.0-2 that represent ranges in the probabilities are not sampled in this study to provide a mean value for the basic events. It is recognized, however, that point estimates are required for the quantification, thus a mid-range (assumes a flat distribution) is used to estimate the likelihoods in the logic tree. Qualitative ranges of probabilities used in this study provide a self-consistent approach in quantifying the relative severity of the phenomenological issues that affect containment performance. For example, the likelihood of containment failure due to the pressure load from one phenomenon (such as HPME for the high RCS pressure PDSs) relative to the pressure load induced by another phenomenon (such as hydrogen burning for low RCS pressure PDSs) can be estimated on a consistent basis. This process assures determination of the significant contributors to containment vulnerabilities as intended by the IPE Generic Letter 88-20.

The logic trees decompose the complex relationships of severe accident progression, containment response, and systems-related effects into basic events that can be quantified using values based on existing information, MAAP 3.0B calculational models, and comparative evaluations. It also ensures that plant specific information, specifically system related issues and containment performance features, is considered in the CET quantification. The resulting quantified containment logic model provides a severe accident model that can be easily understood. Insights derived from this model are not dependent so much on uncertainties in severe accident phenomena as they are on plant specific information with respect to containment parameters and sequence definitions.

To summarize the quantification approach in this study, basic event values (based on NUREG-1150 evaluations) were estimated on the basis of phenomenological conditions, and flags are used for system related events that are determined by the PDSs. The implications of physical processes and the dependencies of the basic events on the occurrence of previous events are also depicted in the logic trees. Generic information is used to quantify highly uncertain phenomenological issues, plant-specific simulation code calculations are used to quantify containment response (i.e., failures) and accident-specific information is used to quantify system related issues affecting containment response.

4.4.2 Logic Tree Basic Events

The basic events included in the logic trees are quantified in this study by considering the phenomenological issues, system related issues, and PDS defined scenario dependencies. The quantification process considers the following general classification:

1. Phenomena related events that are subject to large uncertainties;
2. Phenomena related events that are subject to uncertainties, but are influenced by plant specific features;
3. System related events that are defined by the PDS (determined from Level 1 systems) or human response issues; and
4. Basic events that are either true or false, depending on the success or failure, respectively, of a previous CET event node.

The basic events are provided for each CET top event in Table 4.0-3 and are listed according to these general classifications. The phenomena related events are quantified using reported values from existing severe accident phenomenological analyses and reference plant studies (e.g., NUREG-1116 and NUREG-1150). In general, conservative assumptions are used due to the large uncertainties associated with the first category of basic events. The probabilities of phenomena related events within the second category are generated using qualitative descriptions of probability ranges, e.g., likely (high), indeterminate (medium) or unlikely (low) representing 0.8, 0.5, or 0.2, respectively, supported by plant-specific code (MAAP 3.0B) calculations for St. Lucie and by other industry studies, or judgement. The probabilities used in the quantification are consistent with the qualitative ranges used in NUREG-1150 accident progression event tree analysis (see Table 4.0-2). The basic events within the third category are either zero or one, depending on the PDS definition. These can also be quantified using probabilistic models of system related events that are influenced by systems performance given the accident conditions in progress, human response issues, e.g., emergency operating procedures, or recovery actions. The basic events requiring operator actions may be quantified using the Level 1 PRA probabilistic models. These recovery events are identified in the CETs, consistent with the intent of the IPE Level 2 analysis. However, no credit is given to these events, if no procedures exist to implement the recovery action beyond core damage.

Consequently, the analysis of physical processes to support the quantification of the CETs primarily involve the second category of basic events. The quantification of most of the basic events within the second category requires an assessment of the pressure loads that challenge the containment integrity (see Appendix F). It also requires an estimate of the performance limits of the containment, i.e., ultimate failure capability of the containment (Appendix G). It is important to note, however, that since probabilistic data are not generally available for severe accident phenomena, qualitative estimates of probabilities are used throughout the quantification.

4.4.3 Results of Severe Accident Progression Analysis

The results in terms of key parameters of St. Lucie specific MAAP analyses of dominant plant damage states are provided in Appendix F. These results include timings of the sequence of events and the figures of merit related to the core melt progression and the fission products release and transport. For all the baseline calculations the estimated containment pressure at vessel breach is lower than 95 psig, the containment failure pressure. The MAAP analyses, therefore, indicate that early containment failure is unlikely.

The plant damage states analyzed include those of high RCS pressure, moderate RCS pressure and low RCS pressure scenarios. In the case of high RCS pressure sequences, most of the debris from the core is predicted by MAAP to relocate in the lower or the upper compartment, depending on the debris entrainment option used. In the case of low pressure melt ejection, the molten debris resides in the cavity. The cavity in the St. Lucie containment remains wet, particularly for the period after the vessel breach, except for the V sequence and the IIIH scenarios with the hot leg break. For these cases the RCS water getting into the cavity, after the vessel failure, boils-off rapidly from the residing hot debris. The maximum concrete attack, however, is found to be less than 5 ft. as per the MAAP calculations.

The failure of the containment is found to be dependent on the operation of the ECCs and the CS. In all the cases where the containment sprays are lost, either on EQ limits or due to the recirculation failure, the containment pressure rises gradually due to the continuous steaming in the cavity and the lower compartment, leading to the late containment failure. When the containment spray is available in the recirculation mode, the ECCs and the CS are found to be very effective in maintaining low pressures in the containment for extended periods of time.

The MAAP analyses described in Appendix F (e.g. Table 43) demonstrate the following:

1. Direct Containment Heating (DCH) could cause early containment failures for high pressure scenarios, if a large fraction of the debris is assumed to participate in DCH. This is simulated in MAAP cases TIIIH4 and TIIIH5.

2. Low pressure scenarios do not lead to early containment failures.

3. Without long-term containment heat removal (either ECC or CS), late containment failure is very likely, if the simulation time is extended beyond 50 hours. ECC and CS are lost on EQ limits in some of the scenarios simulated in MAAP. Even with the loss of ECC and CS on EQ limits, the containment is found not to fail within 24 hours in majority of the cases.

4. Hydrogen concentrations in the containment are low enough for any detonation concerns. Complete hydrogen burn is also very unlikely based on the MAAP predictions of hydrogen concentrations of less than 4%. At these levels of hydrogen, local burns due to pocketing will not threaten the containment integrity. Sensitivity calculations show that even complete hydrogen burn at vessel failure time does not produce early containment failure.

5. The maximum depth of concrete attack in scenario in which the cavity dries out early, is predicted in MAAP to be about 4.7 ft. The basemat melt-through is therefore not a concern.

4.5 Conditional Probabilities of CET Event Nodes

The following discussion describes the phenomenological events included in the logic trees. This discussion summarizes the values and bases used for the estimates of generalized event conditional probabilities that appear in several of the St. Lucie CETs. In most instances, the bases will be direct quotes from the reference documents, particularly for those areas where the quantification draws heavily from the expert elicitation process in NUREG-1150.

4.5.1 DP: RCS Depressurized Before Vessel Breach

This event node answers the question regarding the RCS pressure at vessel breach. This would include the entry state to the CET (boundary condition of the PDS) which implies RCS break sufficient to depressurize the vessel (such as a large break LOCA event). It also addresses depressurization of the vessel subsequent to core melt (i.e., severe accident phenomena induced RCS break) that would result in a low RCS pressure prior to vessel head failure.

The issues considered and basic events quantified in this event node for the dominant PDSs and the basic event types are described as follows:

PRHLSLOK — Hot leg and surge line remain intact. This belongs to the second category of basic events, in which the high temperature response of the RCS hot leg or surge lines induces creep rupture, thus depressurizing the RCS. Existing calculations using NRC's Source Term Code Package (STCP) code (NUREG/-CR-4624) indicate that peak primary system gas temperatures can approach 4000 F, threatening the integrity of the hot legs, while MAAP calculations, show temperatures 800 F (EPRI NP-6111) lower than STCP MARCH for similar type accident scenarios. Plant-specific code calculations for St. Lucie for high RCS-pressure scenarios (Appendix F) which account for potentially high zircaloy cladding oxidation in the vessel indicate that high temperatures are possible. Reported probability values from the NUREG-1150 assessments for a four-loop and three-loop Westinghouse design are used as point of reference in determining the probability of hot leg failure for St. Lucie, considering the MAAP calculations performed in this analysis. It is judged that the St. Lucie design would allow natural circulation to occur supporting heat-up of the RCS structures above the core. The heat-up of the hot leg or pressurizer surge line could be similar to that of the reference plant. The basis for this assumption is the RCS hot leg temperature time history calculated by MAAP 3.0B and a comparison of the upper plenum and hot leg/surge line design (in particular, the masses and thermal capacity), of the reference plant (Zion) which are similar to that of St. Lucie plant (see Appendix C). The St. Lucie NSSS consists of two loops. Based on examination of the St. Lucie plant-specific calculations, a probability of maintaining the hot leg integrity, consistent with NUREG-1150 is judged to be similar to the value of 0.3-0.05 used for Surry (NUREG/CR-4551). The probability of 0.278 was assigned to this basic event for high pressure PDSs.

PRHLSLOK1 — Hot leg and surge line remain intact given medium pressure PDS. The probability for medium pressure sequences are judged higher (1.0 is used) than high pressure PDSs consistent with the Surry Analysis (NUREG-CR/4551).

PRHLSLOK2 — Hot leg and surge line remain intact given sequences depressurized subsequent to core damage. The induced medium pressure sequences is judged to result in a slightly lower probability of remaining intact than initially moderate pressure PDS. A value of 0.5 is sued.

PRSGOK — Steam generator tubes do not rupture before hot leg failure occurs. The steam generators tubes, while they have a smaller thermal capacity, generally heat up after the hot legs or surge lines since they are downstream of the hot legs. This is supported by the MAAP 3.0B calculations. This basic event is assigned a probability (0.982) that is higher than the hot leg probability. (NUREG-1150 assigned a range of 0.995 to 0.95 for this basic event).

PRSEALOK — RCP seal remains intact following core damage. This is quantified by considering the degraded conditions during core melt and the particular reactor coolant pump (RCP) seal design of St. Lucie. Since there is no sufficient data to quantify the value of this basic event, a value of 0.5 is used based on NUREG-1150 Analysis for Surry Plant.

QHP (QMP) — Sequence is a high (medium) pressure PDS. This basic event is PDS dependent. For a high pressure PDS, QHP is 1. For a medium pressure PDS, QMP is 1. Otherwise, the value is 0. This is determined implicitly by the PDS definition.

HOP-DP — Operator fails to depressurize RCS. Depressurization of the RCS as a result of human intervention is addressed in this basic event. This is included in the logic model to identify potential recovery measures. For this analysis, a value of 0.02 is used due based on judgement.

PRCSRVS — SRVs are not stuck open. SRVs could stick open if they are cycling or have high temperature induced failure. This value can be determined using the Level 1 analysis of safety/relief valves sticking open, considering a potentially more degraded condition than that prior to core damage. Because of the uncertainty relative to the severely degraded conditions during core damage, the value of 0.9 is used for all PDSs.

4.5.2 REC: In-Vessel Coolant Make-up Recovery

This event node is quantified on the basis of plant specific factors of systems recovery, alternative systems, emergency operating procedures, and human intervention. In this analysis, the potential for successful human intervention is identified but not credited at this time since no existing procedures are in place to direct such a recovery action given failure of the normal systems and degraded conditions. With the exception of high RCS pressure PDSs in which the low pressure injection systems are operating, but the high head precludes injection, depressurization of the RCS in the previous event node would lead to successful LPI injection without operator intervention. In this case, the failure probability of recovery is determined by quantifying the systems models associated with the LP injection systems. AC power recovery is explicitly considered and only AC power recovery within 2 hours is taken credit in the Level 1 analysis.

Two distinct REC logic trees are developed in this event node, depending on the success (REC1) or failure (REC2) of the previous event node, DP: RCS depressurized before vessel breach.

The basic events in this CET event node are quantified using the following guidelines:

SHP-SIS1 — The HPSI system is not recoverable. This basic event identifies a potential recovery action. The probability of 0.5 is assumed for all sequences

SLP-SIS1 — Low pressure safety injection (LPSI) can not be recovered. For PDSs in which the RCS is at high pressure and does not depressurize under event node DP, the probability assigned is 1.0; if the RCS is depressurized in DP, the initiation of the low pressure injection systems is considered possible provided the PDS indicate that the low head injection systems are available, but operation is precluded only due to the RCS high head conditions. The LPSI system recovery considered in this event is the automatic injection from the LPSI system once the high pressure condition in the vessel is removed (i.e., successful depressurization in the previous event node). This is relevant only if the previous event node (DP) is successful. A value of 0.5 (unlikely) is used for PDSs that meet these conditions; otherwise the value is 1.0 and if water is available in the RWST. This basic event is strongly influenced by the PDS.

SALT-SIS1 — Alternative system not recovered during core melt. This basic event is analogous to the previous basic events discussed above. In this case, however, alternative injection paths are considered. A probability of 0.1 is used.

SACPOWER — AC power not restored or available. The probability of AC power recovery prior to vessel breach is determined for station black-out sequences as an extension of the Level 1 analysis. If power is available, as defined by the PDS, this is assumed to be 0.01; otherwise one.

4.5.3 VF: No Vessel Failure

Event VF is a phenomenological issue that is subject to generic considerations of formation of a stable coolable debris configuration in the vessel given recovery of coolant injection in the previous event. The core damage is arrested, precluding vessel head thermal attack and failure. This event node is quantified using guidelines obtained from the reference plant analysis. This event is considered more likely and relevant only if coolant is recovered in vessel (i.e., success in the previous event node REC). Although external cooling through the bottom head may be possible for St. Lucie, (due to the cavity configuration), it is considered indeterminate in this analysis due to the lack of existing information that would support sufficient heat transfer through the bottom head, particularly if the support plate has failed.

The basic events quantified include:

PRCOOLDBIV — Coolable debris bed not formed in-vessel. The probability that a coolable debris bed is not formed in the vessel, given that coolant injection has been recovered in the previous event node, is characterized in the Surry analysis as unlikely (0.1) for induced depressurization sequences. For small LOCAs with AFW, the Surry analysis assigned a probability of 0.30, provided recovery occurs within 1 to 4 hours. Thermal failure of the bottom head at the instrumentation penetration weldments is predicted to occur (IDCOR TSR). The initial contact of the molten material would thermally attack the penetration weldment of the reactor vessel head failing the instrumentation tube and ablating the surrounding structure. The formation of a coolable debris bed may be considered possible provided coolant is recovered prior to relocation of the molten material outside the original core boundary (such as TMI). However, the uncertainties in the characterization of the molten debris formation and flow to the bottom head following core slump would not support a high likelihood of precluding vessel failure. Thus, a non-zero probability may be used for scenarios in which early recovery of injection is possible. The plant-specific analysis of the core uncovery timing and recovery of coolant injection indicate at least one hour is required from core uncovering to core slump for most non-large LOCA scenarios (see Appendix F). MAAP 3.0B, however, does not have adequate core recovery models. MAAP predicts core melting would proceed to vessel breach despite recovery of core coolant injection. Therefore, a probability of 0.1 is used at this time. (This is slightly higher than a similar case in the Surry analysis.) For accident sequences involving a loss of power, the likelihood of arresting core degradation is closely tied into the recovery of power and subsequent recovery of coolant injection. The power recovery curve for St. Lucie is considered.

PR-HT-TRAN[1] No ex-vessel heat transfer established. No ex-vessel heat transfer is established through the bottom head. This is similar to basic event PRCOOLDBIV, although it is considered less likely to be successful. The St. Lucie reactor cavity design assures the reactor bottom head to be submerged provided the RWST water inventory has been discharged to the containment (i.e., sprays are actuated or core cooling system fails upon recirculation). For most PDSs, the sprays are available, therefore the bottom head is going to be fully submerged in a flooded cavity. However, limited information is available at this time to determine the viability of establishing heat transfer through the vessel walls. A value of 0.1 is always used at this time.

4.5.4 CFE: No Early Containment Failure

The logic tree for early containment failure includes several phenomenological issues that are subject to uncertainty. Accordingly, the quantification of the basic events significantly draws from the NUREG-1150 accident progression event tree analysis documented in NUREG/CR-4551 (1989). The likelihoods of each of these events are dependent on the accident sequence (i.e., PDS) in progress. Additionally, the occurrence of a phenomenological event can be dependent on the PDS boundary conditions (for example, overpressure is considered unlikely if the sprays or fan coolers are functional). These dependencies are all considered in the evaluation of event CFE.

[1] These events constitute the gate LWR-HEAD (no lower head cooling via ex-vessel heat removal) that is included in the logic tree to depict a potential recovery mode related to cooling through the vessel walls. Although this is not considered viable in the NUREG-1150 analysis, it is included to portray ongoing analyses relative to this issue. For this analysis, the possibility is included, but not credited in the baseline scenarios. It is treated as a sensitivity issue.

The major source of containment pressure loads considered in this CET event node is the high pressure melt ejection (HPME) of molten core material at vessel breach for high RCS-pressure PDSs. The containment loads associated with HPME are generated by the addition of mass and energy to the containment atmosphere from several sources:

- Blowdown of reactor coolant system steam and hydrogen inventory into the containment;
- Combustion of hydrogen released prior to and during HPME;
- Interactions between molten core debris and water on the containment floor; and
- Direct containment heating.

Although the rapid thermal transient at vessel failure is not calculated by MAAP to induce the extremely high pressures and temperatures that threaten containment integrity in existing reference plant analysis (Zion Task 23.1 Technical Report, EPRI NP-7192 Report on PWR MAAP 3.0B sensitivity analysis), the NUREG-1150 assessments provide insights on the containment loads from HPME at vessel failure.[2] Conservative assumptions relative to the extent of clad oxidation, debris dispersal and coincident hydrogen burning is made in the plant-specific MAAP 3.0B analysis for St. Lucie to quantify the likelihood of containment failure. The assumptions are consistent with NRC's position (GL 88-20) with respect to the hydrogen generation or core blockage issue as modeled in MAAP.

Five major parameters considered for NUREG-1150 are related to this issue. These are:

1. Reactor vessel pressure prior to vessel breach;
2. Fraction of unoxidized zircaloy in the melt;
3. Fraction of molten core debris ejected;
4. Initial size of hole in the reactor vessel lower head; and

[2] Appendix C of NUREG-1150 claims that "uncertainties in containment loads accompanying high pressure melt ejection are not major contributors to the overall uncertainty in risk for any of the three PWRs examined." There is "high confidence that these containments can accommodate the pressure increment accompanying high pressure melt ejection." Also, "accident sequences that have traditionally been considered as 'high pressure' core melt accidents (e.g., a fast station blackout) are estimated to result in a depressurized reactor vessel by the time of reactor vessel breach with a relatively high frequency. Depressurization mechanisms considered in the present analysis include temperature induced hot leg failure and steam generator tube ruptures, reactor coolant pump seal failures, and stuck-open power-operated relief valves (PORVs)."

5. Presence (or lack) of water in the reactor cavity.

The quantification of the phenomenological basic events influenced by HPME for the CFE event node is based on a comparison of the total pressure obtained during the HPME event and the ultimate pressure capacity of the containment. The containment failure pressure for St. Lucie is estimated to be 95 psig (see Appendix G). The MAAP analysis of dispersal of the debris to the upper compartment region in addition to no core blockage (see Appendix F) indicate that the containment integrity will indeed be challenged. A peak pressure greater than 95 psig is obtained. Estimates of the total pressure load may also be obtained by adding the pressure rise to the base pressure in the containment before vessel breach. An estimate of equivalent pressure load may be obtained by scaling the pressure increase from HPME obtained for Surry according to the volume ratio (of the reference plant containment) and thermal power relative to St. Lucie.

Screening probabilities are obtained by performing bounding calculations of the equivalent pressure rise, and appropriate likelihoods are used (either zero or one) if the pressure is likely to be well below the ultimate capacity (zero) or well above (one). The process is illustrated graphically in Figure 4.0-1. In this figure, the pressure challenge to containment integrity as a result of HPME is used as an example. Case 1 demonstrates a situation in which the pressure load is significantly lower than the ultimate pressure capacity of the St. Lucie containment (i.e., pressure load less than or equal to the design pressure). In this case a failure probability of 1.0E-4 (an arbitrary number assigned) is used. Case 2 is a situation where the reverse condition is true, i.e., the pressure load exceeds the ultimate pressure by a significant margin, hence the failure probability used is 1.0 (i.e., certain). For a pressure load that is within the range of failure pressures estimated for a similar containment design relative to design pressure (e.g., $2 \times P_{design} < P_{load} < 3 \times P_{design}$), the uncertainty in the containment failure probability function is considered.

The third case illustrated in Figure 4.0-1 requires a more detailed analysis since the pressure load is close to the failure pressure of the containment, particularly if the sequence is a dominant contributor. In this case, the quantification of containment failure probability considers the uncertainty in the determination of the containment capacity. It is recognized, that due to the uncertainty in characterizing the containment capacity, the probability of containment failure is represented by a distribution of failure pressures (containment fragility curve). In addition, the uncertainty in characterizing severe accident phenomena, results in a range of conditions that would also generate a distribution of pressure loads given a phenomenon (such as HPME pressure loads). Therefore, there is a finite probability that the containment might fail given a pressure load in containment that is lower than the failure pressure of the containment.

A similar approach is used for the pressure challenges of hydrogen burning, steam overpressure and non-condensible gas generation as will be discussed in event CFL: No Late Containment Failure.

The determination of the probability of containment failure conducted for this situation (Case 3) considers the uncertainty in the pressure load and the containment capability of surviving such a load. Figure 4.0-2 illustrates the relationship of the distribution in the pressure load and the containment capacity. The curves are used together by selecting a realistic characterization of the probability that the containment would survive a pressure load imposed by a particular phenomenon. Using these curves, the median estimate of the pressure load (0.5 cumulative probability) is obtained by reading across horizontally in the pressure load distribution. The value obtained is compared against the failure pressure distribution of the containment (by reading vertically upward) to determine the point on the static failure pressure curve that intersects the same value of pressure, then left. This final value of the probability obtained is the probability (by reading left back to the ordinate) that the containment will survive the pressure load. The failure probability of the containment is one minus this value. In summary, a simple and reasonable approach used to estimate the containment failure probability for St. Lucie Units 1 and 2. An interpolated failure probability is used if pressure load is greater than 2 times the containment design pressure (i.e., P_{load} > 80 psig) but less than 3 times the design pressure (i.e., 120 psig). If pressure load is greater than 3 times design pressure the failure probability is assumed to be 1.0.

There are two distinct logic tree structures used to quantify this event node, depending on the success or failure of the previous CET event nodes DP and REC. If the RCS is depressurized, HPME loads are not considered relevant. If the RCS is at low pressure, the loads are significantly reduced and HPME is not a contributor to CFE.

The basic events quantified for CET event node CFE using this approach account for different systems as phenomena-related conditions. The information below for the dominant PDSs is used to quantify the appropriate basic events: (Note the first two events are for High-RCS-Pressure Sequences)

> <u>Containment Pressure Exceeds the Ultimate Pressure Capacity of the Containment Given Wet Cavity</u>. The Surry analysis generated a relationship between the probability of attaining a certain pressure level in the containment versus the ultimate capacity of the containment. This information along with the MAAP calculations are used for screening this event in the logic tree (CFE2).

> <u>Containment Pressure Exceeds the Ultimate Pressure Capacity of the Containment Given Dry Cavity</u>. The relationship is again derived from the NUREG-1150 analysis, modified to reflect a plant specific ultimate capacity.

> <u>Containment Pressure Exceeds the Ultimate Pressure Given Low RCS Pressure Condition at Vessel Breach</u>. The quantification of this event node is similar in concept to the HPME events, although in this case, the pressure loads are principally driven by steam generation following debris pour into the cavity. The pressure rise will vary depending on the PDS being evaluated. Reference documents (e.g., NUREG-1150, NUREG/CR-4551) are used to guide the analyses, and plant specific characterization of the pressure loads are used to supplement the analytical process. These are used to conclude whether the pressure rise at vessel failure (for low pressure sequences) will be higher than the ultimate pressure capability of the containment. For the Surry analysis, the pressure rise at vessel breach was determined to be 19 psi. Plant specific pressure rise for low-RCS pressure PDSs (such as PDS VH or PDS IIIC in which hot leg failure occurs as discussed in Appendix F) are used in the quantification.

Other phenomenological and system related events in the CFE logic tree are quantified using the following guidelines:

PRALPHAL Alpha Event Occurs Given RCS Depressurized. Alpha event is a "very energetic molten fuel-coolant interaction (steam explosion)" that can fail the vessel and generate a missile that can fail containment as well. The opinions expressed from the Steam Explosion Review Group (NUREG-1116) was that this event was very unlikely (10^{-4} to 10^{-2}) and for Surry was assigned a mean value of 0.008 for low pressure sequences. Seabrook assigned a value of 10^{-4} for this event.

PRALPHAH Alpha Event Occurs Given RCS at High pressure. Expert opinion is that Alpha is less likely for high pressure sequences. For the Surry analysis of high pressure sequences, the likelihood was set to be an order of magnitude lower than the low pressure scenario value. Decreasing the value by an order of magnitude is an acceptable way to arrive at this value; both values reflect a very unlikely probability.

PRROCKET Vessel Acts as Rocket and Fails Containment. From NUREG/CR-4551, Volume 3, Appendix A: "The 'rocket' problem has not been well studied. A possible scenario is there is gross failure of the bottom head of the vessel at high pressure. The gas inside the vessel is at about 2500 psia and its escape from the bottom of the vessel accelerates the vessel upwards. The bolts holding down the vessel fail, the hot legs and cold legs are sheared off, and the vessel attains enough momentum to rise clear of the shield wall. Striking the containment wall, the vessel could potentially fail the pressure boundary. Before striking the containment wall or dome, the vessel must dislodge the missile shield and the manipulator crane, and avoid or dislodge the polar crane." Expert opinion is that a rocket event is less likely than a low pressure Alpha event, so a mean value of 0.001 was assigned for Surry, given that a circumferential failure of the vessel occurs (a very unlikely event or a probability of 0.027). Therefore, the probability of this basic event is 0.001 x 0.027 = 2.7E-5. For the St. Lucie Study, 0.001 is used.

PREVSE Ex-Vessel steam explosion occurs. The likelihood of steam explosion in the cavity is considered indeterminate, provided the cavity is wet. Otherwise, the probability is zero.

PRCFEEVSE Containment fails given ex-vessel steam explosion. The cavity design for St. Lucie is essentially isolated. There are no direct pathways to the containment wall that can provide impulse load of the displaced water. Thus the probability of containment failure is assessed to be negligible. (Fuel coolant interactions leading to containment pressurization are considered in overpressure failure basic events.)

PRIMPINGE Containment fails given debris impingement. The containment wall at St. Lucie, as discussed in PRCFEEVSE is essentially isolated from the cavity. Therefore, a probability of 0.0 is assigned.

SNOSPRAY1 — Containment Sprays Do Not Operate before vessel breach. Whether containment sprays are operating, is implicitly defined by the PDS bin. If the sprays are not operating, this probability is 1; otherwise it is zero.

PRCI — Containment Isolation Failure. For independent isolation failures (i.e., not influenced by the PDS), the basic event probability associated with containment isolation failure of 1.0E-3 is obtained from the containment isolation system fault tree models.

4.5.5 DC: Coolable Debris Formation Ex-Vessel

This event node addresses the possibility that core-concrete interactions will not occur if the debris is inherently coolable and there is water to cool the debris. The quantification is dependent on the success or failure of previous event nodes and on the generic issues related to the formation of a coolable debris bed configuration. The St. Lucie cavity configuration is a deep cylinder, which would likely result in formation of a deep molten core material if all of the core mass pours into the cavity. On the other hand, the flow relationship of the cavity to the outer compartment regions of the containment assures a flooded cavity. MAAP 3.0B calculations predict a coolable and quenched debris, provided water is available. However, MCCI studies indicate that concrete attack can still occur despite an overlying water pool. Therefore, coolability is considered to be an uncertain issue even with the likely wet cavity. The following basic events are quantified using the following guidelines:

SNOSPRAY2 — Containment sprays operate after vessel breach. If the sprays are not operating, it is conservatively assumed that water will not be available to replenish boil-off from the cavity and cool the debris. Whether sprays are operating is implicitly defined by the PDS.

SNOLPI — Loss of LPI injection through failed vessel. Low pressure injection by way of a failed vessel head is one other means of providing coolant into the cavity. This is determined by the success of event node REC, followed by failure to arrest core damage (failure of event node VF). Water must be available in the RWST so that only PDSs with failure of spray (or initial failure at injection) credit this condition.

PRCDB-HP — Coolable debris bed doesn't form given HPME. Given high pressure melt ejection, the likelihood that a coolable debris bed does not form was assigned a probability of 0.175 (unlikely), based on NUREG-1150 analysis.

PRCDB-LPSE — Coolable debris bed doesn't form given EVSE occurs and no HPME. Formation of a coolable debris configuration is considered possible, if an ex-vessel steam explosion occurs, given that the vessel fails at low pressure. The probability is 0.175 (unlikely) that a coolable debris bed would not form, based on expert elicitation for the NUREG-1150 analysis.

PREVSE — Ex-vessel steam explosion occurs. The dependence of the formation of a coolable debris bed configuration on ex-vessel steam explosions is modeled by this basic event. The probability that an ex-vessel steam explosion occurs was determined to be indeterminate (0.5) (NUREG-1150).

PRCRUST — Impervious crust forms precluding coolable debris. If the debris is not dispersed by ex-vessel steam explosion nor by high pressure melt ejection, Surry analysis experts (NUREG-1150) determined the probability to be indeterminate (0.5), again provided that water is available. This relates to the formation of an impervious crust that would prevent quenching of debris.

QWETCAV — Cavity is wet. Formation of a coolable debris requires that the cavity is flooded either as a boundary condition prior to vessel breach as considered in this basic event, or by coolant addition as considered in other basic events. An initially flooded condition is determined by the wet cavity flag, defined implicitly by the PDS.

4.5.6 CFL: No Late Containment Failure

This event node is quantified in a manner similar to that for event CFE: No Early Containment Failure. The pressure loads of concern are not unlike those considered under event node CFE, with the exception of HPME. The major source of pressure load at this stage of the severe accident progression is steam generation and hydrogen burning. In the St. Lucie PRA, the conditional probabilities assigned to hydrogen burns failing containment after vessel breach are based on an examination of the pressurization calculations of reference PWR plants obtained from existing literature (e.g., BMI STCP calculations, NUREG-1150 CETs). These probabilities are supplemented by the MAAP analysis by varying parameters associated with hydrogen burns. These estimates accounted for the self-limiting conditions of steam inerting for some accident sequences and upper bound calculations of the attendant pressure rise.

The following basic events are quantified in this event node:

SACSPREC — AC power and Sprays not recovered early. Condition of sprays and power early in the scenario is implicitly defined by the PDS. The probability of 0.01 is assumed for not recovering AC power for all PDSs.

PRHB2 — H2 burn occurs given AC power and sprays recovered. Surry experts determined that if sprays and AC power were restored at the start of CCI, hydrogen burn was "very likely" if minute sparks from electrical equipment were present. A value of 0.973 is used.

PRPR2 — Pressure rise due to H2 burn at start of CCI fails containment. The probability of containment failure given HB2 is estimated using the same approach as HPME loads, i.e., a pressure load is calculated and compared against the ultimate pressure capacity of the containment. If the conservative assumptions yield benign pressure loads, then the failure probability is zero, otherwise a distribution of the containment failure pressure is considered. For baseline St. Lucie study, 2.0E-02 is used.

SACSPRECL — AC power and sprays recovered late. Condition of sprays and power late in the scenario are defined by the PDS.

PRHB3 — Late H2 burn occurs given AC power and sprays recovered. Surry experts determined that if sprays and AC power were restored after CCI, late hydrogen burn was 'unlikely' if minute sparks from electrical equipment were present. The difference between HB2 and HB3 is the timing of hydrogen burning.

PRPR3 — Pressure rise due to late H2 burn > ultimate. The probability of containment failure given HB3 is estimated using the same approach as HPME loads, i.e., a bounding pressure is calculated and compared against the ultimate pressure capacity of the containment. If the conservative assumptions yields benign pressure loads, then the failure probability is zero, otherwise a distribution of the containment failure pressure is considered.

PRHB4 — Late H2 burn occurs (given no sprays). Surry experts determined that if sprays and AC power are not restored, hydrogen burn was "very unlikely." Steam inerted conditions would exist and no ignition source is postulated.

PRPR4 — Pressure rise due to late H2 burn > ultimate. The probability of containment failure given HB4 is estimated using the same approach as HPME loads, i.e., a bounding pressure is calculated and compared against the ultimate pressure capacity of the containment. If the conservative assumptions yields benign pressure loads, then the failure probability is zero, otherwise a distribution of the containment failure pressure is considered.

SNOSPRAY2 — Containment sprays do not operate after vessel breach. Conditions of the sprays are determined from PDS definition.

PRMT1 — Melt through occurs given that the debris is not coolable, with sprays on. This basic event assumes that meltthrough would occur before overpressure failure. Surry experts determined meltthrough, even without coolable debris, was "unlikely". It is extremely unlikely at St. Lucie due to the large depth of concrete before the liner. A value of 0.175 is used for all PDSs.

PRMT2 — Melt through occurs given that the debris is not coolable, without sprays on. This basic event assumes that meltthrough would occur before overpressure failure. Surry experts determined that meltthrough, even without coolable debris, was "indeterminate." The failure probability of this event (and the previous event MT1) is dependent on the concrete type used in the basemat. For Surry, the concrete is basaltic, thus non-condensible gas generation is less than would be expected for the limestone concrete at St. Lucie. A value of 0.5 is used for all PDSs.

4.5.7 FPR: Fission Product Removal

Most of the basic events in this logic tree are dependent on the previous event nodes. These are quantified as follows:

QRCS-RET — Significant RCS retention does not occur. For this analysis it has been conservatively assumed that retention for all cold and hot legs breaks will be based on hot leg break release fractions (i.e., retention in the steam generators is neglected). The residence time within the vessel of the fission products generated during core heat-up is considered to be short, and deposition processes do not act to deplete the airborne aerosol concentrations. A probability of 0.175 is used for all PDS.

SNOSPRAY1 — Containment sprays do not operate before vessel breach. The probability of this event is based on the PDS definition.

SNOSPRAY2 — Containment sprays do not operate after vessel breach. The probability of this event, as in the previous event, is determined implicitly by the PDS.

PRHEATUP — RCS/Containment heat up causes revolatilization and late releases into the containment. The probability of this event is 0.175 for all PDSs.

PRNOPOOL — No overlying pool in cavity. Since all PDSs, the St. Lucie plant cavity is wet, the probability of no overlying pool in the cavity is negligible (1.0E-10 used in the quantification).

4.5.8 CFM: Containment Failure Modes

Quantification of this event node is based on the failure mechanism leading up to containment failure. Success means a small break size occurs (as may be expected for leak-before-break failures). A small leakage rate from the containment is obtained that prevents a further increase in primary containment pressure. A slow release to the environment occurs, extending the release duration, and mitigating potential off-site consequences. A small break also allows natural removal mechanisms to compete with leakage from containment. A small break is assumed very likely for slow pressurization, leading up to containment failure such as steaming or non-condensible gas generation.

Failure of this event is certain for catastrophic failures, where large containment flow rates are obtained. This could result in a rapid (or puff) release of airborne fission products to the environment. A large or catastrophic failure is assumed for energetic pressurization such as alpha or rocket failure modes.

The radionuclide leakage rate (or the duration of release) is significantly affected by containment leakage size. If a driving force for large leakage rates is maintained, the natural removal processes are not capable of reducing fission product releases for those accident classes where the containment is breached during core melt and vessel breach. The duration of release also affects the resulting atmospheric dispersion of the radionuclides released to the environment.

The basic events are quantified in the logic tree using the following guidelines:

- For large and catastrophic failures (as may be expected for steam explosions and rocket failure modes), the failure probability is considered to be certain (1.0).

- For small containment failures (as may be induced by failure mechanisms resulting in penetration seal failure), the failure probability is considered "unlikely." Overpressure failures resulting from steam and non-condensible gas generation are considered "highly likely" to result in small containment failures (IDCOR, TSR). Overpressure failures that are induced by high pressure melt ejection loads and hydrogen burning are judged indeterminate (NUREG/CR 4551).

- Basemat failure is judged to result in small containment failures.

Fission product releases can be significantly reduced by the retention of aerosols within the auxiliary building.[3] In this study, auxiliary building retention after containment failure is conservatively neglected for most of the accident scenarios[4] with the exception of interfacing LOCAs. The characterization of the impact of the auxiliary building retention on the fission product source terms is subject to uncertainties (NUREG-1150). The most significant issue is the strong dependence of the auxiliary building retention effectiveness on the residence time during transport. This, in turn, is determined by the mode of containment failure, the location of containment failure relative to the auxiliary building elevations, the driving force of leakage from the primary containment or the rate of gas production in the primary containment, and the potential for hydrogen burning in the auxiliary building. On the other hand, MAAP calculations and more detailed modeling of the retention in the auxiliary building indicate that significant retention can indeed occur (EPRI NP-6586-L). Therefore, a very large volume of secondary enclosure is available to contain the fission products that escape the primary containment in case the containment fails. Plant specific MAAP calculations neglected the auxiliary building as an additional deposition/retention site. By neglecting the impact of the adjacent buildings, a conservative assessment of the severity of the potential consequences (i.e., fission product release magnitudes) is made in this study.

[3] Auxiliary building retention of fission products following containment failure significantly affects the magnitude of radionuclides released to the environment. For St. Lucie, the surrounding auxiliary and adjacent buildings may provide additional deposition sites, but is not considered in this analysis.

[4] The extent of fission product retention depends on the sequence, the mechanisms for natural processes for aerosol removal (e.g., gravitational settling, impaction, or condensation), scrubbing through an overlying water pool (if the failure site is submerged, as might be the case for interfacing LOCA sequences), and the residence time of aerosols in the building.

4.6 CET End State Probabilities for Dominant PDSs

The various progression paths in the CETs lead to unique end states that characterize the time of containment failure and severity of the potential consequences. The CET quantification for each PDS provides the probability of the end states; the radionuclide release characterization provides the measure of the potential consequences.

Each CET of the progression paths or end states may be grouped in a general classification of containment damage states as follows:

Containment Damage States	Description
A	Recovered in-vessel, no vessel failure
B	Late containment failure, no CCI
C	Late containment failure, core-concrete interaction occurs
D	Early containment failure, no CCI
E	Early containment failure, CCI occurs
NCF	No containment failure

The CET end states that result from early containment failures are more severe than late containment failures, generally resulting in higher releases of fission products to the environment, particularly if no active removal mechanisms, such as sprays, work. These CET end states are characterized as "D" and "E" damage states. The CET end states resulting in late containment failures, characterized as "C" damage states, have releases that are significantly mitigated. These include accident sequences where the containment integrity is maintained long after vessel breach. Sequences in which the core damage is recovered ex-vessel (i.e., no CCI) are characterized as "B" damage states, and those that are recovered in-vessel are included in the "A" damage states classification. Accident scenarios with no containment failures are provided as a separate category in the CETs. The quantified CETs for the dominant PDSs are described below.

Figures 4.0-4A and 4.0-4B illustrate the results of the CET quantification for the dominant PDS accident sequence contributors. This figure provides the conditional probability of the various containment damage states, given a core damage accident. Note that the highest probability containment damage state consists of recovered accident scenarios in which the containment remains intact (72% for St. Lucie Unit 1 and 71% for Unit 2). Early containment failures have a very low conditional probability (only 1%). Late containment failures contribute 15% and 13% for St. Lucie Unit 1 and St. Lucie Unit 2 respectively. Bypass sequences include both steam generator tube rupture and interfacing LOCA scenarios. No credit is taken for Auxiliary Building for interfacing LOCA scenarios. Bypass sequences contribute 12% for St. Lucie unit 1 and 15% for St. Lucie Unit 2.

4.6.1 PDS 3B

The quantified CET for PDS 3B, shown in Figure 4.0-5, indicates that the dominant containment damage states are late containment failures. For the most part, these results are determined by the high likelihood that the cavity will be flooded due to the cavity design of St. Lucie which assures all water injected to the vessel goes to the cavity at vessel breach. If the hot legs fail[5] (as might be expected given the potentially high surface temperatures obtained) the pressure loads at vessel breach are quite low, and the containment does not fail early. In the long term, should the core debris form a coolable configuration, the deep water pool would essentially preclude concrete attack and the steam generated from the quenched debris and water in the cavity is condensed by the ECCs and the sprays. The dominant release modes for PDS 3B are C1-L, A1, B5-L, C3-L, and C4-L.

4.6.2 PDS VB

PDS VB involves small and large LOCAs with injection failure. As indicated in Figure 4.0-6, the dominant release modes for PDS VB are C1-L, C3-L, and C4-L, all late containment failures.

4.6.3 PDS IB

PDS IB involves small-small LOCAs with injection failure. As indicated in Figure 4.0-7, the dominant release modes for PDS IB are C1-L, C3-L, and C4-L, all late containment failures.

4.6.4 PDS 3H

The quantified CET for PDS 3H, as shown in Figure 4.0-8. The dominant release modes for PDS 3H are A1, C1-L, C3-L, and C4-L.

4.6.5 PDS IIA

PDS IIA involves small-small LOCAs with recirculation failure. As indicated in Figure 4.0-9, the dominant release modes for PDS IIA are C1-L, C3-L, and C4-L, all late containment failures.

4.6.6 PDS IIB

The quantified CET for PDS is provided in Figure 4.0-10. The dominant release modes for PDS IIH are C3-L and C4-L, all late containment failures.

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[5] Hot leg failures induced by high-temperature gases during core damage also cause the high pressure and low pressure injection systems to deliver water upon RCS depressurization.

4.6.7 PDS IIE

PDS IIE consists of small-small LOCA sequences in which coolant makeup is lost upon switchover to recirculation (prior to core damage). The containment sprays also fail at recirculation, furthermore, the fan coolers are not operational. Spray injection is assumed to continue until the RWST is depleted (that is, operator action to control spray flow is not considered in extending the time to core damage). Thus, for this PDS, the time to vessel breach and potential for early containment failure is not significantly extended. The results of the CET quantification illustrated in Figure 4.0-11 reflect failure of the sprays during core damage. The cavity is flooded before vessel breach, reducing the potential for high pressure melt ejection pressure loads. The dominant release modes for PDS IIE are NCFV, C5-L, and C6-L.

4.7 Radionuclide Release Characterization

This section describes the CET end state radionuclide release characterization for St. Lucie. The CET analysis described the containment damage states for the spectrum of severe accident progression paths. Inherent in the CET definition of top events are the associated fission product release and removal mechanisms, hence, source terms for the CET end states can be readily characterized. The release mechanisms include in- and ex-vessel release terms. The removal mechanisms include scrubbing by active removal systems (e.g., sprays) as may be implied by the PDS definition, pool decontamination by a flooded cavity, and deposition on structures by natural processes. The CET end states are categorized to denote differences in the estimated release fractions to the environment. Groups of PDS and CET end states combinations may be collapsed further to define the St. Lucie release categories. The basis for collapsing the CET end states into a limited number of release categories is similarity in the estimated radionuclide releases to the environment (i.e., source terms).

4.7.1 Release Fraction Characterization

Each CET end state represents a particular release event or a recovered, degraded core state that may be characterized according to its potential for fission product releases to the atmosphere, its timing of release initiation (relative to time of incipient core damage), and its release duration, all of which are important to the off-site consequence determination. These are referred to in this study as release modes. The term CET end state and release modes may be used interchangeably. CET end states describe the particular containment failure or recovered state; release modes imply fission product release characteristics of the progression paths in the CETs.

Table 4.0-4 summarizes the possible CET release modes for the spectrum of core melt accident sequences. This table lists the various CET release modes as early and late release events (relative to the time of core melting) and containment damage states (i.e., failure modes), including recovered states and release mechanisms (e.g., no CCI). Each release mode represents a release path from the fuel through the RCS and the containment atmosphere to the environment, should the containment ultimately fail. The release path (including the associated removal mechanisms) is related to a particular environmental source term.

The release of fission products to the environment is negligible as long as the containment function is maintained. With the containment integrity maintained, fission product concentration within the containment are reduced by natural removal processes, such that if the containment ultimately fails in the long term, the release to the environment will be significantly mitigated. In-containment natural and active removal processes are likely to reduce airborne fission products released from the fuel. The source terms are low and the release duration is likely to be extended.

Specific release modes provide a point of reference in the binning of these CET end states into release categories.

4.7.2 Source Term Estimates

The source terms for each CET end states are calculated using a combination of plant-specific MAAP and reference plant analysis of fission product releases from the fuel to the environment. Associated with each severe core damage accident are various in-vessel and ex-vessel fission-product release mechanisms. A series of MAAP calculations were examined to determine the retention fractions (or escape fraction) of fission product species along the transport paths from the fuel to the environment. The estimated escape fractions are then applied to similar CET release modes, to minimize the number of MAAP runs for source term determinations. For example, a particular PDS in which the sprays were operational, is used to estimate a spray removal fraction, that is then used for other PDSs not specifically analyzed with MAAP. These are supplemented by available information from NUREG/CR-4881 and NUREG/CR-4551 in order to provide release fractions of fission products into containment. These are applied for the spectrum of CET end states and severe core damage accidents for the various time periods of severe accident progression.

Fission product release from the fuel and retention fractions within the RCS and containment used in this study are derived from the MAAP runs and STCP calculations that include conservative values of the release from the fuel into the RCS and escape to the containment atmosphere. Removal from the containment atmosphere considers passive and active removal mechanisms such as natural deposition and scrubbing due to spray operation. MAAP tends to provide different distributions.

4.7.2.1 Release Mechanisms

A brief discussion of release mechanisms from the fuel and removal factors in containment that were considered in fission product release and transport calculations are provided below.

In-Vessel Release Considerations

- With core uncovery and heat up, oxidation of the zirconium produces heat and hydrogen. Heat produced by the exothermic reaction is on the order of, if not greater than, the amount released by decay heat - if sufficient steam is available.

- Burst releases from cladding rupture -- attributed to heatup of cladding -- are relatively small, mostly volatile fission product releases.

- Diffusional releases, that diffuse from rupture opening through the gaps, are relatively small.

- During melting process, various fission products evaporate at liquid surfaces; both structural and core material can be released.

- If RPV fails quickly, the highly volatile fission products will be released while the less volatile ones are retained.

- If RPV failure occurs late, fewer volatile fission products are released from the containment during the melt and these have less time to settle or be removed from the RPV atmosphere.

- In complete meltdown, releases continue after RPV melt-through.

Conditions important in RCS fission product retention include surface temperatures in the RCS, velocity of gases traveling through the RCS, and overall aerosol generation rate into the RCS.

Ex-Vessel Release Considerations

- Of particular importance in determining ex-vessel releases is the condition at vessel breach. If the RCS is in a pressurized state, significant amounts of aerosols will be generated at vessel breach. The amounts generated are not quantified in NUREG/CR-4881. NUREG/CR-4551 is used to provide information that supports the model.

- Corium-concrete interactions serve to generate more aerosols once the corium contacts the concrete floor. Various factors influence the release during corium-concrete interactions. They are:

  -- The composition and temperature of corium;

  -- The composition of concrete; and

  -- The amount of heat directed downward from the corium and extent of concrete penetration.

Removal In RCS and Containment

- RCS pressure prior to RPV failure. With a low RCS pressure condition, high steam velocities are implied, hence less time is allowed for fission product retention in the RCS. With a high-pressure RCS condition, the steam velocities are typically lower and hence, the fission product retention (not for noble gases) due to gravitational settling is increased.

- Spray operation in containment. Scrubbing of fission products airborne in containment during the in- and ex-vessel release periods from the fuel will significantly reduce the amounts available for release to the environment.

- Water in reactor cavity. The deep overlying water pool in the cavity as much as 16.12 ft. (MAAP maximum is 90% of the equivalent cavity height of 17.9 ft.) serves as a scrubber to remove aerosols. Either action of sprays and water pool would have mitigating effects on release from the fuel during core-concrete interactions.

4.7.2.2 Fission Product Release Calculations

The approach used to calculate the fission product release fractions for each end state is similar to what was used in the NUREG-1150 and NUREG/CR-4881 studies. The method uses fission product results from reference plant MARCH runs for a few specific scenarios, and extrapolates these results to fit the sequences for each end state that has been defined. For example, in the Zion NUREG-1150 studies, MARCH calculations for a transient and three small LOCAs were performed to find the data necessary to extrapolate for the spectrum of core damage and containment failure sequences. For low pressure scenarios, data from the Surry study was used. The data is extrapolated by multiplying the fission products released from the fuel (from the reference analysis) by the fractions associated with the various conditions defined by the sequences (i.e., from sprays, wet cavity, early or late containment failure). The spectrum of CET end states encompass the range of conditions for the various PDSs defined, and will therefore need only be calculated once for a point estimate result.

The calculations have been programmed on a computerized spread sheet that allows variations in CET end state bins, as well as the escape fractions associated with the retention mechanisms. A release fraction is the fraction that is released from the fuel (in- or ex-vessel) without any retention mechanisms. Escape fractions for retention mechanisms are the fraction of fission products released given the retention mechanism is available (i.e., sprays, cavity water pool scrubbing or settling in-containment prior to containment failure).

4.7.3 Approximate Source Term Model Formulation

The correlations used in calculating the release to the environment is based on a series of fission product source and removal terms in the vessel and the containment. The input parameters for release into the containment are derived from reference plant analysis summarized in NUREG/CR-4881 and NUREG/CR-4551 for the various fission product release components. These are applied directly for St. Lucie as appropriate. The removal terms (e.g., retention in the RCS or scrubbing by an overlying pool) are likewise derived from existing analysis. The values used are consistent with the availability of systems (e.g., sprays) defined by the accident progression paths modeled in the CETs. The fission product release model performs a table lookup, applying the appropriate release terms for the specified sequence. Fission-product releases are calculated for each of accident progression time phases and CET end states. To calculate radiological source terms and uncertainty ranges for each of these end states using extensive deterministic calculations could potentially be time consuming and expensive. The simplified methodology for source term uses a relative approach that considerably reduces the calculation requirement to determine the relative severity of the end states. The approach 'adjusts' values calculated for specific scenarios to values that apply for other scenarios that have similar characteristics.

The total release to the atmosphere for any given end state can be separated into four groups:

1. In-vessel;
2. Direct containment heating (DCH) and high pressure melt injection (HPME);
3. Core-concrete interaction (CCI); and
4. Delayed (Iodine and Cesium).

The combined release to atmosphere accounting for all four modes of release is

FCON = EFLEAK * (In-vessel release + Vessel breach release + CCI release + Late release + LIR)

where:

- EFLEAK — Escape fraction for leak containment failure (discriminate between early and late)
- Vessel Breach Release — The release due to high pressure melt ejection (HPME) and release due to direct containment heating (DCH)
- CCI Release — Release to atmosphere due to core-concrete interaction (CCI)
- Late Release — Release of iodine and cesium to atmosphere due to revolatilization
- LIR — Late iodine release from containment due to decomposition
- FCON — Release from containment to atmosphere

The final release for each fission product species (i) to the environment is obtained as the sum of all the release terms from the fuel during the accident progression.

In-Vessel Release. The in-vessel releases are those which are released from the core from the time core melt begins until the vessel fails. There is some retention in the RCS (particularly if the AFW system is operating and the steam generators capture most of the fission products released from the fuel into the vessel), as well as the containment, so everything that is released from the core is not released to the atmosphere at containment failure. The process begins when the core becomes uncovered and heats up, eventually leading to melting. Because the melting occurs on a region-by-region basis, the total release of any given fission product would occur over a period of time. In general, however, the more volatile radionuclides are released in the early heatup and melting. These releases enter the RCS, where a fraction is retained at vessel failure, while the rest enters the containment environment. At containment failure, some settling will have occurred, as well as washing out if sprays are operating, leading to more retention in the containment. The final release to atmosphere from what was released from the core in-vessel is calculated by the following equation:

In-Vessel Release = EFSPR(i)*EFAERCOR(i)*FCONV(i)*FVES(i)*FCOR(i)

where:

- EFSPR Fraction that is not retained (i.e., escape fraction) by sprays (for high and low pressure PDS)
- EFAERCOR Escape fraction for aerosol agglomeration uncertainties
- FCONV Fraction of in-vessel release that is released from the containment to atmosphere (early, late, and late with sprays failures)
- FVES Fraction of material released from fuel that is released from the vessel (high and low pressure)
- FCOR Fraction of initial core inventory released from the fuel prior to vessel failure

The (i) for each variable symbolizes each fission product species (i.e., Iodine, Cesium, etc). For certain end states, some of the variables in the equation are ignored or set equal to one. For example, no credit may be taken for sprays if the PDS defines spray failure; there is no fission product removal and EFSPR(i) is set to one.

Experimental data indicates that CsI will decompose to elemental iodine which will be released from containment, while the Cs will most likely be retained in the RCS. To account for this, FI2D, the amount of CsI decomposition specified, is used to calculate a new FVES for Iodine as follows:

FVES(I) = FI2D + (1-FI2D)*FVES(I)

where:

- FI2D Fraction of CsI decomposition

<u>DCH and HPME Release</u>. If the RCS has not been depressurized at vessel failure, then there could be a contribution to the source term from DCH and HPME. The fuel could be ejected in a process that would cause aerosol generation, as has been shown experimentally. If DCH occurs, the high temperature and fragmented debris will oxidize and lead to additional aerosol and radionuclide release from the fuel. The calculation for the contribution due to these processes is:

Vessel Breach Release = (1-FCOR(i))*FREJ*((EFHPME*RADEJL(i))+EFDCH*FR DH*-RADDHL(i)*(1-RADEJL(i)))

where:

- FREJ Fraction of melt ejected from vessel
- EFHPME Escape fraction for HPME
- RADEJL Radionuclide release fraction for HPME
- EFDCH Escape fraction for DCH
- FRDH Fraction of core participating in DCH
- RADDHL Radionuclide release for DCH

This effect is non-existent in most of the end states, so for all cases except early containment failure with high RCS pressure, FREJ is set equal to zero. It is assumed that if DCH and HPME contributed to a large release, it would also have contributed to a large enough pressure rise to fail containment early, so it is not considered for late containment failure end states. Furthermore, retention in the containment by natural removal processes would reduce the impact of this term.

<u>CCI Release</u>. Following vessel breach it is expected that the molten core debris falls into a concrete cavity and attacks the concrete, releasing decomposition gas products. Aerosols will then be released from the molten mass into the containment atmosphere.

The factors that influence the amount of release are the composition and temperature of the corium as it is released from the vessel. As the corium cools, the fission product release decreases to such an extent that it makes further release negligible. A water pool overlying the corium will retain some of the releases from the interaction, as well as help in cooling the debris.

The equation used to calculate the release due to CCI is:

CCI Release = EFSPRCCI (i) * EFCCIWCAV (i) * EFAERCCI (i) * FCCID (i)*FCONC (i)

where:

- EFSPRCCI Escape fraction for CCI with sprays on.
- EFCCIWCAV Escape fraction for CCI with wet cavity
- EFAERCCI Escape fraction for aerosol agglomeration uncertainties associated with CCI

- FCCID — Initial inventory that is released from the melt during CCI for dry cavity
- FCONC — CCI release that is released from the containment (FCONCE for early and FCONCL late containment failures)

For certain end states, some of the variables in the equation are ignored or set equal to one. For example, no credit may be taken for spray removal if the sprays are failed; hence, EFSPRCCI(i) and EFCCIWCAV(i) are set to one. For St. Lucie, the cavity is likely to be flooded and the deep overlying water pool provides an efficient scrubbing effect. If the end state is defined as having no CCI, then FCCID(i) is set equal to zero. Also the aerosol agglomeration uncertainties (AERCCI) are considered negligible if sprays are operating in the early containment failure end states, and so AERCCI is set to one for those cases.

For most of the accident scenarios identified at St. Lucie, the cavity configuration ensures that all of the water discharged to the containment (either from the RCS, or accumulators, and RWST through the injection and spray systems). As a result, the cavity will most likely be filled with water. MAAP would predict that the debris will be quenched. Therefore, core concrete interaction is precluded. However, there is uncertainty with regard to the possibility that the overlying water pool will not penetrate the core debris, and concrete attack may occur. The values used to estimate the extent of concrete attack, are obtained from reference plant (Zion) analysis, to simulate the CET branch point that considers core-concrete interaction.

If some of the core participated in HPME and DCH, the value of FCONCE should be reduced to reflect that less is available for CCI. The equation to show this is:

FCONCE(i) = (1-((FREJ * FRDH) + RADEJL* (FREJ-(FREJ * FRDH))))

It is assumed that this correction would be more relevant for early containment failures than for late.

<u>Delayed Release</u>. The fission products released from the fuel and deposited in the RCS may be evolved when the temperature of deposition sites increase with time. There are several mechanisms could lead to a delayed release for I and Cs. These include the following:

- Revolatilization from RCS;
- Slow deposition of initial releases from RCS;
- Radioactive decay chains;
- Re-suspension at containment failure; and
- Retention in melt until after vessel failure.

All of these mechanisms are lumped together as a delayed release for I and Cs which is calculated by the following equation:

Delayed Release = FLATE(i)*FCOR(i)*(1-FVES(i))*(1-FI2D)*EFSPRCCI(i) *EFA ERC-CI(i)

where:

- FLATE Late Revolatilization from the RCS.

Similar to the other types of releases mentioned, for specific end states, certain variables in the above equation are ignored or set equal to one. For end states with no sprays operating, EFSPRCCI is set equal to one. If the sprays are considered to be operating, EFAERCCI is set equal to one. To calculate the Cs delayed release, the portion of the equation relating to FI2D (1-FI2D) is ignored by setting FI2D to zero.

Total Release. The total release to atmosphere (i.e., environmental source terms) is calculated by summing the four above contributors. To allow for a lower release for end states with leakage as the containment failure mode, the escape fraction for leakage, EFLEAK or EFLEAKLS is multiplied by the total. In addition, a late Iodine release from containment variable (LIR) is also added to the total for Iodine. The equation is as follows:

Total Release = EFLEAK*(In-vessel release + DCH and HPME release + CCI release + Late release + LIR)

Table 4.0-5 lists the figures of merit for PWR source term calculations obtained from reference studies using MAAP and STCP. These values were used in the calculation of approximate source terms. Table 4.0-6 provides a summary of St. Lucie input constants based on MAAP results if available and suggested parameters that bound most of the accident scenarios derived from the reference studies and NUREG/CR-4551. The values shown are sufficient to calculate releases for the likely PDSs by substituting them into the methodology model formation.

4.7.4 CET Release Mode Source Terms

The fission product estimates for the CET release modes are calculated for St. Lucie using both plant-specific MAAP accident simulation calculations and existing reference plant analyses. The plant-specific feature of the St. Lucie cavity design which assures a very deep overlying pool, compared to that of the reference plant analyses, provide a higher scrubbing factor than was used for the reference plant studies. The higher scrubbing factor is reflected in the results summarized in Table 4.0-7. This table provides the source term estimates used in characterizing the severity of the CET end states. As expected, the source terms are lower for end states in which fission products are scrubbed by containment sprays. The ex-vessel releases are also reduced by an overlying pool.

4.8 Summary and Conclusions

This section summarizes the results of the containment performance analysis for St. Lucie Units 1 and 2. St. Lucie is a 2700 MWth Combustion Engineering PWR with a free-standing steel containment building with a free volume of 2.5 million cubic feet. Its design pressure is 44 psig and its failure pressure is estimated to be 95 psig. The accident progression analysis performed draws from plant-specific MAAP accident simulation calculations and the CET quantification of the likelihoods of containment challenges made extensive use of the results of the NUREG-1150 reference PWR plant with large dry containment.

The results of the analysis are summarized in Figures 4.0-4A and 4.0-4B. These figures display the conditional probability distribution of the various containment damage states; e.g., early containment failure, late containment failure (with and without core-concrete interactions) and no containment failures. On a frequency-weighted basis of the plant damage states, the estimated probability from internally-initiated core damage accidents for early containment failures is approximately 1% for both St. Lucie Units 1 and 2. Late containment failures contribute 15% for St. Lucie Unit 1 and 13% for St. Lucie Unit 2. Containment integrity is maintained for 72% of the core damage sequences for Unit 1 and 71% for Unit 2.

The following observations and conclusions may be drawn from an examination of the key results of this study.

Early Containment Challenge. The potential for early containment failure is important in the St. Lucie risk analysis as it also provides a measure of the potential consequences of a severely degraded core accident. The major contributors to early containment failure for St. Lucie include containment threats due to HPME loads from high RCS pressure core damage accidents, steam explosion events for low pressure sequences, and isolation failures. The threat to HPME loads is reduced for an initially high pressure sequence if RCS depressurization occurs prior to vessel breach as a result of hot-leg failure or seal LOCAs. A sensitivity analysis of HPME loads using MAAP, with very conservative assumptions proved that direct containment heating can pose a threat to containment integrity only if dispersal of the molten material to the upper containment region is assumed. Forcing hydrogen burning during HPME also contributed to significant pressure loads that would likely challenge containment integrity. However, depressurization of the RCS (due to temperature induced hot-leg failure) precluded high pressure melt ejection at vessel failure. This would likely reduce the likelihood of early containment failure. For the baseline cases, no early containment failures were predicted based on MAAP simulation results. Probability of early containment failures of 1% stems from the uncertain phenomological modeling embedded in the CET logic (e.g., ex-vessel steam explosion, vessel acting as a rocket).

Severe Accident Loads. In general, the large dry containment design of St. Lucie proved to be robust, thus pressure loads were not likely to lead to containment failure. The containment failure time, if the containment did fail, was greatly delayed relative to the time of core damage. The major contributor to late containment failures is steam overpressure in long term (hydrogen burning is likely to be precluded due to the steam-inerted containment atmosphere). It is noteworthy that the accident progression analysis performed for this study indicates that the containment floor and cavity configuration for St. Lucie will, in most cases, allow the RCS (and accumulator) water inventory to collect in the cavity thus providing an overlying water pool above the debris following vessel breach. Core concrete interaction is mitigated and debris quenching with attendant boil-off of the water would cause the containment pressure to increase to failure.

Systems-Related Functions. It was found in this study that by far the most effective measures were systems operation either to prevent core melt from progressing to vessel breach, to protect containment integrity, and to mitigate the potential consequences by reducing the source terms. Systems survivability is, of course, crucial in order to continue operation as to preclude containment failure. This principal observation confirms and continues to reinforce conclusions and that containment safeguards availability and continued operation would effectively mitigate, in many cases preclude, containment failure and significant releases to the environment. Availability of the sprays in the long term would result in steam condensation and heat removal from the containment, thus maintaining containment integrity.

The large dry containment of St. Lucie was found to be less susceptible to early containment failures given a severe accident. This study shows a high likelihood of maintaining containment integrity during the early phases of the accident progression. The uncertainties in the determination of these pressure loads, and mitigation of the potential loads due to RCS depressurization, however, are high. The total conditional probability of early containment failure for St. Lucie (1%) is lower than the reference PWR (13%), as the reference PWR Surry study includes containment bypass (contributing the major portion (0.12) of the likelihood of early failures). The Zion study indicates a comparable probability of no containment failure (0.73) to that for St. Lucie. The combined large containment volume (2.6 million cubic feet) and estimated failure pressure (95 psig), provide considerable capability to withstand loads.

This study demonstrated the inherent capability of the St. Lucie containment to survive pressure loads during the early phases of a severe accident and to mitigate the long term effects of severe accident progression. It has provided a valuable means of determining which core damage sequences are likely to be potential risk contributors. It has also provided insights as to the accident conditions, modeling parameters or systems (or operator actions) that can strongly affect the MAAP 3.0B code predictions and thereby ascertaining where a more realistic characterization of the plant is most needed.

The results of the Level 2 analysis, therefore, demonstrate that there are no unique failure mechanisms or vulnerabilities for the St. Lucie Unit 1 and Unit 2 containments.

4.9 References for Section 4.0

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3. PSL PRA 5.0, Revision 0, "*Integration and Quantification Report for St. Lucie Units 1 & 2 Probabilistic Risk Assessment*".

4. NUREG-1150, "*Reactor Risk Reference Document*", U.S.N.R.C., Draft Report, February 1987.

5. NUREG/CR-4881, "*Fission Product Release Characteristics Into Containment Under Design Basis and Severe Accident Conditions*", March 1988.

6. NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*", June 1983, Vols. 1 & 2 and Appendices.

TABLE 4.0-1A
ST. LUCIE UNIT 1 DOMINANT PLANT DAMAGE STATES FREQUENCY

Plant Damage Status (ID)	Mean Frequency Per Year
3B	3.93E-6
VB	2.97E-6
IB	2.76E-6
3H	2.72E-6
CB	1.74E-6
IIA	1.73E-6
IIB	1.54E-6
IIE	1.49E-6
IIF	9.31E-7
VIA	7.01E-7
VIB	6.31E-7
IIR	4.70E-7
VIE	4.22E-7
IR	2.87E-7
IVB	2.38E-7
PDS Total	2.26E-5

TABLE 4.0-1B
ST. LUCIE UNIT 2 DOMINANT PLANT DAMAGE STATES FREQUENCY

Plant Damage Status (ID)	Mean Frequency Per Year
3B	4.46E-6
VB	2.86E-6
CB	2.72E-6
3H	2.64E-6
IB	2.39E-6
IIA	1.86E-6
IIB	1.71E-6
IIE	1.51E-6
VIB	1.08E-6
IIR	7.94E-7
VIA	7.18E-7
IIF	5.34E-7
VIE	4.25E-7
IVH	1.32E-7
PDS Total	2.39E-5

Table 4.0-2

CET QUANTIFICATION PROBABILITY RANGES

Description	Range
Certain	P = 1.0
Highly Likely	1 > P > 0.995
Very Likely	0.995 > P > 0.95
Likely	0.95 > P > 0.70
Indeterminate	0.70 > P > 0.30
Unlikely	0.30 > P > 0.05
Very Unlikely	0.05 > P > 0.005
Highly Unlikely	0.005 > P > 0
Impossible	P = 0

Source: NUREG/CR-4551, draft July 1989

Table 4.0-3

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
DP	HOP-DP	3	Operator fails to depressurize RCS. Quantified as certain on the basis of the lack of emergency operating procedures that direct operators to depressurize the RCS beyond core damage.
	PRHLSLOK	2	Hot leg and surge line remain intact. Determined by creep rupture of hot legs from RCS heat-up. The probability is influenced by the RCS pressure. PDSs with high RCS pressure have the lowest likelihood of not failing the hot leg relative to the medium pressure PDSs (PRSLOK-1), or depressurized high pressure PDSs (PRSLOK2).
	PRHSLOK1	2	Hot leg and surge line remain intact given medium pressure PDS.
	PRHSLOK2	2	Hot leg and surge line remain intact given depressurized (medium pressure) RCS.
	QHP	3	Sequence is a high pressure PDS.
	QMP	3	Sequence is a medium pressure PDS.
	PRCSRVS	2	SRVs are not stuck open. Determined by failure of relief valves as Influenced by degraded conditions.
	PRSEALOK	2	RCP seal remains intact. Determined by the particular seal design, influenced by harsh environment.
	PRSGOK	2	Steam generator tubes do not rupture. Determined by scenario and influenced by accident phenomena.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
REC	SACPOWER	3	AC power not restored or available. For SBO sequences, this is quantified based on the power recovery curve. For non-SBO sequences, this basic event assumes that power is available, hence recovery of systems would depend on human intervention or removal of conditions that preclude operation (e.g., RCS pressure above the pump shutoff pressure).
	SALT-SIS1	3	Alternative systems not recovered during core melt. Determined by scenario.
	SHP-SIS1	3	High pressure ECCS not recovered during core melt. For St. Lucie, there are no high pressure injection systems capable of injecting into the vessel at the SRV setpoint pressure.
	SLP-SIS1	3	Low pressure ECCS not recovered during core melt. Determined by the accident scenario. Considered likely for high pressure PDSs in which the low pressure (LP) injection systems are not failed, and RWSP is not discharged
VF	PR-HT-TRAN	2	No ex-vessel heat transfer established. Considered to be a possible, given the cavity configuration and bottom head design of St. Lucie that ensures flooding. Quanti-fied only if the coolant is recovered in-vessel.
	WETCAVITY	3	Wet cavity flag. (Complement of cavity is dry.) This is determined by the PDS. The cavity and sump arrange-ment assures that the cavity will always be flooded as long the RWSP is discharged.
	PRCOOLDBIV	1	Coolable debris bed not formed in-vessel. This is considered to be a generic uncertain issue.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	SALTCONSYS	3	Alternative systems not available for ex-vessel cooling. Alternative system for flooding the cavity is considered in this basic event. For St. Lucie, the cavity will be flooded provided the RWSP is discharged. This event is not quantified at this time.
CFE	PRCI	3	Containment isolation failure. Quantified by the containment isolation systems models.
	PRALHPAH	1	Alpha event occurs given RCS at high pressure. Considered to be a generic uncertain issue.
	PRALPHAL	1	Alpha event occurs given RCS depressurized. Considered to be a generic uncertain issue.
	PRROCKET	1	Vessel acts as rocket and fails containment. Considered to be a generic uncertain issue.
	PRWSWFLP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., low RCS pressure, fan coolers and sprays on with wet cavity. Determined from scoping calculations of containment pressure.
	PRWSNFLP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., low RCS pressure, no fan coolers and sprays on with wet cavity). Determined from scoping calculations of containment pressure.
	QNOFAN	3	Fan coolers not operating. Quantified as true or false depending on PDS definition.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	SNOSPRAY1	3	Containment sprays not operating before vessel breach. Quantified as true or false depending on PDS definition.
	PRNSWFLP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., low RCS pressure, fan coolers and sprays off with wet cavity. Determined from scoping calculations of containment pressure.
	PRWSNFLP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., low RCS pressure, no fan coolers and sprays on with wet cavity). Determined from scoping calculations of containment pressure.
	PRNSWFHP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., high RCS pressure, fan coolers and sprays off with dry cavity. Determined from scoping calculations of containment pressure.
	PRNSNFHP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., high RCS pressure, no fan coolers and no sprays with wet cavity). Determined from scoping calculations of containment pressure.
	PRWSWFHP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., high RCS pressure, with fan coolers and sprays on with wet cavity). Determined from scoping calculations of containment pressure.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	PRWSNFHP>U	2	Containment pressure > ultimate pressure (P_{ult}) given conditions at vessel breach (e.g., high RCS pressure, no fan coolers and sprays on with wet cavity). Determined from scoping calculations of containment pressure.
	PREVSE	1	Ex-Vessel steam explosion occurs.
	PRCFE_EVSE	2	Containment fails given ex-vessel steam explosion occurs. Influenced by cavity geometry and water pathways to containment wall.
	QWETCAV	3	Wet cavity flag. Quantified as true or false depending on the PDS.
	SNOSPRAY1	3	Containment sprays do not operate before vessel breach.
DC	PRCDB-HP	2	Coolable debris bed doesn't form given HPME. Generic uncertain issue, influenced by plant feature that promotes debris dispersal
	PRCDB-LPSE	2	Coolable debris bed doesn't form given EVSE occurs and no HPME
	PREVSE	1	Ex-vessel steam explosion occurs.
	SNOLPI	3	Loss of low pressure injection (LPI) injection through failed vessel. Relevant only for scenarios in which coolant injection was recovered or available subsequent to vessel breach. Quantified by increasing failure rate of the systems models, reflecting adverse environment and NPSH requirements.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	SNOSPRAY2	3	Containment sprays do not operate after VB. (PDS dependent)
	PRDESTNCFE	3	Sprays failed given no CFE (early containment failure). Relevant only for scenarios in which the sprays are operating subsequent to vessel breach. Quantified by increasing failure rate of the systems models, reflecting adverse environment.
	PRDEST-CFE	3	Sprays failed given CFE (early containment failure). Relevant only for scenarios in which the sprays are operating subsequent to vessel breach. Quantified by considering the possibility that NPSH requirements may be jeopardized and increasing failure rate of the systems models, reflecting adverse environment.
	PRCRUST	1	Impervious crust forms precluding coolable debris. Generic uncertain issue.
	QWETCAVITY	3	Cavity is wet. Flag for wet cavity condition, determined by PDS definitions of RWSP discharge to the containment prior to vessel breach.
CFL	CDHR-PASS	2	Decay heat rate exceeds passive heat transfer to containment heat sinks. Considered relevant for slow-developing accident sequences, in which active heat removal is lost late in the scenario.
	CHGAVB	2	Heat generation exceeds removal after vessel breach. Relevant for accident sequences in which the active containment heat removal systems capacity is reduced, and/or metal water reaction exacerbates heat generation in the debris. Quantified by scoping calculations of heat generation and heat loss through fans or sprays.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	PRHB2	2	H_2 burn occurs given AC power and sprays recovered. Hydrogen generated during core melt in-vessel burns late (long after vessel breach).
	PRPR2	2	Pressure rise due to H_2 burn at start of CCI fails containment (late burning of H2 generated from in-vessel zircaloy oxidation). Scoping calculations determine the pressure rise.
	PRHB3	2	Late H_2 burn occurs given AC power and sprays. Hydrogen generated during CCI is also considered in this basic event. This is influenced by the conditions in containment (e.g., steam inerting and hydrogen concentration).
	PRPR3	2	Pressure rise due to late H2 burn > P_{ult}. Determined by scoping calculations of pressure based on adiabatic combustion.
	PRHB4	2	Late H_2 burn occurs (given no sprays). This includes hydrogen generated during CCI. Steam inerted condition may be precluded with sprays off. This is influenced by the conditions in containment (e.g., steam inerting and hydrogen concentration).
	PRPR4	2	Pressure rise due to late H2 burn > P_{ult}. Determined by scoping calculations of pressure based on adiabatic combustion.
	QNOFAN	3	Fan coolers not operating. Determined by PDS

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
	PRMT1	2	Melt through occurs (before over-pressure failure) given debris not coolable but sprays on. Determined by scoping calculations of concrete decomposition and non-condensible gas generation that would result in containment overpressure failure prior to basemat penetration.
	PRMT2	2	Melt through occurs (before over-pressure failure) given debris not coolable and no sprays. Determined by scoping calculations of concrete decomposition and non-condensible gas generation that would result in containment overpressure failure.
	PRNCG-FAIL	2	Non-condensible gas generation fails containment. Determined by scoping calculations of concrete decomposition and non-condensible gas generation that would result in containment overpressure failure.
	SACSPREC	3	AC power and sprays recovered early
	SACSPRECL	3	AC power and sprays recovered late
	SNOSPRAY2	3	Containment sprays do not operate after VB. Defined by PDS
	QWETCAVITY	3	Cavity is wet. Flag for wet cavity condition, determined by PDS definitions of RWSP discharge to the containment prior to vessel breach.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
FPR	PRHEATUP	2	RCS/containment heatup causes revolatilization. Influenced by the RCS condition. High pressure PRHEATUPHI) or low pressure (PRHEATUPLO) PDSs are considered. Flushing of the RCS due to two effective holes in the vessel (i.e., bottom head failure and (LOCA) is considered in the revolatilization issue for low RCS pressure PDSs.
	QRCS-RET	2,3	RCS retention does not occur. Influenced by PDS definition related to the status of the RCS and steam generators.
	PRNO-POOL	2	No overlying pool. Determined by the wet cavity flag.
	NO-SPR-CM	3	Containment sprays do not operate during core melt. Considered in the removal of volatile fission products released during core melting. Determined by PDS.
	PRNCG-FAIL	2	Non-condensible gas generation fails containment. Determined by scoping calculations of concrete decomposition and non-condensible gas generation that would result in containment overpressure failure.
	SACSPREC	3	AC power and sprays recovered early
	SACSPRECL	3	AC power and sprays recovered late
	SNOSPRAY2	3	Containment sprays do not operate after VB. Defined by PDS
	QWETCAVITY	3	Cavity is wet. Flag for wet cavity condition, determined by PDS definitions of RWSP discharge to the containment prior to vessel breach.

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

CET Event Node	Basic Event Name	Basic Event Type	Description and Comments
FPR	PRHEATUP	2	RCS/containment heatup causes revolatilization. Influenced by the RCS condition. High pressure PRHEATUPHI) or low pressure (PRHEATUPLO) PDSs are considered. Flushing of the RCS due to two effective holes in the vessel (i.e., bottom head failure and (LOCA) is considered in the revolatilization issue for low RCS pressure PDSs.
	QRCS-RET	2,3	RCS retention does not occur. Influenced by PDS definition related to the status of the RCS and steam generators.
	PRNO-POOL	2	No overlying pool. Determined by the wet cavity flag.
	NO-SPR-CM	3	Containment sprays do not operate during core melt. Considered in the removal of volatile fission products released during core melting. Determined by PDS.
	SNOSPRAY2	3	Containment sprays do not operate after VB. Considered in the removal of fission products released subsequent to vessel breach.
CFM	PR-RUPWCFE	1	Containment fails early by rupture due to other than alpha or rocket
	PR-RUPWCFL	1	Containment fails late by rupture mode
	PRALHPAH	1	Alpha event occurs given RCS at high pressure
	PRALPHAL	1	Alpha event occurs given RCS depressurized
	PRROCKET	1	Vessel acts as rocket and fails containment

Table 4.0-3 (continued)

LOGIC TREE BASIC EVENT AND ISSUES

KEY

1. Phenomena related subject to large uncertainties
2. Phenomena related influenced by plant specific features
3. System related issues defined by PDS

Table 4.0-4

DESCRIPTION OF CET RELEASE MODES

Release Modes	CET Sequence Description
NCF(AO)	Recovered in-vessel, no containment failure
A1	Recovered in-vessel, late containment failure, in-vessel fission product release mitigated
A2	Recovered in-vessel, late containment failure, in-vessel fission product release not mitigated
NCF(BO)	Recovered ex-vessel, no containment failure
B1	Recovered ex-vessel, late containment failure, in-vessel fission product release mitigated
B2	Recovered ex-vessel, late containment failure, in-vessel fission product release not mitigated
B3	No CCI, late containment failure, in-vessel fission product release mitigated by sprays
B4	No CCI, late containment failure, in-vessel fission product release not mitigated
B5	No CCI, late containment failure, in-vessel fission product release mitigated by sprays
B6	No CCI, late containment failure, in-vessel and late fission product release not mitigated
NCF(CO)	CCI occurs, no containment failure
C1	CCI occurs, late containment failure, ex-vessel fission product release mitigated by overlying pool, in-vessel release mitigated by sprays
C2	CCI occurs, late containment failure, ex-vessel fission product release mitigated by overlying pool, in-vessel release not mitigated
C3	Significant CCI occurs, late containment failure, in- and ex-vessel fission product release mitigated by sprays
C4	Significant CCI occurs, late containment failure, in- and ex-vessel fission product release not mitigated
C5	Moderate CCI occurs, late containment failure, in- and ex-vessel fission product release mitigated by sprays
C6	Moderate CCI occurs, late containment failure, in- and ex-vessel fission product release not mitigated
D1	No CCI, early containment failure, in-vessel fission product release mitigated
D2	No CCI, early containment failure, in-vessel fission product release not mitigated
D3	No CCI, early containment failure, in-vessel and late fission product release mitigated
D4	No CCI, early containment failure, in-vessel and late fission product release not mitigated

Table 4.0-4 (continued)

DESCRIPTION OF CET RELEASE MODES

Release Modes	Description of CET Release Modes
E1	Significant CCI occurs, early containment failure, in- and ex-vessel fission product release mitigated
E2	Significant CCI occurs, early containment failure, ex-vessel fission product release mitigated by overlying pool, in-vessel fission product release not mitigated
E3	Significant CCI occurs, early containment failure, in- and ex-vessel fission product release mitigated by sprays
E4	Significant CCI occurs, early containment failure, fission product release not mitigated
E5	Moderate CCI occurs, early containment failure, in- and ex-vessel fission product release mitigated by sprays, no late fission product release
E6	Moderate CCI occurs, early containment failure, ex-vessel and late fission product release not mitigated

NOTE: The release modes are further characterized as Leakage (L) or Rupture (R) to indicate the duration of fission product release to the environment.

Table 4.0-5

Release Fractions From Various Studies on Reference Plants

PLANT	SEQUENCE	REPORT	SIMULATION CODE	FRACTION DEFINITION	FISSION PRODUCT RELEASE NG	I	CS	TE	SR
SURRY	AG	NUREG-4881	STCP	FCOR	1	1	1	0.86	0.001
				FVES	1	0.87	0.87	0.83	0.75
				FRCS	1	0.87	0.87	0.71	9.0E4
				FCCI	0	1.5E-4	1.6E-4	0.017	0.09
		BMI 2139	STCP	FCON	0	0.58	0.57	0.47	0.0048
	V	NUREG-4881	STCP	FCOR	1	1	1	0.63	0.0015
				FVES	1	0.62	0.6	0.25	0.35
				FRCS	1	0.62	0.6	0.16	5.0E-4
				FCCI	0	0	0	0.06	0.33
		EPRI 4096	STCP	FCON	1	0.06	0.06	0.02	0.008
ZION	S3DCr	NUREG-4881	STCP	FCOR	0.99	0.99	0.99	0.43	4.0E-4
				FVES	1	0.28	0.28	0.47	0.34
				FRCS	0.99	0.27	0.27	0.2	1.4E-4
				FCCI	0	0.004	0.004	0.22	0.32
				DFPOOL	0	13	14	13	14
		BMI-2139	STCP	FCON	1	2.9E-6	1.6E-4	0.0019	9.0E-4
		EPRI 6111	MAAP	FCON	1	0.006	0.0078	0	2.7E-6
	S2DCr(2)	EPRI 6111	MAAP	FCON	1	0.055	0.11	0	1.5E-4
	S2DCF1	NUREG-4881	STCP	FCOR	0.99	0.99	0.99	0.43	4.5E-4
				FVES	1	0.28	0.28	0.47	0.34
				FRCS	0.99	0.27	0.27	0.2	1.4E-4
				FCCI		0.004	0.004	0.4	0.1
				DFPOOL		2.25	1.5	1.2	2
		BMI-2139	STCP	FCON		0.22	0.22	0.32	0.037
		EPRI 6111	MAAP	FCON	0.95	0.081	0.16	0	4.0E-4
	S2DCF2	NUREG-4881	STCP	FCOR	0.99	0.99	0.99	0.43	4.0E-4
				FVES	1	0.28	0.28	0.47	0.34
				FRCS	0.99	0.27	0.27	0.2	1.4E-4
				FCCS	0	0.0035	0.004	0.28	0.34
				DFPOOL		19.7	17.3	4.8	17
		BMI-2139	STCP	FCON		0.026	0.027	0.034	0.0025
		EPRI 6111	MAAP	FCON	1.6E-4	7.5E-6	1.1E-5	0	1.9E-8
SURRY	TMLB	NUREG-4881	STCP	FCOR	0.98	0.98	0.98	0.46	7.0E-4
				FVES	1	0.22	0.21	0.62	0.26
				FRCS	0.98	0.22	0.21	0.28	2.0E-4
				FCCI	0	0.02	0.02	0.12	0.17
				DFPOOL	2.3	2	1.5	2.4	2.4
			EPRI 4096	FCON	0.004	0.002	0.002	4.0E-4	2.0E-4

Table 4.0-5 (continued)

RELEASE FRACTIONS FROM VARIOUS STUDIES ON REFERENCE PLANTS

PLANT	SEQUENCE	REPORT	SIMULATION CODE	FRACTION DEFINITION	FISSION PRODUCT RELEASE NG	I	CS	TE	SR
ZION	TMLU	NUREG-4881	STCP	FCOR	1	1	1	0.54	0.002
				FVES	1	0.22	0.19	0.47	0.16
				FRCS	1	0.22	0.19	0.25	3.0E-4
				FCCI	0	0.001	0.001	0.17	1.4E-4
				DFPOOL	0	16	9	10	11
		BMI-2139	STCP	FCON		0.0057	0.0064	0.04	9.4E-5
		EPRI 6111	MAAP	FCON	0.99	0.015	0.034	0	0.001
	TMLU(2)	EPRI 6111	MAAP	FCON	0.98	5.1E-4	0.0013	0	2.2E-5

Sequence Description:

AG: A large hot leg break LOCA accompanied by failure of containment heat removal system; the emergency core cooling and containment spray systems are available.

V: Interfacing systems LOCA with containment bypass.

S2DCr: A LOCA initiated by rupture of primary coolant system accompanied by failure of the emergency core cooling injection as well as containment spray recirculation systems. Fan coolers are initially operable, but are assumed to fail at the time of vessel failure. Late overpressure failure has been selected as the containment failure mode.

S2DCr(2): Same as above with early containment failure.

S2DCF1: A LOCA initiated by primary pump seal rupture accompanied by failures of emergency core cooling, containment sprays as well as containment coolers. An early containment failure mode due to hydrogen combustion and/or direct heating.

TMLB: Failure of power conversion and auxiliary feedwater systems given the initiating transient event of loss of off-site AC power.

TMLU: Initiated by a transient and is accompanied by the loss of power conversion, auxiliary feedwater and emergency core cooling systems, both containment coolers and sprays are available. Early containment failure due to direct heating.

TMLU(2): Same as above with late containment failure.

Table 4.0-6

ST. LUCIE RELEASE CALCULATIONS INPUT CONSTANTS

0.73	= FREJ	Fraction of melt ejected from vessel
0.75	= FRDH	Fraction of core participating in DCH
0.8	= EFDCH	Escape fraction for DCH
0.8	= EFHPE	Escape fraction for HPME
0.33	= FI2D	Amount of CsI decomposition
0.05	= LIR	Late iodine release fraction

	NG	I	CS	TE	SR
EFAERCCI	1	0.33	0.33	0.33	0.33
EFAERCOR	1	0.33	0.33	0.33	0.33
EFCCIWCAV	1	0.2	0.167	0.176	0.4
EFSPRLP	1	0.033	0.033	0.033	0.033
EFSPRHP	1	0.33	0.33	0.33	0.33
EFSPRCCI	1	0.033	0.033	0.033	0.033
EFLEAKE	1	1	1	0.4	0.2
EFLEAKL	1	1	1	0.4	0.2
FCOR	1	1	1	0.54	0.0017
FVESH	1	0.27	0.28	0.48	0.16
FVESL	1	0.87	0.95	0.95	0.0
FCONVL	0.8	0.3	0.3	0.3	0.3
FCONVLS	0.8	0.003	0.003	0.003	0.003
FCONVE	1	0.8	0.8	0.8	0.8
FCCID	0	0.002	0.003	0.34	0.05
FCONCE	0	0.7	0.7	0.47	0.7
FCONCL	0	0.03	0.13	0.4	0.2
RADEJL	1	1	1	0.1	0.01
RADDHL	1	1	1	1	0.2
FLATE	-	0.23	0.23	1.0	1

Table 4.0-6 (continued)

INPUT CONSTANTS

Variable Name	Definition
EFAERCCI	Escape fraction for CCI aerosol agglomeration uncertainties
EFAERCOR	Escape fraction for FCOR aerosol agglomeration uncertainties, all FCOR
EFCCIWCAV	Escape fraction for CCI with wet cavity
EFSPRLP	Escape fraction associated with CCI with sprays on
EFSPRHP	Escape fraction for in-vessel release with sprays on (high pressure sequence)
EFSPRCCI	Escape fraction for in-vessel release with sprays on (low pressure sequence)
EFLEAKE	Escape fraction for leakage failure mode (early)
EFLEAKL	Escape fraction for leakage failure mode (late)
FCOR	Fraction of the initial core inventory released from the fuel prior to vessel failure
FVESH	Fraction of in-vessel release which is released from the RCS for high pressure sequences
FVESL	Fraction of in-vessel release which is released from RCS for low pressure sequences
FCONVL	Fraction of in-vessel release which is released to atmosphere for late containment failure
FCONVLS	Fraction of in-vessel release which is released to atmosphere for late containment failure w/sprays
FCONVE	Fraction of in-vessel release which is released to atmosphere for early containment failure
FCCID	Fraction of initial inventory released from the melt during CCI for dry cavity cases
FCONCE	Fraction of CCI release that is released from the containment for early containment failure cases
FCONCL	Fraction of CCI release that is released from the containment for late containment failure cases
RADEJL	Radionuclide release fractions for HPME
RADDHL	Radionuclide release fractions for DCH
FLATE	Late revolatilization fraction from the RCS

Table 4.0-7 Fission Product Releases Associated with St. Lucie Release Modes

RESULTS:

FISSION PRODUCTS RELEASED TO ENVIRONMENT

	NG	I	CS	TE	SR
A1	8.00E-01	2.98E-05	3.10E-05	6.70E-06	0.00E+00
A2	8.00E-01	9.04E-02	9.41E-02	2.03E-02	0.00E+00
B1	8.00E-01	5.07E-02	4.11E-04	6.70E-06	0.00E+00
B2-L	8.00E-01	1.47E-01	9.78E-02	2.03E-02	0.00E+00
B2-R	8.00E-01	1.40E-01	9.78E-02	5.08E-02	0.00E+00
B3-L	8.00E-01	5.07E-02	4.11E-04	6.70E-06	0.00E+00
B3-R	8.00E-01	5.07E-02	4.11E-04	1.68E-05	0.00E+00
B4-L	8.00E-01	1.47E-01	9.78E-02	2.03E-02	0.00E+00
B4-R	8.00E-01	1.47E-01	9.78E-02	5.08E-02	0.00E+00
B5-L	8.00E-01	5.37E-02	5.47E-03	3.39E-06	1.78E-09
B5-R	8.00E-01	5.37E-02	5.47E-03	8.47E-06	8.89E-09
B6-L	8.00E-01	1.38E-01	8.24E-02	1.03E-02	5.39E-06
B6-R	8.00E-01	1.38E-01	2.78E-02	2.57E-02	2.69E-05
C1-L	8.00E-01	5.07E-02	4.13E-04	3.23E-04	2.64E-05
C1-R	8.00E-01	5.07E-02	4.13E-04	8.07E-04	1.32E-04
C2-L	8.00E-01	1.47E-01	9.80E-02	3.83E-02	6.60E-04
C2-R	8.00E-01	1.47E-01	9.80E-02	9.57E-02	3.30E-03
C3-L	8.00E-01	5.07E-02	4.13E-04	3.23E-04	2.64E-05
C3-R	8.00E-01	5.07E-02	4.13E-04	8.07E-04	1.32E-04
C4-L	8.00E-01	1.47E-01	9.80E-02	3.83E-02	6.60E-04
C4-R	8.00E-01	1.47E-01	9.80E-02	9.57E-02	3.30E-03
C5-L	8.00E-01	5.37E-02	5.48E-03	3.19E-04	2.64E-05
C5-R	8.00E-01	5.37E-02	5.48E-03	7.98E-04	1.32E-04
C6-L	8.00E-01	1.38E-01	8.25E-02	2.82E-02	6.65E-04
C6-R	8.00E-01	1.38E-01	8.25E-02	7.05E-02	3.33E-03
D1-L	1.00E+00	7.48E-02	2.55E-02	5.42E-03	0.00E+00
D1-R	1.00E+00	7.48E-02	2.55E-02	1.35E-02	0.00E+00
D2-L	1.00E+00	3.11E-01	2.62E-01	5.42E-02	0.00E+00
D2-R	1.00E+00	3.11E-01	2.62E-01	1.35E-01	0.00E+00
D3-L	1.00E+00	1.89E-01	7.94E-02	1.11E-01	1.85E-02
D3-R	1.00E+00	1.89E-01	7.94E-02	2.77E-01	9.25E-02
D4-L	1.00E+00	5.71E-01	3.90E-01	1.66E-01	1.85E-02
D4-R	1.00E+00	5.71E-01	3.90E-01	4.16E-01	9.26E-02
E1-L	1.00E+00	7.48E-02	2.55E-02	7.53E-03	2.31E-04
E1-R	1.00E+00	7.48E-02	2.55E-02	1.88E-02	1.16E-03
E2-L	1.00E+00	3.12E-01	2.64E-01	1.18E-01	7.00E-03
E2-R	1.00E+00	3.12E-01	2.64E-01	2.95E-01	3.50E-02
E3-L	1.00E+00	7.48E-02	2.55E-02	7.53E-03	2.31E-04
E3-R	1.00E+00	7.48E-02	2.55E-02	1.88E-02	1.16E-03
E4-L	1.00E+00	3.12E-01	2.64E-01	1.18E-01	7.00E-03
E4-R	1.00E+00	3.12E-01	2.64E-01	2.95E-01	3.50E-02
E5-L	1.00E+00	1.89E-01	7.94E-02	1.12E-01	1.86E-02
E5-R	1.00E+00	1.89E-01	7.94E-02	2.79E-01	9.30E-02
E6-L	1.00E+00	5.72E-01	3.90E-01	1.94E-01	2.17E-02
E6-R	1.00E+00	5.72E-01	3.90E-01	4.85E-01	1.08E-01

Analysis steps in quantification of containment failure due to HPME

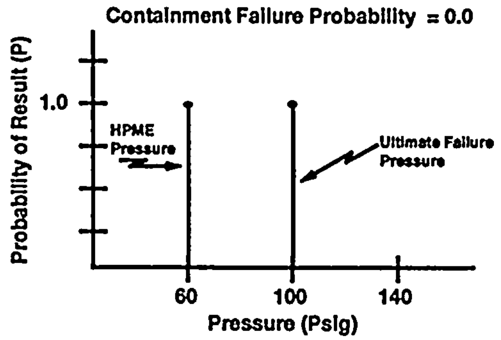
Example Plant Parameters:

- Containment Failure Pressure = 100 psig;
- Containment Design Pressure = 40 psig.

Bounding Calculations of Containment Loads Provide:

CASE 1 (P < 2 Times Design Pressure)

- If Calculated HPME Pressure = 60 psig; then
- Assume Failure Probability = 0.0;

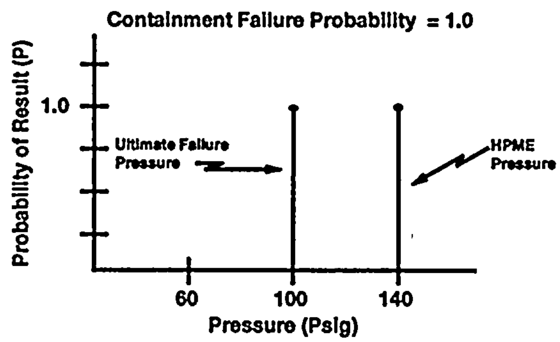


CASE 1 Bounding Assumption: $P_{HPME} << P_{failure}$

OR

CASE 2 (P > 3 Times Design Pressure)

- If Calculated HPME Pressure = 140 psig; then
- Assume Failure Probability = 1.0;

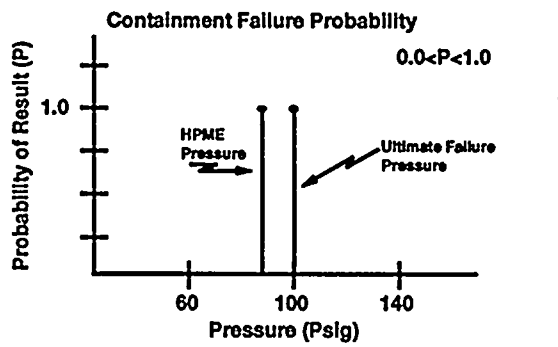


CASE 2 Bounding Assumption: $P_{HPME} > P_{failure}$

OTHERWISE

CASE 3 $(2 * P_d < P < 3 * P_d)$

- If Calculated HPME Pressure = 90 psig $(2 * P_d < P < 3 * P_d)$; Consider Containment Fragility Curve; Uncertainty in HPME Pressure; and $1.0 \leq$ Failure Probability ≤ 0.0



CASE 3 Uncertainty in Containment Failure Due to HPME

F:\USERS\DMS\PSL\VL2\LVL2FIG1.DRW

Figure 4.0-1. CET Quantification: Containment Failure Due to HPME; Cases 1, 2 and 3

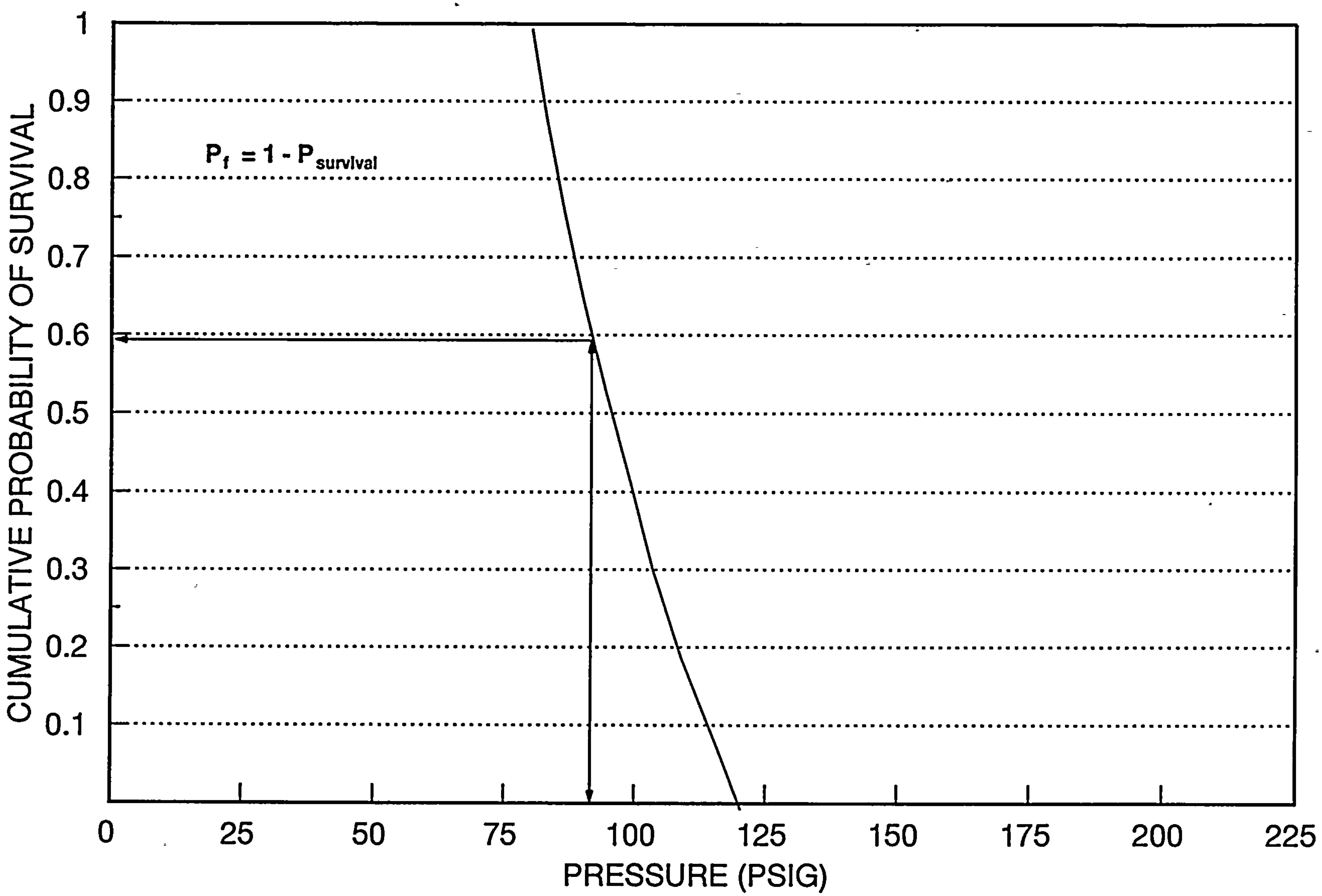


Figure 4.0-2 Example Illustration of Quantifying Probability of Containment Failure

F:\USERS\DMS\PSL\LVL2\LVL2FIG2.DRW

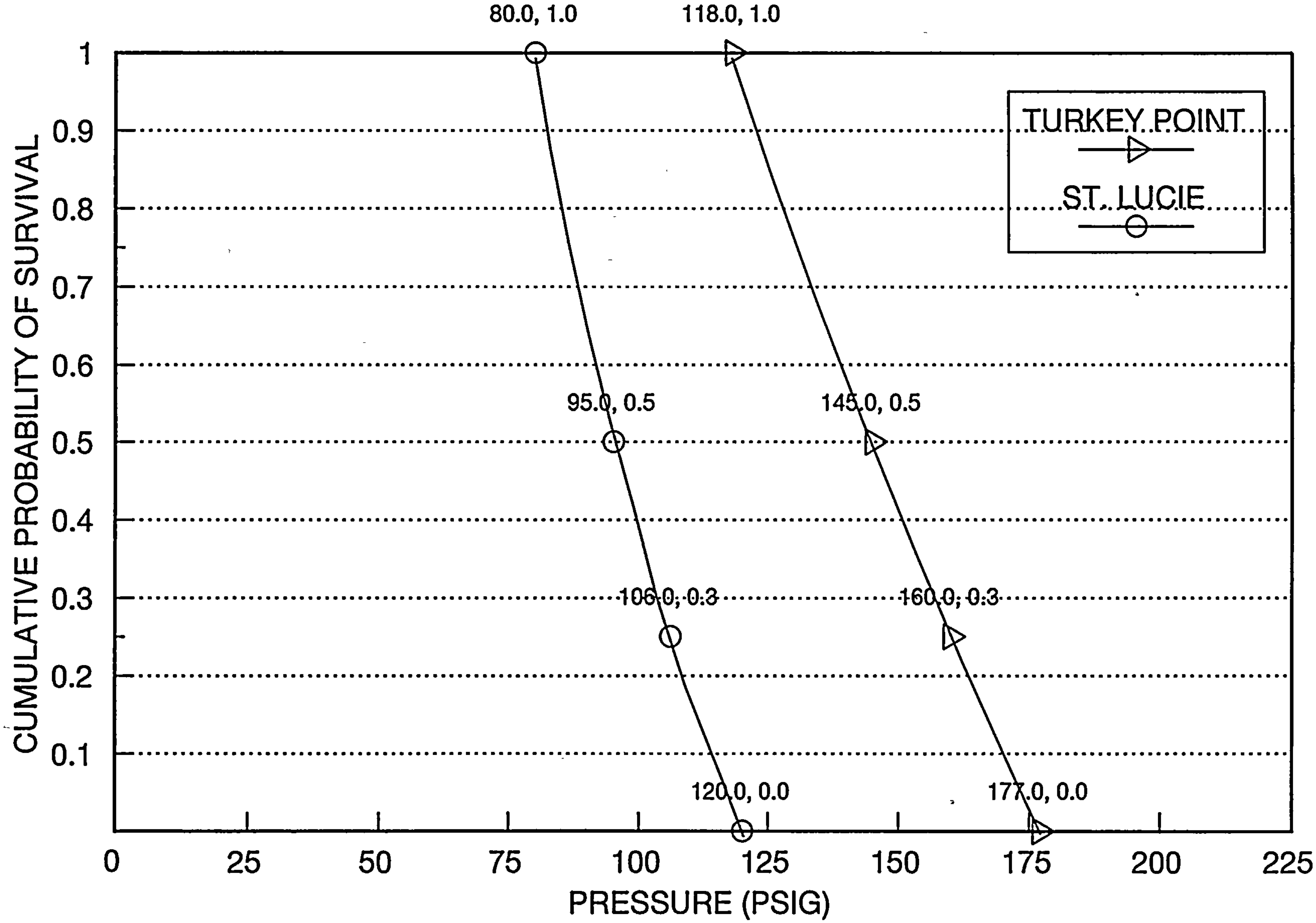


Figure 4.0-3 Adjusted St. Lucie Containment Survival Probability

F:\USERS\DMS\PSL\LVL2\LVL2FIG3.DRW

Figure 4.0-4A
Summary of St. Lucie Unit 1 Containment Analysis Results

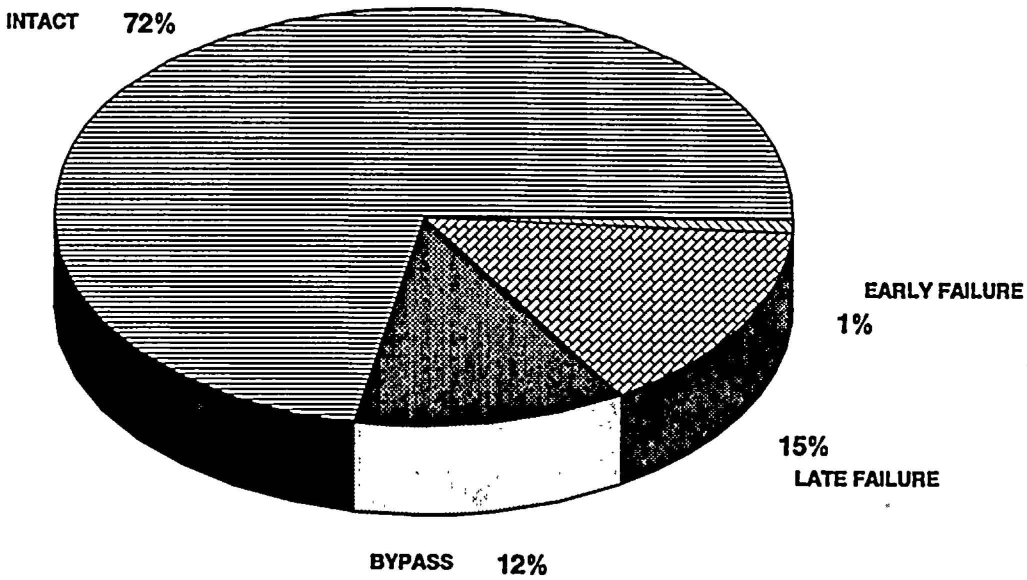
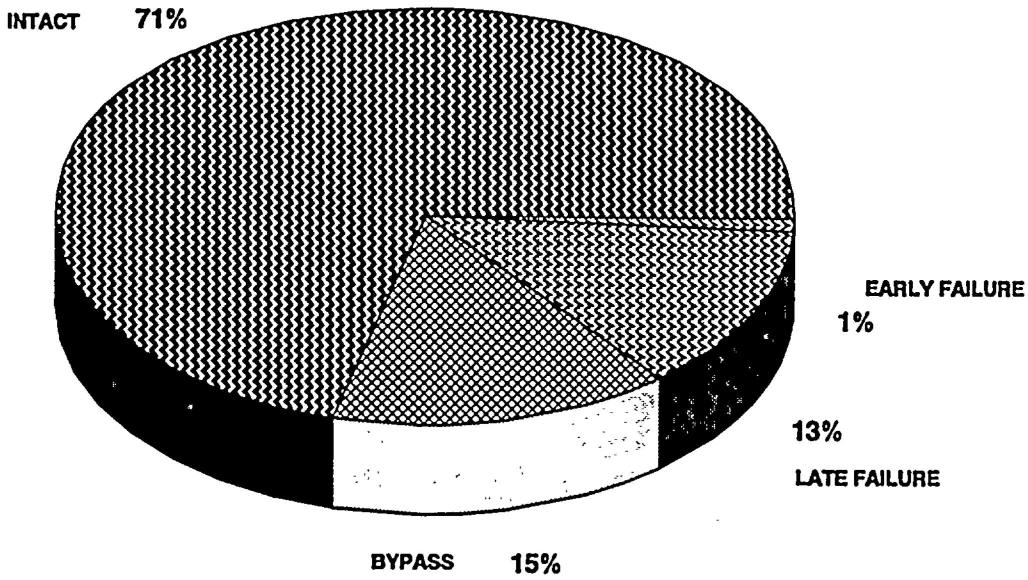


Figure 4.0-4B
Summary of St. Lucie Unit 2 Containment Analysis Results



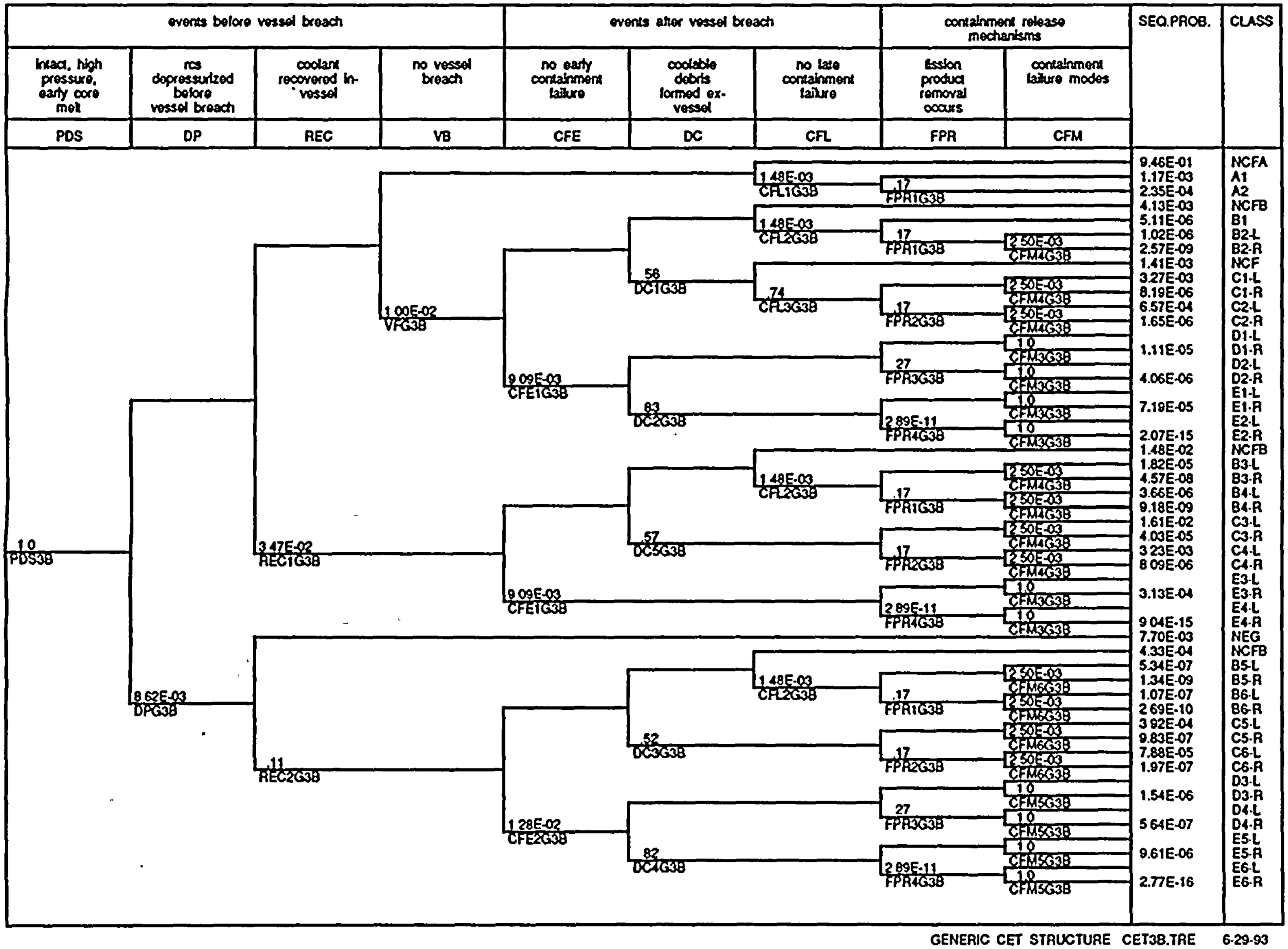


Figure 4.0-5 Quantified CET for PDS 3B

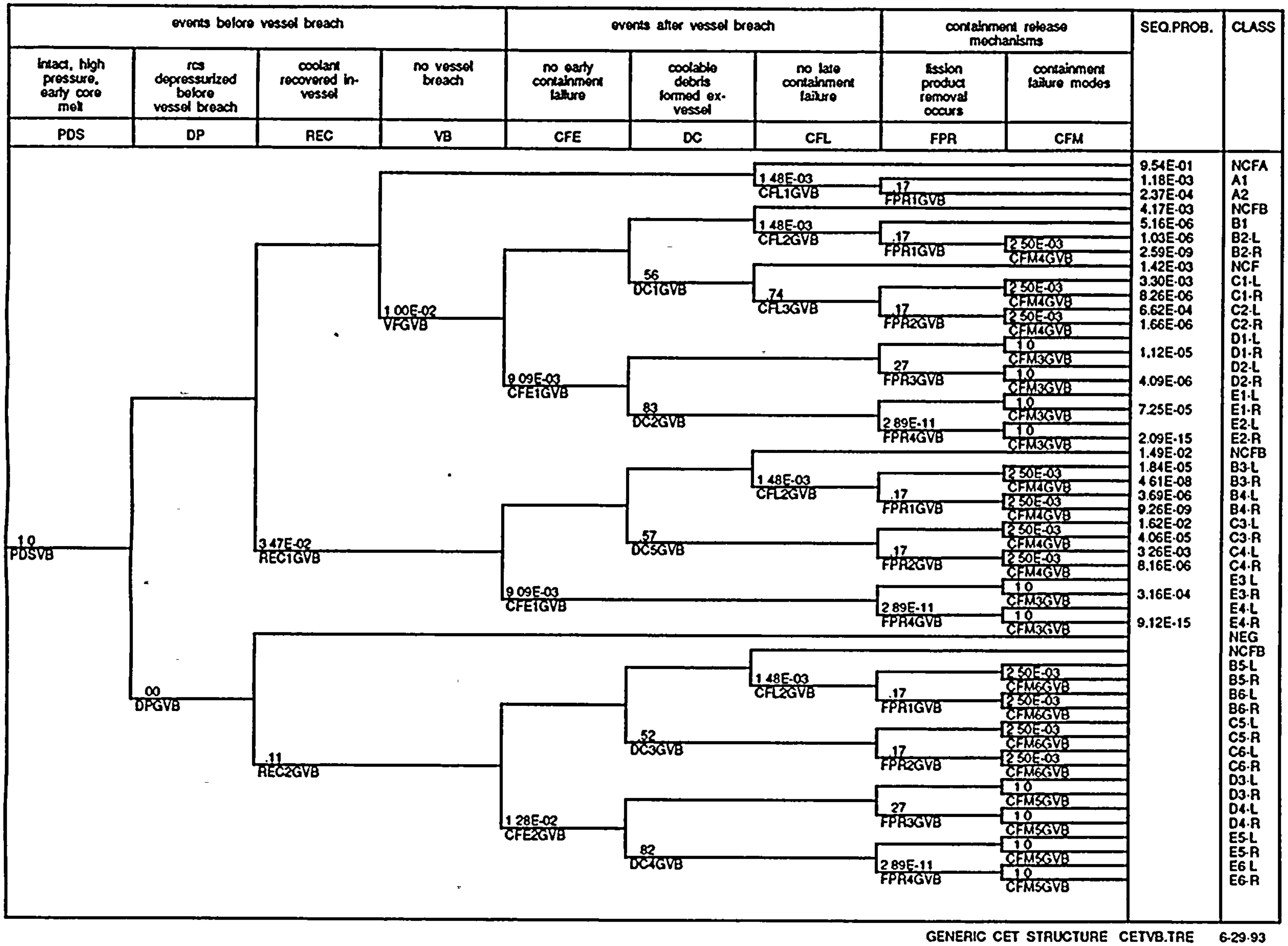


Figure 4.0-6 Quantified CET for PDS VB

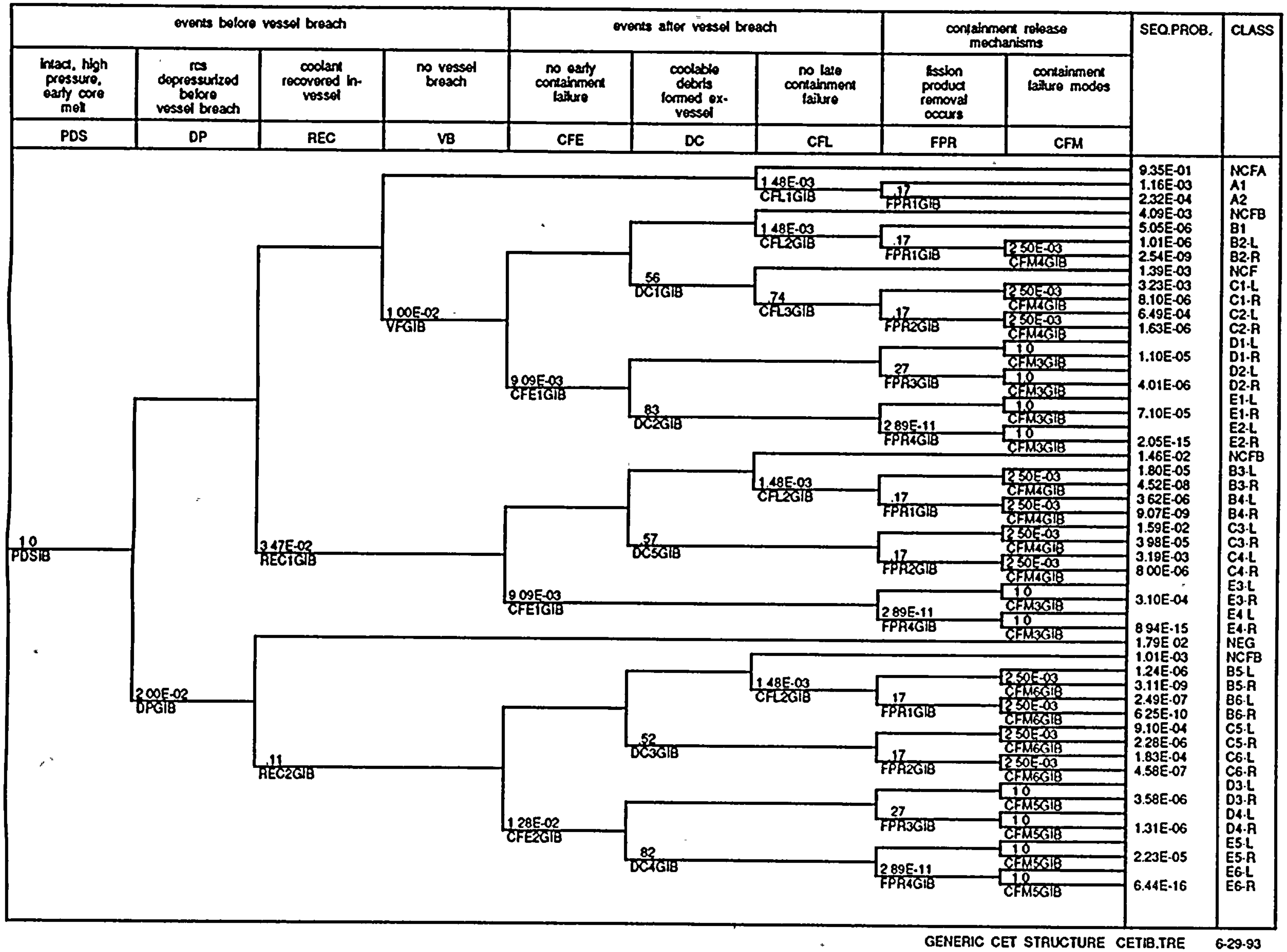
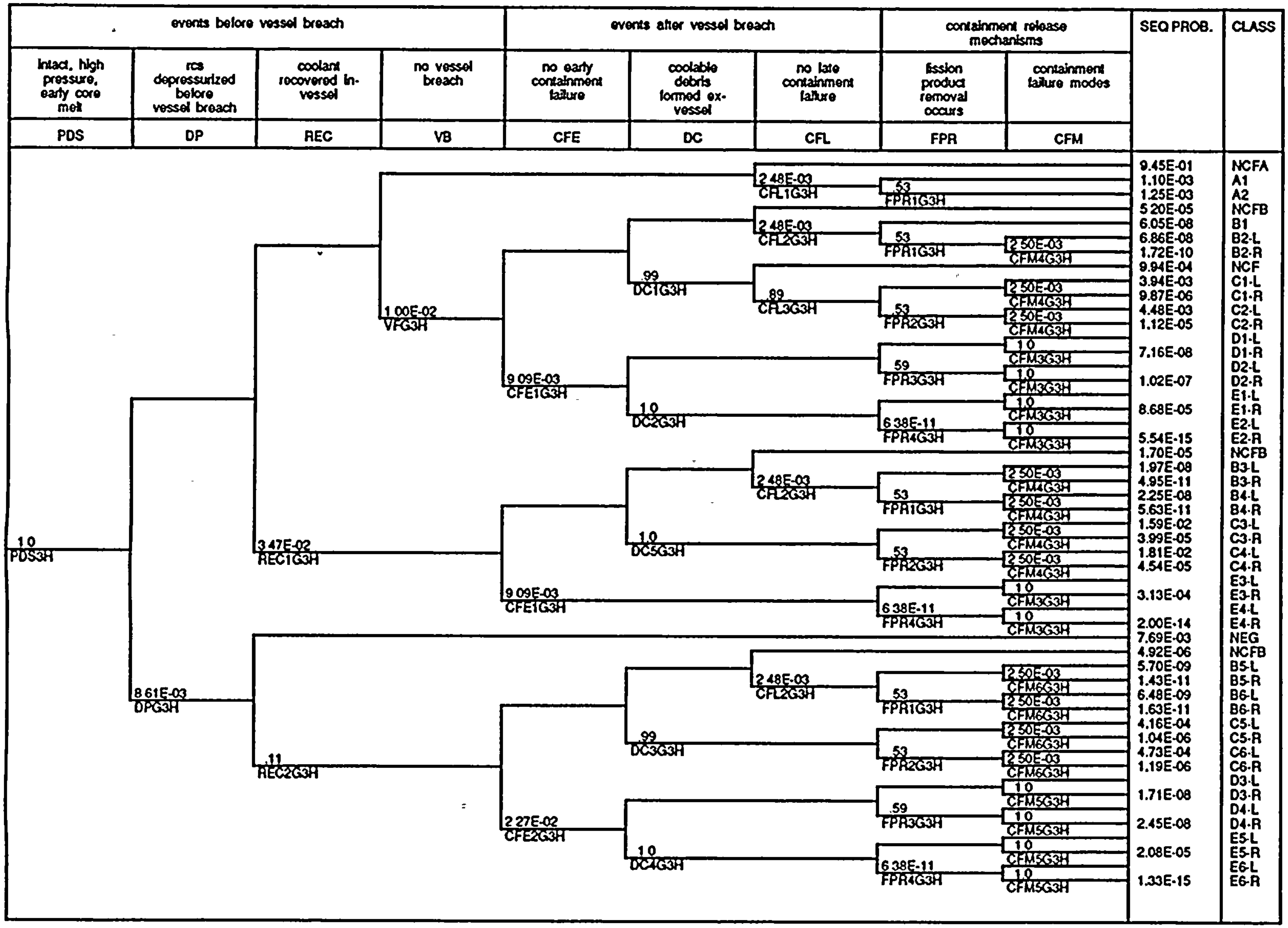


Figure 4.0-7 Quantified CET for PDS IB



GENERIC CET STRUCTURE CET3H.TRE 6-29-93

Figure 4.0-8 Quantified CET for PDS 3H

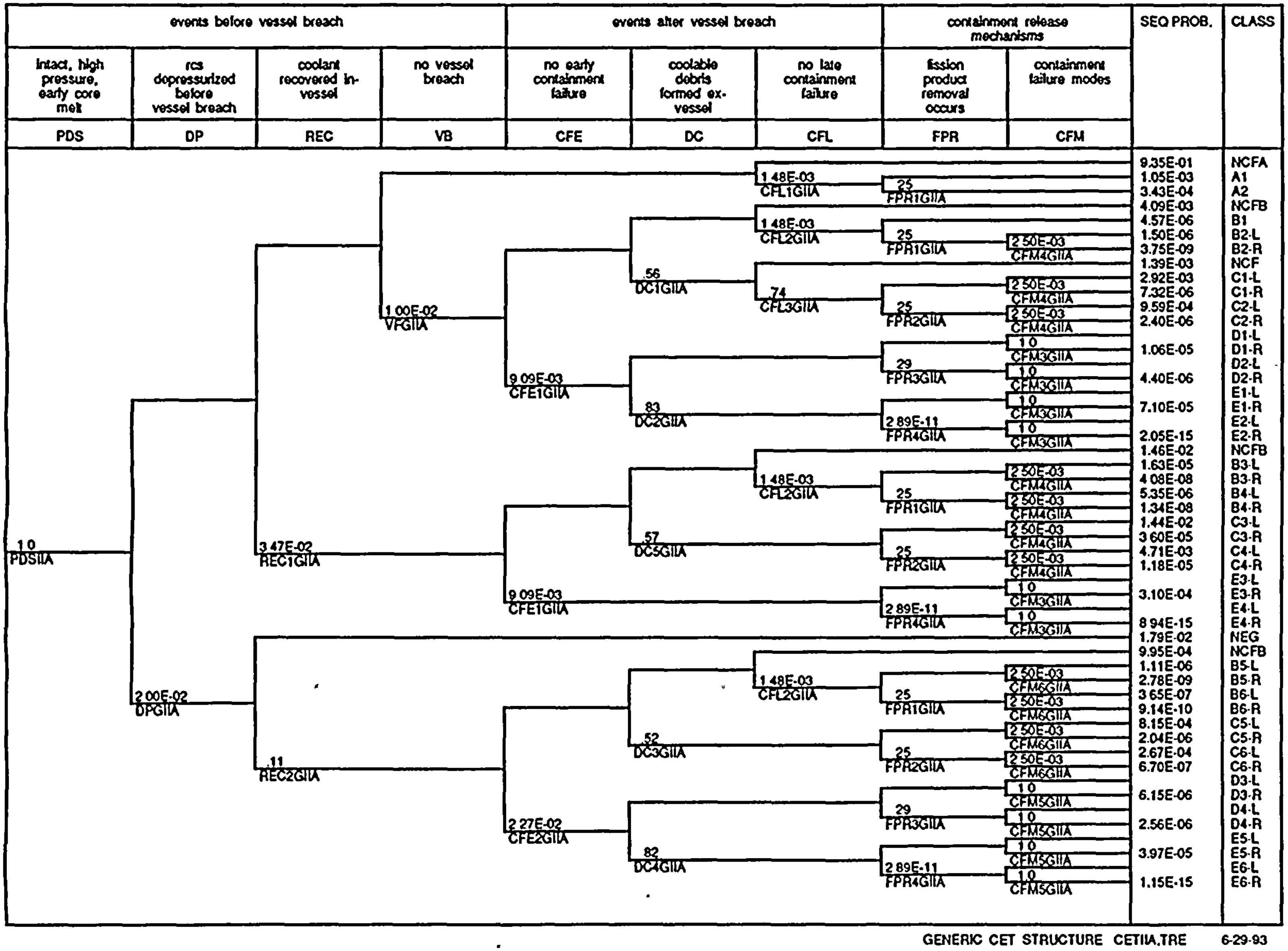


Figure 4.0-9 Quantified CET for PDS IIA

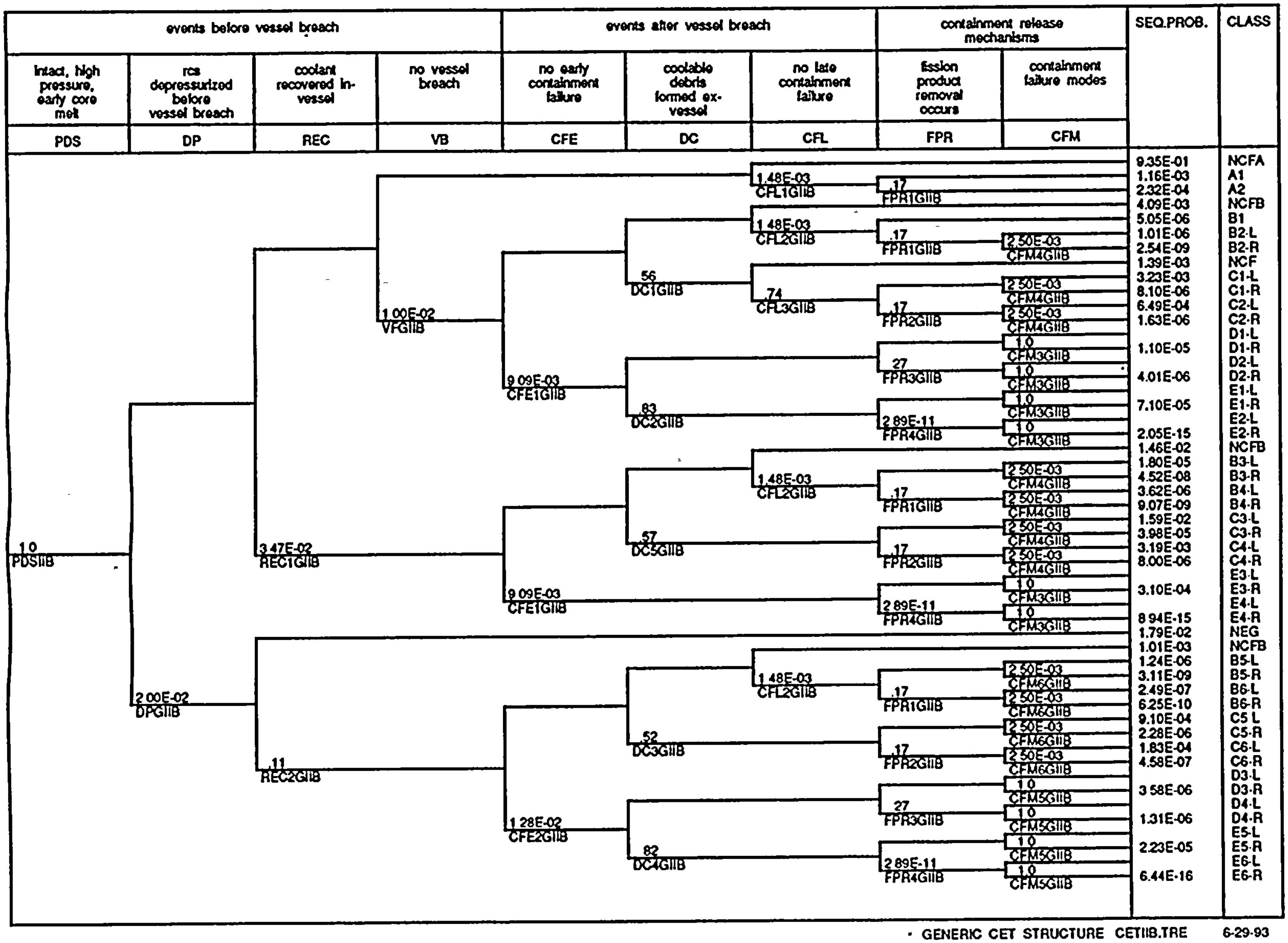


Figure 4.0-10 Quantified CET for PDS IIB

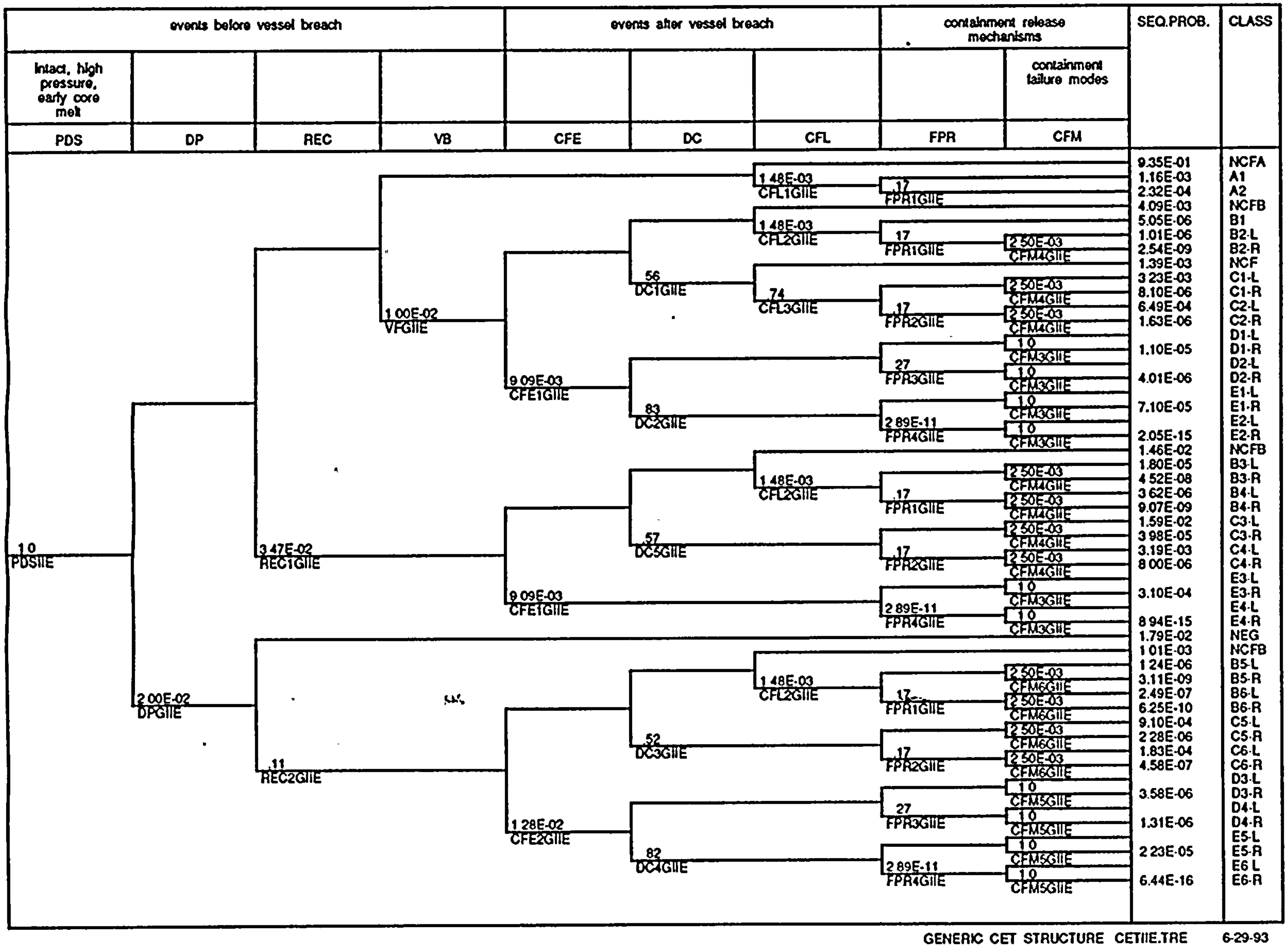


Figure 4.0-11 Quantified CET for PDS IIE

5.0 UTILITY PARTICIPATION AND PROJECT REVIEWS

GL 88-20 requested significant participation by utility personnel in the performance of the IPE. NRC also recommended that an independent review be conducted to assure the accuracy and validity of the results. This section describes the St. Lucie PRA organization, the extent of utility personnel involvement, and the independent reviews that were conducted.

5.1 IPE Program Organization

One of FPL's major objectives in performing the Turkey Point and St. Lucie Probabilistic Risk Assessments was to bring the PRA technology "in-house." To accomplish this objective, FPL established a group of engineers in the Nuclear Engineering Department, responsible for developing and applying the PRA.

Although the PRA development group brought a great deal of nuclear experience to the table, there were certain areas where assistance from outside the development group was solicited and obtained. The Nuclear Fuels department is responsible for nuclear core and RCS thermal-hydraulic analyses. This group brought the MAAP code in-house to support the PRA development. Virtually all of the plant departments, including Operations, Maintenance, Technical, Training and ISEG, provided input to the PRA development. The St. Lucie System Engineers, for example, were helpful in providing details and insights into the operation of their systems. The Operations and Training Departments provided experienced personnel during the analysis to help the team understand the plant and its response to accidents, and to help identify and quantify recovery actions for the accident scenarios.

5.1.1 St. Lucie PRA Development Team Composition

5.1.1.1 General Project Organization

The St. Lucie PRA development was primarily accomplished by the Nuclear Engineering Reliability and Risk Assessment Group (RRAG). ABB Combustion Engineering assisted in preparation (approximately 50%) of the system description notebooks. The RRAG supervisor was responsible as the overall FPL Project Manager for the effort.

For each major task, a RRAG engineer was assigned as the FPL lead. The FPL task leader had the accountabilities to:

1. Develop the necessary plans and procedures for the assigned task. To identify and obtain the necessary personnel resources for the task. To monitor and report on progress of the task.

2. Act as the prime focus for the analytical work.

3. Ensure that reviews of the PRA task output were conducted and comments incorporated into the final PRA work package/report.

4. Prepare and lead task related meetings, identify and obtain technical resources such as drawings, references, reports, etc.

5.1.2 FPL Staff Participation

As discussed above, the FPL Nuclear Division PRA development responsibilities were assigned to the Nuclear Engineering Reliability and Risk Assessment Group (RRAG). A brief discussion of the RRAG team members and their experience before the St. Lucie PRA began follows:

William A. Skelley, (Reliability & Risk Assessment Group (RRAG) Supervisor) - Bill was an officer in the Navy Nuclear Submarine Program. He worked for Bechtel as an Engineer, Manager of Nuclear Discipline, and Site Start-Up Mechanical/Nuclear Engineering Supervisor. With FPL, Bill has served as the Turkey Point Site Engineering Supervisor and is presently Engineering Staff Chief Nuclear Discipline Engineer and the RRAG Supervisor. Bill graduated from the University of Michigan with a Masters in Nuclear Engineering and attained a PE registration in Michigan.

Ching N. Guey, Ph.D., (Lead Engineer - Accident Sequence, Containment Performance) - Dr. Guey is the group's most experienced PRA analyst. His Nuclear Engineering Doctoral thesis focused on expanding the application of PRA methods. He has focused his career on probabilistic methods application with experience gained at several internationally recognized PRA consulting firms. He has performed probabilistic assessments of systems, external hazards, and has participated in full-scale PRAs. While a member of the Brookhaven National Laboratory staff, he reviewed and critically commented on several industry initiatives, including the IDCOR IPE methods and has trained NRC personnel in the use of SETS and IMPORTANCE codes. He was a member of the Turkey Point PRA Development Team. Dr. Guey received his Bachelor's degree from the National Central University (Taiwan), Master's from the University of Wisconsin-Madison and Doctorate from Massachusetts Institute of Technology.

Donald K. James, (Lead Engineer - Systems Analysis) - Don has extensive experience in nuclear plant design and operation. He has been a member of numerous FPL and industry teams whose charter was to improve plant reliability and reduce reactor trip frequency. He has performed design and safety analysis studies, performed startup testing and outage planning/technical support. Don served in the U.S. Naval Nuclear program as an Reactor operator/electronics technician on submarines. He was a member of the Turkey Point PRA Development Team. He held an Atomic Energy Commission reactor operator license and is a graduate of the University of Rhode Island.

M. Brien Vincent, (Lead Engineer - Data Analysis, HRA/Recovery Analysis, Sensitivity Analysis, and Quantification) - Brien has extensive experience in nuclear plant design and maintenance. He was the electrical maintenance assistant superintendent at St. Lucie plant, supervising and planning normal and outage maintenance activities. He has performed and managed electrical design changes to operating nuclear plants, and has served as supervising engineer of the St. Lucie Plant Reliability Group. He was a member of the Turkey Point PRA

Development Team. Brien graduated from the University of Florida with a B.S. in Electrical Engineering.

Kelly J. Korth, (Integration and Quantification) - Kelly was a certified senior instructor for Navy Nuclear Prototype. He was also a shift supervisor for the overhaul and conversion of a naval submarine to a training prototype. Kelly has been a lead engineer for the Nuclear Licensing Discipline in the Turkey Point Production Engineer Group. He graduated from the University of Pittsburgh.

Jack W. Revell, (Lead Engineer - Internal Flooding) - Jack has performed the duties of a non-licensed operator for Florida Power Corp., worked on scheduling and expediting electrical modifications at the Palisades Plant and performed system walkdowns and plant labeling at the Zion Plant. While at Turkey Point, Jack completed the SRO class and worked in the Planned Maintenance Group. Jack graduated from the University of Florida.

5.2 Composition of Project Reviews

To ensure the quality of the St. Lucie PRA, FPL developed a plan to build the quality in "up-front" as well as provide sufficient and diverse reviews of the final project output products.

5.2.1 Project Quality Assurance

Project Quality Assurance measures were established to ensure that the work performed by the PRA project team produced high quality outputs. The PRA development project was classified as a "Safety-Related" activity, subject to the requirements of 10CFR50, Appendix B. This work classification was <u>not</u> a provision of Generic Letter 88-20.

To implement the Safety Related classification, specific task procedures were written. Input documents used by the PRA team were controlled. Independent review of the PRA work products was performed.

5.2.2 Project Reviews

Three levels of review were used for the St. Lucie PRA. The first consisted of normal engineering Quality Assurance carried out by the organization performing the analysis. A qualified individual with knowledge of PRA methods and plant systems performed an independent review of the results for each task. This represents a detailed check of the input to the PRA model and provides a high degree of quality assurance.

The second level of review was performed by plant personnel not directly involved with the development of the PRA model. This consisted of individuals from Operations, Technical, Training, and ISEG groups who reviewed the system description notebooks and accident sequence description. This provided diverse expertise with plant design and operations knowledge to review the system descriptions for accuracy.

The third level of review was performed by PRA experts from ERIN Engineering, FRH, Inc., NUS, and Baltimore Gas & Electric. This review provided broad insights on techniques and results based on experience from other plant PRAs. The review team concentrated on the overall PRA methodology, accident sequence analysis, system fault trees and draft quantification results. The intent was to provide early feedback to the St. Lucie PRA staff concerning the adequacy and accuracy of the reviewed products. It should be noted that the methodologies used for the St. Lucie Level I and Level II analyses were similar to those used for the Turkey Point PRA. The Turkey Point IPE submittal was thoroughly reviewed by the NRC staff and NRC contractors. The NRC review concluded that the process used to develop the Turkey Point PRA was acceptable in meeting the intent of GL-88-20.

5.3 Areas of Review and Major Comments

The general areas of review were as described above. The overall purpose of the review was to ensure the quality of the PRA project and to ensure that the project objectives were being met. The review team found that the project was successfully meeting those objectives with a sound methodology. Specific review comments are not repeated here. However, some of the major comment areas are summarized below:

- The overall methodology reflects the current state of the art for PRAs and will meet the requirements of GL 88-20.
- The system description notebooks were very well organized and very complete.
- The event trees and success criteria used to support the systems analysis interface are consistent with those of other similar analyses.
- CST replenishment should be included for sequences where long-term cooling via AFW may be required. (This was included for Unit 1.)
- Units 1 & 2 data should be combined to formulate the plant specific history. (This was incorporated.)

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

Based on the results discussed in sections 3.7 (Level 1) and 4.0 (Level 2) an effort has been made to gain useful insights relating to St. Lucie Units 1 and 2 safety features and to identify potential plant improvements. Section 6.1 and 6.2 describes insights about the St. Lucie safety features. Section 6.3 provides a summary of important plant features and section 6.4 discusses plant improvements.

6.1 Level 1 Insights and Unique Safety Features

A summary of the Level 1 insights are provided below. Insights gained are both in areas where the plant's design is robust as well as areas where the core damage frequency has been determined to be sensitive.

The largest contributor to the Unit 1 and Unit 2 core damage frequency is a small-small LOCA (1/2" - 3") initiating event. The dominant cutsets are related to common cause failures of motor operated valves and of HPSI pumps to start and run. For Unit 1, there is a common minimum recirculation line for both HPSI pumps. If the valves in this line were to transfer to the closed position, the HPSI pumps may fail. It should be noted that the power to these valves is removed when the unit is on line. It was also assumed that loss of CCW to the HPSI pump seals could prohibit long term pump operation. The assumption that failure to isolate the CCW N-header could fail the CCW system function due to the combined safety related and non-safety related heat loads was shown to be an important contributor. These analyses are conservative in that no operator action was credited for the potential to repair these failed components.

The capability to power one safety related 4kV bus on both units from any of the four diesel generators via the blackout crosstie reduces the likelihood of a long term station blackout for loss of grid scenarios. The dominant failures which result in a blackout are common cause failures of all four diesel generators and the failure of one Unit's diesel generator with failure of the blackout crosstie. Procedures provide direction to the operator as to how to implement this crosstie.

The loss of multiple feedwater systems is required to fail the secondary heat removal function. The main feedwater (MFW) system would normally be available following a unit trip unless there was a loss of offsite power or if MFW had been isolated by a SIAS. Even if isolated by a SIAS, main feedwater flow can be re-established to the steam generators by the operator. On Unit 2, MFW is isolated by AFAS or MSIS. MSIS cannot be overridden unless containment pressure is less than 3.5 psig. If the secondary pressure is lowered to approximately 600 psig, the condensate pumps could also be used to feed the steam generators directly. Each unit also has two motor driven AFW pumps and one turbine driven AFW pump. The turbine driven AFW pump is only dependent on DC power, which can be supplied from either the "A" or "B" train safety related batteries. Although not credited in this analysis, either of the two non-safety related batteries could also be aligned to provide power to the turbine driven AFW pump, or the turbine driven AFW pump could be operated locally without DC power. In the event all of the feedwater systems are lost, secondary heat removal can be provided by once-through-cooling (OTC) using a HPSI pump and both PORVs on Unit 1 or 1-of-2 PORVs on Unit 2. The difference in success criteria is due to the relief capacity of the PORVs (i.e., larger PORVs on Unit 2). The available of both PORV flow paths on Unit 1 is therefore critical to OTC success.

Due to the smaller size of the Unit 1 condensate storage tank (CST) compared to the Unit 2 CST, sequences where long term AFW operation is credited (greater then approximately 10 hours) require that makeup to the Unit 1 CST be provided or the operator must re-align the suction of Unit 1 AFW pumps to the Unit 2 CST.

The dominant contributor to an ISLOCA is two normally closed shutdown cooling suction isolation valves transfering to the open position. These valves are normally locked closed during power operation with power removed and are also interlocked with the RCS pressure to prevent inadvertent opening at power.

The internal flooding analysis demonstrated that there is no credible flood/spray scenario which provides a significant contribution to the overall risk of St. Lucie Unit 1 or Unit 2.

6.2 Level 2 Containment Response Insights

On a frequency-weighted basis of the plant damage states, the estimated probability from internally-initiated core damage accidents for early containment failures is approximately 1% for both St. Lucie Units 1 and 2. Late containment failures contribute 15% for St. Lucie Unit 1 and 13% for St. Lucie Unit 2. Containment integrity is maintained for 72% of the core damage sequences for Unit 1 and 71% for Unit 2.

The following observations and conclusions may be drawn from an examination of the key results of this study.

Early Containment Challenge. The potential for early containment failure is important in the St. Lucie risk analysis as it also provides a measure of the potential consequences of a severely degraded core accident. The major contributors to early containment failure for St. Lucie include containment threats due to HPME loads from high RCS pressure core damage accidents, steam explosion events for low pressure sequences, and isolation failures. The threat to HPME loads is reduced for an initially high pressure sequence if RCS depressurization occurs prior to vessel breach as a result of hot-leg failure or seal LOCAs. A sensitivity analysis of HPME loads using MAAP, with very conservative assumptions, proved that direct containment heating can pose a threat to containment integrity only if dispersal of the molten material to the upper containment region is assumed. Forcing hydrogen burning during HPME also contributed to significant pressure loads that would likely challenge containment integrity. However, depressurization of the RCS (due to temperature induced hot-leg failure) precluded high pressure melt ejection at vessel failure. This would reduce the likelihood of early containment failure. For the baseline cases, no early containment failures were predicted based on MAAP simulation results. Probability of early containment failures of 1% stems from the uncertain phenomological modeling embedded in the CET logic (e.g., ex-vessel steam explosion, vessel acting as a rocket).

Severe Accident Loads. In general, the large dry containment design of St. Lucie proved to be robust, thus pressure loads were not likely to lead to containment failure. The containment failure time, if the containment did fail, was greatly delayed relative to the time of core damage. The major contributor to late containment failures is steam overpressure in long term (hydrogen burning is likely to be precluded due to the steam-inerted containment atmosphere). It is noteworthy that the accident progression analysis performed for this study indicates that the containment floor and cavity configuration for St. Lucie will, in most cases, allow the RCS (and accumulator) water inventory to collect in the cavity thus providing an overlying water pool above the debris following vessel breach.

Systems-Related Functions. It was found in this study that by far the most effective measures were systems operation either to prevent core melt from progressing to vessel breach, to protect containment integrity, and to mitigate the potential consequences by reducing the source terms. Systems survivability is crucial in order to continue operation as to preclude containment failure. This principal observation confirms and continues to reinforce conclusions that containment safeguards availability and continued operation would effectively mitigate, in many cases preclude, containment failure and significant releases to the environment. Availability of the sprays in the long term would result in steam condensation and heat removal from the containment, thus maintaining containment integrity.

The large dry containment of St. Lucie was found to be less susceptible to early containment failures given a severe accident. This study shows a high likelihood of maintaining containment integrity during the early phases of the accident progression. The uncertainties in the determination of these pressure loads, and mitigation of the potential loads due to RCS depressurization, however, are high. The total conditional probability of early containment failure for St. Lucie (1%) is lower than the reference PWR (13%), as the reference PWR Surry study includes containment bypass (contributing the major portion (12%) of the likelihood of early failures). The Zion study indicates a comparable probability of no containment failure (73%) to that for St. Lucie. The combined large containment volume (2.6 million cubic feet) and estimated failure pressure (95 psig), provide considerable capability to withstand loads.

This study demonstrated the inherent capability of the St. Lucie containment to survive pressure loads during the early phases of a severe accident and to mitigate the long term effects of severe accident progression. It has provided a means of determining which core damage sequences are likely to be potential risk contributors. It has also provided insights as to the accident conditions, modeling parameters or systems (or operator actions) that can strongly affect the MAAP 3.0B code predictions and thereby ascertaining where a more realistic characterization of the plant is most needed.

6.3 Summary of Important Plant Features

This section summarizes the significant features that affect multiple sequences and have a major effect on the PRA results.

- Feedwater can be provided by multiple systems - main feedwater (two MFW pumps and three condensate pumps per Unit), and AFW (two motor driven pumps and one turbine driven pump per Unit). Main feedwater remains functional following a unit trip unless failed by the initiator (e.g., loss of grid, feedline break). The decay heat removal capability is also enhanced by the ability to provide once-through-cooling using the HPSI pumps and PORVs.

- St. Lucie Unit 1 and Unit 2 has an automatic switchover for HPSI suction from the RWT to the containment sump.

- The St. Lucie Unit 1 and Unit 2 reactor coolant pump seal design is not susceptible to gross failure if the pumps are secured within 10 minutes following loss of seal cooling. The emergency and off-normal operating procedures direct the operator to trip the RCPs if seal cooling cannot be restored.

- The containment design is such that the reactor cavity is always wet, except for containment bypass sequences. Water first flows to the containment sump which then overflows into the reactor cavity.

- Ex-vessel cooling will occur due to reactor cavity flooding and the low placement of the reactor vessel. This reduces the probability of vessel failure.

- There are no lower head penetrations in the St. Lucie Unit 1 or Unit 2 reactor vessel thus delaying the time for vessel failure.

6.4 Plant Improvements

Based on the final results of the IPE for St. Lucie Units 1 and 2 and the definition of vulnerability provided in Section 3.7, no modifications to either hardware or procedures is required. Although no vulnerabilities were identified, one procedure enhancement was implemented. As discussed in Section 3.7, long term operation of AFW (beyond approximately 10 hours) on Unit 1 would require that the operator initiate makeup to the CST or re-align AFW pump suction to the Unit 2 CST. Existing emergency and off-normal procedures address the potential need for CST makeup and provide guidance to the operator regarding the steps to align the Unit 1 AFW pumps to the Unit 2 CST. While reviewing the various procedures related to long term AFW operation with Operations personnel during the recovery analysis, it was determined that the specific steps required for the operator to initiate makeup to the CST would be included in the water plant operations procedure.

7.0 SUMMARY AND CONCLUSIONS

FPL has performed a Level 1 and limited scope Level 2 PRA for St. Lucie Units 1 & 2 in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities". The objectives for this assessment are consistent with the objectives given in the generic letter and listed in Section 1.1 of this report. FPL personnel have been directly involved in all aspects of the development, quantification, and documentation of the PRA models. The approach included system, procedure, and drawing reviews, discussions with Operations, Training, Technical Staff, and other plant personnel, and independent peer reviews by PRA experts to ensure that the models are consistent with accepted PRA practices.

As a result, the IPE provides a comprehensive and detailed analysis of the severe accident behavior of St. Lucie Units 1 & 2. The overall likelihood of core damage and fission product release from the containment from internally initiated events has been quantified consistent with the guidance provided in GL 88-20. The relative contribution to core damage frequency from the different accident sequence types has been determined.

The following general conclusions have been drawn from the IPE:

- The overall core damage frequency due to internally initiated events for St. Lucie 1 is 2.3×10^{-5}/yr and St. Lucie Unit 2 is 2.6×10^{-5}/yr. This is much less than the NRC safety goal of 1×10^{-4}/yr and illustrates the high level of safety.

- The overall core damage frequency for St. Lucie Units 1 & 2 is within the range of past PRAs performed for PWRs. Thus, the susceptibility to core damage at St. Lucie Units 1 & 2 is not unlike other PWRs.

- A chart of the dominant accident sequences is shown in Figure 1.4-1. It shows that the core melt risk is dominated by small-small (1/2" - 3") LOCAs. Total loss of feedwater events are also important accident sequences for core damage risk. Section 3.7 presents the Level 1 results in more detail.

- St. Lucie has several means of providing feedwater to the steam generators for decay heat removal. No vulnerability related to USI A-45, Decay Heat Removal, has been identified.

- The St. Lucie Units 1 & 2 large dry containment design provides adequate capability to mitigate severe accidents. No unusually poor containment performance has been found. The dominant containment failure modes are described in detail in Section 4.

- The greatest threat to containment integrity is due to a loss of all containment heat removal during an accident where the RCS is at high pressure.

- A key feature of the St. Lucie containment design is that for almost all accident sequences, the reactor cavity is flooded with water. This decreases the likelihood of reactor vessel failure due to ex-vessel cooling and results in lower releases (due to retention of fission products in the RCS and scrubbing of ex-vessel fission products by the water) compared to if the vessel were to fail and the core were to fall on a dry cavity floor.

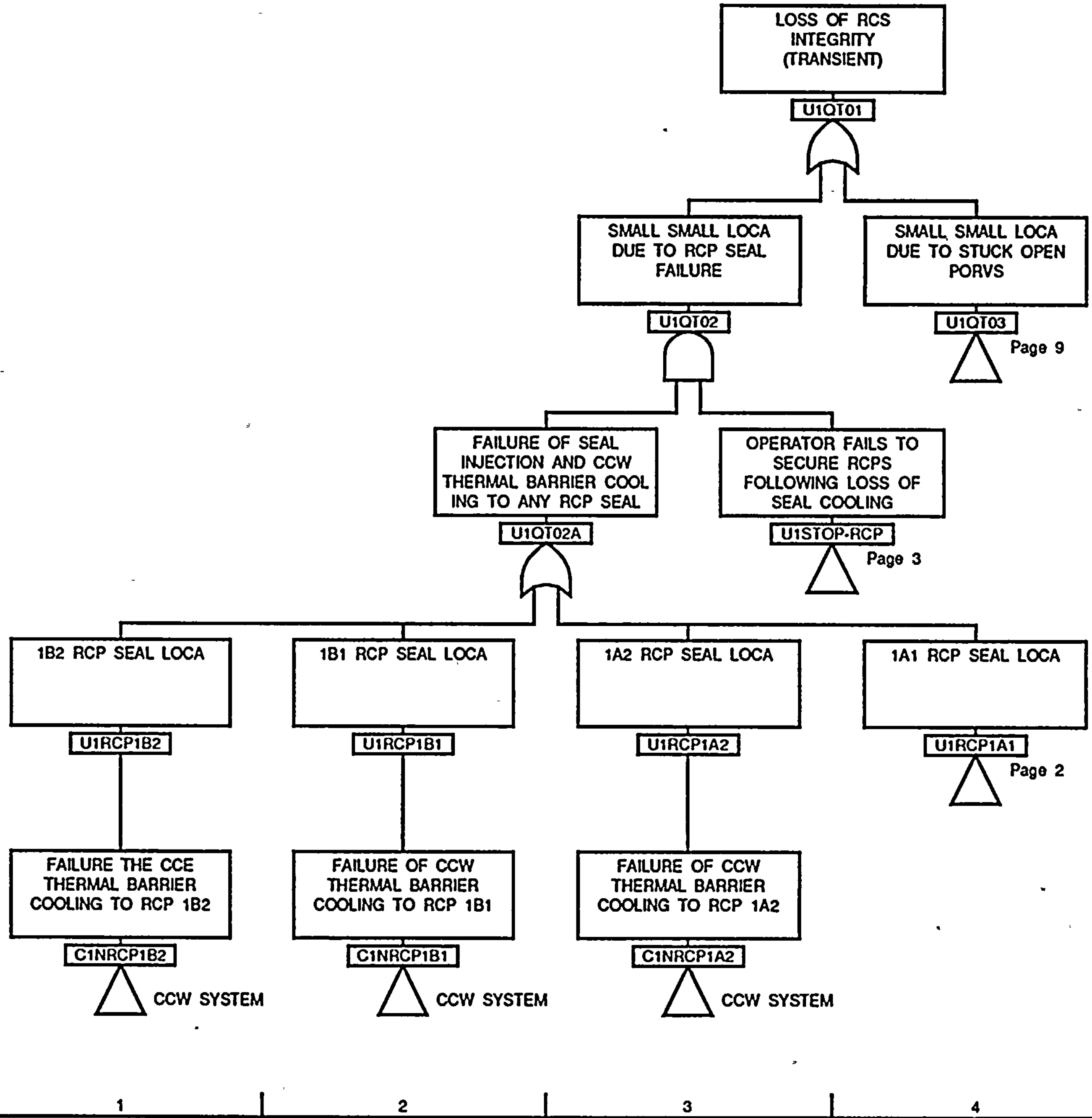
- The open design of the St. Lucie containment means that local hydrogen accumulation (identified in GL 88-20, Supplement 3, containment performance improvement issues) is not a significant contribution to containment failure.

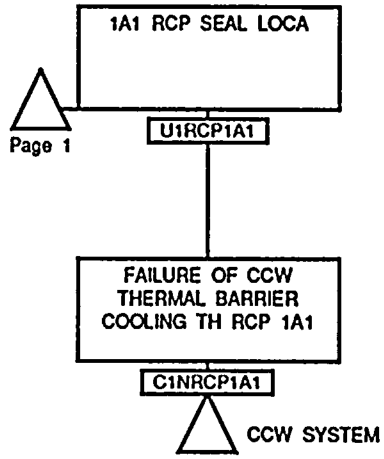
In conclusion, the St. Lucie Units 1 & 2 PRA has been performed in a manner consistent with the objectives stated in GL 88-20 and the results found that there are no plant unique severe accident vulnerabilities.

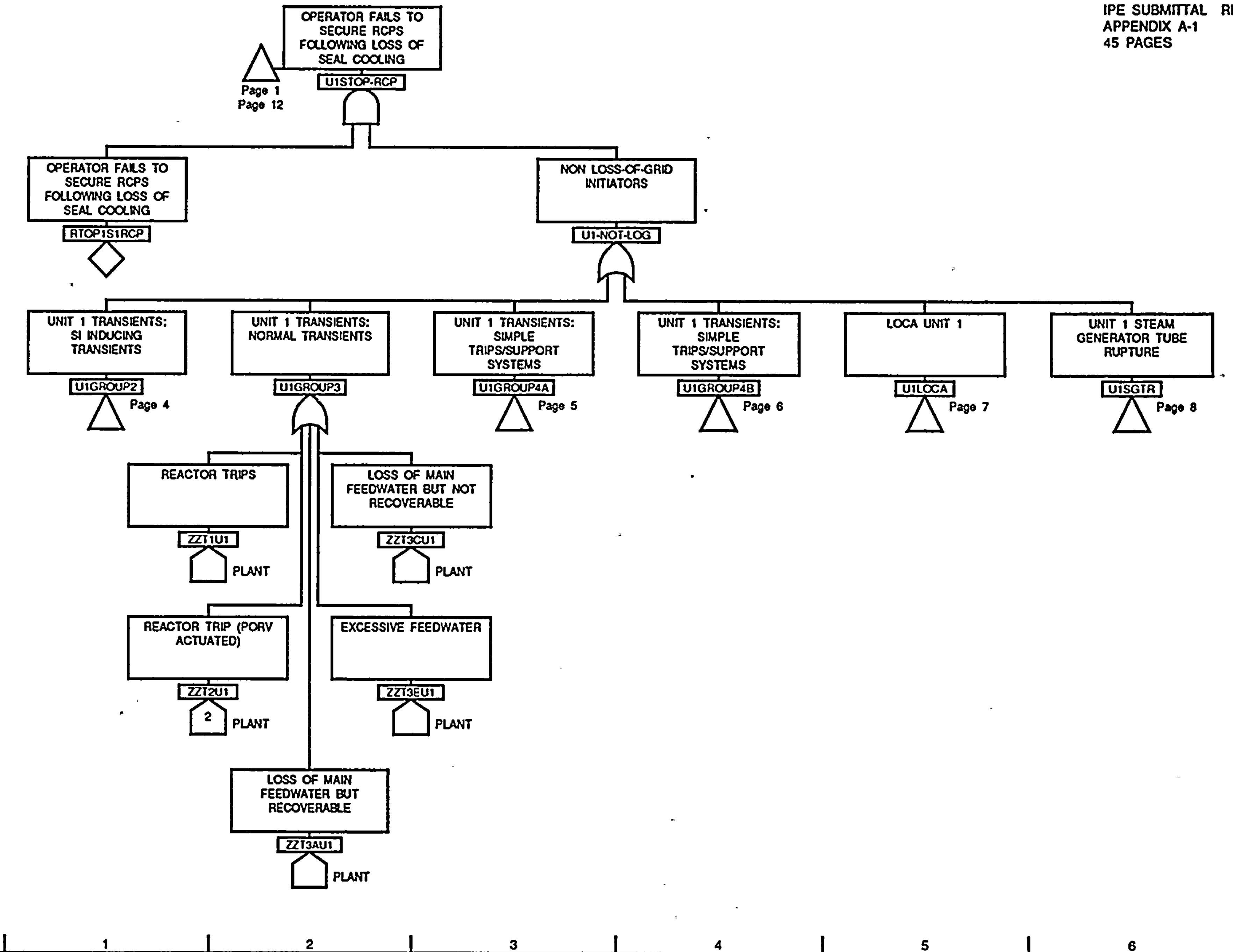
Appendix A

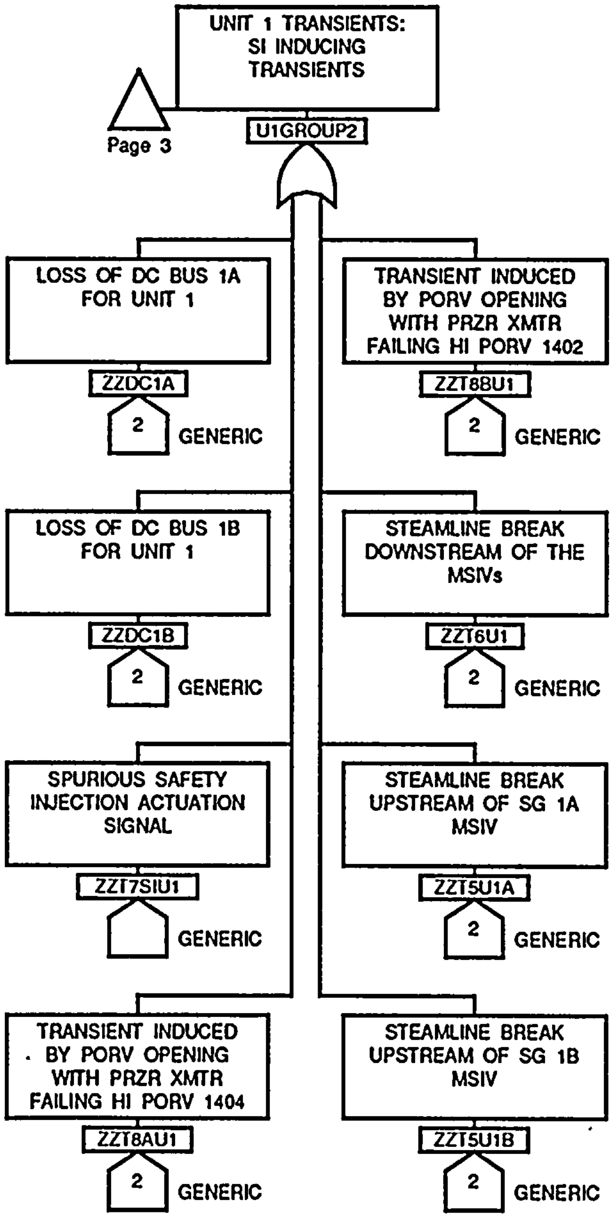
A-1 UNIT 1 TOP LOGIC

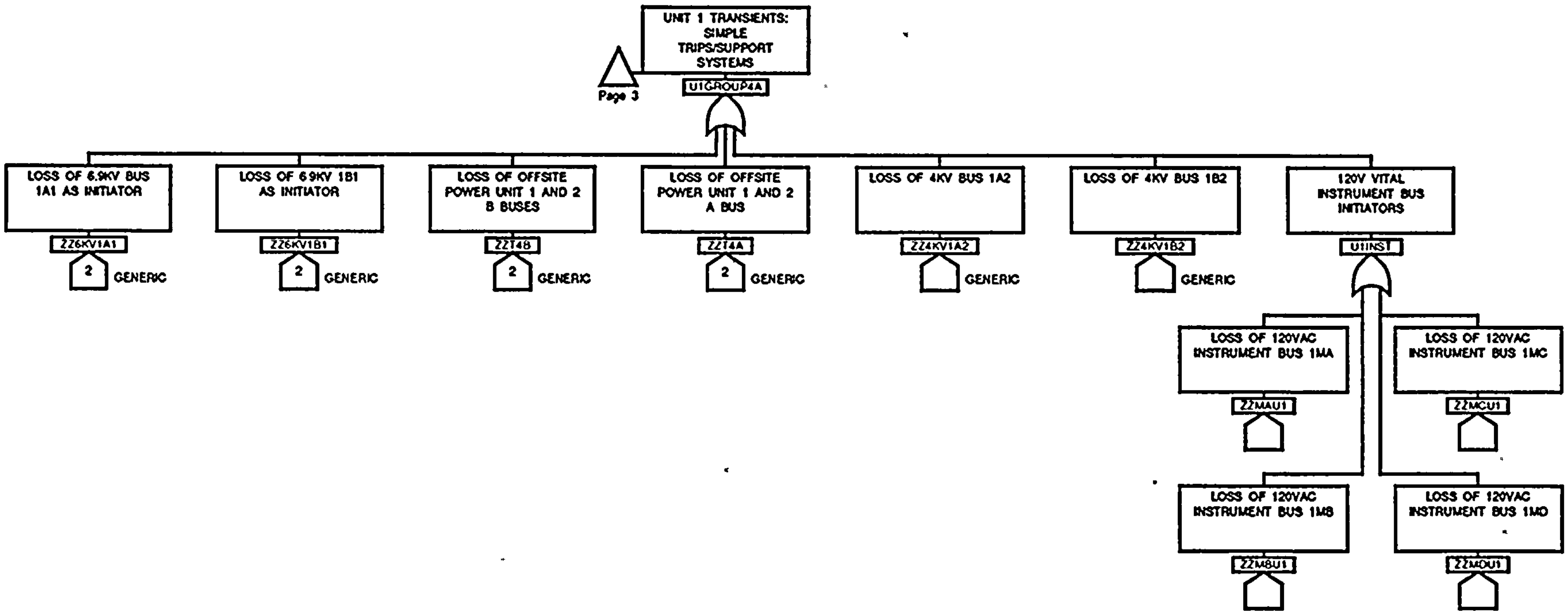
A-2 UNIT 2 TOP LOGIC

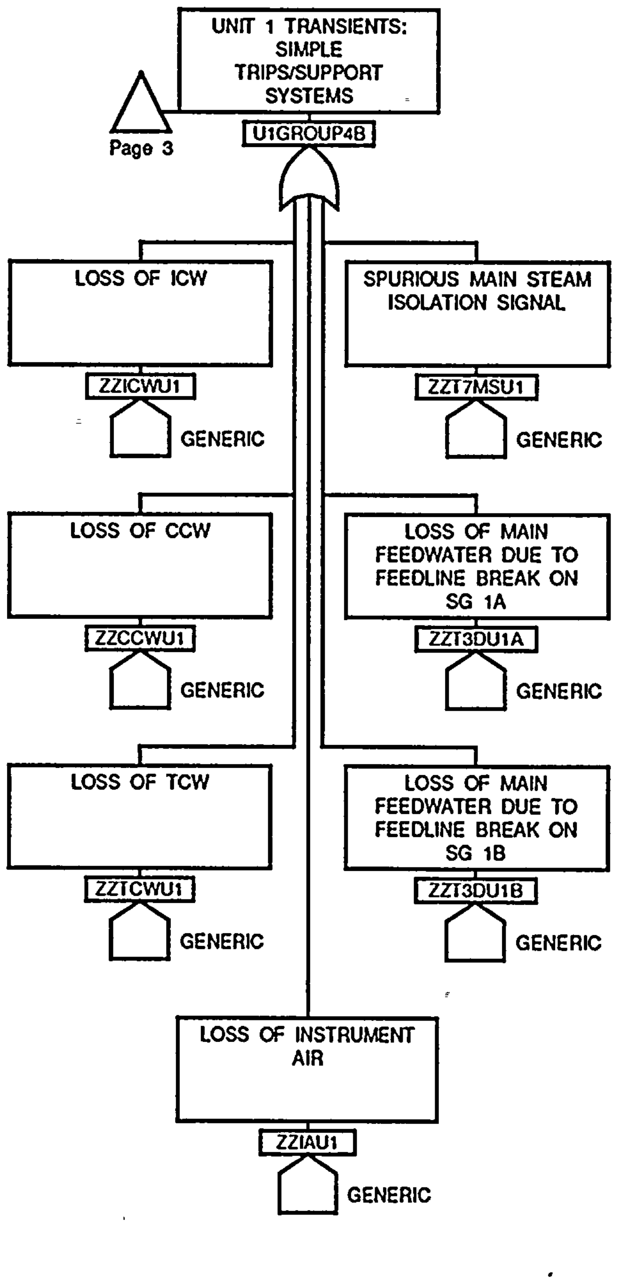

LOSS OF RCS INTEGRITY (TRANSIENT)
U1QT01
SMALL SMALL LOCA DUE TO RCP SEAL FAILURE
U1QT02
SMALL SMALL LOCA DUE TO STUCK OPEN PORVS
U1QT03
Page 9
FAILURE OF SEAL INJECTION AND CCW THERMAL BARRIER COOL ING TO ANY RCP SEAL
U1QT02A
OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U1STOP-RCP
Page 3
1B2 RCP SEAL LOCA
U1RCP1B2
1B1 RCP SEAL LOCA
U1RCP1B1
1A2 RCP SEAL LOCA
U1RCP1A2
1A1 RCP SEAL LOCA
U1RCP1A1
Page 2
FAILURE THE CCE THERMAL BARRIER COOLING TO RCP 1B2
C1NRCP1B2
CCW SYSTEM
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 1B1
C1NRCP1B1
CCW SYSTEM
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 1A2
C1NRCP1A2
CCW SYSTEM
1
2
3
4

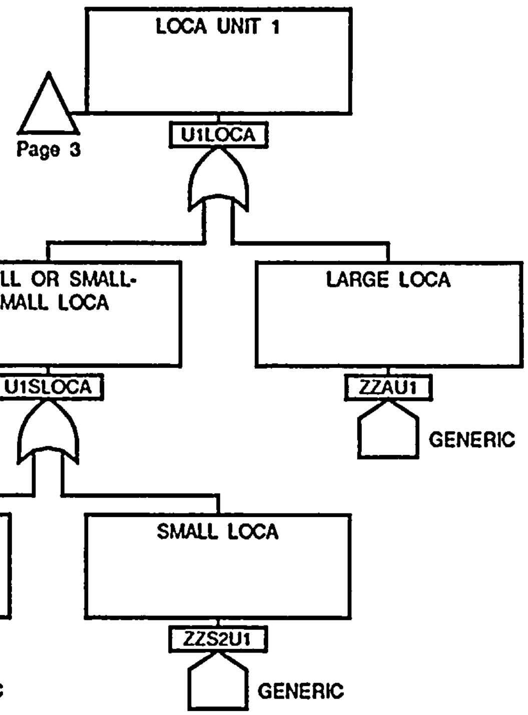

1A1 RCP SEAL LOCA
Page 1
U1RCP1A1
FAILURE OF CCW THERMAL BARRIER COOLING TH RCP 1A1
C1NRCP1A1
CCW SYSTEM

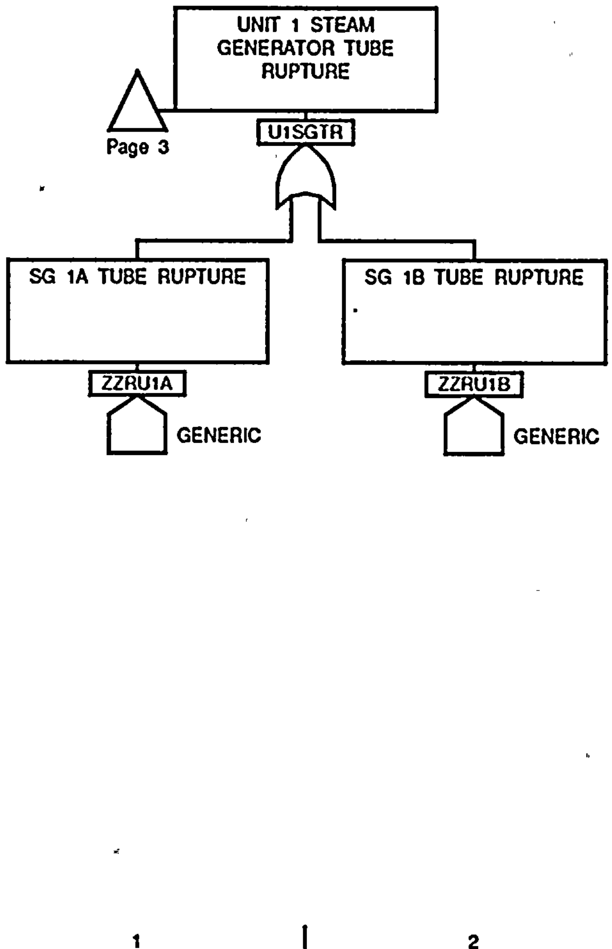

OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U1STOP-RCP
Page 1
Page 12
OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
RTOP1S1RCP
NON LOSS-OF-GRID INITIATORS
U1-NOT-LOG
UNIT 1 TRANSIENTS: SI INDUCING TRANSIENTS
U1GROUP2
Page 4
UNIT 1 TRANSIENTS: NORMAL TRANSIENTS
U1GROUP3
UNIT 1 TRANSIENTS: SIMPLE TRIPS/SUPPORT SYSTEMS
U1GROUP4A
Page 5
UNIT 1 TRANSIENTS: SIMPLE TRIPS/SUPPORT SYSTEMS
U1GROUP4B
Page 6
LOCA UNIT 1
U1LOCA
Page 7
UNIT 1 STEAM GENERATOR TUBE RUPTURE
U1SGTR
Page 8
REACTOR TRIPS
ZZT1U1
PLANT
LOSS OF MAIN FEEDWATER BUT NOT RECOVERABLE
ZZT3CU1
PLANT
REACTOR TRIP (PORV ACTUATED)
ZZT2U1
2
PLANT
EXCESSIVE FEEDWATER
ZZT3EU1
PLANT
LOSS OF MAIN FEEDWATER BUT RECOVERABLE
ZZT3AU1
PLANT
1
2
3
4
5
6

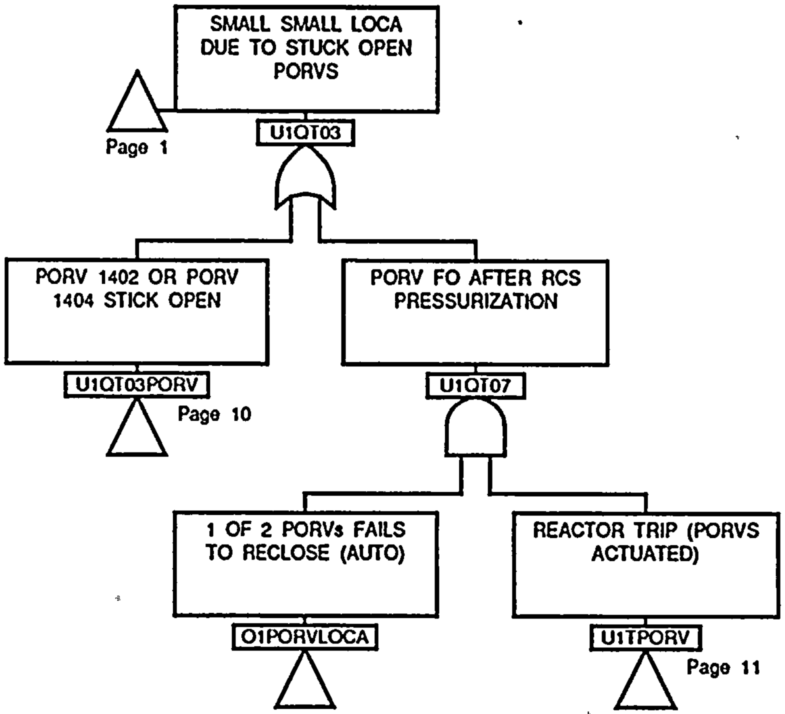

UNIT 1 TRANSIENTS: SI INDUCING TRANSIENTS
Page 3
U1GROUP2
LOSS OF DC BUS 1A FOR UNIT 1
ZZDC1A
2
GENERIC
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1402
ZZT8BU1
2
GENERIC
LOSS OF DC BUS 1B FOR UNIT 1
ZZDC1B
2
GENERIC
STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
ZZT6U1
2
GENERIC
SPURIOUS SAFETY INJECTION ACTUATION SIGNAL
ZZT7SIU1
GENERIC
STEAMLINE BREAK UPSTREAM OF SG 1A MSIV
ZZT5U1A
2
GENERIC
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1404
ZZT8AU1
2
GENERIC
STEAMLINE BREAK UPSTREAM OF SG 1B MSIV
ZZT5U1B
2
GENERIC
1
2


UNIT 1 TRANSIENTS: SIMPLE TRIPS/SUPPORT SYSTEMS
U1GROUP4A
Page 3
LOSS OF 6.9KV BUS 1A1 AS INITIATOR
ZZ6KV1A1
2
GENERIC
LOSS OF 6.9KV 1B1 AS INITIATOR
ZZ6KV1B1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 B BUSES
ZZT4B
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 A BUS
ZZT4A
2
GENERIC
LOSS OF 4KV BUS 1A2
ZZ4KV1A2
GENERIC
LOSS OF 4KV BUS 1B2
ZZ4KV1B2
GENERIC
120V VITAL INSTRUMENT BUS INITIATORS
U1INST
LOSS OF 120VAC INSTRUMENT BUS 1MA
ZZMAU1
LOSS OF 120VAC INSTRUMENT BUS 1MC
ZZMCU1
LOSS OF 120VAC INSTRUMENT BUS 1MB
ZZMBU1
LOSS OF 120VAC INSTRUMENT BUS 1MD
ZZMDU1

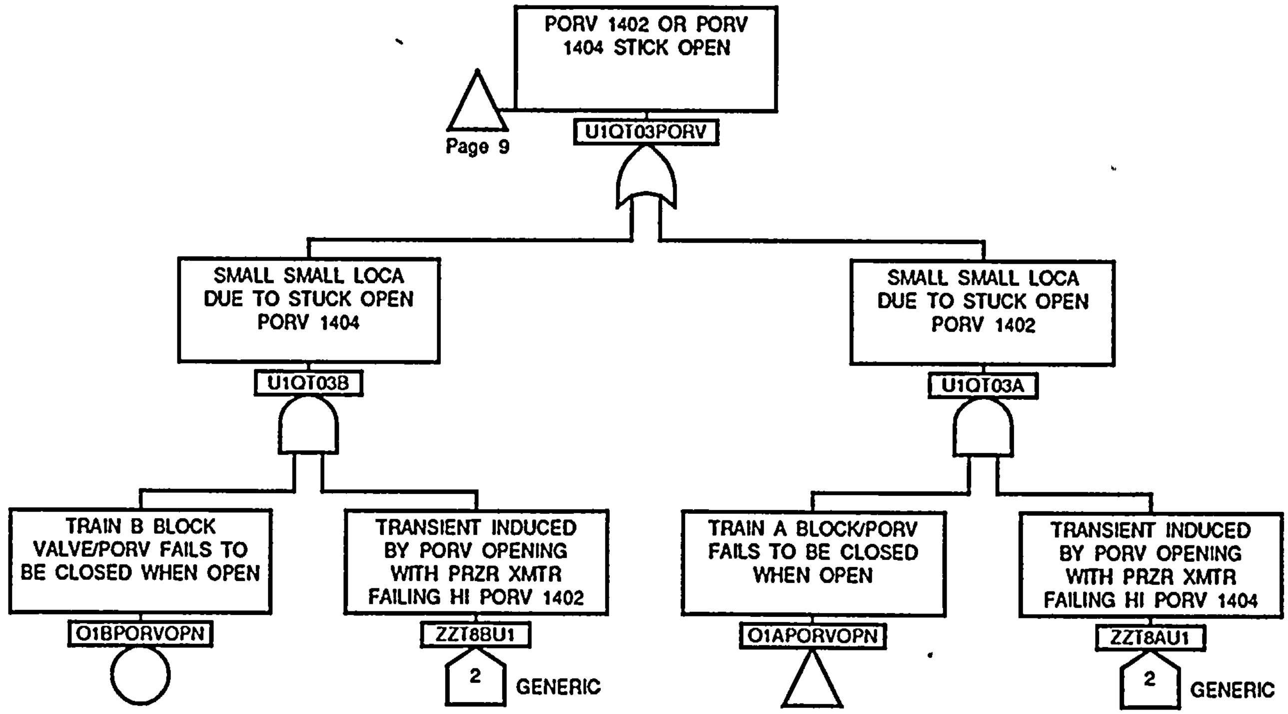

UNIT 1 TRANSIENTS: SIMPLE TRIPS/SUPPORT SYSTEMS
U1GROUP4B
Page 3
LOSS OF ICW
ZZICWU1
GENERIC
SPURIOUS MAIN STEAM ISOLATION SIGNAL
ZZT7MSU1
GENERIC
LOSS OF CCW
ZZCCWU1
GENERIC
LOSS OF MAIN FEEDWATER DUE TO FEEDLINE BREAK ON SG 1A
ZZT3DU1A
GENERIC
LOSS OF TCW
ZZTCWU1
GENERIC
LOSS OF MAIN FEEDWATER DUE TO FEEDLINE BREAK ON SG 1B
ZZT3DU1B
GENERIC
LOSS OF INSTRUMENT AIR
ZZIAU1
GENERIC
1
2

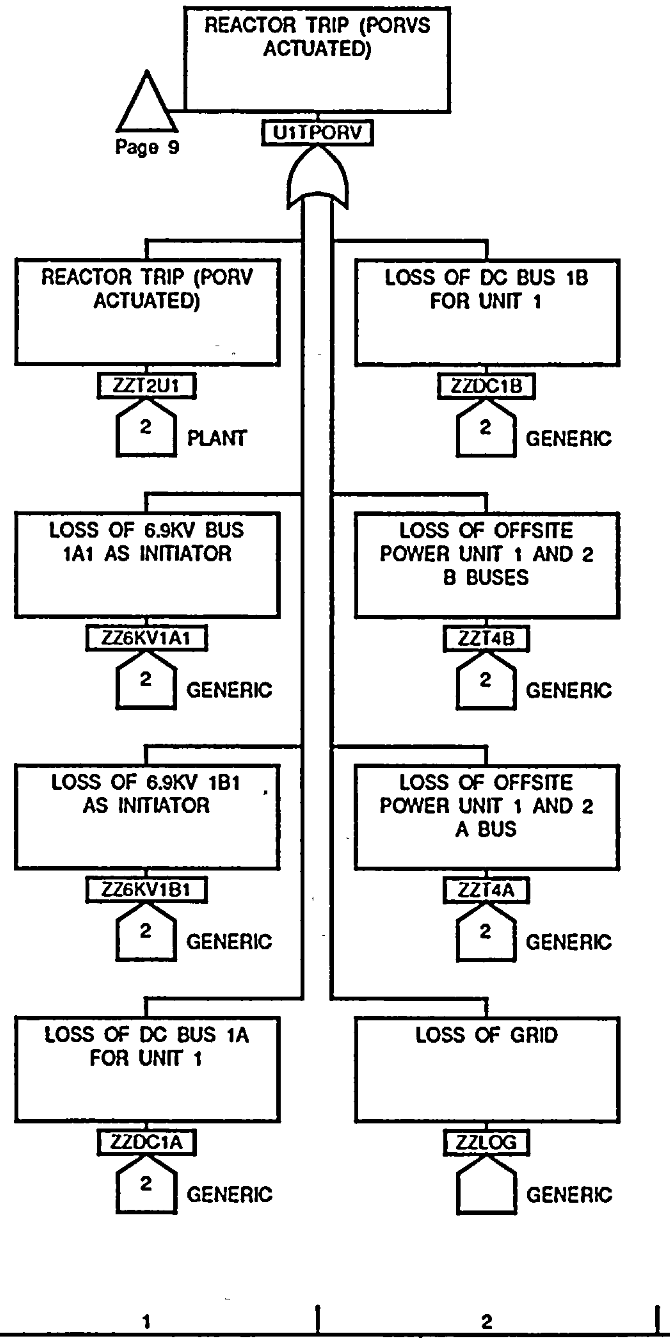

LOCA UNIT 1
Page 3
U1LOCA
SMALL OR SMALL-SMALL LOCA
LARGE LOCA
U1SLOCA
ZZAU1
GENERIC
SMALL-SMALL LOCA
SMALL LOCA
ZZS1U1
GENERIC
ZZS2U1
GENERIC
1
2

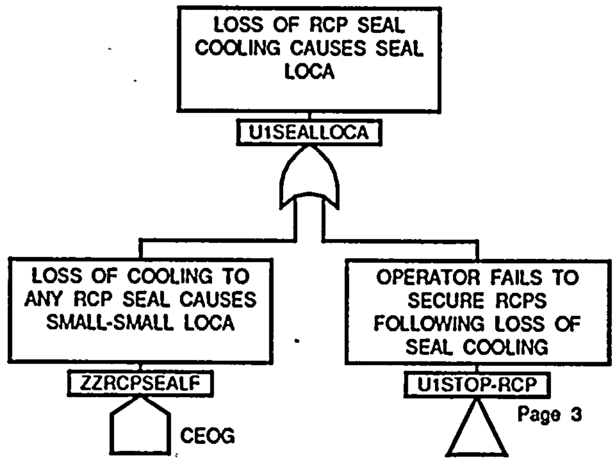

UNIT 1 STEAM GENERATOR TUBE RUPTURE
Page 3
U1SGTR
SG 1A TUBE RUPTURE
SG 1B TUBE RUPTURE
ZZRU1A
GENERIC
ZZRU1B
GENERIC
1
2

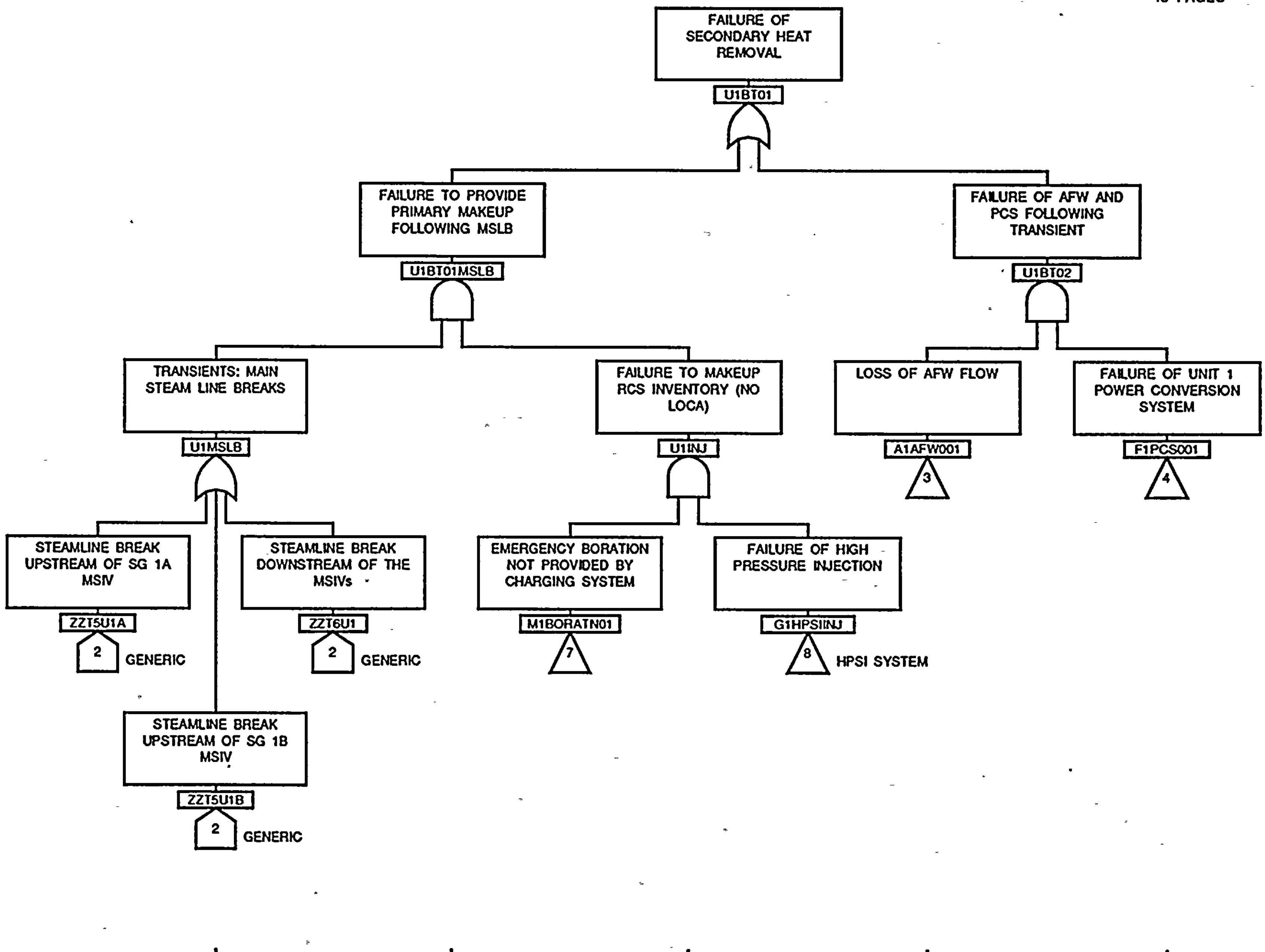

SMALL SMALL LOCA DUE TO STUCK OPEN PORVS
Page 1
U1QT03
PORV 1402 OR PORV 1404 STICK OPEN
U1QT03PORV
Page 10
PORV FO AFTER RCS PRESSURIZATION
U1QT07
1 OF 2 PORVs FAILS TO RECLOSE (AUTO)
O1PORVLOCA
REACTOR TRIP (PORVS ACTUATED)
U1TPORV
Page 11

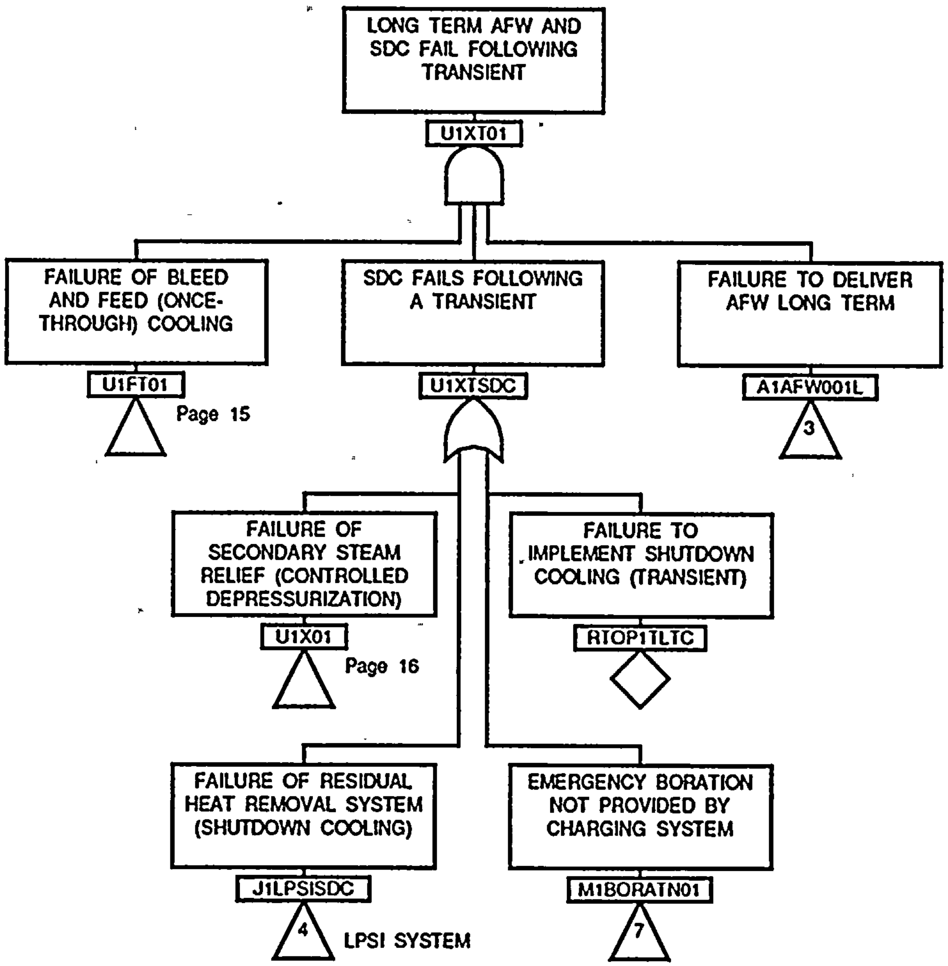
1 2

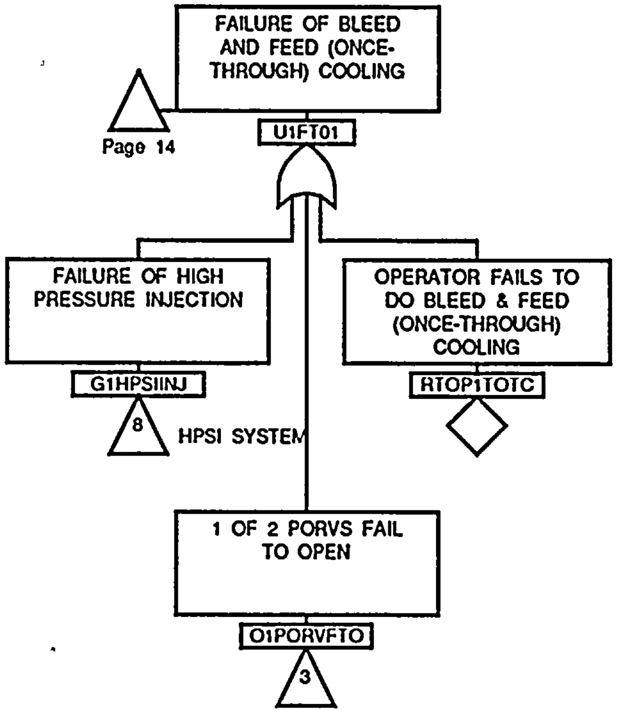

PORV 1402 OR PORV 1404 STICK OPEN
U1QT03PORV
Page 9
SMALL SMALL LOCA DUE TO STUCK OPEN PORV 1404
U1QT03B
SMALL SMALL LOCA DUE TO STUCK OPEN PORV 1402
U1QT03A
TRAIN B BLOCK VALVE/PORV FAILS TO BE CLOSED WHEN OPEN
O1BPORVOPN
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1402
ZZT8BU1
2
GENERIC
TRAIN A BLOCK/PORV FAILS TO BE CLOSED WHEN OPEN
O1APORVOPN
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1404
ZZT8AU1
2
GENERIC
1
2
3
4


REACTOR TRIP (PORVS ACTUATED)
Page 9
U1TPORV
REACTOR TRIP (PORV ACTUATED)
ZZT2U1
2
PLANT
LOSS OF DC BUS 1B FOR UNIT 1
ZZDC1B
2
GENERIC
LOSS OF 6.9KV BUS 1A1 AS INITIATOR
ZZ6KV1A1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 B BUSES
ZZT4B
2
GENERIC
LOSS OF 6.9KV 1B1 AS INITIATOR
ZZ6KV1B1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 A BUS
ZZT4A
2
GENERIC
LOSS OF DC BUS 1A FOR UNIT 1
ZZDC1A
2
GENERIC
LOSS OF GRID
ZZLOG
GENERIC
1
2


LOSS OF RCP SEAL COOLING CAUSES SEAL LOCA
U1SEALLOCA
LOSS OF COOLING TO ANY RCP SEAL CAUSES SMALL-SMALL LOCA
ZZRCPSEALF
CEOG
OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
U1STOP-RCP
Page 3

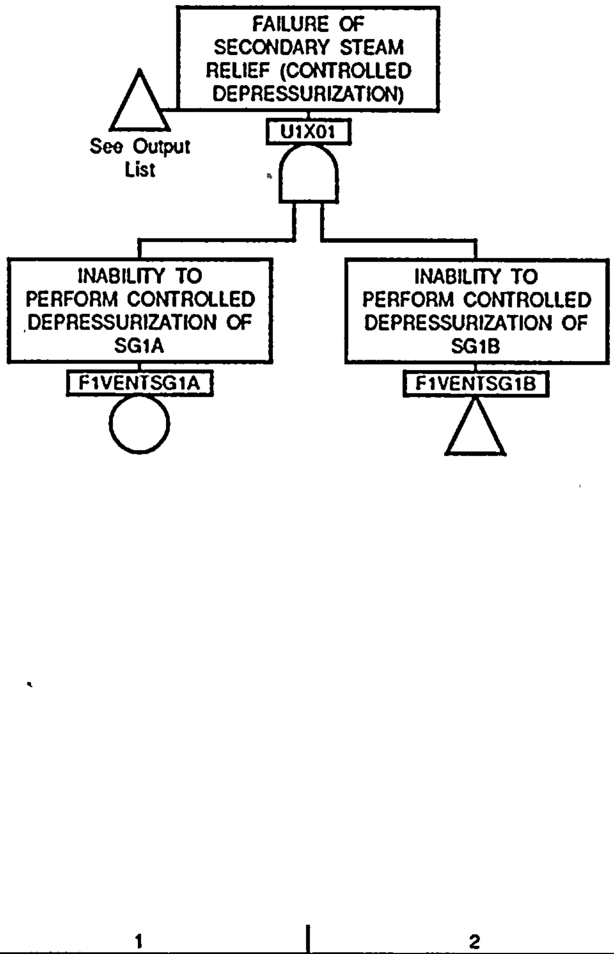

FAILURE OF SECONDARY HEAT REMOVAL
U1BT01
FAILURE TO PROVIDE PRIMARY MAKEUP FOLLOWING MSLB
U1BT01MSLB
FAILURE OF AFW AND PCS FOLLOWING TRANSIENT
U1BT02
TRANSIENTS: MAIN STEAM LINE BREAKS
U1MSLB
FAILURE TO MAKEUP RCS INVENTORY (NO LOCA)
U1INJ
LOSS OF AFW FLOW
A1AFW001
3
FAILURE OF UNIT 1 POWER CONVERSION SYSTEM
F1PCS001
4
STEAMLINE BREAK UPSTREAM OF SG 1A MSIV
ZZT5U1A
2
GENERIC
STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
ZZT6U1
2
GENERIC
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
STEAMLINE BREAK UPSTREAM OF SG 1B MSIV
ZZT5U1B
2
GENERIC
1
2
3
4
5

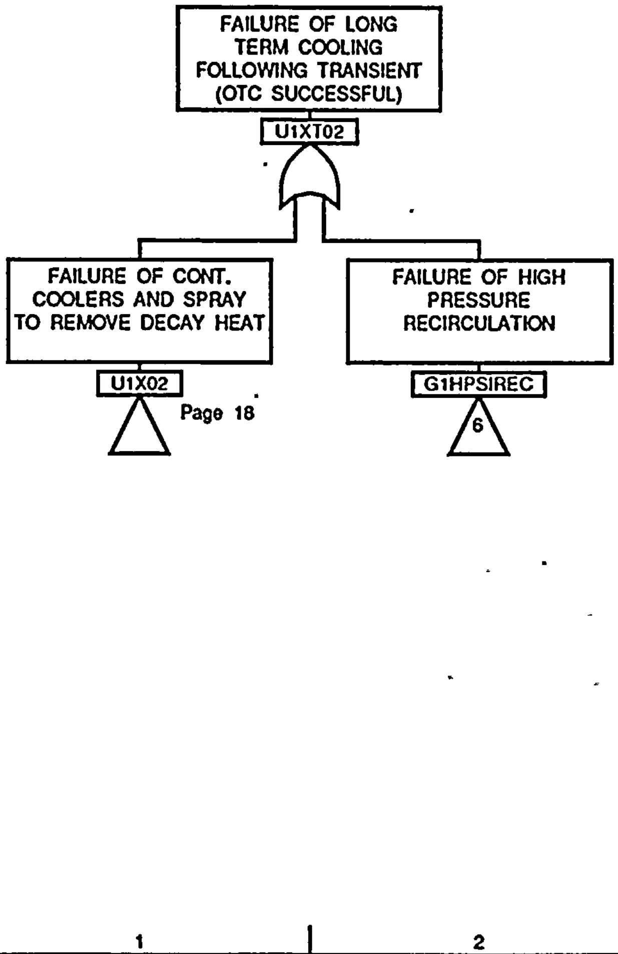

LONG TERM AFW AND SDC FAIL FOLLOWING TRANSIENT
U1XT01
FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U1FT01
Page 15
SDC FAILS FOLLOWING A TRANSIENT
U1XTSDC
FAILURE TO DELIVER AFW LONG TERM
A1AFW001L
3
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U1X01
Page 16
FAILURE TO IMPLEMENT SHUTDOWN COOLING (TRANSIENT)
RTOP1TLTC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J1LPSISDC
4
LPSI SYSTEM
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
1
2
3


FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U1FT01
Page 14
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
OPERATOR FAILS TO DO BLEED & FEED (ONCE-THROUGH) COOLING
RTOP1TOTC
1 OF 2 PORVS FAIL TO OPEN
O1PORVFTO
3

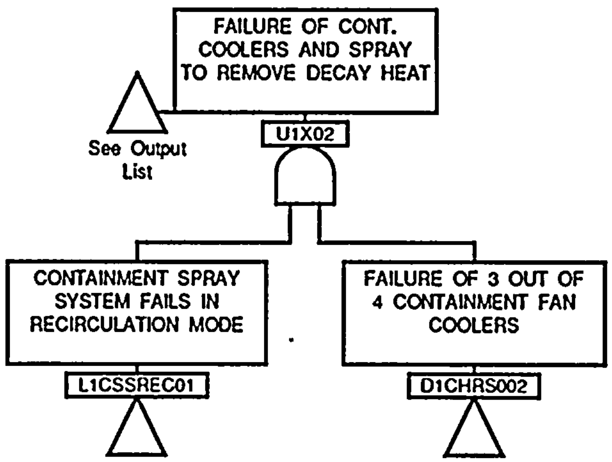
1 2

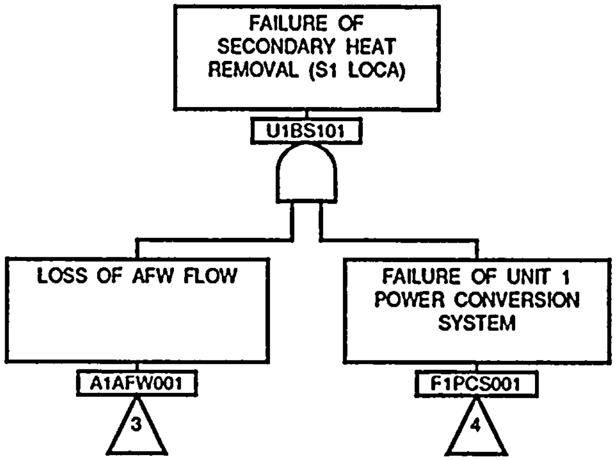
ST. LUCIE UNITS 1 & 2
IPE SUBMITTAL REV. 0
APPENDIX A-1
45 PAGES
U1X01 Outputs:
Page 14, Page 23, Page 31, Page 33, Page 37

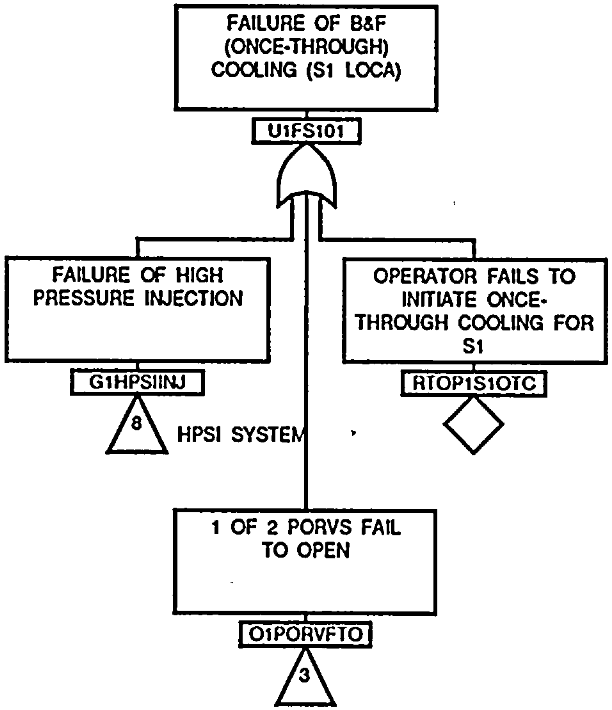


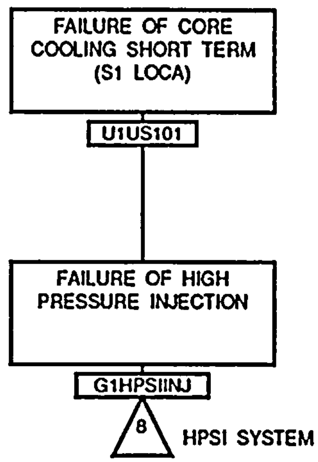

FAILURE OF LONG TERM COOLING FOLLOWING TRANSIENT (OTC SUCCESSFUL)
U1XT02
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
1
2

U1X02 Outputs:
Page 17, Page 22, Page 25, Page 25, Page 27, Page 28, Page 38, Page 43

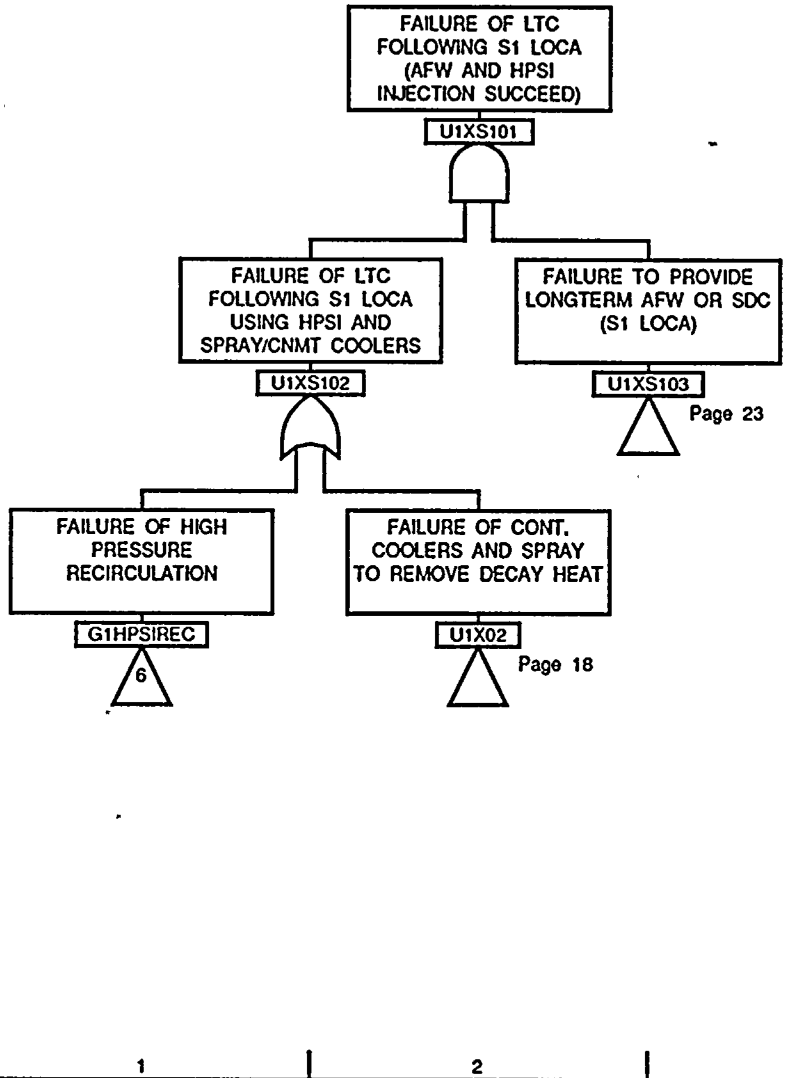


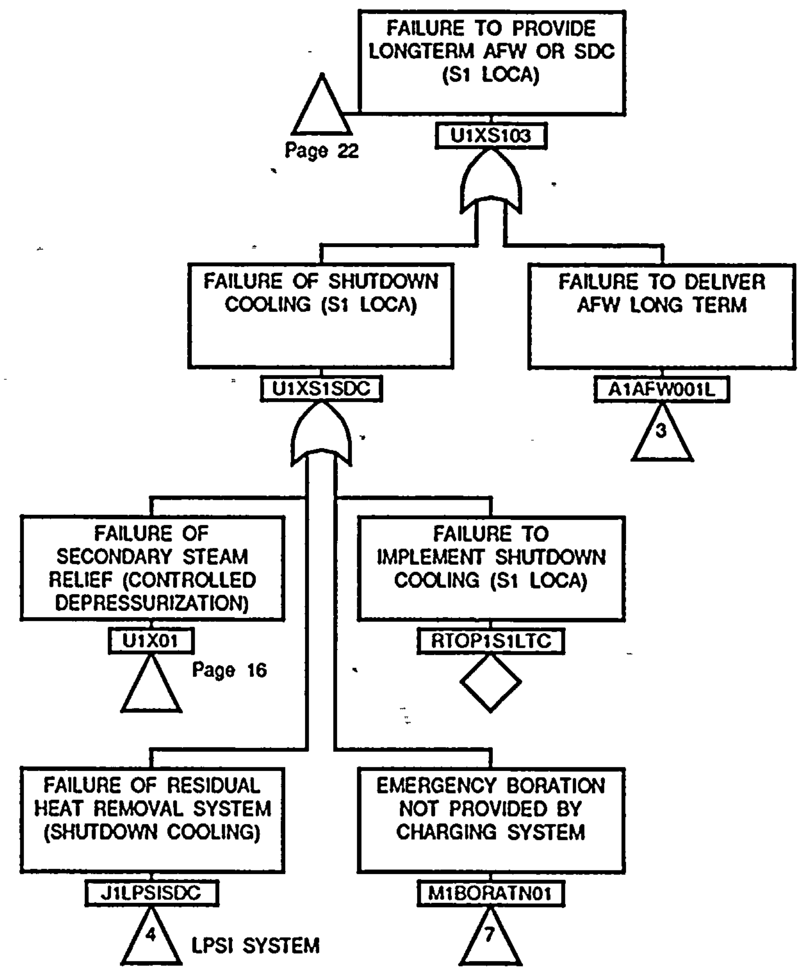

FAILURE OF SECONDARY HEAT REMOVAL (S1 LOCA)
U1BS101
LOSS OF AFW FLOW
A1AFW001
3
FAILURE OF UNIT 1 POWER CONVERSION SYSTEM
F1PCS001
4

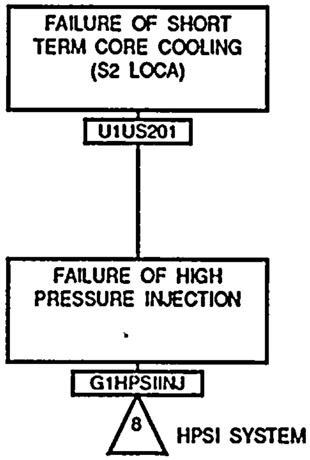

FAILURE OF B&F (ONCE-THROUGH) COOLING (S1 LOCA)
U1FS101
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR S1
RTOP1S1OTC
1 OF 2 PORVS FAIL TO OPEN
O1PORVFTO
3

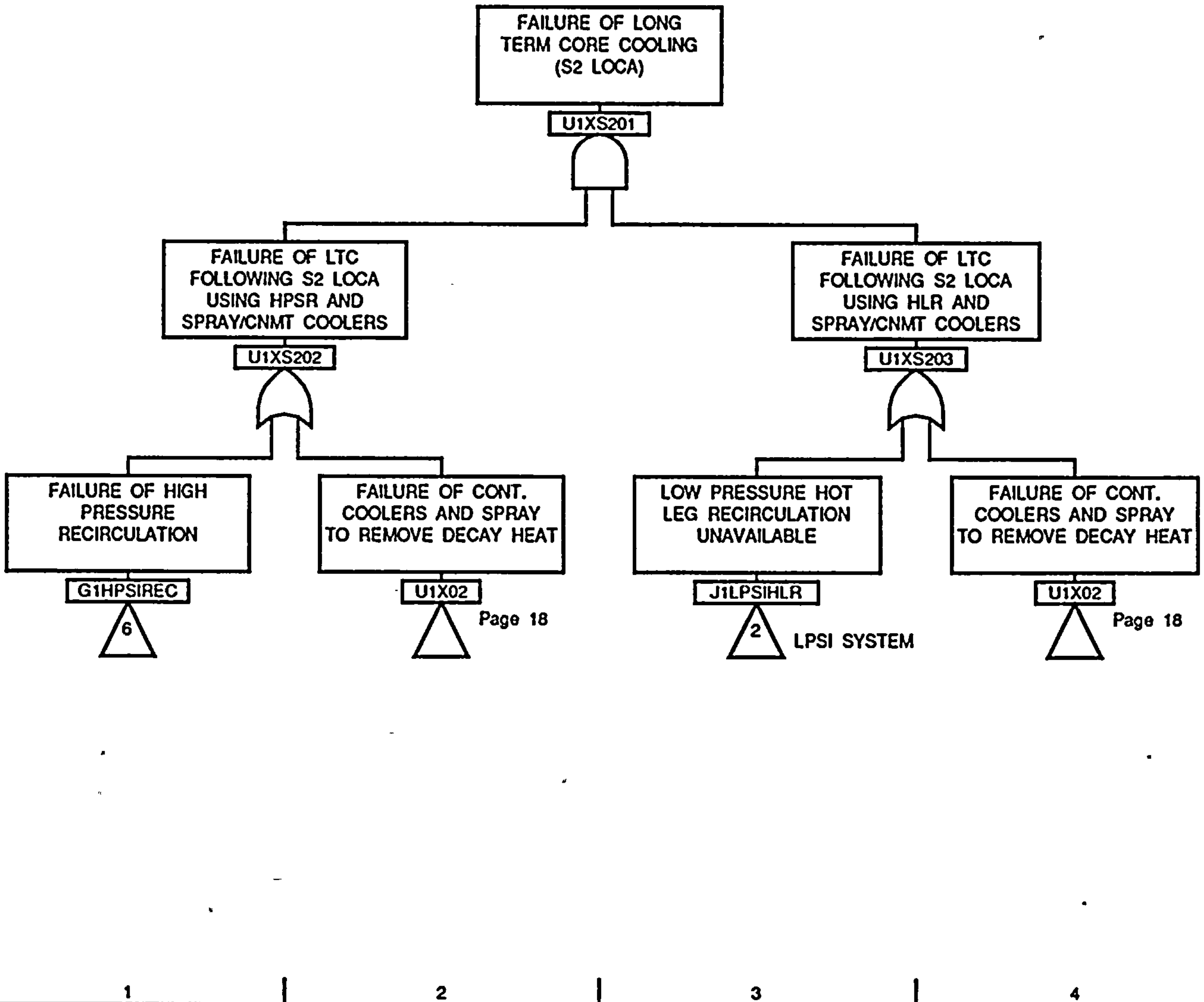

FAILURE OF CORE COOLING SHORT TERM (S1 LOCA)
U1US101
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM

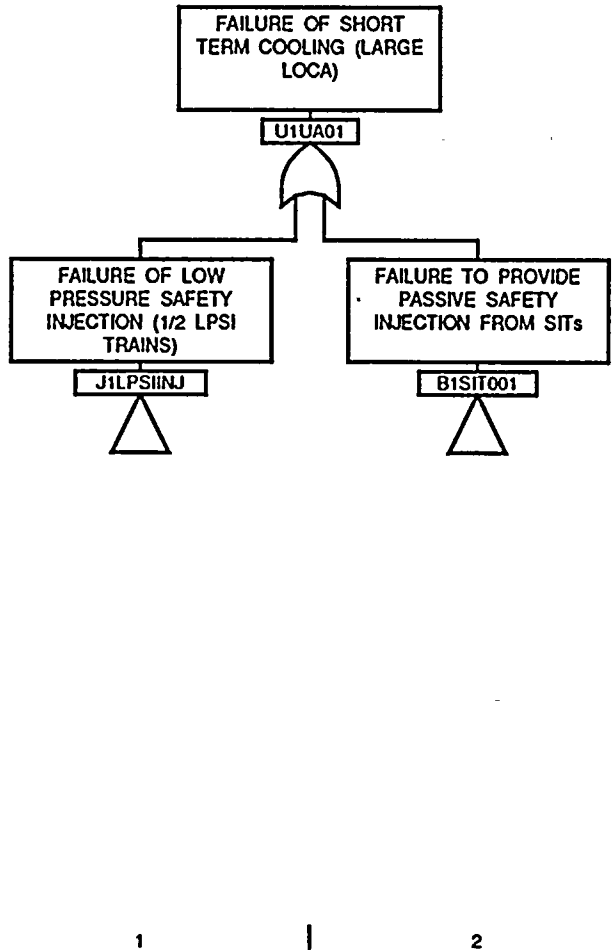
1

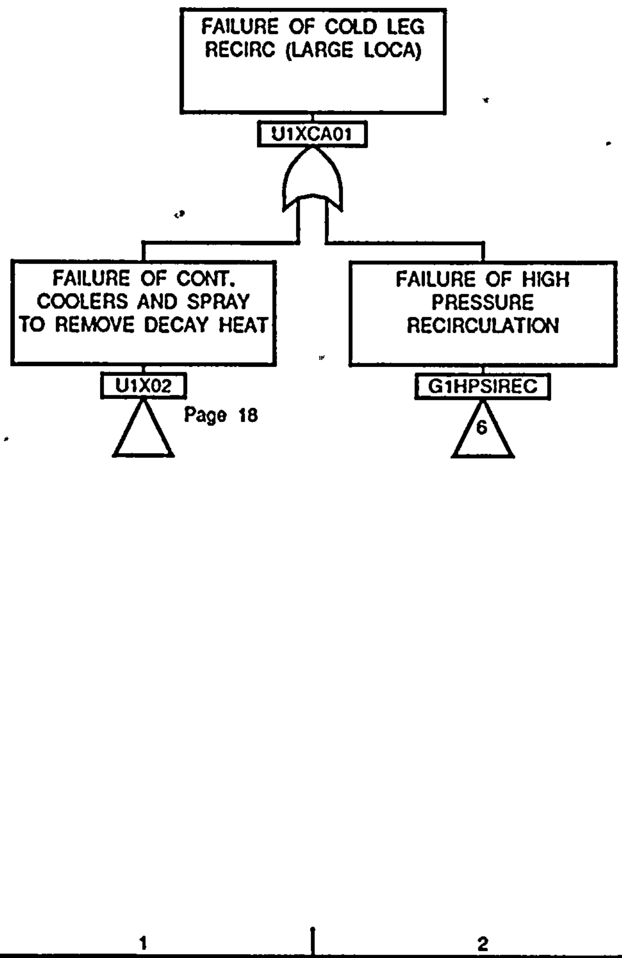

FAILURE OF LTC FOLLOWING S1 LOCA (AFW AND HPSI INJECTION SUCCEED)
U1XS101
FAILURE OF LTC FOLLOWING S1 LOCA USING HPSI AND SPRAY/CNMT COOLERS
U1XS102
FAILURE TO PROVIDE LONGTERM AFW OR SDC (S1 LOCA)
U1XS103
Page 23
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
1
2

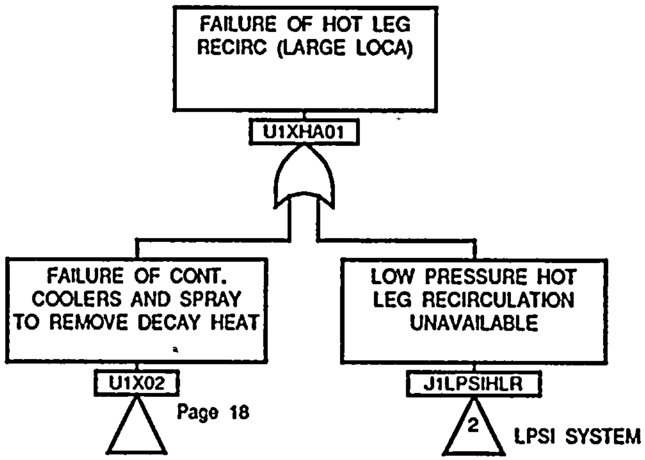

FAILURE TO PROVIDE LONGTERM AFW OR SDC (S1 LOCA)
U1XS103
Page 22
FAILURE OF SHUTDOWN COOLING (S1 LOCA)
U1XS1SDC
FAILURE TO DELIVER AFW LONG TERM
A1AFW001L
3
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U1X01
Page 16
FAILURE TO IMPLEMENT SHUTDOWN COOLING (S1 LOCA)
RTOP1S1LTC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J1LPSISDC
4
LPSI SYSTEM
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
1
2

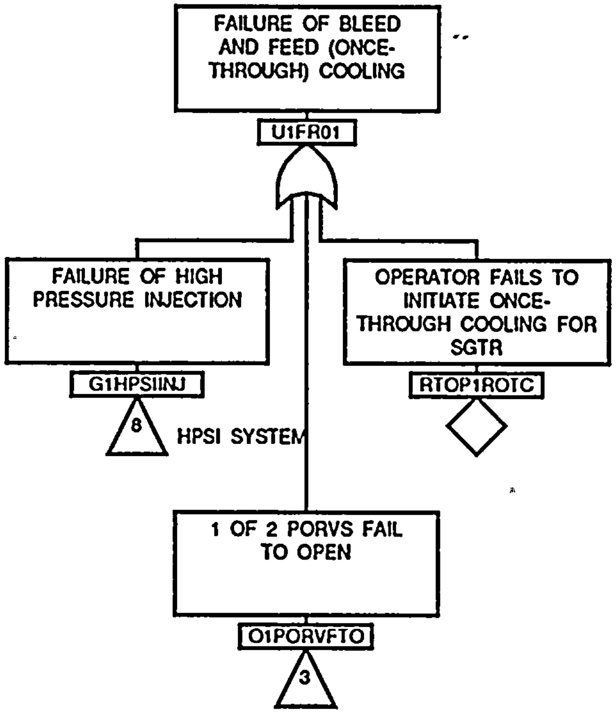

FAILURE OF SHORT TERM CORE COOLING (S2 LOCA)
U1US201
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM

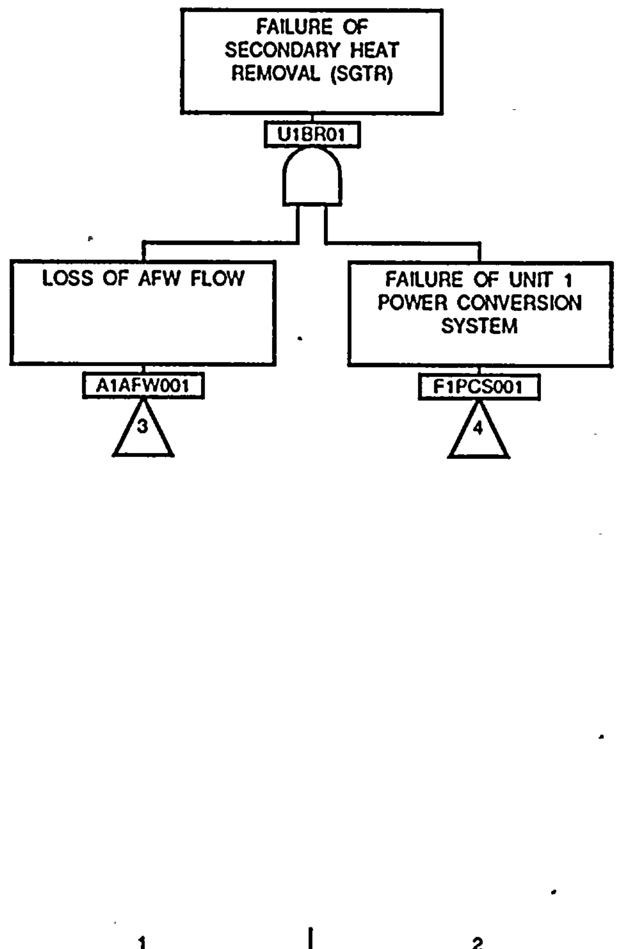

FAILURE OF LONG TERM CORE COOLING (S2 LOCA)
U1XS201
FAILURE OF LTC FOLLOWING S2 LOCA USING HPSR AND SPRAY/CNMT COOLERS
U1XS202
FAILURE OF LTC FOLLOWING S2 LOCA USING HLR AND SPRAY/CNMT COOLERS
U1XS203
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
LOW PRESSURE HOT LEG RECIRCULATION UNAVAILABLE
J1LPSIHLR
2
LPSI SYSTEM
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
1
2
3
4

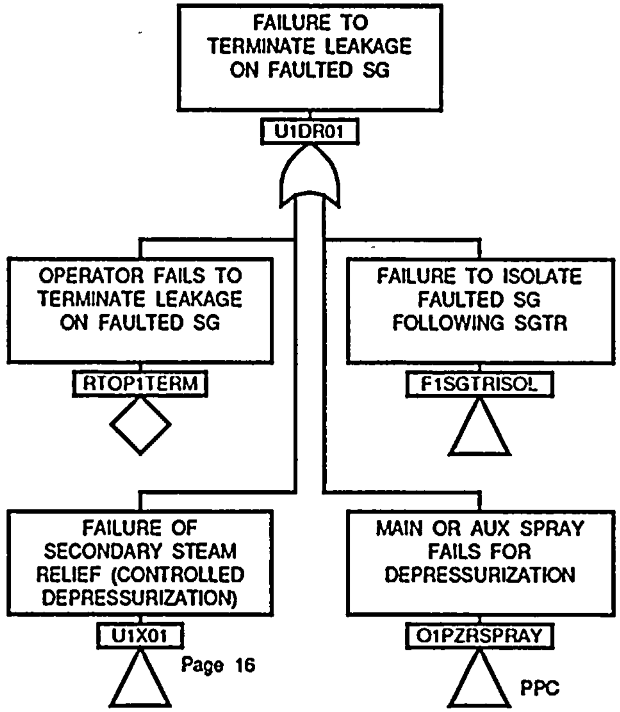

FAILURE OF SHORT TERM COOLING (LARGE LOCA)
U1UA01
FAILURE OF LOW PRESSURE SAFETY INJECTION (1/2 LPSI TRAINS)
J1LPSIINJ
FAILURE TO PROVIDE PASSIVE SAFETY INJECTION FROM SITs
B1SIT001
1
2

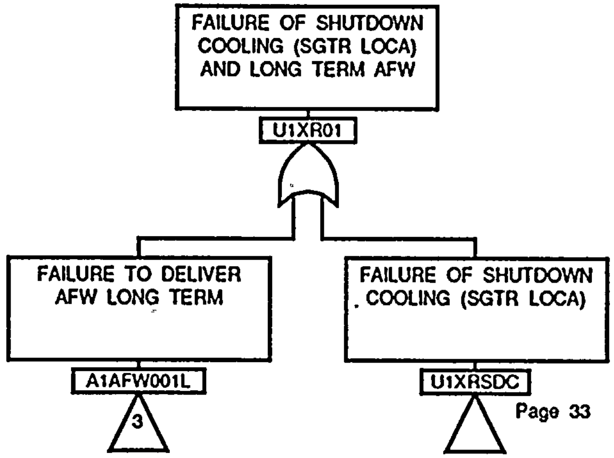

FAILURE OF COLD LEG RECIRC (LARGE LOCA)
U1XCA01
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
1
2


FAILURE OF HOT LEG RECIRC (LARGE LOCA)
U1XHA01
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
LOW PRESSURE HOT LEG RECIRCULATION UNAVAILABLE
J1LPSIHLR
2
LPSI SYSTEM


FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U1FR01
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR
RTOP1ROTC
1 OF 2 PORVS FAIL TO OPEN
O1PORVFTO
3


FAILURE OF SECONDARY HEAT REMOVAL (SGTR)
U1BR01
LOSS OF AFW FLOW
A1AFW001
3
FAILURE OF UNIT 1 POWER CONVERSION SYSTEM
F1PCS001
4
1
2


FAILURE TO TERMINATE LEAKAGE ON FAULTED SG
U1DR01
OPERATOR FAILS TO TERMINATE LEAKAGE ON FAULTED SG
RTOP1TERM
FAILURE TO ISOLATE FAULTED SG FOLLOWING SGTR
F1SGTRISOL
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U1X01
Page 16
MAIN OR AUX SPRAY FAILS FOR DEPRESSURIZATION
O1PZRSPRAY
PPC
1
2


FAILURE OF SHUTDOWN COOLING (SGTR LOCA) AND LONG TERM AFW
U1XR01
FAILURE TO DELIVER AFW LONG TERM
A1AFW001L
3
FAILURE OF SHUTDOWN COOLING (SGTR LOCA)
U1XRSDC
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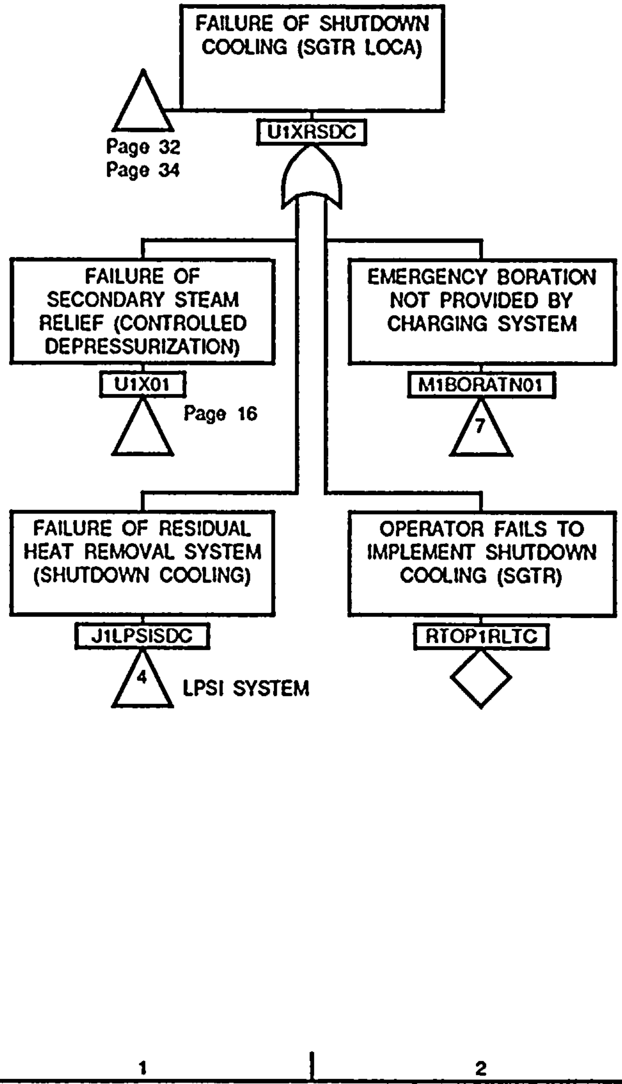
FAILURE OF SHUTDOWN COOLING (SGTR LOCA)
U1XRSDC
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FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U1X01
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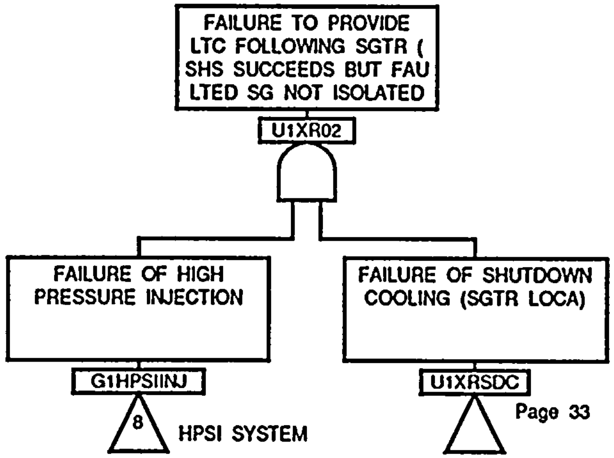
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J1LPSISDC
4
LPSI SYSTEM
OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)
RTOP1RLTC
1
2


FAILURE TO PROVIDE LTC FOLLOWING SGTR (SHS SUCCEEDS BUT FAU LTED SG NOT ISOLATED
U1XR02
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
FAILURE OF SHUTDOWN COOLING (SGTR LOCA)
U1XRSDC
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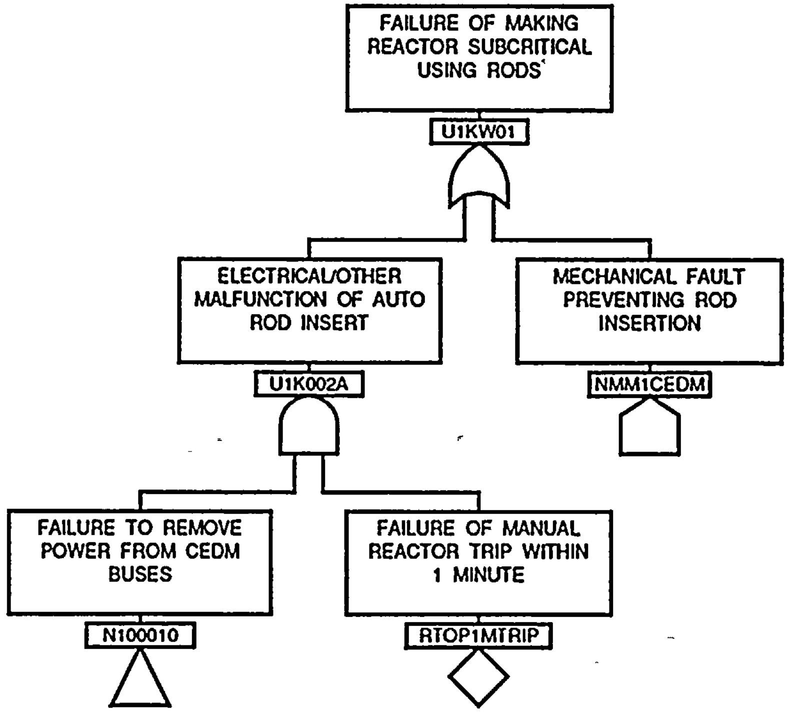
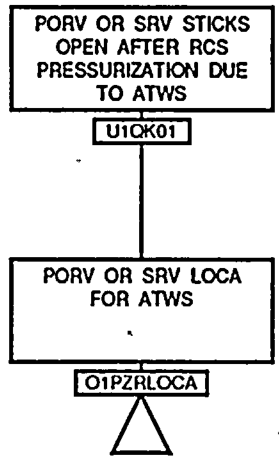
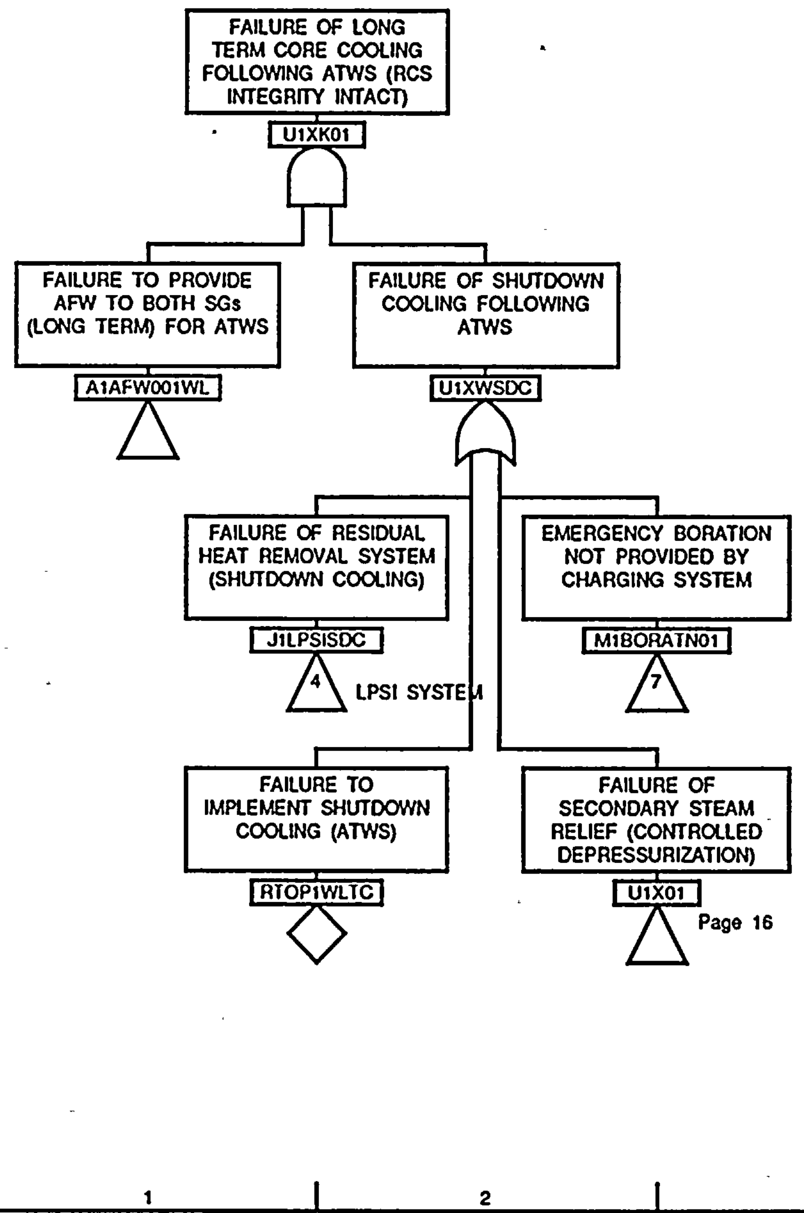

FAILURE OF MAKING REACTOR SUBCRITICAL USING RODS
U1KW01
ELECTRICAL/OTHER MALFUNCTION OF AUTO ROD INSERT
U1K002A
MECHANICAL FAULT PREVENTING ROD INSERTION
NMM1CEDM
FAILURE TO REMOVE POWER FROM CEDM BUSES
N100010
FAILURE OF MANUAL REACTOR TRIP WITHIN 1 MINUTE
RTOP1MTRIP
1
2


PORV OR SRV STICKS OPEN AFTER RCS PRESSURIZATION DUE TO ATWS
U1QK01
PORV OR SRV LOCA FOR ATWS
O1PZRLOCA


FAILURE OF LONG TERM CORE COOLING FOLLOWING ATWS (RCS INTEGRITY INTACT)
U1XK01
FAILURE TO PROVIDE AFW TO BOTH SGs (LONG TERM) FOR ATWS
A1AFW001WL
FAILURE OF SHUTDOWN COOLING FOLLOWING ATWS
U1XWSDC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J1LPSISDC
4
LPSI SYSTEM
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
FAILURE TO IMPLEMENT SHUTDOWN COOLING (ATWS)
RTOP1WLTC
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U1X01
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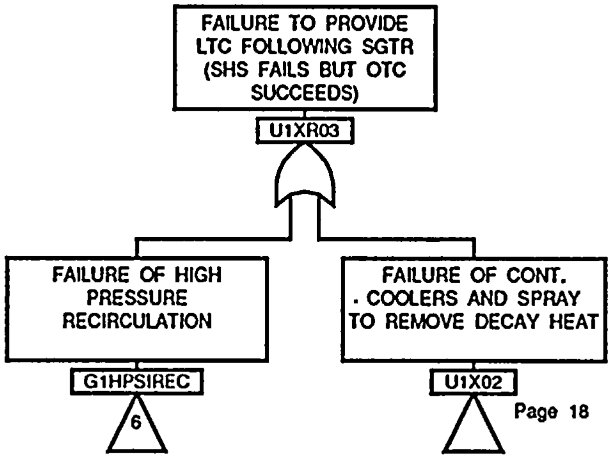
FAILURE TO PROVIDE LTC FOLLOWING SGTR (SHS FAILS BUT OTC SUCCEEDS)
U1XR03
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
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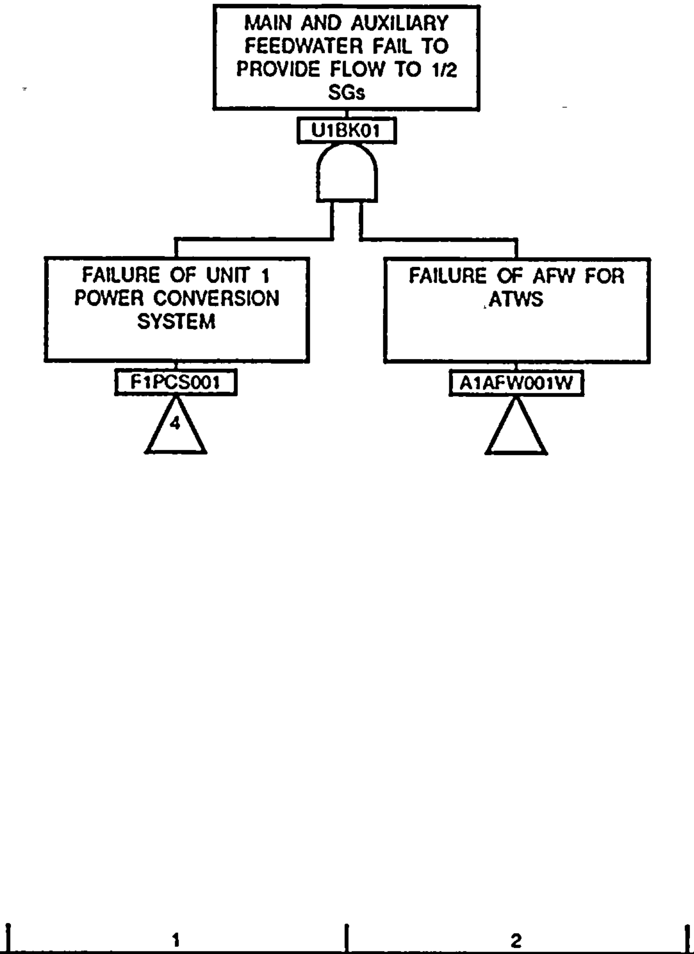
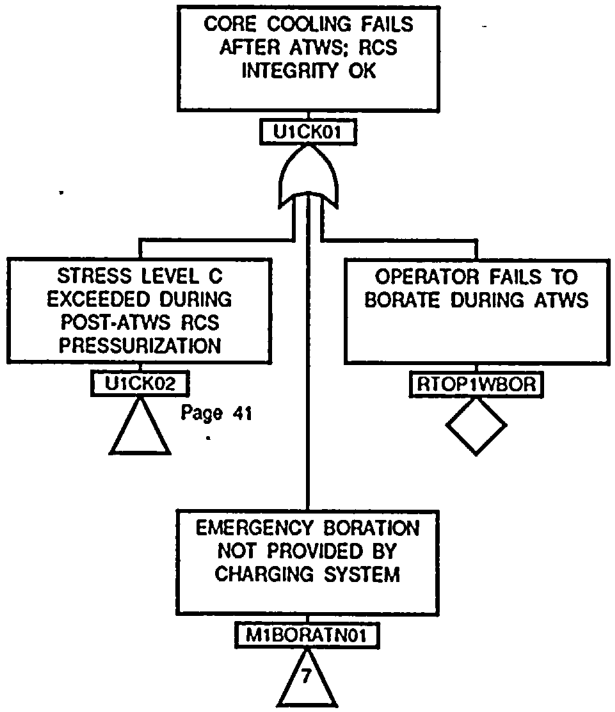
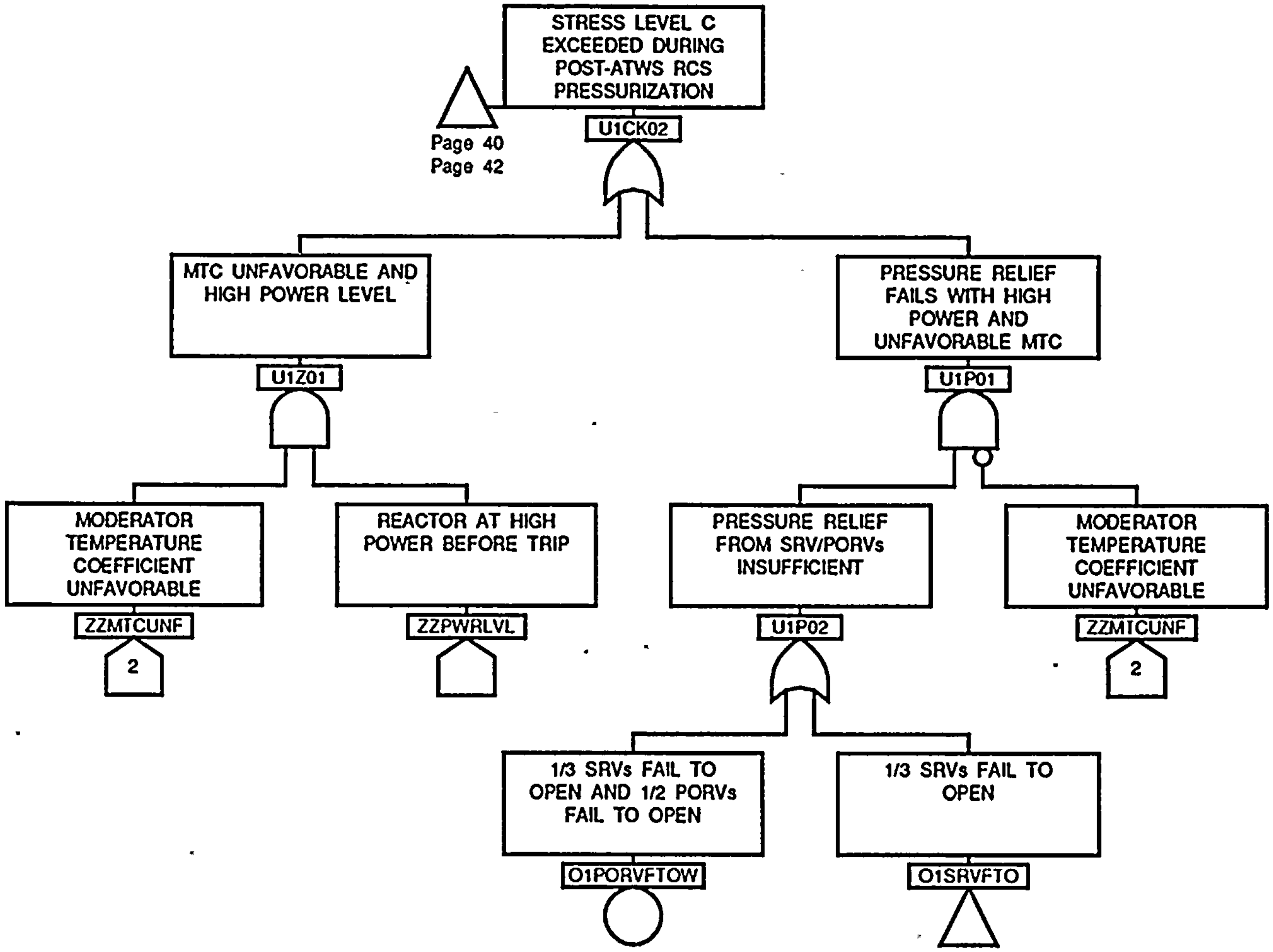

MAIN AND AUXILIARY FEEDWATER FAIL TO PROVIDE FLOW TO 1/2 SGs
U1BK01
FAILURE OF UNIT 1 POWER CONVERSION SYSTEM
F1PCS001
4
FAILURE OF AFW FOR ATWS
A1AFW001W
1
2


CORE COOLING FAILS AFTER ATWS; RCS INTEGRITY OK
U1CK01
STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
U1CK02
Page 41
OPERATOR FAILS TO BORATE DURING ATWS
RTOP1WBOR
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
1
2


STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
U1CK02
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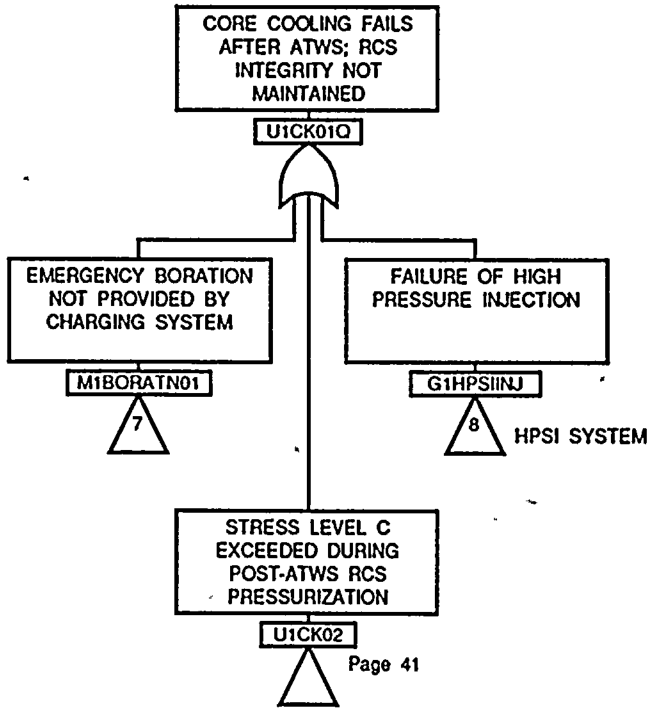
MTC UNFAVORABLE AND HIGH POWER LEVEL
U1Z01
PRESSURE RELIEF FAILS WITH HIGH POWER AND UNFAVORABLE MTC
U1P01
MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
ZZMTCUNF
2
REACTOR AT HIGH POWER BEFORE TRIP
ZZPWRLVL
PRESSURE RELIEF FROM SRV/PORVs INSUFFICIENT
U1P02
MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
ZZMTCUNF
2
1/3 SRVs FAIL TO OPEN AND 1/2 PORVs FAIL TO OPEN
O1PORVFTOW
1/3 SRVs FAIL TO OPEN
O1SRVFTO
1
2
3
4


CORE COOLING FAILS AFTER ATWS; RCS INTEGRITY NOT MAINTAINED
U1CK01Q
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M1BORATN01
7
FAILURE OF HIGH PRESSURE INJECTION
G1HPSIINJ
8
HPSI SYSTEM
STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
U1CK02
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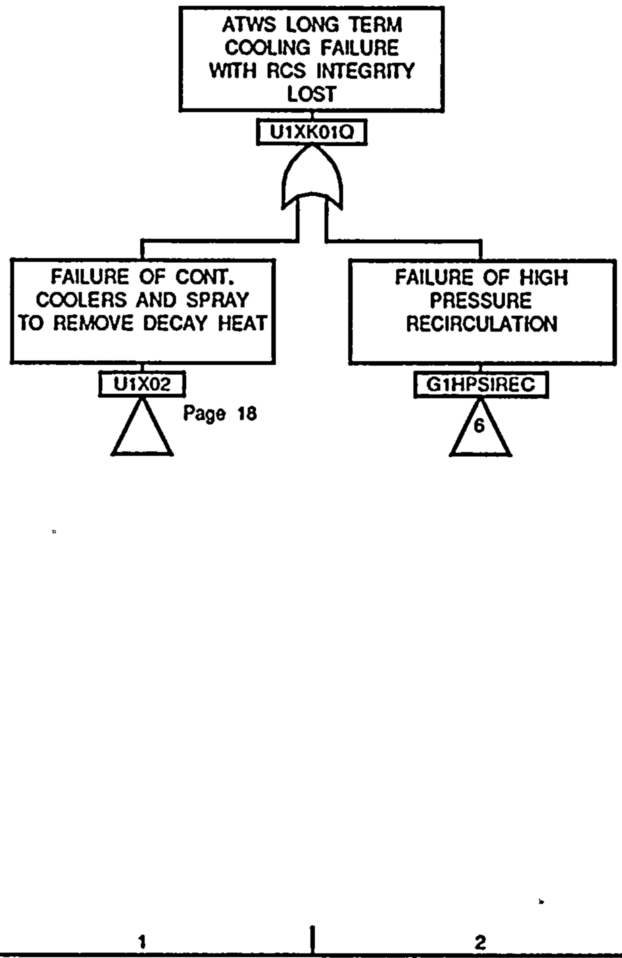
1
2


ATWS LONG TERM COOLING FAILURE WITH RCS INTEGRITY LOST
U1XK01Q
FAILURE OF CONT. COOLERS AND SPRAY TO REMOVE DECAY HEAT
U1X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION
G1HPSIREC
6
1
2

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
A1AFW001	13	5	J1LPSISDC	14	2	U1BS101	19	2	U1QT03PORV	9	1
A1AFW001	19	1	J1LPSISDC	23	1	U1BT01	13	4	U1QT03PORV	10	2
A1AFW001	30	1	J1LPSISDC	33	1	U1BT01MSLB	13	3	U1QT07	9	2
A1AFW001L	14	3	J1LPSISDC	37	2	U1BT02	13	5	U1RCP1A1	1	4
A1AFW001L	23	3	L1CSSREC01	18	1	U1CK01	40	2	U1RCP1A1	2	1
A1AFW001L	32	1	M1BORATN01	13	3	U1CK01Q	42	2	U1RCP1A2	1	3
A1AFW001W	39	2	M1BORATN01	14	3	U1CK02	40	1	U1RCP1B1	1	2
A1AFW001WL	37	1	M1BORATN01	23	2	U1CK02	41	2	U1RCP1B2	1	1
B1SIT001	26	2	M1BORATN01	33	2	U1CK02	42	2	U1SEALLOCA	12	2
C1NRCP1A1	2	1	M1BORATN01	37	3	U1DR01	31	2	U1SGTR	3	6
C1NRCP1A2	1	3	M1BORATN01	40	2	U1FR01	29	2	U1SGTR	8	2
C1NRCP1B1	1	2	M1BORATN01	42	1	U1FS101	20	2	U1SLOCA	7	2
C1NRCP1B2	1	1	N100010	35	1	U1FT01	14	1	U1STOP-RCP	1	4
D1CHRS002	18	2	NMM1CEDM	35	3	U1FT01	15	2	U1STOP-RCP	3	2
F1PCS001	13	6	O1APORVOPN	10	3	U1GROUP2	3	1	U1STOP-RCP	12	2
F1PCS001	19	2	O1BPORVOPN	10	1	U1GROUP2	4	2	U1TPORV	9	3
F1PCS001	30	2	O1PORVFTO	15	2	U1GROUP3	3	2	U1TPORV	11	2
F1PCS001	39	1	O1PORVFTO	20	2	U1GROUP4A	3	3	U1UA01	26	2
F1SGTRISOL	31	2	O1PORVFTO	29	2	U1GROUP4A	5	4	U1US101	21	1
F1VENTSG1A	16	1	O1PORVFTOW	41	3	U1GROUP4B	3	4	U1US201	24	1
F1VENTSG1B	16	2	O1PORVLOCA	9	2	U1GROUP4B	6	2	U1X01	14	2
G1HPSIINJ	13	4	O1PZRLOCA	36	1	U1INJ	13	4	U1X01	16	2
G1HPSIINJ	15	1	O1PZRSPRAY	31	2	U1INST	5	7	U1X01	23	1
G1HPSIINJ	20	1	O1SRVFTO	41	4	U1K002A	35	2	U1X01	31	1
G1HPSIINJ	21	1	RTOP1MTRIP	35	2	U1KW01	35	2	U1X01	33	1
G1HPSIINJ	24	1	RTOP1RLTC	33	2	U1LOCA	3	5	U1X01	37	3
G1HPSIINJ	29	1	RTOP1ROTC	29	2	U1LOCA	7	2	U1X02	17	1
G1HPSIINJ	34	1	RTOP1S1LTC	23	2	U1MSLB	13	2	U1X02	18	2
G1HPSIINJ	42	2	RTOP1S1OTC	20	2	U1P01	41	4	U1X02	22	2
G1HPSIREC	17	2	RTOP1S1RCP	3	1	U1P02	41	3	U1X02	25	2
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G1HPSIREC	43	2	RTOP1WLTC	37	2	U1QT03	1	4	U1X02	43	1
J1LPSIHLR	25	3	U1-NOT-LOG	3	4	U1QT03	9	2	U1XCA01	27	2
J1LPSIHLR	28	2	U1BK01	39	2	U1QT03A	10	4	U1XHA01	28	2
J1LPSIINJ	26	1	U1BR01	30	2	U1QT03B	10	2	U1XK01	37	2

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
U1XK01Q	43	2	ZZMDU1	5	8						
U1XR01	32	2	ZZMTCUNF	41	1						
U1XR02	34	2	ZZMTCUNF	41	4						
U1XR03	38	2	ZZPWRLVL	41	2						
U1XRSDC	32	2	ZZRCPSEALF	12	1						
U1XRSDC	33	2	ZZRU1A	8	1						
U1XRSDC	34	2	ZZRU1B	8	2						
U1XS101	22	2	ZZS1U1	7	1						
U1XS102	22	2	ZZS2U1	7	2						
U1XS103	22	3	ZZT1U1	3	2						
U1XS103	23	2	ZZT2U1	3	2						
U1XS1SDC	23	2	ZZT2U1	11	1						
U1XS201	25	2	ZZT3AU1	3	2						
U1XS202	25	2	ZZT3CU1	3	3						
U1XS203	25	4	ZZT3DU1A	6	2						
U1XT01	14	2	ZZT3DU1B	6	2						
U1XT02	17	2	ZZT3EU1	3	3						
U1XTSDC	14	2	ZZT4A	5	4						
U1XWSDC	37	2	ZZT4A	11	2						
U1Z01	41	2	ZZT4B	5	3						
ZZ4KV1A2	5	5	ZZT4B	11	2						
ZZ4KV1B2	5	6	ZZT5U1A	4	2						
ZZ6KV1A1	5	1	ZZT5U1A	13	1						
ZZ6KV1A1	11	1	ZZT5U1B	4	2						
ZZ6KV1B1	5	2	ZZT5U1B	13	2						
ZZ6KV1B1	11	1	ZZT6U1	4	2						
ZZAU1	7	3	ZZT6U1	13	2						
ZZCCWU1	6	1	ZZT7MSU1	6	2						
ZZDC1A	4	1	ZZT7SIU1	4	1						
ZZDC1A	11	1	ZZT8AU1	4	1						
ZZDC1B	4	1	ZZT8AU1	10	4						
ZZDC1B	11	2	ZZT8BU1	4	2						
ZZIAU1	6	2	ZZT8BU1	10	2						
ZZICWU1	6	1	ZZTCWU1	6	1						
ZZLOG	11	2									
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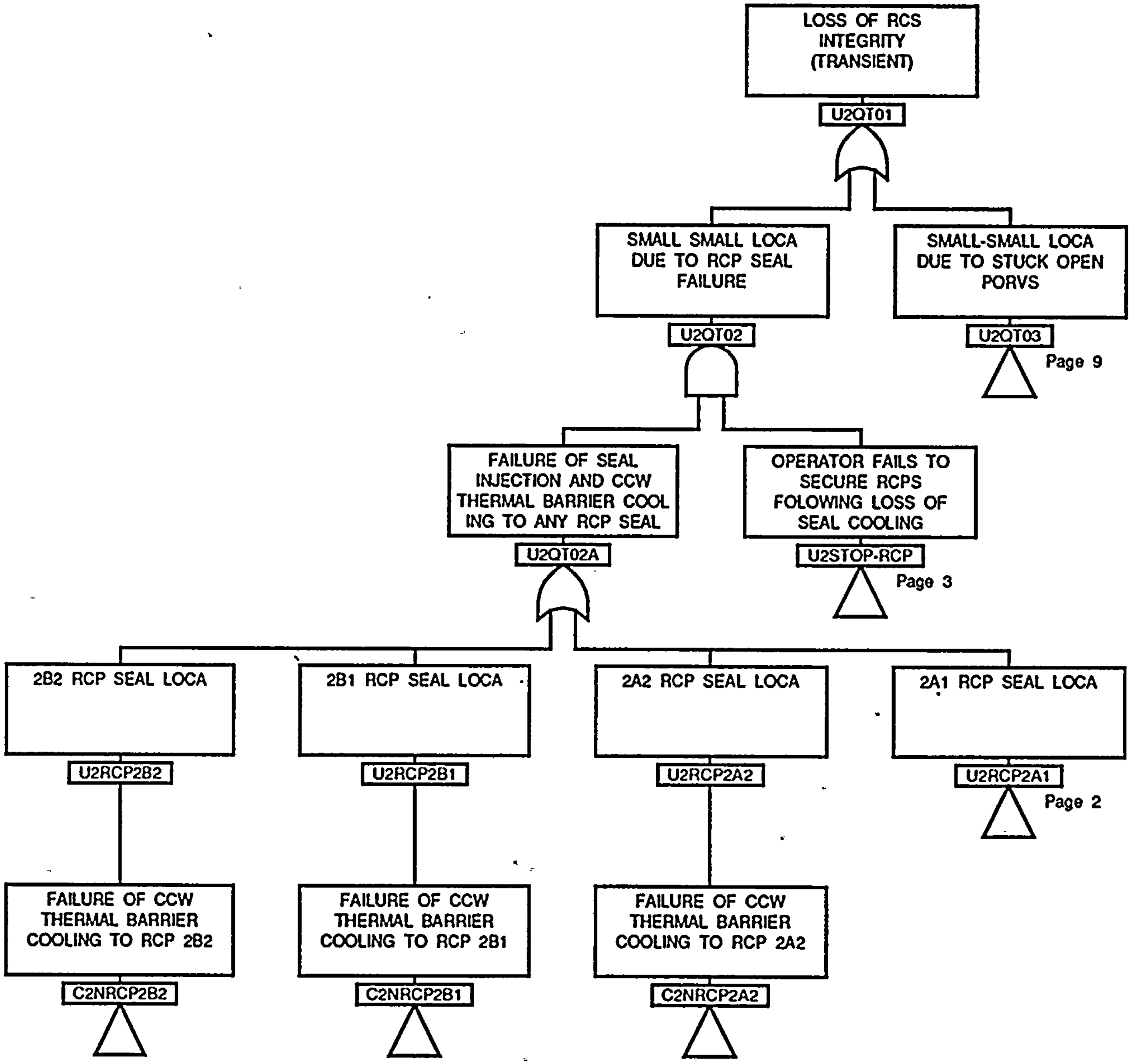
LOSS OF RCS INTEGRITY (TRANSIENT)
U2QT01
SMALL SMALL LOCA DUE TO RCP SEAL FAILURE
U2QT02
SMALL-SMALL LOCA DUE TO STUCK OPEN PORVS
U2QT03
Page 9
FAILURE OF SEAL INJECTION AND CCW THERMAL BARRIER COOL ING TO ANY RCP SEAL
U2QT02A
OPERATOR FAILS TO SECURE RCPS FOLOWING LOSS OF SEAL COOLING
U2STOP-RCP
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2B2 RCP SEAL LOCA
U2RCP2B2
2B1 RCP SEAL LOCA
U2RCP2B1
2A2 RCP SEAL LOCA
U2RCP2A2
2A1 RCP SEAL LOCA
U2RCP2A1
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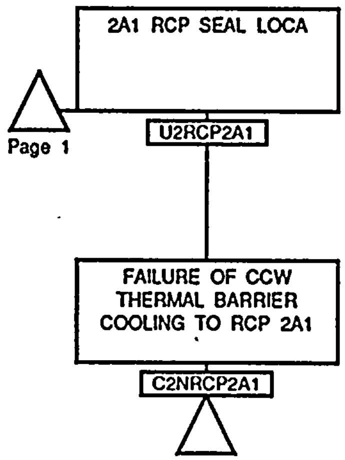
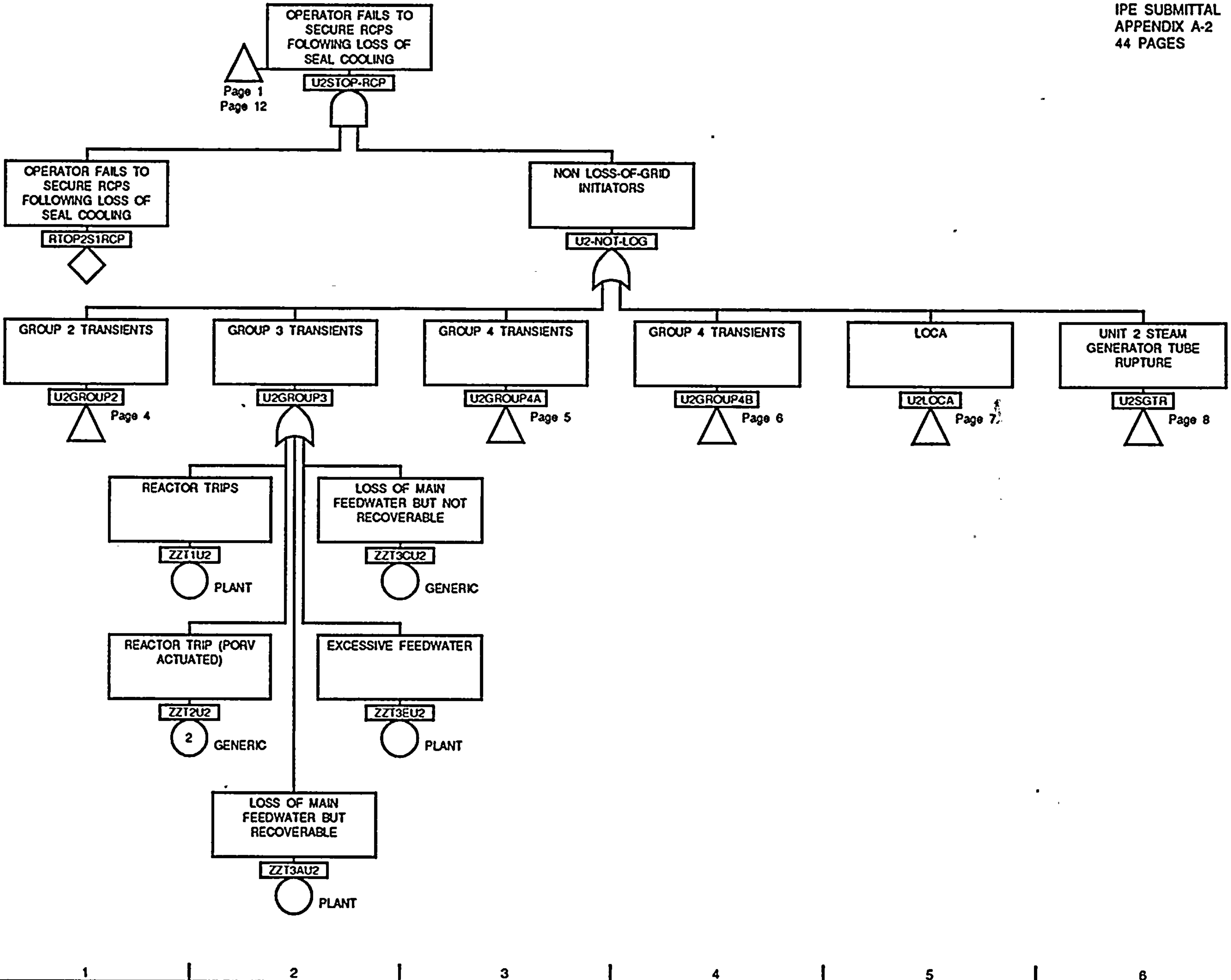
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 2B2
C2NRCP2B2
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 2B1
C2NRCP2B1
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 2A2
C2NRCP2A2

1 2 3 4


2A1 RCP SEAL LOCA
Page 1
U2RCP2A1
FAILURE OF CCW THERMAL BARRIER COOLING TO RCP 2A1
C2NRCP2A1


OPERATOR FAILS TO SECURE RCPS FOLOWING LOSS OF SEAL COOLING
U2STOP-RCP
Page 1
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OPERATOR FAILS TO SECURE RCPS FOLLOWING LOSS OF SEAL COOLING
RTOP2S1RCP
NON LOSS-OF-GRID INITIATORS
U2-NOT-LOG
GROUP 2 TRANSIENTS
U2GROUP2
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GROUP 3 TRANSIENTS
U2GROUP3
GROUP 4 TRANSIENTS
U2GROUP4A
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GROUP 4 TRANSIENTS
U2GROUP4B
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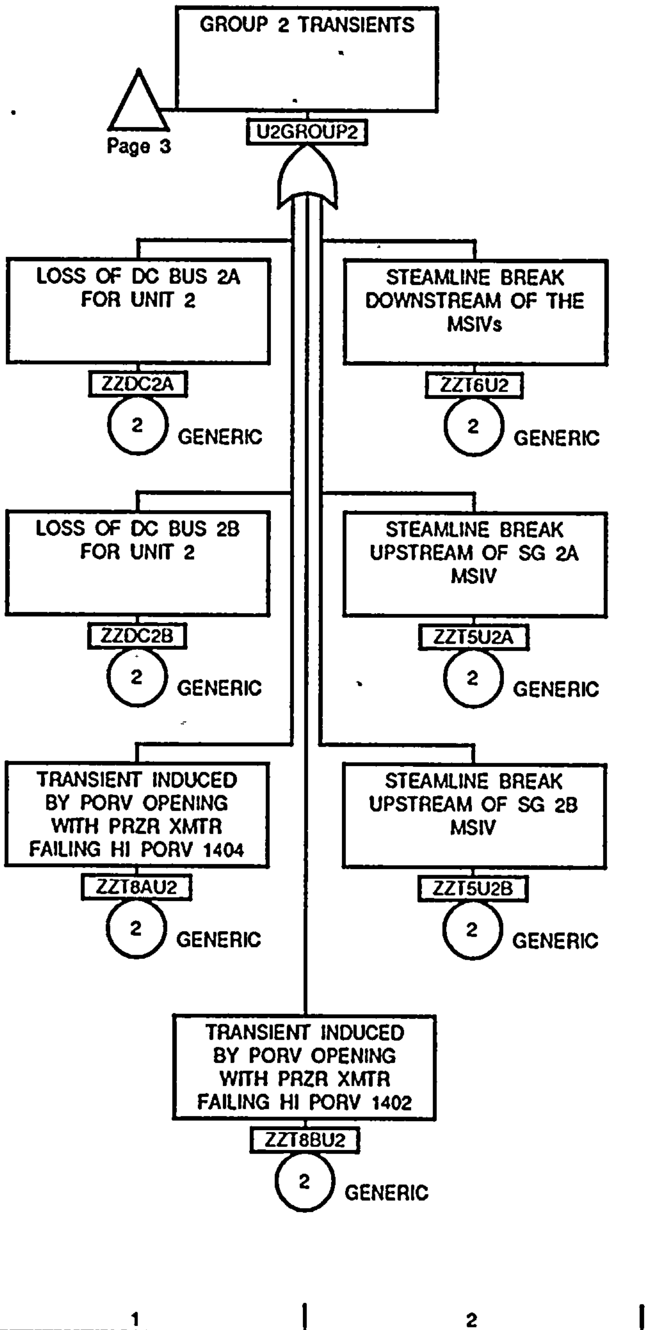
LOCA
U2LOCA
Page 7
UNIT 2 STEAM GENERATOR TUBE RUPTURE
U2SGTR
Page 8
REACTOR TRIPS
ZZT1U2
PLANT
LOSS OF MAIN FEEDWATER BUT NOT RECOVERABLE
ZZT3CU2
GENERIC
REACTOR TRIP (PORV ACTUATED)
ZZT2U2
2
GENERIC
EXCESSIVE FEEDWATER
ZZT3EU2
PLANT
LOSS OF MAIN FEEDWATER BUT RECOVERABLE
ZZT3AU2
PLANT
1
2
3
4
5
6


GROUP 2 TRANSIENTS
Page 3
U2GROUP2
LOSS OF DC BUS 2A FOR UNIT 2
ZZDC2A
2
GENERIC
STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
ZZT6U2
2
GENERIC
LOSS OF DC BUS 2B FOR UNIT 2
ZZDC2B
2
GENERIC
STEAMLINE BREAK UPSTREAM OF SG 2A MSIV
ZZT5U2A
2
GENERIC
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1404
ZZT8AU2
2
GENERIC
STEAMLINE BREAK UPSTREAM OF SG 2B MSIV
ZZT5U2B
2
GENERIC
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1402
ZZT8BU2
2
GENERIC
1
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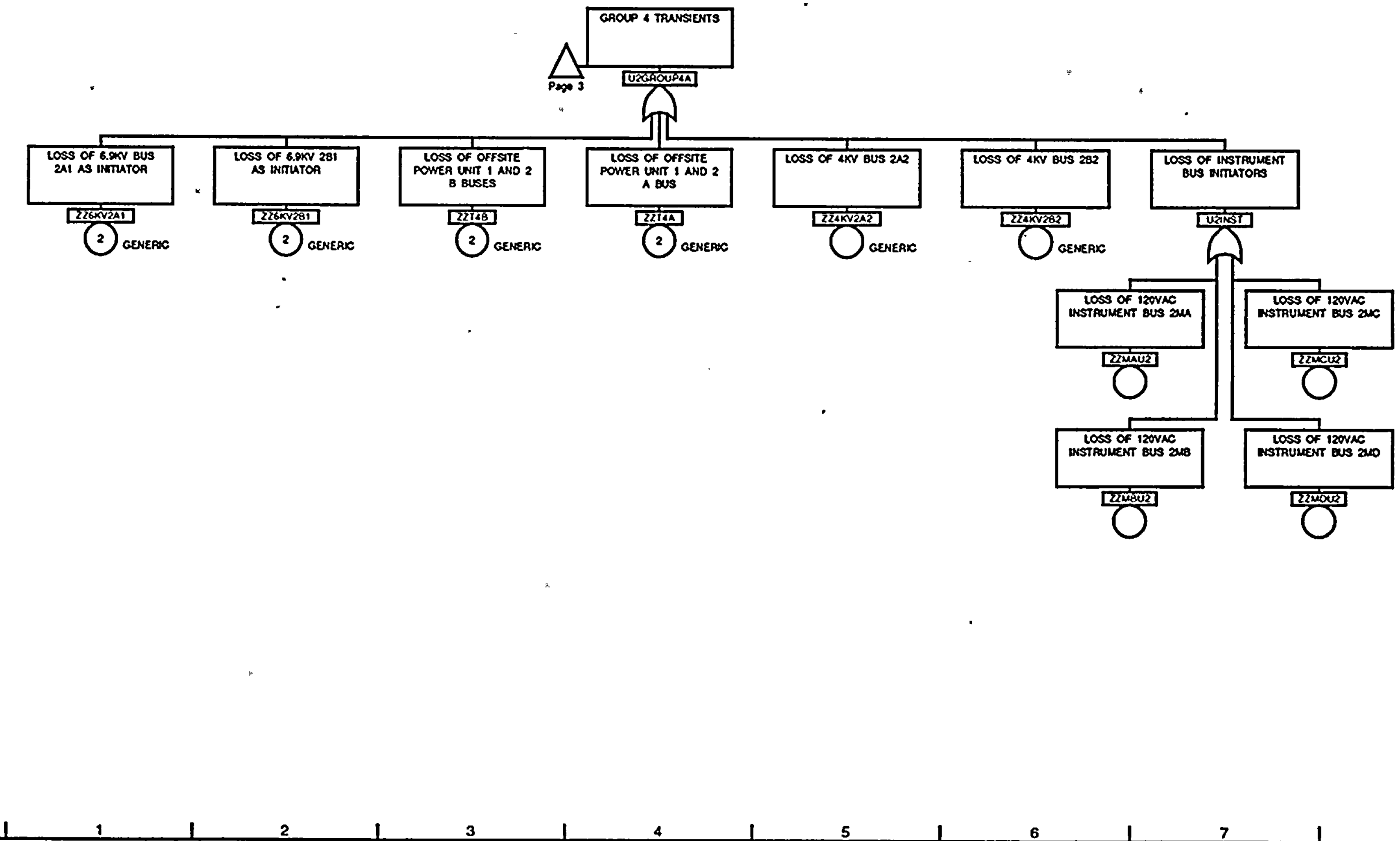
GROUP 4 TRANSIENTS
Page 3
U2GROUP4A
LOSS OF 6.9KV BUS 2A1 AS INITIATOR
ZZ6KV2A1
2
GENERIC
LOSS OF 6.9KV 2B1 AS INITIATOR
ZZ6KV2B1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 B BUSES
ZZT4B
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 A BUS
ZZT4A
2
GENERIC
LOSS OF 4KV BUS 2A2
ZZ4KV2A2
GENERIC
LOSS OF 4KV BUS 2B2
ZZ4KV2B2
GENERIC
LOSS OF INSTRUMENT BUS INITIATORS
U2INST
LOSS OF 120VAC INSTRUMENT BUS 2MA
ZZMAU2
LOSS OF 120VAC INSTRUMENT BUS 2MC
ZZMCU2
LOSS OF 120VAC INSTRUMENT BUS 2MB
ZZMBU2
LOSS OF 120VAC INSTRUMENT BUS 2MD
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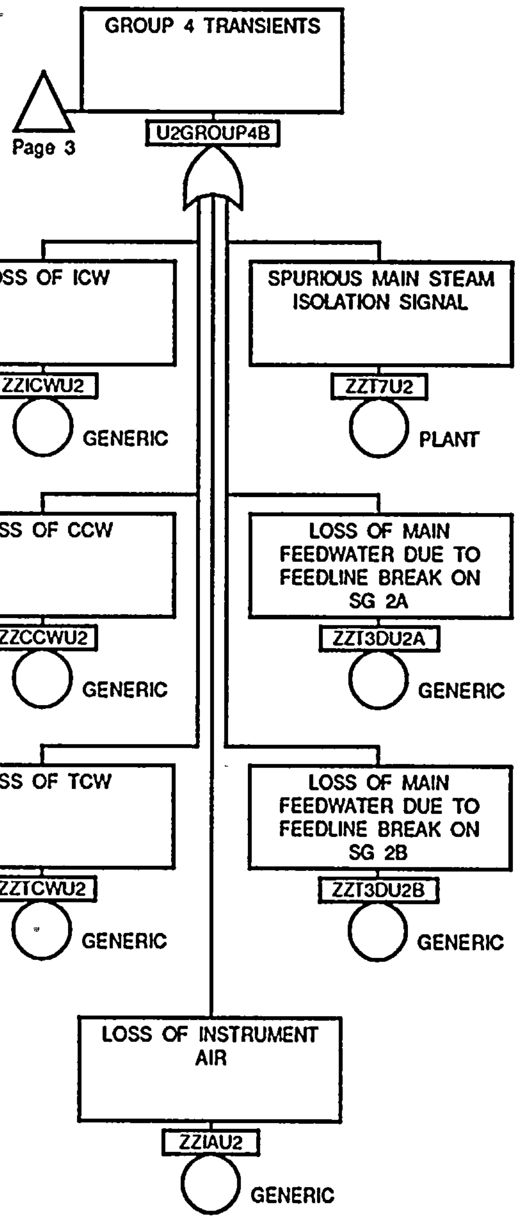
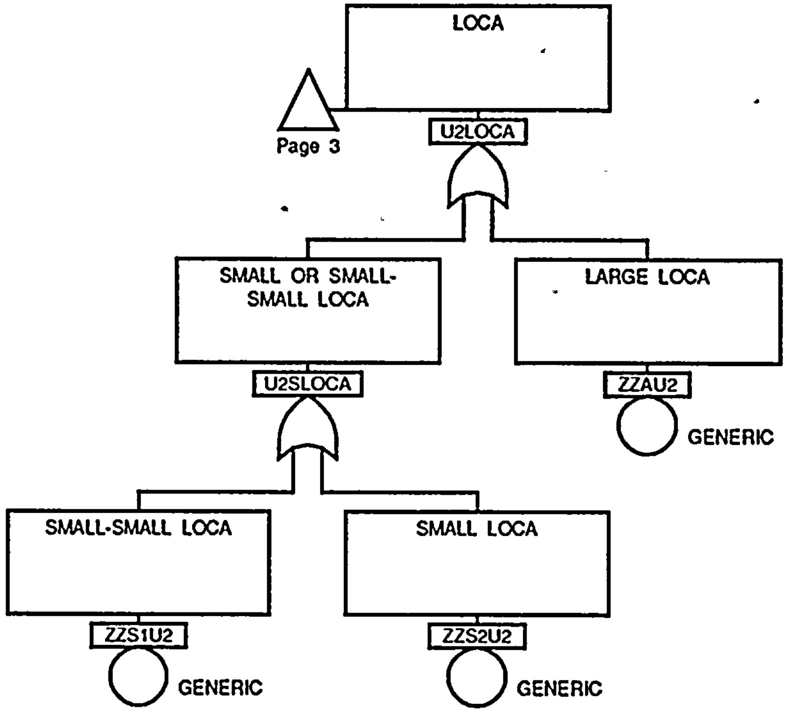
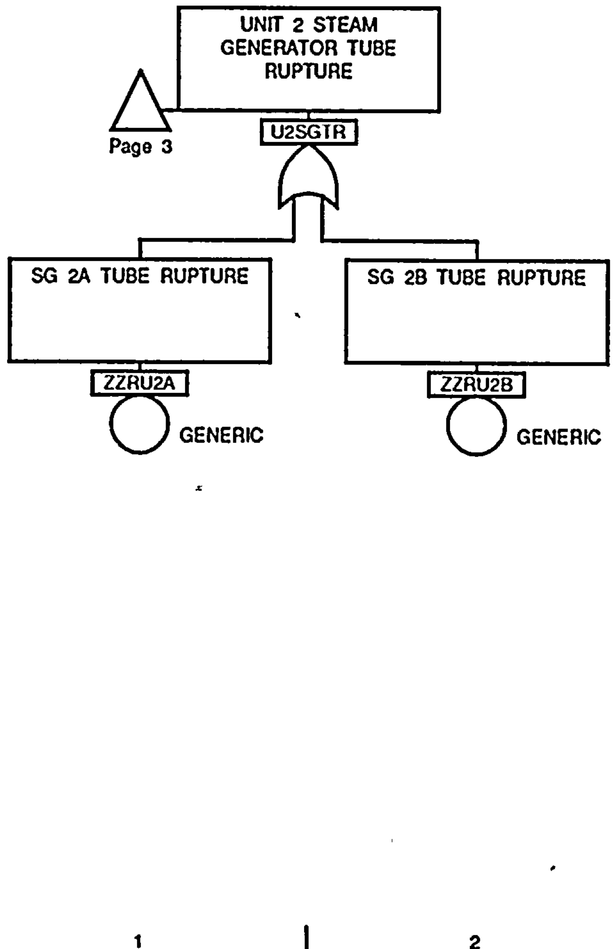
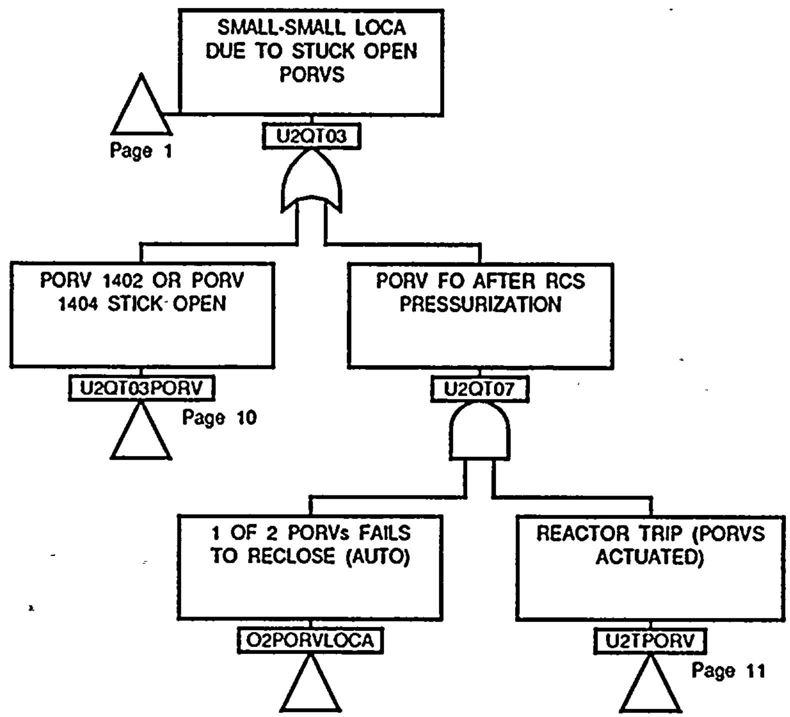
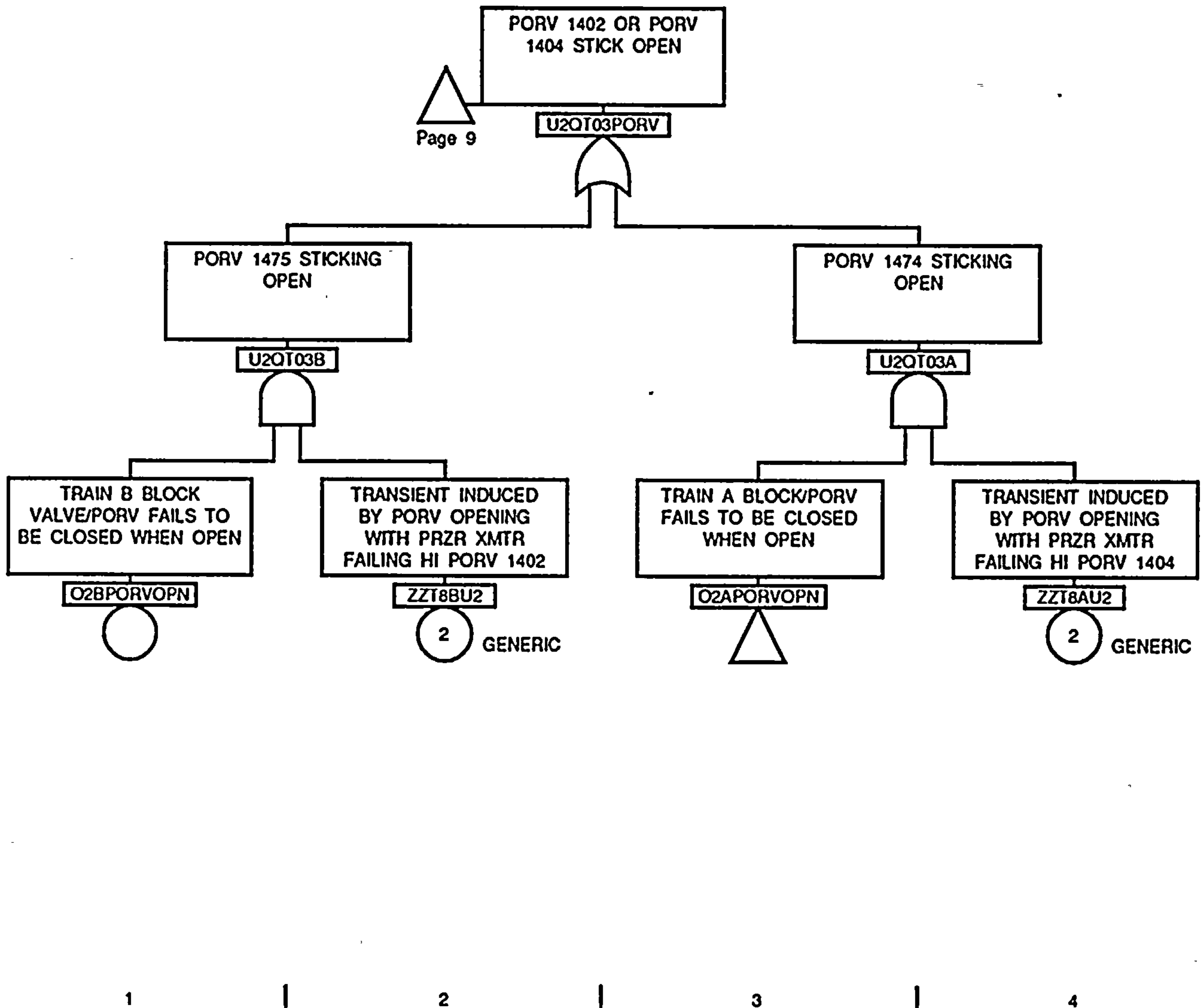
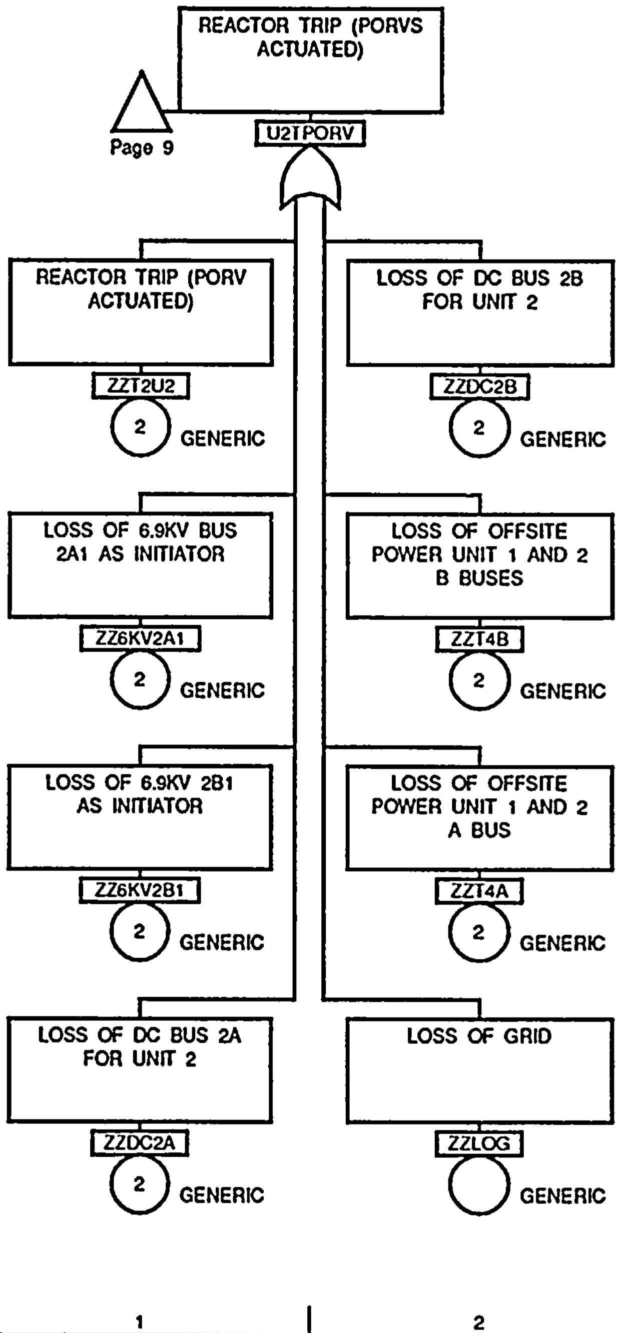

GROUP 4 TRANSIENTS
Page 3
U2GROUP4B
LOSS OF ICW
ZZICWU2
GENERIC
SPURIOUS MAIN STEAM ISOLATION SIGNAL
ZZT7U2
PLANT
LOSS OF CCW
ZZCCWU2
GENERIC
LOSS OF MAIN FEEDWATER DUE TO FEEDLINE BREAK ON SG 2A
ZZT3DU2A
GENERIC
LOSS OF TCW
ZZTCWU2
GENERIC
LOSS OF MAIN FEEDWATER DUE TO FEEDLINE BREAK ON SG 2B
ZZT3DU2B
GENERIC
LOSS OF INSTRUMENT AIR
ZZIAU2
GENERIC
1
2

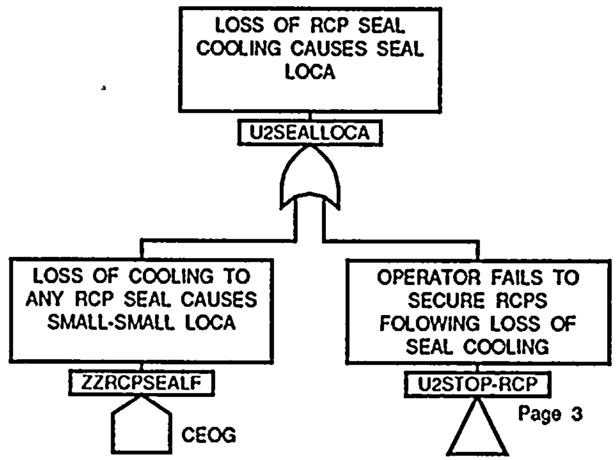

LOCA
Page 3
U2LOCA
SMALL OR SMALL-SMALL LOCA
LARGE LOCA
U2SLOCA
ZZAU2
GENERIC
SMALL-SMALL LOCA
SMALL LOCA
ZZS1U2
GENERIC
ZZS2U2
GENERIC
1
2

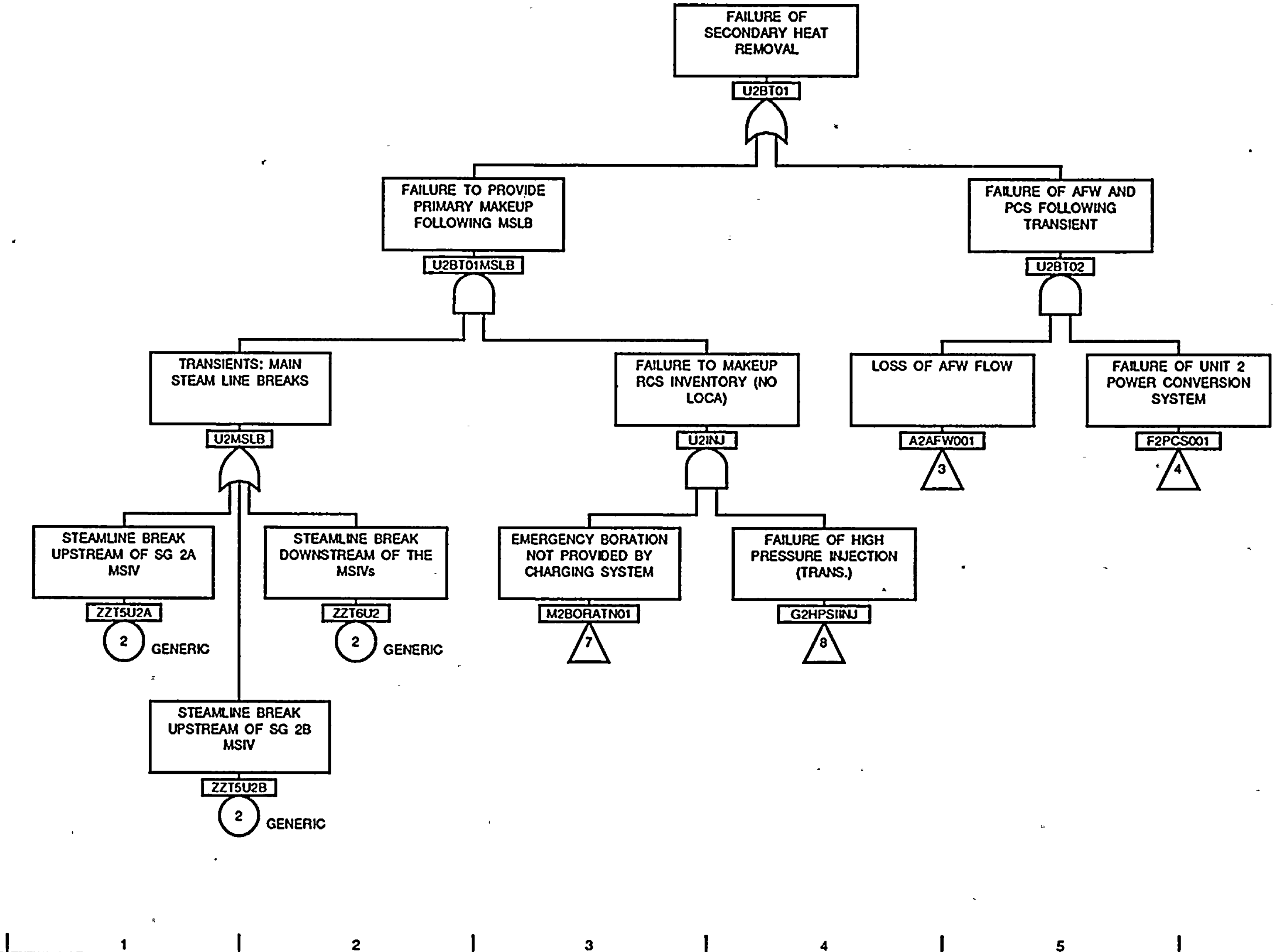

UNIT 2 STEAM GENERATOR TUBE RUPTURE
Page 3
U2SGTR
SG 2A TUBE RUPTURE
SG 2B TUBE RUPTURE
ZZRU2A
GENERIC
ZZRU2B
GENERIC
1
2

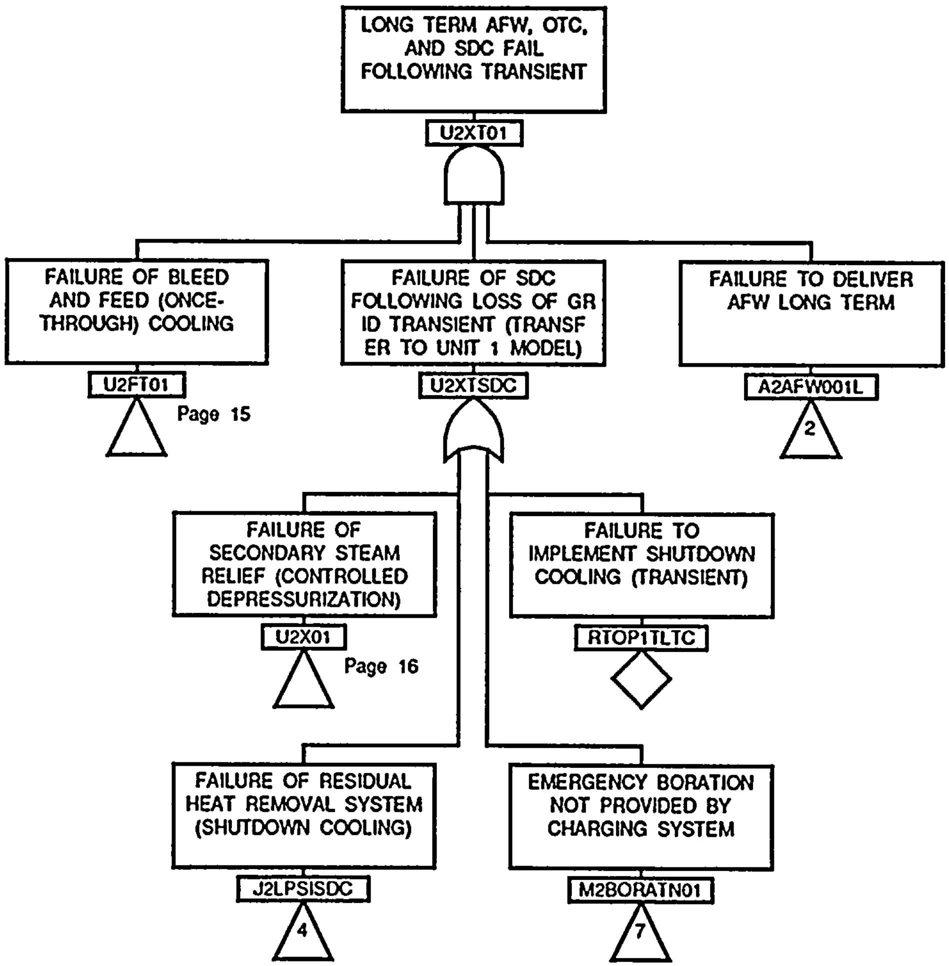

SMALL-SMALL LOCA DUE TO STUCK OPEN PORVS
Page 1
U2QT03
PORV 1402 OR PORV 1404 STICK OPEN
U2QT03PORV
Page 10
PORV FO AFTER RCS PRESSURIZATION
U2QT07
1 OF 2 PORVs FAILS TO RECLOSE (AUTO)
O2PORVLOCA
REACTOR TRIP (PORVS ACTUATED)
U2TPORV
Page 11
1
2

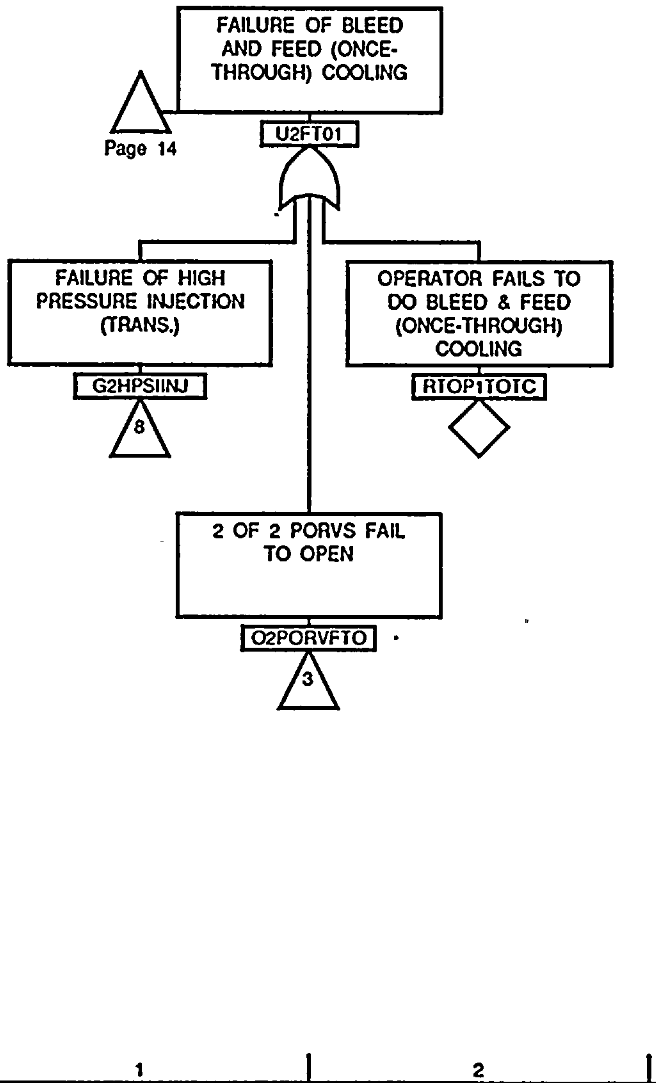

PORV 1402 OR PORV 1404 STICK OPEN
U2QT03PORV
Page 9
PORV 1475 STICKING OPEN
U2QT03B
PORV 1474 STICKING OPEN
U2QT03A
TRAIN B BLOCK VALVE/PORV FAILS TO BE CLOSED WHEN OPEN
O2BPORVOPN
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1402
ZZT8BU2
2
GENERIC
TRAIN A BLOCK/PORV FAILS TO BE CLOSED WHEN OPEN
O2APORVOPN
TRANSIENT INDUCED BY PORV OPENING WITH PRZR XMTR FAILING HI PORV 1404
ZZT8AU2
2
GENERIC
1
2
3
4


REACTOR TRIP (PORVS ACTUATED)
Page 9
U2TPORV
REACTOR TRIP (PORV ACTUATED)
ZZT2U2
2
GENERIC
LOSS OF DC BUS 2B FOR UNIT 2
ZZDC2B
2
GENERIC
LOSS OF 6.9KV BUS 2A1 AS INITIATOR
ZZ6KV2A1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 B BUSES
ZZT4B
2
GENERIC
LOSS OF 6.9KV 2B1 AS INITIATOR
ZZ6KV2B1
2
GENERIC
LOSS OF OFFSITE POWER UNIT 1 AND 2 A BUS
ZZT4A
2
GENERIC
LOSS OF DC BUS 2A FOR UNIT 2
ZZDC2A
2
GENERIC
LOSS OF GRID
ZZLOG
GENERIC
1
2

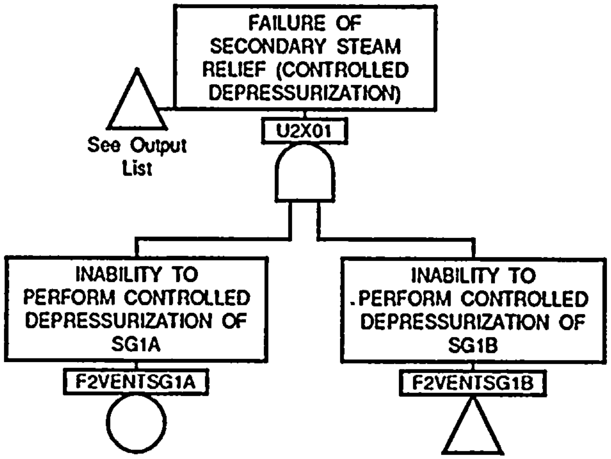

LOSS OF RCP SEAL COOLING CAUSES SEAL LOCA
U2SEALLOCA
LOSS OF COOLING TO ANY RCP SEAL CAUSES SMALL-SMALL LOCA
ZZRCPSEALF
CEOG
OPERATOR FAILS TO SECURE RCPS FOLOWING LOSS OF SEAL COOLING
U2STOP-RCP
Page 3


FAILURE OF SECONDARY HEAT REMOVAL
U2BT01
FAILURE TO PROVIDE PRIMARY MAKEUP FOLLOWING MSLB
U2BT01MSLB
FAILURE OF AFW AND PCS FOLLOWING TRANSIENT
U2BT02
TRANSIENTS: MAIN STEAM LINE BREAKS
U2MSLB
FAILURE TO MAKEUP RCS INVENTORY (NO LOCA)
U2INJ
LOSS OF AFW FLOW
A2AFW001
3
FAILURE OF UNIT 2 POWER CONVERSION SYSTEM
F2PCS001
4
STEAMLINE BREAK UPSTREAM OF SG 2A MSIV
ZZT5U2A
2
GENERIC
STEAMLINE BREAK DOWNSTREAM OF THE MSIVs
ZZT6U2
2
GENERIC
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
STEAMLINE BREAK UPSTREAM OF SG 2B MSIV
ZZT5U2B
2
GENERIC
1
2
3
4
5


LONG TERM AFW, OTC, AND SDC FAIL FOLLOWING TRANSIENT
U2XT01
FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U2FT01
Page 15
FAILURE OF SDC FOLLOWING LOSS OF GRID TRANSIENT (TRANSFER TO UNIT 1 MODEL)
U2XTSDC
FAILURE TO DELIVER AFW LONG TERM
A2AFW001L
2
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U2X01
Page 16
FAILURE TO IMPLEMENT SHUTDOWN COOLING (TRANSIENT)
RTOP1TLTC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J2LPSISDC
4
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7


FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U2FT01
Page 14
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
OPERATOR FAILS TO DO BLEED & FEED (ONCE-THROUGH) COOLING
RTOP1TOTC
2 OF 2 PORVS FAIL TO OPEN
O2PORVFTO
3
1
2

U2X01 Outputs:
Page 14, Page 23, Page 31, Page 33, Page 38



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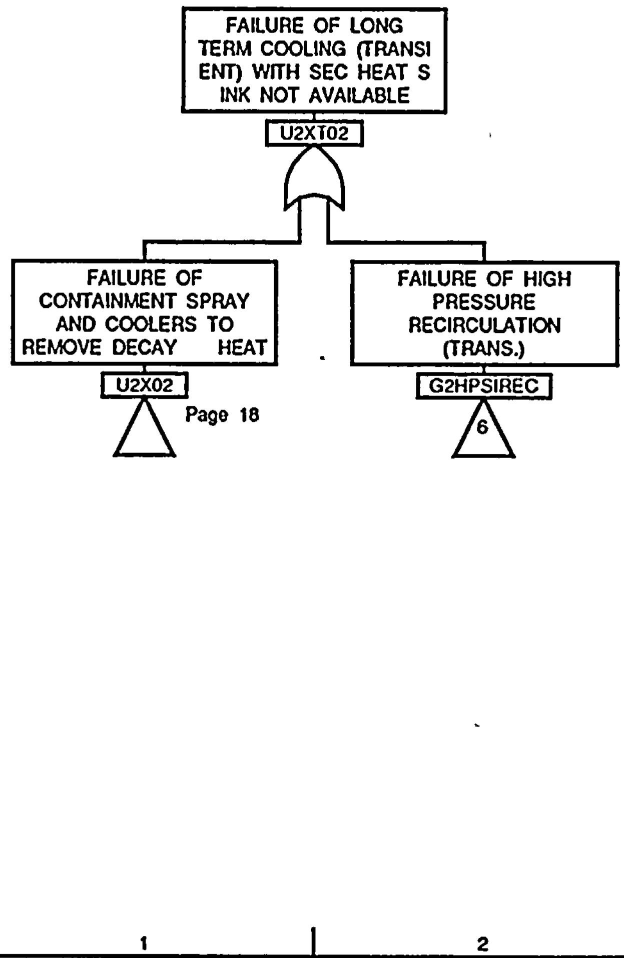
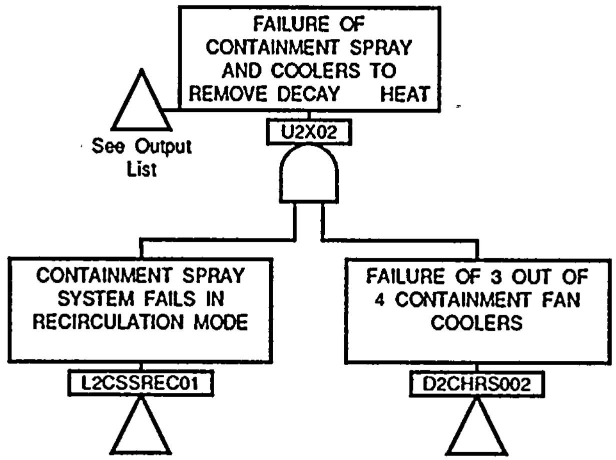
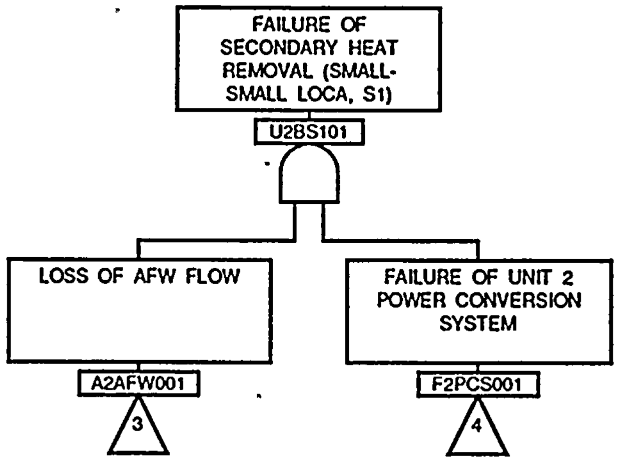
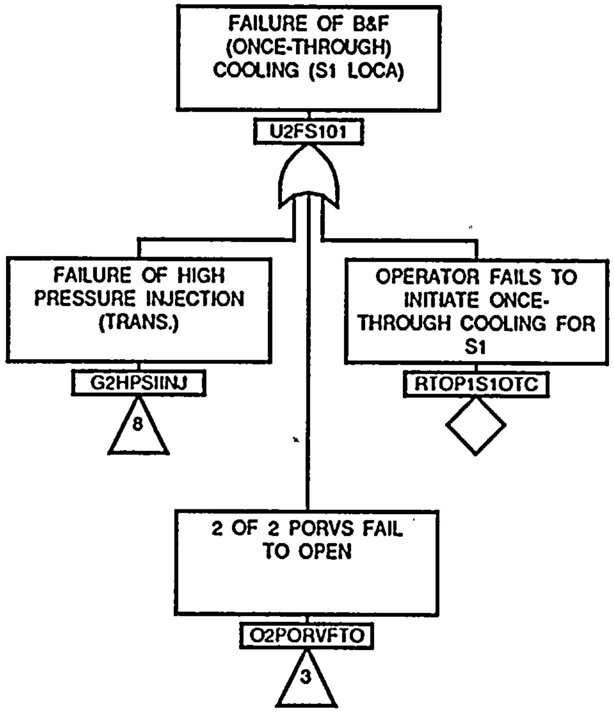

FAILURE OF LONG TERM COOLING (TRANSI ENT) WITH SEC HEAT S INK NOT AVAILABLE
U2XT02
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6
1
2

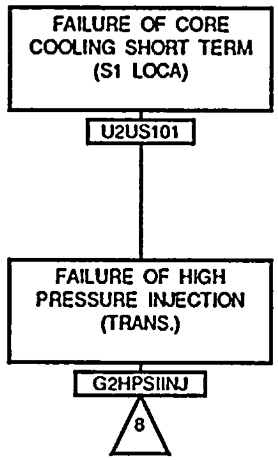
U2X02 Outputs:
Page 17, Page 22, Page 25, Page 25, Page 27, Page 28, Page 37, Page 42



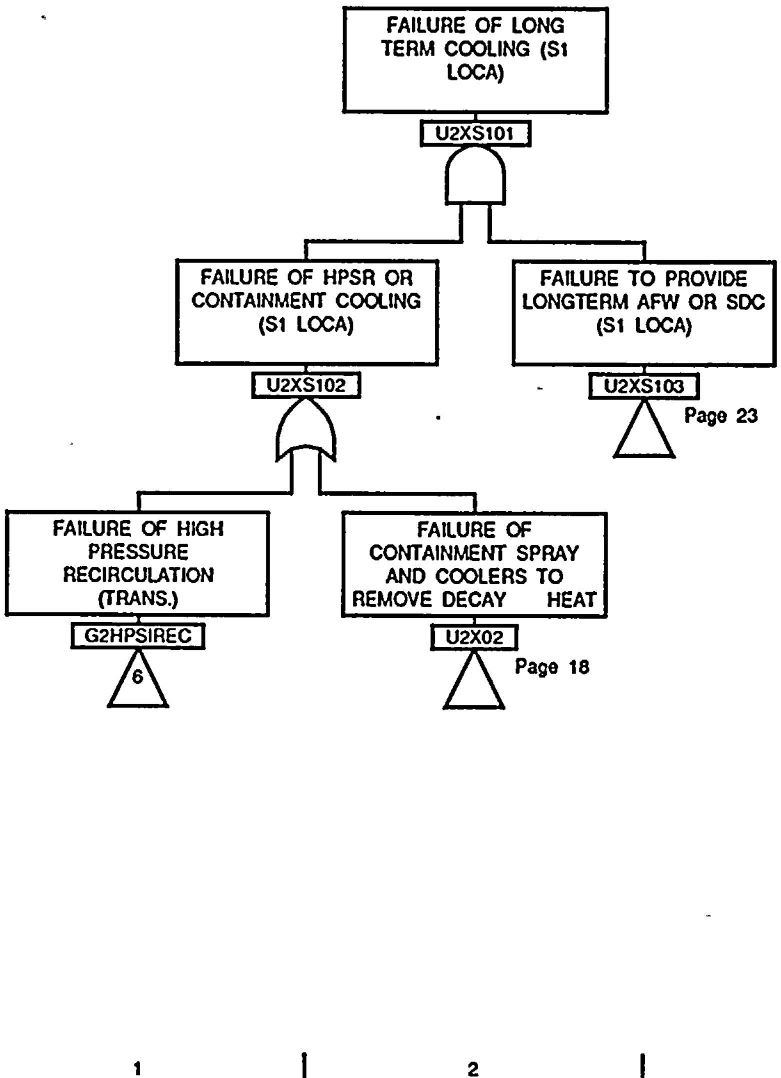

FAILURE OF SECONDARY HEAT REMOVAL (SMALL-SMALL LOCA, S1)
U2BS101
LOSS OF AFW FLOW
A2AFW001
3
FAILURE OF UNIT 2 POWER CONVERSION SYSTEM
F2PCS001
4

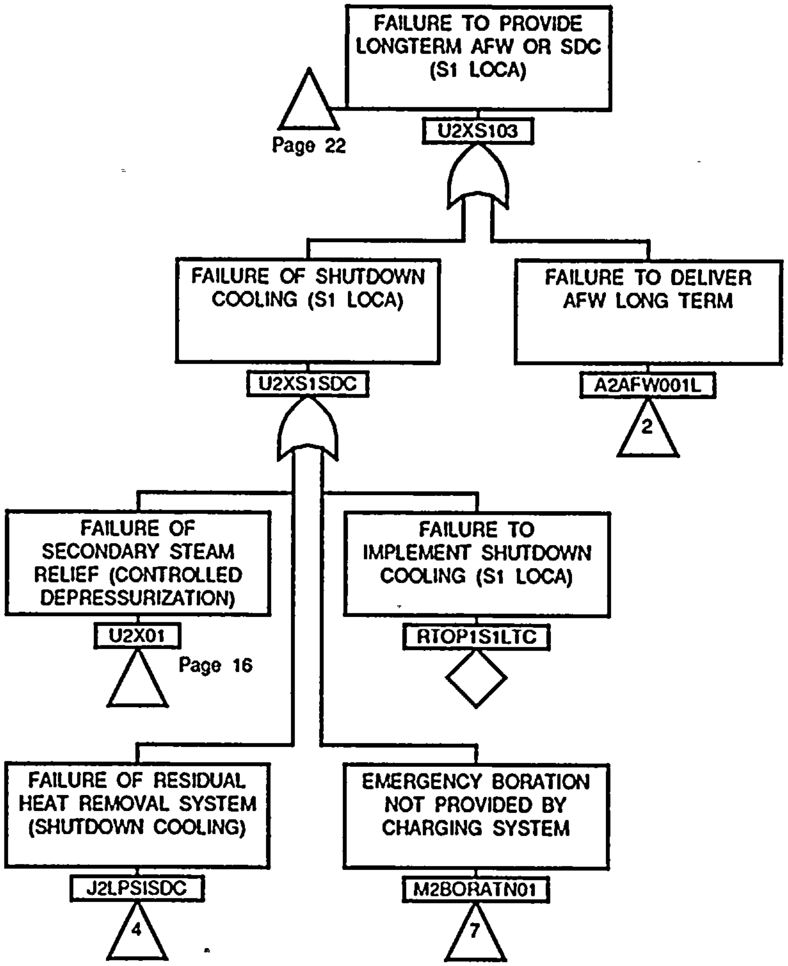
1 2


FAILURE OF B&F (ONCE-THROUGH) COOLING (S1 LOCA)
U2FS101
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR S1
RTOP1S1OTC
2 OF 2 PORVS FAIL TO OPEN
O2PORVFTO
3

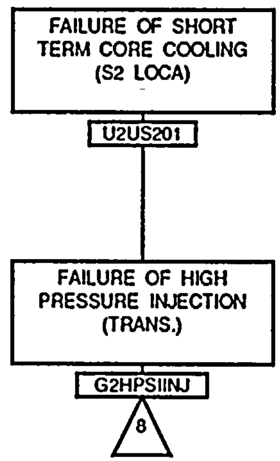

FAILURE OF CORE COOLING SHORT TERM (S1 LOCA)
U2US101
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8

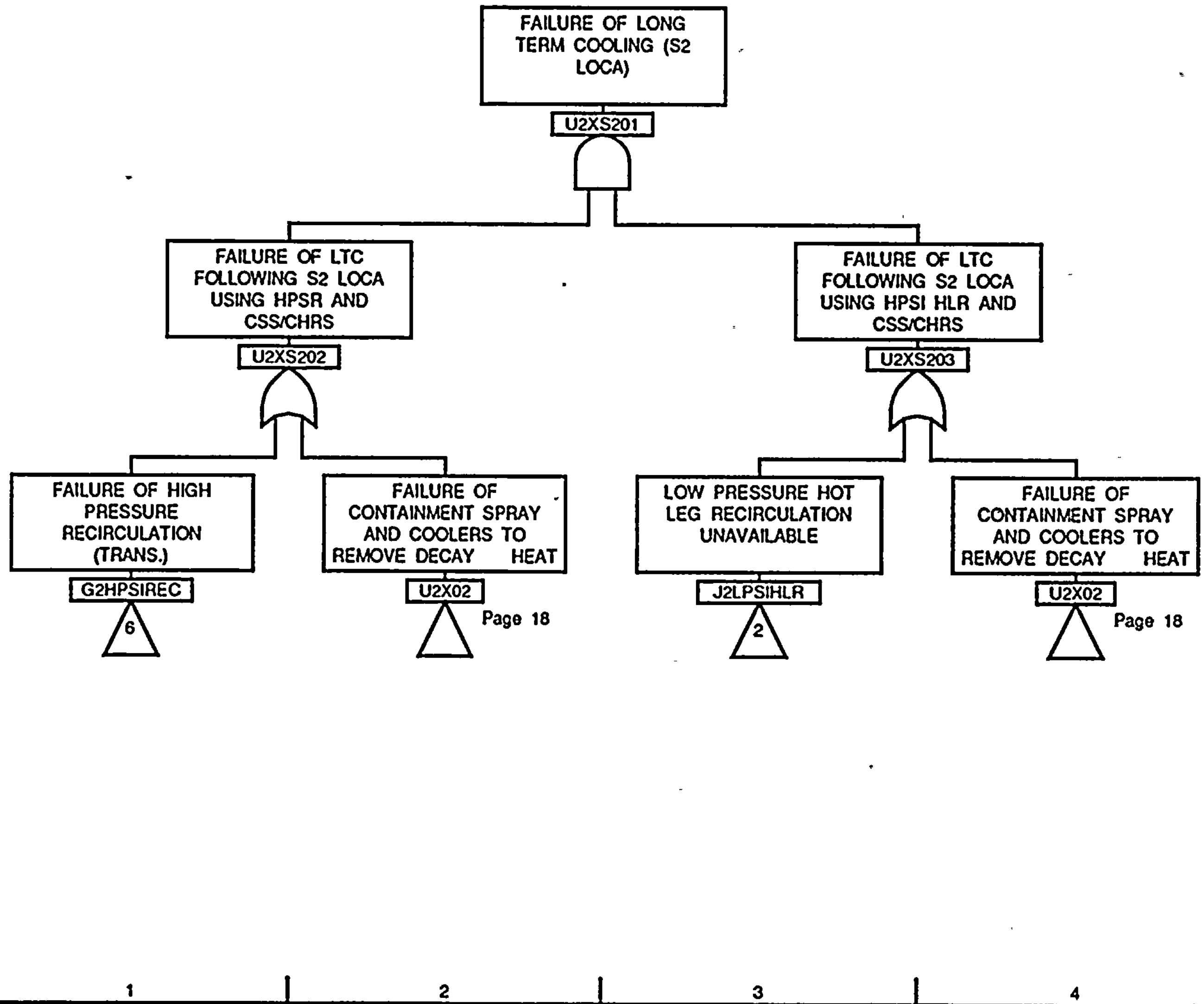
1

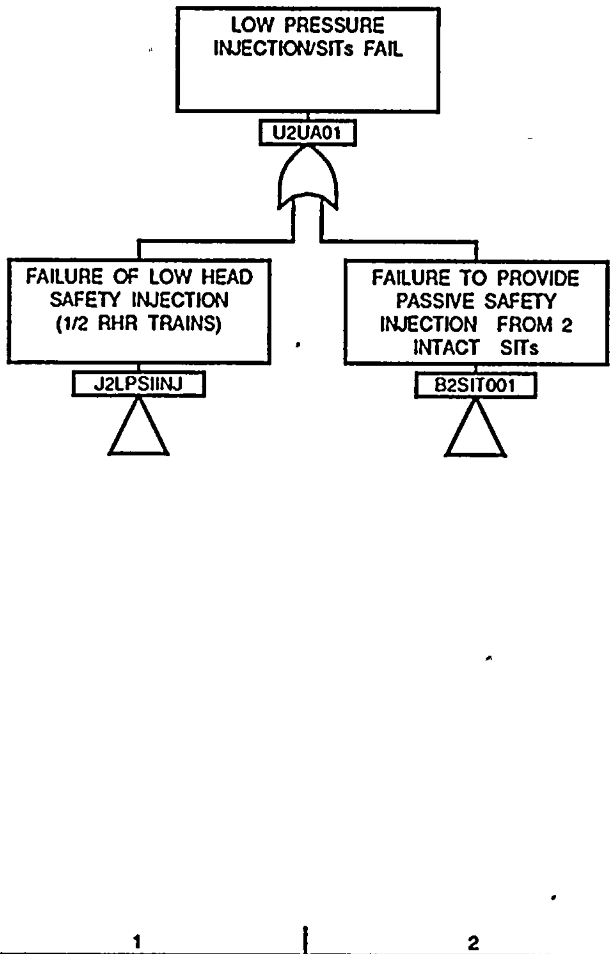

FAILURE OF LONG TERM COOLING (S1 LOCA)
U2XS101
FAILURE OF HPSR OR CONTAINMENT COOLING (S1 LOCA)
U2XS102
FAILURE TO PROVIDE LONGTERM AFW OR SDC (S1 LOCA)
U2XS103
Page 23
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
1
2

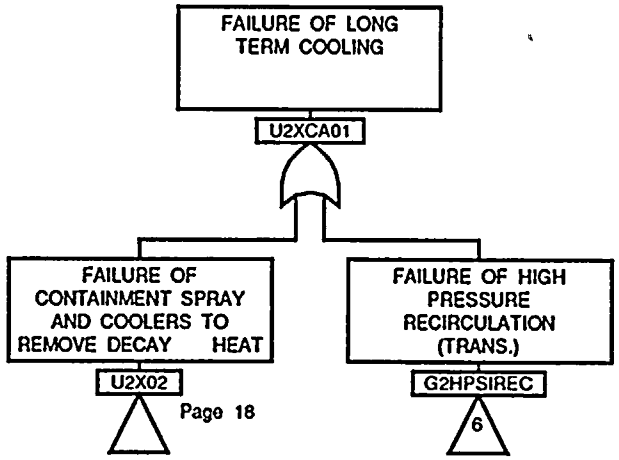

FAILURE TO PROVIDE LONGTERM AFW OR SDC (S1 LOCA)
U2XS103
Page 22
FAILURE OF SHUTDOWN COOLING (S1 LOCA)
U2XS1SDC
FAILURE TO DELIVER AFW LONG TERM
A2AFW001L
2
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U2X01
Page 16
FAILURE TO IMPLEMENT SHUTDOWN COOLING (S1 LOCA)
RTOP1S1LTC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J2LPSISDC
4
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7

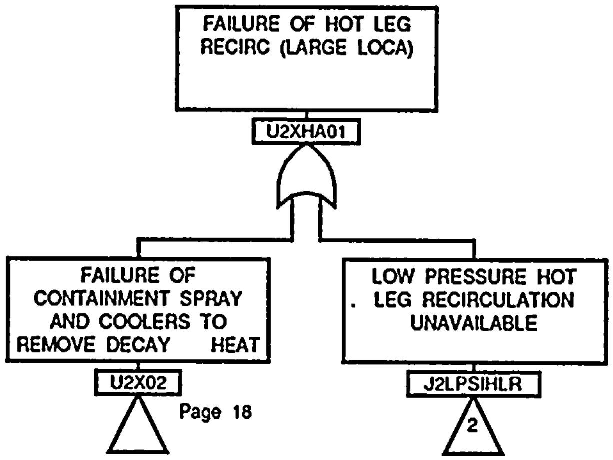
1 2

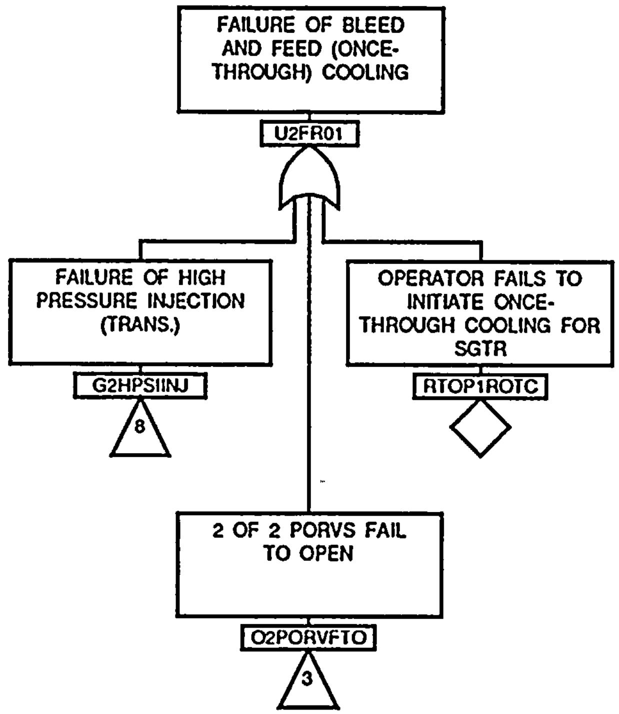

FAILURE OF SHORT TERM CORE COOLING (S2 LOCA)
U2US201
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8

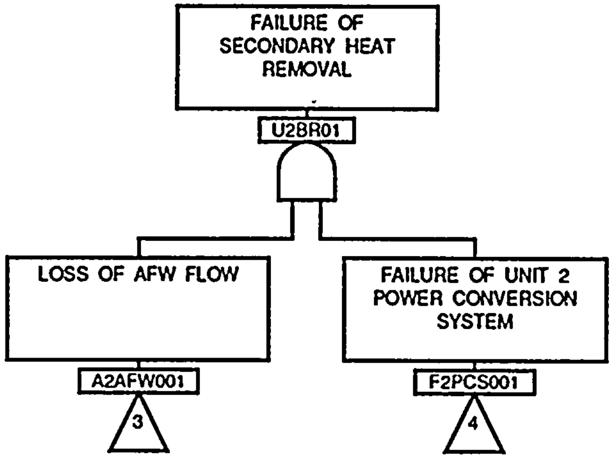

FAILURE OF LONG TERM COOLING (S2 LOCA)
U2XS201
FAILURE OF LTC FOLLOWING S2 LOCA USING HPSR AND CSS/CHRS
U2XS202
FAILURE OF LTC FOLLOWING S2 LOCA USING HPSI HLR AND CSS/CHRS
U2XS203
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
LOW PRESSURE HOT LEG RECIRCULATION UNAVAILABLE
J2LPSIHLR
2
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
1
2
3
4


LOW PRESSURE INJECTION/SITs FAIL
U2UA01
FAILURE OF LOW HEAD SAFETY INJECTION (1/2 RHR TRAINS)
J2LPSIINJ
FAILURE TO PROVIDE PASSIVE SAFETY INJECTION FROM 2 INTACT SITs
B2SIT001
1
2

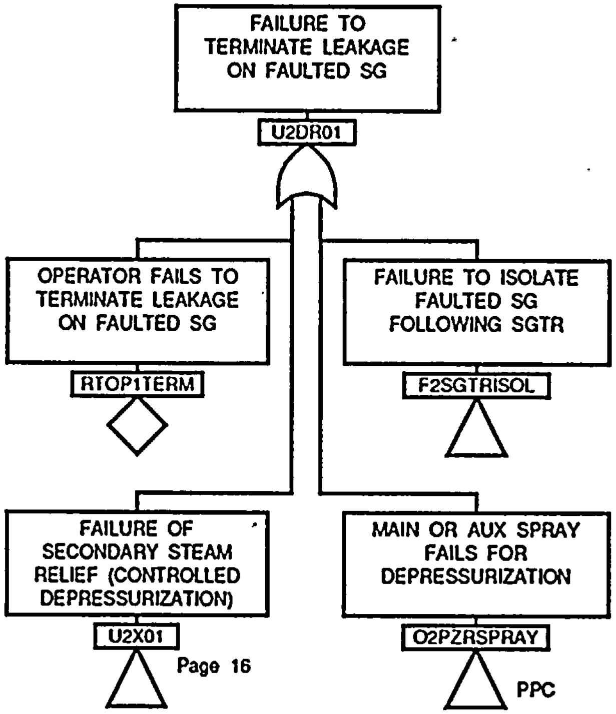

FAILURE OF LONG TERM COOLING
U2XCA01
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6
1
2


FAILURE OF HOT LEG RECIRC (LARGE LOCA)
U2XHA01
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
LOW PRESSURE HOT LEG RECIRCULATION UNAVAILABLE
J2LPSIHLR
2

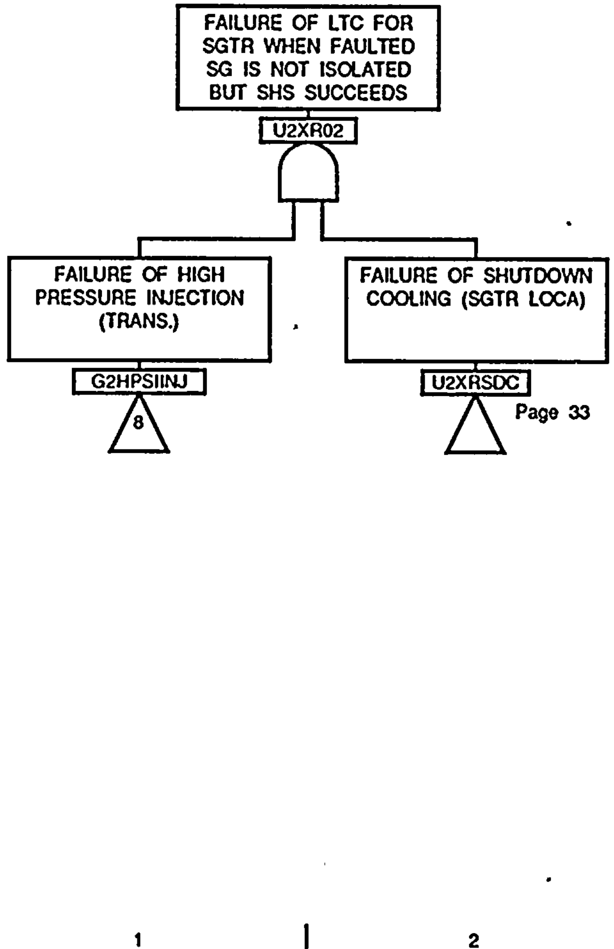

FAILURE OF BLEED AND FEED (ONCE-THROUGH) COOLING
U2FR01
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR
RTOP1ROTC
2 OF 2 PORVS FAIL TO OPEN
O2PORVFTO
3

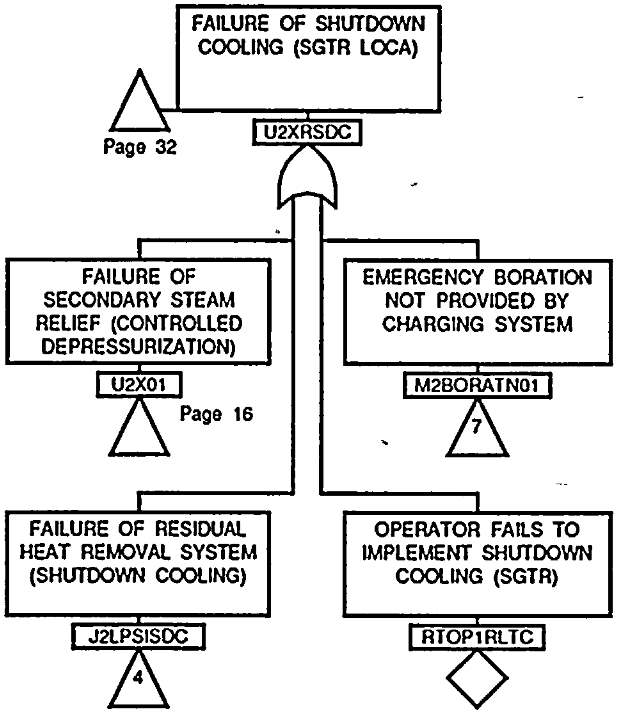

FAILURE OF SECONDARY HEAT REMOVAL
U2BR01
LOSS OF AFW FLOW
A2AFW001
3
FAILURE OF UNIT 2 POWER CONVERSION SYSTEM
F2PCS001
4

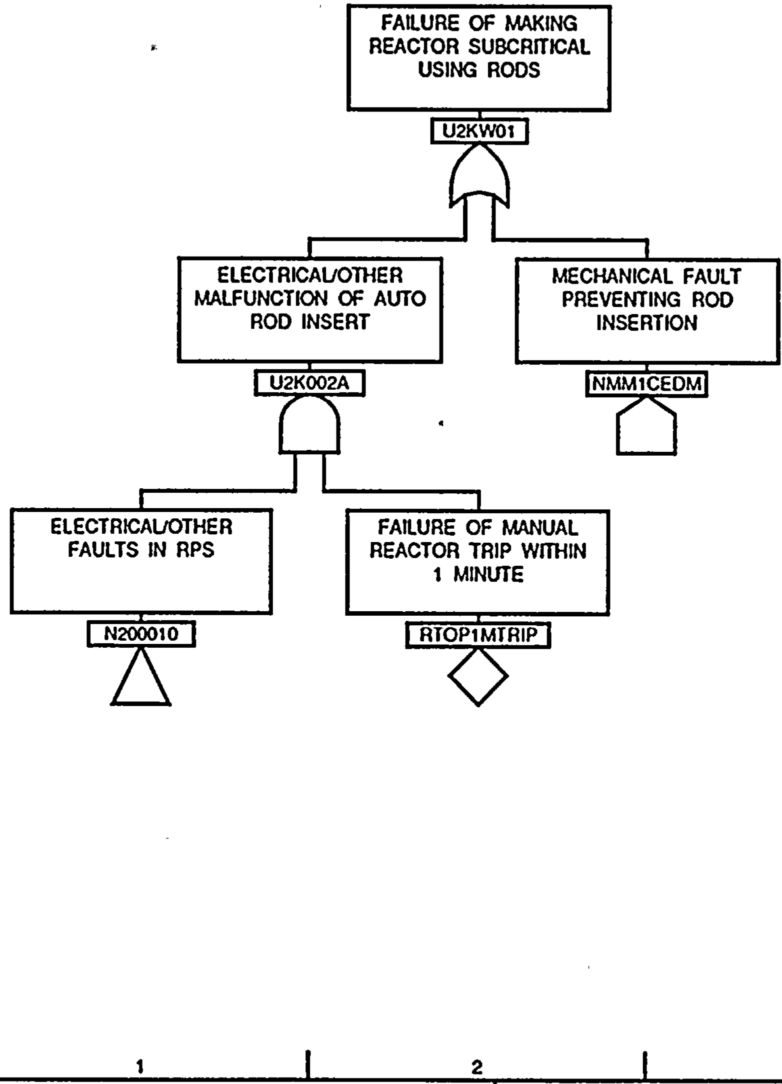
1
2

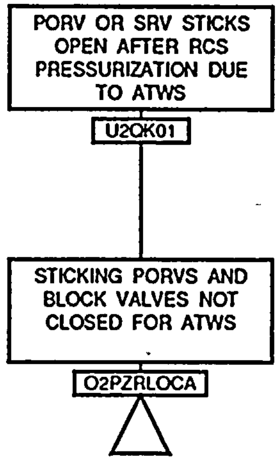

FAILURE TO TERMINATE LEAKAGE ON FAULTED SG
U2DR01
OPERATOR FAILS TO TERMINATE LEAKAGE ON FAULTED SG
RTOP1TERM
FAILURE TO ISOLATE FAULTED SG FOLLOWING SGTR
F2SGTRISOL
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U2X01
Page 16
MAIN OR AUX SPRAY FAILS FOR DEPRESSURIZATION
O2PZRSPRAY
PPC

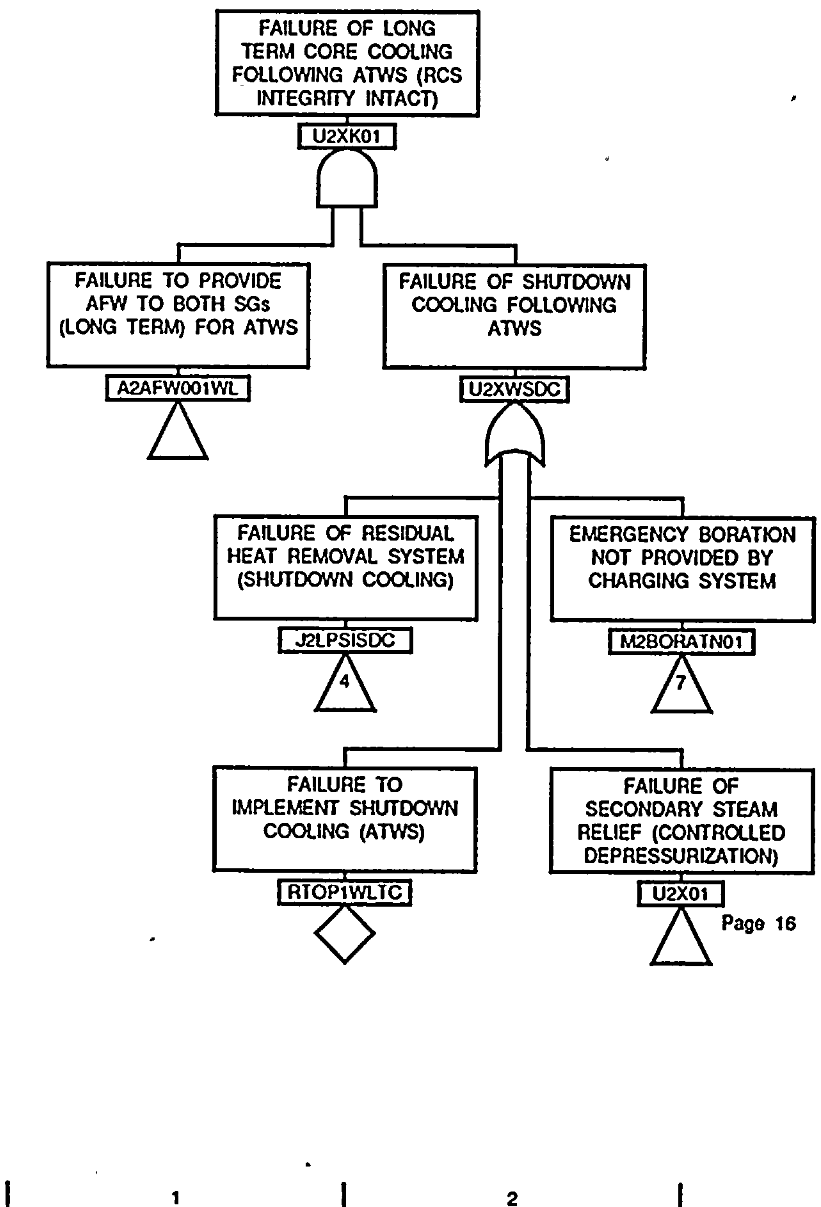
1 2

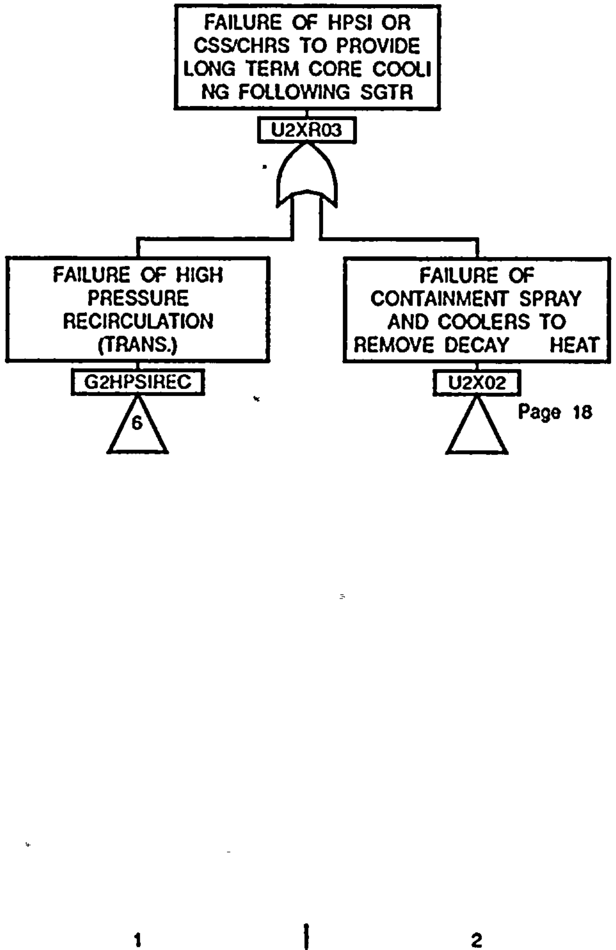

FAILURE OF LTC FOR SGTR WHEN FAULTED SG IS NOT ISOLATED BUT SHS SUCCEEDS
U2XR02
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
FAILURE OF SHUTDOWN COOLING (SGTR LOCA)
U2XRSDC
Page 33
1
2

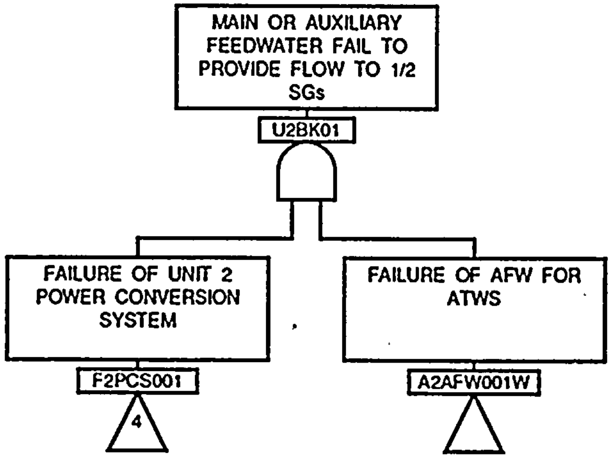

FAILURE OF SHUTDOWN COOLING (SGTR LOCA)
U2XRSDC
Page 32
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U2X01
Page 16
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J2LPSISDC
4
OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)
RTOP1RLTC

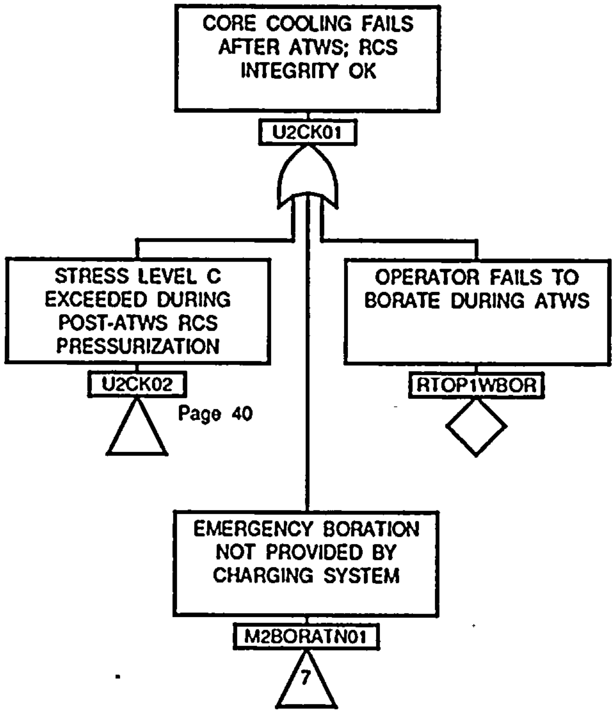

FAILURE OF MAKING REACTOR SUBCRITICAL USING RODS
U2KW01
ELECTRICAL/OTHER MALFUNCTION OF AUTO ROD INSERT
U2K002A
MECHANICAL FAULT PREVENTING ROD INSERTION
NMM1CEDM
ELECTRICAL/OTHER FAULTS IN RPS
N200010
FAILURE OF MANUAL REACTOR TRIP WITHIN 1 MINUTE
RTOP1MTRIP
1
2

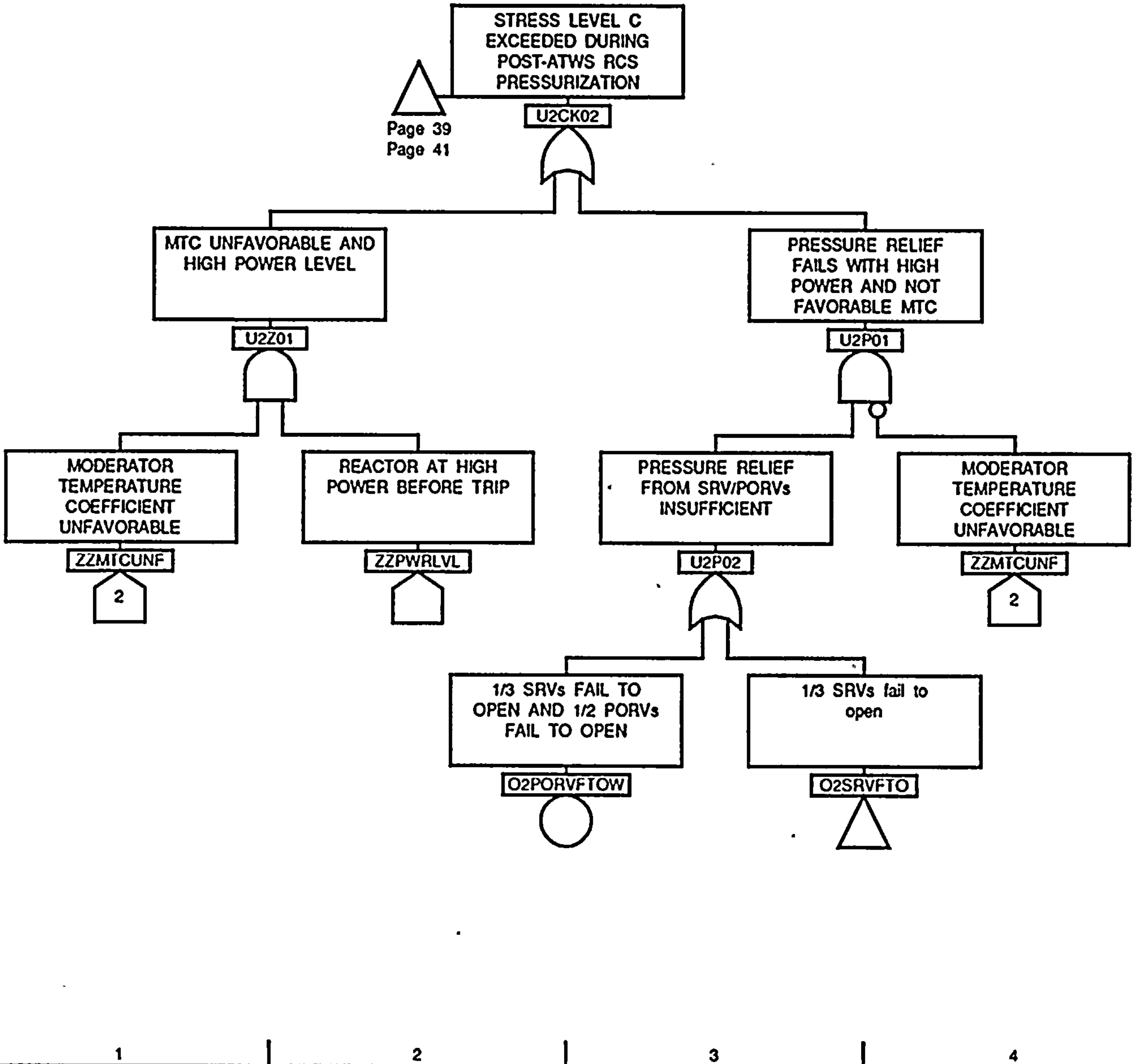

PORV OR SRV STICKS OPEN AFTER RCS PRESSURIZATION DUE TO ATWS
U2QK01
STICKING PORVS AND BLOCK VALVES NOT CLOSED FOR ATWS
O2PZRLOCA

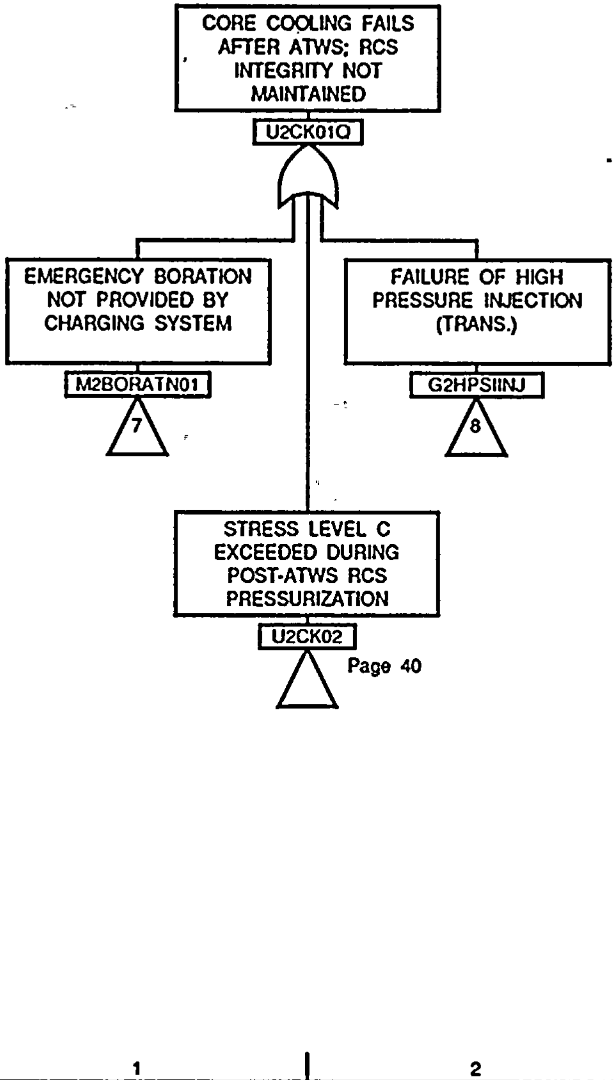

FAILURE OF LONG TERM CORE COOLING FOLLOWING ATWS (RCS INTEGRITY INTACT)
U2XK01
FAILURE TO PROVIDE AFW TO BOTH SGs (LONG TERM) FOR ATWS
A2AFW001WL
FAILURE OF SHUTDOWN COOLING FOLLOWING ATWS
U2XWSDC
FAILURE OF RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING)
J2LPSISDC
4
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7
FAILURE TO IMPLEMENT SHUTDOWN COOLING (ATWS)
RTOP1WLTC
FAILURE OF SECONDARY STEAM RELIEF (CONTROLLED DEPRESSURIZATION)
U2X01
Page 16
1
2

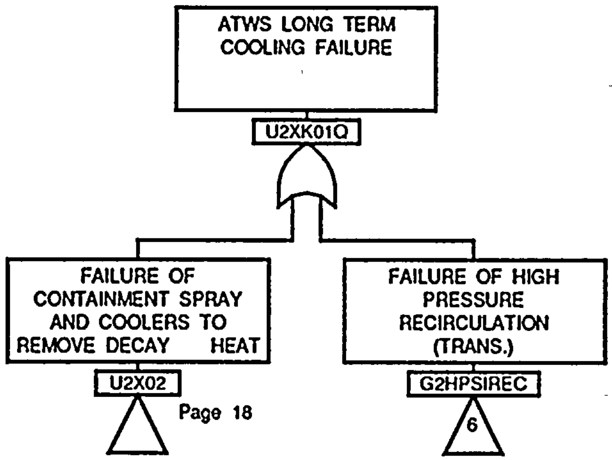

FAILURE OF HPSI OR CSS/CHRS TO PROVIDE LONG TERM CORE COOLI NG FOLLOWING SGTR
U2XR03
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
1
2


MAIN OR AUXILIARY FEEDWATER FAIL TO PROVIDE FLOW TO 1/2 SGs
U2BK01
FAILURE OF UNIT 2 POWER CONVERSION SYSTEM
F2PCS001
4
FAILURE OF AFW FOR ATWS
A2AFW001W


CORE COOLING FAILS AFTER ATWS; RCS INTEGRITY OK
U2CK01
STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
U2CK02
Page 40
OPERATOR FAILS TO BORATE DURING ATWS
RTOP1WBOR
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7


STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
Page 39
Page 41
U2CK02
MTC UNFAVORABLE AND HIGH POWER LEVEL
U2Z01
PRESSURE RELIEF FAILS WITH HIGH POWER AND NOT FAVORABLE MTC
U2P01
MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
ZZMTCUNF
2
REACTOR AT HIGH POWER BEFORE TRIP
ZZPWRLVL
PRESSURE RELIEF FROM SRV/PORVs INSUFFICIENT
U2P02
MODERATOR TEMPERATURE COEFFICIENT UNFAVORABLE
ZZMTCUNF
2
1/3 SRVs FAIL TO OPEN AND 1/2 PORVs FAIL TO OPEN
O2PORVFTOW
1/3 SRVs fail to open
O2SRVFTO
1
2
3
4


CORE COOLING FAILS AFTER ATWS; RCS INTEGRITY NOT MAINTAINED
U2CK01Q
EMERGENCY BORATION NOT PROVIDED BY CHARGING SYSTEM
M2BORATN01
7
FAILURE OF HIGH PRESSURE INJECTION (TRANS.)
G2HPSIINJ
8
STRESS LEVEL C EXCEEDED DURING POST-ATWS RCS PRESSURIZATION
U2CK02
Page 40
1
2


ATWS LONG TERM COOLING FAILURE
U2XK01Q
FAILURE OF CONTAINMENT SPRAY AND COOLERS TO REMOVE DECAY HEAT
U2X02
Page 18
FAILURE OF HIGH PRESSURE RECIRCULATION (TRANS.)
G2HPSIREC
6

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
A2AFW001	13	5	J2LPSISDC	23	1	U2BT01	13	4	U2QT03PORV	10	2
A2AFW001	19	1	J2LPSISDC	33	1	U2BT01MSLB	13	3	U2QT07	9	2
A2AFW001	30	1	J2LPSISDC	36	2	U2BT02	13	5	U2RCP2A1	1	4
A2AFW001L	14	3	L2CSSREC01	18	1	U2CK01	39	2	U2RCP2A1	2	1
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A2AFW001W	38	2	M2BORATN01	14	3	U2CK02	39	1	U2RCP2B1	1	2
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C2NRCP2B1	1	2	M2BORATN01	41	1	U2FS101	20	2	U2SLOCA	7	2
C2NRCP2B2	1	1	N200010	34	1	U2FT01	14	1	U2STOP-RCP	1	4
D2CHRS002	18	2	NMM1CEDM	34	3	U2FT01	15	2	U2STOP-RCP	3	2
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F2VENTSG1B	16	2	O2PORVLOCA	9	2	U2GROUP4B	6	2	U2X01	14	2
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G2HPSIREC	25	1	RTOP1TOTC	15	2	U2QT01	1	4	U2X02	27	1
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J2LPSISDC	14	2	U2BS101	19	2	U2QT03PORV	9	1	U2XK01Q	42	2

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
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U2XS102	22	2	ZZS2U2	7	2						
U2XS103	22	3	ZZT1U2	3	2						
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U2XS1SDC	23	2	ZZT2U2	11	1						
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ZZAU2	7	3	ZZT6U2	13	2						
ZZCCWU2	6	1	ZZT7U2	6	2						
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ZZDC2A	11	1	ZZT8AU2	10	4						
ZZDC2B	4	1	ZZT8BU2	4	2						
ZZDC2B	11	2	ZZT8BU2	10	2						
ZZIAU2	6	2	ZZTCWU2	6	1						
ZZICWU2	6	1									
ZZLOG	11	2									
ZZMAU2	5	7									
ZZMBU2	5	7									
ZZMCU2	5	8									
ZZMDU2	5	8									
ZZMTCUNF	40	1									
ZZMTCUNF	40	4									

APPENDIX B

ST. LUCIE UNITS 1 & 2

SYSTEM DESCRIPTIONS

TABLE OF CONTENTS

NOTE: In the following system descriptions, Unit 2 specific features are generally contained between brackets as [ ].

B1 SAFETY INJECTION TANKS

B1.1 Function

The primary function of the Safety Injection Tank system is to rapidly reflood and cool the core during the time period between the occurrence of a large LOCA and the time at which flow from the low pressure safety injection pumps can actually reach the core. In the case of a LOCA with concurrent loss of offsite power, this delay may be as long as 30 seconds.

The tank gas/water fractions, gas pressure, and outlet pipe size are selected to allow the tanks to re-cover the core before significant clad melting or zirconium-water reaction can occur.

The Safety Injection Tank system components are designed to withstand design basis earthquake loads without loss of function. They are also designed to withstand post-accident environmental conditions without loss of function, and to permit inspection/testing at appropriate intervals to assure system availability and functional capability.

B1.2 Configuration

Figures B1.1 and B1.2 depict the SIT system configuration for Units 1 and 2, respectively. The modeled SIT system consists of the following components for each unit:

A. Four Safety Injection Tanks (1A1, 1A2, 1B1, 1B2 [2A1, 2A2, 2B1, 2B2])

B. One discharge motor-operated isolation valve (V-3614/24/34/44)

C. One SIT discharge check valve (V-3215/25/35/45)

D. One RCS isolation check valve (V-3217/27/37/47)

B1.3 Success Criteria

The Safety Injection Tanks discharge their contents into the RCS during large break LOCA events when the RCS pressure falls below the tank pressure. Depending on the break location, the contents of a single tank may be lost through the break with the inventory falling into the containment sump. The function of this system is considered successful when the contents of at least three tanks are discharged into the core following large break LOCA events.

B1.4 Operation

B1.4.1 Normal Operation

During normal power operation, the safety injection tanks are in standby mode filled and pressurized, with the discharge MOVs in the open position with their breakers maintained open, and the loop check valves closed due to RCS pressure.

During startup operation, the motor-operated isolation valves on the SIT discharge piping are interlocked with pressurizer pressure to open the valves automatically as system pressure is increased, and to prevent inadvertent closure prior to or during an accident. When the RCS pressure increases as indicated by the pressurizer pressure, the operator repressurizes the SITs. From this point on, the SITs are on standby. Although the SIT isolation valves are normally open with power removed (breakers off) during power operations, they receive a confirmatory signal to open on SIAS to ensure SIT injection during LOCAs..

Prior to initiation of shutdown cooling operations, the SITs are vented through the vent valves and SIT isolation valves are positioned in the closed position to prevent inadvertent dumping of contents into the core.

B1.4.2 Emergency Operation

As discussed earlier, the SITs are operated only during emergency post-LOCA conditions. No operator action or actuation signal is required for operation. The contents of the SITs are discharged into the cold legs of the RCS when the RCS pressure falls below the tank pressure. Adequate borated water is supplied to rapidly cool the core, with the contents of one tank assumed in the safety analysis to be discharging through the break.

Figure B1.1
Safety Injection Tanks Simplified Schematic
St. Lucie Unit 1

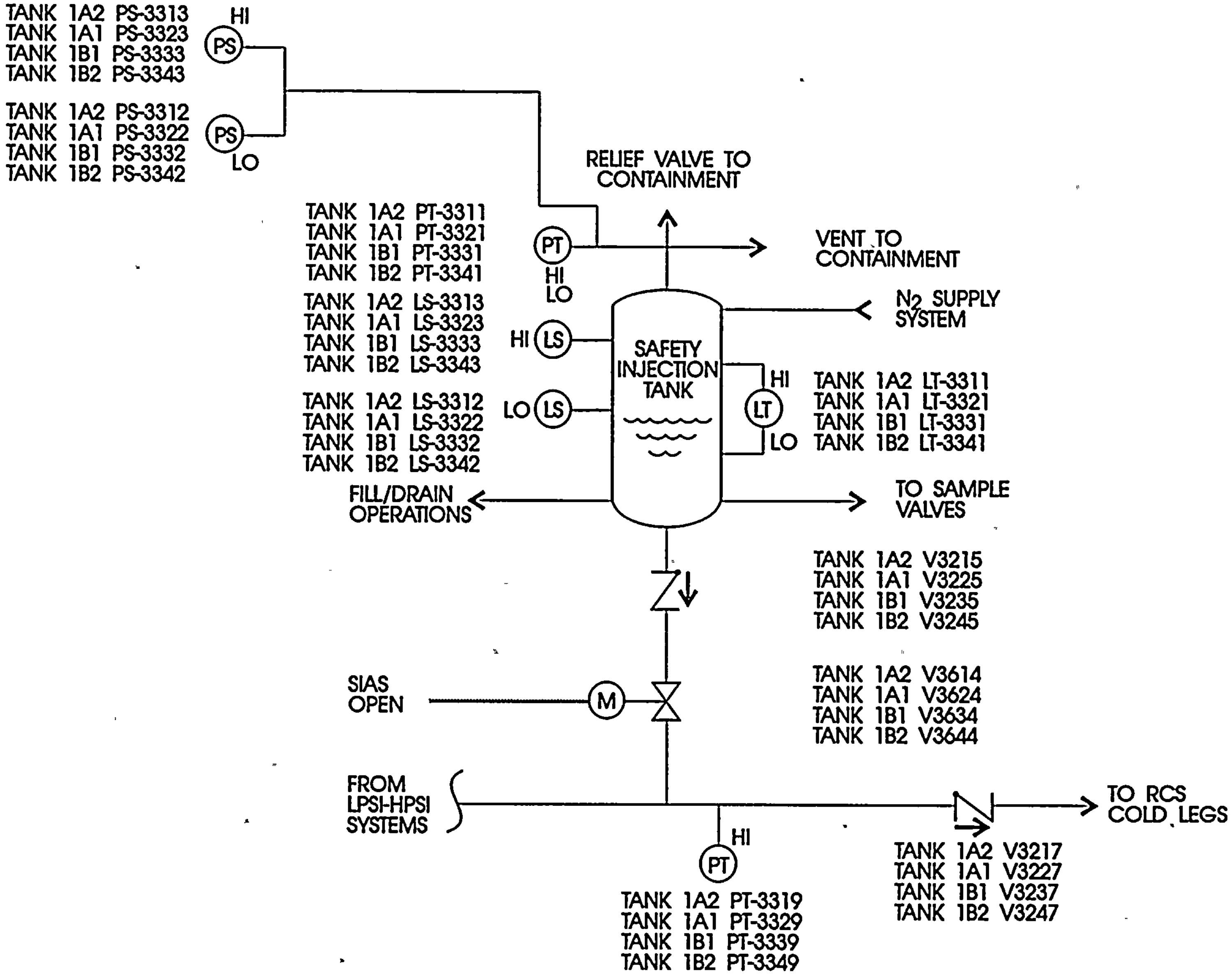
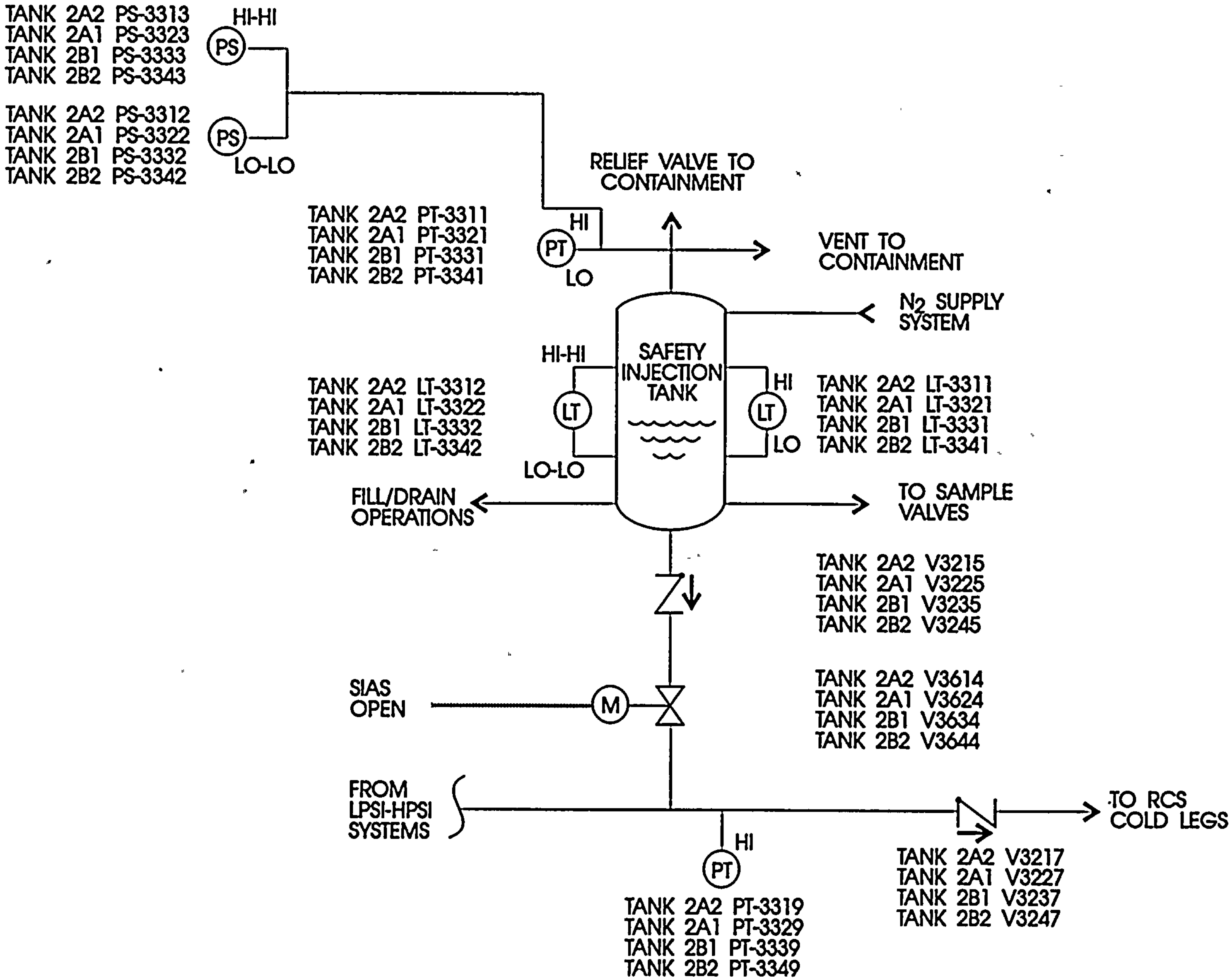


Figure B1.2
Safety Injection Tanks Simplified Schematic
St. Lucie Unit 2



B2 AUXILIARY FEEDWATER SYSTEM

B2.1 Function

The primary function of the Auxiliary Feedwater (AFW) system is to ensure a sufficient supply of cooling water to at least one of the steam generators when main feedwater (MFW) is not available. This ensures removal of sensible and decay heat from the reactor coolant system during normal or off-normal cooldown operation. The AFW system should also provide sufficient feedwater capacity to permit plant cooldown to 325°F (shutdown cooling conditions).

B2.2 Configuration

Figures B2.1 and B2.2 depict the AFW system standby configuration for Units 1 and 2, respectively.

The AFW system includes the electric and steam-driven pumps, valves, condensate storage tanks, AFW piping and associated instrumentation/controls. Pumps, valves and instrumentation include their associated electric power supplies (with circuit breakers).

The Auxiliary Feedwater system at each St. Lucie unit consists of: two full flow capacity motor-driven pumps at 325 [300] gpm per pump, one steam-driven pump at 600 [570] gpm, one Condensate Storage Tank (CST) with a minimum Tech. Spec. required volume of 116,000 gallons [307,000 gal. - this includes 149,600 gallons for Unit 2 for hot standby and cooldown operations, and 125,000 gallons in reserve for Unit 1], and an arrangement of manual, check, [solenoid,] and motor-operated valves.

Each motor-driven pump supplies water to the associated steam generator upon receipt of an auto start signal from the Auxiliary Feedwater Actuation System (AFAS). A cross connection path is provided to enable the routing of the flow of both motor-driven pumps to a single steam generator. These pumps are powered from the 4160 VAC 1A3[2A3] and 1B3[2B3] vital busses. These busses receive power from offsite and from the emergency generators in case of a loss of offsite power.

The steam-driven pump is driven by a noncondensing steam turbine. The turbine receives steam from the main steam lines upstream of the main steam isolation valves and exhausts to atmosphere. This pump is capable of supplying auxiliary feedwater flow to both steam generators over the total expected range of steam generator pressures. The pump on Unit 1 has an electro-hydraulic governor system which gives the operator variable speed control over the design range. The electro-hydraulic control system is powered from the emergency DC power system. [The Unit 2 pump has a mechanical-hydraulic governor system to maintain turbine speed within the design range. The governor setpoint, however, cannot be varied by the control room operator. Turbine speed is adjusted to the desired setpoint by a local speed control. The governor valve is normally set to accelerate the turbine and maintain constant speed over varying load conditions.]

The AFW pumps (electrical and steam-driven) take suction from the CST, and discharge to the steam generators. The CST provides sufficient quantity of water to allow for decay heat removal and cooldown of the Nuclear Steam Supply System (NSSS) to 325°F following a reactor trip. The total storage capacity of this tank is 271,200 [400,000] gallons. The normal Tech Spec level is 116,000 [307,000] gallons. The higher capacity in the Unit 2 CST is an additional reserve for the unlikely event that a tornado missile ruptures the Unit 1 CST and the water contained therein is unavailable to Unit 1. [The Unit 2 CST is surrounded by a structural barrier which provides missile protection for the tank. Should a missile disable the Unit 1 CST, the Unit 1 operators will be alerted of the loss of auxiliary feedwater by level alarms and indicators in the control room. Once alerted, the Unit 1 operators will initiate procedures to obtain auxiliary feedwater via the Unit 1/Unit 2 cross-tie.]

B2.3 Success Criteria

The AFW system is automatically actuated following a low steam generator level condition which can result from main steam line break (MSLB), loss of feedwater (LOF), or loss of offsite power (LOOP) events. The AFW system can also be manually actuated from the control room.

For all of these events, success of the AFW system in the PRA models is defined as the ability to inject cooling inventory to at least one steam generator from any available AFW flow path from the condensate storage tank. For an ATWS in particular, success requires delivery of AFW flow to both steam generators.

B2.4 Operation

B2.4.1 Normal Operation

During normal operation of the plant, the AFW system is not in operation and feedwater is supplied to the steam generators by the main feedwater system. However, the AFW system is on standby ready to inject CST inventory upon receipt of an AFAS.

During plant startup, the AFW pumps provide the steam generators with water until the main feedwater pumps can be put into operation. Once MFW is in operation, the AFW pumps are secured and the control switches placed in the AUTO position. During plant cooldown the main feedwater pumps are secured, and the AFW system provides the means of removing decay heat to bring the RCS temperature to 325°F (the Shutdown Cooling system entry temperature).

B2.4.2 Accident Operation

If the main feedwater system is unavailable due to loss of feedwater or offsite power, the steam generator feedwater levels will decrease. The AFAS system is provided with sensor and control instrumentation to enable the system to automatically respond to a loss of steam generator inventory. Should the steam generator level decrease to the low steam generator level trip setpoint, an alarm is sounded in the control room and the AFAS time delay is actuated. If the AFAS time delay expires while the steam generator level is below the AFAS low level setpoint, an AFAS will be generated. Once an AFAS is generated, the AFW pumps and associated valves are automatically actuated and cooling inventory is delivered to the steam generators. Once the steam generator levels are restored, the AFW regulating valves automatically reclose. Also, if a steam generator is faulty due to leakage, it is isolated per emergency procedures. Cooling is then provided to the plant through the unaffected generator.

Figure B2.1 AFW System Simplified Diagram (Unit 1)

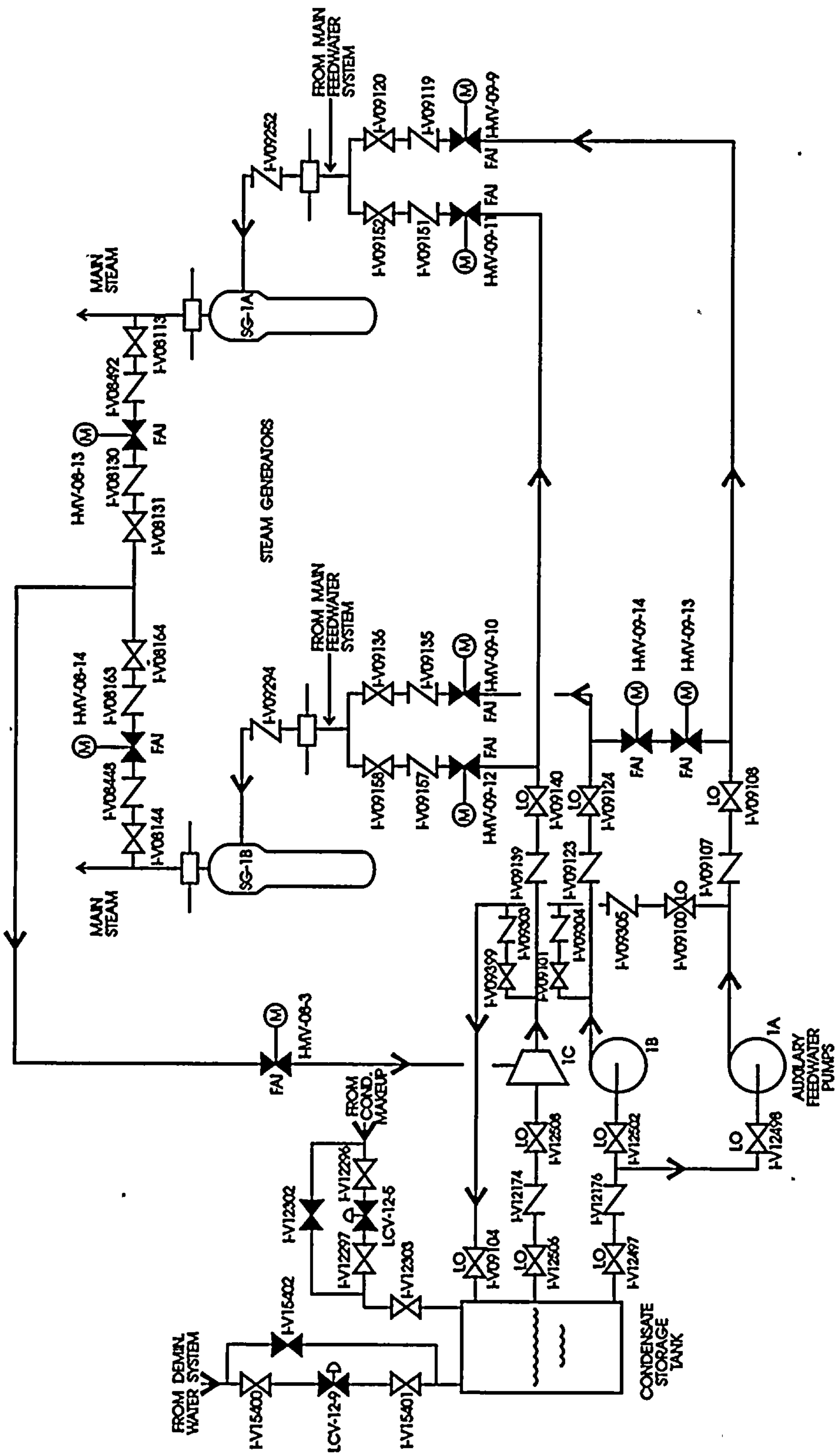
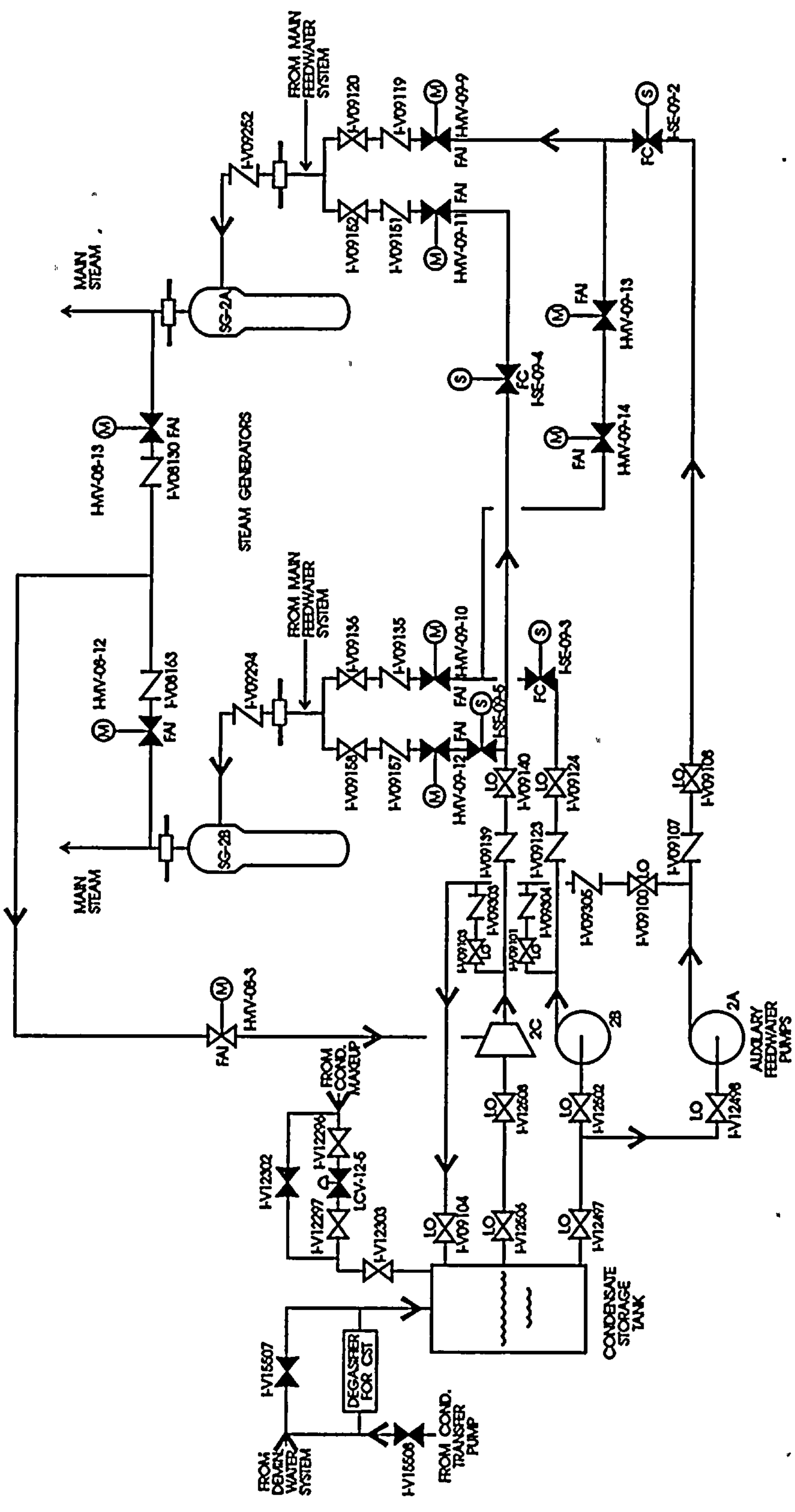


Figure B2.2 AFW System Simplified Diagram (Unit 2)



B3 COMPONENT COOLING WATER SYSTEM

B3.1 Function

The function of the Component Cooling Water (CCW) system is to provide cooling to various components and systems during normal and accident conditions. The CCW system also functions as a buffer between potentially radioactive systems and the Intake Cooling Water (ICW) system. During normal operation, cooling water is provided to the A and B essential headers and the non-essential header. The individual loads on each of these headers are listed below. During accident conditions, the CCW system only provides cooling water to the essential headers since the flow paths to the non-essential header are automatically isolated.

Header A	Header B	Header N
Shutdown heat exchanger 1A [2A]	Shutdown heat exchanger 1B [2B]	Unit 1 Fuel Pool Heat Exchanger
Containment fan coolers 1A/1B [1A/1B]	Containment fan coolers 1C/1D [1C/1D]	Sample System Heat Exchangers
High Pressure Safety Injection Pump 1A [2A]	High Pressure Safety Injection Pump 1B [2B]	Boric Acid Concentrators
Low Pressure Safety Injection Pump 1A	Low Pressure Safety Injection Pump 1B	Waste Concentrator
Containment Spray Pump 1A	Containment Spray Pump 1B	Waste Gas Compressors
[Unit 2 Fuel Pool Heat Exchangers (alternate supply)]	[Unit 2 Fuel Pool Heat Exchangers (normal supply)]	Letdown Heat Exchangers
[Unit 2 Control Room A/C 3A & 3C normal supply, 3B alternate supply]	[Unit 2 Control Room A/C 3B normal supply, 3A & 3C alternate supply]	CEDM Air Coolers
		Reactor Coolant Pumps and Motors
		Blowdown Radiation Monitoring and Sampling
		Unit 1 Containment Air Compressors
		Unit 1 Quench Tank Cooling
		[Unit 2 Sample Cooler on Condensate Recovery Tank Radiation Monitor]
		Condensate Recovery System Conductivity Sample Cooler
		[Post Accident Sample System]

B3.2 Configuration

The CCW system for Unit 1 and 2 is shown on Figures B3.1 through B3.4. The system consists of three CCW pumps, two heat exchangers, a surge tank, a chemical addition tank and associated piping and valves. The CCW system is a closed loop cooling water system that utilizes demineralized water with a corrosion inhibitor to cool various components.

The CCW system is arranged as two redundant essential headers designated A and B, each with the capability to supply the minimum cooling requirements during plant shutdown or LOCA conditions. The non-essential header, designated the N-header, is connected to both essential headers during normal operations. Following a Safety Injection Actuation Signal (SIAS), the N-header is automatically isolated from the essential supply and return headers by automatic isolation valve closure. This N-header isolation also acts to split the essential headers following an SIAS.

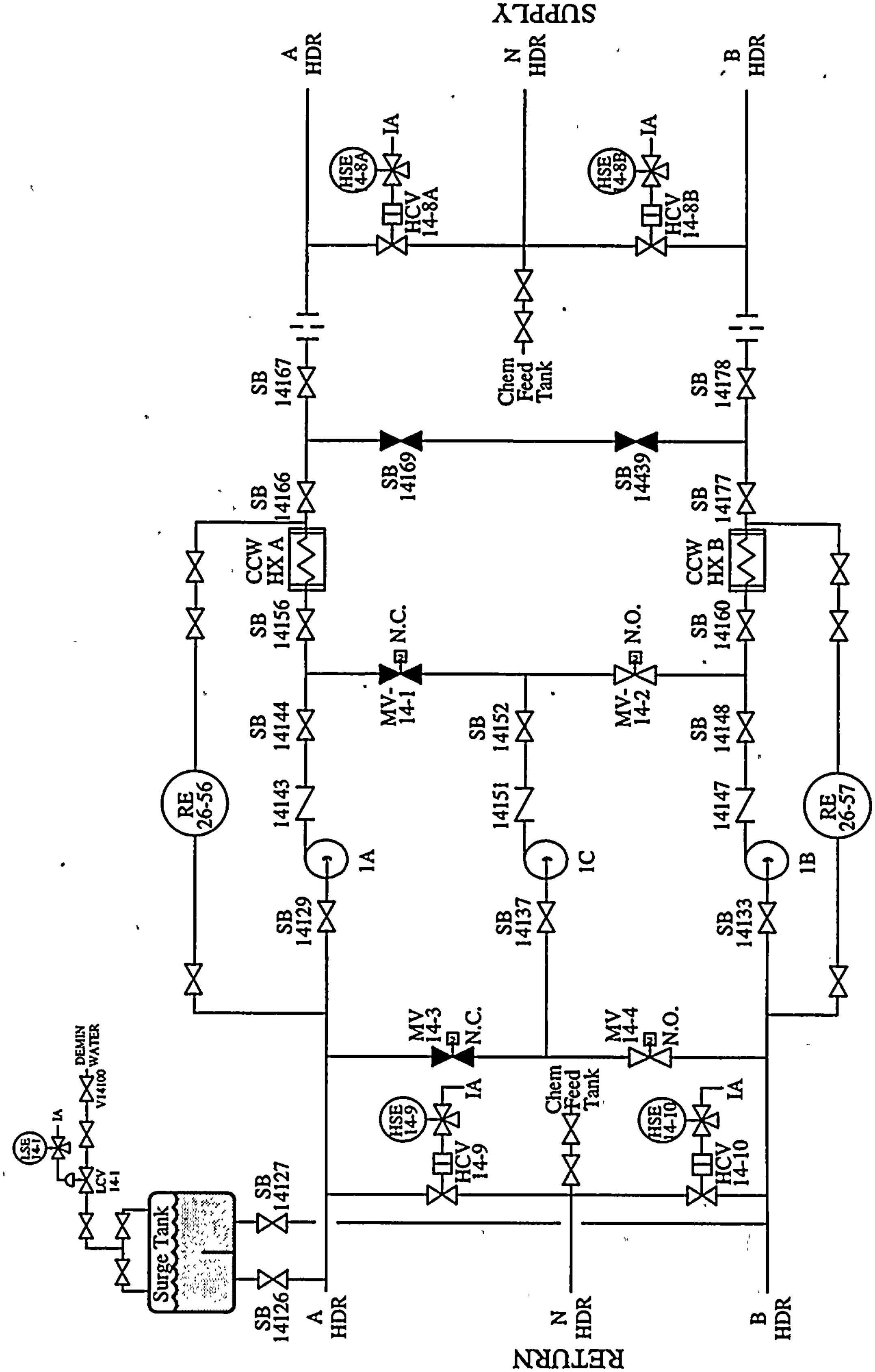
During normal operation, Pump 1A [2A] supplies the A header and Pump 1B [2B] supplies the B header. Pump 1A [2A] and 1B [2B] are powered from 4.16 kV buses 1A3 [2A3] and 1B3 [2B3] respectively. CCW Pump 1C [2C] can be aligned to either header A or B by realigning the pump suction and discharge cross-tie isolation valves (MV-14-1, MV-14-2, MV-14-3 and MV-14-4). These pump suction and discharge cross-tie valves are motor operated valves whose positions are administratively controlled. Likewise, the Pump 1C [2C] power supply, 4.16 kV Bus 1AB [2AB], can be aligned to 4.16 kV Bus 1A3 [2A3] or 1B3 [2B3]. This allows for the flexibility of aligning Pump 1C [2C] to either header during failure, test or maintenance of pump A or B. Normally the 1C [2C] pump is aligned to the B [A] header and the 4.16 kV Bus 1AB [2AB] is powered from 4.16kV Bus 1B3 [2A3].

For Unit 1, the idle 1C pump, if available, will start automatically to supply CCW to the header to which it is aligned following a SIAS, if the breaker to the pump that normally supplies that header is not closed or does not remain closed. [For Unit 2, Pump 2C, if available, will only start following an SIAS when aligned to the B header if the pump 2B breaker has been racked out and when aligned to the A header, if the pump 2A breaker is racked out or its selector switch is in ISOLATE.]

Component Cooling Water flows through the pumps, the pump discharge check valves, the pump discharge isolation valves and the heat exchanger isolation valves to the shell of the CCW heat exchangers. ICW flow rate through the CCW heat exchanger is automatically controlled to maintain CCW temperature. The cooling water then flows to the essential headers directly and to the N header via the N header isolation valves and on to the individual components cooled by CCW. CCW is returned to the pumps via the return headers which have a similar configuration.

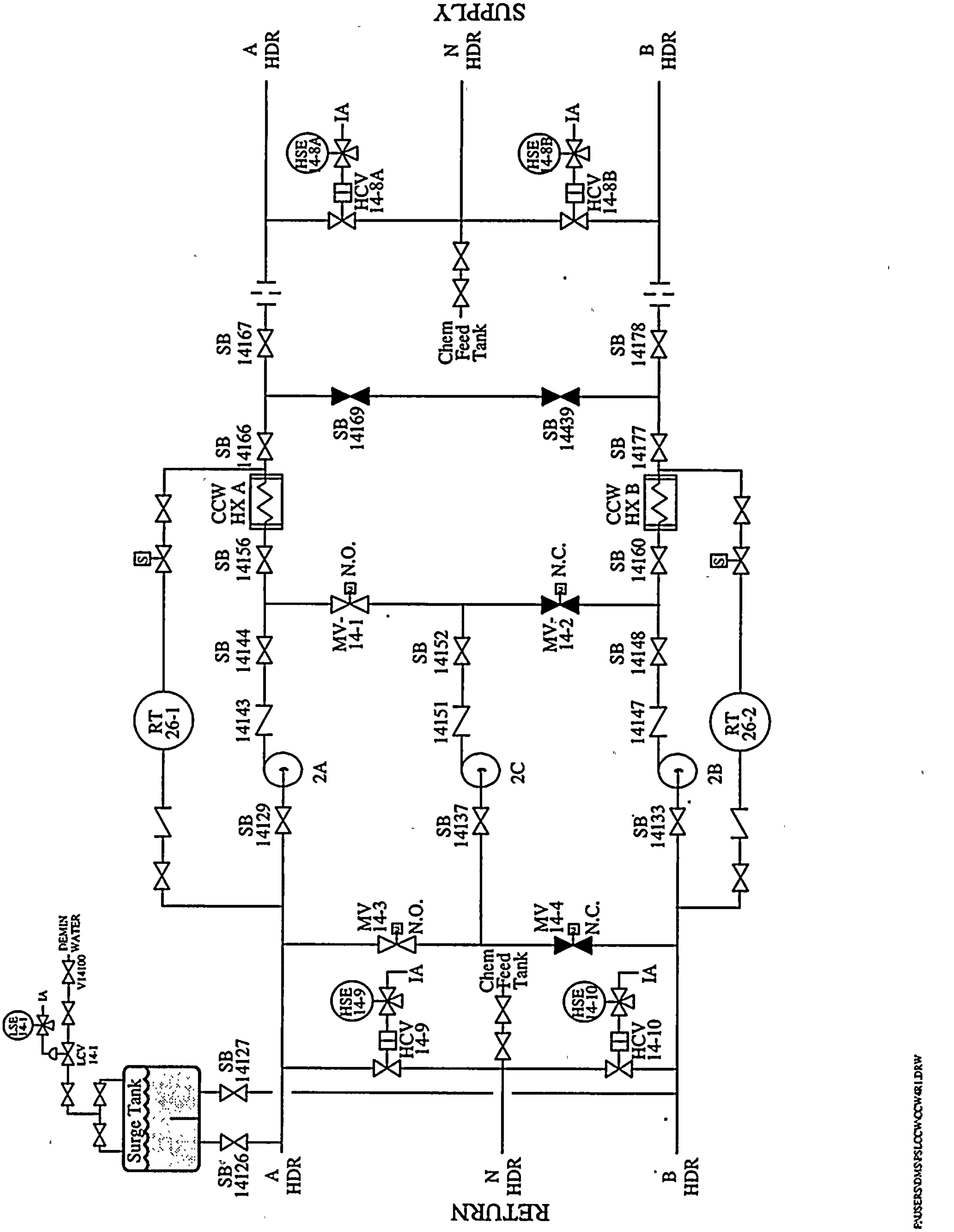
A CCW Surge Tank is provided to maintain sufficient net positive suction head to the pumps, to allow a surge volume for thermal expansion and contraction and to provide a convenient point to add make-up water. The surge tank has a divider plate to separate the supply to the essential headers. The surge tank is connected to the essential CCW return headers. A Chemical Feed Tank is also provided for the addition of corrosion inhibitors. This tank is normally isolated during operation of the system.

Figure B3.1 Unit 1 CCW Normal Operation



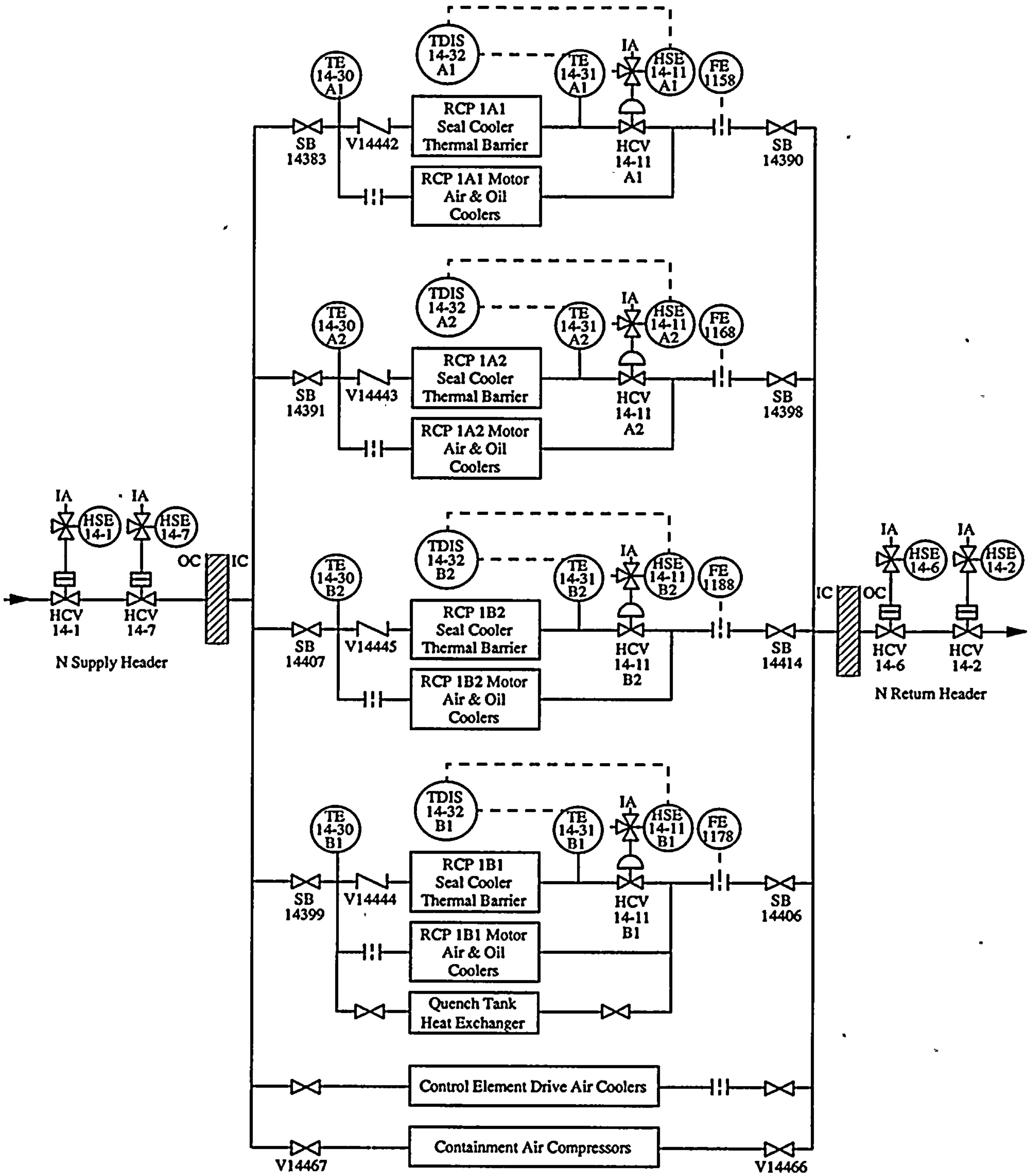
E:\CCW\CCW3.DRW

Figure B3.2 Unit 2 CCW Normal Operation



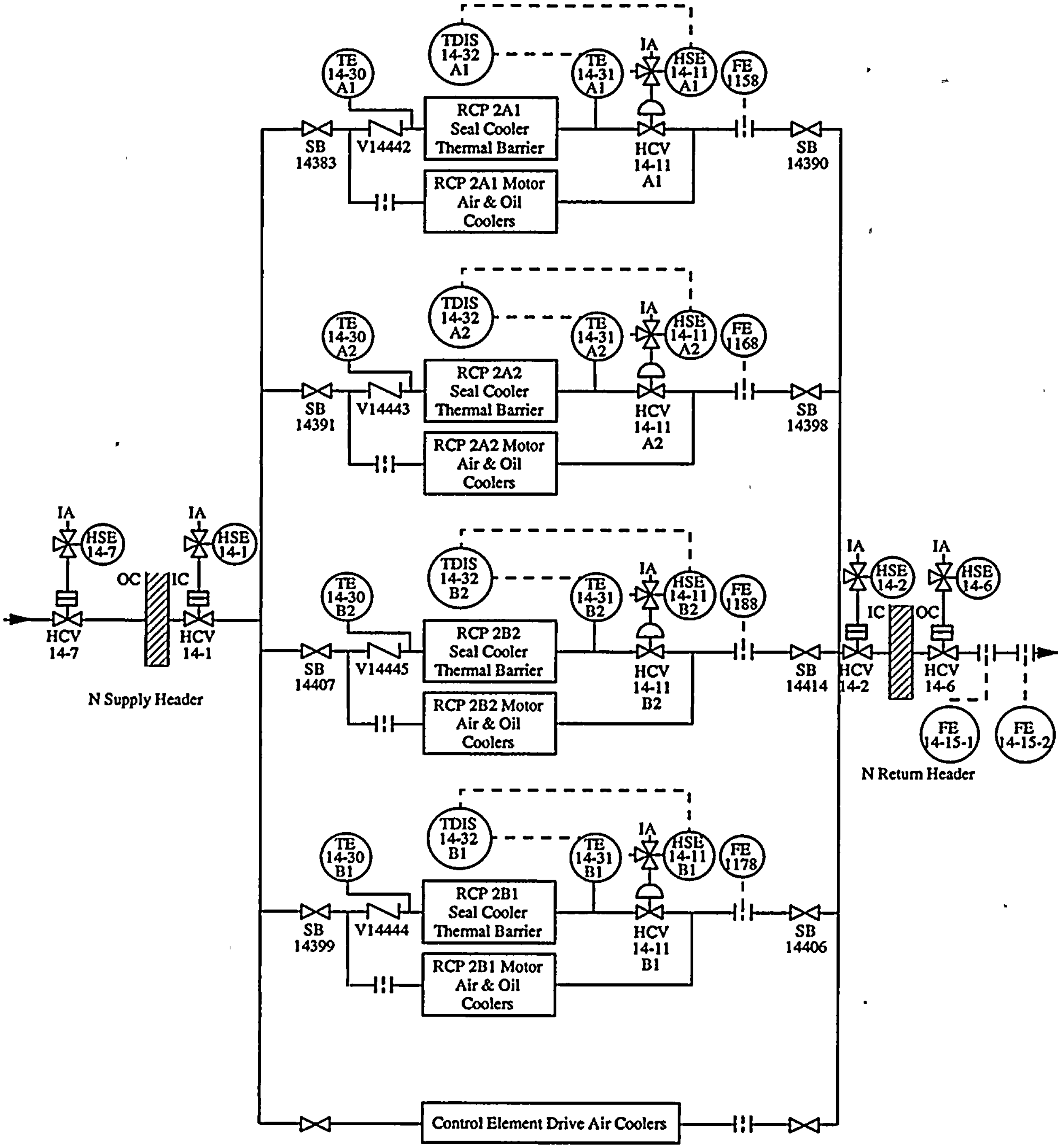
F:\USERS\DMS\PSL\CCW\CCW4R1.DRW

Figure B3.3 Unit 1 CCW to RCP's Normal Operation



F:\USERS\DMS\PSLCCW\CCW1R2.DRW

Figure B3.4 Unit 2 CCW to RCP's Normal Operation



F:\USERS\DMS\PSLCCW\CCW2R2.DRW

B3.3 Success Criteria

The CCW system must provide sufficient cooling capability to cool the safety related components following a Design Basis Accident. The minimum requirement to mitigate the design basis accident is one CCW pump supplying cooling water to one CCW heat exchanger with CCW from that header isolated from the non-essential header.

During normal operation (i.e., no SIAS and RCPs are running) CCW must supply cooling water to all Reactor Coolant Pumps and the Unit 1 Containment Air Compressors. That is, the N header receives flow from either header A or header B and flow through the N-header and RCPs is uninterrupted, returning back to the CCW pumps.

B3.4 Operation

B3.4.1 Normal Operation

The normal CCW system lineup is with pump A supplying flow to the A heat exchanger and pump B supplying flow to the B heat exchanger. Downstream of the heat exchangers the A and B headers are cross-connected through the N header isolation valves. These headers deliver flow to the CCW cooled components and return to the CCW pump suctions through their respective return headers. The return headers are again cross-connected through the N header isolation valves. Pump C is idle and is normally lined up to the B [A] train both electrically (4160 VAC Bus 1AB powered from 4160 VAC Bus 1B3 [2AB powered from 2A3]) and mechanically (MV-14-2 and MV-14-4 open and MV-14-1 and MV-14-3 closed [MV-14-1 and MV-14-3 open and MV-14-2 and MV-14-4 closed]). The CCW outlet temperature is regulated by automatically adjusting ICW flow through the CCW heat exchanger.

The CCW surge tank is connected to the A and B return headers through separate lines. Each line originates from opposite sides of the surge tank divider plate. Level in the surge tank is automatically maintained. Each compartment has a separate low level alarm and a common high level alarm. The surge tank high level alarm annunciator is combined with the compartment A low level alarm annunciator. [Additionally in Unit 2, N-header isolation valves from the affected essential header, HCV-14-8A/9 or HCV-14-8B/10, will automatically close when the associated surge tank compartment level reaches its setpoint.] The surge tank vent for Unit 1 is normally aligned to the Chemical Drain Tank. [For Unit 2, the tank is normally vented to atmosphere. If high radiation levels are detected in the CCW system, the vent automatically shifts to the Chemical Drain Tank.]

CCW is supplied to the Reactor Coolant Pumps and other non-essential loads inside containment through N header Containment Isolation valves HCV-14-1 and HCV-14-7 and is returned through Containment Isolation valves HCV -14-2 and HCV-14-6. At each RCP, CCW flow is split between the pump thermal barrier/lower seal cooler and the motor air coolers/oil coolers. Air operated valves on the outlet of the pump thermal barrier/lower seal cooler CCW return line automatically close on high temperature to isolate CCW should a reactor coolant pump seal heat exchanger leak occur. Isolation on the inlet is provided by a check valve. This line is protected from overpressurization by a relief valve [2 relief valves] when isolated. [In Unit 2, a low CCW flow to the RCPs (measured downstream of the of the N-return header Containment Isolation valves) for 10 minutes will initiate an automatic reactor trip (2-out-of-4 logic).]

B3.4.2 Accident Operation

Following a Safety Injection Actuation Signal, the N-header isolation valves (HCV-14-8A, -8B, -9 and -10) close to secure CCW flow to the non-essential header. The N-header Containment Isolation Valves also automatically close. The CCW pumps receive an automatic start signal. These automatic actions are modelled in the CCW fault tree.

Automatic actions outside the CCW boundary include the opening of HCV-14-3A and HCV-14-3B in the CCW return line from the Shutdown Heat Exchangers and for Unit 2, the closing of the Fuel Pool Heat Exchanger supply valves.

B4 CONTAINMENT ISOLATION SYSTEM

B4.1 Function

The primary function of the Containment Isolation System (CIS) is to prevent the release of gaseous or airborne radioactivity from the containment atmosphere to the outside environment. At the same time, the CIS must allow the passage of essential fluids across the containment boundary to mitigate the consequences of the accident. Prevention of liquid releases from closed systems outside containment or operating ESF systems is not a containment isolation function.

B4.2 Configuration

There are two basic types of containment penetrations: piping penetrations and integral barriers. Piping penetrations allow the passage of fluids across the containment boundary. For the most part, these penetrations rely on active closure for the containment isolation function. Integral barriers on the other hand, are passive barriers. These barriers maintain rather than change state to effect isolation. The descriptions below indicate whether certain penetrations are included or not included in the analysis.

B4.2.1 Piping Penetrations

There are 71 piping penetrations on Unit 1 and 73 on Unit 2. Most of these penetrations are provided with two containment isolation valves in series. These include manual valves, check valves, motor operated valves (MOV), and air operated valves (AOV). In some cases, however, a single isolation valve is used if the piping functions as a closed system.

The piping penetrations are classified as follows:

Class A: Penetrations that Connect Directly to the Containment Atmosphere

For penetrations in Class A, valves and/or piping or ductwork represent the only barriers between the containment atmosphere and the outside environment. These penetrations are either open directly to the containment atmosphere and connected to non-seismic piping or ductwork outside the containment or connected to non-seismic piping on both sides of the containment.

There are two categories of Class A penetrations. Class A1 includes penetrations that are normally open, or may be open, during power operation. Class A2 includes penetrations that are normally closed and are not opened during power operation.

Penetration Nos. 23 and 24, RCP Cooling Water Supply and Return, are not included because the RCP cooling water supply and return lines are neither connected directly to the RCS nor open to the containment atmosphere. Component cooling water to the RCPs is supplied from the non-essential CCW header "N" which is designed as a non-seismic, non-safety class system. While these lines may fail from seismic, jet impingement, or pipe whip forces, the RCP cooling lines are assumed to remain intact during all LOCA initiating events. Failure probabilities associated with seismic or dynamic LOCA effects are not included.

Unit 1 Penetration Nos. 48A, 48C, 51A, and 51C, H_2 Sample Lines To and From H_2 Analyzers, are not included because the piping forms a closed system outside containment. The piping is seismically qualified and is not subject to jet impingement forces or LOCA generated missiles. As a result, the failure probability for these penetrations is considered insignificant.

Class B: Penetrations that Connect Directly to the RCS

For penetrations in Class B, valves and/or piping represent the only barriers between the reactor coolant and reactor coolant exposed systems outside containment. Reactor coolant exposed systems include chemical and volume control, safety injection, shutdown cooling, and the sample system.

There are two categories of Class B penetrations. Class B1 includes penetrations that are normally open, or may be open, during power operation. Class B2 includes penetrations that are normally closed and never opened during power operation.

Penetration Nos. 40 and 64, Shutdown Cooling Suction from Loops B and A, are not included because they are connected to closed seismic Class I piping outside containment. Penetration Nos. 69 and 70, Hot Leg Injection Lines, are not included because they are required to open to mitigate certain accidents.

Class C: Penetrations that Connect to Closed Systems

For penetrations in Class C, a closed piping system inside containment and a single isolation valve represent the only barriers between the containment atmosphere and the outside environment. Closed systems inside containment that function as containment barriers include component cooling water, main steam, feedwater, and steam generator blowdown. The main steam and blowdown system inside containment are considered to be closed for all events except a main steam line break or a steam generator tube rupture.

Class C penetrations are not included because the closed system piping is considered a permanent, passive barrier that is not postulated to fail. That is, the component cooling water, main steam, feedwater, and steam generator blowdown systems are seismically qualified inside containment and designed for a higher pressure than the containment design pressure. These systems are also protected from the dynamic effects of pipe rupture. As a result, the probability of a piping failure is deemed insignificant.

The main steam, feedwater, and steam generator blowdown piping inside containment function as closed systems for all events except a main steam line break or a steam generator tube rupture. When such an event occurs, failure to close the associated containment isolation valves (i.e., isolate the affected penetrations) is included in the Power Conversion System (PCS) and Auxiliary Feedwater System (AFW) fault tree models.

Class D: Instrument Sensing Line Penetrations

The penetrations in Class D are for containment pressure instrument sensing lines. For these penetrations, a single isolation valve and a closed piping system outside containment represent the only barriers between the containment atmosphere and the outside environment. These lines are provided with either an automatic isolation valve or a remote manual valve located outside containment. A self-actuated excess flow check valve is considered an automatically actuated valve.

Class D penetrations are not included because the closed system piping (or instrument tubing) is not postulated to fail. The instrument tubing is seismically qualified and designed for a higher pressure than the containment design pressure. Since the pressure transmitters and interconnecting tubing are located outside containment, these penetrations are also protected from the dynamic effects of pipe rupture. As a result, the probability of failure of an instrument sensing line penetration is considered insignificant.

Class E: Engineered Safety System Penetrations

Penetrations in Class E (other than 48 and 51, H_2 Sample Lines) are designed to be open during a design basis event. Consequently, the containment isolation valves for these penetrations do not provide a barrier against the release of radioactivity during ESF system operation. During ESF system operation, containment integrity is maintained by a water seal established by the flow of water into containment and the volume of water collected in the containment sump.

Unit 2 Penetration Nos. 48A, 48B, 51A, and 51B, H_2 Sample Lines To and From H_2 Analyzers, are considered to be a special case of Class E. While these lines are not designed to open during a design basis event for accident mitigation, they are required to operate intermittently post-accident. When these lines are opened for H_2 sampling, containment integrity is maintained by a closed system outside containment.

Unit 2 Penetration Nos. 48A, 48B, 51A, and 51B are not included because they are connected to closed seismic Class I piping outside containment. Failure of the containment isolation valves will not result in a release of radioactivity to the outside environment because the closed system will contain any leakage past the isolation valves. There are no credible failure modes that will cause the closed system outside containment to be breached.

Containment isolation valve closure for Class E penetrations (other than 48 and 51) is considered to be a recovery action resulting from an ESF system failure. As a result, these penetrations are not included.

B4.2.2 Integral Barrier Penetrations

Integral barrier penetrations function as an integral part of, or an extension of, the containment vessel. They include large access openings, electrical penetrations, spare penetrations, and the fuel transfer penetration. Integral barrier penetrations are typically sealed with a single, passive barrier.

These barriers rely on seal welds, resilient seals, or a combination of both, for containment isolation. Due to the low probability of a seal weld failure, only the degradable mechanical seals (or resilient seals) are considered. When a resilient seal (such as an O-ring or gasket) is incorporated as part of the integral barrier, a redundant seal is included in the design for leak testing purposes. Double gaskets and concentric O-rings are examples of this. This design feature allows the space between the redundant seals to be pressurized for verification of proper sealing. Both seals must fail in order to create a release path to the outside environment, so both seals are considered.

Integral barrier penetrations are classified as follows:

Large Access Openings

Large access openings are provided in the containment vessel for equipment installation or removal and personnel access. A large diameter (28'-0") equipment hatch and a smaller diameter (12'-0") maintenance hatch are provided for transporting equipment and material across the containment boundary. The large diameter equipment hatch is seal welded closed and the smaller diameter maintenance hatch is sealed with a double gasketed, flanged and bolted cover. The large diameter equipment hatch is not included.

Two containment air locks are provided for personnel access to the containment vessel. Each lock has two double gasketed doors in series. Provision is made to pressurize the space between the gaskets for leak testing. These air locks maintain containment integrity while providing a path into and out of containment. Each air lock consists of two doors in series that are mechanically interlocked to assure that one door is closed at all times. The inside containment door provides the first barrier and the outside containment door provides the second barrier. Each door is equipped with quick acting ball valves for equalizing pressure across the doors. The doors will not be operable unless the pressure is equalized. The air lock equalization valves are also part of the containment isolation barrier. One of the valves is located on the air lock bulkhead inside containment and the other is located on the bulkhead outside containment. The valves for the two doors are interlocked so that only one valve can be opened at one time, and only when the opposite door is closed and sealed. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidently left open it can be closed by remote control.

Electrical Penetrations

Canisters or header plate penetration assemblies are used for all electrical conductors for the continuation of electrical circuits through the containment vessel, the annulus and the shield building. Sufficient cable slack is provided in the annulus to allow for differential expansion between the containment vessel and the shield building. Cable protection sleeves are provided to give support and protection to the cables in the annular space.

The primary containment penetrations feature hermetic cable sealing achieved by a ceramic, glass or high temperature thermoplastic material bonding to a metal flange. The flange is welded to a header plate or secured by screw threads and a ferrule assembly to a header plate, which in turn is welded to the penetration nozzle. The secondary seal is achieved by either epoxy resin or thermoplastic material forming a continuous seal between the metal canister pipe and all conductors. Both sets of seals provide a containment barrier and are therefore included. All penetration assemblies are provided with means to pressurize the primary canisters for monitoring of leakage rates.

The primary containment penetration is inserted in a containment vessel nozzle and is field welded inside the steel vessel to form the sealing weld. The secondary seal is inserted in a nozzle embedded in the concrete shell of the shield building aligned with the containment vessel nozzle. The secondary seal is field welded to the nozzle in the shield building. These welds do not provide a containment barrier; therefore, they are not included.

Spare Penetrations

Spare piping penetrations consist of short sections of pipe that pass through the containment vessel. They are typically sealed closed with pipe caps on both sides of the containment. The pipe caps may be either threaded onto the pipe or seal welded. In some cases, however, gasketed blind flanges are used.

The spare piping penetrations are not included because the pipe caps are not postulated to fail. For the purpose of this analysis, they are considered to be part of, or an extension of, the containment liner with regard to quality assurance, design testing, and missile protection.

Fuel Transfer Penetration

A fuel transfer penetration is provided to transport fuel between the refueling transfer canal and the spent fuel pool during refueling operations of the reactor. The penetration consists of a 36 in. diameter stainless pipe installed inside a 48 in. pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment vessel and provision is made for testing welds essential to the integrity of containment. Bellows expansion joints are provided on the pipe to compensate for building settlement and differential seismic motion between the Reactor Building and the Fuel Handling Building. The bellows expansion joints are considered as they are flexible metal surfaces which are susceptible to cyclic failure.

B4.3 Success Criteria

Success of the CIS consists of maintaining the integrity of non-essential piping penetrations and integral barrier penetrations that rely on valves, resilient seals, or flexible metal surfaces.

B4.3.1 Piping Penetrations

For piping penetrations, containment integrity is maintained as long as one of the redundant isolation barriers is closed. Each piping penetration has at least two isolation valves in series, which serve as isolation barriers, so that no single failure can prevent containment isolation.

B4.3.2 Integral Barrier Penetrations

For integral barrier penetrations, containment integrity is maintained by passive barriers that rely on mechanical sealing components. Success for these penetrations requires that the sealing mechanism remains intact (i.e. leak tight).

B4.4 Operation

The primary function of the containment isolation system is to prevent the release of gaseous or airborne radioactivity from the containment atmosphere to the outside environment. This is primarily a post-accident function. Containment integrity is established prior to Unit startup and is maintained under normal conditions, while allowing the passage of required fluids into containment to support plant operation.

B5 CONTAINMENT SPRAY SYSTEM

B5.1 Function

The primary function of the Containment Spray System (CSS) is heat removal from the reactor containment building following a Loss of Coolant Accident (LOCA) to prevent the containment pressure from exceeding its design value.

The CSS has two modes of operation:

a) The initial injection mode, during which the system sprays borated water from the refueling water tank into the containment; and

b) The recirculation mode, which is automatically initiated by the recirculation actuation signal (RAS) after low level is reached in the refueling water tank. During this mode of operation, suction for the spray pumps is from the containment sump.

The containment heat removal function can be fulfilled by either the CSS or the Containment Cooling System (CCS), or a combination of both.

B5.2 Configuration

Figures B5.1 and B5.2 depict the CSS standby configuration for Units 1 and 2, respectively.

The CSS for each unit consists of two independent and redundant trains (subsystems). The heat removal capacity of either of the two trains is adequate to keep the containment pressure and temperature below design values. Either train will bring the containment pressure below a predetermined value within 24 hours after any size break in the reactor coolant system piping up to and including a double-ended break of the largest reactor coolant pipe, assuming unobstructed discharge from both ends.

Containment spray is automatically initiated by the containment spray actuation signal (CSAS) which is a coincidence of the safety injection actuation signal (SIAS) and the high-high containment pressure signal.

Each CSS train includes the following:

a) A normally open spray pump suction path from the refueling water tank (closes on RAS)

b) A normally closed spray pump suction path from the containment sump (opens on RAS)

c) A containment spray pump

d) A normally open spray pump discharge path through a Shutdown Heat Exchanger, and

e) A normally closed air-operated valve which opens on CSAS to direct flow to an independent full capacity containment spray header.

The refueling water tank (RWT) is an aluminum [stainless steel] tank which provides a reservoir of 525,000 [554,000] gallons of water borated to a minimum of 1720 ppm. The RWT is sized to contain sufficient water to fill the refueling cavity, refueling canal, and the transfer tube to a depth of 24' above the reactor vessel flange joint. While operating in the injection mode, the RWT must supply enough water to allow operation of all Engineering Safety Features (ESF) pumps (including CSS pumps) for at least 20 minutes. The volume required for the injection mode is 305,600 [330,000] gallons. A total required tank volume of 401,800 [417,100] gallons has been established as the Technical Specification minimum tank volume.

The containment spray pumps are single stage centrifugal pumps located in separate compartments in the Reactor Auxiliary Building. Component cooling water is required to cool the seals [not required for Unit 2]. The motors are powered from safety related 4160 volt buses 1A3 [2A3] and 1B3 [2B3]. Maximum flow is 3425 gpm [3450 gpm] while taking suction from the containment sump. On a loss of off-site power the containment spray pumps are powered by the emergency diesels.

CSS valves include the following:

a) Refueling water tank outlet valves (MV-07-1A and B) are normally-open motor-operated valves which close on RAS;

b) Containment sump valves (MV-07-2A and B) are normally-closed motor-operated valves which open on RAS; and

c) CS flow control valves (FCV-07-1A and B) are normally-closed air-operated valves which open on CSAS and also fail open on loss of power or loss of air. These are the only valves required to open for initial injection of spray water, and also provide manual control of spray flow.

Each CSS train has a supply header and four spray nozzle rings located with 178 nozzles per header [178 nozzles in one header and 179 nozzles in the other]. The spray nozzles are of the open throat design and are not subject to clogging. The spray nozzles are located approximately 70 feet above the tops of the steam generators.

The containment sump is a large collecting reservoir provided to supply water to the Containment Spray and Safety Injection systems for recirculation. Located in the containment, the structurally protected containment sump receives all containment drains. The containment sump is provided with a primary and secondary debris filtration system to minimize the possibility of hindering safety injection and spray pump operation. Both sets of screens have sufficient flow area or are oriented to preclude flow restriction to the sump recirculation lines. Particulates under 1/4 in. which manage to pass through the pumps will flow through the system. Containment spray system nozzles are the non-clog type and have openings of 3/8 in. There is no mechanism by which valves or other fittings between the pump and nozzles will retain any of these particulates.

The only portions of the containment spray system which will be subjected to the containment environment associated with a LOCA are the spray headers, check valves and piping. The remaining portions of the system are located outside the containment in the reactor auxiliary building where the environmental conditions are essentially the same as those prior to the postulated accident.

B5.3 Success Criteria

The Containment Heat Removal System (CHRS) is designed to remove containment heat following an accident to prevent the containment pressure from exceeding its design value. Any of the following CHRS combinations provides the minimum required heat removal capability:

1) Operation of all four fan coolers (100% capacity).
2) Either of the two containment spray trains (100% capacity).
3) One spray train in conjunction with two fan coolers (150% capacity)[100%].

Containment spray is automatically initiated by the CSAS which is a coincidence of the SIAS and the high-high containment pressure signal. Success of the CSS requires that upon a CSAS one CSS train delivers a minimum of 2700 gpm to the reactor containment building. Initially, spray is delivered from the refueling water tank (injection mode). Following a RAS (low refueling water tank level), spray is delivered from the containment sump (recirculation mode) and the requirement is added that the spray be cooled by a shutdown cooling heat exchanger.

B5.4 Operation

B5.4.1 Normal Operation

During normal operation, the CSS is in the standby mode as shown in Figures B5.1 and B5.2, and is operated only for testing.

B5.4.2 Accident Operation

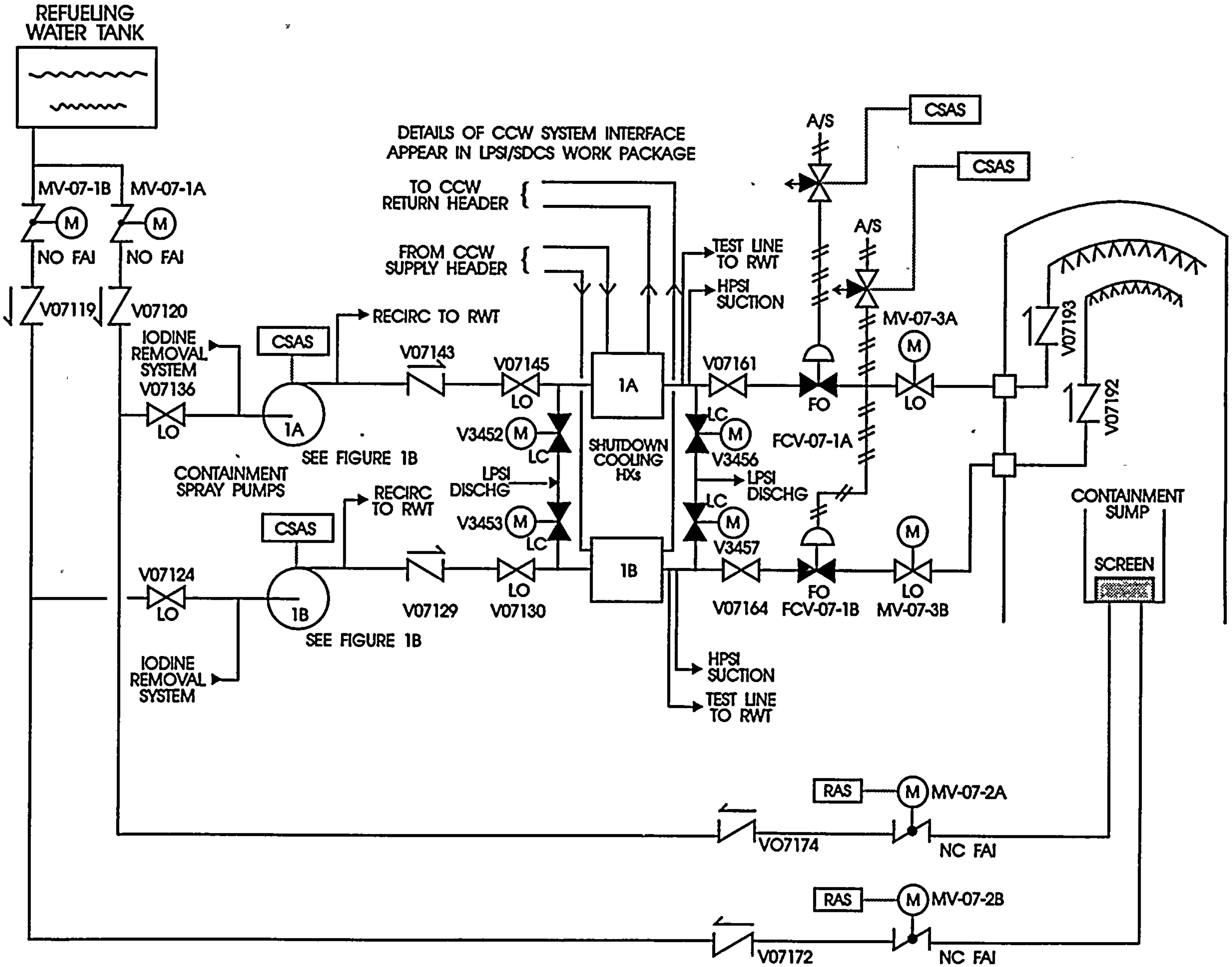
Following a Loss of Coolant Accident, the CSS operates sequentially in two modes to control containment pressure and temperature, as follows:

a) Injection Mode: A CSAS is generated upon coincidence of an SIAS and high-high containment pressure causing CSS pumps to start and flow control valves to open, initiating borated water flow to the containment spray headers. The containment spray pumps initially take suction from the refueling water tank. When low level is reached in the refueling water tank, sufficient water has been transferred to the containment to allow for the recirculation mode of operation.

b) Recirculation Mode: Upon low refueling water tank level, a RAS is generated and spray pump suction is automatically realigned to the containment sump. Automatic realignment of suction requires opening the valves in the recirculation lines. Closing the valves at the outlets of the refueling water tank is automatically initiated but not required for successful CSS operation. To assure adequate supply of water for the pumps during suction transfer, the sump valves are designed to be fully open within 30 seconds and the tank valves to be fully closed at 90 seconds following RAS. If offsite power is unavailable, CSS operation will be delayed until the pumps are loaded onto the emergency buses.

In both injection and recirculation modes the operator can manually actuate and control the system from the control room.

Figure B5.1 Simplified Schematic for Unit 1 CS System



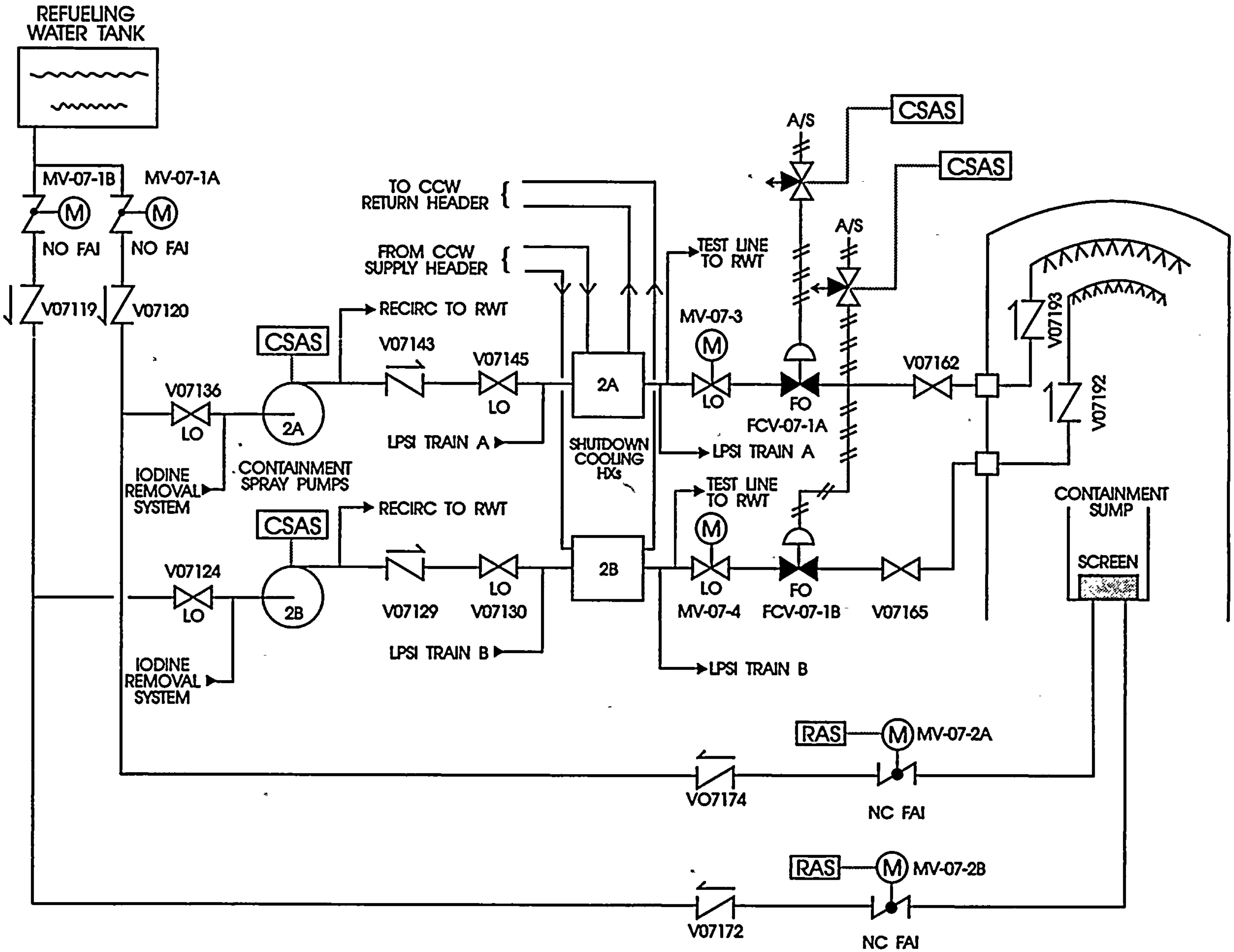


Figure B5.2 Simplified Schematic for Unit 2 CS System

B6 CHEMICAL AND VOLUME CONTROL SYSTEM

B6.1 Function

The Chemical & Volume Control System (CVCS) is designed to perform the following functions which may have an impact on plant risk:

1. Provide a means of injecting concentrated boric acid into the RCS to effect an emergency shutdown of the plant when a Safety Injection Actuation Signal (SIAS) is present. This includes providing charging flow to the RCS during LOCA events.

2. Provide auxiliary pressurizer spray for operator control of pressure in the RCS during shutdown and to allow pressurizer cooling.

B6.2 Configuration

Figures B6.1, B6.2, B6.3, and B6.4 show simplified schematics for the charging portion of the CVCS. The charging portion of the CVCS consists of charging and boric acid makeup pumps, control valves, a heat exchanger, and tanks.

<u>Charging Pumps</u>

Three charging pumps (1A, 1B, and 1C [2A, 2B, 2C]), triplex positive displacement plunger type pumps, are provided for each plant. The pump control switches are located in the control room. Each charging pump has a design flow rate of 44 gpm and is capable of developing a discharge head pressure of 2735 psig.

The combined capacity of two pumps is sufficient to match the RCS contraction rate at the design maximum rate of cooldown. The pumps are located in the reactor auxiliary building, and they take suction from the Volume Control Tank (VCT) and return the purification flow to the RCS during normal operation.

<u>Regenerative Heat Exchanger (RGHX)</u>

This vertical shell and tube heat exchanger, located inside the containment building, conserves RCS thermal energy by transferring heat from the letdown stream to the charging stream. The heat exchanger is designed to maintain a letdown outlet temperature below 450°F under all normal operating conditions. The charging fluid flows in the outer shell of the heat exchanger, extracting heat from the letdown flow.

Figure B6.1 Simplified Schematic for CVCS St. Lucie Unit 1

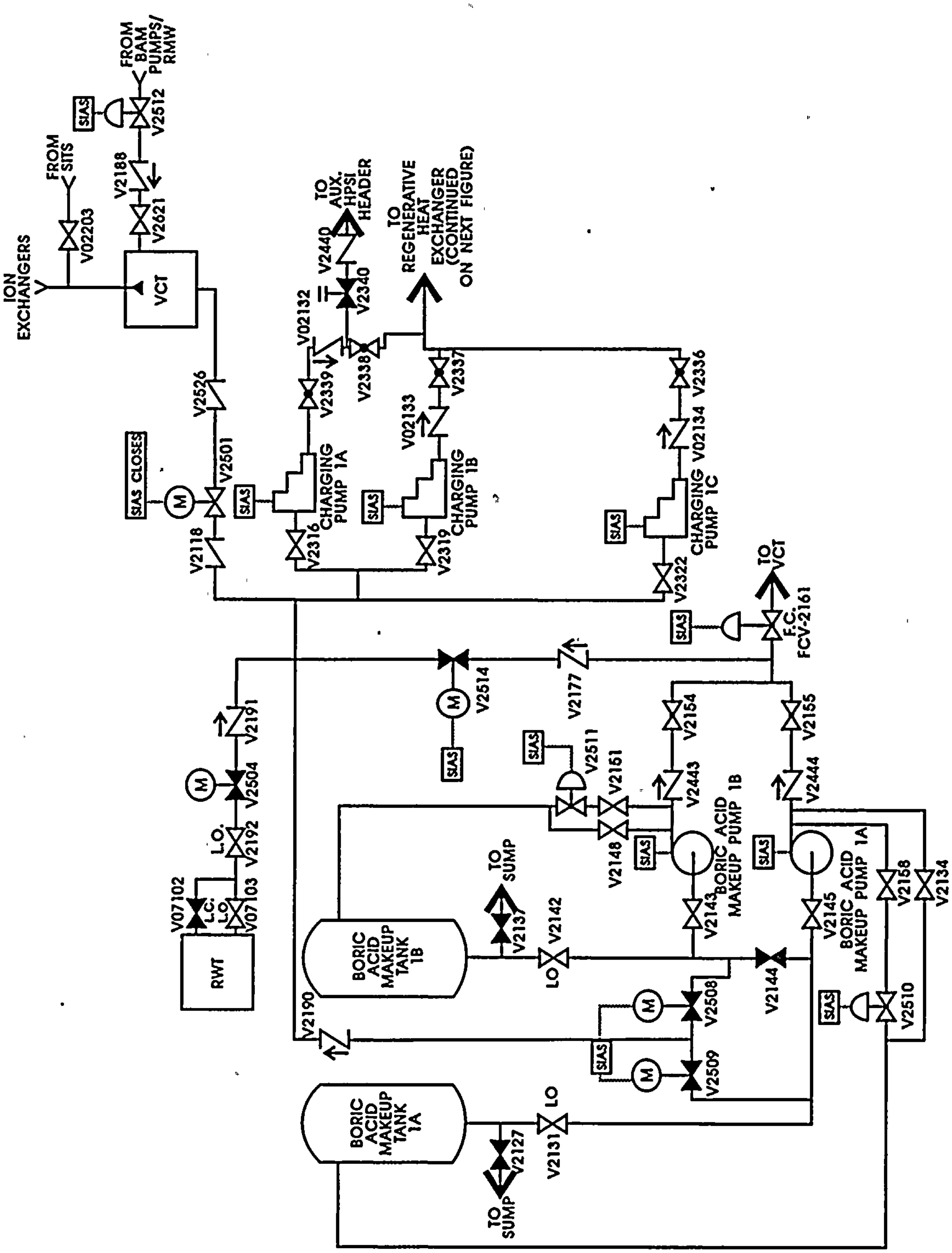


Figure B6.2 Simplified Schematic for CVCS St. Lucie Unit 1

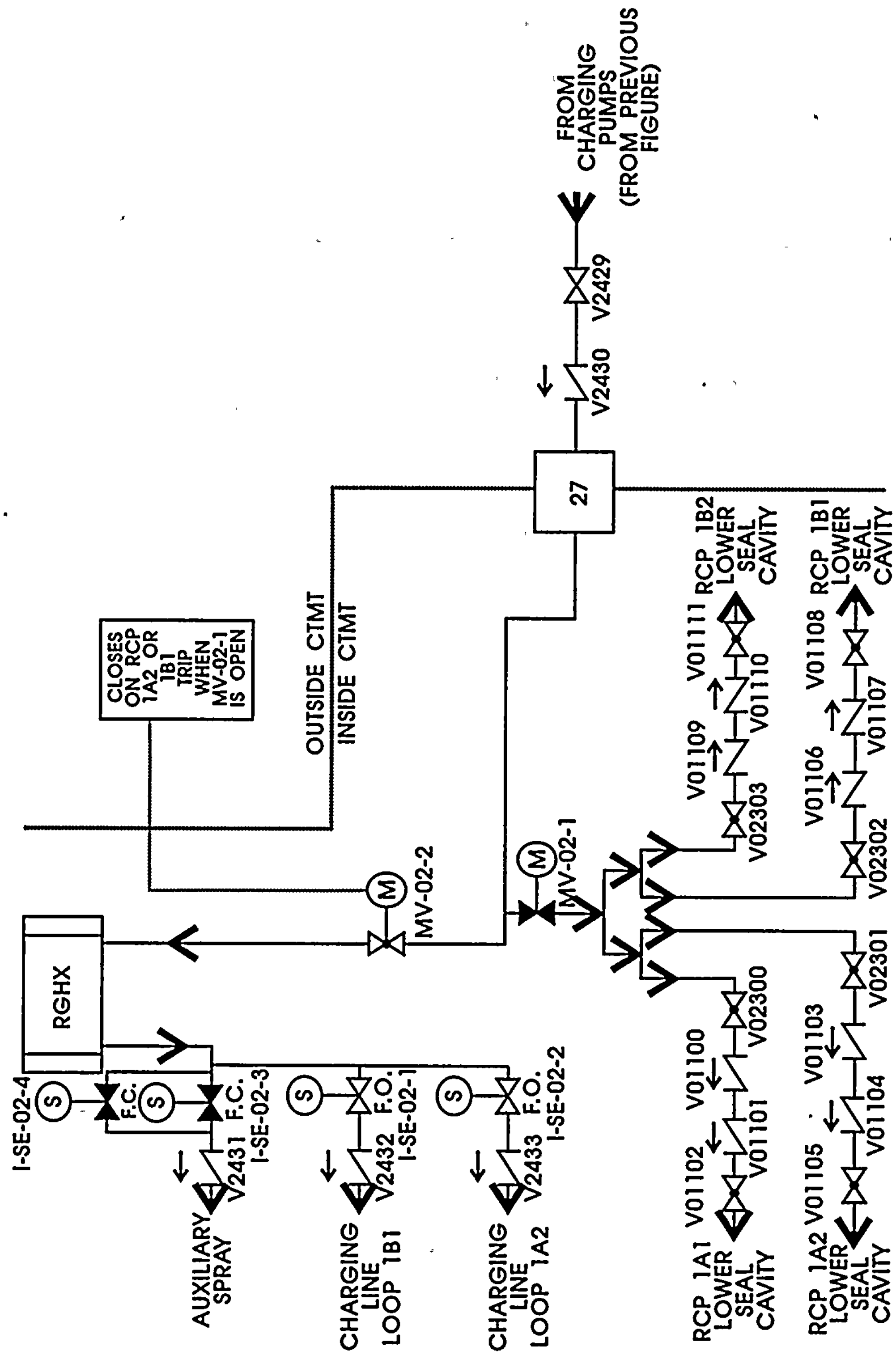


Figure B6.3 Simplified Schematic for CVCS St. Lucie Unit 2

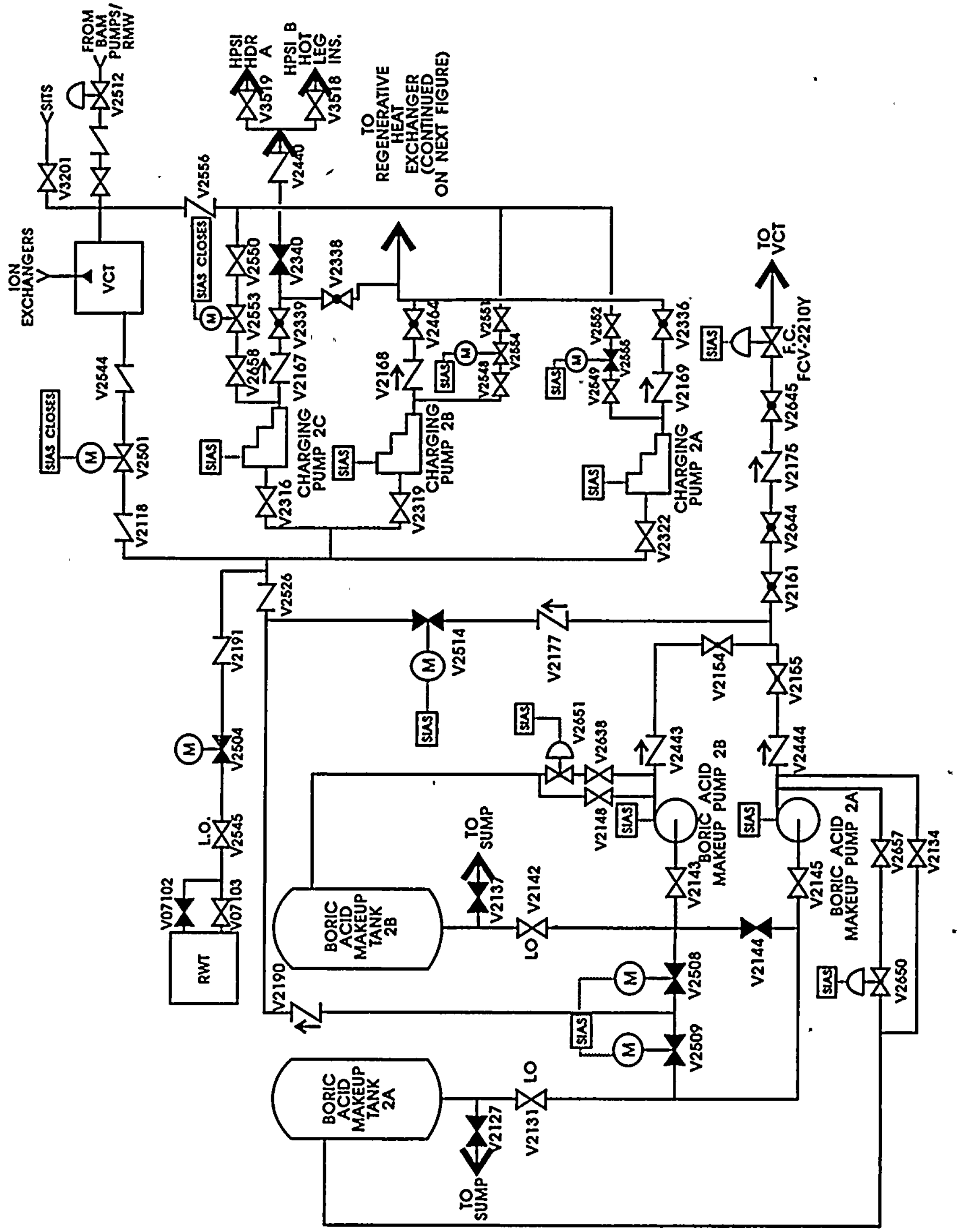
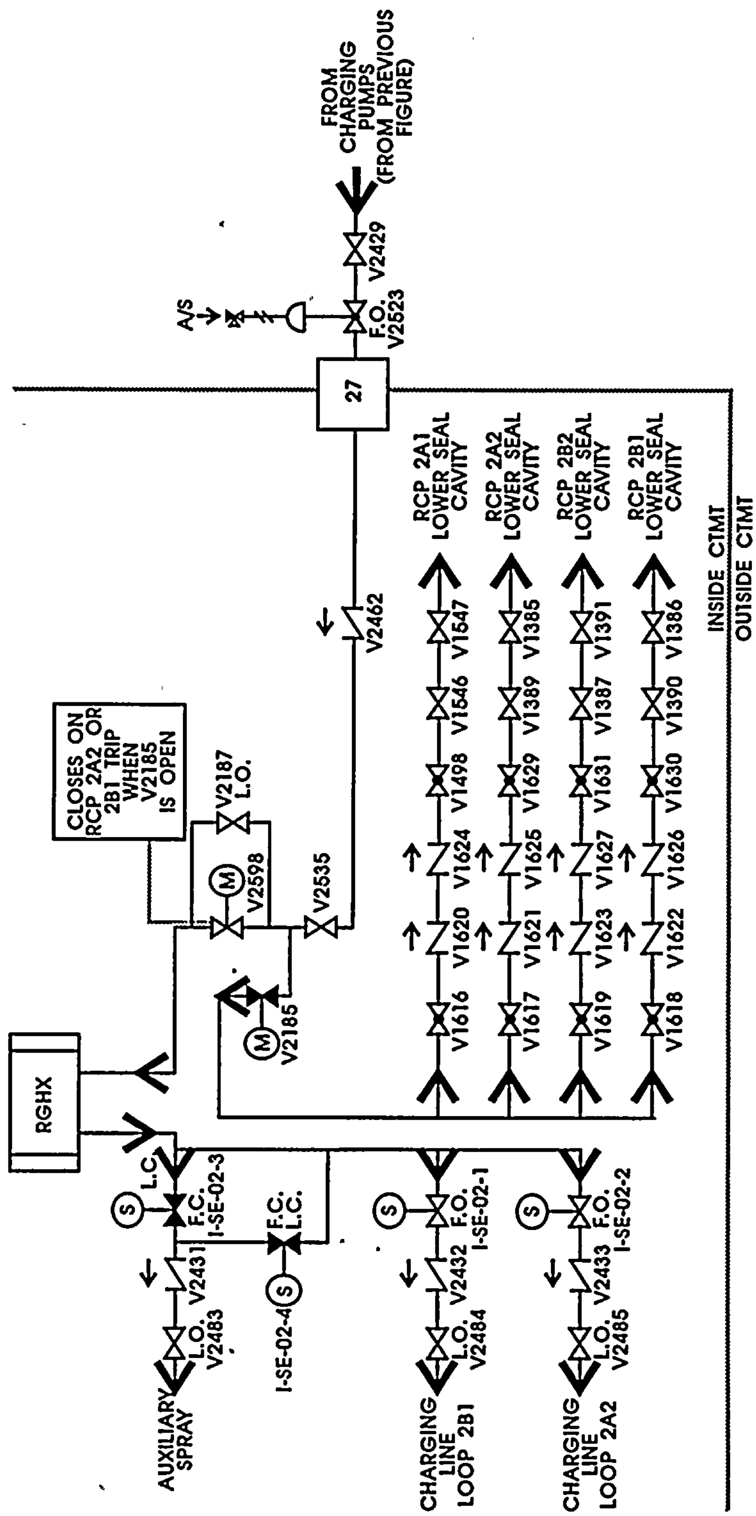


Figure B6.4 Simplified Schematic for CVCS St. Lucie Unit 2



Volume Control Tank

A single vertical, cylindrical VCT per plant is located in the Reactor Auxiliary Building. It is used to accumulate letdown flow from the RCS to provide a reservoir of reactor coolant for the charging pumps, and to maintain a desired hydrogen concentration in the RCS. Hydrogen and nitrogen supplies and a vent to the waste management system are provided to enable venting of hydrogen, nitrogen, and fission product gases.

Boric Acid Makeup Tanks

Two vertical cylindrical boric acid makeup (BAM) tanks are provided for each plant and they are located in the Reactor Auxiliary Building. These tanks provide a source of boric acid solution for injection into the RCS. Each tank is insulated and has redundant electrical strip heaters. One of the heaters has been de-energized. The combination of the BAM tanks and the refueling water tank (RWT) contain sufficient boric acid to bring the plant to a cold shutdown condition following a loss of letdown at operating conditions.

Boric Acid Makeup Pumps

Two BAM pumps located in the Reactor Auxiliary Building are provided, both of which take suction from the overhead BAM tanks and provide boric acid to the makeup subsystem and to the charging pump suction header. The capacity of each pump is greater than the combined capacity of all three charging pumps. The BAM pumps are also used to recirculate makeup tank contents, to pump from one makeup tank to the other, and to supply makeup to the RWT. The pumps are single stage centrifugal pumps with mechanical seals and liquid/vapor leakage collection connections. The pumps are insulated and the heat tracing has been de-energized due to boric acid concentration reduction.

System Valves

Inventory to the charging pumps from the VCT is controlled by motor-operated valve MV-2501 which closes when an SIAS is actuated. Check valves are located upstream and downstream of this valve. Between these valves and the charging pumps there are only manual valves (1 for each pump) which are used to isolate the pumps for maintenance. Makeup inventory from each of the two BAM tanks to the charging line (downstream of VCT check valve V2118) can be accomplished in three different ways: manual boration, emergency boration, or gravity feed. In the first two cases, the BAM tank inventory flows down through independent flow paths each containing manual isolation valves, BAM pumps, and check valves to protect the pumps from pressure transients. Both lines then converge into a single header. In the manual boration, the flow is directed into the boric acid strainer through the air-operated isolation valve FCV-2161 [manual isolation valve V-2161]. The manual boration valve V-2174 [V-2647] is opened and boric acid inventory flows into the charging header. In the emergency boration mode (initiated by an SIAS), flow is directed from the BAM pump through the motor-operated valve MV-2514 and into the charging pumps' common header. Also, during emergency boration the BAM pump recirculation flow is reduced by automatic SIAS-initiated closure of air-operated valves V2510 and V2511 [V2650 and V2651]. Gravity feed is automatically initiated by SIAS by opening the motor-operated valves V-2508 and

V-2509 and closing V-2501. The combined flow then enters the charging pump suction via a portion of the emergency boration header.

Motor-operated valve MV-02-2 [V2598] is interlocked with the Reactor Coolant Pumps RCP 1A2 or 1B1 [2A2 or 2B1]. This interlock closes the valve when any of these two RCPs trip and the valve MV-02-1 [V2185] is open. In Unit 1, the motor-operated valve MV-02-2 throttles to a preset position allowing flow to the charging header. Control room position indication is provided for these valves. [In Unit 2, valve V2598 closes and charging flow is not interrupted due to the bypass valve V2187 which is locked open.]

The flowpath downstream of valve MV-02-1 splits into two flow paths, and then each of these split into two more flow paths for a total of four. [The flow path downstream of valve V2185 splits into four flow paths.] Each of these lines feeds a single RCP seal after passing through 2 [4] manual valves in series for each RCP.

The flow path from the RGHX to the pressurizer auxiliary spray has two flow control solenoid-operated valves in parallel (I-SE-02-3, 4), and a check valve (V2431) downstream to prevent reverse flow. [In Unit 2, a locked open manual valve, V2483, is also present in the flow path.] This flow path injects directly into the RCS pressurizer spray line coming from loop 1B1 [2B1].

The flow path from the RGHX to the charging lines has two flow lines connected to RCS loops 1B1 [2B1] and 1A2 [2A2]. Each line has a flow control solenoid-operated valve (I-SE-02-1 & 2), a check valve (V2432 & V2433) to prevent reverse flow, [and a locked open manual valve (V2484 & V2485)]. These flow paths directly inject into the respective RCS loop.

B6.3 Success Criteria

The charging portion of the CVCS is considered to be successful if the following criteria are satisfied:

<u>Emergency Boration</u>

a. Either one BAM pump or one gravity feed MOV and associated tank/flowpath to the charging pump suction header must be available. After 1 hour, charging pump suction is shifted from the BAMTs to the RWT. For successful gravity feed, VCT outlet valve V2501 must shut. For successful BAM pump operation, BAM pump recirculation valves V2510 [V2650] and V2511 [V2651] must shut.

b. One charging pump and associated flowpath to the regenerative heat exchanger must be available.

c. One charging line to either RCS cold leg 1A2 [2A2] or 1B1 [2B1] must be available.

Pressurizer Auxiliary Spray

a. Either one BAM pump or one gravity feed MOV and associated tank/flowpath to the charging pump suction header must be available. After 1 hour, charging pump suction is shifted from the BAMTs to the RWT. For successful gravity feed, VCT outlet valve V2501 must shut. For successful BAM pump operation, BAM pump recirculation valves V2510 [V2650] and V2511 [V2651] must shut.

b. One charging pump and associated flowpath to the regenerative heat exchanger must be available.

c. One of two auxiliary spray solenoid valves must open.

B6.4 Operation

B6.4.1 Normal Operation

During power operations, the CVCS maintains the required volume in the RCS by compensating for coolant contraction or expansion due to plant load changes, and also maintains the purity and chemistry of the RCS inventory within specified limits.

During normal operations one charging pump is in operation taking suction from the VCT and one letdown control valve is controlled by the pressurizer level control program to maintain an exact balance between letdown flow rate plus RCS bleedoff rate and charging flow rate. The other two charging pumps are placed in AUTO and they are automatically started (on SIAS) or started/stopped as determined by pressurizer level programming due to plant loading transients. [In Unit 2, all non-running charging pumps are in AUTO with the selector switch to a running pump and one backup pump.] Operator actions for valve lineup and the starting of the pumps are provided in the operating procedure "Charging and Letdown".

A low-low level signal automatically closes the outlet valve on the VCT and switches the charging pump suction to the Refueling Water Tank (RWT) so that operation of the charging pumps continues without interruption.

[In Unit 2, on the start of one or more of the charging pumps, the charging pump bypass loop is used to minimize thermal transients to the charging piping. When the charging pump starts, the bypass is open. Charging flow does not enter the discharge pipe but flows back to the VCT. At the start of a charging pump signal, the throttle valve in the bypass line closes. As the valve closes, the pressure in the bypass line increases causing flow to begin divert to the charging header. When the charging pump receives a stop signal, the throttling valve begins to open and flow diverts through the bypass line to the VCT resulting in charging pump stopping.]

The charging flow from the charging pumps proceeds as follows: the flow passes through the shell side of the regenerative heat exchanger for recovery of heat from the letdown flow before being returned to the RCS. After flowing through the RCS, coolant flow returns to the CVCS via cold leg loop 1B1 [2B1], and then passes through the tube side of the regenerative heat exchanger for an initial temperature reduction. The cooled fluid is reduced to the operating pressure of the letdown heat exchanger by one of the two control valves. A final pressure and temperature reduction occurs at the letdown heat exchanger and one of two letdown backpressure valves. The flow then is purified by passing through a prefilter, one of three ion exchangers, a strainer, and an after-filter before is sprayed into the VCT where hydrogen gas is absorbed by the reactor coolant.

An automatic system maintains the water level in the VCT. The letdown flow is automatically diverted to the waste management system when the highest permissible water level is reached in the VCT.

A makeup system provides for changes in the RCS boron concentration and for RCS chemistry control. Concentrated boric acid solution, prepared in an electrically heated batching tank, is stored in two BAM tanks. Two BAM pumps are used to transfer the concentrated boric acid for mixing with reactor makeup water in a predetermined ratio to produce the desired boron concentration. The controlled boric acid solution is then directed into the VCT. A chemical addition tank and metering pump are used to transfer chemical additives to the suction of the charging pumps. Boric acid recovered from the waste management system (WMS) boric acid concentrator is returned to the BAM tanks. Operating procedure "Boron Concentration Control - Normal Operation" provides instructions for establishing a method of operation to supply water to the RCS, safety injection system, and RWT at a desired boron concentration for the following modes of control: Borate, Dilute, Manual, Automatic, and Shutdown Cooling Boron Concentration Control.

The operation of the automatic makeup to the VCT is started by a VCT low level signal which causes a preset solution of boric acid and reactor makeup water to be injected into the VCT. However, since the manual mode (instead of this automatic makeup of the VCT) is the preferred means of supplying the VCT with boron solution, and since valves V2512, V2525, and FCV-2161 [FCV-2210Y] are normally closed (unless open for VCT manual filling), a diversion flow to the VCT during emergency boration is not possible.

Also during normal operations, flow to the RCP seals is not provided, and valve MV-02-1 [V-2185] is in the closed position. Likewise, auxiliary spray flow to the pressurizer is not provided and solenoid valves I-SE-02-3 and I-SE-02-4 are kept closed.

B6.4.2 Accident Operation

Safe plant shutdown can be achieved without letdown flow, so automatic isolation of the letdown portion of the CVCS is provided for accident situations. Automatic letdown flow isolation is initiated from a regenerative heat exchanger high outlet temperature of 400°F, which closes letdown isolation valve V2515 and high differential pressure across the regenerative heat exchanger of 275 psid which closes letdown isolation valve V-2516. The letdown isolation valves, V2515 and V2516 also shut on an SIAS or CIS [V2522 closes on CIS only]. An accident resulting in a letdown line rupture outside containment without a coincident SIAS or CIS will be automatically isolated by the closure of V2516. This action is initiated by pressure differential switch PDIS-02-1 [PDIS-2216] which will sense a high flow through the regenerative heat exchanger. Reactor coolant pump controlled bleedoff is also isolated in accident conditions by automatic closure of the containment isolation valves I-SE-01-1 [V2524] and V2505, initiated by a CIS.

The charging pumps are used to inject concentrated boric acid into the RCS. With one pump normally in operation, the other two [one] charging pumps are automatically started by the pressurizer level control or by an SIAS. On SIAS, the stopping of the pumps due to pressurizer level or low suction trip is overridden [in Unit 2, only the pressurizer level trip is overridden by SIAS]. [In Unit 2, the running pump continues to operate and only the standby pump automatically starts on SIAS.] The capability of boration and makeup is required for safe shutdown and is assured by automatic valve action and pump control. On SIAS, the charging pump suction is switched from the VCT (V2501 shuts) to the BAM pump discharge. This action prevents a loss of water supply to the charging pump(s) normally running before the SIAS. The following automatic actions occur on SIAS:

1. Automatic closure of the following:
   a. Boron load control valve V2525
   b. Blend valve V2512
   c. BAMT recirculation valves V2510 & V2511 [V2650 & V2651]
   d. Boric acid strainer inlet valve FCV-2161 (Unit 1 only)
   e. [Boric acid flow control valve FCV-2210Y]
   f. VCT outlet valve V2501
2. Automatic opening of emergency borate valve V2514
3. Automatic starting of both boric acid makeup pumps
4. Auto start of standby charging pumps.

Also, on SIAS the BAM pump recirculation valves (V2510, V2511 [V2650, V2651]) are automatically closed. Should the pumped boric acid supply be unavailable, the charging pumps are also lined up for gravity feed from the BAM tanks by the SIAS initiated opening of gravity valves V2508 and V2509. This prevents a loss of water supply to the charging pumps.

During a loss of offsite power (LOOP) event with an SIAS, any running charging pump is stopped upon LOOP. Pumps 1A & 1B do not shed on LOOP (if previously running), and they will immediately reload when power is available. The pump 1C sheds on undervoltage, and then automatically loads onto the diesel generators after a time delay if the 1A or 1B pump, depending on "AB" bus alignment, does not load. [In Unit 2, during SIAS with offsite power available, the normally running charging pump would remain running and a second charging pump would automatically start. However, for an SIAS with LOOP, any running charging pumps are stopped upon LOOP. Two charging pumps are automatically loaded onto the EDGs (one per EDG) after a time delay.]

Should the charging line inside the containment be inoperable for any reason, the line may be isolated outside the containment, and the charging flow may be injected via the Safety Injection System A HPSI header.

As discussed above, for an SIAS with a loss of offsite power any running charging pump is stopped; two charging pumps are then automatically loaded onto the diesel generators. The charging pumps, BAM pumps, and all related automatic control valves are connected to one of the emergency busses. Each bus is supplied in turn from one of two sources; the power grid or a designated diesel generator.

B7 CONTAINMENT HEAT REMOVAL SYSTEM

The Containment Heat Removal System (CHRS) at St. Lucie consists of the Containment Spray System (CSS) and the Containment Cooling System (CCS). The CSS and CCS are analyzed as separate systems; this description applies only to the CCS.

B7.1 Function

The primary function of the CCS is to act as an independent means of containment heat removal during a LOCA, and to remove containment heat during normal operations. During normal operation three of the four fan-cooler units operate to maintain ambient containment temperature at less than 120°F. The heat removal capacity of the CCS is adequate to keep containment pressure below a predetermined value within 24 hours after any size LOCA.

The CCS alone is designed to remove containment heat post-LOCA to reduce containment pressure and temperature. However, this system can be used in conjunction with the Containment Spray System to provide the same results.

B7.2 Configuration

Figure B7.1 shows a simplified schematic for the containment coolers. Figures B7.2 and B7.3 depict the Component Cooling Water system valves for the CCS.

The CCS consists of four fan-coil cooling units, a ducted air distribution system, and the associated instrumentation and controls. The heat removal capacity of the coolers alone is adequate to keep the containment pressure and temperature below design values and to bring the containment pressure below a predetermined value within 24 hours after any size LOCA. The coolers are also designed to operate during a main steam line break (MSLB) inside containment.

Each fan cooler consists of two banks of 3 [4] copper cooling coils, casing, fan, and motor. The cooling coils are designed to remove 7.9×10^5 BTU/hr [1×10^6 BTU/hr] during normal conditions and 60×10^6 BTU/hr [61.6×10^6 BTU/hr] during accident conditions. Cooling water is supplied to the cooling coils by the Component Cooling Water system (CCW) through supply motor-operated valves MV-14-5 and 6, and the return lines are controlled by MV-14-7 and 8. [In Unit 2 the supply valves are MV-14-9, 11, 13, 15 and the return valves are MV-14-10, 12, 14, 16.] These valves are not closed by a containment isolation signal.

The Unit 1 fans are centrifugal type, direct-driven, with backwardly curved airfoil blades to provide a non-overloading characteristic. The fan motors are single speed, water cooled AC induction motors. Cooling water is supplied by the Component Cooling Water system. [The Unit 2 fans use vane axial flow fans which consist of a multi-bladed rotor assembly mounted directly to the motor shaft. The fan-rotor is of the adjustable pitch type so that air flow can be mechanically adjusted. The two-speed fan motors are not cooled by Component Cooling Water. The fan motors are cooled by air that has been through the cooling coils.]

The containment fan coolers are located outside the secondary shield wall in different quadrants of the containment. This arrangement provides separation and minimizes recirculation between units. During normal operation three of four fan-cooler units operate to supply the normal containment building cooling. The fans are powered from 480V load centers, and are automatically actuated on a Safety Injection Actuation Signal (SIAS) during post-LOCA conditions. Upon receipt of an SIAS, the standby fan cooler unit will automatically start [and the units will switch from fast to slow speed] and each of the four fans will supply post-accident heat removal air at a flow rate of 58,000 [39,600] cfm. Each fan motor is a 460 volt induction type with integral air to water heat exchanger. Each unit is sized to remove one-third of the normal heat load or one-fourth of the accident load. Containment ambient temperature under non-accident conditions is limited to 120°F when the units are supplied with CCW at 100°F.

The duct distribution system is arranged to promote mixing of the containment air and includes a common ring header to assure continuity of design air flows at all outlets. Ducts are of welded construction, reinforced and provided with pressure relief dampers to withstand LOCA induced pressure transients. The ring header is designed to attenuate high pressure transmission from the steam generator area through the duct by having blowout panels in the ductwork from the header to the steam generator and cavity cooling system. There are also gravity dampers at the point of juncture between the ring header and the ducts to prevent negative pressure in the ducts.

B7.3 Success Criteria

The CHRS is designed to remove containment heat following an accident to prevent the containment pressure from exceeding its design value. Any of the following CHRS combinations provides the minimum required heat removal capability:

1) Operation of all four fan coolers (100% capacity).
2) Either of the two containment spray subsystems (100% capacity).
3) One spray subsystem in conjunction with two fan coolers (150 % capacity)[100%].

Based on MAAP runs, however, the particular success criteria chosen for the CCS is that at least two out of four coolers must operate when an SIAS is generated due to LOCA or MSLB inside containment. [For Unit 2 the fan coolers must switch to slow speed for accident conditions.]

B7.4 Operation

During normal operation of the plant, three of the four containment fan coolers are in operation. The fourth fan is automatically started upon receipt of an SIAS following a loss of coolant accident or MSLB inside containment. [In Unit 2, all fans automatically switch to their low speeds when the SIAS is generated.] The fans can also be manually started from the control room.

Figure B7.1. Containment Cooling Unit Simplified Schematic

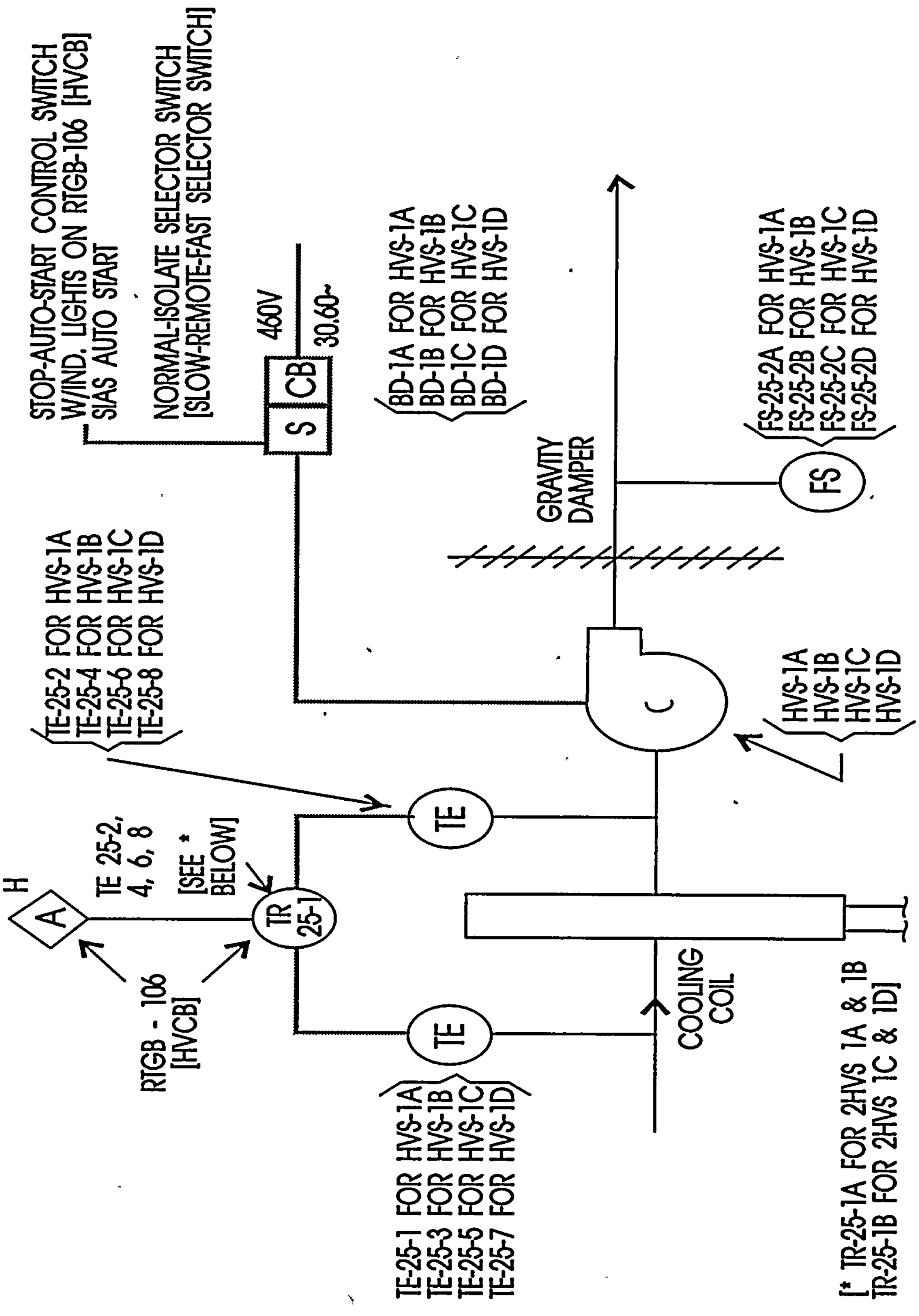


Figure B7.2 CCW Flow to Containment Coolers (Unit 1)

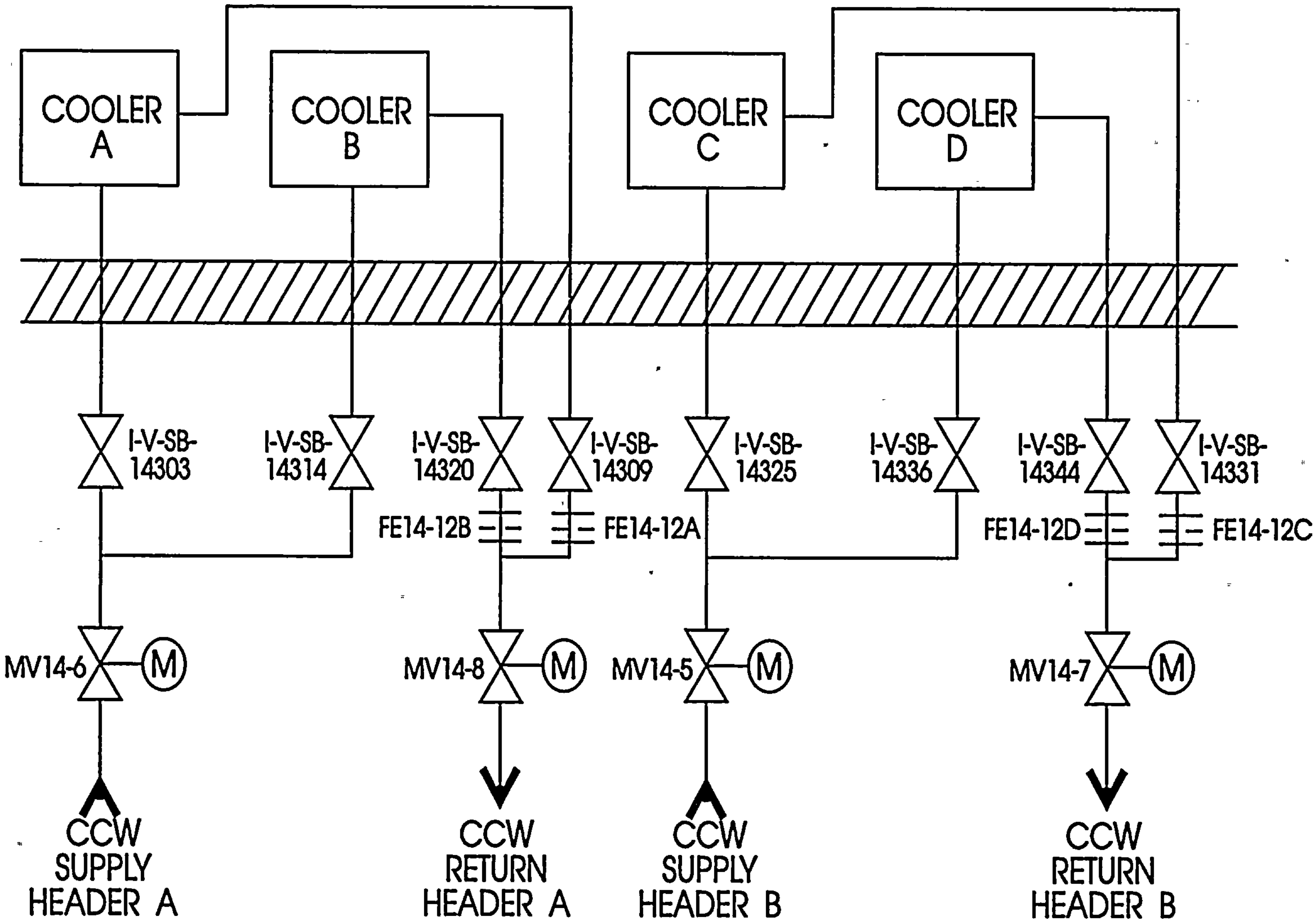
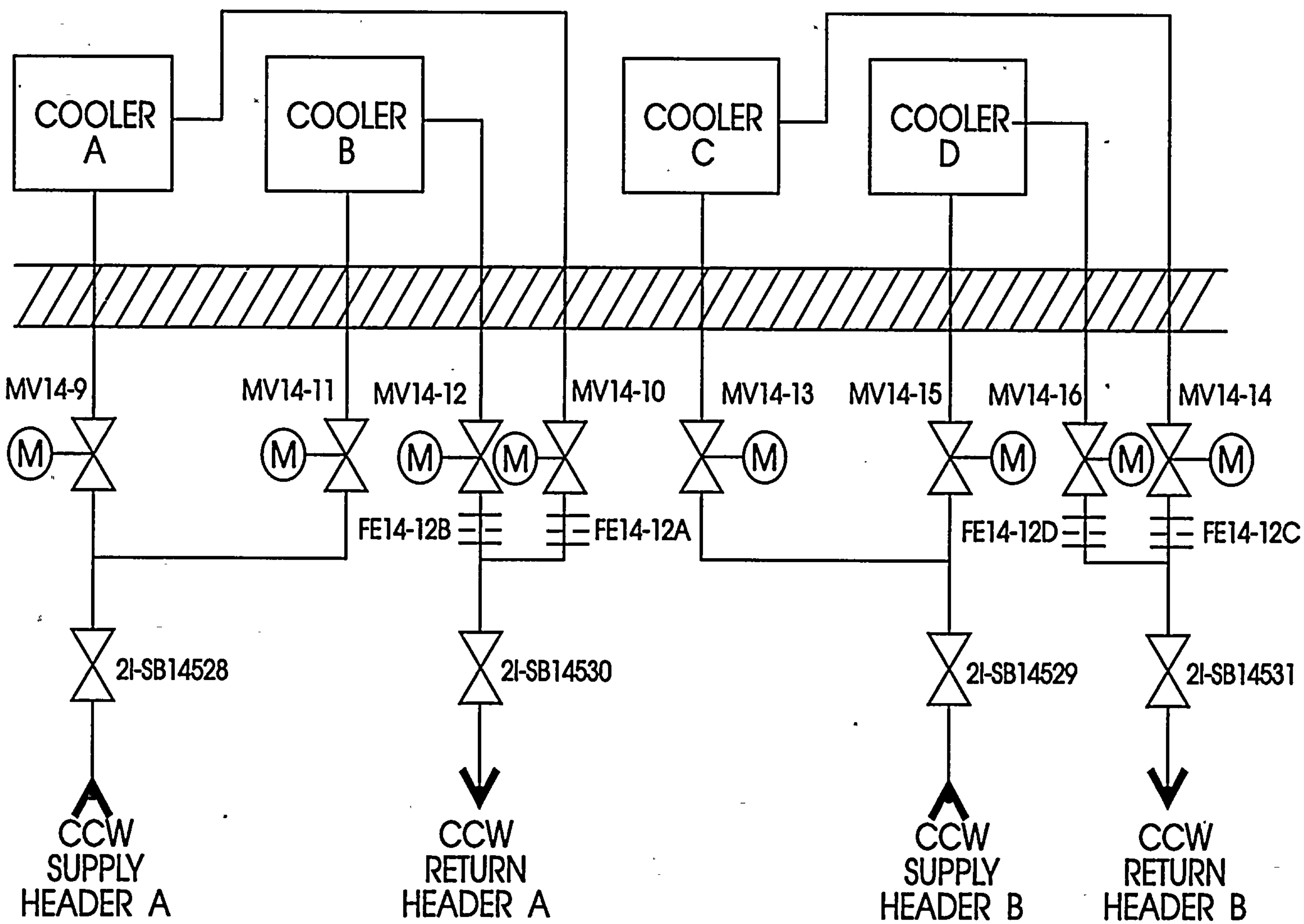


Figure B7.3 CCW Flow to Containment Coolers (Unit 2)



B8 ELECTRIC POWER SYSTEM

B8.1 Function

The Electric Power System (EPS) provides plant systems with both AC and DC power for motive, control, indicator and annunciator functions during all modes of plant operation. The EPS includes both safety related and non-safety related AC and DC subsystems. The safety related subsystems supply systems required to safely shutdown the reactor and limit the release of radioactive material following a Design Basis Accident. The configuration and operation of the safety related portions of the EPS will be described. Discussions regarding non-safety related subsystems will only be included where necessary to show support for non-safety systems included in the PRA analysis.

B8.2 Configuration

In order to simplify the following discussion, the Electric Power System will be broken into the following subsystems:

1. 240kV
2. 22kV
3. 6.9kV
4. 4.16kV
5. 480V
6. 120/208 Volt
7. 125VDC
8. 120V Vital Instrument AC
9. Emergency Diesel Generators

240kV SUBSYSTEM (See Figure B8.1)

The 240kV switchyard consists of two full capacity operating buses (East Bus and West Bus). The switchyard is further divided into four bays. A bay consists of three circuit breakers tied between the East and West buses with a tap feeding one of three 240kV transmission lines (or a local distribution station line) and a tap from one of the main generators or to one of the startup transformers.

The offsite power system provides the following reliability and flexibility:

1. Any one transmission line may be interrupted with the remaining two circuits being capable of carrying full station output.

2. Any circuit can be switched under normal conditions without affecting another circuit.

3. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.

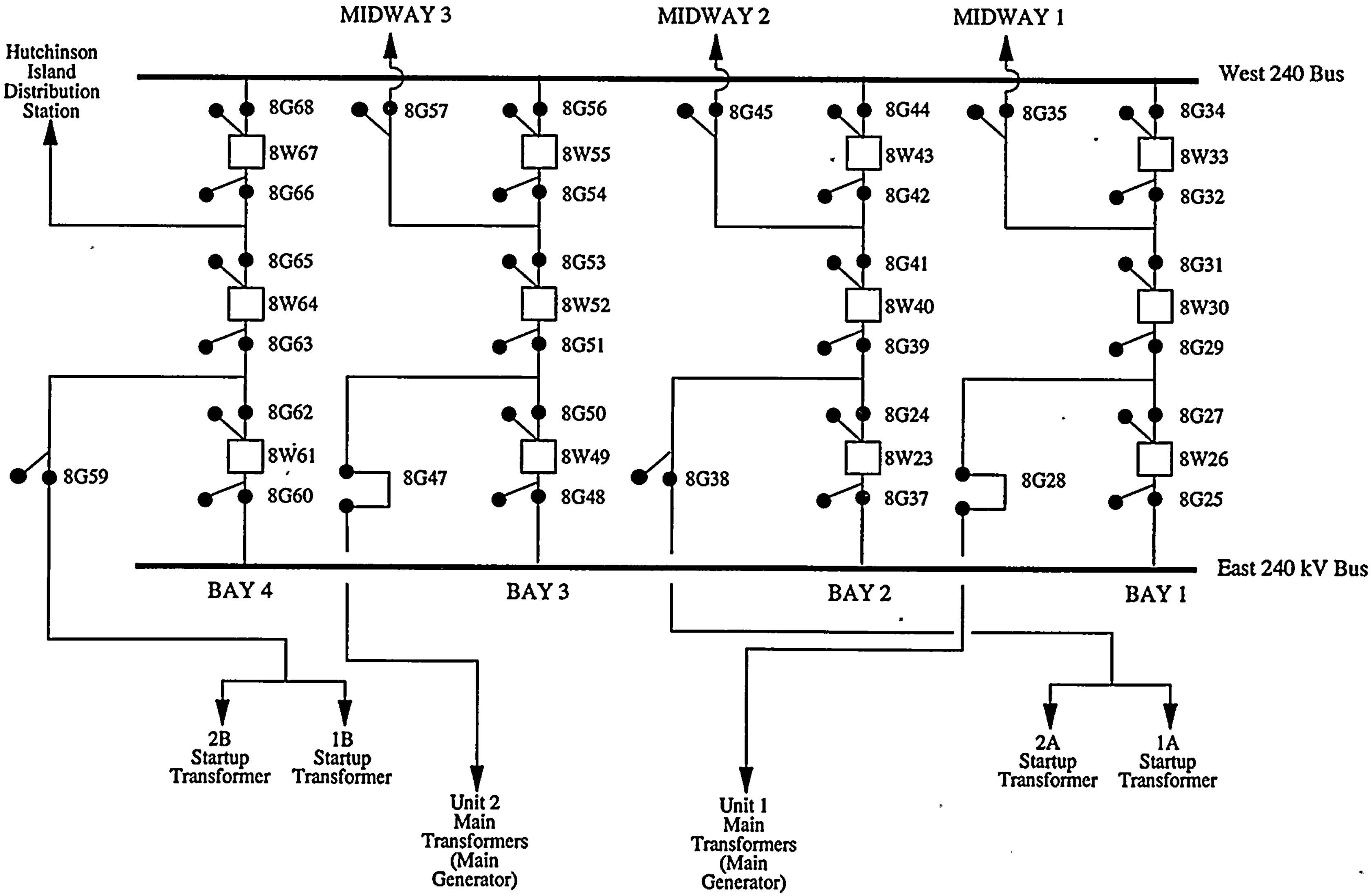


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Figure B8.1 St. Lucie Units 1 & 2 Switchyard

4. Short circuits in a single main bus will be isolated without interrupting service to any circuit.

5. Short circuit failure of a single bay breaker will not result in the permanent loss of any transmission line or any startup transformer.

6. Physical independence of power for the startup transformer is achieved by separating its switchyard connection into two different bays.

Two startup transformers are provided for each unit (1A (2A) and 1B (2B)). Each transformer steps down the voltage from 240kV to 6.9kV and 4.16kV for distribution to 6.9kV buses 1A1 (2A1) and 1B1 (2B1) and 4.16kV buses 1A2 (2A2) and 1B2 (2B2). A single startup transformer is sized to accommodate the auxiliary loads of both units provided operating procedures limit load sufficiently to prevent overloading.

<u>22kV SUBSYSTEM</u> (See Figures B8.2 and B8.3)

The Unit 1 (2) main generator supplies electrical power at 22kV through an isolated phase bus to the Unit 1 (2) parallel connected main transformers (1A (2A) and 1B (2B)) which step up the voltage to 240kV for transmission to the switchyard. The isolated phase bus also supplies power to the Unit 1 (2) auxiliary transformers (1A (2A) and 1B (2B)) which step down the voltage to 6.9kV and 4.16kV for distribution to 6.9kV buses 1A1 (2A1) and 1B1 (2B1) and 4.16kV buses 1A2 (2A2) and 1B2 (2B2).

<u>6.9kV SUBSYSTEM</u> (See Figures B8.2 and B8.3)

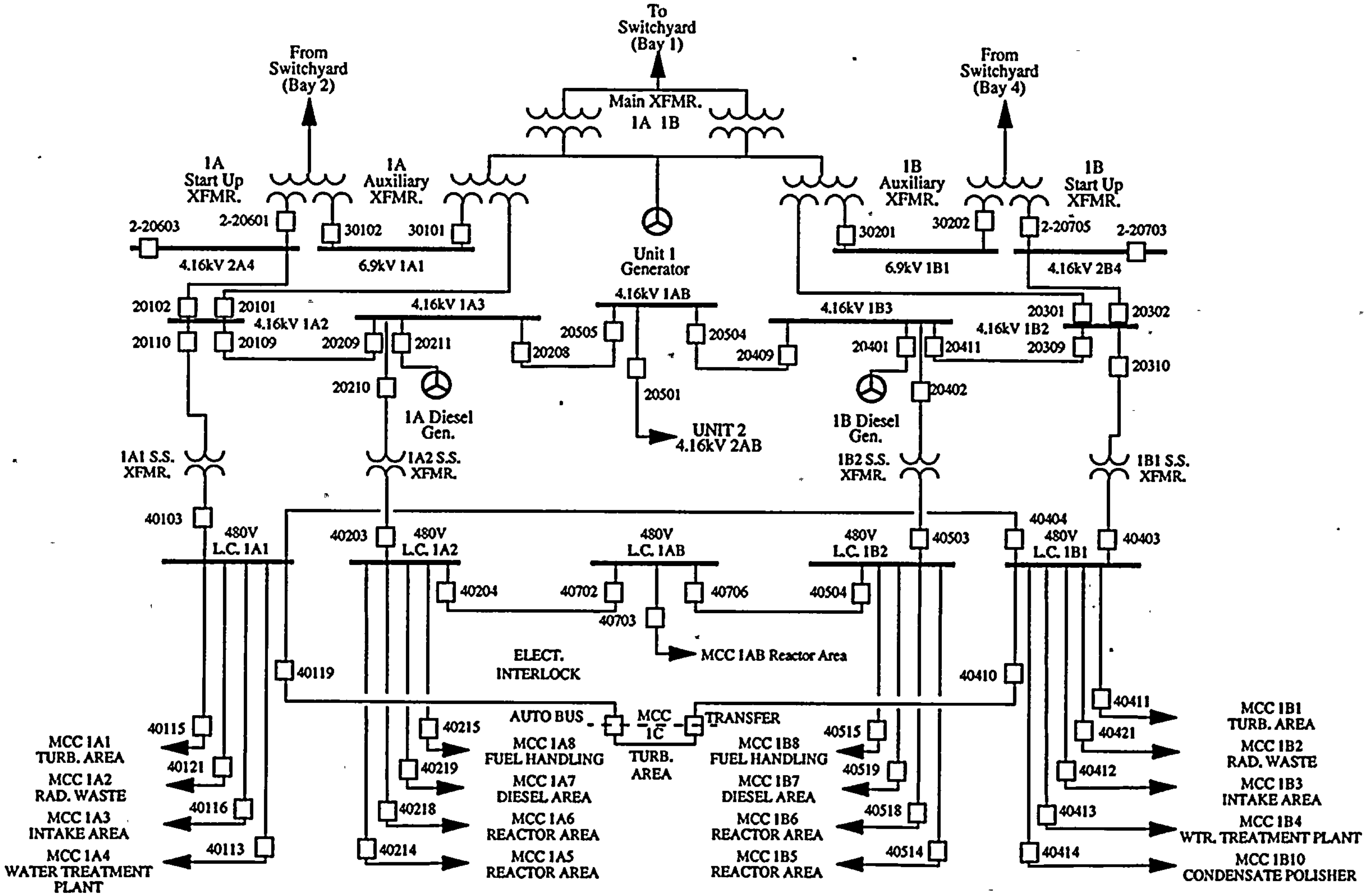
There are two 6.9kV buses per Unit (1A1 (2A1) and 1B1 (2B1)). These buses are non-safety related and power the main feedwater pumps and reactor coolant pumps. Remote operation of the breakers requires 125VDC control power.

<u>4.16kV SUBSYSTEMS</u> (See Figures B8.2 and B8.3)

There are five 4.16kV buses per unit (1A2 (2A2), 1B2 (2B2), 1A3 (2A3), 1B3 (2B3), and 1AB (2AB)). 4.16kV buses 1A2 (2A2) and 1B2 (2B2) are non-safety related and provide power to the condensate pumps, circulating water pumps, heater drain pumps, and non-safety related 480V load centers. 4.16kV buses 1A3 (2A3) and 1B3 (2B3) are safety related and power safe shutdown loads. Power from the startup transformers to buses 1A3 (2A3) and 1B3 (2B3) is via 4.16kV buses 1A2 (2A2) and 1B2 (2B2), respectively.

Upon an undervoltage on safety related buses 1A3 (2A3) and 1B3 (2B3), the tie breakers between the non-safety related (1A2 (2A2) and 1B2 (2B2)) buses and safety related buses automatically open and the Emergency Diesel Generators (1A (2A) and 1B (2B)) automatically provide power to the safety related buses (1A3 (2A3) and 1B3 (2B3)).

Figure B8.2 St. Lucie Unit 1 Electrical Distribution System



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

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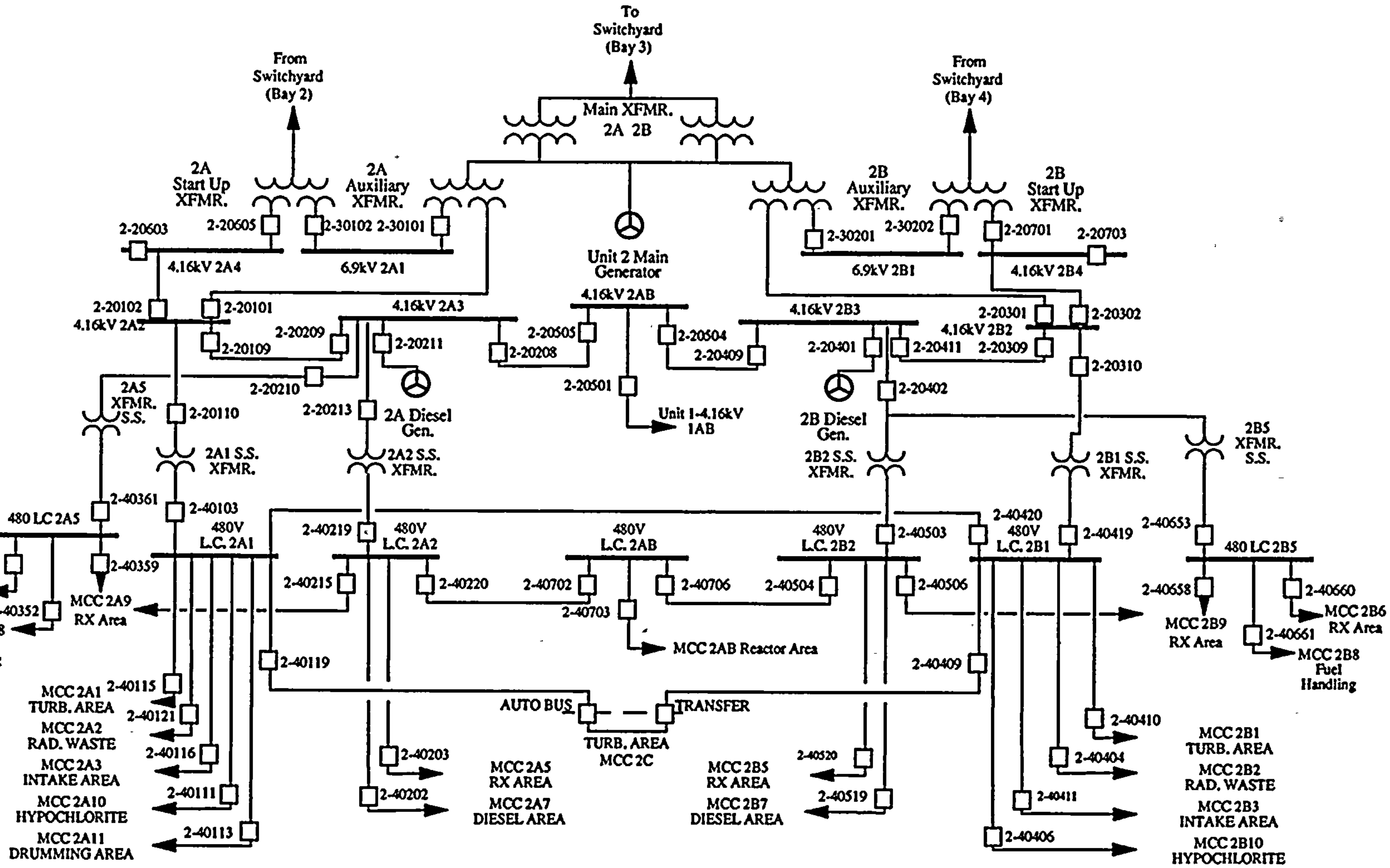
Figure B8.3 St. Lucie Unit 2 Electrical Distribution System



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

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Undervoltage protection for Unit 1 safety related 4.16kV buses 1A3 and 1B3 is provided in two levels. The first level utilizes two undervoltage definite-time delay relays, in a 2-out-of-2 coincident logic, for loss-of-voltage protection. These relays function to initiate source disconnection, load shedding, diesel generator starting, and load sequencing on the affected bus. The second level of undervoltage protection utilizes two sets of two undervoltage definite-time delay relays in a 2-out-of-2 coincident logic for degraded grid protection. Each set of relays has different voltage and time delay settings which define two distinct points on the voltage/time curve required for equipment protection.

Bus undervoltage protection for Unit 2 safety related 4.16kV buses 2A3 and 2B3 is also provided in two levels. The first level detects a loss of offsite power. One inverse time voltage relay is provided for safety related buses 2A3 and 2B3. Upon detection of a loss of voltage condition, this relay automatically initiates diesel generator starting and disconnection of the offsite source. A second level of undervoltage protection is also provided for buses 2A3 and 2B3 which utilizes a coincident logic protection scheme consisting of three definite time relays. The relay logic actuates control room annunciation to alert the operator to a degraded voltage condition and aligns the circuitry associated with the undervoltage logic such that subsequent occurrence of a safety injection actuation signal (SIAS) separates the safety related buses from the offsite power system automatically.

There is one "swing" 4.16kV bus (1AB (2AB)) per unit. This bus can be manually aligned to either the 1A3 (2A3) or 1B3 (2B3) 4.16kV bus. This bus powers the "C" intake cooling water pump and "C" component cooling water pump.

4.16kV buses 2A4 and 2B4 provide the capability of paralleling the 4.16kV buses of one power train ("A" or "B") of Units 1 and 2. This allows safety buses 1A3 and 2A3 or 1B3 and 2B3 to be powered from a single startup transformer or diesel generator. This alignment can only be performed manually and under strict administrative controls.

A blackout crosstie provides the capability of powering one vital 4.16kV bus on one unit from an EDG on the other unit through the "AB" 4.16kV buses.

All 4.16kV breakers require 125VDC control power for remote operation. Breakers can also be operated manually.

480V SYSTEM (See Figures B8.2 and B8.3)

480V Load Centers

There are two non-safety related 480V load centers (LC) per unit (1A1 (2A1) and 1B1 (2B1)), three safety related load centers on Unit 1 (1A2, 1B2, and 1AB), and five safety related load centers on Unit 2 (2A2, 2A5, 2B2, 2B5, and 2AB). With the exception of 1AB (2AB), each load center is powered from a 4.16kV bus through a 4.16kV/480V station service transformer. Load center 1AB (2AB) is powered from either the 1A2 (2A2) or 1B2 (2B2) load center. Re-alignment must be performed manually. Remote breaker operation requires 125VDC control power. Load center breakers may also be operated manually.

480V load centers power motors above 100 hp. (and generally below 300 hp.) and 480V motor control centers.

480V Motor Control Centers

There is a total of nineteen 480V motor control centers (MCC's) on Unit 1 - twelve non-safety related (1A1, 1B1, 1C, 1A2, 1B2, 1A3, 1B3, 1A4, 1B4, 1A8, 1B8, and 1B10), and seven safety related (1A5, 1A6, 1B5, 1B6, 1A7, 1B7, and 1AB). On Unit 2, there is a total of twenty-one MCC's divided twelve non-safety related (2A1, 2B1, 2C, 2A2, 2B2, 2A3, 2B3, 2A8, 2B8, 2A10, 2B10, and 2A11) and nine safety related (2A5, 2B5, 2A6, 2B6, 2A7, 2B7, 2A9, 2B9, and 2AB). MCC 1B9 powers the SGBTF.

120/208 Volt System

Power is provided for normal lighting and other plant loads requiring an unregulated power supply by the 120/208V system. This system consists of distribution panels and transformers fed from 480V MCC's. These panels feed both safety related and non-safety related loads.

125VDC SYSTEM (See Figures B8.4, B8.5, B8.6, B8.7)

The 125VDC system provides DC power for plant control and instrumentation and for operation of DC motor operated equipment. There are two non-safety related DC buses (1C (2C) and 1D (2D)) and three safety related buses (1A, (2A), 1B (2B), and 1AB (2AB)) per unit.

Buses 1C (2C) and 1D (2D) are each powered from a charger (1C (2C) and 1D (2D)) and a battery (1C (2C) and 1D (2D)). Safety related buses 1A (2A) and 1B (2B) are each powered from two battery chargers connected in parallel (1A (2A) and 1AA (2AA), 1B (2B) and 1BB (2BB)), and a battery (1A (2A), 1B (2B)). Each charger is sized to carry normal DC load and to recharge a battery. A fifth safety related charger (1AB (2AB)) on the 1AB (2AB) DC bus provides a backup for the four operating safety related chargers. The 1AB (2AB) DC bus is powered from either the 1A (2A) or 1B (2B) DC bus. Re-alignment of the 1AB DC bus to its alternate source is manual.

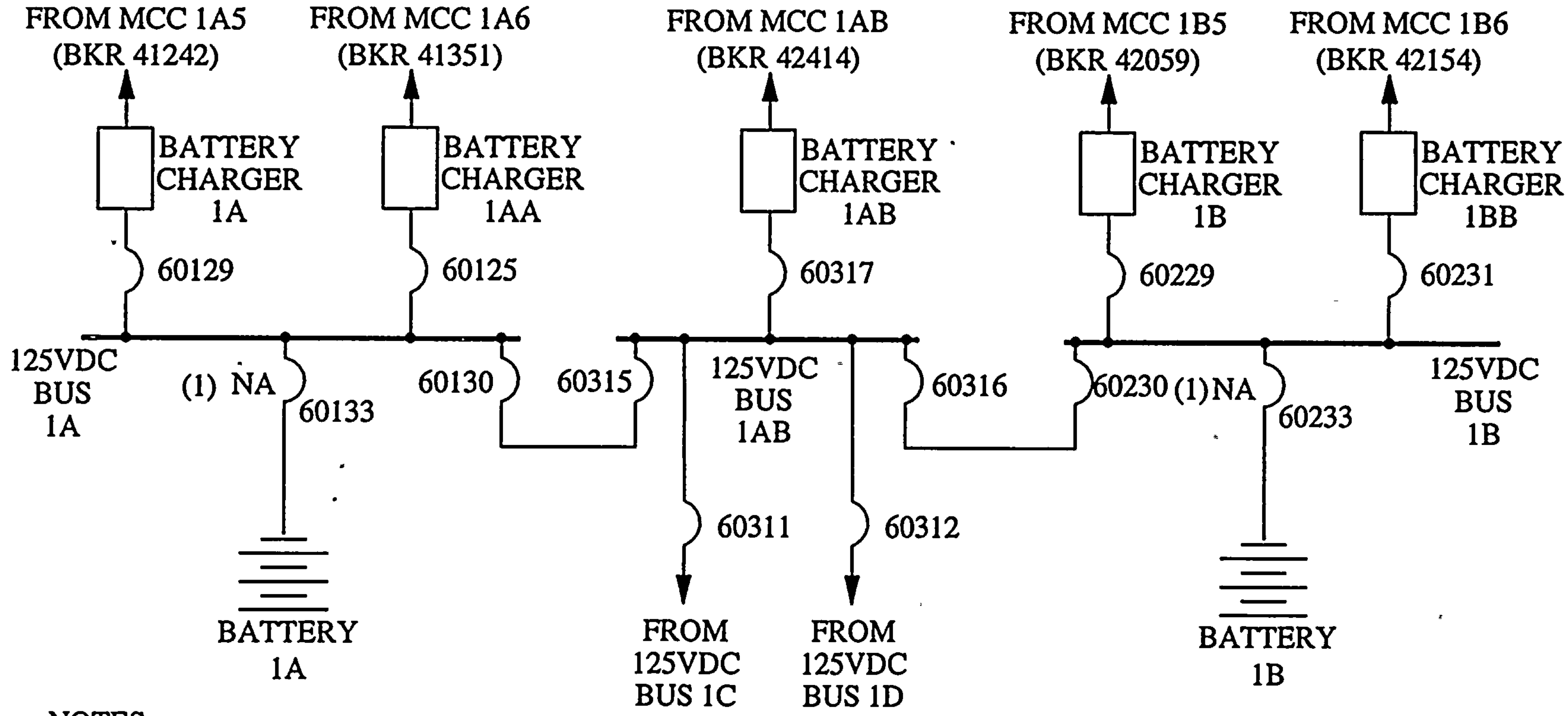
Each of the Unit 1 safety related batteries have sufficient capacity for an 8 hour emergency period without assistance from a battery charger with all non-emergency loads disconnected within one hour. Safety related battery chargers are automatically loaded onto the diesel generators approximately 40 seconds following a loss of offsite power. The Unit 2 safety related batteries have the same 8 hour discharge rating as the Unit 1 batteries. This rating is sufficient to supply DC loads following a loss of offsite power until the chargers are powered by the diesel generators (approximately 40 seconds).

120VAC VITAL INSTRUMENT AC SYSTEM

Unit 1 (See Figure B8.8)

Four redundant 120VAC single phase instrument power buses (1MA, 1MB, 1MC, and 1MD) provide power to essential instrumentation and control loads. Each bus is supplied separately from

Figure B8.4 St. Lucie Unit 1 Vital 125VDC



NOTES:
(1) NA DENOTES NON-AUTOMATIC BREAKER

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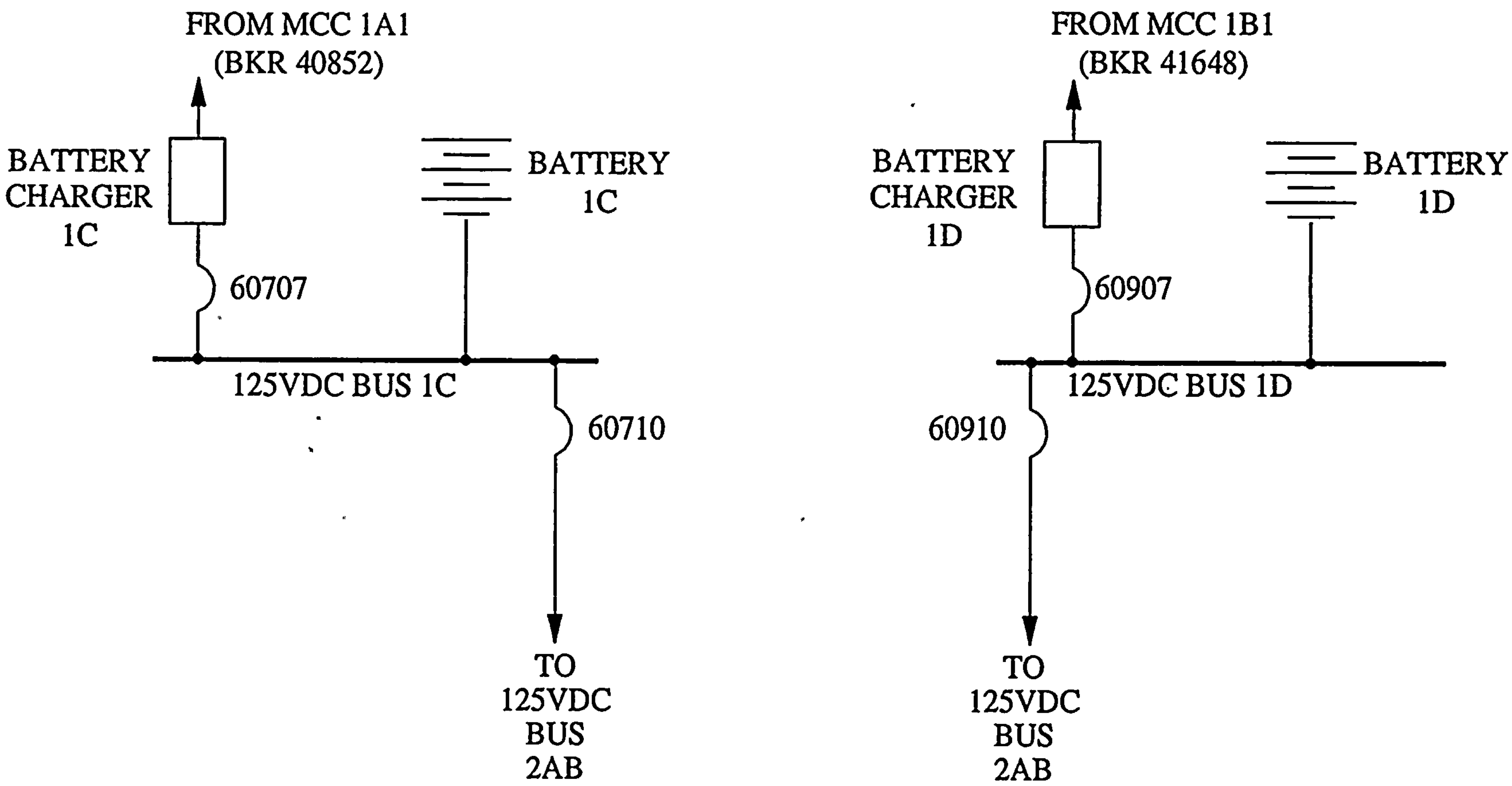

Figure B8.5 St. Lucie Unit 1 Non-Safety 125VDC

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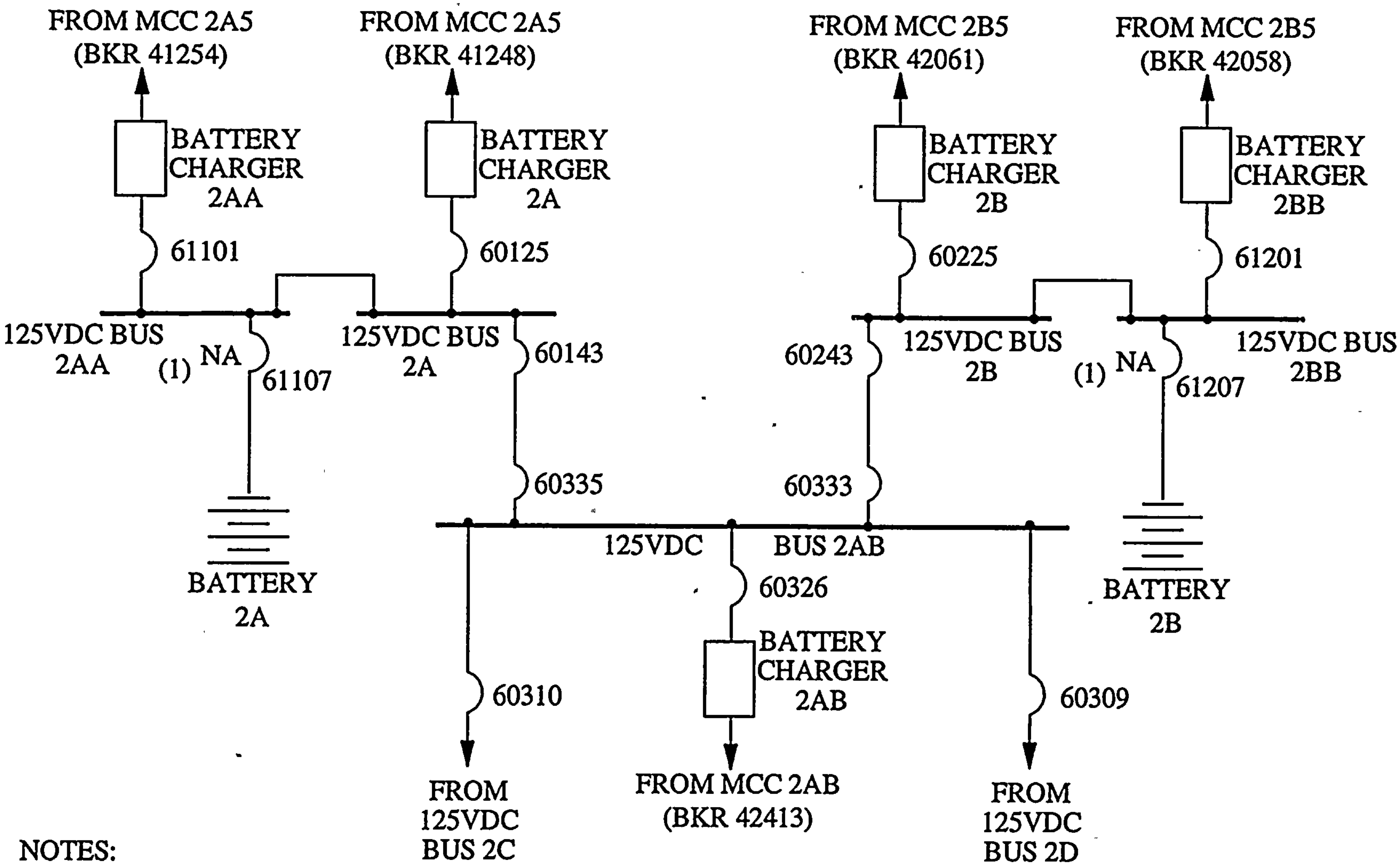

Figure B8.6 St. Lucie Unit 2 Vital 125VDC

NOTES:

(1) NA DENOTES NON-AUTOMATIC BREAKER

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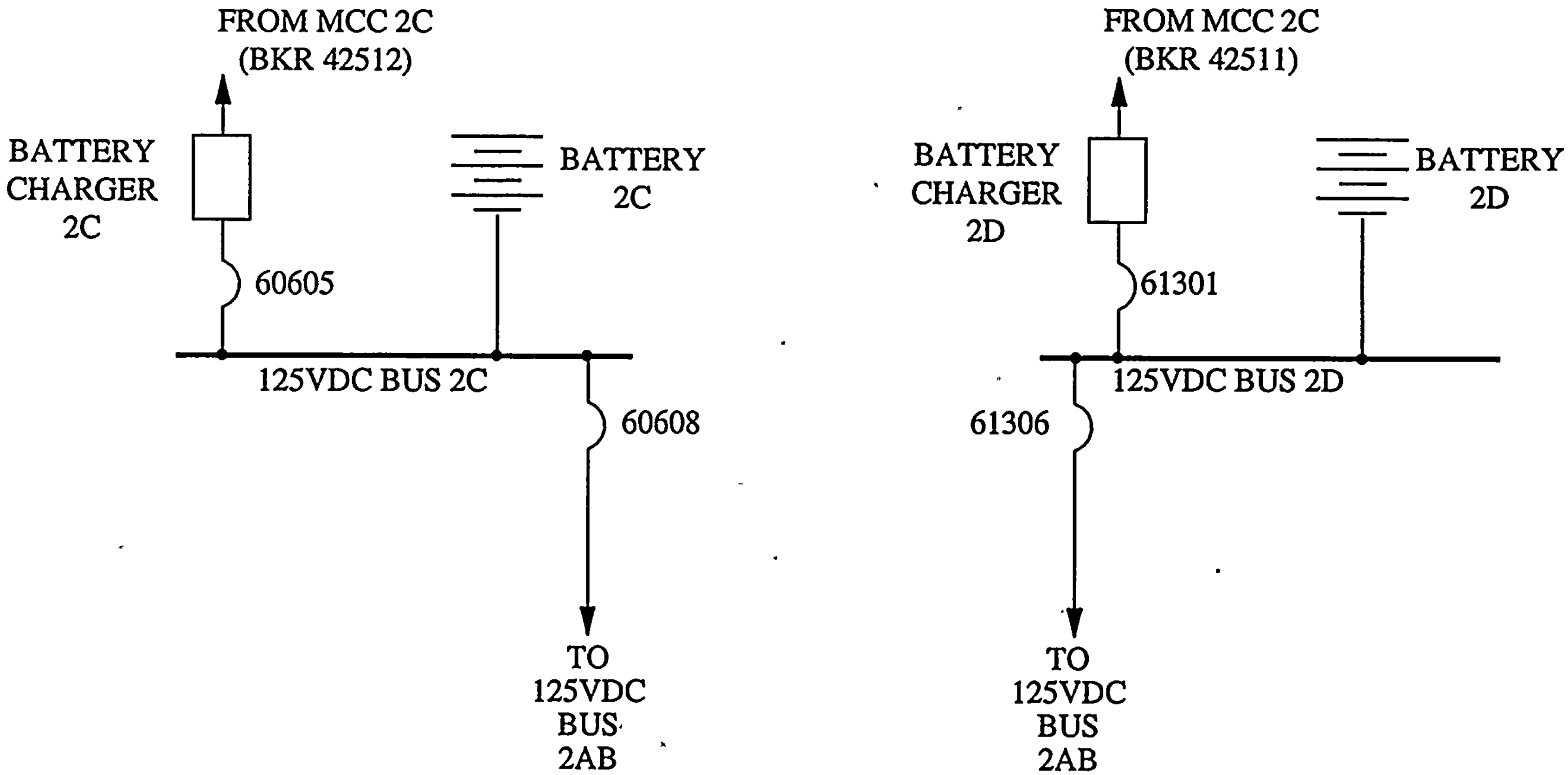
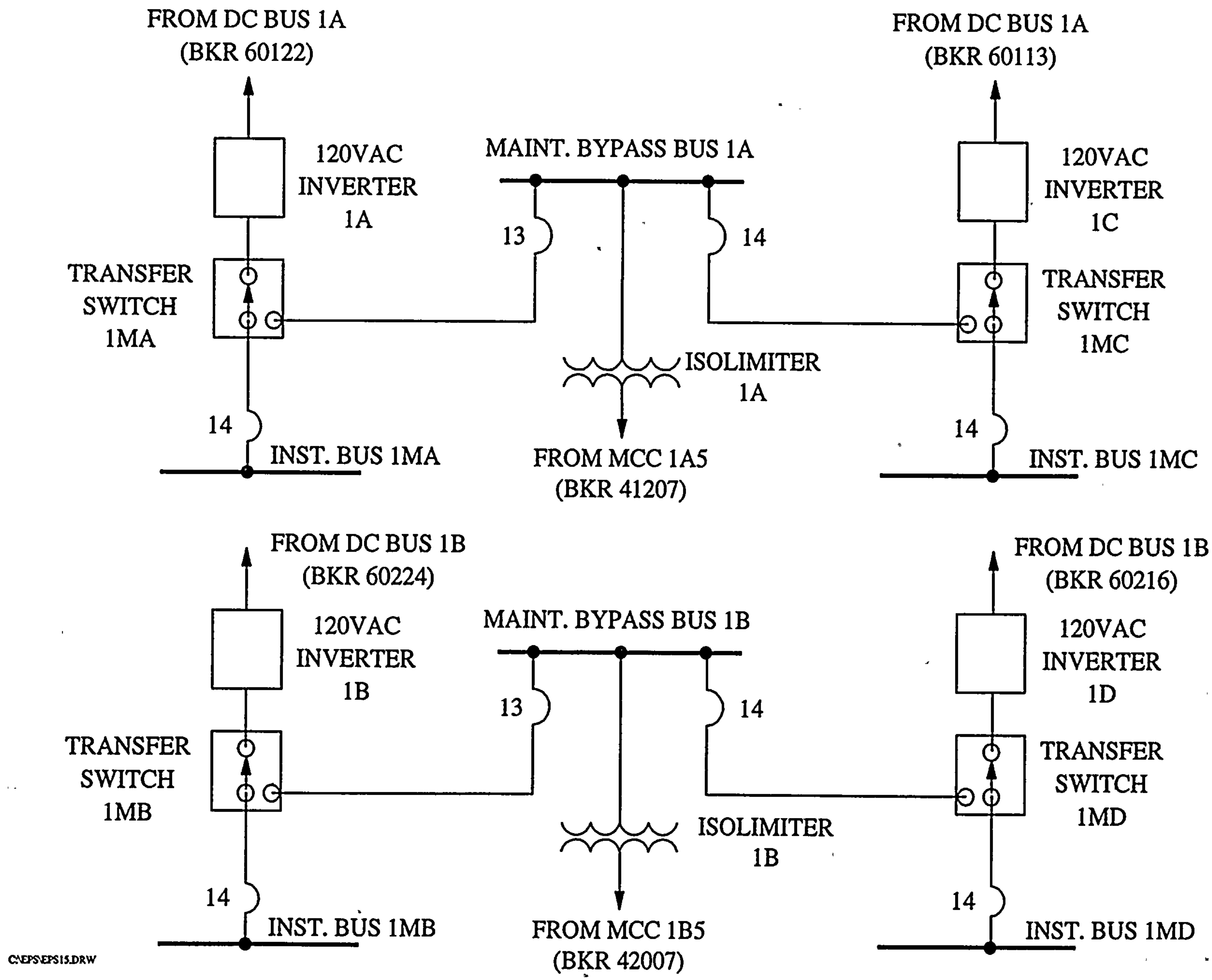


Figure B8.7 St. Lucie Unit 2 Non-Safety 125VDC

Figure B8.8 St. Lucie Unit 1 120V Vital Instrument AC



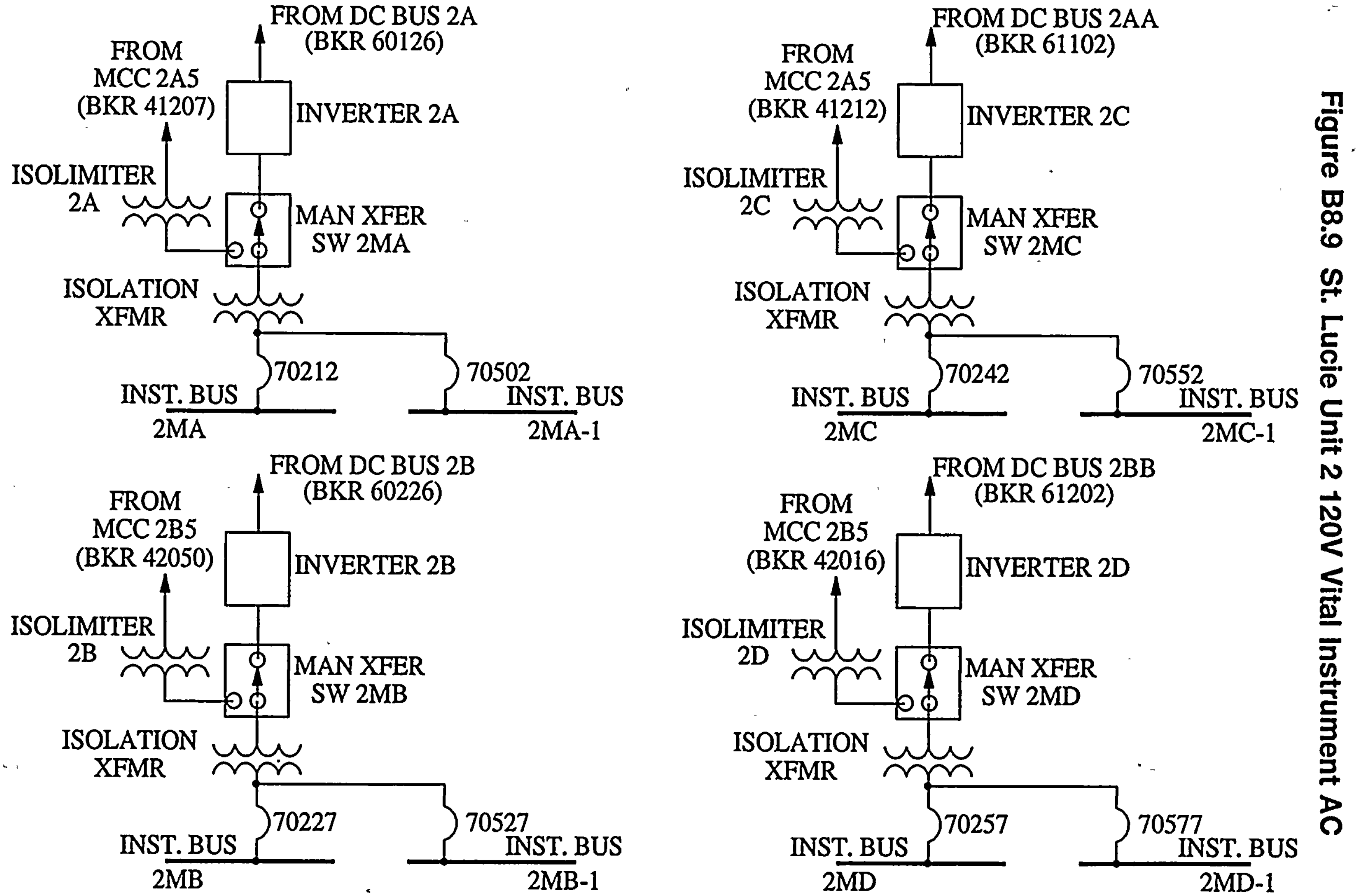


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Figure B8.9 St. Lucie Unit 2 120V Vital Instrument AC an inverter (1A, 1B, 1C, and 1D) powered from one of the two vital 125VDC buses (1A and 1B). To permit maintenance of any inverter without de-energizing the corresponding instrument bus, two redundant maintenance bypass buses (1A and 1B) powered by isolimiters (1A and 1B) through "make before break" transfer switches are provided. Breaker interlocks are provided to prevent simultaneous connection of more than one instrument bus to a maintenance bypass bus.

Unit 2 (See Figure B8.9)

Four pairs of 120VAC single phase instrument buses (2MA, 2MA-1, 2MB, 2MB-1, 2MC, 2MC-1, 2MD and 2MD-1) provide uninterruptible power to Engineered Safety Features Actuation (ESFAS) and Reactor Protection System (RPS) instrumentation. Each pair of instrument buses is supplied from an inverter (2A, 2B, 2C, and 2D) connected to one of the vital 125VDC buses. To permit maintenance without disabling the corresponding instrument bus, maintenance bypass transformers and voltage regulators are provided for each inverter system.

EMERGENCY DIESEL GENERATORS

The standby AC power supply consists of two emergency diesel generator (EDG) sets (1A and 1B, 2A and 2B), their associated air starting and fuel supply systems, and automatic control circuitry. The EDG's supply power to those electrical loads which are required to achieve safe shutdown of the unit or to mitigate the consequences of a LOCA in the event of a coincident loss of the normal AC power supply. Each EDG consists of two diesel engines mounted in tandem with a 3500 kw (3800 kw) generator coupled directly between the engines.

Each engine has a self-contained cooling system which consists of a forced circulation cooling water system which cools the engine directly and an air cooled radiator system which removes heat from the cooling water. The cooling system requires no external source of power and does not depend on any plant cooling system.

The engines of each EDG have a self-contained lube oil system consisting of a lube oil sump located at the base of the engine, an engine driven lube oil pump, piping, and a heat exchanger. The lube oil heat exchanger is served by the EDG cooling water system and thus no external source of power or other plant type systems are required.

Each EDG has an independent air starting system. Each EDG is provided with air receivers which have sufficient air to start a cold EDG five (5) times. The air starting system does not depend on normal plant electrical power except for 125VDC control power.

The diesel generator fuel oil system is used to transfer diesel fuel oil from the onsite storage tanks to the day tanks which supply the EDG's. Two subsystems, one for each EDG, are provided. Each subsystem consists of a diesel oil storage tank, transfer pump, two day tanks, and associated piping and valves. The day tanks have sufficient capacity for 75 minutes of EDG operation on Unit 1 and a minimum of 2 hours of operation on Unit 2. The diesel oil storage tanks allow for operation at post-accident power output for four days on Unit 1 and seven days on Unit 2. The fuel oil transfer pumps transfer oil from the oil storage tanks to the day tanks. Electrical power required for operation of each subsystem is provided by the associated EDG.

B8.3 Success Criteria

The Electric Power System must maintain the minimum required AC and DC buses energized to support front line system operation. The specific requirements are dependent on the postulated scenario.

An EDG must start upon an SI and/or loss of voltage signal and, with a loss of bus voltage, automatically close its output breaker.

B8.4 Operation

B8.4.1 Normal Operation (Steady-State 100% Power Operation Assumed) (See Figures B8.10 and B8.11)

240kV SYSTEM

Both the East and West 240kV operating buses are energized via the output from the Unit 1 (2) main transformers. The outgoing transmission lines to the grid are in turn energized from the switchyard. The Unit 1 (2) startup transformers are energized via the switchyard and are in standby.

22kV SYSTEM

The main generator is on line supplying power to vital 4.16kV buses 1A3 (2A3) and 1B3 (2B3) through 4.16kV buses 1A2 (2A2) and 1B2 (2B2) via the auxiliary transformers and to the 240kV switchyard via the main transformers.

6.9kV SYSTEM

6.9kV buses 1A1 (2A1) and 1B1 (2B1) are powered from auxiliary transformers 1A (2A) and 1B (2B).

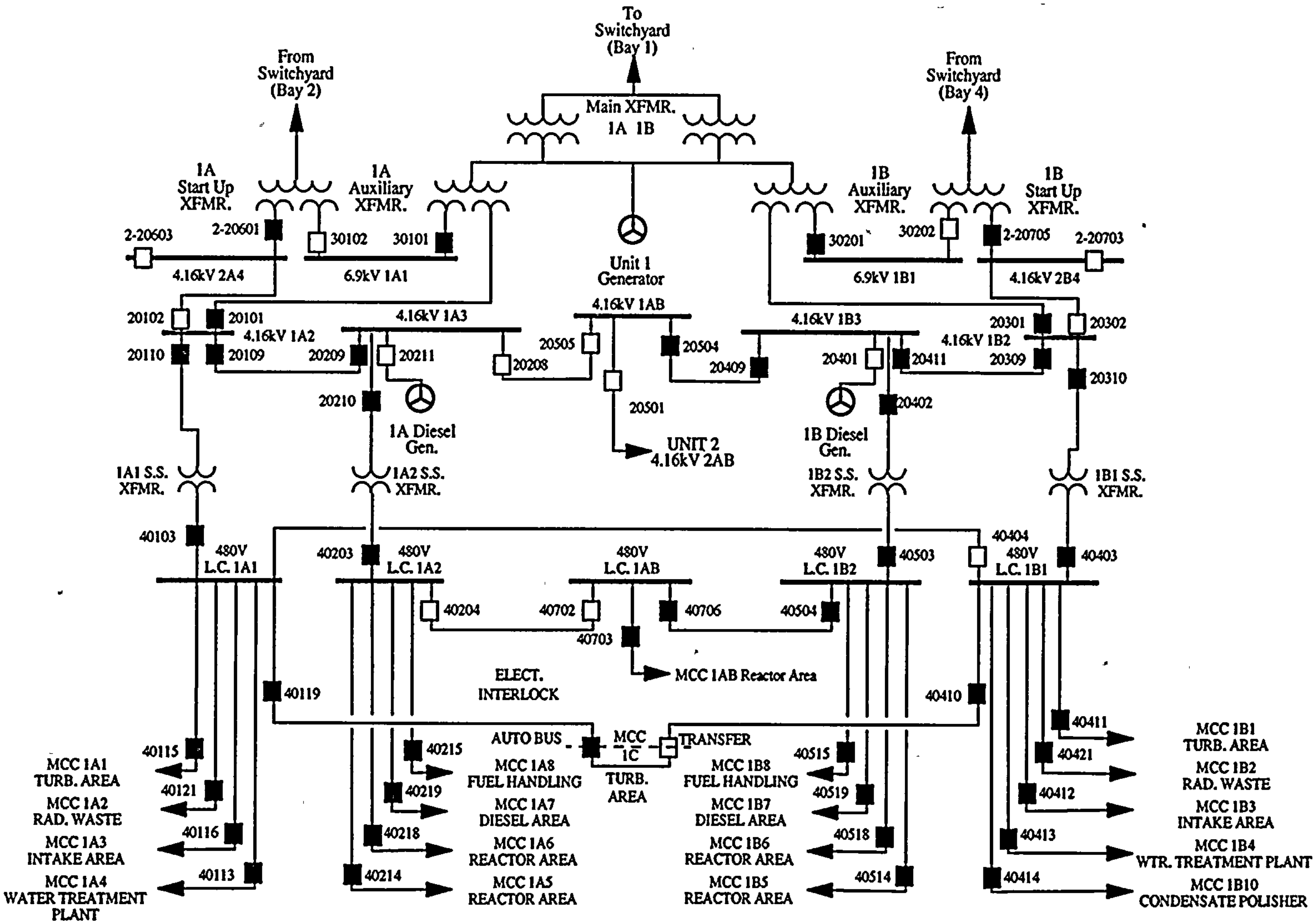
4.16kV SYSTEM

Vital 4.16kV buses 1A3 (2A3) and 1B3 (2B3) are powered from auxiliary transformers 1A (2A) and 1B (2B) via non-safety related 4.16kV buses 1A2 (2A2) and 1B2 (2B2). Swing 4.16kV bus 1AB (2AB) is aligned to either the 1A3 (2A3) or 1B3 (2B3) bus.

480V SYSTEM

Vital 480V load centers (LC's) 1A2 and 1B2 are powered from 4.16kV buses 1A3 and 1B3. Vital 480V LC's 2A2, 2A5, 2B2, and 2B5 are powered from 4.16kV buses 2A3 and 2B3. Swing 480V LC 1AB (2AB) is powered from either 480V LC 1A2 (2A2) or 1B2 (2B2). Vital MCC's are aligned to their associated LC as shown in Figures B8.10 and B8.11. Non-safety related LC's 1A1 (2A1) and 1B1 (2B1) are powered from 4.16kV buses 1A2 (2A2) and 1B2 (2B2).

Figure B8.10 St. Lucie Unit 1 Electrical Distribution System Normal Operation



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

■ - DENOTES CLOSED BREAKER

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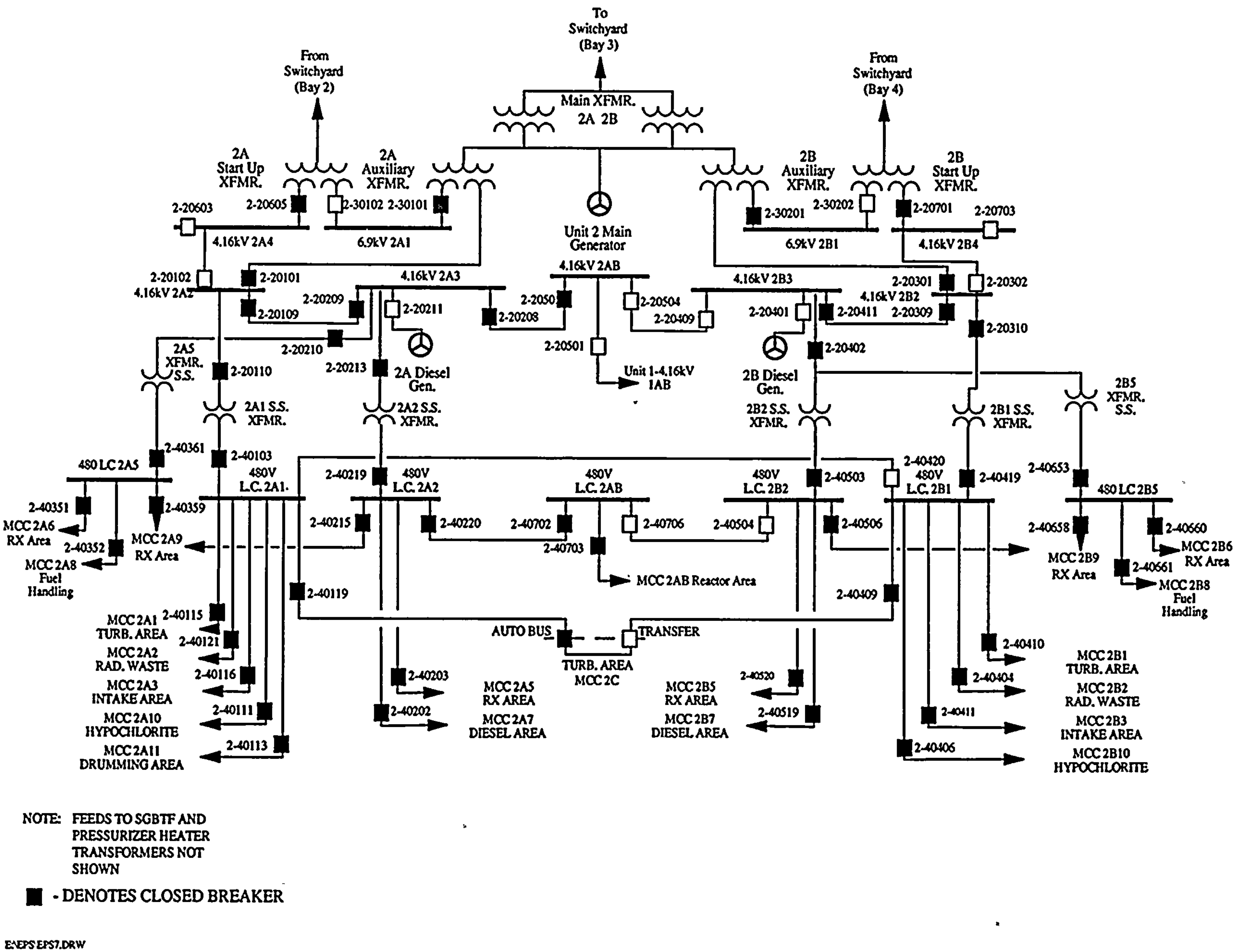
Figure B8.11 St. Lucie Unit 2 Electrical Distribution System Normal Operation



120/208 VOLT SYSTEMS

Power and lighting panels are fed from their associated 480V MCC's.

125VDC SYSTEM

The vital battery chargers carry the vital DC loads and maintain a float charge on the vital batteries. The non-vital battery chargers carry the non-vital DC loads and maintain a float charge on the non-vital batteries.

120VAC VITAL INSTRUMENT AC

Each vital 120VAC inverter is powered from its associated 125VDC bus and powers its associated instrument bus.

EMERGENCY DIESEL GENERATORS

The EDG's are in standby (not running) with their output breakers open unless they are being run for surveillance.

B8.4.2 Accident Operation (See Figures B8.12, B8.13, B8.14, B8.15)

240kV SYSTEM

A safety injection signal results in a generator lockout (trip) and in turn loss of the unit's output to the switchyard. The switchyard operating buses remain energized via the 240kV transmission lines unless a system disturbance results in a loss of grid condition.

If, while the unit is on line, the switchyard were to be isolated from the grid (loss of all transmission lines), the unit would trip due to the loss of load and the inability to successfully runback the generator output to sustain only plant auxiliary loads. This situation would result in loss of power to all startup transformers.

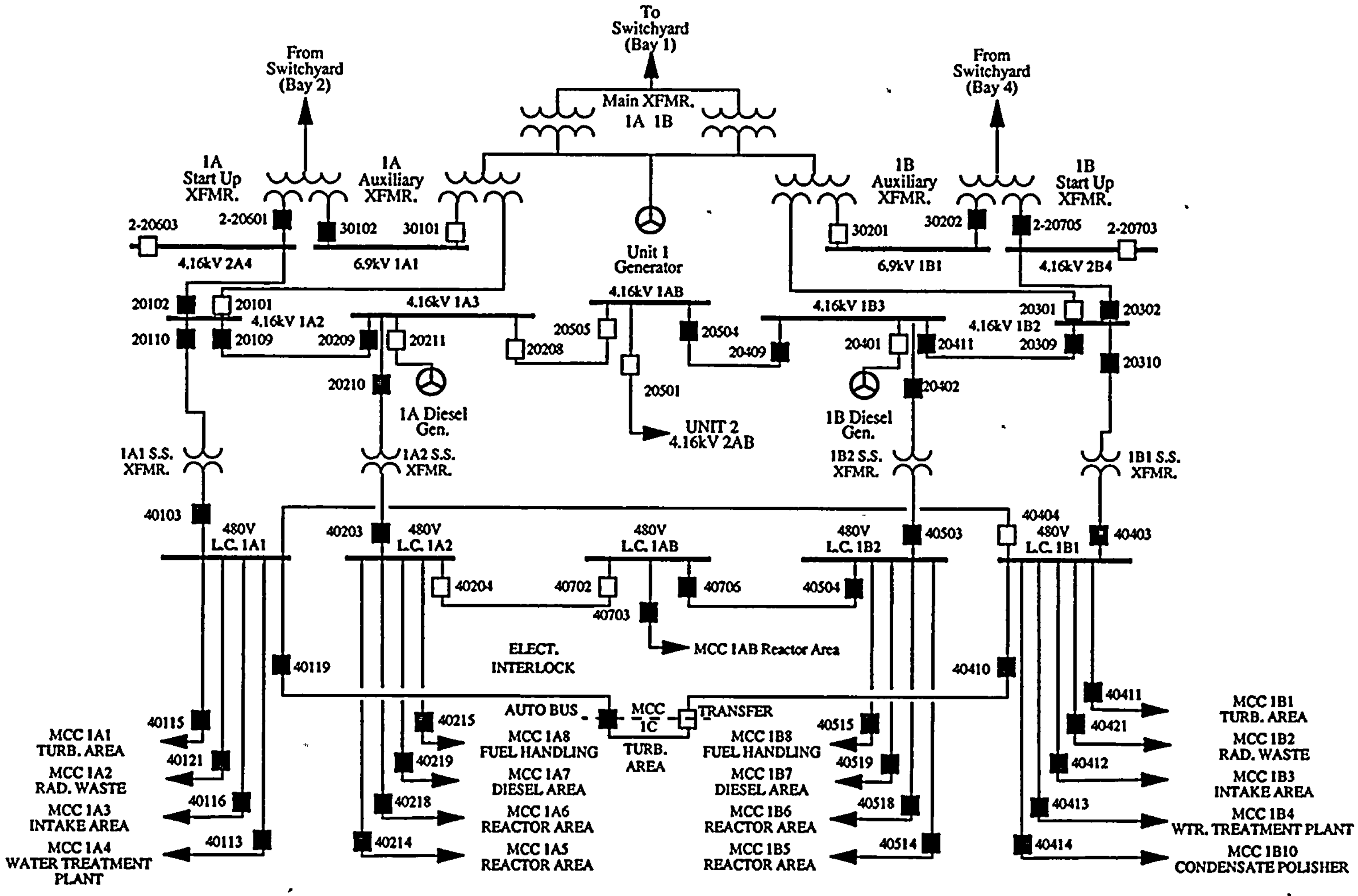
22kV SYSTEM

A generator trip de-energizes the iso-phase bus, main transformers, and auxiliary transformers. 4.16kV buses 1A2 (2A2) and 1B2 (2B2), and 6.9 kV buses 1A1 (2A1) and 1B1 (2B1) transfer to the startup transformers (if available).

6.9kV SYSTEM

Upon receipt of a main generator lockout signal, supply breakers 30101 (2-30101) and 30201 (2-30201) to buses 1A1 (2A1) and 1B1 (2B1) from auxiliary transformers 1A (2A) and 1B (2B) automatically open, and supply breakers 30102 (2-30102) and 30202 (2-30202) from startup transformers 1A (2A) and 1B (2B) automatically close.

Figure B8.12 St. Lucie Unit 1 Electrical Distribution System Accident Operation Offsite Power Available



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

■ - DENOTES CLOSED BREAKER

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Figure B8.13 St. Lucie Unit 2 Electrical Distribution System Accident Operation Offsite Power Available

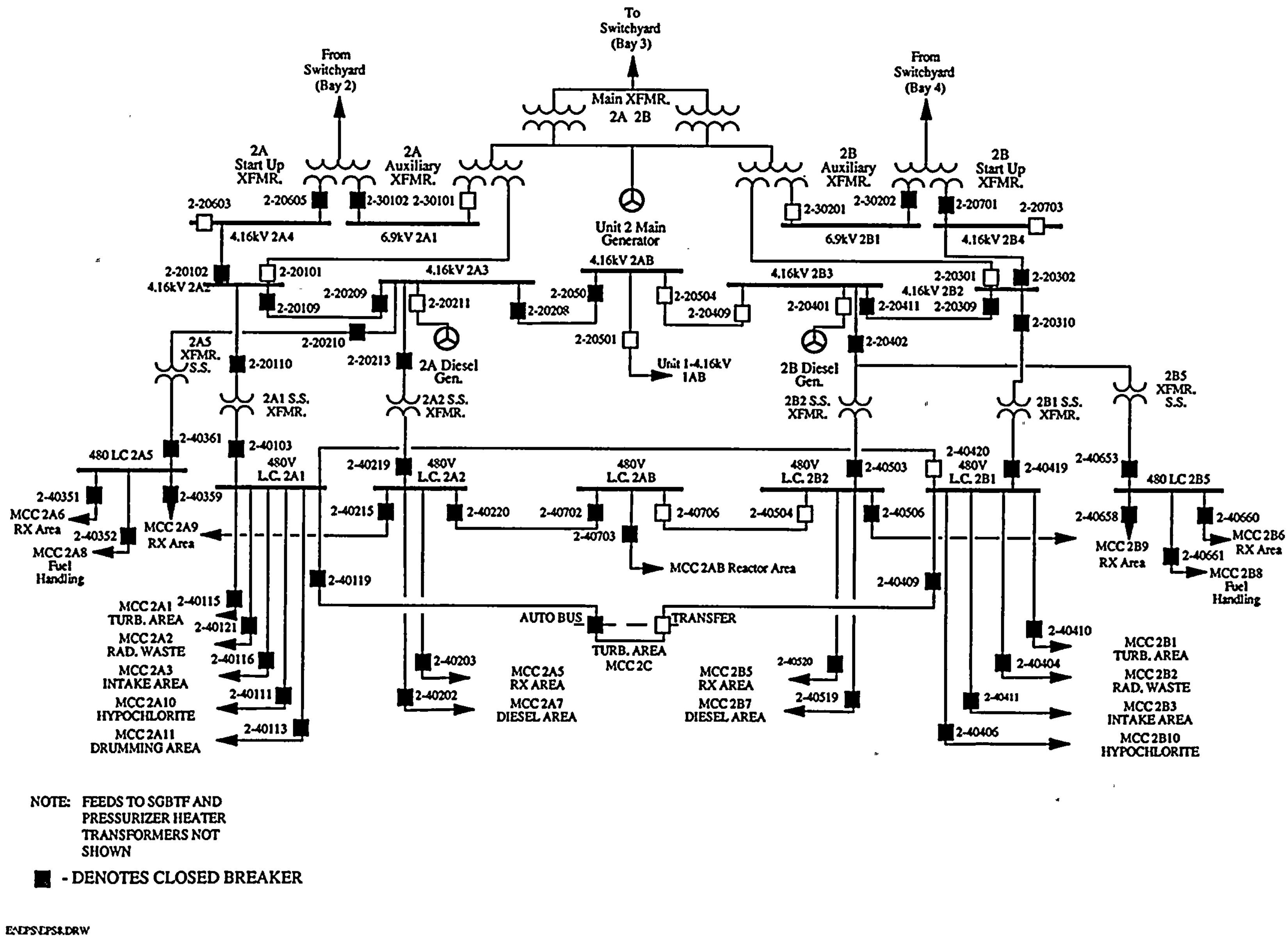
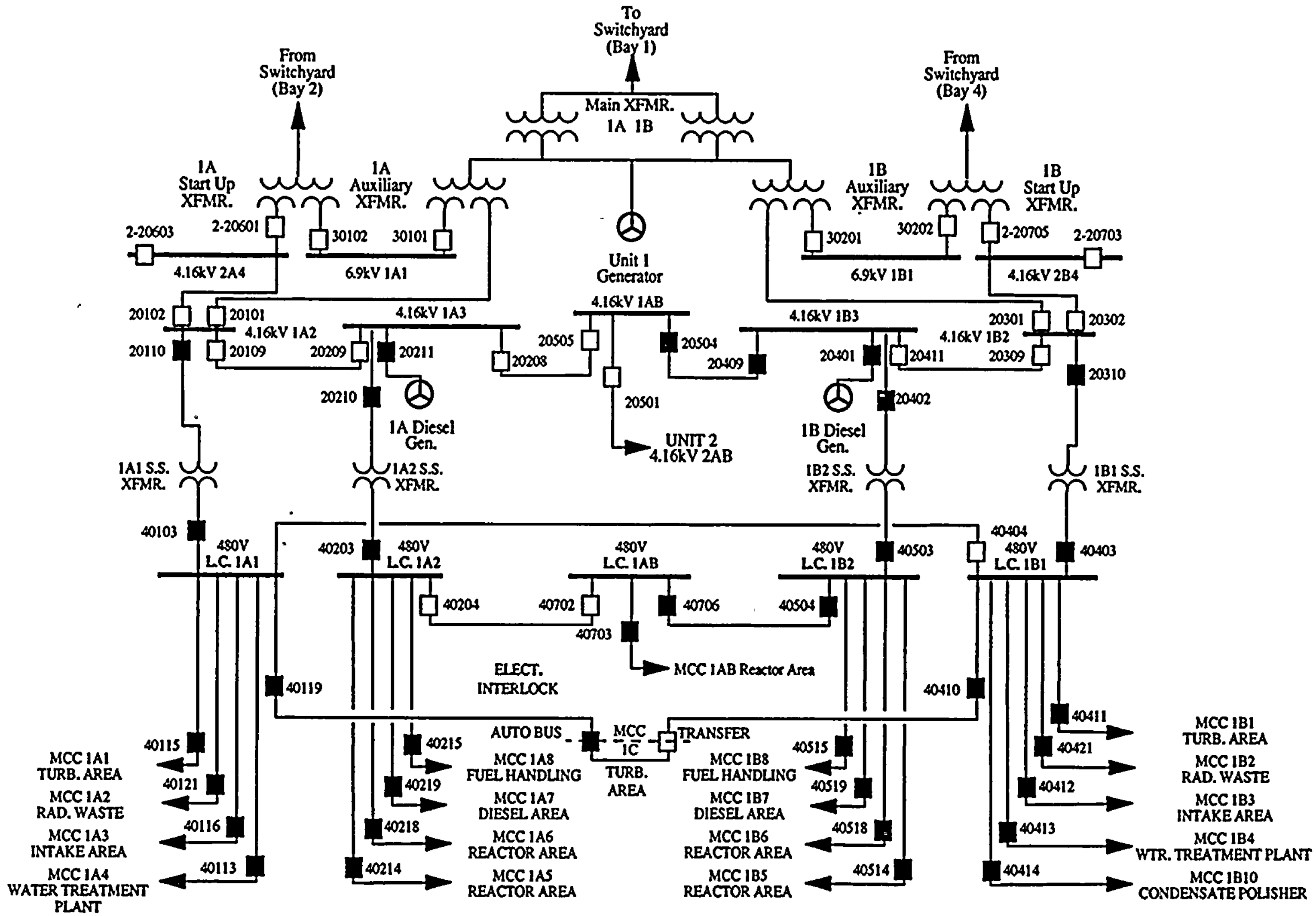


Figure B8.14 St. Lucie Unit 1 Electrical Distribution System Accident Operation Loss of Offsite Power



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

■ - DENOTES CLOSED BREAKER

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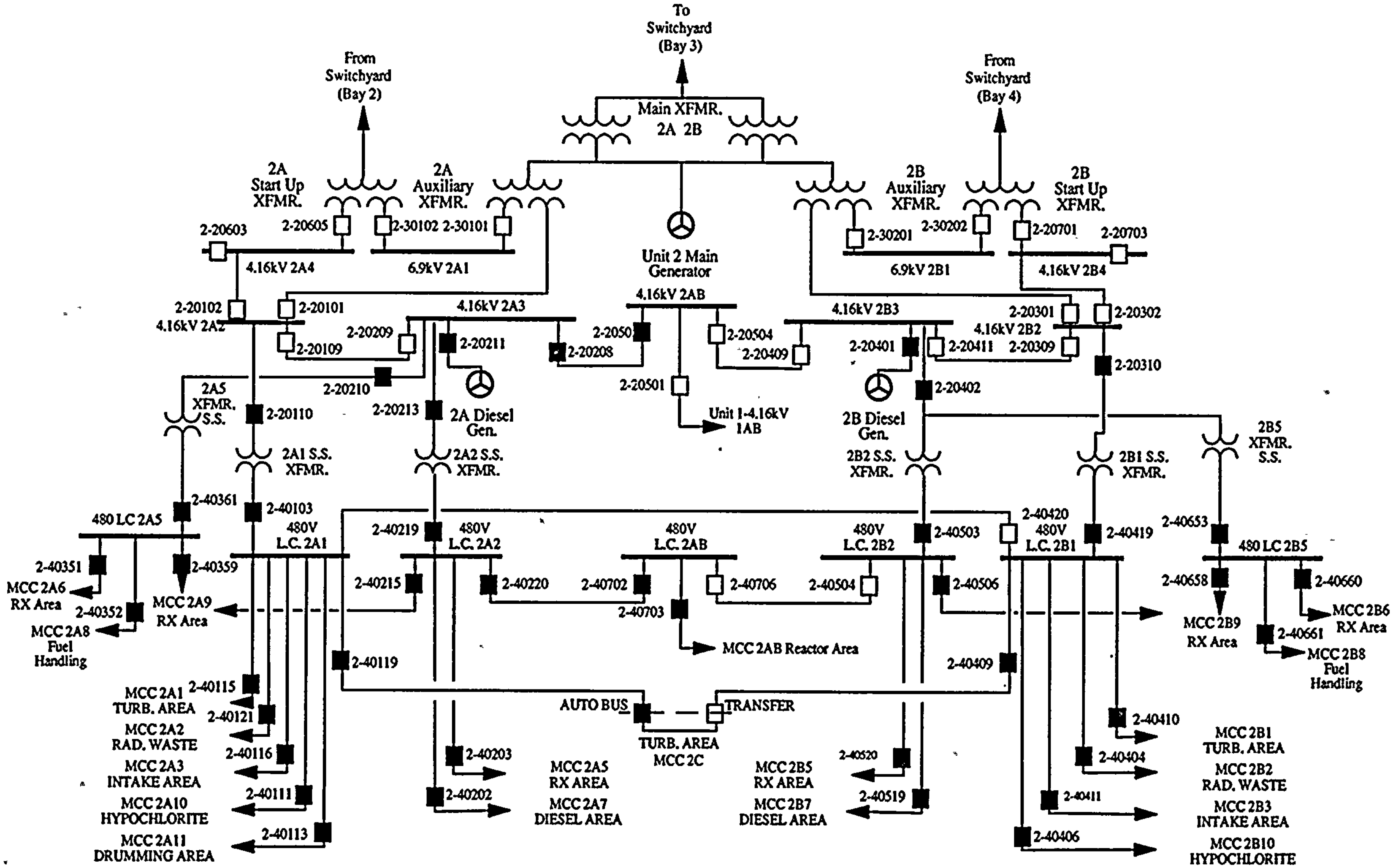
Figure B8.15 St. Lucie Unit 2 Electrical Distribution System Accident Operation Loss of Offsite Power



NOTE: FEEDS TO SGBTF AND PRESSURIZER HEATER TRANSFORMERS NOT SHOWN

■ - DENOTES CLOSED BREAKER

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4.16kV SYSTEM/EMERGENCY DIESEL GENERATORS

Upon receipt of a main generator lockout signal, supply breakers 20101 (2-20101) and 20301 (2-20301) to buses 1A2 (2A2) and 1B2 (2B2) from auxiliary transformers 1A (2A) and 1B (2B) automatically open and supply breakers 20102 (2-20102) and 20202 (2-20202) from startup transformers 1A (2A) and 1B (2B) automatically close if the startup transformer is available.

EDG 1A (2A) starts upon receipt of a safety injection signal and/or an undervoltage signal on 4.16kV bus 1A3 (2A3). EDG 1B (2B) starts upon receipt of a safety injection signal and/or an undervoltage signal on 4.16kV bus 1B3 (2B3). If the EDG's were started in response to a safety injection signal with no undervoltage condition, the output breakers remain open and the EDG's continue to run until manually shutdown. If there is a loss of offsite power, bus tie breakers 20109 (2-20109) and 20209 (2-20209), and 20309 (2-20309) and 20411 (2-20411) open, all load breakers (except for feeds to LC's) on 4.16kV buses 1A3 (2A3) and 1B3 (2B3) open, and the EDG output breakers close thus energizing the vital 4.16kV buses. Required loads then sequence on in their specified load blocks. 4.16kV bus 1AB (2AB) operates as an extension of the 4.16kV bus to which it is aligned (1A3 (2A3) or 1B3 (2B3)). Re-alignment of 4.16kV bus 1AB (2AB) is accomplished manually.

480V SYSTEM

Following a loss of offsite power, vital LC's 1A2 (2A2), 1B2 (2B2), 2A5, and 2B5 are powered from the EDG's via their associated 4.16kV buses. These LC's in turn power vital MCC's. LC 1AB (2AB) remains aligned to its pre-accident source (LC 1A2 (2A2) or LC 1B2 (2B2)) unless it is manually re-aligned. Non-vital load centers 1A1 (2A1) and 1B1 (2B1) are powered from their associated 4.16kV buses if available.

120/208 VOLT SYSTEM

120/208V panels are fed from their associated MCC's.

125VDC SYSTEM

Upon loss of AC power to the vital battery chargers, DC bus loads are automatically supplied from the vital batteries (1A (2A) and 1B (2B)). The vital battery chargers are re-energized from the EDG's and then power the DC bus loads and recharge the batteries.

Upon loss of AC power to the non-vital battery chargers, non-vital DC bus loads are supplied by the non-vital batteries (1C (2C) and 1D (2D)). The non-vital chargers will be re-energized only if their associated non-vital MCC's are re-energized.

120V VITAL INSTRUMENT AC

Operation of the vital instrument 120VAC system does not change during a safety injection with or without a loss of offsite power since the 120VAC instrument buses are supplied by inverters which are powered from vital 125VDC buses. Manually aligned alternate feeds are provided to the 120VAC instrument buses from vital MCC's through step-down transformers in case of an inverter failure.

B9 ENGINEERED SAFETY FEATURES ACTUATION AND REACTOR PROTECTION SYSTEM

B9.1 Function

The function of the Reactor Protection System (RPS) is to protect the fuel, fuel cladding and the reactor coolant pressure boundary during anticipated operational occurrences. RPS monitors selected NSSS parameters and initiates a reactor trip when unsafe operating conditions are detected.

The Engineered Safeguards Features Actuation System (ESFAS) functions to mitigate the consequences of an accident by initiating systems designed to provide core cooling, establish containment isolation, and protect containment integrity.

The function of the Auxiliary Feedwater Actuation System (AFAS) is to initiate equipment designed to provide an emergency source of feedwater to the steam generators when low-low steam generator level is detected and the pressure boundary is determined to be intact.

B9.2 Configuration

<u>RPS</u>

RPS is a four channel, two train protection system designed to interrupt power to the control element drive mechanisms when the coincidence logic of any one of the input parameters is satisfied. The RPS trip logic consists of the following basic parts:

1) NSSS parameter measurement channels
2) Bistable and Auxiliary trip units
3) Coincidence logic matrices
4) CEDM power trip paths

Each of these parts has its particular function in the overall system. The measurement channels supply information to the RPS on the selected NSSS conditions. The bistable trip units compare this information to reference setpoints and provide trip signals to the Protection System Logic if the measured NSSS condition deviates from normal. The coincidence logic matrices provide trip signals to the reactor trip circuit breakers when coincident trip signals from the bistable and auxiliary trip units are present.

With the exception of measurement channel sensors and signal conditioners, reactor trip switchgear, and some auxiliary nuclear instrumentation equipment, the RPS is housed in a four bay cabinet in the control room.

As shown in Figure B9.1, the RPS consists of four protection channels - A, B, C and D. Each of these protection channels monitors the following NSSS parameters:

1) Power Level;
2) Rate of Change of Power;
3) Reactor Coolant Flow;
4) Steam Generators Low Water Level;
5) Steam Generators Steam Pressure;
6) Pressurizer Pressure;
7) Thermal Margin/Low Pressure;
8) Loss of Load;
9) Containment Pressure;
10) Local Power Density;
11) Reactor Coolant Pump - Component Cooling Water Flow (Unit 2 only)

The signal output from each measurement channel is fed to the input of an auxiliary or bistable trip unit. Forty trip units, 10 [11 parameters and 44 trip units in Unit 2] for each of the four protective channels, monitor all NSSS parameters providing an input to the Reactor Protection System. The four measurement channels monitoring each NSSS parameter are completely independent and isolated from each other. Each protection channel in the protection system cabinet assembly is housed in separate cabinet bays to provide channel isolation.

The trip units have their output contacts arranged in six logic matrices, identified as AB, AC, AD, BC, BD and CD to represent all possible two-out-of-four combinations of trip signals. Each logic matrix, when tripped, trips four matrix relays, which in turn provide trip signals to each of four trip paths. Each trip path opens two associated reactor trip circuit breakers to deenergize the magnetic coils that hold the control element assemblies (CEA's).

The outputs of the CEDM motor generator sets are paralleled to prevent CEA release on a single power supply failure and to permit system testing during reactor operation. Manual trip action bypasses all logic and trips the breakers directly by interrupting the DC control power to the trip switchgear.

<u>ESFAS</u>

ESFAS is also a four channel, two train protection system with selective two-out-of-four actuation logic. ESFAS is designed to actuate safety related equipment associated with the following signals:

1) Safety Injection Actuation Signal (SIAS)
2) Recirculation Actuation Signal (RAS)
3) Containment Spray Actuation Signal (CSAS)
4) Main Steam Isolation Signal (MSIS)
5) Diverse Scram System (DSS)
6) Containment Isolation Actuation Signal (CIAS)

The ESFAS circuitry includes redundant initiating variable measurement devices, trip bistables, coincidence logic matrices, actuation modules, output relays, manual and automatic test circuitry, and separated channel cabinets for housing the components.

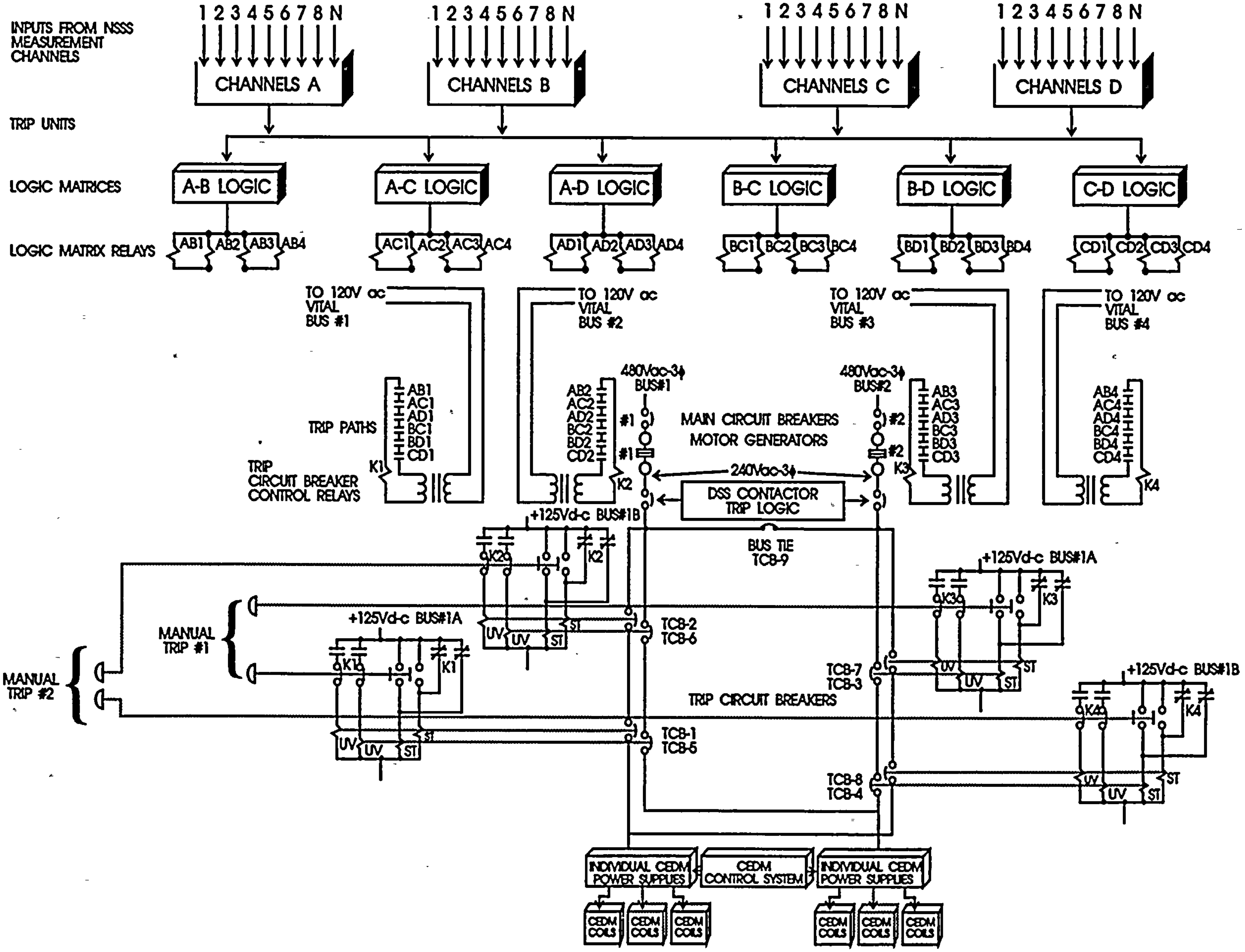


Figure B9.1 Simplified Functional Diagram of the Reactor Protection System

ESFAS is arranged into four measurement channels (MA, MB, MC, and MD) and two actuation trains (SA and SB). Two separate actuation cabinets are provided for each of the actuation trains. The four measurement channel cabinets are located between the two actuation cabinets. The engineered safeguards cabinets are located in the control room.

The variables measured by the ESFAS are containment pressure, containment radiation, refueling water tank level, pressurizer pressure, and steam generator pressure. Each detector develops an analog signal which is fed into signal conditioning circuits in the four measurement channels. In the case of containment radiation, the signal is developed from an interface module within the radiation monitoring system.

Bistable modules in the measurement cabinet accept the signal from the signal conditioning circuits and provide an output to indicators on the measurement channel cabinet. The output of the bistable is split and fed into isolation modules which are the entry points for actuation trains SA and SB. These output signals are then fed into the 2-out-of-4 matrices of the actuation modules. The actuation modules are arranged to supply the SIAS, CIAS, CSAS, MSIS, RAS and DSS functions. Note, however, that the DSS (Diverse Scram System) is not an ESFAS signal but a diverse reactor trip designed to protect against an ATWS.

An auto test insertion (ATI) panel is located in the engineered safeguards cabinets. The ATI automatically generates signals that test the bistable trip modules without actually causing a trip.

AFAS

The AFAS is a four-channel safety grade system utilizing two-out-of-four coincidence logic. When AFW is required, the AFAS circuitry will generate an AFAS-1 signal to supply AFW to S/G-A and/or an AFAS-2 signal to supply AFW to S/G-B.

Each S/G is monitored for water level, steam generator pressure and feedwater header pressure. The pressure inputs are used in the fault detection circuitry to prevent feeding a ruptured steam generator. The AFAS actuation setpoint is lower than the Reactor Protection System (RPS) reactor trip setpoint and incorporates a time delay to prevent unnecessary actuation.

The AFAS actuation circuits are similar to the RPS circuits described above. One significant difference is the use of both cycling and latching actuation relays. Cycling relay contacts are used in the valve control circuits to isolate AFW once generator level has been restored. Latching relay contacts are used to keep the AFW pumps running throughout an entire event while steam is continuously dumped and generator level is periodically restored.

B9.3 Success Criteria

RPS must initiate a reactor trip, by removing power to both CEDM busses, whenever a condition is detected which could be detrimental to the fuel, fuel cladding or RCS pressure boundary integrity.

ESFAS must initiate safety injection when low pressurizer pressure or high containment pressure is detected.

ESFAS must initiate containment isolation when high containment radiation or high containment pressure is detected. Containment isolation must also be initiated when safety injection is initiated.

ESFAS must initiate containment spray when high-high containment pressure is detected and safety injection has been initiated.

ESFAS must initiate main steam isolation when low steam generator pressure is detected. For Unit 2 only, MSIS is also initiated when high containment pressure is detected.

ESFAS must initiate recirculation when low refueling water tank level is detected.

DSS must trip the control element drive motor generator set output contactor when high-high pressurizer pressure is detected.

AFAS must initiate auxiliary feedwater to either or both steam generators when low-low level is detected and the generator pressure boundary is determined to be intact.

B9.4 Operation

B9.4.1 Normal Operation

The RPS trip unit relays are normally energized. Contacts from the trip unit relays are normally closed to hold the matrix relays energized. Contacts from the matrix relays are normally closed to hold the trip path relays energized. Normally closed contacts from the trip path relays hold the reactor trip switchgear breaker undervoltage coils energized, and normally open trip path relay contacts prevent energization of the breaker shunt trip coils. The CEDM gripper coils are powered via the normally closed reactor trip switchgear breakers, which provide a selective two-out-of-four trip logic.

The ESFAS actuation logic consists of four stages: the bistable modules, the isolation modules, the 2-out-of-4 actuation modules and finally the actuation relays. Where the same input parameter is used in multiple actuation logic circuits a separate bistable is used for each function. The SIAS, CIAS, and MSIS actuation relays in the ESFAS cabinets are all normally energized. The RAS and CSAS actuation relays are all normally deenergized.

The AFAS actuation logic is similar to RPS. The AFAS bistable relays are normally energized, and their contacts hold the matrix relays energized. Normally closed contacts from the matrix relays hold the initiation relays energized. The initiation relay contacts hold the interposing relays energized. Normally closed interposing relay contacts hold the lockout, cycling and latching relays energized.

B9.4.2 Accident Operation

When any RPS input parameter exceeds the trip unit bistable setpoint the three trip unit relays will all deenergize. This will open one contact in each of the three matrix relay circuits associated with the tripped channel. If a trip is experienced in one or more of the other channels for that same input parameter the remaining contact in at least one matrix relay circuit will open to deenergize the four matrix relays. Contacts from the four matrix relays open all four trip paths. The undervoltage coils of all eight reactor trip switchgear breakers will deenergize and the shunt trip coils will energize. All eight breakers open to drop all CEAs.

If pressurizer pressure exceeds the HI-HI DSS trip setpoint on two or more of the four channels, the SA and SB actuation relays will energize and open the output contactors of CEDM motor generator sets 1 and 2 respectively.

When any ESFAS input parameter exceeds the bistable setpoint associated with a given actuation function, trip logic signals are sent to all corresponding actuation modules in both SA and SB via the isolation modules. If a trip is experienced in one or more of the other three channels for that parameter and that function, a second trip signal is sent to all corresponding actuation modules. The actuation modules incorporate a two-out-of-four trip logic. When this trip logic is met the associated actuation relays will be deenergized (or energized for CSAS and RAS only).

The AFAS accident response is similar to RPS. The bistable, matrix, initiation, and interposing relays will deenergize. However, after the matrix relay contacts open a time delay relay holds the initiation relays energized for 225 seconds [210 seconds in Unit 2]. If generator level is restored during this delay time the system will automatically reset. An interposing relay contact deenergizes the lockout, cycling and latching relays. The lockout relay ensures the latching relays remain deenergized until AFAS is reset by the operator. The cycling relays will energize and deenergize as determined by the steam generator level bistable setpoint and hysteresis.

B10 HIGH PRESSURE SAFETY INJECTION SYSTEM

B10.1 Function

The High Pressure Safety Injection system (HPSI) is a component of the Emergency Core Cooling system (ECCS) which also includes the Safety Injection Tanks (SIT) and the Low Pressure Safety Injection system (LPSI). The function of the ECCS is to prevent significant alteration of core geometry, preclude fuel melting, limit the cladding metal-water reaction and remove the energy generated in the core during various postulated accident situations. The ECCS also functions to maintain the reactor subcritical by injecting borated water into the Reactor Coolant System (RCS) and to provide for long term cooling of the core by recirculating borated water from the containment sump. The HPSI system functions to automatically inject borated water from the Refueling Water Tank (RWT) into the RCS cold legs following a Safety Injection Actuation Signal (SIAS), to automatically recirculate water from the containment sump to the RCS cold legs following a Recirculation Actuation Signal (RAS) and, in Unit 2, to recirculate water from the sump to the RCS hot legs.

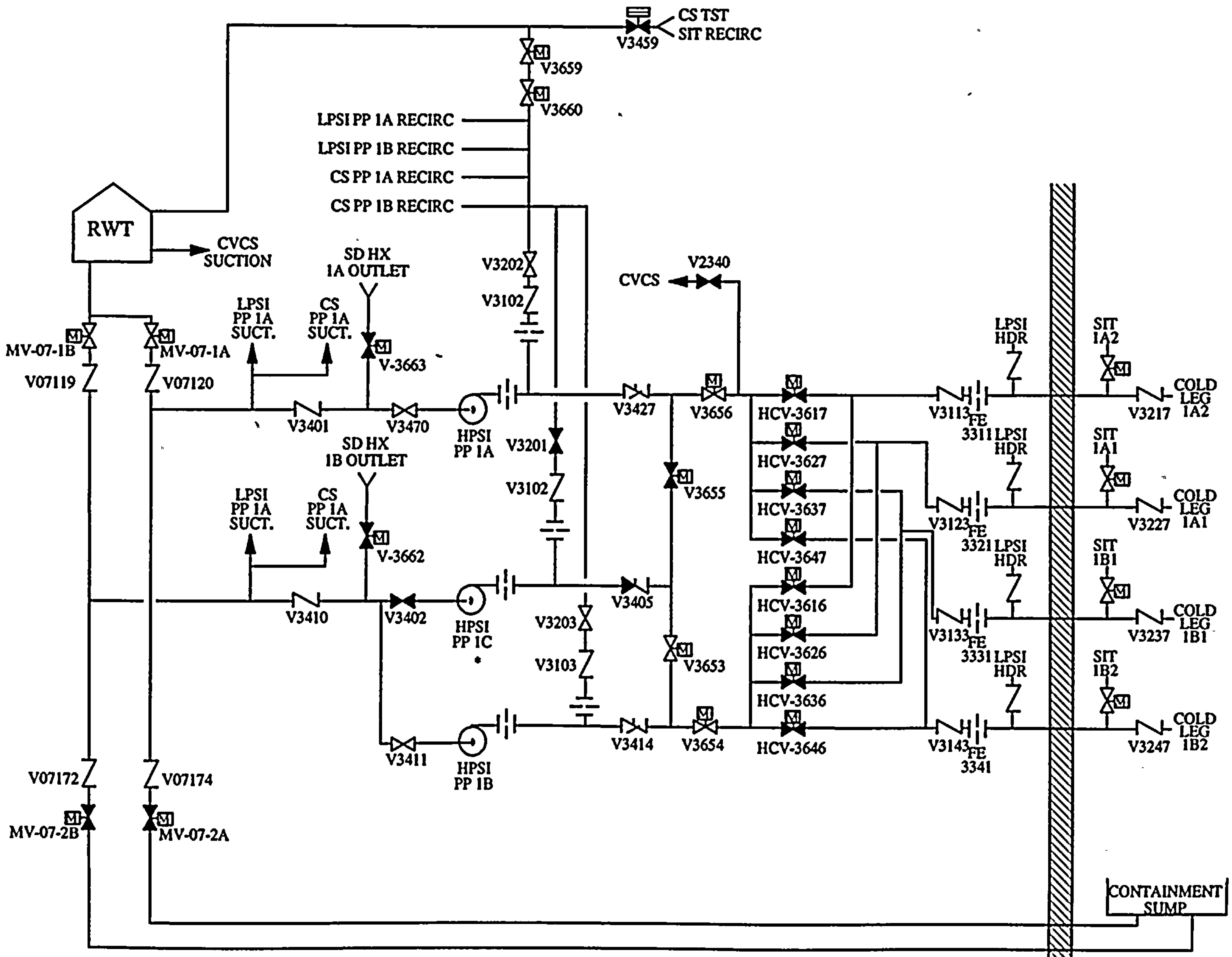
B10.2 Configuration

The standby configurations of the HPSI systems for Units 1 and 2 are shown on Figures B10.1 and B10.2. The system consists of two HPSI pumps (the 1C pump has been abandoned), a Refueling Water Tank, a containment sump and associated piping and valves. The HPSI system is arranged as two redundant trains, each with the capability to supply the minimum flow requirements during accident conditions.

While in standby and during the Post-Accident Injection phase, the pumps are aligned to take suction from the RWT. The RWT contains an inventory of borated water which is a common supply for the HPSI, LPSI and Containment Spray (CS) systems. The configurations of the HPSI system during the injection phase are shown in Figures B10.3 and B10.4. During the Post-Accident Recirculation phase, the HPSI pumps (as well as the CS and LPSI pumps if required) take suction from the containment sump. Following a Loss of Coolant Accident (LOCA), water from the RCS as well as the injected water from the RWT will accumulate in the sump. Once the RWT has been depleted, the suction of the HPSI pumps will automatically shift to the sump following a RAS. In both cases two suction lines are provided for redundancy, one for the train A ECCS pumps and one for the train B ECCS pumps. During Recirculation in Unit 1, a portion of the HPSI pump suction can also be provided from the outlet of the Containment Spray pumps that has passed through the Shutdown Cooling heat exchangers, thereby providing additional core cooling. The configurations of the HPSI system during recirculation phases are shown in Figures B10.5, B10.6, and B10.7.

Unit 1 was originally provided with three HPSI pumps. The 1C pump has been abandoned and is maintained isolated with the motor disconnected. Even though there are pump discharge cross tie valves, a cross tie valve is kept locked closed. Unit 2 has two pumps that cannot be cross connected at the discharge. Each pump has a portion of its discharge cooled by CCW and circulated through the pump seal.

Figure B10.1 Unit 1 HPSI System Standby Configuration



* IC HPSI PUMP IS NO LONGER FUNCTIONAL

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Figure B10.2 Unit 2 HPSI System Standby Configuration

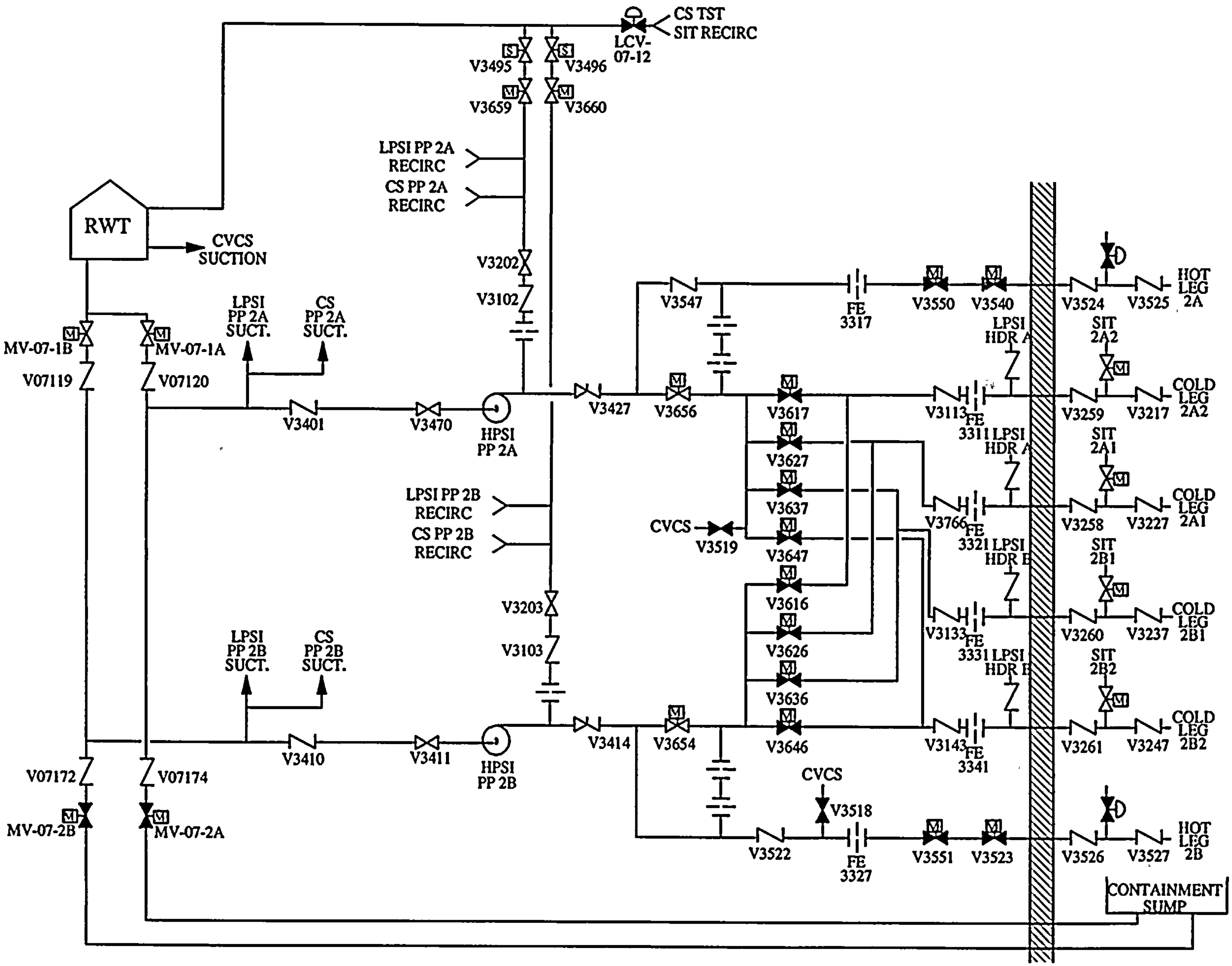

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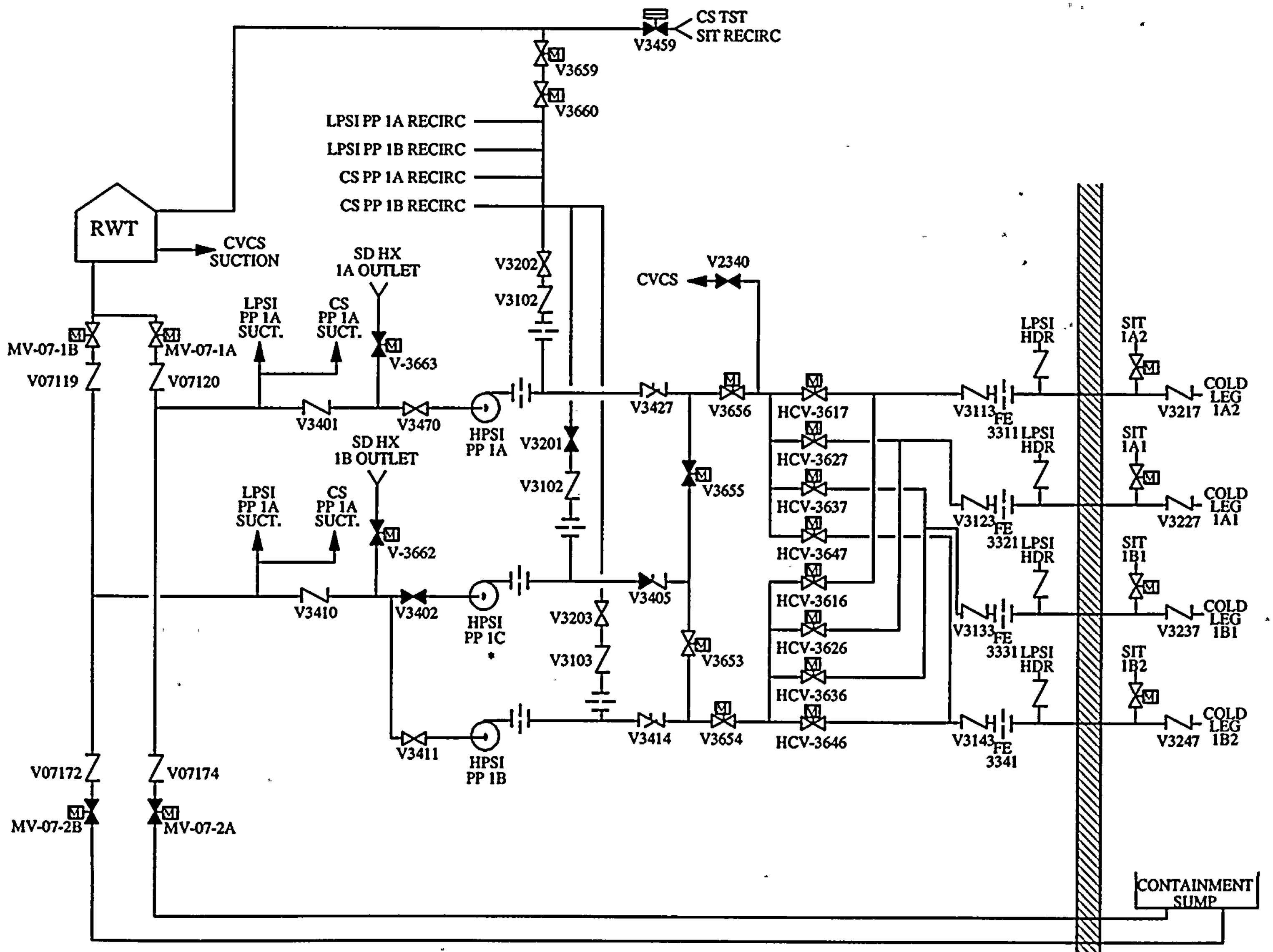
Figure B10.3 Unit 1 HPSI System Injection Phase



* 1C HPSI PUMP IS NO LONGER FUNCTIONAL

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Figure B10.4 Unit 2 HPSI System Injection Phase

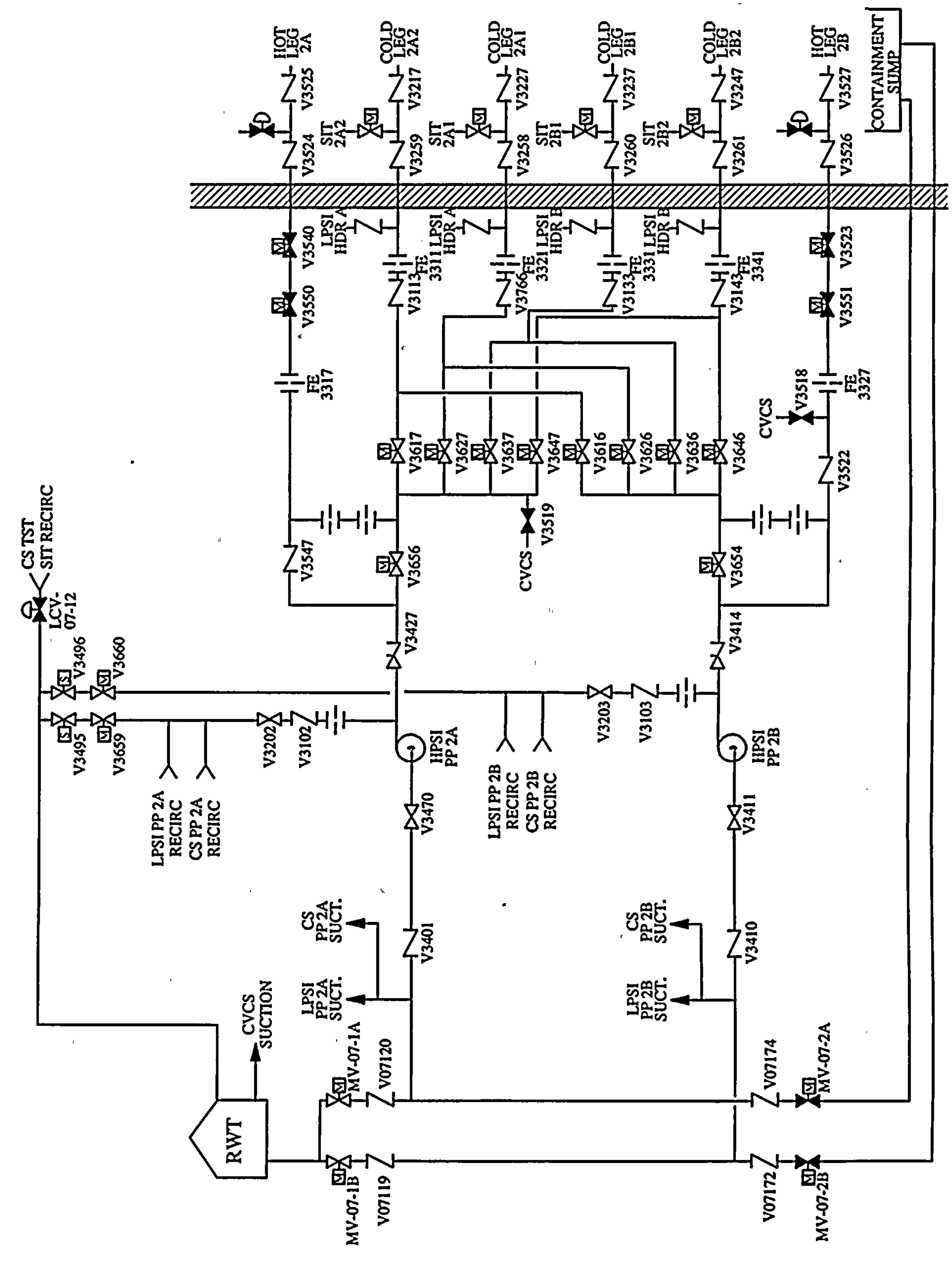
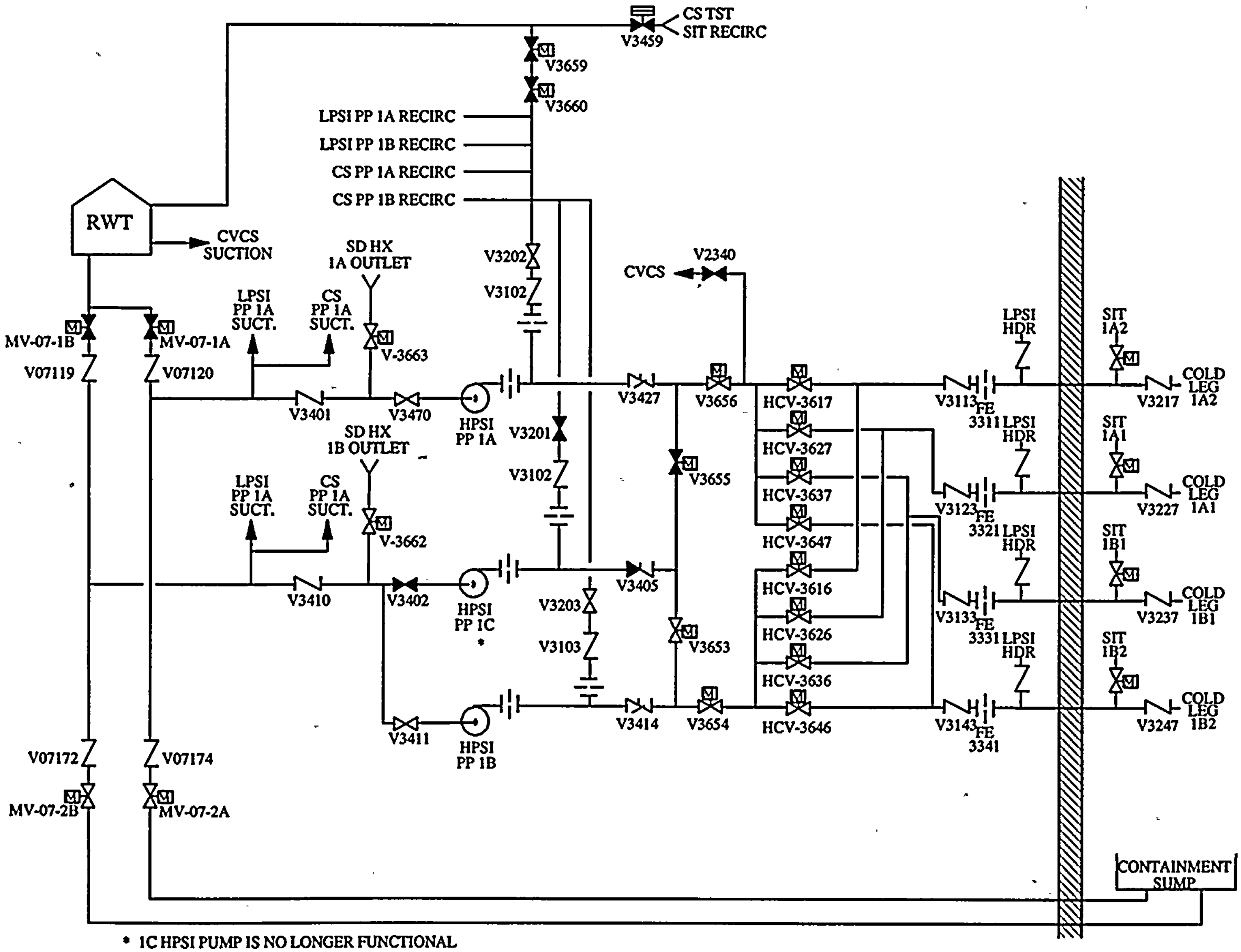


Figure B10.5 Unit 1 HPSI System Cold Leg Recirculation



* 1C HPSI PUMP IS NO LONGER FUNCTIONAL

F:\USERS\DMS\PSL\HPSI\HPSI7R2.DRW

Figure B10.6 Unit 2 HPSI System Cold Leg Recirculation

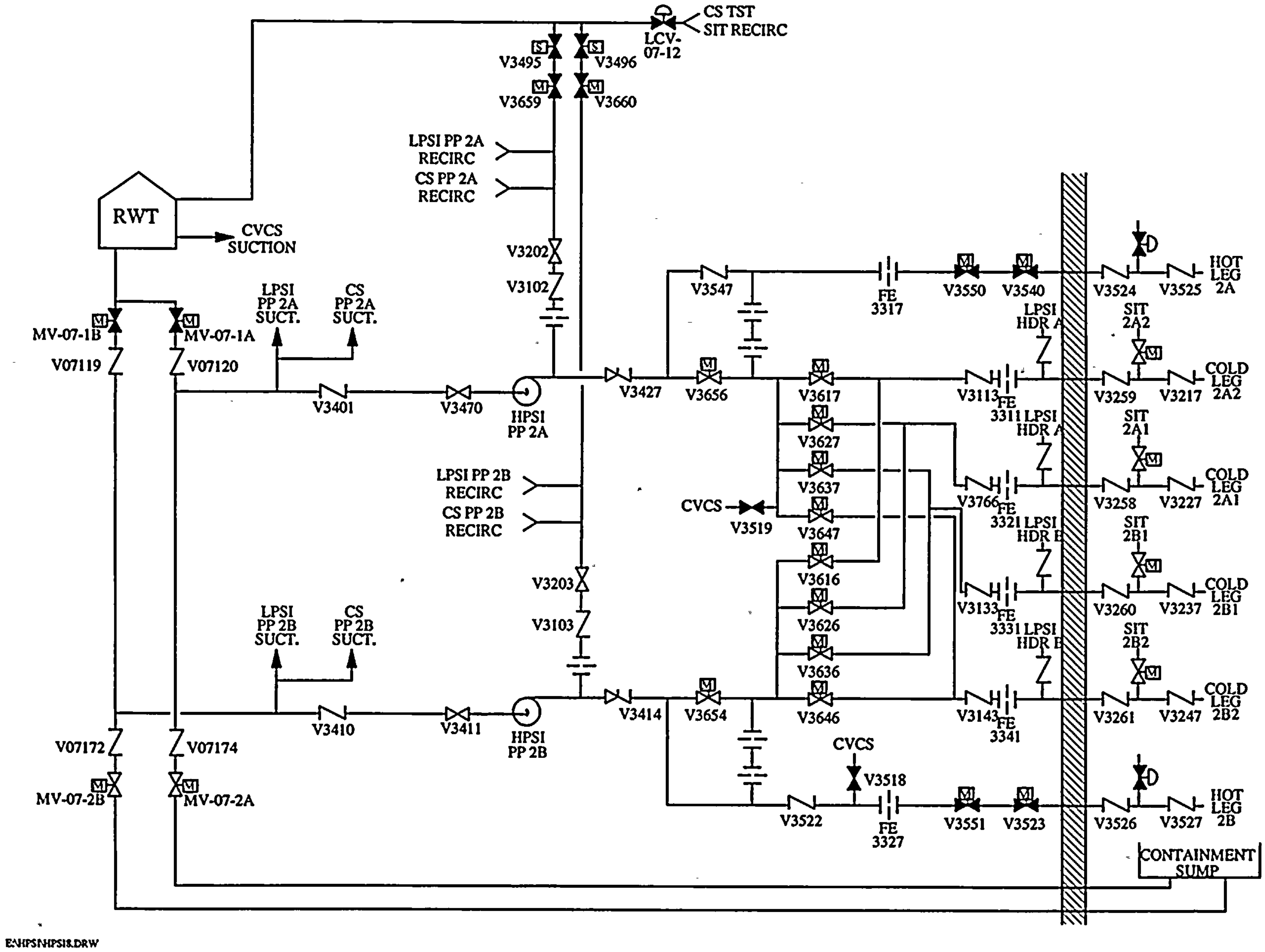
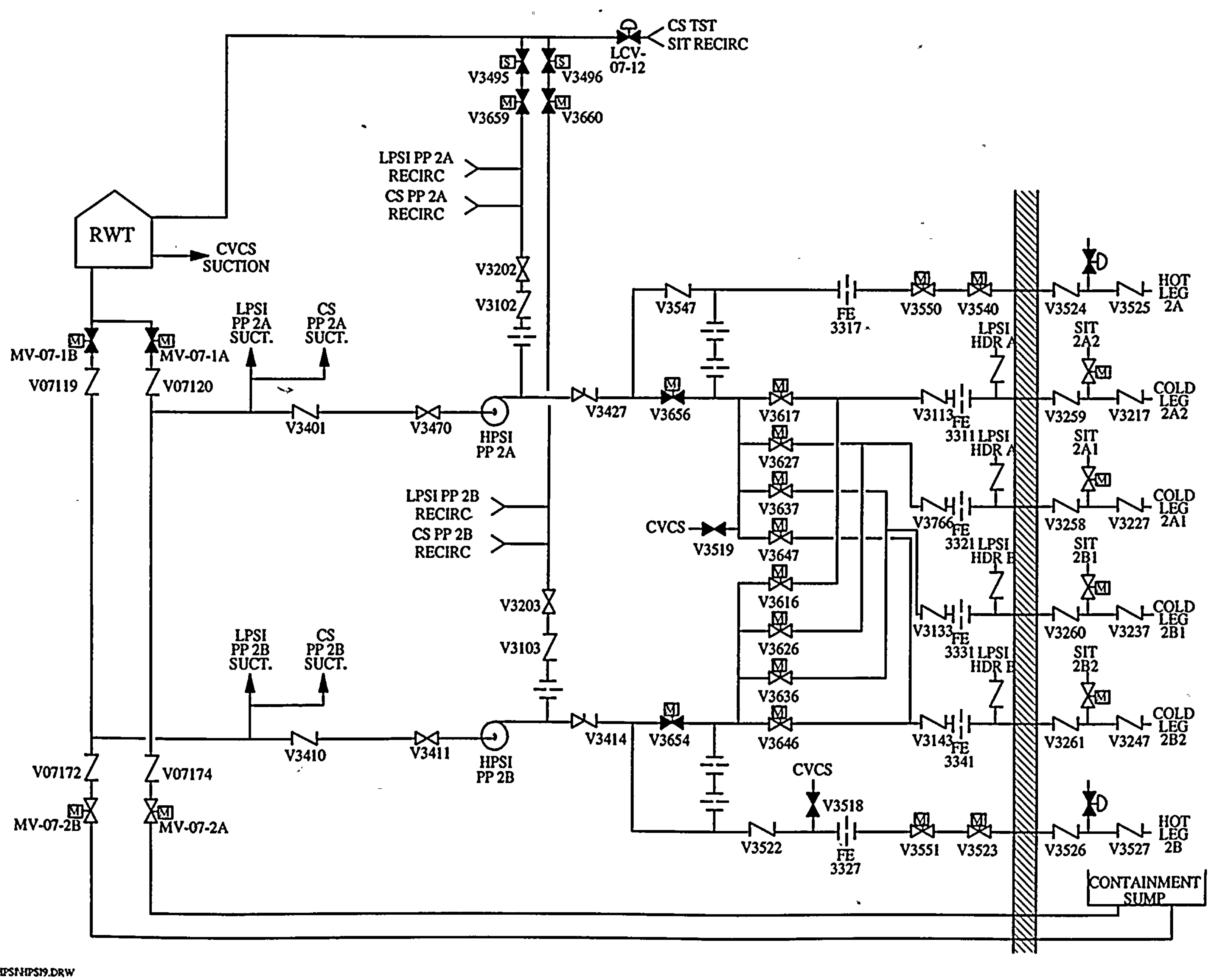


Figure B10.7 Unit 2 HPSI System Hot Leg/Cold Leg Recirculation



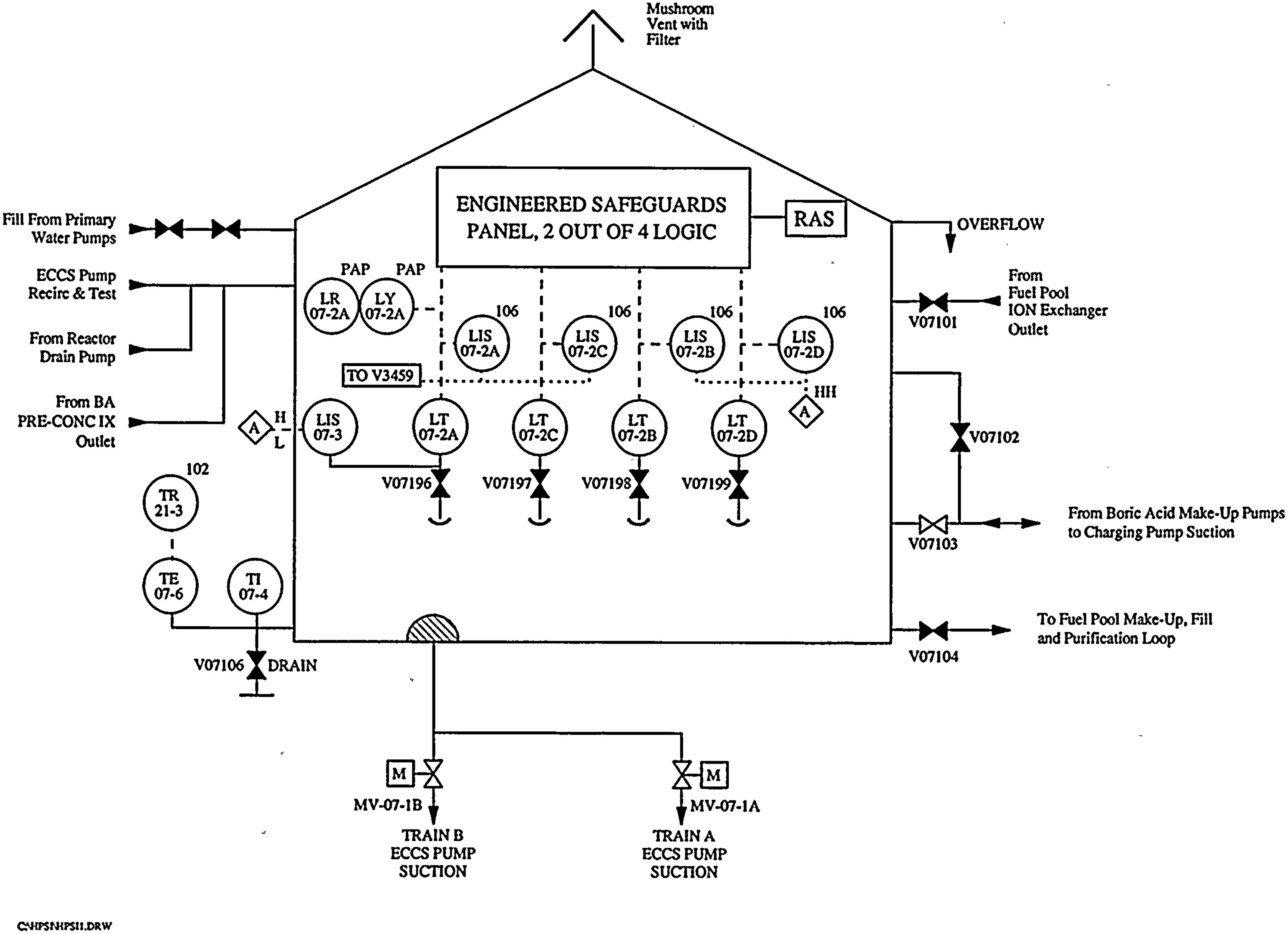
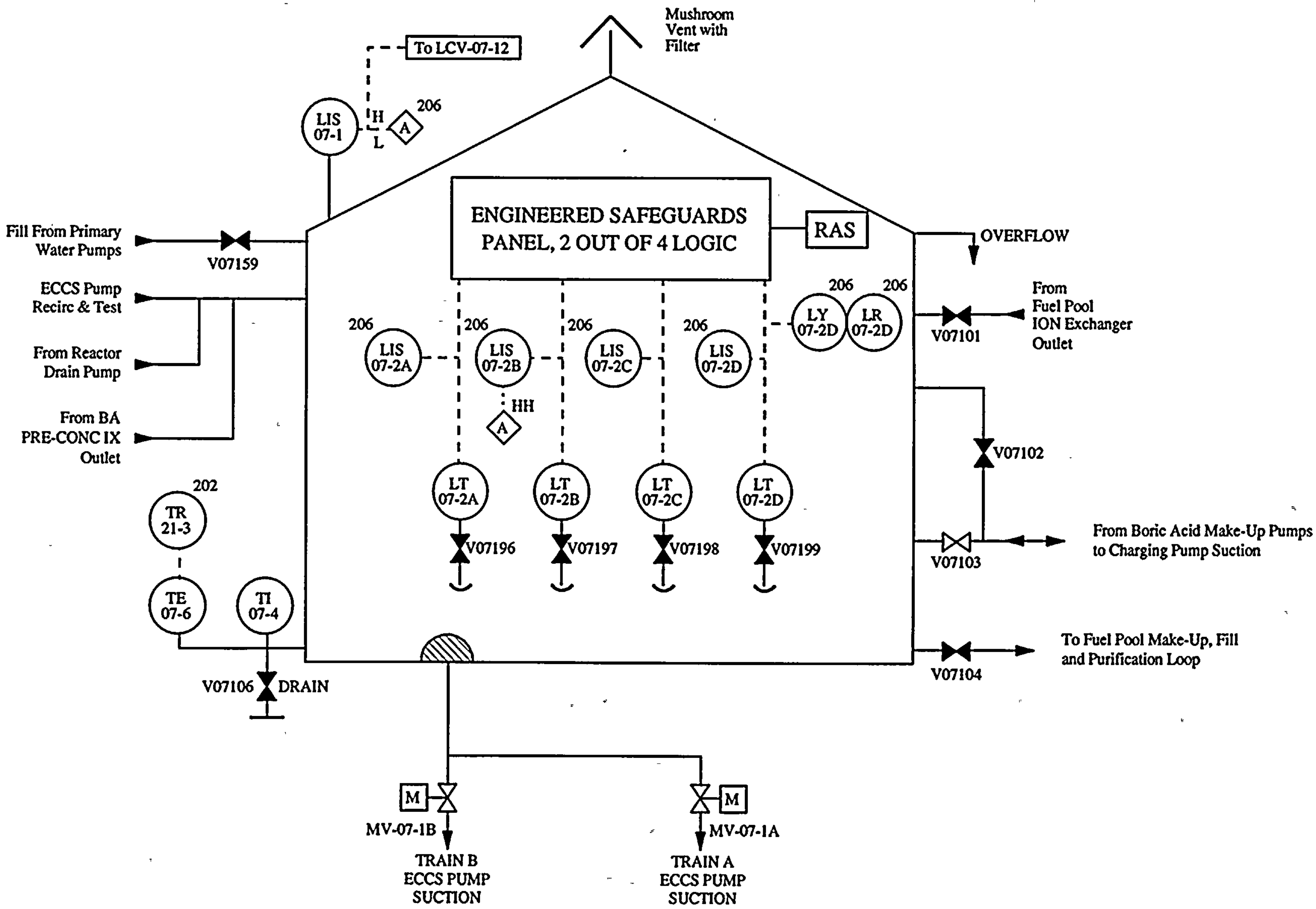


Figure B10.8 Unit 1 RWT



CNHPSNHPSI4.DRW

Figure B10.9 Unit 2 RWT

Following an SIAS, HPSI pump 1A [2A] automatically starts and supplies the Auxiliary High Pressure header [A HP header] and HPSI pump 1B [2B] automatically starts and supplies the High Pressure header [B HP header]. HPSI pump 1A [2A] and 1B [2B] are powered from 4.16 kV buses 1A3 [2A3] and 1B3 [2B3] respectively.

Each pump is protected from operating for extended periods at shutoff head by a minimum recirculation line. In Unit 1, the mini-recirc lines from each pump combine with that of the other HPSI, LPSI and CS pumps to form a common return line to the RWT. The common line contains two normally open motor operated valves which will go shut following a RAS. [In Unit 2, each ECCS recirculation line from the respective train combines to form a common minimum recirculation header (i.e. the HPSI, LPSI and CS A pumps' recirculation lines combine into a single line and those of the B pumps combine into a separate line). Each minimum recirculation line has two valves in series; a solenoid valve (V3495 and V3496) and a motor operated valve (V3659 and V3660). These normally open valves also receive a closing signal following a RAS.]

Borated water flows through the pumps, the pump discharge stop check valves and the HPSI header isolation valves. Each HPSI header splits into four lines, each line containing a normally closed motor operated isolation valve. These valves receive an open signal upon SIAS. Each line from the Auxiliary HP header [A HP header] combines with one line from the HP header [B HP header] to form a combined line, one for each RCS Cold Leg. HPSI flow passes through a check valve and then combines with the associated LPSI discharge and SIT outlet. [In Unit 2, an additional check valve separates the common HPSI/LPSI discharge from the SIT outlet.] The combined flow enters the individual RCS Cold Legs through the SI check valves.

The Refueling Water Tank is an aluminum [stainless steel] tank containing a minimum of 401,800 [417,000] gallons of borated water at a minimum boron concentration of 1720 ppm [1720-2100 ppm]. The normal configurations of the RWTs are shown in Figures B10.8 and B10.9. Each tank is sized such that all ECCS pumps can operate at runout for 20 minutes and not deplete the tank. This volume will provide a water level inside containment which will meet the minimum net positive suction head of the ECCS pumps during recirculation. The RWT also acts as the water source for the Charging pumps following a LOCA.

B10.3 Success Criteria

B10.3.1 Injection Phase:

The success criteria for the Injection phase of all accident sequences is one-out-of-two HPSI pumps supplying flow from the RWT to one-out-of-four unfaulted cold legs.

B10.3.2 Cold Leg Recirculation Phase:

The success criteria for the Cold Leg Recirculation phase of all accident sequences that require cold leg recirculation is one-out-of-two HPSI pumps supplying flow from the sump to one-out-of-four cold legs.

B10.3.3 Hot Leg/Cold Leg Recirculation Phase:

Failure of Hot Leg Recirculation is not considered a core damage sequence due to the low boron concentrations. If assumed to be required, the success criteria for the Hot Leg/Cold Leg Recirculation phase (Unit 2) would be one-out-of-two HPSI pumps supplying flow from the sump to one-out-of-two hot legs.

B10.4 Operation

B10.4.1 Normal Operation

During normal operation, the HPSI system is configured to automatically inject borated water into the RCS cold legs following an SIAS. Specifically, the HPSI pumps are idle with CCW supplied to them, the RWT outlet motor operated valves are open, the minimum recirculation line isolation valves are open (and power is removed from them in Unit 1), the sump outlet motor operated isolation valves are closed and the individual HPSI header flow control valves are closed. [In Unit 2, the Hot Leg Recirculation isolation valves are closed.] For Unit 1, the HPSI pump 1C suction, discharge and minimum recirc manual isolation valves are closed. The RWT is filled to a minimum level of 401,800 [417,100] gallons with a minimum boron concentration of 1720 ppm [between 1720 and 2100 ppm].

B10.4.2 Accident Operation

Following a Safety Injection Actuation Signal, the HPSI pumps will automatically start and the HPSI header flow control valves will automatically open to provide a flow path from the RWT to the RCS cold legs. When sufficient water has been transferred from the RWT to the Containment sump an automatic Recirculation Actuation Signal will be generated. The ESFAS system receives RWT level signals and upon 2 out 4 level transmitters reaching 4 [6] feet, a RAS will be generated. In Unit 1, prior to receiving a RAS, the operators manually restore power to the minimum recirculation isolation valves, ensure a Containment Spray pump is operating (start the pump manually if required) and manually open the High Pressure Suction Crossover valves (V3662 and V3663). This configuration will provide containment sump water cooled by the Shutdown Cooling heat exchanger to the suction of the HPSI pumps for additional core cooling. The RAS will cause the sump outlet valves to open in 30 seconds and the RWT outlet valves to shut in 90 seconds thereby ensuring uninterrupted flow to the ECCS pumps. The RAS will also cause the ECCS pump minimum recirculation valves to close.

In Unit 1, for those LOCA break sizes for which Shutdown Cooling cannot be entered prior to 10 hours following the accident, simultaneous Hot Leg/Cold Leg Recirculation is manually initiated. The primary means for Hot Leg Recirculation is via the LPSI pumps. However, if the LPSI pumps are unavailable, Hot Leg recirculation can be accomplished by directing the discharge of the HPSI pumps through the cross connect to the CVCS system (normally closed manual valve V2340) and on to the Auxiliary Spray valves. [In Unit 2, for those LOCA break sizes for which Shutdown Cooling cannot be entered prior to 6 hours following the accident, simultaneous Hot Leg/Cold Leg Recirculation is manually initiated. This involves shutting the HPSI header isolation valves (V3654 and V3656) and opening the Hot Leg Recirculation Line isolation valves (V3550, V3540, V3551 and V3523).]

B11 HEATING, VENTILATION AND AIR CONDITIONING SYSTEM

B11.1 Function

Electrical Equipment and Cable Spreading Rooms Ventilation Subsystem

The electrical equipment and cable spreading rooms ventilation subsystem has the following function whose failure may contribute to plant risk:

> Provide cooling air supply and exhaust to the rooms containing essential electrical equipment. Ventilation is provided to permit proper functioning of the equipment during normal plant operations and accident conditions.

ECCS Area Ventilation Subsystem

The ECCS area ventilation subsystem has the following functions:

1. Provide air exhaust from the rooms containing the HPSI, LPSI, containment spray pumps, and the shutdown heat exchangers. Ventilation is provided to permit proper functioning of this equipment during accident conditions, and

2. Limit the temperature rise in the ECCS pump rooms, shutdown heat exchanger rooms, and penetration areas.

Turbine Switchgear Ventilation Subsystem

The turbine switchgear area subsystem provides ventilation to permit proper functioning of equipment in the turbine switchgear room during normal and accident plant operations. Equipment normally enclosed in this room includes 480 volt switchgears 1A1 & 1B1 [2A1 & 2B1], as well as 4.16 kv switchgears 1A2 & 1B2 [2A2 & 2B2] and 6.9 kv switchgears 1A1 & 1B1 [2A1 & 2B1]. The 4.16 and 6.9 kv switchgears provide power to components such as circulating water, feedwater, and condensate pumps, and RCPs.

Intake Structure Ventilation Subsystem

There is no intake structure ventilation subsystem for Unit 1. [The intake structure ventilation subsystem provides ventilation to assure a controlled thermal environment in the intake cooling water pump enclosure during normal plant operation and accident conditions.]

B11.2 Configuration

Electrical Equipment and Cable Spreading Rooms Ventilation Subsystem

The electrical equipment rooms (EER) 1A, 1B, and the cable spreading room (CSR) [A, B and cable spreading room], are ventilated by a common air supply header and individual room exhaust fans. The air supply header consists of louvered intake, filters, 2 centrifugal supply fans in parallel (HVS-5A, 5B [2HVS-5A, 5B]), and a duct distribution system. Air is supplied from the outside to the rooms and then exhausted back to the outside atmosphere by the following fans:

Room	Boundary
EER 1A	RV-3, RV-4 (both with hoods)
EER 1B	HVE-12 [followed by motor-operated louver L-10]
Cable Spreading Room	HVE-11 [followed by motor-operated louver L-9]

The cable spreading room is also provided with supplemental cooling from non-safety related air conditioners ACC-4 and ACC-5, and air is recirculated within the room by air handlers HVA-4 and HVA-5, respectively. [Unit 2 only has air conditioner 2ACC-4/2HVA-4.]

The electrical equipment and cable spreading rooms supply/exhaust fans are connected to separate emergency buses. This system is safety related, since it is required for proper functioning of the emergency electrical distribution equipment. The Unit 1 EER fans are not automatically loaded on the diesel generator, so manual actions are taken during Loss Of Offsite Power (LOOP).

ECCS Area Ventilation Subsystem

This subsystem consists of two sections: the supply and the exhaust headers.

The air supply header of the ECCS area ventilation subsystem is also the auxiliary building main ventilation supply header. This subsystem consists of two 100% capacity centrifugal fans (HVS-4A & 4B [2HVS-4A & 4B]), wall louvers, gravity and motor-operated dampers, roughing filters, and a ductwork distribution system. This subsystem provides the ambient cooling requirements for the low and high pressure safety injection pumps, containment spray pumps, and shutdown cooling heat exchangers. Ventilation rate is sized to achieve the design ambient maximum temperature of 104^0F in the equipment areas with an outside air temperature of 93^0F.

Exhaust air from the ECCS rooms is normally directed to the plant stack via the main exhaust system (not modeled). This system consists of ductwork, pre-filters, HEPA filters, air-operated/gravity dampers, and two 100% capacity exhaust fans HVE-10A and HVE-10B [2HVE-10A and 2HVE-10B]. HVE-10A and HVE-10B [2HVE-10A and 2HVE-10B] receive a stop signal on SIAS.

The emergency air exhaust header consists of two 100% capacity fans (HVE-9A, & 9B [2HVE-9A & 9B]), HEPA and charcoal filter banks, ductwork, dampers, instrumentation, and controls. The exhaust air is normally vented to the outside atmosphere. Unit 1 is provided with a flow monitor and isokinetic sampling units located downstream of the exhaust fans.

The redundant exhaust fans, dampers, and controls serve redundant safety related components requiring ventilation. Each of the exhaust fans, dampers, and controls are supplied from the same electrical source as the component which they serve. The ECCS ventilation system maintains a slightly negative pressure in the ECCS area with respect to surrounding areas in the RAB. Access to the ECCS area is through gasketed, self-closing or locked closed doors. Opening of locked doors is under administrative controls. Piping penetrations into the ECCS area are provided with flexible rubber seals. This permits pipe movement due to thermal expansion or seismic motion while limiting the amount of air in-leakage.

Turbine Switchgear Ventilation Subsystem

The turbine building is an open structure with no mechanical ventilation system for the open equipment areas. The enclosed portions (the switchgear room and chemical storage area) are provided with HVAC systems.

The turbine switchgear room is provided with a filtered air supply and with wall mounted exhaust louvers for ventilation. The air supply is provided by two centrifugal fans (HVS-18 & 19 [2HVS-18 & 19]). Each train includes a filter housing with a louver, pre-filters, high efficiency filters, a centrifugal fan, and a backdraft gravity damper. Each supply fan provides filtered supply air into a common ductwork system. This arrangement allows a single fan train to supply air to both the A and B sides of the switchgear room. [Pressure relief dampers are provided on the east side of the switchgear room. These dampers are selected to allow the exhausting of the air while maintaining a light positive pressure within the room. One train is in operation at all times in order to provide filtered supply air to the switchgear room and to provide positive pressure. Once the supplied air flows through the rooms picking up equipment heat, the air is exhausted to the atmosphere through wall mounted louvers provided with protective hoods.

[Intake Structure Ventilation Subsystem

The intake structure ventilation system consists of two redundant 100% capacity propeller exhaust fans (2HVE-41A & 41B), two pressure dampers (PD-1, PD-2), and two screened openings.

Air is drawn to the structure through the screened openings and exhausted by the propeller fans to the atmosphere. Each fan is powered from an independent redundant bus of the emergency electrical distribution system. Missile protection and pressure dampers are provided in the exhaust opening to protect the exhaust fans from external missiles and excessive wind conditions.]

B11.3 Success Criteria

The Heating, Ventilation, and Air Conditioning subsystems discussed are designed to remove heat from the electrical equipment/cable spreading rooms, the ECCS area, the turbine switchgear room, [and intake structure room] during normal and post-accident conditions.

The particular success criteria for the each of these subsystems is as described below.

Electrical Equipment and Cable Spreading Rooms Ventilation Subsystem

The electrical equipment and cable spreading rooms ventilation subsystem provides cooling and ventilation to two electrical equipment rooms (1A, 1B, and one cable spreading room [2A, 2B, cable spreading room]). These rooms are supplied with outside air by a pair of supply fans (HVS-5A, HVS-5B). Both of these supply fans must be operating in addition to the exhaust fan(s) or ventilator(s) of the individual rooms. Therefore, the success criteria for Electrical Equipment and Cable Spreading Rooms are as follows:

The Electrical Equipment Room 1A [2A] does not require forced ventilation during the assumed 24 hr. PRA mission time.

Successful operation of the Electrical Equipment Room 1B Ventilation subsystem requires the operation of one of the supply fans (HVS-5A or 5B). [For Unit 2, successful operation of the Electrical Equipment Room 2B requires exhaust to the cable spreading room. Hence, the success criterion for the cable spreading room is applicable.]

Successful operation of Unit 1 cable spreading room ventilation subsystem requires the operation of one of the supply fans (HVS-5A or 5B). Operation of the air conditioning units is not required for success of the system. [Successful operation of Unit 2 cable spreading room subsystem requires the operation of one supply fan (HVS-5A or 5B) or one of the two exhaust paths. Each exhaust path contains a motor-operated fan (HVE-11 & HVE-12) and a gravity damper (GD-19 & GD-20).]

The static inverter room in Unit 1 receives air supply from the Electrical Equipment Room 1B. Also, air exhausted from the static inverter room flows into EER 1B. Therefore, the success criterion for the static inverter room is the same as for EER 1B. [In Unit 2, air supply and exhaust is similar to that of the cable spreading room. Therefore, the success criterion for the static inverter room is the same as for the cable spreading room.]

ECCS Area Ventilation Subsystem

The ECCS Room Ventilation subsystem consists of a main supply portion, a main exhaust portion, an ECCS exhaust portion, dampers and ductwork to direct air to and exhaust air from various ECCS areas during ECCS and non-ECCS operations. Successful operation of this HVAC subsystem during ECCS pump operation requires the operation of one of the two supply fans (HVS-4A or HVS-4B), or one of two ECCS exhaust paths, and proper alignment of motor-operated dampers. Each ECCS path consists of two motor-operated dampers (D-13, D-14 and D-15, D-16), filters, a motor-operated fan (HVE-9A, HVE-9B), and a motor-operated louver (L-7A, L-7B). During non-ECCS operation, one of the two supply fans and one of the two main exhaust paths operate to provide ventilation of the ECCS area. Each main exhaust path consists of filters, pressure dampers (DPR-25-10A, 10B), a motor-operated fan (HVE-10A, 10B), and a gravity damper. In the event the main exhaust system is available, the operator would use the ECCS exhaust system by manually actuating the ECCS exhaust fans. HVE-10A and HVE-10B [2HVE-10A and 2HVE-10B] receive a stop signal on SIAS.

Turbine Switchgear Ventilation Subsystem

The Turbine Switchgear Room Ventilation subsystem consists of two supply trains. Each train contains a supply fan (HVS-18 [2HVS-18] or HVS-19 [2HVS-19]), filters, a fixed louver, and a gravity damper. One train is assumed for this HVAC subsystem to be successful.

[Intake Structure Ventilation Subsystem]

[Intake Structure Ventilation subsystem success is assumed to require one fan in the exhaust paths to operate, since each fan train has a 100% capacity. Each path consists of a motor-operated fan (HVE-41A & HVE-41B) and a pressure/gravity damper (PD-1 & PD-2).]

B11.4 Operation

B11.4.1 Normal Operation

The following table shows the number of operating fans in each room during normal plant operations:

TABLE B11.1 - HVAC FANS OPERATING DURING POWER OPERATIONS

FAN/ VENTILATOR	NO. IN OPERATION UNIT 1	NO. IN OPERATION UNIT 2
HVS-4A, 4B	1	1
HVE-9A, 9B	0 (2 ON SIAS)	0 (2 ON SIAS)
HVE-10A, 10B	1 (0 ON SIAS)	1 (0 ON SIAS)
HVS-5A, 5B	2	1
RV-1, 2	2	2
RV-3, 4	2	2
HVE-11, 12	2	1
HVA/ACC-4, 5	HVA/ACC-4 ONLY	0
HVS-18, 19	2	2
HVE-41A, 41B	N/A	2

Electrical Equipment and Cable Spreading Rooms Ventilation Subsystem

During normal operation with all supply and exhaust fans operating (and one non-safety grade air conditioner in Unit 1 in operation), the ventilator air flow rates for the electrical equipment rooms, static inverter room and cable spreading room are sized to achieve required temperatures. In the event that the air conditioners are not in operation (both in Unit 1), the ventilators' air flow rates are sufficient to maintain adequate cooling in all the rooms].

During normal operations, two air supply fans (HVS-5A and 5B) and two exhaust fans in EER 1A (RV-3 and 4) are operating as well as both fans HVE-11 and HVE-12 along with one air conditioner (HVA/ACC-4). [In Unit 2, one air supply fan (2HVS-5A or 5B) is operating, with both exhaust propeller fans (2RV-3 or 4), and one centrifugal fan exhaust (2HVE-11 or 12). Air conditioner 2HVA/ACC-4 is not in operation.]

ECCS Area Ventilation Subsystem

During normal operations, the auxiliary building main ventilation supply and exhaust system provides the necessary ventilation of the ECCS pump rooms, and the ECCS area ventilation subsystem is maintained in a standby condition. During normal operations, only one supply fan of the main supply system and one exhaust fan in the main exhaust system are in operation. Operation of the auxiliary building main supply fans (HVS-4A & 4B [2HVS-4A & 4B]) is administratively controlled in plant procedure 1-1900020 [2-1900020]. According to this procedure, one fan is started and the second one is placed in AUTO mode for normal operations. All dampers to the exhaust fans HVE-9A and 9B [2HVE-9A and 9B] are closed and these fans are not operating during normal operations.

Turbine Switchgear Ventilation Subsystem

During normal operations, fans HVS-18 and 19 [2HVS-18 and 19] are in operation. [In Unit 2, this ventilation subsystem has been designed to operate with either one or both fans. One train should be in operation at all times in order to provide filtered supply air to the switchgear room and provide a positive pressure.]

[Intake Structure Ventilation Subsystem

During normal plant operations, one of the two intake structure 100% capacity fans is operating to maintain the temperature of the intake cooling water pump room below 104°F.]

B11.4.2 Accident Operation

Electrical Equipment and Cable Spreading Rooms Ventilation Subsystem

During an emergency condition that involves a loss of offsite power (LOOP), the temperature in the rooms will increase until the fans are restarted, and quickly reduce temperature. Administrative controls ensure the fans will be manually restarted prior to exceeding 120^0F in any of the rooms, since high temperature alarms are available to the operators with a setpoint of 110^0F.

ECCS Area Ventilation Subsystem

Under accident conditions, when several or all of the safety injection pumps are operating, the air supply to the nonessential section of the auxiliary building is directed to the ECCS pump rooms to provide the additional cooling air requirement. The idle auxiliary building main supply fan is started by an SIAS, and motor-operated dampers are positioned automatically on SIAS to provide the proper flow path for supply air to the ECCS area. Simultaneously, the ECCS exhaust fans are energized and dampers in the exhaust ductwork are positioned to allow the fans to draw all exhaust air from the area through the HEPA and charcoal filter bank before discharge to the atmosphere. Motor-operated dampers are interlocked with the exhaust fans, so that they are actuated (open or close) when the fans are actuated by an SIAS. HVE-10A and HVE-10B [2HVE-10A and 2HVE-10B] also receive a stop signal on SIAS.

Turbine Switchgear Ventilation Subsystem

During accident conditions, filtered supply air is provided to the switchgear rooms. The system is not available during a loss of offsite power condition.

[Intake Structure Ventilation Subsystem

During accident conditions, the two 100% capacity propeller exhaust fans operate to serve the safety related equipment and assure a controlled environment in the structure.]

B12 INSTRUMENT AIR SYSTEM

B12.1 Function

The function of the Instrument Air system is to provide motive power and control air to plant system components during normal and accident operations. This system is designed to provide clean (oil, dirt, and contaminant free) dry air.

The minimum system pressure is 85 psig measured downstream of the instrument air dryers. This pressure is required for proper operation of the MSIVs. Note that this is an operational requirement and does not relate to any safety function.

B12.2 Configuration

Figures B12.1 through B12.9 are simplified schematics of the Unit 1 and Unit 2 Instrument Air systems.

The Instrument Air system provides a reliable supply of dry, oil-free air at the required pressure for pneumatic instruments and controls and pneumatically operated valves both inside and outside containment. The system utilizes oil free compressors, air dryers, and air filters located at each dryer. The system serves no safety function since it is not required to achieve safe shutdown or mitigate the consequences of a LOCA.

The Unit 1 Instrument Air system is comprised of a system outside containment, including 4 instrument air compressors located outside containment, as well as a system including 2 instrument air compressors located inside containment and dedicated to supplying instrument air inside containment. Note that the numbering of the compressors can cause confusion as the inside containment compressors and the outside containment backup compressors are numbered 1A and 1B. The Unit 1 inside and outside containment Instrument Air systems are linked by a containment penetration line and pressure control valve PCV-18-5. A check valve allows flow only from the outside containment Instrument Air system to the inside containment Instrument Air system. [All of the Unit 2 air compressors are located outside containment with headers leading to components both inside and outside the containment. Therefore, the term 'outside' containment in regard to Unit 2 will refer to the entire Unit 2 Instrument Air system.]

The outside containment Instrument Air system incorporates two full capacity (1C, 1D) [(2C, 2D)] and two half capacity (1A, 1B) [(2A, 2B)] compressors. The instrument air compressors discharge to a single header connected to an air receiver and two full capacity air dryer and filter assemblies. The various air operated valves and pneumatic instruments and controls are supplied from the header.

The Instrument Air header is divided into branch lines supplying various areas/components.

Unit 1 and Unit 2 Instrument Air systems may be cross-connected. The cross connection consists of normally closed pressure regulating valves which are actuated when system pressure in either unit decreases to 85 psig. The Unit 1 and Unit 2 Service Air system may also be manually cross connected during off-normal or emergency conditions.

The 1C and 1D [2C and 2D] air compressors function as the primary source of instrument air and are each capable of meeting the full requirement of the plant instrument air usage. The 1C and 1D [2C and 2D] compressors can operate either fully loaded or half loaded. During operation, when the instrument air receiver pressure decreases, the in-service compressor loads to 50% capacity and remains at that rating until system air pressure increases. If the system air pressure continues to decrease, the in-service compressor loads to 100% capacity. The other air compressor is in standby and starts automatically if system air pressure falls to 100 psig.

The second set of air compressors (1A and 1B [2A and 2B]), which are not full capacity units, are normally placed in 'OFF', but may be manually aligned for use under abnormal operating conditions (e.g., whenever air compressors are required with only vital power available to meet the instrument demand during a loss of offsite power event). The 1A and 1B compressors will maintain the instrument air receiver pressure between 92-98 psig.

The Unit 1 inside containment compressed air system is comprised of two full capacity compressors each having separate aftercoolers, prefilters, moisture separators, air dryers, and afterfilters. The compressors discharge to a single header which supplies the containment instrument air requirements. One compressor is normally operating with the other in standby, in order to maintain air receiver pressure. [Note that Unit 2 does not have a separate inside containment Instrument Air system.]

Compressors

I. Inside Containment Compressors (1A, 1B)

The Unit 1 inside-Reactor Containment Building (RCB) Instrument Air system has two rotary air compressors in parallel. These compressors are cooled, lubricated and sealed with CCW, take suction on containment atmosphere, and discharge through seal water separators, which are part of the compressor package, to a common RCB Instrument Air Receiver. The receiver outlet is routed to two fully automatic, dual tower desiccant dryers equipped with pre/post-filters. Both drying tower sets discharge to a common line feeding the RCB Instrument Air Ring Header.

II. Outside Containment Compressors (1A, 1B , 1C, 1D [2A, 2B, 2C, 2D])

The outside-RCB Instrument Air systems each have four air compressors: two 400 scfm non-lubricated air compressors which are in normal use and two smaller capacity, 162 [230.9] scfm, non-lubricated air compressors.

Instrument Air Compressors A and B are normally in 'OFF' with their compressor discharge valves shut. All controls are fully functional, but A and B are used only for backups to the C and D compressors. The A and B compressors are powered from the Safety-Related, non-essential 480 V AC buses and can be used during a Loss Of Off-Site Power (LOOP) as a source of air. Compressors C and D are powered from non-Safety-Related buses and cannot be powered during a LOOP.

The Turbine Cooling Water system (TCW) supplies cooling water for the Instrument Air compressor water jackets. The A and B Instrument Air compressors (NOT C and D) and Service Air compressor can be connected to a separate, ambient air cooled system, called the Instrument Air Emergency Cooling System, as an emergency measure, if the TCW pumps become disabled. An overload on either the Emergency Cooling System water pump or cooling fan causes the Control Room alarm INSTRUMENT AIR COMPRESSOR EMERGENCY COOLING SYSTEM OVERLOAD TRIP.

Inter and Aftercoolers

Aftercoolers are located at the outlet of each of the Instrument and Service Air Compressors, to cool and condense moisture in the air. The aftercoolers are tube and shell type and are cooled by TCW. The shell side contains water and the tube side contains the air.

Intercoolers are located at the discharge of the Instrument Air Compressor C & D first stage(s) to cool and condense the interstage moisture in the air. The intercoolers are shell and tube heat exchangers which are cooled by Turbine Cooling Water (TCW). In addition, Instrument Air Compressors C & D also have TCW supplied to a lube oil heat exchanger to cool the oil system and cylinder heads.

Receivers

There are two (2) Unit 1 instrument air receivers, one for the Instrument Air system outside the RCB and one for the Instrument Air system inside the RCB. [There is one Unit 2 instrument air receiver which is located outside the RCB.]

Filters and Air Dryers

The Unit 1 Instrument Air inside the RCB is processed through a receiver and dual train of prefilters, desiccant dryers and afterfilters to upgrade instrument air downstream of the receiver prior to use. Solenoid operated control valves FCV-18-1A and FCV-18-1B on the outlet of the aftercoolers are open only when the respective dryers are in operation. The dryers automatically cycle through drying and regeneration/purge modes with timer sequenced solenoid valves. A local indicator on the dryer panels shows the operating status of the dryer and position of the associated FCV.

The Containment Instrument Air Dryers consist of two sets of dual drying chambers with attendant controls, prefilters, and postfilters. The dryers are equipped with postfilters to stop any desiccant 'FINES' or other particles.

Instrument Air outside of the RCB is processed through a prefilter, dual chamber instrument air dryer, and an afterfilter, downstream of the Instrument Air Receiver.

System Valves

PCV-18-5 (Unit 1 Only)

This valve links the Unit 1 inside and outside containment Instrument Air systems. The valve is set at 80 psig such that when the inside containment Instrument Air system pressure drops below 80 psig, it is fed by the outside containment Instrument Air system. Note that check valve V18195 allows flow only from the outside containment Instrument Air system to the inside containment Instrument Air system.

FCV-18-1A, FCV-18-1B

These solenoid operated flow control valves are on the outlet of the Unit 1 Instrument Air inside containment aftercoolers and are open only when the respective dryers are in operation.

Unit 1 & 2 Instrument Air Cross Connection

Cross connection capability exists between the St. Lucie Unit 1 and Unit 2 Instrument Air systems. The cross connect lines have 2 normally closed pressure regulating valves which are actuated by a decrease in pressure on either unit. Note that all valves and instruments in the cross connection system are considered to be Unit 2 components.

Unit 1 Supply from Unit 2

Valve PCV-18-5 is the Unit 1 Supply from Unit 2 IA Cross Connect Valve. PS-18-47 controls flow from Unit 2 to Unit 1 such that PCV-18-5 opens at Unit 1 pressure of 85 psig and closes at Unit 1 pressure of 95 psig. PIC-18-5 closes PCV-18-5 if Unit 2 air pressure drops below 85 psig to prevent simultaneous loss of Unit 1 and Unit 2 Instrument Air.

Unit 2 Supply from Unit 1

Valve PCV-18-6 is the Unit 2 Supply from Unit 1 IA Cross Connect Valve. PS-18-46 controls flow from Unit 1 to Unit 2 such that PCV-18-6 opens at Unit 2 pressure of 85 psig and closes at Unit 2 pressure of 95 psig. PIC-18-6 closes PCV-18-6 if Unit 1 air pressure drops below 85 psig to prevent simultaneous loss of Unit 1 and Unit 2 Instrument Air.

Figure B12.1 Simplified Schematic for Containment Instrument Air System Supply [Inside Containment] Unit 1

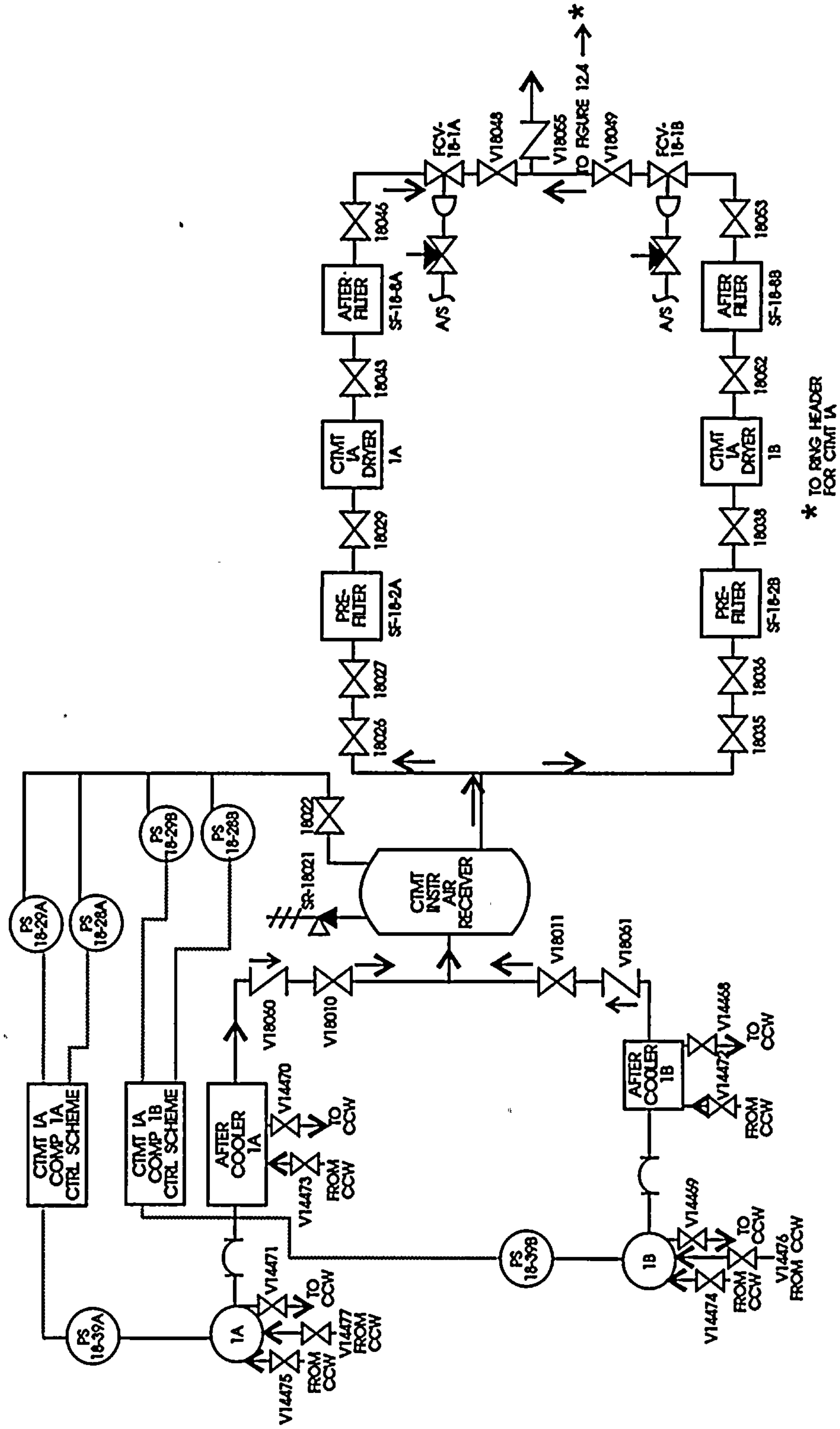


Figure B12.2 Simplified Schematic for Instrument Air System Supply [Outside Containment] Unit 1

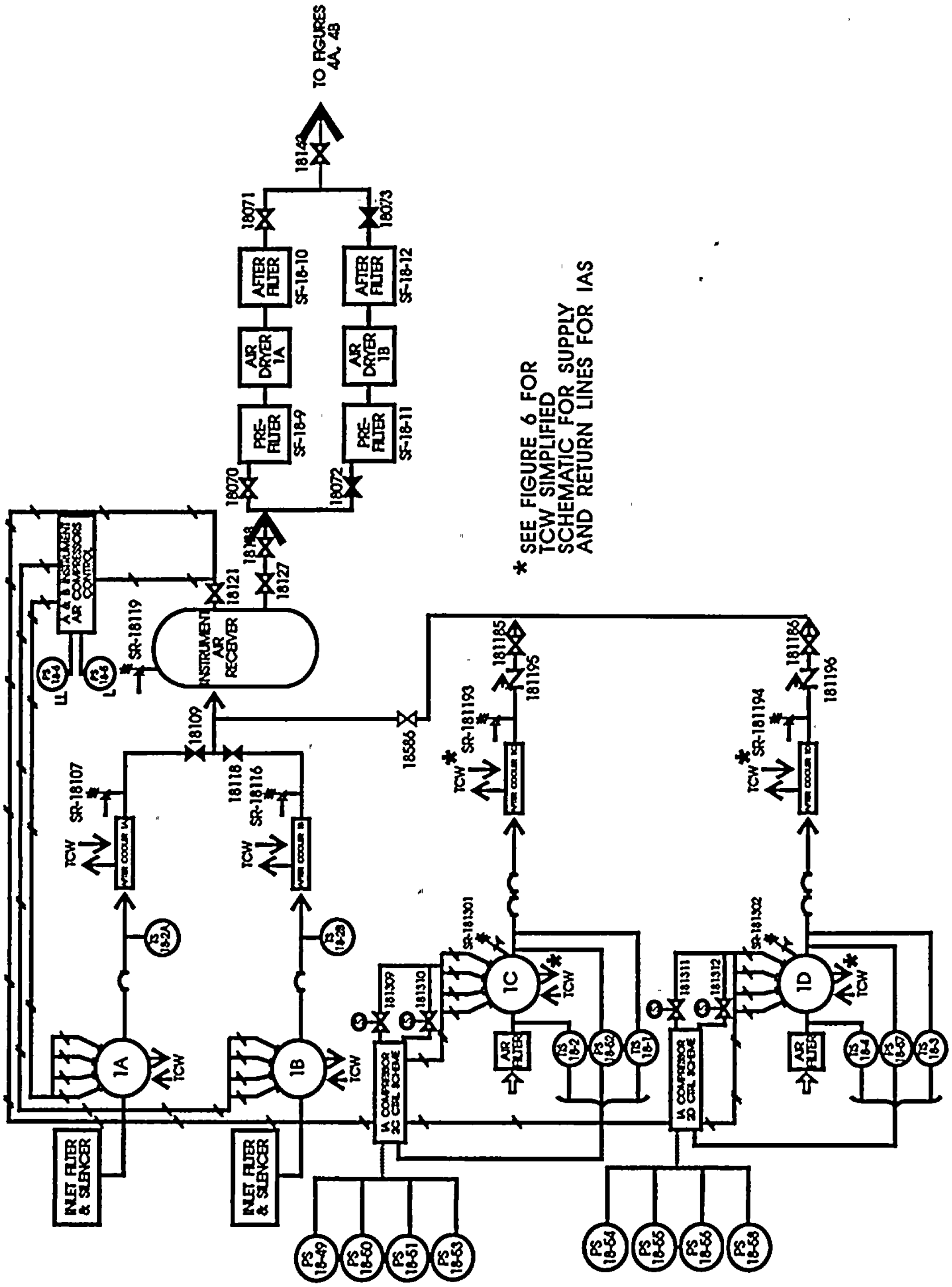
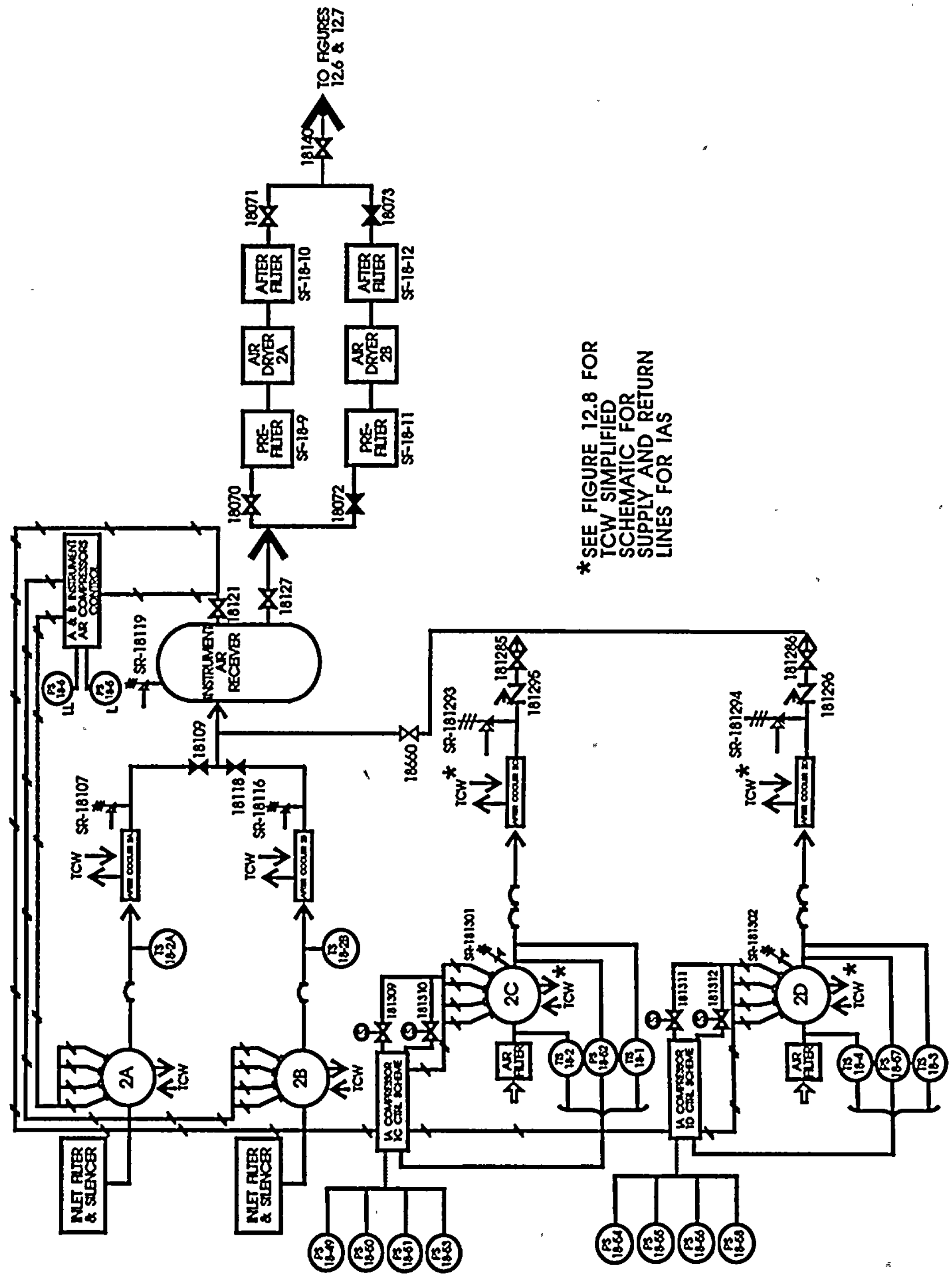


Figure B12.3 Simplified Schematic for Instrument Air System Supply (Unit 2)



*SEE FIGURE 12.8 FOR TCW SIMPLIFIED SCHEMATIC FOR SUPPLY AND RETURN LINES FOR IAS

Figure B12.4 Simplified Schematic for Instrument Air System Unit 1

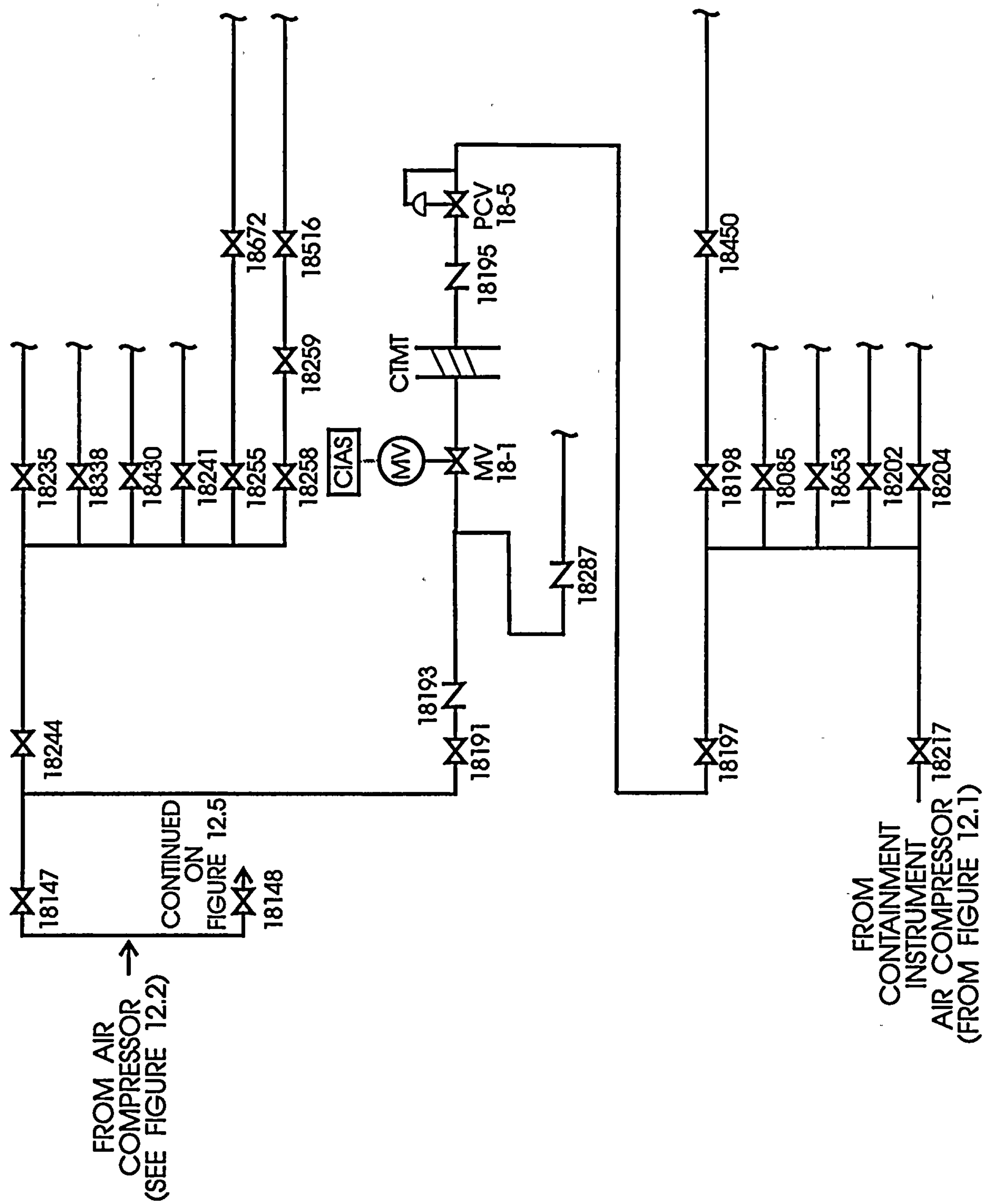


Figure B12.5 Simplified Schematic for Instrument Air System Unit 1

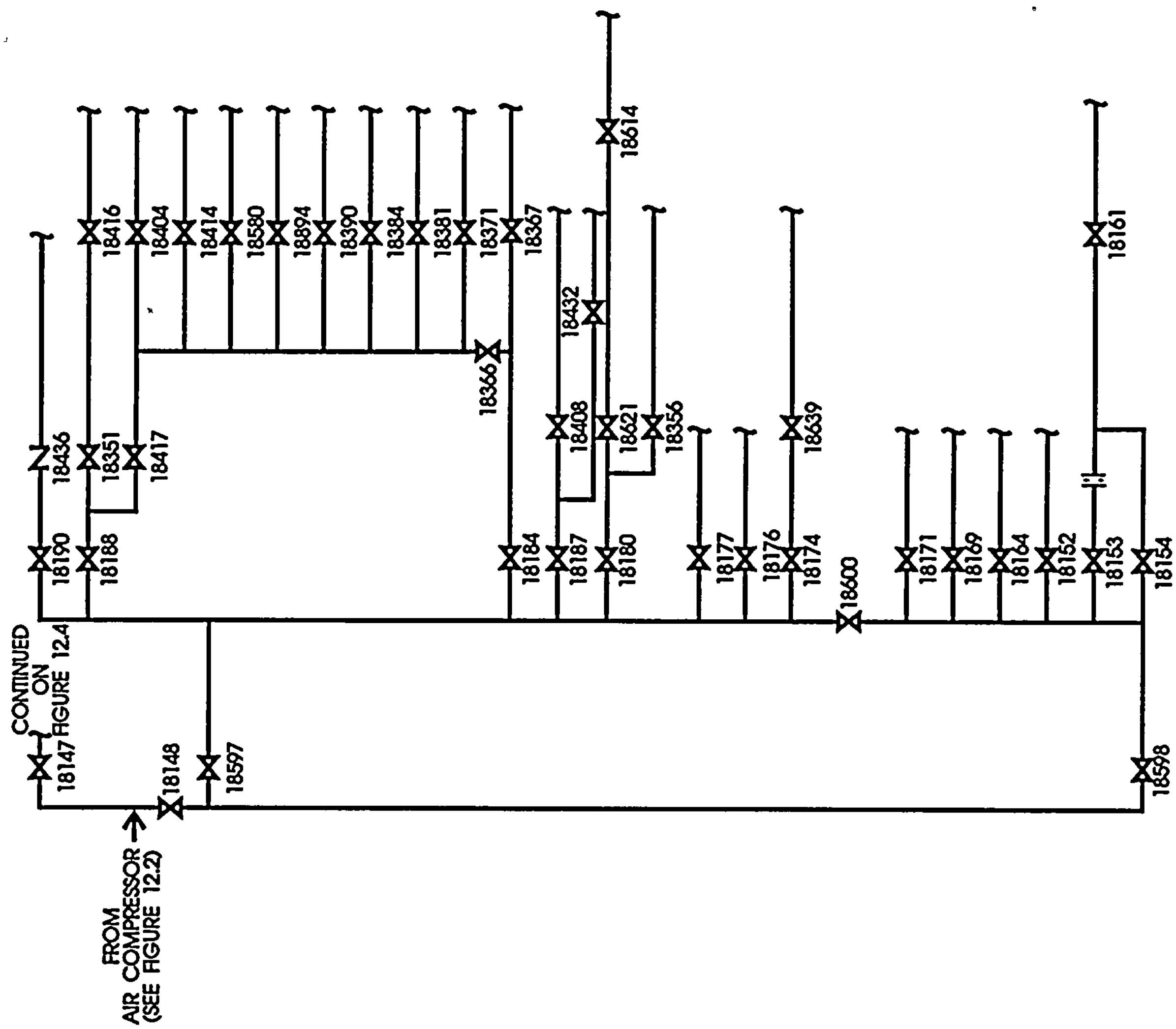


Figure B12.6 Simplified Schematic for Instrument Air System Unit 2

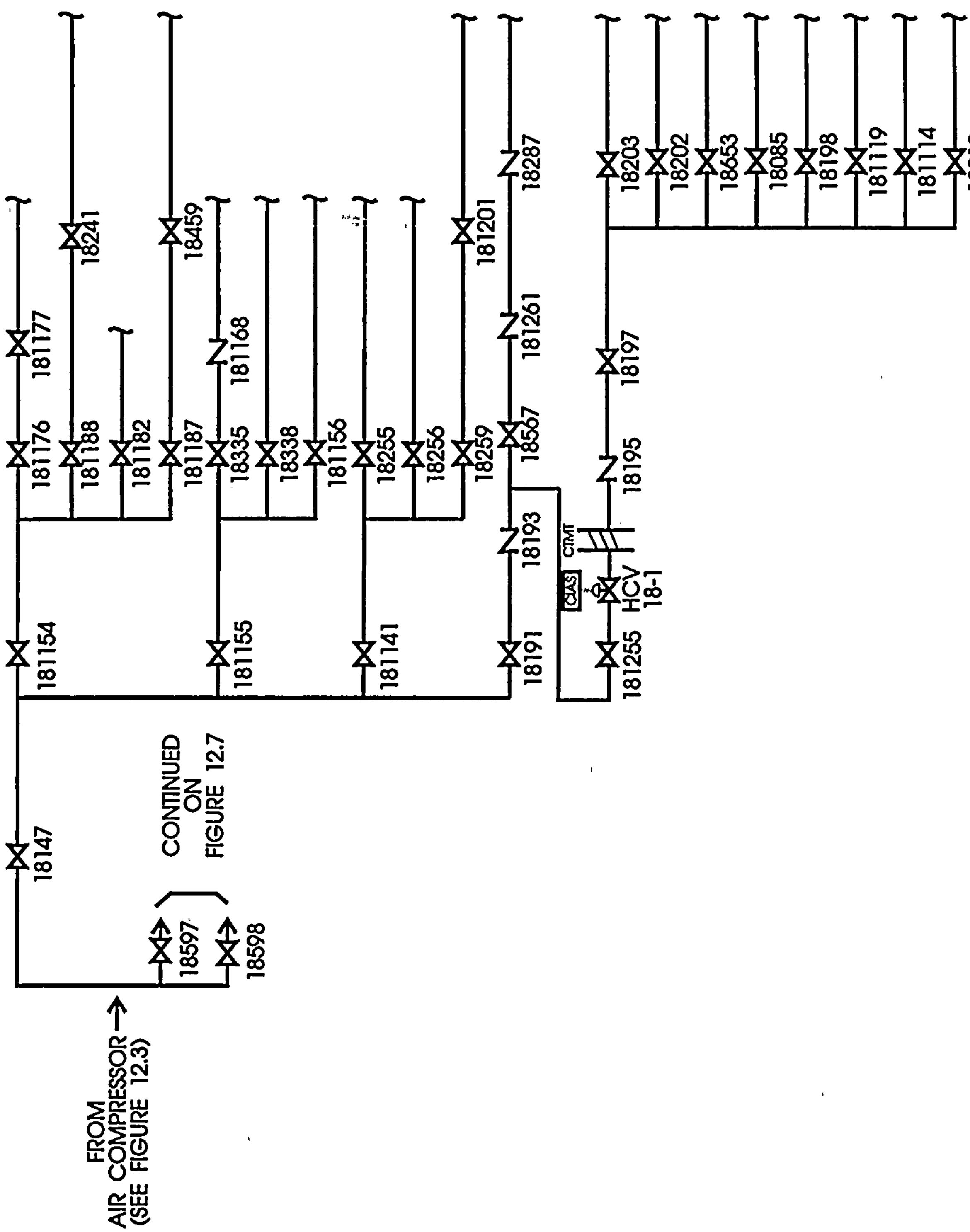


Figure B12.7 Simplified Schematic for Instrument Air System Unit 2

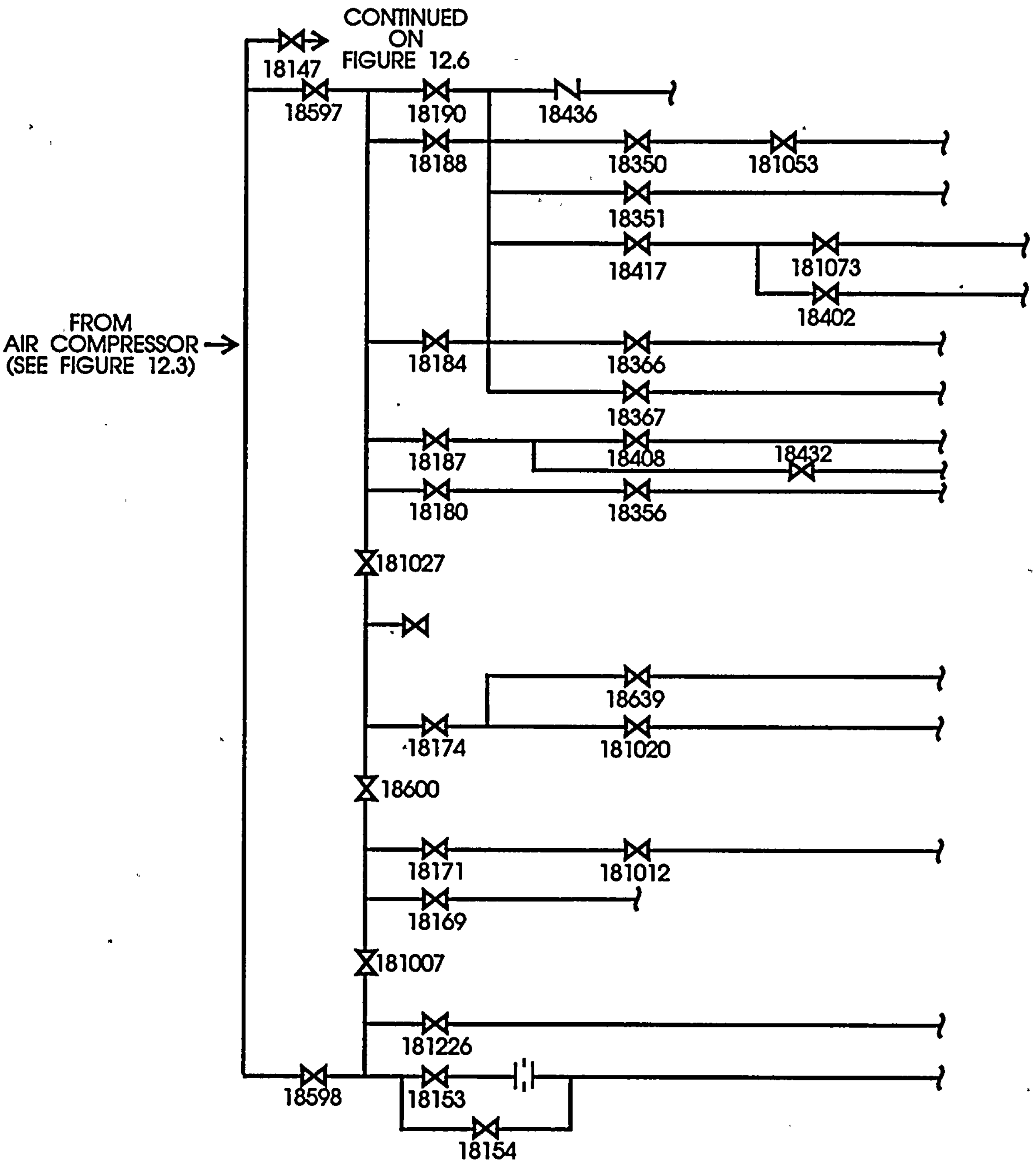
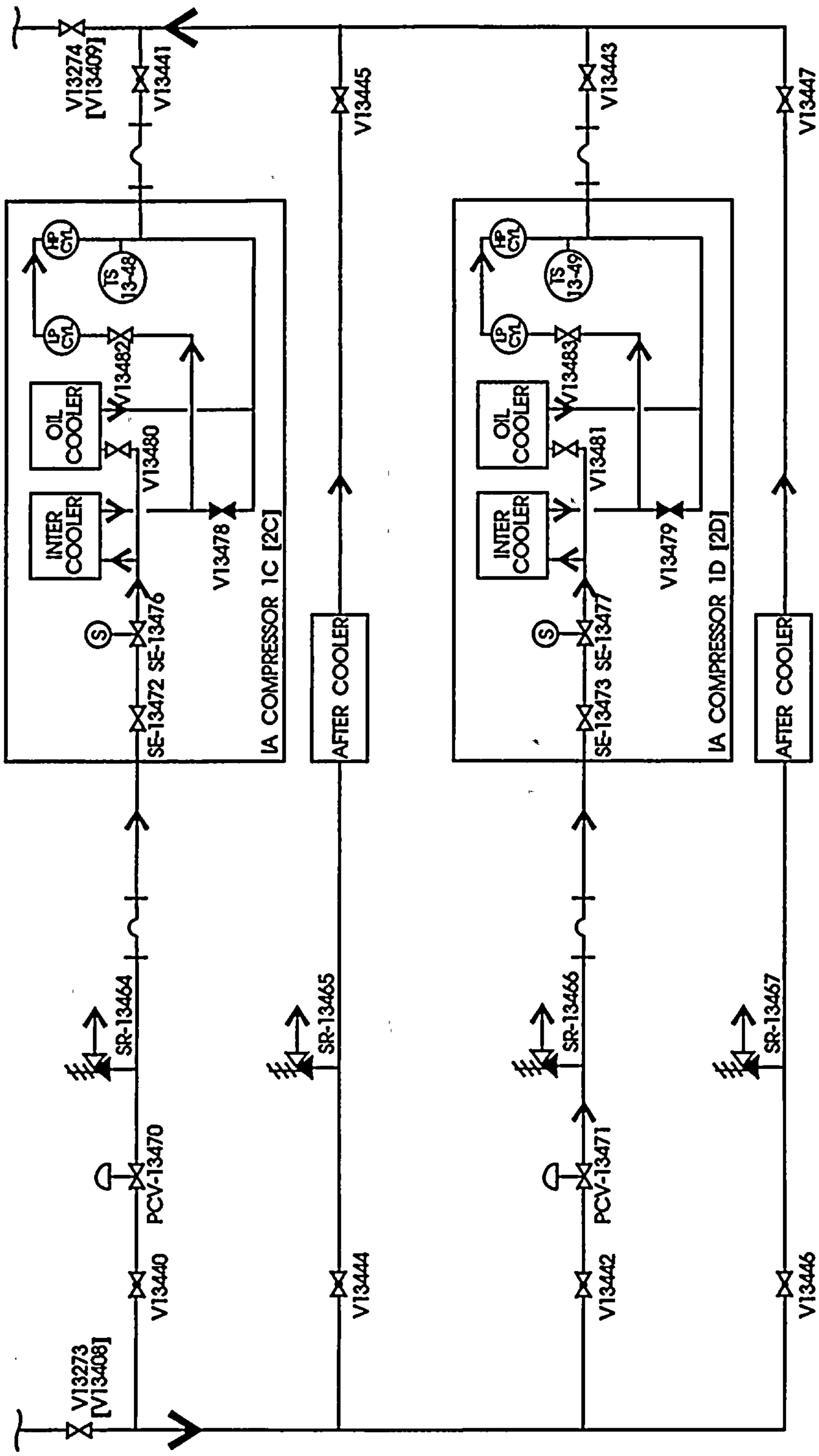


Figure B12.8 TCW Supply to Instrument Air Compressors Units 1 & 2



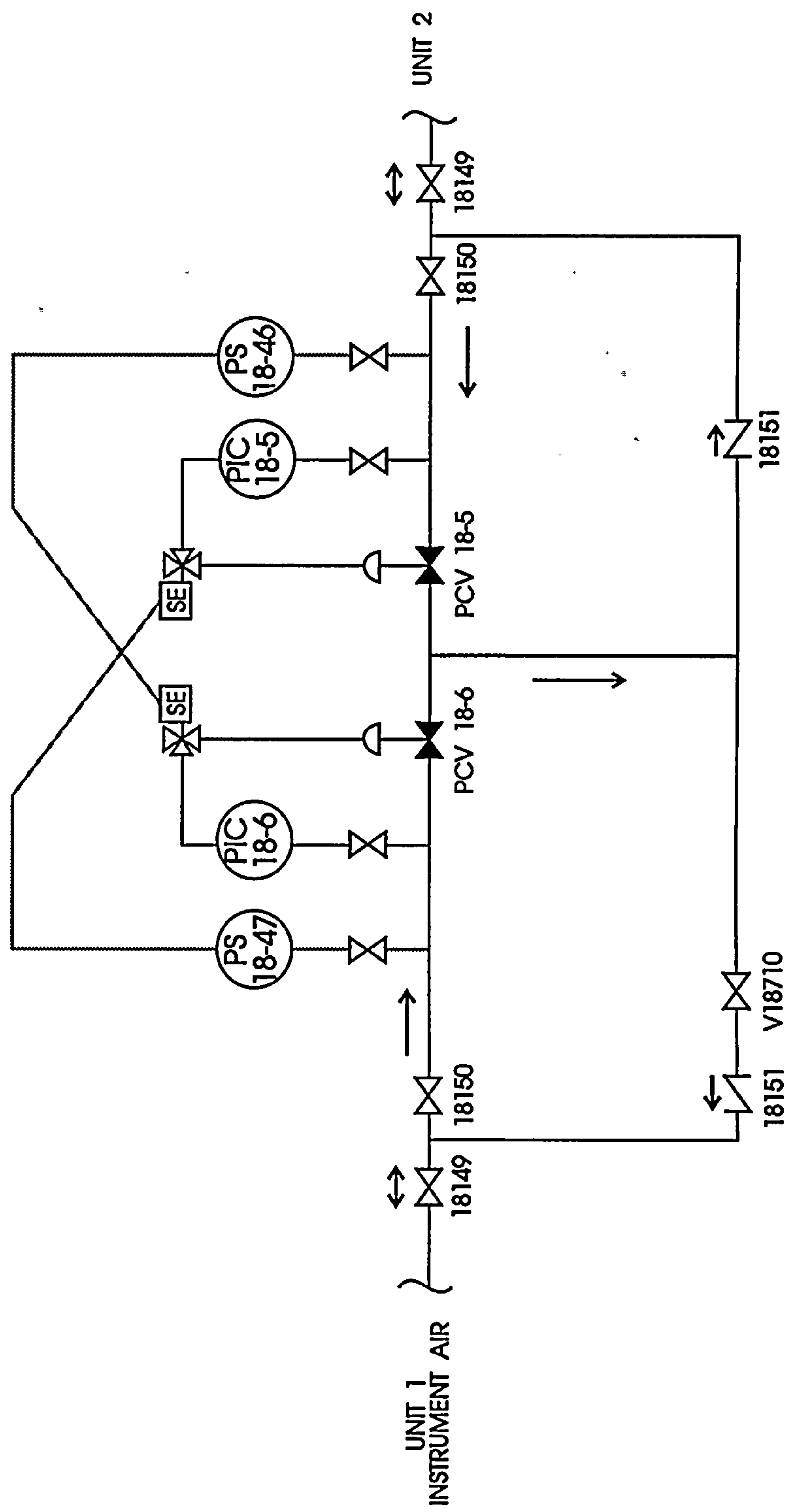


Figure B12.9 Instrument Air Cross Connection Units 1 & 2

B12.3 Success Criteria

The Instrument Air system modeled as discussed is successful if the following criteria are met:

a) One outside air compressor and associated path to its air receiver is available,

b) Containment isolation valves are open,

c) Valves in flowpath from operating air compressor to the component being supplied with IA remain open.

B12.4 Operation

B12.4.1 Normal Operation

The operation of the Instrument Air system is continuous and is required during all plant modes of operation (including normal transients). During normal Unit 1 operation, one inside and one outside containment air compressor are running as described below. [During normal Unit 2 operation, one outside containment air compressor is running.]

During normal operation, one outside containment instrument air compressor is in operation (1C [2C] or 1D [2D]) to maintain air receiver pressure between 110-120 psig, with the other compressor (1C [2C] or 1D [2D]) starting automatically if the instrument air receiver pressure falls below 105 psig. Compressors 1A [2A] and 1B [2B] remain off, available for use under abnormal operating conditions. One of the two 100% capacity desiccant air dryers will normally operate and the other will serve as a standby independent of which air compressor is operating.

During normal operation one Unit 1 inside containment compressor is normally operating with the other in standby, in order to maintain air receiver pressure between 100 and 105 psig.

B12.4.2 Accident Operation

B12.4.2.1 Loss of Air Compressor Cooling Water

Component Cooling Water: CCW supplies the compressor ring and seal water to the Unit 1 inside-RCB instrument air compressor. These compressors should be shut down upon loss of Unit 1 CCW or CCW 'N' header isolation to avoid damage to the seals and rotating elements.

Turbine Cooling Water: TCW cools the compressor cylinder jackets, oil coolers, intercoolers and aftercoolers of the C & D instrument air compressors. The Instrument Air Emergency Cooling system cooling fan and recirculation pump may be valved in and started, to allow operation of the A & B instrument air compressors, until normal TCW service is restored. Note that the C & D instrument air compressors can NOT be cooled with the Instrument Air Emergency Cooling system.

B12.4.2.2 Loss of Off-Site Power (LOOP)

In a Loss of Offsite Power Event, the instrument air compressors will stop. The A & B instrument air compressors may be reenergized from Diesel-powered, Safety-Related (non-essential) buses. The C & D instrument air compressors can not be reenergized until off-site power is restored.

After Instrument Air Emergency Cooling Water is established to the A & B instrument air compressors, the operator can reset the local handswitch and manually start the A & B instrument air compressors.

B12.4.2.3 Total Loss of AC Power / Station Blackout

In a Total Loss of AC Power Event, Instrument Air will be lost until AC power is restored or until some other supply of Instrument Air can be obtained (e.g., oil-free diesel air compressors).

B13 INTAKE COOLING WATER SYSTEM

B13.1 Function

The function of the Intake Cooling Water (ICW) system is to provide cooling water to the Component Cooling Water (CCW) system, the Turbine Cooling Water (TCW) system and the Steam Generator Open Blowdown Cooling (SGOBD) system during normal operation. During accident conditions, the ICW system only provides cooling water to the CCW system since the flow paths to the TCW and SGOBD systems are automatically isolated. The ICW system also functions to provide Lube Water for the Circulating Water pumps during normal operation. During accident conditions, ICW-supplied Lube Water is automatically isolated from the Circulating Water pumps.

B13.2 Configuration

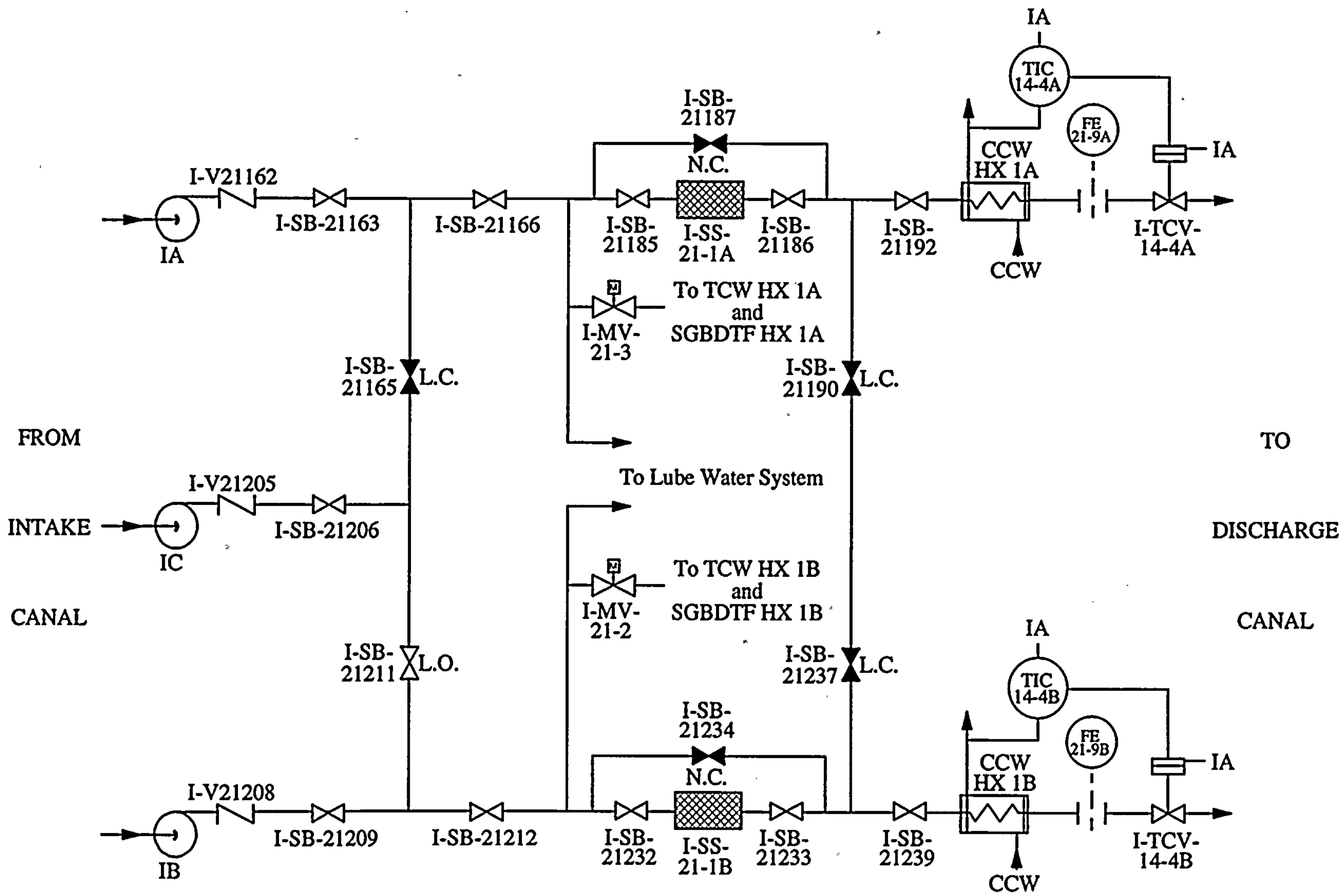
The ICW systems for Units 1 and 2 are shown on Figures B13.1 through B13.4. The system consists of three ICW pumps and associated piping and valves. ICW pumps take suction from the Intake Structure which provides a salt water supply from the Atlantic Ocean via the intake canal. This cooling water is discharged from the pumps through check valves to the supply headers.

The ICW system is divided into two redundant supply headers designated A and B. During normal operation, Pump 1A supplies the A header and Pump 1B supplies the B header. Pump 1A and 1B are powered from 4.16 kV buses 1A3 and 1B3 respectively. ICW Pump 1C, if available, can be aligned to either header A or B by realigning the pump discharge cross-tie isolation valves, SB-21165 and SB-21211. The pump discharge cross-tie valves are manual valves whose positions are administratively controlled. Likewise, the Pump 1C power supply, 4.16 kV Bus 1AB, can be aligned to 4.16 kV Bus 1A3 or 1B3. This allows for the flexibility of aligning Pump 1C to either header during failure, test or maintenance of Pump 1A or 1B. Normally the 1C [2C] pump is aligned to the B [A] header and the 4.16 kV Bus 1AB [2AB] is powered from 4.16kV Bus 1B3 [2A3].

For Unit 1, the idle 1C pump, if available, will start automatically to supply ICW to the header to which it is aligned following a Safety Injection Actuation Signal (SIAS), if the breaker to the pump that normally supplies that header is not shut or does not remain shut. [For Unit 2, Pump 2C will only start following a SIAS when aligned to the B header if the pump 2B breaker has been racked out and when aligned to the A header, if the pump 2A breaker is racked out or its selector switch is in ISOLATE.]

Each supply header is divided into two branches. One branch feeds the Component Cooling Water heat exchanger and the other branch feeds the TCW and SGOBD heat exchangers (non-essential header). Each non-essential header is automatically isolated on an SIAS by the automatic closure of MV-21-2 (Header B) and MV-21-3 (Header A).

Figure B13.1 Unit 1 ICW Normal Operation



ENCWICW1.DRW

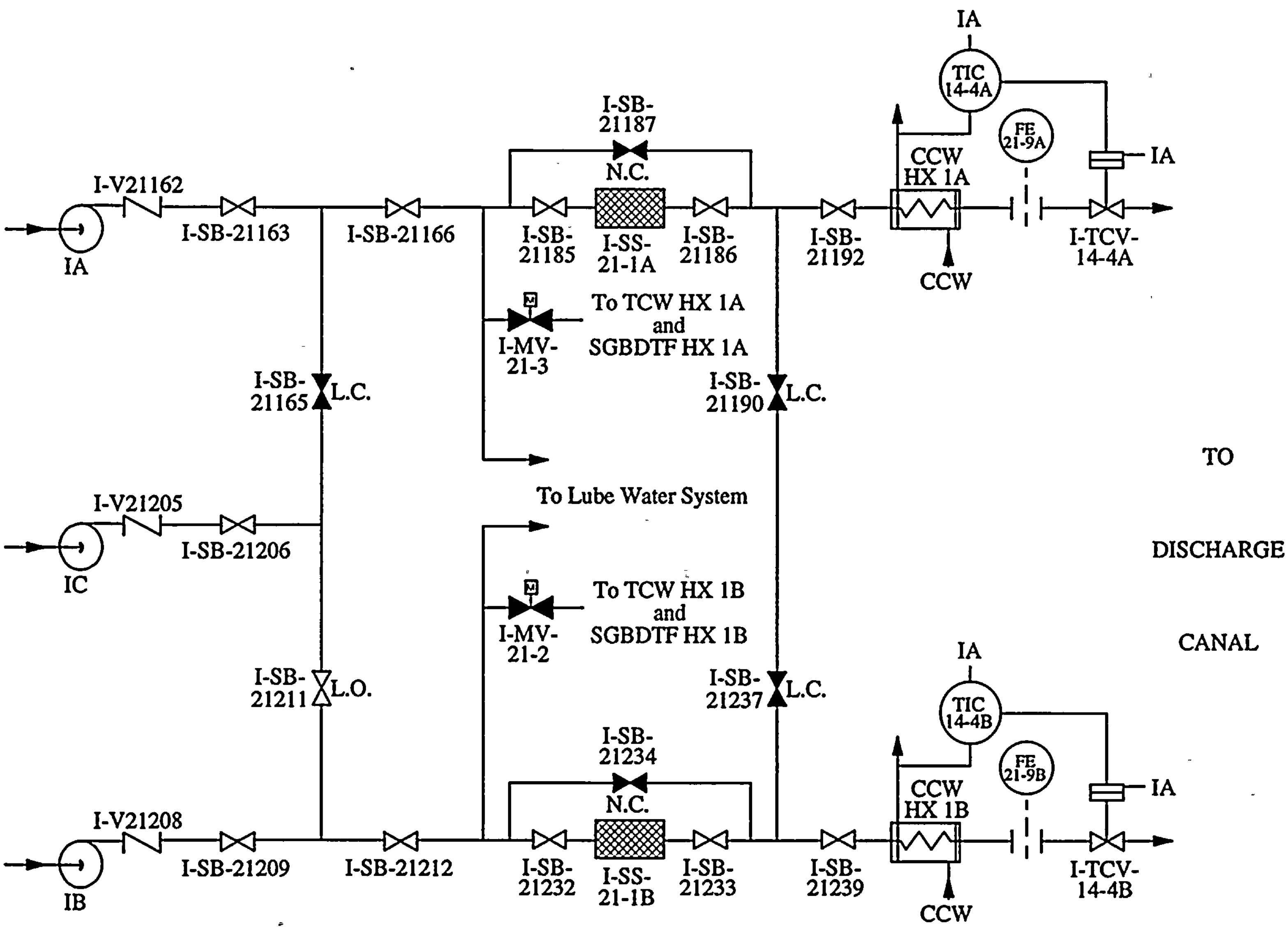
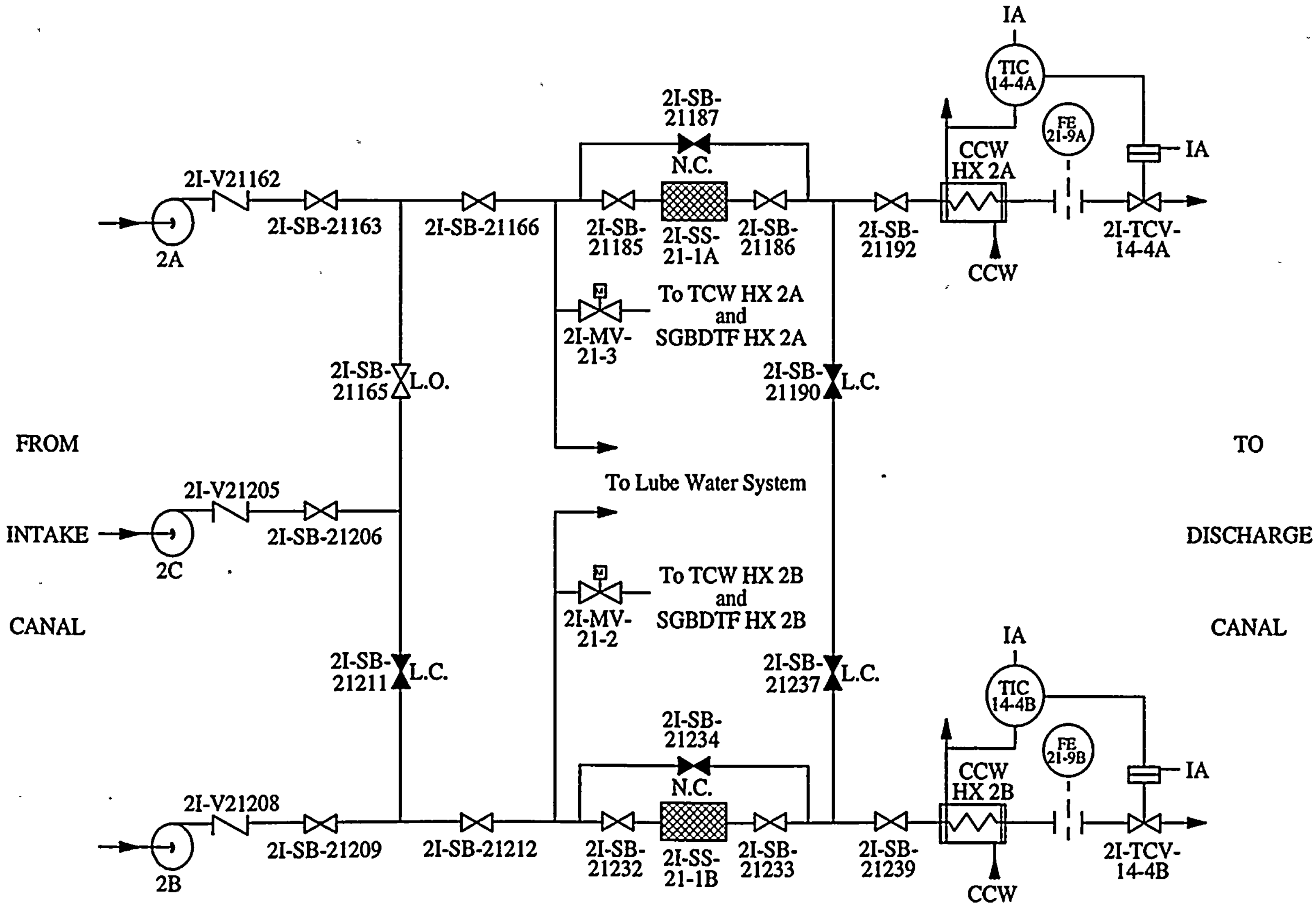


Figure B13.2 Unit 1 ICW Post-Accident Operation

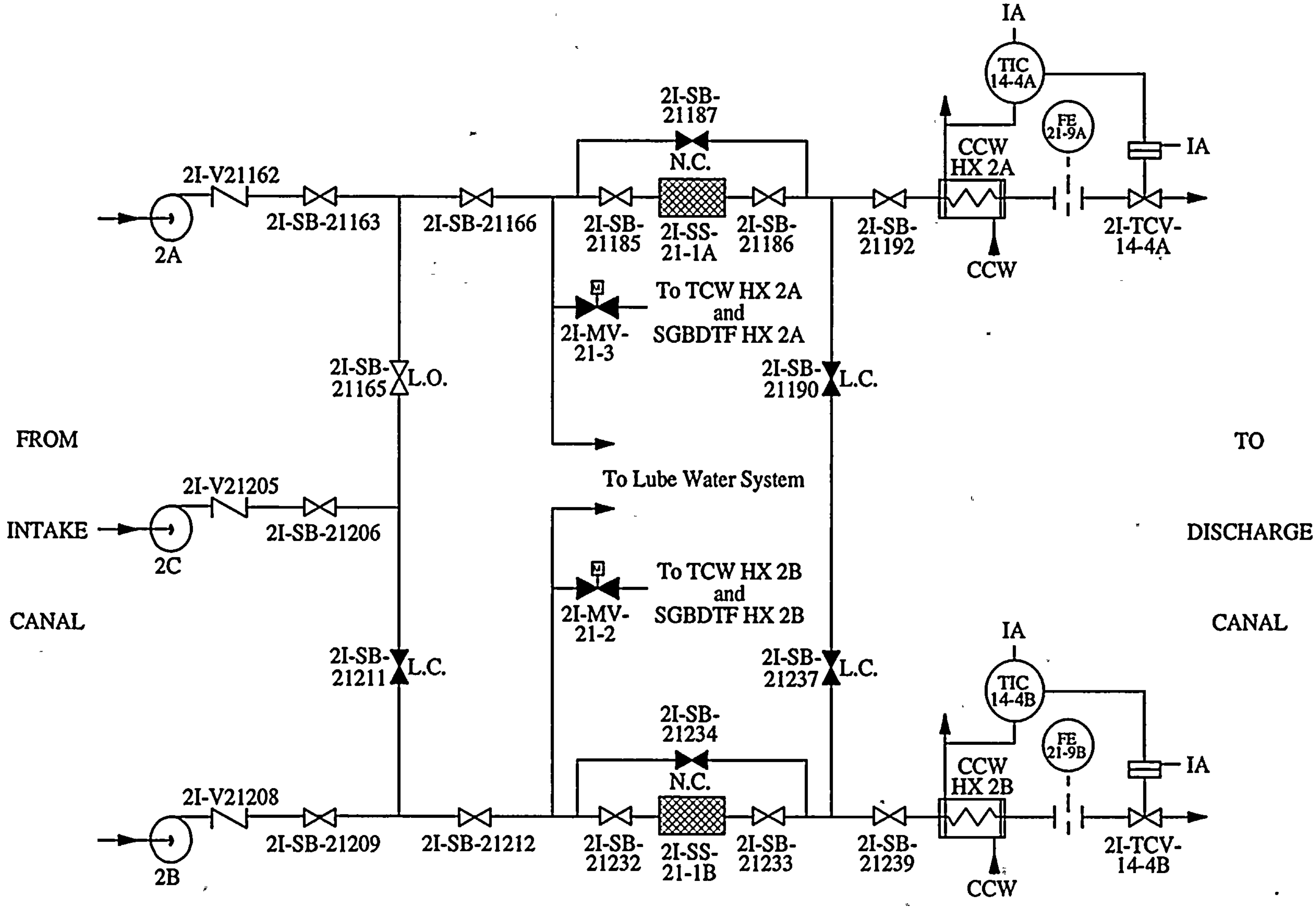
ENCWICW1A.DRW

Figure B13.3 Unit 2 ICW Normal Operation



F:\USERS\DMS\PSL\ICW\ICW2R1.DRW

Figure B13.4 Unit 2 ICW Post-Accident Operation



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Intake Cooling Water flows through basket strainers located at the inlet to the individual heat exchangers, passes through the tube side of the heat exchangers and flows to the discharge canal. The ICW flow rate through the CCW heat exchange is automatically controlled by temperature control valves (TCV-14-4A and TCV-14-4B) at the outlet of the individual heat exchangers. These valves are air-operated valves. Temperature indicating controllers (TIC-14-4A and TIC-14-4B) sense CCW temperature at the outlet of the CCW heat exchangers and send a pneumatic signal to the control valves to regulate ICW flow.

The ICW system provides the normal water supply for the Lube Water system, which lubricates the bearings of the Circulating Water pumps.

B13.3 Success Criteria

The ICW system must maintain cooling water to the CCW heat exchangers to remove the design basis accident heat load from the CCW system. The minimum requirement to mitigate the design basis accident is one ICW pump supplying cooling water to one CCW heat exchanger with ICW to the associated TCW/SGOBD systems isolated.

B13.4 Operation

B13.4.1 Normal Operation

The normal ICW system lineup is with pump A supplying the A header and pump B supplying the B header in a split (non-cross connected) configuration. Pump C is idle and, if available, is normally lined up to the B [A] train both electrically (4160 VAC Bus 1AB powered from 4160 VAC Bus 1B3 [2AB powered from 2A3]) and mechanically (I-SB-21211 locked open and I-SB-21165 locked closed [2I-SB-21165 locked open and 2I-SB-21211 locked closed]). Each header contains a CCW heat exchanger, a TCW heat exchanger and a Steam Generator Open Blow Down heat exchanger. Flow through the CCW heat exchanger is automatically controlled by a temperature control valve that throttles open or shut to maintain a constant CCW outlet temperature.

B13.4.2 Accident Operation

Following a Safety Injection Actuation Signal, MV-21-2 and MV-21-3 close to secure ICW flow to the non-essential headers, thereby removing cooling water to the TCW and SGOBD heat exchangers. ICW-supplied Lube Water to the Circulating Water pumps is also automatically secured.

B14 LOW PRESSURE SAFETY INJECTION SYSTEM

B14.1 Function

The Low Pressure Safety Injection system (LPSI) is a component of the Emergency Core Cooling system (ECCS) which also includes the Safety Injection Tanks (SIT) and the High Pressure Safety Injection system (HPSI). The function of the ECCS is to prevent significant alteration of core geometry, preclude fuel melting, limit the cladding metal-water reaction and remove the energy generated in the core during various postulated accident situations. The ECCS also functions to maintain the reactor subcritical by injecting borated water into the Reactor Coolant System (RCS) and to provide for long term cooling of the core by recirculating borated water from the containment sump. The LPSI system functions to automatically inject borated water from the Refueling Water Tank (RWT) into the RCS cold legs following a Safety Injection Actuation Signal (SIAS) and, in Unit 1, to recirculate water from the sump to the RCS hot legs when required to prevent boron precipitation. The LPSI system is also used to provide long term shutdown cooling (SDC) by circulating reactor coolant from the RCS hot legs through the SDC heat exchangers and returning it to the RCS cold legs.

B14.2 Configuration

The standby configurations of the LPSI systems for Units 1 and 2 are shown on Figures B14.1 and B14.2. The system consists of two LPSI pumps and associated piping and valves. The LPSI system receives water from the Refueling Water Tank (RWT) or the containment sump and supplies water to each RCS cold leg through safety injection lines shared with the Safety Injection Tanks and the HPSI system. The RWT and containment sump are modeled as part of the HPSI system. In Unit 1, the LPSI system is arranged with two parallel pumps supplying a common header which supplies flow to each of the RCS cold legs. In Unit 2, the LPSI system is arranged as two completely separate redundant trains, each supplying flow to its associated RCS cold legs.

While in standby and during the Post-Accident Injection phase, the pumps are aligned to take suction from the RWT. The RWT contains an inventory of borated water which is a common supply for the HPSI, LPSI and Containment Spray (CS) systems. During the Post-Accident Recirculation phase, the ECCS pump suctions are automatically switched to the containment sump. In both cases the suction line for the ECCS Train A pumps are separate from the suction line for the Train B pumps beyond the RWT suction MOVs.

The configuration of the LPSI system during the post-accident injection phase is shown on Figures B14.3 and B14.4. In Unit 1 following an SIAS, each LPSI pump automatically starts and supplies the Low Pressure header. [In Unit 2 following an SIAS, LPSI pump 2A automatically starts and supplies LP header A and LPSI pump 2B automatically starts and supplies LP header B.] LPSI pump 1A [2A] and 1B [2B] are powered from 4.16 kV buses 1A3 [2A3] and 1B3 [2B3] respectively. Each Unit 1 pump has a portion of its discharge cooled by CCW and circulated through the pump seal. The Unit 2 pumps do not require CCW seal cooling.

Figure B14.1 Unit 1 LPSI System Standby Configuration

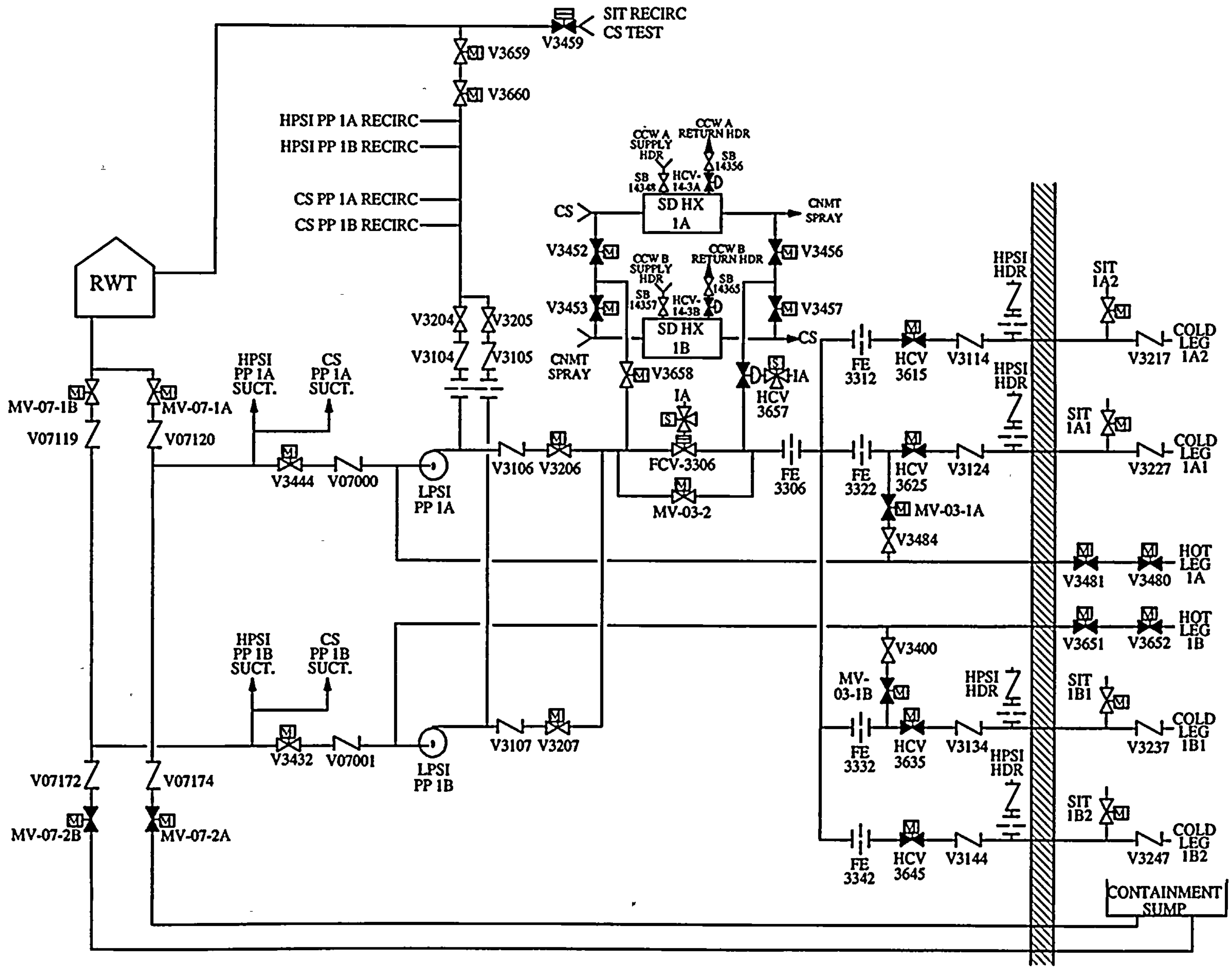

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Figure B14.2 Unit 2 LPSI System Standby Configuration

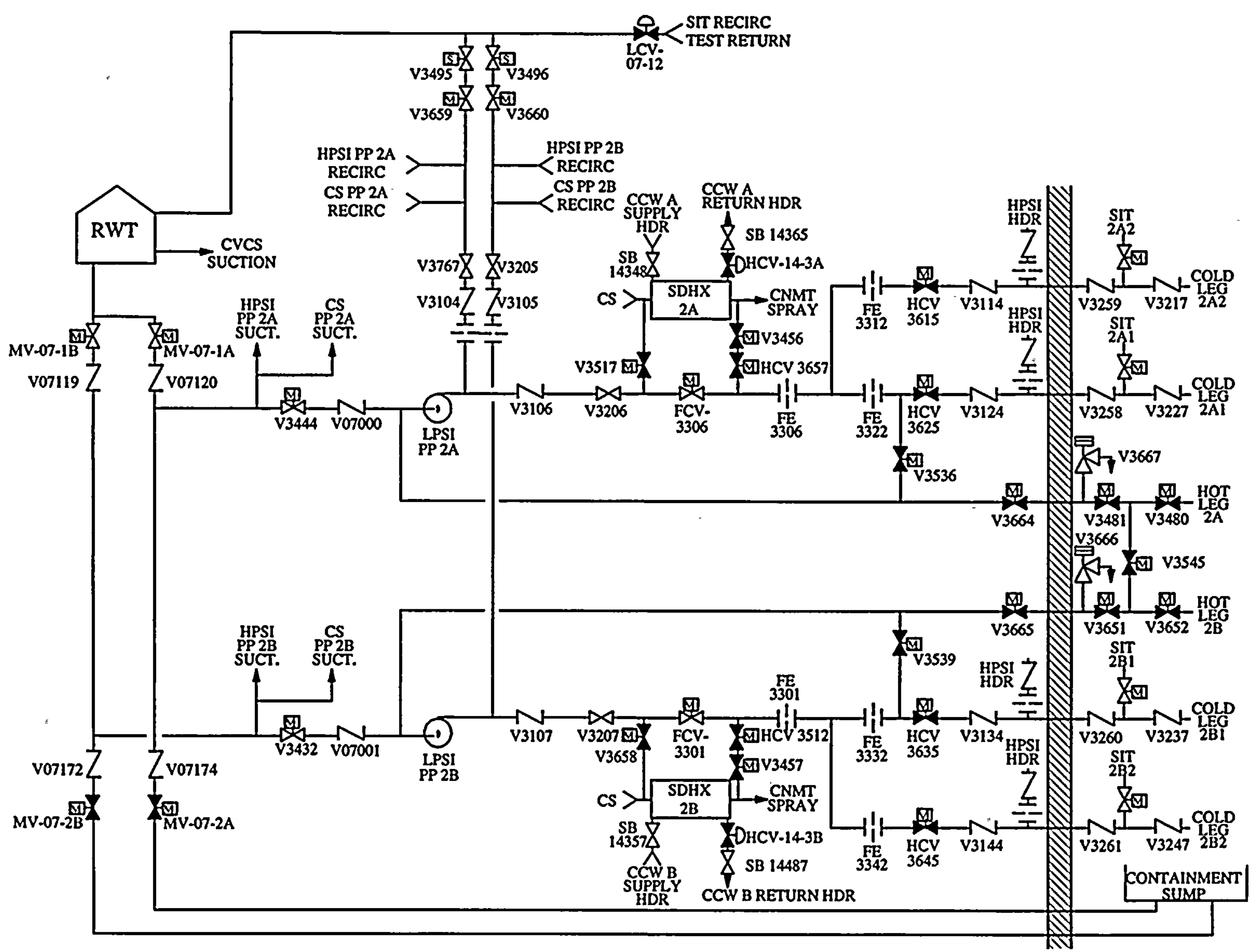

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Figure B14.3 Unit 1 LPSI System Injection Phase

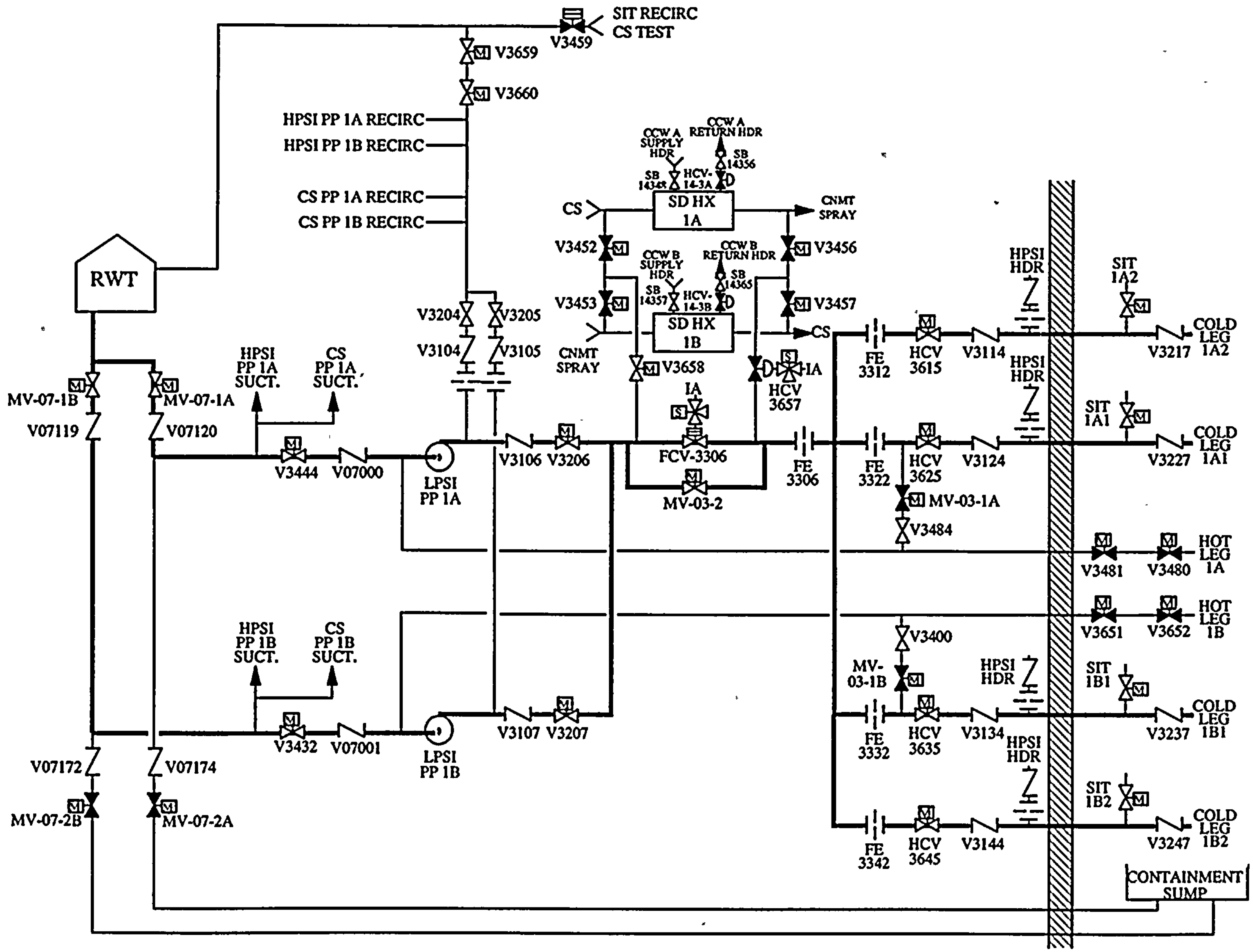

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Figure B14.4 Unit 2 LPSI System Injection Phase

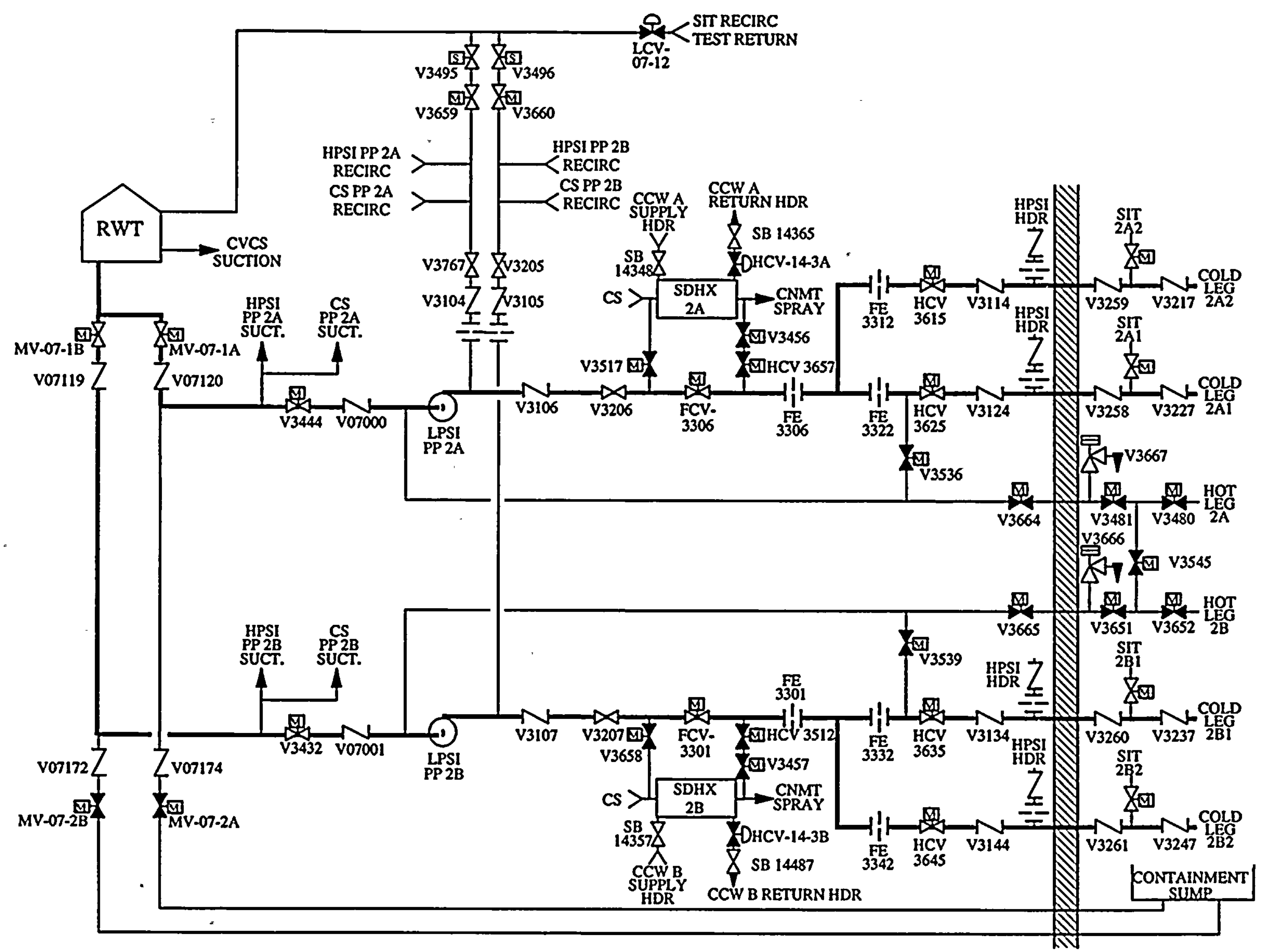

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Figure B14.5 Unit 1 LPSI System
Hot Leg Recirculation Using LPSI Pump 1A

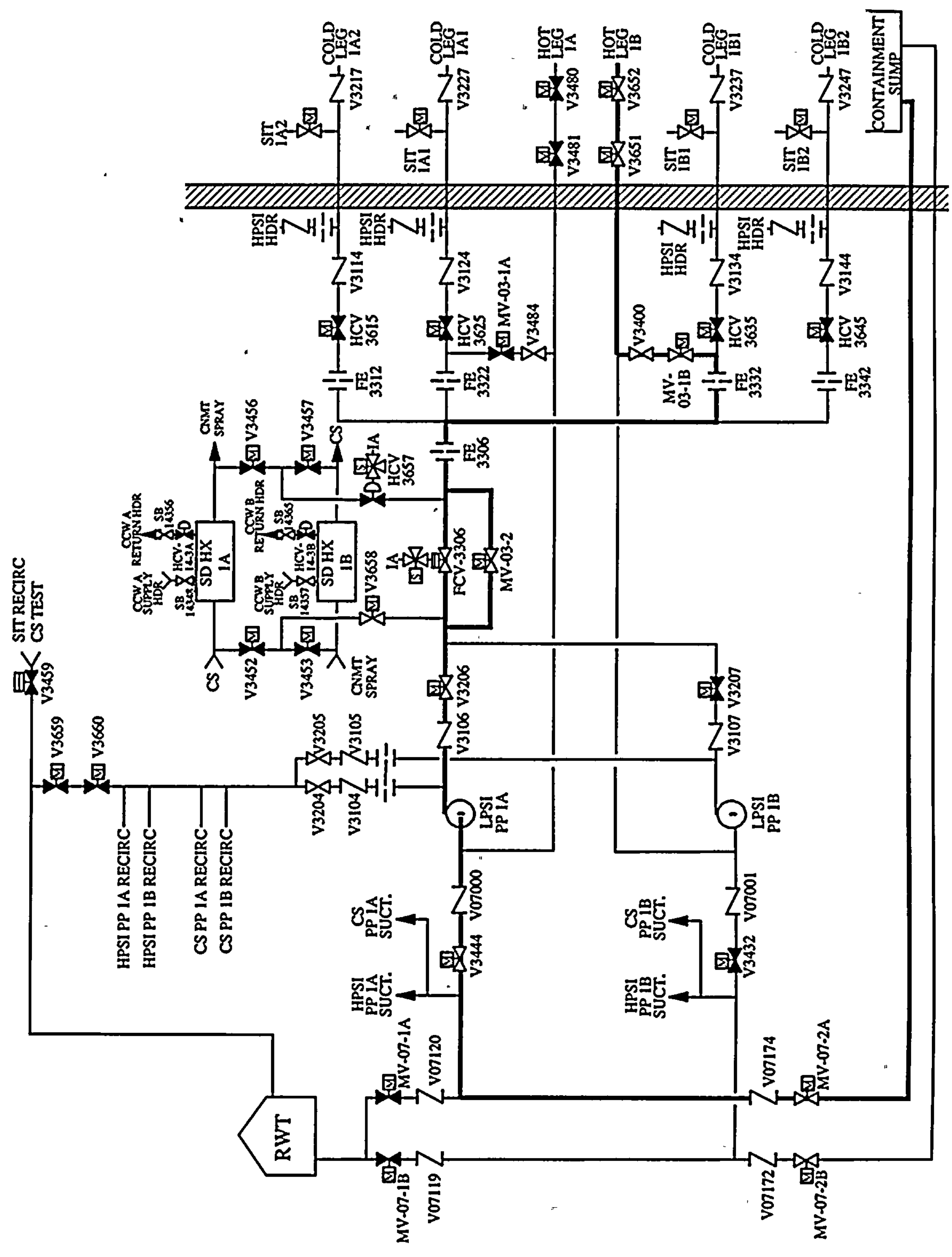

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Figure B14.6 Unit 1 LPSI System
Hot Leg Recirculation Using LPSI Pump 1B

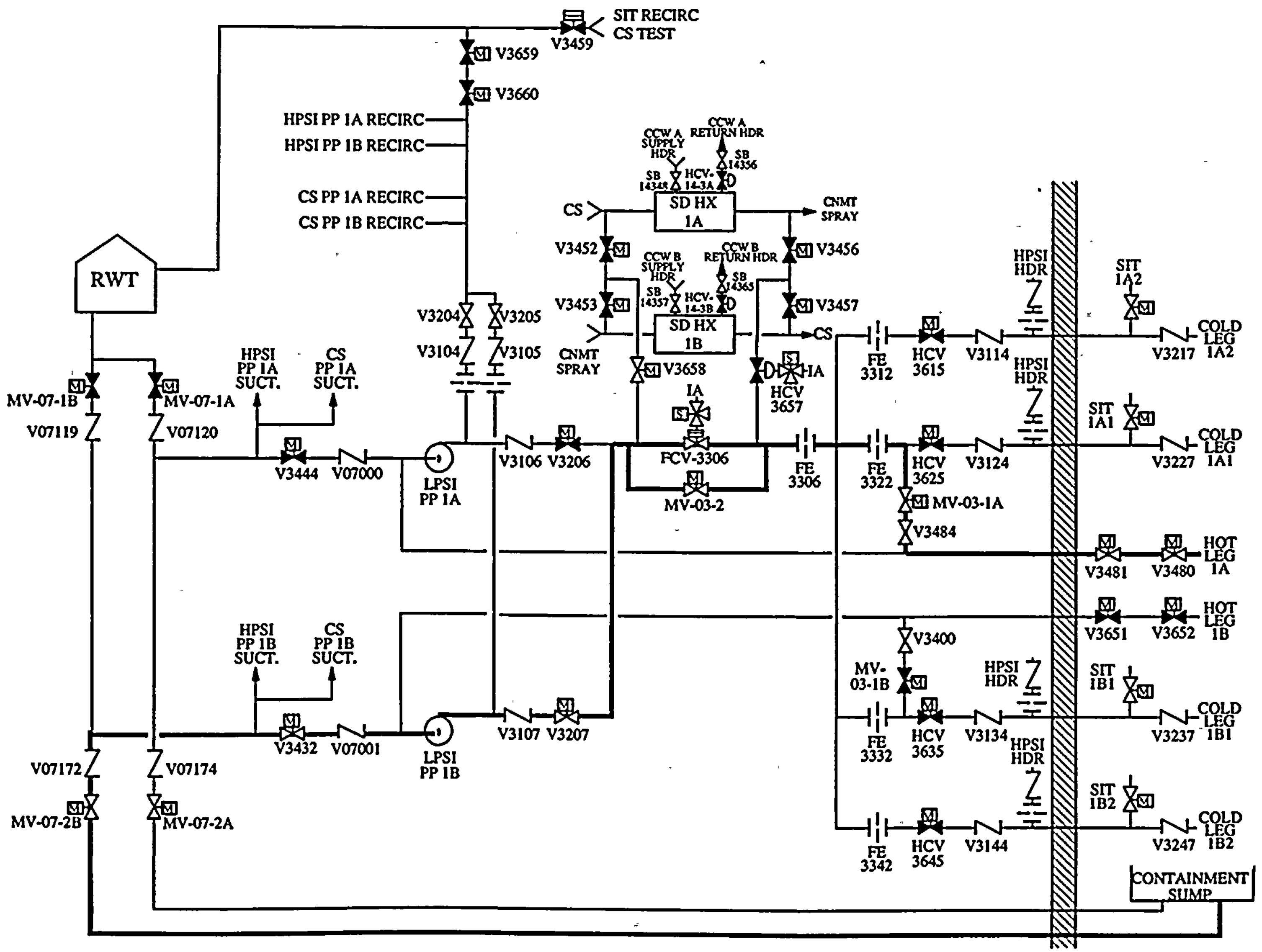

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Figure B14.7 Unit 1 LPSI System Shutdown Cooling

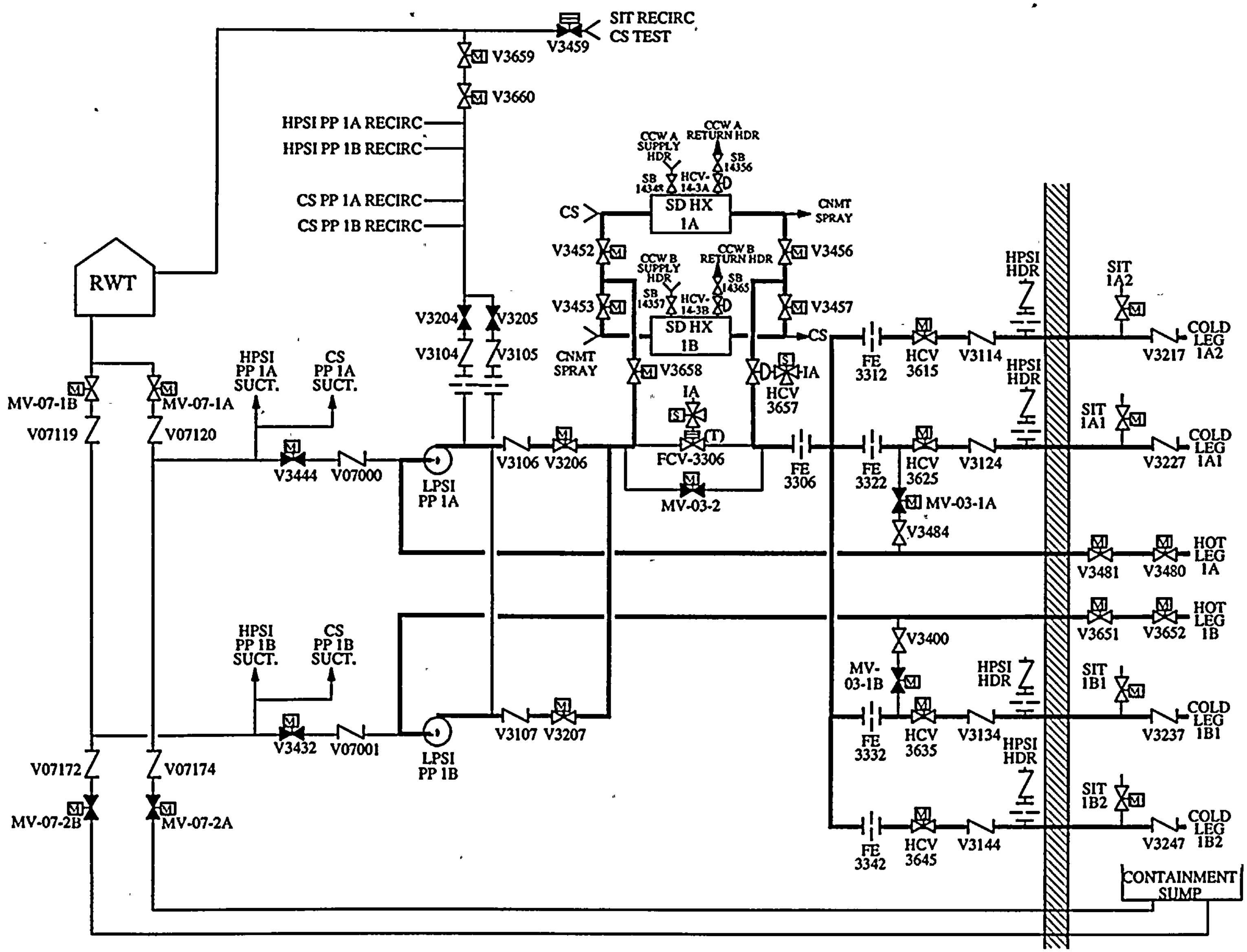

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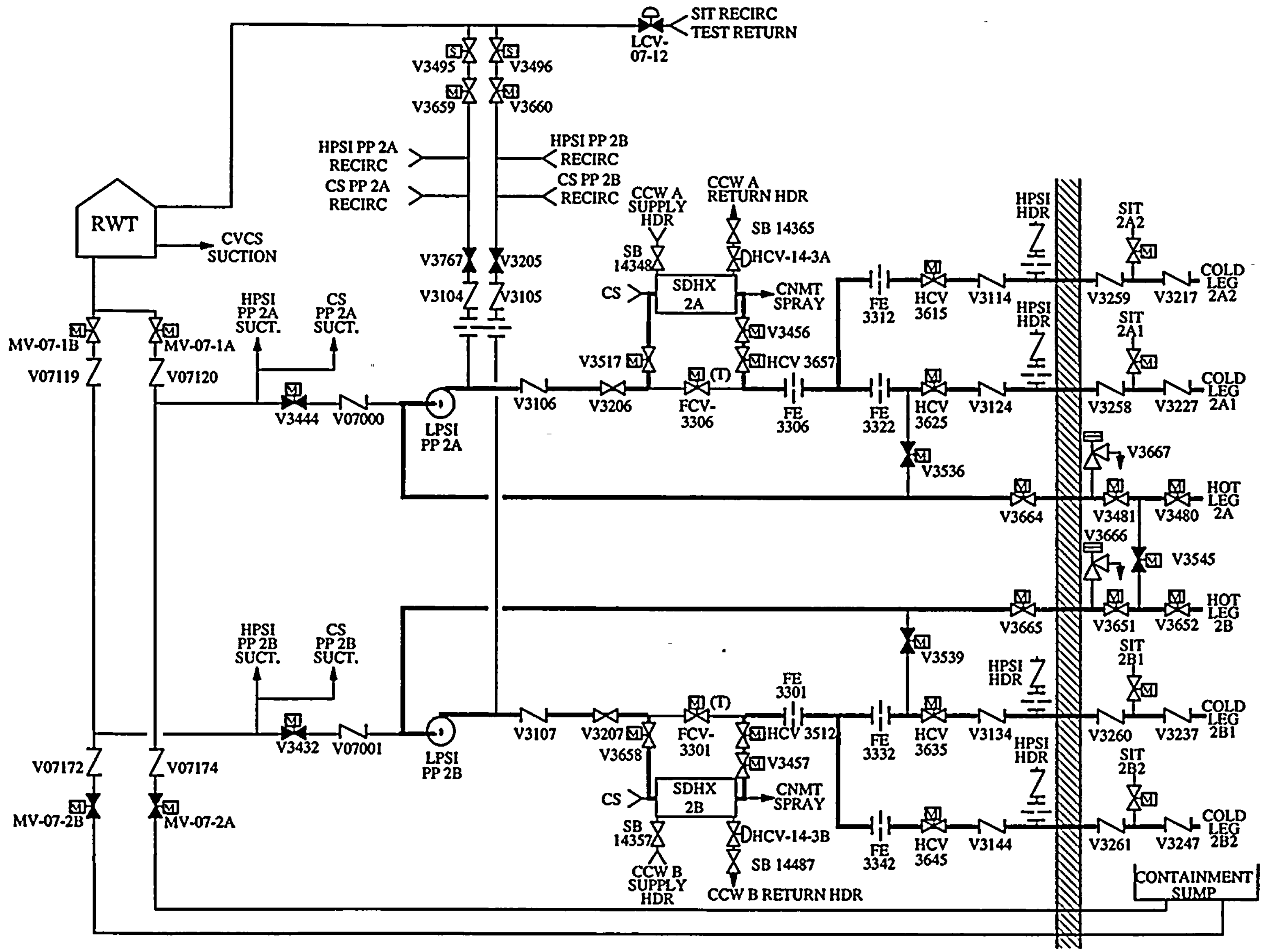
Figure B14.8 Unit 2 LPSI System Shutdown Cooling



E:\LPSI\LPSI7.DRW

Each pump is protected from operating for extended periods at shutoff head by a minimum recirculation line. In Unit 1, the mini-recirc lines from each pump combine with the other HPSI, LPSI and CS pumps to form a common return line to the RWT. The common line contains two normally open motor operated valves which will close following a RAS. [In Unit 2, each ECCS pump's discharge from the respective train combine to form a common minimum recirculation header (i.e. the HPSI, LPSI and CS A pumps' discharges combine into a single line and the B pumps' discharges combine into a separate line). Each minimum recirculation line has two valves in series: a solenoid valve (V3495 and V3496) and a motor operated valve (V3659 and V3660). These normally open valves also receive a closing signal following a RAS.]

Water flows from the suction line through the pump suction motor operated isolation valve, the pump suction check valve, the pump, the pump discharge check valve, and the pump discharge motor operated [manual] isolation valve. In Unit 1, the pump discharges form a common header which contains an air operated flow control valve (FCV-3306) and its bypass (MV-03-2). The LPSI header then splits into 4 lines, each containing a normally closed motor operated isolation valve. [In Unit 2, each pump discharge supplies a separate header containing a motor operated flow control valve (FCV-3306 for header A and FCV-3301 for header B). Each of these headers then splits into 2 lines, each containing a normally closed motor operated isolation valve. These valves, as well as the Unit 1 valves, receive an open signal upon SIAS.] LPSI flow passes through a check valve and then combines with the associated HPSI discharge and SIT outlet. [In Unit 2, an additional check valve separates the common HPSI/LPSI discharge from the SIT outlet.] The combined flow enters the individual RCS cold legs through the SI check valves.

In Unit 1, the LPSI system also functions to recirculate water from the containment sump to the RCS hot legs to prevent boron precipitation. The configuration of the LPSI system during this mode of operation is shown on Figures B14.5 and B14.6. This function can be accomplished by using either LPSI pump 1A or 1B. When using LPSI pump 1A (1B), the pump 1B (1A) suction and discharge valves are closed and the pump draws a suction directly from the containment sump. The pump discharge is directed to the RCS hot leg 1B (1A) Shutdown Cooling suction line via the B (A) warmup line. All the LPSI injection valves are closed.

The LPSI system also functions to provide long term Shutdown Cooling (SDC). The configuration of the LPSI system during SDC is shown on Figures B14.7 and B14.8. In the SDC mode of operation, each pump takes a suction from its associated RCS hot leg through normally closed motor operated hot leg isolation valves located inside containment. The LPSI pump suction isolation valves and the pump minimum recirculation manual isolation valves are closed during SDC. [Unit 2 is provided with an additional normally closed motor operated isolation valve in each hot leg suction line located outside containment. Additionally, Unit 2 has a hot leg suction cross connect line with a normally shut motor operated isolation valve and a large capacity relief valve on each hot leg suction line inside containment.]

In Unit 1 during SDC, the discharge of the LPSI pumps is directed to the Shutdown Cooling heat exchangers by throttling shut the flow control valve, FCV-3306, to control overall cooling flow rate and shutting the flow control valve bypass valve, MV-03-2. The SDC flow passes through the normally open, common heat exchanger inlet valve, V3658. This valve is a motor operated valve with the motor disconnected. The SDC flow can then be directed to either or both of the SDC heat exchangers through the normally closed, motor operated heat exchanger inlet isolation valves, V3452 and V3453, and outlet isolation valves, V3656 and V3657. The piping is then joined into a common return header containing a normally closed, air operated, hand control valve, HCV-3657, which is throttled to control the RCS cooldown rate. The SDC cooling flow then returns to the RCS via the Low Pressure cold leg injection lines.

In Unit 2 during shutdown cooling, the flow path is similar to Unit 1 except that each train's piping to the SDC heat exchangers and back to the RCS cold legs is completely separate. The discharge of each pump is directed to its associated SDC heat exchanger by throttling shut the flow control valve (FCV-3306, FCV-3301), opening the motor operated inlet valve (V3517, V3658), opening the motor operated outlet valve (V3456, V3457) and throttling open the motor operated hand control valve (HCV-3657, HCV-3512). Flow is then returned to the RCS loop A or B cold legs through the normal Low Pressure header A or B injection path. When both SDC loops are used simultaneously or when power is available to only one electric train, the hot leg suction cross connect valve is opened.

Component Cooling Water is supplied to each of the SDC heat exchangers from its associated CCW essential header. Each SDC heat exchanger has manual, normally open, CCW inlet and outlet valves. CCW flow through the heat exchanger is automatically initiated following an SIAS or manually, by opening the air operated, normally shut, SDC heat exchanger CCW outlet valves (HCV-14-3A for heat exchanger A and HCV-14-3B for heat exchanger B).

Prior to initiation of shutdown cooling, the piping and heat exchangers are warmed by circulating water with the LPSI pumps. A warmup line is provided between the hot leg suction line downstream of the hot leg isolation valves and the cold leg injection lines upstream of the injection line isolation valves. A normally closed motor operated valve in the warmup line is provided to allow for this circulation path. With the flow path to the SDC heat exchangers established, the cold leg injection valves shut, the hot leg suction isolation valves shut and the LPSI pump suction valves open, the LPSI pumps are started, providing the heat to warmup the lines.

B14.3 Success Criteria

B14.3.1 <u>Injection Phase:</u>

The success criteria for the Injection phase of all accident sequences is one-out-of-two LPSI pumps supplying flow from the RWT to one out of four intact cold legs.

B14.3.2 <u>Hot Leg Recirculation Phase (Unit 1 only):</u>

Although discussed here, failure of hot leg recirculation is not considered a core damage sequence due to the low boron concentrations. If assumed to be required, the success criteria for the Hot Leg Recirculation phase would be one-out-of-two LPSI pumps supplying flow from the sump to one-out-of-two hot legs.

B14.3.3 Shutdown Cooling:

The success criteria for Shutdown Cooling in Unit 1 is one of two LPSI pumps supplying flow to one of four cold legs via one of two SDC heat exchangers. [For Unit 2, the success criteria is one of two SDC trains; that is, one LPSI pump supplying flow to one of two of its associated cold legs via its associated SDC heat exchanger.]

B14.4 Operation

B14.4.1 Normal Operation

During normal operation at power, the LPSI system is configured to automatically inject borated water into the RCS cold legs following an SIAS. Specifically, the LPSI pumps are idle with CCW supplied to them (for Unit 1 only), the RWT outlet motor operated valves are open, the minimum recirculation line isolation valves are open (and power is removed from them in Unit 1), the sump outlet motor operated isolation valves are closed, the individual loop LPSI injection valves are closed, the flow control valve[s] are locked open, the flow control bypass valve for Unit 1 is locked open and the SDC heat exchangers and hot leg suction lines are isolated. CCW to the SDC heat exchangers is aligned such that the outlet valves will automatically open to supply CCW following an SIAS.

B14.4.2 Accident Operation

Following a Safety Injection Actuation Signal, the LPSI pumps will automatically start and the LPSI header injection valves will automatically open to provide a flow path from the RWT to the RCS cold legs. The CCW outlet valves from the SDC heat exchanger automatically open following an SIAS to provide CCW flow. The SDC heat exchangers are used to the cool Containment Spray discharge for containment temperature and pressure control.

When sufficient water has been transferred from the RWT to the containment sump, an automatic Recirculation Actuation Signal (RAS) will be generated. In Unit 1, prior to receiving a RAS, the operators manually restore power to the minimum recirculation isolation valves. The RAS will cause the sump outlet valves to open and the RWT outlet valves to shut, switching ECCS pump suction to the sump. The RAS will also cause the ECCS pump minimum recirculation valves to close and the LPSI pumps to stop.

In Unit 1, for those LOCA break sizes for which Shutdown Cooling cannot be entered prior to 10 hours following the accident, simultaneous Hot Leg/Cold Leg Recirculation is manually initiated. [Simultaneous Hot Leg/Cold Leg Recirculation utilizes the HPSI system in Unit 2.] The primary means of hot leg recirculation is via the LPSI pumps, but hot leg recirculation can also be accomplished by the HPSI pumps through the CVCS system and on to the Auxiliary Spray valves or by the Containment Spray pumps via the SDC hot leg suction lines. Hot leg recirculation is initiated by securing and isolating one LPSI pump while the remaining LPSI pump takes a suction from the sump and discharges to the opposite train hot leg suction line via the associated warmup line isolation valve.

B14.4.3 Shutdown Cooling Operation

Shutdown cooling is the normal method of cooldown below 325°F and long term decay heat removal. It is also utilized after certain accidents following RCS depressurization to remove decay heat (e.g. Small-Small LOCA). In either case, the same procedure for SDC initiation is followed. With the hot leg suction isolation valves shut, the LPSI injection valves shut, the LPSI pump suction valves open and the SDC heat exchanger isolation valves open, the LPSI pumps are started and the warmup line isolation valves are opened to recirculate the water in the SDC lines to warm up the piping and components. When the SDC lines have been warmed and Pressurizer pressure verified to be below 265 psia [275 psia] and temperature below 325°F, SDC can be initiated. The warmup line isolation valves are shut, the LPSI pump minimum recirculation manual valves are shut, the LPSI pump suction valves are shut, the hot leg suction isolation valves are opened and the LPSI injection valves are opened. Total SDC flow is automatically [manually] controlled by the flow control valve[s] while flow to the SDC heat exchangers (and thereby the RCS cooldown rate) is manually controlled by the SDC hand control valve[s]. CCW flow is likewise initiated by opening the SDC heat exchanger CCW outlet valves, HCV-14-3A and HCV-14-3B.

B15 POWER CONVERSION SYSTEM

B15.1 Function

For the purposes of the St. Lucie PRA, the Power Conversion System (PCS) is considered to be a single system consisting of the major elements of the Main Steam, Condensate, and Main Feedwater systems. Only those components important to the PRA are included. In addition to its normal function of supplying the motive force for the generation of electricity, the PCS has the following functions of importance to the PRA:

a) The primary function of the main steam system is to convey steam generated in the two Steam Generators through the containment vessel to the turbogenerator for power generation. The main steam portion of PCS also provides steam to the steam-driven auxiliary feedwater pump when main feedwater is not available.

b) The primary function of the condensate and feedwater system is to supply heated condensate to the Steam Generators for steam production during normal and off-normal operations.

c) The PCS provides the capability for manual isolation of a faulted Steam Generator following Steam Generator Tube Rupture (SGTR) and Main Steam Line Break (MSLB) events and isolates non-safety related portions of the PCS.

d) The PCS prevents uncontrolled blowdown of both Steam Generators in the event of a MSLB.

e) The PCS provides overpressure protection of the Steam Generators and main steam lines.

f) The PCS dissipates heat generated in the reactor coolant system following normal and abnormal plant transients.

g) The PCS permits 45% load rejection without turbine or reactor trip.

B15.2 Configuration

The PCS configuration is shown in Figures B15.1 through B15.4. The system includes the main steam, feedwater, and condensate systems. Also included in the PCS model are the steam generator blowdown and sample lines for isolation of a Steam Generator following a SGTR event. Turbine cooling water outlet and inlet valves to the main feedwater and condensate pumps are also included within the PCS model, as well as the circulating water pumps, as means to provide cooling to the condenser.

The PCS is designed to accept step load increases of 10%, and ramp changes of 5% per minute, within the load range of 15% and 100% without a reactor trip. Load rejections of 45% or less can be accommodated through the use of the steam dump system.

The PCS consists of the following main components:

a) Atmospheric Dump Valves (2 [4]; 1 [2] per Steam Generator)

b) Main Steam Safety Valves (16; 8 per Steam Generator)

c) Main Steam Isolation Valves (2; 1 per Steam Generator)

d) Steam Dump and Bypass Valves (5)

e) Main Condenser (1A, 1B [2A, 2B])

f) Condensate Pumps (3)

g) Main Feedwater Pumps (2)

h) Main Feedwater Isolation Valves (2 [4]; 1 [2] per Steam Generator)

i) Main Feedwater Regulating Valves (2; 1 per Steam Generator)

j) 5% Feedwater Regulating Bypass Valves (2; 1 per Steam Generator)

k) Feedwater Pump Discharge Valves (2; 1 per pump)

l) Feedwater Recirculation Valves (2; 1 per pump)

m) SG Blowdown Isolation and Sample Valves (4 of each; 2 sets per Steam Generator)

n) Turbine Cooling Water Inlet (4)/Outlet Valves (8)

o) Circulating Water Pumps (4; 2 per Condenser)

All components listed above are located outside the containment with the exception of two SG Blowdown Isolation Valves (1 per SG). The steam supply piping for the Auxiliary Feedwater Pump is located upstream of the Main Steam Isolation Valves.

Atmospheric Dump Valves (HCV-08-2A, 2B [MV-08-18A,B, MV-08-19A,B])

The Unit 1 Atmospheric Dump Valves (ADVs) are 90° angle plug-type valves with reverse-acting air operators which discharge to the atmosphere. Each Steam Generator is equipped with one ADV. The ADVs are designed to provide decay heat removal with sufficient margin to start a

75°F/hour cooldown to 325°F within 3.5 hours following a shutdown. The Unit 1 ADVs are normally operated from the Control Room.

[The Unit 2 ADVs are 90° angle plug-type motor-operated valves. The Atmospheric Dump Valve system consists of four Drag-Type valves (two per Steam Generator). Using two valves, the plant can be cooled down to 350°F in about 3.5 hours. The Atmospheric Dump Valve system has remote-manual capability from the control room in order to bring the plant from hot standby conditions to Shutdown Cooling system entry temperature.]

The ADVs in both units are kept in manual.

Main Steam Safety Valves (V-8201 through V-8216)

Eight Main Steam Safety Valves (MSSVs) tap off each main steam line downstream of the Atmospheric Dump Valves, but upstream of the MSIVs. The Main Steam Safety Valves protect the Steam Generators and main steam piping from overpressurization. The 16 valves can pass steam flow at a rate equivalent to an NSSS power level of 2700 MW. The capacity of the safety valves ensures overpressure protection even without the availability of the Atmospheric Dump Valves and the bypass control system valves. The reason for having eight MSSVs per main steam header rather than one or two is so that, if one valve fails to reseat, the resultant cooldown of the reactor coolant can be maintained within safe limits.

Main Steam Safety Valve characteristics ensure design overpressure protection. In each header, four MSSVs are set to relieve pressure at 985 psig and the other four valves are set at 1025 psig. An accumulation specification of 3% means the valves will be fully open at 103% of the setpoint. With a blowdown specification of 4%, the valves are designed to reseat 4% below the lifting setpoint. The 985 psig valves are designed to relieve saturated steam to the atmosphere.

Main Steam Isolation Valves (HCV-08-1A, 1B)

Each main steam line is equipped with a Main Steam Isolation Valve (MSIV). The MSIV is an air-operated, stop valve [Y-type bi-directional balanced stop valve] (butt welded to a check valve in Unit 1 only) positioned such that steam flow tends to close and seat the valve. These valves are designated as safety-related and must close within 6 [6.75] seconds as specified in the Technical Specifications. The MSIVs will close automatically on a Main Steam Isolation Signal (MSIS) and function to isolate the main steam lines in order to mitigate the consequences of a main steam line break (MSLB). In the event of a Steam Generator tube rupture (SGTR), the operator must manually close the MSIVs.

The MSIV closes to prevent steam flow from the Steam Generator to the turbine inlet manifold and to prevent backflow from the intact Steam Generator to the affected Steam Generator if the SG pressure drops below the turbine inlet manifold pressure. In Unit 1, the check valve portion of the MSIV assembly prevents backflow from the intact SG to the affected SG. [On Unit 2, the MSIVs are de-energized to prevent backflow even with full differential pressure across the valve].

The Unit 1 MSIV electro-pneumatic control system, which functions to open or close the MSIV, differs from the Unit 2 control system. For the Unit 1 control system actuation air is made available from the instrument air supply with backup air coming from two banks of high pressure air bottles. The Unit 2 control system is wholly dependent on availability of instrument air. The Unit 1 and 2 MSIV air actuation systems also have accumulators for each MSIV which can keep the MSIVs open for eight hours after a loss of instrument air. Note that the accumulators and backup air supply only exist to prevent spurious closure of the MSIVs on a momentary loss of Instrument Air. There is no safety function which requires the MSIVs to remain open.

The MSIVs in both units will fail open with loss of power to the solenoid valves and fail closed with a loss of instrument air supply.

The safety functions of the MSIVs are:

a) To provide containment isolation in the event of a loss-of-coolant-accident.

b) To ensure that no more than one Steam Generator will blowdown in the event of a steam line rupture. This restriction is required to: 1) minimize the positive reactivity effects of the reactor coolant system cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment.

Steam Dump and Bypass Valves to Condenser (PCV-8801, 8802, 8803, 8804, 8805)

The main function of the Steam Dump Bypass system is to limit the pressure rise in the Steam Generators to preclude the opening of the Main Steam Safety Valves. If the turbine cannot accept all of the steam being produced in the Steam Generators, for example in the event of a turbine trip or partial loss of electrical load on the generator, an alternate heat removal path is provided to remove the sensible heat in the reactor coolant as well as the reactor decay heat in order to limit the pressure rise in the Steam Generators. Steam dump and bypass valves, located downstream of the Main Steam Isolation Valves, connect the main steam header outside containment directly to the main condenser and are programmed to bypass steam directly to the condenser if such a high pressure condition should arise. PCV-8801 is a 5% capacity turbine bypass valve while valves PCV-8802 through 8805 are 10% capacity allowing a 45% steam load divergence to the condenser. Therefore, the system is designed to enable the plant to accept a loss of electrical load on the generator up to 45% of full power, without tripping the turbine. Note that the Steam Dump Bypass system is dependent upon availability of the condenser and instrument air.

Main Condenser

The condenser is of the deaerating type and is sized to condense exhaust steam from the main turbine under full load conditions. The condenser consists of two 50 percent capacity, divided-water-box, surface condensers of the single pass type with tubes arranged perpendicular to the turbine shaft. Non-condensable gases are removed from the condenser by the two hogging ejectors, a steam jet air ejector with associated inter- and after-condensers, manifolds, valves, and piping.

The circulating water system is required to establish and maintain the condenser as a heat sink. The condenser hotwell is a storage reservoir for the deaerated condensate which supplies the condensate pumps. The storage capacity of the hotwell can provide sufficient feedwater for four minutes of operation at maximum throttle flow with some additional volume for surge protection. The hotwell supply of condensate is backed up by the condensate storage tank from which condensate may be admitted into the condenser for deaeration.

The condenser also receives various drains, steam dumps, and relief valve discharges from the main steam system, main turbine system, feedwater system, and feedwater heater extractions drain system as well as the condensate system.

Condensate Pumps (COND PP 1A, 1B, 1C [2A, 2B, 2C])

There are three condensate pumps in the condensate system. Each pump is an eight-stage [seven-stage], vertical, centrifugal pump capable of pumping up to 60% of total full power condensate flow requirements. Pump 1A takes a suction on the 1A condenser and pump 1B takes a suction on the 1B condenser. Condensate pump 1C suction piping branches to tap off both condensers. [The Unit 2 configuration consists of a main header off both condensers which branches to each of the condensate pumps 2A, 2B, and 2C.] Both units' piping configurations allow maintenance of one condensate pump without limiting the total output capability of the unit. Seal water to the pump seals is normally supplied from the pump discharge header. Backup seal water is supplied from the condensate transfer pump discharge [condensate storage tank static head] for initial pump start. The pump bearings are radial sleeve type and are cooled by condensate flow through the pump.

The motors for condensate pumps 1A [2A] and 1B [2B] are powered from the 4160V buses 1A2 [2A2] and 1B2 [2B2], respectively. Condensate pump 1C [2C] is powered from either 1A2 [2A2] or 1B2 [2B2], depending upon the lineup of the condensate pump transfer switch. The switch selects the pump to be fed from the 4160V breaker and the pump to be controlled from the control room. The operation of the pump control switch will start-stop either 1A [2A] pump or 1C [2C] pump in one instance, and 1B [2B] pump or 1C [2C] pump in the other instance. The transfer switch is manually operated and is located on the southeast quadrant of the turbine building ground floor. Each condensate pump motor has a thrust bearing and two radial sleeve bearings that are oil cooled and lubricated. The oil is cooled in a coil-type cooler by turbine cooling water.

Main Feedwater Pumps (FW PP 1A and 1B [2A and 2B])

Each unit is equipped with two main feedwater pumps. Each feedwater pump has a rated capacity of 60% of rated full power feedwater flow. Each pump is directly connected to a constant speed motor. Each feedwater pump motor has two radial sleeve bearings, which are lubricated and cooled by oil from the feed pump oil system. Each feed pump's oil system has a shell-and-tube heat exchanger for oil cooling. The heat exchanger is supplied with cooling water from the turbine cooling water system. 6.9kV power is supplied to the 1A and 1B [2A and 2B] pumps from Buses 1A1 and 1B1 [2A1 and 2B1], respectively.

Feedwater Pump Discharge Valves (MV-09-1, 2)

Each feedwater pump has an associated motor operated discharge valve (MV-09-1 and 2) which opens and closes with the pump's start and stop signal.

Main Feedwater Isolation Valves (MV-09-7,8 [HCV-09-1A,B and HCV-09-2A,B])

In Unit 1, one motor-operated Main Feedwater Isolation Valve is located in each Steam Generator feed line just before the pipe enters the containment building. These valves receive an auto close signal on MSIS or SIAS. Note that the Unit 1 Main Feedwater Pump Discharge MOVs (MV-09-1, 2) also close on MSIS or SIAS. Also, Unit 1 check valves V09248 and V09280 located on the Steam Generator feed lines provide isolation capability.

[In Unit 2, each feedwater line is provided with two redundant feedwater isolation valves. The isolation valves are provided with electro-hydraulic operators which enable fast closure during accident conditions and slow closure during normal operation. The Feedwater Isolation Valves (FWIVs) receive an auto fast closure signal on MSIS or AFAS. The AFAS signal can be overridden by placing the control switches in the CLOSE/OVERRIDE position, then back to open. Any subsequent AFAS will close the valves again.]

Main Feedwater Regulating Valves (FCV-9011, 9021)

The main feedwater regulating valves are electro-pneumatic piston-operated angle globe valves. The feedwater regulating valves can be operated in three ways. Local manual operation can be accomplished with the manual handwheel. Pneumatic operation is accomplished by means of a signal from either the feedwater regulating system or a remote manual operator in the control room. With loss of control power or loss of air to the valve, a mechanism is in place for the valve to remain in its position at the time of the control failure/loss.

Feedwater Regulating Bypass Valves (LCV-9005, LCV-9006)

Bypass flow around the feedwater regulating valves is accomplished by shutting the feedwater regulating valve inlet block valves, MV-09-5 and 6, and assuming control of the 15% bypass valves (LCV-9005, 9006). Note that closure of the block valves is not required if the feedwater regulating valve is closed.

The 15% bypass valves are electro-pneumatic diaphragm-operated valves which are controlled manually from the control room or automatically by the Feedwater Control System. These valves are used for normal plant operations and during low flow conditions such as plant startups and shutdowns. If a turbine trip signal is generated, the 15% bypass valves will automatically open to 5% flow and the feedwater regulating valves will shut. This flow amount will allow for decay heat removal without causing excessive cooldown of the plant.

Feedwater Pump Recirculation Valves (FCV-09-1A2, 1B2)

The recirculation flow control valves, FCV-09-1A2, 1B2, are controlled by flow transmitters FT-09-1A1 and 1B1, respectively. When the feedwater regulating valves decrease the feedwater flow into the SGs, the flow transmitters will sense the change and send a signal to open the recirculation valve. When the feedwater regulating valves open, the increased flow will be sensed and the recirculation valves will modulate in the close direction. Once the main feedwater pump minimum flow requirements are satisfied, the recirculation valves will be fully shut. These flow transmitters are also used to generate an electrical signal for the annunciator ("FW Pump 1A and 1B [2A and 2B] Low Flow") in the control room.

Feedwater Regulating Control System

During steady-state normal operating conditions, steam generator water level is maintained by keeping the rate of feed flow equal to the combined steam and blowdown flow rates. During load transients, when steam demand is changing, the water level changes until feed flow is again matched to steam flow. Consequently, a control system is needed to regulate the feedwater flow rate and thereby control the water level. This is accomplished by the feedwater regulating control system.

Steam Generator Blowdown Isolation and Sample Valves (FCV-23-3, 5, 7, 9)

Each blowdown line has an air-actuated isolation valve outside the containment and one inside the containment. The inside isolation valves for SG 1A [2A] and 1B [2B] are FCV-23-4 and FCV-23-6, respectively. FCV-23-3 and FCV-23-5 are the outside isolation valves for SG 1A [2A] and 1B [2B]. All four isolation valves shut on loss of control air, on loss of control power, or high radiation signal. [For Unit 2, inside containment blowdown valves do not close on high radiation.] The outside isolation valves, FCV-23-3 and FCV-23-5, also close automatically on a Containment Isolation Signal (CIS). The isolation valves are powered from separate power supplies to provide redundant isolation protection. Valves FCV-23-3 and FCV-23-5 are powered from emergency bus A while valves FCV-23-4 and FCV-23-6 are powered from emergency bus B. The opening air supply is throttled to each valve such that the opening time, full closed to full open, is approximately two minutes. This long stroke time minimizes the effects of water hammer and transient thermal loads.

Each pair of inside and outside containment isolation valves is controlled by a switch in the control room. An override feature associated with the CLOSED/OVERRIDE switch position for the outside containment isolation valves is only installed on Unit 1. This feature allows the operator to override the automatic close signal generated by CIS or high radiation, thus allowing blowdown of both S/Gs as called for in certain emergency operating procedures (1/2-EOP-04 - SGTR, and 1/2-EOP-15 - Functional recovery). Open or closed indication is also provided.

Figure B15.1 Simplified Schematic for PCS Main Steam Portion (Unit 1)

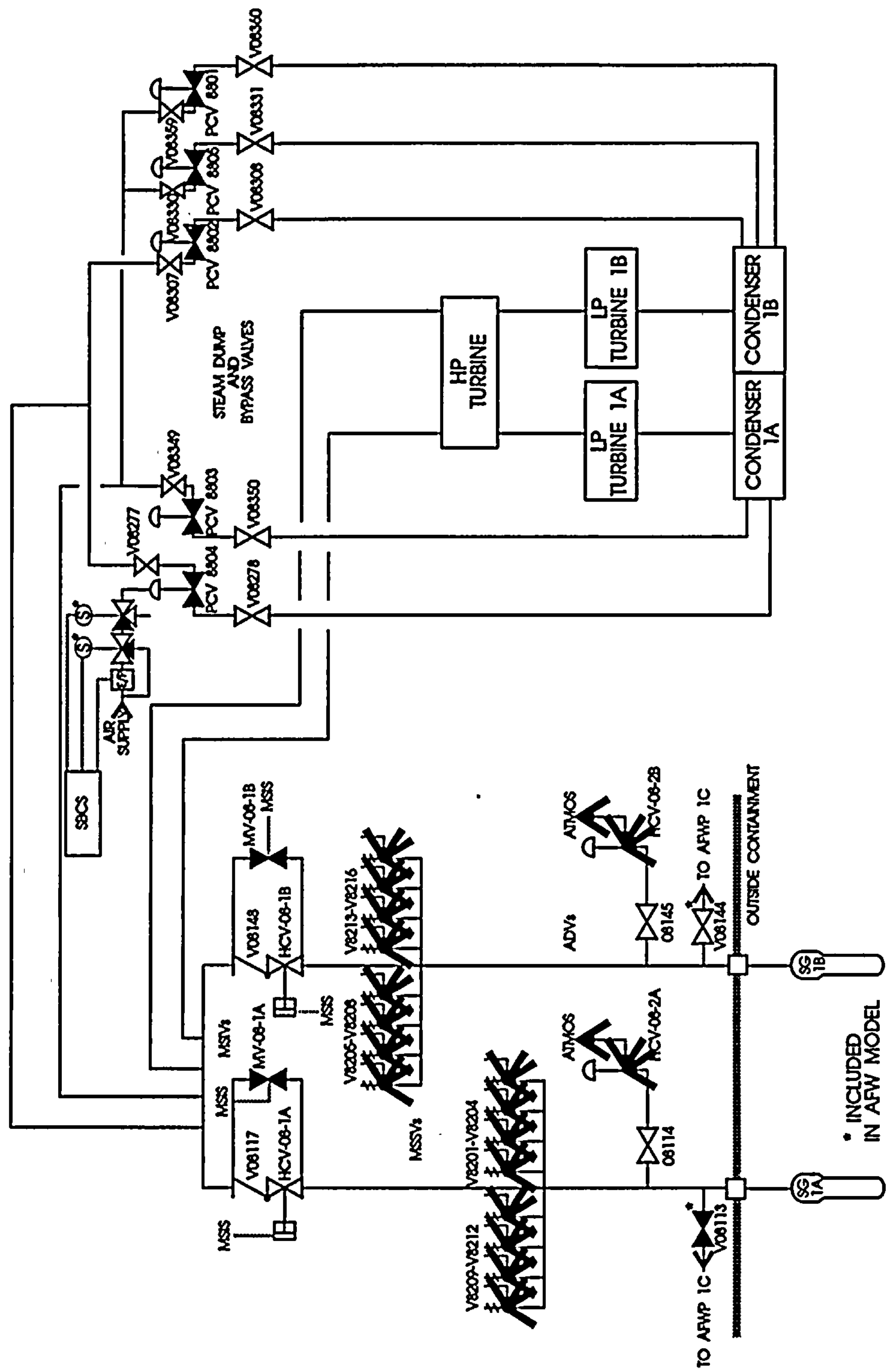


Figure B15.2 Simplified Schematic for PCS Main Steam Portion (Unit 2)

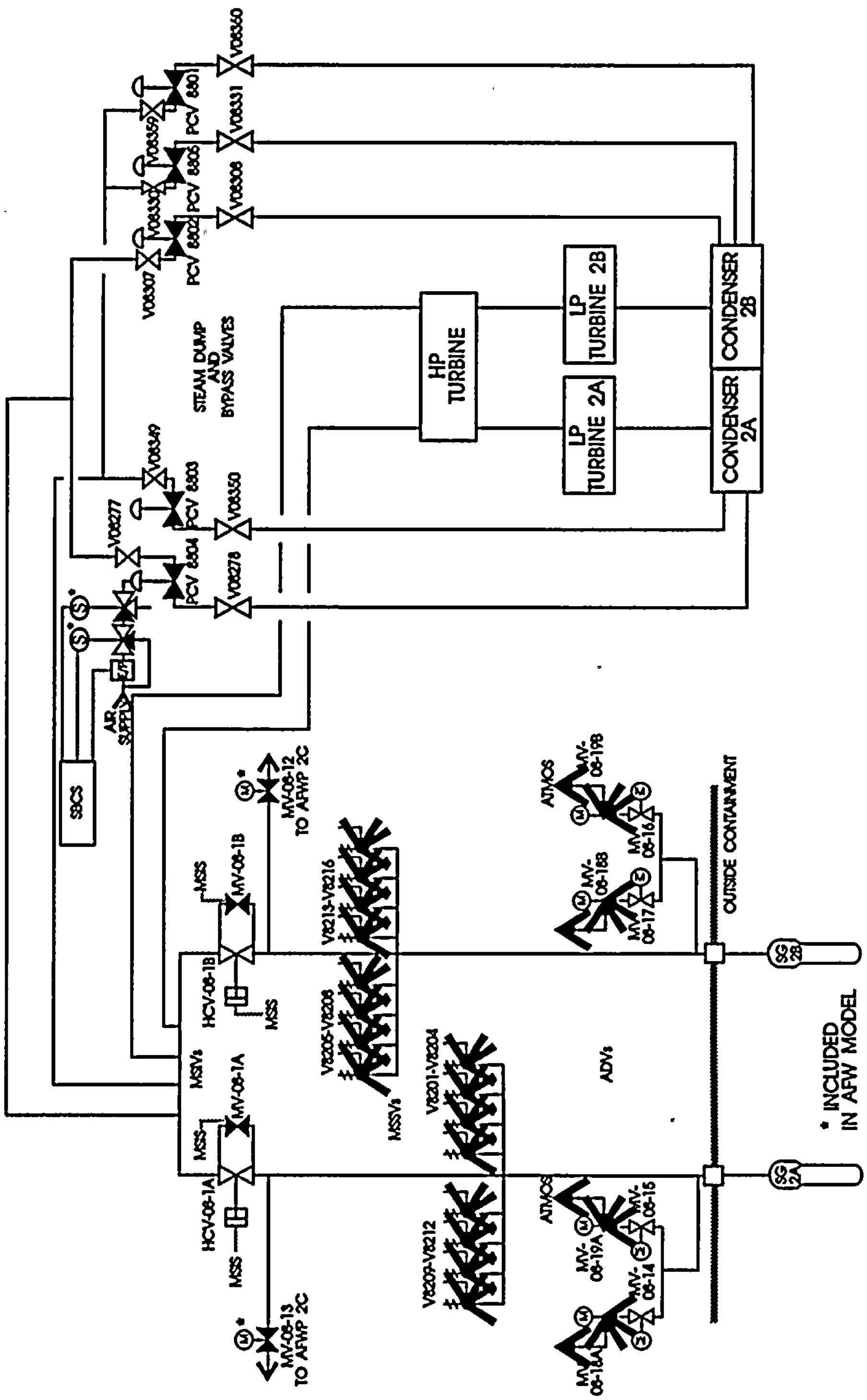


Figure B15.3 Simplified Schematic For PCS Condensate & Feedwater Portion (Unit 1)

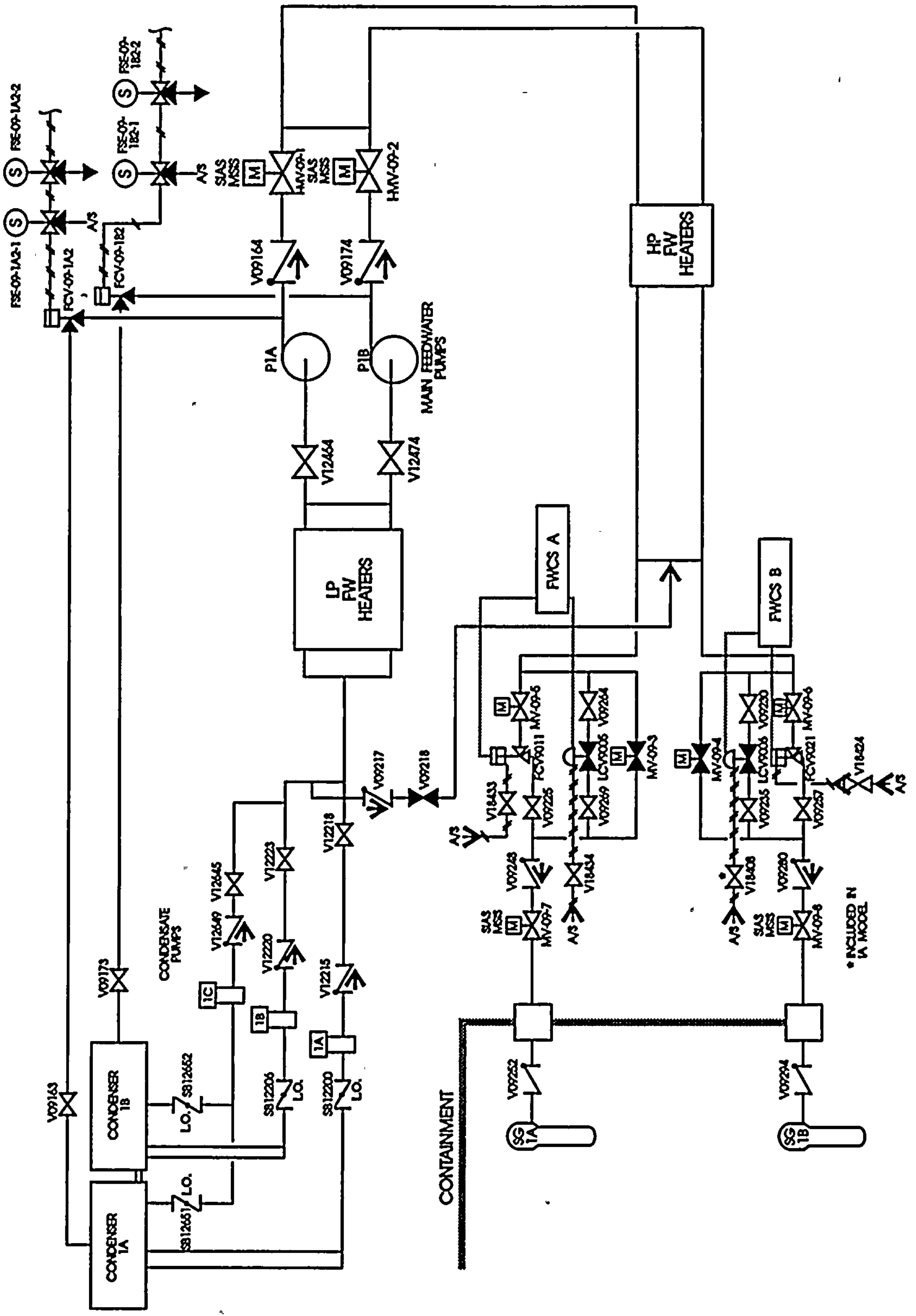
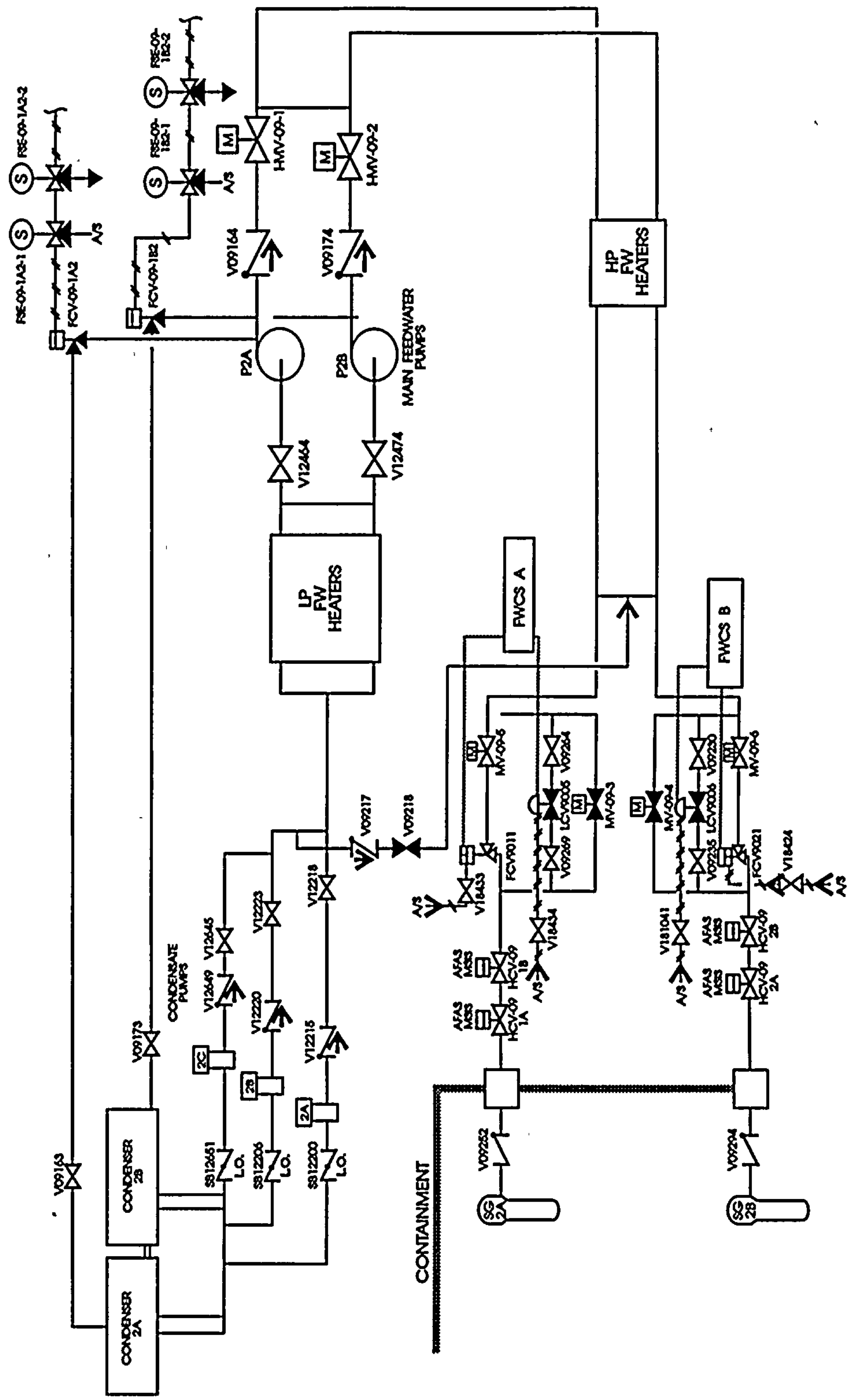


Figure B15.4 Simplified Schematic for PCS Condensate & Feedwater Portion (Unit 2)



The sample lines, like the blowdown lines, are also provided with isolation valves outside containment. FCV-23-7 isolates the sample line of Steam Generator 1A [2A], and FCV-23-9 isolates the sample line of Steam Generator 1B [2B]. The sample line isolation valves are actuated and close automatically on a CIS, high radiation, or manually. Valve position indication, open and closed, is provided.

Turbine Cooling Water Inlet/Outlet Valves

The turbine cooling water inlet and outlet manual valves provide cooling water to the condensate pump motor bearing coolers and the feedwater pump oil coolers.

Circulating Water Pumps

The circulating water pumps and associated flow paths provide support to the condensers. The circulating water pumps provide means to remove heat from the main condensers under normal operating and shutdown conditions. Each unit is provided with four pumps (two pumps per condenser), four condenser discharge valves (two per condenser), two discharge tunnels, a seal well, and the discharge canal. The four circulating water pumps are single stage, vertical removable element, mixed flow. Each pump has a capacity of 121,000 [122,650] gpm and head of 40 ft. The motors are of constant speed.

B15.3 Success Criteria

The Power Conversion system (steam and feedwater subsystems) is designed to convert thermal energy in the form of steam into electricity by means of the regenerative cycle turbine generator.

The success criterion assumed for the PCS (following a reactor trip) in developing the fault tree models is defined as the ability to remove decay and sensible heat from the RCS using at least one intact Steam Generator. This involves delivery of main feedwater to the generator(s) and the discharge of steam to either the condenser or to the atmosphere. The following describes the individual success criteria for the PCS subsystems.

B15.3.1 Main Steam Safety Valves

Each steam generator is equipped with eight MSSVs. A minimum of two operable safety valves per Steam Generator ensures that sufficient relieving capacity is available for removing decay heat. However, failure of an MSSV to reseat results in loss of the PCS.

B15.3.2 Atmospheric Dump Valves

Each Steam Generator is equipped with one [two] ADV[s] capable of relieving approximately 4% [5%] of the total rated steam flow from each Steam Generator. One ADV is sufficient for success of the PCS. During rapid cooldown, however, two ADVs are required to open.

B15.3.3 Main Steam Isolation Valves

Successful operation of the MSIVs requires both valves to close when demanded by an MSIS and the associated MSIV bypass valves must remain closed. The MSIV associated with the affected or most affected Steam Generator is also closed manually to isolate the generator following a SGTR event.

B15.3.4 SG Blowdown Valves

Following a SGTR, the flow paths from the affected SG must be isolated. To prevent radiation leakage via the SG Blowdown system, both the sample isolation valve and at least one of the blowdown isolation valves on the affected SG must close.

B15.3.5 Steam Dump Bypass System

Although there are five Steam Dump and Bypass flowpaths to the condenser, the successful opening of any one path will provide sufficient capacity (5% for PCV-8801 or 10% for the others) to remove decay heat from the RCS, provided the condenser is available. Note that the SDBVs are not available to remove decay heat following transients which involve loss of offsite power or loss of instrument air. The SDBVs are also unavailable following any event that leads to closure of both MSIVs.

B15.3.6 Main Feedwater

Successful delivery of main feedwater to at least one intact Steam Generator is required for removal of decay heat from the RCS. One main feedwater pump (with delivery through the 5% bypass valves) can supply sufficient feedwater to the Steam Generator(s). Suction to the feedwater pumps is provided by one condensate pump. One circulating water pump is required for successful cooling of the condenser.

B15.4 Operation

B15.4.1 Normal Operation

The PCS includes the steam system, turbine generator, main condenser, and auxiliary subsystems. The PCS is designed to convert thermal energy in the form of steam, as produced in the two Steam

Generators, into electrical energy by means of a regenerative cycle turbine-generator. The turbine consists of a high pressure turbine element, four moisture-separator/reheater assemblies, and two low pressure turbine elements all aligned in tandem. After expanding in the turbine, the steam is condensed in the main condenser and the energy which is unusable in the thermal cycle is rejected to the Circulating Water system. The condensate is collected in a hotwell. Non-condensable gases in the steam are removed by the steam jet air ejectors.

The condensate is returned to the Steam Generators by means of two condensate pumps and two steam generator feedwater pumps. The feedwater passes through five stages of heat exchangers (i.e., high and low pressure heaters) arranged in two parallel trains where it is heated by steam extracted from various stages of the turbine. The drains from the first three stages of low pressure heaters are eventually cascaded back to the condenser hotwell, and the drains from the fourth stage low pressure heaters and fifth stage high pressure heaters are returned to the feedwater system by two heater drain pumps.

Heat produced in the reactor core is transferred from the reactor coolant to the water in the Steam Generators producing steam for use in the turbine. In the event of a turbine trip, the heat transferred from the reactor coolant to the Steam Generators is dissipated through the steam dump and bypass system to the condenser and/or through the atmospheric dump valves and the main steam safety valves.

During normal operation the system lineup is as follows (described according to flow direction):

Main Steam Portion

All ADVs are closed and in manual mode to preclude automatic opening on high SG pressure; isolation valves that provide steam to the AFW turbine pump are closed (but will open on AFAS); main steam safety valves are closed due to normal operating main steam pressures; both MSIVs are in the open position with their respective bypass valves (MV-08-1A/1B) closed; and steam dump and bypass valves are in the closed position. This valve alignment ensures main steam flow to the turbine for power generation, and down to the condenser for exhaust heat extraction.

Feedwater Portion

Two condensate pumps are normally operating taking suction from both condensers; all condensate pump suction and discharge manual valves are open; both main feedwater pumps are running with their respective pump recirculation valves closed; FW pump discharge motor-operated and manual valves are open; FW isolation and regulating valves are open; and FW regulating bypass valves are in the closed position. The Feedwater Regulating Control System monitors and regulates flow to the SGs from 15% to full power under normal operation. The Feedwater Bypass Control System maintains Steam Generator level under 15% power.

B15.4.2 Accident Operation

The steam and power conversion system is designed to meet its safety design bases under conditions postulated to exist for each of the abnormal incidents for which it must perform a safety function such as steam line breaks, feedline breaks, and loss of offsite power.

Steam Line Break

The main steam system is designed to prevent blowdown of both Steam Generators in the event of a postulated steam line break accident. Following a steam line break event, the following automatic actions are performed by the PCS:

- Both main steam line isolation valves and main feedwater isolation valves will receive a closure signal upon MSIS actuation from either Steam Generator. The FW pump discharge valves also receive a closure signal for Unit 1. The system is designed such that no single failure will cause both isolation valves to remain open.

- The air-operated FW pump recirculation valves open when the feedwater pumps are tripped. For Unit 1, the FW recirculation valves also receive a turbine trip signal to open.

- The Steam Dump and Bypass capability is lost because the MSIVs are closed. Therefore, there will be an initial pressure spike in which the MSSVs will open (if needed) and then reseat. The operator then can take control by opening the ADV associated with the affected SG and vent steam directly to the atmosphere.

Transients

Following an uncomplicated transient (i.e., Offsite Power and Instrument Air are available), the PCS performs the following actions:

- Feedwater regulating valves close, and the feedwater regulating bypass valves open, reducing feedwater flow to the SGs down to 5% of full power flow. This operation is accomplished by the Feedwater Bypass Control System (FBCS) which monitors and regulates flow to the SGs.

- The Steam Bypass Control System (SBCS) monitors and regulates steam flow to the condenser via the steam dump and bypass valves (SDBVs). If the condenser is unavailable, these SDBVs will receive a signal to close.

During plant shutdown with off-site power available, the required number of valves may be manually positioned to remove reactor decay heat, pump heat, and reactor coolant system sensible heat to reduce the reactor coolant temperature at the design cool down rate until shutdown cooling is initiated.

For plant shutdown without off-site power, the atmospheric dump valves are used to remove reactor decay and sensible heat by venting steam from the Steam Generators directly to the atmosphere. Both MSIVs will close automatically on a MSIS. The Unit 1 MSIS is initiated by two-out-of-four low pressure signals from either steam generator. [The Unit 2 MSIS is initiated by two-out-of-four low pressure signals from either steam generator and/or upon high containment pressure.]

The MSIVs will fail in the open position on loss of electric power to the solenoid valve and in the closed position on loss of air supply. Air accumulator tanks are provided to hold the valves open for at least 8 hours after a loss of normal air supply, unless the valves are tripped or closed. The valves have limit switches for valve operation and open/close position indication in the control room. A pressure switch will initiate an alarm in the control room in the event of low pressure in the air accumulator system.

The total capacity of the SBCS steam dump valves and turbine bypass valve is 40 percent and 5 percent, respectively, of reactor full power. This flow is sufficient to control the secondary steam pressure following a turbine trip at full power and thus avoid lifting the spring-loaded safety valves.

On a load rejection the steam dump and bypass valves are modulated in sequence to control main steam pressure to a set point that is programmed with load, i.e., steam flow. A quick opening signal is generated as a function of the magnitude and rate of change of the load rejection determined by monitoring the steam flow. The duration of the quick opening signal is proportional to the flow magnitude and rate of change. Once the signal is removed the valves revert back to modulation control.

The steam dump and bypass system may also be used to remove reactor decay heat following a reactor shutdown or during hot standby conditions.

On a reactor trip the steam dump valves are positioned by the reactor coolant average temperature while the bypass valve remains on main steam pressure control. The quick opening signal on a reactor trip is generated when the reactor coolant average temperature is above the value corresponding to the maximum valve opening demand. The valves are designed to close on loss of instrument air actuator power, or control signal. In the event of loss of condenser vacuum the valves close automatically. Redundancy is provided in the design to prevent a single equipment failure or operator error from opening more than one valve. The system controls are designed for either automatic or remote manual control.

In the event that the steam dump and bypass valves fail to open on complete loss of turbine generator load with offsite power available, the turbine trip will result in an increase in steam pressure in the Steam Generators. The Steam Generator pressure rise is terminated by opening of the main steam safety valves. Main steam safety valves continue to release to the atmosphere until the over-pressure condition is relieved or either the atmospheric dump valves are opened or the steam dump and bypass system valves are restored to the operating open position to discharge steam to the condenser.

The turbine is equipped with an automatic stop and emergency trip system which trips the throttle and governor valves to a closed position in the event of turbine overspeed, low bearing oil pressure, low vacuum, or thrust bearing failure. An electric solenoid trip valve is provided for remote manual trips and for various automatic trips. In addition, a turbine trip initiates a main generator lockout to prevent generator damage. Upon occurrence of a turbine trip, a signal is supplied to the Reactor Protection System to trip the reactor.

The turbine generator is provided with two overspeed protection systems:

a) Overspeed protection controller (OPC).

b) Mechanical overspeed protection system.

The OPC system and the mechanical system do not share any sensing devices.

SGTR

Following a Steam Generator Tube Rupture (SGTR), the SG blowdown sample valve and the outside containment isolation valve associated with the affected SG receive a signal to close. Both SG blowdown isolation valves [SG blowdown valve outside containment only] and the sample valve associated with the affected SG also receive a signal from the radiation monitor to close.

B16 PRIMARY PRESSURE CONTROL SYSTEM

B16.1 Function

The Primary Pressure Control System (PPCS) functions to control the Reactor Coolant System (RCS) pressure so that the design limits are not exceeded. For the purposes of the St. Lucie PRA, only the pressurizer Safety Relief Valves (SRV), Power Operated Relief Valves (PORV), and the Primary and Auxiliary Spray Valves are modeled.

The SRVs function to relieve pressure during 100% load rejection to maintain RCS pressure within design limits. The PRA also models SRV function during an ATWS. The SRVs must open and reclose.

The PORV function is to prevent SRV challenges by opening and relieving RCS pressure at a setpoint lower than the SRV setpoint. The PORVs also must be opened by the operator to bleed RCS pressure during once-through-cooling (Feed and Bleed) operations. Additionally the PORVs also must be capable of opening and reclosing during an ATWS.

The Spray Valves function to provide cooling to the pressurizer steam space, limiting pressure increases via steam condensation. Each primary spray valve is sized to prevent PORV actuation during normal load-following transients.

The two Auxiliary Spray valves are also modeled as backups to the normal spray valves. Charging flow to Loop A2 downstream of the regenerative heat exchanger provides spray flow when primary spray flow is unavailable.

B16.2 Configuration

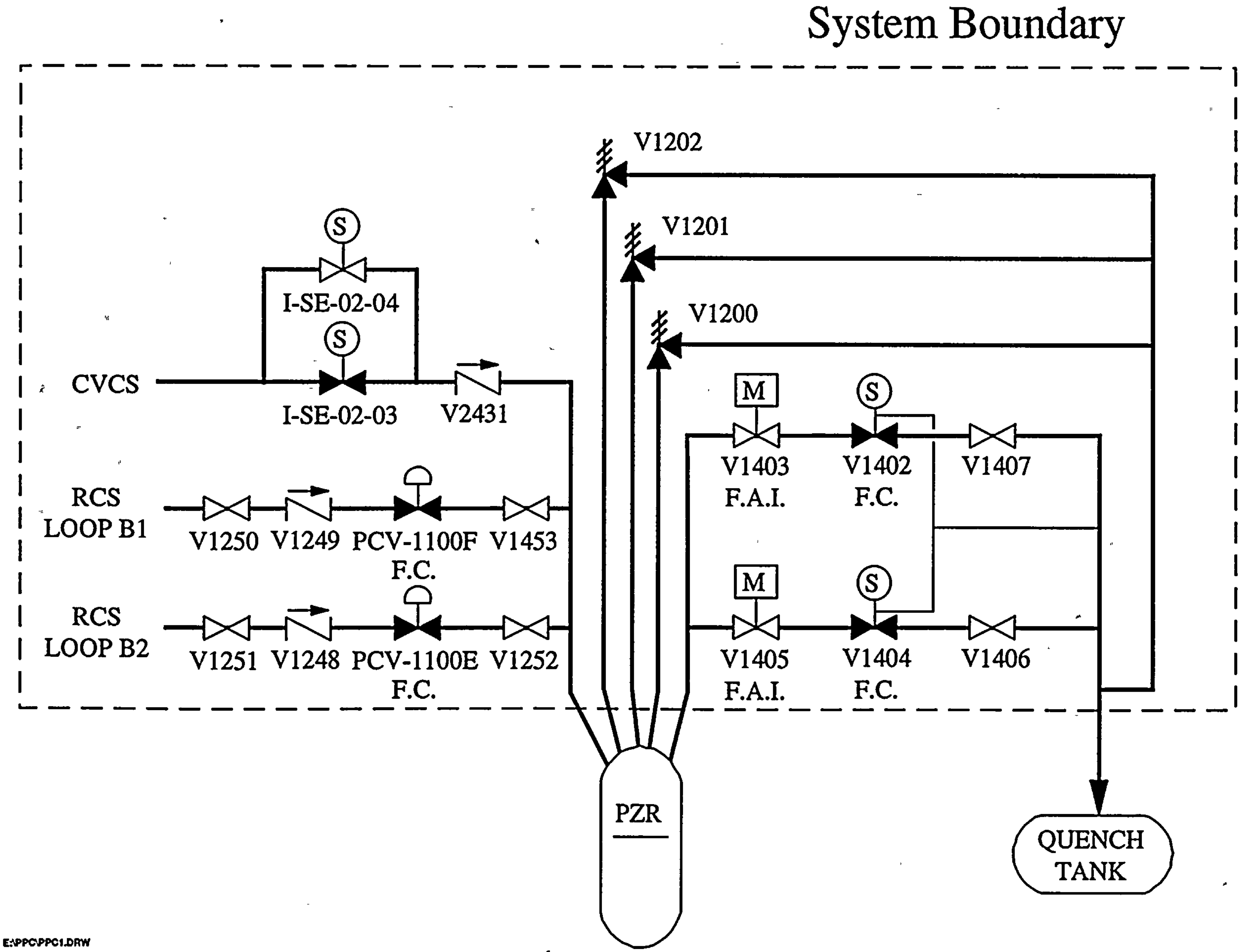
Figures B16.1 and B16.2 show the PPCS configuration and boundary. All PPCS components are located within the reactor containment building.

The PPCS is a subsystem of the RCS. The PPCS ties into the RCS at the pressurizer dome. Therefore, the PPCS is also a part of the RCS pressure boundary. Three spring loaded safety valves and two power operated relief valves provide overpressure protection. These valves are located on separate lines from the pressurizer. The PORVs share one pressurizer nozzle, while each SRV has a separate pressurizer nozzle. All five valves link to a common discharge line which is directed to the Quench Tank inside containment. In addition, two motor operated block valves are included in each PORV branch line.

The primary spray valves (in conjunction with the pressurizer heaters) provide for pressure control under normal operating conditions. Each primary spray valve is supplied from a different RCS cold leg, but both valves share a common inlet pipe and nozzle at the pressurizer. Additionally, two solenoid-operated auxiliary spray valves can provide cooling spray to the pressurizer from the charging pumps. These valves can also be operated from the Hot Shutdown Control Panel.

The spray valves are included in the model primarily to facilitate recovery operations (depressurization of the RCS).

Figure B16.1 St. Lucie Unit 1 PPCS



E:\PPC\PPC1.DRW

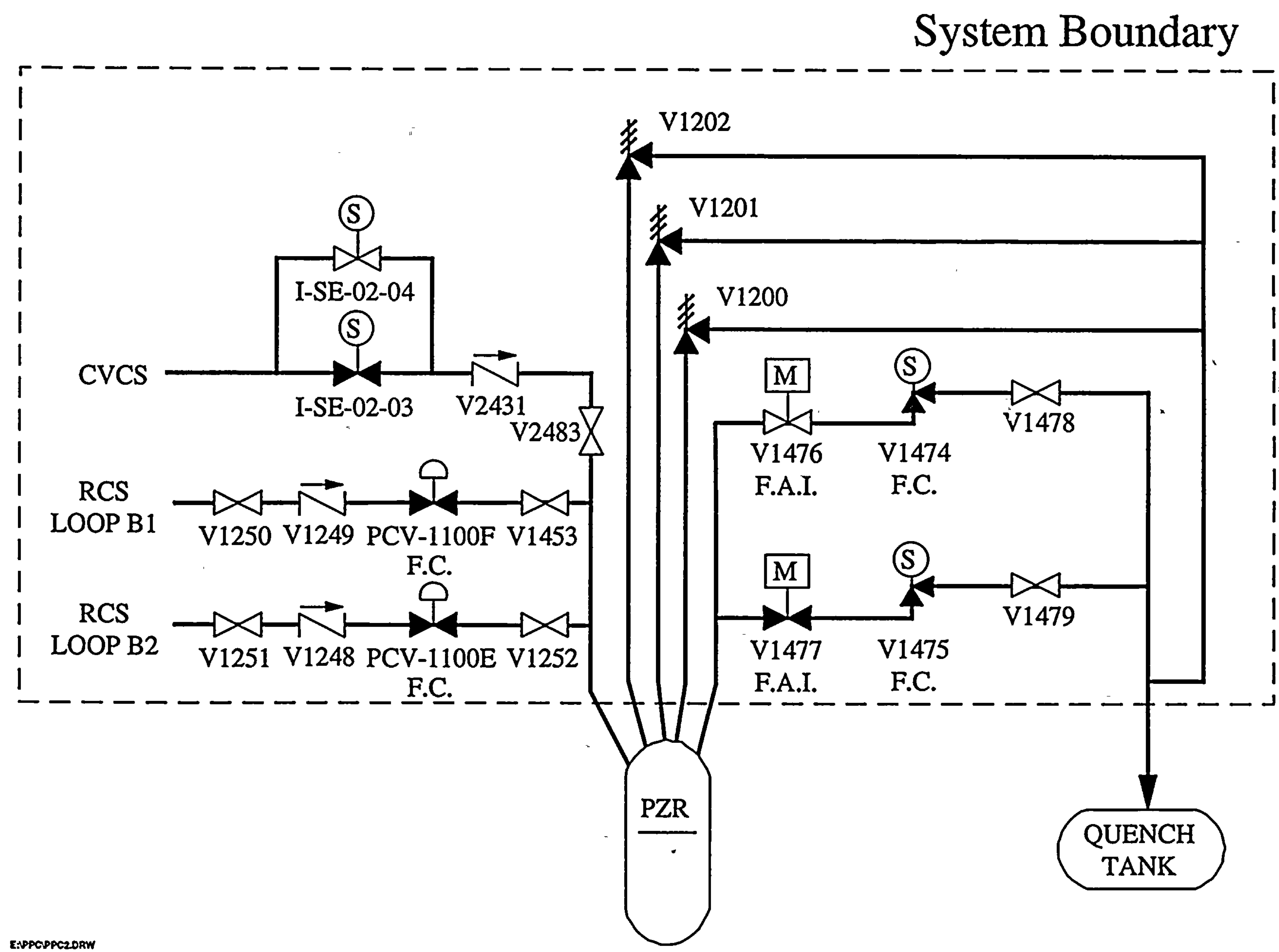


Figure B16.2 St. Lucie Unit 2 PPCS

B16.3 Success Criteria

Success criteria for the PPCS are dependent on plant conditions. For over-pressure conditions, a Power Operated Relief Valve or all three Safety Relief Valves are required to open then reclose when pressure decreases below their respective setpoints. For once-through cooling (Feed-and-Bleed) operations, both [one] PORV trains are required to be operable.

B16.4 Operation

B16.4.1 Normal Operation

During normal operation all three pressurizer SRVs are closed but operable, as required by the unit Technical Specifications. Both PORVs are normally closed with their associated block valves open (Unit 2 has 1 block valve closed). However, as the PORVs are not considered to be safety class 1E equipment, one or both block valves may be closed to control PORV leakage.

The primary spray valves modulate as necessary to control RCS pressure and as determined by HIC-1100 on RTGB 103. The valves are normally operated in parallel and are fully shut at 2300 psia and ramp to fully open at 2335 psia. The auxiliary spray valves are normally closed with their key-locked switches in the RESET/LOCKED CLOSED position.

B16.4.2 Accident Operation

Under accident conditions, the PORVs and SRVs function to limit RCS pressure to less than 110 percent of system design (2750 psia) following a complete loss of load without simultaneous reactor trip. At 2400 [2370] psia, both [one] PORVs will open to relieve RCS pressure to the Quench Tank. If pressure continues to rise, the SRVs will open at 2500 psia. Both the PORVs and the SRVs are sized to accommodate a full load rejection without concurrent reactor trip.

Under certain LOCA scenarios RCS pressure can 'hang-up' at pressures beyond the shut-off head of the high pressure safety injection (HPSI) pumps. For these scenarios, the operators must open both PORVs and/or use the spray valves (primary or auxiliary) to reduce RCS pressure, allowing the HPSI pumps to provide core cooling. The auxiliary spray valves can be operated from the Hot Shutdown Control Panel.

B17 SHIELD BUILDING VENTILATION SYSTEM

B17.1 Function

The Shield Building Ventilation System (SBVS) performs the following functions:

1. Limit the pressure rise in the shield building annulus following a LOCA so as not to exceed the shield building internal design pressure.

2. Establish and maintain a subatmospheric pressure in the shield building annulus following a LOCA to ensure that offsite doses resulting from post-accident leakage from the containment are reduced by routing through the shield building filters.

B17.2 Configuration

Figure B17.1 [B17.2] shows the configuration of the SBVS. The SBVS consists of two full capacity redundant fan and filter subsystems which share a common shield building duct intake and a common plant vent. Each filter subsystem consists of a demister, two electric heating coils, two HEPA filters, and a charcoal adsorber enclosed in a common casing.

The SBVS annulus air intake consists of a ring duct with inlets located at each quadrant and at the top of the shield building. Two separate lines from the ring duct penetrate the shield building walls to connect to their corresponding filter subsystems. The fan and filter subsystems are located in the reactor auxiliary building. Outside air lines, each isolated by a check valve and a motor-operated valve in series, are connected to the intake of the filter subsystems to provide cooling air to the filters when required. A line with an isolating butterfly valve cross connects the filter subsystems downstream of the filter banks and upstream of the fans to maintain flow through the filters in the event of failure of a fan. A gravity damper and motor-operated control valve (D-23, D-24) are located at the discharge of each fan.

<u>SBVS Fans HVE-6A & 6B [2HVE-6A & 6B]</u>

The shield building exhaust fans are 480V AC centrifugal fans. They are powered from 480V AC emergency motor control centers. Both fans are automatically started on a Containment Isolation Actuation Signal (CIAS) (unless in ISOLATE). Additionally, if one fan is running, the standby fan will automatically start on a low flow. The exhaust fans have a motor overload trip.

<u>SBVS Filter Subsystems</u>

The SBVS filter subsystems are composed of a demister, HEPA filters, a charcoal adsorber, and electric heaters.

Demisters

The demisters (one per filter train) are located in the upstream portion of the filter subsections, between the electric heaters. Their function is to remove moisture contained in the intake air to increase the efficiency of the downstream charcoal adsorbers.

HEPA Filters

Each filter train is provided with two high efficiency particulate absorber (HEPA) filter banks, each located on either side of the charcoal adsorber. Their function is to reduce the particulate activity in the air being exhausted from the shield building to the atmosphere.

Charcoal Adsorbers

Each filter train contains 18 charcoal adsorber banks for the removal of fission product iodine.

Electrical Heaters

Two electric heaters, one rated at 30 kW (EHC-[2]HVE-6B1, 6A1) and the other at 1.5 kW (EHC-[2]HVE-6B2, 6A2), are located in each of the filter trains. Their function is to provide a humidity of less than 70% maximum to ensure maximum efficiency of the charcoal adsorbers. The heaters are controlled by separate temperature controllers from each filter train. Both sections of the heaters will be energized whenever both fans are running. If one fan is shutdown or trips, only the high power heater (30kW) automatically de-energizes. This will provide sufficient temperature/humidity control in the deactivated filter train because of the reduced air flow. The high power heaters are automatically energized whenever the corresponding exhaust fan is started. The low power heater for each train is energized when the alternate fan is started.

<u>Flow Control Dampers (D-23, D-24)</u>

Motor-operated dampers D-23 and D-24 are located downstream of fans A and B, respectively. Their function is to control the flow rate through each SBVS train. The dampers are controlled by the annulus-to-outside differential pressure. The dampers are fully open at a differential pressure of 1 in. w.g. negative. In this condition (shortly after a LOCA), the maximum system flow rate is provided. The dampers are partially closed at a differential pressure of 2 in. w.g. negative. In this condition, a continuous flow rate of 6,000 cfm is provided.

[The Unit 2 dampers can also be controlled by FIC-25-20-A1 and FIC-25-20-B1 located in the control room. This gives the operators a remote control capability for long-term operation of the system.]

Outside Air Valves (FCV-25-11, FCV-25-12)

Motor-operated flow control valves FCV-25-11 and 12 are designed to provide outside cooling air to the SBVS filters after operation of the system has evacuated the annulus. The valves are controlled from the control room by AUTO/OVERRIDE switches. In Auto, the valves will open at an annulus-to-outside differential pressure of - 1 in. w.g.. The valves are also interlocked with their associated exhaust fan so that the valve will not open unless the fan is running. Overload alarms are provided in the control room to warn of the loss of outside air valves.

Crosstie Valve FCV-25-13

The motor-operated valve FCV-25-13 is a normally open valve that connects both trains of the SBVS and it is located downstream of the filters and upstream of the fans. With one train operating and the second one inoperable, a flow diversion analysis shows that 300 cfm (from the total of 6,000 cfm) is drawn through the inoperable train via this valve. This flow diversion is low enough so that failure of the operating train does not occur.

[SBVS Isolation Valves (FCV-25-32 & FCV-25-33)

Unit 2 also has motor-operated annulus isolation valves to isolate the annulus from the fuel handling building in the event of a high radiation signal in the fuel handling building (FHB). These valves, FCV-25-32 and FCV-25-33, are controlled by switches in the control room. The valves are opened automatically by CIAS and are closed automatically on a FHB high radiation signal. The valves are normally open during normal operations.]

B17.3 Success Criteria

The Shield Building Ventilation System (SBVS) is designed to reduce the shield building annulus pressure following a Loss Of Coolant Accident (LOCA). Since the SBVS consists of two full capacity redundant fan and filter subsystems (trains) which share a common shield building duct intake and a common plant vent, the successful operation of the SBVS following a LOCA requires operation of one of the two trains.

B17.4 Operation

B17.4.1 Normal Operation

During normal plant operations, the shield building ventilation system is lined up for operation with the fans in automatic (standby).

Figure 17.1 Simplified Schematic of Shield Building Ventilation For Unit 1

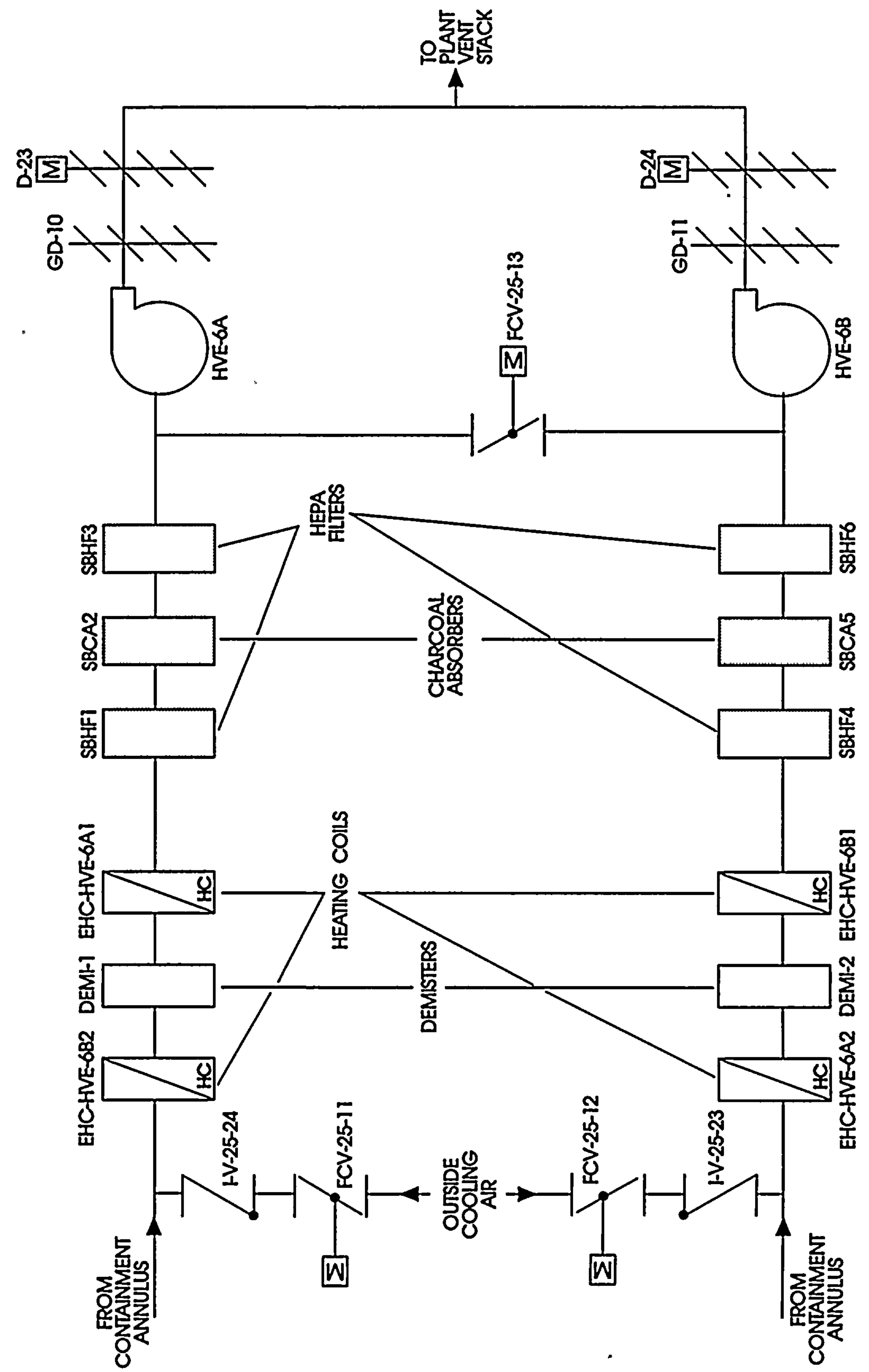
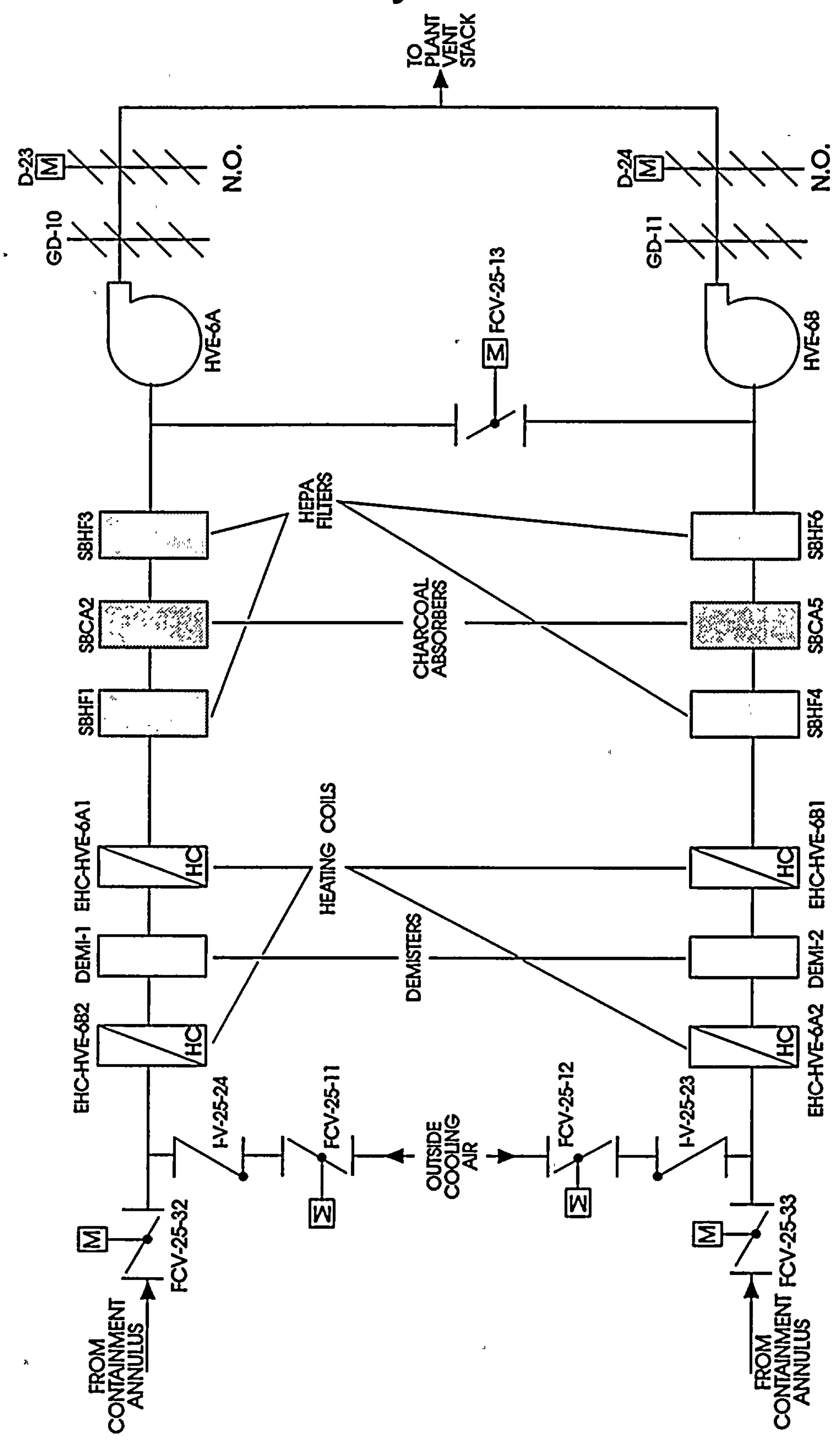


Figure 17.2 Simplified Schematic for Shield Building Ventilation System For Unit 2



B17.4.2 Accident Operation

Following a LOCA, the resultant pressure and temperature induced expansion of the containment vessel and the heat transfer through the vessel walls cause a decrease in the shield building volume and an increase in pressure. The annulus pressure is rapidly drawn down by the SBVS. Both fans are automatically started upon receipt of a CIAS. [The motorized dampers downstream of the fans are normally open.] Once initiated, one subsystem can be manually shutdown from the control room and placed in the standby mode. The standby subsystem will automatically restart if the operating system should fail. In the event of a failed subsystem, the cross-connection valve is opened from the control room to assure adequate cooling air flow through the failed system. During operation of the system, charcoal adsorber temperatures and HEPA differential pressures are carefully monitored. Within 120 [310] sec. after a LOCA, the shield building annulus pressure is below atmospheric. The fan continues exhausting at a decreasing rate until the pressure in the shield building is -2 in. w.g. with respect to atmospheric. At this point, the motorized damper at the fan discharge closes to a preset position to throttle air flow to a continuous 6,000 cfm. In order to provide adequate cooling air flow through the filters (which become hot due to the decay of adsorbed fission products), additional air is taken from the outside atmosphere through a makeup cooling air line located outside the annulus upstream of the filter train. The outside air makeup motor-operated butterfly valve is opened automatically when the annulus differential pressure is -1 in. w.g.. The check valve in the cooling line is designed to open at -1 in. w.g. to provide vacuum control in the system and to allow outside air to cool the filters.

C.1 INTRODUCTION

This appendix documents the features of the St. Lucie reactor coolant system, emergency core cooling and containment which strongly influence the progression of severe accident sequences and their potential consequences. These discussions include a comparison of the reactor features among the different designs and of containment parameters and performance limits among reference plant designs.

C.2 SPECIFIC PLANT FEATURES

Core and containment features specific to St. Lucie that influence the core damage sequence progression and could have significant impact on the consequences of the accident are described in this section. Most of the features include active and passive systems which maintain core cooling and mitigate pressure/thermal loads that challenge containment integrity. The normal operating conditions and component information of the RCS and containment for St. Lucie and reference plants are summarized in Table C-1.

St. Lucie-specific features are compared to reference PWR large dry containment designs in Table C-1. The key differences that strongly influence severe accident progression include:

- Reactor Coolant System;
- Emergency Core Cooling System;
- Containment configuration/design; and
- Containment safeguards systems.

Specific features and impact on the containment response are discussed below. The items listed in Table C-1 are described qualitatively in terms of specific functions that influence accident progression and implications on the containment response under severe accident loadings. The most important plant characteristic is the containment ultimate pressure capability, which determines the capacity of the containment to withstand pressure loads during a severe accident.

C.2.1 Reactor Core/Coolant System

The reactor core assembly is the part of the PWR NSSS in which the controlled nuclear fission chain reaction is established. The reactor core is light-water cooled and moderated and fueled with slightly enriched uranium dioxide.

The St. Lucie Plant Reactor Coolant System is illustrated in Figure C-1. The reactor is designed to operate at power with all four reactor coolant pumps in operation. The pressurizer (PZR) serves to control RCS pressure and as a surge volume to limit RCS pressure transients during power operation. It normally operates partially full of water, with a steam bubble in its remaining volume.

The Steam Generators supply saturated steam to the turbine and provide a barrier to prevent fission products from entering the Main Steam System. The St. Lucie Steam Generators are vertical, U-tube and shell heat exchangers.

The reactor and RCS volumes, pressures, temperatures, and setpoints are required in the calculations for the RCS response throughout degraded core accident scenarios. The core information is used to determine the magnitude of energy released due to metal water reactions, and is used in the calculations of fission product released into containment. The primary system metal mass and thermal capacitance are used to determine the amount of decay heat removed through passive heat transfer. The volumes and flow areas in the reactor vessel are used in the calculation of core uncovery and vessel breach event timings.

C.2.2 Emergency Core Cooling System

The capacity and availability of the ECCS during both injection and recirculation modes of operation is important in determining event timings for core uncovery, vessel breach and the type of sequence (e.g., high-pressure RCS breach). The status of the containment atmosphere, debris bed coolability, potential for preventing vessel breach, and decay heat removal are also contingent upon ECCS capabilities.

During the injection phase, the ECCS provides injection of borated water to the RCS to ensure reactor shutdown and adequate cooling. For large LOCA scenarios, the RCS is rapidly depressurized and makeup is provided by the safety injection tanks as RCS pressure drops below the Safety Injection Tank (SIT) pressure (e.g., 200 psig for Unit 1 and 600 psig for Unit 2). The high- and low-pressure safety injection systems (HPSI and LPSI) take suction from the RWST and deliver makeup water through the cold legs. For accident scenarios with the RCS at high pressure, the makeup water will be provided by the HPSI. For some plants, these pumps operate at pressures up to the primary safety valve setpoint. St. Lucie, however, has an HPSI pump shutoff head of 1225 psig and are therefore not able to provide makeup at full RCS pressure. For St. Lucie, RCS makeup at high pressure is limited to the normal charging pumps.

After the RWST water supply has been depleted, the ECCS is placed in a recirculation mode. Both LPSI pumps are stopped by the recirculation actuation system (RAS).

C.2.3 Containment Configuration/Structural Design

The containment structure is a steel containment vessel surrounded by a reinforced concrete shield building. The two structures are separated by an annular air space. The containment vessel consists of a cylindrical steel shell with hemispherical dome and ellipsoidal bottom. The containment forms a low-leakage barrier against the release of radioactivity from the reactor core. It is designed to perform this function under the post-accident environmental conditions resulting from a postulated LOCA. The shield building protects the containment vessel from environmental conditions resulting from severe natural phenomena (e.g., tornados, etc.), provides biological shielding and provides a means of controlling radioactive fission products that leak from containment if an accident should occur.

APPENDIX C

ST. LUCIE UNITS 1 & 2

CONTAINMENT PERFORMANCE FEATURES

TABLE OF CONTENTS

The containment internals consist of the reactor cavity, steam generator compartments, and a fuel transfer canal located above the reactor cavity. The steam generator compartment houses the steam generators, reactor coolant pumps, and the pressurizer. The primary function of the steam generator compartment walls is to serve as secondary shield walls and to resist jet loads due to pipe rupture. The cavity walls are designed to withstand the jet force coincident with the pressure load resulting in the pipe rupture. The reactor cavity is heavily reinforced to support the reactor core and the primary shield wall. The basemat also supports the steam generator and their shield structures.

The containment system principal performance objectives are to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining material properties, residual, steady-state and transient stresses, and size of flaws.

The containment is designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any design basis LOCA. This margin reflects consideration of:

1. The effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded ECCS functioning.

2. The limited experience and experimental data available for defining accident phenomena and containment responses.

3. The conservatism of the calculational model and input parameters.

The containment is designed to withstand loads from the following:

- Design internal pressure;
- Design external pressure;
- Design internal temperature (accident); and
- Design internal temperature (normal).

C.2.4 Containment Safeguard Systems

The operation of the containment is supported by a variety of containment-related auxiliary systems that perform the following functions:

- Containment isolation;
- Containment heat removal;
- Containment purge; and
- Combustible gas control.

C.2.4.1 Containment Isolation

The primary function of the containment isolation system is to prevent the release of gaseous or airborne radioactivity from the containment atmosphere to the outside environment. At the same time, the containment isolation system must allow the passage of essential fluids across the containment boundary to mitigate the consequences of the accident. Prevention of liquid releases from closed systems outside containment or operating ESF systems is not a containment isolation function.

There are two basic types of containment penetrations; piping penetrations and integral barriers. Piping penetrations allow the passage of fluids across the containment boundary. For the most part, these penetrations rely on active closure for the containment function. Integral barriers on the other hand, are passive barriers. These barriers maintain rather than change state to affect isolation.

Piping Penetrations

There are 71 piping penetrations on Unit 1 and 73 on Unit 2. Most of these penetrations are provided with two containment isolation valves in series. These include manual valves, check valves, motor operated valves (MOV), and air operated valves (AOV). In some cases, however, a single isolation valve is used if the piping functions as a closed system.

Class A: Penetrations that Connect Directly to the Containment Atmosphere

For penetrations in Class A, valves and/or piping or ductwork represent the only barriers between the containment atmosphere and the outside environment. These penetrations are either open directly to the containment atmosphere and connected to non-seismic piping or duct work outside the containment or connected to non-seismic piping on both sides of the containment.

There are two categories of Class A penetrations. Class A1 includes penetrations that are normally open, or may be open, during power operation. Class A2 includes penetrations that are normally closed and are not opened during power operation.

Class B: Penetrations that Connect Directly to the RCS

For penetrations in Class B, valves and/or piping represent the only barriers between the reactor coolant and reactor coolant exposed systems outside containment. Reactor coolant exposed systems include chemical and volume control, safety injection, shutdown cooling, and the sample system.

There are two categories of Class B penetrations. Class B1 includes penetrations that are normally open, or may be open, during power operation. Class B2 includes penetrations that are normally closed and never opened during power operation.

Class C: Penetrations that Connect to Closed Systems

For penetrations in Class C, a closed piping system inside containment and a single isolation valve represent the only barriers between the containment atmosphere and the outside environment. Closed systems inside containment that function as a containment barrier include component cooling water, main steam, feedwater, and steam generator blowdown. The main steam and blowdown system inside containment is considered to be closed for all events except a main steam line break or a steam generator tube rupture.

Class D: Instrument Sensing line Penetrations

The penetrations in Class D are for containment pressure instrument sensing lines. For these penetrations, a single isolation valve and a closed piping system outside containment represent the only barriers between the containment atmosphere and the outside environment. These lines are provided with either an automatic isolation valve or a remote manual valve located outside containment. A self actuated excess flow check valve is considered an automatically actuated valve.

Class E: Engineered Safety System Penetrations

Penetrations in Class E (other than 48 and 51) are designed to be open during an design basis event. Consequently, the containment isolation valves for these penetrations do not provide a barrier against the release of radioactivity during ESF system operation. During ESF system operation, containment integrity is maintained by a water seal established by the flow of water into containment and the volume of water collected in the containment sump.

Penetration Nos. 48A (Unit 2), 48B (Unit 2), 51A (Unit 2), and 51B (Unit 2) are considered to be a special case of Class E. While these lines are not designed to open during a design basis event for accident mitigation, they are required to operate intermittently post-accident. When these lines are opened for H_2 sampling, containment integrity is maintained by a closed system outside containment.

Integral Barrier Penetrations

Integral barrier penetrations function as an integral part of, or an extension of, the containment vessel. They include large access openings, electrical penetrations, spare penetrations, and the fuel transfer penetration. Integral barrier penetrations are typically sealed with a single, passive, barrier. These barriers rely on seal welds, resilient seals, or a combination of both, for containment isolation. Due to the low probability of a seal weld failure, only the degradable mechanical seals (or resilient seals) are included in the fault tree models. When a resilient seal (such as an O-ring or gasket) is incorporated as part of the integral barrier, a redundant seal is included in the design for leak testing purposes. Double gaskets and concentric O-rings are examples of this. This design feature allows the space between the redundant seals to be pressurized for verification of proper sealing.

Large Access Openings

Large access openings are provided in the containment vessel for equipment installation or removal and personnel access. A large diameter (28'-0") equipment hatch and a smaller diameter (12'-0") maintenance hatch are provided for transporting equipment and material across the containment boundary. The large diameter equipment hatch is seal welded closed and the smaller diameter maintenance hatch is sealed with a double gasketed flanged and bolted cover. The large diameter equipment hatch is not included in the fault tree models.

Two containment air locks are provided for personnel access to the containment vessel. Each lock has two double gasketed doors in series. Provision is made to pressurize the space between the gaskets for leak testing. These air locks maintain containment integrity while providing a path into and out of containment. Each air lock consists of two doors in series that are mechanically interlocked to assure that one door is closed at all times. The inside containment door provides the first barrier and the outside containment door provides the second barrier. Each door is equipped with quick acting ball valves for equalizing pressure across the doors. The doors will not be operable unless the pressure is equalized. The air lock equalization valves are also part of the containment isolation barrier. One of the valves is located on the air lock bulkhead inside containment and the other is located on the bulkhead outside containment. The valves for the two doors are properly interlocked so that only one valve can be opened at one time, and only when the opposite door is closed and sealed. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidently left open it can be closed by remote control.

Electrical Penetrations

Canisters or header plate penetration assemblies are used for all electrical conductors for the continuation of electrical circuits through the containment vessel, the annulus and the shield building. Sufficient cable slack is provided in the annulus to allow for differential expansion between the containment vessel and the shield building. Cable protection sleeves are provided to give support and protection to the cables in the annular space.

The primary containment penetrations feature hermetic cable sealing achieved by a ceramic, glass or high temperature thermoplastic material bonding to a metal flange. The flange is welded to a header plate or secured by screw threads and a ferrule assembly to a header plate, which in turn is welded to the penetration nozzle. The secondary seal is achieved by either epoxy resin or thermoplastic material forming a continuous seal between the metal canister pipe and all conductors. Both sets of seals provide a containment barrier and are therefore included in the fault tree models. All penetration assemblies are provided with means to pressurize the primary canisters for monitoring of leakage rates.

The primary containment penetration is inserted in a containment vessel nozzle and is field welded inside the steel vessel to form the sealing weld. The secondary seal is inserted in a nozzle embedded in the concrete shell of the shield building aligned with the containment vessel nozzle. The secondary seal is field welded to the nozzle in the shield building. These welds do not provide a containment barrier therefore they are not included in the model.

Spare Penetrations

Spare piping penetrations consist of a short section of pipe that passes through the containment vessel. They are typically sealed closed with a pipe cap on both sides of the containment. The pipe caps may be either threaded on to the end pipe or seal welded. In some cases, however, gasketed blind flanges are used. Based on the limited amount of information associated with these penetrations, it is assumed that they do not pass through the shield building.

Fuel Transfer Penetration

A fuel transfer penetration is provided to transport fuel rods between the refueling transfer canal and the spent fuel pool during refueling operations of the reactor. The penetration consists of a 36 in. diameter stainless pipe installed inside a 48 in. pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment vessel and provision is made for testing welds essential to the integrity of containment. Bellows expansion joints are provided on the pipe to compensate for building settlement and differential seismic motion between the Reactor Building and the Fuel Handling Building.

C.2.4.2 Containment Heat Removal

The Containment Heat Removal System at St. Lucie consists of the Containment Spray System (CSS) and the Containment Cooling System (CCS).

The Containment Cooling System provides an independent means of heat removal following a LOCA. The Containment Heat Removal function can be fulfilled by either the CSS or CCS, or a combination of both.

C.2.4.2.1 Containment Spray System (CSS)

The primary function of the CSS is heat removal from the reactor containment building following a Loss of Coolant Accident (LOCA) to prevent the containment pressure from exceeding its design value.

The CSS has two modes of operation:

a) The initial injection mode, during which the system sprays borated water from the refueling water storage tank into the containment; and

b) The recirculation mode, which is automatically initiated by the recirculation actuation signal (RAS) after low level is reached in the refueling water tank. During this mode of operation, suction for the spray pumps is from the containment sump.

The containment spray system for each Unit consists of two independent and redundant trains (subsystems). The heat removal capacity of either of the two trains is adequate to keep the containment pressure and temperature below design values and to bring the containment pressure below 10 psig within 24 hours after any size break in the reactor coolant system piping up to and including a double-ended break of the largest reactor pipe, assuming unobstructed discharge from both ends.

Containment spray is automatically initiated by the containment spray actuation signal (CSAS) which is a coincidence of the safety injection actuation signal (SIAS) and the high-high containment pressure signal.

Each CSS train includes:

a) A normally open spray pump suction path from the refueling water tank (closes on RAS)

b) A normally closed spray pump suction path to the containment sump (opens on RAS)

c) A containment spray pump

d) A normally open spray pump discharge path through a Shutdown Heat Exchanger to

e) A normally closed air-operated valve which opens on CSAS to direct flow to an independent full capacity containment spray header.

The refueling water tank RWT is an aluminum [stainless steel] tank which provides a reservoir of 525,000 [554,000] gallons of water borated to a minimum of 1720 ppm. The RWT is sized to contain sufficient water to fill the refueling cavity, refueling canal, and the transfer tube to a depth of 24' above the reactor vessel flange joint. While operating in the injection mode, the refueling water tank must supply enough water to allow operation of all Engineering Safety Features (ESF) pumps (including CSS pumps) for at least 20 minutes. The volume required for the injection mode is 305,600 [330,000] gallons. A total required tank volume of 401,800 [417,100] gallons has been established as the Technical Specification minimum tank volume.

Each CSS train has a supply header and four spray nozzle rings located with 178 nozzles per header [178 nozzles in one header and 179 nozzles in the other]. The spray nozzles are of the open throat design and are not subject to clogging. The spray nozzles are located approximately 70 feet above the top of the steam generators.

The containment sump is a large collecting reservoir provided to supply water to the Containment Spray and Safety Injection Systems for recirculation. Located in the containment, the structurally protected containment sump receives all containment drains. The containment sump is provided with a primary and secondary debris filtration system to minimize the possibility of hindering safety injection and spray pump operation. Both sets of screens have sufficient flow area or are oriented to preclude flow restriction to the sump recirculation lines. Particulates under 1/4 in. which manage to pass through the pumps will flow right through the system. Containment spray system nozzles are the non-clog type and have openings of 3/8 in. There is no mechanism by which valves or other fittings between the pump and nozzles will retain any of these particulates.

The only portions of the containment spray system which will be subjected to the containment environment associated with a LOCA are the spray headers, check valves and piping. The remaining portions of the system are located outside the containment in the reactor auxiliary building where the environmental conditions are essentially the same as those prior to the postulated accident.

C.2.4.2.2 Containment Cooling System

The primary function of the Containment Cooling System is to act as an independent means of containment heat removal during a LOCA, and to remove containment heat during normal operations. During normal operation three of the four fan-cooler units operate to maintain ambient containment temperature at less than 120°F. The heat removal capacity of the Containment Cooling System is adequate to keep containment pressure below 10 psig within 24 hours after any size LOCA [Ref. 1].

The Containment Cooling System alone is designed to remove containment heat post-LOCA to reduce containment pressure and temperature. However, this system can be used in conjunction with the Containment Spray System to provide the same results.

The CCS consists of four fan-coil cooling units, a ducted air distribution system, and the associated instrumentation and controls. The heat removal capacity of the coolers alone is adequate to keep the containment pressure and temperature below design values and to bring the containment pressure below 10 psig within 24 hours after any size LOCA. The coolers are also designed to operate during a main steam line break (MSLB) inside containment.

Each fan cooler consists of two banks of 3 [4] copper cooling coils, casing, fan, and motor. The cooling coils are horizontal tube, vertical plate-fin and are mounted on a structural frame. The cooling coils are designed to remove 7.9×10^5 BTU/hr [1×10^6 BTU/hr] during normal conditions and 60 x 10^6 BTU/hr [61.6 x 10^6 BTU/hr] during accident conditions. They are designed to withstand a pressure of 225 psig. Cooling coils are provided with individual drain pans and drain piping to prevent flooding of lower coils by condensed water cascading from upper coils. Condensation is drained to the reactor cavity sump. Cooling water is supplied to the cooling coils by the Component Cooling Water System (CCW) through supply motor-operated valves MV-14-5 and 6, and the return lines are controlled by MV-14-7 and 8. [In Unit 2 the supply valves are MV-14-9, 11, 13, 15 and the return valves are MV-14-10, 12, 14, 16.] These valves are not closed by a containment isolation signal.

The unit 1 fans are centrifugal type, direct-driven, with backwardly curved airfoil blades to provide a non-overloading characteristic. The fan motors are single speed, water cooled AC induction motors. Cooling water is supplied by the component cooling water system. [The unit 2 fans use vane axial flow fans which consist of a multi-bladed rotor assembly mounted directly to the motor shaft. The fan-rotor is of the adjustable pitch type so that air flow can be mechanically adjusted. The two-speed fan motors are not cooled by component cooling water. The fan motors are cooled by air that has been through the cooling coils.]

The containment fan coolers are located outside the secondary shield wall in different quadrants of the containment: three on elevation 45 ft, and one on the operating floor at elevation 62 ft. This arrangement provides separation and minimizes recirculation between units. During normal operation three of four fan-cooler units operate to supply the containment building with 60,000 cfm per fan cooler. The fans are powered from 480V load centers, and are automatically actuated on a Safety Injection Actuation Signal (SIAS) during post-LOCA conditions. Upon receipt of SIAS, the standby fan cooler unit will automatically start [and the units will switch from fast to slow speed] and each of the four fans will supply post-accident heat removal air at a flow rate of 58,000 [39,600] cfm. Each fan motor is a 150 hp [125/83 hp], 460 volt induction type with integral air to water heat exchanger. Each unit is sized to remove one-third of the normal heat load or one-fourth of the accident load. Containment ambient temperature under non-accident conditions is limited to 120°F when the units are supplied with CCW at 100°F.

The duct distribution system is arranged to promote mixing of the containment air and includes a common ring header to assure continuity of design air flows at all outlets. Ducts are of welded construction, reinforced and provided with pressure relief dampers to withstand LOCA induced pressure transients. The ring header is designed to attenuate high pressure transmission from the steam generator area through the duct by having blowout panels in the ductwork from the header to the steam generator and cavity cooling system. There are also gravity dampers rated at 2 psi differential at the point of juncture between the ring header and the ducts to prevent a negative pressure in the duct.

C.2.4.3 Containment Purge System

The containment purge systems in both units exhaust containment air to the atmosphere when an extended containment entry is required. The purge system in each unit removes residual iodine and particulate activity by filtration and due to the fact that the purging evolution replaces containment air with outside air, the concentrations or radioactive noble gases and tritium are reduced. See Figure C-6 for Simplified Schematic.

Unit 2 has an additional purge system as described later in continuous containment/hydrogen purge system for Unit 2. The continuous containment purge is used intermittently during normal operation to counteract containment pressure buildup due to instrument air leakage.

The prefilter racks are fitted with high efficiency prefilters (90-95% ASHRAE efficiency). The HEPA racks are fitted with V-bank carbon absorber cells. The carbon filters are used to improve elemental iodine removal.

The suction side of the purge system is connected through a 48" x 48" duct containing automatic damper DM-25-5A to containment cooling system ring duct header to assure uniform purging of the containment. A 36" x 14" branching duct from forty air inlets located above the water line in the refueling cavity is also connected to the purge system through automatic damper DM-25-5B. These dampers are pneumatically-operated and are controlled by the PURGE-REFUELING SELECTOR SWITCH on RTGB 106 [HVCB] which energizes the actuating solenoid when in REFUELING. DM-25-5A is 100% open during purge and partially open during refueling. DM-25-5B is closed during purge and 100% open during refueling.

When in use, air is drawn from the containment and/or refueling cavity through butterfly isolation valves FCV-25-4, 5, and 6, a debris screen, then into a filter housing that is common to the two parallel, 100% capacity exhaust fans. The exhaust fans, HVE-8A and 8B, exhaust to the plant stack. The filter housing contains a set of medium efficiency prefilters and a bank of HEPA filters. The three flow control valves are pneumatically-operated and are operated by the HVE-8A or 8B control switches on RTGB 106 [HVCB] which controls the actuating solenoids for each valve. The are automatically closed upon receipt of a CIS. Red-open and green-closed indication for FCV-25-4, 4, and 6 is provided on RTGB 106 [HVCB].

The air makeup side of the purge system includes a 12' x 10' air intake louver, a bank of medium efficiency filters, and three 48" diameter butterfly isolation valves designated FCV-25-1, 2, and 3, and a debris screen.

These flow control valves are pneumatically operated and are controlled by the exhaust fan control switches on RTGB 106 [HVCB] and differential pressure between containment and outside. An exhaust fan must be operating and there must be a 0.5-inch differential pressure between containment and outside to open the flow control valves. This prevents unfiltered backflow through the air makeup valves. FCV-25-1, 2, and 3 are automatically closed on a CIS. Red-open and green-closed indication of FCV-25-1, 2, and 3 is also provided on RTGB 106 [HVCB].

<u>Purge Fans HVE-8A and HVE-8B</u>

The purge fans are 480 VAC, belt-driven fans that discharge to the plant vent stack. Each fan is rated at 42,000 cfm and 10.5 inches water gage static pressure. HVE-8A is powered from 480 VAC MCC 1A5 and HVE-8B is powered from 480 VAC MCC 1B5.

The fans are controlled by STOP/START Switches that spring return to neutral (auto) on RTGB 106 [HVCB]. When the switch is taken to START, exhaust butterfly valves FCV-25-4, 5, and 6 open and, through valve limit switches, the fan is started when the dampers have all completely opened. In neutral after stop, the fan is in standby; it will start on a low flow condition at the outlet of running fan. The fan inlet dampers are automatically opened on a fan start signal and automatically closed on a fan stop signal. The purge fans are interlocked to trip on a high containment to outside differential pressure of 0.15 psid. Ten seconds after a low flow condition or motor overload the standby fan starts and actuates the CONTAINMENT PURGE HVE-8A(B) LOW FLOW MOTOR OVERLOAD alarm on RTGB 106 [HVCB].

C.2.4.4 Combustible Gas Control

The hydrogen build-up inside containment is controlled by the containment combustible gas control system. This system provides the capability to monitor and maintain hydrogen concentrations within safe limits after a LOCA and consists of the following subsystems: containment hydrogen analyzers (also called hydrogen sampling), containment hydrogen recombiners and containment hydrogen purge.

C.2.4.4.1 Containment Hydrogen Analyzer Subsystem

The containment hydrogen analyzer system consists of two redundant subsystems, consisting of the sample and return piping, associated valves, hydrogen analyzer, grab sample cylinder, sample pump, moisture separator, cooler, instruments, calibration gasline and reagent gasline.

Each of the redundant subsystems is physically separate and operates independently of the other. Failure of one train is annunciated in the control room. The system is initiated by manual operator action from the control room. No action outside the control room is necessary for system operation.

Air samples are drawn from any of the following sample points:

a) Containment dome

b) Upper Containment

c) Pressurizer enclosure

d) Vicinity of reactor coolant pump (RCP) 1A1, [2A1]

e) Vicinity of reactor coolant pump 1A2, [2A2]

f) Vicinity of reactor coolant pump 1B1, [2B1]

g) Vicinity of reactor coolant pump 1B2, [2B2]

These points provide broad coverage of the containment for hydrogen monitoring and constitute a redundant independent H_2 Sampling System. Sampling lines originating from the containment dome, pressurizer, RCP 1A1 [2A1] and RCP 1A2 [2A2] areas constitute one independent train of the H_2 Sampling System. The other train consists of sampling lines originating from the upper containment, RCP 1B1 [2B1] and RCP 1B2 [2B2] areas. Each train of the sampling lines has a common header inside the containment and penetrates the containment in a separate penetration assembly.

There is adequate mixing of containment atmosphere so that local stratification or pocketing of hydrogen does not occur. The analyzer cubicles are located at elevation 43.0 ft of the Reactor Auxiliary Building (RAB). The analyzer system control panel is located in the control room.

A grab sample cylinder located at elevation 43.0 ft of the RAB is provided to permit hydrogen concentration measurement independent of the containment hydrogen analyzer detector.

C.2.4.4.2 Containment Hydrogen Recombiner Subsystems

The containment hydrogen recombiners control hydrogen in containment by using heat to cause recombination of liberated hydrogen with free oxygen in the air to form water. Two stationary thermal recombiners are located in containment next to the steam generators at a floor elevation of 62 ft.

The hydrogen recombiner system is described in Westinghouse Topical Report WCAP 7709-L. Supplement 1 through 4 of WCAP 7709-L were accepted by NRC on May 1, 1976. It is designed seismic Category I and Quality Group B requirements.

Each recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of containment air to a temperature which is sufficient to cause a reaction between the hydrogen and the oxygen in the air. The recombiner is provided with an outer enclosure to provide protection from water spray coming from the containment spray system. The recombiner consists of an inlet preheater section, a heater-recombination section, a mixing chamber, and a cooling/exhaust section. Mixing of containment air is by the containment fan coolers and their associated ductwork by the turbulence introduced by the containment sprays and by the process of natural diffusion of combustible gas with the containment air. There are no moving parts in the recombiners. Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. This accomplishes the dual functions of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150°F to 1400°F causing recombination to occur. The flow then enters the cooling/exhausting section where the stream is mixed and diluted with cooler containment air in order to discharge the stream back into the containment atmosphere at a lower temperature.

Each hydrogen recombiner system has a removal capacity which is sufficient to limit concentrations of gases within the containment to safe concentrations, i.e., concentrations below the flammability limits. After a three-hour startup period, the recombiner efficiency is 99-100 percent and the effluent does not exceed 100°F above ambient.

The recombiners are located on the elevation 62.0 ft of the containment. They are inaccessible following a LOCA, and as such there is no sharing of recombiners among St. Lucie Units 1 and 2 or with other facilities. The hydrogen recombiners are designed for 40 years normal and one year post LOCA conditions.

The recombiner is started by the operator by manual action from the control room. The operator is alerted when the containment H_2 level reaches three volume percent as signaled by the redundant Class IE alarms of the containment hydrogen analyzer system. Plant procedures require the operator to start the recombiner within 24 hours following a LOCA.

C.2.4.4.3 Containment Hydrogen Purge System

Hydrogen Purge System (Unit 1)

The hydrogen purge system consists of two parallel, 100% capacity fans, a common filter train comprised of a demister, two HEPA filters, two charcoal adsorbers, a motor-operated flow control valve, a motor-operated cooling air control valve, two motor-operated dampers, and the associated ductwork as shown on Figure C-7. This system is non-safety-related except for the containment penetrations and the isolation valves.

Containment air is drawn from the top of the dome through two 3 inch purge lines. These lines are reduced to 2 inches at the penetration of the containment. One of these lines is connected to the filter train through the motor-operated flow control valve. The other line bypasses the filter train and connects to the suction of the purge fans. Both lines are isolated by two normally locked closed isolation valves. These valves function as part of containment isolation.

The filter train consists of a demister to remove moisture from the air to prevent loss of efficiency of the charcoal adsorbers. The demister has an efficiency of 99% for removing entrained water particles of 1 to 5 microns. A thermocouple is located downstream of the demister to provide indication of air flow temperature by a temperature recorder on RTGB 106. The HEPA filters consist of a single cell mounted in a structural frame. Each HEPA filter is factory tested to meet or exceed an efficiency of 99.97% when tested with 0.3 micron dioctylpthalate (DOP) smoke. A moisture detector is located in the air flow downstream of the first HEPA filter. If the relative humidity exceeds 70% a HYDROGEN PURGE SYSTEM HIGH HUMIDITY alarm on RTGB 106, annunciator P-30, will actuate.

The first charcoal adsorber consists of six cells arranged 1 wide by 6 high. The second charcoal adsorber consists of three cells arranged 1 wide by 3 high. Each cell consists of two 2-inch thick flat iodine-impregnated charcoal beds. Iodine-impregnated charcoal is capable of removing 99.9% minimum of iodine with 10% in the form of methyl iodide (CH_3I), when operating at 70% relative humidity. Each charcoal bed is provided with a thermocouple to monitor the bed temperature through a recorder on RTGB 106. A HYDROGEN PURGE/FUEL POOL EXHAUST CHARCOAL ADSORBERS HIGH TEMPERATURE alarm, annunciator P-10, on RTGB 106 actuates at 200°F. A thermocouple is located downstream of the second charcoal adsorber to provide air flow temperature indication.

A 6-inch line permits drawing air directly from outside of the Reactor Auxiliary Building for mixing with the purge flow to allow removal of decay heat generated in the charcoal adsorbers. A locally-operated, motor-operated modulating valve regulates the outside air flow consistent with the purge rate. This valve is interlocked to open when the purge fans start. In case of failure of the motor-operated valve which fails as is, a check valve is in the line to prevent backflow.

The hydrogen purge fans, HVE-7A and HVE-7B, are single speed, centrifugal, vane-axial type fans powered from 480 VAC motor control centers 1A6 and 1B6, respectively. Each fan has the capacity to deliver 500 cfm of flow through the charcoal adsorber which can effectively remove the maximum temperature. A flow switch is located after the gravity damper for each fan to provide a HYDROGEN PURGE FANS LOW FLOW/MOTOR OVERLOAD alarm on RTGB 106, annunciator P-50, at 1125 cfm.

To prevent vacuum buildup in containment when operating the hydrogen purge system a 2" air makeup line is provided. This line allows filtered air from the main containment purge intake plenum to be drawn into containment. A check valve is provided in this line to prevent outflow from containment. The makeup line is isolated from containment, when not in use, by two normally locked closed isolation valves that are part of containment isolation.

<u>Continuous Containment/Hydrogen Purge System (Unit 2)</u>

The continuous containment/hydrogen purge system consists of two parallel 100% capacity fans, a common filter train comprised of a demister, and electric heater, a medium efficiency filter, two HEPA filters, and a charcoal adsorber, three air-operated isolation valves, three motor-operated flow control valves, and the associated ductwork, as shown on Figure C-8. The terms continuous containment purge and hydrogen purge refer to two different air flow paths through the ducting illustrated on Figure C-8. The hydrogen purge path bypasses the filters and is directed to the shield building ventilation system to be filtered prior to discharge to atmosphere. Bypass valve FCV-25-28 is interlocked with FCV-25-35 such that when FCV-25-28 is open, FCV-25-35 is shut thus preventing an unfiltered discharge to the stack. The continuous containment purge flow path passes through FCV-25-20, FCV-25-21, FCV-25-9, the filter assembly, and out through FCV-25-35 to the vent stack. The hydrogen purge flow is a backup method of hydrogen removal. The hydrogen recombiners are the preferred hydrogen removal method. The continuous containment purge system has a misleading name in that system operation is limited to 1000 hours per year and as such is used intermittently.

The continuous containment purge system in Unit 2 provides a pressure reduction flow path from the containment to the atmosphere to compensate for the tendency of the containment pressure to increase due to instrument air system leakage. The system is not needed in Unit 1 because the air compressor supplying containment instrument air takes its suction from the containment atmosphere whereas in Unit 2 the instrument air is piped in from outside of containment. Therefore, in Unit 1, the rise of containment pressure caused by instrument air leakage is counteracted by the operation of the containment instrument air compressor, and the leakage in Unit 2 requires intermittent purging.

Containment air is drawn from the top of the dome and the 55 ft. elevation of the containment by the system exhaust fan through the continuous containment purge filter train. The air is filtered and discharged to either the vent stack or to the shield building ventilation system. The filter train may be bypassed through a motor-operated flow control valve. Containment isolation is provided by two air-operated flow control valves. Each valve, one inside the containment and the other outside the shield building, has an air accumulator that allows valve operation after a loss of instrument air. Valves FCV-25-20 and FCV-25-21 are operated by air via a solenoid valve which is controlled from the HVCB. Each valve has a CLOSE/OPEN switch with red-open and green-closed indicating lights located above each switch. A CNTNS CNTMNT H2 PURGE ISOL VLV CIS OVRRD alarm on HVCB, annunciator X-23, is activated when the switch is in the OPEN position.

A makeup air line is used to prevent drawing a vacuum in the containment during the operation of the continuous containment hydrogen purge system. An air-operated flow control valve inside and outside the shield building provide containment isolation for the makeup line. These valves each have an air accumulator that allows valve operation in the event of loss of instrument air.

The continuous purge isolation valves and fans are interlocked to ensure that the valves will only open when a negative pressure differential is established in the RCB and the purge fans are running. This prevents the possibility of venting the RCB out through the supply line and challenging the RCB vacuum relief system in the event that the supply valves are not opened.

C.2.5 Reactor Cavity and Basemat

The significance of the reactor cavity is related to the determination of water availability and the ultimate disposition of the molten core after vessel breach. The plant-specific details of the cavity configuration were evaluated to determine whether water can accumulate during core damage. The presence (or lack) of water affects pressure loads at vessel breach and debris coolability in the longer term. Without water on the cavity floor, and the debris remaining on the cavity floor forming a pool more than a few inches deep, concrete attack and basemat penetration is likely. Geometric configuration that allows a shallow pool to form can result in a coolable configuration despite a dry cavity scenario. If water is available on the cavity floor (and replenished continuously), concrete attack can be mitigated. For high RCS pressure scenarios, for which high-pressure melt ejection (HPME) of the molten material occurs, the debris can be dispersed to the containment atmosphere, producing pressure loads that can threaten containment integrity. A wet cavity has the potential to mitigate pressure loads during HPME or preclude core-concrete interaction; should the debris configuration not be coolable, an overlying pool can also scrub fission products released during core-concrete interaction.

The St. Lucie configuration is similar to Figure C-3. This cavity has the potential for the reactor to be submerged at vessel breach. A submerged vessel affects vessel bottom head coolability and has the potential to mitigate ex-vessel steam explosions and pressure loads associated with direct containment heating (DCH) during HPME. The volume of the cavity region is 7212 ft^3, with a projected floor area of 458 ft^2. The maximum depth of an overlying pool is 21.0 ft assuming that all of the RWST water inventory is discharged into the containment. The height of the vessel bottom head relative to the cavity floor is 5.14 ft.

The St. Lucie Reactor Cavity contains a containment sump (different from recirculation sumps). The containment sump is the collection facility for various drains inside of containment. It is located in the reactor cavity at approximately the -7' elevation. Equipment drains and floor drains throughout containment are routed by drain piping to drain collection headers.

Any component leakage onto the containment floor would be collected by the drain collection headers. Abnormal sump level increases could be due to leakage from the RCS, CVCS, CCW, main steam, feedwater, or primary water supplies inside of containment. During normal operations, water condensed from the containment atmosphere by the containment cooling unit coils is the principal source of water to the containment sump.

The reactor cavity is formed by concrete walls surrounding the reactor vessel on all sides and below. The floor and walls of the cavity are lined with stainless steel. Extending below and to the outside of the primary shield wall is the reactor cavity sump. Refer to Figure C-4.

Two sump pumps provide a means of removing any water collected in the reactor cavity sump. The sump pumps are started and stopped by magnetrol level switches. The switches are float switches and serve to maintain sump level within specified limits by starting and stopping the pumps. The upper switch will cause the first of the two sump pumps to start at an elevation of -5'4", the second pump starts at -4'8". The lower switches stops both pumps when the reactor cavity sump level drops to an elevation of -6'0".

Associated with the levels previously mentioned are two annunciators. The elevation of -5'4" will trigger the REACTOR CAVITY SUMP HIGH LEVEL alarm on annunciator N-29. An elevation of -4'8" will actuate the REACTOR CAVITY SUMP HIGH-HIGH LEVEL alarm on annunciator N-21.

The reactor cavity sump is provided with a level and flow indication system. Flow into the sump is measured using a weir tank and associated instrumentation. The tank is a ten-gallon tank with a triangular hole cut in the side. As the tank fills, the water flows out of the hole until an equilibrium is established where flow in is equal to flow out. For this level in the tank, there is a unique flow rate associated with it. This allows the tank level to be calibrated to indicate a certain flow. Now if a level reading is taken on the tank a flow signal can be generated.

Weir tank level is measured by two methods. The first method of measuring level is the use of a bubbler system which looks at the back pressure generated by the height of water above the air inlet into the tank and generates a signal proportional to tank level and converts that signal to a flow signal. Flow indication is provided on RTGB 105 [205] by the bubbler system. The recorder provides an alarm signal which annunciates on N-35, REACTOR CAVITY LEAKAGE HIGH, when the flow rate reaches 1 gpm.

A second method of level measurement is accomplished using an ultrasonic level sensor. The flow signal developed by the ultrasonic sensor triggers the REACTOR CAVITY LEAKAGE HIGH alarm on annunciator N-46 when the flow rate reaches 1 gpm.

Reactor cavity sump level indication is provided in the control room by a sump level transmitter. The level transmitter develops a sump level signal which provides sump level indication on RTGB 105 [205].

The St. Lucie cavity configuration is shown in Figures C-4 and C-5. For this configuration the possibility of entrapment exists, which would be similar to Zion.

Table C-1

ST. LUCIE AND REFERENCE CONTAINMENT INFORMATION TABLE

Parameter Description	Units			
PLANT NAME TYPE OF REACTOR MANUFACTURER DATE OF COMMERCIAL OPERATION		ZION PWR Westinghouse 1973	TURKEY POINT PWR Westinghouse 1972 & 1973	ST. LUCIE 1 & 2 CE 1976 & 1983
Reactor Core				
Nominal	MWth	3,236	2,200	2,700
Number of Fuel Assemblies		193	157	217
Core Weight Total	lb	267,350	225,000	
Uranium Dioxide	lb	216,600	181,021	207,186
Zircaloy	lb	44,600	42,704	58,737
Miscellaneous	lb	6,150	-	
Reactor Vessel				
Vessel Diameter	in	173	155.5	172
Water Capacity w/Core & Int. in place	ft^3	-	3,622	
Reactor Coolant System				
RCS Volume (Nominal, including PZR)	ft^3	13,000	9,750	
Water in System (Nominal)	lb	-	9230 ft^3	
Operating Temperature (Nominal)	°F	562	574	572.5
Operating Pressure (Nominal)	psia	2,265	2,250	2,250
Number of PORVs		2	2	2
Lowest PORV Setpoint	psia	2,335	2,350	2,400 [2,370]
Number of SRVs		3	3	3
Lowest SRV Setpoint	psia	2,485	2,485	2,500
Number of Reactor Coolant Pumps		4	3	4
Number of Steam Generators		4	3	2
Containment		Large Dry	Large Dry	Steel
Containment Inside Diameter	ft	141	116	140
Containment Maximum Inside Height	ft	189	169	232
Free Volume	ft^3	2.86E6	1.55E6	2.5E6

Table C-1 (continued)

ST. LUCIE AND REFERENCE CONTAINMENT INFORMATION TABLE

Parameter Description	Units				
PLANT NAME TYPE OF REACTOR MANUFACTURER DATE OF COMMERCIAL OPERATION		ZION PWR Westinghouse 1973	TURKEY POINT PWR Westinghouse 1972 & 1973	ST. LUCIE 1 & 2 CE 1976 & 1983	
Design Leak Rate	%Vol/Day	0.1	0.25	.05	
Design Pressure	psig	47	59	39.6 [44]	
Operating Pressure	psig	14.7	14.7	14.7	
Operating Temperature	°F	100	100	120	
Construction	Type	Prestressed	Prestressed	Steel	Reinforced Concrete
Wall Thickness	ft	-	3.75	2"	3 ft
Dome Thickness	ft	-	3.25	1"	2.5
Basemat Thickness	ft	3.5	10.5	10 ft	10 [12]
Floor Thickness	ft	-	8.33		
Pressure Boundary	Type	Prestressed	Prestressed	Steel	
Liner Thickness, Walls	in	0.376	0.25	-	
Liner Thickness, Dome	in	-	0.25	-	
Liner Thickness, Floor	in	0.375	0.25	-	
Liner Thickness, Cavity	in	-	0.25	-	
Atmosphere, Nitrogen	lb-moles	-	-		
Atmosphere, Oxygen	lb-moles	-	-		
Annular Cavity Radius	ft	-	9.5	11'	
Concrete Type		Limestone	Limestone	Limestone	
In-Core Instrument Room	ft x ft	-	10.5' x 9		
Floor Area (Cavity & ICIR)	ft^2	399	495	458	
Water Capacity (Cavity & ICIR)	ft^3	7,940	7,974	7,212	
Refueling Water Storage Tank	gal	350,000	335,000	335,000	
Recirculation Spray Pumps			(spray)		
Number of Recirculation Spray Pumps		3	2	2	
Design Flow (each)	gpm	2,164	1,450	2,750	
Design Head	ft(psia)	-	470 (315)	470	

Table C-1 (continued)

ST. LUCIE AND REFERENCE CONTAINMENT INFORMATION TABLE

Parameter Description	Units	Reference Plants		
PLANT NAME TYPE OF REACTOR MANUFACTURER DATE OF COMMERCIAL OPERATION		ZION PWR Westinghouse 1973	TURKEY POINT PWR Westinghouse 1972 & 1973	ST. LUCIE 1 & 2 CE 1976 & 1983
Recirculation Spray Heat Exchangers			RHR	RHR
Number of Recirculation Spray		2	2	2
Heat Exchangers				
Design Capacity (each)	Btu/hr.	28E6	29.4E6	
Auxiliary Feedwater System				
Type Drive		Motor	Motor	Motor
Number of Pumps		2	0	2
Capacity	gpm@psig	450 @ 1,343	0	325 [300]
Auxiliary Feedwater System				
Type Drive		Turbine	Turbine	Turbine
Number of Pumps		1	3	1
Capacity	gpm@psig	900 @ 1,343	600 @ 1,203	600[570]
Charging System				
Number of Pumps		3	3	3
Capacity	gpm@psig	150 @ 2,800	77 @ 3,200	44
Capacity @ PORV S.P.	gpm@porv	150	77	44
High Pressure Injection System				
Number of Pumps		2	4	2
Capacity		400 @ 1,084	300 @ 1,000	345 @ 2,500
Capacity @ PORV S.P.	gpm@porv	0	0	0
Accumulators				
Number of Accumulators		4	3	4
Pressure	psig	600	600	200 [600]
Water Capacity (Total)	ft^3	3,400	2625	4,552 [6,000]
Source of Information		NUREG/CR-4551 MAAP User's Manual	MAAP Parameter File PTN FSAR, Rev. 7	MAAP Parameter File and FSAR

Table C-2

VARIOUS CONTAINMENT STRENGTHS

Plant	Type of Containment	Free Volume	I.D.	Design Pressure	Analysis	Failure Pressure	Ratio of Failure to Design Pressure	Dominant Failure
Zion	Large Dry; concrete cylinder w/steel liner (prestressed)	2.86E6 ft^3	141 ft.	47 psig	NUREG-1150 IDCOR 10.1	108-180 psig 149 psig	2.3-3.8 3.17	Leak/rupture in cylinder wall or basemat/wall intersection Hoop/tendon strain
Surry	Sub-atmospheric; concrete with steel liner (reinforced)	1.3E6 ft^3	126 ft.	45 psig	NUREG-1150	95-155 psig	2.1-3.5	Leak/rupture near dome/wall intersection
Indian Point	Large Dry; concrete cylinder with steel liner (reinforced)	2.61E6 ft^3	135 ft.	47 psig	IDCOR 10.1	126 psig	2.7	Hoop rebar yield/cylinder shell near spring line
Seabrook	Large Dry; concrete cylinder with steel liner (pre-stressed)	2.76E6 ft^3	140 ft.	60 psig	Seabrook Risk Management Study 1985	211 psig (wet seq.) 190 psig (dry seq.)	3.5 3.1	Feedwater penetration
Oconee Unit 3	Large Dry; concrete cylinder with steel liner (prestressed)	1.91E6 ft^3	116 ft.	59 psig	Oconee PRA	162.5 psig	2.7	2.5 x design pressure, rupture of prestress tendons and liner failure
Yankee Rowe	Large Dry; bare steel sphere	1.02E6 ft^3	125 ft.	34 psig	IDCOR 10.1	84 psig	2.5	Hoop yield/steel sphere
St. Lucie - 1	Large Dry; steel cylinder	2.50E6 ft^3	140 ft.	44 psig(1)	NUREG/CR-4870	95 psig	2.2	Twice Yield Strain
Sequoyah - 1	Ice condenser with steel cylinder	1.26E6 ft^3	115 ft.	10.8 psig	NUREG-1150 IDCOR 10.1 NUREG/CR-4870	40-95 psig 58 psig 60 psig	3.7-8.8 5.4 5.6	Gross rupture in the containment or rupture in the lower compartment Hoop Yield Twice Yield Stress
McGuire - 1	Ice condenser with concrete cylinder and steel liner		115 ft.	28 psig	NUREG/CR-4870	84 psig	3	Twice Yield Stress
Watts Bar - 1	Ice condenser with steel cylinder		115 ft.	15 psig	NUREG/CR-4870	98 psig	6.5	Twice Yield Stress
Turkey Point	Large Dry: Concrete cylinder with steel liner (prestressed)	1.55E6	116 ft.	59 psig	PTN IPE Submittal	146 psig	2.47	Liner Tearing

(1) 44 psig based on LOCA ECCS performance analyses results, not on structural integrity.

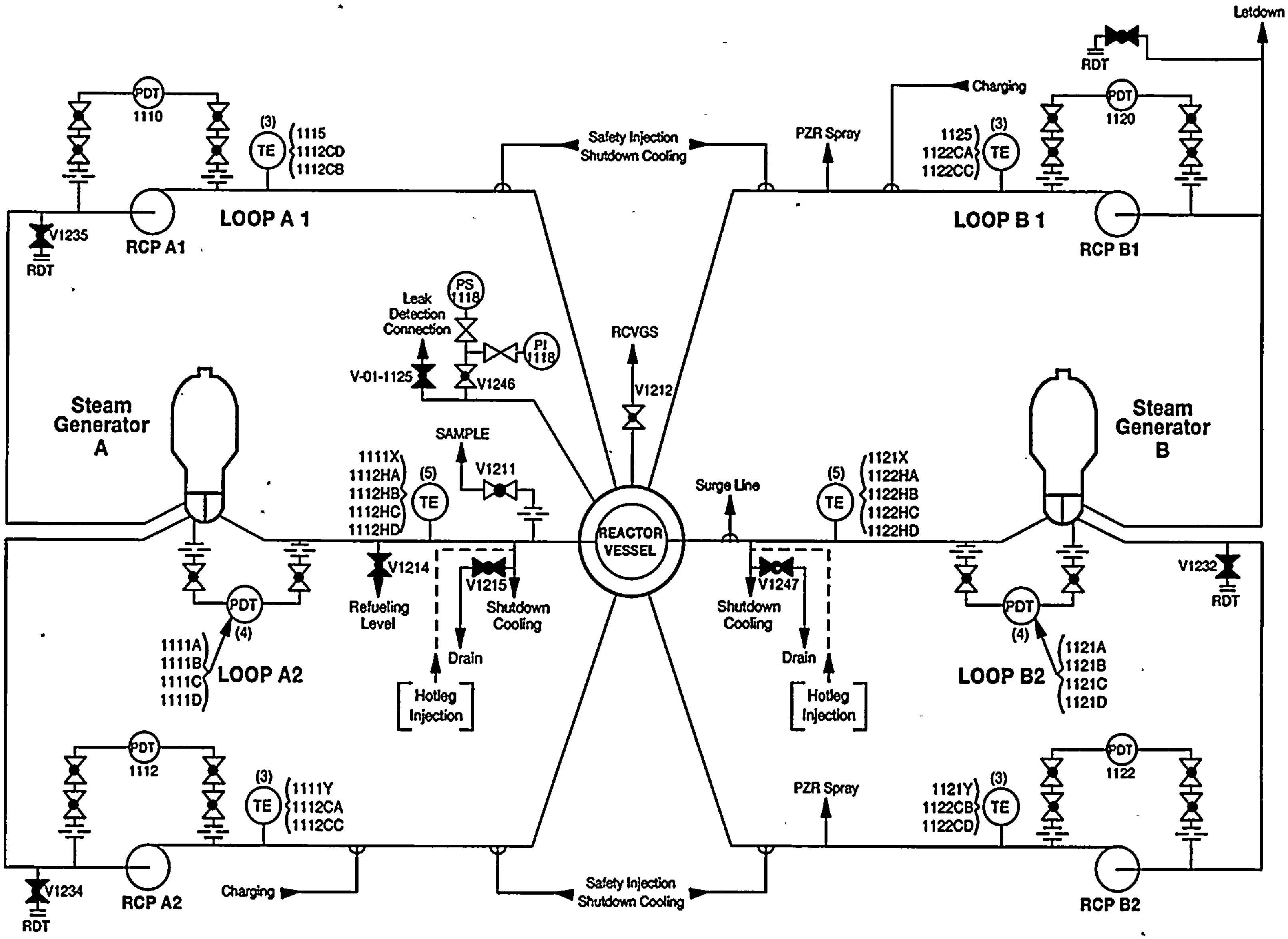
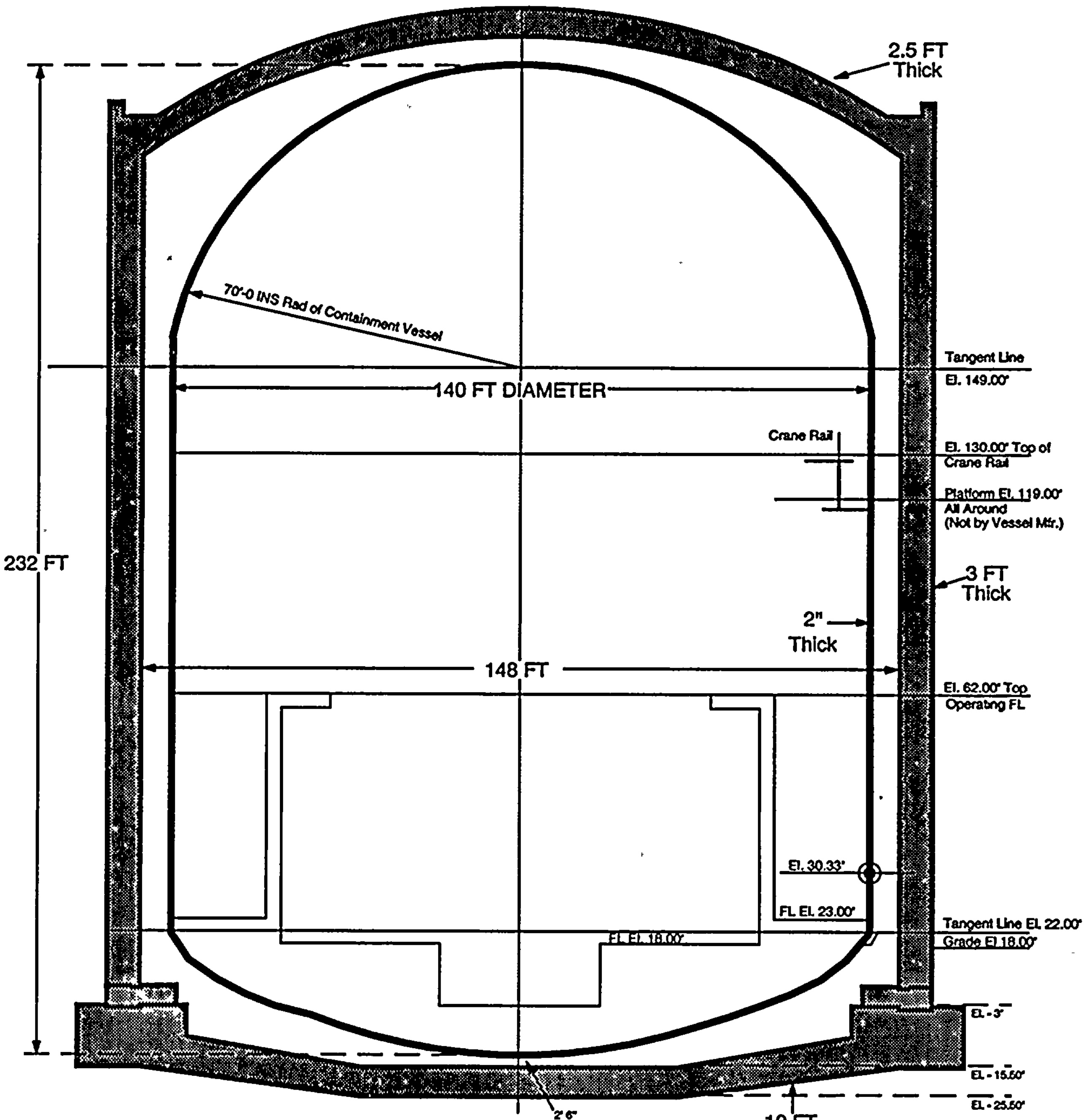


Figure C-1 RCS Simplified Diagram

C:\LEVEL2\T7FIGA1.DRW

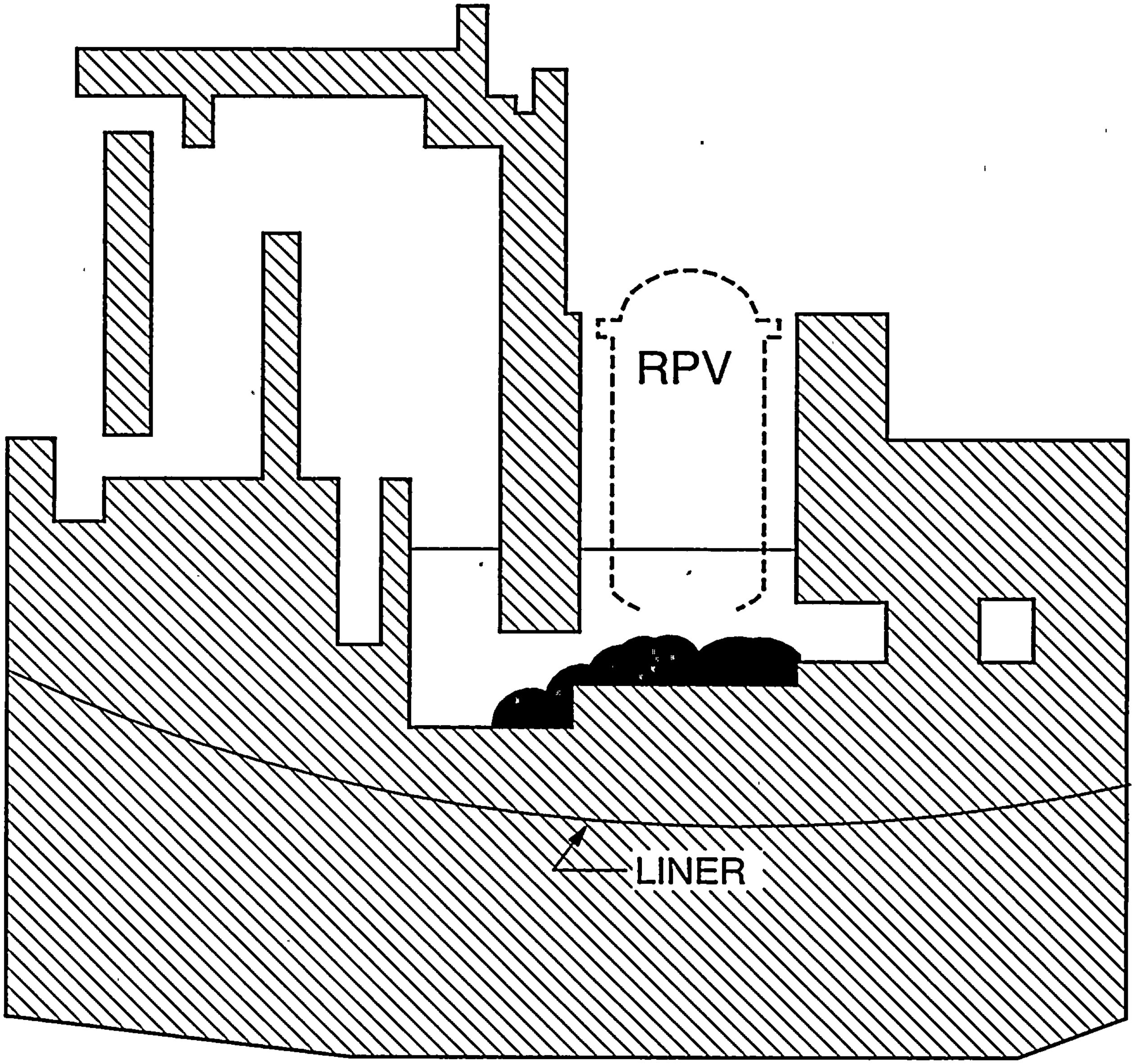
Figure C-2 St. Lucie Unit 1 Containment Structure



NOTE: DRAWING NOT TO SCALE

C:\LEVEL2\PSLCNTU1.DRW

Figure C-3 Cavity Showing Potential for Reactor to be Submerged at Vessel Breach



C:\LEVEL2\T7FIGA3.DRW

Figure C-4 St. Lucie Reactor Cavity Sump

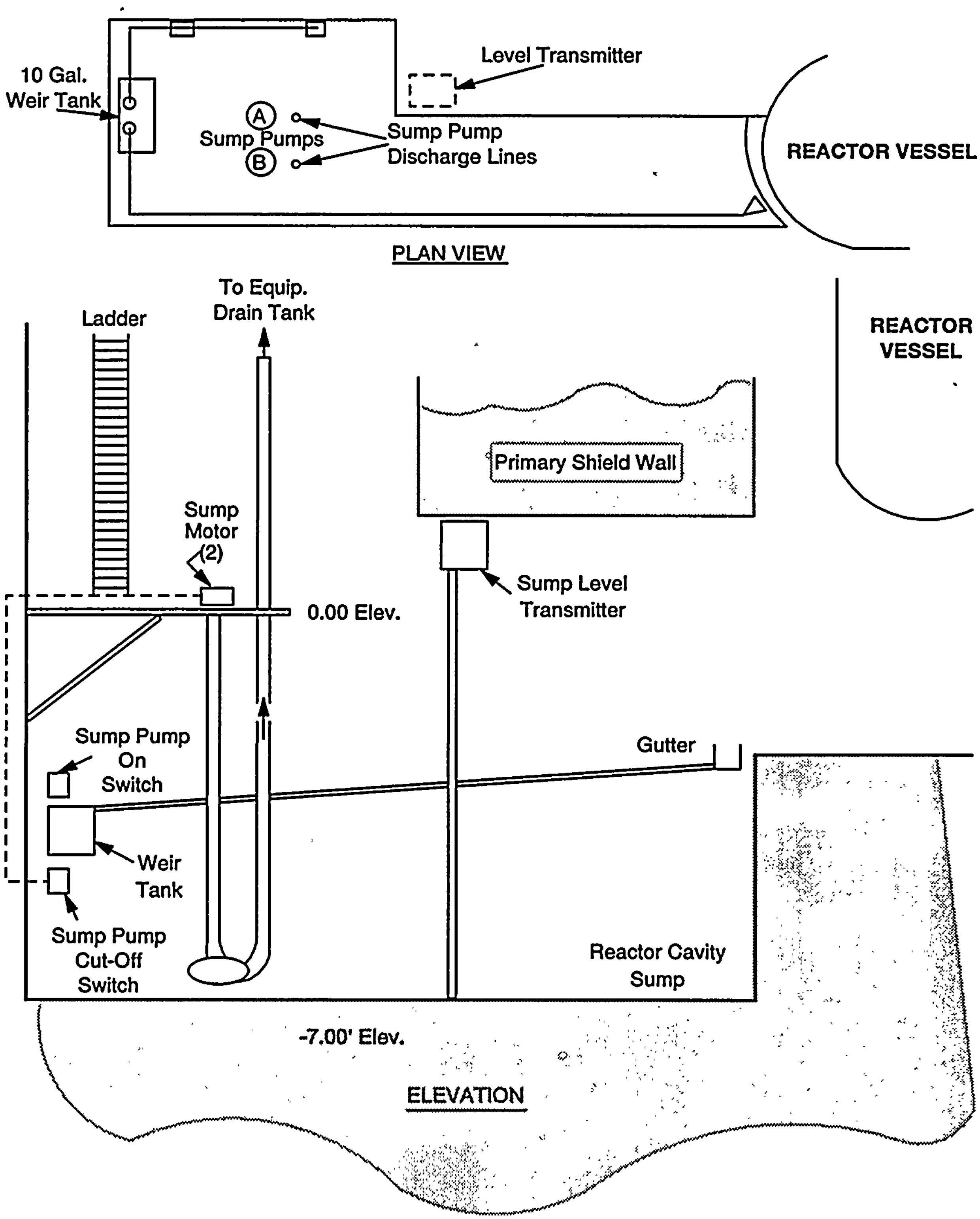


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Figure C-5 St. Lucie Reactor Cavity Dimensions

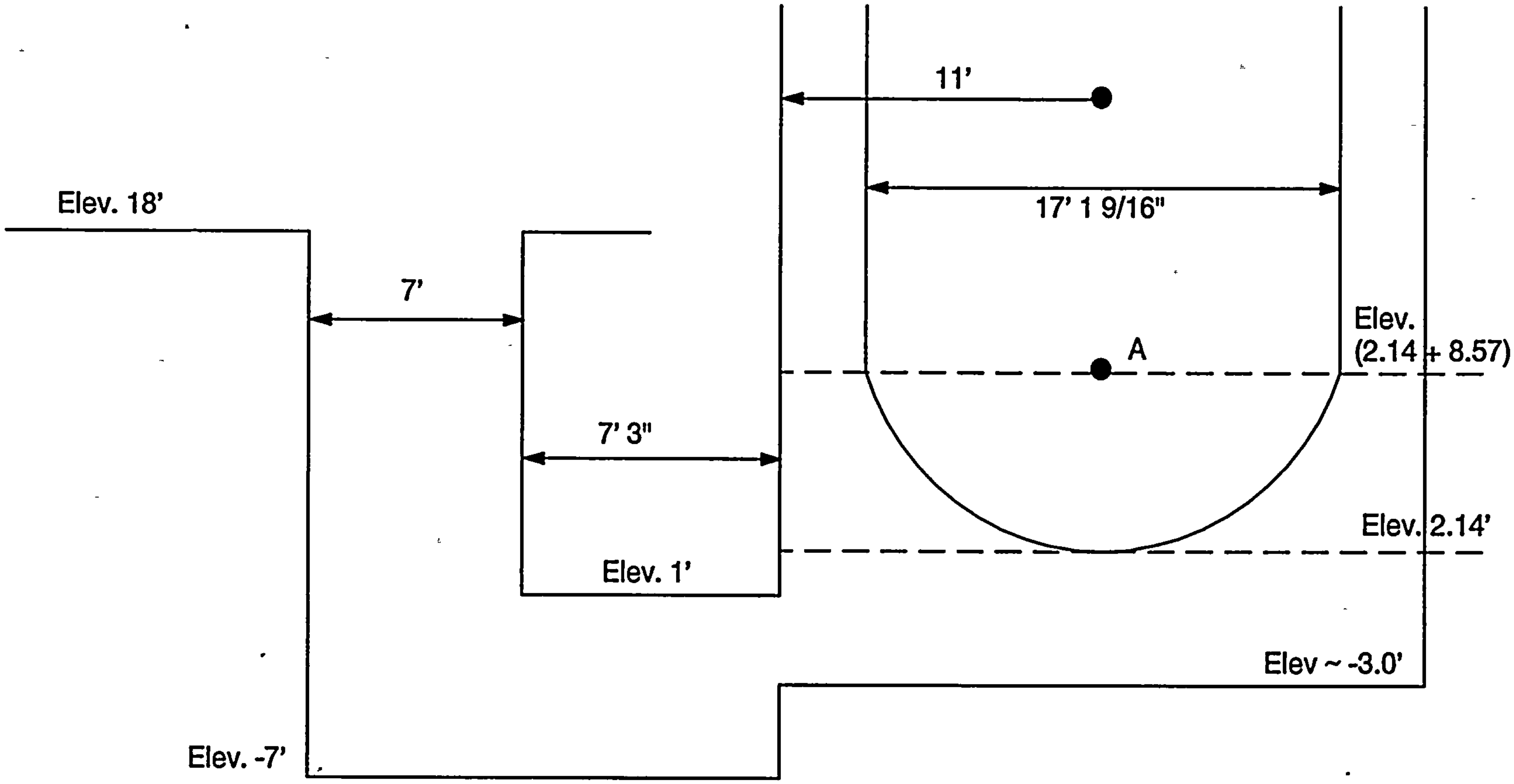

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Figure C-6 St. Lucie Containment Purge System

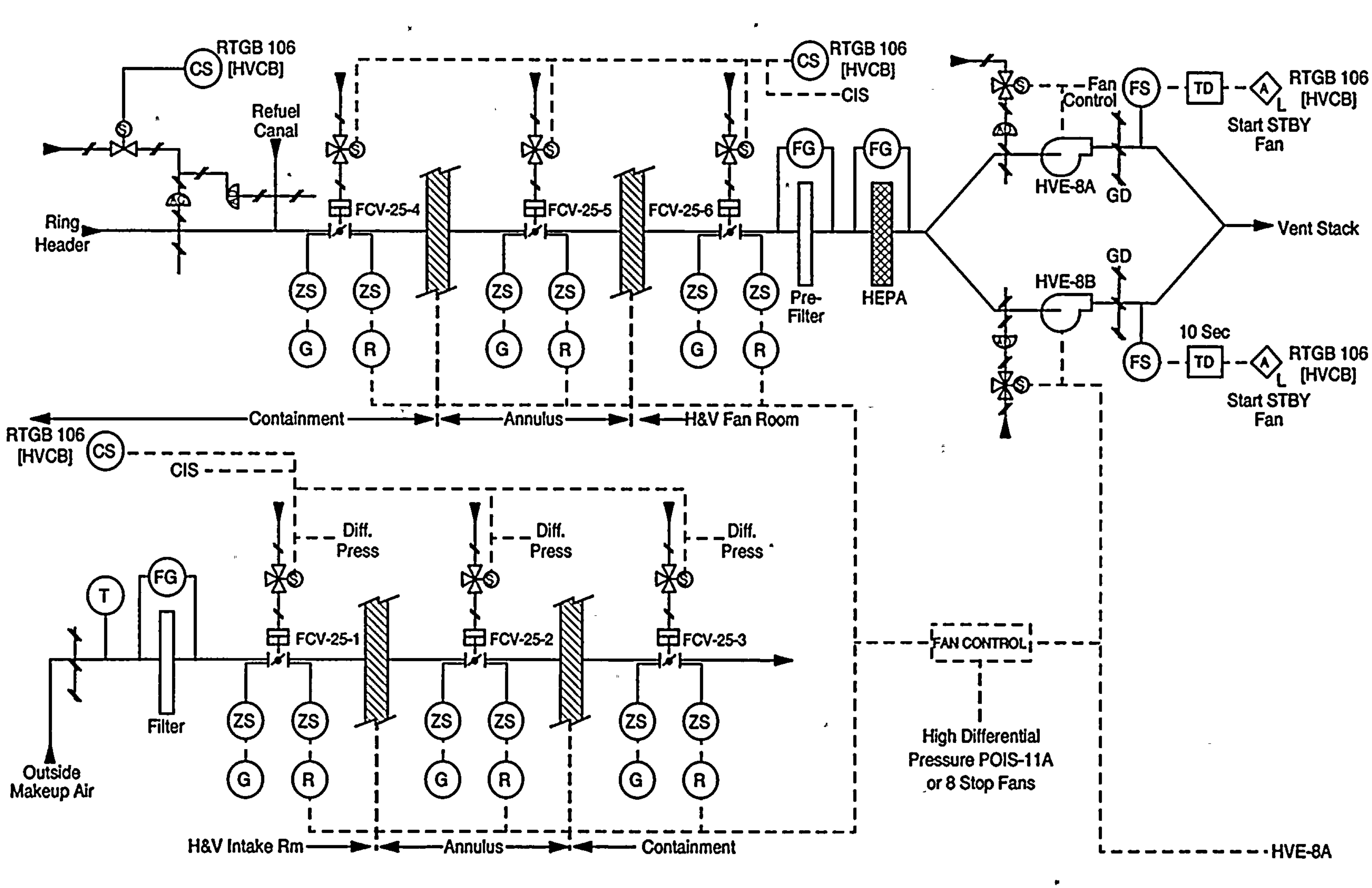

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Figure C-7 St. Lucie Unit 1 Hydrogen Purge System

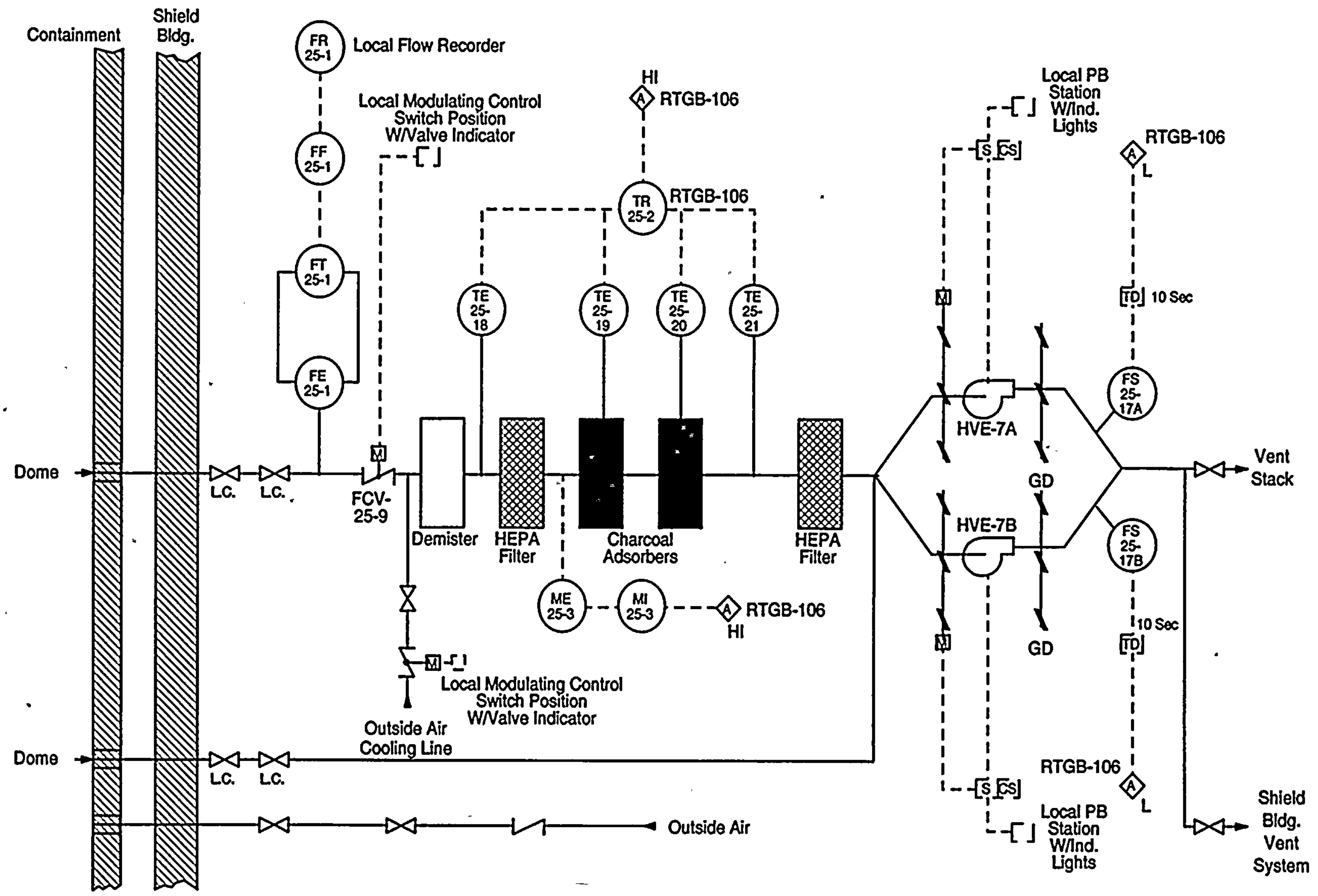

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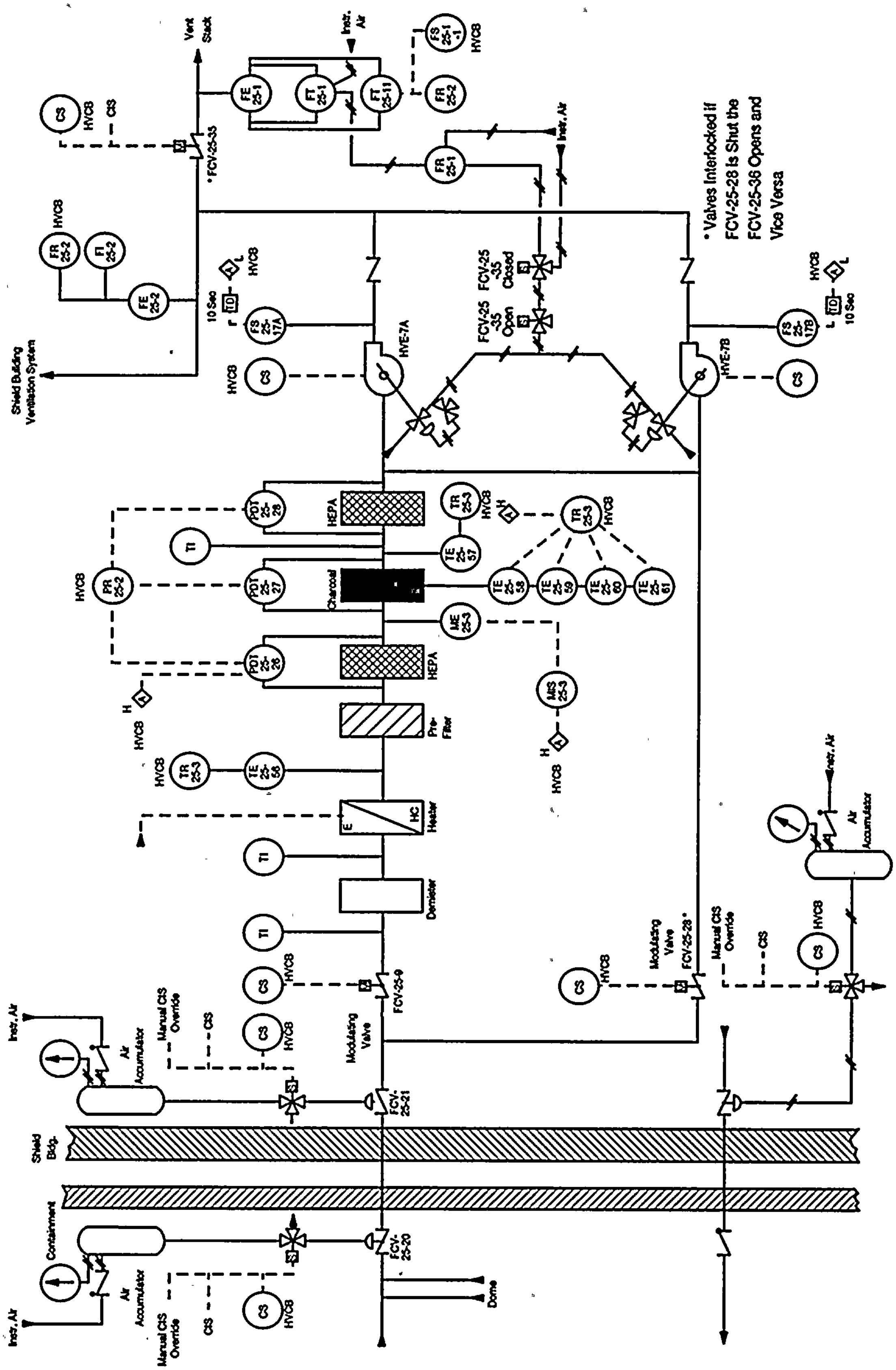
Figure C-8 St. Lucie Unit 2 Hydrogen Purge System



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APPENDIX D

ST. LUCIE UNITS 1 & 2

PLANT DAMAGE STATES BINNING CRITERIA

TABLE OF CONTENTS

D.1 INTRODUCTION

This section groups accident sequences identified in Level 1 study into plant damage states (PDSs). A plant damage state, by definition, is a group of core damage sequences, that have similar characteristics with respect to the severe accident progression and containment performance. Binning of the core damage sequences into a few PDS, is based on the following general characteristics:

- Reactor coolant system (RCS) condition during core degradation;
- Containment condition before/during core degradation; and
- Containment safeguards system performance.

The objective of grouping the large number of accident sequences identified in the Level 1 analysis is to collapse the spectrum of core damage accident scenarios into a manageable set of representative PDSs. Within each of the PDSs, a single assessment of the containment response and fission product release pathways can be made, for which source terms can be estimated.

The containment performance analysis starts with a set of PDSs that define the boundary conditions of the core damage sequences identified in the Level 1 analysis (front-line sequence characteristics). These core damage bins, along with containment systems conditions, provide the entry point to the containment event trees (CETs). The PDSs are used in the evaluation of the containment response to severe accident phenomena.

The physical parameters with similar implications for the containment response and release of fission products to the environment provide the attributes for binning. Binning of core damage sequences into PDSs is conducted systematically through the use of a bridge tree. The development of the bridge tree and the criteria used for defining PDSs are described in this section.

D.2 PLANT-DAMAGE STATE BINNING ATTRIBUTES

There are primarily a few core and containment conditions that have important implications in the determination of containment performance during core-damage progressions. The plant-damage state binning criteria include unique combinations of the core-damage sequence (i.e., reactor/containment conditions) and the containment safeguards systems performance that affect containment response to loads and fission product release. The factors considered in defining plant damage state bins are described below:

D.2.1 Core Melt Timing

The time the core melt starts after shutdown determines the decay heat power level directly affecting the rate of core melt and energy loads in containment. It is also a key parameter in the determination of potential consequences (i.e., time of release of fission products to the environment). The distinction in the time of core melt initiation for determining the rate of core melting is important only for large time differences (i.e., 1 hour or 1 day). However, for purposes of characterizing the time of release and its implications on potential off-site consequences, three time periods are defined:

1. Less than 2 hours;
2. Greater than 2 but less than 6 hours; and
3. Greater that 6 hours.

These times are selected on the basis of rapid coolant loss for large LOCAs (< 2 hours), or transient (cycling SRVs) or small LOCA events without emergency feedwater (2-6 hours) when coupled with emergency core cooling system (ECCS) injection failure. Failure of the ECCS at recirculation for either LOCAs or transient events would likely extend the time to core melt, significantly exceeding 6 hours, especially if the water inventory for injection (such as the refueling water tank (RWT)) is significant and competing flows (i.e., to containment sprays) from the RWT do not deplete the water source for injection.

D.2.2 RCS Pressure

The RCS pressure at reactor pressure vessel melt-through is a key parameter in determining debris dispersal and attendant Direct Containment Heating (DCH). Moreover, loss of integrity of the RCS pressure boundary leading to depressurization of the primary system prior to vessel breach (e.g., hot leg or surge line failure at high temperature) is also influenced by the reactor vessel pressure. (According to the Surry PWR analysis for NUREG-1150, a vessel pressure at the SRV setpoint of 2500 psig would most likely lead to hot leg or surge line piping failure). A pressure of less than 200 psig in the reactor vessel at the time of vessel breach has been selected as the threshold for significant debris ejection (NUREG-1150). Consequently, pressure ranges corresponding to transient events (SRV setpoint) (> 2000 psig) and small-small LOCAs (less than 2000 but greater than 200 psi) are defined. For small and large LOCAs, the primary system pressure can substantially drop below 200 psig. A pressure below 230 psia [600 psia] in the primary system represents a cut-off for Safety Injection Tank (SIT) actuation, for which another PDS may be defined. Actuation of the SIT can delay core uncovery, given ECCS injection failure. Based on this discussion, three reactor pressure ranges (> 2000 psig, 200-2000 psig and < 200 psig) were selected to distinguish core-damage sequences which could affect subsequent events in the accident progression.

D.2.3 Containment Pressure Boundary Status

The containment status at the time of core-damage directly influences the potential for fission product release. An initially impaired containment (i.e., bypassed or unisolated containment) would lead to early release of radioactive material from the containment. If the containment is isolated, it would meet the design leakage criteria, thus precluding early release of fission products to the environment. In this case, containment challenges imposed by the core melt sequence potentially leading to failure will be evaluated. Containment isolation failures determine the leakage level from the containment which in turn affects subsequent pressure loads.

Containment penetrations that are directly connected to the containment atmosphere (or interface with the RCS) provide a potential for discharge of fission products released from the fuel directly to the environment. Should these penetrations fail to isolate, fission products are released early and the magnitudes are likely to be significant. The important penetrations considered are those that do not connect to closed systems; those that do connect to closed systems tend to mitigate the consequences of a release if these penetrations fail to isolate. Isolation failures may be grouped into two classes: large or small. Large diameter piping penetrations would likely preclude containment pressurization while small diameter piping penetrations may not preclude containment pressure challenges. In either case, early releases of fission products from containment would occur, and the size of penetration failure would determine the rate of release (and ultimately the duration of fission product release to the environment). Ideally, models of containment isolation systems should classify isolation failures as small, medium or large to allow a distinction to be made regarding the leakage rate of fission products from the containment (source-term characterization) for off-site consequence assessments. This classification was not performed due to large uncertainties associated with failure mechanisms of isolation valves or penetrations beyond design conditions.

D.2.4 Containment Safeguards Status

The availability of containment heat removal functions and fission product removal systems would be particularly relevant for core-damage states where the containment is initially intact. The decay heat removal systems mitigate the pressure loads during core coolant boiloff and uncovery. The resultant containment pressure during core melting affects the likelihood for failure for a given pressure load imposed on the containment, particularly during vessel breach. The initial pressure level is usually an outcome of the core-damage sequence. A slightly elevated pressure is possible, for example, for a core-damage sequence that is initiated by loss of core cooling at recirculation (and coincident failure of the heat removal function). A moderate pressure rise in containment as a result of vessel blowdown at vessel breach could potentially challenge containment integrity if the initial pressure is elevated.

Several combinations of the containment safeguards performance are considered in the binning process:

- Containment sprays are operating before core-damage and vessel breach, and continue to operate upon recirculation;
- Containment sprays successfully inject before core-damage but fail at recirculation;
- Containment sprays are failed, and do not actuate when called upon; and

- Containment sprays are available, but are not called upon before core-damage (i.e., containment spray set-point is not exceeded). Sprays will inject at vessel breach when spray setpoint is reached.

Each spray status is combined with and without the fan coolers successfully operating. Thus, there could potentially be eight combinations of the containment safeguards states that will be coupled with each of the RCS and containment states. Containment isolation failures were not explicitly included in the containment safeguards status; but included in the containment event tree top event logic.

D.2.5 Water Availability in Cavity

The presence (or lack) of water in the cavity is a function of both the cavity configuration of St. Lucie Units 1 & 2 and the particular scenario in progress. For plant designs such as St. Lucie that would allow water to accumulate in the cavity provided RWT water is injected to the containment, the cavity would likely be filled with water (wet cavity) at vessel breach. Other plant configurations, on the other hand, may not allow overflow to the cavity of the water discharged from the RCS to the containment floor. For such a cavity configuration, spray actuation may be required to deliver sufficient water to the cavity. A wet or dry cavity is important in the determination of containment challenges at vessel breach, and the likelihood of containment failure. Since this is dependent on systems actuation, it is used in the characterization of PDS definitions.

Water availability in the reactor cavity at vessel breach is implied by injection of the RWT water inventory to the containment leading to flooding of the containment sumps and overflow to reactor cavity. This is implied either by ECCS recirculation failure or sprays actuation prior to vessel breach. If both of these conditions are not true, then the implication is that the PDS has a dry cavity.

D.2.6 RCS Retention Capability

For core-damage sequences involving a LOCA, fission product aerosols carried by the hot gases exiting the core to the containment can either be deposited on cooler RCS surfaces or be swept away without significant attenuation. The extent of deposition on the RCS structural surfaces would depend on the location of the break relative to the steam generators. If the break occurs in the hot leg, fission product aerosols are not likely to be deposited significantly. Conversely, if the break is in the cold leg, significant deposition in the steam generators is expected. Moreover, if secondary side cooling is available, retention in the RCS is certain given cold leg breaks. RCS retention during core melting can be influenced by the break location as modeled in previous PRAs (e.g., Oconee PRA):

- Cold leg break (leakage path through steam generators with secondary side cooling); and

- Hot leg break (leakage bypasses steam generator or steam generators boil dry) similar to cycling or stuck open SRV flows where the leakage path bypasses steam generators.

An RCS break in the cold leg through the steam generator would allow more effective retention than would be obtained for hot leg break LOCAs. However, no distinction is made in the modeling and quantification of the event sequences (Level 1) for hot or cold leg breaks.

The binning criteria developed for the St. Lucie PRA conservatively combines cold leg breaks with the hot leg break LOCAs. A distinction in the characterization of the fission product retention in the vessel during core-damage is not made. It is assumed that all LOCAs occur in the hot leg of the RCS coolant loop.

D.3 CORE DAMAGE FUNCTIONAL SEQUENCE BINNING

Each of the core damage functional sequences identified in Level I were characterized based on the binning factors presented above. For each functional event tree (Figures 3.1-1 through 3.3-6) the core damage sequence(s) bins are discussed below. Table D-2 describes the core damage bins.

D.3.1 Binning for Transient Event Tree

Sequence 2 represents a scenario in which secondary heat removal is available initially but long term cooling fails. The sequence is classified as core damage bin III.

Sequence 4 represents a scenario in which secondary heat removal is not available but once through cooling is available for the short-term with long term cooling failure. The sequence is classified as core damage bin IV.

Sequence 5 represents a scenario in which secondary heat removal and once-through cooling are not available and the core uncovery is assumed to occur within 2 hours after the accident and with corresponding RCS pressure at greater than 2000 psig. This sequence is classified as core damage bin III.

Sequences 7, 8, 10 and 11 are transient-induced small-small LOCAs and are similar to sequences 2, 3, 5 and 6 of the next section (D.3.2), respectively.

D.3.2 Binning for Small-Small LOCA Event Tree

Sequence 2 involves a Small-Small LOCA in which secondary heat removal and early core cooling are available but long term core cooling is not successful. MAAP calculations indicate that core uncovery occurs between 3 and 13 hours and the corresponding RCS pressure at the time of core uncovery varies between 350 and 1000 psig. This sequence is classified as core damage bin II.

Sequence 3 involves early failure of core cooling, but secondary heat removal is available. The core uncovery is calculated to occur in less than 2 hours. Corresponding RCS pressure of 1000 psia is calculated. This sequence is classified as core damage bin I.

Sequence 5 is a Small-Small LOCA with no heat sink (i.e. no steam generator cooling) with early bleed and feed cooling but no long term core cooling. Sequence 5 is classified as core damage bin II.

Sequence 6 is a Small-Small LOCA with no heat sink and no bleed and feed cooling. Core uncovery occurs within 2 hours with corresponding RCS pressure less than 2000 psig depending on the size of the break. Sequence 6 is classified as core damage bin I.

D.3.3 Binning For Small LOCA Event Tree

Sequence 2 represents a Small LOCA in which early core cooling is available, but long term core cooling fails. The time at which core uncovery occurs is determined by the time when recirculation occurs and is calculated to be between 2 and 6 hours. The RCS pressure at which core damage occurs is less than 200 psig. Sequence 2 is classified as core damage bin VI.

Sequence 3 is a Small LOCA in which no early core cooling is available. Core uncovery is calculated to occur within 2 hours and the corresponding RCS pressure is below 200 psig. This sequence is classified as core damage bin V.

D.3.4 Binning for Large LOCA Event Tree

Sequence 2 is similar to sequence 1 except cold leg recirculation failure occurs within 2 hours and is classified as core damage bin VI.

Sequence 3 is a Large LOCA with early core cooling failure. The time at which core recovery occurs is assumed to be within 2 hours with corresponding RCS pressure of less than 200 psig. This sequence is classified as core damage bin V.

D.3.5 Binning for Steam Generator Tube Rupture

Sequence 2 involves SGTR with successful secondary heat removal initially but failure of long term core cooling. The sequence is classified as a core damage bin IIR.

Sequence 4 involves SGTR with successful secondary heat removal but failure to isolate ruptured steam generator. Although high pressure safety injection is available long term core cooling fails. This scenario is classified as a core damage bin IIR.

Sequence 5 represents SGTR with no heat sink (i.e. no secondary heat removal) but with short term HPSI and long term core cooling fails, leading to core uncovery. This sequence is a core damage bin IIR.

Sequence 7 is SGTR involving early failures of heat sink and is classified as core damage bin IR.

D.3.6 Binning for Anticipated Transient Without Scram

Sequence 3 involves ATWS with successful initial response (i.e. overpressure avoided by turbine trip, emergency boration, PORV/SRV opening and appropriate initial core conditions such as favorable moderator temperature coefficient) but failure of early core cooling. The core damage is assumed to occur within 2 hours with corresponding RCS pressure at above 2000 psig.

Sequence 4 represents ATWS with no secondary heat removal. The core damage is assumed to occur within 2 hours with corresponding RCS pressure at above 2000 psig.

Sequence 6 involves ATWS with initial PORV/SRV opening and subsequent failure of any of these valves to reclose. Early core cooling is available but long term core cooling fails. The core damage is assumed to occur between 2 hours and 6 hours with corresponding RCS pressure between 600 psig and 2000 psig.

Sequence 7 is similar to sequence 6 except early core cooling is not available. The core damage is assumed to occur within 2 hours with corresponding RCS pressure of above 2000 psig.

All ATWS sequences are classified as core damage bin III.

D.4 BRIDGE TREE DEVELOPMENT

The relationship of the functional sequences identified in the Level 1 portion of the PRA (RCS and core cooling status) and the containment safeguards status (see Figure D-1) are developed in a bridge tree. The entry state to the bridge tree is the set of core-damage sequences evaluated in the Level 1 analysis. The bridge tree maps out the relationship of the core damage bins and the containment safeguards state discussed in the previous section. The structure of the bridge tree is similar to that of an event tree, for which the top events would include core/containment status and containment safeguard systems-related questions. These generally consider the dependency of the containment safeguard systems to the entry state conditions of the functional sequence. This allows a structured approach to cross check the core-damage sequence into PDSs. Table D-1 provides a list of nodal questions used for developing PDSs. The PDSs in the bridge tree denote a core-damage sequence ID with an added dimension of containment and phenomenological parameters judged to be important to the release characterization.

The bridge tree is essentially a post-core-damage event tree that is structured to display the relationships of the various physical parameters and systems described in the previous section. The entry state of the bridge tree is a functional sequence leading to core melt (defined in terms of the cut sets determined in the Level 1 analysis). The subsequent event nodes are questions related to the attributes of the physical parameters described above. These attributes are considered in the development of the bridge tree for binning accident sequences into unique PDSs. The advantage of using a bridge tree (versus defining a matrix of attributes) is principally the ability to account for the dependencies of the various component failures and conditions considered to be important in severe accident analysis within a logic tree framework. The cut sets defining the core-damage state and the system fault trees developed for some of the containment mitigating systems can be tracked in an integrated logic model.

D.5 PLANT-DAMAGE STATES DEFINITION

The spectrum of core and containment conditions following core-damages are portrayed in the bridge tree as a set of end states which are identified with a sequence ID. The bridge tree end states that have similar implications on the containment response and radionuclide release characterization are called PDSs. The sequence ID's follow a simplified nomenclature to categorize the front-line sequence characteristics of the core-damage sequences (i.e., I, II, III, IV, V and VI) that is coupled with an added identifier for the containment safeguards state (A, B, C, D, E, F, G, and H). They are briefly described in Tables D-2 and D-3. The criteria for binning the core-damage sequences into PDSs are principally:

- RCS pressure (as implied by the initiator type):
  - A Large LOCA.
  - S_2 Small LOCA.
  - S_1 Small-small LOCA
  - T Transient event (no RCS breach).
- Time of core melt initiation:
  - E Early (<2 hours)
  - D Delayed (2 to 6 hours).
  - L Late (>6 hours).
- RCS pressure at vessel breach:
  - H High (>2000 psig).
  - M Moderate (200-2000 psig).
  - L Low (<200 psig).
- Initial containment condition prior to core-damage:
  - I Intact (leakage within design).
  - U Unisolated (leakage exceeds design e.g., unisolated or failed).

- B Bypassed (leakage from RCS bypasses containment).

- F Failed (overpressure failure due to loss of containment heat removal and steam generation prior to core-damage).

• Containment mitigating systems availability:

  - Containment sprays operating during core degradation, injection and/or recirculating mode.

  - Containment sprays not operating during core degradation, but may be actuated due to pressure rise at vessel breach.

  - Containment sprays failed without potential for recovery during core degradation.

• Fan cooler availability during core degradation.

The attributes of the PDSs are summarized in Table D-4. The final PDS state definitions are determined based on the binning criteria indicated above and the quantification of the bridge tree. In addition, specific PDSs are defined for the following bridge tree end states:

1. Containment bypass events where releases occur directly to the environment or auxiliary building; and

2. Loss of AC power; where the time-dependent recovery of power could affect subsequent recovery of core cooling or containment systems following core-damage.

The two bridge tree end states above require a unique characterization of the severe accident progression due to the nature of release paths and recovery options. In the final analysis, containment performance is evaluated only for the dominant PDSs.

The binning criteria which follow and scoping calculations using the MAAP code provide unique combinations that are physically and functionally possible for St. Lucie. Table D-5 define the PDSs developed for the St. Lucie PRA. More detailed representation of the PDSs defined for the St. Lucie PRA allowed less complicated yet realistic CETs to be developed and quantified, principally focusing analysis on the important phenomena, containment response, and fission product releases.

Table D-1

BRIDGE TREE NODAL QUESTIONS FOR CROSS-CHECKING PDS BINS

Node	Event Description/Question	Comments	
CM	Entry state		End state of functional sequence/core-damage states.
AC	Is AC power available?	Yes	Implies availability of other systems.
		No	Time dependent recovery of power following core degradation can lead to significant mitigation of the accident without active operator intervention.
RCS	Is the RCS break sufficient to depressurize the vessel?	Yes	Implies low vessel pressure at vessel breach (large LOCAs)
		No	Implies high pressure sequence likely at core melt (transient, medium and small LOCAs are modeled as it could potentially affect systems setpoints and RCS fission product release flow paths).
CB	Is the containment not bypassed?	Yes	Releases occur within containment.
		No	Release bypass containment.
CI	Is the containment isolated?	Yes	Containment is initially intact with potential for failure due to pressure and thermal loading as a result of the severe accident progression, unless mitigating system are available.
		No	Containment is initially impaired and leakage rates may preclude pressurization. Retention within containment would not be as effective, although mitigation of releases is still possible.
CF	Is containment failed before core damage?	Yes	Containment failure occurs due to steam over-pressure prior to loss of coolant makeup and core damage (implies potential for uncontained releases during core degradation).
		No	Similar in nature to isolated containment, although the initial containment pressure may be elevated.

Table D-1 (continued)

BRIDGE TREE NODAL QUESTIONS FOR CROSS-CHECKING PDS BINS

Node	Event Description/Question		Comments
SG	Is the steam generator[1] along the RCS leak path with secondary cooling?	Yes	Implies effective RCS retention and potential for delaying core damage. Long term heatup of the primary system generally result in revolatilization of initially deposited fission product aerosols. This is particularly important for small LOCAs with the steam generator in the release pathway.
		No	Implies degraded RCS radionuclide deposition and heat removal capability for transient and LOCA initiated events. For transient events, there is also a potential for induced steam generator tube rupture, a contributor to bypass events.
ECI	Is core coolant makeup not available initially?	Yes	Core damage occurs shortly after initiating event. (For purposes of the potential consequences, a time frame of less than two hours is defined for early damage.)
		No	Core damage occurs upon recirculation failure or loss of coolant makeup is slow. (Two time frames may be defined, $2<t<6$ hours and $t>6$ hours. However, if the conditional probability is not significant, the time period may be lumped into greater than two hours.) Failure at this branch implies delayed core melt.
FC	Do the fan coolers actuate?	Yes	Implies containment heat removal function. This also implies forced circulation of the containment atmosphere which may mitigate localized concentrations of hydrogen.
		No	Implies no containment heat removal and containment challenge could potentially be more severe.
CS	Are the containment sprays available?	Yes	Implies decay heat/fission product removal capability and availability of water in containment sump/reactor cavity. It could also affect RWT water inventory depletion for use in providing core coolant makeup during ECCS injection. Three possibilities exist if the sprays are available: 1) the sprays actuate before vessel breach; 2) they continue operating through recirculation phase; and 3) the containment conditions do not exceed spray set before vessel breach.
		No	Implies lack of water in containment sump/reactor cavity and decay heat removal function.

1 In the bridge tree, cold leg breaks are combined with the hot leg breaks. The retention capability of the steam generators is conservatively neglected.

Table D-2

CORE DAMAGE BIN (SEQUENCE CHARACTERISTICS)

Core Damage Bin[2]	Scenario Description
I	RCS pressure and leakage rates associated with small-break LOCAs, with early melting of the core (e.g., within about 2 hours after the break occurs).
IR[2]	Similar to Bin I except for the radioactivity release through the steam generator.
II	RCS Pressure and leakage rates associated with small-break LOCAs, with late melting of the core (e.g., during recirculation).
IIR[2]	Similar to Bin II except for the radioactivity release path through the steam generator.
III	High RCS pressure and leakage rates associated with boil-off of the reactor coolant through cycling pressurizer relief valves, with early core melting (within about 2 hours).
IV	High RCS pressure and leakage rates associated with boiloff of the reactor through cycling relief valves with late melting of the core.
V	Large rates of leakage from the RCS and low pressures associated with large break LOCAs and failure of coolant injection, resulting in early melting of the core.
VI	Large-break LOCA conditions with failure of coolant recirculation and late melting.
CB	Containment bypass sequences in which the RCS leakage bypass the containment (e.g., interfacing systems LOCA or steam generator tubes ruptures).
SBO	Station blackout sequences in which the engineered safeguards systems may be recoverable.

2 Based upon Nuclear Safety Analysis Center, Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3, NSAC-60, June 1984. For the Turkey Point PRA Core Damage Bins, IR and IIR bins are conservatively combined with I and II respectively.

Table D-3

CONTAINMENT SAFEGUARDS BINS (SEQUENCE CHARACTERISTICS)

Containment Safeguard Bins	Fan Coolers	Containment Sprays
A	On	Injection Mode Only
B	On	Recirculating Mode
C	On	Available, but not on[3]
D	On	Failed
E	Failed	Injection Mode Only
F	Failed	Recirculating Mode
G	Failed	Available, but not on[3]
H	Failed	Failed

3 Containment sprays are available (not failed) but do not actuate prior to vessel breach; the containment pressure does not exceed the required actuation setpoint.

Table D-4

BRIDGE TREE CONTAINMENT STATES SEQUENCE DEFINITIONS

PLANT DAMAGE BINS	CORE MELT TIMING	RCS PRESSURE	CONTAINMENT CONDITIONS MITIGATING SYSTEMS (e)		COMMENTS AND ASSUMPTIONS
ID (a)	(E,D,L) (b)	(H,M,L) (c)	FAN COOLERS	SPRAYS	
V/A	EARLY	LOW	X	INJ	Core damage sequences (LOCA) within the bridge tree end states V/A - V/H differ in terms of the combinations of the containment safeguard systems. The RCS conditions, however, are similar.
V/B	E	L	X	REC	
V/C (f)	E	L	X	AVAIL	
V/D	E	L	X		
V/E	E	L		INJ	
V/F	E	L		REC	
V/G (f)	E	L		AVAIL	
V/H	E	L			
VI/A	DELAYED	L	X	INJ	This grouping is similar to above sequences V/A-H except for the time of core melt and cavity condition. In all the core damage bins, the RWT is discharged to the containment, flooding the cavity. Failure of ECC at recirculation could delay core melt.
VI/B	LATE	L	X	REC	
VI/C	L	L	X	AVAIL	
VI/D	D	L	X		
VI/E	D	L		INJ	
VI/F	L	L		REC	
VI/G	L	L		AVAIL	
VI/H		L			
IR/IIR (g)					Core damage bins (small (LOCA) assigned to this PDS involve failure at injection. Differences between the PDSs include sprays actuation, which can lead to a flooded cavity.
I/A	E	M	X	INJ	
I/B	E	M	X	REC	
I/C	E	M	X	AVAIL	
I/D	E	M	X		
I/E	E	M		INJ	
I/F	E	M		REC	
I/G	E	M		AVAIL	
I/H	E	M			
II/A	D	M	X	INJ	Core damage bins (small LOCA) assigned to this PDS involve failure at recirculation. Difference between PDSs I and II, with failure of ECC at recirculation is that RWT is injected to the containment, leading to a flooded cavity for sequences assigned to the PDS II.
II/B	D	M	X	REC	
II/C	L	M	X	AVAIL	
II/D	L	M	X		
II/E	D	M		INJ	
II/F	D	M		REC	
II/G	L	M		AVAIL	
II/H	L	M			

Table D-4 (continued)

BRIDGE TREE CONTAINMENT STATES SEQUENCE DEFINITIONS

PLANT DAMAGE BINS	CORE MELT TIMING	RCS PRESSURE	CONTAINMENT CONDITIONS		COMMENTS AND ASSUMPTIONS
			MITIGATING SYSTEMS (e)		
ID (a)	(E,D,L) (b)	(H,M,L) (c)	FAN COOLERS	SPRAYS	
III/A III/B III/C III/D III/E III/F III/G III/H III/HR	E E E E E E E E E	H H H H H H H H H	X X X X	INJ REC AVAIL INJ REC AVAIL	Core damage bins (transients) assigned to this PDS involve failure at injection. Differences between PDSs include sprays actuation, which can lead to a flooded cavity. Included in this PDS group are station blackout sequences (PDS IIIHR), in which containment safeguards are not available, but recoverable.
IV/A IV/B IV/C IV/D IV/E IV/F IV/G IV/H	D D L L D D L L	H H H H H H H H	X X X X	INJ REC AVAIL INJ REC AVAIL	Core damage bins (transients) assigned to this PDS group involve recirculation failure. This leads to a flooded cavity for all containment safeguards states.
CF CB	L E	H M,L			Core damage sequences with loss of CHR only and leading to containment failure prior to core damage are assigned to CF. Bypass sequences (interfacing LOCAs or SGTRs are assigned to CB).

NOTES:

a) The sequence ID (e.g. V/A) takes on the form of a bridge tree numbering scheme coupled with the core damage bins similar to Oconee PRA. For example, the bridge tree sequence end state "A" will have similar containment system states. The difference lies in the core damage boundary conditions (V).

b) Core melt timing is measured from the time of shutdown. It generally depends on the time when coolant makeup is lost as defined by the core damage state. Three relative

Table D-4 (continued)

BRIDGE TREE CONTAINMENT STATES SEQUENCE DEFINITIONS

NOTES (continued):

time phases are used. They are: 1) Early for times less than 2 hours; 2) Delayed for times between 2 to 6 hours; and 3) Late for times greater than 6 hours.

c) RCS pressure is determined at the time of core damage and vessel breach. Three levels of pressure are generally defined. They are: 1) High for pressures above the threshold of debris dispersal which impacts DCH; 2) Moderate for pressures encompassing small and medium LOCAs which is also within the SIT setpoint during which coolant makeup may be provided intermittently by passive sources; and 3) Low for pressures at which debris dispersal is not significant.

d) Intentionally deleted.

e) Fan coolers actuation (X) and failure are considered. Containment sprays actuation is denoted prior to vessel breach during INJection, or RECirculating mode, AVAILable, but not actuated, and available.

f) The low CS setpoint would preclude this condition for St. Lucie. If sprays are available, it is likely to be actuated before core damage.

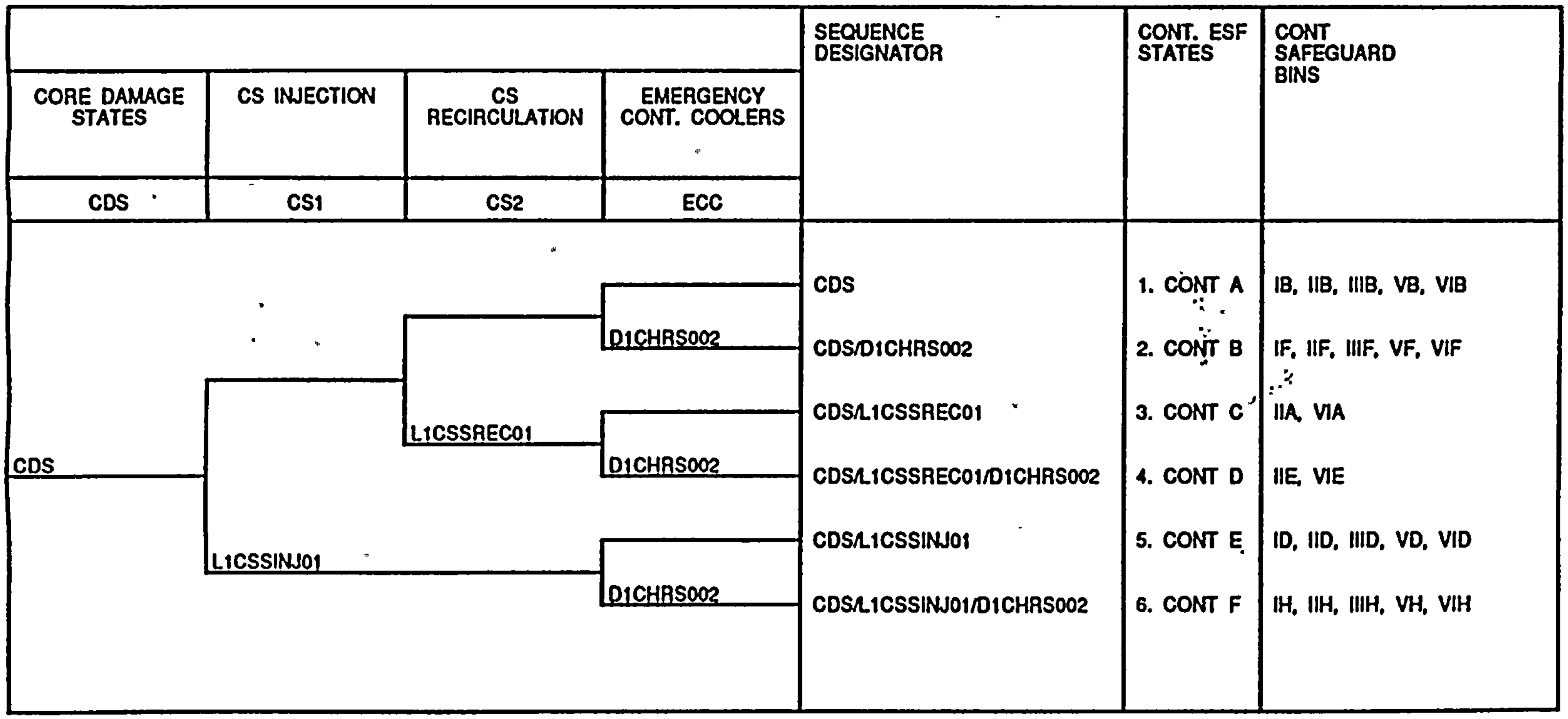
g) These end states involve cold leg breaks, conservatively binned with the hot leg breaks.

Table D-5

SUMMARY OF CONDITIONS FOR CORE DAMAGE STATES AND PLANT DAMAGE STATES

CDS DESCRIPTION	RCS PRESSURE AT CM	CONTAINMENT SAFEGUARD STATES			POSSIBLE PLANT DAMAGE STATES	IMPOSSIBLE STATES
		CS	ECC	CI		
I. S1 LOCA With AFW; Early Failure	MOD/High 200-2000 psi	1. Available for injection only 2. Available 3. Available but not actuated 4. Not Available	YES OR NO	YES OR NO	IA, IB, ID, IE, IF, IH	IC, IG: CS will be actuated
II. S1 LOCA with AFW; Late Failure	MOD/High 200-2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	IIA, IIB, IID, IIE, IIF, IIH	IIC, IIG: CS will be actuated
III. Transient & S1 LOCA w/o AFW; Early Failure	High-High > 2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	3B, 3D, 3F, 3H	3A, 3C, 3E, 3G: CS will be actuated with or without ECCs.
IV. TRANSIENT & S1 LOCA w/o AFW; Late Failure	High-High > 2000 psi OR MOD/High 200-2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	IVB, IVD, IVF, IVH	IVA, C, E, G: CS will be actuated with or without ECCs.
V. Large & Small (S2) LOCA; Early Failure	Low < 200 psi	1, 2, 4	YES OR NO	YES OR NO	VA, VB, VD, VE, VF, VH	VC, G: CS will actuate if available.
VI. Large & Small (S2) LOCA; Late Failure	Low < 200 psi	1, 2, 4	YES OR NO	YES OR NO	VIA, VIB, VID, VIE, VIF, VIH	VIC, G: CS will actuate if available.

Assumptions:
1. S2 LOCAs do not require AFW for heat removal; leakage rates from the RCS is sufficient for depressurization, allowing low head injection; and
2. Long term transients and S1 LOCAs are not considered since Bleed and Feed (B&F) is not given credit.



SIMPLIFIED CONTAINMENT SAFEGUARD TREE C:\ETA-INCON8.TRE 9-02-93

Figure D.1 Containment Safeguards Bridge Tree

APPENDIX E

ST. LUCIE UNITS 1 & 2

CONTAINMENT EVENT TREE ANALYSIS

E.0 INTRODUCTION

This appendix describes the development of the St. Lucie containment event trees (CETs), the rationale for the simplified approach, and the implementation of the methodology for creating the overall containment logic model. The CETs provide a systematic framework for displaying the sequence of events and spectrum of containment damage states of a severe accident progression. The containment damage states (i.e., failure and recovered CET end states) provide a measure of the potential fission product releases or source terms for the dominant accident sequences as represented by the Plant Damage States (PDSs). Linking the plant damage states with the source terms through the CETs provides an overall containment logic model for the PRA. The plant damage states are described in Appendix D, the accident progression analysis conducted to estimate the timing and determine the source terms is discussed in Appendix F and Section 4 describes the quantification of the CETs.

E.1 CET DEVELOPMENT PHILOSOPHY

The CETs developed for St. Lucie include all the important phenomenological and systems-related events identified in the NUREG-1150 reference PWR accident progression analysis and previous PRAs examined in this study (Zion, Surry, Oconee). Generic information is used to guide the development process, and plant-specific information is used to form conclusions regarding the impact of phenomenological issues on the plant. The overall philosophy adopted for the Level 2 study is to create a simplified model, so that key results and insights are readily apparent.

The common characteristics of the plant damage states form the basis for the development of the generic CETs.[1] Core damage sequences in that the containment is successfully isolated (i.e., not bypassed) would have different modeling requirements with respect to containment challenges as compared to those sequences where the containment is bypassed. Accident sequences within the core damage bins involving containment isolation consider the initial and boundary conditions so that the containment may be challenged early or late. Thus, plant damage bins (such as core damage bins I, II, III, or IV) are defined for those accident sequences in which the reactor is at pressure. These PDSs would likely offer high pressure challenges to containment integrity at vessel breach as a result of the blowdown forces or direct containment heating. On the other hand, core damage bins V and VI generally involve a sequence progression while the Reactor Coolant System (RCS) is at low pressure. This condition is likely to preclude high pressure blowdown and severe pressure and thermal challenges during vessel breach. The net result of this approach is a CET structured for each of the generic accident classes (core damage bins). The same CET structure is used for each of the PDSs within a core damage bin, but the specific effects of the PDSs appear in the quantification of the CET event nodes. Plant-specific considerations are incorporated during the CET quantification through the logic trees that address accident-specific system failure, phenomenological response, and accident conditions by examining the sequence cut-sets.

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1 The characteristics of the PDSs are described in Appendix D.

E.2 TIME PHASING OF CETs

Radionuclide release magnitudes are generally greatest if the containment integrity is lost early. Therefore, physical phenomena and systems-related events are considered at critical phases of the accident. The time phases used in this study include:

- Core degradation as defined by the PDS sequence characteristics;
- Time period during core melt prior to vessel breach;
- Time period between vessel breach and short-term response following initial debris disposition outside the vessel; and
- Time period involving the long-term response after vessel breach subsequent to initial debris disposition and core-concrete interactions in the cavity.

The events that contribute to early containment failure and releases of fission products from the containment are grouped into key CET event nodes that either occur early or late relative to the time of core melt and fission product release into the containment. Relative time phases are conceptually defined with respect to core melt and vessel breach in order to classify plant behavior into early or late CET events. These definitions serve as a starting point for CET development by allowing relevant phenomenological and systems-related events to be identified within a consistent time frame relative to core damage (as implied in the PDS classification as well).

E.3 CET LOGIC MODEL

In this PRA, the objective was to define a limited number of CET event nodes in order to convey the full spectrum of the accident progression in a single event tree. The issues affecting the top event nodes are dissected into logic trees that portray the relationship of severe accident phenomena, systems and recovery measures in a "fault tree" framework. The "fault trees" are referred to in this appendix as phenomenological logic trees.

A series of phenomenological or functional events is identified within each of the CET time phases. The combinations of events determine:

- Reactor and containment status;
- Coolant injection systems status;
- Containment safeguards status; and
- Magnitude and mechanisms of radionuclide release.

The effectiveness of the containment in removing radionuclides from the containment atmosphere is significantly influenced by the time of containment failure. Previous source term calculations using the NRC suite of codes (NUREG-0956) and MAAP (Task 23.1 Technical Reports) indicate that, as long as the containment integrity is maintained after fission products enter the containment, natural removal mechanisms act to deplete airborne concentrations of radionuclides in containment so that releases to the environment become very small should the containment ultimately fail. More recent studies published using MAAP 3.0 code (EPRI sensitivity studies) and MARCH code (BMI 2104, 2469) calculations for large, dry PWRs examined in this study show that the containment integrity is generally maintained in the long term, despite loss of all containment safeguards.

The top event nodes that appear in the CETs are defined by combining the phenomenological and systems-related events, as well as the containment failure modes and locations. The rationale for developing simplified CETs (versus using detailed CETs developed for the reference plant) is based on similarities in the outcome of interactions and dependencies between the front-line and containment systems, the core melt physical processes, and fission product transport behavior. Thus, physical processes and systems-related events that result in a similar impact in containment response (i.e., resulting in similar containment challenges) can be collapsed into a single CET top event node.

E.4 CET STRUCTURE

This section discusses the CET structures developed in this analysis. A brief description of the CET top event nodes and the supporting logic trees are also provided. These are also characterized as being early and late relative to the time of fission product release from the fuel to the containment system. Containment failure modes are described with respect to their impact on the magnitude of fission products released to the environment. Detailed discussion of specific systems-related events and phenomenological considerations are provided in the quantification of the logic trees.

Attachment C-1 presents a typical CET for plant damage states considered in this study. The CET structures represent the two possible containment states during a severe accident:

1. The containment is not bypassed during core melt; or

2. The containment is initially impaired (bypassed) during core melt and fission products released from the fuel bypass the containment.[2]

A single CET structure is used in this analysis since the important top events are all similar, with respect to the containment challenges at key phases of the accident (i.e., early or late containment failures). The difference lies in the mechanisms that could challenge containment integrity and potential recovery measures associated with the RCS condition (i.e., high or low pressure). For an initially impaired containment, the CET is significantly simplified since containment failure is not an issue. At issue is the fission product release mechanisms and potential mitigation measures. The generic CETs are further developed for each PDS using logic trees to break down the top events into phenomenological, systems, or operator actions. These are generally plant-specific

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[2] Isolation failures are included in the determination of early containment failures in the CET logic trees.

issues that determine the probability of each possible sequence of events that may lead to a release to the environment.

E.5 GENERIC CET TOP EVENTS

The following discussion defines the top events in the CET (related to the physical processes and system availabilities) that strongly influence the core melt accident progression and fission product transport behavior within the containment system. Table E-1 summarizes the phenomenological and systems-related events included in the CETs. The following fundamental dependencies between the CET top events are assumed in the development of the CET. This results in further simplification of the CET structures:

1. During a sequence in which the RCS is not recovered in-vessel, it is assumed that the vessel bottom head will fail; hence event VF is not developed as this branch will always be a failure. This is due to inadequate water being available to form a coolable debris configuration, as to preclude vessel head attack. Ex-vessel heat transfer for a submerged bottom head is not assumed sufficient to preclude vessel head failure in the long term, due to lack of published calculations of effective heat transfer through the bottom head that can preclude vessel head attack. Hence, by definition of the "no vessel breach" logic tree, a lack of coolant recovered in-vessel implies failure at this time. Similarly, sequences in which the RCS is depressurized and coolant is not recovered in-vessel, vessel failure is conservatively assumed to occur.

2. High RCS pressure sequences with coolant recovered in-vessel are considered very unlikely and are not developed further. The high pressure injection systems at St. Lucie cannot inject at the system pressure, requiring depressurization of the RCS to allow coolant makeup. Recovery of an initially failed high pressure injection system requires another condition (i.e., depressurized RCS) which is considered in another branch of the CET.

3. For sequences in which the RCS is depressurized, in which coolant is recovered in-vessel and no vessel breach occurs, coolable debris formation in the ex-vessel branch is not applicable. This is true whether or not early containment failure occurs. No vessel breach implies a coolable in-vessel debris is formed, hence an ex-vessel coolable debris question has no significance.

4. All branches in which the containment is assumed to fail early, the "no late containment failure" top event is not applicable.

5. If containment survives (both the early and late top events), it is assumed that fission product removal occurs and the containment failure modes are not applicable.

The rationale and basis for including each of these events in the CET are described in this section. The CET top event logic fault trees are presented in Attachment C-2.

E.5.1 Event DP: RCS Depressurized Before Vessel Breach

The question asked in this top event node is related to depressurization of the RCS prior to vessel breach. Success in this branch implies that RCS pressure is reduced either through the capability of the operator to depressurize the reactor or through a phenomenological condition that could induce RCS depressurization. This event node is considered for high pressure PDSs to indicate a potential recovery or mitigating condition during core melt prior to vessel breach. For accident sequences with the RCS at pressure (and low pressure coolant injection initially unable to deliver makeup to the vessel due to the high pressure in the vessel), depressurization of the RCS can mean either of the following:

- The condition that precludes successful low pressure coolant injection is removed, and coolant makeup is likely to occur.
- High RCS pressure that could exacerbate containment challenges at vessel breach (such as direct containment heating) is removed.

This event node directly impacts the likelihood of the subsequent CET event nodes related to in-vessel recovery and early containment challenge.

The issues considered here include:

- Initial RCS state as determined by initiating event (e.g., LOCAs or transient events);
- Active operator action to depressurize the RCS before vessel breach; and
- Severe accident induced LOCA due to high RCS piping temperature.

Event node DP applies only to transient initiated high-pressure accident sequences represented by PDSs III and IV, and the moderately high RCS pressure small LOCA initiated events represented by PDS I and II.

E.5.2 Event REC: Coolant Recovered In-Vessel Before Reactor Vessel Breach

The question asked in this top event node is related to recovery of coolant injection after core degradation, prior to vessel breach. This event node addresses the vessel injection recovery measures, that have the potential for arresting core melting and subsequent thermal failure of the reactor vessel. It considers the possibility of low pressure injection systems working once the RCS is depressurized by phenomena given in top event node DP. The PDSs define the boundary conditions under which these systems could operate. The dominant accident sequence contributors within each PDS determine whether these systems are initially failed due to hardware problems or are principally unavailable because reactor conditions (e.g., high pressure shutoff head) preclude their successful operation.

For PDSs where the containment is initially intact and the RCS is at pressure, core damage might be induced by lack of coolant make-up due to failure of the high pressure injection systems. However, the low pressure injection systems may be available, but coolant injection is prevented by conditions that preclude pump operation (i.e., RCS pressure exceeding the shut-off head). Once the high pressure condition is removed, (as modeled in the previous event node, DP), coolant injection would most likely be recovered. Success in this branch is judged likely provided low pressure injection systems are not isolated by the operator, in which case, human intervention would be considered. This event node considers this possibility along with successful recovery of alternative systems that may have failed prior to core damage, but could potentially succeed given additional time for operator action. Success at this branch implies fission product releases from the fuel would be mitigated and establishment of a heat transfer cycle can assure maintenance of containment integrity.

For PDSs involving low RCS pressure (i.e., large LOCAs with failure to provide adequate coolant makeup), success at this event node is not likely, as implied by the accident sequence definition. The accident sequence cut-sets include human intervention in providing alternative injection systems; core damage occurs given failure to recover. Because of the short time constants for core heat-up for large LOCAs, it is judged that the additional time from core damage to vessel breach has no significant effect on the success of human intervention.

This tree also addresses small break LOCA and transient sequences where the RCS was not depressurized by events in DP. In this case, both LPI and HPI are still required for successful cooling to avoid vessel breach. Similar to the large LOCA events, success at this event node is unlikely although slightly longer times are available between core damage to vessel breach than for large LOCAs. This event node directly affects the subsequent event node relative to arresting core melting and precluding thermal failure of the vessel bottom head.

The issues considered in the top event logic tree include the following:

- Initiation of coolant injection upon depressurization;
- Active operator action to recover alternative injection sources; and
- Availability of AC power.

The conditional probability associated with this event node is principally determined by the recovery of coolant injection systems as defined by the accident sequence definition. For example, for PDSs, where loss of coolant makeup may be caused by loss of power, recovery of the systems are considered plausible, with likelihoods that are comparable to those estimated in the Level 1 analysis. At issue is the reliability of equipment under potentially adverse environment and the time available for operator recovery action subsequent to core damage prior to vessel breach. Although accident- and plant-specific information are considered in the quantification of these issues (particularly with respect to alternative systems that may be available), generic information is required as to the controlling phenomena that determines timing of core degradation and vessel breach.

In this PRA, the conditions that exist during degraded core accident progression (such as systems availability, RCS conditions, etc.) are implicit in the PDSs definitions. Therefore, the conditional failure probability of event REC for core damage bins III and IV is strongly dependent upon specific boundary conditions of dominant accident sequence contributors (i.e., dominant cut-sets) to these PDSs. With the exception of loss of power or station blackout scenarios, the recovery measures are identified but not explicitly quantified. Because of the uncertainties with regard to operator action under potentially severe accident conditions, the quantification process would require plant-specific deterministic analysis to support timing assessments and human response analysis.[3]

E.5.3 Event VF: No Vessel Failure

The question raised in this event node addresses recovery of core degradation within the vessel, which prevents vessel head thermal attack. In-vessel recovery is considered only to the extent that coolant make-up has been successful in the previous event node. This event physically signifies that the core degradation process leading to vessel failure is successfully terminated, thus arresting core melt and precluding significant fission product release to the environment. Due to the short time constants associated with the core meltdown progression once melting is initiated and core geometry is lost, the time available for repair of failed equipment and installation of alternative core cooling systems is generally short.

For purposes of this assessment, core debris cooling and termination of core degradation prior to vessel head attack is not considered possible once incipient melting and core slump has occurred. However, the logic tree shown in Figure E-2 provides some insights into how core debris cooling within the vessel might be established to preclude vessel breach. The potential for terminating melt progression prior to vessel breach is represented by a simplified model of debris cooling from within the vessel, or vessel bottom head cooling from outside the vessel, thus terminating head attack. The former would depend on the success of the previous event node, REC, and phenomenological considerations of coolable debris bed formation. The latter may be achieved by filling the cavity with water above the vessel bottom head, allowing heat transfer to be established through the vessel walls. However, no calculations exist that indicate that sufficient cooling would be available to maintain vessel head integrity, although this is considered as a recovery action for some BWRs to achieve core cooling under certain accident conditions (Rev. 3 BWROG EOPs). Success of Event VF is essentially determined by the physical processes controlling core melting and material relocation to the lower plenum. A good understanding is very important in order to determine possibilities of arresting vessel head attack once significant core degradation has occurred. Exact modeling is not always possible; therefore, simplifying assumptions are made to approximate the analytic treatment of the physical process involved. There is considerable uncertainty in determining the formation of a coolable configuration once significant core geometry deformation and melting has occurred. There are recognized limitations in the existing PRA analytical models (MAAP and MARCH) in predicting coolable debris bed formation within the vessel. The limiting factor, therefore, is the time available between core vulnerability

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[3] This can be performed by FP&L to supplement this analysis should this study indicate unacceptable results; a process that is beyond the scope of this PRA.

(i.e., conservatively defined as core uncovery, the end state of a core damage sequence) and core melting (i.e., peak temperatures exceeding core material eutectic temperature of 4130°F).

MAAP calculations conducted by IDCOR indicate that the time period between core uncovery and onset of core melting is typically one half hour to several hours (TSR, Task 23 Technical Reports) depending on the accident sequence. MAAP calculations also indicate that core degradation cannot be arrested once fuel melting has started and core reflood (as determined by the success branch of event REC) would not preclude support plate failure and vessel breach. MARCH STCP calculations, on the other hand, indicate that vessel breach can be arrested provided coolant injection is recovered before significant core melting has occurred. Because of the phenomenological uncertainty associated with this event, the limiting factor used for the conditional probability of success is the time period between core uncovery and core melting compared with the time period between core uncovery and core collapse. Plant-specific MAAP calculations conducted for St. Lucie, as described in Appendix F, are consistent with the previous assessments of the Zion reference plant.

E.5.4 Event CFE: No Early Containment Failure

This functional event is included in the CET to signify that the containment integrity is maintained during the early phases of core degradation and release of fission products from the fuel up to vessel breach. The fission products released from the fuel are contained within the primary containment system so that natural removal mechanisms can effectively act to deplete airborne concentrations in containment.

Failure at event CFE (see Attachment C-2) is defined as loss of containment integrity early in the accident sequence. Several failure mechanisms are postulated for this top event node as illustrated in the logic tree. These include containment challenges resulting from:

- High pressure melt ejection loads generated by phenomena, such as combustion of hydrogen released prior to and at vessel breach, and direct containment heating;

- Pressure spikes occurring due to blowdown at RPV failure with the RCS at high pressure; and

- Fuel-coolant interaction resulting in rapid steam generation within the vessel at core slump or in the reactor cavity at vessel breach.

The failure mechanisms identified above that, individually or in combination, result in loss of containment integrity early in the accident are considered in the logic tree. Uncertainties in containment loads at vessel breach arise from the nonstochastic nature of some of these events (e.g., hydrogen burns), as well as a poor understanding of the phenomena governing others (e.g., direct containment heating). Although more experimental and analytical information regarding direct containment heating has been generated, substantial uncertainties persist and the phenomenon continues to generate controversy. Pressure loads from high pressure melt ejection and fuel-coolant interaction are further discussed below.

High Pressure Melt Ejection (HPME) Loads. The potential for pressure rise as a result of the high pressure melt ejection of molten debris from the vessel to the cavity (and containment atmosphere) is considered in this issue. These loads are generated by a combination of severe accident phenomena, dominated by direct containment heating (DCH) and attendant hydrogen burning. DCH is used to define the series of physical and chemical processes that is postulated to accompany the ejection of the melt when the vessel fails at high pressure. If a large fraction of the molten core debris is dispersed into the containment as fine particles, a substantial portion of the core material sensible heat is transferred to the atmosphere. The containment pressure rise depends strongly on the reactor cavity geometry and the mass of material dispersed. The pressure rise can also be augmented by the release of chemical energy associated with the oxidation of metals as it is transported through the containment atmosphere. The containment pressure rise accompanying direct containment heating depends on reactor cavity geometry, the mass of material dispersed by reactor vessel blowdown, and several other parameters described later.

There are several parameters included in the logic tree to model the dependencies of the containment challenge resulting from DCH:

1. Provided the reactor pressure prior to vessel breach is sufficiently high to transport molten material and hot gases to the upper regions of the containment, the pressure rise is probably insensitive to reactor pressure. The cut-off pressure (reactor pressure threshold below which DCH does not occur) is still not clear-cut based on Sandia experiments of debris dispersal. In this assessment, DCH is regarded possible if the pressure is above 200 psi (NUREG-1150).

2. The fraction of core melt ejected from the vessel at the time of vessel breach determines the amount of material that can participate in DCH. This is governed by the model used to represent core melting and, to some extent, the accident sequence definition.

3. Unoxidized metal content in the melt is among the important contributors to containment loads as it determines the energy release associated with the oxidation of unreacted metals (zircaloy).

4. The availability of water in the reactor cavity at the time of vessel breach is considered, because it can influence DCH. The water can interrupt the pathway for debris dispersal following vessel breach as it is displaced only after a fraction of the debris is injected to the cavity, or the water can be co-dispersed, in which case the droplets can continue to quench the debris.

DCH resulting from the dispersal of molten core debris can induce hazards that challenge containment in the short term. Oxidation of the metallic components of the melt can generate chemical energy that can increase the pressure and temperature of the containment. Additionally, hydrogen is produced from this exothermic reaction, and at this point could burn (in the event that the containment is not steam inerted and sufficient oxygen is present). If the amount of debris involved in this process is significant, extremely high pressure and thermal loads can indeed fail containment.

In an HPME, combustion of sufficient hydrogen to generate a substantial pressure rise is subject to physical requirements regarding minimum hydrogen concentrations, oxygen availability, and maximum inerting gas concentrations. Hydrogen concentrations in containment prior to vessel breach depend upon in-vessel core melt progression (primarily the fraction of the core Zircaloy oxidized before vessel breach), and the type of accident scenario being considered. Hydrogen burning alone can also induce over-pressure failure. Since the St. Lucie containment is not inerted, rapid thermal transient and associated pressure rise due to hydrogen burning are considered possible.

Pressure Spikes due to RCS Blowdown. The pressure rise in containment associated with the blowdown of the RCS at the time the vessel fails is determined to be significantly less than the pressure associated with HPME. MAAP calculations at St. Lucie indicate that the total pressure in containment at vessel breach is limited to the remaining RCS water and steam inventory (and hydrogen that may be generated during core degradation) that is discharged during RCS depressurization. RCS blowdown alone is not considered sufficient to challenge containment integrity.

Fuel Coolant Interaction (FCI). The consequences associated with the rapid transfer of thermal energy from fuel-coolant interaction in the vessel or ex-vessel can be risk-significant as it poses a plausible threat to containment integrity. There is a possibility that in certain accident sequences, molten material can flow into a pool of water in-vessel (reactor vessel lower plenum) or ex-vessel (reactor cavity) leading to steam explosion failing containment. It is also noted that ex-vessel steam explosion may result not only in an impulse load, but also in a quasistatic pressure load on the containment structures (NUREG-1150).

In-vessel steam explosion was first assessed in the Reactor Safety Study, WASH 1400. Since then, numerous similar tests (which provided data) and related analytical models have been developed. However, some uncertainty still exists regarding this issue, much of that is related to the applicability of the small- and intermediate-scale tests to reactor scales and geometries. In this assessment, the likelihood of in-vessel steam explosion is dependent on the RCS pressure. Some experiments indicate that high ambient reactor pressures would reduce the likelihood of triggering steam explosions. Therefore, the assessment here is considered only for accident sequences where the reactor is at low pressure. Generic Letter 88-20 indicates that this issue has been resolved and provides a very low probability (if not negligible) for this event.[4] The draft NUREG-1150 assessments also indicate a low likelihood of containment failure due to steam explosions relative to other failure modes.

Ex-vessel steam explosions for the Zion and Surry containments were not assessed to threaten containment integrity (NUREG-1150, 1989). The containment structures would not have significant vulnerability to the impulse loads generated since the water in the cavity would not directly contact structures that are essential to the containment function. For St. Lucie, the reactor cavity (one of

4 This has been revised in the subsequent publication of NUREG-1150 (NUREG/CR-4551 draft, July 1989), that considered the assessment of the panel of experts comprising the Steam Explosion Review Group (SERG). Their assessment is that this issue provides a slightly higher probability for this event for Surry.

the compartments within the reactor building internals) is a reinforced concrete structure. The cavity walls are heavily reinforced concrete to support the reactor core and the primary shield wall (see Appendix C). The cavity configuration (if filled with water), not unlike that of Zion or Surry, would not allow water to present vulnerability to containment structures from impulse loads generated during ex-vessel steam explosion. Potential ex-vessel interactions between core debris and water is of concern only for accident scenarios during which water covers the reactor cavity floor prior to vessel breach.[5] In general, this implies the successful operation of containment sprays or a LOCA or both. The effect of pressure spikes resulting from high pressure blowdown and rapid steam production, although subject to some uncertainty, can be bound by conservative assumptions regarding steam generation.

The logic tree developed for this event considers the time-dependency of the various phenomenological events that contribute to early containment challenges during a severe accident. Because of the importance of this event, several functional and phenomenological events were considered in quantifying early containment failures. These include systems-related events, such as induced containment isolation failure, (which may occur given the combustible gas control procedures in place at St. Lucie), as well as physical processes that contribute to providing the pressure loads that challenge containment.

E.5.5 Event DC: Coolable Debris Formed Ex-Vessel

This event is included in the CET to signify the termination of the core melt progression subsequent to vessel breach. The success branch at this CET node means that a coolable debris bed is formed, terminating concrete attack, and thus precluding ex-vessel fission product releases from core-concrete interaction. Following the success branch also implies that containment overpressure challenges from non-condensible gas generation is precluded, thus containment integrity is likely to be maintained in the long term. For example, for PDSs where the low pressure injection systems were previously unable to inject due to high RCS pressure, these systems would likely start to deliver coolant when the vessel is breached. Coolant injection could potentially quench the debris. This condition can also establish a heat transfer cycle from the debris to the environment in the subsequent event node.

Failure at this branch implies that concrete attack occurs in the cavity, the core debris remains hot and sparging of the concrete decomposition products through the melt releases the less volatile fission products to the containment atmosphere. This condition is considered more likely if a deep core debris bed is formed in the cavity and, absent coolant addition, the debris is not able to effectively dissipate the decay heat to the surroundings. Should an impervious crust form, coolant addition would not likely terminate concrete attack, although the released fission product aerosols are scrubbed by the overlying water pool.

MAAP 3.0 calculations for St. Lucie, as in the reference large, dry PWR containment (such as Zion) indicate that dispersal and entrainment of molten core material outside the cavity region into

[5] For sequences where the core debris is retained in-vessel, this event node is not considered relevant.

the containment occurs in most accident sequences where the vessel fails with the RCS at pressure. The extent of debris dispersal could vary depending on the amount of core debris molten at the time of core vessel failure. St. Lucie cavity and instrumentation tunnel geometry is similar to that of the reference Zion PWR. The principal flow path is the instrumentation tunnel between the reactor cavity and the lower compartment region. The STCP model used for the NUREG-1150 supporting calculations for large, dry PWRs does not model dispersal following vessel breach at pressure. Separate effects calculation of DCH (such as CONTAIN analysis for Surry), indicate dispersal is likely to occur even with the more restrictive design of the Surry cavity. Should debris dispersal occur, formation of a coherent debris bed is not likely. Conversely, formation of a more coherent debris bed is considered more likely for accident sequences with the RCS depressurized prior to vessel breach. The phenomenological uncertainty associated with debris bed coolability given water injection is the formation of an impervious crust that precludes water ingress into the debris.

This event considers the formation of an uncoolable geometry in the cavity implying significant core-concrete attack that could challenge containment integrity. The ability to cool the debris after vessel breach is determined by the possibility of water ingress or, absent water addition to the cavity, formation of a coolable corium geometry. The important issues include:

- Phenomenological considerations of crust formation;
- Sequence dependencies related to corium dispersion at vessel breach (i.e., high RCS pressure);
- Geometric configuration dependencies allowing formation of a shallow debris bed (cavity projected floor area) or accumulation of an overlying water pool; and
- Systems-oriented considerations related to water availability (e.g., ECC injection post vessel breach or sprays actuation).

The conditional probabilities that are plant-specific in nature are obtained from an examination of the cut-sets, as indicators of the availability of emergency core cooling (ECC) injection following vessel breach; the PDS definitions, which determine debris dispersal; and calculation of the spreading that might occur in the cavity. Coolability is at issue if an impervious crust is formed. It is recognized that there is some disagreement in this area among experts in severe accident phenomena. However, experimental information and calculations are available in the literature and can be used to provide guidelines in the quantification of the likelihood of crust formation for this event node. The boundary conditions will be obtained from the plant-specific assessments, e.g., depth of debris and coolant injection availability.
Accumulation of water in the cavity as a result of injection of coolant through the breached vessel or overflow from the containment floor of RWST water discharged as a result of ECC injection or spray actuation provide the most effective way of mitigating concrete attack (absent crust formation) and of scrubbing vaporized fission product aerosols. With regard to injection through the failed vessel, explicit procedures for the operator are usually not in place to assure a high reliability for the recovery of this injection mode as a post-core melt recovery action.

The systems-related events that contribute to the success branch depend heavily on a passive actuation of the accumulators, initiation of low pressure injection systems following RPV breach, and continued spray operation. If the low pressure systems are initially available, as in most PDSs involving high RCS pressure, coolant addition to the debris is likely. The limiting factor is the long-term reliability of the systems under adverse conditions.

E.5.6 Event CFL: No Late Containment Failure

This event is included in the event tree to address the potential loss of containment integrity in the long term, after vessel breach and core-concrete interaction (CCI) in the cavity. Potential failure modes considered at this stage of the core damage sequence include overpressure failure of the containment pressure boundary or basemat concrete attack. The conditions that contribute to containment overpressurization include boil-off of steam from the reactor cavity, given loss of heat removal function; pressure and thermal challenges from hydrogen burning, resulting from the long-term combustible gas formation in containment; non-condensible gas generation; or the less likely thermal failure of penetrations under high temperature conditions in containment. The latter can result from the long-term, decay heat energy radiation to the containment atmosphere and structures, given dispersal of the debris outside the reactor cavity into the surrounding lower containment volumes. This condition is considered unlikely given the cavity and instrumentation tunnel geometry. The long-term radiative heat transfer from the dispersed debris, if left unchecked, can potentially lead to high temperatures in the containment. However, thermal-induced failure is not likely to occur before pressure loads exceed the ultimate capacity of the containment.

This event is included in the CET to characterize the long-term behavior of the containment after core melt and vessel breach. Event CFL includes such events as overpressure failure of the primary containment, loss of containment integrity due to overtemperature[6] (a less likely condition), or basemat penetration. The success path here depends strongly on the recovery of systems that establish a complete heat transfer cycle from the core debris to the environment. One of the most important considerations is related to the time taken for gradual pressure build-up in the containment following vessel breach and ultimate disposition of the molten corium in the cavity and the containment floor.

The long-term containment pressurization is strongly influenced by the availability of decay heat removal systems (DHR), and this is included in the logic trees for event CFL. This event is related directly to the long-term reliability or recovery of the containment heat rejection function of the low pressure systems, given that the core is recovered in-vessel or the debris is quenched ex-vessel. This event implies that a direct heat transfer cycle is established from the core to the environment, such that containment pressure rise is controlled. The implication of failure at this event node is that the containment pressure would increase and reach the ultimate capacity of the containment,

6 The impact of elevated temperatures on the containment capacity after vessel breach is considered in this event. This can either result in a reduction of the equivalent ultimate capacity of the containment given a high temperature environment or penetration seal failure. (This is contrary to event CFE, where the ultimate capacity is principally determined by the pressure loads alone.)

challenging containment integrity. Should this occur, the recovered state of the core melt sequence either in-vessel or ex-vessel may also be jeopardized. The success criteria for decay heat removal is determined from the assessment of the containment systems available for St. Lucie.

The difference between this event node and that for CFE lies in the long-term process. The pressurization rates are not expected to be as rapid, as one might expect during RPV blowdown. However, the overpressure failure mechanisms would likely include similar mechanisms, such as hydrogen burning, long-term steaming and non-condensible gas generation from concrete attack. (DCH is not an issue in the longer term.) Another failure mechanism considered includes basemat penetration due to concrete attack.

Hydrogen burning and pressure rises as a result of this phenomenon are dependent on several important issues, such as:

- The stoichiometric ratio, and concentration of hydrogen and oxygen is required;
- A steam inerted atmosphere can preclude hydrogen burn (containment sprays tend to de-inert the containment atmosphere allowing for hydrogen burn); and
- An ignition source is required to start the hydrogen burn, the availability of AC power is a source for ignition.

The dependencies on the availability of containment sprays and AC power for hydrogen burn are modeled in the CFL logic tree. Based on the NUREG-1150 analysis if containment sprays and AC power are recovered/available during CCI, hydrogen burn is considered likely. In all cases, the pressure rise is determined from the concentration and quantity of hydrogen available at the time of recovery.

E.5.7 Event FPR: Fission Product Removal

This event is included in the CET in order to characterize releases from the fuel (in-vessel and ex-vessel) into the containment, the fission product retention processes, and the potential release magnitudes to the environment should the containment fail. The question raised in this event node is related to the airborne fission product removal mechanisms within the containment system. Success implies reduction of the fission product release magnitudes to the environment. Failure implies that most of the fission products are ultimately released to the environment from the fuel and the containment without mitigation. The release magnitudes are likely to be relatively high should the containment fail early.

The issues considered in determining the success branch of this event node include mitigating the release mechanisms from the fuel (in-vessel or ex-vessel recovery) or ensuring in-containment removal processes. The capability of the containment to reduce release magnitudes is measured through availability of active systems (e.g., containment sprays), passive capabilities for natural depletion as a result of the long time period from release to containment failure, or scrubbing afforded by an overlying water pool. Success or failure of this event depends on previous event nodes in the CET and PDS boundary conditions defined by the accident sequence cut sets. The St. Lucie containment design has mitigating features that ensure that fission products released from the fuel are contained within, if not permanently removed from, the containment atmosphere. These features include the containment sprays, to a lesser extent, the fan coolers.

This event node models the in-containment fission product removal process that might occur prior to containment failure. The active systems include scrubbing of radioactive aerosols from the containment atmosphere by the sprays. Passive removal includes natural processes (e.g., gravitational settling, thermophresis, or diffusiophoresis) that act on the radioactive airborne materials. The effectiveness of these natural removal processes in reducing the fission product concentrations depends on the length of time that the containment integrity is maintained after release of fission products from the fuel.

E.5.8 Event CFM: Containment Failure Modes

This event is included in the CET to characterize the impact of containment failure modes and locations as they affect the timing (i.e., duration) and mitigation in the secondary enclosure of the fission product source terms. In addition to the events summarized above, a number of issues are identified in this event node that apply to processes occurring after a containment failure. If core damage occurs before containment failure, containment pressure and temperature loads during a severely degraded core accident could ultimately lead to loss of integrity and release of fission products to the environment if left unchecked. The primary containment provides an important barrier for release of fission products to the environment, thus containment failure modes and resulting release paths are examined.

Following primary containment loss of integrity, the auxiliary building could provide yet another barrier for mitigating release of radioactive materials to the environment. The auxiliary building surrounds the primary containment of the St. Lucie reactor at key locations where failure might occur. While its ability to withstand the pressure loads resulting from containment failure is considered marginal, the auxiliary building does present substantial deposition sites for aerosol removal, thus could significantly mitigate fission products released from the primary containment.

For every containment failure, the location of the break could also be important in determining the atmospheric source term. The mode of release of the fission products following containment failure is considered in the CETs. The issues considered in modeling these scenarios are described in the next section.

Success of this event implies that the release duration is extended with mitigation of the fission products along the release path from the containment to the environment. The containment failure modes modeled consider the following issues:

- Containment Break Size. The manner in which a containment fails relative to the depressurization that ensues is characterized by the containment break size. Failure size can be characterized as large (rupture) or small (leakage) depending on the failure mode and flow rates that are obtained during depressurization following containment failure.

- Containment Break Location. The containment break can be above ground directly to the atmosphere or through the auxiliary building, or below ground (basemat failure). There are no "success" or "failure" states for this event. The three specified locations are defined in order to establish potential radionuclide removal mechanisms along the exit path from containment for containment overpressure and basemat failures.

Retention of fission product aerosols in the auxiliary building reduces the magnitude of the release to the environment. Successful fission product retention implies that the full impact of deposition sites in the secondary building is realized by virtue of the containment failure flow paths and leakage rates. Failure implies that the containment failure location directs the exit gases to the environment, thus bypassing the auxiliary building deposition surfaces, or the containment failure results in leakage rates that are substantially higher than the time constants for deposition.

The dependencies of primary containment break location and leakage rates (break size and driving force) on the auxiliary building effectiveness is included in the model to the extent that the likelihood of significant building retention accounts for the effect of primary containment failure modes on residence times. The leakage rate from the containment and potential hydrogen burning within the auxiliary building are among the contributing factors that could negate efficient auxiliary building aerosol deposition.

It is recognized that there is significant uncertainty associated with the determination of the containment failure mode, the failure size (leak before break or catastrophic rupture), and the failure location. In this analysis, no formalized structural analysis was conducted, but an attempt is made to incorporate insights from previous analysis to characterize implications of the containment failure on the release magnitudes to the environment.

E.6 ST. LUCIE CET

The generic CET described in Section E.5 was reviewed and modified to incorporate St. Lucie plant-specific features. The following major changes are reflected in the St. Lucie CET and associated top event logic shown in Attachment C-2.

1. The St. Lucie CET top events are redefined based on the CET conditions. For example, top events CFE, DC, CFL, FPR, CFM are subdivided to explicitly take into account for the prevailing event tree conditions.

2. The generic CET top event logic includes all possible contributors under all conditions; the St. Lucie CET top event logic considers only those that are applicable under a specific condition dictated by the event tree nodes.

3. The generic CET top event logic basic events includes conditions defined by the event tree (those prefixed by B...). The St. Lucie CET top event logic removes these basic events because the CET structure and the definition of associated top event already incorporates such dependencies.

The Basic events were changed by adding suffixes of plant damage state names to distinguish among various damage states. This facilitates generating a very general scoping model using RMQS program to accommodate future sensitivity studies.

E.7 REFERENCES

E-1. "Evaluation of Severe Accident Risks and Potential for Risk Reduction," Sandia National Laboratories, NUREG/CR-4551, Volumes 1 to 7, 1990.

E-2. "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," NRC Generic Letter 88-20, November 23, 1988.

E-3. "Oconee 3 Probabilistic Risk Assessment," Duke Power Company and Nuclear Safety Analysis Center/EPRI, NSAC/60, June 1984.

Table E-1

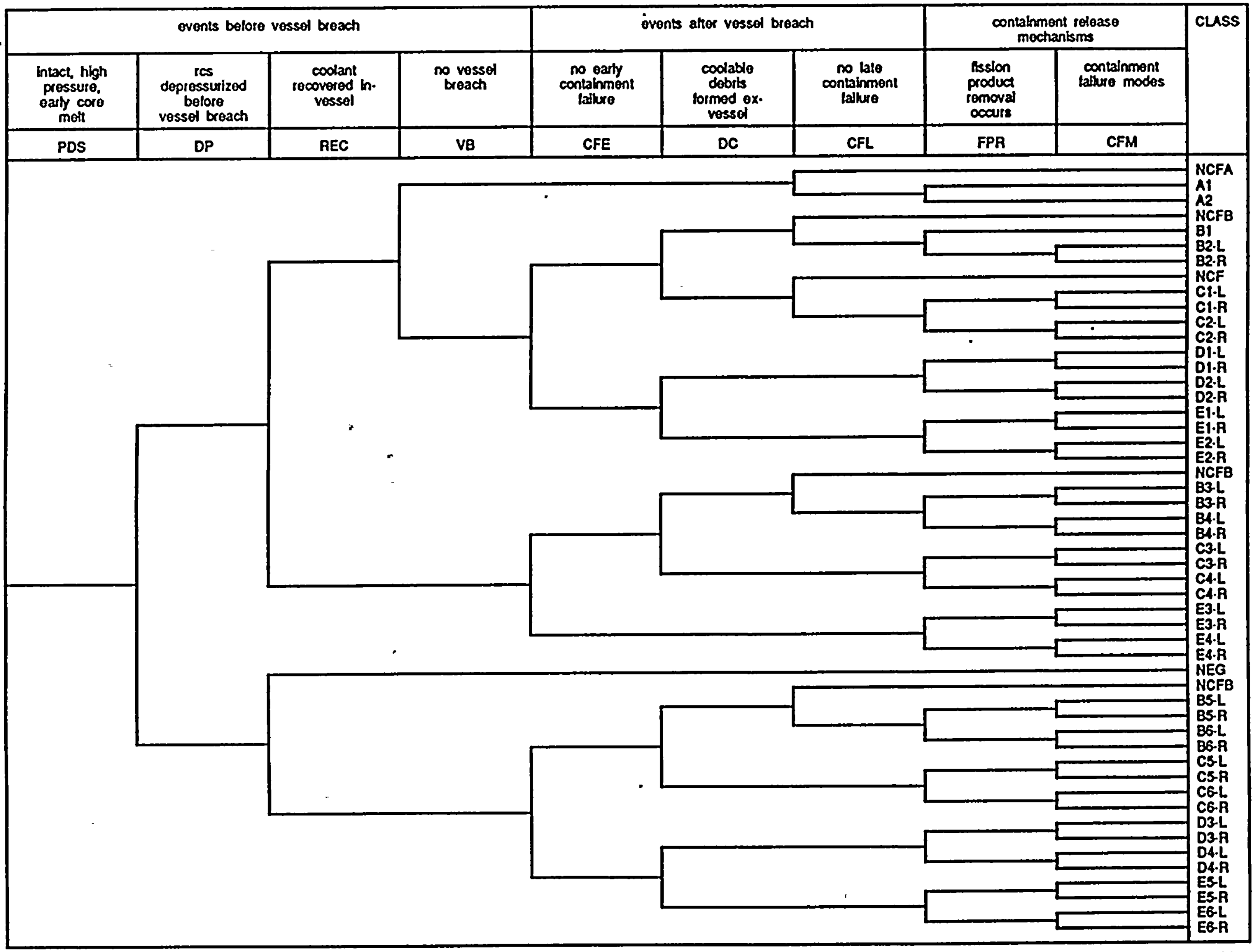
SUMMARY OF CONTAINMENT EVENT TREE NODAL QUESTIONS

Event Node Description and Issues Addressed	
PDS	Plant Damage State. This is the entry state to the core melt progression defining boundary conditions of core damage.
DP	RCS depressurized before vessel breach. This is generally defined by the entry state, operator actions or phenomena subsequent to core vulnerability
REC	Coolant recovered in-vessel. This node may be initiated by operator recovery action or a passive actuation should the conditions that preclude initial operation be removed in the previous event.
VF	No vessel failure. This implies arrest of core melt progression, terminating within the vessel. A coolable debris configuration is formed.
CFE	No early containment failure. This node implies that challenges to containment integrity are insufficient to fail containment early.
DC	Coolable debris formed ex-vessel. This node implies formation of a coolable configuration outside the vessel, precluding significant core-concrete interaction.
CFL	No Late Containment Failure. This node implies that long-term containment challenges are mitigated or do not occur as to breach containment pressure boundary.
FPR	Fission Product Removal. This event node is used to characterize potential fission product release magnitudes. This considers the mitigation of releases from the fuel in or outside the vessel and in-containment removal processes.
CFM	Containment Failure Modes. As in the previous event node, this top event node is used to characterize implications of containment failure on the magnitudes and duration of fission product release to the environment.

Attachment E-1 Typical CET For A Plant Damage State

1 Page

ATTACHMENT E-1
TYPICAL CET FOR A PLANT DAMAGE STATE



GENERIC CET STRUCTURE .\ETREE\CET.TRE 9-02-93

Attachment E-2 St. Lucie CET Top Event Logic

51 Pages

ST. LUCIE UNITS 1 & 2
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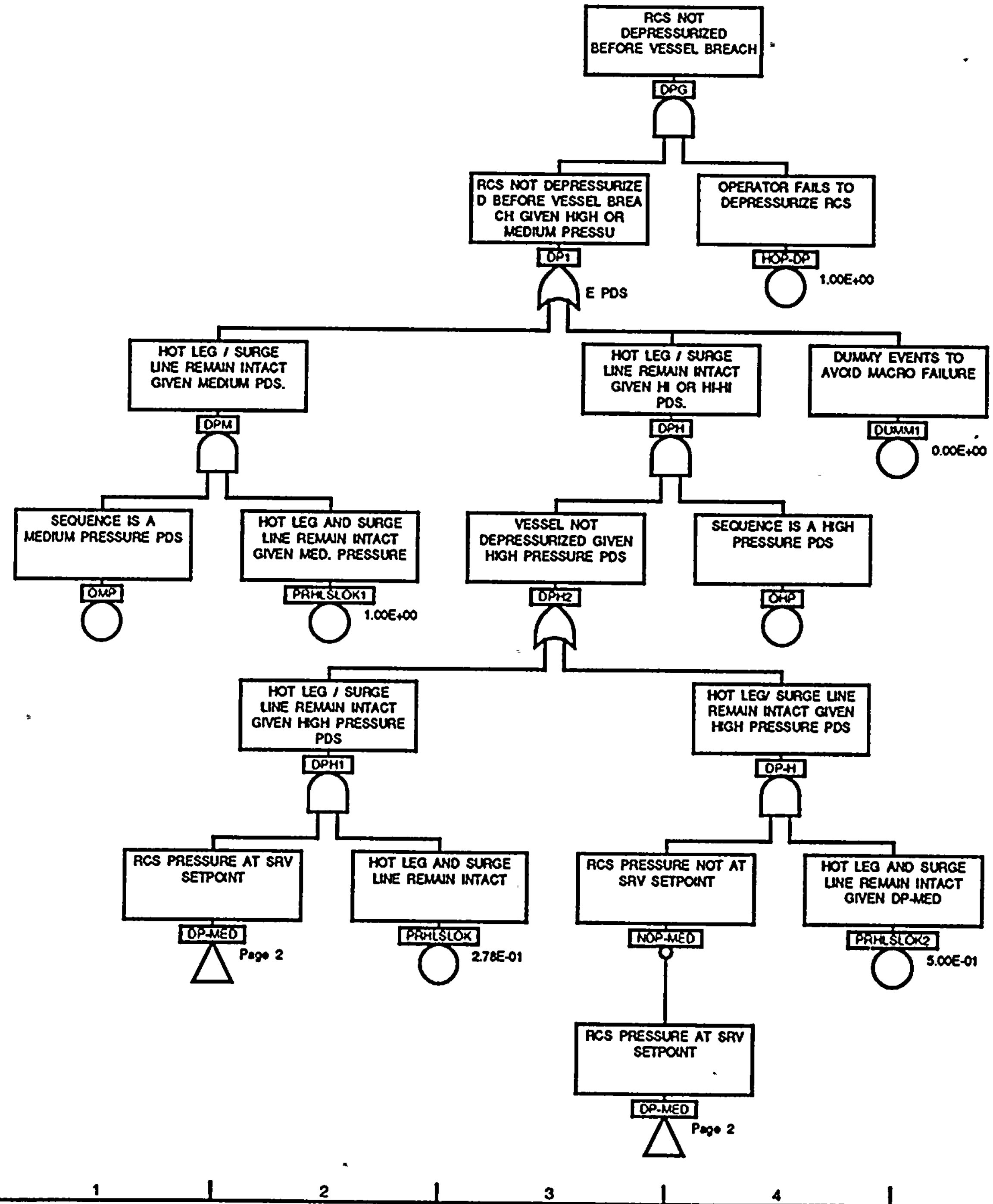
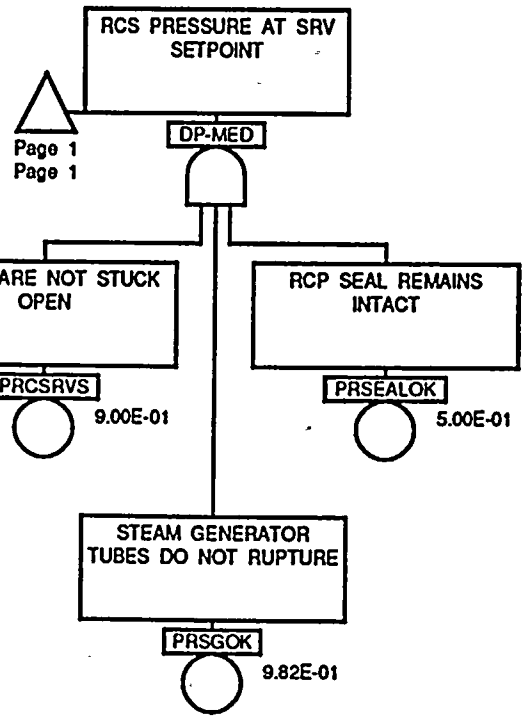
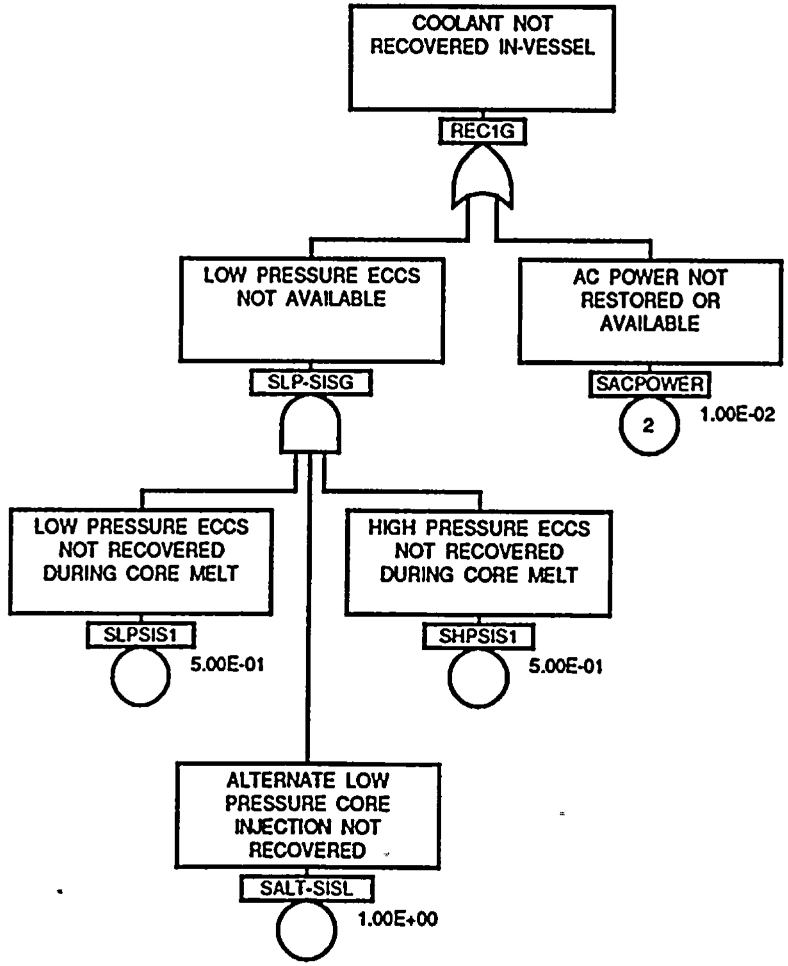
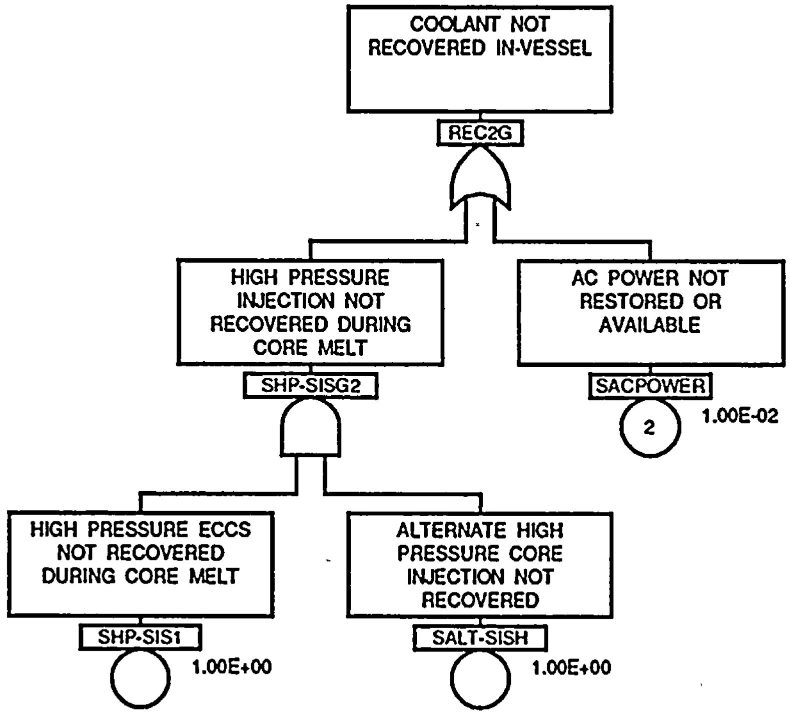
APPENDIX E-2




RCS PRESSURE AT SRV SETPOINT
Page 1
Page 1
DP-MED
SRV s ARE NOT STUCK OPEN
PRCSRVS
9.00E-01
RCP SEAL REMAINS INTACT
PRSEALOK
5.00E-01
STEAM GENERATOR TUBES DO NOT RUPTURE
PRSGOK
9.82E-01


COOLANT NOT RECOVERED IN-VESSEL
REC1G
LOW PRESSURE ECCS NOT AVAILABLE
SLP-SISG
AC POWER NOT RESTORED OR AVAILABLE
SACPOWER
2
1.00E-02
LOW PRESSURE ECCS NOT RECOVERED DURING CORE MELT
SLPSIS1
5.00E-01
HIGH PRESSURE ECCS NOT RECOVERED DURING CORE MELT
SHPSIS1
5.00E-01
ALTERNATE LOW PRESSURE CORE INJECTION NOT RECOVERED
SALT-SISL
1.00E+00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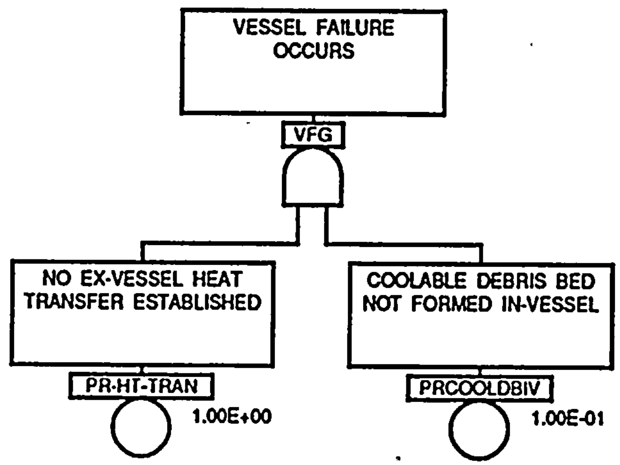
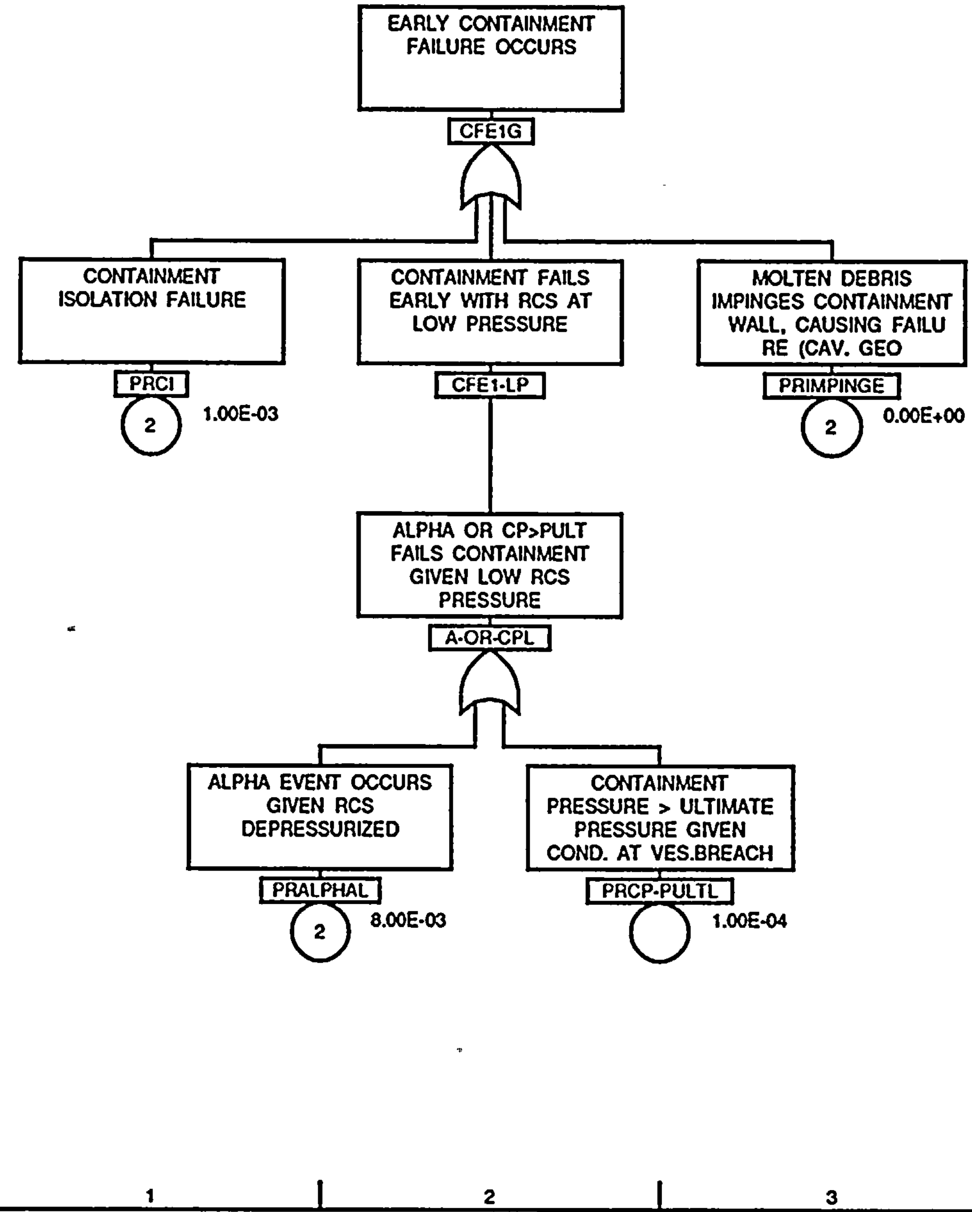
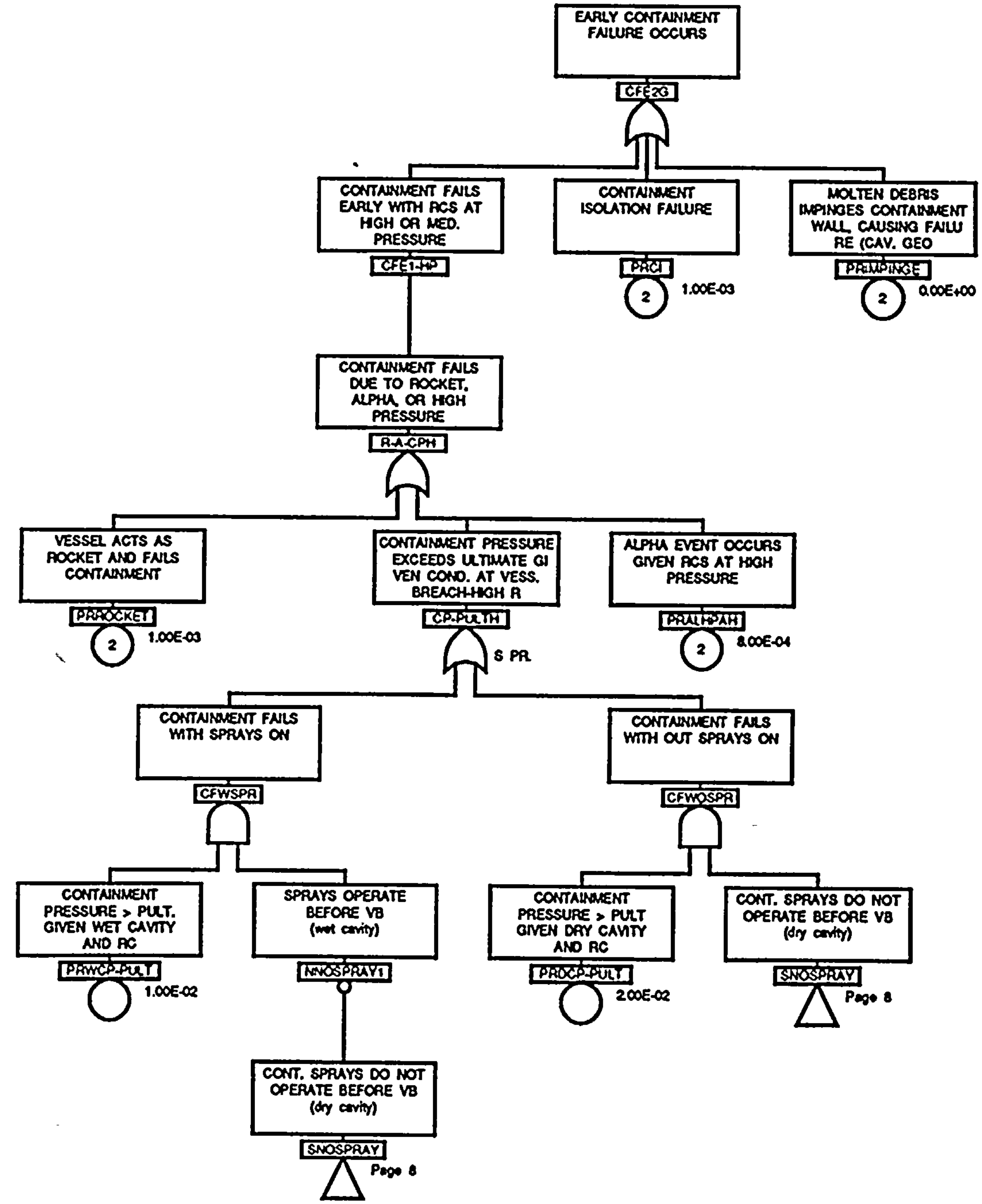
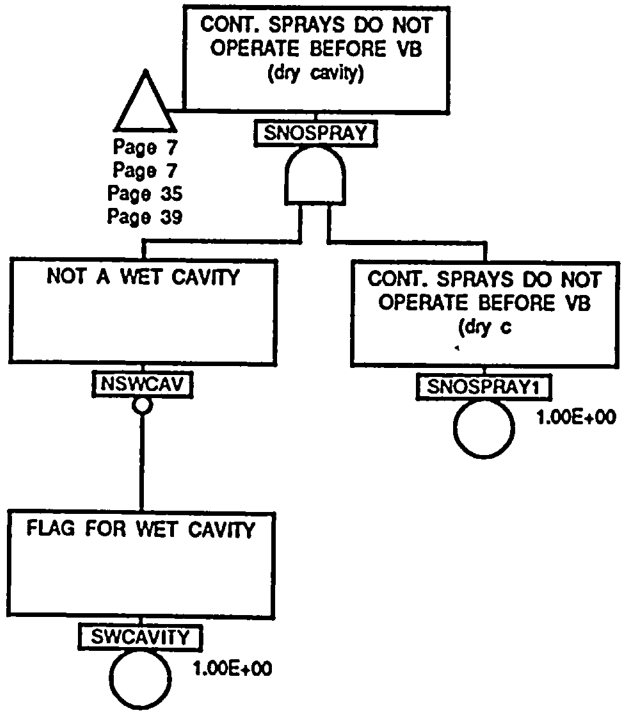
REC2G
HIGH PRESSURE INJECTION NOT RECOVERED DURING CORE MELT
SHP-SISG2
AC POWER NOT RESTORED OR AVAILABLE
SACPOWER
2
1.00E-02
HIGH PRESSURE ECCS NOT RECOVERED DURING CORE MELT
SHP-SIS1
1.00E+00
ALTERNATE HIGH PRESSURE CORE INJECTION NOT RECOVERED
SALT-SISH
1.00E+00

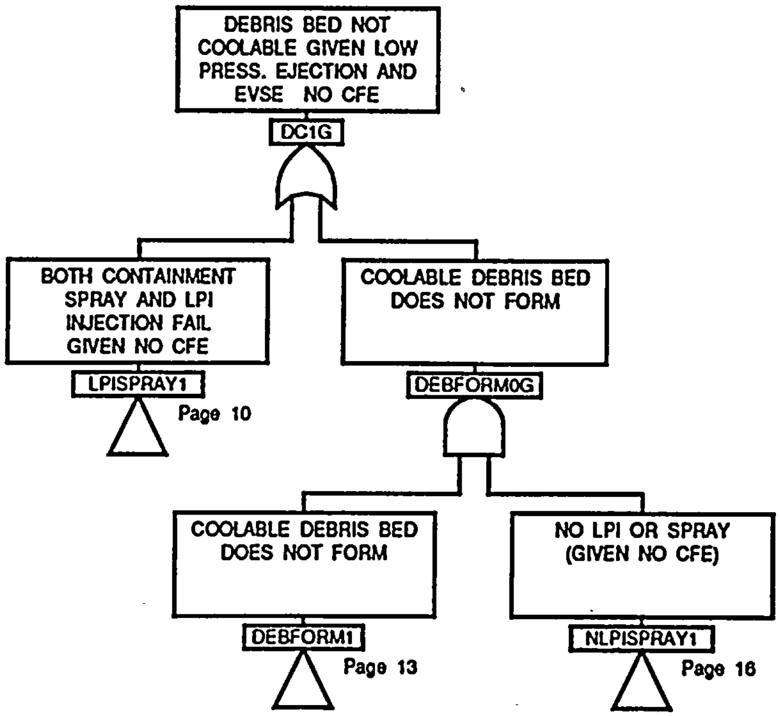

VESSEL FAILURE OCCURS
VFG
NO EX-VESSEL HEAT TRANSFER ESTABLISHED
PR-HT-TRAN
1.00E+00
COOLABLE DEBRIS BED NOT FORMED IN-VESSEL
PRCOOLDBIV
1.00E-01


EARLY CONTAINMENT FAILURE OCCURS
CFE1G
CONTAINMENT ISOLATION FAILURE
PRCI
2
1.00E-03
CONTAINMENT FAILS EARLY WITH RCS AT LOW PRESSURE
CFE1-LP
MOLTEN DEBRIS IMPINGES CONTAINMENT WALL, CAUSING FAILU RE (CAV. GEO
PRIMPINGE
2
0.00E+00
ALPHA OR CP>PULT FAILS CONTAINMENT GIVEN LOW RCS PRESSURE
A-OR-CPL
ALPHA EVENT OCCURS GIVEN RCS DEPRESSURIZED
PRALPHAL
2
8.00E-03
CONTAINMENT PRESSURE > ULTIMATE PRESSURE GIVEN COND. AT VES.BREACH
PRCP-PULTL
1.00E-04
1
2
3

ST. LUCIE UNITS 1 & 2
IPE SUBMITTAL REV. 0
APPENDIX E-2




CONT. SPRAYS DO NOT OPERATE BEFORE VB (dry cavity)
SNOSPRAY
Page 7
Page 7
Page 35
Page 39
NOT A WET CAVITY
NSWCAV
CONT. SPRAYS DO NOT OPERATE BEFORE VB (dry c
SNOSPRAY1
1.00E+00
FLAG FOR WET CAVITY
SWCAVITY
1.00E+00
1
2


DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND EVSE NO CFE
DC1G
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN NO CFE
LPISPRAY1
Page 10
COOLABLE DEBRIS BED DOES NOT FORM
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COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM1
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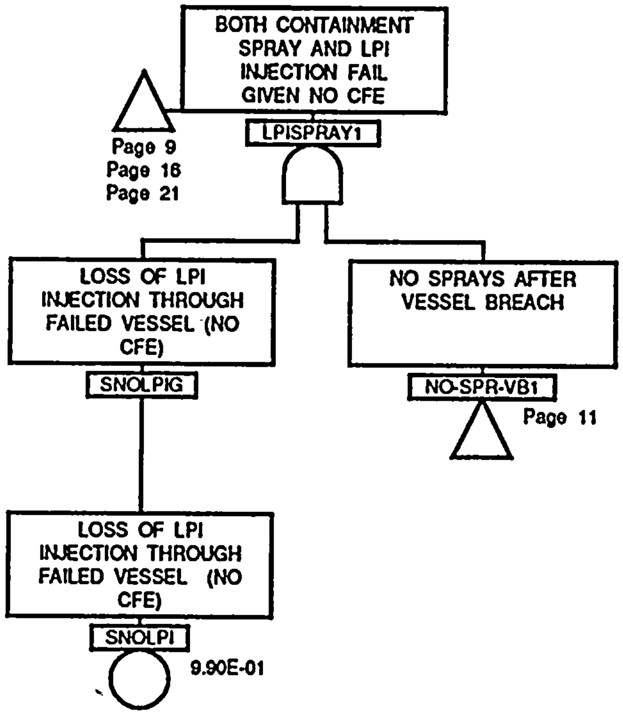
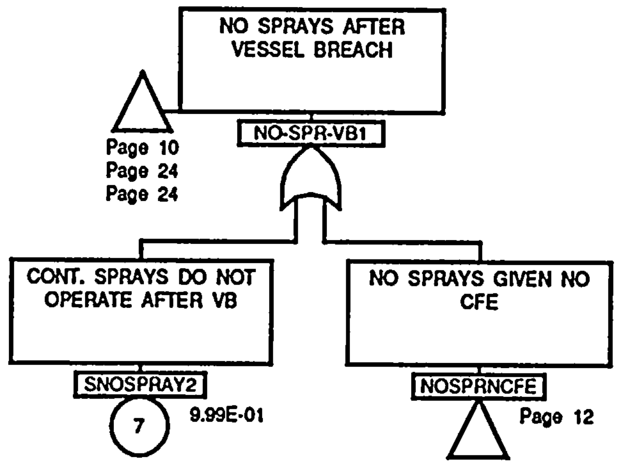
NO LPI OR SPRAY (GIVEN NO CFE)
NLPISPRAY1
Page 16


BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN NO CFE
LPISPRAY1
Page 9
Page 16
Page 21
LOSS OF LPI INJECTION THROUGH FAILED VESSEL (NO CFE)
SNOLPIG
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB1
Page 11
LOSS OF LPI INJECTION THROUGH FAILED VESSEL (NO CFE)
SNOLPI
9.90E-01
1
2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Page 24
Page 24
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NOSPRNCFE
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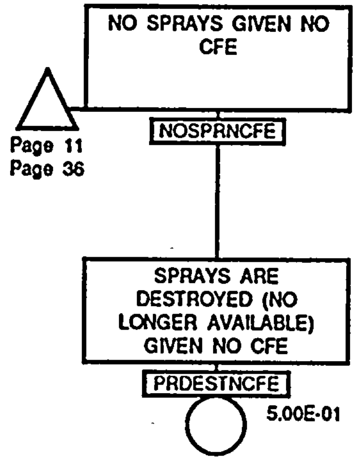
1
2


NO SPRAYS GIVEN NO CFE
NOSPRNCFE
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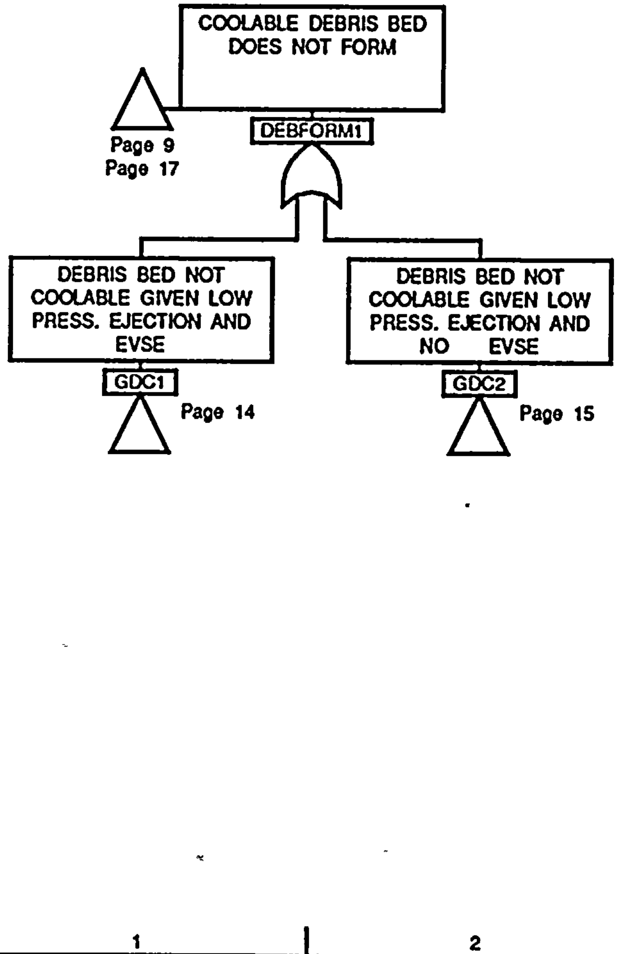
Page 36
SPRAYS ARE DESTROYED (NO LONGER AVAILABLE) GIVEN NO CFE
PRDESTNCFE
5.00E-01


COOLABLE DEBRIS BED DOES NOT FORM
Page 9
Page 17
DEBFORM1
DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND EVSE
GDC1
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DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND NO EVSE
GDC2
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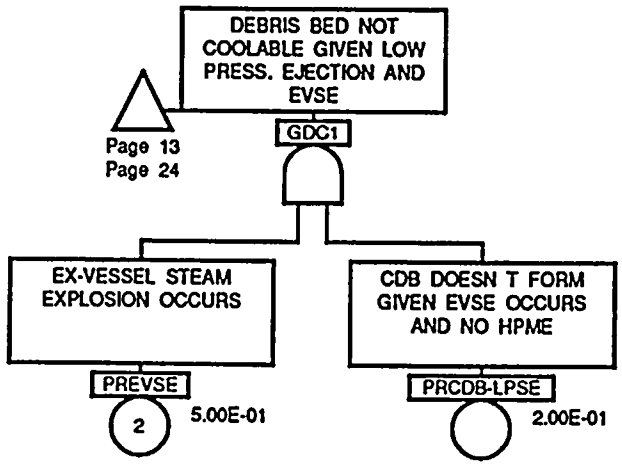
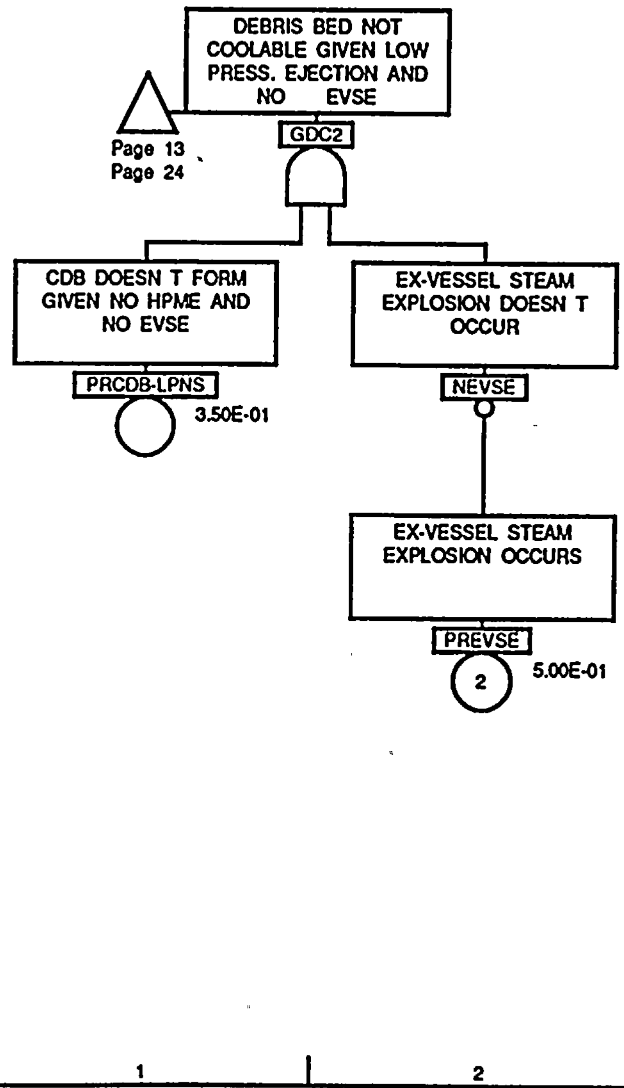
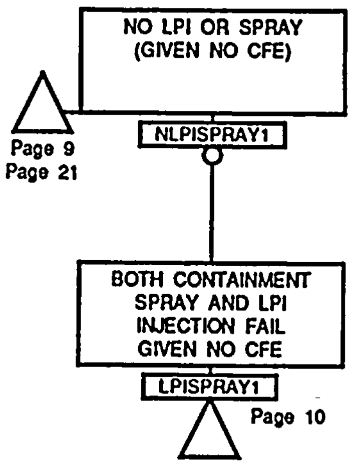
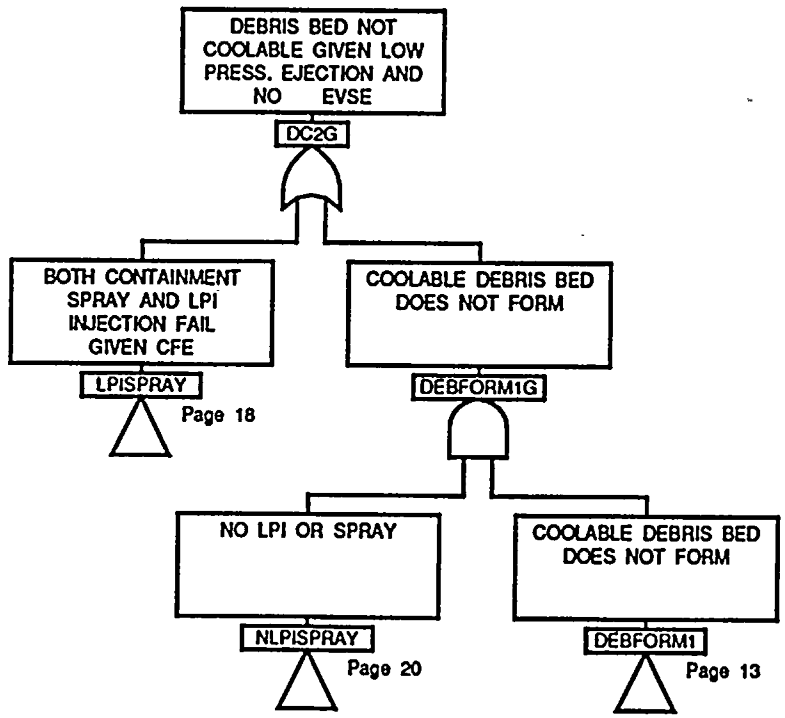
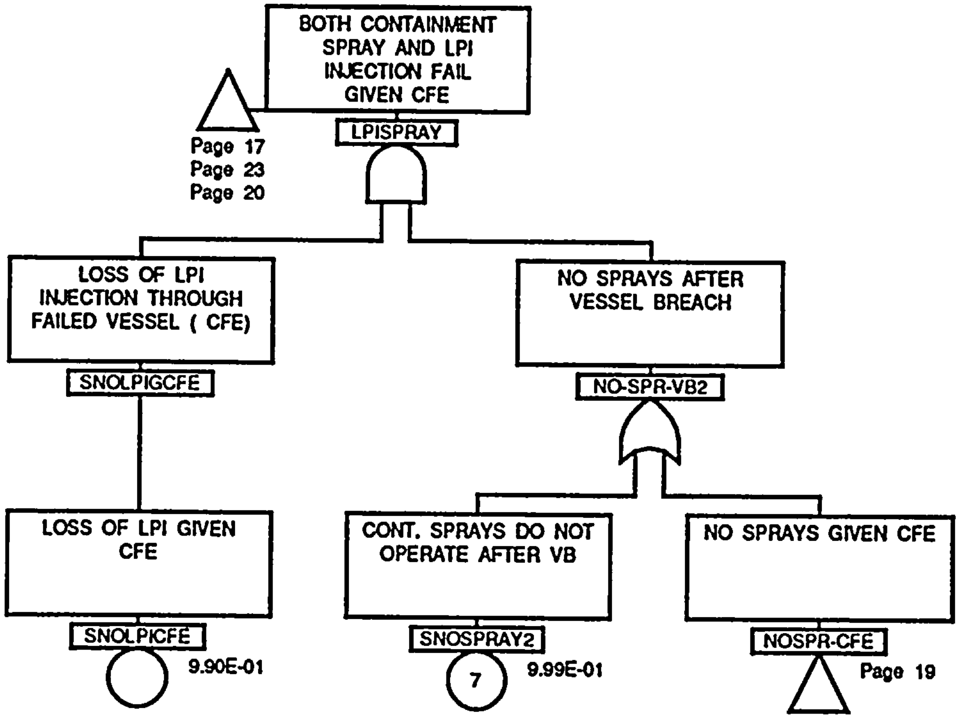
1
2

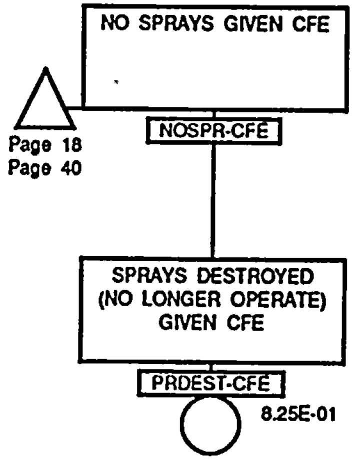

DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND EVSE
Page 13
Page 24
GDC1
EX-VESSEL STEAM EXPLOSION OCCURS
PREVSE
2
5.00E-01
CDB DOESN T FORM GIVEN EVSE OCCURS AND NO HPME
PRCDB-LPSE
2.00E-01


DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND NO EVSE
GDC2
Page 13
Page 24
CDB DOESN T FORM GIVEN NO HPME AND NO EVSE
PRCDB-LPNS
3.50E-01
EX-VESSEL STEAM EXPLOSION DOESN T OCCUR
NEVSE
EX-VESSEL STEAM EXPLOSION OCCURS
PREVSE
2
5.00E-01
1
2

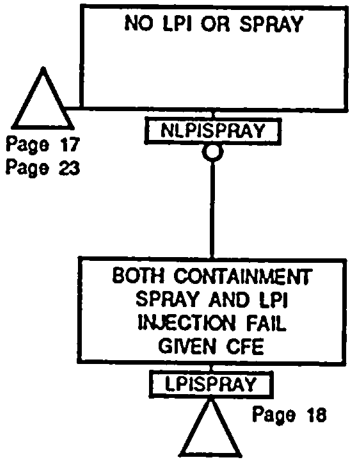

NO LPI OR SPRAY (GIVEN NO CFE)
NLPISPRAY1
Page 9
Page 21
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN NO CFE
LPISPRAY1
Page 10
1

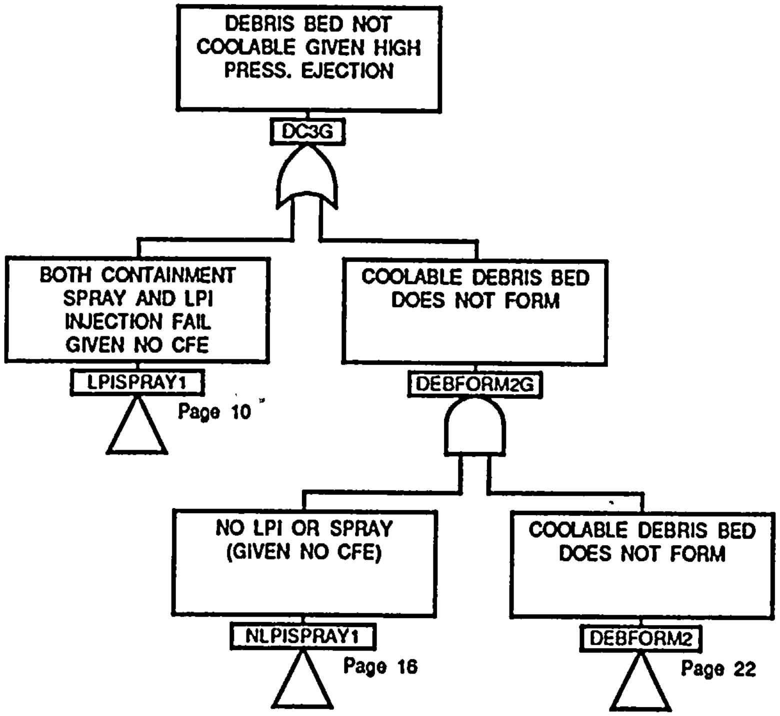

DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND NO EVSE
DC2G
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN CFE
LPISPRAY
Page 18
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM1G
NO LPI OR SPRAY
NLPISPRAY
Page 20
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM1
Page 13

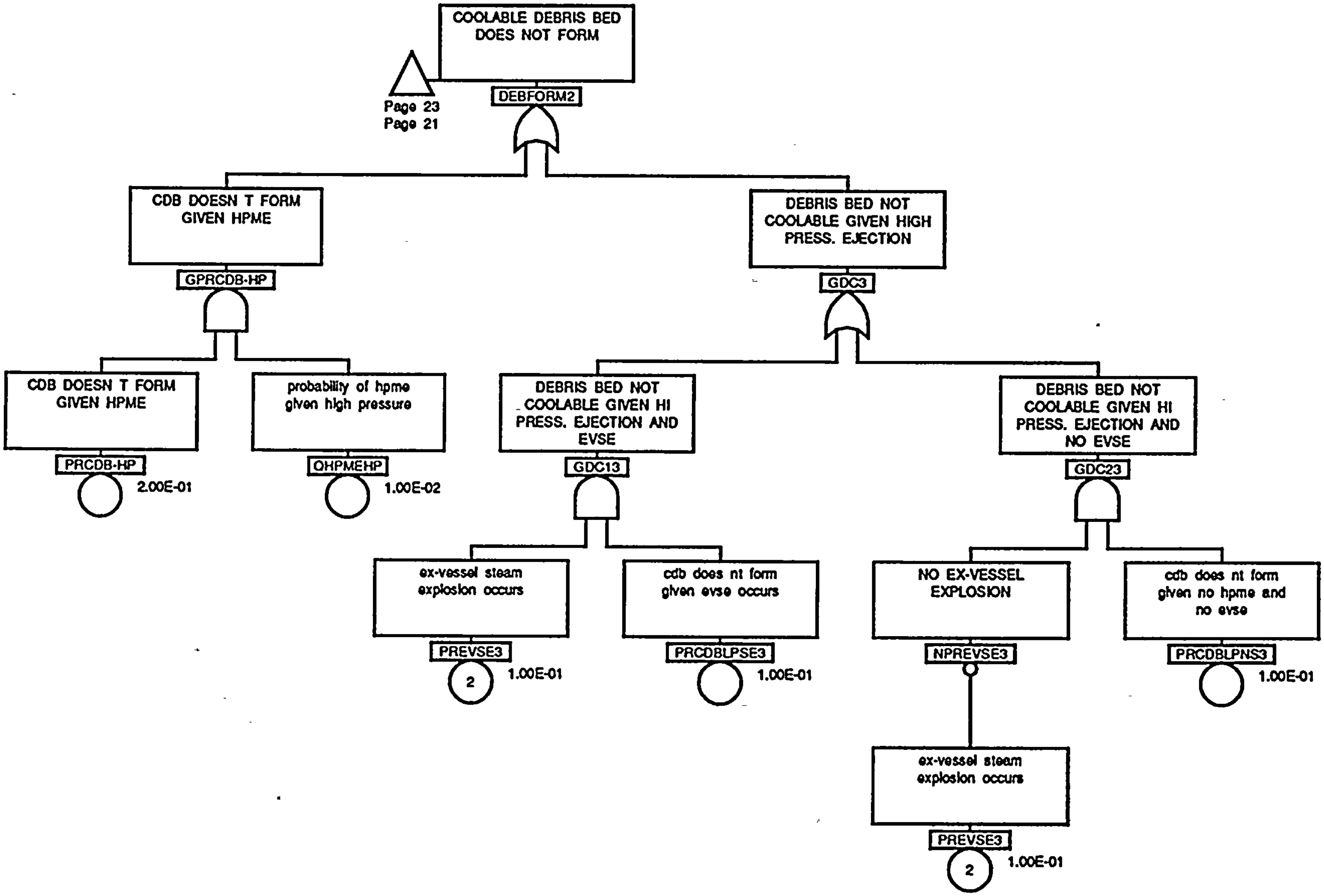

BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN CFE
LPISPRAY
Page 17
Page 23
Page 20
LOSS OF LPI INJECTION THROUGH FAILED VESSEL (CFE)
SNOLPIGCFE
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB2
LOSS OF LPI GIVEN CFE
SNOLPICFE
9.90E-01
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
NO SPRAYS GIVEN CFE
NOSPR-CFE
Page 19

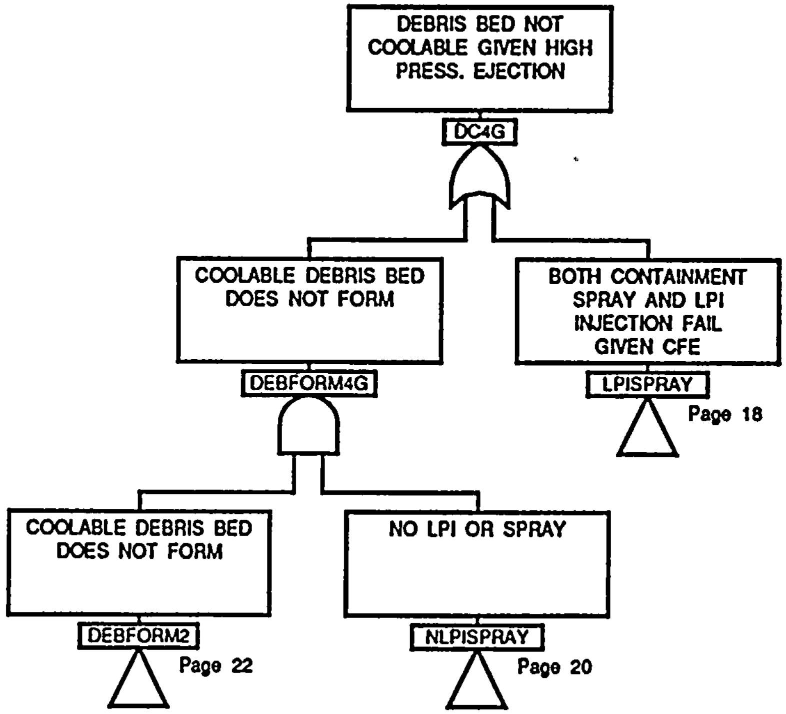

NO SPRAYS GIVEN CFE
NOSPR-CFE
Page 18
Page 40
SPRAYS DESTROYED (NO LONGER OPERATE) GIVEN CFE
PRDEST-CFE
8.25E-01

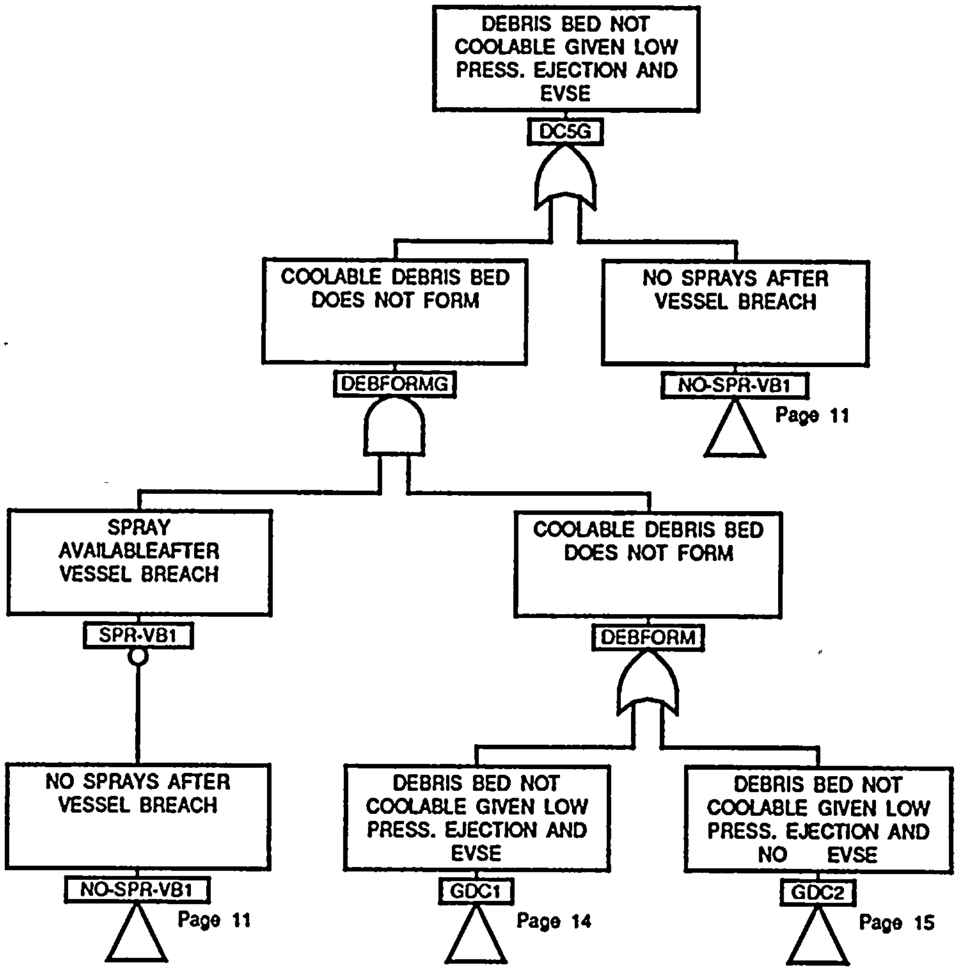
1

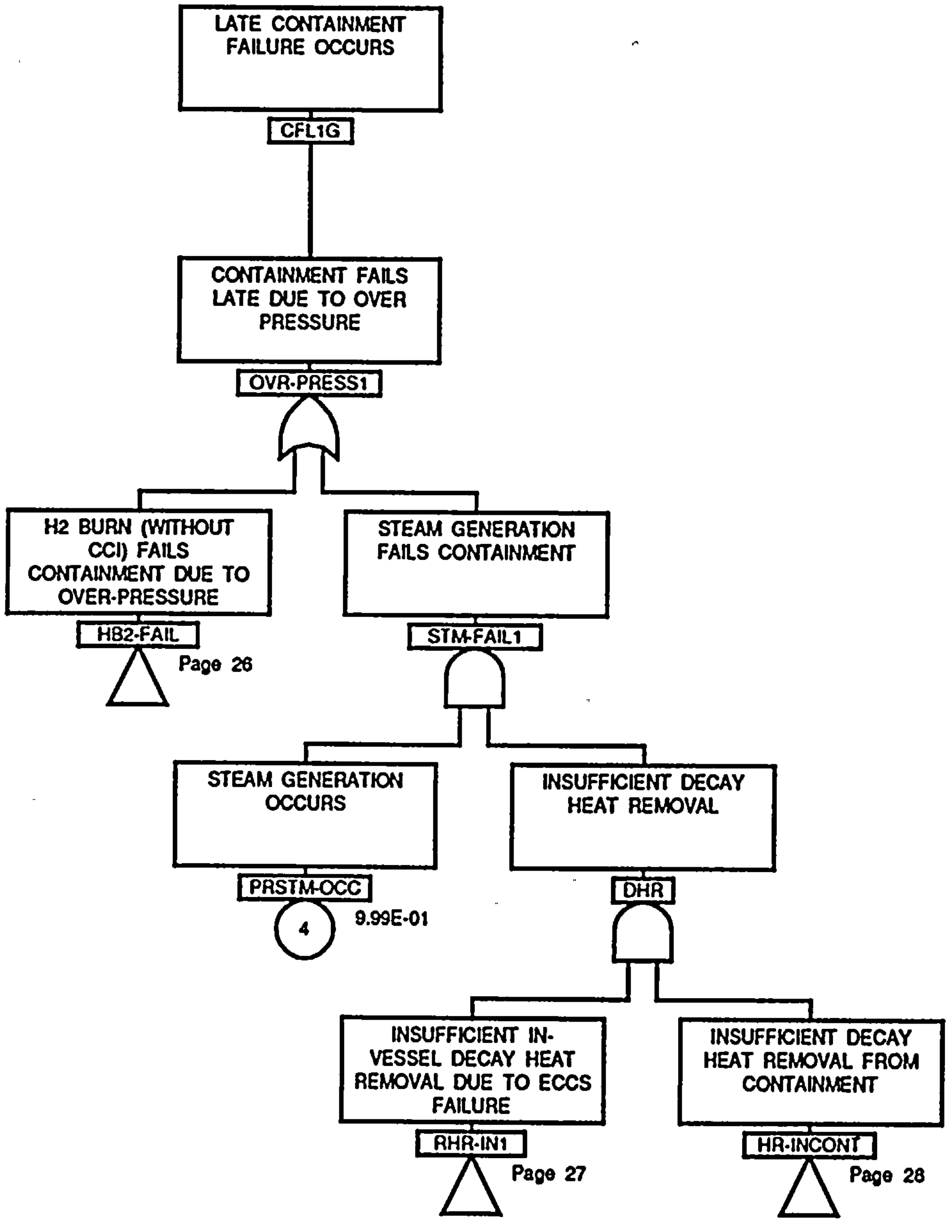

NO LPI OR SPRAY
NLPISPRAY
Page 17
Page 23
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN CFE
LPISPRAY
Page 18

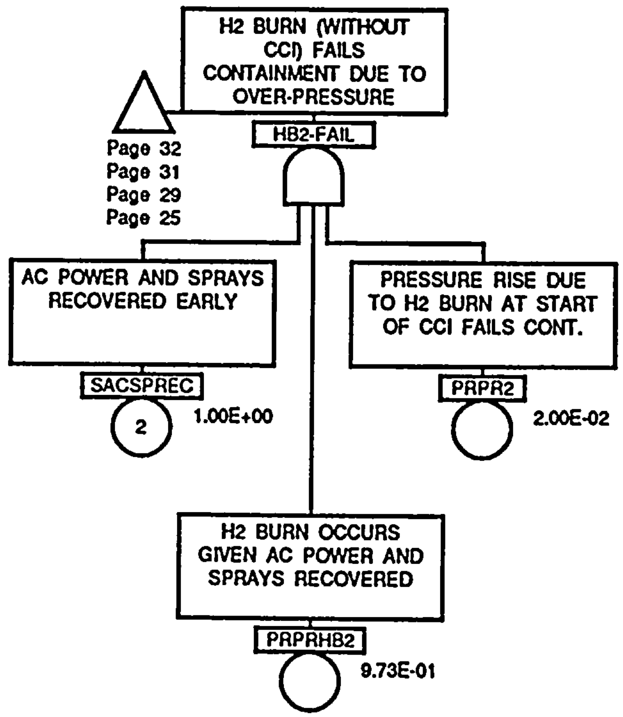

DEBRIS BED NOT COOLABLE GIVEN HIGH PRESS. EJECTION
DC3G
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN NO CFE
LPISPRAY1
Page 10
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM2G
NO LPI OR SPRAY (GIVEN NO CFE)
NLPISPRAY1
Page 16
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM2
Page 22

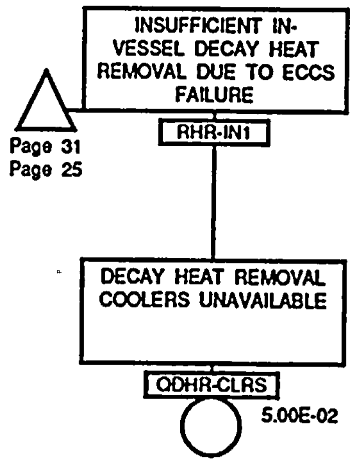

COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM2
Page 23
Page 21
CDB DOESN T FORM GIVEN HPME
GPRCDB-HP
DEBRIS BED NOT COOLABLE GIVEN HIGH PRESS. EJECTION
GDC3
CDB DOESN T FORM GIVEN HPME
PRCDB-HP
2.00E-01
probability of hpme given high pressure
QHPMEHP
1.00E-02
DEBRIS BED NOT COOLABLE GIVEN HI PRESS. EJECTION AND EVSE
GDC13
DEBRIS BED NOT COOLABLE GIVEN HI PRESS. EJECTION AND NO EVSE
GDC23
ex-vessel steam explosion occurs
PREVSE3
2
1.00E-01
cdb does nt form given evse occurs
PRCDBLPSE3
1.00E-01
NO EX-VESSEL EXPLOSION
NPREVSE3
cdb does nt form given no hpme and no evse
PRCDBLPNS3
1.00E-01
ex-vessel steam explosion occurs
PREVSE3
2
1.00E-01
1
2
3
4
5

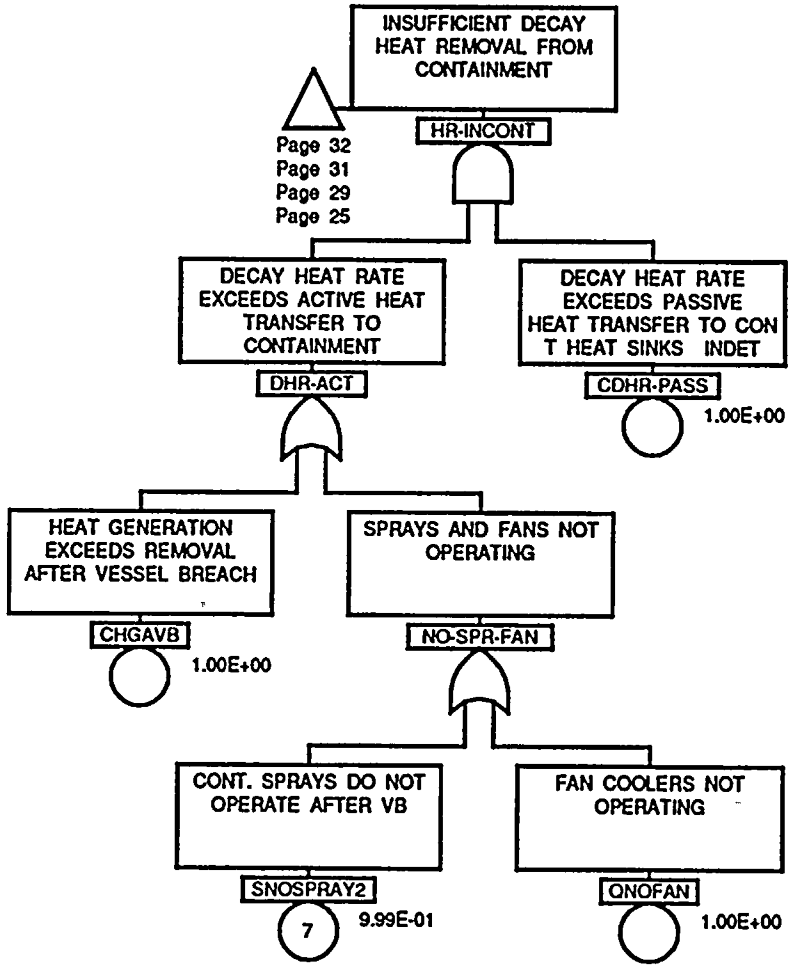

DEBRIS BED NOT COOLABLE GIVEN HIGH PRESS. EJECTION
DC4G
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM4G
BOTH CONTAINMENT SPRAY AND LPI INJECTION FAIL GIVEN CFE
LPISPRAY
Page 18
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM2
Page 22
NO LPI OR SPRAY
NLPISPRAY
Page 20

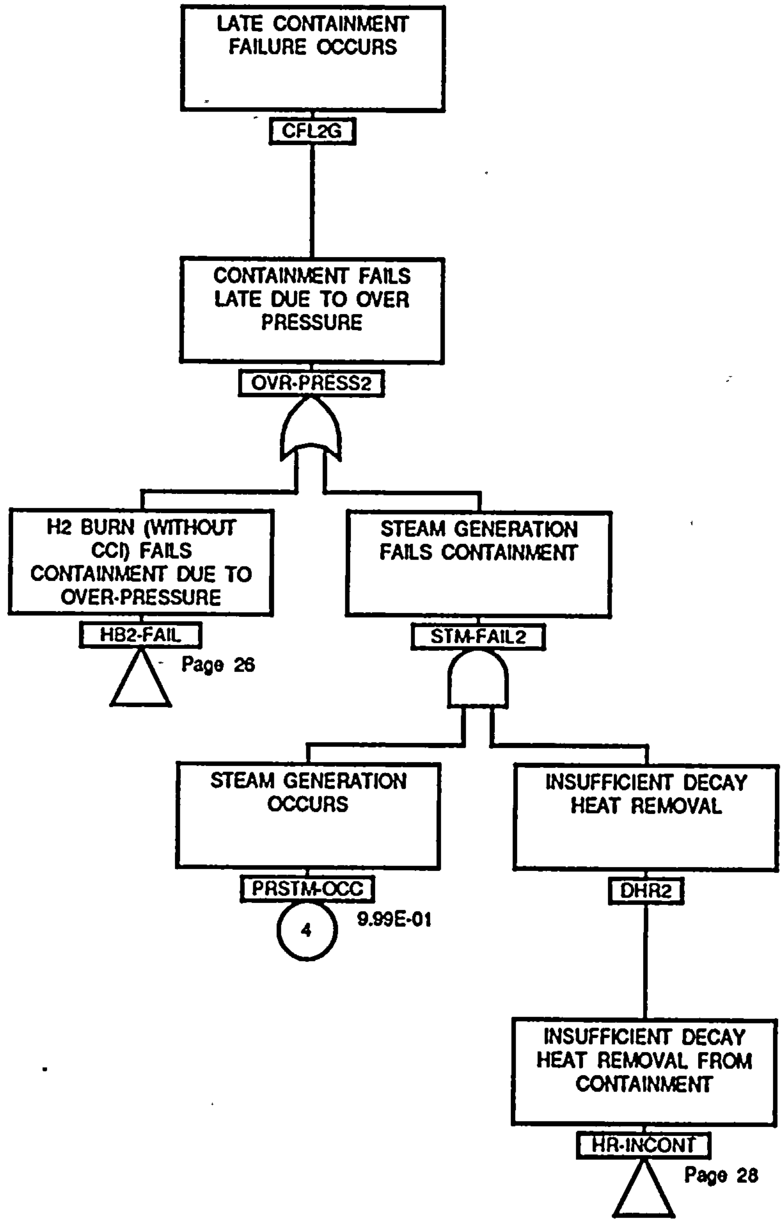

DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND EVSE
DC5G
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORMG
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB1
Page 11
SPRAY AVAILABLEAFTER VESSEL BREACH
SPR-VB1
COOLABLE DEBRIS BED DOES NOT FORM
DEBFORM
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB1
Page 11
DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND EVSE
GDC1
Page 14
DEBRIS BED NOT COOLABLE GIVEN LOW PRESS. EJECTION AND NO EVSE
GDC2
Page 15

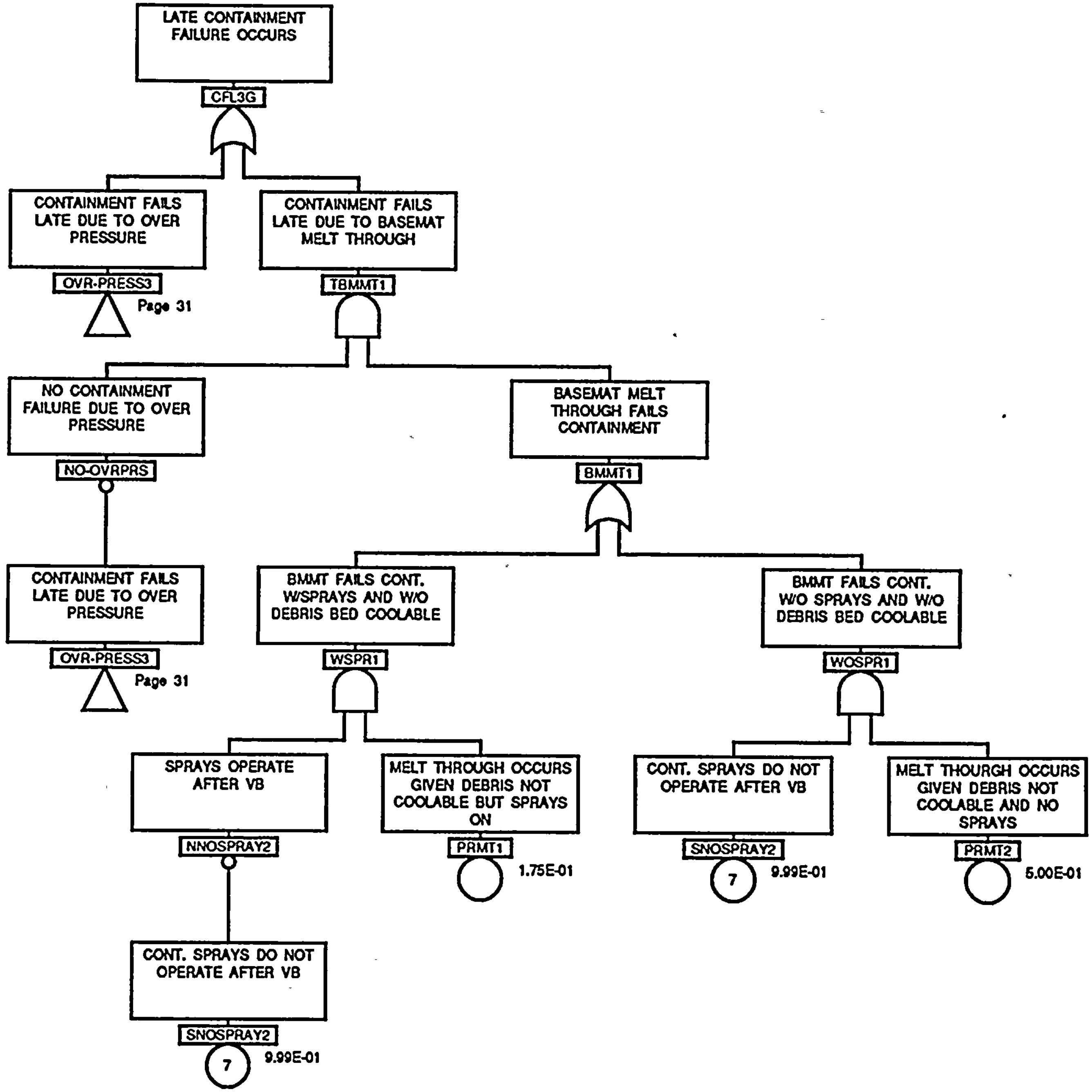

LATE CONTAINMENT FAILURE OCCURS
CFL1G
CONTAINMENT FAILS LATE DUE TO OVER PRESSURE
OVR-PRESS1
H2 BURN (WITHOUT CCI) FAILS CONTAINMENT DUE TO OVER-PRESSURE
HB2-FAIL
Page 26
STEAM GENERATION FAILS CONTAINMENT
STM-FAIL1
STEAM GENERATION OCCURS
PRSTM-OCC
4
9.99E-01
INSUFFICIENT DECAY HEAT REMOVAL
DHR
INSUFFICIENT IN-VESSEL DECAY HEAT REMOVAL DUE TO ECCS FAILURE
RHR-IN1
Page 27
INSUFFICIENT DECAY HEAT REMOVAL FROM CONTAINMENT
HR-INCONT
Page 28


H2 BURN (WITHOUT CCI) FAILS CONTAINMENT DUE TO OVER-PRESSURE
HB2-FAIL
Page 32
Page 31
Page 29
Page 25
AC POWER AND SPRAYS RECOVERED EARLY
SACSPREC
2
1.00E+00
PRESSURE RISE DUE TO H2 BURN AT START OF CCI FAILS CONT.
PRPR2
2.00E-02
H2 BURN OCCURS GIVEN AC POWER AND SPRAYS RECOVERED
PRPRHB2
9.73E-01
1
2

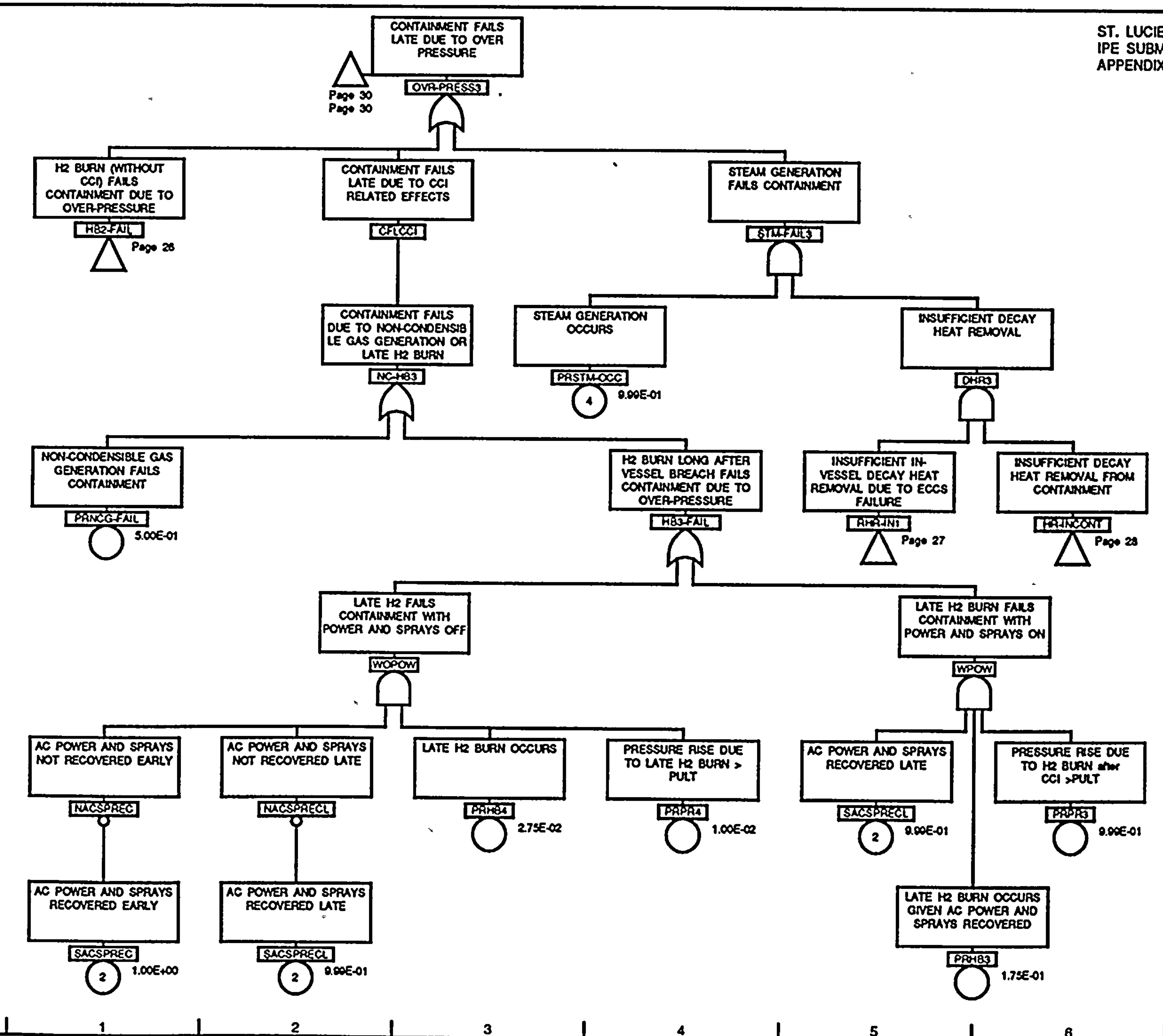

INSUFFICIENT IN-VESSEL DECAY HEAT REMOVAL DUE TO ECCS FAILURE
RHR-IN1
Page 31
Page 25
DECAY HEAT REMOVAL COOLERS UNAVAILABLE
QDHR-CLRS
5.00E-02

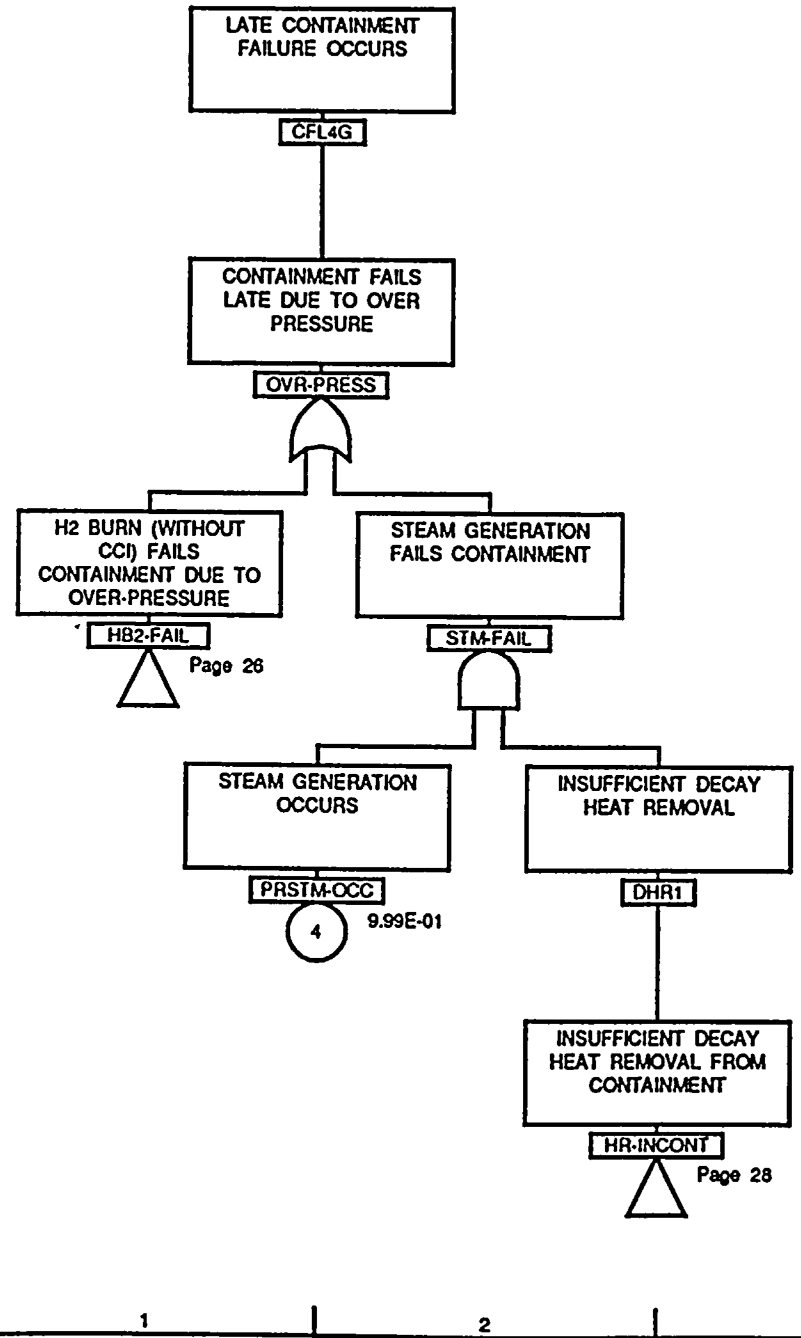

INSUFFICIENT DECAY HEAT REMOVAL FROM CONTAINMENT
HR-INCONT
Page 32
Page 31
Page 29
Page 25
DECAY HEAT RATE EXCEEDS ACTIVE HEAT TRANSFER TO CONTAINMENT
DHR-ACT
DECAY HEAT RATE EXCEEDS PASSIVE HEAT TRANSFER TO CON T HEAT SINKS INDET
CDHR-PASS
1.00E+00
HEAT GENERATION EXCEEDS REMOVAL AFTER VESSEL BREACH
CHGAVB
1.00E+00
SPRAYS AND FANS NOT OPERATING
NO-SPR-FAN
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
FAN COOLERS NOT OPERATING
QNOFAN
1.00E+00

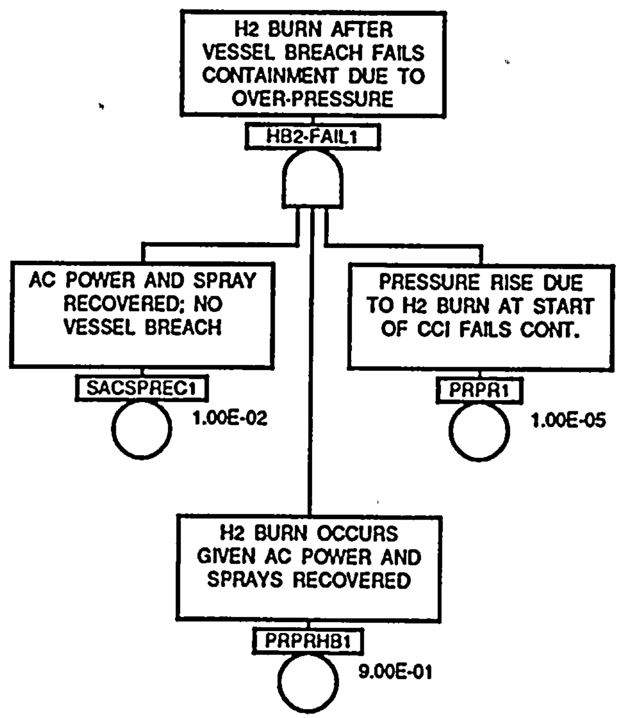

LATE CONTAINMENT FAILURE OCCURS
CFL2G
CONTAINMENT FAILS LATE DUE TO OVER PRESSURE
OVR-PRESS2
H2 BURN (WITHOUT CCI) FAILS CONTAINMENT DUE TO OVER-PRESSURE
HB2-FAIL
Page 26
STEAM GENERATION FAILS CONTAINMENT
STM-FAIL2
STEAM GENERATION OCCURS
PRSTM-OCC
4
9.99E-01
INSUFFICIENT DECAY HEAT REMOVAL
DHR2
INSUFFICIENT DECAY HEAT REMOVAL FROM CONTAINMENT
HR-INCONT
Page 28
1
2

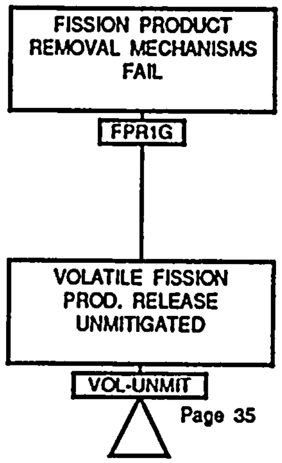

LATE CONTAINMENT FAILURE OCCURS
CFL3G
CONTAINMENT FAILS LATE DUE TO OVER PRESSURE
OVR-PRESS3
Page 31
CONTAINMENT FAILS LATE DUE TO BASEMAT MELT THROUGH
TBMMT1
NO CONTAINMENT FAILURE DUE TO OVER PRESSURE
NO-OVRPRS
CONTAINMENT FAILS LATE DUE TO OVER PRESSURE
OVR-PRESS3
Page 31
BASEMAT MELT THROUGH FAILS CONTAINMENT
BMMT1
BMMT FAILS CONT. W/SPRAYS AND W/O DEBRIS BED COOLABLE
WSPR1
BMMT FAILS CONT. W/O SPRAYS AND W/O DEBRIS BED COOLABLE
WOSPR1
SPRAYS OPERATE AFTER VB
NNOSPRAY2
MELT THROUGH OCCURS GIVEN DEBRIS NOT COOLABLE BUT SPRAYS ON
PRMT1
1.75E-01
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
MELT THOURGH OCCURS GIVEN DEBRIS NOT COOLABLE AND NO SPRAYS
PRMT2
5.00E-01
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
1
2
3
4

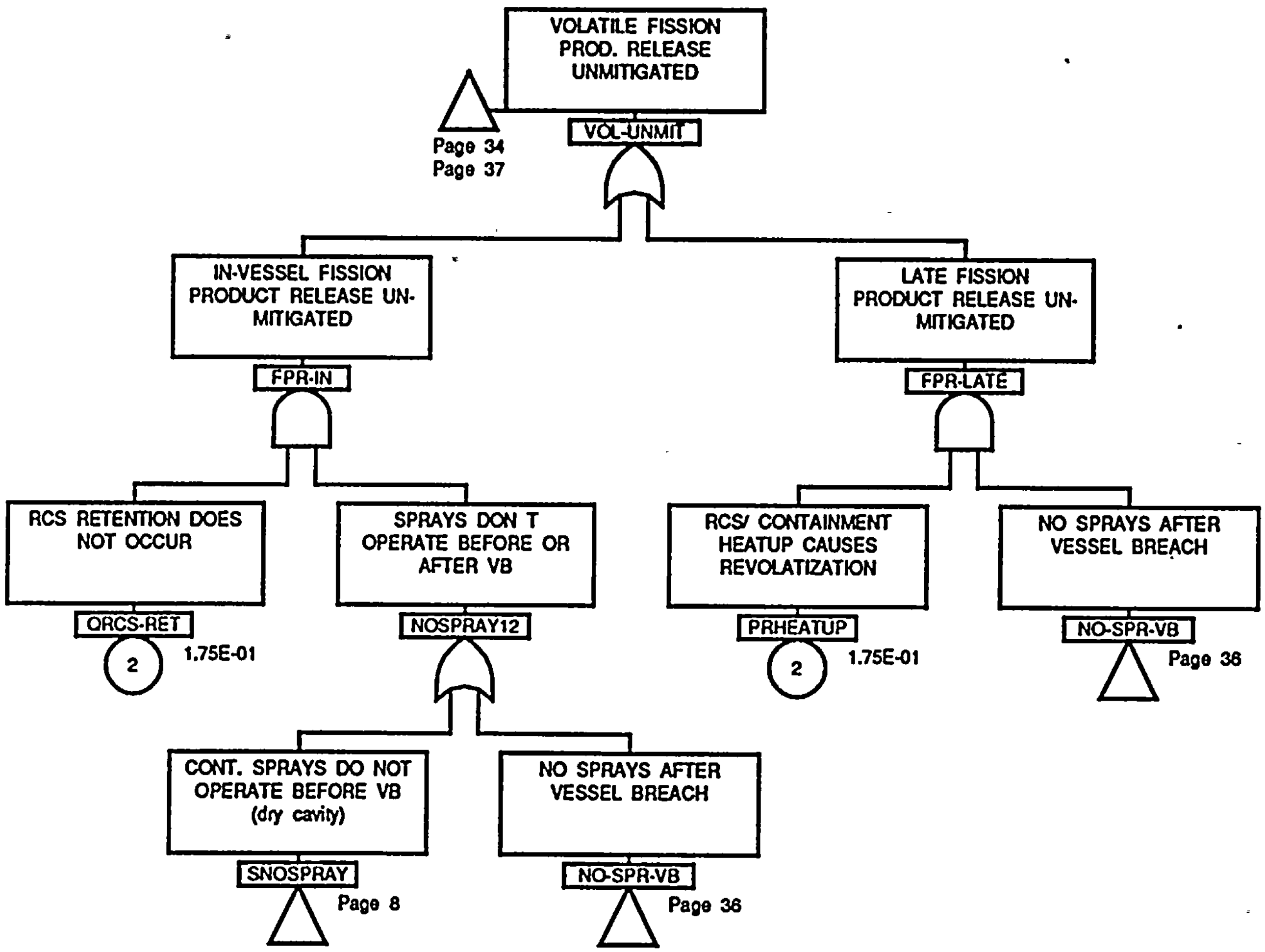
ST. LUCIE UNITS 1 & 2
IPE SUBMITTAL REV. 0
APPENDIX E-2

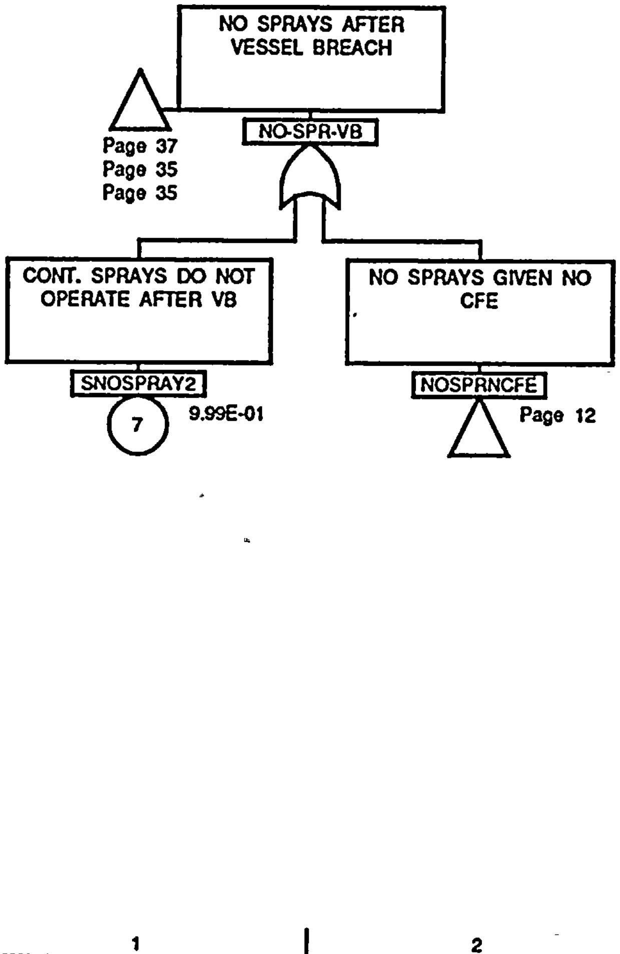


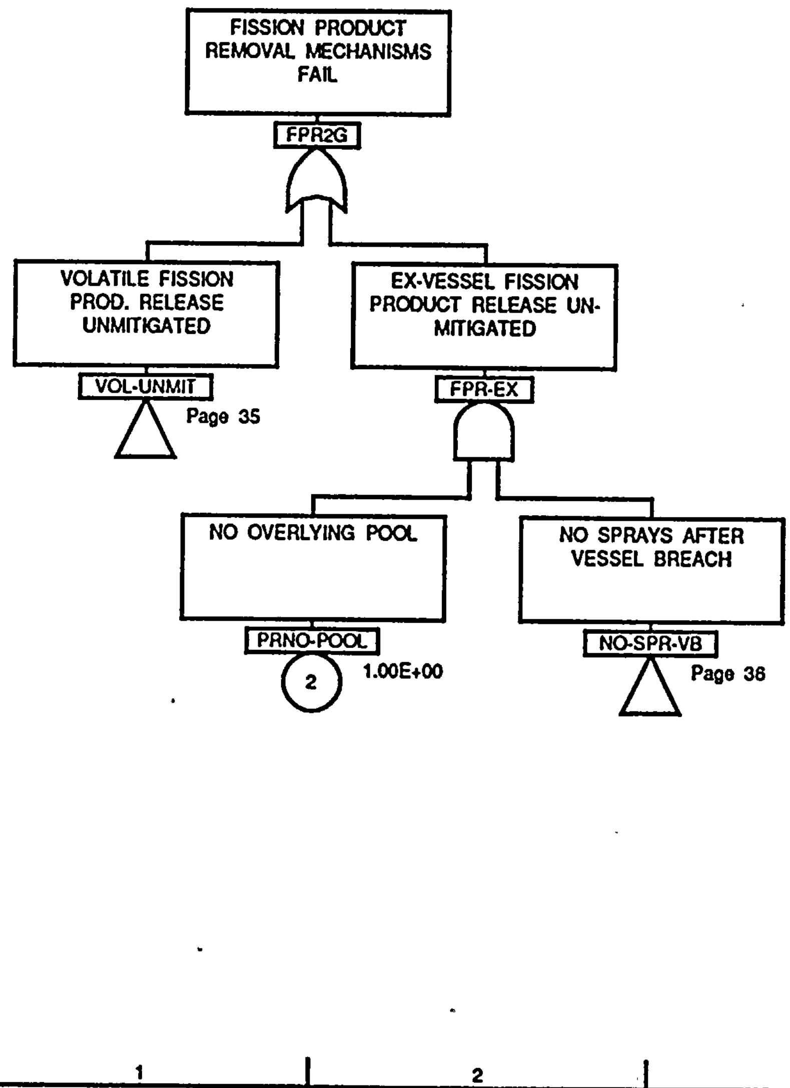

LATE CONTAINMENT FAILURE OCCURS
CFL4G
CONTAINMENT FAILS LATE DUE TO OVER PRESSURE
OVR-PRESS
H2 BURN (WITHOUT CCI) FAILS CONTAINMENT DUE TO OVER-PRESSURE
HB2-FAIL
Page 26
STEAM GENERATION FAILS CONTAINMENT
STM-FAIL
STEAM GENERATION OCCURS
PRSTM-OCC
4
9.99E-01
INSUFFICIENT DECAY HEAT REMOVAL
DHR1
INSUFFICIENT DECAY HEAT REMOVAL FROM CONTAINMENT
HR-INCONT
Page 28
1
2

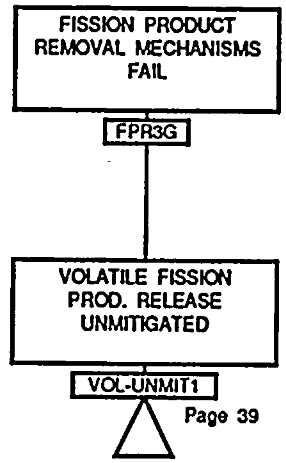

H2 BURN AFTER VESSEL BREACH FAILS CONTAINMENT DUE TO OVER-PRESSURE
HB2-FAIL1
AC POWER AND SPRAY RECOVERED; NO VESSEL BREACH
SACSPREC1
1.00E-02
PRESSURE RISE DUE TO H2 BURN AT START OF CCI FAILS CONT.
PRPR1
1.00E-05
H2 BURN OCCURS GIVEN AC POWER AND SPRAYS RECOVERED
PRPRHB1
9.00E-01

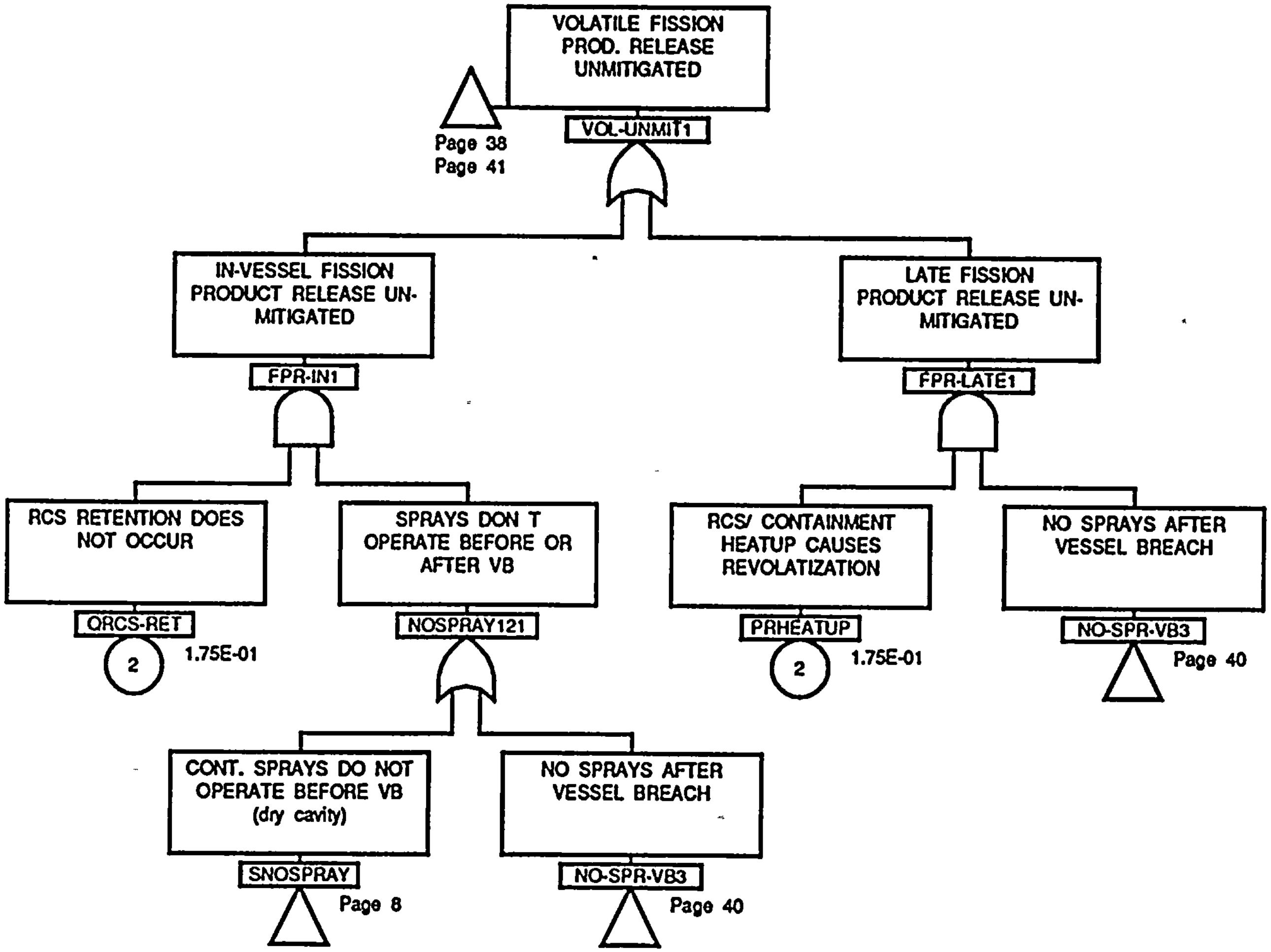

FISSION PRODUCT REMOVAL MECHANISMS FAIL
FPR1G
VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT
Page 35
1

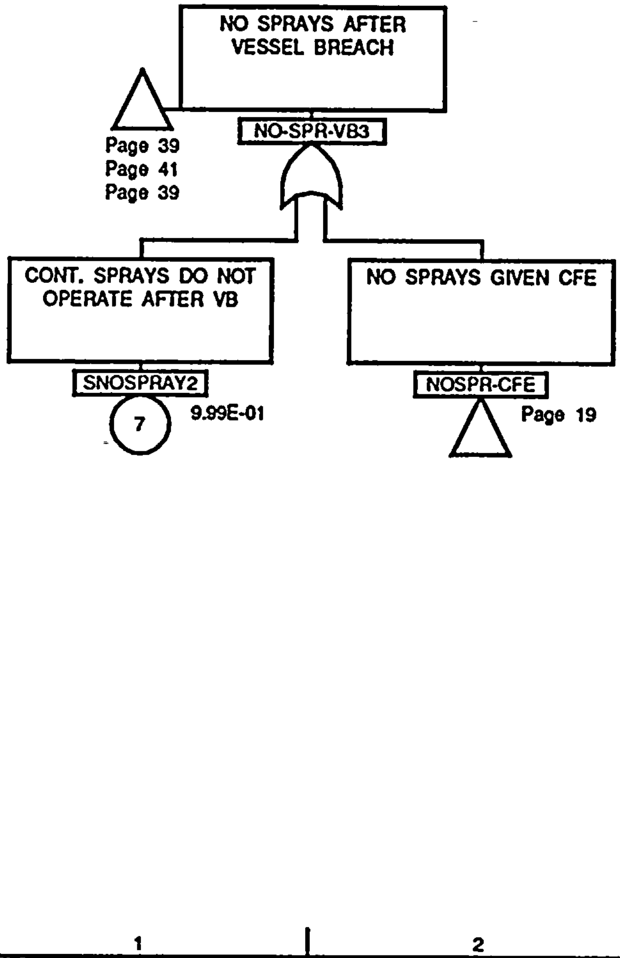

VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT
Page 34
Page 37
IN-VESSEL FISSION PRODUCT RELEASE UN-MITIGATED
FPR-IN
LATE FISSION PRODUCT RELEASE UN-MITIGATED
FPR-LATE
RCS RETENTION DOES NOT OCCUR
QRCS-RET
2
1.75E-01
SPRAYS DON T OPERATE BEFORE OR AFTER VB
NOSPRAY12
RCS/ CONTAINMENT HEATUP CAUSES REVOLATIZATION
PRHEATUP
2
1.75E-01
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB
Page 36
CONT. SPRAYS DO NOT OPERATE BEFORE VB (dry cavity)
SNOSPRAY
Page 8
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB
Page 36

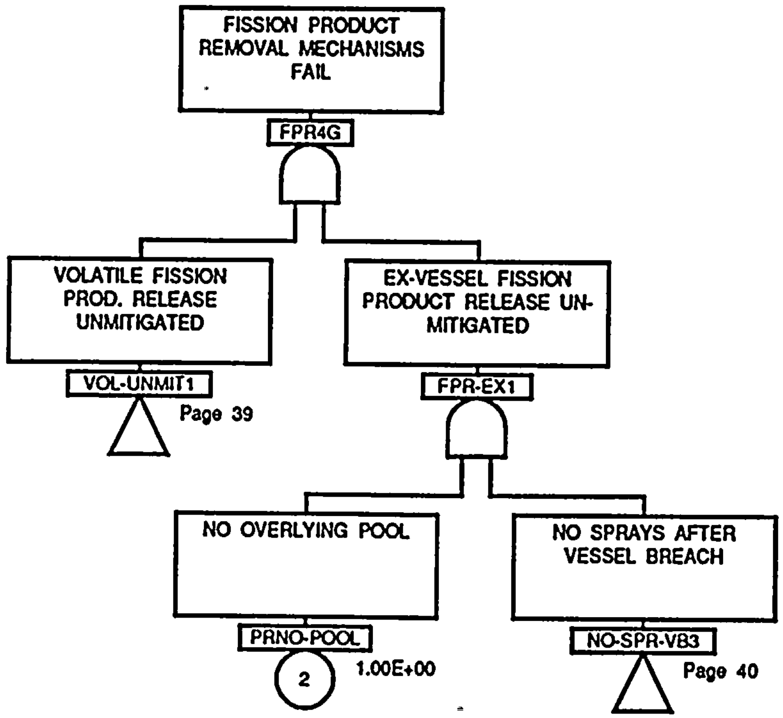

NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB
Page 37
Page 35
Page 35
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
NO SPRAYS GIVEN NO CFE
NOSPRNCFE
Page 12
1
2

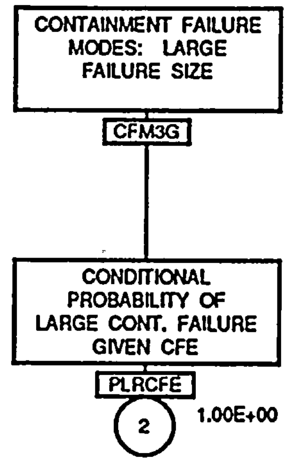

FISSION PRODUCT REMOVAL MECHANISMS FAIL
FPR2G
VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT
Page 35
EX-VESSEL FISSION PRODUCT RELEASE UN-MITIGATED
FPR-EX
NO OVERLYING POOL
PRNO-POOL
2
1.00E+00
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB
Page 36
1
2

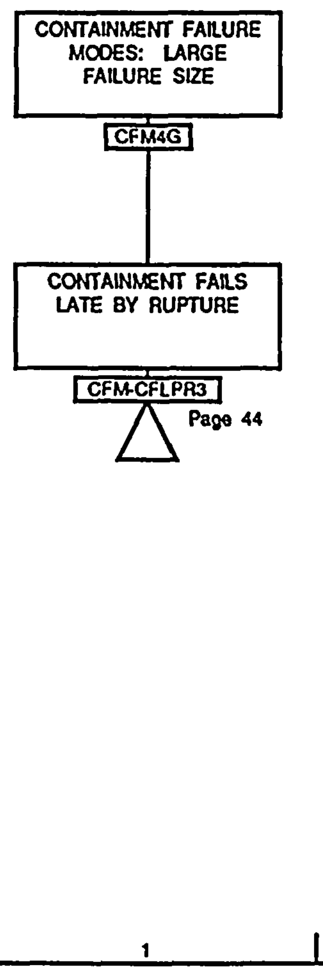

FISSION PRODUCT REMOVAL MECHANISMS FAIL
FPR3G
VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT1
Page 39

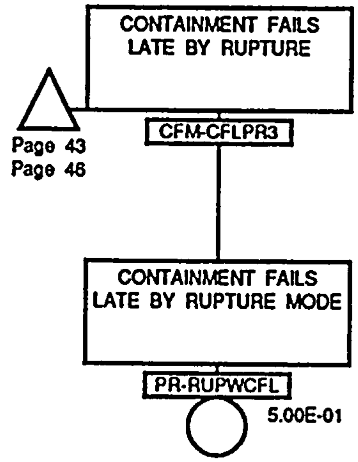

VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT1
Page 38
Page 41
IN-VESSEL FISSION PRODUCT RELEASE UN-MITIGATED
FPR-IN1
LATE FISSION PRODUCT RELEASE UN-MITIGATED
FPR-LATE1
RCS RETENTION DOES NOT OCCUR
QRCS-RET
2
1.75E-01
SPRAYS DON T OPERATE BEFORE OR AFTER VB
NOSPRAY121
RCS/ CONTAINMENT HEATUP CAUSES REVOLATIZATION
PRHEATUP
2
1.75E-01
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB3
Page 40
CONT. SPRAYS DO NOT OPERATE BEFORE VB (dry cavity)
SNOSPRAY
Page 8
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB3
Page 40

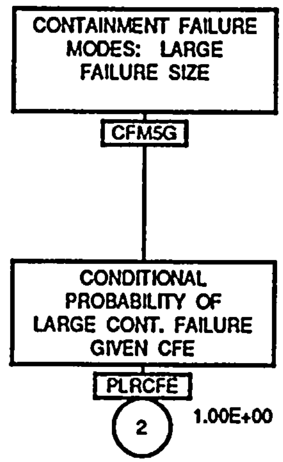

NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB3
Page 39
Page 41
Page 39
CONT. SPRAYS DO NOT OPERATE AFTER VB
SNOSPRAY2
7
9.99E-01
NO SPRAYS GIVEN CFE
NOSPR-CFE
Page 19
1
2

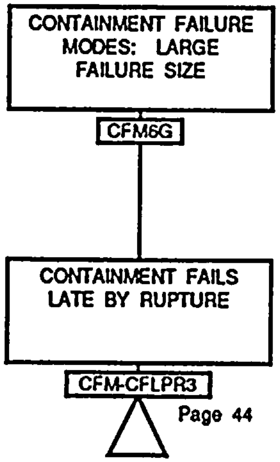

FISSION PRODUCT REMOVAL MECHANISMS FAIL
FPR4G
VOLATILE FISSION PROD. RELEASE UNMITIGATED
VOL-UNMIT1
Page 39
EX-VESSEL FISSION PRODUCT RELEASE UN-MITIGATED
FPR-EX1
NO OVERLYING POOL
PRNO-POOL
2
1.00E+00
NO SPRAYS AFTER VESSEL BREACH
NO-SPR-VB3
Page 40
1
2

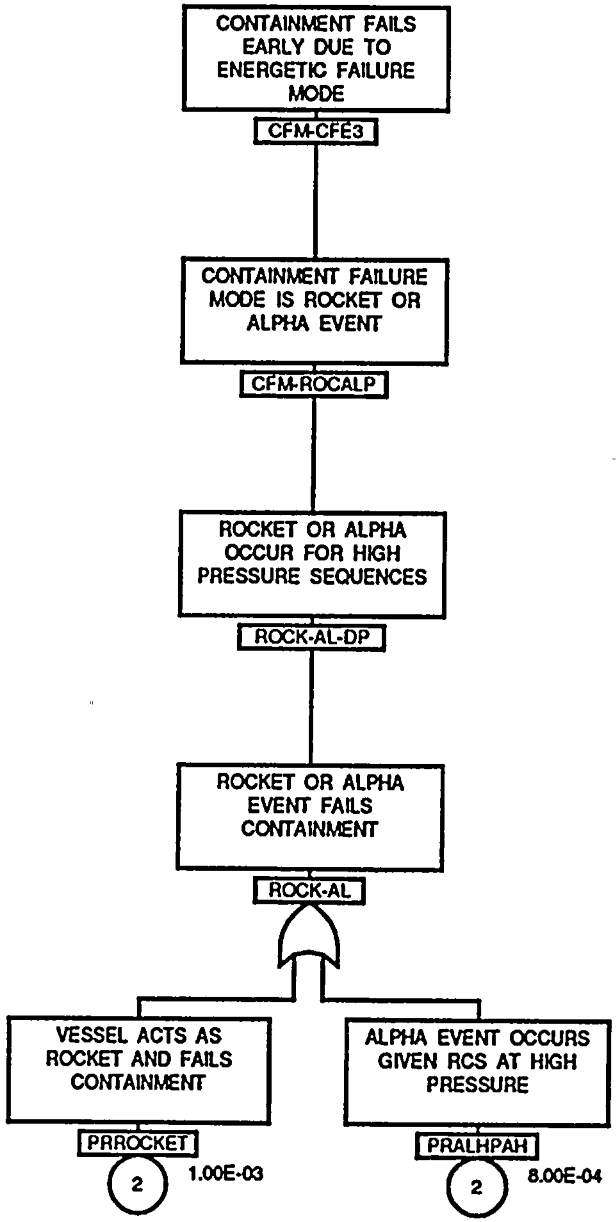

CONTAINMENT FAILURE MODES: LARGE FAILURE SIZE
CFM3G
CONDITIONAL PROBABILITY OF LARGE CONT. FAILURE GIVEN CFE
PLRCFE
2
1.00E+00
1


CONTAINMENT FAILURE MODES: LARGE FAILURE SIZE
CFM4G
CONTAINMENT FAILS LATE BY RUPTURE
CFM-CFLPR3
Page 44
1

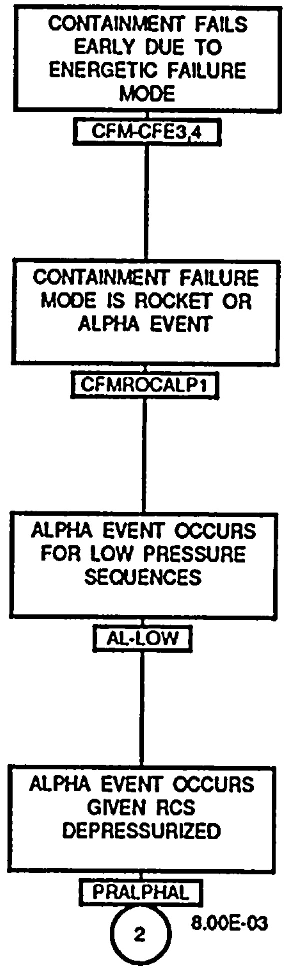

CONTAINMENT FAILS
LATE BY RUPTURE
Page 43
Page 46
CFM-CFLPR3
CONTAINMENT FAILS
LATE BY RUPTURE MODE
PR-RUPWCFL
5.00E-01


CONTAINMENT FAILURE MODES: LARGE FAILURE SIZE
CFMSG
CONDITIONAL PROBABILITY OF LARGE CONT. FAILURE GIVEN CFE
PLRCFE
2
1.00E+00
1

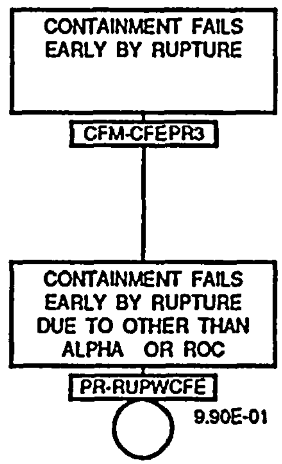

CONTAINMENT FAILURE MODES: LARGE FAILURE SIZE
CFM6G
CONTAINMENT FAILS LATE BY RUPTURE
CFM-CFLPR3
Page 44


CONTAINMENT FAILS EARLY DUE TO ENERGETIC FAILURE MODE
CFM-CFE3
CONTAINMENT FAILURE MODE IS ROCKET OR ALPHA EVENT
CFM-ROCALP
ROCKET OR ALPHA OCCUR FOR HIGH PRESSURE SEQUENCES
ROCK-AL-DP
ROCKET OR ALPHA EVENT FAILS CONTAINMENT
ROCK-AL
VESSEL ACTS AS ROCKET AND FAILS CONTAINMENT
PRROCKET
2
1.00E-03
ALPHA EVENT OCCURS GIVEN RCS AT HIGH PRESSURE
PRALHPAH
2
8.00E-04

1 2


CONTAINMENT FAILS EARLY DUE TO ENERGETIC FAILURE MODE
CFM-CFE3,4
CONTAINMENT FAILURE MODE IS ROCKET OR ALPHA EVENT
CFMROCALP1
ALPHA EVENT OCCURS FOR LOW PRESSURE SEQUENCES
AL-LOW
ALPHA EVENT OCCURS GIVEN RCS DEPRESSURIZED
PRALPHAL
2
8.00E-03

1


CONTAINMENT FAILS EARLY BY RUPTURE
CFM-CFEPR3
CONTAINMENT FAILS EARLY BY RUPTURE DUE TO OTHER THAN ALPHA OR ROC
PR-RUPWCFE
9.90E-01

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
A-OR-CPL	6	2	DEBFORM1	17	3	GDC2	13	2	NNOSPRAY1	7	2
AL-LOW	48	1	DEBFORM1G	17	2	GDC2	15	2	NNOSPRAY2	30	1
BMMT1	30	3	DEBFORM2	21	3	GDC2	24	3	NO-OVRPRS	30	1
CDHR-PASS	28	3	DEBFORM2	22	3	GDC23	22	5	NO-SPR-FAN	28	2
CFE1-HP	7	2	DEBFORM2	23	1	GDC3	22	4	NO-SPR-VB	35	3
CFE1-LP	6	2	DEBFORM2G	21	2	GPRCDB-HP	22	2	NO-SPR-VB	35	4
CFE1G	6	2	DEBFORM4G	23	2	HB2-FAIL	25	1	NO-SPR-VB	36	2
CFE2G	7	3	DEBFORMG	24	2	HB2-FAIL	26	2	NO-SPR-VB	37	3
CFL1G	25	2	DHR	25	3	HB2-FAIL	29	1	NO-SPR-VB1	10	2
CFL2G	29	2	DHR-ACT	28	2	HB2-FAIL	31	1	NO-SPR-VB1	11	2
CFL3G	30	2	DHR1	32	3	HB2-FAIL	32	1	NO-SPR-VB1	24	1
CFL4G	32	2	DHR2	29	3	HB2-FAIL1	33	2	NO-SPR-VB1	24	3
CFLCCI	31	2	DHR3	31	6	HB3-FAIL	31	4	NO-SPR-VB2	18	2
CFM-CFE3	47	2	DP-H	1	4	HOP-DP	1	4	NO-SPR-VB3	39	3
CFM-CFE3,4	48	1	DP-MED	1	2	HR-INCONT	25	3	NO-SPR-VB3	39	4
CFM-CFEPR3	49	1	DP-MED	1	4	HR-INCONT	28	2	NO-SPR-VB3	40	2
CFM-CFLPR3	43	1	DP-MED	2	2	HR-INCONT	29	3	NO-SPR-VB3	41	3
CFM-CFLPR3	44	1	DP1	1	3	HR-INCONT	31	6	NOPIG	4	1
CFM-CFLPR3	46	1	DPG	1	4	HR-INCONT	32	3	NOSPR-CFE	18	3
CFM-ROCALP	47	2	DPH	1	4	LPISPRAY	17	1	NOSPR-CFE	19	1
CFM3G	42	1	DPH1	1	2	LPISPRAY	18	2	NOSPR-CFE	40	2
CFM4G	43	1	DPH2	1	3	LPISPRAY	20	1	NOSPRAY12	35	2
CFM5G	45	1	DPM	1	2	LPISPRAY	23	3	NOSPRAY121	39	2
CFM6G	46	1	DUMM1	1	5	LPISPRAY1	9	1	NOSPRNCFE	11	2
CFMROCALP1	48	1	FPR-EX	37	2	LPISPRAY1	10	2	NOSPRNCFE	12	1
CFWOSPR	7	4	FPR-EX1	41	2	LPISPRAY1	16	1	NOSPRNCFE	36	2
CFWSPR	7	2	FPR-IN	35	2	LPISPRAY1	21	1	NPREVSE3	22	5
CHGAVB	28	1	FPR-IN1	39	2	NACSPREC	31	1	NSWCAV	8	1
CP-PULTH	7	3	FPR-LATE	35	4	NACSPRECL	31	2	OVR-PRESS	32	2
DC1G	9	2	FPR-LATE1	39	4	NC-HB3	31	2	OVR-PRESS1	25	2
DC2G	17	2	FPR1G	34	1	NDP-MED	1	4	OVR-PRESS2	29	2
DC3G	21	2	FPR2G	37	2	NEVSE	15	2	OVR-PRESS3	30	1
DC4G	23	2	FPR3G	38	1	NLPISPRAY	17	2	OVR-PRESS3	30	1
DC5G	24	2	FPR4G	41	2	NLPISPRAY	20	1	OVR-PRESS3	31	3
DEBFORM	24	2	GDC1	13	1	NLPISPRAY	23	2	PLRCFE	42	1
DEBFORM0G	9	2	GDC1	14	2	NLPISPRAY1	9	3	PLRCFE	45	1
DEBFORM1	9	2	GDC1	24	2	NLPISPRAY1	16	1	PR-HT-TRAN	5	1
DEBFORM1	13	2	GDC13	22	3	NLPISPRAY1	21	2	PR-RUPWCFE	49	1

Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone	Gate/Event Name	Page	Zone
PR-RUPWCFL	44	1	PRPR3	31	6	SHP-SISG2	4	2	WPOW	31	6
PRALHPAH	7	4	PRPR4	31	4	SHPSIS1	3	2	WSPR1	30	2
PRALHPAH	47	2	PRPRHB1	33	2	SLP-SIS1	4	2			
PRALPHAL	6	2	PRPRHB2	26	2	SLP-SISG	3	2			
PRALPHAL	48	1	PRROCKET	7	1	SLPSIS1	3	1			
PRCDB-HP	22	1	PRROCKET	47	1	SNOLPI	10	1			
PRCDB-LPNS	15	1	PRSEALOK	2	2	SNOLPICFE	18	1			
PRCDB-LPSE	14	2	PRSGOK	2	2	SNOLPIG	10	1			
PRCDBLPNS3	22	6	PRSTM-OCC	25	2	SNOLPIGCFE	18	1			
PRCDBLPSE3	22	4	PRSTM-OCC	29	2	SNOSPRAY	7	2			
PRCI	6	1	PRSTM-OCC	31	3	SNOSPRAY	7	4			
PRCI	7	3	PRSTM-OCC	32	2	SNOSPRAY	8	2			
PRCOOLDBIV	5	2	PRWCP-PULT	7	1	SNOSPRAY	35	2			
PRCP-PULTL	6	3	QDHR-CLRS	27	1	SNOSPRAY	39	2			
PRCSRVS	2	1	QHP	1	4	SNOSPRAY1	8	2			
PRDCP-PULT	7	3	QHPMEHP	22	2	SNOSPRAY2	11	1			
PRDEST-CFE	19	1	QMP	1	1	SNOSPRAY2	18	2			
PRDESTNCFE	12	1	QNOFAN	28	3	SNOSPRAY2	28	2			
PREVSE	14	1	QRCS-RET	35	1	SNOSPRAY2	30	1			
PREVSE	15	2	QRCS-RET	39	1	SNOSPRAY2	30	3			
PREVSE3	22	3	R-A-CPH	7	2	SNOSPRAY2	36	1			
PREVSE3	22	5	REC1G	3	2	SNOSPRAY2	40	1			
PRHB3	31	6	REC2G	4	2	SPR-VB1	24	1			
PRHB4	31	3	RHR-IN1	25	2	STM-FAIL	32	2			
PRHEATUP	35	3	RHR-IN1	27	1	STM-FAIL1	25	2			
PRHEATUP	39	3	RHR-IN1	31	5	STM-FAIL2	29	2			
PRHLSLOK	1	3	ROCK-AL	47	2	STM-FAIL3	31	4			
PRHLSLOK1	1	2	ROCK-AL-DP	47	2	SWCAVITY	8	1			
PRHLSLOK2	1	5	SACPOWER	3	3	TBMMT1	30	2			
PRIMPINGE	6	3	SACPOWER	4	3	VFG	5	2			
PRIMPINGE	7	4	SACSPREC	26	1	VOL-UNMIT	34	1			
PRMT1	30	2	SACSPREC	31	1	VOL-UNMIT	35	2			
PRMT2	30	4	SACSPREC1	33	1	VOL-UNMIT	37	1			
PRNCG-FAIL	31	1	SACSPRECL	31	2	VOL-UNMIT1	38	1			
PRNO-POOL	37	2	SACSPRECL	31	5	VOL-UNMIT1	39	2			
PRNO-POOL	41	2	SALT-SISH	4	2	VOL-UNMIT1	41	1			
PRPR1	33	2	SALT-SISL	3	2	WOPOW	31	3			
PRPR2	26	2	SHP-SIS1	4	1	WOSPR1	30	4			

APPENDIX F

ST. LUCIE UNITS 1 & 2

SEVERE ACCIDENT PROGRESSION ANALYSIS

F.1 PURPOSE

This appendix documents the severe accident progression analysis and accident simulation code calculations conducted for the St. Lucie Level 2 PRA. This analysis focuses on the RCS behavior and containment response for identified dominant accident scenarios. The accident analysis consists of several deterministic calculations using the MAAP accident simulation code. The MAAP code was originally developed by IDCOR which subsequently was enhanced and is currently maintained by EPRI. The enhancements include modeling improvements in response to the IDCOR/NRC technical resolution process and EPRI-sponsored design review comments. The code version used in these calculations is MAAP 3.0B Rev 19 [Ref. F-1]. It is reviewed by NRC, in anticipation of its application by utilities for IPE back-end analysis. The objective of the deterministic analysis documented in this appendix is to provide the plant-specific RCS and containment response to severe accident phenomena in support of the CET analysis for the St. Lucie PRA study.

F.2 METHODOLOGY

The analytical approach used in this study was to select accident scenarios that would characterize the dominant plant damage states for PSL PRA (Level 1 accident sequence analysis report). The accident sequence is defined in MAAP with a set of intervention codes or actions that either turn front-line or containment safeguards systems on or off automatically; or as would be defined from the Level 1 analysis, systems are forced off to reflect failure to operate. A "baseline" scenario is selected to represent the most likely conditions within the sequences cutsets binned into the plant damage states (PDSs). The PDSs analyzed in this appendix include PDS IB, IIA, IIB, IIE, IIF, IIIB, IIIH, IVD, VB, VIA, VIB, etc. (Appendix D provides a description of the range of possible PDSs.) A brief overview of the important PDS characteristics is provided in this appendix to provide a better understanding of the results of the accident simulation code.

The baseline scenarios also include the current "understanding" of the phenomenological issues that are important in severe accident risk assessments as implemented in MAAP. Uncertainties delineated in GL 88-20 and evaluated in the NUREG-1150 reflecting NRC's positions on key issues are addressed in the sensitivity analyses. The reference accident scenario is analyzed in some detail first, and is then used as the basis against which variations in the input data could be made. Next, variations in the system performance and phenomenological uncertainties in our understanding of degraded core accident progression were also analyzed (such as forcing DCH for certain high pressure scenario). The baseline scenario is used to quantify the most likely progression within the CETs, and the sensitivity of results in these cases were used to provide insights in the CET qualification of the various progression paths in the CET. The uncertain issues related to the modeling variables of physical processes strongly influence severe accident progression. Examples of these include the following:

- In-vessel core melt progression and relocation;
- Ex-vessel corium water interactions and core-concrete interactions;
- Fission product release from the fuel and transport within the containment system;

- Containment performance (i.e., failure modes, failure locations or failure size), and

- Active operator actions involving systems operations and actuation (e.g., steam generator depressurization).

Some combinations of model and input data parameter changes were also defined to investigate possible synergistic effects.

MAAP 3.0B includes a set of model parameters that can be varied within defined ranges about the best estimate (or default) values to address the phenomenological issues. These are provided as either scalar multipliers of calculated functions or logic switches which allow the user to select another basis for the model calculation. The ranges are judged by the code developers (MAAP Users Manual) as the appropriate values within the constraints of the phenomenological modeling in MAAP.

F.3 ACCIDENT SIGNATURES

The assessment of containment response and source term predictions of MAAP 3.0B vary according to the importance of the various phases of the severe accident progression. The variations in the results, as represented by the figures of merit and accident signatures, are influenced by key elements of the accident progression. The elements of the severely degraded core accident sequence resulting in challenges to containment integrity or a release of fission products to the environment include:

- Core damage and relocation in the reactor coolant system:
  - Core uncovery up to initial core geometry changes;
  - Core melting and relocation to the lower plenum; and
  - Core material thermal attack of the reactor pressure vessel.
- Core materials interaction in the containment:
  - Exit and deposition from the pressure vessel;
  - Initial interactions with the containment atmosphere, concrete, and/or water; and
  - Long-term interactions with concrete and/or water.

- Containment Performance:
  - Containment challenge and response; and
  - Containment leakage area and location.
- Radionuclide Release and Transport:
  - In-vessel release, transport, retention and revaporization; and
  - In-containment release, transport, retention and revaporization.

The phenomenological issues that have significant influence in the evaluation of St. Lucie's containment performance (and source term) can be associated with the uncertainties in the modeling of the different phases of severe accident progression and radionuclide release. The impact of a physical process or phenomenon and modeling of an event on the overall risk in some of these phases is more significant than others. For example, containment performance is recognized as a major factor in the determination of the timing and magnitude of release of the fission product species. If containment integrity is maintained, fission products released from the fuel would be contained and the consequences to the environment would be insignificant. Early loss of containment integrity would generally lead to a potentially significant release of fission products to the environment. Containment safety systems actuation such as containment sprays, can improve upon the performance of the containment in mitigating the pressure loads and the fission product release. These variations in the sequence definition were investigated by modeling operator actions to obtain the required systems performance.

It should be stressed, however, that this is not a rigorous deterministic analysis of all the possible combinations of systems and phenomena interactions. It only reflects the changes in the magnitudes in the context of the model parameter variations and systems status over their possible conditions within the PDS definition. This does, of course, provide a strong insight into the magnitudes of the uncertainties, which are then used as measures of the likelihoods of the CET progression paths.

The key elements described above are similar to the eight technical areas of uncertainty identified in NUREG-0956 [Ref. F-3] affecting current severe accident progression analysis (Table F-1). Outstanding technical issues were also defined in IDCOR/NRC technical exchange meetings which resulted in a list formulated by the NRC staff comprising of eighteen different items, many of which involve the same or similar phenomena identified above. The IDCOR/85 Program addressed major technical issues identified by NRC and worked towards resolving these issues. Not all of the significant technical issues were addressed in this analysis. Focus was principally on the areas involving core melt progression within the vessel, debris dispersion and hydrogen generation, as these determine the containment loads that could challenge containment integrity early, and core-concrete interactions. In particular, Direct Containment Heating (DCH), hydrogen generation/combustion and debris coolability are considered in the present analysis.

F.4 DESCRIPTION OF ACCIDENT SCENARIOS

This section discusses the representative PDS accident scenarios and the results obtained using the best estimate input values included in the MAAP 3.0B parameter files. A brief description of the reference cases is provided, including the MAAP modeling assumptions relative to the reactor coolant system (RCS) and the containment system. The results of the baseline calculations are summarized in terms of the key figures of merit. These are further discussed in the next section, with respect to the thermal-hydraulic response of the plant and potential containment loads. The fission product release and transport within the RCS and the containment system are provided, but not discussed in any detail as these are not used in the CET quantification.

Reference F-2 recommends a set of sensitivity analyses for MAAP-based back-end study and was used as described below. A discussion of the MAAP sensitivity analyses is provided later in this appendix.

The base case MAAP runs are made with the following assumptions:

1.	EQ Limit (for ECC and CS)	44 psig 264°F ;
2.	Containment Failure Area (ACFPR)	0.1 m^2; for Bypass Sequence use 2E-3 m^2
3.	Vessel Failure Time (TTRX)	30 min.;
4.	Cavity Flooding Coefficient for debris dispersal (FENTR)	1.0;
5.	Debris Cooling CHF Constant (FCHF)	0.1;
6.	No Blockage (FCRBCK)	0.0; modeling option; to maximize H_2 generation
7.	2 Sided Oxidation (FAOX)	2.0; modeling option; to maximize H_2 generation
8.	Melt Ejected from Compartment C to Compartment B (FCMDA)	0; modeling option;
9.	Debris Fraction in DCH (FCMDCH)	0.03;

Each of the above assumptions is briefly described below:

1. EQ Limit

   A conservative set of values of 44 psig and 264°F was used to terminate the ECC and CS. In general, the temperature limit appears to be reached first. It is also noted that the temperature in the lower compartment (compartment B) is higher than that in the upper compartment (compartment A). Although not expected to be affected by the EQ limit, CS is conservatively made unavailable when EQ limits are reached. The EQ limits are considered reached when the upper compartment pressure and/or the lower compartment temperature exceed the specified limits for a continuous period of 10 minutes.

2. Containment Failure Area (ACFPR)

   For sequences in which containment fails due to rapid pressurization, e.g. Hydrogen burns or direct containment heating, a large failure area of order 0.1 square meter appears reasonable. For late containment failures, where steam generation or lack of decay heat removal would cause slow pressurization, it is expected that smaller leak area is more likely.

Because of the known uncertainty in MAAP models and to bound the fission product release estimates it is conservatively assumed that for all containment failures, the failure area is 0.1 square meter.

3. <u>Vessel Failure Time (TTRX)</u>
Reference F-2 recommends using 1 minute as a value for the time required to fail the RPV lower head for plants with lower head penetrations and 30 minutes for plants without such penetrations. Since St. Lucie does not have lower head in-core instrument penetrations, vessel failure time of 30 minutes is used.

4. <u>Cavity Flooding Coefficient for Debris Dispersal (FENTR)</u>
This parameter is a multiplier on FFLOOD (Kutateladze number used in flooding calculations) used to adjust for effect of cavity geometry on debris dispersal velocity threshold. Reference F-2 recommends that a value of 1 be used for Zion-like plants, with smooth, sloping instrument tunnels. For St. Lucie Units 1 and 2, it is expected that the debris dispersal threshold velocity is about the same as that for Zion, if not higher (because of more tortuous path to disperse into lower or upper compartment). Thus a value of 1.0 for FENTR is used for baseline runs.

5. <u>Debris Cooling CHF Constant (FCHF)</u>
This parameter is the coefficient (Kutateladze number) used in critical heat flux formula for calculating debris coolability. A default value of 0.1 is used. Considering the high level of uncertainty associated with debris coolability, and to conform with the requirements of the Generic Letter, a sensitivity run with FCHF=0.02 is made (Case TIIIH12).

6. <u>No Blockage (FCRBLK)</u>
This parameter is set to 0 (deactivate the blockage model) to de-emphasize the effects of core degradation on clad oxidation as recommended by Reference F-2.

7. <u>2-Sided Oxidation (FAOX)</u>
This parameter is a multiplier on clad surface area used for oxidation calculation. A conservative assumption leading to higher hydrogen generation based on two sided oxidation is used to maximize the containment load.

8. <u>Melt Ejected from Compartment C to Compartment B (FCMDA)</u>
Based on the cavity and the containment configuration, the debris dispersal is modeled from cavity to lower compartment. Sensitivity runs were made to investigate the effects of debris dispersal to upper compartment.

9. <u>Debris Fraction in DCH (FCMDCH)</u>
This parameter is the fraction of debris which is transported from the reactor cavity as finely fragmented droplets rather than as a film or wave. Reference F-2 recommends that a low value of the debris fragmentation fraction, e.g. 0.03 or so be used as a best-estimate for most plants in which dispersal mainly occurs to the lower compartment.

F.4.1 High Pressure Plant Damage States

The accident simulation calculations described in this section were performed for selected accident scenarios for scoping analysis of potentially risk-dominant high RCS pressure accident sequences. PRAs have historically identified accidents involving transient events in which the pressure in the vessel remains high (such as LOSP sequences) as potentially risk-dominant contributors for which the containment may be challenged early. The St. Lucie PRA has likewise identified accident scenarios in which the reactor pressure is high, much like LOSP or transient-initiated accident sequences. Variations of accident scenarios that were evaluated for St. Lucie include station blackout sequences with and without the attendant seal LOCA that results from loss of seal cooling.

The high-pressure scenario calculations provide insights for the CET analysis by meeting the following objectives:

- Verify accident sequence progression of a class of accident sequences that lead to core damage in which the RCS is not breached (transient events). These generally involve loss of heat removal capability through the steam generators precluding coolant injection.

- Support determination of PDSs falling into core damage bins I, II, and primarily III.

- Determine containment pressure loads for the baseline scenarios of PDSs and define accident conditions that would bound containment challenges (i.e., no safeguards).

- Evaluate RCS conditions and phenomena that affect containment loads that may challenge containment early (i.e., HPME loads).

- Evaluate long term containment response.

Within these objectives, it is necessary to investigate the containment performance under various safeguards conditions for the core damage bins identified above.

The conditions that were examined include typical accident signatures, i.e., containment pressure and temperatures, major event timings, etc. The RCS conditions monitored include the pressure in the reactor and the surface temperatures of the hot leg or the surge line prior to vessel breach. In addition, the dispersal of the molten core material at vessel breach and its effect on containment pressure under various conditions were examined. The distribution of hydrogen within the containment building, and potential for accumulation of localized high concentrations in sub-compartments were also examined.

The discussion that follows describe the accident sequences selected for evaluation. It also provides an overview of the sequence progression and variations in the sequence definition of the transient scenarios addressed in this appendix. The transient analysis is, for the most part, focused on high RCS pressure sequences, such as a station blackout scenario, in which all systems that require power for operation are rendered unavailable. Variation of this baseline scenario include scenario in which the RCP seals are postulated to fail. The auxiliary feedwater system is assumed to operate for two hours. Additional variations of these scenarios were also defined to simulate uncertainties in the phenomenology, such as in-vessel clad oxidation and high pressure melt ejection (HPME), in order to determine challenges to containment integrity early in the scenario.

F.4.1.1 Plant Damage State IIIB

a) Loss of Feedwater (LIIIB0)

This base case accident scenario simulates loss of main and auxiliary feedwater. With the loss of RCS heat sink, the core remains at high pressure until the vessel breach. The containment cooling is by 1 CS and 2 ECCs. The simulation is terminated at 50 hours after the initiation of the transient.

b) Loss of Grid (LIIIB1)

This accident scenario simulates loss of main and auxiliary feedwater. The reactor coolant pumps are tripped at 3.0 seconds into the transient. With the loss of secondary heat sink, the core remains at high pressure until the vessel breach. The containment cooling is by 1 CS and 1 ECCs. The simulation is terminated at 50 hours after the initiation of the transient.

F.4.1.2 Plant Damage State IIIE

This accident scenario (CIIIE1) simulates loss of CCW. AFW is not available, leading to the loss of secondary heat removal. The RCPs are tripped at 5 minutes and ADVs are opened to reduce the steam generator pressure to 400 psia. ECCs are not available and 1 CS is available only in the injection mode. The simulation is terminated at 10 hours after the containment failure time.

F.4.1.3 Plant Damage State IIIF

This scenario CIIIF0 is similar to the above scenario CIIIE1 with respect to the primary and secondary system performance. However, 1 containment spray pump is available in both the injection and the recirculation modes. The simulation is terminated at 10 hours after the containment failure time.

F.4.1.4 Plant Damage State IIIH

a) Station Blackout - No Seal LOCA (TIIIH0)

This accident scenario simulates loss of both off and on-site power. The reactor is shut down, core cooling is not successful and the RCS heat sink (i.e., heat removal through the steam generators) is not maintained. The auxiliary feedwater is postulated to be lost after two hours. ECCs and containment sprays are not available. The simulation time is 50 hours.

b) Station Blackout - Seal LOCA (TIIIH1)
This scenario is similar to the station blackout without Seal LOCA scenario, except that a Seal LOCA of 1/4 in. size for each RCP is assumed to occur at 90 minutes into the transient. With no secondary heat removal subsequent to the loss of AFW, the reactor vessel fails at high pressure. The simulation is terminated at 50 hours into the transient.

F.4.1.5 Plant Damage State IVD

This accident scenario (LIVD0) is similar to the LIIIB0 scenario initiated by the loss of main feedwater. The auxiliary feedwater, however, is available till the CST gets depleted. Containment sprays are made unavailable and the containment cooling is provided by 2 ECCs. The scenario is run for 50 hours.

F.4.2 Moderate Pressure Plant Damage States

The analysis documented in this section is focused on small-small LOCA initiated events. Variations of small-small LOCA accident scenarios that were evaluated for St. Lucie include sequences with and without the safety injection, and with various combinations of the containment safeguards systems (containment sprays and fan coolers).

The objectives of the small-small LOCA analysis include the following:

- Verify accident sequence progression of a class of accident sequences that lead to core damage in which the RCS is breached through a small break (seal LOCAs greater than 1/4" but less than 1.5" are also considered small-small LOCAs).

- Support determination of PDSs falling into core damage bins I and II.

- Determine containment pressure loads for the baseline scenarios of PDSs and define accident conditions that would bound containment challenges (i.e., no safeguards).

- Evaluate RCS conditions and phenomena that affect containment loads that may challenge containment early (i.e., HPME loads).

- Evaluate long term containment response.

Within these objectives, it is necessary to investigate the containment performance under various safeguards conditions for the core damage bins identified above.

The discussion that follows describes the accident sequences selected for evaluation. It also provides an overview of the sequence progression and variations. The analysis described in this subsection is, for the most part, focused on a small-small LOCA scenario in which make-up water may or may not be available with containment sprays competing with the injection flow. Variations of this baseline scenario include scenarios in which the HPSI pumps are assumed failed and the containment sprays are inoperable in the recirculation mode.

F.4.2.1 Plant Damage State IB

This accident sequence (LCIB0) is initiated by a 2" cold leg break LOCA. The auxiliary feedwater is available until the CST is depleted. The high pressure and the low pressure injection systems are assumed to fail, leading to the early core uncovery. 1 ECC and 1 CS are available from the start of the transient, which is run for 50 hours.

F.4.2.2 Plant Damage State IIA

This accident scenario (LCIIC2) simulates a 3" cold leg break LOCA. Safety injection (1 HPSI and 1 LPSI) is available in the injection mode only until the RWT is depleted. 1 CS (injection mode only) and 1 ECC are available until the EQ limits are reached. The auxiliary feedwater is available similar to the LCIB0 scenario. The simulation is terminated at 10 hours after the containment failure.

F.4.2.3 Plant Damage State IIB

a) 3" Break LOCA (LCIIB0)
Case (a) of this accident PDS (LCIIB0) is initiated by a 3" cold leg break LOCA and is similar to the LCIIC2 scenario of PDS IIA, except that the containment spray is available also in the recirculation mode. The run is terminated at 50 hours.

b) 2" Break LOCA (LCIIB1)
Case (b) of this accident PDS (LCIIB1) is initiated by a 2" cold leg break LOCA and is run identical to the case (a) scenario.

F.4.2.4 Plant Damage State IIE

This accident scenario (LCIIE0) simulates a 3" cold leg break LOCA similar to the scenario LCIIC2. However, ECCs are not available in this scenario, whereas 1 containment spray pump is available in the injection mode only. The auxiliary feedwater is available and the safety injection (1 HPSI and 1 LPSI) is made available in the injection mode until the RWT is depleted. The end of simulation is 10 hours from the containment failure time.

F.4.2.5 Plant Damage State IIF

This PDS scenario LCIIF1 is similar to the scenario LCIIE0 with respect to the accident initiation and the availability of the injection systems. However, in this accident scenario the containment sprays (1 CS) are available in both the injection and the recirculation modes. The simulation time for this scenario is 50 hours.

F.4.2.6 Steam Generator Tube Rupture

a) SGTR (SGTR01)
This scenario is initiated by a single steam generator tube rupture (double-ended) of area equal to 4.335E-4 M^2. The high pressure injection is made unavailable. The MSIVs are closed at 10 minutes and the intact steam generator is depressurized to 400 Psia. Auxiliary feedwater is available to the intact steam generator. 1 CS and 2 ECCs remove the containment heat after the vessel failure. The simulation is terminated at 50 hours.

b) SGTR (SGTR02)
This accident scenario is similar to the above scenario SGTRC1, except that the auxiliary feedwater is assumed to fail.

F.4.3 Low Pressure Plant Damage States

The plant damage states with RCS pressure at 200 psi or lower (i.e., small, medium, and large LOCAs) fall into this category. Only the large LOCA initiated events are evaluated using MAAP. These PDS runs are described below.

F.4.3.1 Plant Damage State VB

This accident scenario (LCVB0) simulates a 18 in. break (cold leg) LOCA. The auxiliary feedwater is made unavailable, although its effect on heat removal from the RCS is not significant. The reactor shuts down on low RCS pressure and neither the high nor the low pressure injection is available. One containment spray pump and 2 fan coolers are available to remove the heat from containment. Three safety injection tanks provide water to reflood the core after the initial blowdown. The simulation for this case is terminated at 50 hours from the initiation of the transient.

F.4.3.2 Plant Damage State VIA

In this accident scenario (LCVIA0), which is initiated by a 18 in. LOCA, safety injection (1 HPSI and 1 LPSI) is available until the RWT gets depleted. Two ECCs are available along with 1 CS (injection mode only) for containment heat removal. The simulation is run for 50 hours, similar to the LCVC1 scenario.

F.4.3.3 Plant Damage State VIB

This accident scenario (LCVIB0) is similar to the scenario LCVIA0, except that the containment spray is available in both the injection and the recirculation modes.

F.4.3.4 V Sequence

This accident scenario simulates a 10.25" break interfacing system LOCA into the auxiliary building (RHR hot leg shutdown cooling line). Low pressure injection is lost, whereas 1 HPSI is available to inject water from the RWT. 1 CS and 2 ECCs are available. This scenario is run for 24 hours.

F.5 SEVERE ACCIDENT PROGRESSION RESULTS

Selected results of the calculations are provided as key figures of merit in evaluating the severe accident progression for the CET analysis. The figures of merit include:

- Timings of major events, such as time of core uncovery, time to melt the core, time to vessel breach, and containment failure time.
- Thermal-hydraulic conditions such as pressure and temperature of the gases and structures in the RCS or the containment.
- Potential challenges to containment integrity as measured in terms of the HPME loads at vessel breach, hydrogen burning, long term pressurization from steaming and non-condensible gas generation, or concrete penetration leading to basemat failure.
- Fission product releases to the environment, in the event that the containment fails. In addition, the distribution of the fission product inventories within the containment system (or in certain runs the adjacent auxiliary buildings) may also be used to determine habitability relative to the feasibility of operator intervention. In the current scope of the Level 2 analysis, operator intervention is not considered, therefore, the fission product distribution is not developed in this work package.

The results for PDS baseline scenarios are described in this section, while the results for the variations examined are described in the next section.

F.5.1 PDS IB Baseline Scenario

The PDS IB, dominated by small-small LOCA scenarios, is simulated by a 2" break in the cold leg (LCIB0). The reactor scrams and the MSIVs are closed just after 1 minute. The decay heat is removed by the break energy and the secondary side heat transfer. With the safety injection assumed failed, the core uncovers at ~0.9 hours. The mass and energy flows out the break increase the containment pressure, actuating the containment sprays at 0.76 hours. The RWT inventory is depleted at 2.16 hours and the vessel fails at 3.33 hours. Most of the corium relocates to the lower compartment as it is ejected out of the vessel. 1 ECC and 1 CS are effective in removing the containment heat. The containment does not fail in this simulation which is run for 50 hours. The results of the key figures of merit are presented in Tables F-3 and F-12.

F.5.2 PDS IIA Baseline Scenarios

Case LCIIC2 of PDS IIA begins with a Loss of Coolant Accident caused by a 3" break in the cold leg. The safety injection systems (1 HPSI, 1 LPSI) are working, but are not available in the recirculation mode. Reactor coolant inventory is lost through the cold leg break and upon recirculation failure, is boiled off. The containment spray is actuated early in the transient at approximately 14 minutes and is lost at 1.5 hours when the RWT gets depleted. The actuation of CS has a significant effect on the accident progression through depletion of the RWT water.

The vessel fails at 8.38 hours with the RCS pressure at about 500 psia. At vessel failure, the molten debris is ejected into the cavity, some of which gets relocated to the lower plenum. With the ECC turned off on EQ limits (8.23 hours) and CS lost on recirculation failure, the containment pressure increases gradually due to the steaming of water in the cavity and the lower compartment. The containment eventually fails at 44.3 hours. The results are summarized in Tables F-3 and F-13.

F.5.3 PDS IIB Baseline Scenarios

a) 3" Break LOCA (LCIIB0)
In the PDS IIB case (a) scenario (LCIIB0), the transient begins with a 3" cold leg break. The reactor trips, and the safety injection signal actuates the high pressure and the low pressure injection pumps at ~45 seconds. The containment pressure reaches the ECC and CS setpoints within the first 15 minutes of the transient.

The scenario is similar to the above PDS IIA scenario, except that the CS is available in the recirculation mode. With the containment spray operating, the EQ limits are not reached in the containment. The vessel fails at 7.94 hours, when the debris in the vessel bottom head gets ejected into the cavity. The gas velocities are sufficiently high to disperse some of the molten core material into the lower compartment. The containment pressure remains well below the failure pressure when the simulation is terminated at 50 hours. The key timings and the figures of merit are in Tables F-4 and F-15.

b) 2" Break LOCA (LCIIB1)
The scenario LCIIB1 is initiated with a 2" cold leg break LOCA with no injection in the recirculation mode, similar to the above LCIIB0 scenario. The RWT depletes at ~2 hours and the vessel fails at 5.53 hours. Subsequent to the vessel failure, the molten debris gets ejected out of the vessel lower head. The RCS pressure at the vessel failure time is about 950 psia. The debris ejection at this pressure leads to most of the corium being relocated to the lower compartment in the containment.

The containment spray and the fan cooler get actuated early in the transient, at approximately 45 minutes. The operation of 1 ECC and 1 CS provides sufficient cooling for the containment to prevent any large pressure increases which would threaten the containment integrity. The key event timings and the selected figures of merit are provided in Tables F-3 and 14, respectively.

F.5.4 PDS IIE Baseline Scenarios

The dominant PDS IIE involves a small-small LOCA, which is simulated in MAAP by a 3" cold leg break (LCIIE0). This scenario is similar to the PDS IIA scenario, except that the ECCs are not available. Tables F-4 and F-16 summarize the key event timings and the figures of merit. With the ECCs unavailable, EQ limits are reached at 3.91 hours. The containment spray is lost when the RWT depletes at 1.49 hours. The vessel fails at 8.41 hours and the containment fails at 32.7 hours, due to the gradual pressurization from the continued steam production in the cavity and the lower compartment.

F.5.5 PDS IIF Baseline Scenarios

The scenario LCIIF1, for the PDS IIF, simulated in MAAP is similar to the above scenario LCIIB0, except that the fan coolers are not available in LCIIF1. The sequence of events is similar. The RWT depletes at 1.49 hours and the injection into the RCS is lost upon the recirculation failure. The vessel fails at 7.64 hours. One containment spray is sufficient to provide the necessary containment cooling, even after the vessel failure. The containment pressure at the end of the simulation is less than 20 psia. The results of key parameters are presented in Tables F-4 and F-17.

F.5.6 PDS IIIB Baseline Scenarios

a) Loss of Feedwater (LIIIB0)
Case (a) of this PDS is simulated in MAAP by a loss of feedwater transient. The reactor trips at 26.2 seconds on low steam generator level, and the steam generators dry out in about 22 minutes. The reactor coolant pumps trip on high void fraction at 51 minutes into the transient. With the loss of secondary heat sink, the RCS boils-off and the core uncovers, initiating fuel heatup and core degradation. The vessel fails at high pressure (2355 psia) at 2.92 hours.

The ECC and CS get actuated within the first 50 minutes and keep the containment pressure low until the vessel failure time. The containment pressure rises after the vessel breach from 18 psia to 34 psia as the hot debris and gases get ejected from the vessel. The fan coolers and the containment spray are effective in controlling the containment pressure and temperature so as not to exceed the EQ limits. The distribution of corium and water in the containment determines the long term pressurization. In LIIIB0 scenario, most of the debris get relocated to the lower compartment. With the ECC and the CS operating, the containment pressure remains well below the failure pressure at the end of the simulation time of 50 hours. The results of this accident analysis are summarized in Tables F-5 and F-19.

b) Loss of Grid (LIIIB1)
This accident scenario is initiated by the loss of a grid, resulting in the loss of main feed at time=0. The auxiliary feedwater is made unavailable and the reactor coolant pumps are tripped at 3 seconds. The progression of events in this accident sequence is similar to that in LIIIB0. Since the reactor and the reactor coolant pumps trip early, the timings of key

events differ. The steam generators dry out at 1.28 hours and the vessel fails at 4.12 hours. The timings of major events and the figures of merit are presented in Tables F-5 and F-18.

F.5.7 PDS IIIE Baseline Scenarios

In this loss of CCW accident scenario (CIIIE1), the reactor coolant pumps are tripped at 5 minutes, leading to the reactor scram. The steam generator ADVs are opened to reduce the secondary side pressure to 400 psia. The main feedwater shuts off at 0.44 hours and, with auxiliary feedwater not available, the steam generators dry out at 1.69 hours. The primary system inventory boils-off and the top of core uncovers at 3.1 hours. The vessel fails at 5.61 hours with the RCS pressure at 2340 psia. The containment spray comes on at 2.87 hours and is lost at 4.25 hours when the RWT depletes. The containment fan coolers are assumed to have failed.

The containment pressure increases from 24 psia to 41 psia just after the vessel breach. The high pressure melt ejection relocates most of the molten corium to the lower compartment. With the containment spray inoperable, the steaming of water in the lower compartment and in the cavity gradually pressurizes the containment, which eventually fails at 26.7 hours. The figures of merit are summarized in Tables F-7 and F-22.

F.5.8 PDS IIIF Baseline Scenarios

This PDS scenario (CIIIF0) is run similar to the case CIIIE1. However, the containment spray is made available also in the recirculation mode. The containment spray (CS), therefore, continues to operate after the RWT depletes. With the CCW unavailable, the containment spray is not effective in removing the containment heat. The containment pressure and temperature continue to rise till the CS is turned off on EQ limits at 12.2 hours. The vessel fails at 5.59 hours. The continued production of steam in the cavity and the lower compartment fails the containment at 27.3 hours. The major timings and the figures of merit are presented in Tables F-7 and F-23.

F.5.9 PDS IIIH Baseline Scenarios

Case (a) scenario TIIIH0 is the most likely accident scenario analyzed for PDS IIIH. It begins with a loss of on and off-site AC power. The main coolant pumps stop, the main steam isolation valve (MSIV) closes and the reactor trips. Reactor coolant inventory is boiled off and steam is lost through the cycling pressurizer relief valves to the quench tank. The active injection systems fail to provide coolant makeup because of the power failure and the passive systems (i.e., safety injection tanks) do not dump because of high RCS pressure. Pump seal LOCA is not assumed to occur. The auxiliary feedwater is available initially, but is lost after two hours into the accident.

The timings of key accident events for the PDS IIIH baseline scenario are summarized in Table F-6. The various figures of merit are summarized in Table F-20. The important parameters that affect the accident progression are:

1. Possible pump seal LOCA, decreasing the pressure in the primary system below the PORV system setpoint;

2. Loss of secondary cooling (AFW failure) resulting in steam generator (SG) dryout and subsequent pressurization of the RCS;

3. Failure of the reactor vessel at high pressure, potentially dispersing the molten debris to the lower compartment. After the initial blowdown, some of the melting core material drops to the wet cavity region;

4. Continued containment pressurization (after vessel breach) due to the subsequent boil-off of the water accumulated in the reactor cavity, and the heating of the atmosphere in the lower compartment by the relocated debris, challenging the containment integrity in the long term prior to concrete attack; and

5. Potential containment failure at a pressure of 95 psig with an assumed large failure area of $0.1m^2$. The large failure area is selected consistent with MARCH calculations for the reference plant reflecting the uncertainties associated with containment failure. (A smaller failure area is considered to be more likely.)

No pump seal LOCA is postulated to occur in scenario TIIIHO. Steam generator secondary cooling is lost after the dryout at 3.80 hours resulting in repressurization of the RCS. The vessel is at a pressure of 2300 psia when the top of the core is uncovered after 4.76 hours into the transient, thus initiating fuel heatup and core degradation. The support plate fails at 7.13 hours, and a few minutes later (user input), the vessel fails in the bottom head. The pressure in the primary system remains elevated during core melt and is about 2350 psia at RPV failure.

The pressure in the containment essentially remains at 14.7 psia during the initial boiloff and increases slightly because of quench tank rupture disc failure (3.61 hours). The containment pressure increases to 39 psia during core melting. When the vessel fails, the pressure increases to 55 psia.

The distribution of the water and corium between the lower compartment and the cavity affects the long term containment pressurization subsequent to vessel breach. The amount of water in contact with the corium is important in the determination of containment failure time following vessel breach. The pressure increases rapidly in the containment as the vessel fails, which is followed by a gradual increase in pressure as the water in the cavity (and the lower compartment) is boiled off. The continued steam production in the cavity causes the containment pressure to increase to 95 Psia when the simulation ended at 50 hours. The cavity configuration of St. Lucie and the flow paths from the containment floor to the cavity assure water accumulation in the cavity. Most of the debris, in PDS TIIIHO scenario, gets relocated to the lower compartment, thus precluding early cavity dryout (concrete attack does not occur according to MAAP modeling assumptions). The cavity dryout, in fact, is not calculated by MAAP to occur.[1]

Case (b) scenario (TIIIH1) for the PDS IIIH is run similar to the above TIIIH0 scenario. However, a pump seal LOCA, equivalent to a 1/4" break per reactor coolant pump, is initiated at 90 minutes into the transient. Subsequent to the loss of AFW at 2 hours, the steam generators dry out at 3.8 hours. The core begins to melt at 5.6 hours leading to the vessel failure at 7.62 hours. The distribution of corium and water in the containment and the containment response is quite similar to that of scenario TIIIH0, with the containment pressure reaching ~95 psia at the end of simulation time of 50 hours.

F.5.10 PDS IVD Baseline Scenario

The baseline scenario (LIVD0) in PDS IVD is initiated by a loss of main feedwater. The auxiliary feedwater (AFW) is available till the condensate storage tank gets depleted at 7.35 hours. Subsequent to the loss of AFW, the steam generators dry out at 8.46 hours. The top of the core uncovers at 9.54 hours, initiating core heatup. The core melt begins at 10.9 hours, leading to the vessel failure at 12.8 hours. The primary system remains at high pressure, until the vessel fails at 2365 psia. The reduction in pressure at vessel breach initiates safety injection from the refueling water tank (RWT). The SITs empty at 12.8 hours and the RWT depletes at 14.0 hours .

The containment cooling in this scenario is provided by 2 fan coolers (ECC). The ECCs actuate at 9.16 hours and turn off on EQ limit about 10 minutes after the vessel breach. The high RCS pressure at vessel failure (HPME) leads to most of the corium being relocated to the lower compartment. The continued steam production in the lower compartment and in the cavity causes the containment pressure to increase gradually. The containment finally fails at 46.2 hours. Tables F-8 and F-24 summarize the results of the key timings and the figures of merit.

F.5.11 PDS VB Baseline Scenarios

The PDS VB scenario (LCVB0) begins with a LOCA caused by a 18 in. break in the cold leg. The main steam isolation valves close when the reactor trips on low RCS pressure. The safety injection systems are assumed to be not available. The primary system inventory reduces as it is lost through the break. The boiling off and the loss of the inventory leads to the top of core uncovery at 0.26 hours. The core begins to melt at 0.54 hours and the vessel fails at 2.05 hours. The RCS pressure at the vessel failure time is calculated to be 20 psia.

The steam flow through the break increases the containment pressure, actuating the fan coolers and the containment spray within the first 12 seconds. The CS depletes the RWT at 1.41 hours, but continues to operate in the recirculation mode with suction from the containment sump. The containment pressure at the vessel failure time is 20 psia. After the vessel breach (low pressure

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1 Uncertainty in the formation of a coolable debris bed in the cavity despite an overlying water pool is considered in the CETs.

failure), 99% of the debris resides in the cavity. The CS and ECCs are very effective in holding the containment pressure low in the presence of continuous steaming in the cavity. The results are summarized in Tables F-9 and F-25.

F.5.12 PDS VIA Baseline Scenarios

In this PDS scenario (LCVIA0), initiated by a 18 in. break LOCA, the safety injection systems are working, but are not available in the recirculation mode. Containment cooling is provided by 2 ECCs and 1 CS (injection mode only). The reactor trips on the low RCS pressure at 1.5 seconds. The setpoints for the actuation of ECCs and CS are reached within the first 12 seconds of the transient. The SITs provide inventory to reflood the core after the initial blowdown. The RWT depletes at 0.68 hours, turning off the injection and the containment spray. The core melt begins at 1.95 hours and the vessel fails at 4.06 hours with the RCS pressure at 24.5 psia.

The containment pressure at the vessel breach time is 24 psia. The containment pressure increases abruptly to about 40 psia after the vessel breach, as molten debris gets ejected out of the vessel. The pressure reduces subsequently due to the heat removal by the heat sinks and the ECCs. Tables F-9 and F-26 present the key timings and the figures of merit, respectively.

F.5.13 PDS VIB Baseline Scenarios

This PDS scenario (LCVIB0) is run similar to the case LCVIA0, except that the CS is available also in the recirculation mode. The progression of the accident events is similar and the containment pressure remains low throughout the transient. The vessel fails at 3.77 hours, ejecting the molten debris into the cavity. Tables F-9 and F-27 summarize the timings of key events and the figures of merit.

F.5.14 SGTR Baseline Scenarios

a) In the SGTR01 scenario, the auxiliary feedwater is made available to the intact steam generator. The transient begins with a single (double-ended) steam generator tube rupture. The reactor scrams at 249 seconds and the auxiliary feedwater is actuated at 268 seconds. The MSIVs are closed and the secondary side pressure in the intact steam generator is reduced to 400 psia by opening the atmospheric dump valves. The AFW is lost at 7.45 hours when the condensate storage tank gets depleted. The intact steam generator dries out at 8.26 hours. With the high pressure safety injection assumed failed, the core uncovers at 11.1 hours and the vessel fails at 15.1 hours. At the vessel failure time, the RCS pressure is 1615 psia and the containment pressure is 16 psia. The pressure in the containment rises abruptly to ~30 psia as hot gases and debris get ejected out of the vessel lower head, actuating the fan coolers and the containment spray. The containment pressure reduces due to the cooling provided by the ECCs and the CS. After the vessel failure, most of the debris is predicted by MAAP to reside in the lower compartment. The results of this analysis are tabulated in Tables F-10 and F-28.

b) The scenario SGTR02 is initiated by a single steam generator tube rupture. The auxiliary feedwater is made unavailable and the ADVs are opened to reduce the intact steam generator pressure to 400 psia. With no AFW, the intact generator dries out at 0.52 hours. In the absence of high pressure safety injection flow, the top of core uncovers at 2.65 hours leading to the vessel failure at 4.99 hours. The containment pressure rise accompanying the core melt ejection actuates the fan coolers and the containment spray. The containment pressure remains low and well below the containment failure pressure. The key timings and the figures of merit are presented in Tables F-10 and F-29.

F.5.15 V Sequence

The V Sequence scenario analyzed involves a failure of pressure isolation function for the 10.25" RHR shutdown cooling suction line. A hot leg break of 10.25" is therefore postulated to be the initiator for this LOCA (into the auxiliary building) sequence. Safety injection gets actuated at 9.5 seconds into the transient when 1 high pressure injection pump begins injecting water into the core from the RWT (with 30 seconds delay). Safety Injection Tanks provide additional water inventory into the core, till they get depleted at 409 seconds. The RWT depletes at 8.64 hours and the vessel eventually fails at 12.5 hours. Tables F-11 and F-30 summarize the timings of key events and the figures of merit, respectively, for the V Sequence.

F.6 SELECTED ACCIDENT SEQUENCE VARIATIONS

The accident sequence variations involving phenomenological model parameter changes which affect containment response and challenges by the key technical issues discussed in Section 2 are presented in this section. The sensitivity cases cover the range of input variables which define the phenomenological issues, such as core melt and debris dispersal, hydrogen burn, DCH and debris coolability. These combinations of input variables and model parameter changes may be grouped into major areas of uncertainties or phenomenological issues which significantly affect the code predictions, hence the likelihoods associated with the CET branch points. This section describes the sequence variations which were defined to examine the effects on containment loads as sensitivity ranges about the baseline scenario.

F.6.1 PDS IIIH Variations

Variations of the base case scenario, for the station blackout transient, are defined with model parameter changes as described below.

<u>Case TIIIH2</u> This scenario is similar to the base case scenario TIIIH0 with respect to the model parameters assumptions. However, a hot leg break is initiated at the surge line when the temperature reached 900^{0}K. In the MAAP run this occurred at 5.85 hours, leading to low RCS pressure at vessel breach. The containment pressure rises after the hot leg rupture, reaching ~65 psia at the vessel failure time. The base case scenario has most of the debris dispersed and relocated to the lower compartment. Due to the low pressure vessel failure, the gas velocities are not high enough to entrain the molten debris, which therefore resides

in the cavity in scenario TIIIH2. The vessel fails at 10.1 hours, and the slow pressurization of the containment leads to the containment failure at 39.6 hours. After the cavity dryout, the corium in the cavity attacks the concrete in the basemat, resulting in a depth of penetration of 4.62 feet at 49.6 hours into the transient. Tables F-31A and F-32 summarize the timing of key events and figures of merit for Case TIIIH2.

Case TIIIH3 In this scenario the high pressure melt ejection is modeled to allow the debris to disperse and relocate in the upper compartment at vessel breach. The model parameter FCMDA is set to 1.0, while the other parameters were the same as in the base case scenario TIIIH0. The peak containment pressure at vessel breach increased to 71 psia. Slow subsequent pressurization resulted in the containment failure at 45.6 hours. Tables F-31A and F-33 summarize the timing of key events and figures of merit for Case TIIIH3.

Case TIIIH4 This accident scenario is similar to the base case scenario TIIIH0, where the debris relocates to the lower compartment at vessel breach. However, the fraction of debris which participates in the DCH is changed from 0.03 to 1.0. The progression of events is identical to that of the base case up to the vessel failure time (7.63 hours). With all the debris assumed to be fragmented and dispersed at vessel breach, the DCH effects are very strong. The containment pressure rises abruptly and exceeds the failure pressure of 110 psia. The amount of hydrogen burned was 700 kg in the MAAP simulation. The peak pressure was about 120 psia. This calculation is overly conservative with respect to the debris dispersal. From the cavity configuration, it can be seen that FCMDCH=1 is very unlikely due to the flow path restrictions, precluding any early containment failure. Tables F-31A and F-34 summarize the timings of key events and figures of merit for case TIIIH4.

Case TIIIH5 For this scenario, variations from the base case scenario (TIIIH0) include the parameter changes (FCMDA=1) to model the debris dispersal and relocation to the upper compartment. The fraction of debris fragmented and participating in DCH (FCMDCH=1) is changed to 1.0. This fraction was 0.03 in the base case scenario. The containment pressure response in this case is similar to the case TIIIH4 and the containment fails early after the vessel breach. The peak containment pressure in MAAP was 188 psia, the amount of hydrogen burned being 1260 kg. As stated in case TIIIH4, the modeling of debris dispersal with FCMDCH=1 is very conservative for the PSL cavity configuration. The results are presented in Tables F-31A and F-35.

Case TIIIH8 This accident scenario is similar to the base case scenario, except that the inputs related to the fuel melt are changed so that the hydrogen production due to the clad oxidation is increased. The fuel melt temperature is increased to 5120^{0}K from 2500^{0}K and the latent heat of melting is reduced from 2.5E6 J/kg to 2000 J/kg. The debris dispersal is modeled to the upper compartment (FCMDA=1). The vessel failed at 6.52 hours. The fraction of clad reacted in-vessel was 0.671 compared to 0.326 in the base case. Even with the increased clad oxidation, the hydrogen concentration in the containment was less than 6%, well below the detonation limits. The effect of hydrogen burn is evaluated separately. Tables F-31B and F-36 summarize the timings of key events and figures of merit.

Case TIIIH9 The scenario TIIIH9 is similar to the case TIIIH3, except that the hydrogen is forced to burn at vessel breach time by changing the related parameter values. No hydrogen burn is calculated by MAAP for the scenarios TIIIH0 (base case) and TIIIH3. The changes to the input are:

TAUTO = 400°K
FLPHI = 10.

In this accident sequence, the debris is dispersed and relocated to the upper compartment by setting FCMDA=1. A total of 410 kg of hydrogen is burned. The containment did not fail early. However, the rise in pressure due to the hydrogen burn resulted in the containment failure by the subsequent pressurization at 22.6 hours. Tables F-31B and F-37 provide the results of the key timings and the figures of merit.

Case TIIIH10 The variation of the base case scenario TIIIH0, to force the hydrogen burn at vessel breach, is simulated in the MAAP case TIIIH10, by the following changes,

TAUTO = 400°k
FLPHI = 10.

The amount of hydrogen burned is 430 kg. Although the containment pressure increased due to the hydrogen burn, the containment did not fail when the simulation ended at 50 hours. Tables F-31B and F-38 summarize the timings of key events and figures of merit for Case TIIIH10.

Case TIIIH11 This variation of the base case scenario is similar to that of TIIIH9, except that the hydrogen burn is forced at 4 hours after the vessel breach rather than at vessel breach. The amount of hydrogen burned is 411 kg. The pressure rise at the burn time did not fail the containment. The containment failed late at 46.6 hours. The results of this case are summarized in Tables F-31C and F-39.

Case TIIIH12 The variation of the base case sequence, similar to the case TIIIH2, is simulated in the MAAP run TIIIH12, where the debris heat transfer coefficient is reduced by changing the value of the constant FCHF from 0.1 to 0.02. The progression of events is similar to that of TIIIH2. The hot leg failed at 5.85 hours. The reduction of the debris heat flux leads to an increased heat flow downward into the basemat. However, since the cavity dries out early, the effect of reducing the heat flux has no major impact on this sequence. The depth of penetration of concrete attack was seen to be less than 5 ft. The containment failed at 40.3 hours. A summary of the key event timings and the figures of merit is presented in Tables F-31C and F-40.

Case TIIIH13 As per the recommendation in Reference F-2, an additional MAAP simulation is performed with FCRDR=0.8. This parameter defines the limit for the fraction of the original core below which the remaining core is dumped into the cavity. The default value for this parameter in the base case simulations is 0.1. Since core "dumping" is predicted in the base case calculations, this variation was found to have no impact on the results. The results of TIIIH13 simulation are therefore identical to those of case TIIIH0 as presented in Tables F-31C and F-41.

F.6.2 Direct Containment Heating (DCH)

The cavity configuration of the St.Lucie plant (PSL) does not provide a free and clear path for the debris entrainment out of the cavity. The best estimate Zion-like recommended value of 0.03 for the MAAP parameter FCMDCH is therefore appropriate for PSL to model DCH. This value is used in all the base case simulations. With this value of 0.03 for FCMDCH, the DCH effects in MAAP calculations are found to be not strong enough to fail the containment early. However, for sensitivity purposes, the value of this parameter is increased to 1.0 in scenarios TIIIH4 and TIIIH5. The high gas velocities, corresponding to the flow areas out of the cavity, result in the fragmented particles to be held suspended in the containment atmosphere for an extended period of time. With all the debris participating in DCH, the effects of DCH are therefore intense. The containment fails early after vessel breach in scenarios TIIIH4 and TIIIH5.

The value of FCMDCH=1, as stated earlier, is overly conservative. It is, therefore, unlikely that the containment in PSL will fail early due to DCH.

F.6.3 Hydrogen Combustion

In the base case station blackout scenario, the hydrogen concentration in the containment remained below the flammability limit of ~4%. The total amount of hydrogen was about 430 kg and no hydrogen burn occurred.

In the variations TIIIH4 and TIIIH5, where the DCH effects were conservatively augmented by allowing all the debris to be fragmented and dispersed, the hydrogen production increased to more than 900 kg. The concentrations, however, were still below 6% to eliminate any detonation concerns. In these cases the temperatures were high enough to initiate hydrogen burn and fail the containment early after the vessel breach.

The effect of hydrogen burn, without overly conservative DCH modeling, is studied in MAAP calculations of TIIIH9, TIIIH10 and TIIIH11. In cases TIIIH9 and TIIIH10 the hydrogen was forced to burn at the vessel breach time, whereas in TIIIH11 hydrogen was forced to burn at 4 hours after the vessel failure time. None of these scenarios produced high containment pressures to fail the containment early. The amount of hydrogen burned was about 410-430 kg.

Hand calculations, to estimate adiabatic pressure rise due to hydrogen burn, are performed using the containment conditions after vessel breach from the base case scenario TIIIH0. The amount of hydrogen burned is varied to see the effect on the containment pressure. The conditions from the case TIIIH0 after vessel breach are:

P = 51 psia
T = 350^{0}F
Mass of Steam = 71215 kg (157000 lbm)
Mass of Oxygen = 16100 kg (35500 lbm)
Mass of Nitrogen = 52844 kg (116500 lbm)

H_2 Burned = 432 kg (953 lbm)

For this case, the pressure rise ratio is estimated to be ~1.72, giving the final pressure as 87 psia. This adiabatic pressure rise compares well with the MAAP prediction of ~74 psia, which takes into account the heat sink effects. The containment does not fail at these pressures.

H_2 Burned = 907 kg (2000 lbm)

For this case, the adiabatic pressure rise ratio after the hydrogen burn is estimated to be ~2.47. The final pressure of 126 psia is above the failure pressure of 110 psia.

In conclusion, the hydrogen concentrations in the containment are low enough for any detonation concerns. Additionally, complete hydrogen burn is very unlikely based on the MAAP predictions of low hydrogen concentrations. However when hydrogen is forced to burn completely at vessel breach, the pressure rise does not result in early containment failure. Finally, no credit is taken for the hydrogen recombiners.

F.6.4 Core-Concrete Interaction

The core concrete interaction is studied in MAAP runs TIIIH2 and TIIIH12. In these station blackout scenarios, the vessel fails at low pressure due to the postulated surge line break. Low pressure melt ejection results in most of the debris being resident in the cavity. The cavity dries out early at about 10 to 11 hours into the transient. The debris heat flux constant FCHF is set to a value of 0.1 for the case TIIIH2 and a reduced value of 0.02 in simulation TIIIH12. The maximum depth of penetration of concrete attack is seen to be about 4.7 ft at 50 hours of simulation time. Basemat melt-through is therefore considered to be not a major concern.

F.7 SUMMARY OF SEVERE ACCIDENT PROGRESSION ANALYSIS

The results in terms of key parameters such as core uncovery time and containment pressure of St. Lucie specific MAAP analyses of dominant plant damage states are summarized in Table F-42. For all the baseline runs the estimated containment pressure at vessel breach is lower than 95 psig, the containment failure pressure. This indicates that early containment failure is unlikely.

A sensitivity study based on varying modeling parameters (e.g. Table F-43) demonstrates the following:

1. Direct Containment Heating (DCH) could cause early containment failures for high pressure scenarios, if a large fraction of the debris is assumed to participate in DCH. This is simulated in MAAP cases TIIIH4 and TIIIH5.

2. Low Pressure scenarios do not lead to early containment failures.

3. Without long-term containment heat removal (either ECC or CS), late containment failure is very likely, if the simulation time is extended beyond 50 hours. ECC and CS are lost on EQ limits in some of the scenarios simulated in MAAP. Even with the loss of ECC and CS on EQ limits, the containment is found not to fail within 24 hours in majority of the cases.

4. Hydrogen concentrations in the containment are low enough for any detonation concerns. Complete hydrogen burn is also very unlikely based on the MAAP predictions of hydrogen concentrations of less than 4%. At these levels of hydrogen, local burns due to pocketing will not threaten the containment integrity. Sensitivity calculations show that even complete hydrogen burn at vessel failure time does not produce early containment failure.

5. The maximum depth of concrete attack in scenario in which the cavity dries out early, is predicted in MAAP to be about 4.7 ft. The basemat melt-through is therefore not a concern.

F.8 REFERENCES

F-1. EPRI NP-7071-CCML, "MAAP-3.0B-Modular Accident Analysis Program for LWR Power Plants", November 1990.

F-2. "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B", M. A. Kenton and J. R. Gabor.

F-3. NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms", M. Silberberg, et.al., July 1986.

Table F-1

MAJOR AREAS OF UNCERTAINTY

1.	Natural circulation in RCS
2.	Core melt progression and hydrogen generation
3.	Steam explosion
4.	High pressure melt ejection
5.	Core-concrete interactions
6.	Hydrogen combustion
7.	Iodine chemical form
8.	Fission product revaporization

Table F-2

SUMMARY OF CONDITIONS FOR CORE DAMAGE STATES AND PLANT DAMAGE STATES

CDS DESCRIPTION	RCS PRESSURE AT CM	CONTAINMENT SAFEGUARD STATES CS	ECC	CI	POSSIBLE PLANT DAMAGE STATES	IMPOSSIBLE STATES
I. S1 LOCA With AFW; Early Failure	MOD/High 200-2000 psi	1. Available for injection only 2. Available 3. Available but not actuated 4. Not Available	YES OR NO	YES OR NO	IA, IB, ID, IE, IF, IH	IC, IG: CS will be actuated
II. S1 LOCA with AFW; Late Failure	MOD/High 200-2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	IIA, IIB, IID, IIE, IIF, IIH	IIC, IIG: CS will be actuated
III. Transient & S1 LOCA w/o AFW; Early Failure	High-High > 2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	3B, 3D, 3F, 3H	3A, 3C, 3E, 3G: CS will be actuated with or without ECCs.
IV. TRANSIENT & S1 LOCA w/o AFW; Late Failure	High-High > 2000 psi OR MOD/High 200-2000 psi	1, 2, 3, 4	YES OR NO	YES OR NO	IVB, IVD, IVF, IVH	IVA, C, E, G: CS will be actuated with or without ECCs.
V. Large & Small (S2) LOCA; Early Failure	Low < 200 psi	1, 2, 4	YES OR NO	YES OR NO	VA, VB, VD, VE, VF, VH	VC, G: CS will actuate if available.
VI. Large & Small (S2) LOCA; Late Failure	Low < 200 psi	1, 2, 4	YES OR NO	YES OR NO	VIA, VIB, VID, VIE, VIF, VIH	VIC, G: CS will actuate if available.

Assumptions:
1. S2 LOCAs do not require AFW for heat removal; leakage rates from the RCS is sufficient for depressurization, allowing low head injection; and
2. Long term transients and S1 LOCAs are not considered since Bleed and Feed (B&F) is not given credit.

TABLE F-3 MODERATE PRESSURE PDS (SMALL-SMALL LOCAs)

CALCULATED TIMINGS OF KEY EVENTS IN HOURS			
EVENT	LCIB0	LCIIC2	LCIIB1
PDS	IB	IIA	IIB
Primary System Break	0.0	0.0	0.0
Reactor Scram	0.2	0.1	0.02
Reactor Coolant Pumps Trip	0.52	0.26	0.69
Fan Coolers On	0.19	0.07	0.19
Steam Generator Dryout	----	----	----
Top of Core Uncovery	0.89	0.5	1.88
Containment Sprays On	0.76	0.23	0.72
RWST Water Depletion	2.16	1.5	2.02
Support Plate Failure	2.83	7.88	5.03
Vessel Failure	3.33	8.38	5.53
Accumulator (SITs) Water Depletion	3.35	8.40	5.55
Lower Compartment Floor Dryout	----	----	----
Cavity Dryout	----	----	----
Containment Failure	----	44.25	----
End of Simulation	50	54.25	50.0
RCS Pressure at Vessel Breach (psia)	948.44	497.28	1005.88
Containment P at Vessel Breach (psia)	21.02	37.126	21.24
Containment P Just After VB (psia)	30.12	42.16	28.94

TABLE F-4 MODERATE PRESSURE PDS (SMALL-SMALL LOCAs)

CALCULATED TIMINGS OF KEY EVENTS IN HOURS			
EVENT	LCIIB0	LCIIE0	LCIIF1
PDS	IIB	IIE	IIF
Primary System Break	0.0	0.0	0.0
Reactor Scram	0.01	0.01	0.01
Reactor Coolant Pumps Trip	0.26	0.26	0.26
Fan Coolers On	0.07	----	----
Steam Generator Dryout	----	----	----
Top of Core Uncovery	0.50	0.52	0.52
Containment Sprays On	0.23	0.22	0.22
RWST Water Depletion	1.50	1.49	1.49
Support Plate Failure	7.44	7.91	7.14
Vessel Failure	7.94	8.41	7.64
Accumulator (SITs) Water Depletion	7.95	8.43	7.65
Lower Compartment Floor Dryout	----	----	----
Cavity Dryout	----	----	----
Containment Failure	----	32.70	----
End of Simulation	50.0	42.70	50.0
RCS Pressure at Vessel Breach (psia)	478.6	479.36	499.03
Containment P at Vessel Breach (psia)	22.0	60.67	24.31
Containment P Just After VB (psia)	26.95	65.69	28.78

TABLE F-5 LOSS OF ALL FEEDWATER SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS IN HOURS		
EVENT	LIIIB1	LIIIBO
PDS	IIIB	IIIB
Loss of Feedwater	0.0	0.0
Reactor Scram	0.0008	0.0073
Reactor Coolant Pumps Trip	0.0008	0.84
Fan Coolers On	1.57	0.75
Steam Generator Dryout	1.28	0.37
Top of Core Uncovery	1.87	0.93
Quench Tank Rupture Disk Failure	0.90	0.40
Containment Sprays On	1.67	0.81
RWST Water Depletion	3.07	2.21
Support Plate Failure	3.62	2.42
Vessel Failure	4.12	2.92
Accumulator (SITs) Water Depletion	4.14	2.94
Lower Compartment Floor Dryout	----	----
Cavity Dryout	----	----
Containment Failure	----	----
End of Simulation	50.0	50.0
RCS Pressure at Vessel Breach (psia)	2354.9	2355.6
Containment P at Vessel Breach (psia)	18.6	18.0
Containment P Just After VB (psia)	34.11	33.78

TABLE F-6 STATION BLACKOUT SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS (PDS IIIH) IN HOURS		
EVENT	TIIIH0	TIIIH1
Loss of AC Power	0.0	0.0
Loss of DC Power	0.0	0.0
Loss of CCW	0.0	0.0
Pump Seal LOCA	----	1.50
Reactor Scram	0.0	0.0
Reactor Coolant Pumps Trip	0.0	0.0
Fan Coolers On	----	----
Steam Generator Dryout	3.80	3.80
Top of Core Uncovery	4.76	4.64
Quench Tank Rupture Disk Failure	3.61	4.26
Containment Sprays On	----	----
RWST Water Depletion	----	----
Support Plate Failure	7.13	7.12
Vessel Failure	7.63	7.62
Accumulator (SITs) Water Depletion	----	----
Lower Compartment Floor Dryout	----	----
Cavity Dryout	----	----
Containment Failure	----	----
End of Simulation	50.0	50.0
RCS Pressure at Vessel Breach (psia)	2351.2	2099.3
Containment P at Vessel Breach (psia)	39.05	38.70
Containment P Just After VB (psia)	54.81	54.05

TABLE F-7 LOSS OF CCW SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS IN HOURS		
EVENT	CIIIE1	CIIIF0
PDS	IIIE	IIIF
Loss of CCW	0.0	0.0
Pump Seal LOCA	----	----
Reactor Scram	0.083	0.083
Reactor Coolant Pumps Trip	0.083	0.083
Fan Coolers On	----	----
Steam Generator Dryout	1.69	1.69
Top of Core Uncovery	3.10	3.10
Quench Tank Rupture Disk Failure	1.75	1.75
Containment Sprays On	2.87	2.87
RWST Water Depletion	4.25	4.25
Support Plate Failure	5.11	5.09
Vessel Failure	5.61	5.59
Accumulator (SITs) Water Depletion	5.64	5.61
Lower Compartment Floor Dryout	----	----
Cavity Dryout	----	----
Containment Failure	26.68	27.27
End of Simulation	36.68	37.27
RCS Pressure at Vessel Breach (psia)	2340.9	2352.1
Containment P at Vessel Breach (psia)	24.05	23.70
Containment P Just After VB (psia)	40.42	39.35

TABLE F-8 LOSS OF FEEDWATER

CALCULATED TIMINGS OF KEY EVENTS (PDS IVD) IN HOURS	
EVENT	LIVD0
Loss of Feedwater	0.0
Reactor Scram	0.0073
Reactor Coolant Pumps Trip	9.39
Fan Coolers On	9.16
Steam Generator Dryout	8.46
Top of Core Uncovery	9.54
Quench Tank Rupture Disk Failure	8.79
CST Depletion	7.35
RWST Water Depletion	14.00
Support Plate Failure	12.28
Vessel Failure	12.78
Accumulator (SITs) Water Depletion	12.81
Lower Compartment Floor Dryout	----
Cavity Dryout	----
Containment Failure	46.21
End of Simulation	50.0
RCS Pressure at Vessel Breach (psia)	2365.0
Containment P at Vessel Breach (psia)	21.89
Containment P Just After VB (psia)	38.71

TABLE F-9 LOW PRESSURE PDS (LARGE LOCA)

CALCULATED TIMINGS OF KEY EVENTS IN HOURS			
EVENT	LCVB0	LCVIA0	LCVIB0
PDS	VB	VIA	VIB
Primary System Break	0.0	0.0	0.0
Reactor Scram	0.00042	0.00042	0.00042
Reactor Coolant Pumps Trip	0.0059	0.0059	0.0059
Fan Coolers On	0.00094	0.00094	0.00094
Steam Generator Dryout	----	----	----
Top of Core Uncovery	0.26	1.50	1.30
Containment Sprays On	0.0031	0.0031	0.0031
RWST Water Depletion	1.41	0.68	0.68
Support Plate Failure	1.55	3.56	3.27
Vessel Failure	2.05	4.06	3.77
Accumulator (SITs) Water Depletion	0.048	0.053	0.053
Lower Compartment Floor Dryout	----	----	----
Cavity Dryout	----	----	----
Containment Failure	----	----	----
End of Simulation	50.0	50.0	50.0
RCS Pressure at Vessel Breach (psia)	20.02	24.36	19.95
Containment P at Vessel Breach (psia)	19.55	23.93	19.52
Containment P Just After VB (psia)	23.19	~40.00	20.60

TABLE F-10 STEAM GENERATOR TUBE RUPTURE

CALCULATED TIMINGS OF KEY EVENTS IN HOURS		
EVENT	SGTR01	SGTR02
Primary System Break (1 Tube 0.654")	0.0	0.0
Reactor Scram	0.069	0.069
Reactor Coolant Pumps Trip	11.10	2.19
Fan Coolers On	15.14	4.99
Steam Generator Dryout (unbroken)	8.26	0.52
Top of Core Uncovery	12.06	2.65
CST Depletion	7.45	----
Containment Sprays On	15.14	4.99
RWST Water Depletion	15.84	5.69
Support Plate Failure	14.64	4.49
Vessel Failure	15.14	4.99
Accumulator (SITs) Water Depletion	15.16	5.01
Lower Compartment Floor Dryout	----	----
Cavity Dryout	----	----
Containment Failure	----	----
End of Simulation	50.0	50.0
RCS Pressure at Vessel Breach (psia)	1615.4	1395.9
Containment P at Vessel Breach (psia)	15.77	15.50
Containment P Just After VB (psia)	28.04	26.84

TABLE F-11 V SEQUENCE

CALCULATED TIMINGS OF KEY EVENTS (Containment Bypass) IN HOURS	
EVENT	VSEQC0
Primary System Break	0.0
Reactor Scram	0.0013
Reactor Coolant Pumps Trip	0.020
Fan Coolers On	12.76
Steam Generator Dryout	----
Top of Core Uncovery	0.038
Containment Sprays On	----
RWST Water Depletion	8.64
Support Plate Failure	12.03
Vessel Failure	12.53
Accumulator (SITs) Water Depletion	0.11
Lower Compartment Floor Dryout	----
Cavity Dryout	12.76
Containment Failure	----
End of Simulation	24.0
RCS Pressure at Vessel Breach (psia)	19.5
Containment P at Vessel Breach (psia)	15.26
Containment P Just After VB (psia)	19.71

Table F-12

PDS IB FIGURES OF MERIT
(EVENT LCIB0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.89
2	Time of reactor pressure vessel (RPV) breach, hours	3.33
3	Mass of water in cavity at vessel breach, (lbm)	3.777E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.820E+06
5	Fraction of Zr reacted at vessel breach	0.293
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.917
8	Fraction of UO_2 in cavity	0.082
9	Fraction of CsI in RPV	0.708
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	5.073E-04
12	Fraction of SRO released ex-vessel	3.193E-04
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	6.418E-06
15	Fraction of La_20_3 released ex-vessel	5.626E-06
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	3.587E-02
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-13

PDS IIA FIGURES OF MERIT
(EVENT LCIIC2)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.50
2	Time of reactor pressure vessel (RPV) breach, hours	8.38
3	Mass of water in cavity at vessel breach, (lbm)	3.750E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	2.397E+06
5	Fraction of Zr reacted at vessel breach	0.312
6	Fraction of UO_2 in RPV	0.279
7	Fraction of UO_2 in lower compartment	0.586
8	Fraction of UO_2 in cavity	0.135
9	Fraction of CsI in RPV	0.397
10	Fraction of CsI released to environment	8.997E-04
11	Fraction of SRO released in-vessel	2.789E-02
12	Fraction of SRO released ex-vessel	4.820E-06
13	Fraction of SRO released from containment	1.870E-05
14	Fraction of La_20_3 released in-vessel	0.518
15	Fraction of La_20_3 released ex-vessel	0.0
16	Fraction of La_20_3 released from containment	1.567E-04
17	Fraction of Te_2 released ex-vessel	1.230E-03
18	Fraction of Te_2 released from containment	1.166E-05
19	Fraction of Noble Gases Released from Containment	0.908

End of Simulation Time = 54.25 hours

Table F-14

PDS IIB FIGURES OF MERIT
(EVENT LCIIB1)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	1.88
2	Time of reactor pressure vessel (RPV) breach, hours	5.53
3	Mass of water in cavity at vessel breach, (lbm)	3.804E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.825E+06
5	Fraction of Zr reacted at vessel breach	0.305
6	Fraction of UO_2 in RPV	0.120
7	Fraction of UO_2 in lower compartment	0.818
8	Fraction of UO_2 in cavity	0.062
9	Fraction of CsI in RPV	0.670
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	1.312E-03
12	Fraction of SRO released ex-vessel	2.988E-04
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	1.041E-03
15	Fraction of La_20_3 released ex-vessel	4.850E-06
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	3.179E-02
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50.0 hours

Table F-15

PDS IIB FIGURES OF MERIT
(EVENT LCIIB0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.50
2	Time of reactor pressure vessel (RPV) breach, hours	7.94
3	Mass of water in cavity at vessel breach, (lbm)	3.799E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.922E+06
5	Fraction of Zr reacted at vessel breach	0.298
6	Fraction of UO_2 in RPV	0.175
7	Fraction of UO_2 in lower compartment	0.694
8	Fraction of UO_2 in cavity	0.130
9	Fraction of CsI in RPV	0.377
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	2.670E-02
12	Fraction of SRO released ex-vessel	7.471E-05
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	6.454E-01
15	Fraction of La_20_3 released ex-vessel	1.980E-07
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	1.275E-02
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-16

PDS IIE FIGURES OF MERIT
(EVENT LCIIE0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.52
2	Time of reactor pressure vessel (RPV) breach, hours	8.41
3	Mass of water in cavity at vessel breach, (lbm)	3.739E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	2.317E+06
5	Fraction of Zr reacted at vessel breach	0.274
6	Fraction of UO_2 in RPV	0.205
7	Fraction of UO_2 in lower compartment	0.643
8	Fraction of UO_2 in cavity	0.151
9	Fraction of CsI in RPV	0.375
10	Fraction of CsI released to environment	1.328E-03
11	Fraction of SRO released in-vessel	3.441E-02
12	Fraction of SRO released ex-vessel	0.0
13	Fraction of SRO released from containment	4.699E-05
14	Fraction of La_20_3 released in-vessel	5.408E-01
15	Fraction of La_20_3 released ex-vessel	0.0
16	Fraction of La_20_3 released from containment	3.996E-04
17	Fraction of Te_2 released ex-vessel	1.200E-04
18	Fraction of Te_2 released from containment	1.052E-05
19	Fraction of Noble Gases Released from Containment	0.938

End of Simulation Time = 42.7 hours

Table F-17

PDS IIF FIGURES OF MERIT
(EVENT LCIIF1)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.52
2	Time of reactor pressure vessel (RPV) breach, hours	7.64
3	Mass of water in cavity at vessel breach, (lbm)	3.782E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.900E+06
5	Fraction of Zr reacted at vessel breach	0.275
6	Fraction of UO_2 in RPV	0.216
7	Fraction of UO_2 in lower compartment	0.646
8	Fraction of UO_2 in cavity	0.136
9	Fraction of CsI in RPV	0.400
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	3.426E-02
12	Fraction of SRO released ex-vessel	3.856E-05
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	4.591E-01
15	Fraction of La_20_3 released ex-vessel	9.899E-08
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	7.291E-03
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-18

PDS IIIB FIGURES OF MERIT
(EVENT LIIIB1)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	1.87
2	Time of reactor pressure vessel (RPV) breach, hours	4.12
3	Mass of water in cavity at vessel breach, (lbm)	3.763E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.803E+06
5	Fraction of Zr reacted at vessel breach	0.344
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.917
8	Fraction of UO_2 in cavity	0.082
9	Fraction of CsI in RPV	0.820
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	3.060E-04
12	Fraction of SRO released ex-vessel	7.398E-04
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	1.949E-03
15	Fraction of La_20_3 released ex-vessel	3.705E-05
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	7.929E-02
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-19

PDS IIIB FIGURES OF MERIT
(EVENT LIIIB0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.93
2	Time of reactor pressure vessel (RPV) breach, hours	2.92
3	Mass of water in cavity at vessel breach, (lbm)	3.754E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.872E+06
5	Fraction of Zr reacted at vessel breach	0.317
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.932
8	Fraction of UO_2 in cavity	0.067
9	Fraction of CsI in RPV	0.811
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	2.470E-04
12	Fraction of SRO released ex-vessel	1.340E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	5.395E-06
15	Fraction of La_20_3 released ex-vessel	8.947E-05
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	1.220E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-20

PDS IIIH FIGURES OF MERIT
(EVENT TIIIH0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.896
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.893
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	4.662E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	3.762E-04
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	4.242E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-21

PDS IIIH FIGURES OF MERIT
(EVENT TIIIH1)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.64
2	Time of reactor pressure vessel (RPV) breach, hours	7.62
3	Mass of water in cavity at vessel breach, (lbm)	7.067E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.911E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.897
8	Fraction of UO_2 in cavity	0.097
9	Fraction of CsI in RPV	0.925
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	2.844E-04
12	Fraction of SRO released ex-vessel	5.237E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	4.801E-06
15	Fraction of La_20_3 released ex-vessel	4.499E-04
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	4.762E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-22

PDS IIIE FIGURES OF MERIT
(EVENT CIIIE1)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	3.10
2	Time of reactor pressure vessel (RPV) breach, hours	5.61
3	Mass of water in cavity at vessel breach, (lbm)	3.734E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	2.080E+06
5	Fraction of Zr reacted at vessel breach	0.360
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.971
8	Fraction of UO_2 in cavity	0.028
9	Fraction of CsI in RPV	0.629
10	Fraction of CsI released to environment	9.937E-02
11	Fraction of SRO released in-vessel	2.627E-04
12	Fraction of SRO released ex-vessel	3.555E-04
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	5.708E-06
15	Fraction of La_20_3 released ex-vessel	1.275E-05
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	3.873E-02
18	Fraction of Te_2 released from containment	4.833E-06
19	Fraction of Noble Gases Released from Containment	0.962

End of Simulation Time = 36.68 hours

Table F-23

PDS IIIF FIGURES OF MERIT
(EVENT CIIIF0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	3.10
2	Time of reactor pressure vessel (RPV) breach, hours	5.59
3	Mass of water in cavity at vessel breach, (lbm)	3.761E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.773E+06
5	Fraction of Zr reacted at vessel breach	0.343
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.903
8	Fraction of UO_2 in cavity	0.097
9	Fraction of CsI in RPV	0.767
10	Fraction of CsI released to environment	8.852E-02
11	Fraction of SRO released in-vessel	2.566E-04
12	Fraction of SRO released ex-vessel	5.037E-04
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	5.428E-06
15	Fraction of La_20_3 released ex-vessel	2.437E-05
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	5.322E-02
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 37.27 hours

Table F-24

PDS IVD FIGURES OF MERIT
(EVENT LIVD0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	9.54
2	Time of reactor pressure vessel (RPV) breach, hours	12.78
3	Mass of water in cavity at vessel breach, (lbm)	1.650E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.556E+05
5	Fraction of Zr reacted at vessel breach	0.364
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.933
8	Fraction of UO_2 in cavity	0.066
9	Fraction of CsI in RPV	0.881
10	Fraction of CsI released to environment	3.259E-02
11	Fraction of SRO released in-vessel	4.362E-04
12	Fraction of SRO released ex-vessel	3.818E-03
13	Fraction of SRO released from containment	1.446E-05
14	Fraction of La_20_3 released in-vessel	6.896E-06
15	Fraction of La_20_3 released ex-vessel	3.120E-04
16	Fraction of La_20_3 released from containment	1.138E-06
17	Fraction of Te_2 released ex-vessel	3.392E-01
18	Fraction of Te_2 released from containment	1.222E-03
19	Fraction of Noble Gases Released from Containment	0.943

End of Simulation Time = 50 hours

Table F-25

PDS VB FIGURES OF MERIT
(EVENT LCVB0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.26
2	Time of reactor pressure vessel (RPV) breach, hours	2.05
3	Mass of water in cavity at vessel breach, (lbm)	3.726E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	2.028E+06
5	Fraction of Zr reacted at vessel breach	0.264
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.006
8	Fraction of UO_2 in cavity	0.990
9	Fraction of CsI in RPV	0.232
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	6.205E-04
12	Fraction of SRO released ex-vessel	1.702E-02
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	3.145E-03
15	Fraction of La_20_3 released ex-vessel	3.423E-03
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	7.127E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-26

PDS VIA FIGURES OF MERIT
(EVENT LCVIA0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	1.50
2	Time of reactor pressure vessel (RPV) breach, hours	4.06
3	Mass of water in cavity at vessel breach, (lbm)	3.752E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	2.586E+06
5	Fraction of Zr reacted at vessel breach	0.258
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.018
8	Fraction of UO_2 in cavity	0.980
9	Fraction of CsI in RPV	0.416
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	6.097E-04
12	Fraction of SRO released ex-vessel	5.999E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	2.997E-02
15	Fraction of La_20_3 released ex-vessel	8.046E-04
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	3.435E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-27

PDS VIB FIGURES OF MERIT
(EVENT LCVIB0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	1.30
2	Time of reactor pressure vessel (RPV) breach, hours	3.77
3	Mass of water in cavity at vessel breach, (lbm)	3.776E+05
4	Mass of water in lower compartment of vessel breach, (lbm)	1.984E+06
5	Fraction of Zr reacted at vessel breach	0.253
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.028
8	Fraction of UO_2 in cavity	0.970
9	Fraction of CsI in RPV	0.363
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	6.398E-04
12	Fraction of SRO released ex-vessel	1.052E-02
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	4.279E-03
15	Fraction of La_20_3 released ex-vessel	1.542E-03
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	5.226E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-28

PDS SGTR FIGURES OF MERIT
(EVENT SGTR01)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	12.06
2	Time of reactor pressure vessel (RPV) breach, hours	15.14
3	Mass of water in cavity at vessel breach, (lbm)	0.0
4	Mass of water in lower compartment of vessel breach, (lbm)	0.0
5	Fraction of Zr reacted at vessel breach	0.342
6	Fraction of UO_2 in RPV	0.119
7	Fraction of UO_2 in lower compartment	0.857
8	Fraction of UO_2 in cavity	0.022
9	Fraction of CsI in RPV	0.676
10	Fraction of CsI released to environment	2.030E-02
11	Fraction of SRO released in-vessel	5.928E-04
12	Fraction of SRO released ex-vessel	4.338E-03
13	Fraction of SRO released from containment	1.084E-05
14	Fraction of La_20_3 released in-vessel	6.649E-06
15	Fraction of La_20_3 released ex-vessel	1.901E-04
16	Fraction of La_20_3 released from containment	1.320E-07
17	Fraction of Te_2 released ex-vessel	3.747E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.764

End of Simulation Time = 50 hours

Table F-29

PDS SGTR FIGURES OF MERIT
(EVENT SGTR02)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	2.65
2	Time of reactor pressure vessel (RPV) breach, hours	4.99
3	Mass of water in cavity at vessel breach, (lbm)	0.0
4	Mass of water in lower compartment of vessel breach, (lbm)	0.0
5	Fraction of Zr reacted at vessel breach	0.322
6	Fraction of UO_2 in RPV	0.115
7	Fraction of UO_2 in lower compartment	0.879
8	Fraction of UO_2 in cavity	0.004
9	Fraction of CsI in RPV	0.502
10	Fraction of CsI released to environment	5.563E-02
11	Fraction of SRO released in-vessel	4.350E-04
12	Fraction of SRO released ex-vessel	8.545E-03
13	Fraction of SRO released from containment	2.530E-05
14	Fraction of La_20_3 released in-vessel	9.022E-03
15	Fraction of La_20_3 released ex-vessel	7.578E-04
16	Fraction of La_20_3 released from containment	3.130E-04
17	Fraction of Te_2 released ex-vessel	5.573E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.919

End of Simulation Time = 50 hours

Table F-30

V SEQUENCE FIGURES OF MERIT
(EVENT VSEQC0)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	0.038
2	Time of reactor pressure vessel (RPV) breach, hours	12.53
3	Mass of water in cavity at vessel breach, (lbm)	0.0
4	Mass of water in lower compartment of vessel breach, (lbm)	0.0
5	Fraction of Zr reacted at vessel breach	0.280
6	Fraction of UO_2 in RPV	~0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.977
9	Fraction of CsI in RPV	0.127
10	Fraction of CsI released to environment	8.69E-01
11	Fraction of SRO released in-vessel	7.86E-04
12	Fraction of SRO released ex-vessel	8.55E-02
13	Fraction of SRO released from containment	2.06E-02
14	Fraction of La_20_3 released in-vessel	6.22E-06
15	Fraction of La_20_3 released ex-vessel	1.14E-02
16	Fraction of La_20_3 released from containment	2.57E-03
17	Fraction of Te_2 released ex-vessel	9.97E-01
18	Fraction of Te_2 released from containment	3.79E-01
19	Fraction of Noble Gases Released from Containment	0.996

End of Simulation Time = 24 hours

TABLE F-31A SENSITIVITY STUDY OF STATION BLACKOUT SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS IN HOURS				
EVENT	TIIIH2	TIIIH3	TIIIH4	TIIIH5
Loss of AC Power	0.0	0.0	0.0	0.0
Loss of DC Power	0.0	0.0	0.0	0.0
Loss of CCW	0.0	0.0	0.0	0.0
Pump Seal LOCA	----	----	----	----
Reactor Scram	0.0	0.0	0.0	0.0
Reactor Coolant Pumps Trip	0.0	0.0	0.0	0.0
Fan Coolers On	----	----	----	----
Steam Generator Dryout	3.80	3.80	3.80	3.80
Top of Core Uncovery	4.76	4.76	4.76	4.76
Quench Tank Rupture Disk Failure	3.61	3.61	3.61	3.61
Containment Sprays On	----	----	----	----
RWST Water Depletion	----	----	----	----
Support Plate Failure	9.61	7.13	7.13	7.13
Vessel Failure	10.11	7.63	7.63	7.63
Accumulator (SITs) Water Depletion	----	7.65	----	----
Lower Compartment Floor Dryout	----	----	----	----
Cavity Dryout	10.19	----	----	----
Containment Failure	39.63	45.58	7.63	7.63
End of Simulation Time	49.63	50.0	17.63	17.63
RCS Pressure at Vessel Breach (psia)	68.7	2351.2	2351.2	2351.2
Containment P at Vessel Breach (psia)	64.84	39.05	39.05	39.05
Containment P Just After VB (psia)	~69.05	~71.23	~120.92	~187.72

TABLE F-31B SENSITIVITY STUDY OF STATION BLACKOUT SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS IN HOURS			
EVENT	TIIIH8	TIIIH9	TIIIH10
Loss of AC Power	0.0	0.0	0.0
Loss of DC Power	0.0	0.0	0.0
Loss of CCW	0.0	0.0	0.0
Pump Seal LOCA	----	----	----
Reactor Scram	0.0	0.0	0.0
Reactor Coolant Pumps Trip	0.0	0.0	0.0
Fan Coolers On	----	----	----
Steam Generator Dryout	3.80	3.80	3.80
Top of Core Uncovery	4.76	4.76	4.76
Quench Tank Rupture Disk Failure	3.61	3.61	3.61
Containment Sprays On	----	----	----
RWST Water Depletion	----	----	----
Support Plate Failure	5.98	7.13	7.13
Vessel Failure	6.52	7.63	7.63
Accumulator (SITs) Water Depletion	6.55	7.65	7.65
Lower Compartment Floor Dryout	----	----	----
Cavity Dryout	----	----	----
Containment Failure	---	22.57	----
End of Simulation	50.0	32.57	50.0
RCS Pressure at Vessel Breach (psia)	2225.6	2351.2	2351.2
Containment P at Vessel Breach (psia)	38.33	39.05	39.05
Containment P Just After VB (psia)	~51.0	~75.83	~74.20

TABLE F-31C SENSITIVITY STUDY OF STATION BLACKOUT SEQUENCES

CALCULATED TIMINGS OF KEY EVENTS IN HOURS			
EVENT	TIIIH11	TIIIH12	TIIIH13
Loss of AC Power	0.0	0.0	0.0
Loss of DC Power	0.0	0.0	0.0
Loss of CCW	0.0	0.0	0.0
Pump Seal LOCA	----	----	----
Reactor Scram	0.0	0.0	0.0
Reactor Coolant Pumps Trip	0.0	0.0	0.0
Fan Coolers On	----	----	----
Steam Generator Dryout	3.80	3.80	3.80
Top of Core Uncovery	4.76	4.76	4.76
Quench Tank Rupture Disk Failure	3.61	3.61	3.61
Containment Sprays On	----	----	----
RWST Water Depletion	----	----	----
Support Plate Failure	7.13	9.61	7.13
Vessel Failure	7.63	10.11	7.63
Accumulator (SITs) Water Depletion	7.65	----	----
Lower Compartment Floor Dryout	----	----	----
Cavity Dryout	----	10.85	----
Containment Failure	46.57	40.29	----
End of Simulation	50.0	50.0	50.0
RCS Pressure at Vessel Breach (psia)	2351.2	60.52	2351.2
Containment P at Vessel Breach (psia)	39.05	60.2	39.05
Containment P Just After VB (psia)	~71.23	~65.41	~54.81

Table F-32

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH2)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	10.11
3	Mass of water in cavity at vessel breach, (lbm)	1.489E+03
4	Mass of water in lower compartment of vessel breach, (lbm)	3.340E+05
5	Fraction of Zr reacted at vessel breach	0.442
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.986
9	Fraction of CsI in RPV	0.481
10	Fraction of CsI released to environment	3.667E-03
11	Fraction of SRO released in-vessel	1.362E-03
12	Fraction of SRO released ex-vessel	7.898E-03
13	Fraction of SRO released from containment	2.410E-05
14	Fraction of La_20_3 released in-vessel	2.460E-02
15	Fraction of La_20_3 released ex-vessel	1.775E-03
16	Fraction of La_20_3 released from containment	2.323E-05
17	Fraction of Te_2 released ex-vessel	9.029E-01
18	Fraction of Te_2 released from containment	1.366E-02
19	Fraction of Noble Gases Released from Containment	0.894

End of Simulation Time = 49.6 hours

Table F-33

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH3)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.906
10	Fraction of CsI released to environment	9.924E-04
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	4.506E-04
13	Fraction of SRO released from containment	7.230E-06
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	2.343E-05
16	Fraction of La_20_3 released from containment	3.300E-08
17	Fraction of Te_2 released ex-vessel	4.185E-01
18	Fraction of Te_2 released from containment	8.183E-02
19	Fraction of Noble Gases Released from Containment	0.871

End of Simulation Time = 50 hours

Table F-34

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH4)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.903
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.776
10	Fraction of CsI released to environment	6.930E-02
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	9.639E-06
13	Fraction of SRO released from containment	8.434E-06
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	1.897E-05
16	Fraction of La_20_3 released from containment	1.234E-05
17	Fraction of Te_2 released ex-vessel	9.587E-01
18	Fraction of Te_2 released from containment	6.371E-01
19	Fraction of Noble Gases Released from Containment	0.839

End of Simulation Time = 17.6 hours

Table F-35

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH5)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	7.638E-04
9	Fraction of CsI in RPV	0.737
10	Fraction of CsI released to environment	1.433E-01
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	0.0
13	Fraction of SRO released from containment	3.615E-06
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	0.0
16	Fraction of La_20_3 released from containment	4.949E-08
17	Fraction of Te_2 released ex-vessel	9.986E-01
18	Fraction of Te_2 released from containment	3.565E-02
19	Fraction of Noble Gases Released from Containment	0.969

End of Simulation Time = 17.6 hours

Table F-36

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH8)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	6.52
3	Mass of water in cavity at vessel breach, (lbm)	1.092E+03
4	Mass of water in lower compartment of vessel breach, (lbm)	1.770E+05
5	Fraction of Zr reacted at vessel breach	0.671
6	Fraction of UO_2 in RPV	----
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.0
9	Fraction of CsI in RPV	0.880
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	2.907E-03
12	Fraction of SRO released ex-vessel	0.0
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	2.405E-03
15	Fraction of La_20_3 released ex-vessel	0.0
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	0.0
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-37

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH9)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.897
10	Fraction of CsI released to environment	6.328E-03
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	1.051E-03
13	Fraction of SRO released from containment	1.181E-04
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	1.765E-05
16	Fraction of La_20_3 released from containment	3.135E-07
17	Fraction of Te_2 released ex-vessel	4.795E-01
18	Fraction of Te_2 released from containment	4.729E-02
19	Fraction of Noble Gases Released from Containment	0.904

End of Simulation Time = 32.6 hours

Table F-38

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIII10)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.896
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.866
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	4.492E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	3.611E-04
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	4.174E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

Table F-39

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIIIH11)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.906
10	Fraction of CsI released to environment	8.997E-04
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	4.579E-04
13	Fraction of SRO released from containment	6.025E-06
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	2.346E-05
16	Fraction of La_20_3 released from containment	3.300E-08
17	Fraction of Te_2 released ex-vessel	3.820E-01
18	Fraction of Te_2 released from containment	7.220E-02
19	Fraction of Noble Gases Released from Containment	0.867

End of Simulation Time = 50 hours

Table F-40

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIII12)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	10.11
3	Mass of water in cavity at vessel breach, (lbm)	1.440E+03
4	Mass of water in lower compartment of vessel breach, (lbm)	3.322E+05
5	Fraction of Zr reacted at vessel breach	0.441
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.0
8	Fraction of UO_2 in cavity	0.987
9	Fraction of CsI in RPV	0.480
10	Fraction of CsI released to environment	3.291E-03
11	Fraction of SRO released in-vessel	1.357E-03
12	Fraction of SRO released ex-vessel	9.965E-03
13	Fraction of SRO released from containment	2.289E-05
14	Fraction of La_20_3 released in-vessel	7.705E-06
15	Fraction of La_20_3 released ex-vessel	2.509E-03
16	Fraction of La_20_3 released from containment	6.863E-06
17	Fraction of Te_2 released ex-vessel	9.204E-01
18	Fraction of Te_2 released from containment	1.142E-02
19	Fraction of Noble Gases Released from Containment	0.893

End of Simulation Time = 50 hours

Table F-41

FIGURES OF MERIT (SENSITIVITY STUDY)
(EVENT TIII13)

Figure of Merit	Description	Calculated Values
1	Time top of core is uncovered, hours	4.76
2	Time of reactor pressure vessel (RPV) breach, hours	7.63
3	Mass of water in cavity at vessel breach, (lbm)	8.165E+02
4	Mass of water in lower compartment of vessel breach, (lbm)	1.843E+05
5	Fraction of Zr reacted at vessel breach	0.326
6	Fraction of UO_2 in RPV	0.0
7	Fraction of UO_2 in lower compartment	0.896
8	Fraction of UO_2 in cavity	0.096
9	Fraction of CsI in RPV	0.893
10	Fraction of CsI released to environment	0.0
11	Fraction of SRO released in-vessel	3.976E-04
12	Fraction of SRO released ex-vessel	4.662E-03
13	Fraction of SRO released from containment	0.0
14	Fraction of La_20_3 released in-vessel	6.484E-06
15	Fraction of La_20_3 released ex-vessel	3.762E-04
16	Fraction of La_20_3 released from containment	0.0
17	Fraction of Te_2 released ex-vessel	4.242E-01
18	Fraction of Te_2 released from containment	0.0
19	Fraction of Noble Gases Released from Containment	0.0

End of Simulation Time = 50 hours

TABLE F-42 BEST ESTIMATE ACCIDENT CHRONOLOGY Time in Hours								
Plant Damage State	Core Uncovery	Vessel Failure	Cavity Dryout	Sprays On	Containment Failure	Containment P at VB	RCS P at VB	MAAP Run Name
IB (2"LOCA)	0.89 hr.	3.33 hr.	---	0.76 hr.	---	~30 psia	950 psia	LCIB0
IIA (3"LOCA)	0.50 hr.	8.38 hr.	---	0.23 hr.	44.25 hr.	~45 psia	500 psia	LCIIC2
IIB (3"LOCA)	0.50 hr.	7.94 hr.	---	0.23 hr.	---	~30 psia	480 psia	LCIIB0
IIB (2"LOCA)	1.88 hr.	5.53 hr.	—	0.72 hr.	---	~30 psia	1006 psia	LCIIB1
IIE (3"LOCA)	0.52 hr.	8.41 hr.	---	0.22 hr.	32.70 hr.	~66 psia	480 psia	LCIIE0
IIF (3"LOCA)	0.52 hr.	7.64 hr.	---	0.22 hr.	---	~30 psia	500 psia	LCIIF1
IIIB (LOFW)	0.93 hr.	2.92 hr.	---	0.81 hr.	---	~35 psia	2355 psia	LIIIB0
IIIB (LOGR)	1.87 hr.	4.12 hr.	---	1.67 hr.	---	~35 psia	2355 psia	LIIIB1
IIIE (CCW)	3.10 hr.	5.61 hr.	---	2.87 hr.	26.68 hr.	~41 psia	2340 psia	CIIIE1
IIIF (CCW)	3.10 hr.	5.59 hr.	---	2.87 hr.	27.27 hr.	~40 psia	2350 psia	CIIIF0
IIIH (NO SL)	4.76 hr.	7.63 hr.	---	—	---	~55 psia	2350 psia	TIIIH0
IIIH (S LOCA)	4.64 hr.	7.62 hr.	---	---	---	~55 psia	2100 psia	TIIIH1
IVD (LOFW)	9.54 hr.	12.78 hr.	—	—	46.21 hr.	~40 psia	2365 psia	LIVD0
VB (18"BRK)	0.26 hr.	2.05 hr.	---	0.003 hr.	---	~25 psia	20 psia	LCVB0
VIA (18"BRK)	1.50 hr.	4.06 hr.	—	0.003 hr.	---	~35 psia	25 psia	LCVIA0

TABLE F-42
BEST ESTIMATE ACCIDENT CHRONOLOGY
Time in Hours

Plant Damage State	Core Uncovery	Vessel Failure	Cavity Dryout	Sprays On	Containment Failure	Containment P at VB	RCS P at VB	MAAP Run Name
VIB (18"BRK)	1.30 hr.	3.77 hr.	___	0.003 hr.	---	~25 psia	20 psia	LCVIB0
SGTR	12.06 hr.	15.14 hr.	___	15.14 hr.	---	~30 psia	1620 psia	SGTR01
SGTR	2.65 hr.	4.99 hr.	---	4.99 hr.	---	~30 psia	1400 psia	SGTR02
VSEQ	0.038 hr.	12.53 hr.	12.76 hr.	---	---	~20 psia	20 psia	VSEQC0
IIIH	4.76 hr.	10.11 hr.	10.19 hr.	---	39.63 hr.	~70 psia	70 psia	TIIIH2
IIIH	4.76 hr.	7.63 hr.	___	___	45.58 hr.	~75 psia	2350 psia	TIIIH3
IIIH	4.76 hr.	7.63 hr.	___	___	7.63 hr.	~121 psia	2350 psia	TIIIH4
IIIH	4.76 hr.	7.63 hr.	___	___	7.63 hr.	~188 psia	2350 psia	TIIIH5
IIIH	4.76 hr.	6.52 hr.	___	___	---	~55 psia	2225 psia	TIIIH8
IIIH	4.76 hr.	7.63 hr.	___	___	22.57 hr.	~80 psia	2350 psia	TIIIH9
IIIH	4.76 hr.	7.63 hr.	___	___	---	~75 psia	2350 psia	TIIIH10
IIIH	4.76 hr.	7.63 hr.	---	___	46.57 hr.	~75 psia	2350 psia	TIIIH11
IIIH	4.76 hr.	10.11 hr.	10.85 hr.	___	40.29 hr.	~65 psia	60 psia	TIIIH12
IIIH	4.76 hr.	7.63 hr.	---	---	---	~55 psia	2350 psia	TIIIH13

Table F-43 Key Assumptions for Various Cases of MAAP runs

LCIB0	Base Case PDS IB 2" LOCA, No inj., AFW on, 1 ECC, 1 CS (Inj. & Recirc)
LCIIC2	Base Case PDS IIA 3" LOCA, Inj. on (No Recirc), AFW on, 1 ECC, 1 CS (No Recirc)
LCIIB0	Base Case PDS IIB 3" LOCA, Inj. on (No Recirc), AFW on, 1 ECC, 1 CS (Inj. & Recirc)
LCIIB1	Base Case PDS IIB 2" LOCA, Inj. on (No Recirc), AFW on, 1 ECC, 1 CS (Inj. & Recirc)
LCIIE0	Base Case PDS IIE 3" LOCA, Inj. on (No Recirc), AFW on, ECC off, 1 CS (No Recirc)
LCIIF1	Base Case PDS IIF 3" LOCA, Inj. on (No Recirc), AFW on, ECC off, 1 CS (Inj. & Recirc)
LIIIB0	Base Case PDS IIIB LOFW, No AFW, 2 ECCs, 1 CS (Inj. & Recirc), PORVs auto
LIIIB1	Base Case PDS IIIB Loss Of Grid, RCPs off 3.0 sec., LOFW, No AFW, 1 ECC, 1 CS (Inj. & Recirc), PORVs auto
CIIIE1	Base Case PDS IIIE CCW failure, No AFW, No Seal LOCA, ECC off, 1 CS (No Recirc), ADVs 400 Psia
CIIIF0	Base Case PDS IIIF CCW failure, No AFW, No Seal LOCA, ECC off, 1 CS (Inj. & Recirc), ADVs 400 Psia
TIIIH0	Base Case PDS IIIH Loss Of Power (Station Blackout), No Seal LOCA, AFW 2 hours, ECC off, CS off
TIIIH1	Base Case PDS IIIH Loss Of Power (Station Blackout), 1/4" per RCP Seal LOCA, AFW 2 hours, ECC off, CS off
TIIIH2	PDS IIIH (Sensitivity Run) 10.126" Dia Hot Leg Break at 900°K, FCMDA=0, FCMDCH=0.03
TIIIH3	PDS IIIH (Sensitivity Run) Debris to Upper Compartment (A), FCMDA=1, FCMDCH=0.03
TIIIH4	PDS IIIH (Sensitivity Run) Debris Fraction in DCH increased to 1.0, FCMDA=0, FCMDCH=1.0
TIIIH5	PDS IIIH (Sensitivity Run) Debris to Upper Compartment (A), Fraction in DCH 1.0, FCMDA=1, FCMDCH=1.0
TIIIH8	PDS IIIH (Sensitivity Run) H_2 increased by Fuel Melt Input, LHEU=2000 J/Kg, TEU=5120°K, FCMDA=1, FCMDCH=1.0
TIIIH9	PDS IIIH (Sensitivity Run) Force H_2 Burn at VB, Debris to A, TAUTO=400°K, FLPHI=10.0, FCMDA=1, FCMDCH=0.03

TIIIH10	PDS IIIH (Sensitivity Run) Force H_2 Burn at VB, TAUTO=400°K, FLPHI=10.0, FCMDA=0, FCMDCH=0.03
TIIIH11	PDS IIIH (Sensitivity Run) Force H_2 Burn 4 Hrs. after VB, Debris to A, TAUTO=400°K, FLPHI=10.0, FCMDA=1
TIIIH12	PDS IIIH (Sensitivity Run) 10.126" Dia Hot Leg Break at 900°K, Reduce Debris Heat Flux, FCHF=0.02, FCMDA=0
TIIIH13	PDS IIIH (Sensitivity Run) Increase Core Dump Fraction, FCRDR=0.8, FCMDA=0, FCMDCH=0.03
LIVD0	Base Case PDS IVD LOFW, AFW available, 2 ECCs, CS off
LCVB0	Base Case PDS VB 18" LOCA, No inj., No AFW, 2 ECCs, 1 CS (Inj. & Recirc), 3 SITs
LCVIA0	Base Case PDS VIA 18" LOCA, Inj. on (No Recirc), No AFW, 2 ECCs, 1 CS (No Recirc), 3 SITs
LCVIB0	Base Case PDS VIB 18" LOCA, Inj. on (No Recirc), No AFW, 2 ECCs, 1 CS (Inj. & Recirc), 3 SITs
SGTR01	Base Case SGTR SG Tube Rupture, AFW to intact SG, HPSI off, ADVs 400 Psia, 2 ECCs, 1 CS (Inj. & Recirc)
SGTR02	Base Case SGTR SG Tube Rupture, AFW off, HPSI off, ADVs 400 Psia, 2 ECCs, 1 CS (Inj. & Recirc)
VSEQC0	Base Case V Sequence 10.25" LOCA into Aux Bldg., LPSI off, 1 HPSI on, 2 ECC and 1 CS available

APPENDIX F-1

29 PAGES

PPS	=	Primary System Pressure
PBS	=	Broken Steam Generator Pressure
PUS	=	Intact Steam Generator Pressure
PA	=	Containment Upper Compartment Pressure
PB	=	Containment Lower Compartment Pressure
PC	=	Containment Cavity Pressure
PD	=	Containment Annular Compartment Pressure

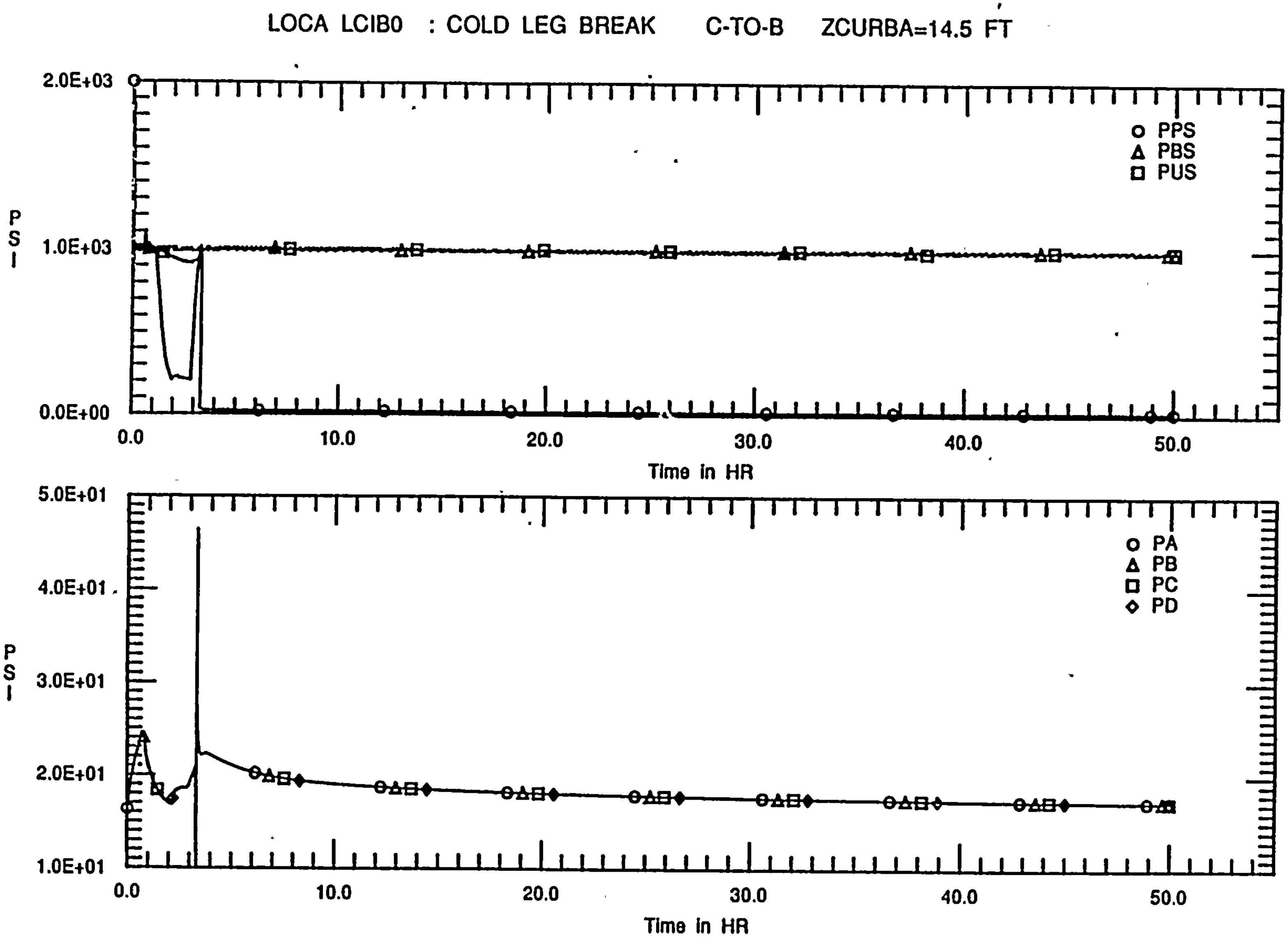

Figure F-1 Pressure Variations for PDS IB (Case LCIBO)

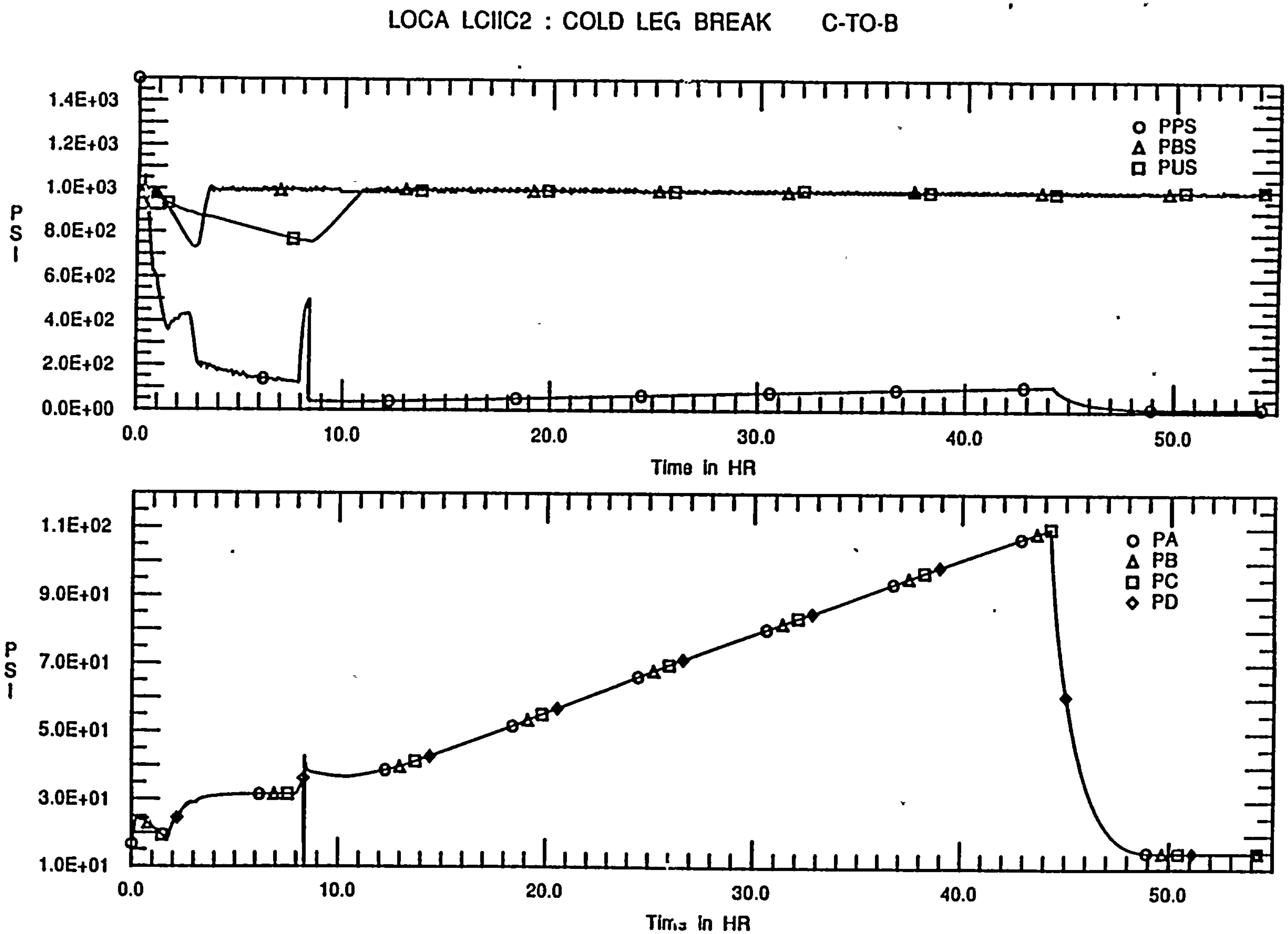

Figure F-2 Pressure Variations for PDS IIA (Case LCIIC2)

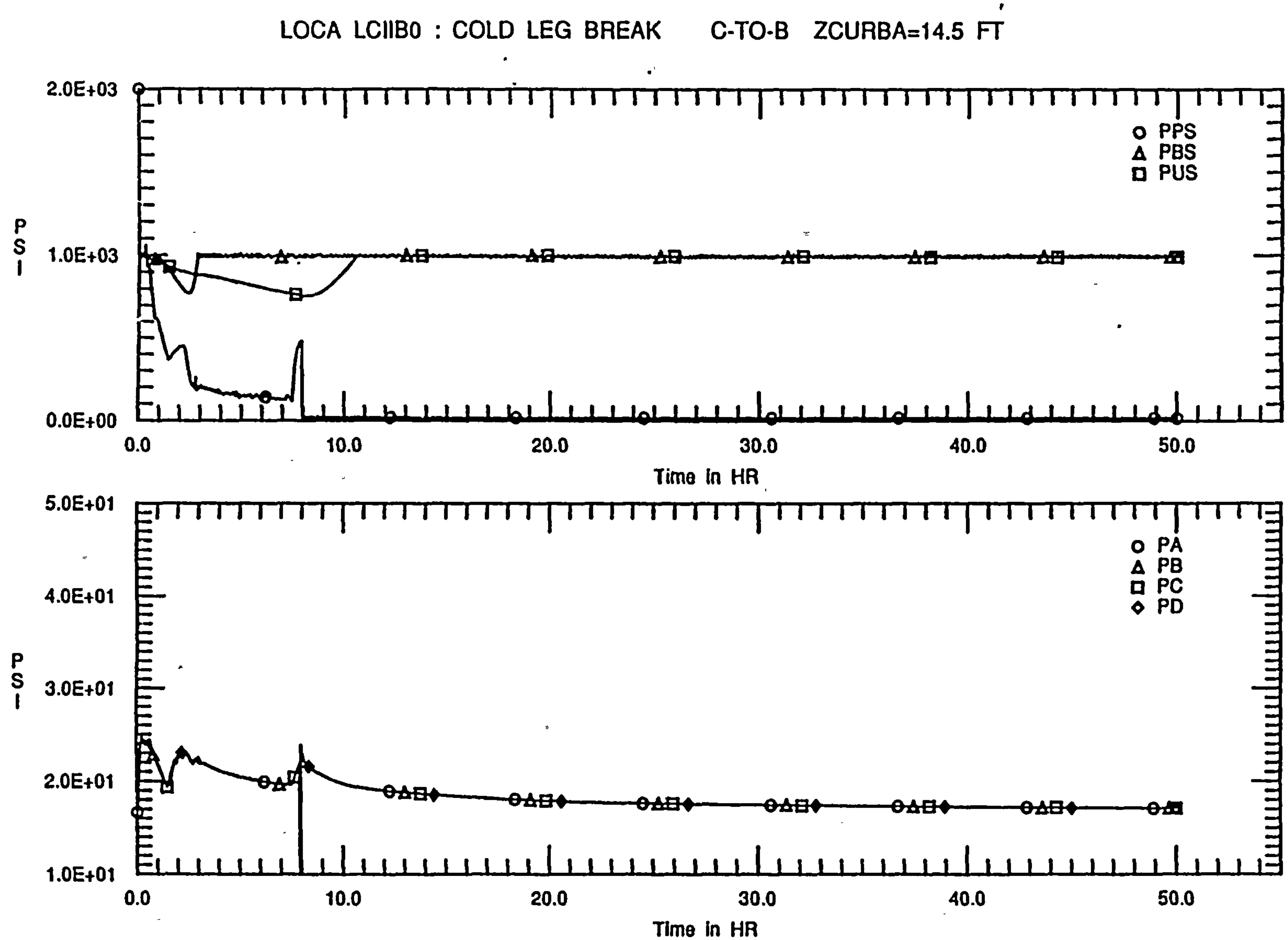

Figure F-3 Pressure Variations for PDS IIB (Case LCIIB0)

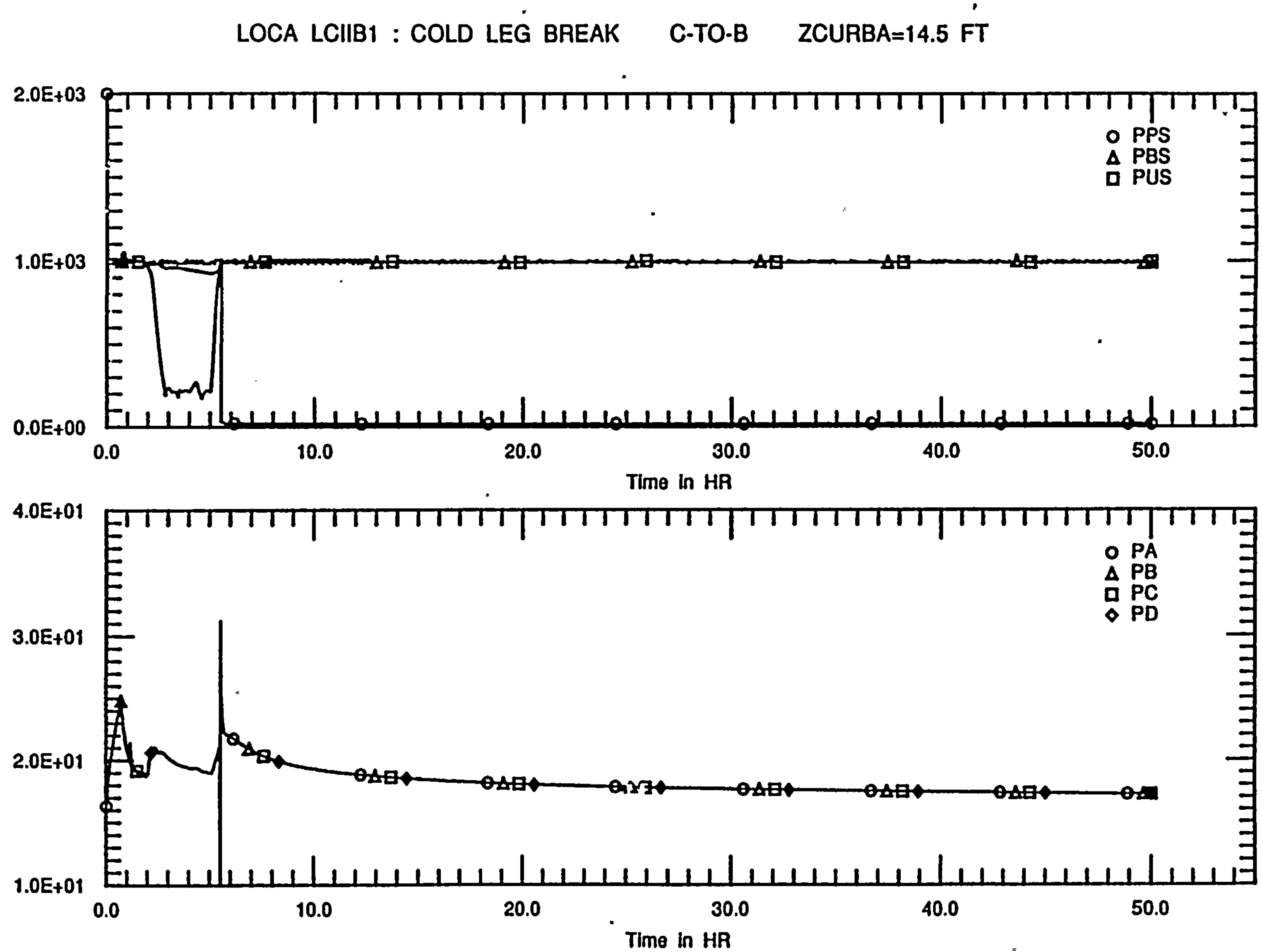

Figure F-4 Pressure Variations for PDS IIB (Case LCIIB1)

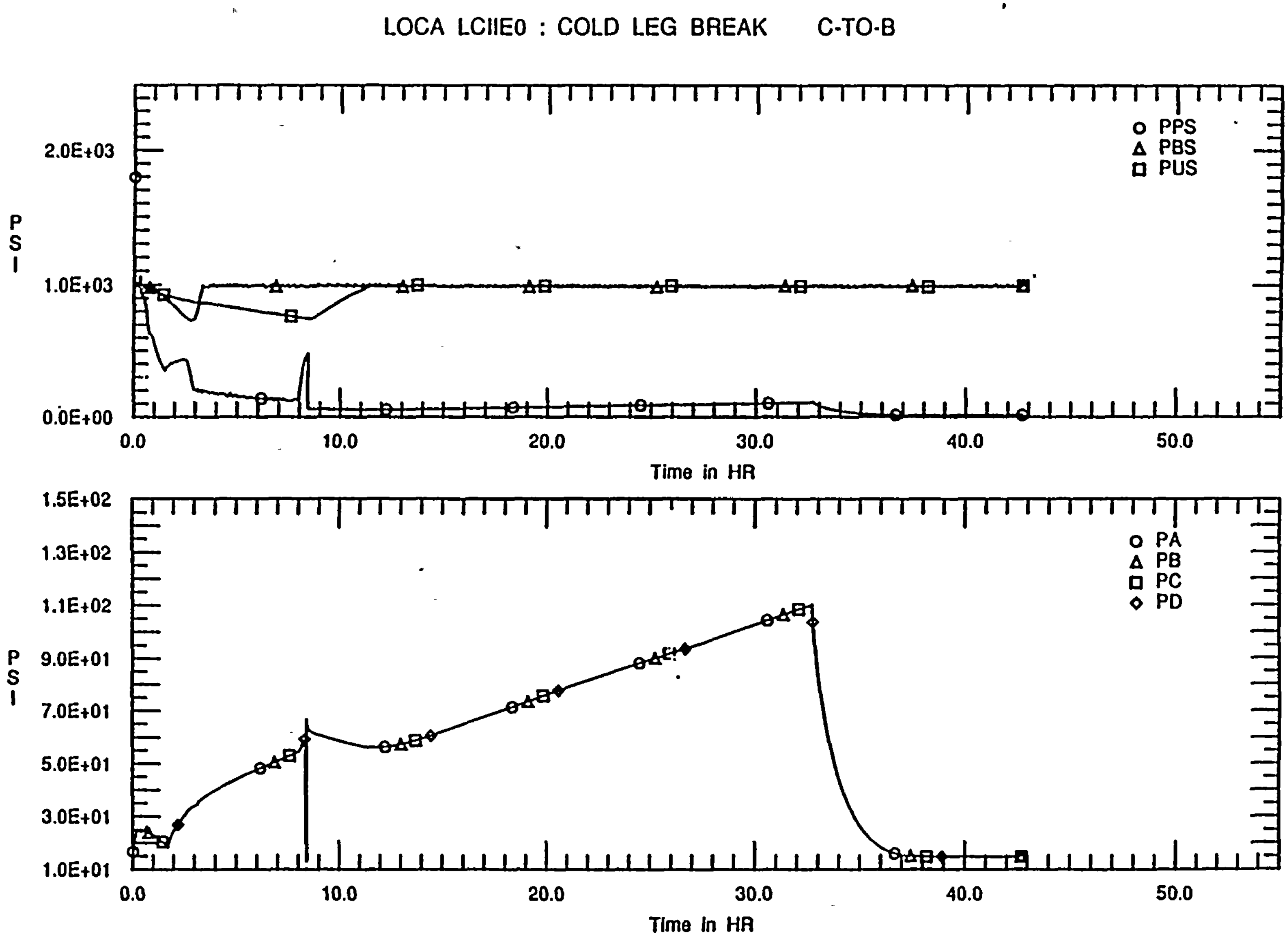

Figure F-5 Pressure Variations for PDS IIE (Case LCIIE0)

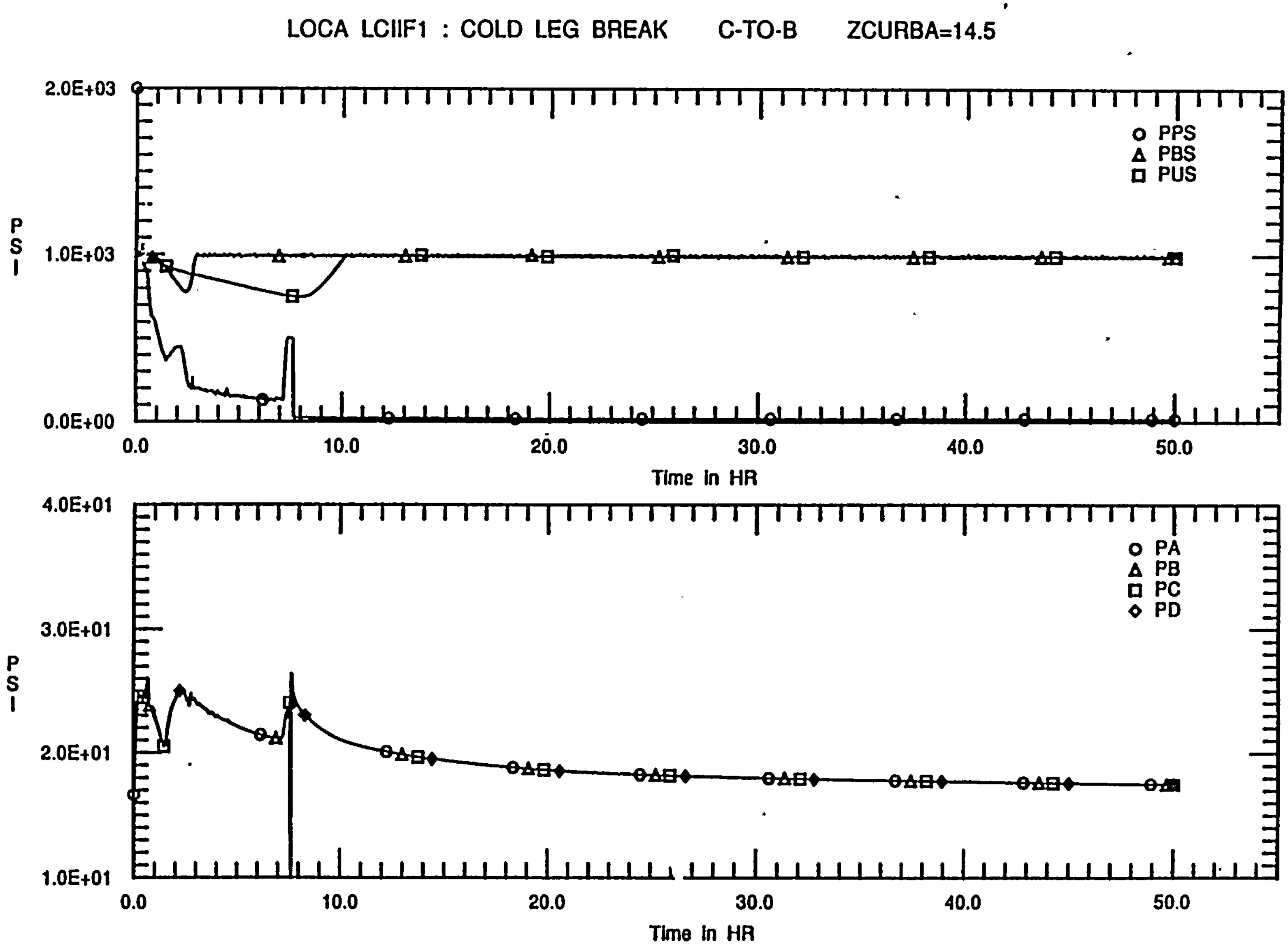

Figure F-6 Pressure Variations for PDS IIF (Case LCIIF1)

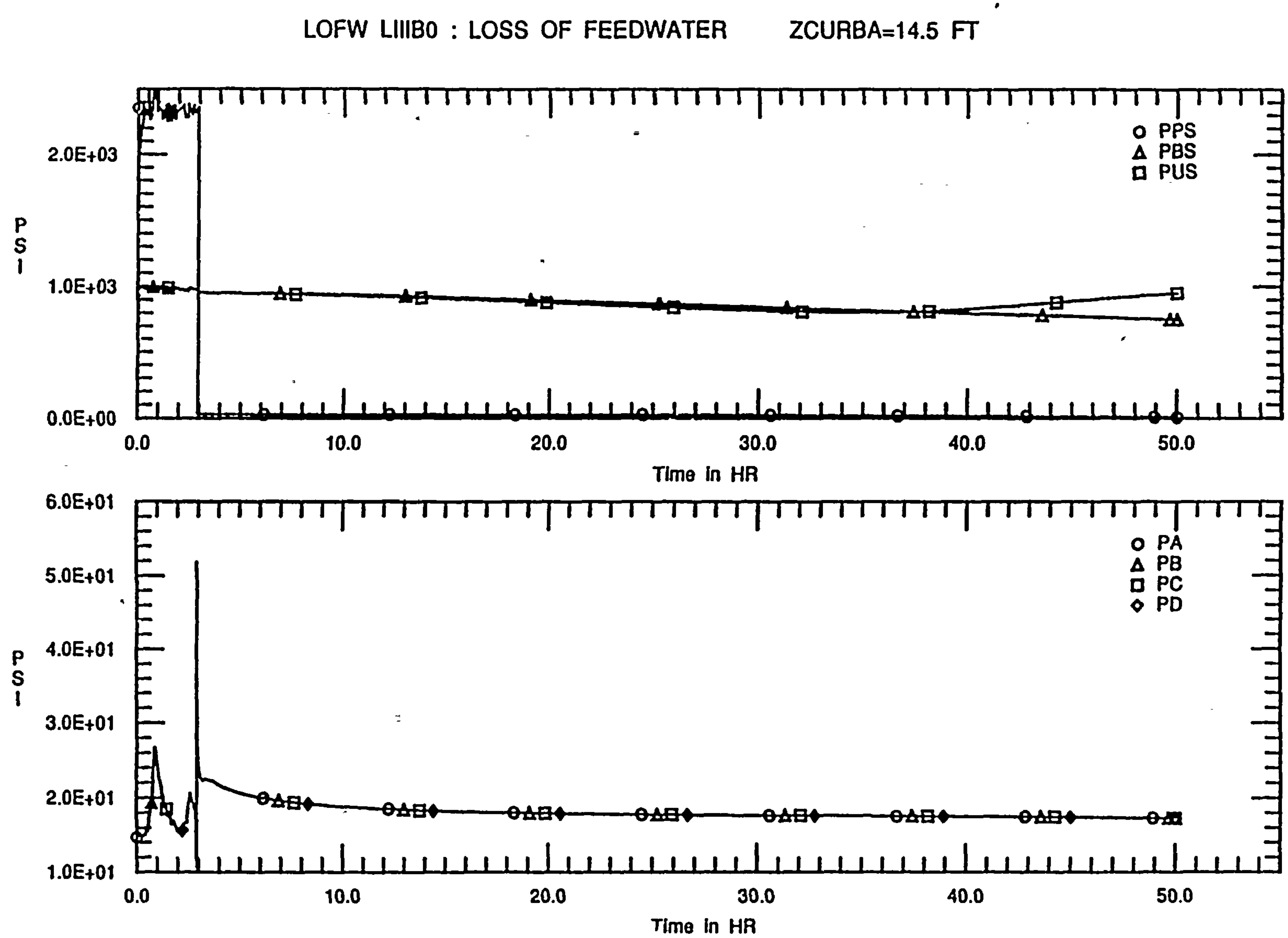

Figure F-7 Pressure Variations for PDS IIIB (Case LIIIBO)

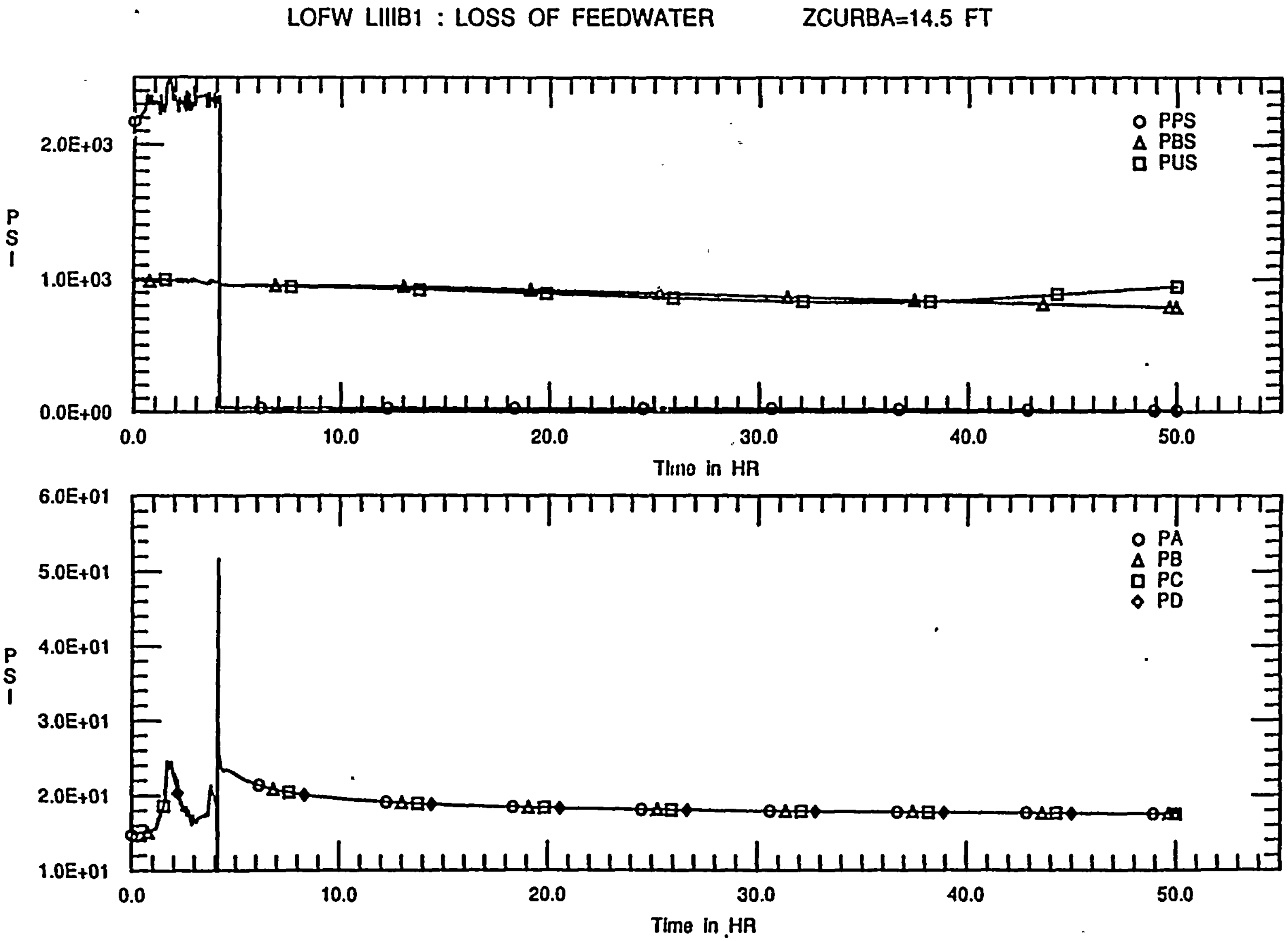

Figure F-8 Pressure Variations for PDS IIIB (Case LIIIB1)

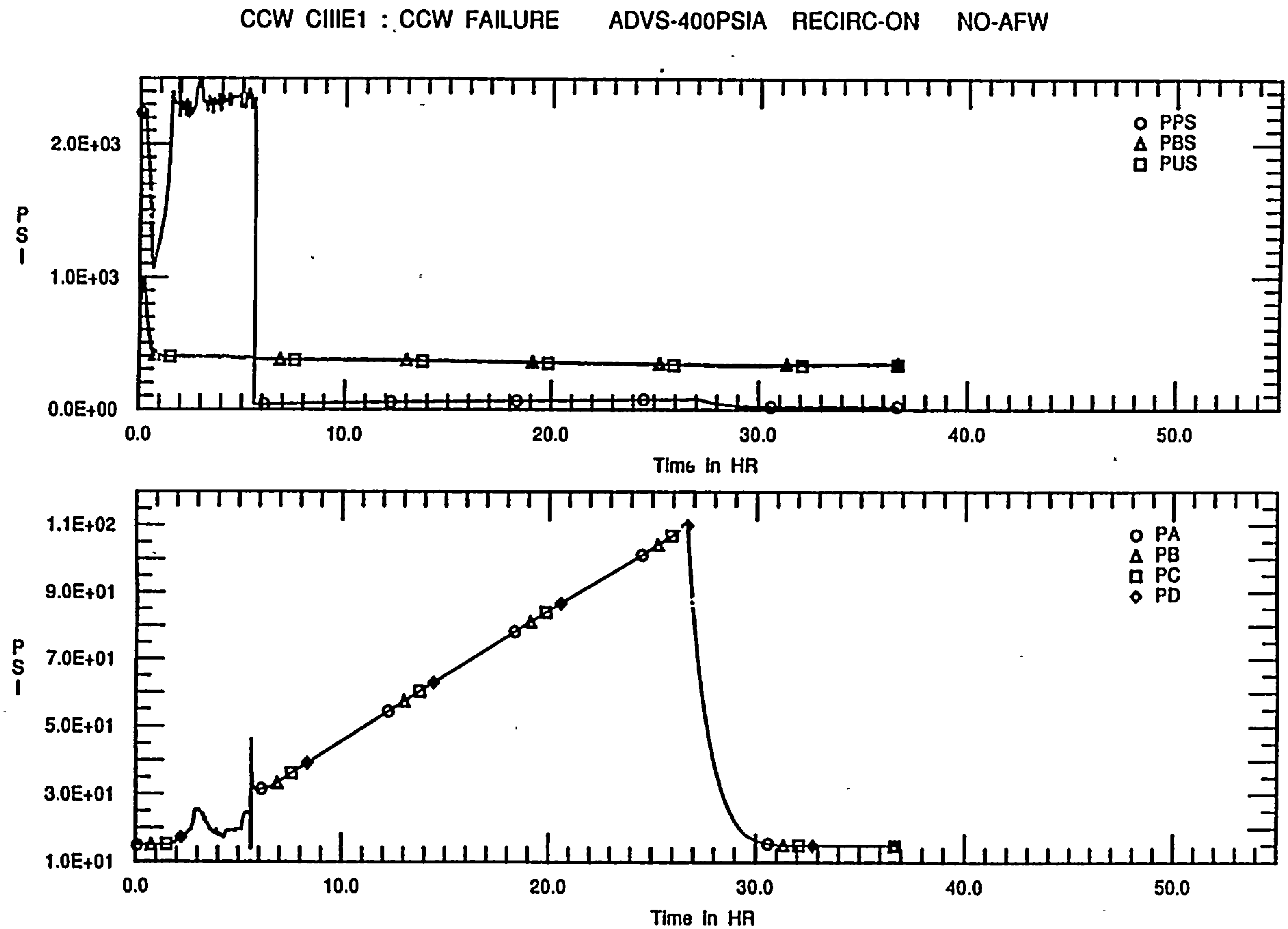

Figure F-9 Pressure Variations for PDS IIIE (Case CIIIE1)

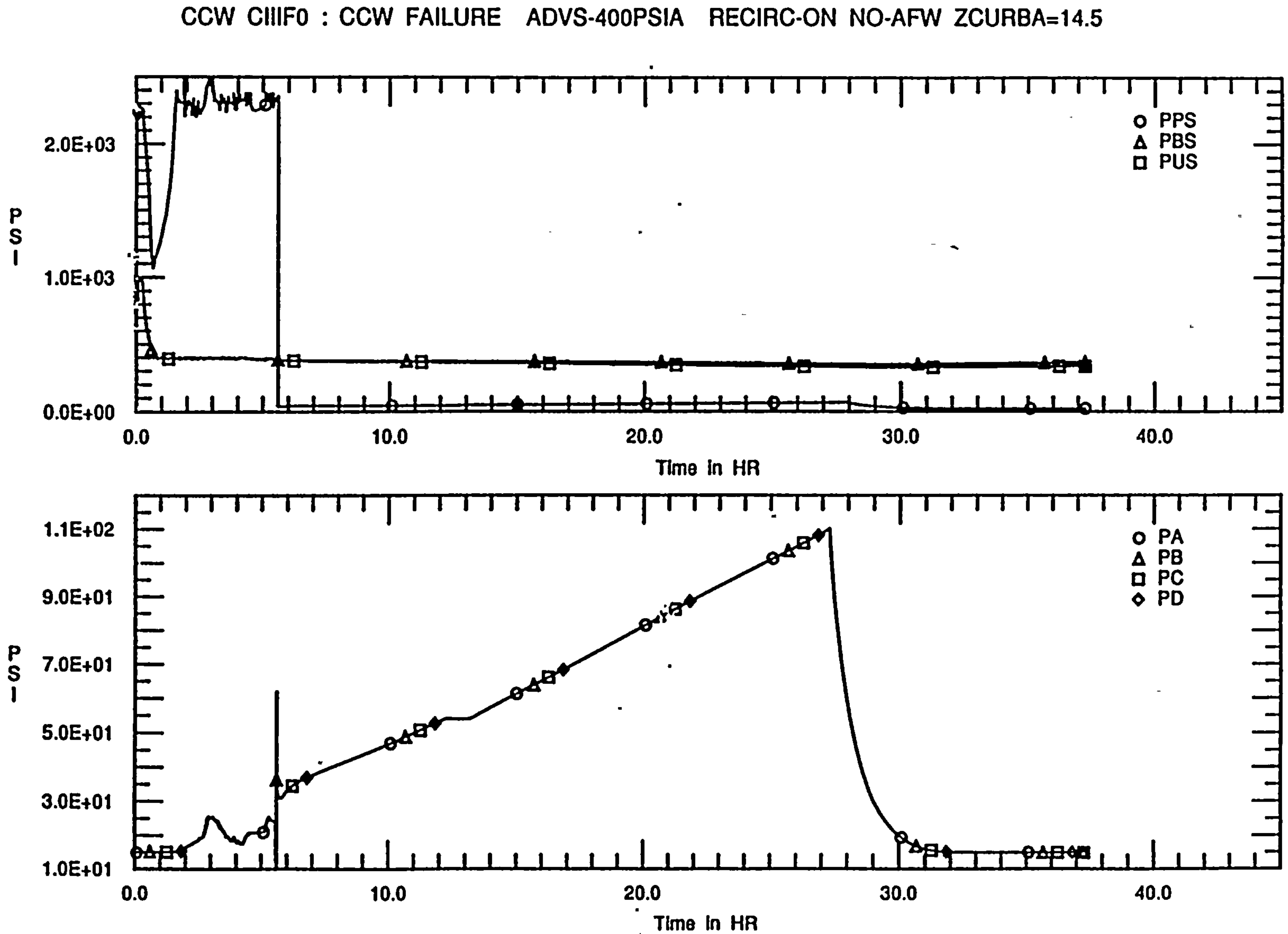

Figure F-10 Pressure Variations for PDS IIIF (Case CIIIF0)

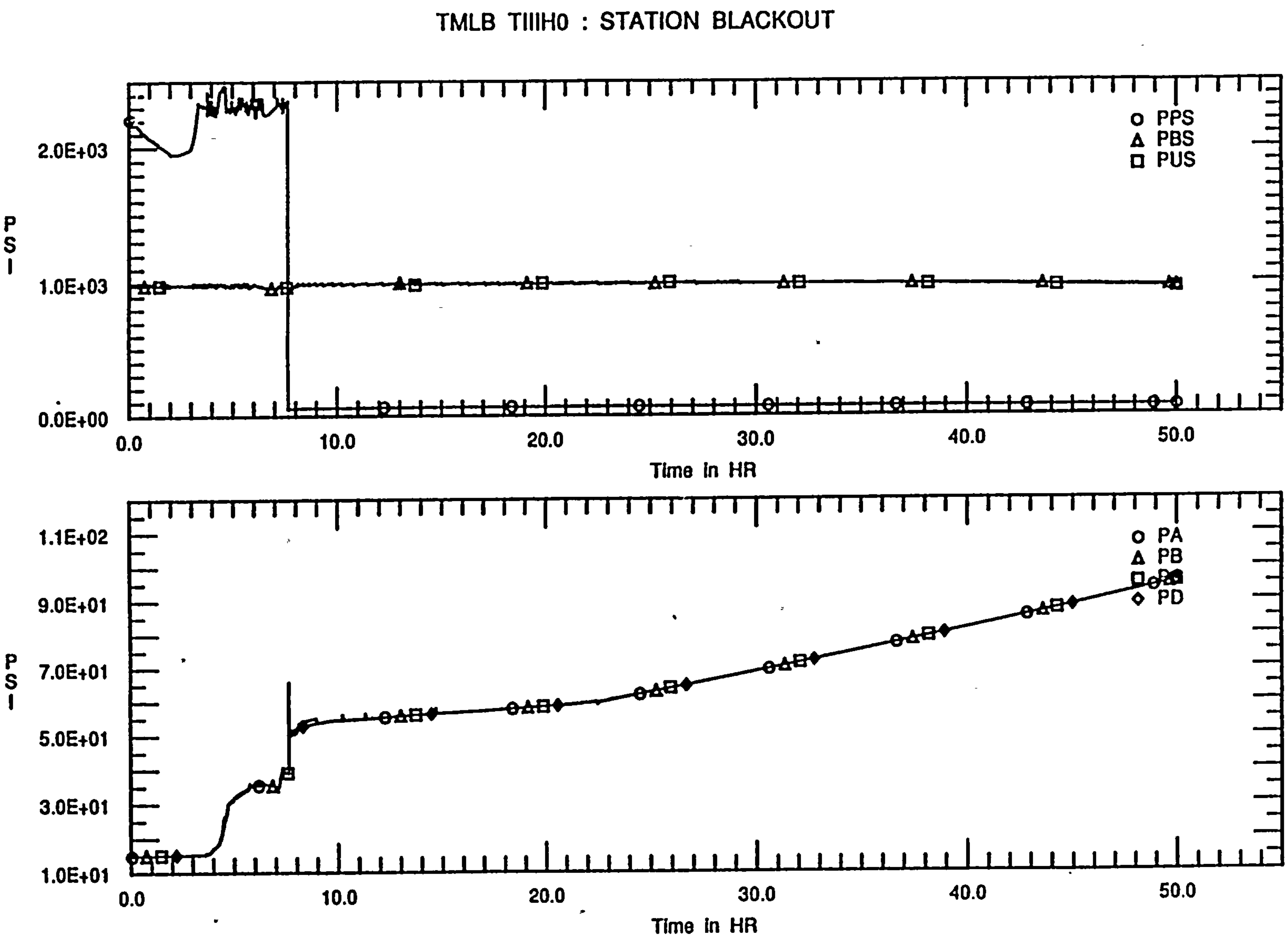

Figure F-11 Pressure Variations for PDS IIIH (Case TIIIH0)

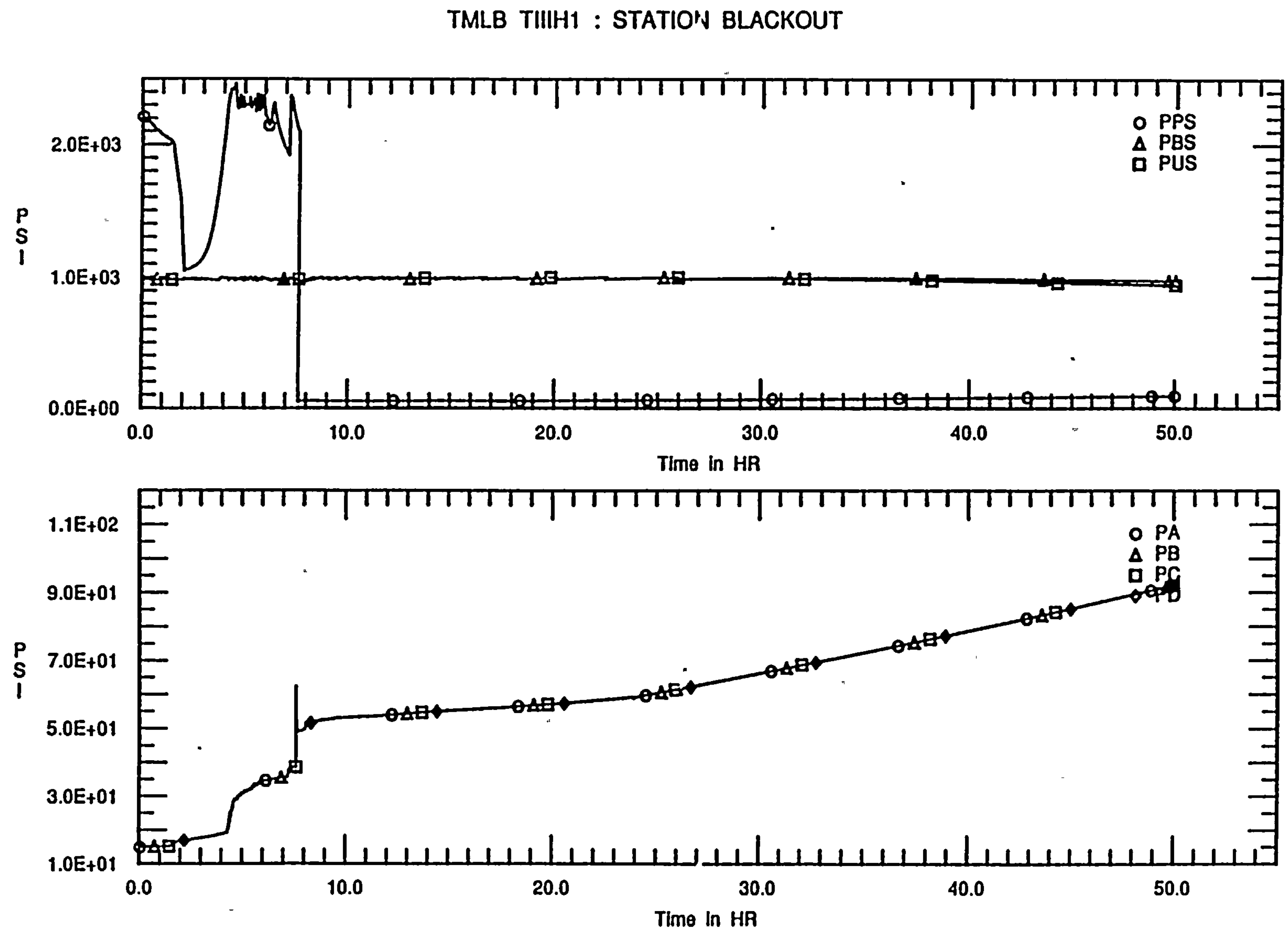

Figure F-12 Pressure Variations for PDS IIIH (Case TIIIH1)

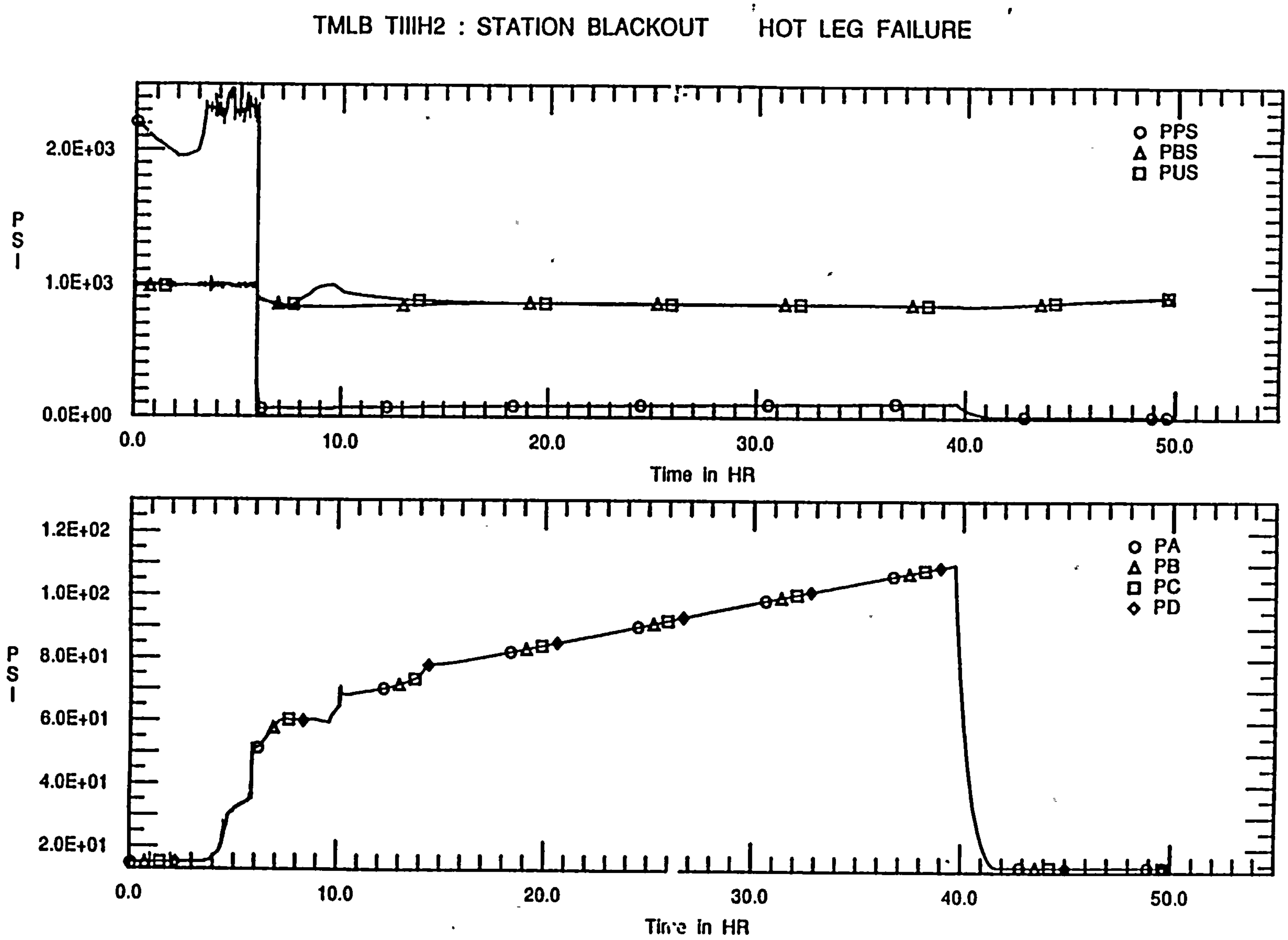

Figure F-13 Pressure Variations for PDS IIIH (Case TIIIH2)

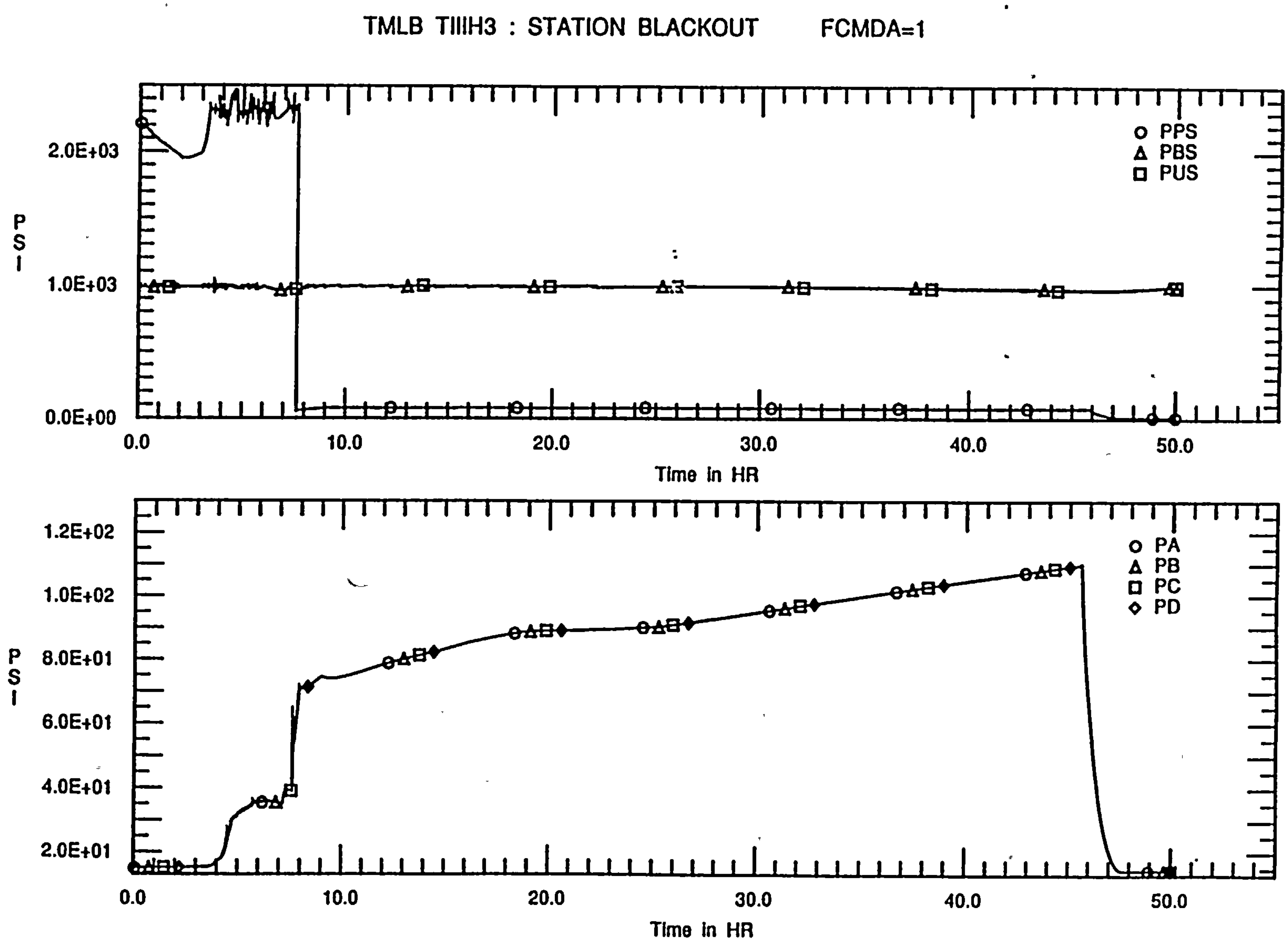

Figure F-14 Pressure Variations for PDS IIIH (Case TIIIH3)

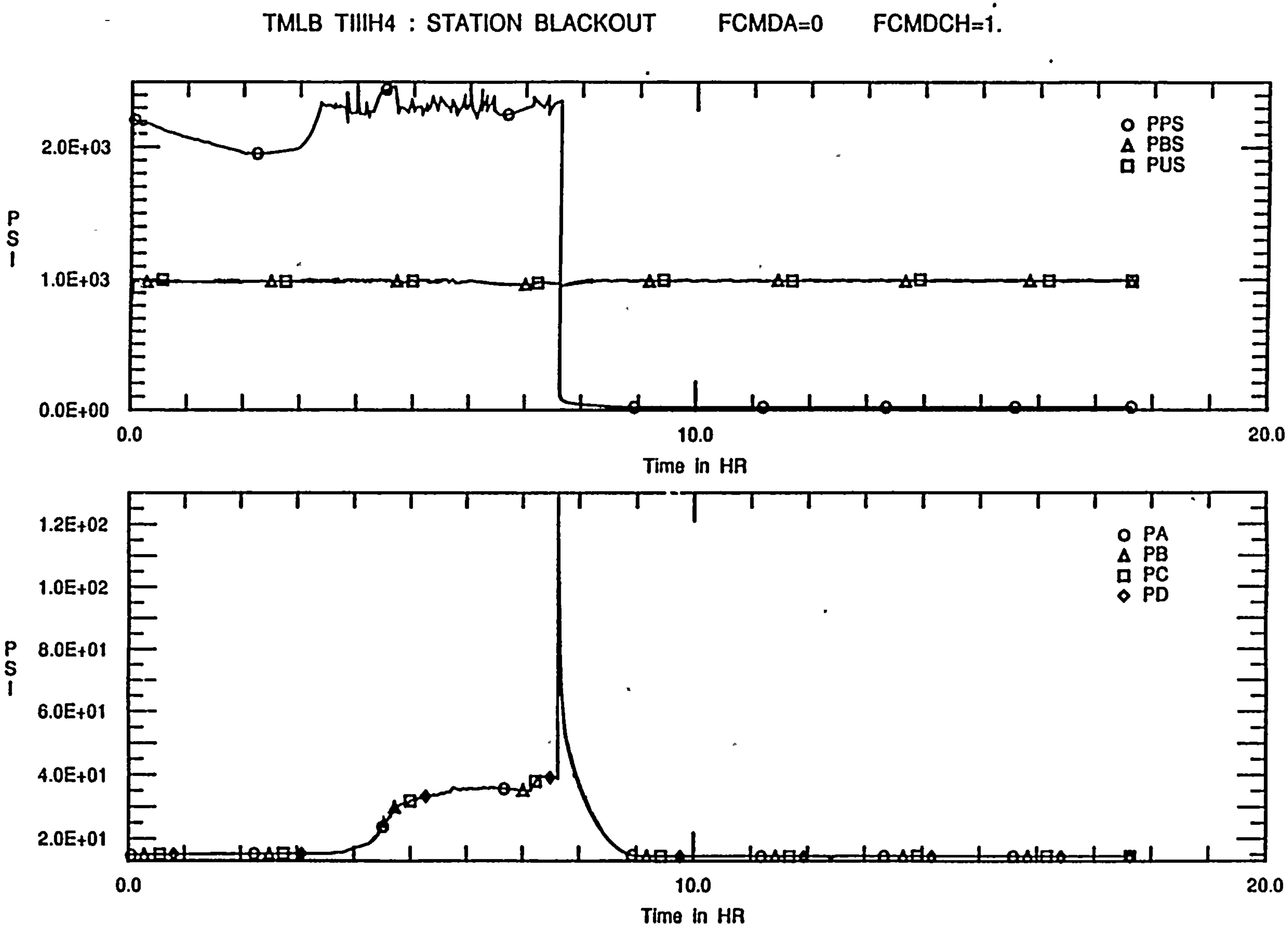

Figure F-15 Pressure Variations for PDS IIIH (Case TIIIH4)

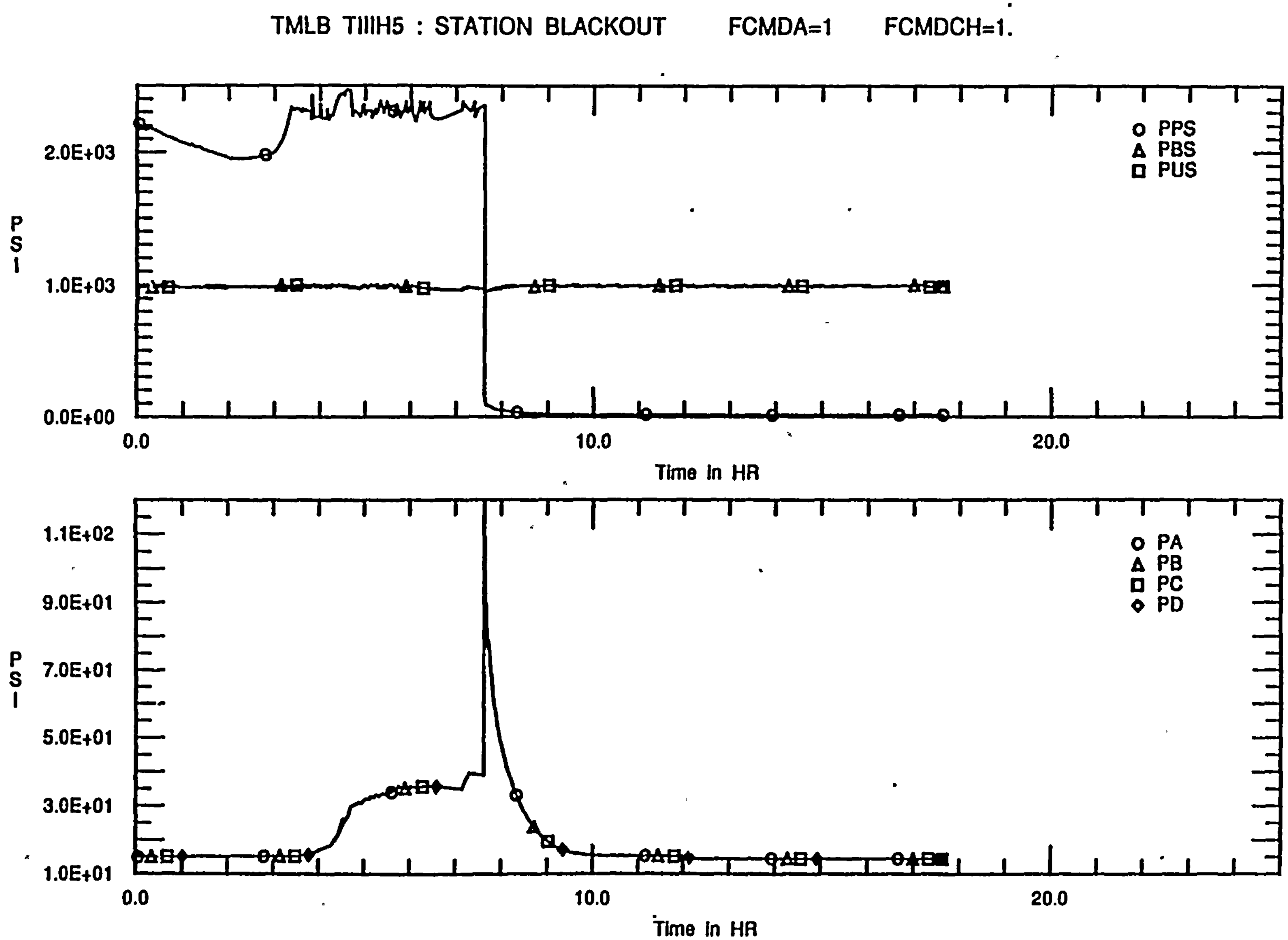

Figure F-16 Pressure Variations for PDS IIIH (Case TIIIH5)

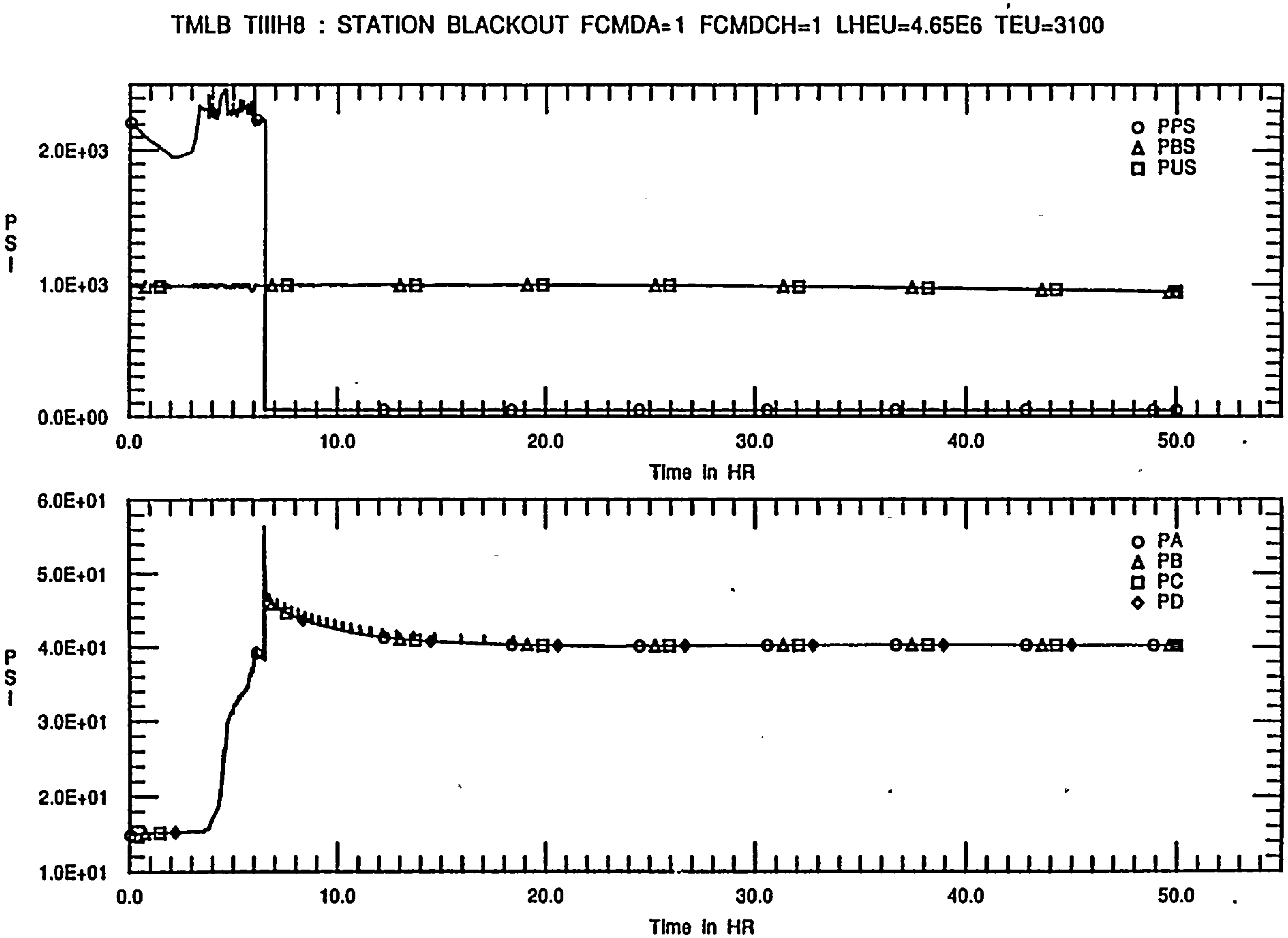

Figure F-17 Pressure Variations for PDS IIIH (Case TIIIH8)

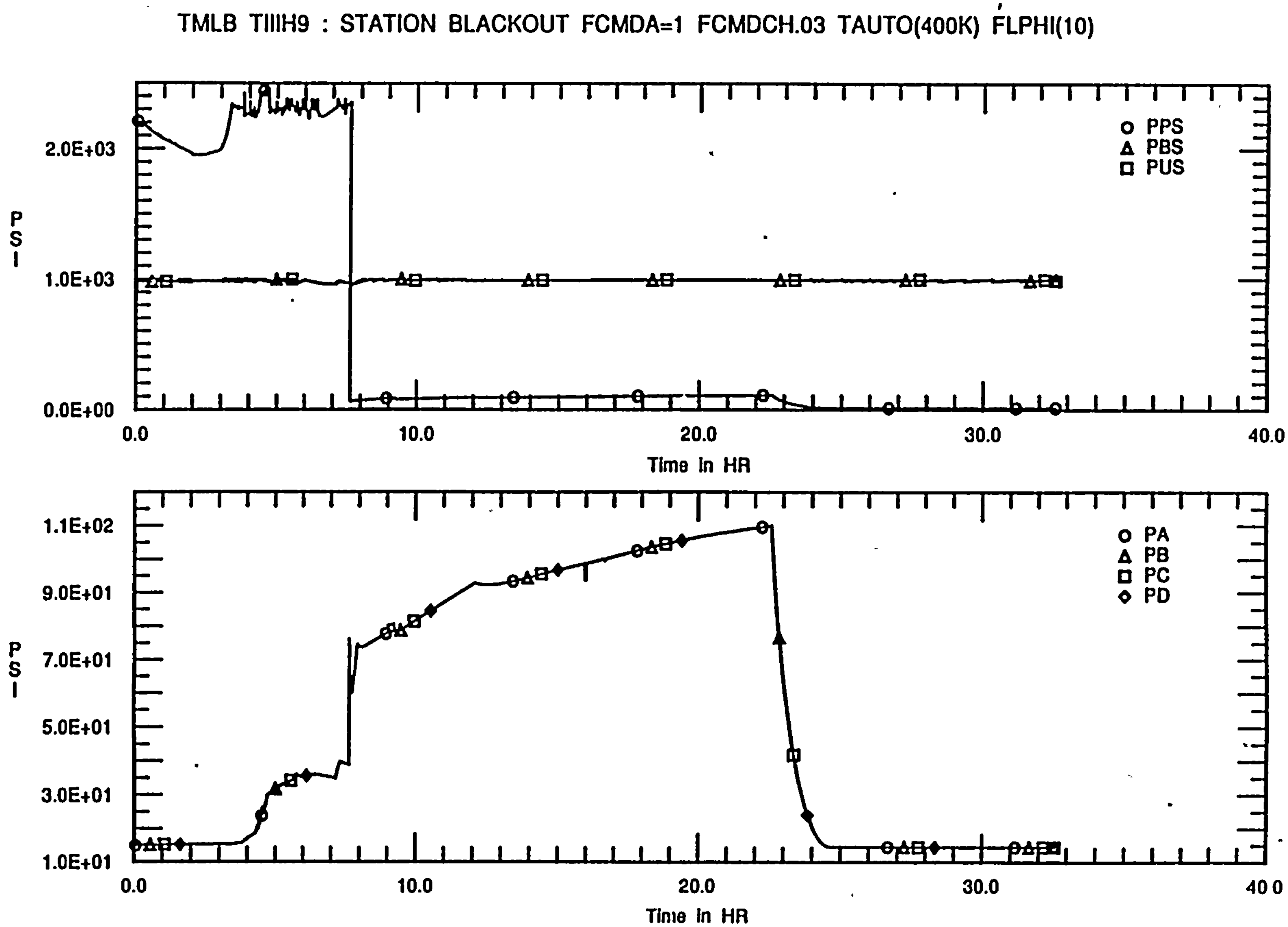

Figure F-18 Pressure Variations for PDS IIIH (Case TIIIH9)

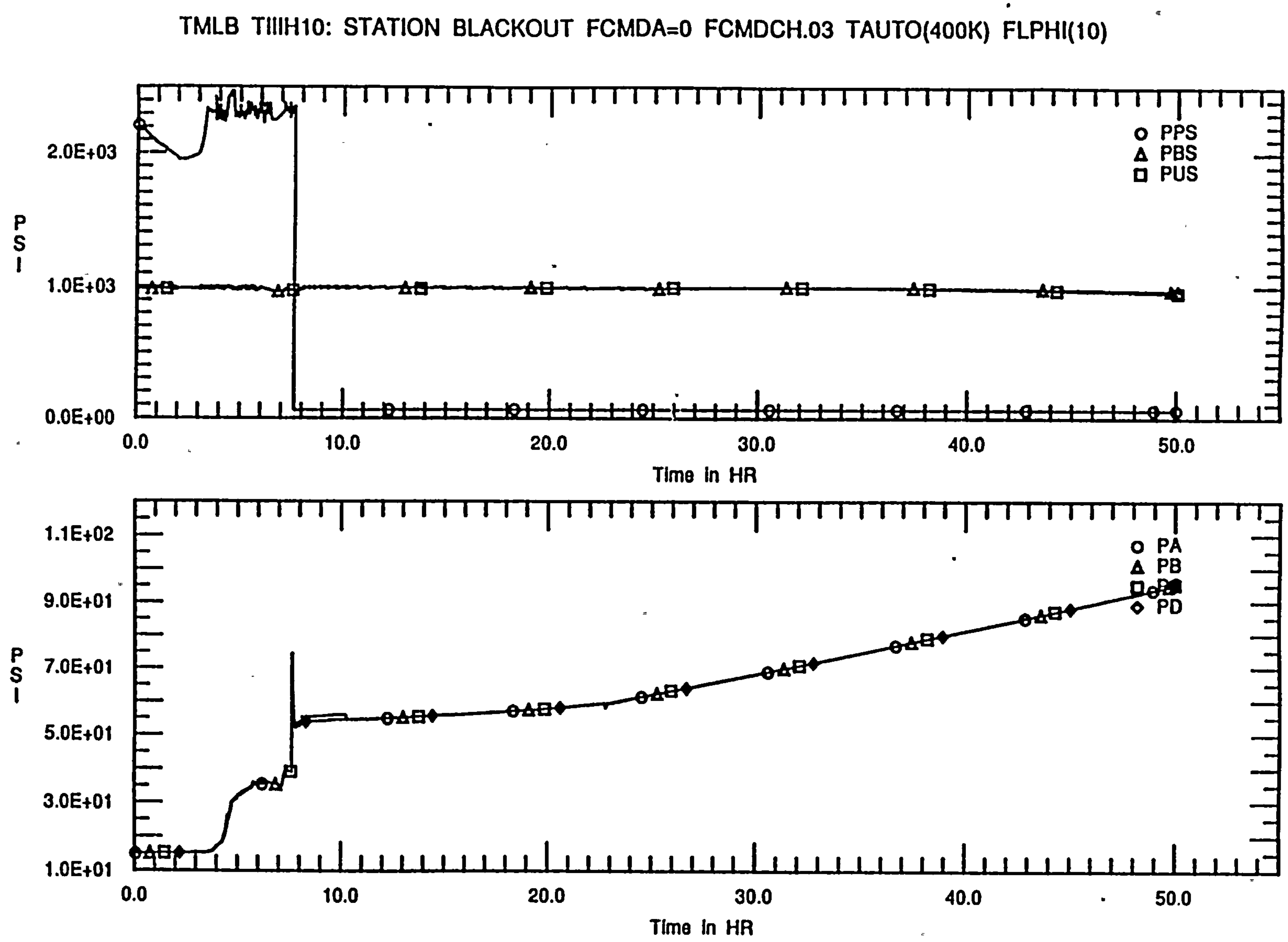

Figure F-19 Pressure Variations for PDS IIIH (Case TIIIH10)

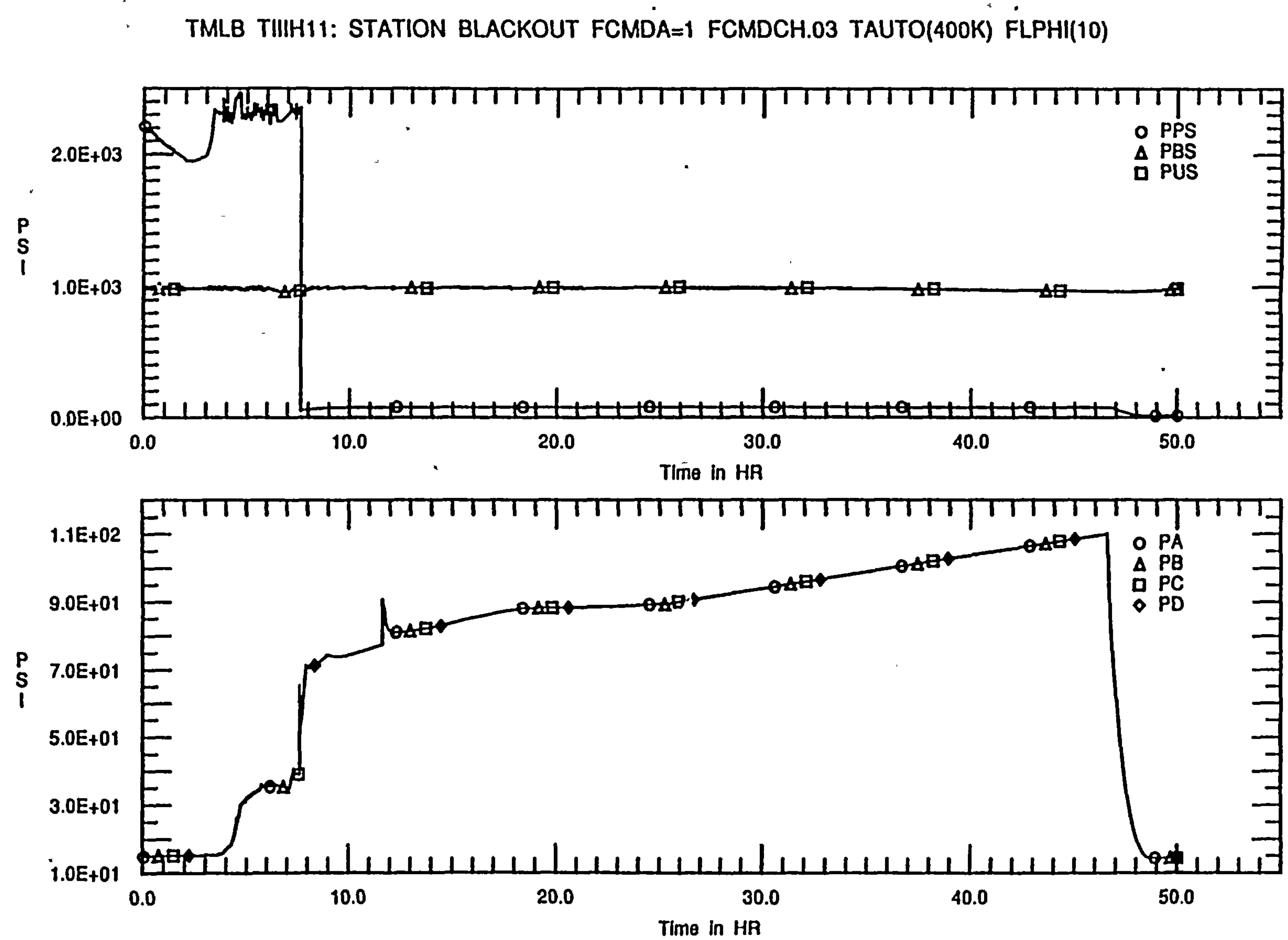

Figure F-20 Pressure Variations for PDS IIIH (Case TIIIH11)

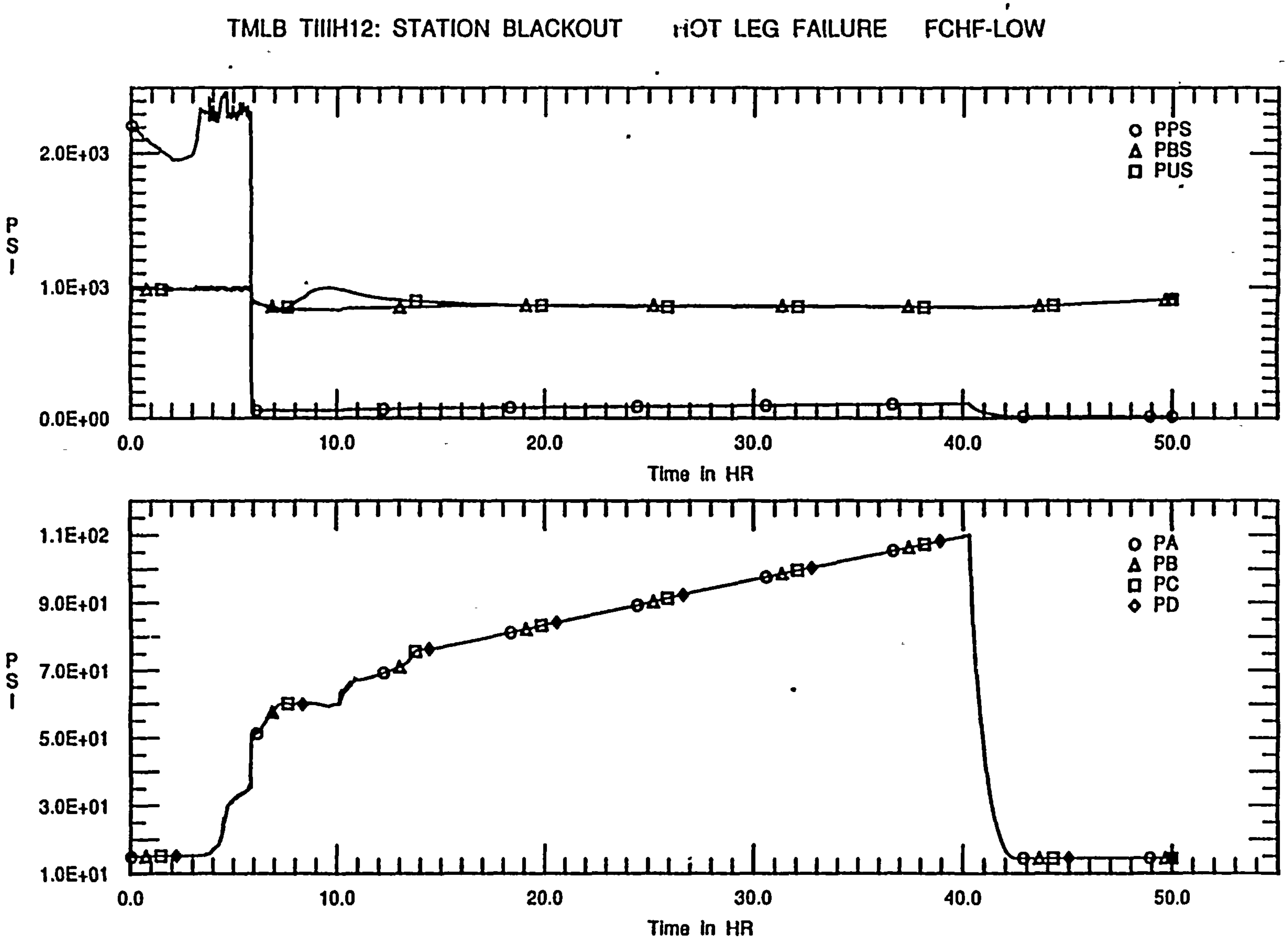

Figure F-21 Pressure Variations for PDS IIIH (Case TIIIH12)

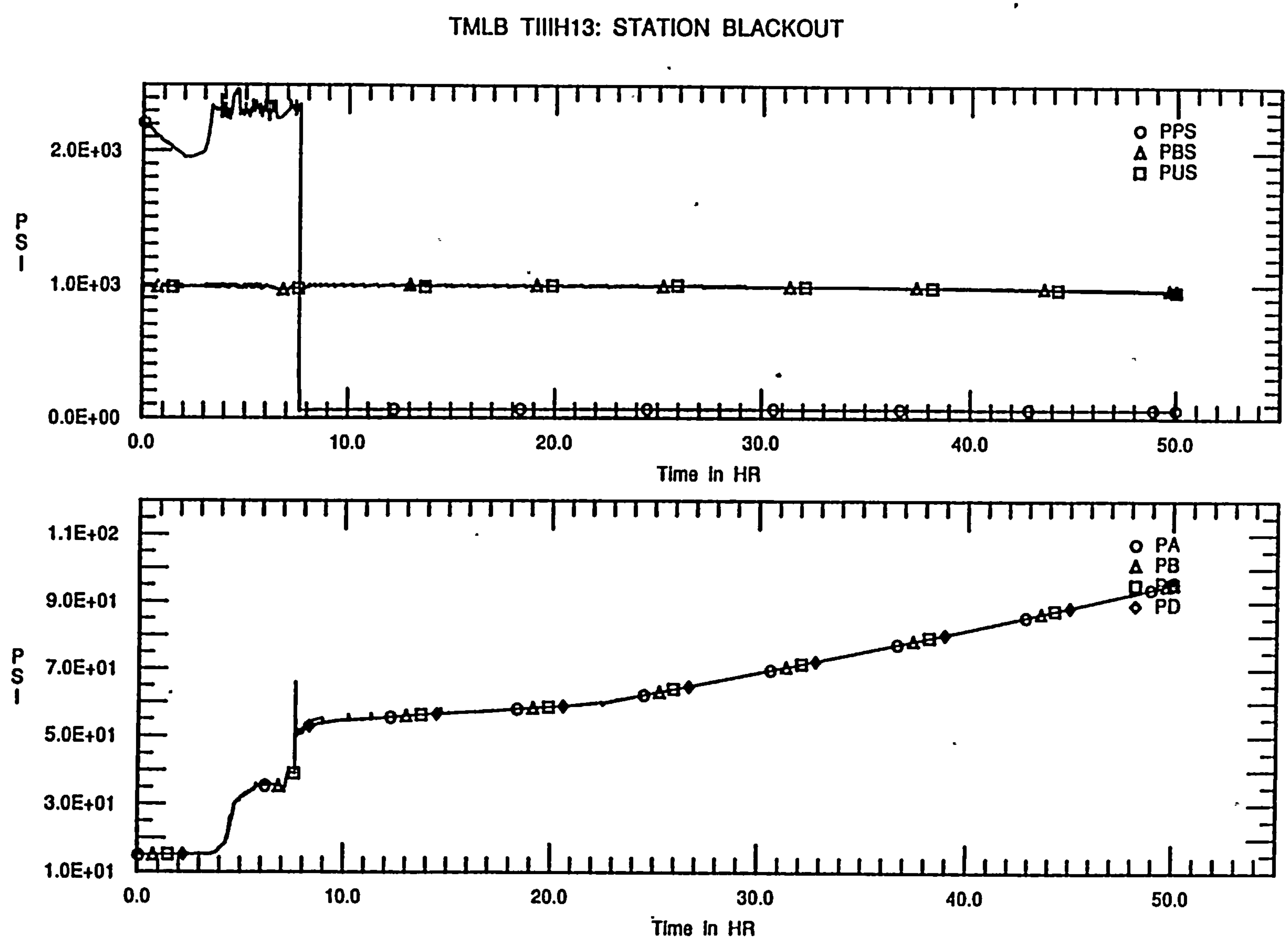

Figure F-22 Pressure Variations for PDS IIIH (Case TIIIH13)

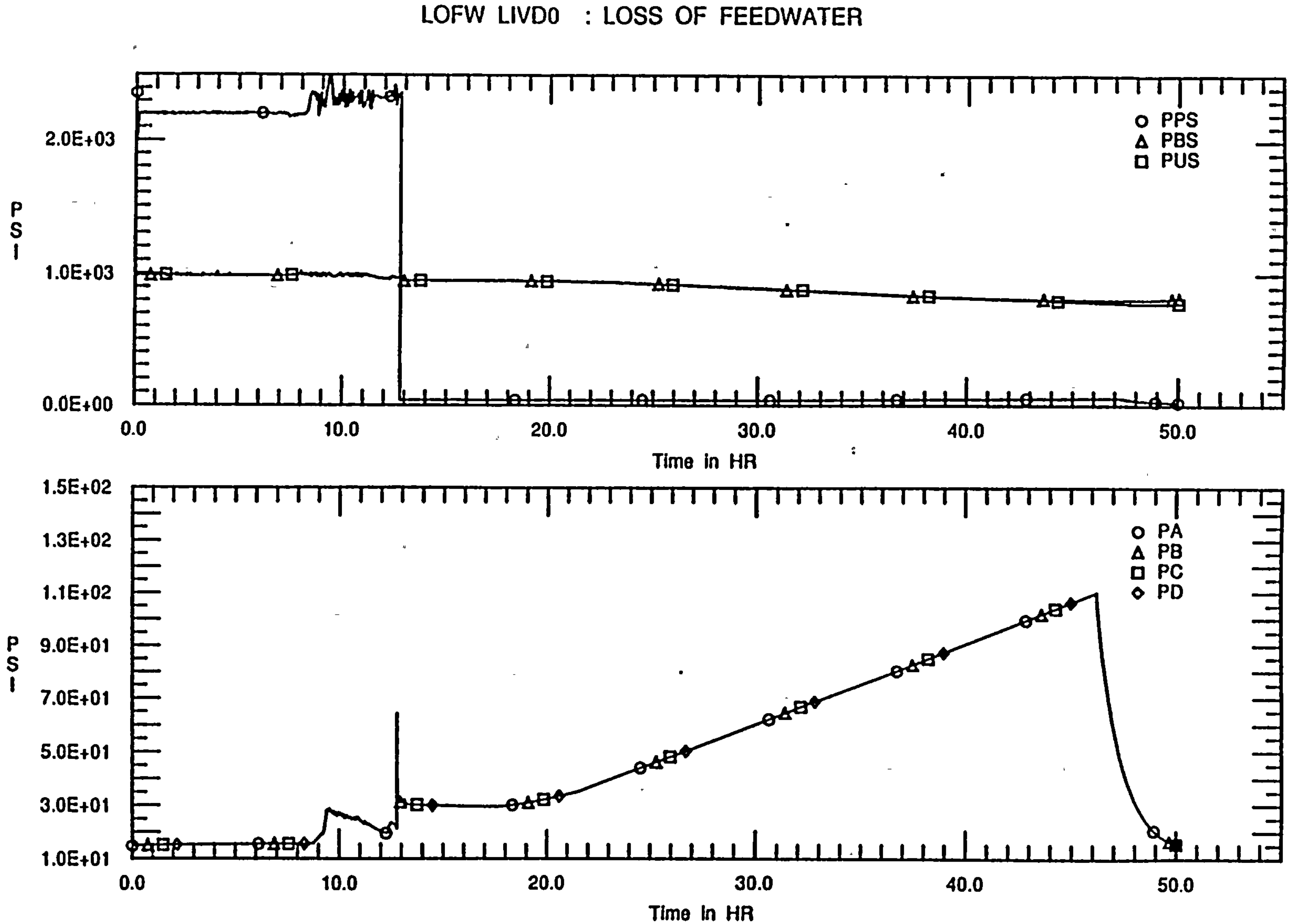

Figure F-23 Pressure Variations for PDS IVD (Case LIVD0)

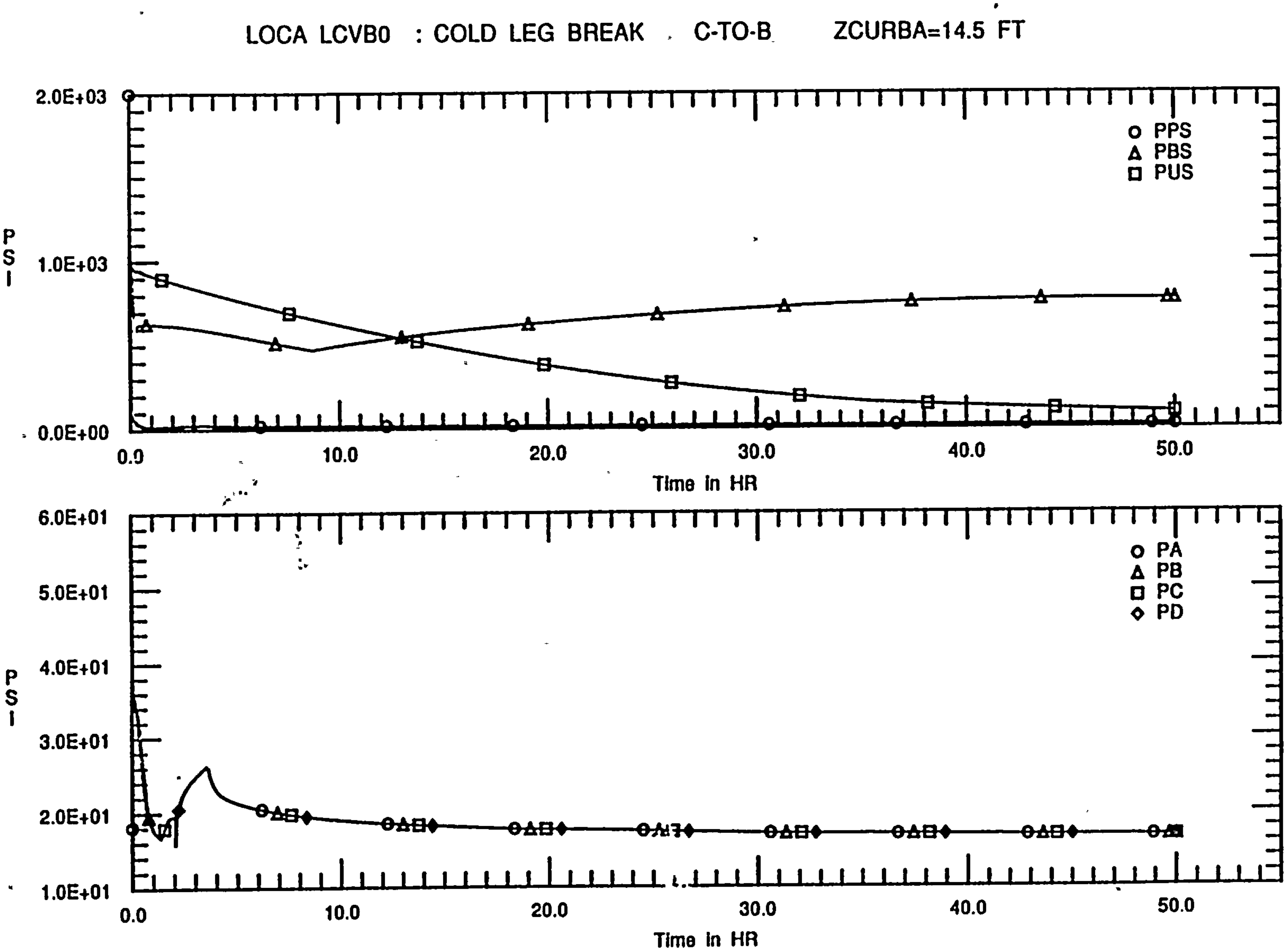

Figure F-24 Pressure Variations for PDS VB (Case LCVB0)

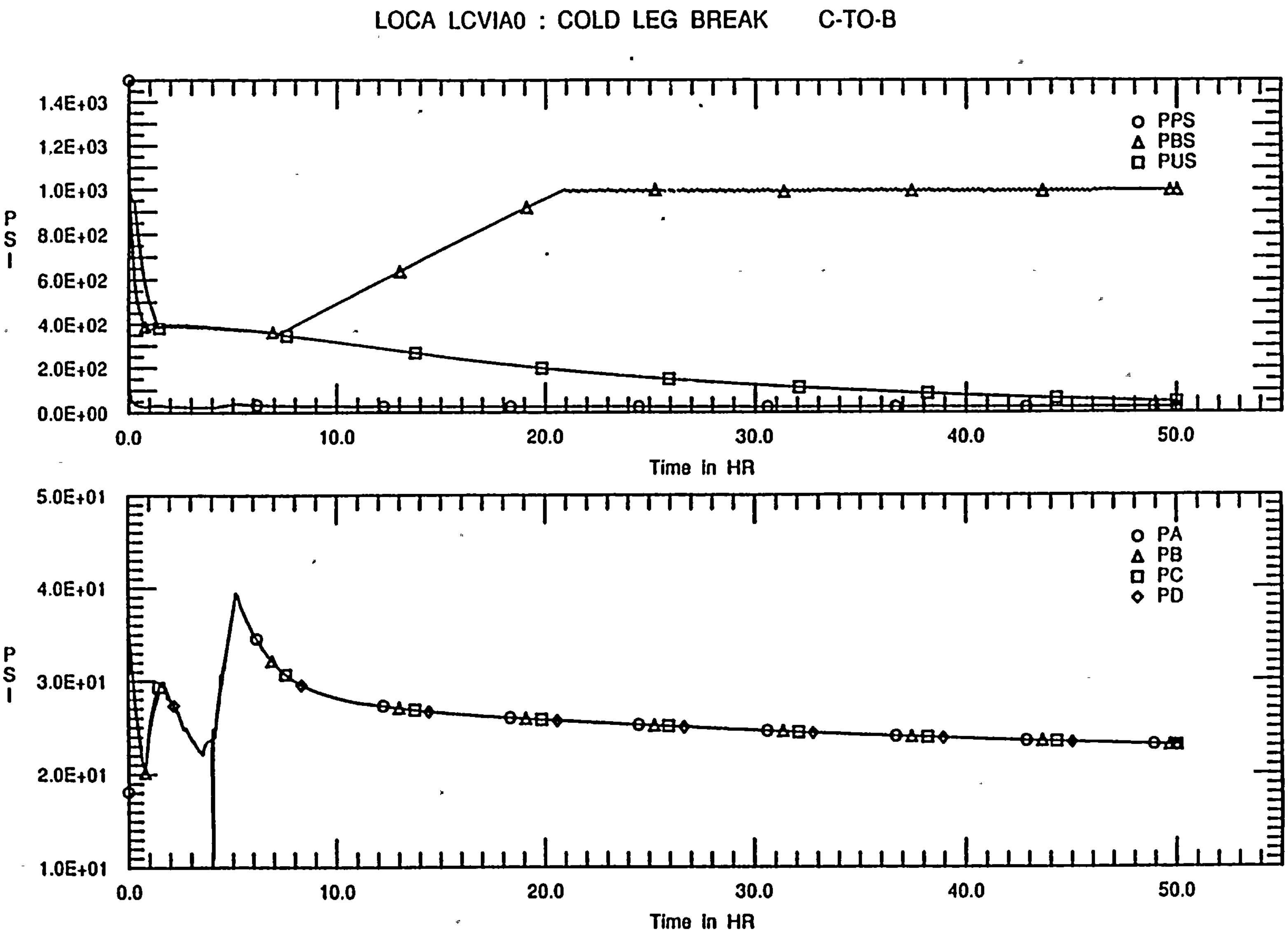

Figure F-25 Pressure Variations for PDS VIA (Case LCVIA0)

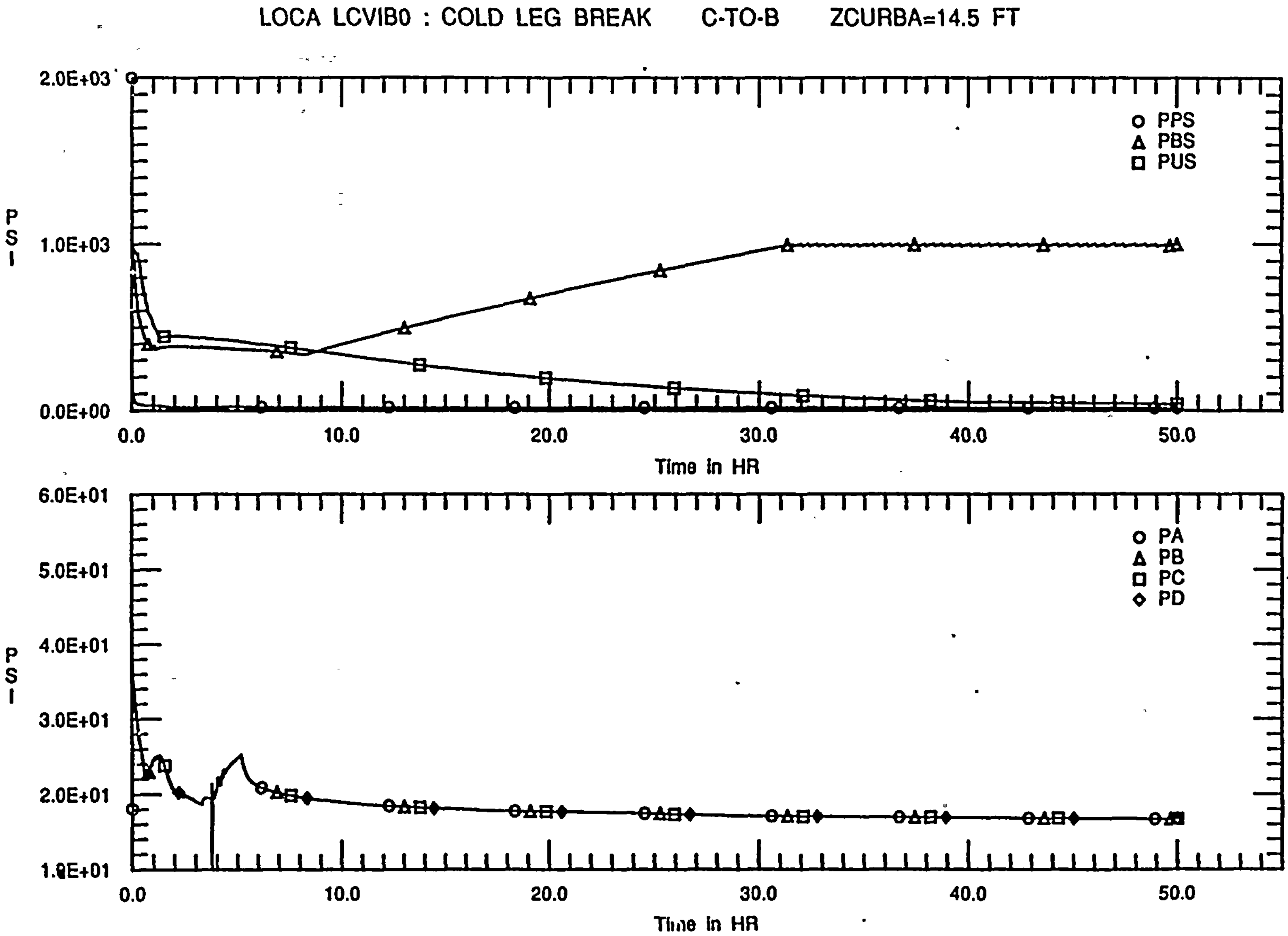

Figure F-26 Pressure Variations for PDS VIB (Case LCVIB0)

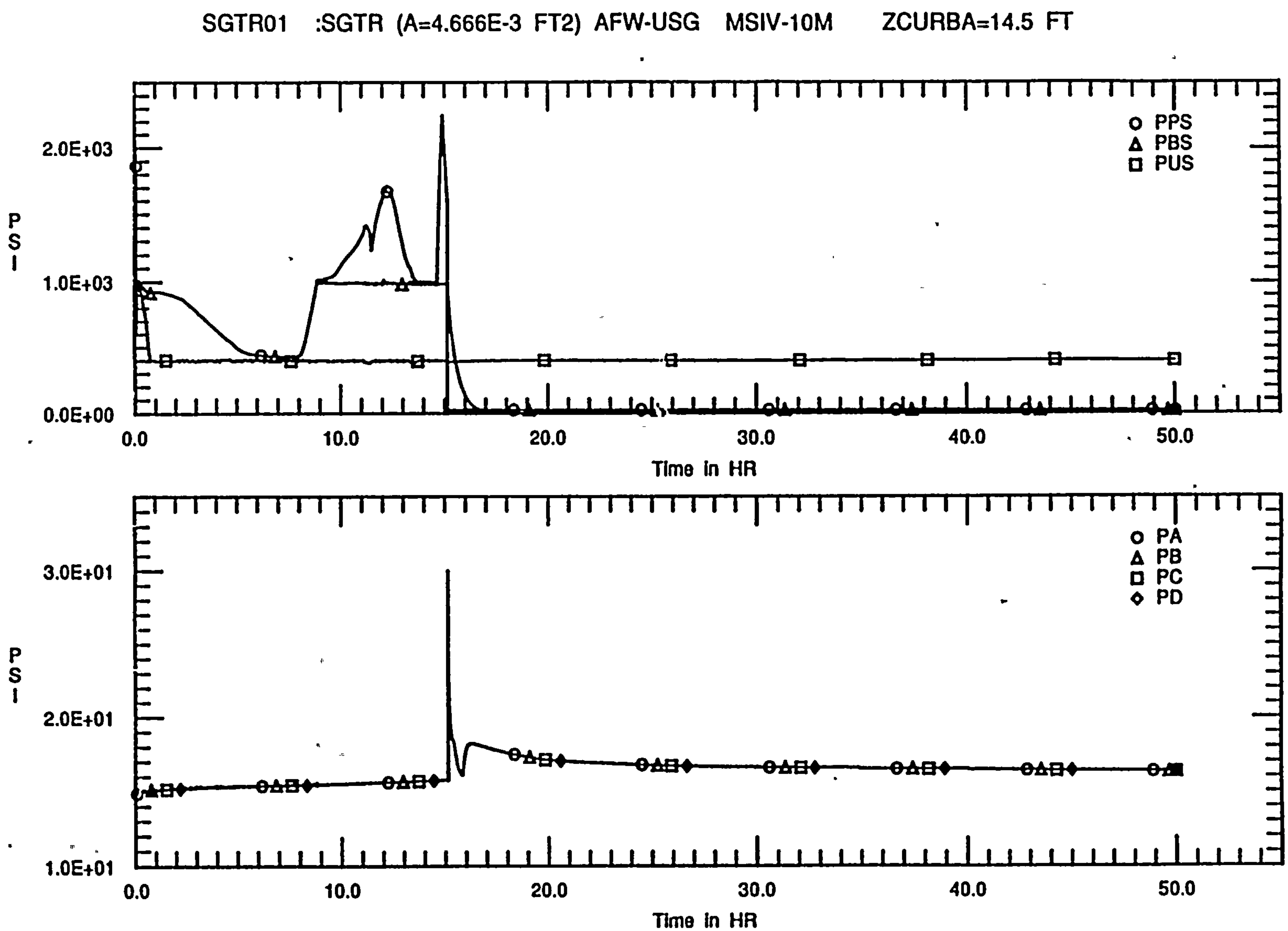

Figure F-27 Pressure Variations for SGTR (Case SGTR01)

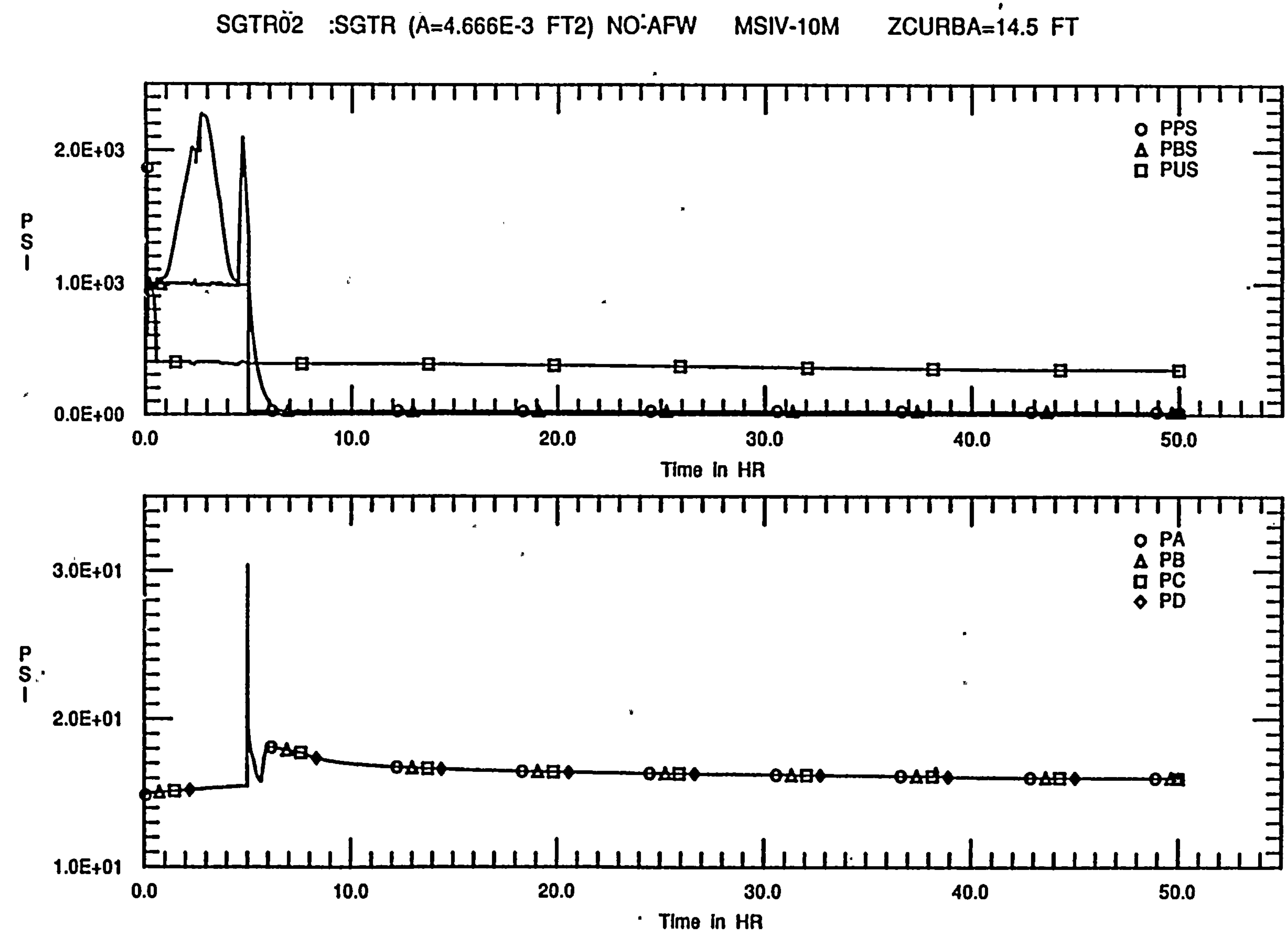

Figure F-28 Pressure Variations for SGTR (Case SGTR02)

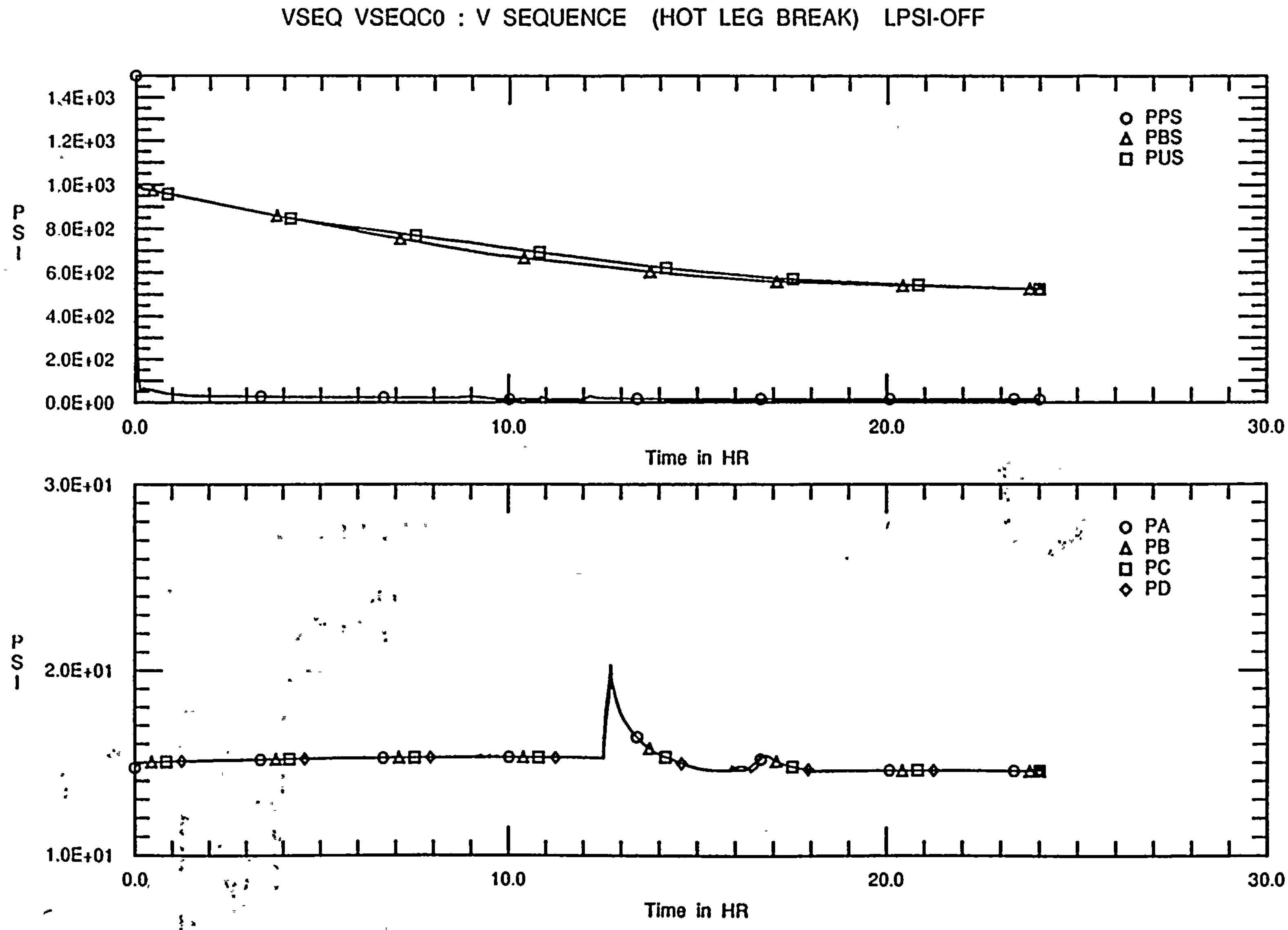

Figure F-29 Pressure Variations for V Sequence (Case VSEQC0)

APPENDIX G

ST. LUCIE UNITS 1 & 2

CONTAINMENT FAILURE PRESSURE CHARACTERIZATION

TABLE OF CONTENTS

G.1 INTRODUCTION

This appendix describes the St. Lucie Units 1 and 2 containment failure characterization to support the back-end analysis. The purpose of the containment in a nuclear power plant is to prevent the release of radioactivity which may be present in the containment atmosphere as a result of core degradation. The performance limits or failure pressure of the containment determines the containment behavior under internal pressure as would be expected during severe accidents. The assessment of the containment failure for the St. Lucie Units 1 and 2 PRA includes a set of hand calculations to determine the ultimate structural capability and comparisons against existing studies on the structural capability of similar containments. The former is a simple force balance equation intended to improve upon the latter approach of scaling to determine the failure pressure relative to the design pressure (using the results obtained from similar containment designs).

G.2 CONTAINMENT FAILURE OVERVIEW

Containments subject to static internal pressure loads can fail in a number of failure modes. Failure, in risk assessments is defined as the point at which leakage occurs, and the containment can no longer perform its intended function. The failure of the containment pressure boundary in any one of these locations can permit leakage:

- Cylindrical vessel
- Floor liner junction
- Penetrations and reinforced openings
- Equipment hatch or personnel air lock
- Valves, expansion joints, bellows or seals
- Basemat.

The failure modes addressed in this section is related to the global failure of the cylindrical vessel or onset of leakage due to localized tearing at discontinuities such as the penetrations, equipment hatch, and small valve penetrations. It also provides an assessment of the failure probability of the various containment isolation systems.

G.3 CONTAINMENT DESCRIPTION

The containment vessel is an all steel right circular cylinder with a hemispherical dome and an ellipsoidal bottom. The shield building is a reinforced concrete right circular cylinder with a dome roof surrounding the containment vessel. The shield building functions:

1. as a biological shield during normal operation and after any accident within the steel containment;

2. as a low leakage structure;

3. as a shield for the primary steel containment against adverse atmospheric conditions due to tornadoes and hurricane winds, external missiles and flooding.

The containment vessel and shield building are supported by the reactor building base slab. Major dimensions of the containment vessel and shield building are as follows:

DIMENSION	CONTAINMENT VESSEL	SHIELD BUILDING
Inside diameter	140 ft.	148 ft.
Inside height	232 ft.	240 ft.
Vertical wall thickness	2 in.	3 ft.
Dome thickness	1 in.	2.5 ft.
Internal free volume	2.5×10^{6} ft.^{3}	5.43×10^{5} ft.^{3} annulus
Foundation thickness	10 ft.	10 ft. [12 ft.]

Access to containment is provided by an equipment hatch and two personnel air locks. The large maintenance hatch is welded shut. Provision has been made for seventy mechanical penetrations of the containment vessel. Figure G-1 shows containment penetrations for the plant.

The internal structures of the containment consist of concrete and steel components. The major concrete structures are the primary and secondary shield walls, refueling cavity, operating deck and the enclosures around the pressurizer and steam generators. The major steel components are the RCS supports and refueling cavity liner. The internal structures are supported on the concrete floor fill that is placed in the bottom of the steel containment. Figure G-2 shows the general arrangement plan for structures within the containment. Figure G-3 shows St. Lucie containment building and major components inside.

G.4 CONTAINMENT VESSEL STRUCTURAL FAILURE LIMIT

The St. Lucie containment, as described above is a free-standing carbon steel shell. With the exception of the embedment at the base, the stresses in the cylindrical shell and dome due to internal pressure are essentially membrane. Analysis methods relating to global stress and strain conditions in the containment structural elements are contained in "Technical Report 10.1, Containment Structural Capability of Light Water Nuclear Power Plants" (IDCOR, 1983) [Ref. G-1]. NUREG/CR-2442 [Ref. G-2] was used as the basis for the containment strength and containment failure pressure estimates.

The St. Lucie containment is designed as an essentially leak-tight pressure vessel under design pressure conditions, with a design pressure of 44 psig. Similar containments have demonstrated structural failure limits in the range of 2.2 to 8.8 times the design pressure as summarized in Table

G-1. Primary internal stresses due to pressure consist of membrane hoop stresses. The hoop force generated by the internal pressure load may be calculated as a statically determinate problem with simple hand calculations.

The following equation and terminology for the pressure ratio at yield is contained in an analysis of a similar structure contained in "Containment Analysis Techniques A State-of-the-Art Summary" (NUREG/CR-3653) [Ref. G-3]. The values reflect the St. Lucie containment being analyzed as described in NUREG/CR-2442.

The equilibrium condition is:

$$P = \frac{f_y A}{R_i}$$

where:

P = Internal pressure at defined failure criterion of 2 times the yield strain.

R_i = Containment inside radius

f_y = 110% of steel shell yield strength

A = Steel shell area (per foot of circumference)

In this equation, the inside radius is used; in reality R may be defined as the average of the inside and outside radius of the containment.

For the St. Lucie containment, the following input data are used:

R_i = 70 ft.

f_y = 41.8 ksi (110% yield strength of SA 516 Gr 70 Steel).

A = 1.918" X 12"/ft. = 23.02 in^2/ft

Substituting the numerical values selected to represent the St. Lucie containment:

$$P = \frac{41.8 \ X \ (23) \ X \ 1000}{70 \ X \ 144 \ inch^2/ft^2}$$

P = 95.4 psig

The sources of each value specifically based on the St. Lucie containment structure are identified below:

f_y Yield strength value was taken from NUREG/CR-2442 material property for St. Lucie.

A The area for the containment wall is based on a 1.918" thick shell and a 70 ft. radius.

Based on the above calculation, the design pressure of containment does not play a role in the determination of the containment failure pressure. Only the type of steel used, the thickness of the steel plate, and the radius of the containment vessel affect the containment failure pressure.

The study [Ref. G-2] on which St. Lucie containment failure pressure is based on conducted a comparison of the simplified method presented above and the finite element techniques. Results indicated good comparison when applied to axisymmetric, stiffened shells. The penetrations of the containment are reinforced according to ASME requirements insuring a higher resistance than that for the unpenetrated shell. Failure in the vessel penetration intersection or in the penetration wall is certainly possible, but it is assumed to be much less probable than the basic shell modes. Hence, the shell modes were the only failure modes considered in this work.

G.5 PHENOMENA-RELATED FAILURE MODES

The failure modes considered in the assessment of the containment's response to severe accident progression loads are consistent with the failure modes provided in Table G-2 (derived from Table 2.2 of NUREG-1335). The evaluation of the potential failure modes induced by phenomena requires the determination of the pressure and thermal loads in containment at key time points in the accident (e.g., at vessel breach). These are included in the logic trees that support the determination of the probability of containment failure in the CET. The pressure load that challenges containment integrity is obtained as the sum of the containment pressure prior to the occurrent of the phenomenon (base pressure) and the pressure rise as DCH (for early containment failures), and noncondensible (and combustible) gas generation from concrete attach (for late containment failures). Other failure causes, such as basemat melt-through or liner melt-through from molten debris impingement, are addressed by examining the plant configuration relative to existing analyses such as those conducted in NUREG-1150 and IDCOR integrated analyses. Important mitigating features of the St. Lucie containment configuration are incorporated for plant-specific evaluation. For example, the very thick concrete foundation and basement would preclude basemat melt-through from core-concrete interactions. Additionally, the isolated configuration of the reactor cavity would likely preclude direct impingement of the containment shell by molten corium. These features are reflected in the development and quantification of the probability of containment failure.

G.6 SUMMARY

For purposes of the containment performance evaluation, failure of the St. Lucie containment is postulated to occur at a pressure of 95 psig. A range of failure pressures consistent with the assessed failure limits from reference studies is used to determine the sensitivity of the results. Although uncertainty in the failure pressure is broad, loss of containment integrity is considered more likely at the lower range. However, the failure characterization conducted would not support such details in the analysis. Although several failure locations are possible, only two distinct selected failure locations are addressed: pre-existing containment isolation failure and interfacing LOCA (containment bypass). Overpressure failure is assumed to occur at the springline.

References

G-1. Technology for Energy Corporation, *Technical Report 10.1, Containment Structural Capability of Light Water Nuclear Power Plants*, IDCOR (Industry Degraded Core Rulemaking Program) Atomic Industrial Forum, Inc., Bethesda, MD, July 1983.

G-2. NUREG/CR-2442, *Reliability Analysis of Steel Containment Strength*, Ames Laboratory, Iowa State University, June 1982.

G-3. U.S. Nuclear Regulatory Commission, *Containment Analysis Techniques: A State-of-the-Art Summary*, by Sandia National Laboratories, NUREG/CR-3653, April 1984.

Table G-1

VARIOUS CONTAINMENT STRENGTHS

Plant	Type of Containment	Free Volume	I.D.	Design Pressure	Analysis	Failure Pressure	Ratio of Failure to Design Pressure	Dominant Failure
Zion	Large Dry; concrete cylinder w/steel liner (prestressed)	2.86E6 ft^3	141 ft.	47 psig	NUREG-1150 IDCOR 10.1	108-180 psig 149 psig	2.3-3.8 3.17	Leak/rupture in cylinder wall or basemat/wall intersection Hoop/tendon strain
Surry	Sub-atmospheric; concrete with steel liner (reinforced)	1.3E6 ft^3	126 ft.	45 psig	NUREG-1150	95-155 psig	2.1-3.5	Leak/rupture near dome/wall intersection
Indian Point	Large Dry; concrete cylinder with steel liner (reinforced)	2.61E6 ft^3	135 ft.	47 psig	IDCOR 10.1	126 psig	2.7	Hoop rebar yield/cylinder shell near spring line
Seabrook	Large Dry; concrete cylinder with steel liner (pre-stressed)	2.76E6 ft^3	140 ft.	60 psig	Seabrook Risk Management Study 1985	211 psig (wet seq.) 190 psig (dry seq.)	3.5 3.1	Feedwater penetration
Oconee Unit 3	Large Dry; concrete cylinder with steel liner (prestressed)	1.91E6 ft^3	116 ft.	59 psig	Oconee PRA	162.5 psig	2.7	2.5 x design pressure, rupture of prestress tendons and liner failure
Yankee Rowe	Large Dry; bare steel sphere	1.02E6 ft^3	125 ft.	34 psig	IDCOR 10.1	84 psig	2.5	Hoop yield/steel sphere
St. Lucie - 1[2]	Large Dry; steel cylinder	2.50E6 ft^3	140 ft.	[44 psig]	NUREG/CR-4870	95 psig [95 psig]	[2.2]	Twice Yield Strain
Sequoyah - 1	Ice condenser with steel cylinder	1.26E6 ft^3	115 ft.	10.8 psig	NUREG-1150 IDCOR 10.1 NUREG/CR-4870	40-95 psig 58 psig 60 psig	3.7-8.8 5.4 5.6	Gross rupture in the containment or rupture in the lower compartment Hoop Yield Twice Yield Stress
McGuire - 1	Ice condenser with concrete cylinder and steel liner		115 ft.	28 psig	NUREG/CR-4870	84 psig	3	Twice Yield Stress
Watts Bar - 1	Ice condenser with steel cylinder		115 ft.	15 psig	NUREG/CR-4870	98 psig	6.5	Twice Yield Stress
Turkey Point	Large Dry: Concrete cylinder with steel liner (prestressed)	1.55E6	116 ft.	59 psig	PTN IPE Submittal	146 psig	2.47	Liner Tearing

Figure G-1 Containment Penetrations

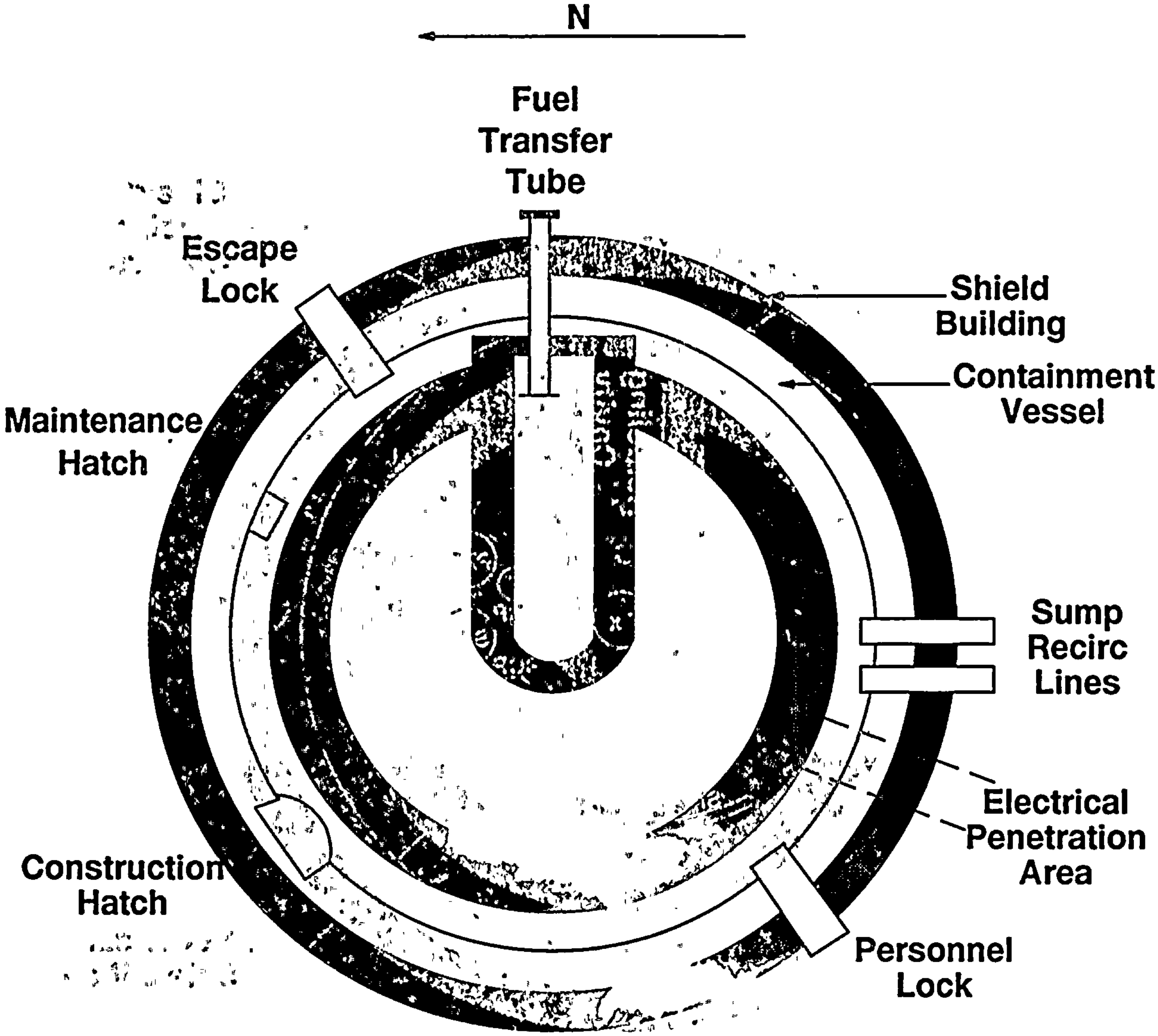

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Table G-2

POTENTIAL CONTAINMENT FAILURE MODES AND MECHANISMS

Failure Mode	Comment
Direct bypass	Systems-related
Failure to isolate	Systems-related
Vapor explosions Missile Generation Quasi-static pressure rise	Phenomena-related Subject to large uncertainties
Overpressurization Steam Non-condensible gases	Phenomena-related Estimated for plant
Combustion processes (hydrogen, carbon monoxide, methane) Blast Quasi-static pressure rise	Phenomena-related Estimated for plant
Core-concrete interaction Basemat penetration Structural failure and tear-out of penetrations	Phenomenal-related Influenced by plant design (e.g., concrete type)
Blowdown forces Vessel thrust force	Phenomena-related Subject to uncertainties
Melt-through Direct contact of containment shell with fuel debris	Phenomena-related
Thermal attach of containment penetrations	Phenomena-related Influenced by plant design (e.g., seal materials)

Source: NUREG-1335, Table 2-2

Figure G-2: Internal Structures Arrangement Plan

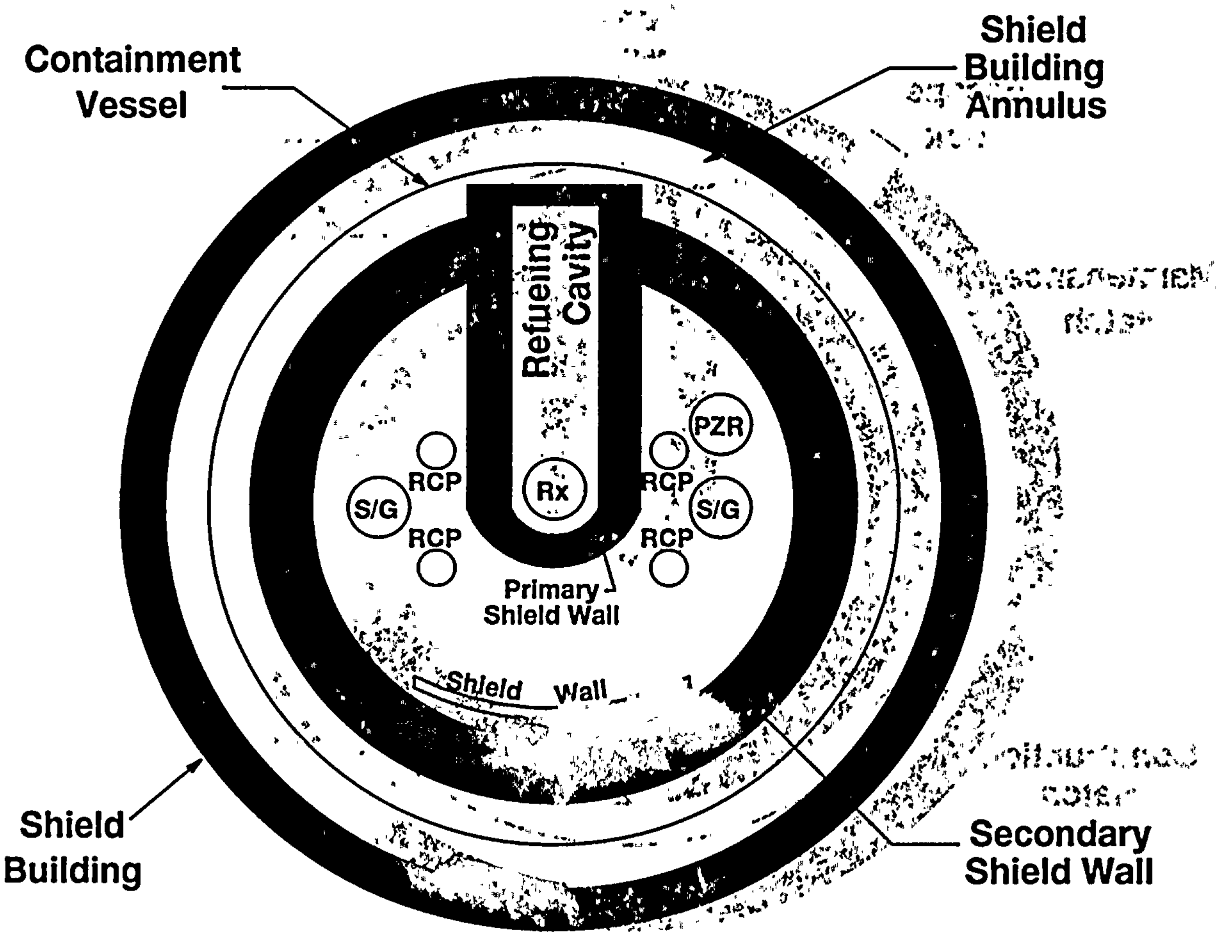

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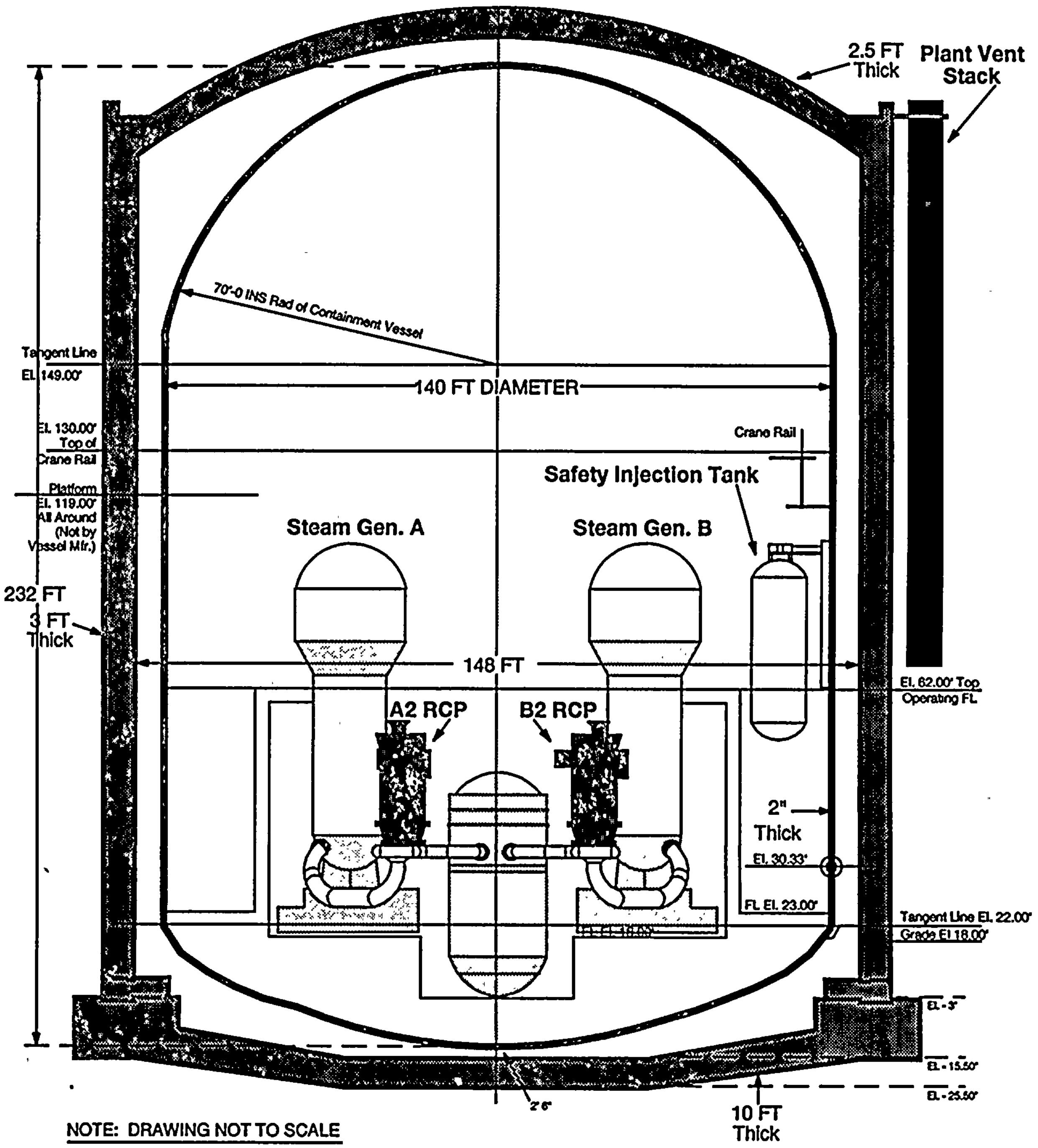
Figure G-3 St. Lucie Containment Building



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