DONALD C. COOK NUCLEAR PLANT
UNITS 1 AND 2

INDIVIDUAL PLANT EXAMINATION
SUMMARY REPORT

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TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd.)

LIST OF TABLES

LIST OF TABLES (Cont'd.)

LIST OF FIGURES

LIST OF FIGURES (Cont'd.)

1.0 EXECUTIVE SUMMARY

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

In August 1989, the NRC issued Supplement 1 to Generic Letter 88-20, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," accompanied by NUREG-1335, "Individual Plant Examination Guidance," (Reference 20) which provided guidance for the information to be reported back to the NRC. In July 1990, the NRC issued Supplement 3 to Generic Letter 88-20, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use In the Individual Plant Examination for Severe Accident Vulnerabilities," which stated that licensees with ice condenser containments are expected to evaluate the vulnerability to interruption of power to the hydrogen igniters as part of the IPE.

This report provides the requested information for the Donald C. Cook Nuclear Plant, Units 1 and 2.

1.1 Background and Objectives

In its Severe Accident Policy Statement (50FR43621, Reference 32) issued in 1985, the NRC concluded that operating nuclear plants pose no undue risk to the public health and safety. However, recognizing that these generic conclusions were derived from a diverse but smaller sample of the existing plants, the NRC requested that all licensees perform a "limited-scope accident safety analysis" to determine if there might be any unique plant-specific vulnerabilities leading to a core damage accident or to poor containment performance given a core damage event.

In responding to Generic Letter 88-20 and its Supplement 1, American Electric Power Service Corporation (AEPSC) established five specific objectives for the severe accident issue resolution program for the Cook Nuclear Plant. The program objectives were:

1. To identify, evaluate, and resolve the severe accident issues germane to the Cook Nuclear Plant in a realistic, technically acceptable manner with emphasis on the prevention of such accidents.

2. To identify and develop input to decision making processes relative to potential enhancements to plant design and/or operation aimed at reduction of risk from severe accidents.

3. To evolve a realistic, documented, auditable Probabilistic Risk Assessment (PRA) for the Cook Nuclear Plant which could be readily used and maintained and which would be suitable for ongoing use.

4. To address the existing NRC information request in Generic Letter 88-20 and those information requests anticipated in the near future on closely related topics.

5. To familiarize AEPSC and Cook Nuclear Plant staffs with the basis and methodology of PRA so that, in the future, the PRA could be independently maintained and updated as necessary.

AEPSC has completed and documented a full scope Level III PRA for the Cook Nuclear Plant that has completely met these objectives. This report, containing a summary of the methods, results, and conclusions, fully complies with the NRC request for information contained in Generic Letter 88-20, Supplement 1. In addition, the entire PRA was conducted according to the applicable sections of 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." AEPSC has retained all supporting analyses, descriptions and files pertaining to the PRA. These are available at AEPSC offices for NRC review as necessary.

As a result of AEPSC's decision to perform a Level III PRA including an External Events Analysis, the scope of the AEPSC PRA program goes beyond that described in Generic Letter 88-20. To assist NRC reviewers, this submittal is structured to contain the Internal Events Analysis (often referred to as a Level I PRA), the Containment Performance Analysis (referred to as a Level II PRA), and the Consequence Analysis (a Level III PRA). The External Events Analysis, including internal fire, seismic PRA, and other external events analysis are described in separate submittal documents.

1.2 Plant Familiarization

The AEPSC PRA program for the Cook Nuclear Plant involved an extensive plant familiarization effort because the undertaking of a full-scope realistic Level III PRA required careful analysis of the as-built, as-operated plant. To accomplish this familiarization, several data collection and documentation activities were undertaken during the initial phase of the project. System notebooks were prepared for modelled systems after plant walkdowns and analyst review of drawings, system descriptions, the Updated Final Safety Analysis Report (UFSAR, Reference 1), technical specifications (References 2 and 3) and applicable plant procedures. The plant walkdowns were conducted to verify the design of the systems, to become familiar with the physical layout of the plant and to visualize restorative actions or alternative systems. Plant records were reviewed to develop plant specific behavioral characteristics such as component failure rates and initiating event frequencies. Also, as part of the familiarization effort, the analysts identified any differences between the systems for Units 1 and 2. Only Unit 1 was specifically modelled as the base analysis.

1.3 Overall Methodology

The AEPSC IPE was performed by conducting a full scope, realistic Level III PRA. In performing the IPE, standard PRA systems analysis practices such as those outlined in the PRA Procedures Guide (NUREG/CR-2300, Reference 26) were used.

The Cook Nuclear Plant IPE is a full scope investigation of the plant systems and operator response. The focus of investigation was on the performance of the realistic assessment of the plant response to potential accident sequences. The models of plant systems are detailed and explicitly include the performance of all key components. The success criteria used to determine whether or not plant systems achieve their intended safety function were realistically determined for each important accident sequence. Rather than relying solely on Reference 1 to set the success criteria for system and operator performed actions, the success criteria definition involved consideration of both system capability and timing. This, in turn, involved the analysis of plant response to a variety of accident scenarios using the Modular Accident Analysis Program (MAAP) code (Reference 13) and other information as well.

Well known approaches for common cause failure and human error were adopted for the Cook Nuclear Plant IPE. In determining the parametric values to be used in the quantification, available industry data bases were scrutinized to assure events and failure modes appropriate for the Cook Nuclear Plant and its equipment were utilized. For common cause analyses, the Multiple Greek Letter (MGL) method was used. Human Reliability Analysis (HRA) was performed using Technique for Human Error Rate Prediction (THERP) (Reference 22) methodology. Realism was achieved through detailed modeling of operator actions and thorough treatment of operator recovery.

Because the AEPSC Cook Nuclear Plant is a dual unit plant, special attention was paid to the consideration of dual unit issues. The interactions of the two Cook Nuclear Plant units' systems were modeled explicitly. In certain cases, conservative assumptions were made in conducting the dual unit analysis in order to provide a perspective of the entire plant's response. The analysis was performed in a manner such that the systems and resources at one unit may play a key role in managing the response to an accident affecting its companion unit.

Special attention was also paid to the interface between the traditional systems analysis and containment systems analysis portions of the IPE. Proper integration of these portions of the analyses was basically accomplished through the establishment of the event trees wherein both system performance and operator

actions that affected Level I and II results were modelled. The tree structures and the use of the MAAP code allowed proper interpretation and assessment of various recovery options. Key success criteria and timing were established with these tools.

1.4 Summary of Major Findings

This section summarizes the major findings of the Cook Nuclear Plant PRA. First, the results of the core damage frequency quantification are presented. Second, the dominant contributors leading to core damage for significant initiating events are described. The containment failure frequency and mechanism are then described. Finally, the offsite dose consequences are reviewed.

The core damage frequency for the Cook Nuclear Plant was found to be 6.26E-5 per year considering internal initiating events, including internal flooding. This value is similar to the results of other PRAs for similarly designed plants (Reference 66.)

For the initiating events with the largest contribution to the core damage frequency, the dominant system failure contributors are described. The order in which the initiators are presented in this section represents their descending contribution to overall core damage frequency.

The initiating event with the largest contribution to core damage frequency at 2.96E-5 per year was found to be Small LOCA (SLO). Failure of the Emergency Core Cooling System (ECCS) during either the cold leg injection or recirculation phases produced the two dominant sequences for this event. In turn, common mode failure of the safety injection (SI) pumps (part of the ECCS) and failure of the Engineered Safety Features (ESF) system to actuate the ECCS dominated these two sequences. The third leading sequence was a functional failure to cool the reactor coolant system (RCS) followed by failure of primary bleed and feed cooling. Hardware and common mode failures in the compressed air system, which supplies air to the pressurizer and steam generator pressure relief valves (PORVs), were the primary contributors to these failures.

The Loss of Component Cooling Water (CCW) event, with a core damage frequency contribution of 1.38E-5 per year, was dominated by three sequences. The first sequence was solely dominated by the failure of the operator to trip the running reactor coolant pumps (RCPs) after seal cooling from CCW is lost, thus leading to gross seal failure. The second sequence was dominated by failure of ECCS cold leg recirculation due to common mode failure of the SI pumps. The third sequence involved the functional failure to restore reactor inventory after CCW was restored (which allows restoration of the ECCS charging pumps). This latter failure was dominated by operator error and ESF signal failure.

The core damage frequency initiated by a Steam Generator Tube Rupture (SGR) was 7.07E-6 per year and it consisted of several significant sequences. The dominant failures associated with these sequences were hardware and common mode failures of the compressed air system and failures of ESF signals. This event is particularly significance since containment may be bypassed and fission products released directly to the environment.

All other internal initiating events were very small contributors to the overall core damage frequency, and displayed no significant vulnerabilities in addition to those previously discussed in this section. The redundancy afforded by the auxiliary feedwater system (two motor-driven pumps and one turbine-driven pump) was significantly beneficial for the transient events. The extremely reliable electric power grid, of which Cook Nuclear Plant is a part, greatly influenced the initiating event frequencies for the Loss of Offsite Power and Station Blackout events, thus directly influencing their small contributions to core damage frequency.

Evaluation of the success criteria used in the event tree analysis showed that containment failure could occur prior to core melt. It was also found that ECCS flow and secondary heat removal can prevent or substantially delay containment failure in those sequences where core melt is prevented regardless of whether containment sprays are operated or not.

The overall frequency of containment failure, which includes isolation failures and containment bypass, was found to be small at 9.1E-6 per year. Given that a core melt accident has occurred, there is approximately an 85% chance that containment integrity will be maintained. Failure of the containment due to slow overpressurization caused by steaming was found to occur only 3.3% of the time following core damage, with the rest of the containment failures involving containment bypass sequences. It is important to note that containment overpressurization failure does not occur if containment spray injection and recirculation are available. Of the bypass sequences, only steam generator tube ruptures occur with sufficient frequency and source term to be considered important.

Failure of containment due to hydrogen generation and combustion was found to be unlikely even for those cases where all power was lost to the hydrogen igniters. Even for a total station blackout, the worst sequence with respect to hydrogen accumulation, it was found to be unlikely that sufficient hydrogen could accumulate to challenge the containment ultimate pressure.

Containment failure due to overpressurization is expected to occur due to shear failure of the concrete at the cylinder wall/basemat junction. This failure mechanism is believed to be such that very few accident mitigation actions would be possible after failure due to the inability to maintain water inventory inside the containment. Because accident mitigation and recovery actions would be very limited, the long term consequences of this failure would be severe and prevention of this failure is considered very important.

Using containment atmospheric release source terms from the Level II analysis, offsite consequences were calculated using the MELCOR Accident Consequence Code System (MACCS). The short term offsite consequences of the dominant sequences from the Level II analysis were addressed considering three emergency planning scenarios (evacuation to 10 miles, no evacuation, and evacuation to 2 miles with sheltering from 2 to 10 miles). In addition, MACCS was used to determine the long term (chronic) affects. The SGR50 sequence (steam generator tube rupture) source terms dominate early and cancer fatalities and whole body doses. SGR50 is a containment bypass sequence for which the containment also fails on overpressure. Virtually all of the noble gases and > 10% of the volatile fission products are released from containment in this scenario. An emergency operating procedure change is being investigated for this sequence to maintain water level in the affected steam generator in the case of a tube rupture. This is expected to greatly reduce the offsite dose consequences of this sequence.

The Cook Nuclear Plant PRA was developed to meet the requirements of 10CFR50 Appendix B. This approach forced detailed review beyond that addressed in Generic Letter 88-20 (Reference 9). AEPSC is confident that this analysis adequately meets the intent of Generic Letter 88-20, including Supplements 1 and 3. No major plant vulnerabilities have been identified which require immediate action or significant hardware changes. Changes to both procedures and hardware, however, are being considered to address minor vulnerabilities. These are identified in Section 6.0.

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

The Cook Nuclear Plant IPE has been performed to identify and resolve plant specific severe accident issues. In accomplishing this task, AEPSC has performed a full scope Level III PRA.

AEPSC has conducted the PRA in full compliance with the requirements of the NRC Generic Letter 88-20, including Supplements 1 and 3. AEPSC's approach to the IPE has been to perform realistic evaluations of Cook Nuclear Plant's capabilities, with emphasis on the prevention of severe accidents and on the need to effectively respond to accident sequence progression in the event of a severe accident. AEPSC's evaluations were also carried out in a manner that supports decisions regarding potential enhancements to plant design and/or operation aimed at reasonable, cost-effective reduction of risk from severe accidents.

The Cook Nuclear Plant Internal Events (Level I PRA) Program consisted of the following 12 major tasks:

1. Project Management
2. Data Analysis
3. Internal Initiating Event Analysis
4. Event Tree Analysis
5. Systems Analysis
6. Systems Interaction
7. Human Reliability Analysis
8. Internal Flooding Analysis
9. Fault Tree and Accident Sequence Quantification
10. Recovery Actions
11. Sensitivity and Importance Analysis
12. Training and Technology Transfer

The Cook Nuclear Plant Containment Performance Analysis (Level II PRA) Program consisted of the following four major tasks:

1. Containment Systems Analysis
2. Containment Structural Capability Review
3. Containment Event Tree Analysis
4. Source Term Analysis

The Cook Nuclear Plant Offsite Consequences Analysis (Level III PRA) Program consisted of the following three major tasks:

1. Site Model Development
2. Offsite Consequences Analysis
3. Consequence Estimation

The models developed in the IPE represent the as-built, as-operated Cook Nuclear Plant. Efforts were taken to ensure that formal procedures for which the operators were trained to use have been credited. In addition, operator interviews were conducted to determine when credit could be taken for the knowledge gained by operators through their daily activities. The value of equipment or procedural improvements and insights were investigated through sensitivity studies.

2.2 Conformance with Generic Letter and Supporting Material

Generic Letter 88-20 requested each utility to perform an Individual Plant Examination for the purpose of:

(1) developing an appreciation of severe accident behavior,

(2) understanding the most likely severe accident sequences that could occur at its plant,

(3) gaining a more quantitative understanding of the overall probabilities of core damage and fission product releases, and if necessary,

(4) reducing the overall probabilities of core damage and fission product releases.

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

(1) The utility staff should be used to the maximum extent possible in the performance of the IPE to insure that they:

- understand the plant procedures, design, operation, maintenance and surveillance,
- understand the quantification of the expected sequence frequencies,
- determine the leading contributors to core damage and unusually poor containment performance,
- identify proposed plant improvements for prevention and mitigation,
- examine each of the proposed improvements, and
- identify which proposed improvements will be implemented and their schedule.

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

(3) The method of examination should either be a PRA that follows the PRA procedures described in References 25, 26 and 29, plus a containment performance analysis that follows the guidance of Appendix 1 to Generic Letter 88-20 or the Industry Degraded Core (IDCOR) front-end method with NRC enhancements, or another systematic method that is acceptable to the staff.

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

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

(6) The utility should report the results of the IPE to the NRC consistent with the criteria provided in the Generic Letter and subsequent guidance provided in Reference 20.

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

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

In response to the Generic Letter, AEPSC issued a letter on October 24, 1989 stating its intent to perform a full scope Level III PRA considering both internal and external events for the Cook Nuclear Plant in order to identify, evaluate, and resolve severe accident issues germane to the plant.

AEPSC has invested substantial personnel time (in excess of 23,000 man-hours) in addition to financial resources for the efforts of a contractor (Individual Plant Evaluation Partnership) in the performance of an IPE that meets or exceeds the NRC directives listed in Generic Letter 88-20. A permanently assigned core staff, knowledgeable in the design and operation of the Cook Nuclear Plant, has been involved in all aspects of the IPE. Other AEPSC personnel have been intensively involved in various aspects of the evaluation as needed. In addition, a substantial training effort was undertaken to insure that AEPSC personnel who had a need for understanding of the evaluation or parts thereof developed an appreciation for the risk significance of the results and the plant response and an understanding of the bases of the IPE.

Finally, AEPSC has and is continuing to review the results of the IPE for areas where plant improvements can be effectively made to reduce the likelihood of core damage. These efforts are being made despite the findings that the overall results of the Cook Nuclear Plant IPE indicate that the core damage frequency and containment performance are within the expected limits.

2.3 General Methodology

The Cook Nuclear Plant IPE program, as previously identified, consisted of 19 major tasks covering the full scope Level III PRA. The IPE was conducted using standard systems analysis practices such as those outlined in References 25 and 26. A comprehensive task breakdown was developed for the Cook Nuclear Plant PRA in order to organize the work to be accomplished. An overview of each of the tasks is provided below. Guidebook instructions were developed for each of the key technical tasks.

Level I PRA Tasks

1. Project Management - Development and monitoring of detailed project planning and scheduling provided necessary technical direction of project analyses and proper review of results.

2. Data Analysis - Plant-specific information was collected from a variety of job orders, control room logs, and completed surveillance test procedures for the period from January 1, 1983 to August 1, 1989 to identify and examine plant-specific component failure, testing, and maintenance data and data related to initiating events that have led to reactor trips. Most of the data used in the Cook Nuclear Plant PRA utilized plant-specific data to calculate failure rates through classical means or through the use of Bayesian techniques. In some instances, generic data from IEEE-500, "IEEE Guide to the Collection and Presentation of Electrical Electronic Sensing Component and Mechanical Reliability Data for Nuclear Power Plant Generating Stations" (Reference 11), Reference 29 or other sources were used to supplement plant data.

3. Internal Initiating Events Analysis - The selection of accident initiating events for the Cook Nuclear Plant PRA considered both actual plant trip data and results of previous PRAs.

   Cook Nuclear Plant trip data was collected from scram reports and plant control room operating logs to identify actual trip events, power level at which the trip occurred, and the failure which caused the trip.

   The Cook Nuclear Plant accident initiating events also included large LOCA, medium LOCA, small LOCA, steam generator tube rupture (SGTR), loss of offsite power, station blackout, steamline/feedline breaks, ATWS, and transients. Most transient initiators were evaluated as either with or without the power steam conversion system being available. Special initiators that were considered included loss of essential service water, loss of component cooling water, loss of control air, loss of 120VAC and loss of a 250VDC bus.

Several methods were employed to determine initiating event frequencies for the relevant initiators. For those events having sufficient Cook Nuclear Plant data, each event was categorized as identified above and the frequency determined by the number of occurrences of each event in the category. For events such as LOCAs, the initiating event frequency was developed from generic data or from the results of previous PRAs or similarly designed plants. As an example, the small, medium, and large break LOCA initiating event frequencies were taken from WASH-1400, "Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants" (Reference 36). Loss of offsite power was determined from a detailed study of the AEPSC grid reliability, and SGTR was determined from the Westinghouse plant population experience data base. The initiating event frequency for the special initiators was determined through plant specific fault tree analysis.

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

5. Systems Analyses - The Cook Nuclear Plant systems were modeled with fault trees. For each system, the complete system analysis included the development of detailed system notebooks describing the system, its operation, the effect of accident conditions (success criteria, initiator impact, etc.), its operating history, the system models and assumptions, quantification, and analyst insights. The relationship between the two units and differences in system designs were also carefully examined and noted.

   The development of the fault tree models was done from the top event down. Fault tree development was accomplished through the generation of simplified flow diagrams and fault tree modules which simplified and standardized the fault tree layouts. The fault trees were developed and quantified using the Westinghouse fault tree GRAFTER Code System (Reference 10).

   The fault tree models incorporated equipment failure, test, maintenance, human reliability modeling, and common cause analysis where appropriate. As the Cook Nuclear Plant PRA utilized the fault tree linking approach to quantification, the appropriate support systems are also included in the fault tree models.

6. Systems Interaction - Possible system interactions were identified by conducting detailed system walkdowns, a control room evaluation and interviews with plant operators. A dependency matrix was also developed to identify the interactions between front-line and support systems.

7. Human Reliability Analysis - Detailed models were developed to represent the interaction of operators and other plant staff with plant systems and equipment during normal operation and during transient and accident conditions. The Technique for Human Error Rate Prediction (THERP) methodology (Reference 22) was used for the human reliability analysis.

8. Internal Flood Analysis - A separate analysis was performed to determine areas in the Cook Nuclear Plant that are susceptible to flooding, equipment in those areas whose failures could cause a plant shutdown or result in a failed safety system, and the contribution to core damage from flooding of those areas. The appropriate event trees from the other internal event initiators were used to quantify the contribution of flooding to core damage frequency.

9. Fault Tree and Accident Sequence Quantification - The Cook Nuclear Plant system fault trees and event tree accident sequences were integrated and quantified to obtain accident sequence cutsets, frequencies for all accident sequences resulting in core damage, and to identify dominant accident sequences among all event tree results. The Westinghouse WLINK Code System (Reference 46) was used to perform the accident sequence quantification. Results of this analysis are the essential data input to the Level II PRA.

10. Recovery Actions - Recovery actions were identified and their respective failure probabilities quantified.

11. Sensitivity and Importance Analyses - The response of the core damage frequency to changes in input parameters and modeling assumptions for the core damage frequency dominant contributors was examined to identify important actions and equipment and to study the sensitivity to those assumptions.

12. Training and Technology Transfer - Training was conducted by contractor employees for utility personnel to provide the in-house ability to understand, evaluate, modify, and update the PRA study to reflect proposed or actual changes in the plant design and operation. Training included initial orientation to PRA technology, training sessions on each major task, and discussion of analysis-specific guidebooks.

Level II PRA Tasks

1. Containment Systems Analysis - Quantitative models for containment systems failures and containment bypass events were developed and quantified as part of the Level I PRA. The results of these analyses provide information for use in the Level II analysis regarding the state of the plant systems, the physical state of the core, and the reactor coolant system. A model for failure of containment isolation was developed and quantified independently of the other models. The failure probability of containment isolation was used for the containment event tree quantification.

2. Containment Structural Capability Review - Existing and updated structural analyses were used to determine the containment ultimate pressure capability and potential failure locations.

3. Containment Event Tree Analysis - A containment event tree (CET) was developed to provide a systematic method for integrating the Level I results with the Level II analysis. The CET describes the containment response to a core melt accident and accounts for system interactions, operator actions, and key phenomenological issues by defining a functional set of top events and their success and failure states.

4. Source Term Analysis - Source terms were developed by analyzing the dominant accident sequences that led to containment failure using the MAAP code (Reference 13). Source terms were binned into release categories based on the type, timing, and magnitude of the release.

Level III PRA Tasks

1. Site Model Development - A Cook Nuclear Plant site model was developed using available information on the demography and meteorology in the region of the Cook Nuclear Plant site.

2. Offsite Consequences Analysis - The offsite consequences associated with each of the fission product release categories identified in the source term analysis were determined using the MACCS computer code (Reference 14). The MACCS computer code outputs used in the analysis included acute fatalities, latent fatalities, and total population exposure. The analysis accounted for emergency action plans, including evacuation, sheltering, and decontamination.

3. Consequence Estimation - The offsite consequences for the three identified outputs (acute fatalities, latent fatalities, and total population dose) were developed based on multiple atmospheric dispersion analyses and presented in the form of complementary cumulative distribution functions, which showed a graphical representation. The analyses yielded the expected consequence level and the probability of exceeding that level. The estimated probabilities were based on the assumption that the release had occurred. Therefore, the actual probability of offsite consequence is equal to the containment release probability times the consequence probability.

2.4 Information Assembly

A tremendous amount of information was needed to perform the detailed Cook Nuclear Plant IPE study. The project team reviewed and assembled information from plant specific sources, similar plant studies, and generic sources. Plant walkdowns were a part of the data collection effort. Information was assembled to familiarize the analyst with the plant, determine the important initiating events and quantify their frequency, determine the component and system failure rates, perform various supporting analyses, conduct the evaluation of internally initiated flooding events, and develop plant layout insights through the use of plant walkdowns. Walkdowns were specifically used to search for plant characteristics that could impact the transport of radionuclides in the containment and auxiliary building. Table 2.4-1 provides a list of the important sources of information that were reviewed for the Level I analysis. The complete lists of all individual references used are documented in the Cook Nuclear Plant IPE project notebooks.

The Cook Nuclear Plant IPE team modelled the Cook Nuclear Plant as-built condition as it existed on August 1, 1989. No major changes to plant operation or design have been identified since August 1, 1989, that would be expected to significantly affect the PRA results.

Much of the information was collected at the outset of the project. All information used in the project is available at the AEPSC offices in Columbus, Ohio. Copies of some information are also housed at the Westinghouse office in Monroeville, PA and the Fauske and Associates (FAI) office in Burr Ridge, Illinois.

Detailed system notebooks were developed for 14 major systems and several miscellaneous systems that were expected to have an influence on the Cook Nuclear Plant IPE results. In addition, notebooks were developed for major analyses of the IPE project (e.g., initiating events, internal flooding, etc.). Plant information sources identified in Table 2.4-1 were used to develop system descriptions and models. Both plant specific and generic sources identified were used to define component availabilities, initiating events and initiating event frequency, important accident sequences, potentially important modeling features, common cause failure rates, and human reliability data. Subsequent sections of this report provide a more detailed discussion of the use of the information collected.

Plant walkdowns were conducted by all members of the AEPSC IPE team and some representatives from the IPEP team who were responsible for the evaluation of a specific plant system, the containment and/or its systems, or the evaluation of internal flooding. The walkdown teams were led by Cook Nuclear Plant personnel who were knowledgeable about the plant systems and the containment and their detailed arrangement.

Walkdowns were conducted for the systems and plant environment of most concern to the PRA. These areas are contained primarily in the Auxiliary Building and the Containment; however, several other buildings or areas were examined because important systems and components are located therein. The areas or buildings in which walkdowns were made are:

- Containment
- Auxiliary Building
- Turbine Building
- Service Water Screen House
- Control Room
- Outside Grounds Including Switchyards

General arrangement drawings of these areas are contained in the UFSAR.

The walkdowns that were conducted during the IPE project are summarized below.

System Walkdowns - The AEPSC system fault tree analysts conducted initial walkdowns of the systems modelled within the Cook Nuclear Plant PRA from March 13 to March 15, 1990. Walkdowns were conducted in Units 1 and 2. Cook Nuclear Plant operations personnel assisted the team in becoming familiar with equipment locations, system operations, and test/maintenance practices.

Containment Walkdowns - The PRA team members assigned to the Containment Performance Analysis task performed a containment walkdown of the Unit 1 containment on July 16 and 17, 1990, to verify that the phenomenological models accurately reflected the condition of the plant. In addition, containment isolation capability and potential containment bypass flow paths were also examined for Units 1 and 2 by walkdowns of the Auxiliary Building during this time.

Operator Interviews - The execution of the human reliability analysis involved both the identification and evaluation of plant procedures and a discussion of the pertinent steps therein with plant operators. The interviews, which were conducted on March 14 and 15, 1991, provided the analysts with insights into the complexity of the tasks, the familiarity of the operators with the required task steps, the time constraints involved, and the extent of training conducted by the plant. The analysts directly involved in the human reliability analysis performed the interviews.

Internal Flooding Walkdown - Walkdowns were performed on March 13 through 15, 1990, and July 18 and 19, 1990, primarily to gain an understanding of the special relationships of components and equipment to the various specific hazards presented by internal flooding sources. Analysts assigned to this task were the primary participants in these walkdowns.

Table 2.4-1

Cook Nuclear Plant IPE Information Sources

<u>SOURCE</u>

<u>Plant Specific</u>

Donald C. Cook Nuclear Plant Updated Final Safety Analysis Report, American Electric Power Service Corporation, July 1989. (Reference 1)

Donald C. Cook Nuclear Plant Units 1 and 2 Technical Specifications, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, Amendment 127 Unit 1, Amendment 113 Unit 2. (References 2 and 3)

Donald C. Cook Nuclear Plant Units 1 and 2 System Descriptions, American Electric Power Service Corporation.

Plant System Flow Diagrams.

Plant Arrangement Drawings.

Plant Electrical One-Line Diagrams and Elementary Diagrams.

Emergency Operating Procedures.

Normal Operating Procedures.

Maintenance Procedures

System Surveillance Test Procedures

Donald C. Cook Nuclear Plant Facility Data Base.

Donald C. Cook Nuclear Plant Setpoint Document.

Nuclear Test Schedule, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, September 1989.

AEPSC Calculations.

Donald C. Cook Control Room Logs

Donald C. Cook Job Order Records

Table 2.4-1 Cook Nuclear Plant IPE Information Sources (Cont'd)

<u>SOURCE</u>

<u>Generic Sources</u>

NUREG-1032, Evaluation of Station Blackout Accidents at Nuclear Power Plants, June 1988. (Reference 19)

NUREG/CR-1174, "Evaluation of System Interactions in Nuclear Power Plants," August 1989. (Reference 21)

NUREG/CR-1278, "Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983. (Reference 22)

NUREG-0909, "January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," April 1982. (Reference 18)

NUREG/CR-4142, "A Review of the Millstone 3 Probabilistic Safety Study," April 1986. (Reference 28)

WASH-1400, "Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants," October 1975. (Reference 36)

NUREG/CR-2300, "PRA Procedures Guide," January 1983. (Reference 26)

NUREG/CR-2678, "Flood Risk Analysis Methodology Development Project Final Report," June 1982. (Reference 23)

NUREG/CR-2815, "Probabilistic Safety Analysis Procedure Guide," Rev. 1, August 1985. (Reference 25)

NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," EG&G Idaho, Inc., May 1985. (Reference 27)

NUREG/CR-4550, "Analysis of Core Damage Frequency from Internal Events," Volumes 1-4, September 1987. (Reference 29)

NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Volume 1, February 1988 and Volume 2, January 1989. (Reference 30)

EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," June 1985. (Reference 7)

EPRI NP-3583, "Systematic Human Reliability Procedure (SHARP)," June 1984. (Reference 5)

NSAC-144, "Loss of Offsite Power at U.S. Nuclear Power Plants," April 1989. (Reference 17)

NSAC-108, The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants. (Reference 16)

IEEE-500, IEEE Guide to the Collection and Presentation of Electrical Electronic Sensing Component and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations. (Reference 11)

IDCOR Technical Reports.

Table 2.4-1

Cook Nuclear Plant IPE Information Sources (Cont'd)

SOURCE

INPO SOER 85-5, "Internal Flooding of Power Plant Buildings," December 1985. (Reference 12)

NUREG/CR-5536, "Mitigation of Direct Containment Heating and Hydrogen Combustion Events in Ice Condenser Plants," October 1990. (Reference 31)

EPRI NP-3878, "Large Scale Hydrogen Combustion Experiments," October 1988. (Reference 6)

Westinghouse WCAPs

WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988. (Reference 41)

WCAP-12078, "Input and Output Parameters for the Accident Analyses Performed for Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1," December 1988. (Reference 43)

WCAP-9600, "Report on Small Break Analysis for Westinghouse NSSS Systems," Volume 1, June 1979. (Reference 37)

WCAP-12135, "Donald C. Cook Nuclear Plant Units 1 and 2, Rerating Engineering Report," September 1989. (Reference 44)

WCAP-10541, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Rev. 2, November 1986. (Reference 39)

WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs," December 1988. (Reference 42)

WCAP-10858-PA, "AMSAC Generic Design Package," Rev. 1, July 1987. (Reference 40)

WCAP-9914, "PORV Sensitivity Study for LOFW-LOCA Analyses," July 1981. (Reference 38)

2.5 Treatment of Dual Units

The Cook Nuclear Plant is a dual unit site. Both units are Westinghouse four-loop pressurized water reactors with ice condenser containments. Unit 1 was explicitly analyzed in the Cook Nuclear Plant IPE. Both units were examined and the Unit 1 analysis was determined to be bounding for Unit 2.

The multi-unit methodology used for the Cook Nuclear Plant IPE consisted of five key analysis areas. These areas were: 1) plant familiarization, 2) initiating event analysis, 3) support system analysis, 4) front line system analyses, and 5) containment analyses.

Plant familiarization involved the collection and evaluation of plant documentation on the design and operation of each unit and identification of dependencies between front line systems and support systems. Plant walkdowns were conducted to support the plant documentation review and to look for dependencies and other as-built information that was not evident from the plant documentation, including differences in the configuration of the same systems in the different units. No differences which would have had an impact in the IPE results were identified.

Data collected to determine the relevant initiating events and the system dependencies were examined to identify intersystem dependencies between units that could result from particular initiators. No inter-unit dependencies were identified which would have had an impact on the internal initiating events analysis.

Support systems were most crucial in determining the correct plant response to an initiating event because some support systems are shared or have the potential to be cross connected between units. The AEPSC method for capturing the effects of the second unit of Cook Nuclear Plant was to develop fault tree models for the shared support systems. These fault tree models were explicitly included in the overall accident sequence quantification through fault tree linking.

Frontline systems analyses used a comparative method for those systems that were found to be completely independent between units, while those systems that were found to be shared or partially shared were modeled to include all components of the system or systems to the extent of influence. For the front line systems, a detailed comparison of the two units was made to identify the differences and commonalities. No differences which would have had an impact on the IPE results were identified.

A final step in the dual unit methodology resulted in the examination of the event trees developed for the first unit to determine whether any differences identified in the review of the second unit would cause different or additional events to be necessary to accurately represent the second unit. No significant differences were found.

Finally, the level 2 analysis which was conducted for Unit 1 was found to be bounding for Unit 2.

In summary, the Cook Nuclear Plant IPE analyzed the design and operation of Unit 1. Unit 2 was examined and no differences from Unit 1 were identified which would have impacted the IPE results. The IPE, therefore, may be applied to either unit.

3.0 FRONT-END ANALYSIS

3.1 Accident Sequence Delineation

3.1.1 Initiating Events

All internal initiating events, including internal flooding, analyzed in the Cook Nuclear Plant IPE are listed in Table 3.1-1. Internal initiating events cause sequences of events that can result in insufficient core cooling. Insufficient core cooling can be caused by either a loss of primary coolant (LOCA) or insufficient heat removal by secondary side systems. A more detailed discussion of the grouping of initiation events follows.

3.1.1.1 Loss of Coolant Accidents

The general category of initiating events referred to as LOCAs includes all accidents that result in a reduction of primary coolant system water inventory. This category of events was further divided into subcategories on the basis of the leak path and size. These subcategories are described below.

3.1.1.1.1 Large LOCA

The large LOCA category includes ruptures inside containment in the size range from a double-ended cold leg guillotine (DECLG) pipe severance down to a six-inch equivalent diameter hole in the reactor coolant system. This range was chosen because it is consistent with the size range for large LOCAs analyzed in Reference 1.

3.1.1.1.2 Medium LOCA

The medium LOCA range of breaks represents all reactor coolant system ruptures inside containment of equivalent diameter from 2 inches to 6 inches. The flow area of a pressurizer safety valve is 3.644 square inches, therefore, failure of one or all pressurizer safety valves would fall within this category of LOCA. The flow area of a pressurizer PORV is 2.0 square inches. Failure of two or more pressurizer PORVs would be included within this category, however, the random failure of two components is not considered credible and, therefore, will not be considered further.

This range was chosen because it is below the lower bound of the large LOCA analysis in Reference 1 and above the size hole where accumulator injection would occur. This size range includes the most limiting size break of the small break LOCA analysis in Reference 1, a 3-inch cold leg break.

3.1.1.1.3 Small LOCA

This category of events comprises breaks inside containment in the range of 2-inch to 3/8-inch equivalent diameter holes. Also included are RCP seal failures, control rod ejections and single failures of PORVs. The upper bound of this event was chosen because, for holes less than 2 inches in diameter, no accumulator flow is required to keep the core covered. The lower bound is the size hole for which normal charging flow can maintain liquid inventory. For breaks in this size range, heat removal by the secondary systems in addition to the heat removal by the ECCS systems would be required to prevent core damage.

3.1.1.1.4 Steam Generator Tube Rupture

Although this category can be included in the small LOCA category, it is separated due to its unique effects on the plant and the environment. A steam generator tube rupture may result in direct bypass of the containment boundary, if steam generator safety or relief valves lift or the steam generator is not isolated.

This category includes all abnormal leakages including multiple tube ruptures from the reactor coolant system (RCS) into one steam generator in excess of charging pump makeup capacity and which would be expected to result in actuation of the ECCS.

3.1.1.1.5 Breaks Beyond ECCS Capability

Two classes of LOCAs that may be beyond the capacity of ECCS have been identified: simultaneous rupture of two or more large pipes and a catastrophic reactor vessel rupture.

3.1.1.1.6 Interfacing Systems LOCA

This category considers RCS supporting systems that have direct piping connections between the RCS and systems outside the containment. Piping and/or valve failures associated with these systems have the potential to cause a LOCA that could disable the ECCS functions and bypass the containment. The limiting factors in this type of event are possible loss of primary coolant outside via a direct release path to the environment.

3.1.1.2 Transients

Ten transient initiating event categories were analyzed. They were grouped into categories based on plant response, signal actuation, systems required for mitigation and subsequent plant-related effects. The following sections provide a general description for each transient initiating event category.

3.1.1.2.1 Transients With the Steam Conversion System Available

This category includes events and support system losses not evaluated separately which cause a reactor trip and would occur with the steam conversion system available to remove decay heat. Practically, this means events in which main feedwater is able to supply the steam generators.

3.1.1.2.2 Transients Without the Steam Conversion System Available

This category includes events and any support system failures not evaluated separately as special initiators that would occur with the steam conversion system not available to remove decay heat. Practically, this means that main feedwater is not available to supply the steam generators.

3.1.1.2.3 Large Steam Line/Feedline Break

This event includes main feedwater breaks and main steam line breaks both inside and outside containment and any spurious valve openings that could result in a large reactor power increase due to a secondary side steam demand increase. This event includes those unanticipated transients that require rapid secondary side isolation and ECCS actuation. The plant response is modelled as if the break occurs inside containment because a break inside containment presents the greatest challenge to safety systems.

3.1.1.2.4 Loss of Offsite Power

This event results from a complete loss of the offsite grid power accompanied by a turbine trip. Following the initial loss of AC power, at least one diesel generator would, by definition, come on line to supply electrical power. Events where both diesel generators fail are included under the station blackout event.

3.1.1.2.5 Station Blackout

This event results from the loss of offsite power accompanied by the loss of the onsite emergency AC power distribution system.

3.1.1.2.6 Anticipated Transient Without Scram

This event involves the failure of the RPS system to trip the reactor following an anticipated transient. The event could be initiated by any event which requires a reactor trip to mitigate the event. This event is basically a transient described in either section 3.1.1.2.1 or 3.1.1.2.2 above combined with the failure probability of the RPS system.

3.1.1.2.7 **Loss of Essential Service Water**

This event involves the complete loss of ESW cooling to one unit's components for any reason other than support system failures. A total loss of ESW would cause rising temperatures in the CCW system. With the loss of CCW cooling, RCP seal temperatures and bearing temperatures would increase and the operators would be expected to initiate a reactor trip. Following the reactor trip, components requiring ESW cooling, including the CCW system, would not be available for accident mitigation until ESW cooling is recovered. In addition, a consequential RCP seal LOCA must be assumed because cooling to the seals would be lost.

3.1.1.2.8 **Loss of Component Cooling Water**

This event involves the complete loss of CCW cooling to one unit for any reason other than support system failures. This event is analyzed separately from the loss of ESW event because, for this event, the containment spray system would be available to prevent containment failure if core damage occurs. As in the loss of ESW event, the operators would be expected to initiate a reactor trip on high RCP bearing and seal temperatures.

3.1.1.2.9 **Loss of 250 VDC**

Loss of a single train of 250 VDC would cause a loss of power to the RCP undervoltage and underfrequency sensing relays causing the reactor protection system to sense a low flow condition. Concurrent loss of two trains of 250 VDC is not evaluated because the initiating event frequency is very small. Following the resulting reactor trip, the train which lost DC power will not have control power available to the safety and non safety equipment. Lack of control power will prevent the automatic starting of standby equipment necessary to mitigate the event.

3.1.1.2.10 **Internal Flooding**

The only internal flooding scenario of significance involved an ESW discharge line break that resulted in flood of the turbine building sub-basement. The NESW pumps are housed in the sub-basement and are postulated to fail due to the submergence of the motors. These pumps provide cooling water for the control air compressors and plant air compressors which in turn are postulated to fail. Failure of the compressors will result in closure of the feedwater regulating valves which subsequently cause a reactor trip. This event was judged to be bounded by the transient without the steam conversion system available event tree discussed in Section 3.1.1.2.2.

TABLE 3.1-1

SUMMARY OF INTERNAL INITIATING EVENT FREQUENCIES

Category	Title	Frequency per Calendar Year	Variance (per Year Squared)
1	Large LOCA	3.00E-04	5.5E-07
2	Medium LOCA	9.17E-04	3.9E-06
3	Small LOCA	6.8E-03	6.3E-05
4	Steam Generator Tube Rupture	7.2E-03	2.9E-05
5	Breaks Beyond ECCS Capability	3.0E-07	5.5E-13
6	Interfacing Systems LOCA (V-Sequence)	6.7E-07	2.5E-13
7	Transients With the Steam Conversion System Available	3.8	8.9
8	Transients Without the Steam Conversion System Available	1.2E-01	0.28
9	Large Steamline/Feedline Break	3.3E-04	5.5E-07
10	Loss of Offsite Power	4.0E-02	9.0E-04
11	Station Blackout	1.40E-05	1.11E-10
12	ATWS	4.67E-05	1.22E-09
13	Loss of Essential Service Water	3.73E-05	6.04E-10
14	Loss of Component Cooling Water	8.71E-04	1.14E-06
15	Loss of 250 VDC	1.16E-02	4.42E-05
16	Internal Flooding	3.00E-03	5.04E-06

3.1.2 Front-Line Event Trees

Event trees were developed for each of the initiating events described above and are shown in Figures 3.1-1 - 3.1-16. The event tree analysis for each initiating event was developed to present the most important events and systems necessary to mitigate the event. These events and systems modelled within the event trees are referred to as top events and may be generally divided into two categories. The first category includes systems needed to provide adequate core cooling to prevent severe core damage. The second category includes systems used to mitigate impact on containment integrity following severe core damage. Some top events fit each category depending on the accident sequence.

Support systems required for the success of the top events were not modelled within the event tree. Support systems were, however, modelled within the fault trees and factored into the accident sequences through the fault tree linking process of the accident sequence quantification.

The accident progression was analyzed with the purpose of preventing severe core damage or mitigating the containment transient for a 24 hour period. The 24 hour period was based on the assumption that extraordinary and generally unquantifiable operator actions can be taken by 24 hours to mitigate the consequences of most accidents. This assumption was consistent with past PRAs and was specified in Reference 20.

For the ECCS and containment spray systems, operation was modelled in two phases: the injection phase and the recirculation phase. Although the length of each phase varied depending on the number of pumps running in each phase, the total time modelled for any accident scenario was 24 hours. In order to simplify the modelling within the fault trees for these systems, the injection phase was modelled for one-half hour and the recirculation phase for 24 hours. Using these mission times was conservative in that a total of 24.5 hours of run time was modelled when only 24 hours was required. In addition, all important operator actions needed to transition from injection to recirculation were included as well as a pump start for the beginning of each phase.

Because this project integrated the Level I and Level II analyses, the success criteria for top events considered the effect of the systems on the containment. In developing the event tree success criteria, prevention of core damage was assumed to be possible only if containment overpressurization was prevented. If the containment pressure increased above the ultimate capacity, then gross containment failure was assumed and all inventory available for ECCS recirculation was assumed to be lost. It should be noted, therefore, that when an accident sequence was indicated as successful, neither severe core damage nor gross containment failure would have occurred. Because modelling containment isolation within the system event trees would have increased the complexity of the event trees, failure to isolate the lines penetrating containment was modelled within the Level II containment event tree.

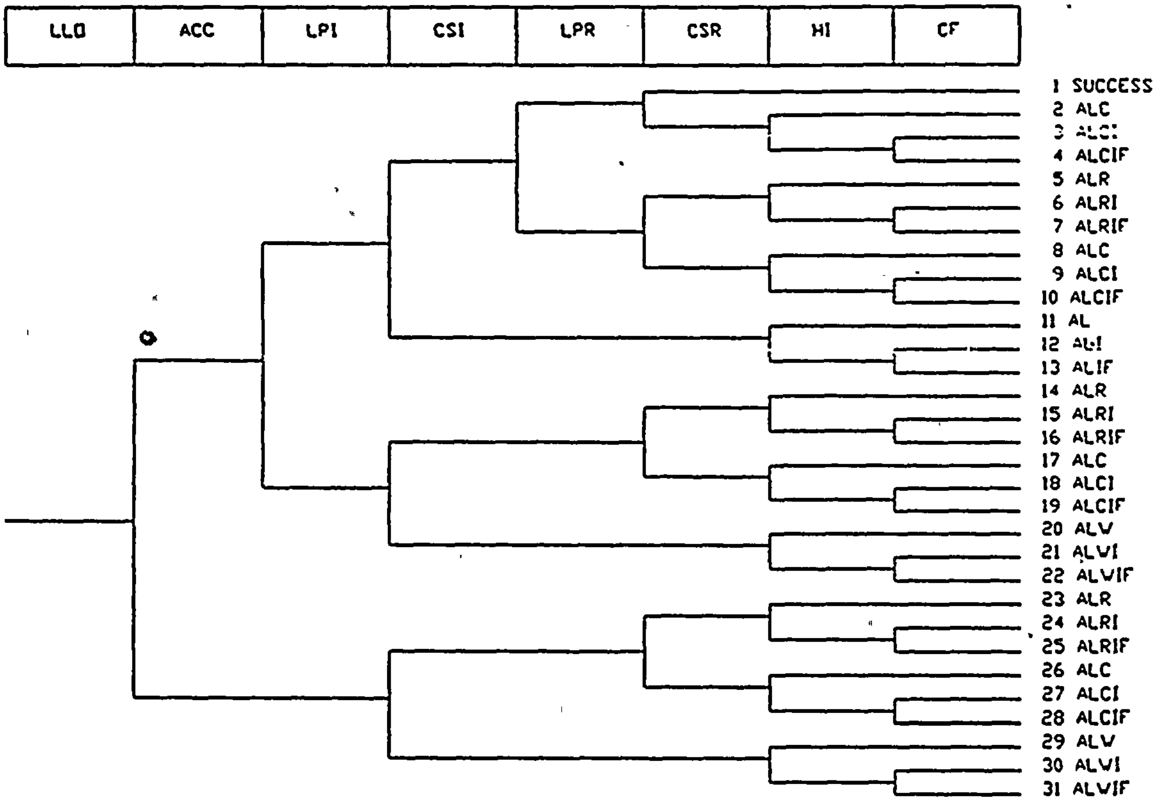
For accident sequences where severe core damage occurred, containment protection systems were modelled as top events to show the effect of support system interactions on the systems. In addition, this modelling provided a quantified assessment of the state of the containment protection systems for use in the Level II PRA. The success or failure of a top event following core damage, however, did not imply that the containment either remained intact or failed. The specific containment response to the success or failure of containment systems was modelled as appropriate in the Level II analysis.

For all internal initiating event tree accident sequences, the ice condenser was assumed to function as designed to mitigate the effect of RCS or steam generator blowdown on containment. As a result, the ice condenser was not modelled as a top event for internal initiating events. The ice condenser is a completely passive system with no support system interrelationships with other systems. An analysis of the probability of ice condenser failure following an internal initiating event was performed. This analysis concluded that, if the ice condenser was demanded following an internal initiating event, the frequency of occurrence of such a sequence would be sufficiently low that the core damage frequency for the sequence would be below the cutoff frequency specified in Reference 20 as requiring further evaluation.

Success criteria for top events were taken from many sources. Where possible, the equipment requirements from the analyses in Reference 1 were used. Because of their inherent conservatism, analyses in Reference 1 were considered to present the greatest challenge to plant systems. If an analysis was not available in Reference 1 to support development of success criteria, then equipment success criteria were selected based on the emergency operating procedures or their background documents and the success criteria was verified using the MAAP computer code (Reference 13). In some cases, success criteria were defined in the event tree to determine the subsequent accident progression. The success of these top events, however, did not cause or prevent core damage. The basis for success criteria determination is defined and referenced in Tables 3.1-2 - 3.1-17.

The initial condition of the NSSS was usually assumed to be normal operating temperature, pressure and pressurizer level with the reactor at 100% power. There were exceptions to these conditions when the initial conditions would impose more limiting success criteria on the top events. For example, the success criteria for the steam line rupture event tree were taken from the Reference 1. The initial operating condition of the reactor assumed for this analysis was hot zero power with the reactor critical in the source range. In all cases, however, the success criteria were taken to be the most limiting in order to bound all initial Mode 1 or 2 operating conditions of the reactor.

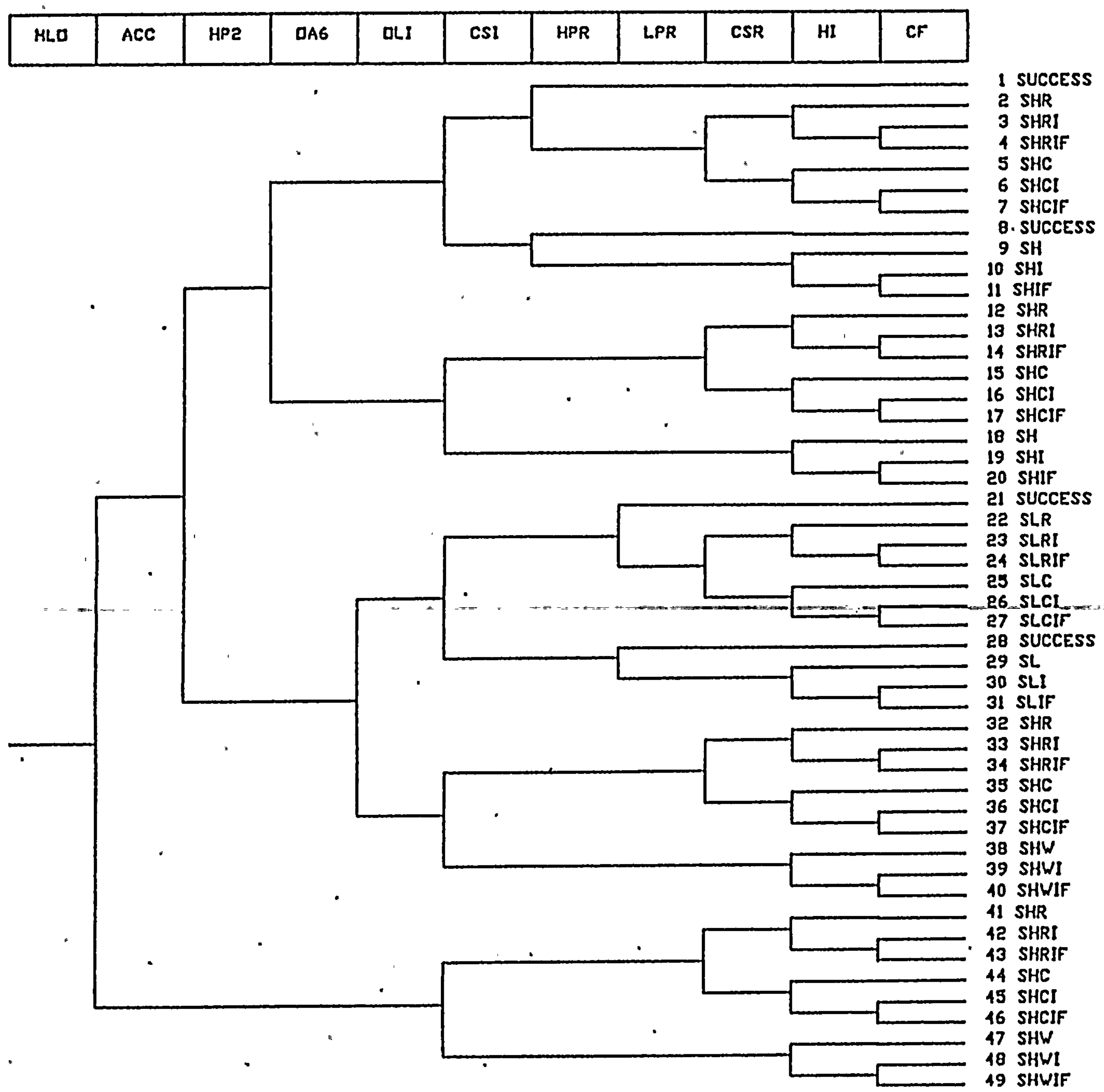
Consequential failures, which would transform the accident sequence from one initiating event category to another, were not explicitly modelled within the event trees. Rather, consequential failures were bounded by the initiating event categorization and frequency development. For example, a transient event could lead to core damage because of a loss of essential service water which, in turn, would cause a consequential RCP seal LOCA. Rather than model the effects of a consequential RCP seal LOCA within the transient tree, a separate initiating event and accident sequence analysis was performed to quantify the effects of a loss of essential service water. All consequential failures caused by the loss of essential service water were considered bounded by the loss of essential service water event tree.



EVENT	EVENT NAME
LLO	LARGE LOCA INITIATING EVENT
ACC	ACCUMULATORS
LPI	RHR (LOW PRESSURE) INJECTION
CSI	CONTAINMENT SPRAY INJECTION
LPR	RHR (LOW PRESSURE) RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
ALC	LARGE LOCA, LOW PRESSURE, CSR FAILS
ALCI	LARGE LOCA, LOW PRESSURE, CSR & HI FAIL
ALCIF	LARGE LOCA, LOW PRESSURE, CSR,HI,CF FAIL
ALR	LARGE LOCA, LOW PRES, CT HEAT REMOVAL
ALRI	LG LOCA CT HEAT REMOVAL SUCCESS HI FAILS
ALRIF	LG LOCA, HI, CF FAIL, CT HEAT REMOVED
AL	LARGE LOCA, LOW PRESSURE
ALI	LARGE LOCA, LOW PRESSURE, HI FAILS
ALIF	LARGE LOCA, LOW PRESSURE, HI & CF FAIL
ALV	LARGE LOCA, LOW PRESSURE, CT DRY
ALVI	LARGE LOCA, LOW PRESS CT DRY HI FAILS
ALVIF	LG LOCA, LOW PRES, CT DRY, HI & CF FAIL

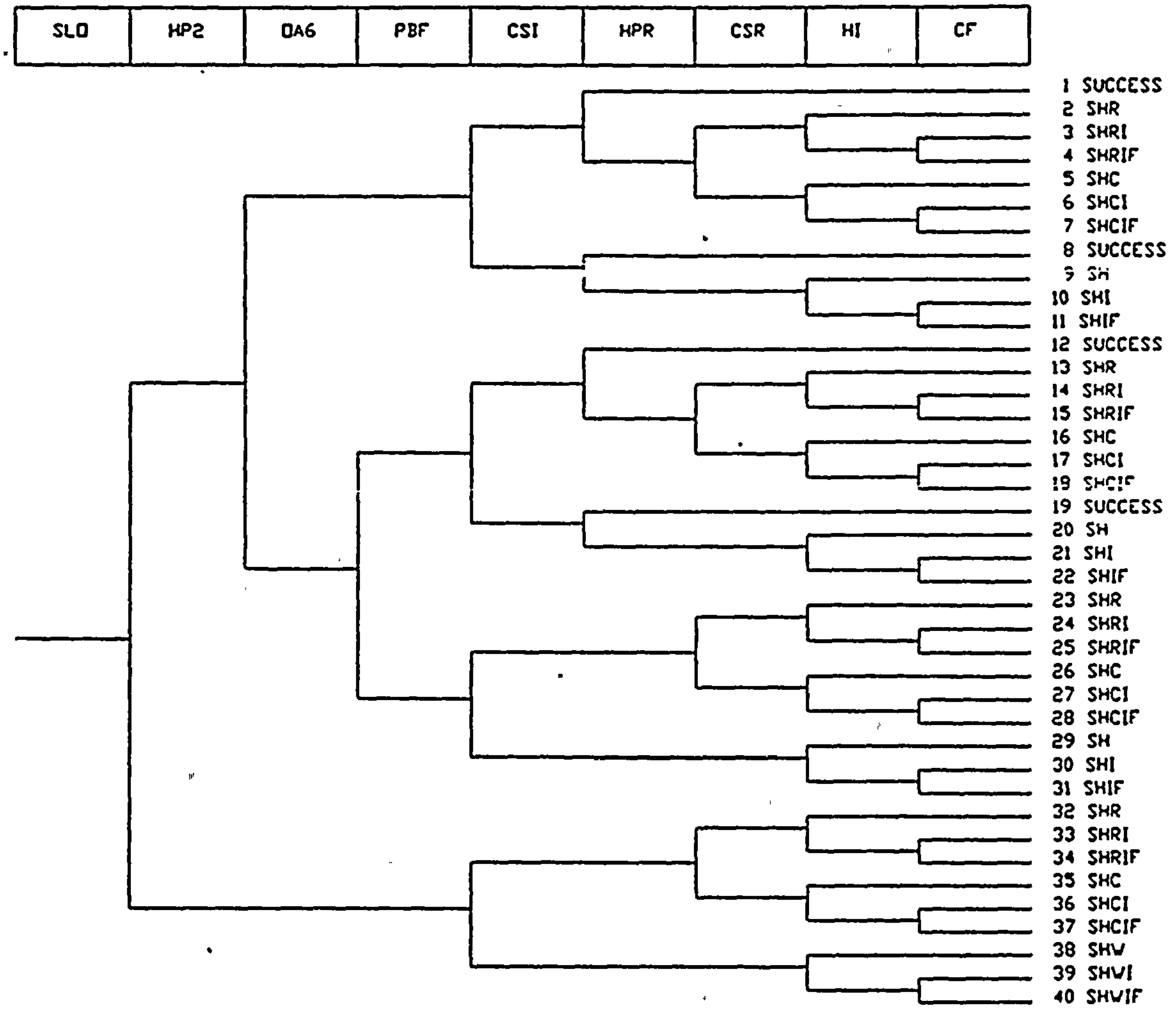
Figure 3.1-1
Large LOCA Event Tree



EVENT	EVENT NAME
HLO	MEDIUM LOCA INITIATING EVENT
ACC	ACCUMULATORS
HP2	ECCS (HIGH PRESSURE) INJECTION
OA6	RCS COOLDOWN USING AFW AND STEAM DUMP
OLI	DEPRESSURIZATION AND LOW PRES. INJECTION
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
LPR	RHR (LOW PRESSURE) RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
SHR	SMALL LOCA, HIGH PRESS, CT HEAT REMOVAL
SHRI	SM LOCA CT HEAT REMOVAL SUCCESS HI FAILS
SHRIF	SM LOCA, HI, CF FAIL, CT HEAT REMOVED
SHC	SMALL LOCA, HIGH PRESSURE, CSR FAILS
SHCI	SMALL LOCA, HIGH PRESSURE, CSR & HI FAIL
SHCIF	SM LOCA, HIGH PRESSURE, CSR, HI, CF FAIL
SH	SMALL LOCA, HIGH PRESSURE
SHI	SMALL LOCA, HIGH PRESSURE, HI FAILS
SHIF	SMALL LOCA, HIGH PRESSURE, HI & CF FAIL
SLR	SMALL LOCA, LOW PRESS, CT HEAT REMOVAL
SLRI	SM LOCA, CT HEAT REMOVAL, HI FAILS
SLRIF	SM LOCA, HI, CF FAIL, CT HEAT REMOVAL
SLC	SMALL LOCA, LOW PRESSURE, CSR FAILS
SLCI	SMALL LOCA, LOW PRESSURE, CSR & HI FAIL
SLCIF	SMALL LOCA, LOW PRESSURE, CSR,HI,CF FAIL
SL	SMALL LOCA, LOW PRESSURE
SLI	SMALL LOCA, LOW PRESSURE, HI FAILS
SLIF	SMALL LOCA, LOW PRESSURE, HI & CF FAIL
SHW	SMALL LOCA, HIGH PRESSURE, CT DRY
SHWI	SMALL LOCA, HIGH PRESS, CT DRY, HI FAILS
SHWIF	SM LOCA, HIGH PRES, CT DRY, HI & CF FAIL

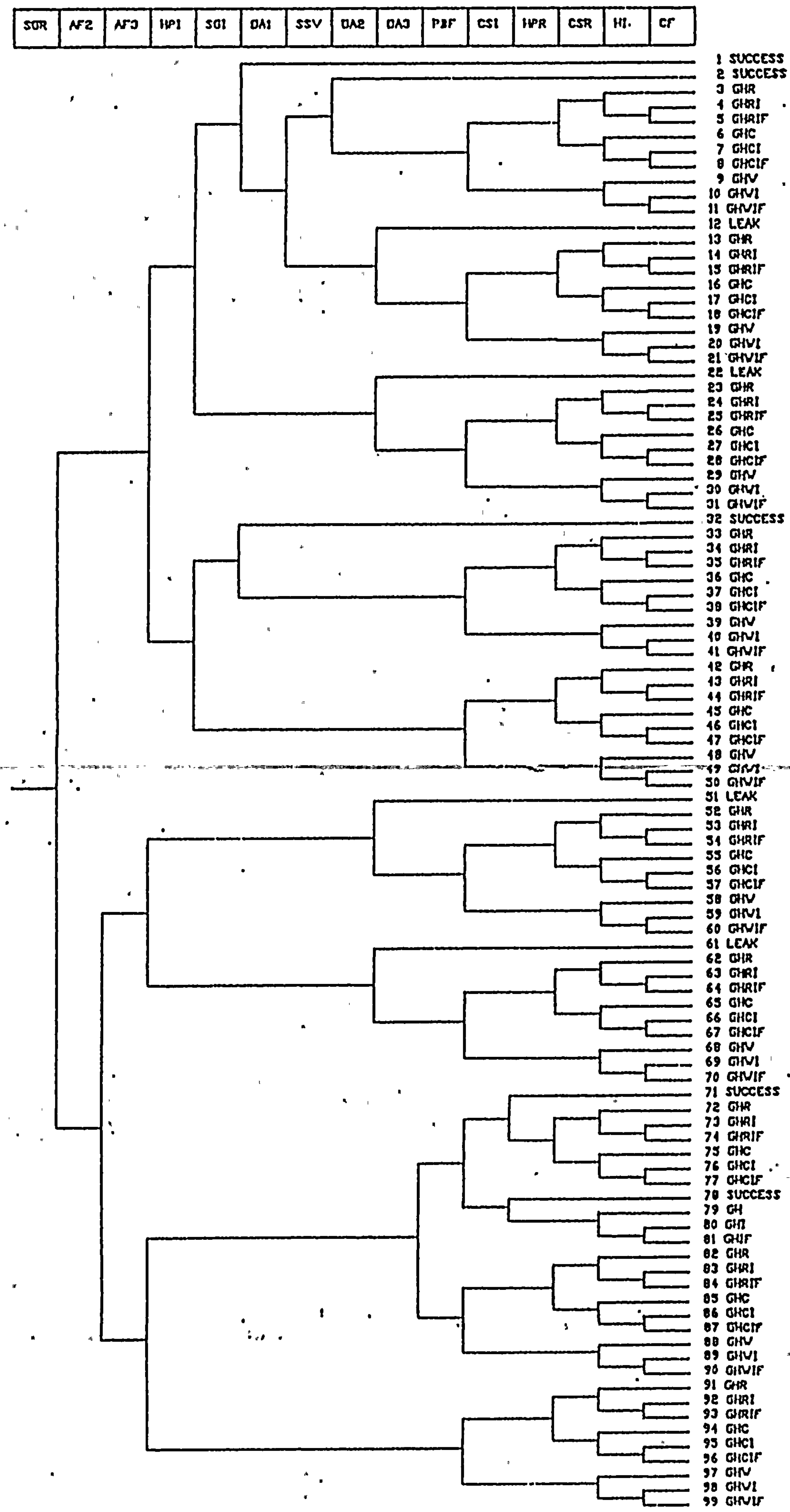
Figure 3.1-2
Medium LOCA Event Tree



EVENT	EVENT NAME
SLO	SMALL LOCA INITIATING EVENT
HP2	ECCS (HIGH PRESSURE) INJECTION
OA6	RCS COOLDOWN USING AFW AND STEAM DUMP
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
SHR	SMALL LOCA, HIGH PRESS, CT HEAT REMOVAL
SHRI	SH LOCA CT HEAT REMOVAL SUCCESS HI FAILS
SHRIF	SH LOCA, HI, CF FAIL, CT HEAT REMOVED
SHC	SMALL LOCA, HIGH PRESSURE, CSR FAILS
SHCI	SMALL LOCA, HIGH PRESSURE, CSR & HI FAIL
SHCIF	SH LOCA, HIGH PRESSURE, CSR, HI, CF FAIL
SH	SMALL LOCA, HIGH PRESSURE
SHI	SMALL LOCA, HIGH PRESSURE, HI FAILS
SHIF	SMALL LOCA, HIGH PRESSURE, HI & CF FAIL
SHV	SMALL LOCA, HIGH PRESSURE, CT DRY
SHVI	SMALL LOCA, HIGH PRESS, CT DRY, HI FAILS
SHVIF	SH LOCA, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-3
Small LOCA Event Tree



EVENT	EVENT NAME
SGR	STEAM GENERATOR TUBE RUPTURE
AF2	AUXILIARY FEEDWATER TO THE INTACT SG(S)
AF3	AUXILIARY FEEDWATER TO THE FAULTED SG
HPI	ECCS (HIGH PRESSURE) INJECTION
SGI	SG ISOLATION BY MSIV CLOSURE
DA1	DEPRESSURIZE AND COOLDOWN BEFORE FILLING
SSV	INTEGRITY MAINTAINED OR RESTORED IN SG
DA2	DEPRESSURIZE AND COOLDOWN AFTER FILLING
DA3	COOLDOWN AND DEPRESSURIZE PER ECA-3.1/.2
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

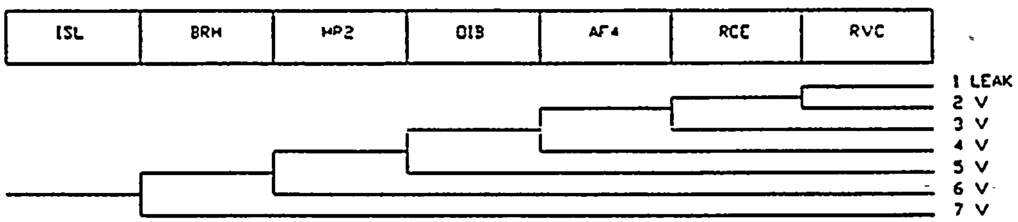
CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
GHR	SGTR, HI PRESSURE, CT HEAT REMOVAL
GHRI	SGTR, CT HEAT REMOVAL SUCCESS, HI FAILS
GHRIF	SGTR, HI, CF FAIL, CT HEAT REMOVED
GHC	SGTR, HI PRESSURE, CSR FAILS
GHCI	SGTR, HI PRESSURE, CSR & HI FAIL
GHCIF	SGTR, HI PRESSURE, CSR, HI, CF FAIL
GHV	SGTR, HI PRESSURE, CT DRY
GHVI	SGTR, HI PRESSURE, CT DRY, HI FAILS
GHVIF	SGTR, HI PRESSURE, CT DRY, HI & CF FAIL
LEAK	NO CORE DAMAGE, PRIMARY COOLANT RELEASED
GH	SGTR, HI PRESSURE
GHI	SGTR, HI PRESSURE, HI FAILS
GHIF	SGTR, HI PRESSURE, HI & CF FAIL

Figure 3.1-4
Steam Generator Tube Rupture Event Tree

SI
APERTURE
CARD

Also Available On
Aperture Card

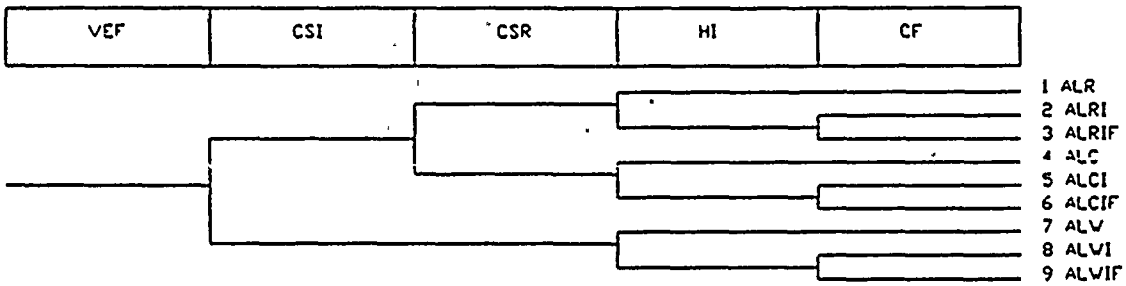
9205050333-02



EVENT	EVENT NAME
ISL	INTERFACING SYSTEMS LOCA
BRH	RHR SYSTEM BREACH
HP2	ECCS (HIGH PRESSURE) INJECTION
OIB	OPERATOR ACTION TO ISOLATE RHR SEAL LOCA
AF4	AUXILIARY FEEDWATER ACTUATION
RCE	RCS COOLDOWN AND RWST CONSERVATION
RVC	RHR RELIEF VALVE CLOSURE

CATEGORY	DESCRIPTION
LEAK	NO CORE DAMAGE, PRIMARY COOLANT RELEASED
V	CORE DAMAGE, INTERFACING SYSTEM LOCA

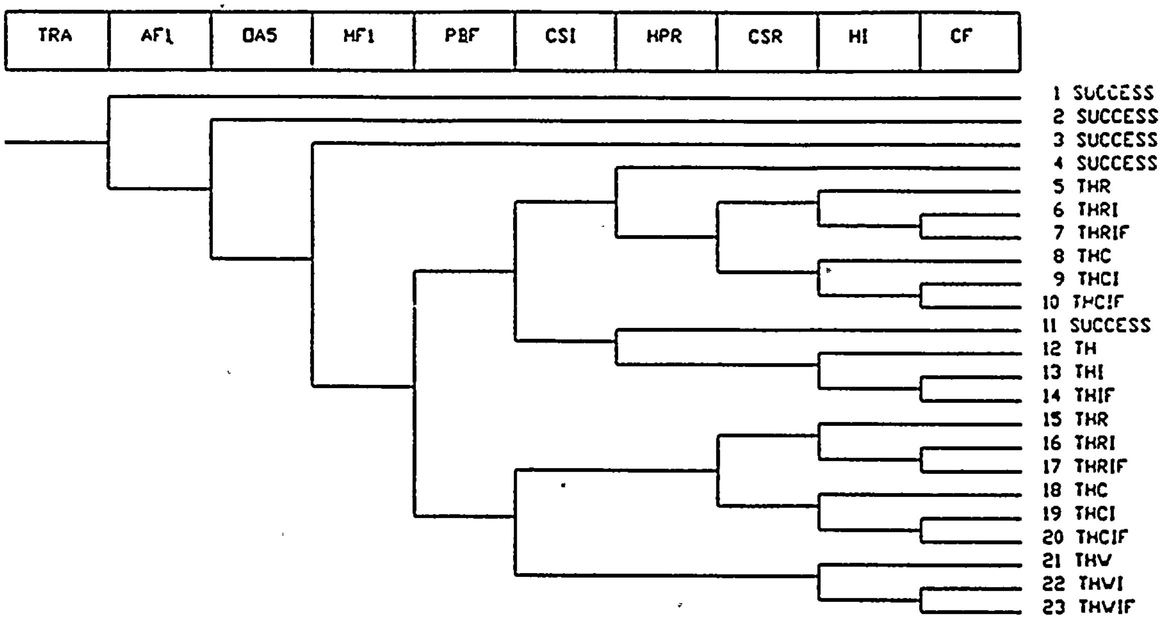
Figure 3.1-5
Interfacing Systems LOCA Event Tree



EVENT	EVENT NAME
VEF	BREAK BEYOND ECCS CAPABILITY INITIATING
CSI	CONTAINMENT SPRAY INJECTION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
ALR	LARGE LOCA, LOW PRES, CT HEAT REMOVAL
ALRI	LG LOCA CT HEAT REMOVAL SUCCESS HI FAILS
ALRIF	LG LOCA, HI, CF FAIL, CT HEAT REMOVED
ALC	LARGE LOCA, LOW PRESSURE, CSR FAILS
ALCI	LARGE LOCA, LOW PRESSURE, CSR & HI FAIL
ALCIF	LARGE LOCA, LOW PRESSURE, CSR,HI,CF FAIL
ALV	LARGE LOCA, LOW PRESSURE, CT DRY
ALVI	LARGE LOCA, LOW PRESS. CT DRY, HI FAILS
ALVIF	LG LOCA, LOW PRES, CT DRY, HI & CF FAIL

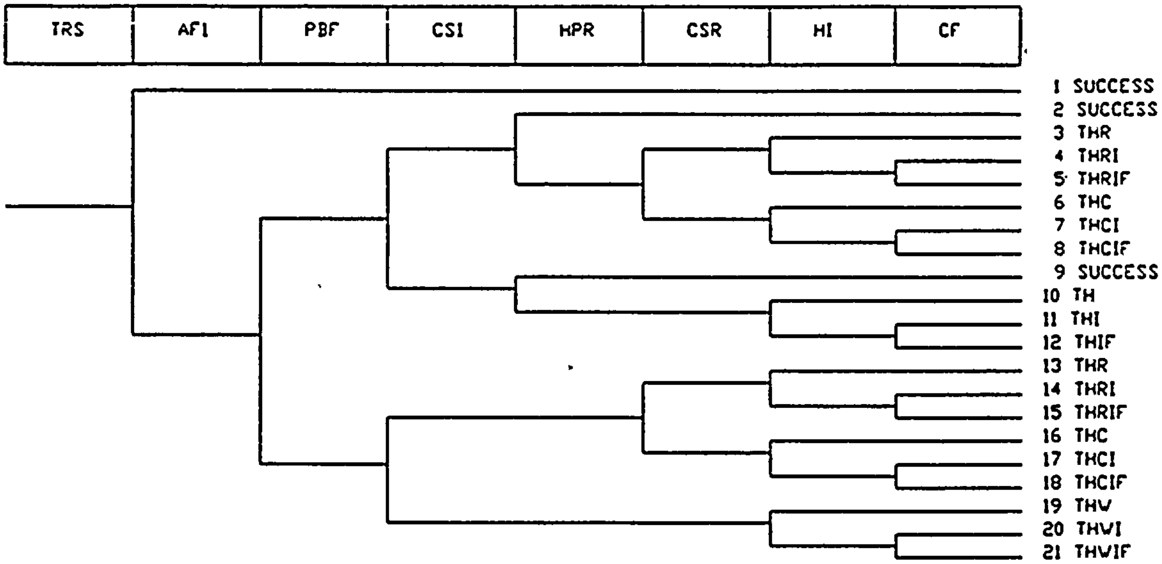
Figure 3.1-6
Breaks Beyond ECCS Capability Event Tree



EVENT	EVENT NAME
TRA	TRANSIENTS W/STEAM CONV. SYSTEMS AVAIL.
AF1	AUXILIARY FEEDWATER ACTUATION
OA5	OA TO DEPRESSURIZE A STEAM GENERATOR
MF1	MAIN FEEDWATER
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

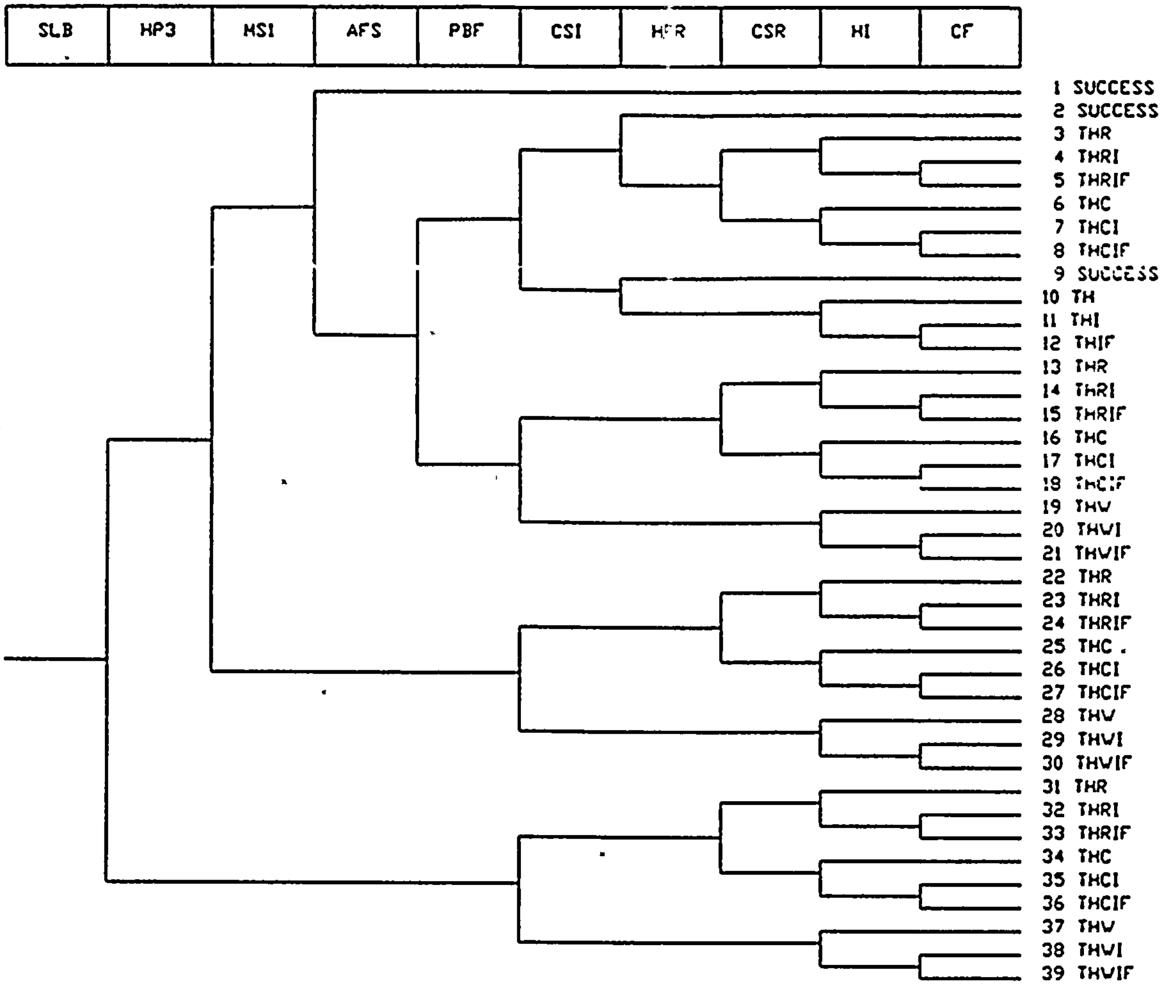
Figure 3.1-7
Transients With Steam Conversion Systems Available Event Tree



EVENT	EVENT NAME
TRS	TRANSIENTS W/O STEAM CONVERSION SYSTEMS
AFI	AUXILIARY FEEDWATER ACTUATION
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

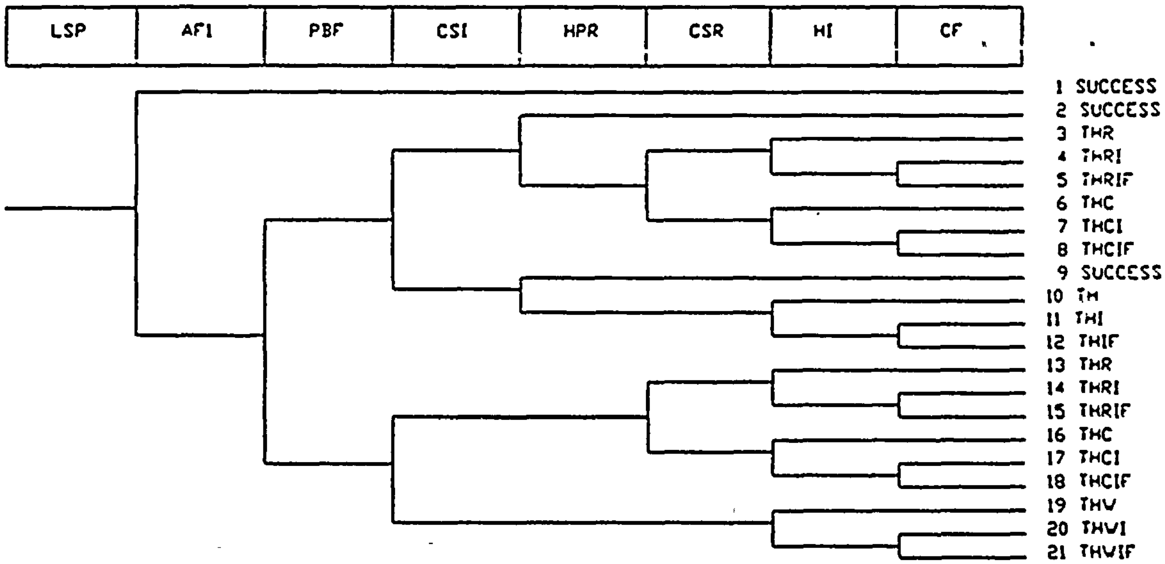
Figure 3.1-8
Transients Without Steam Conversion Systems Available Event Tree



EVENT	EVENT NAME
SLB	STEAM LINE BREAK INITIATING EVENT
HP3	ECCS (CHARGING PUMP) INJECTION/BORATION
MSI	SECONDARY SIDE ISOLATION
AFS	AUXILIARY FEEDWATER ACTUATION W/SLB
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-9
Large Steam Line/Feedline Break Event Tree



EVENT	EVENT NAME
LSP	LOSS OF OFFSITE POVER
AFI	AUXILIARY FEEDVATER ACTUATION
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

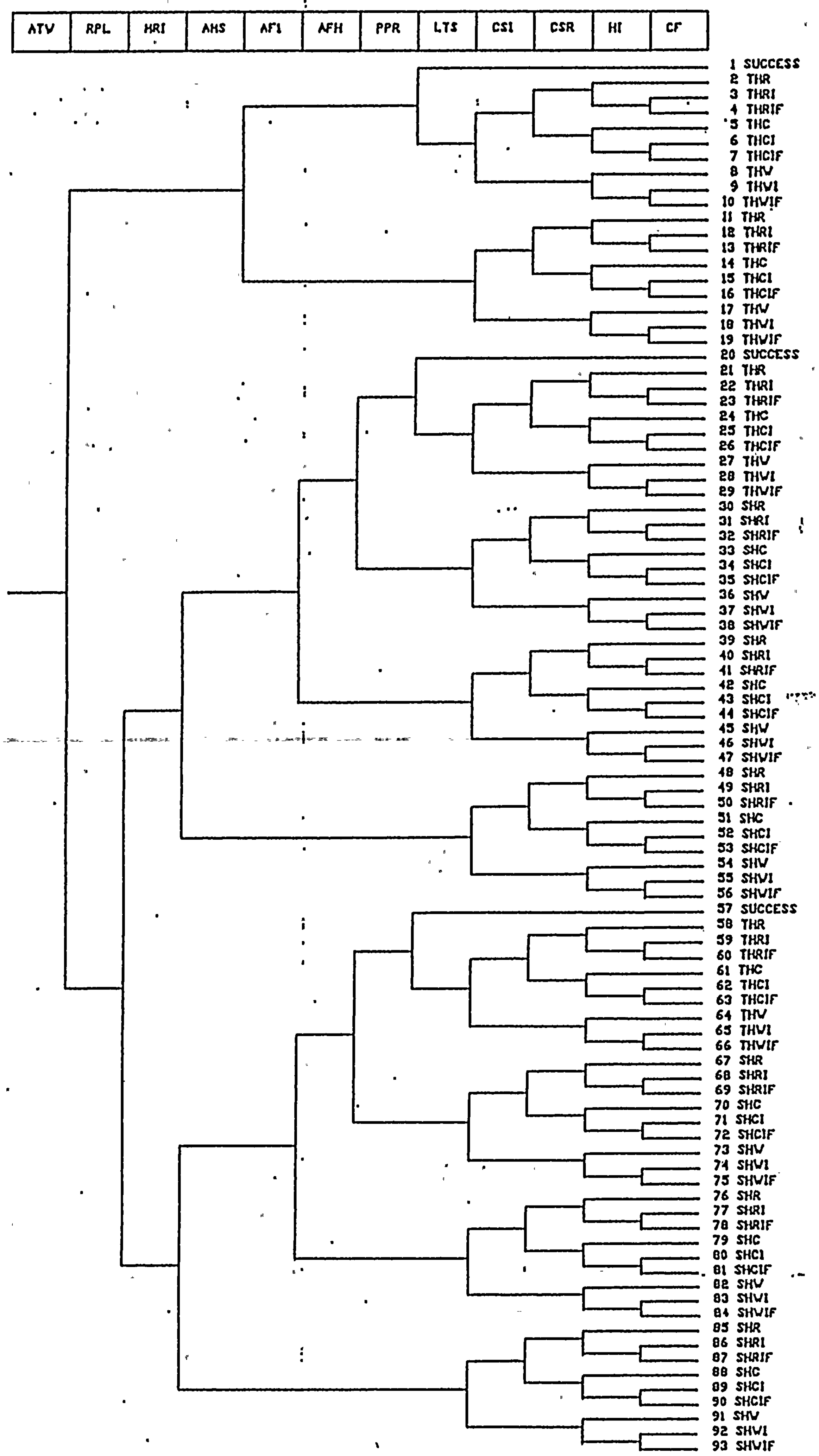
CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-10
Loss of Offsite Power Event Tree

EVENT	EVENT NAME
SBO	STATION BLACKOUT EVENT INITIATING EVENT
AFT	TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP
RCD	RCS COOLDOWN
AFC	AUXILIARY FEEDWATER CONTINUES
XHR	POWER RESTORED IN X HOURS
CNU	CORE NOT UNCOVERED
RRI	RESTORE RCS INVENTORY
AF1	AUXILIARY FEEDWATER ACTUATION
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THW	TRANSIENT, HIGH PRESSURE, CT DRY
THWI	TRANSIENT, HIGH PRESS. CT DRY, HI FAILS
THWIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-11
Station Blackout Event Tree



EVENT	EVENT NAME
ATV	ANTICIPATED TRANSIENT WITHOUT SCRAM
RPL	REDUCED POWER LEVEL (LESS THAN 40%)
HRI	MANUAL ROD INSERTION
AHS	ATWS MITIGATION SYSTEM ACTUATION CKTRY
AFI	AUXILIARY FEEDWATER ACTUATION
AFH	900 GPM AUX FEEDWATER FLOW
PPR	PRIMARY PRESSURE RELIEF
LTS	LONG TERM SHUTDOWN
CSI	CONTAINMENT SPRAY INJECTION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITORS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS. CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL
SHR	SMALL LOCA, HIGH PRESS, CT HEAT REMOVAL
SHRI	SM LOCA CT HEAT REMOVAL SUCCESS HI FAILS
SHRIF	SM LOCA, HI, CF FAIL, CT HEAT REMOVED
SHC	SMALL LOCA, HIGH PRESSURE, CSR FAILS
SHCI	SMALL LOCA, HIGH PRESSURE, CSR & HI FAIL
SHCIF	SM LOCA, HIGH PRESSURE, CSR, HI, CF FAIL
SHV	SMALL LOCA, HIGH PRESSURE, CT DRY
SHVI	SMALL LOCA, HIGH PRESS. CT DRY, HI FAILS
SHVIF	SM LOCA, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-12
Anticipated Transient without Scram Event Tree

SI
APERTURE
CARD

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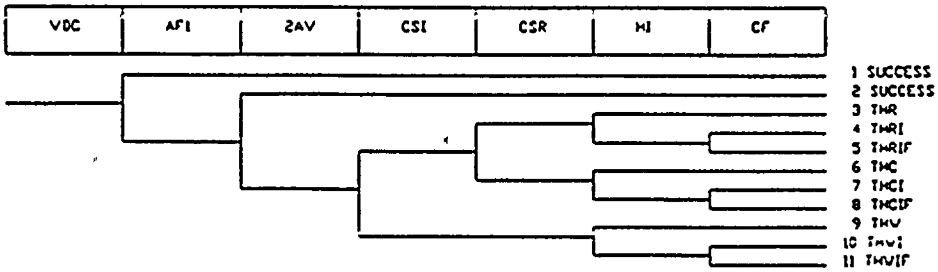
Aperture Card

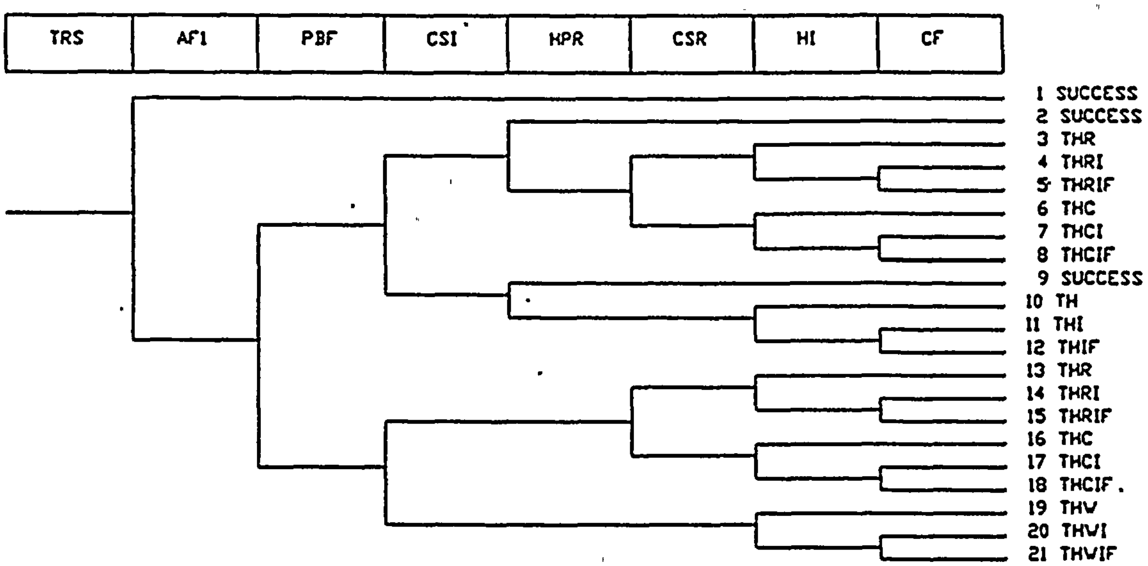
9205050899



EVENT	EVENT NAME
VDC	LOSS OF SINGLE TRAIN OF 250 V DC
AFI	AUXILIARY FEEDWATER ACTUATION
2AV	OPERATOR ACTION TO PROVIDE UNIT 2 AFV
CSI	CONTAINMENT SPRAY INJECTION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-15
Loss of a Single Train of 250 VDC Event Tree



EVENT	EVENT NAME
TRS	TRANSIENTS W/O STEAM CONVERSION SYSTEMS
AF1	AUXILIARY FEEDWATER ACTUATION
PBF	PRIMARY BLEED AND FEED
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THV	TRANSIENT, HIGH PRESSURE, CT DRY
THVI	TRANSIENT, HIGH PRESS, CT DRY, HI FAILS
THVIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-16
Internal Flooding Event Tree

TABLE 3.1-2

LARGE LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Accumulators (ACC)	3 of 3 accumulators inject to the intact cold legs	None	None	--	1
RHR (Low Pressure) Injection (LPI)	1 of 2 RHR Pumps to 1 of 3 intact cold legs	Electric Power, Component Cooling Water, SI Signal	None	0.5	1
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1
RHR (Low Pressure) Recirculation (LPR)	1 of 2 RHR trains switched from the RWST to the recirculation sump and restarted to 1 of 3 intact legs	Electrical Power, Component Cooling Water	Manual valve changes in RHR system, pumps restarted	24	4, 33

TABLE 3.1-2 (Cont'd.)

LARGE LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1, 4
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-3

MEDIUM LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Accumulators (ACC)	3 of 3 accumulators inject to the intact cold legs	None	None	--	33
ECCS (High Pressure) Injection (HP2)	1 of 2 CCPs and 1 of 2 SI pumps to 1 of 3 intact loops (SI cross-tie assumed shut)	Electric Power, SI Signal, Component Cooling Water	None	0.5	1
RCS Cooldown Using AFW and Steam Dump (OA6)	450 gpm of AFW to at Least 2 of 4 Steam Generators at Least 2 of 4 S/G PORVS Opened	Electric Power, Control Air, Main Steam	Initiate Cooldown within 2 Hours following accident. initiation	24	33

TABLE 3.1-3 (Cont'd.)

MEDIUM LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Depressurization and Low Pressure Injection (OLI)	RCS depressurized by dumping steam from at least 2 of 4 S/G PORVs, and opening 2 of 3 pressurizer PORVs, 1 of 2 RHR Pumps injecting to 1 of 3 intact cold legs, 450 gpm of AFW to 2 of 4 S/Gs	Electric Power, Component Cooling Water, Control Air, Main Steam, Pressurizer PORVs	Depressurize RCS within 20 minutes of accident initiation, verify RHR injection	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1
High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33

TABLE 3.1-3 (Cont'd.)

MEDIUM LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
RHR (Low Pressure) Recirculation (LPR)	1 of 2 RHR trains switched from the RWST to the recirculation sump and restarted to 1 of 3 intact legs	Electrical Power, Component Cooling Water	Manual valve changes in RHR system, pumps restarted	24	4, 33
Containment Spray Recirculation. (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electric Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-4

SMALL LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
ECCS (High Pressure) Injection (HP2)	1 of 2 CCPs and 1 of 2 SI pumps to 1 of 3 intact loops (SI cross-tie assumed shut)	Electric Power, SI Signal, Component Cooling Water	None	0.5	1
RCS Cooldown Using AFW and Steam Dump (OA6)	450 gpm of AFW to at Least 2 of 4 Steam Generators at Least 2 of 4 S/G PORVS Opened	Electric Power, Control Air, Main Steam	Initiate Cooldown within 2 Hours following accident initiation	24	33
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power Hi-Hi Containment pressure signal	None	0.5	1

TABLE 3.1-4 (Cont'd.)

SMALL LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1, 4
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-5

STEAM GENERATOR TUBE RUPTURE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater to the Intact S/G (AF2)	1 of 3 aux feed pumps to 1 intact S/G	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	45
Auxiliary Feedwater to the Faulted S/G (AF3)	1 of 2 aux feed pumps to the faulted S/G	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	45
ECCS (High Pressure) Injection (HPI)	1 of 4 SI or CCPs to 1 of 4 Cold Legs	Electric Power, Component Cooling Water, Start Signal	None	24	1
S/G Isolation by MSIV Closure (SGI)	Closure of MSIV on faulted S/G	Electric Power	Closure of MSIV on faulted S/G	N/A	4

TABLE 3.1-5 (Cont'd.)

STEAM GENERATOR TUBE RUPTURE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Cooldown & Depressurization Before Faulted S/G Fills (OA1)	RCS pressure about faulted S/G pressure within 30 min using 2 of 4 SG PORVs. Start cooldown within 20 min.	HPSI, AFW, Main Steam, Electric Power, Control Air, CCW	Cooldown and Depressurize, Terminate SI	24	1, 4
Integrity Maintained or Restored in Faulted Steam Generator (SSV)	All secondary relief valves in faulted S/G remain closed	None	None	N/A	Engineering Judgement
Cooldown & Depressurization After Faulted S/G Fills (OA2)	RCS pressure about faulted S/G pressure within 1 hr. using 2 of 4 SG PORVs. Start cooldown within 30 min.	HPSI, AFW, Electric Power, Control Air, CCW, Main Steam	Cooldown and Depressurize, Terminate SI	24	1, 4

TABLE 3.1-5 (Cont'd.)

STEAM GENERATOR TUBE RUPTURE SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Cooldown & Depressurization per ECA-3.1/3.2 (OA3)	Faulted S/G and RCS reduced to atmospheric pressure using 2 of 4 S/G PORVs and 1 of 3 pressurizer PORVs prior to draining the RWST	HPSI, AFW, Electric Power, CCW, Control Air, Main Steam	Cooldown, Depressurize, Reduce SI	24	1, 4
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1

TABLE 3.1-5 (Cont'd.)

STEAM GENERATOR TUBE RUPTURE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	1, 4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water,	Manual valve changes in CS system, pumps restarted	24	1
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal CCW	None	24	Engineering Judgement

TABLE 3.1-6

INTERFACING SYSTEM LOCA SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
RHR System Breach (BRH)	RHR piping does not fail	None	None	N/A	34
ECCS High Pressure Injection (HP2)	1 of 2 CCPs and 1 of 2 SI Pumps to 1 of 3 Intact Cold Legs	Electric Power, SI Signal, Component Cooling Water	None	0.5	33
Operator Action to Isolate the RHR Seal LOCA (OIB)	Closure of RHR Suction Valves IMO-310 and IMO-320	Electric Power	Closure of both MOVs	N/A	4
Auxiliary Feedwater Actuation (AF4)	450 gpm of AFW to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on on depletion of CST	24	33

TABLE 3.1-6 (Cont'd.)

INTERFACING SYSTEM LOCA
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
RCS Cooldown and RWST Conservation (RCE)	Cooldown and depressurize using 2 of 4 SG PORVs and 1 of 3 Pressurizer PORVs	Main Steam, Electric Power, Control Air, Pressurizer PORVs	Perform cooldown and depressurization to less than 450 psig prior to RWST depletion, Stop CTS pumps, Terminate SI Realign normal charging	24	4, 33
Relief Valve Closure (RVC)	RHR relief valve closes	None	None	N/A	Engineering Judgement

TABLE 3.1-7

BREAKS BEYOND ECCS CAPABILITY SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	Engineering Judgement
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in system, pumps restarted	24	Engineering Judgement
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-8

TRANSIENTS WITH STEAM CONVERSION SYSTEMS AVAILABLE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	1
Main Feedwater (MF1)	1 out of 2 main feedwater pumps to 2 of 4 SGs- or 450 gpm AFW from the opposite Unit to 2 of 4 SGs	Electric Power, Condensate, Main Steam, AFW	Defeat MFW pump trip and MFW isolation signals Start pumps, Manual valve changes to align systems	24	1, 4
Operator Action to Depressurize Steam Generators (OA5)	Supply at least 2 steam generators from at least one condensate booster pump within 30 minutes	Electric Power, Condensate, Main Steam, Control Air, Main Steam	Depressurize SGs to allow condensate flow	0.5	1, 4

TABLE 3.1-8 (Cont'd.)

TRANSIENTS WITH STEAM CONVERSION SYSTEMS AVAILABLE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-8 (Cont'd.)

TRANSIENTS WITH STEAM CONVERSION SYSTEMS AVAILABLE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-9

TRANSIENTS WITHOUT STEAM CONVERSION SYSTEMS AVAILABLE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	1
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1

TABLE 3.1-9 (Cont'd.)

TRANSIENTS WITHOUT STEAM CONVERSION SYSTEMS AVAILABLE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1, 4
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal CCW	None	24	Engineering Judgement

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-10

LARGE STEAM LINE/FEEDLINE RUPTURE
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
ECCS (Charging Pump) Injection/ Boration (HP3)	1 of 2 CCPs to 1 of 4 loops with Boration from BIT	Electric Power, SI Signal, Component Cooling Water	None	0.5	1
Secondary Side Isolation (MS1)	Shut 3 of 4 MSIVs	Electric Power, Isolation Signal	None	N/A	1 & Engineering Judgement
Auxiliary Feedwater Actuation with Steam Line Rupture	600 gpm AFW flow to 2 of 3 intact steam generators AFW isolated to faulted steam generator	Electric Power, Start Signal	Provide additional water supply on depletion of CST, Ý(AFS) Isolate faulted S/G	24	1
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33

TABLE 3.1-10 (Cont'd.)

LARGE STEAM LINE/FEEDLINE BREAK
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-10 (Cont'd.)

LARGE STEAM LINE/FEEDLINE BREAK
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-11

LOSS OF OFFSITE POWER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	1
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1

TABLE 3.1-11 (Cont'd.)

LOSS OF OFFSITE POWER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water Electrical Power RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-12

STATION BLACKOUT
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Turbine-Driven Auxiliary Feedwater Pump (AFT)	Turbine-driven pump delivers flow to 2 Sgs	N-Train Battery, ESFAS Signal	Verification or manual starting of pump	4	39
RCS Cooldown (RCD)	RCS cooled down by 2 of 4 steam generators PORVs - Start within 60 min.	Main Steam, Nitrogen	Open S/G PORV's, depressurize S/Gs to 200 psig within 2 hours after SBO	24	33, 39
Auxiliary Feedwater Continues (AFC)	Turbine-driven pump continues to deliver flow to 2 of 4 S/Gs for an additional 2 hours past AFT mission time.	None	Verification	2	39

TABLE 3.1-12 (Cont'd.)

STATION BLACKOUT
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Power Restored Within X Hours (XHR)	Power restored to AC power system within X hours, X is defined in Reference 47.	None	Verification or manual starting of diesel generator, various manual breaker alignments, restart motors	N/A	33, 39
Core Not Uncovered (CNU)	Core not uncovered after power is restored	None	None	N/A	39
Restore RCS Inventory (RRI)	Restore safeguards systems, initiate SI with 1 of 4 SI or CC pumps delivering to the RCS	Electric Power, CCW	Restore systems, start pumps	24	1, 39
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power	Provide additional water supply on depletion of CST, Start pumps	24	33

TABLE 3.1-12 (Cont'd.)

STATION BLACKOUT
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power	Start system	0.5	1
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water Electrical Power RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-12 (Cont'd.)

STATION BLACKOUT
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-13

ANTICIPATED TRANSIENT WITHOUT SCRAM
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Reduced Power Level (RPL)	Transient Initiated From Less Than 40% Power	None	None	N/A	42
Manual Rod Insertion (MRI)	One Minute of Manual Rod Insertion Within One Minute	Electric Power	Manually Insert Control Rods	N/A	42
AMSAC (AMS)	Trip Turbine and Generate a Signal to Start AFW Pumps	Electric Power	None	N/A	42
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	42

TABLE 3.1-13 (Cont'd.)

ANTICIPATED TRANSIENT WITHOUT SCRAM
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
900 gpm Aux Feedwater Flow (AFH)	900 gpm AFW Flow to 2 of 4 Steam Generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	42
Primary Pressure Relief (PPR)	No unfavorable exposure time, 3 of 3 Safety Valves, and either: a) 3 of 3 PORVs if MRI is successful, or: b) 1 of 3 PORVs if MRI fails.	Electric Power, Control Air	Open Block Valves if Necessary	24	42
Long Term Shutdown (LTS)	1 of 2 CCPs or establish subcriticality by other than boration	Electric Power, CCW	Trip Control Rod Motor Generator Sets, Locally Open Reactor Trip Breakers, Manually insert all control rods fully, or initiate boration	24	4, 42

TABLE 3.1-13 (Cont'd.)

ANTICIPATED TRANSIENT WITHOUT SCRAM
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1, 4
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

TABLE 3.1-14

LOSS OF ESSENTIAL SERVICE WATER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Trip RCPs (RCP)	All RCPs tripped following loss of seal cooling	None	Trip RCPs upon loss of seal cooling	N/A	39 & Engineering Judgement
Auxiliary Feedwater Actuation (AF1)	450 gpm of AFW flow to 2 out of 4 steam generators	Electric Power	Provide additional water water supply on depletion of CST	24	39
Main Feedwater (MF1)	1 out of 2 main feedwater pumps to 2 of 4 SGs- or 450 gpm AFW from the opposite Unit to 2 of 4 SGs	Electric Power Condensate, Main Steam	Defeat MFW pump trip and MFW isolation signals Start pumps, Manual valve changes to align systems	24	4, 39
Restore ESW System Within One Hour (EH1)	Restoration of flow to one ESW header	None	Restore system (one train)	1	4, 39

TABLE 3.1-14 (Cont'd.)

LOSS OF ESSENTIAL SERVICE WATER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
RCS Cooldown (RCD)	RCS cooldown by 2 of 4 S/G PORVs - start within 60 minutes	Electric Power, Control Air Main Steam	Open S/G PORVs, depressurize S/G to initiate RCS cooldown	24	33, 39
Restore ESW System Within Eight Hours (EH8)	Restoration of flow to one ESW header	None	Restore system (one train)	8	4, 39
Core Not Uncovered (CNU)	Core not uncovered after ESW cooling is restored	None	None	N/A	39
Restore RCS Inventory (RRI)	Restore safeguards systems using 1 of 4 SI/CC pumps to 3/4 RCS cold legs	Electric Power, CCW	Restore systems, start pumps	24	4, 39
Containment Spray Injection (CSI)	1 of 2 CS pumps	Electric Power, Hi-Hi Containment Pressure Signal	None	0.5	1

TABLE 3.1-14 (Cont'd.)

LOSS OF ESSENTIAL SERVICE WATER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
* High Pressure Cold Leg Recirculation (HPR)	1 of 4 SI or CCP pumps 1/2 RHR trains	Electric Power, CCW, RHR	Manual valve changes in SI system, isolate RWST, align cooling water to HX, open sump valves, start pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CTS pumps Sump & HX	Electric Power, Essential Service Water, Hi-Hi Containment Pressure Signal	Manual valve changes, pumps restarted	24	1
Hydrogen Igniters (HI)	1/2 trains operating	Electric Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1/2 Containment Recirculation Fan Trains	Electric Power, Hi-Hi Containment Pressure Signal, CCW	None	24	Engineering Judgement

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-15

LOSS OF COMPONENT COOLING WATER SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Trip RCPs (RCP)	All RCPs tripped following loss of seal cooling	None	Trip RCPs upon loss of seal cooling	N/A	39 & Engineering Judgement
Auxiliary Feedwater Actuation (AF1)	450 gpm of AFW flow to 2 out of 4 steam generators	Electric Power	Provide additional water water supply on depletion of CST	24	39
Main Feedwater (MF1)	1 out of 2 main feedwater pumps to 2 of 4 SGs- or 450 gpm AFW from the opposite Unit to 2 of 4 SGs	Electric Power Condensate, Steam	Defeat MFW pump trip and MFW isolation signals Start pumps, Manual valve changes to align systems	24	4, 39
Restore CCW System Within One Hour (CH1)	Restoration of flow to one CCW header	None	Restore system (one train)	1	4, 39

TABLE 3.1-15 (Cont'd.)

LOSS OF COMPONENT COOLING WATER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
RCS Cooldown (RCD)	RCS cooldown by 2 of 4 S/G PORVs - start within 60 minutes	Electric Power, Control Air, Main Steam	Open S/G PORVs, depressurize S/G to initiate RCS cooldown	24	33, 39
Restore CCW System Within Eight Hours (CH8)	Restoration of flow to one CCW header	None	Restore system (one train)	8	4, 39
Core Not Uncovered (CNU)	Core not uncovered after CCW cooling is restored	None	None	N/A	39
Restore RCS Inventory (RRI)	Restore safeguards systems using 1 of 4 SI/CC pumps to 3/4 RCS cold legs	Electric Power, CCW	Restore systems, start pump	24	4, 39
Containment Spray Injection (CSI)	1 of 2 CS pumps	Electric Power, Hi-Hi Containment Pressure Signal,	None	0.5	1

TABLE 3.1-15 (Cont'd.)

LOSS OF COMPONENT COOLING WATER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
* High Pressure Cold Leg Recirculation (HPR)	1 of 4 SI or CCP pumps 1/2 RHR trains	Electric Power, CCW, RHR	Manual valve changes in SI system, isolate RWST, align cooling water to HX, open sump valves, start pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CTS pumps Sump & HX	Electric Power, Essential Service Water, Hi-Hi Containment Pressure Signal	Manual valve changes, pumps restarted	24	1
Hydrogen Igniters (HI)	1/2 trains operating	Electric Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1/2 Containment Recirculation Fan Trains	Electric Power CCW	None	24	Engineering Judgement

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

TABLE 3.1-16

LOSS OF SINGLE TRAIN OF 250V DC ELECTRIC POWER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start signal	Provide additional water supply on depletion of CST	24	1
Unit 2 AFW Crosstie (2AV)	450 gpm Unit 2 AFW flow to 2 of 4 steam generators	Electric Power	Open crosstie valve, isolate Unit 2 S/Gs, start Unit 2 AFW pump	24	1, 4
Containment Spray Injection (CSI)	1 of 2 trains	Electrical Power, HI HI containment pressure signal	None	0.5	1
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical power Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1

TABLE 3.1-16 (Cont'd.)

LOSS OF SINGLE TRAIN OF 250V DC ELECTRIC POWER
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical power Hi-Hi containment pressure signal, CCW	None	24	Engineering Judgement

TABLE 3.1-17

INTERNAL FLOODING
SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
Auxiliary Feedwater Actuation (AF1)	450 gpm AFW flow to 2 of 4 steam generators	Electric Power, Start Signal	Provide additional water supply on depletion of CST	24	1
Primary Bleed and Feed (PBF)	Manually open 2 of 3 PORVs and block valves, 1 of 2 SI and 1 of 2 CC pumps	Electric Power, CCW to SI and CC pumps, Control Air, Pressurizer PORVs	Open 2 of 3 PORVs and block valves. Start pumps or verify pumps running	0.5	4, 33
Containment Spray Injection (CSI)	1 of 2 Trains	Electrical Power, Hi-Hi Containment pressure signal	None	0.5	1

TABLE 3.1-17 (Cont'd.)

INTERNAL FLOODING SYSTEM SUCCESS CRITERIA

Event Tree System	Equipment Success Criteria	System Dependencies	Operator Actions	Mission Time(hr)	References
* High Pressure Cold Leg Recirculation (HPR)	1 of 2 RHR trains supplying 1 of 2 SI and 1 of 2 CC pumps, to 1 of 3 intact legs	Component Cooling Water, Electrical Power, RHR	Manual valve changes in SI system Isolate RWST Open sump valves Restart pumps	24	4, 33
Containment Spray Recirculation (CSR)	1 of 2 CS trains switched from the RWST to the recirculation sump	Electrical Power, Essential Service Water	Manual valve changes in CS system, pumps restarted	24	1, 4
Hydrogen Igniters (HI)	1 of 2 trains in both upper and lower containment	Electrical Power	Energize Igniters	24	Engineering Judgement
Containment Recirculation Fans (CF)	1 of 2 trains operating	Electrical Power, Hi-Hi Containment Pressure Signal CCW	None	24	Engineering Judgement

* Conservatively assumed only three cold legs available to minimize the number of fault trees to be developed. This success criteria is consistent with the LOCA success criteria.

3.1.3 Special Event Trees

Several special event trees were modelled to accommodate consequential events and support system failures. These event trees are included in the above discussion.

3.1.4 Support System Event Tree

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

3.1.5 Sequence Grouping and Back-End Interface

The primary focus of the event tree analysis was to provide a quantifiable model of the safeguards and containment protection systems for the Level II analysis. Each sequence in the event tree that resulted in core damage was identified by a series of descriptors that indicated the type of event, state of the primary system, and state of containment protection systems. These descriptors were then used to group similar core damage states for the Level II analysis.

The first character in the descriptor is one of the following:

A Large LOCA - Characterized by rapid depressurization of the RCS and subsequent core uncovery.

S Small LOCA - Characterized by a small breach of the RCS pressure boundary resulting in a direct release path to the containment. RCS depressurization is expected to be limited by the size of the break and may even stall at relatively high pressure levels.

T Transient - An event which is not initiated by a breach of the RCS pressure boundary. Events in this category are characterized by the opening of pressurizer safety or relief valves with the associated loss of primary coolant subsequent to the event initiation.

G Steam Generator Tube Rupture - This event is initiated by the failure of the primary to secondary pressure boundary of one steam generator. In this event, primary coolant may be lost through steam conversion systems resulting in a direct release of radionuclides to the environment.

V Interfacing Systems LOCA - This event is characterized by a breach of the RCS pressure boundary which directly bypasses the containment (other than the steam generator tube rupture). Since primary coolant is lost outside the containment, no recirculation capability exists and no water is in the containment to provide for fission product scrubbing following core melt. This results in a large release of radionuclides outside the containment building.

The second character is one of the following:

H RCS at High Pressure - Indicating that, when vessel failure occurs following core damage, the RCS is at high pressure, generally taken to be above 200 psig.

L RCS at Low Pressure - Indicating that, when vessel failure occurs following core damage, the RCS is at low pressure, generally taken to be below 200 psig.

Following the first two characters, the following characters may be used either alone or in combination to classify the status of the containment and the containment protection systems:

- W The RWST inventory has not been emptied into containment indicating that the reactor cavity is dry.
- C Containment spray injection was successful, however, containment spray recirculation was disabled.
- R Containment spray injection and recirculation were successful.
- I Hydrogen Igniter System failure.
- F Containment Recirculation Fan system failure.

Those sequences that do not result in core damage are identified by one of the following two descriptors:

- SUCCESS Core damage was prevented and the containment either remained intact or was not challenged.
- LEAK Core damage was prevented, however, a breach of the containment resulted in a loss of primary coolant to the outside atmosphere.

No binning of plant damage states was done in the front end portion of the analysis other than to assign the descriptors described above. All binning of plant damage states was done in the Level 2 portion of the analysis. The process of binning is described in Section 4.6 of this report.

3.2 Systems Analysis

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

3.2.1 System Descriptions

To ensure that the IPE accurately represented how the plant's systems contribute to the overall risk profile, a thorough understanding of key frontline and support systems was essential. Prior to the development of the fault tree logic models, a comprehensive collection, evaluation, and documentation of information was performed for each system. This information was consolidated into a single reference notebook for each system. The outline used for each of these system notebooks is given in Table 3.2-1. The first six sections of the system notebooks contain the essential plant design and operational information needed to develop the fault trees. Included in these sections are the important dependencies reflected in the matrices described in Section 3.2.3 of this document, instrumentation and control requirements, and the results of a comprehensive review of equipment maintenance and surveillance practices. The plant walkdown, described in Section 2.4 of this report, was used to verify that the system model accurately reflected the system configuration and operating conditions. The results of a thorough operating experience review are also documented. This information was used to be sure that plant specific operating experience is reflected in the model development and in the quantification of system and component performance parameters. Section 7 contains the logic models and the assumptions used to construct the fault trees. Section 8 provides the results of the model quantification and Section 9 identifies the insights related to that system that were developed during the course of the IPE.

Because the Cook Nuclear Plant is a dual unit site, a careful examination of the documentation for both unit's systems was performed. Any key differences were identified and explicitly documented in the system notebooks. Shared systems or components were identified, including the degree of sharing and the systems' preferential alignments. Any unit-to-unit cross ties, along with the normal alignment and emergency alignment capabilities, were identified. Plant operating, surveillance and maintenance procedures were reviewed.

Table 3.2-2 lists the systems modeled in the Cook Nuclear Plant IPE. The remainder of this section provides a brief description of these systems and their important design features from a risk perspective. The symbols used in the simplified flow diagrams for these systems are shown in Figure 3.2-1.

TABLE 3.2-1

OUTLINE FOR SYSTEM NOTEBOOKS

1.0 SYSTEM FUNCTION

2.0 SYSTEM DESCRIPTION
- 2.1 Support Systems
- 2.2 Instrumentation and Controls
- 2.3 Technical Specification Limitations
- 2.4 Test and Maintenance
- 2.5 Component Location
- 2.6 Comparison of Units

3.0 SYSTEM OPERATION

4.0 PERFORMANCE DURING ACCIDENT CONDITIONS
- 4.1 Success Criteria
- 4.2 Initiator Impact on the System

5.0 OPERATING EXPERIENCE

6.0 INITIATING EVENT REVIEW

7.0 SYSTEM LOGIC MODELS
- 7.1 Assumptions and Boundary Conditions
- 7.2 Fault Tree Models

8.0 QUANTIFICATION AND RESULTS

9.0 SYSTEM INSIGHTS

10.0 REFERENCES

APPENDICES

A. Common Cause Calculations
B. Computer Code Input and Output
C. Supporting Technical Information

TABLE 3.2-2

SYSTEMS MODELLED IN THE COOK NUCLEAR PLANT IPE

Section 3.2.1.1	Auxiliary Feedwater System
Section 3.2.1.2	Containment Spray System
Section 3.2.1.3	Component Cooling Water System
Section 3.2.1.4	Essential Service Water System
Section 3.2.1.5	Nonessential Service Water System
Section 3.2.1.6	Compressed Air (Control Air) System
Section 3.2.1.7	Emergency Core Cooling System (Including safety injection/charging, residual heat removal, and accumulators)
Section 3.2.1.8	Electric Power System (Including 4160 VAC, 600 VAC, 120 VAC and 250 VDC)
Section 3.2.1.9	Main Feedwater and Condensate Systems
Section 3.2.1.10	Main Steam System
Section 3.2.1.11	Pressurizer Power Operated Relief Valve and Safety Valve System
Section 3.2.1.12	Containment Air Recirculation and Hydrogen Skimmer System
Section 3.2.1.13	Hydrogen Igniter (DIS) System
Section 3.2.1.14	Refueling Canal Drains

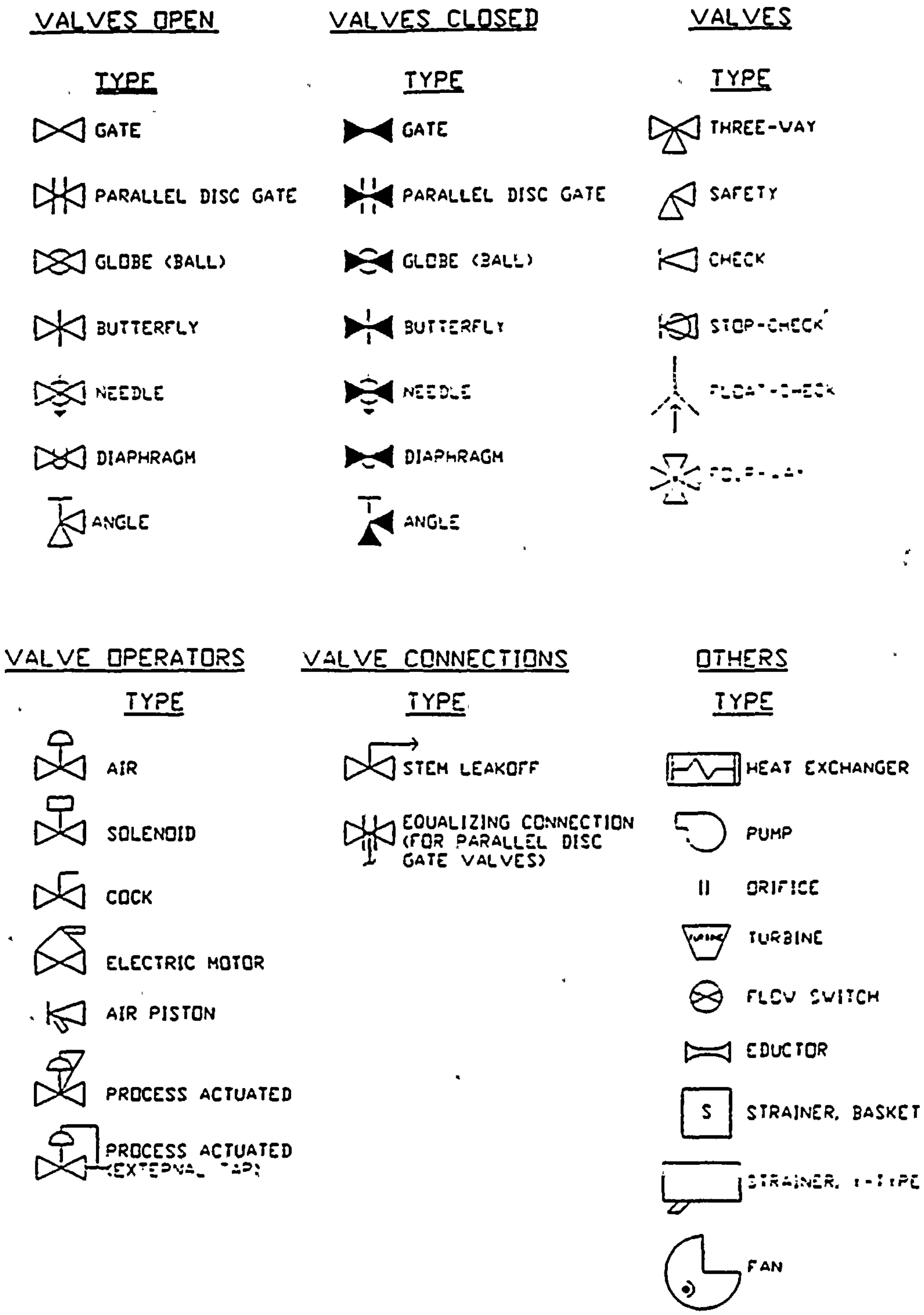


Figure 3.2-1
Flow Diagram Component Symbols

3.2.1.1 Auxiliary Feedwater System

The primary function of the auxiliary feedwater (AFW) system is to provide sufficient makeup to the steam generators to maintain a minimum heat transfer area to prevent loss of primary water through the pressurizer safety or relief valves when normal feedwater supply is not available. For the IPE analysis, the AFW system is the primary source of water to the steam generators when the steam generators are used to remove latent or decay heat from the reactor coolant system (RCS) and the core. Additionally, the AFW system is used to supply the steam generators during startup and shutdown when insufficient steam is available for the main feedwater pumps, however, this function is not modelled in this study. Figure 3.2-2 shows a simplified flow diagram of the system.

The primary source of water for the system is the unit's 500,000 gallon condensate storage tank (CST). An emergency backup supply is available from the opposite unit's CST through normally closed condensate storage tank cross-tie valve, 12-CRV-51. If both unit's CSTs are unavailable, then the system may be aligned to take suction from the essential service water (ESW) system but, this is only used after all other sources of condensate have been exhausted.

AFW flow is delivered to the steam generators by three pumps, one turbine-driven auxiliary feedwater pump (TDAFP), and two motor-driven auxiliary feedwater pumps (MDAFP). Each pump has its own suction strainer and discharge isolation and check valves. The TDAFP is capable of supplying all four steam generators. The east MDAFP supplies only steam generators two and three, and the west MDAFP supplies only steam generators one and four.

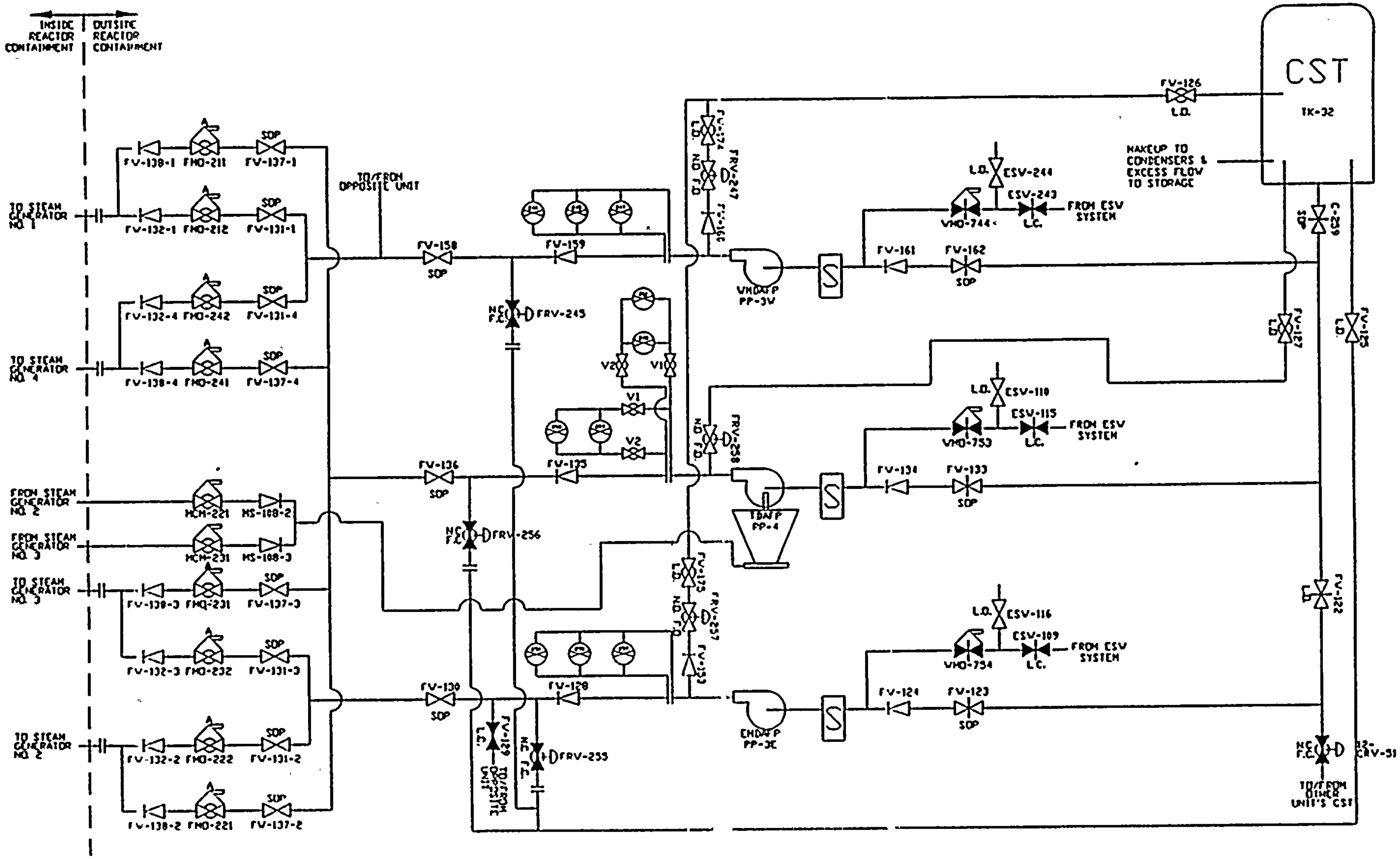
A cross tie between units is provided between the east MDAFP discharge isolation and check valves. This line, normally isolated by locked closed valve, FW-129, provides the ability for one unit's east MDAFP to supply the opposite unit's steam generators normally supplied by the west MDAFP and vice versa.

There are nine models (fault trees) associated with the Auxiliary Feedwater System. Each model represents the unavailability of this system in response to different accident events.

- AF1 - This system model defines the logic associated with the response of the AFW system to provide at least 450 gpm of flow to at least two of four steam generators.
- AFS - This system model defines the logic associated with the response of the AFW system to provide at least 600 GPM of flow to at least two of the three intact steam generators following a large steam line/feed line rupture event.
- AFT - This system model defines the logic associated with the response of the AFW system to provide flow from the TDAFP to at least two of four steam generators during a station blackout event.
- AFH - This system model defines the logic associated with the response of the AFW system to provide at least 50% capacity flow to the steam generators.
- AF2 - This system model defines the logic associated with the response of the AFW system during a SGTR event to provide flow from at least one of three pumps to at least one of three intact steam generators.
- AF3 - This system model defines the logic associated with the response of the AFW system during a SGTR event to provide flow from one of two pumps to the faulted steam generator.
- AFC - This system model defines the logic associated with the response of the AFW system to provide flow to at least two of four steam generators for an additional two hours after the N-Train battery is assumed lost.

AF4 - This system model defines the logic associated with the response of the AFW system to provide at least 450 gpm of flow to at least two of four steam generators during LOCA scenarios.

2AF - This system model defines the logic associated with the response of the AFW system to provide at least 450 gpm of AFW flow to at least two of four steam generators through use of the unit cross-tie connections.



NOTES

1. VALVE NOTATION DEFINITIONS
   a. NO=NORMALLY OPEN
   b. NC=NORMALLY CLOSED
   c. F.O.=FAILS OPEN
   d. F.C.=FAILS CLOSED
   e. L.O.=LOCKED OPEN
   f. L.C.=LOCKED CLOSED
   g. SOP=SEALED OPEN
2. A=VALVE OPENS AUTOMATICALLY ON AUTOMATIC START OF ITS ASSOCIATED AFW PUMP

REFERENCE DRAWINGS: OP-1-5106A-16
OP-2-5106A-15
OP-1-5107-35
OP-2-5107-30
OP-1-5107A-17
OP-2-5107A-23

Figure 3.2-2
Auxiliary Feedwater System Simplified Flow Diagram

3.2.1.2 Containment Spray System

The containment spray (CTS) system is required in the event of a loss of coolant accident (LOCA) or a main steam line break (MSLB) accident inside containment where containment pressure rises above 2.9 psig. The primary functions of the system include:

1) limiting the peak pressure inside the containment to below the containment design pressure of 12 psig, and

2) removal of radioactive isotopes from the containment atmosphere.

The CTS system performs the post-accident containment depressurization and long-term cooling function following a LOCA or MSLB, and it removes fission products from the containment atmosphere to reduce the radiological consequences of an accident. The CTS system, consisting of two 100% capacity, independent flow trains, is initiated only during an accident and delivers chemically-treated water to the spray ring headers in both lower and upper containment. Figures 3.2-3, 3.2-4, and 3.2-5 show simplified flow diagrams of the CTS system during standby conditions, the injection phase and the recirculation phase, respectively.

Each train of the containment spray system consists of a containment spray pump, a containment spray heat exchanger, valves, piping, spray headers in both the upper and lower containment volumes, and all necessary controls and instrumentation. The single refueling water storage tank (RWST) and spray additive tank serve the two trains.

The operation of the CTS system consists of two sequential modes, the injection phase and the recirculation phase. During the injection phase, a portion of the RWST is sprayed into the containment atmosphere via the CTS pumps. By means of an eductor, the NaOH solution in the spray additive tank is mixed with the spray. The recirculation phase begins after the pump suction has been switched over manually, by the control room operator, to the containment recirculation sump. Water is recirculated from the containment sump through a containment spray heat exchanger and back to the ring headers.

There are two fault trees associated with the CTS system. Each model represents the unavailability of this system as it responds to accident events.

CSI - This system model (fault tree) defines the response of the CTS system to provide coolant from the RWST to the upper and lower CTS ring headers. This model is also referred to as the "injection model".

CSR - This system model defines the response of the CTS system to recirculate coolant from the containment recirculation sump through the CTS heat exchangers and the upper and lower CTS ring headers. This model is also referred to as the "recirculation model."

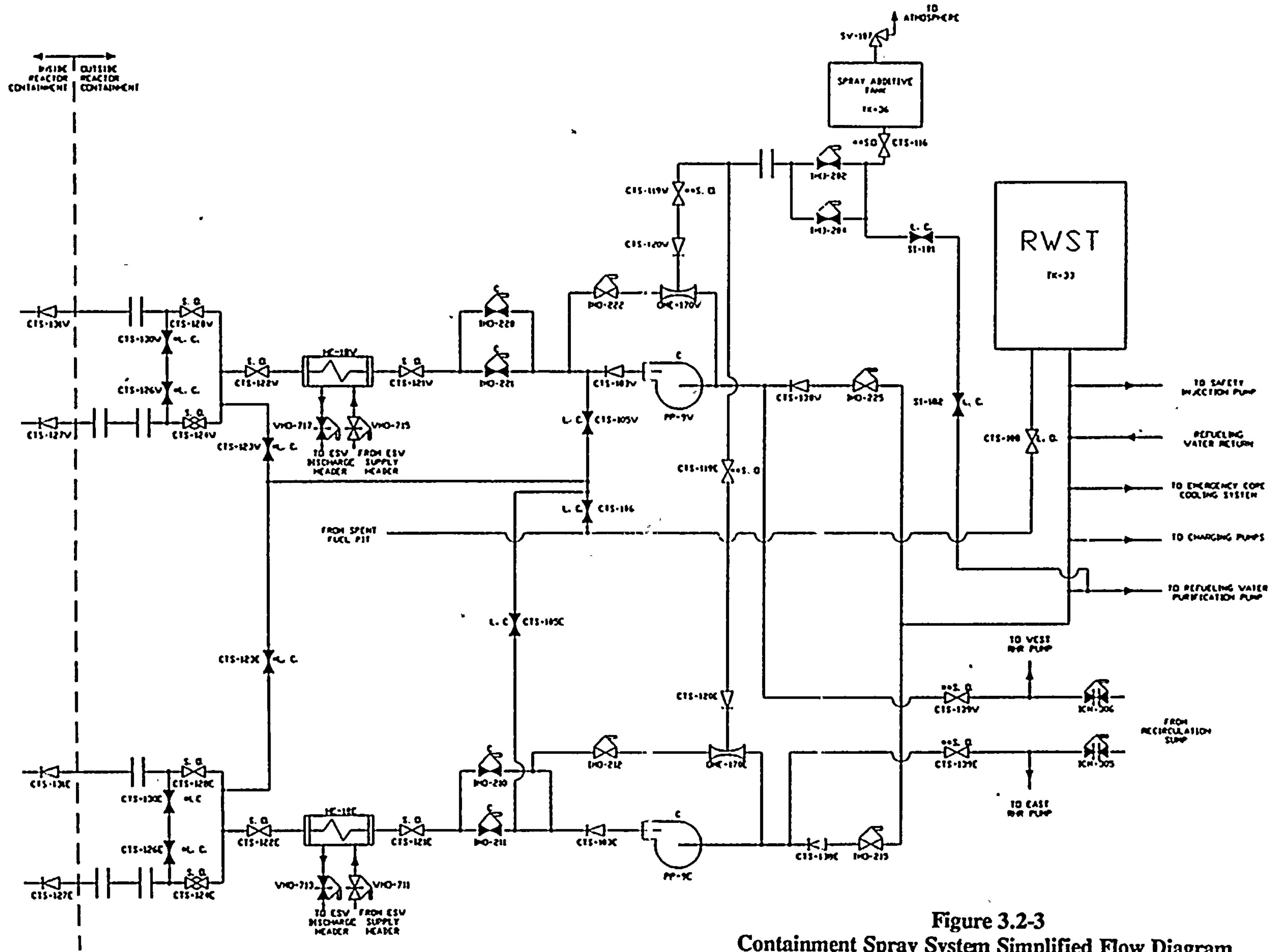


Figure 3.2-3
Containment Spray System Simplified Flow Diagram
- Standby Conditions

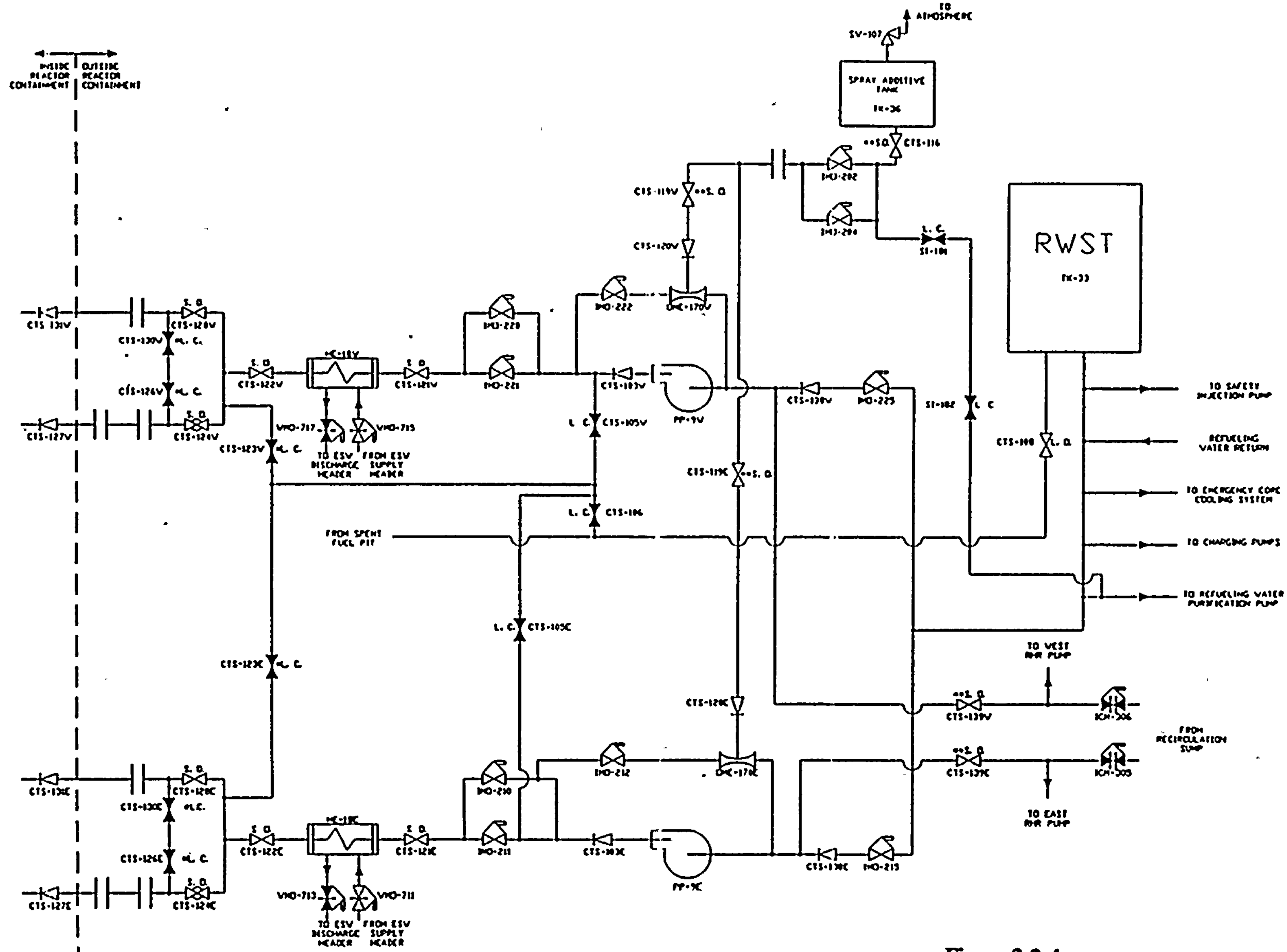
*REFERENCE 13 DESCRIBES THESE VALVES AS LOCKED CLOSED. REFERENCE 3A DOES NOT.

**REFERENCE 13 DESCRIBES THESE VALVES AS SEALED OPEN. REFERENCE 3A MODELS THEM AS LOCKED OPEN.

REFERENCE DRAWINGS:
OP-1-5144-10
OP-2-5144-13
OP-1-5143-20
OP-2-5143-19
OP-1-5113-20
OP-2-5113-25

NOTES
1. VALVE NOTATION DEFINITIONS
a) L. O. = LOCKED OPEN
b) L. C. = LOCKED CLOSED
c) S. O. = SEALED OPEN

2. C = COMPONENT ACTUATED BY A MI-MS



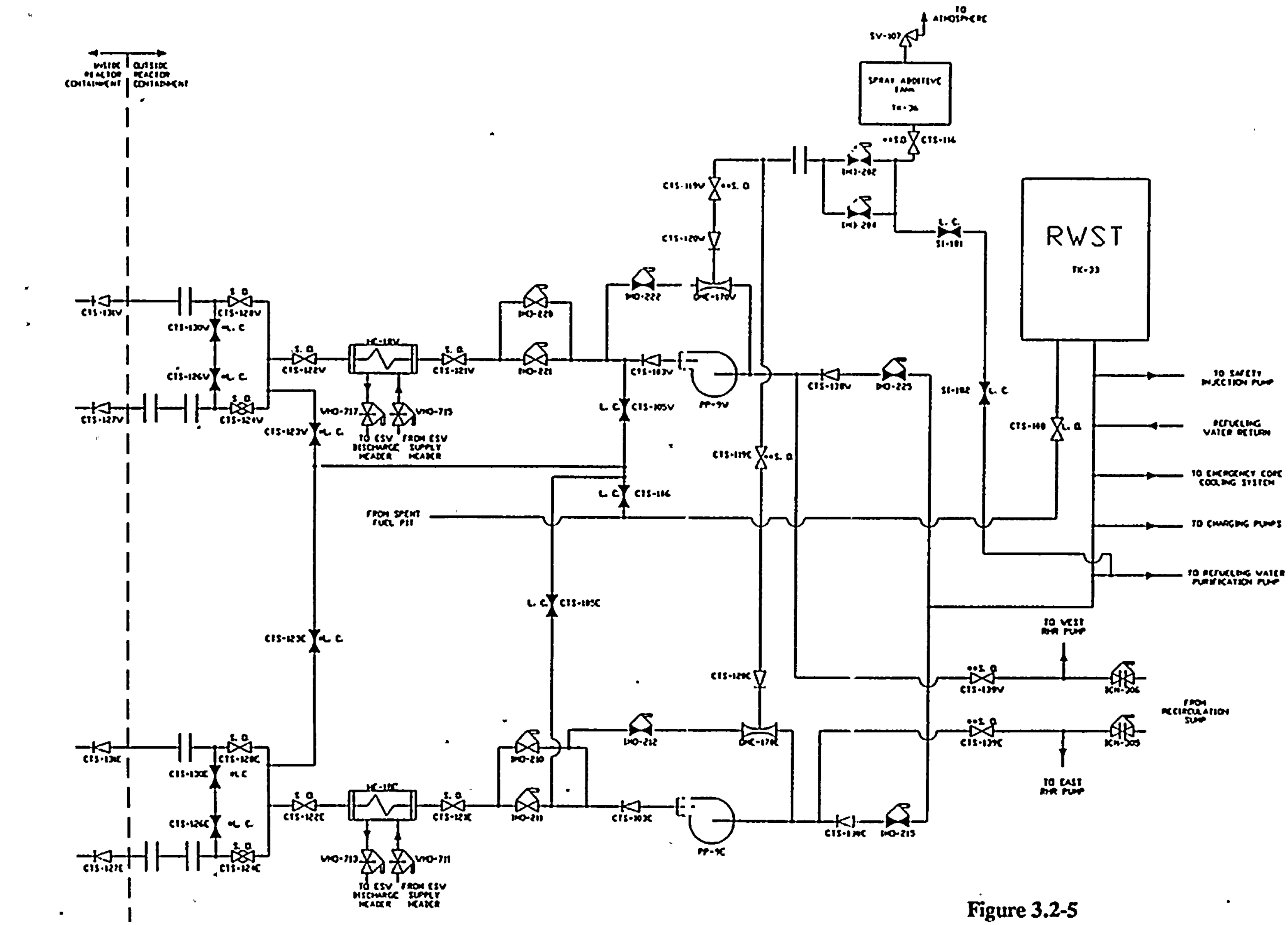
*REFERENCE 13 DESCRIBES THESE VALVES AS LOCKED CLOSED REFERENCE 3A DOES NOT.

**REFERENCE 13 DESCRIBES THESE VALVES AS SEALED OPEN REFERENCE 3A MODELS THEM AS LOCKED OPEN

REFERENCE DRAWINGS:
OP-1-5144-10
OP-2-5144-13
OP-1-5143-20
OP-2-5143-19
OP-1-5113-20
OP-2-5113-25

NOTES
1. VALVE NOTATION DEFINITIONS
a) L. O. = LOCKED OPEN
b) L. C. = LOCKED CLOSED
c) S. O. = SEALED OPEN

Figure 3.2-4
Containment Spray System Simplified Flow Diagram - Injection Phase



*REFERENCE 13 DESCRIBES THESE VALVES AS LOCKED CLOSED REFERENCE 3A DOES NOT.

**REFERENCE 13 DESCRIBES THESE VALVES AS SEALED OPEN REFERENCE 3A MODELS THEM AS LOCKED OPEN

REFERENCE DRAWINGS:
DP-1-5144-10
DP-2-5144-13
DP-1-5143-20
DP-2-5143-19
DP-1-5113-20
DP-2-5113-25

NOTES
1. VALVE NOTATION DEFINITIONS
a) L. O. = LOCKED OPEN
b) L. C. = LOCKED CLOSED
c) S. O. = SEALED OPEN

Figure 3.2-5
Containment Spray System Simplified Flow Diagram - Recirculation Phase

3.2.1.3 Component Cooling Water System

The primary functions of the component cooling water (CCW) system are to:

a) remove residual and sensible heat from the reactor coolant system (RCS) via the residual heat removal (RHR) system during plant shutdown

b) cool the spent fuel pool water and the letdown flow to the chemical and volume control system (CVCS) during power operation

c) provide cooling to dissipate waste heat from various primary plant components

d) provide cooling for safeguards equipment

The CCW system is a support system that provides an intermediate loop between the RCS or other potentially radioactive heat sources and the lake water (essential service water) to ensure that the components receiving cooling water do not leak radioactive fluid outside the plant. The CCW system also provides cooling for each of two safeguard headers following a safety injection signal. Figure 3.2-6 shows a simplified flow diagram of the CCW system for Cook Nuclear Plant Unit 1.

The CCW system is a closed cooling water loop consisting of two component cooling water pumps, two component cooling water heat exchangers, a surge tank, a chemical addition tank and associated piping, valves, instrumentation, and the equipment being cooled. An installed spare maintenance pump is available as a replacement for either unit's CCW system. This pump is placed in service by realigning its associated valves and by switching the pump into the out-of-service pump's power supply and control circuitry.

Component cooling water is normally circulated by one of the two CCW pumps through the shell side of one of the two heat exchangers. Heat is removed from the CCW system by the essential service water which flows through the tubes of the heat exchanger. Component cooling water is supplied to three headers of service:

1) The west safeguards header
2) The east safeguards header
3) The miscellaneous services header

The CCW surge tank is connected to the suction side of the pumps and accommodates the change in volume due to thermal expansion and contraction. By monitoring the surge tank level, leakage into and out of the system can be detected. Make-up to the system is supplied to the surge tank from the demineralized water system.

The CCW system chemistry is controlled by chemical addition to the surge tank through its own charging system and, if required, by system blowdown. Chemical control limits are specified in plant procedures.

The CCW system is a support system to the engineered safeguards system and is required for post-accident removal of decay heat from the reactor. Two completely independent, parallel headers are available for heat removal of the safeguards equipment. The installed spare maintenance pump is capable of completely replacing, both mechanically and electrically, any of the other pumps on either Unit 1 or Unit 2. The two headers are normally cross-tied through the miscellaneous services header, but after an accident occurs, the headers are separated by the operator and function as independent systems.

Power for each CCW pump is provided from a separate electrical bus. The motor-operated valves associated with a given pump's safeguards header, including its supply to the miscellaneous services header, are provided by the same bus. Each bus is fed from both normal and emergency diesel generator power sources. The power supplies are arranged so that one pump and its associated motor-operated valves are served by each emergency diesel generator.

The CCW system is required to be in service at all times. During normal power operation, one header of CCW is utilized. The second pump is on stand-by and will start automatically upon a low discharge pressure from the in-service pump or a safety injection signal. The CCW heat exchanger outlet valve in the stand-by header is normally closed. Both supply valves to the miscellaneous services header are normally open. This alignment provides for a cooling water supply to both safeguards headers and a supply of cooling water to all the miscellaneous services header equipment. There is no need to manually realign the system should the stand-by pump be required to start.

When a safety injection signal is generated, the standby CCW pump receives a start signal, the ESW return valves from the CCW heat exchangers (WMO-733,737) receive a signal to go to a pre-set position, and the CCW heat exchanger discharge valve in the standby header receives an open signal.

Upon switchover of the ECCS to cold leg recirculation following a LOCA, the CCW discharge crosstie valves (CMO-412, -414) are closed to isolate the two CCW headers. This prevents the failure of one header from failing the other header and it also allows the balancing of heat loads. The closure of these valves also isolates any radioactive contamination of the CCW system. Isolation of the two headers does not affect CCW operation.

Upon a total loss of AC power (station blackout), the normally running CCW pump stops operating due to a loss of motive power. When AC power is recovered (either from loading the diesel generators or recovery of offsite power), the CCW pump(s) automatically restart. (If the pump switch is in the pull-to-lock position, the pumps will be manually restarted.)

During cooldown of the reactor coolant system, two CCW pumps and two CCW heat exchangers are used. If a pump fails under these conditions, the cooldown rate may be slowed or even stopped for the time period required to align the spare pump for service.

During cold shutdown, two CCW pumps and two CCW heat exchangers are used. It should be noted that the spent fuel pit heat exchanger can be removed from service periodically or served by the other unit, therefore enabling a single CCW pump and heat exchanger to sufficiently handle the heat load for the shutdown unit.

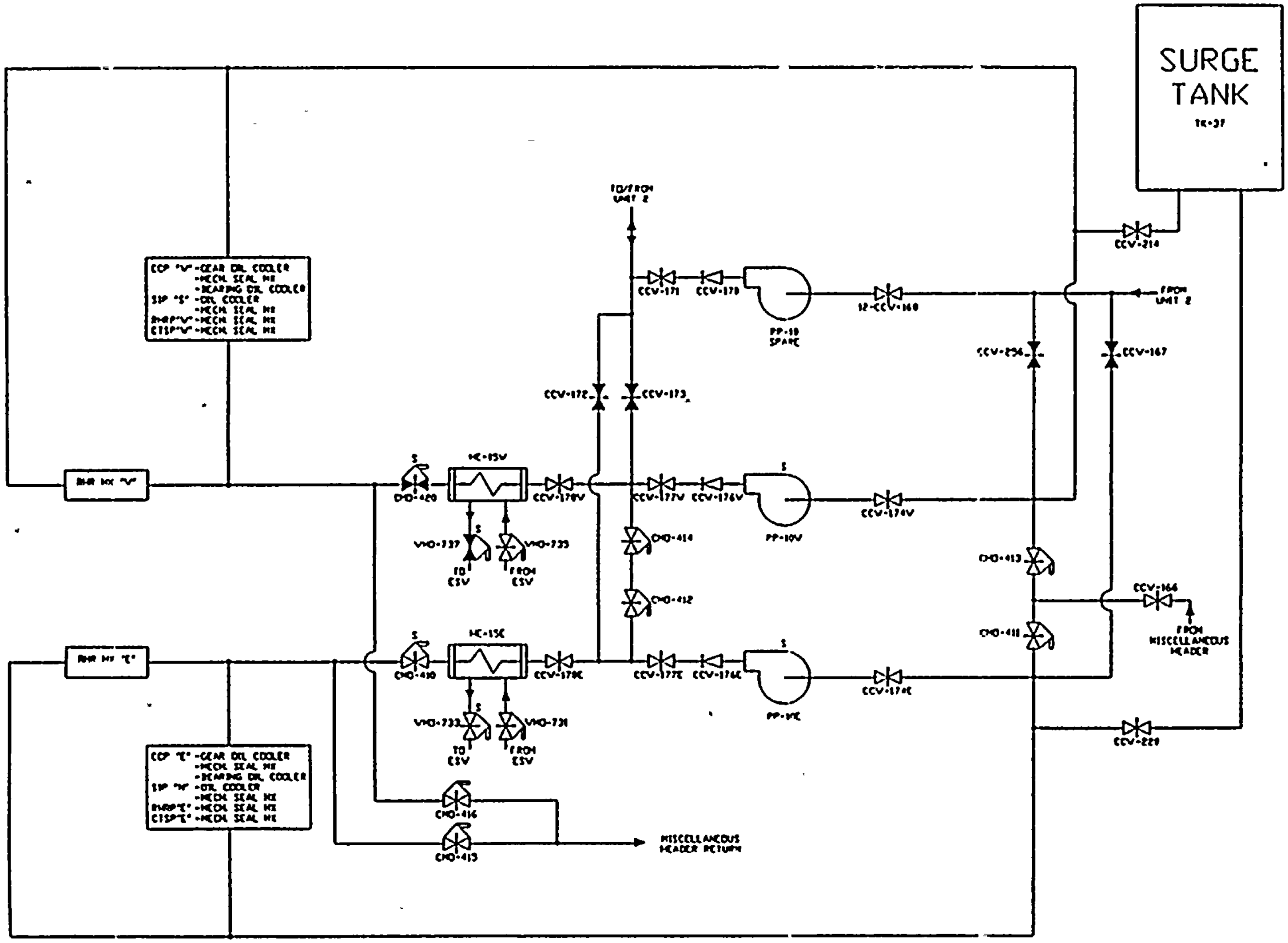
There are four models (fault trees) associated with the CCW system. Each model represents the unavailability of a single CCW header for two different plant conditions. The models are designated as follows:

CCWE - This system model defines the unavailability of the CCW east header to provide sufficient flow to the east RHR heat exchanger and other essential loads to mitigate the effects of a LOCA (both during injection and recirculation). This is for normal AC power conditions.

CCWW - This system model defines the unavailability of the CCW west header to provide sufficient flow to the west RHR heat exchanger and other essential loads to mitigate the effects of a LOCA (both during injection and recirculation). This is for normal AC power conditions.

CCWEL - This system model defines the unavailability of the CCW east header to provide sufficient flow to the east RHR heat exchanger and other essential loads after AC power is recovered following a station blackout.

CCWWL - This system model defines the unavailability of the CCW west header to provide sufficient flow to the west RHR heat exchanger and other essential loads after AC power is recovered following a station blackout.



NOTES
1. VALVE NOTATION DEFINITIONS
a) F. C. = FAILS CLOSED
2. S = COMPONENT ACTUATED BY A SAFETY INJECTION SIGNAL
3. C = COMPONENT ACTUATED BY A CONTAINMENT ISOLATION SIGNAL

REFERENCE DRAWINGS: CP-1-5133-18 CP-2-5133-17 CP-1-5133A-23 CP-2-5133A-18 CP-1-5133B-12 CP-2-5133B-13 CP-1-5113-28 CP-2-5113-25

Figure 3.2-6
Component Cooling Water System Simplified Flow Diagram

3.2.1.4 Essential Service Water System

The essential service water (ESW) system provides the cooling water requirements for the component cooling water (CCW) heat exchangers, the emergency diesel generator coolers, the containment spray (CTS) heat exchangers, and the control room air conditioning condensers and air handling units. It also provides an emergency supply of water to the auxiliary feed pumps in the event that the condensate storage tanks are emptied or otherwise lost as a source of water.

The ESW system, shared by both units, consists of four ESW pumps, each with an automatic backwashing duplex strainer and associated piping, valves, and instrumentation. The system is comprised of two identical main headers. Each header is served by two pumps and each header, in turn, serves half of the system load in each unit. The two headers are arranged such that a rupture in either header will not jeopardize the safety functions of the system. Two pumps are sufficient to supply all service water requirements for unit operation, shutdown, refueling, or post-accident operation, including a loss-of-coolant accident (LOCA) on one unit and a simultaneous hot shutdown in the other. However, a third pump is normally started under the shutdown and refueling operations. All pumps receive a start signal (SI) in the event of an accident. Figure 3.2-7 is a simplified flow diagram for ESW for Unit 1 showing cross-ties to Unit 2.

During normal operation, water is supplied from the lake, through the circulating water intake pipes to the ESW pumps suction well located in the screenhouse. An alternate water supply is available by opening the slide gates (WMO-17 and -27) between either discharge tunnel vault and the forebay, ensuring a water supply in the event that the intakes are unavailable.

During start-up, normal system operation or shutdown, ESW is supplied to the CCW heat exchangers and the control room air conditioning condensers. The remaining heat exchangers are supplied during a loss of offsite power (LOOP) and/or a safety injection (SI).

Since the ESW system is relied upon by such engineered safeguards systems as the containment spray and emergency core cooling systems, it is designed as a Seismic Class I system with the redundancy and other criteria associated with safeguards systems.

There are four models (fault trees) associated with the ESW system. Each of these models represent the unavailability of this system in response to different accident events.

ESWW - This fault tree models the Unit 1 (U-1) west header. If utilizing the west header (standby header), the U-2 east pump would have to remain on-line to supply the U-2 ESW needs and prevent any flow diversion from U-1 to U-2 while the U-1 west pump supplies U-1. The cross connect valves are not modeled since failure would add success to any flow diversion.

ESWE - This fault tree models the Unit 1 east header. With the U-1 east pump train supplying flow to the east U-1 header, either the U-2 west pump train would have to pick up the U-1 ESW loads after it started on the safety injection signal (if the U-1 east pump train fails) or no other flow diversion can be allowed (this flow diversion is modeled in the form of valves inadvertently opening). With the U-2 west header in a standby state, no ESW loads from U-2 would be placed upon it. Thus, the U-2 west pump train can provide full flow to the U-1 east header (barring any of the mentioned flow diversions). The cross connect valves (WMO-706 and -707) are modeled since their failure prevents U-2 flow from going to U-1.

ESWWL - This fault tree models the U-1 west header in response to a loss of power event. The loss of power event is assumed to affect both Units. This case is modeled the same as in the ESWW model except now all pumps must be restarted (and checks valves opened) instead of the standby pump starting as before. For the formerly operating pump, the discharge

motor operated valve does not close on a loss of power and thus does not have to reopen when the pump restarts.

ESWEL - This fault tree models the U-1 east header in response to a loss of power event. As in the ESWWL model, the loss of power affects both units. Also, the treatment of pumps and valves is the same as in the ESWWL model.

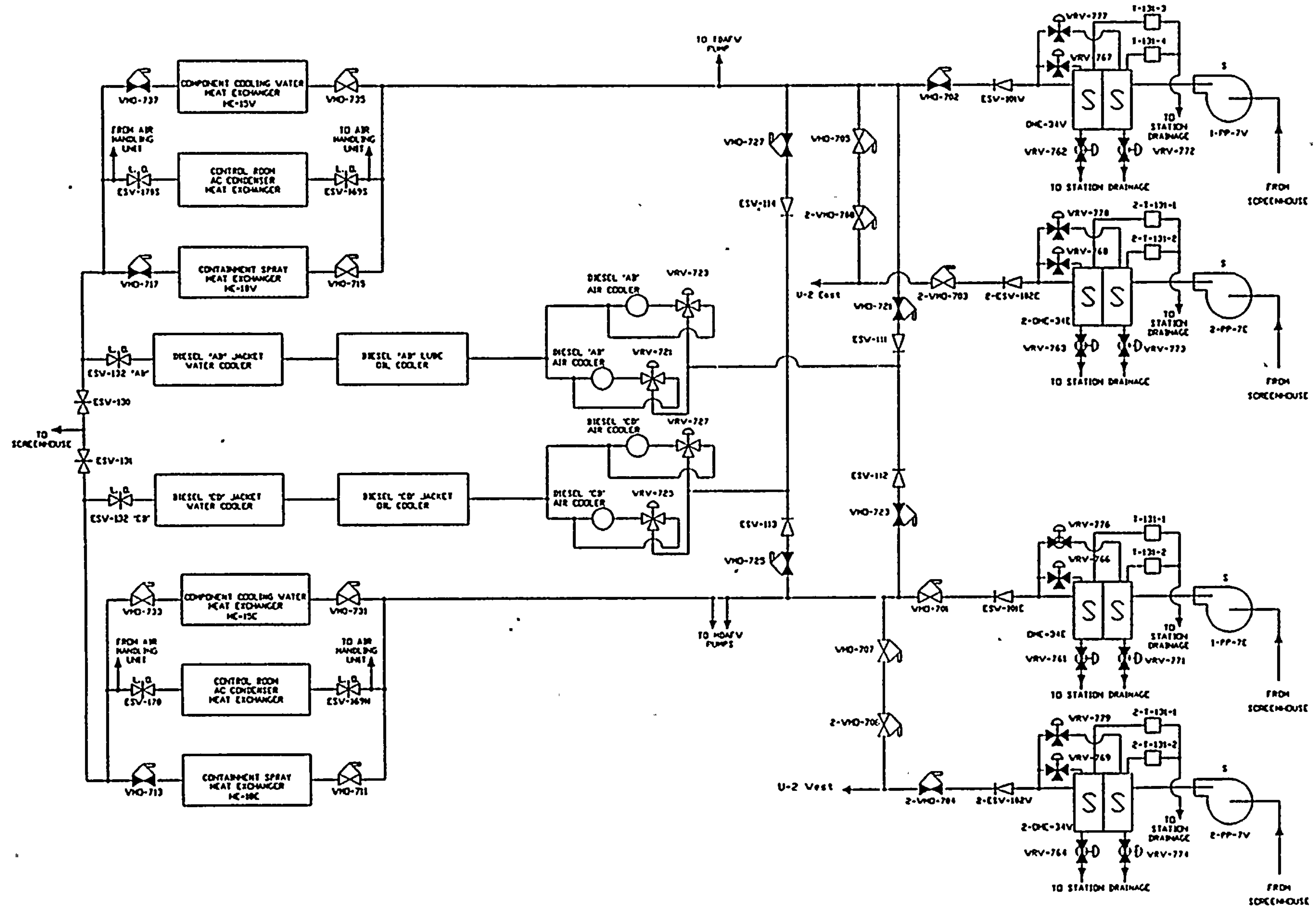


Figure 3.2-7
Essential Service Water System Simplified Flow Diagram

NOTES
1. VALVE NOTATION DEFINITIONS
   a) L.O. = LOCKED OPEN
2. STRAINER BACKWASH VALVES (VRV) FAIL CLOSED UPON LOSS OF AIR OR POWER.
3. ☐ = VENT VALVE

REFERENCE DRAWINGS:
OP-1-5113-20 OP-2-5113-25

3.2.1.5 Nonessential Service Water System

The Nonessential Service Water (NESW) System is a shared system which provides lake water for cooling and makeup water to numerous plant systems and components, including upper and lower containment ventilation units and the plant and control air compressors.

The system consists of four NESW supply pumps, each with an automatic backwashing duplex strainer. During normal two unit full power operation, the NESW system is shared between Unit 1 and Unit 2 with three of four NESW pumps needed to supply required flow. The fourth pump and strainer combination is provided for standby service as an installed spare.

Two NESW pumps are located in each unit. Each pump normally takes suction from its respective unit's circulating water intake tunnel. If the source of water to the circulating water intake is interrupted, all four NESW pumps are capable of taking suction from their respective circulating water discharge tunnel and, by means of a cross-tie line, each can also take suction from the other unit's circulating water intake or discharge tunnel. Thus, nonessential service water supply to both units is assured, even if the tunnels of one unit are out of service. The return water flows to the circulating water discharge tunnel which discharges into Lake Michigan.

Figure 3.2-8 is a simplified flow diagram of the NESW system during normal operating conditions. Figure 3.2-9 is a simplified flow diagram of the NESW system during station blackout conditions.

There are two models (fault trees) associated with the NESW system. Each model represents the unavailability of this system in response to different accident events.

NESW1 - This system model defines the response of the NESW system following the receipt of a safety injection signal.

NESW2 - This system model defines the response of the NESW system following a station blackout (SBO). For the station blackout case, only Unit 1 is assumed to be undergoing the station blackout (thus the diesel generators in Unit 2 come on-line.)

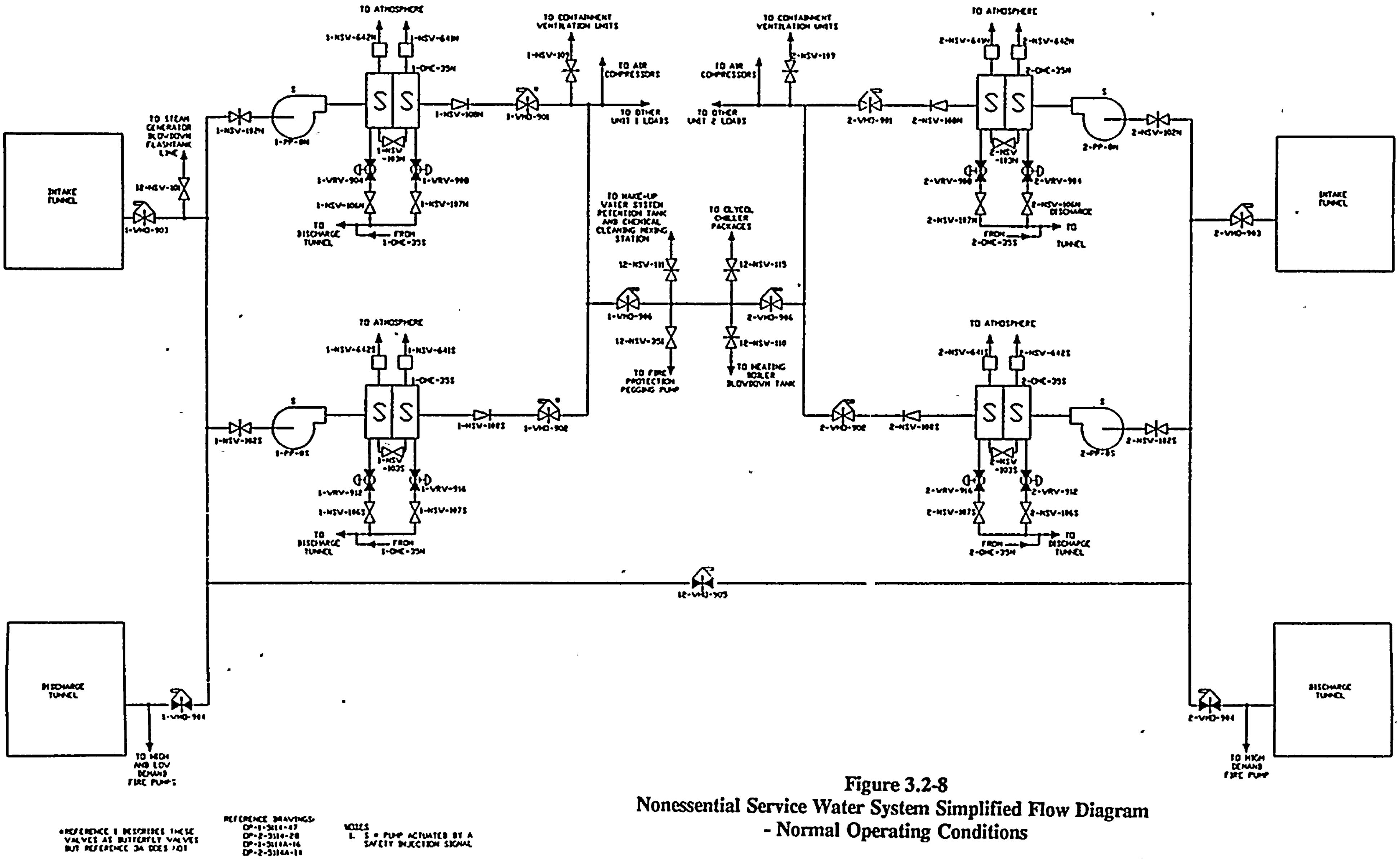


Figure 3.2-8
Nonessential Service Water System Simplified Flow Diagram - Normal Operating Conditions

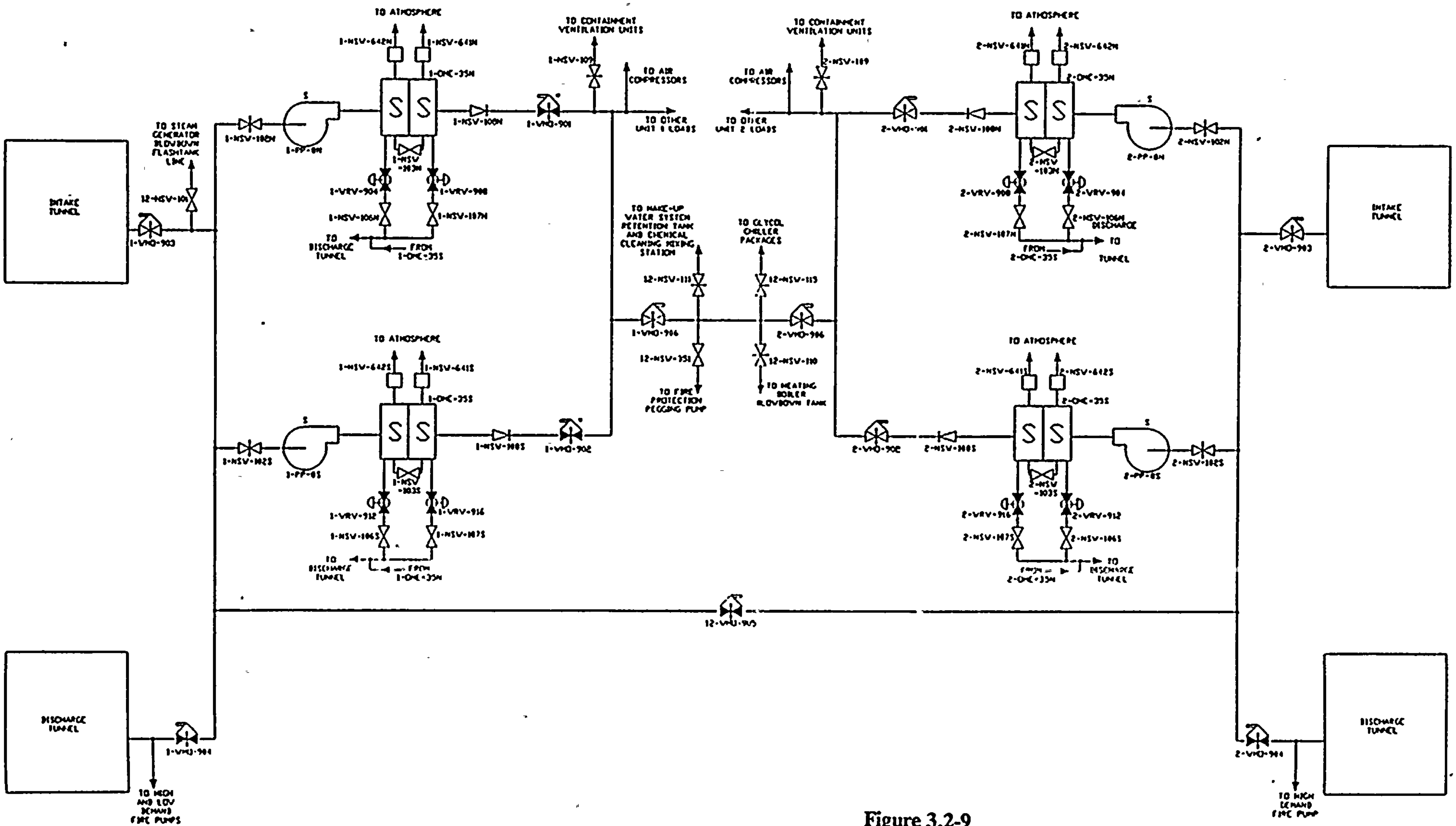


Figure 3.2-9
Nonessential Service Water System Simplified Flow Diagram - Station Blackout

3.2.1.6 Compressed Air (Control Air) System

The control air system, which is a part of the compressed air system (CAS) provides filtered and dried air for instrument and control usage in the plant (both inside and outside of containment).

Compressed air is supplied to Units 1 and 2 through an air distribution system located in the turbine, auxiliary and containment buildings. This distribution system consists of a shared plant air ring header extending throughout the turbine building, a pair of parallel plant air headers in the auxiliary building, and a plant control air header in each containment.

The CAS has two Plant Air Compressors (PACs), each of which is capable of supplying the total compressed air demand for both units. One PAC, plant air receiver, and plant air after cooler is located in each unit. Each plant air receiver supplies its own plant air header in the auxiliary building, however, the shared turbine building plant air header may be supplied from either plant air receiver. Additionally, each unit has a control air compressor (CAC), which is capable of supplying that unit's control air (instrument air) needs.

During normal CAS operation, only one PAC is operating and supplies it's compressed air to the turbine building ring header and to it's own plant air header in the auxiliary building. The turbine building ring header, in turn, supplies compressed air to the standby plant air header in the auxiliary building. Since these distribution headers are parallel and have hose connections in essentially the same location, one unit's header may be isolated from the CAS without affecting the availability of plant air to the auxiliary building. Compressed air is then delivered to the header branches that feed the services requiring compressed air (e.g., screen house, service, turbine and auxiliary buildings).

Compressed air from the turbine room plant air header is filtered and dried for instrument and control usage (control air system). Each unit has its own wet control air receiver. If there is a unit instrument and control air demand and both plant air compressors are unavailable for service, low air pressure at each unit's wet control air receiver will automatically start both CACs. The CACs will run on constant speed regulation until stopped by the operator.

Compressed air to be used for control, instrumentation, containment integrated leak rate testing, or containment penetration and weld channel pressurization is dried to a dew point below the minimum temperature expected at its point of use. Each dryer has a prefilter and afterfilter. The prefilter prevents contamination of the dryer desiccant from moisture carry-over or scale. The afterfilter protects the CAS downstream of the dryers from desiccant dusting. The use of copper and stainless steel for control air piping minimizes pipe scale carry-over to instruments and controls. Each unit has a redundant string of control air prefilters, dryers, afterfilters and dry control air receivers located downstream of the wet control air receiver. Figure 3.2-10 is the simplified flow diagram of the CAS.

There are ten models (fault trees) associated with the compressed air system. Each model represents the unavailability of the compressed air system or portions of it.

CONAIR1 - This fault tree models the Unit 1 PAC operating and the Unit 2 PAC in standby (ready for automatic start). Both PACs feed into the turbine building plant air ring header, which can be isolated into sections by valves PRV-10,-11 (Unit 1) and PRV-20,-21 (Unit 2). The compressed air then feeds the control air dryer strings for filtering and drying. This model ends at the point where the output of the dryer trains meet.

CONAIRLS - This fault tree closely resembles that of CONAIR1 except that the PACs have to start and run following a loss of power. Also, valves PRV-10,-11,-20 and -21 have to be reopened since these valves close when deenergized.

CONAIR2 - This fault tree models control air being supplied to the header inside of containment that supplies compressed air to the Pressurizer PORVs. The

air flow is through containment isolation valves XCR-100,-101 and past regulator XRV-186. Fault tree CONAIR1 is called in as a sub-tree.

CONAIR2L -	This fault tree closely resembles the CONAIR2 tree except that valves XCR-100,-101 have to be reopened after a loss of power (shut when deenergized). The CONAIRLS tree is called in as a sub-tree. No IPE modeled components exist down stream of valves XCR-102-103. Thus, these valves are not modeled.
CONAIR3, CONAIR4 and CONAIR5 -	These fault trees model the pressure reduction of 100 lb control air (tied in as sub-tree (CONAIR1) to 85 lb air (CONAIR3), 50 lb air (CONAIR4) and 20 lb air (CONAIR5). The pressure reducers and filters associated with each path are modeled.
CONAIR3L, CONAIR4L and CONAIR5L -	These fault trees are the same as the original trees except that CONAIRLS is now called in as a sub-tree to account for control air response to a loss of power.

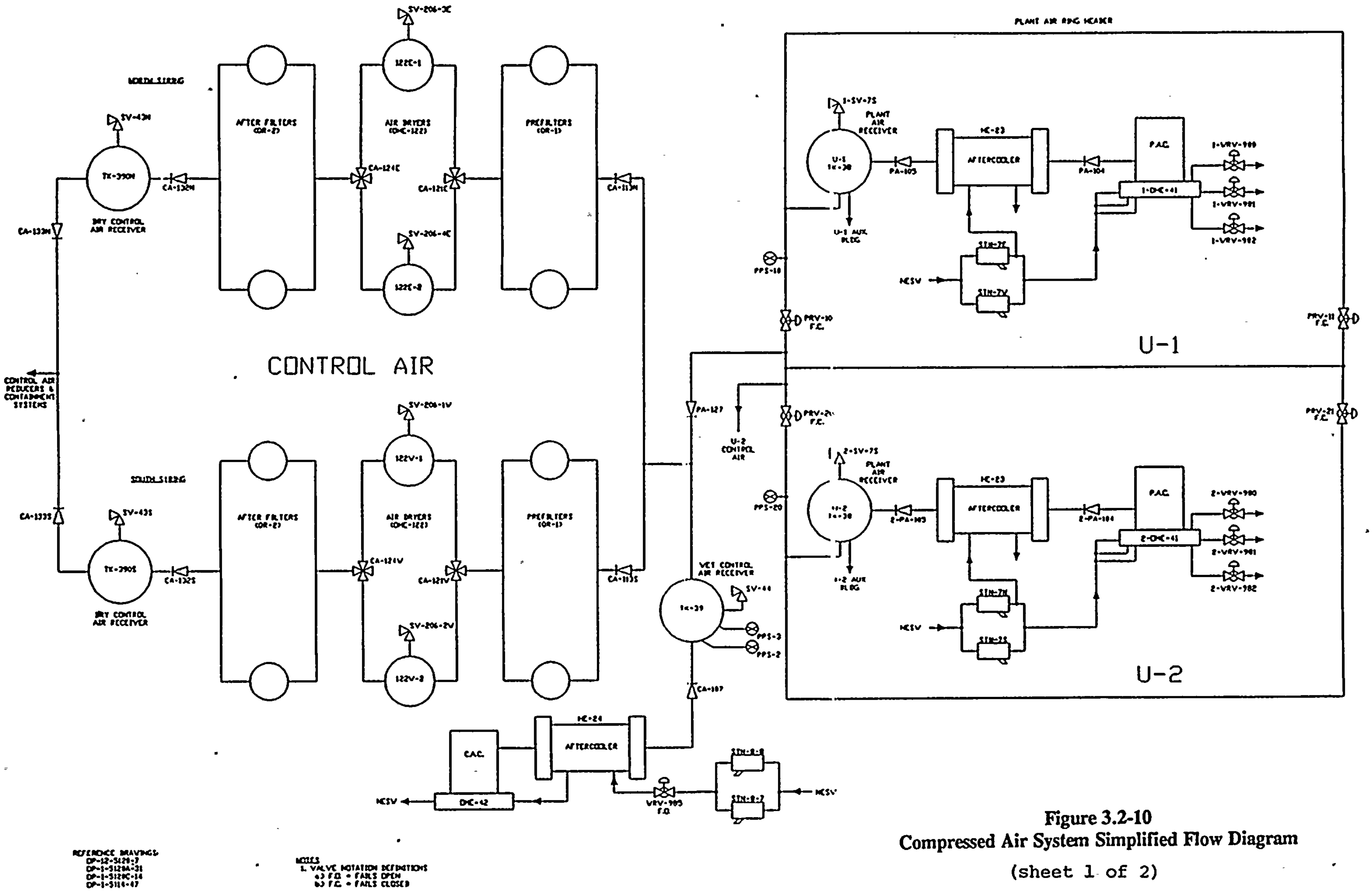
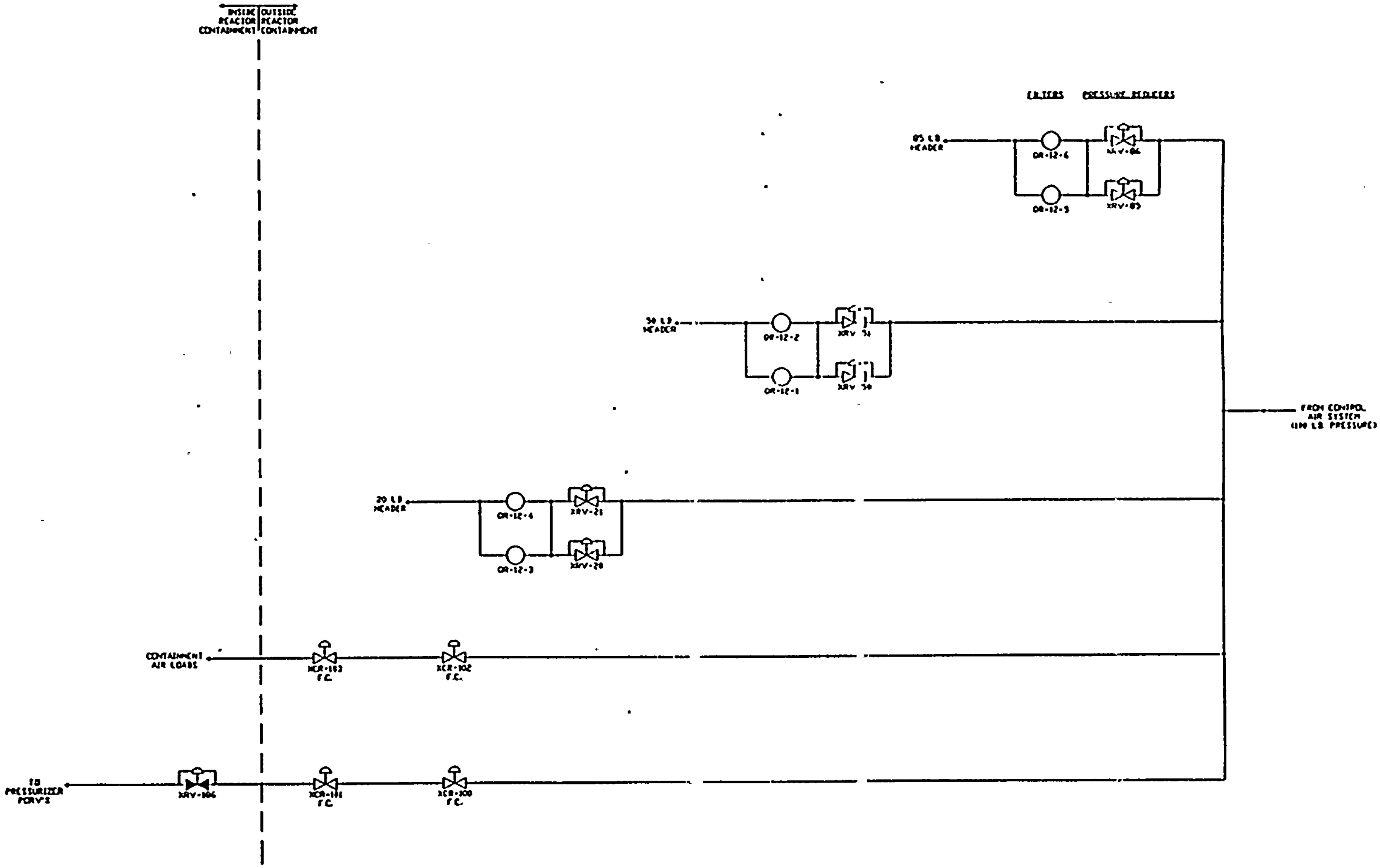


Figure 3.2-10
Compressed Air System Simplified Flow Diagram
(sheet 1 of 2)



REFERENCE DRAWINGS
OP-1-5120C-14
OP-1-5120B-3

NOTES
1. VALVE NOTATION DEFINITIONS
a.) F.C. = FAILS CLOSED

2. XCR-100, -101, -102, -103 OPERATED FROM 50 LB. HEADER

Figure 3.2-10
Compressed Air System Simplified Flow Diagram

(sheet 2 of 2)

3.2.1.7 Emergency Core Cooling System

At the Cook Nuclear Plant, emergency core cooling water is supplied by the accumulators, safety injection pumps, centrifugal charging pumps, and the residual heat removal pumps. For convenience, the system descriptions and operations have been broken down into the functions of: accumulators, low head system operation, and high head system operation.

3.2.1.7.1 Accumulators

The accumulator tanks are designed to operate in the case of a large size break in the reactor coolant system and to rapidly inject the borated water stored in the tanks if the RCS pressure drops below the pressure in the accumulator.

There are four accumulators per unit. The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. One accumulator is attached to each of the four RCS cold legs. Figure 3.2-11 shows the simplified flow diagram for the accumulator system.

The accumulator system is evaluated in the context of a large and medium LOCA where the pressure in the RCS decreases sufficiently to allow rapid injection of coolant. Since the accumulator system is passive and no manual actions are required, injection occurs when the RCS pressure drops to about 620 psig. When this occurs, the borated water from each accumulator flows through its respective normally open, motor-operated isolation valve and continues through its two check valves into the attached RCS cold leg.

There is one model (fault tree) associated with the accumulator system. This model represents the unavailability of this system in response to different accident events.

ACC - This system model defines the unavailability of the accumulator system to inject the contents of three accumulator tanks into the three intact RCS legs given that a large or medium LOCA has occurred on the fourth RCS loop.

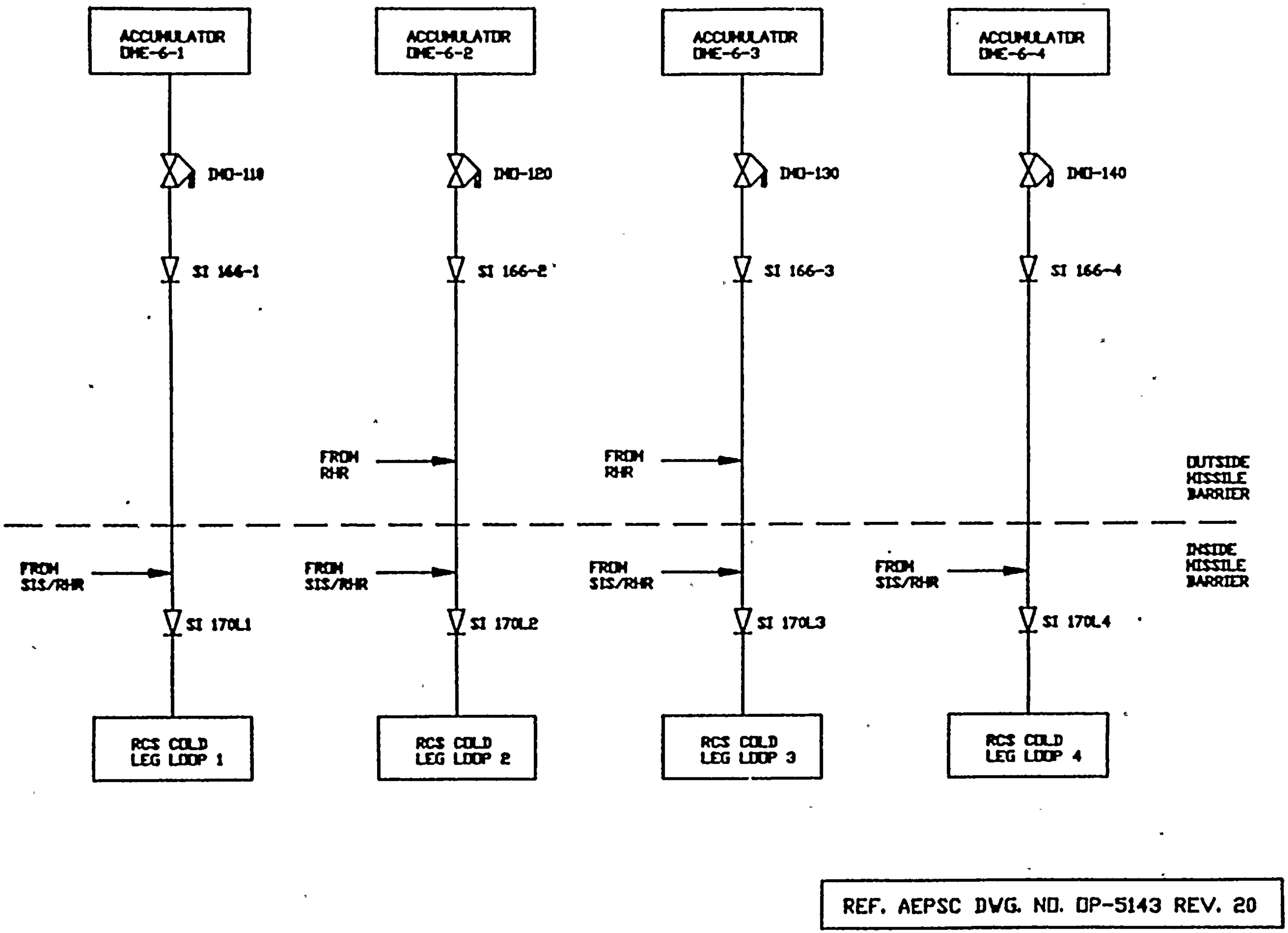


Figure 3.2-11
Accumulator System Simplified Flow Diagram

3.2.1.7.2 ECCS Low Head Cooling System

One of the functions of the Residual Heat Removal (RHR) System is to provide the long term post-accident cooldown requirements of the Emergency Core Cooling System (ECCS). In this function, the RHR subsystem of the ECCS provides a low pressure, high volumetric flowrate water source to the RCS by supplying water from the Refueling Water Storage Tank (RWST) during the injection phase and from the containment sump during the recirculation phase. The RHR system also has a major function during normal plant operations in that the system removes decay heat from the RCS during the second phase of normal plant cooldown when it is no longer practical to cool the RCS to the cold shutdown condition by use of the steam generators.

The description presented herein is limited to that in which the RHR system supports the post accident cooldown requirements of the ECCS in the event of a LOCA. The RHR system is evaluated in the context of a large break LOCA; however, the system is also required for recirculation during small and medium break LOCAs for both cooling and flow to the safety injection (SI) and charging pumps (CC).

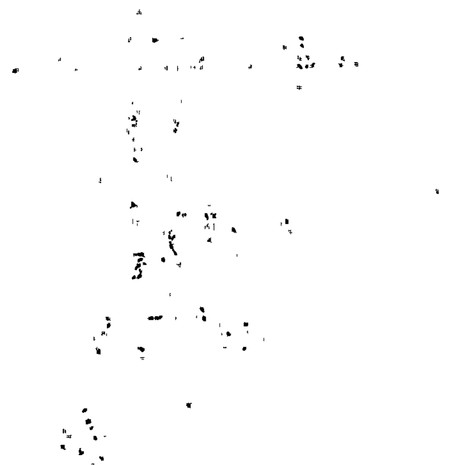
The RHR system consists of two identical trains per unit. Each train consists of an RHR heat exchanger, an RHR pump, valves and associated piping. A simplified flow diagram of the RHR system for the injection phase of ECCS operation is shown in Figure 3.2-12 and a simplified flow diagram of the RHR system for the recirculation phase of ECCS operation is shown in Figure 3.2-13.

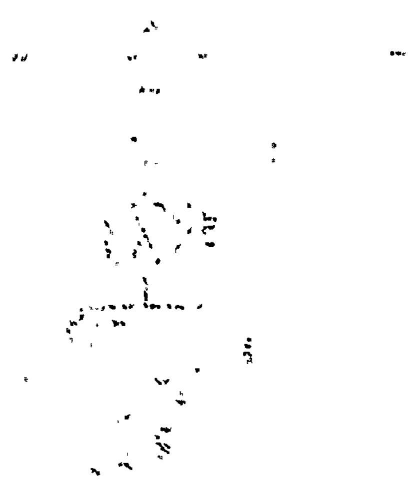
In performing the post accident long term cooling function of the ECCS, the RHR system has two phases of operation: injection and recirculation. The shutoff head of the RHR pumps is less than 205 psig.

Following a LOCA, the RHR system is supplied with water from the RWST during the injection phase. During the recirculation phase, which begins after the RWST is drained, the RHR pumps take water from the containment sump and inject water directly into the RCS cold legs if RCS pressure is below the RHR pump shutoff head, and they also provide adequate suction head for the high pressure SI pumps and the centrifugal charging pumps. In addition, the RHR system provides cooled water to the upper containment RHR spray headers if containment pressure rises above 8 psig.

There are two fault trees associated with the RHR system. Each model represents the unavailability of this system as it responds to accident events.

LPI	-	This system model defines the logic associated with the response of the RHR system to provide coolant from the RWST to the RCS cold legs. This model is also referred to as the "injection model."
LPR	-	This system model defines the logic associated with the response of the RHR system to take coolant from the containment sump and inject it into the RCS cold legs. This model is referred to as the "recirculation model."





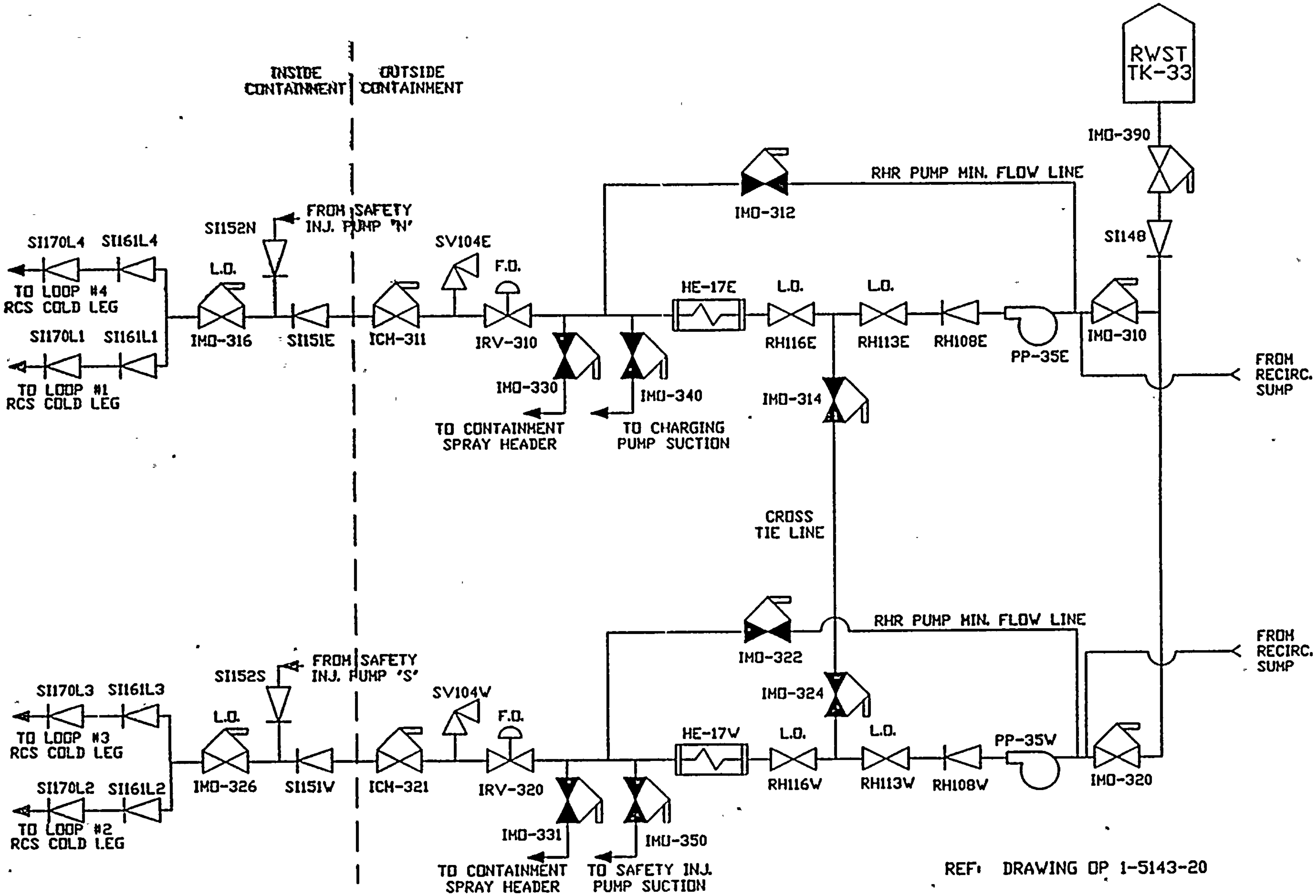
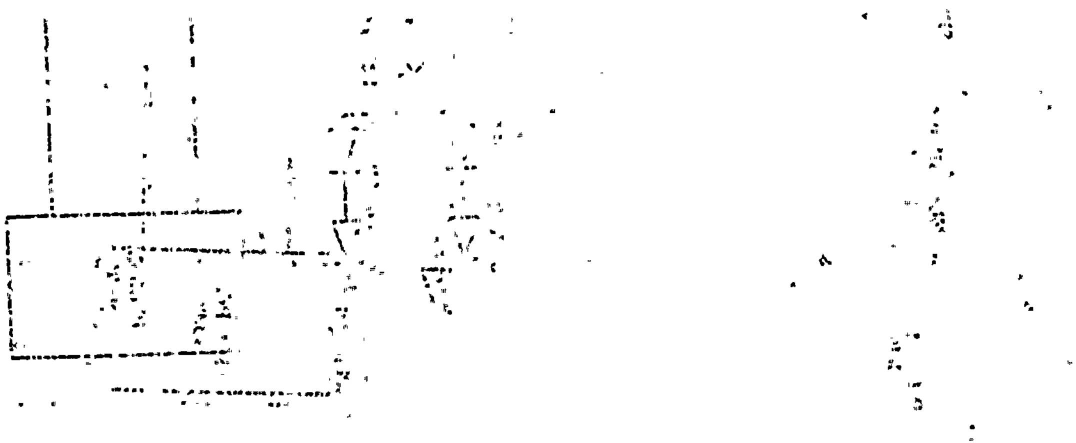


Figure 3.2-12
Low Head Cooling System Simplified Flow Diagram
- Injection Phase



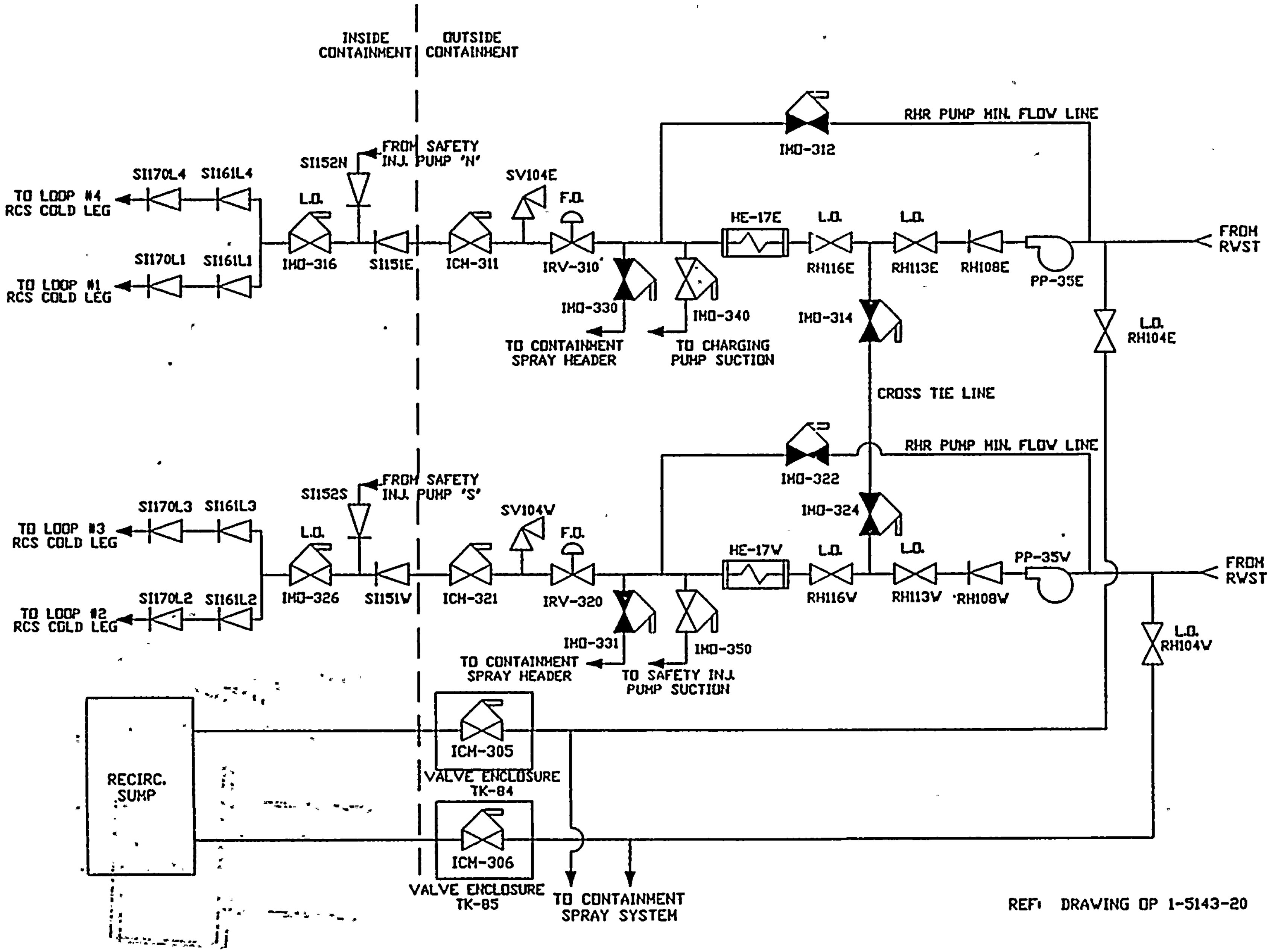


Figure 3.2-13
Low Head Cooling System Simplified Flow Diagram
- Recirculation Phase

3.2.1.7.3 ECCS High Head Cooling System

The high head ECCS provides emergency core cooling and helps to maintain reactor coolant inventory in the event of a LOCA or a main steam line break.

The major components of the high head ECCS are: two centrifugal charging (CC) pumps, a boron injection tank (BIT), two safety injection (SI) pumps, and their associated valves. The primary source of water is the refueling water storage tank (RWST) (injection phase); the secondary source of water is from the containment sump via the RHR pumps (recirculation phase).

The high head ECCS consists of three separate subsystems: one containing the charging pumps and the BIT, one containing the SI pumps, and the last containing the RHR pumps and heat exchangers.

During normal plant operation, a continuous requirement for fluid to the reactor coolant pump seals requires that one charging pump, either centrifugal or positive displacement, be run whenever reactor coolant pumps are operated. During normal operation, the safety injection pumps and the residual heat removal pumps are not operating. A flow diagram of the ECCS alignment during normal plant operation is illustrated in Figure 3.2-14.

The high head ECCS has two phases of operation following an accident: injection and recirculation.

Following a LOCA or steamline break event, the high head system is supplied with coolant from the RWST during the injection phase. This phase ends when the contents of the RWST are nearly depleted. In the recirculation phase, the water that is spilled from the break collects in the containment sump and flows through mesh screens into the recirculation sump. The CC and SI pumps then take suction from the RHR pump's discharge which in turn takes suction from the recirculation sump. During the recirculation phase, both the RHR and SI pumps' discharge line crosstie valves are shut. This provides two separate trains in the recirculation phase should a pipe or pump failure occur, which is in keeping with the plant's licensing basis that pipe breaks must be considered as a possible passive failure during the recirculation phase. The configuration of the ECCS during the injection phase is illustrated in Figure 3.2-15 and the recirculation phase is illustrated in Figure 3.2-16.

There are five models (fault trees) associated with the high head ECCS system.

HPI	-	This system model defines the unavailability of the ECCS system to provide sufficient flow to three of four RCS cold legs. (SI crosstie valves are open.)
HP2	-	This system model defines the unavailability of the ECCS system to provide sufficient flow to one of three intact cold legs. (SI crosstie valves are closed.)
HP3	-	This system model defines the unavailability of the ECCS system to provide sufficient flow to one of four RCS cold legs.
HP5	-	This system model defines the unavailability of the ECCS system to provide sufficient flow to four of four RCS cold legs.
HPR	-	This system model defines the unavailability of the ECCS system to provide sufficient flow to one of three intact RCS cold legs during recirculation.

EVENT	EVENT NAME
SWS	LOSS OF ESW INITIATING EVENT
RCP	REACTOR COOLANT PUMPS TRIPPED
AF1	AUXILIARY FEEDWATER ACTUATION
MF1	MAIN FEEDWATER
EH1	ESW SYSTEM RESTORED WITHIN 1 HOUR
RCD	RCS COOLDOWN
EH8	ESW SYSTEM RESTORED WITHIN 8 HOURS
CNU	CORE NOT UNCOVERED
RRI	RESTORE RCS INVENTORY
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

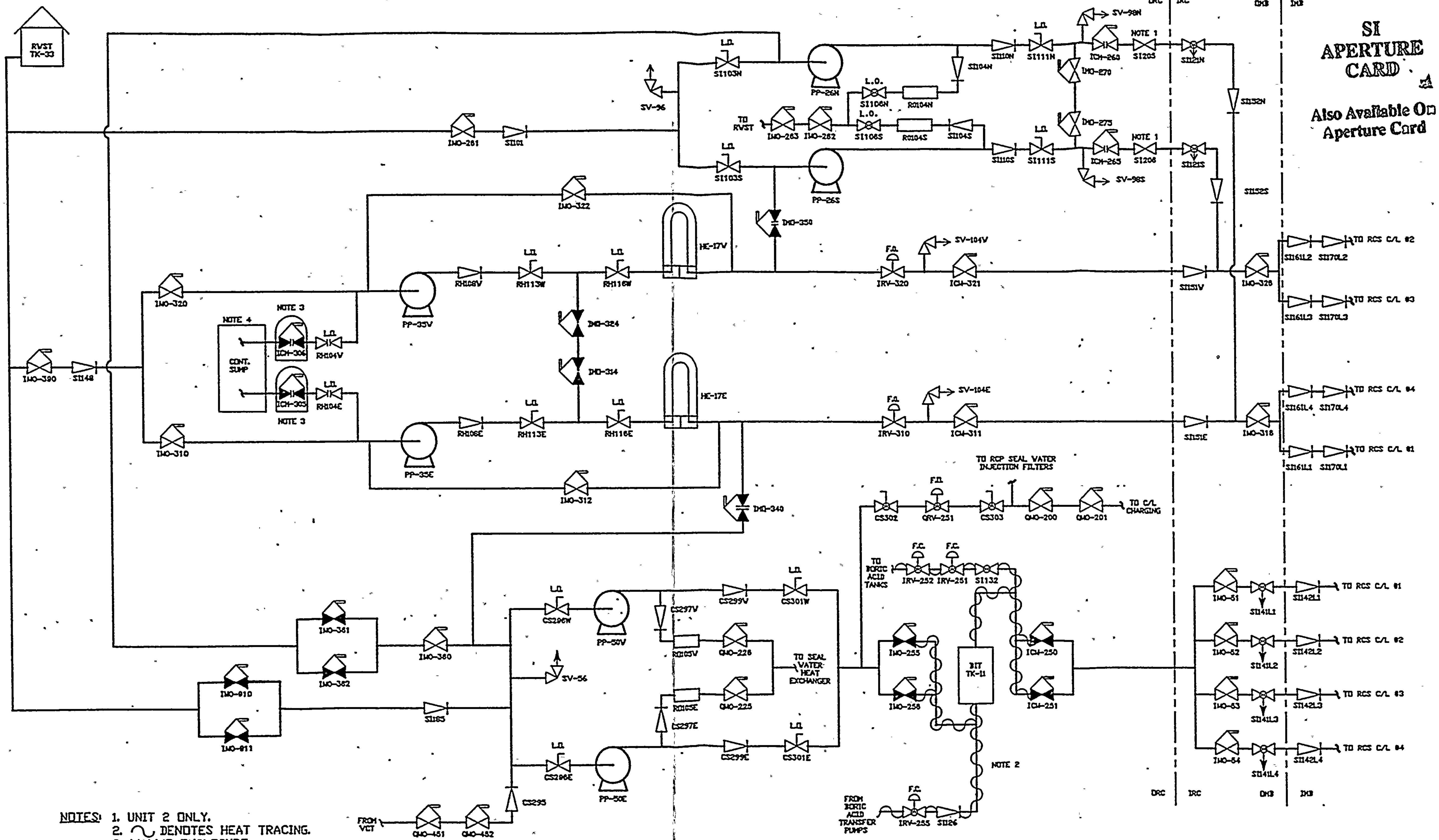
CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THW	TRANSIENT, HIGH PRESSURE, CT DRY
THWI	TRANSIENT, HIGH PRESS. CT DRY, HI FAILS
THWIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-13
Loss of Essential Service Water Event Tree

EVENT	EVENT NAME
CCW	LOSS OF CCW INITIATING EVENT
RCP	REACTOR COOLANT PUMPS TRIPPED
AF1	AUXILIARY FEEDWATER ACTUATION
MF1	MAIN FEEDWATER
CH1	CCW SYSTEM RESTORED WITHIN 1 HOUR
RCD	RCS COOLDOWN
CH8	CCW SYSTEM RESTORED WITHIN 8 HOURS
CNU	CORE NOT UNCOVERED
RRI	RESTORE RCS INVENTORY
CSI	CONTAINMENT SPRAY INJECTION
HPR	HIGH PRESSURE COLD LEG RECIRCULATION
CSR	CONTAINMENT SPRAY RECIRCULATION
HI	HYDROGEN IGNITERS
CF	CONTAINMENT RECIRCULATION FANS

CATEGORY	DESCRIPTION
SUCCESS	NO CORE DAMAGE, NO CONTAINMENT FAILURE
THR	TRANSIENT, HIGH PRESS, CT HEAT REMOVAL
THRI	TRANS, CT HEAT REMOVAL SUCCESS, HI FAILS
THRIF	TRANS, HI & CF FAIL, CT HEAT REMOVED
THC	TRANSIENT, HIGH PRESSURE, CSR FAILS
THCI	TRANSIENT, HIGH PRESSURE, CSR & HI FAIL
THCIF	TRANS, HIGH PRESSURE, CSR, HI, CF FAIL
TH	TRANSIENT, HIGH PRESSURE
THI	TRANSIENT, HIGH PRESSURE, HI FAILS
THIF	TRANSIENT, HIGH PRESSURE, HI & CF FAIL
THW	TRANSIENT, HIGH PRESSURE, CT DRY
THWI	TRANSIENT, HIGH PRESS. CT DRY, HI FAILS
THWIF	TRANS, HIGH PRES, CT DRY, HI & CF FAIL

Figure 3.1-14
Loss of Component Cooling Water Event Tree



NOTES: 1. UNIT 2 ONLY.
2. ∿ DENOTES HEAT TRACING.
3. VALVE ENCLOSURE:
TK-84 ⟶ ICM-305
TK-85 ⟶ ICM-306.
4. INSIDE REACTOR CONTAINMENT.
5. L.O. = LOCKED OPEN.

VEC-0447
5-29-91
JAH

Figure 3.2-14
High Head Cooling System Simplified Flow Diagram
- Normal Operating Conditions

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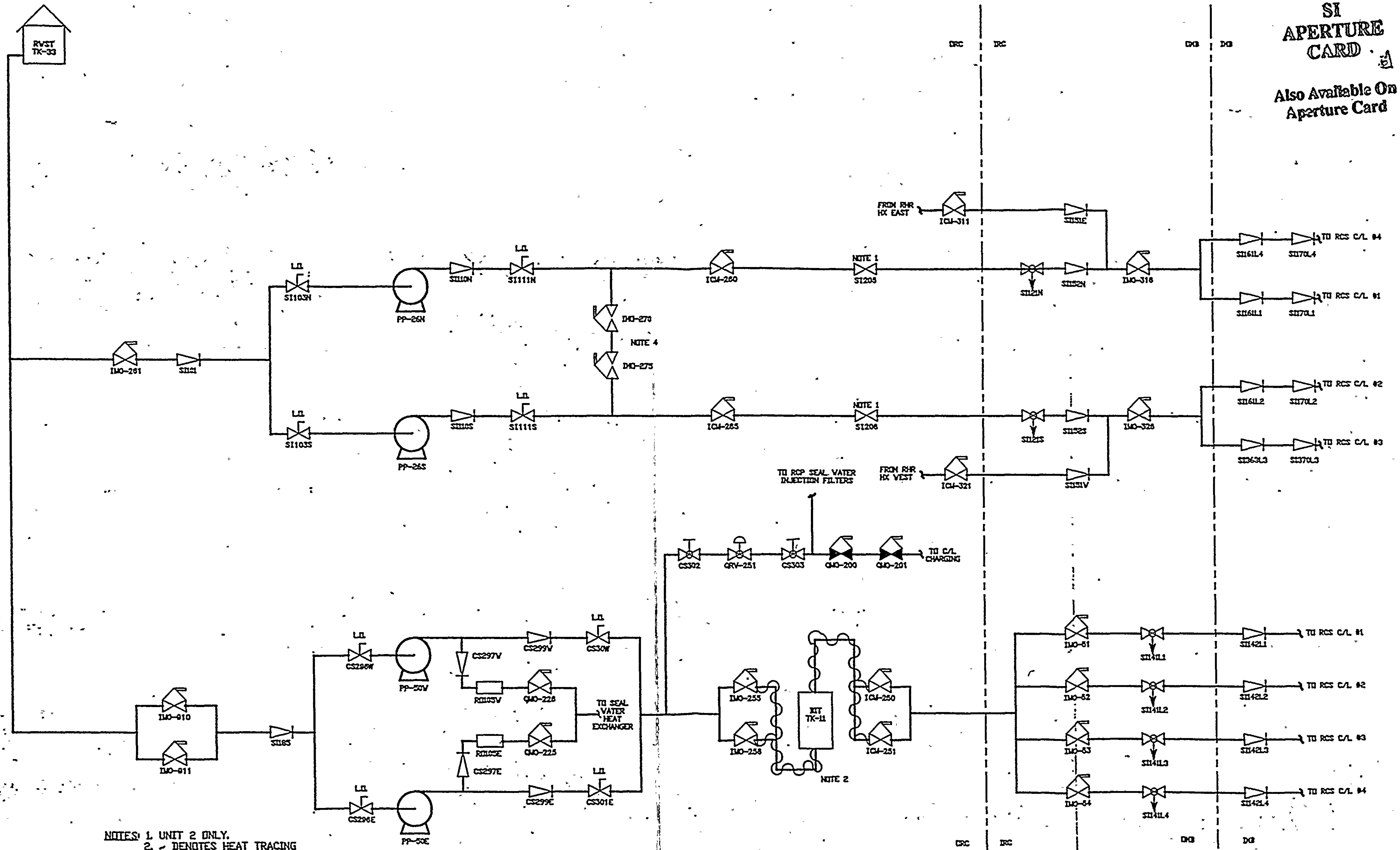

NOTES: 1. UNIT 2 ONLY.
2. ~ DENOTES HEAT TRACING
3. L.O. = LOCKED OPEN
4. CLOSED FOR HP2 ONLY

WEC-0448
5-29-91
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Figure 3.2-15
High Head Cooling System Simplified Flow Diagram
- Injection Phase

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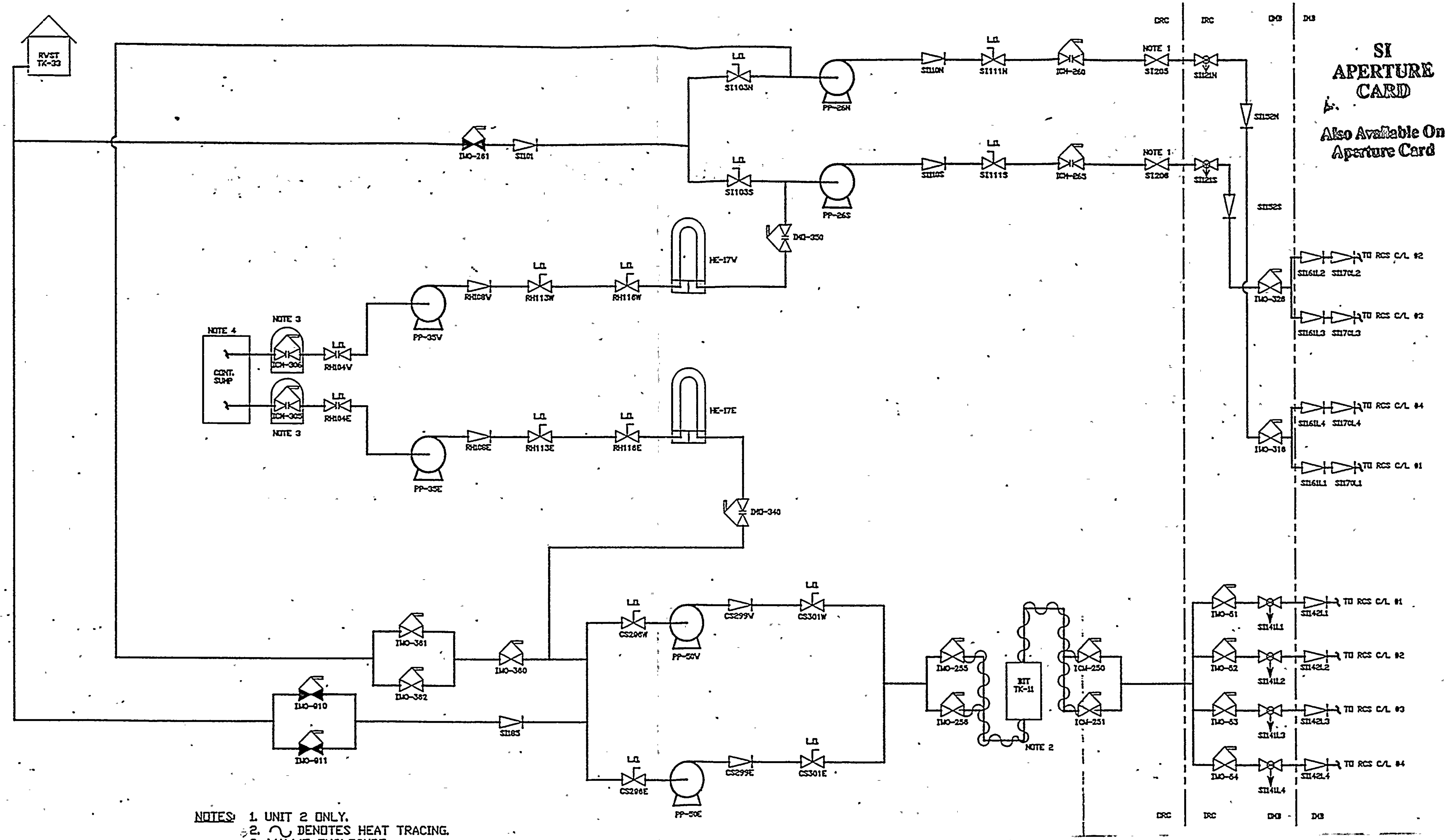

NOTES: 1. UNIT 2 ONLY.
2. ∿ DENOTES HEAT TRACING.
3. VALVE ENCLOSURE:
TK-84 ⟶ ICM-305
TK-85 ⟶ ICM-306.
4. INSIDE REACTOR CONTAINMENT.
5. L.O. = LOCKED OPEN.

WEC-0449
5-29-91
JAH

Figure 3.2-16
High Head Cooling System Simplified Flow Diagram
- Recirculation Phase

3.2.1.8 Electric Power System

The Cook Nuclear Plant's electrical systems are designed to ensure a continuous supply of electrical power to all essential plant equipment during normal operation and under abnormal conditions.

The electric power system (EPS) is made up of the following systems: 4160 and 600 V AC, 250 V DC, and 120 V AC. Each of these subsystems are described in this section.

3.2.1.8.1 4160 V AC System

The 4160 V AC system includes the 600 V AC power supply, offsite power, and the diesel generator systems.

The primary functions of the 4160 V AC electric power system are to:

- Provide a reliable source of motive power to all electric motors rated at 400 hp or larger.
- Provide a reliable source of electric power to the 600 V AC buses via a 2000 kVA, 4160/480 V transformer for essential (emergency) equipment, and a 1500 kVA, 4160/600 V transformer for nonessential 600 V equipment.
- Provide a reliable source of electric power to the 480 V AC buses via two 1000 kVA 4160/600 V transformers.

The primary functions of the 600 V AC electric power system are to:

- Provide a reliable source of motive power to electric motors rated up to 400 hp including emergency equipment required in the event of a failure of normal power supplies.
- Provide a back up power source for essential instrumentation and reactor protection circuits via a 600/120 V transformer.
- Provide a power source for certain interruptible 120 V loads.

The EPS, which provides power to and controls the operation of electrically-driven plant auxiliary equipment, can be fed from any of four electric power sources: Plant Turbine Generators, Preferred Offsite Power, Alternate Offsite Power, and Emergency Diesel Generators.

During normal plant operation, all auxiliary power is supplied from the generator terminals through the normal unit auxiliary transformers (1AB and 1CD). Figure 3.2-18 shows a simplified one-line diagram. Figure 3.2-17 is a summary of symbols and their meanings used in the one-line diagrams. The reserve auxiliary transformers (RATs) also provide power to the 4160 V switchgear busses during startup or shutdown operations.

Upon turbine-generator trip, the station auxiliaries are automatically and instantaneously transferred to the Preferred Offsite Power Source reserve auxiliary transformers (101AB and 101CD) to assure continued power to equipment when the main generator is off the line.

The Preferred Offsite Power auxiliary system is arranged so that the 345 MVA tertiary winding of transformer No. 4 supplies transformers 101AB and 101CD. An alternate supply source for transformers 101AB and 101CD is from the 150 MVA 345/34.5 kV transformer No. 5 which is a full power alternate to transformer No. 4.

The Alternate Offsite Power source is the 69 kV Derby/Hoover/Ugine-Bridgman circuit which supplies power to the two 7500 kVA emergency power transformers (EPTs) located in their own station near the plant. (See

Figure 3.2-19.) One of the 7500 kVA transformers (12-EP-1) supplies power directly to the safety buses without connection to the non-safety buses through circuit breaker 1EP located in the station. The other transformer (12-EP-2) supplies power to the Visitor's Center, Roadway Lighting, Plant Security System and other miscellaneous loads. The second transformer (12-EP-2) is also a backup unit for the first transformer (12-EP-1) and may be switched in place of the first if required. There are no other switching arrangements for this power source.

Each unit also has two 4160 V, 3 phase, 60-cycle, 3500 kW emergency diesel generators (EDG) which are individually capable of supplying power to operate the engineered safety features and protection systems required to safely shutdown the plant and avoid undue risk to public health and safety.

The 4160 V switchgear is arranged in eight bus sections per unit as follows: Buses 1A, 1B, 1C, and 1D are non-safety class while T11A, T11B, T11C, and T11D are safety class buses.

Upon loss of power to a 4160 V safety bus, the associated diesel generator starts automatically and begins to accept load within 10 seconds. The load shed circuitry will automatically open the circuit breaker which normally supplies power to the 4160 V safety bus from the 4160 V non-safety bus and trip all 4160 V safety bus breakers and nonessential 600 V loads. The circuit breaker from the EDG is then automatically closed to re-energize the 4160 V safety buses when the EDG reaches rated speed and voltage. The EDGs will then supply all equipment which must operate under emergency conditions. The EDGs are arranged so that EDG-1AB supplies power to safety busses T11A and T11B while EDG-1CD supplies power to safety busses T11C and T11D. The 4160/600 V AC transformers are energized first, then the 4160 V safety related motors and the 600 V Nonessential Service Water Pump motors are time sequenced back onto their respective buses by individual timer relays in the pump motor circuit breakers.

The 600 V system consists of six buses which are fed individually from the 4160 V AC system through six 4160/600 V transformers, TR11A, TR11B, TR11C, TR11D, TR1BMC and TR11CMC. Four of the 600 V buses feed safety and non-safety related motors up to 400 hp. Each motor 100 hp or more is fed through a 600 V circuit breaker while motors 100 hp or less are fed via motor control centers (MCCs). The other two 600 V buses feed non-safety related loads less than 100 hp via MCCs. The 600 V buses are labeled as follows: Buses 11A, 11B, 11C, and 11D service safety class equipment while Buses 11BMC and 11CMC are non-safety class.

If required, each 600 V bus can be electrically connected to another bus via the appropriate bus crosstie circuit breaker. The 600 V bus crosstie breakers close automatically when the 4160/600 V transformer differential or ground overcurrent relays operate. This automatic closure can only occur when one of the associated 600 V feeder breakers is open and both of the breakers which connect the associated 4 kV buses to the EDGs are open. The crosstie breakers can also be manually closed using control switches on the Station Auxiliary Panel in the Control Room. The crosstie breakers are automatically tripped if either EDG starts. The breakers are as follows: Buses 11A and 11C are cross connected via breaker 11AC and buses 11B and 11D are connected via breaker 11BD.

There are a total of 18 models (fault trees) associated with the 4160 V EPS and they are designated as follows:

T11A, T11B, T11C and T11D	-	These models define the logic associated with the unavailability of the four 4.16 kV safety buses during all postulated accidents when offsite power is available.
T11AP, T11BP, T11CP and T11DP	-	These models define the logic associated with the unavailability of the four 4.16 kV safety buses when the preferred offsite power source is lost.
11A, 11B,	-	These models define the logic associated with the unavailability of the four

11C and 11D	600 V AC safety buses during all postulated accidents when offsite power is available.
11AZ, 11BZ - 11CZ and 11DZ	These models, which define the logic associated with the unavailability of the four 600 V AC safety buses during all postulated accidents when offsite power is available, are used to prevent circular logic when two 600 V AC buses are crosstied. In addition, these models are used to define the system logic during a loss of offsite power when automatic bus crosstie is prevented to ensure that the diesels are not run in parallel.
1AB and 1CD -	These models define the logic associated with the starting and operation of the emergency diesel generators.

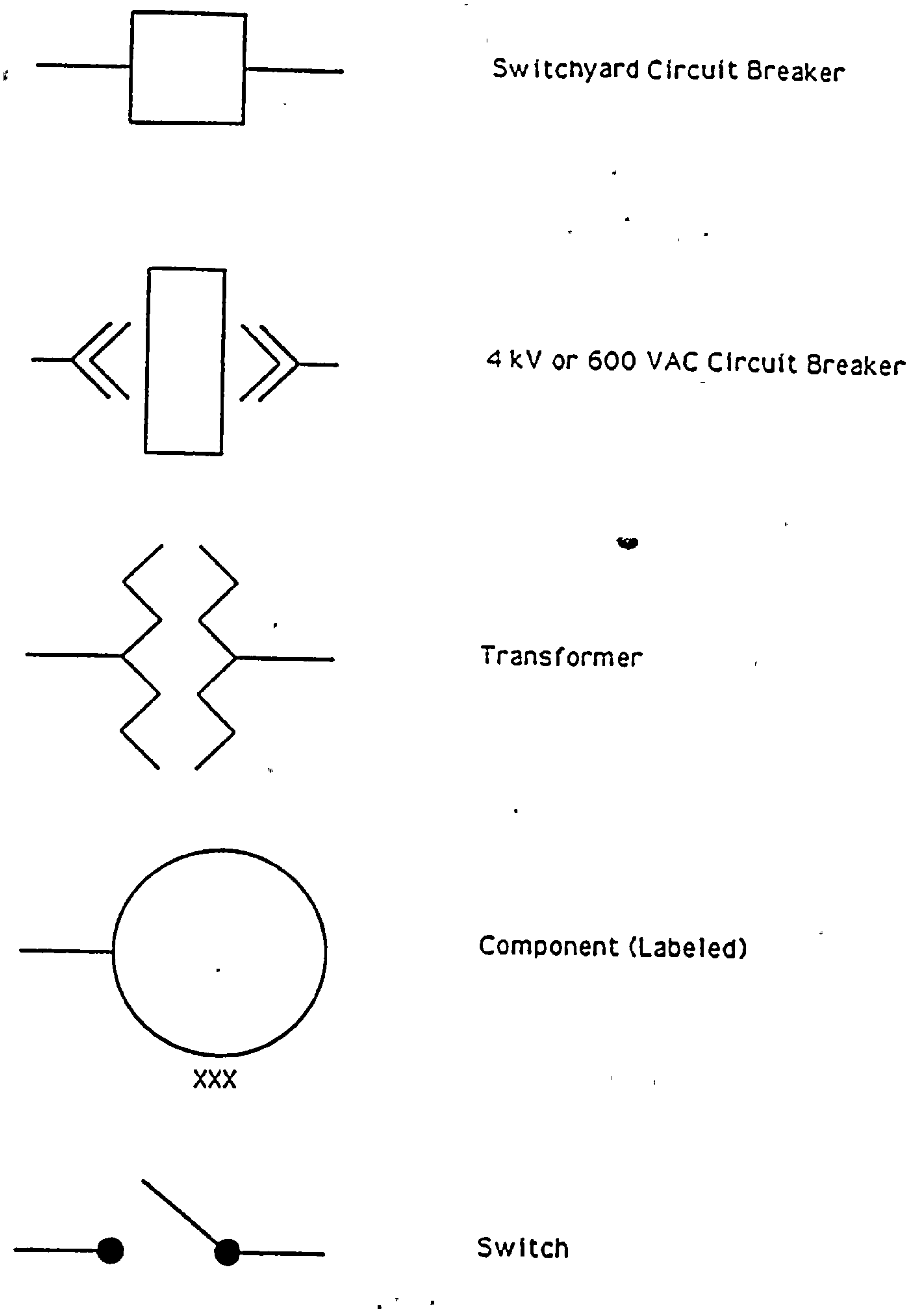

Figure 3.2-17
Electric Power Symbols

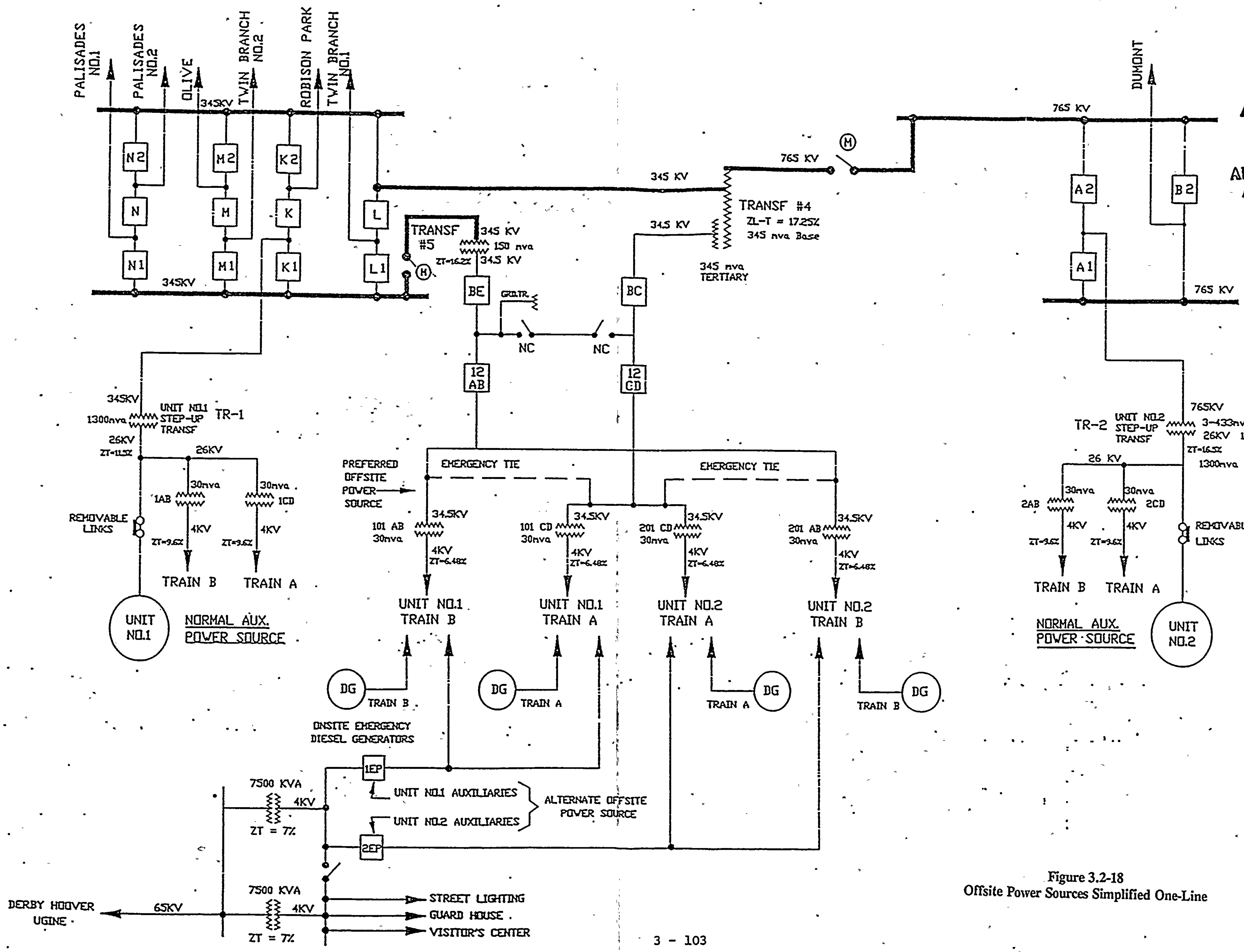

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Figure 3.2-18
Offsite Power Sources Simplified One-Line

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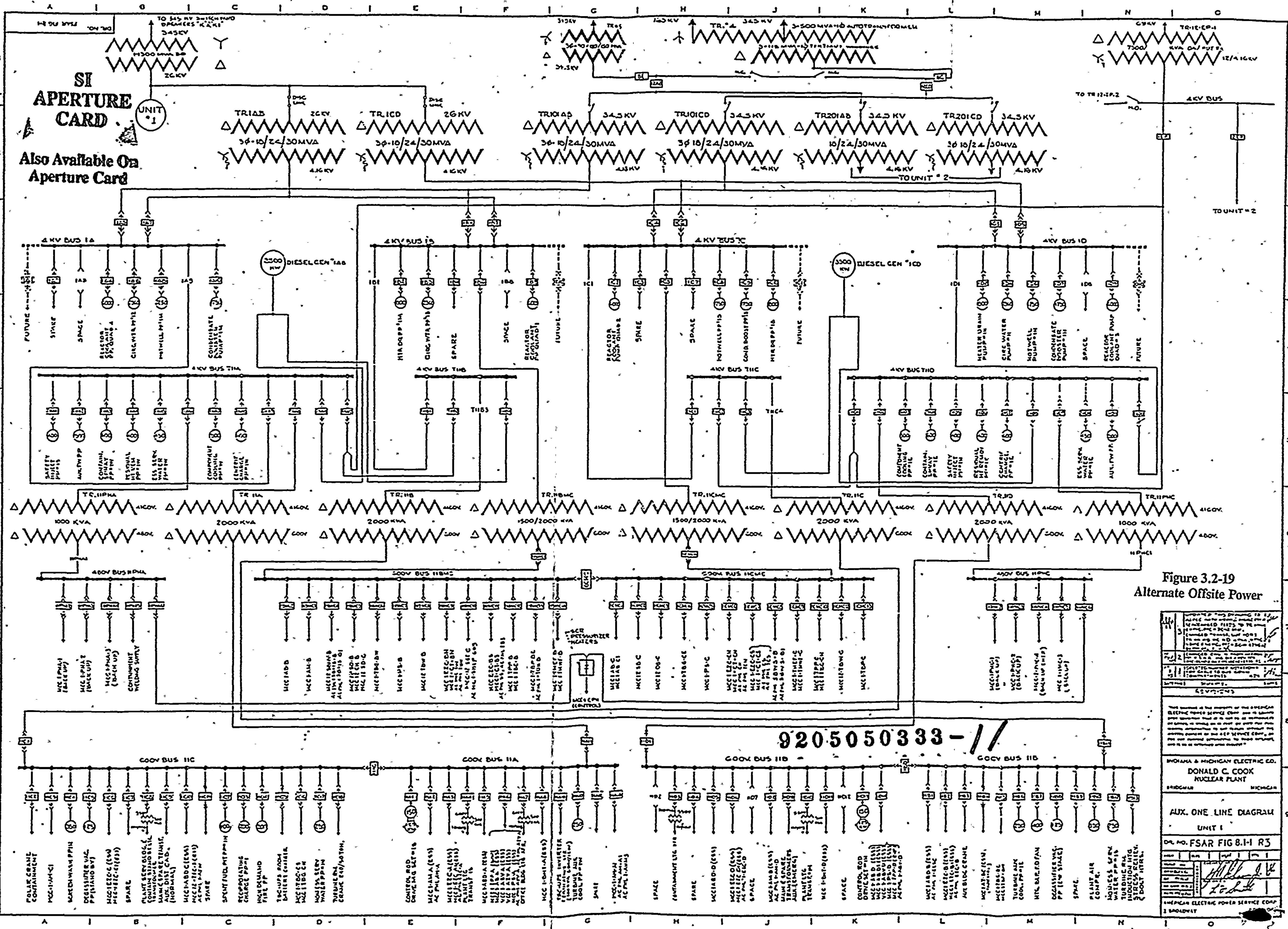


Figure 3.2-19
Alternate Offsite Power

3.2.1.8.2 250VDC System

The primary function of the 250 V DC system is to provide a reliable source of continuous electric power for supply and control of plant safety systems. Included in these safety systems are the reactor trip system, engineered safety features, and auxiliary support features.

The 250 V DC system consists of three major battery groups for each unit:

1) Station or Plant Battery Train A (Bus CD or Green Train)

2) Station or Plant Battery Train B (Bus AB or Red Train)

3) N Train Battery (Turbine Driven Auxiliary Feedwater Control or Brown Train)

Power is supplied from 116 (plant battery) or 117 (N-train battery) individual cells which make up each battery system. Each battery group has two individual battery chargers that maintain each train at its rated stored capacity during normal operation.

The battery chargers are normally energized using available AC station power. In the event of a loss of normal station and offsite system power, the A & B train chargers are energized by the emergency diesel generators.

A single battery charger is normally in operation providing power to the appropriate buses and battery group. The remaining battery charger is in standby. There exists no automatic switchover between battery chargers. In the event of a loss of the working battery charger, manual actions are necessary to align the standby battery charger.

The distribution system for the 250 V DC System is comprised of various direct current switchgear, distribution panels, bus, and individual feeders. The DC switchgear consists mainly of disconnect switches and fuses plus a limited number of molded case circuit breakers.

A number of different DC plant loads are served from each of the main and transfer distribution cabinets. The plant loads include control circuits (switchgear & annunciators), static inverters, valve control centers, emergency lighting, and motor control centers plus vital bus inverters, fire protection control, and main turbine lube oil pumps. The N Train Battery supplies the turbine driven auxiliary feedwater (TDAFW) control bus, the ATWS mitigation system actuation circuitry (AMSAC) inverter, and a valve control center.

The only manual actions associated with this system are the starting of the standby battery charger and the closing of manual switches in order to allow the feeding of one train from the opposite train.

Figure 3.2-20 shows a simplified one-line diagram representing the 250 V DC Electric Power System.

There are four models (fault trees) associated with the 250 V DC electric power system. Each model represents the unavailability of a particular 250 V DC transfer cabinet or distribution cabinet. The models are designated as follows:

DCB - This system model defines the unavailability of 250 V DC Train B Transfer Cabinet TDAB and Distribution Cabinets MCAB and MDAB under normal AC power conditions.

DCA - This system model defines the unavailability of 250 V DC Train A Transfer Cabinet TDCD and Distribution Cabinets MDCD and MCCD under normal AC power conditions.

DCN - This system model defines the unavailability of 250 V DC N Train Transfer Distribution Cabinet 1-DCN under normal AC power conditions.

DCNSBO - This system model defines the unavailability of 250 V DC N Train Transfer Distribution Cabinet 1-DCN under station blackout conditions.

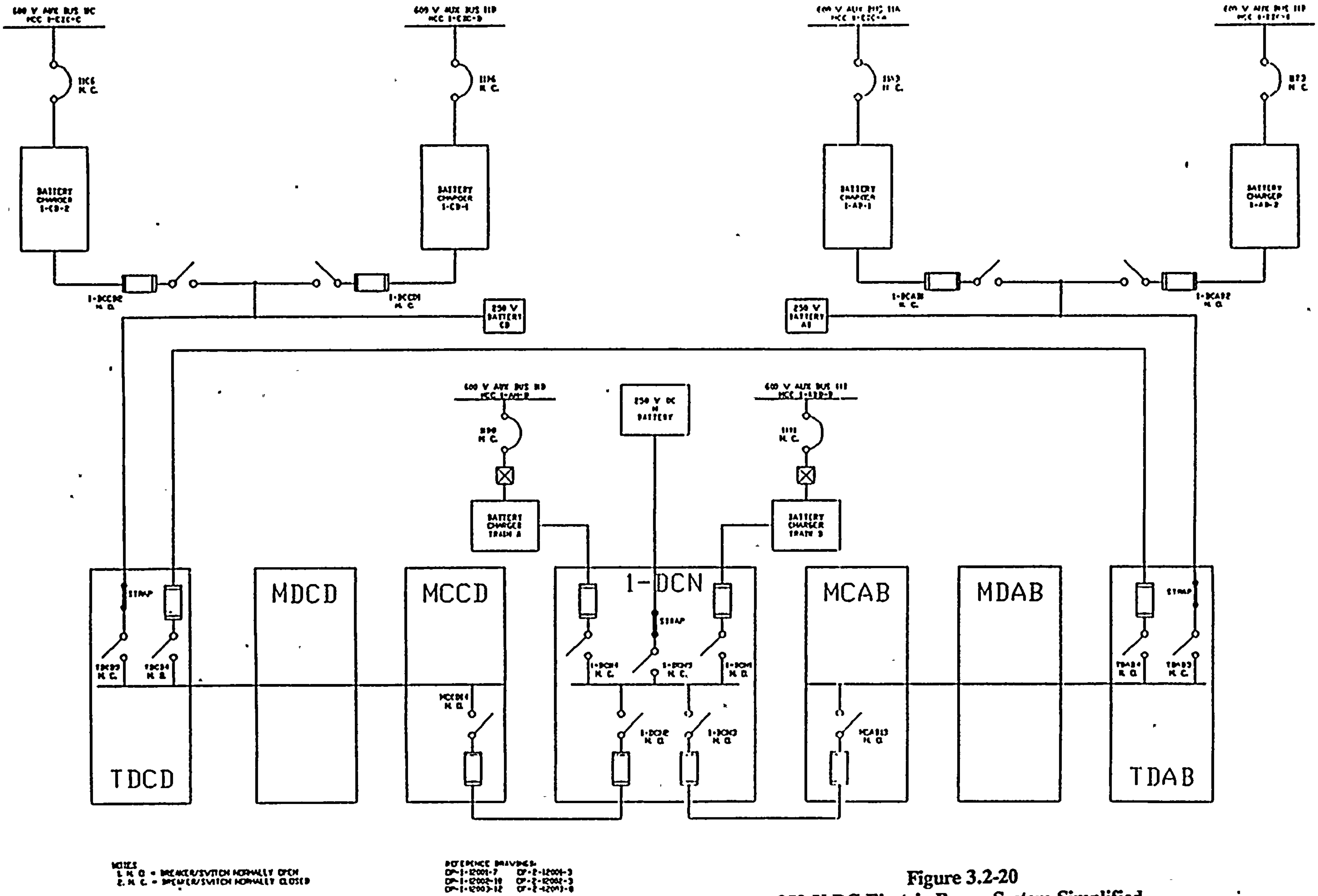


Figure 3.2-20
250 V DC Electric Power System Simplified One-Line Diagram

3.2.1.8.3 120VAC System

The primary function of the 120 V AC electric power system is to provide a reliable source of electric power to essential instrumentation, the reactor protection system (RPS), the engineered safety features protection system, and the hydrogen igniters.

There are four 120 V AC vital bus instrument systems in each unit. Each distribution system, commonly called a CRID (control room instrument distribution), consists of a static inverter, a regulating transformer, and a distribution panel as shown in the simplified one-line diagram in Figure 3.2-21. Each CRID has four sources of power, any one of which can supply sufficient power for CRID operation:

1) The output of a plant battery charger

2) 250 V DC station battery

3) 600 V AC Motor Control Center

4) 120/208 V AC distribution cabinet CRP-3

The Class 1E power sources (Battery charger output, 250 V DC station battery, and 600 V AC motor control center) for two of the four CRIDs are Train A associated while the other two are Train B associated. The third source of power, the 600 V AC motor control center, feeds the regulating transformer but is not uninterruptible. The motor control center is, however, Class 1E. The fourth source of power, 120/208 V AC distribution cabinet CRP-3, is non-Class 1E and feeds directly into the distribution panels, bypassing the inverters and regulators.

Transfers between power sources are automatic (except for the non-Class 1E power source) and do not disturb vital bus voltage and frequency.

Also included as part of the 120 V AC electric power system are distribution panels AFW and ELSC. These panels distribute 120 V AC power to the hydrogen igniters and various plant instrumentation. Each panel uses its own individual transformer to convert 600 V AC power from a 600 V Auxiliary Bus to 120 V AC power. Figure 3.2-22 shows the simplified one-line diagram representing such a panel.

The 120 V AC electric power system is required to be in service at all times to provide a continuous source of power to essential instrumentation, the RPS and engineered safety features protection system. Under normal conditions, the CRID distribution panels are fed from the output of the static inverters through the transfer and bypass switches. Normal power to the inverters is supplied by the 250 V DC system. During normal conditions, the station battery chargers supply the necessary power. In the event of a loss of AC power (for any reason), the station batteries provide the 250 V DC power needed by the inverters.

An alternate source of power to the CRID distribution panels is obtained from the regulating transformers. These transformers take 600 V AC power from the motor control centers, transform it to 120 V AC, and feed it to the distribution panels through the transfer and bypass switches. This alternate source of power is engaged automatically through the transfer switch with no disturbance of bus voltage and/or frequency whenever power is lost to the static inverters.

If the alternate power source, the 600 V AC motor control center, is unavailable, the CRID distribution panels can be energized by manually operating the mechanically interlocked main panel circuit breaker, thus feeding the distribution panels from the Balance of Plant 120/208 V AC distribution cabinet CRP-3.

Distribution panels AFW and ELSC are required to operate constantly to supply power for both trains of the hydrogen igniters. The source of power to their respective transformers is 600 V AC Buses 11C and 11B, respectively.

There are six models (fault trees) associated with the 120 V AC electric power system. Each model represents the unavailability of a particular CRID or other distribution panel. The models are designated as follows:

CRID1	-	This system model defines the unavailability of CRID I under all AC power conditions.
CRID2	-	This system model defines the unavailability of CRID II under all AC power conditions.
CRID3	-	This system model defines the unavailability of CRID III under all AC power conditions.
CRID4	-	This system model defines the unavailability of CRID IV under all AC power conditions.
120AFW	-	This system model defines the unavailability of distribution panel AFW under all AC power conditions.
ELSC	-	This system model defines the unavailability of distribution panel ELSC under all AC power conditions.

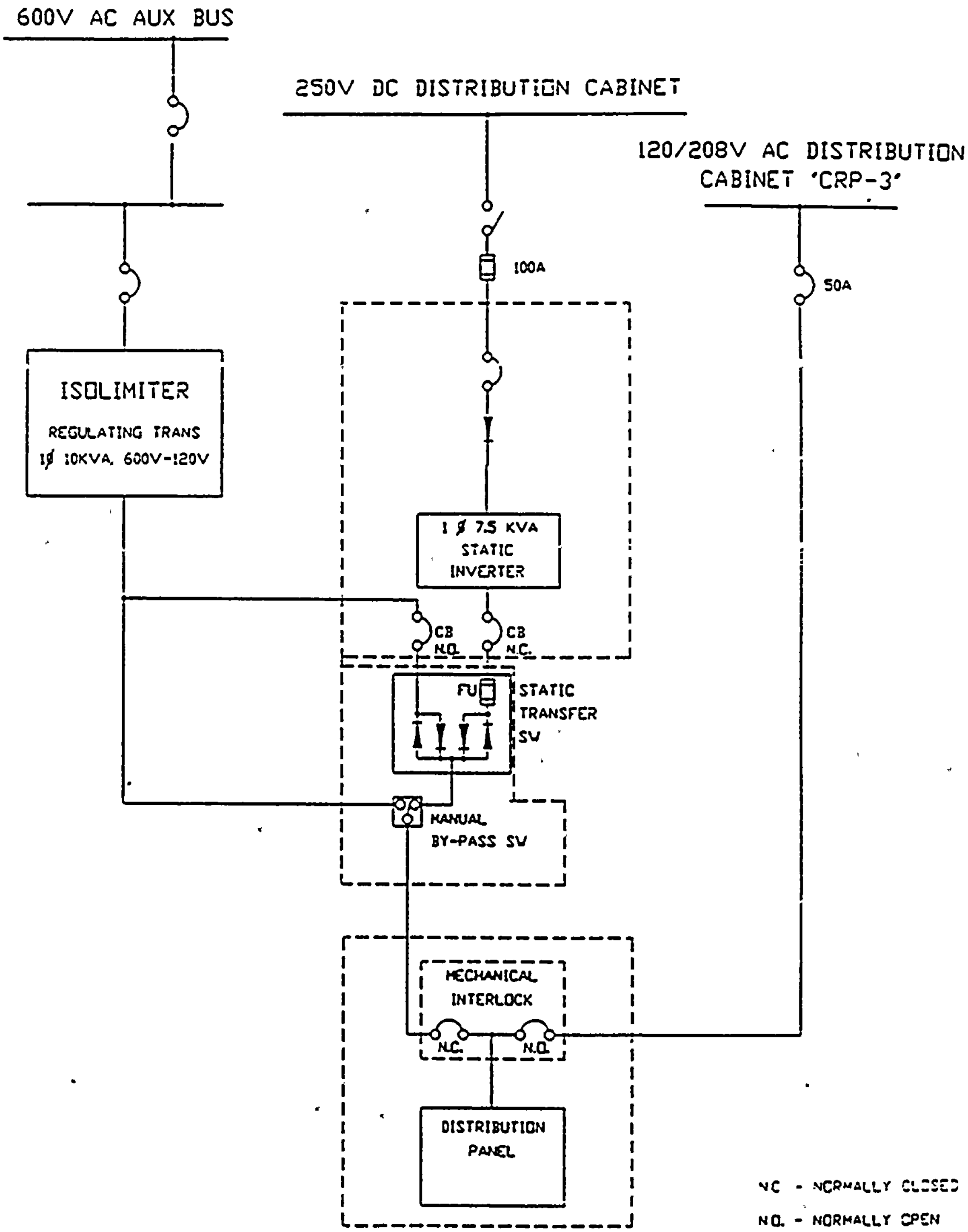


Figure 3.2-21
Typical 120 V AC Vital Bus Instrument System Simplified One-Line Diagram

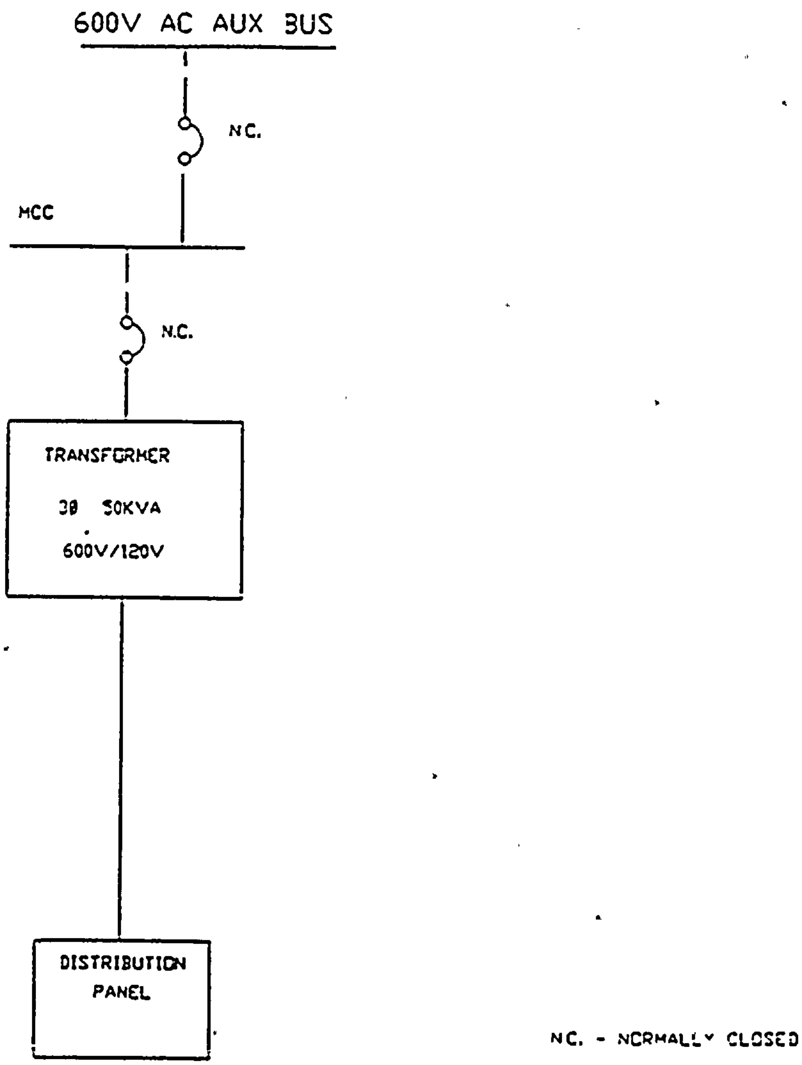


Figure 3.2-22
Typical 120 V AC Panel (ELSC, AFW)

3.2.1.9 Main Feedwater and Condensate Systems

The feedwater system (FW), in conjunction with the condensate system, returns the condensed steam from the turbine condensers and the feedwater heater drains to the steam generators while maintaining water inventory throughout the cycle. These systems automatically maintain the water level of the steam generators during normal unit operation.

The condensate/feedwater system provides a continuous flow of water at uniform temperature to all steam generators under normal operating conditions. The system, during load changes, maintains sufficient fluid capacity to accommodate changes due to expansion and contraction resulting from thermal and pressure effects on the steam generator fluid inventory. Simplified flow diagrams of the systems including the supporting circulating water system are shown in Figures 3.2-23 and 3.2-24.

Condensate collects in the main condenser hotwells after being condensed in the main condenser shells (A, B and C) and the feed pump turbine condensers (which gravity feed to main condenser shell B).

Condensate is then withdrawn from the condenser hotwells by three half-capacity motor driven vertical hotwell pumps. The pumps discharge into a common header which carries the condensate through four parallel steam jet air ejector condensers to the suction of three half-capacity motor driven horizontal condensate booster pumps. The condensate flow is then pumped by the booster pumps through three parallel strings of heaters (each string consists of a separate external drain cooler and a low pressure extraction feedwater heater). Downstream of these heaters are two parallel strings of three stages of low pressure feedwater heaters with integral drain coolers. The condensate from the No. 4 heaters is then routed to two half-capacity main feed pumps via a common header.

There are two turbine driven variable speed main feed pumps installed in parallel, each has its own suction strainer. Minimum flow through each pump is maintained by emergency leakoff valves with lines which terminate as spray pipes in the condensers. The recirculation valves open sequentially as the flow decreases below 4000 and 2000 gpm. The feedwater from the main feed pumps is discharged through two parallel strings of high pressure feedwater heaters, each string consisting of a No. 5 and a No. 6 heater. After discharging from the No. 6 heaters, the feedwater is distributed to the four steam generators through individual feedwater control valves.

Upon receipt of a feedwater isolation signal, the feed pump turbines are tripped and the main feedwater control valves and the feed pump discharge valves close.

There are three models (fault trees) associated with the feed and condensate systems. Each of these models represents the unavailability of this system in response to different accident events.

12FEED - This fault tree modeled the feedwater system from the suction of the main feedwater pumps to the discharge of the high pressure feedwater heaters. Fault tree 13CONDMF is called in as a sub-tree to account for condensate feeding the feedwater system. Following a reactor scram, the feedwater pumps are tripped and the feed pump discharge valves are shut. This model included circulating water flow to the feedwater pump turbine condensers and the main condensers. Success is achieved if one feedwater pump flow path is brought on-line.

13COND - This fault tree modeled the scenario in which the condensate system is feeding depressurized steam generators through idle feedwater pumps. This model included the feedwater control valves and did not require the support of the circulating water system. Following a reactor scram, the two operating booster pumps and the two operating hotwell pumps would be recirculating to the main condenser. Success requires one hotwell pump

(operating or the standby pump starting) feeding one booster pump (operating or standby starting) feeding two of four steam generators.

13CONDMF - This fault tree is similar to the 13COND tree except this tree models the normal lineup where the hotwell and booster pumps feed operable feedwater pumps. Again, the feedwater control valves were included in this tree.

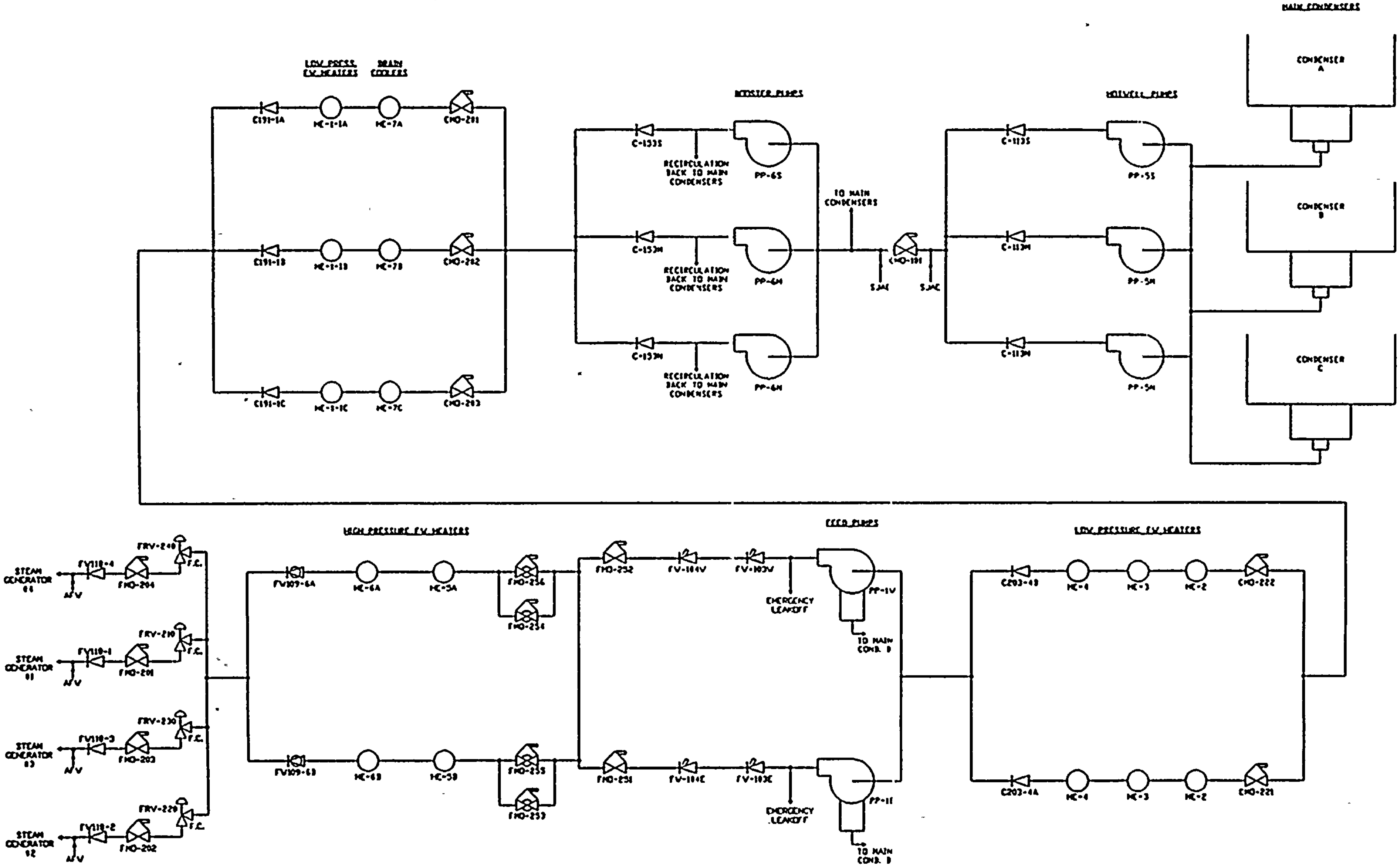


Figure 3.2-23
Feedwater/Condensate System Simplified Flow Diagram

REFERENCE DRAWINGS-
OP-1-5106-23
OP-1-5107-35
OP-1-5107A-17
OP-1-51033-3

NOTES
1. VALVE NOTATION DEFINITIONS
a) F.C. = FAILS CLOSED

2. SJAE = STEAM JET AIR EJECTORS

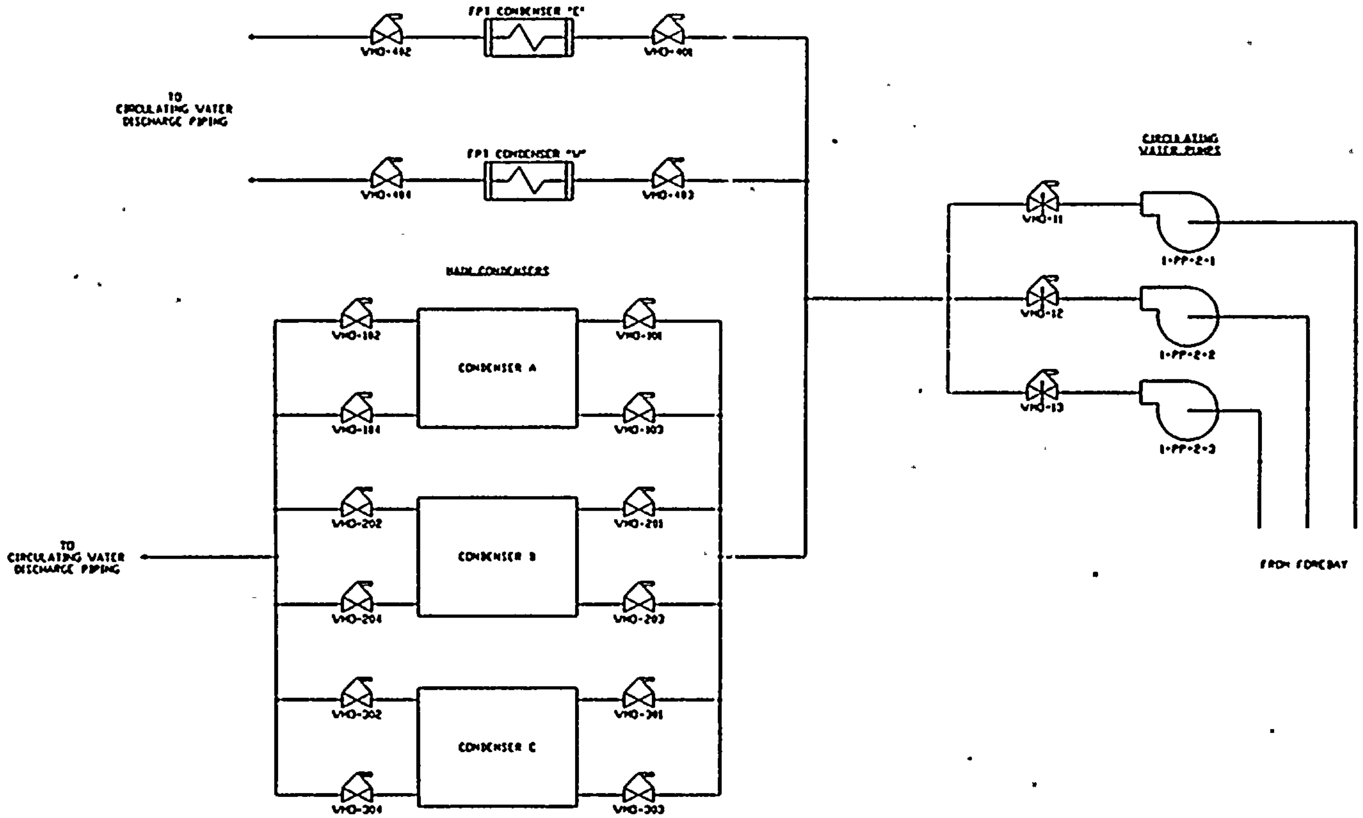


Figure 3.2-24
Circulating Water System Simplified Flow Diagram

REFERENCE DRAWINGS:

3.2.1.10 Main Steam System

The main steam system transports steam generated in the steam generators inside containment to equipment utilizing main steam located outside containment. For this study, the main steam system will be used to remove primary decay heat, and isolate the steam generators in the event of a steam line rupture or a steam generator tube rupture. The main steam system also supplies steam to drive the turbine driven auxiliary feedwater pump; however, this function is modelled within the auxiliary feedwater fault trees.

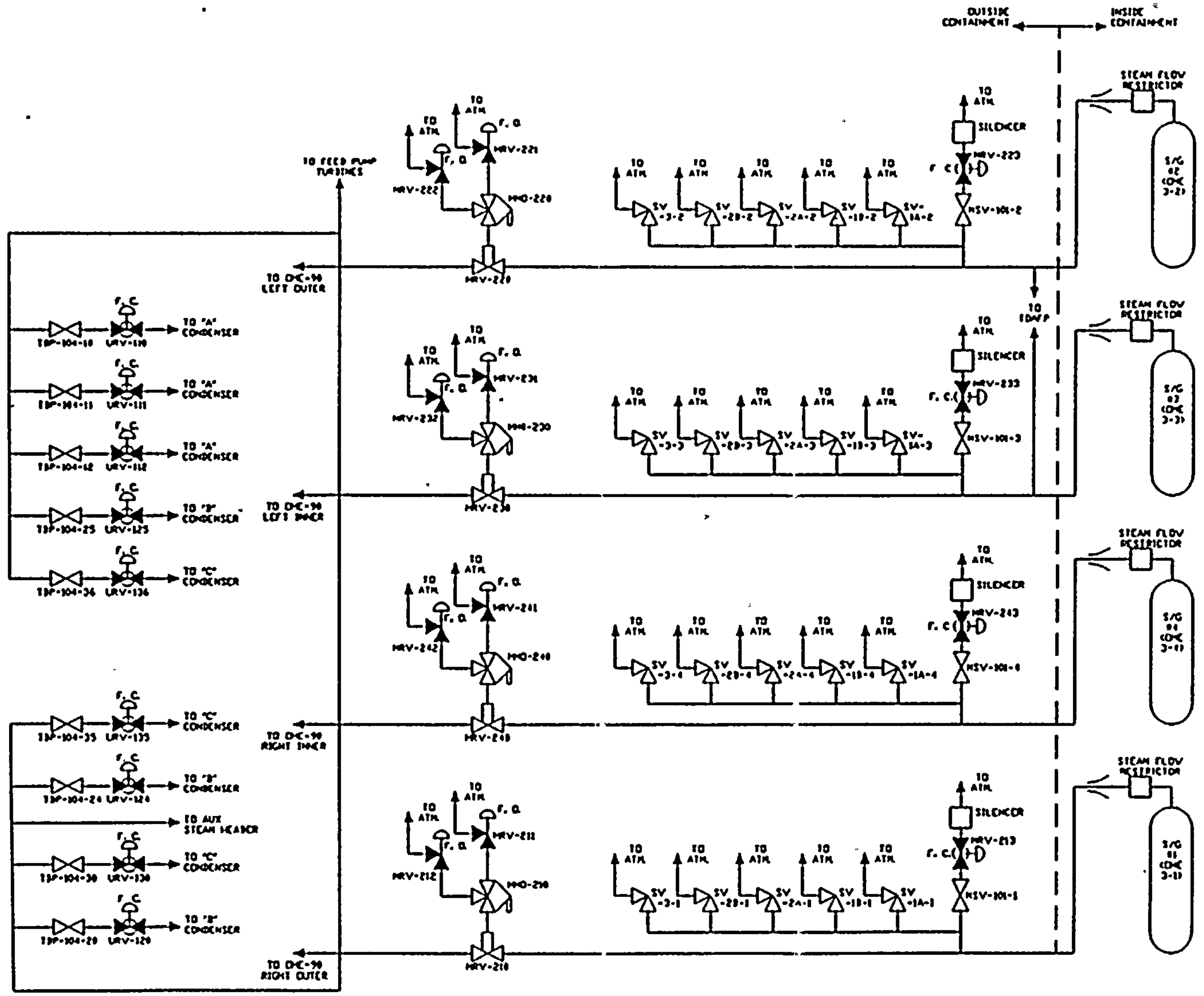
The main steam system is used to remove steam from the four steam generators. The steam is used to provide core decay heat removal. In addition, the main steam is used for the motive force to shut the steam generator stop valves for emergency isolation of the steam generators.

Steam flows through a separate line from each steam generator through a steam flow restrictor through a steam generator stop valve. On each steam generator, there are the five steam generator safety valves and one power operated relief valve. These are located between the steam flow restrictor and the steam generator stop valve.

Following the steam generator stop valves, the individual steam leads combine into a common header which equalizes pressure before the steam flows into the turbine admission valves. This header is also connected to the steam dump system. The steam dump system consists of nine valves which relieve steam to the condenser. This system can be used to control reactor coolant system temperature and pressure and remove core decay heat following a turbine trip. Figure 3.2-25 represents a simplified one-line diagram of the system.

There are four models (fault trees) associated with the Main Steam System. Each model represents the unavailability of this system in response to different accident events.

MS1	-	This system model defines the logic associated with the response of the main steam system to isolate at least three of four steam generators during a steam line rupture event.
SSV	-	This system model defines the logic associated with the response of the main steam system to maintain integrity of the faulted steam generator during a steam generator tube rupture event given that the operator fails to stabilize RCS pressure less than steam generator safety valve setpoints.
SGI	-	This system model defines the logic associated with the response of the main steam system to isolate at least one steam generator during a steam generator tube rupture event.
SGPORV	-	This system model defines the logic associated with the response of the main steam system to an operator initiated RCS cooldown. Success of this system model requires at least two of four steam generator PORVs to open.



NOTES
1. VALVE NOTATION DEFINITIONS
a) F. O. - FAILS OPEN
b) F. C. - FAILS CLOSED

REFERENCE DRAWINGS OP-1-5103-23
OP-1-5103A-18 OP-1-5103B-21 OP-1-5103B-3

Figure 3.2-25
Main Steam Sytem Simplified Flow Diagram

3.2.1.11 Pressurizer Power Operated Relief Valve and Safety Valve System

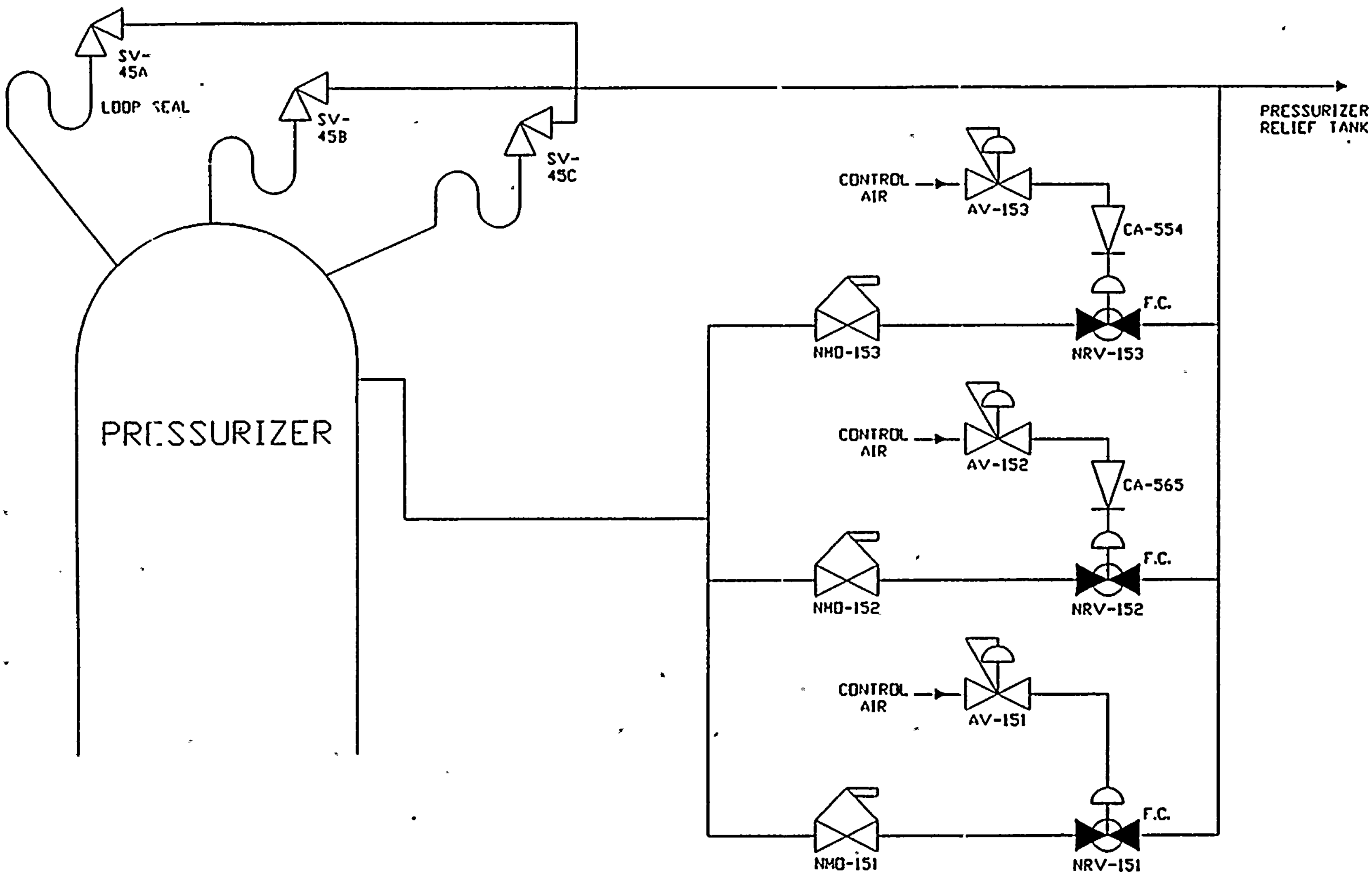
The pressurizer (PZR) is equipped with 3 power-operated relief valves (PORVs), which limit system pressure for a large power mismatch and thus lessen the likelihood of a reactor trip on high pressurizer pressure. The operation of these valves also limits the opening of the spring-loaded pressurizer safety valves.

A six-inch relief line is attached to the upper head of the pressurizer. The line divides into three parallel three-inch lines, each containing a power operated relief valve and a motor operated isolation valve. The relief valves are actuated by signals from the pressurizer pressure instrumentation. Actuation is set to prevent operation of the pressurizer safety valves. A motor operated isolation valve is located upstream of each relief valve and is closed in order to isolate an inoperable relief valve or to remove from service a relief valve with excessive leakage. Downstream of the power operated relief valves, the three three-inch lines recombine and discharge with the pressurizer safety valves into the pressurizer relief tank. Figure 3.2-26 represents a simplified one-line diagram of the system.

The PORVs rely on control air for actuation and 250 VDC power for the solenoids that actuate the control air to the PORVs. The MOVs upstream of each PORV receive their control power by a transformation from the 600V power that operates the MOVs.

There are four models (fault trees) associated with the PZR PORV and safety valve system. Each model represents the unavailability of this system in response to different accident events.

PORVP	-	This fault tree, which is part of a primary feed and bleed event, models two out of three PZR PORVs responding to operator action to open them or failing to stay open once opened. The motor operated blocking valves also have to be opened (also two of three) for the PORVs to be effective.
PZRSAFE	-	This fault tree models failure of all three safety valves and all three PZR PORVs to open in response to an anticipated transient without scram (ATWS) event if manual rod insertion is NOT successful.
PZRSAFE1	-	This fault tree models failure of all three safety valves and one of the PZR PORVs to open in response to an ATWS event if manual rod insertion is successful.
13PORV	-	This fault tree very closely resembles the PORVP tree except only one out of the three PORVs is required to open. This tree accommodates PORV response to a steam generator tube rupture (SGTR) event.



REFERENCE DRAWINGS:
OP-1-5128A-27
OP-1-5120D-3

Figure 3.2-26
Pressurizer PORV and Safety Valves Simplified Flow Diagram

3.2.1.12 Containment Air Recirculation and Hydrogen Skimmer System

The containment air recirculation and hydrogen skimmer system has two basic functions. One is the general recirculation of the containment atmosphere between upper and lower containments following a loss-of-coolant accident (LOCA). The second is the prevention of the accumulation of hydrogen in restricted areas within containment following a LOCA.

The containment air recirculation and hydrogen skimmer system consists of two redundant independent systems which includes fans, backdraft dampers, valves, piping and ductwork. The system is normally in standby and is actuated automatically by a hi-hi containment pressure signal (2.9 psig) or manually from the control room. Both air recirculation systems have an air recirculation fan located in the upper containment. Each system has a total capacity of 41,800 cfm. The fans discharge via the annular space between the crane wall and the containment liner into the lower containment. The fans are provided with backdraft dampers on the discharge to prevent backflow during initial LOCA blowdown into the lower containment.

Each air recirculation fan has its own intake system which includes three separate headers. The three separate headers perform the following functions:

(a) Draw 39,000 cfm from the upper containment in the immediate vicinity of the fan.

(b) Draw 1,000 cfm from the upper containment at the top of the dome.

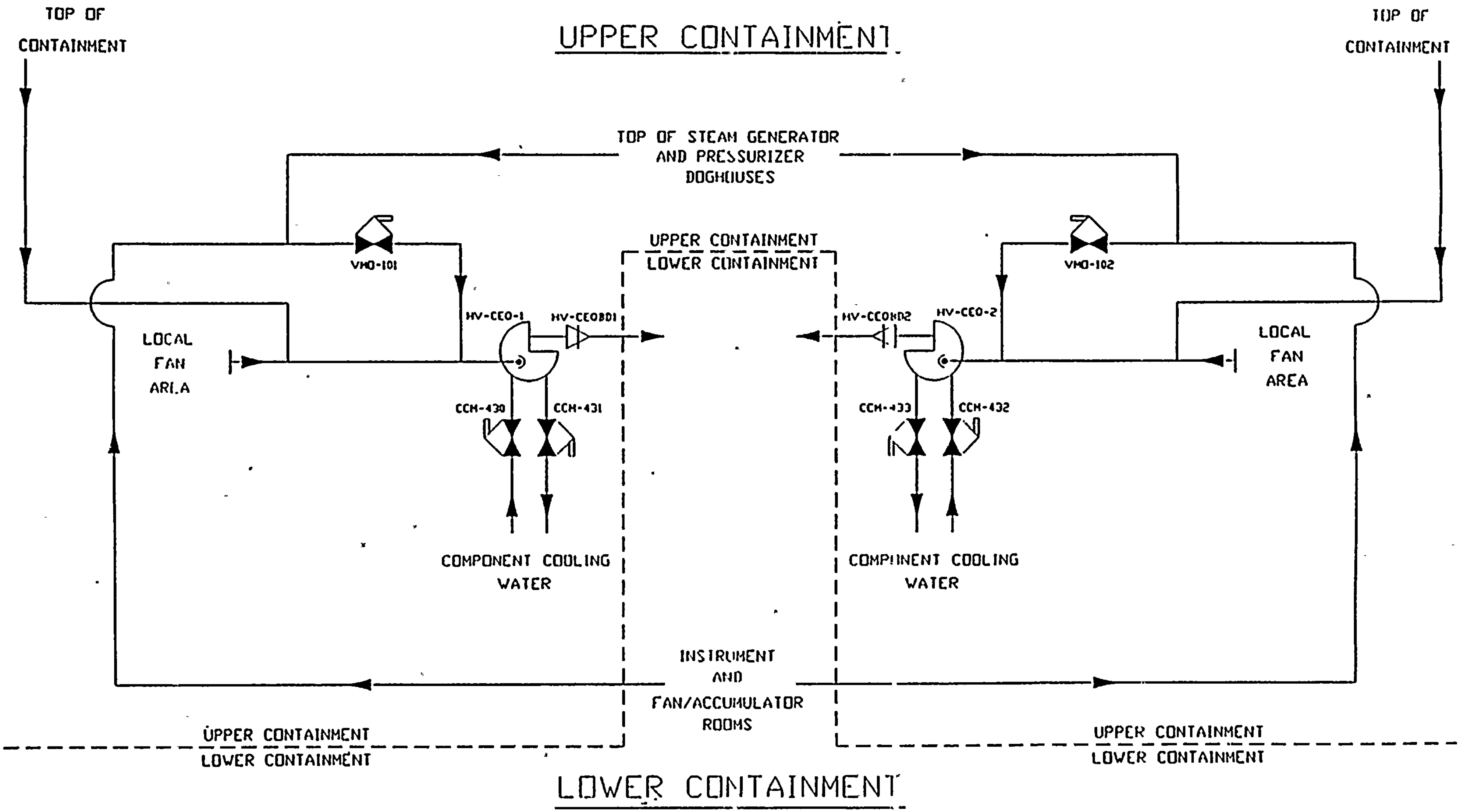
(c) Draw 1,800 cfm from the potential hydrogen pockets in the lower containment via a hydrogen skimmer system.

This continuous air flow (once initiated) is of such a rate as to limit potential local hydrogen concentrations. The potential areas of hydrogen pocketing are: the three rooms (i.e., east and west fan room and instrument room) in the annular space between the crane wall and the liner, the top of the steam generator and pressurizer enclosures and the top of the containment dome.

The containment air recirculation and hydrogen skimmer system is shown in Figure 3.2-27.

There is one model (fault tree) associated with the containment air recirculation and hydrogen skimmer system.

CF - The containment air recirculation and hydrogen skimmer system fault tree model is comprised of both trains of containment air recirculation (CEQ) fan systems responding to a Phase B containment isolation signal. Success is achieved if one of the two trains becomes operational. Each train consists of the inlet shutoff valve (VMO valve), the outlet backdraft damper, the CEQ fan itself and the associated component cooling water (CCW) valves that allow cooling water to the CEQ fan motor air cooler. The motor-operated CCW valves that supply CCW to the miscellaneous header are also included in this model instead of in the CCW models. A failure of these valves (CMO-415, -416, -413, -411) does not affect the operability of the CCW trains.



REFERENCE DRAWINGS:
OP-1-5147A-22
OP-1-5135B-12

NOTES
1. CCM-430, -431, -432, -433 ARE OUTSIDE OF CONTAINMENT
2. CMO-416, -415, -413, -411 ARE NOT SHOWN

Figure 3.2-27
Containment Air Recirculation and Hydrogen Skimmer System Simplified Flow Diagrams

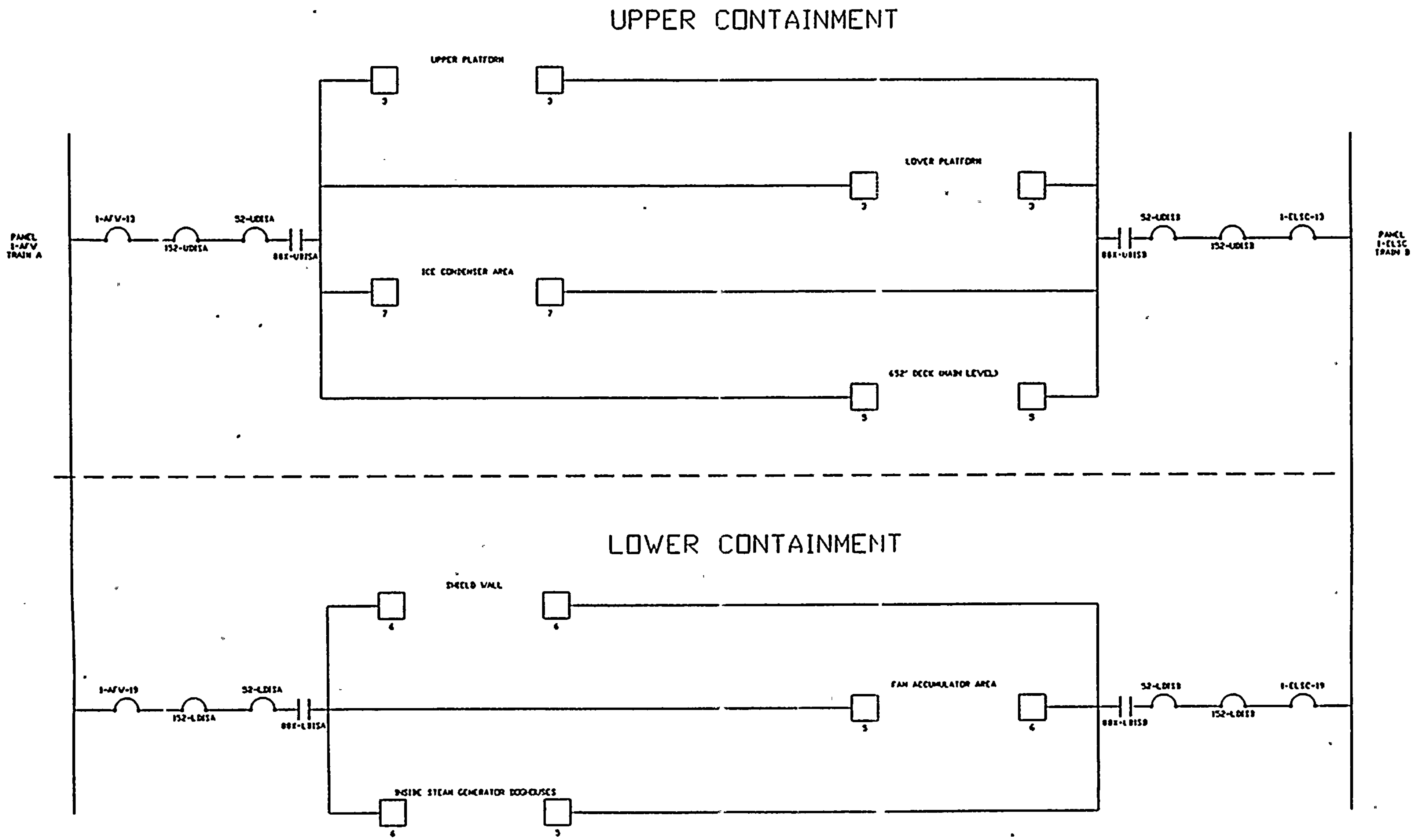
3.2.1.13 Hydrogen Igniter System

A distributed ignition system (DIS) (also known as the hydrogen igniter system) is provided to ensure adequate hydrogen control in containment during a degraded core cooling event. The DIS utilizes electrical resistance heating elements (glow plugs) located throughout the containment building. The DIS will be manually actuated from the control room when called upon by the Emergency Operating Procedures (EOPs).

The DIS is a two-train system employing a total of 70 igniter assemblies located throughout the containment building. The system was installed in response to post-Three Mile Island containment hydrogen control concerns and is meant, in general, to limit post-accident hydrogen concentrations. Each train of 35 igniter assemblies is further divided into two groups: one group of 17 assemblies is in the lower volume area and the second group of 18 assemblies is in the upper volume area (including the ice condenser upper plenum volume). Igniters are located above the maximum floodup levels and are placed in regions throughout containment to promote combustion of lean hydrogen/air/steam mixtures. The system is shown in Figure 3.2-28.

There is one model (fault tree) associated with the DIS. This model represents the system unavailability as it responds to actuation.

HI	-	This fault tree models the DIS system such that a failure occurs if both Train A and Train B igniters in either upper or lower containment fail to actuate. Failure of individual glow plugs is not included.



REFERENCE DRAWINGS:
PSI-97469-8
PSI-97470-8
OP-1-90264-3

NOTES

1. NUMBER = NUMBER OF IGNITERS
2. □ = SAMPLE IGNITER BOX
3. ⌒ = BREAKER
4. || = CONTACTOR

Figure 3.2-28
Hydrogen Igniters (Distributed Ignition System)
Simplified Fow Diagrams

3.2.1.14 Refueling Canal Drains

Three refueling canal drains are used to return containment spray water injected into upper containment back to the lower containment sump. The consequences of failure of the refueling canal drains is severe in that it would eventually prohibit emergency core cooling system (ECCS) and containment spray system (CTS) recirculation.

The two 12" and one 10" refueling canal drain connections are basically open pipes in the floor of the refueling canal and contain no valves but are equipped with a flange on the refueling canal side. During refueling outages, these drains are covered to allow filling the refueling canal for fuel transport from the reactor to the spent fuel pool. Prior to startup, the covers are removed.

The drains are approximately 7.36' apart, center-to-center, and each has a 1" lip above the floor elevation. They are located in the refueling canal, just outside of the removable gate separating the reactor cavity from the refueling canal.

3.2.2 System Analysis

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

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

The approach used to develop the fault tree models is consistent with the guidance provide in Reference 26. Cook Nuclear Plant Fault Tree guidelines were developed to ensure that a consistent approach was used in establishing modeling assumptions and in structuring the models. They provide guidance in areas such as the selection of random hardware failures to model, treatment of test and maintenance outages, modeling of operator errors, and common cause failure analysis. The following provides an overview of the fault tree construction process.

STEP 1 Develop Simplified Flow Diagram

A simplified flow diagram was developed from the detailed plant drawings of each modelled system to provide the level of detail required for the modeling of the system. Support system interfaces, normal component position, etc., were identified on the simplified diagram. The plant drawings were simplified through the elimination of flow paths not directly related with the main process (such as fill and sampling lines). Small diverted flow paths which did not cause failure of the system were removed. The original Cook Nuclear Plant drawings from which the simplified diagrams were derived were identified on the simplified flow diagrams.

STEP 2 Develop Fault Tree

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

Step 2.2 Use fault tree modules to develop fault tree - Fault tree modules served as logic building blocks in the construction of fault trees. In addition, they were used to simplify and standardize fault tree development layout. Modules were defined for the system level, the node level, the segment level, the component level, and

the component interface level (actuation, electrical, etc.). The system level module was used to relate the system success criteria to the fault logic. The node level modules served as input into the system level module and were applied to totally define the fault logic associated with the segments. Once the node level logic was developed and constructed, the next step was to establish the fault logic associated with each individual segment. This was accomplished using segment level modules which related components to the segment. Finally, component level modules were used to further define fault contributions related to failure mode elements of each component identified in the segment level module. They related to hardware failures, test and maintenance outages, operator error, actuation system failure, and support system interfaces (e.g., electrical, cooling).

Procedures were used to define the step-by-step process in the development of the fault trees using the fault tree modules. Rules were applied to determine the node level modules to be used based on the system success criteria and flow requirements. The fault tree was developed graphically with the Westinghouse GRAFTER Code System (Reference 10).

STEP 3 Quantify Fault Tree

The fault trees were quantified using the GRAFTER Code System (Reference 10) to determine an initial system failure probability and to obtain the minimal cutsets. The calculational methods for quantifying the basic event probabilities that were input into the fault tree quantification were specified in the procedures mentioned above. Calculational methods were described for hardware failures (both demand and time dependent), maintenance unavailabilities, test unavailabilities, human errors and common cause failures. A discussion of system mission times was provided and a component identification format was provided to maintain consistency within the analyses.

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

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

Step 3.3 Calculate common cause failure probabilities - Once a fault tree for a system was developed, the important common cause component groups were identified for inclusion in the fault trees. The common cause attributes that were used for the identification of common cause failures were:

- Component Type
- Component Use/Function (system isolation, flow modulation, etc.)
- Component initial conditions (i.e., normally closed, initially running, etc.)
- Component failure mode

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

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

STEP 4 Document Process

The entire process of fault tree development including key assumptions, boundary conditions, and other important information was documented in the fault tree section of the system notebook. The quantification of the fault tree was documented in the system notebook in the quantification section. The dominant contributors to system failure and key insights were identified and documented in the system notebook.

3.2.3 System Dependencies

Table 3.2-3 identifies the front-line systems or functions that are dependent on the support systems and Table 3.2-4 identifies which support systems are dependent upon each other. The support systems modeled include AC power, DC power, component cooling water, essential service water, nonessential service water, control air and initiation signals.

TABLE 3.2-3

FRONTLINE SYSTEM DEPENDENCY ON SUPPORT SYSTEMS

Front-Line System/ Support System	4160 VAC	600 VAC	250 VDC	120 VAC	Control Air	Signals	CCW	ESW	NESW	DGs
Auxiliary Feedwater	X	X	X	X	X	X		X		
High Pressure Injection	X	X	X			X	X			
High Pressure Recirculation	X	X	X			X	X			
Low Pressure Injection	X		X			X				
Low Pressure Recirculation	X	X	X			X	X			
Containment Spray Injection	X	X	X			X				
Containment Spray Recirculation	X	X	X			X		X		
Main Steam			X			X				
SG PORVs				X	X					
Hydrogen Igniters				X						
Pzr PORVs & Safeties		X	X	X	X					
Containment Fans		X				X	X			
Main Feed		X			X	X				
Condensate		X	X		X					

TABLE 3.2-4

SUPPORT SYSTEM DEPENDENCY ON SUPPORT SYSTEMS

Support System/ Support System	4160 VAC	600 VAC	250 VDC	120 VAC	Control Air	Signals	CCW	ESW	NESW	DGs
4160 VAC	-		X							X
600 VAC	X	-	X							
250 VDC		X	-							
120 VAC		X	X	-						
Control Air		X	X		-				X	
Signals				X		-				
CCW	X	X	X			X	-	X		
ESW	X	X	X		X	X		-		
NESW		X	X		X				-	
DGs		X	X			X		X		-

3.3 Sequence Quantification

3.3.1 List of Generic Data

3.3.1.1 Component Hardware Data

Generic data formed the basis for many of the component failure rates used in the Cook Nuclear Plant PRA. If a shortage of plant-specific data existed, generic values were utilized as either the actual failure rates or as the prior distributions for Bayesian updates. Generic failure rates were used for the majority of electrical components modeled in the Cook Nuclear Plant PRA. Table 3.3-1 lists all of the generic component failure rates. References 11, 24, 25, 29 and 36 were used for generic data. Reference 29 was, for the most part, the base for generic data used in the Cook Nuclear Plant PRA, including fans used for equipment cooling. The other sources listed above were used when Reference 29 did not contain data for component failure modes of interest.

3.3.1.2 Initiating Event Data

Initiating events data used in the Cook Nuclear Plant IPE was taken from both plant specific and generic sources. Generic data was used to calculate the initiating event frequency of those events not expected to occur during the life of the plant. For LOCA events, Reference 36 was used as the source of pipe break frequencies. Calculation of valve failure frequencies leading to LOCAs utilized the component failure data discussed above. The frequencies of steam generator tube ruptures and reactor coolant pump seal LOCAs were determined using proprietary data provided by Westinghouse Electric Corporation in References 68 and 69, respectively.

3.3.2 Plant-Specific Data and Analysis

3.3.2.1 Component Hardware Data

Plant specific data was collected very early in the PRA project. A list of components for which data was to be collected was generated based on the results of industry PRAs and on the opinions of utility and contractor personnel. This list was used to focus the data collection effort. Most of the mechanical and testing data used in the Cook Nuclear Plant PRA utilized plant-specific data to calculate failure rates through classical means or through the use of Bayesian techniques. Table 3.3-1 lists all of the plant-specific component failure rates and their respective calculational techniques. Those failure rates calculated using Bayesian techniques used generic prior failure rates from the sources identified in Section 3.3.1.

3.3.2.2 Initiating Event Data

An analysis of all unit trips since the start of commercial operation was completed and provided the frequency of transient events for the Cook Nuclear Plant.

For special initiating events such as loss of CCW, the frequency of occurrence was calculated using fault tree analysis techniques and component failure data from the master data file.

TABLE 3.3-1

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
026	Centrifugal Charging Pump	/D	3.0E-03	5.48E-05	0	750	6.66E-04	4.88E-07	Bayesian
027	Centrifugal Charging Pump	/Hr	3.0E-05	5.48E-09	1	75272	1.33E-5	9.94E-11	Plant
028	Motor Driven AFW Pump	/D	3.0E-03	5.48E-05	0	179	1.32E-03	2.92E-06	Bayesian
029	Motor Driven AFW Pump	/Hr	3.0E-05	5.48E-09	0	88.6	3.0E-05	5.48E-09	Generic
*030	Turbine Driven AFW Pump	/D	3.0E-02	5.48E-03	*	*	4.40E-02	1.09E-03	*See Note
031	Turbine Driven AFW Pump	/Hr	5.0E-03	1.52E-04	0	52.1	2.86E-03	1.79E-05	Bayesian
032	Containment Spray Pump	/D	3.0E-03	5.48E-05	0	250	1.15E-03	1.99E-06	Bayesian
033	Containment Spray Pump	/Hr	3.0E-05	5.48E-09	0	187.2	3.0E-05	5.48E-09	Generic
034	Essential Service Water Pump	/D	3.0E-03	5.48E-05	0	412	9.09E-04	1.08E-06	Bayesian
035	Essential Service Water Pump	/Hr	3.0E-05	5.48E-09	1	75248	1.33E-05	9.94E-11	Plant
036	Safety Injection Pump	/D	3.0E-03	5.48E-05	1	238	2.98E-03	7.71E-06	Bayesian
037	Safety Injection Pump	/Hr	3.0E-05	5.48E-09	2	73.4	3.0E-05	5.48E-09	Generic
038	Non-Essential Service Water Pump	/D	3.0E-03	5.48E-05	1	35	8.85E-03	1.16E-04	Bayesian
039	Non-Essential Service Water Pump	/Hr	3.0E-05	5.48E-09	2	225738	8.86E-06	4.41E-11	Plant

*Note: Calculation includes Bayesian updated failure rates of trip/throttle valve and pump/turbine

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
040	Component Cooling Water Pump	/D	3.0E-03	5.48E-05	0	372	9.55E-04	1.23E-06	Bayesian
041	Component Cooling Water Pump	/Hr	3.0E-05	5.48E-09	0	75236	6.65E-06	4.87E-11	Bayesian
042	Residual Heat Removal Pump	/D	3.0E-03	5.48E-05	0	275	1.10E-03	1.77E-06	Bayesian
043	Residual Heat Removal Pump	/Hr	3.0E-05	5.48E-09	1	123.4	3.0E-05	5.48E-09	Generic
044	Circulating Water Pump	/D	3.0E-03	5.48E-05	1	--	3.0E-03	5.48E-05	Generic
045	Circulating Water Pump	/Hr	3.0E-05	5.48E-09	1	260937	3.83E-06	8.26E-12	Plant
046	Feedwater Pump	/D	3.0E-02	5.48E-03	0	--	3.0E-02	5.48E-03	Generic
047	Feedwater Pump	/Hr	5.0E-03	1.52E-04	0	150492	1.82E-03	7.41E-11	Bayesian
048	Hotwell Pump	/D	3.0E-03	5.48E-05	0	--	3.0E-03	5.48E-05	Generic
049	Hotwell Pump	/Hr	3.0E-05	5.48E-09	2	150492	1.33E-05	9.94E-11	Plant
050	Condensate Booster Pump	/D	3.0E-03	5.48E-05	0	--	3.0E-03	5.48E-05	Generic
051	Condensate Booster Pump	/Hr	3.0E-05	5.48E-09	0	150492	4.44E-06	1.82E-11	Bayesian
067	Diesel Driven Fire Pump	/D	3.0E-02	5.06E-04	--	--	3.0E-02	5.06E-04	Generic
068	Diesel Driven Fire Pump	/Hr	8.0E-04	3.90E-06	--	--	8.0E-04	3.90E-06	Generic
069	Air Compressor (Fails to Start)	/D	8.0E-02	3.6E-03	--	--	8.0E-02	3.6E-03	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
070	Air Compressor (Fails to Run)	/Hr	2.0E-04	2.44E-07	--	--	2.0E-04	2.44E-07	Generic
076	Motor Operated Valve (Fail To Actuate)	/D	3.0E-03	5.48E-05	19	12559	1.51E-03	1.28E-06	Plant
077	Motor Operated Valve (Failure to Remain Closed)	/Hr	5.0E-07	1.52E-12	--	--	5.0E-07	1.52E-12	Generic
078	Motor Operated Valve (Fail to Remain Open)	/Hr	1.0E-07	5.62E-15	--	--	1.0E-07	5.62E-15	Generic
079	Solenoid Operated Valve (Failure To Operate)	/D	2.0E-03	2.25E-06	--	--	2.0E-03	2.25E-06	Generic
080	Hydraulic Valve	/D	2.0E-03	2.25E-06	--	--	2.0E-03	2.25E-06	Generic
081	Air Operated Valve	/D	2.0E-03	2.25E-06	0	2246	7.89E-04	1.68E-07	Bayesian
082	Check Valve (Fails to Open)	/D	1.0E-04	5.62E-09	--	--	1.0E-04	5.62E-09	Generic
083	Check Valve (Fails to Close)	/D	1.0E-03	5.62E-07	--	--	1.0E-03	5.62E-07	Generic
084	Power Operated Relief Valve (PORV) (Fail to Open)	/D	2.0E-03	2.25E-06	0	48	2.0E-03	2.25E-06	Generic
085	Power Operated Relief Valve (Fail to Reclose Once Open)	/D	2.0E-03	2.25E-06	--	--	2.0E-03	2.25E-06	Generic
086	Pressure Regulating Valve (Fails to Open)	/D	2.0E-03	2.25E-06	--	--	2.0E-03	2.25E-06	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
087	Relief Valve (Fails to Open)	/D	3.0E-04	5.48E-07	--	--	3.0E-04	5.48E-07	Generic
088	Main Steam Isolation Valve (MSIV)	/D	2.0E-03	2.95E-06	1	216	2.41E-03	2.50E-06	Bayesian
089	Relief Valve (Fails to Reclose)	/D	2.0E-02	2.25E-04	--	--	2.0E-02	2.25E-04	Generic
090	Air Operated Valve (Spurious Closure)	/Hr	1.0E-07	5.62E-15	--	--	1.0E-07	5.62E-15	Generic
091	Air Operated Valve (Spurious Open)	/Hr	5.0E-07	1.52E-12	--	--	5.0E-07	1.52E-12	Generic
092	Manual Valve (Fails to Open)	/D	1.0E-04	5.62E-09	--	--	1.0E-04	5.62E-09	Generic
093	Check Valve (Disc Rupture)	/Hr	1.0E-07	5.62E-15	--	--	1.0E-07	5.62E-15	Generic
094	Motor Operated Valve (Disc Rupture)	/Hr	1.0E-07	5.62E-15	--	--	1.0E-07	5.62E-15	Generic
095	Relief Valve (Fails to Open)	/D	3.0E-04	5.48E-07	--	--	3.0E-04	5.48E-07	Generic
096	Relief Valve (Fails to Reclose)	/D	2.0E-02	2.25E-04	--	--	2.0E-02	2.25E-04	Generic
097	Stop Check Valve (Fails to Open)	/D	1.0E-04	5.62E-09	--	--	1.0E-04	5.62E-09	Generic
098	Safety Valve (Opens Prematurely)	/Hr	3.0E-06	5.48E-11	--	--	3.00E-06	5.48E-11	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
099	SCRAM System (Failure to SCRAM)	/D	3.0E-05	5.06E-10	--	--	3.0E-05	5.06E-10	Generic
100	Pressurizer Safety Valve (Prematurely Opens)	/Hr	3.0E-06	5.06E-12	1	225738	3.23E-06	4.02E-12	Bayesian
126	Fan (Fails to Run)	/Hr	1.0E-05	5.62E-11	--	--	1.0E-05	5.62E-11	Generic
127	Fan (Fails to Start)	/D	3.0E-04	5.06E-08	--	--	3.0E-04	5.06E-08	Generic
128	Fan Corrective Maintenance	/D	2.0E-03	2.44E-05	--	--	2.0E-03	2.44E-05	Generic
152	Orifice (Plugged)	/D	3.00E-04	5.06E-08	--	--	3.00E-04	5.06E-08	Generic
153	Orifice (Plugged)	/Hr	6.00E-07	2.02E-13	--	--	6.00E-07	2.02E-13	Generic
154	Orifice (Rupture)	/Hr	3.0E-08	5.06E-16	--	--	3.0E-08	5.06E-16	Generic
155	Damper (Fail to Open)	/D	3.0E-03	5.48E-05	--	--	3.0E-03	5.48E-05	Generic
156	Damper (Fail to Operate)	/Hr	1.0E-06	6.09E-12	--	--	1.0E-06	6.09E-12	Generic
157	Air Cooler (Fail to Operate)	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
158	Heat Exchanger (Rupture)	/Hr	3.0E-06	5.48E-11	--	--	3.0E-06	5.48E-11	Generic
159	Heat Exchanger (Blockage)	/Hr	5.7E-06	1.98E-10	--	--	5.7E-06	1.98E-10	Generic
160	Strainer/Filter (Plugged)	/Hr	3.0E-05	5.48E-09	--	--	3.0E-05	5.48E-09	Generic
161	Pipe Plug (Boron Precipitation)	/Hr	7.7E-07	3.33E-13	--	--	7.7E-07	3.33E-13	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
162	Tank Plug (Boron Precipitation)	/Hr	7.7E-07	3.33E-13	--	--	7.7E-07	3.33E-13	Generic
163	Tank (Rupture)	/Hr	8.48E-10	5.10E-17	--	--	8.48E-10	5.10E-17	Generic

Lines 251-376 represent unavailabilities, not failure rates. The unavailability is calculated from the product of the failure rate and the average duration.

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
251	OHP 4030.STP.002V	/D					5.03E-03	1.42E-05	Plant
253	OHP 4030.STP.007E	/D					8.33E-03	3.90E-05	Plant
255	OHP 4030.STP.007W	/D					7.85E-03	3.47E-05	Plant
257	OHP 4030.STP.011	/D					3.39E-03	6.45E-06	Plant
265	OHP 4030.STP.017E	/D					6.10E-03	2.09E-05	Plant
269	OHP 4030.STP.017T	/D					4.98E-03	1.39E-05	Plant
271	OHP 4030.STP.017W	/D					3.32E-03	6.21E-06	Plant
273	OHP 4030.STP.019F	/D					9.35E-04	4.92E-07	Plant
275	OHP 4030.STP.019P	/D					3.91E-04	8.60E-08	Plant
277	OHP 4030.STP.020E	/D					2.68E-03	4.04E-06	Plant
279	OHP 4030.STP.020W	/D					9.41E-03	4.98E-05	Plant
281	OHP 4030.STP.022E	/D					5.28E-03	1.57E-05	Plant
283	OHP 4030.STP.022W	/D					1.69E-02	1.60E-04	Plant
285	THP 4030.STP.209	/D					7.48E-04	3.14E-07	Plant

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
289	OHP 4030.STP.050E	/D					4.00E-03	8.97E-06	Plant
291	OHP 4030.STP.050W	/D					1.45E-03	1.19E-06	Plant
293	OHP 4030.STP.051N	/D					2.36E-03	3.14E-06	Plant
295	OHP 4030.STP.051S	/D					2.70E-03	4.09E-06	Plant
297	OHP 4030.STP.052E	/D					3.31E-03	6.15E-06	Plant
299	OHP 4030.STP.052W	/D					3.14E-03	5.56E-06	Plant
301	OHP 4030.STP.053A	/D					7.02E-03	2.77E-05	Plant
303	OHP 4030.STP.053B		No Duration Presently Available				0.0		Plant
304	OHP 4030.STP.018	/D					1.20E-03	8.09E-07	Plant
305	PP-1 Corrective Maintenance	/D	No Duration Available				0.0	--	--
307	PP-1 Preventive Maintenance	/D					4.57E-03	1.17E-05	Plant
309	PP-2 Corrective Maintenance	/D					4.02E-04	9.08E-08	Plant
310	PP-2 Preventive Maintenance	/D					2.01E-04	2.27E-08	Plant
311	PP-3 Preventive Maintenance	/D					4.12E-04	9.54E-08	Plant
313	PP-3 Corrective Maintenance	/D	No Duration Available				0.0	--	--

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
315	PP-4 Preventive Maintenance	/D					1.53E-03	1.32E-06	Plant
317	PP-4 Corrective Maintenance	/D					1.15E-03	7.43E-07	Plant
319	PP-5 Corrective Maintenance	/D					7.67E-04	3.31E-07	Plant
321	PP-5 Preventive Maintenance	/D					3.04E-04	5.19E-07	Plant
323	PP-6 Corrective Maintenance	/D	No Duration Available				0.0	--	--
325	PP-6 Preventive Maintenance	/D					1.75E-02	1.72E-04	Plant
327	PP-7 Corrective Maintenance	/D	No Duration Available				0.0	--	--
329	PP-7 Preventive Maintenance	/D					1.48E-03	1.23E-06	Plant
331	PP-8 Corrective Maintenance	/D					1.98E-03	2.20E-06	Plant
332	PP-8 Preventive Maintenance	/D					2.91E-02	4.76E-04	Plant
333	PP-9 Corrective Maintenance	/D	No Duration Available				0.00E+00	--	--
334	PP-9 Preventive Maintenance	/D					3.12E-05	5.47E-10	Plant

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
335	PP-10 Preventive Maintenance	/D					3.03E-03	5.16E-06	Plant
337	PP-10 Corrective Maintenance	/D	No Duration Available				0.0	--	--
339	PP-26 Preventive Maintenance	/D					3.52E-04	6.96E-08	Plant
341	PP-26 Corrective Maintenance	/D	No Duration Available				0.0	--	--
343	PP-35 Preventive Maintenance	/D					7.39E-04	3.07E-07	Plant
345	PP-35 Corrective Maintenance	/D	No Duration Available				0.0	--	--
347	PP-50 Corrective Maintenance	/D	No Duration Available				0.0	--	--
349	PP-50 Preventive Maintenance	/D					2.85E-03	4.57E-06	Plant
350	Air Compressor Corrective Maintenance	/D	2.0E-03	2.44E-05	--	--	2.0E-03	2.44E-05	Generic
351	Air Compressor Preventive Maintenance	/D	No Data Available				0.0	--	--
359	FW Pump Turbine Condenser Preventive Maintenance	/D					2.84E-04	4.53E-08	Plant
365	PORV Corrective Maintenance	/D					8.75E-02	4.30E-03	Plant

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
366	PORV Preventive Maintenance	/D	No Duration Available				0.00E+00	--	--
369	SG PORV Preventive Maintenance	/D					7.44E-03	3.11E-05	Plant
371	Air Operated Valves Corrective Maintenance	/D	No Duration Available				0.0	--	--
372	Air Operated Valves Preventive Maintenance	/D					3.72E-06	7.78E-12	Plant
373	MOV Corrective Maintenance	/D					2.22E-05	2.77E-10	Plant
374	MOV Preventive Maintenance	/D					4.95E-05	1.38E-09	Plant
375	Aux. Building Exhaust Fan Corrective Maintenance	/D	No Duration Available				0.0	--	--
376	Aux. Building Exhaust Fan Preventive Maintenance	/D					1.66E-03	1.55E-06	Plant
400	Diesel Generator	/D	3.0E-02	5.06E-04	4	537	7.45E-03	3.12E-05	Plant
401	Diesel Generator	/Hr	2.0E-03	2.42E-05	1	1107.1	9.03E-04	4.58E-07	Plant
402	Hydrogen Recombiner	/D			0	51	9.80E-03	5.40E-05	Plant
403	Hydrogen Recombiner	/Hr			0	19.5	2.56E-02	3.68E-04	Plant
405	Manual Switch (Fails to Operate)	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
406	Circuit Breaker (Spurious Open)	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
407	Circuit Breaker (Fails to Transfer)	/D	3.0E-03	5.48E-05	--	--	3.0E-03	5.48E-05	Generic
408	Fuse (Opens Spuriously)	/Hr	3.0E-06	5.06E-12	--	--	3.0E-06	5.06E-12	Generic
409	Bus (Fails All Modes)	/Hr	1.0E-07	1.60E-14	--	--	1.00E-07	1.60E-14	Generic
410	Current Transformer	/Hr	6.0E-07	2.02E-13	--	--	6.0E-07	2.02E-13	Generic
411	Potential Transformer	/Hr	6.0E-07	2.02E-13	--	--	6.0E-07	2.02E-13	Generic
412	Power Transformer	/Hr	6.0E-07	2.02E-13	--	--	6.0E-07	2.02E-13	Generic
413	Process Transformer	/Hr	6.0E-07	2.02E-13	--	--	6.0E-07	2.02E-13	Generic
414	Relay Contacts (Fail to Transfer)	/Hr	1.00E-06	5.62E-13	--	--	1.00E-06	5.62E-13	Generic
415	Relay Coil (Open or Short)	/Hr	3.0E-06	5.06E-12	--	--	3.00E-06	5.06E-12	Generic
416	Time Delay Relay (Premature Transfer)	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
417	Bimetallic Time Delay Relay	/Hr	1.0E-05	5.62E-11	--	--	1.0E-05	5.62E-11	Generic
418	Battery	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
419	Battery Charger	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
420	Battery Charger	/D	3.0E-04	5.48E-07	--	--	3.0E-04	5.48E-07	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
421	DC Motor Generator	/Hr	3.0E-06	5.48E-11	--	--	3.0E-06	5.48E-11	Generic
422	Invertor	/Hr	1.0E-04	5.62E-09	--	--	1.0E-04	5.62E-09	Generic
423	Wiring Fails (Open Circuit)	/Hr	1.0E-05	5.62E-11	--	--	1.0E-05	5.62E-11	Generic
424	Wiring Fails (Short to Ground)	/Hr	1.0E-06	5.62E-13	--	--	1.0E-06	5.62E-13	Generic
425	Wiring Fails (Short to Power)	/Hr	3.0E-08	5.06E-16	--	--	3.0E-08	5.06E-16	Generic
426	Solid State Device (High Power)	/Hr	3.0E-06	5.48E-11	--	--	3.0E-06	5.48E-11	Generic
427	Solid State Device (Low Power)	/Hr	3.0E-06	5.48E-11	--	--	3.0E-06	5.48E-11	Generic
428	Solid State Device (Bistable)	/Hr	3.0E-07	5.48E-13	--	--	3.0E-07	5.48E-13	Generic
429	Terminal Board (Open Circuit)	/Hr	3.0E-07	5.48E-13	--	--	3.0E-07	5.48E-13	Generic
430	Terminal Board (Short to Adjacent Circuit)	/Hr	3.0E-07	5.48E-13	--	--	3.0E-07	5.48E-13	Generic
431	Torque Switch (Fail to Operate)	/Hr	2.0E-07	2.25E-14	--	--	2.0E-07	2.25E-14	Generic
432	Limit Switch (Fail to Operate)	/Hr	6.0E-06	2.02E-11	--	--	6.0E-06	2.02E-11	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
433	Pressure Switch (Fail to Operate)	/Hr	2.0E-07	2.25E-14	--	--	2.0E-07	2.25E-11	Generic
434	Flow Switch (Fails to Operate)	/D	4.0E-08	8.99E-16	--	--	4.0E-08	8.99E-16	Generic
435	Flow Switch (Fails to Operate)	/Hr	3.67E-06	7.57E-11	--	--	3.67E-06	7.57E-11	Generic
436	Relay (Fails to Open)	/D	2.09E-06	2.46E-12	--	--	2.09E-06	2.46E-12	Generic
437	Relay (Fails to Close)	/D	2.00E-06	2.25E-12	--	--	2.00E-06	2.25E-12	Generic
438	Limit Switch (Fails to Open)	/D	1.00E-04	5.62E-09	--	--	1.00E-04	5.62E-09	Generic

Lines 451-487 represent unavailabilities, not failure rates. The unavailability is calculated from the product of the failure rate and the average duration.

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
439	Limit Switch (Fails to Close)	/D	1.00E-04	5.62E-09	--	--	1.00E-04	5.62E-09	Generic
451	Hydrogen Recombiner Test	/D					7.91E-05	3.51E-09	Plant
453	Diesel Generator Corrective Maintenance	/D					2.26E-04	2.87E-08	Plant
454	Diesel Generator Preventive Maintenance	/D					4.37E-04	1.07E-07	Plant
459	OHP 4030.STP.013A	/D					3.35E-04	6.30E-06	Plant
461	OHP 4030.STP.013B	/D					2.69E-04	4.06E-08	Plant
485	OHP 4030.STP.027AB	/D					6.01E-03	2.03E-05	Plant

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
487	OHP 4030.STP.027CD	/D					5.33E-03	1.60E-05	Plant
654	Generic HEP #8	/D	1.30E-02	8.8E-05	--	--	1.30E-02	8.8E-05	Generic
655	Generic HEP #9	/D	1.30E-02	1.1E-03	--	--	1.30E-02	1.1E-03	Generic
656	Generic HEP #16	/D	3.80E-03	7.9E-06	--	--	3.80E-03	7.9E-06	Generic
657	Generic HEP #17	/D	1.30E-02	8.8E-05	--	--	1.30E-02	8.8E-05	Generic
658	Generic HEP #18	/D	8.10E-02	1.0E-02	--	--	8.10E-02	1.0E-02	Generic
659	Generic HEP #19	/D	1.60E-02	4.2E-04	--	--	1.60E-02	4.2E-04	Generic
660	Generic HEP #20	/D	3.80E-03	7.9E-06	--	--	3.80E-03	7.9E-06	Generic
661	Generic HEP #21	/D	1.30E-03	8.8E-07	--	--	1.30E-03	8.8E-07	Generic
662	Generic HEP #22	/D	1.30E-03	1.1E-05	--	--	1.30E-03	1.1E-05	Generic
663	Generic HEP #23	/D	1.30E-03	1.1E-05	--	--	1.30E-03	1.1E-05	Generic
664	Generic HEP #27	/D	2.70E-03	4.3E-05	--	--	2.70E-03	4.3E-05	Generic
665	Generic HEP #39	/D	2.70E-04	4.3E-07	--	--	2.70E-04	4.3E-07	Generic
666	Generic HEP #40	/D	1.60E-03	1.60E-05	--	--	1.60E-03	1.6E-05	Generic
667	Generic HEP #51	/D	3.75E-03	7.90E-06	--	--	3.75E-03	7.90E-06	Generic

TABLE 3.3-1 (Cont'd.)

DATA PROCESSING TABLE

#	DESCRIPTION	UNITS	GEN. PROB	GEN. VAR	N[1]	M[2]	CALC. PROB	CALC. VAR	COMMENTS[3]
668	Generic HEP #52	/D	7.50E-03	3.16E-05	--	--	7.50E-03	3.16E-05	Generic
671	Generic HEP #26	/D	2.70E-04	4.3E-07	--	--	2.70E-04	4.3E-07	Generic

[1] N = Number of failures. In some cases a dash (--) appears which means no data was collected for this entry.

[2] M = Number of demands/hours of operation. In some cases a dash (--) appears which means no data was collected for this entry.

[3] Bayesian = Bayesian Update, Plant = Plant Specific

3.3.3 Human Failure Data

3.3.3.1 General Approach

The general methodology used in quantifying human error probabilities (HEPs) was a step-by-step task analysis.. Each general operator action was broken down into a series of subtasks (discrete physical or mental steps). In addition, specific activities which could serve to disable the system or which needed to be performed during the accident were also explicitly modeled. These subtasks were then quantified by assigning conditional HEPs associated with the specific subtask. The conditional HEPs were assigned using Technique for Human Error Rate Prediction (THERP) methodology (from Reference 22) to analyze individual tasks defined in task analysis. The HEP associated with the general operator action was quantified using engineering calculations or fault tree models (provides same calculational results as THERP trees), as appropriate.

The overall steps in performing this evaluation for the quantification of the Human Error Probabilities associated with the Cook Nuclear Plant PRA were as follows:

3.3.3.1.1 Define Operator Actions and Other Human Interactions

<u>Define Operator Actions in Event Trees</u>

For operator actions appearing in event trees, the following information for each operator action was obtained:

a. Action Identifier;
b. Description of Action;
c. Time Window Available for Action;
d. Applicable Procedures;
e. Indication of Whether the Action Was Simulated in Training.

Note that at this point, the analyst established the definition, scope and success criteria for the operator actions modeled in the event trees; no human reliability modeling was performed yet.

This task was analogous to the task of defining the success criteria, mission time, and system boundary of other event tree nodes to prepare for the fault tree analysis.

<u>Define "OTHER" Human Interactions</u>

For other human errors needed in fault tree analysis, either the fault tree analyst provided his/her own calculation for a HEP, using data from the master HEP data bank supplied by the Human Reliability Analysis (HRA) analyst, or the fault tree analyst requested the HRA analysts to calculate and document such an HEP.

3.3.3.1.2 Model Each Operator Action/Human Interaction

In this task, the human actions defined in step 3.3.3.1.1 were broken down into smaller subtasks for which potential human errors and associated HEPs could be assigned. (Table 3.3-2 lists generic HEPs obtained from Reference 22 which were used as a basis for calculation of human error.) Key subtasks were identified and retained, others were eliminated. Plant specific considerations (e.g., operator training, existence of procedures, operator stress levels, etc.) were then determined and, collectively, were used to

calculate a Performance Shaping Factor (PSF). The PSF was used as a multiplier on a generic HEP to calculate a plant specific value. Guidelines used in calculating PSFs are identified in Reference 22 and in Table 3.3-3.

Each model is either:

a. a simple algebraic sum/product model; or
b. a fault tree.

3.3.3.1.3 Visit Plant and Interview Operators

As part of the Quantification of Human Error Probabilities, a plant visit to interview the operators and other plant personnel about the human actions that are modeled in step 3.3.3.1.2 was made. The objectives of this task were as follows:

a. Verify the models and assumptions through discussions with the plant personnel.

b. Assess the additional PSFs that may modify the nominal HEPs in the quantification task.

c. Identify those actions that may be subject to a screening process without further quantification.

3.3.3.1.4 Quantify HEPS for each Operator Action/Human Interaction

In this step, the analyst performed a detailed quantification of the HEPs for the operator actions, using a master data bank obtained from Reference 22 and applying the PSFs. If not already identified, the PSFs were determined at this point. For convenience, tables from the master data bank are provided in Table 3.3-2.

Detailed HEP Quantification

For this task, the operator actions were quantified by using the rules of THERP methodology utilizing the following steps:

a. Generation of the Human Error Probability Data Bank

   This task involved the generating of a master data bank for operator actions. This data bank contains HEPs obtained from applicable references, such as Reference 22. The HEPs are to be mean values and a variance may also be provided. To convert the median values in the data bank of Reference 22 to mean values, a lognormal distribution was assumed using the error factor times the median as the 95th percentile.

   Portions of this data bank were also made available to analysts for calculation of HEPs that may appear as simple basic events in fault trees.

b. Identification of Operator Action Subtasks

c. Estimation of Performance Shaping Factors (PSF)

   This task included the following for the generation of a combined PSF for each specific subtask:

   - Consideration of each subtask separately while being sensitive to its place in the overall scheme of the operator action.

- Identification and consideration of clues that may help the operator carry out the subtask (including warning lights and alarms).

- Identification and consideration of conditions that may impede the operator in carrying out the task.

- Identification and consideration of dependencies between tasks. (Table 3.3-4 lists the equations used to calculate probabilities conditional on success or failure of previous task.)

d. Quantification of HEPs Defined in PRA

This task involved the cumulative quantification of the HEPs for each human action, using the data bank, the models generated for the operator actions, the performance shaping factors identified, and the THERP method.

3.3.3.2 General Assumptions

It was generally assumed that there were no dependencies between subtasks within a particular operator action. However, if a strong dependency between tasks clearly existed, such cases were accounted for individually. A detailed review of each operator action was performed to ensure that dependencies were appropriately addressed.

For many operator actions, step-by-step procedures are available to guide the operator during an accident. Credit for the availability of these procedures was provided in the selection of appropriate HEPs from the HEP Data Bank. For other operator actions, no detailed procedures are available, and only general direction is provided to the operator. In such cases, the success of the operator action depends largely on the training and memory of the operator, and the calculated HEP may be unreasonably high. Interviews with the operators were conducted to determine if the operators had the knowledge and training to deal with these situations. Estimates of HEPs were made based on interviews with operators and reflect the availability of written procedures.

Significant improvements have occurred within the past decade pertaining to emergency response procedures. The procedures in place today for most commercial nuclear power plants (including the Cook Nuclear Plant) are more symptomatic in nature. Unless specific diagnosis difficulties were noted, it was assumed that the EOPs currently in place for the Cook Nuclear Plant were adequate to address the transient symptoms and ensure that the operator provides the correct functional response.

Due to the emphasis placed on following procedures at the Cook Nuclear Plant, it was assumed that the failure probability for the operator to fail to enter the reactor trip procedure following transient initiation is statistically insignificant.

3.3.3.3 Results

The HEPs used in the system fault trees were generally calculated using hand calculations. The HEPs used in event trees were generally calculated using individual fault trees. Table 3.3-5 summarizes the operator errors analyzed, the fault trees utilized, and the resultant human error probabilities that were calculated.

TABLE 3.3-2

HEP DATA

NO.	HEP DESCRIPTION	HEP(MEAN)	VARIANCE
	DIAGNOSIS BY CONTROL ROOM PERSONNEL OF AN ABNORMAL EVENT		
1	Within 10 minutes of a compelling signal	2.7E-01	4.3E-01
2	Within 20 minutes of a compelling signal	2.7E-02	4.3E-03
3	Within 30 minutes of a compelling signal	2.7E-03	4.3E-05
4	Within 60 minutes of a compelling signal	8.5E-04	5.1E-05
5	Within 1500 minutes of a compelling signal	8.5E-05	5.1E-07
	FAILURE OF ADMINISTRATIVE CONTROL		
6	Carry out a plant policy or scheduled tasks such as periodic tests or maintenance performed weekly, monthly, or at long intervals	1.6E-02	4.2E-04
7	Initiate a scheduled shiftly checking or inspection function	1.3E-03	8.8E-07
	Use written operations procedures under:		
8	normal operating conditions	1.3E-02	8.8E-05
9	abnormal operating conditions	1.3E-02	1.1E-03
10	Use a valve change or restoration list	1.3E-02	8.8E-05
11	Use written test or calibration procedures	8.1E-02	1.0E-02
12*	Use written maintenance procedures	3.9E-01	1.1E-01
13*	Use a checklist properly	5.5E-01	5.8E-02
	ERRORS OF OMISSION		
	When procedures with checkoff provisions are correctly used (HEP per item of instruction):		
14	Short List, < 10 items	1.3E-03	8.8E-07
15	Long List, > 10 items	3.8E-03	7.9E-06

TABLE 3.3-2 (Cont'd.)

HEP DATA

NO.	HEP DESCRIPTION	HEP(MEAN)	VARIANCE
	When procedures without checkoff provisions are used, or when checkoff provisions are incorrectly used (HEP per item of instruction):		
16	Short list, < 10 items	3.8E-03	7.9E-06
17	Long list, > 10 items	1.3E-02	8.8E-05
18	When written procedures are available and should be used but are not used	8.1E-02	1.0E-02
19	Same as item 18, but is judged to be a "second nature"	1.6E-02	4.2E-04
ERRORS OF COMMISSION			
	Select wrong control from an array of similar-appearing controls		
20	identified by labels only	3.8E-03	7.9E-06
21	from a functionally grouped set of controls	1.3E-03	8.8E-07
22	from a panel with clearly drawn mimic lines	1.3E-03	1.1E-05
	Turn control in wrong direction:		
23	when there is no violation of populational stereotypes	1.3E-03	1.1E-05
24	when design violates a strong populational stereotype and operating conditions are normal	8.1E-02	1.0E-02
25	when design violates a strong populational stereotype and operation is under high stress	5.5E-01	5.8E-02
26	Turn a two-position switch in wrong direction: no violation of populational stereotypes	2.7E-04	4.3E-07
27	Set a multiposition selector switch to an incorrect setting (this error is a function of the clarity with which indicator position can be determined: designs of switch knobs and their position indications vary greatly)	2.7E-03	4.3E-05

TABLE 3.3-2 (Cont'd.)

HEP DATA

HEP DESCRIPTION		HEP(MEAN)	VARIANCE
28	Failure to complete change of state of a component if switch must be held until change is completed	3.8E-03	7.9E-06
29	Select wrong circuit breaker in a group of circuit breakers	6.2E-03	2.2E-05
30	Improperly mate a connector (this includes failures to seat connectors completely and failure to test locking features of connectors for engagement)	3.8E-03	7.9E-06
CHECKER FAILS TO DETECT ERROR			
31	Checking routine tasks, checker using written materials (includes over-the-shoulder inspections, verifying position of locally operated valves, switches, circuit breakers, connectors, etc., and checking written lists, tags, or procedures for accuracy)	1.6E-01	4.2E-02
32	Same as above but without written materials	3.2E-01	1.7E-01
33	Special short-term, one-of-a-kind checking with alerting factors	8.1E-02	1.0E-02
34	Checking that involves active participation, such as special measurements	1.6E-02	4.2E-04
35*	Given that the position of a locally operated valve is checked (item 31 above), checking that it is completely opened or closed after being changed or restored	5.5E-01	5.8E-02
36*	Checking by reader/checker of the task performer in a two-man team, or checking by a second checker, routine task (no credit for more than 2 checkers)	5.5E-01	5.8E-02
37	Checking of status of equipment if that status affects one's safety when performing his tasks	1.6E-03	4.2E-06
38	An operator checks change or restoration tasks performed by a maintainer	Divide HEP in No. 37 by 2	

TABLE 3.3-2 (Cont'd.)

HEP DATA

NO.	HEP DESCRIPTION	HEP(MEAN)	VARIANCE
FAILURE TO RESPOND TO ONE OF MANY ANNUNCIATORS ALARMING CLOSELY IN TIME			
39	1 annunciator alarming	2.7E-04	4.3E-07
40	2 annunciators alarming	1.6E-03	1.6E-05
41	3 annunciators alarming	2.7E-03	4.3E-05
42	4 annunciators alarming	5.3E-03	1.7E-04
43	5 annunciators alarming	8.0E-03	3.9E-04
44	6 annunciators alarming	1.3E-02	1.1E-03
45	7-10 annunciators alarming	1.3E-01	1.1E-01
46	>10 annunciators alarming	3.6E-01	1.3E-01
FAILURE TO RESPOND TO AN ANNUNCIATED LEGEND LIGHT			
47	Resume attention to a legend light within 1 minute after an interruption (sound and blinking canceled before interruption)	1.3E-03	8.8E-07
48	Respond to a legend light if more than 1 minute elapses after an interruption (sound and blinking canceled before interruption)	1.0	
49	Respond to a steady-on legend light during initial audit	1.0	
50	Respond to a steady-on legend light during other hourly scans	1.0	
ERRORS OF COMMISSION IN READING AND RECORDING QUANTITATIVE INFORMATION FROM UNANNUNCIATED DISPLAYS:			
51	Analog Meter	3.8E-03	
52	Chart Recorder	7.5E-03	

Note:
* = Median HEPs are used, instead of mean values.

TABLE 3.3-3

DESCRIPTIVE HRA SCALING GUIDES

(Extracted from NUREG/CR-1278)

Activity	PSF
STRESS LEVEL APPLICATION	
Extremely High (threat stress, e.g., ATWS, Station Blackout, Loss of Indicators during Loss of 250 VDC - early in transient) (conservative w.r.t. recommended value of 5)	10
Typical Transient - Step-by-step, Moderately High Stress (conservative w.r.t. recommended value of 2 - Assumes Experienced Operators during off-normal conditions - Appropriate for diagnostic activities early in the transient)	5
Typical Transient - Longer Time Frame for Response - Step-by-step, Moderately High Stress - Appropriate for activities performed after the initial diagnosis since there would be expected to be only a limited amount of confusion in the control room at the Cook Nuclear Plant	2
Very Low Task Load	2
Optimum Stress Level - for activities performed in a non-transient situation where the operator is very familiar with the activity	1
OTHER CONDITIONS	
General value for response for operators who are well-trained in the appropriate procedures (this also applies to the failure to follow procedures once entered).	0.1
Medium Time Frame for Response	0.1
Absence of a Direct Indicator of Change	2
Availability of Multiple Supportive Indicators	0.1

TABLE 3.3-3 (Cont'd.)

DESCRIPTIVE HRA SCALING GUIDES

Activity	PSF
Failure to Enter Detailed Procedures at Cook Nuclear Plant by Following the Initial Steps of E-0 due to Emphasis Placed on Following Procedures (applies typically to the diagnosis phase of the transient).	0.01
Selection of Wrong Control when a Relatively Long Time Frame is Involved.	1.0
Selection of Wrong Control when Controls are Very Clearly Marked with Mimic Lines	0.1
Special Value for Cases where the Operator is Especially Well-Trained in a Specific Diagnosis - At the Cook Nuclear Plant the operators are especially well-trained on entrance into the EOPs (e.g., Steam Generator Tube Rupture)	0.01
Memorized Procedure	0.1

TABLE 3.3-4

DEPENDENCE LEVEL DEFINITIONS

Equations for conditional probabilities of success and failure on Task "N", given success or failure on previous Task "N-1", for different levels of dependence

Level of Dependence DEP	Success Equations	Failure Equations
1=ZD	$Pr[F_{"N"}S_{"N-1"}ZD]=1$	$Pr[F_{"N"}S_{"N-1"}ZD]=Q$
2=LD	$Pr[F_{"N"}S_{"N-1"}LD]=19Q/20$	$Pr[F_{"N"}S_{"N-1"}LD]=(1+19Q)/20$
3=MD	$Pr[F_{"N"}S_{"N-1"}MD]=6Q/7$	$Pr[F_{"N"}S_{"N-1"}MD]=(1+6Q)/7$
4=HD	$Pr[F_{"N"}S_{"N-1"}HD]=Q/2$	$Pr[F_{"N"}S_{"N-1"}HD]=(1+Q)/2$
5=CD	$Pr[F_{"N"}S_{"N-1"}CD]=0.0$	$Pr[F_{"N"}S_{"N-1"}CD]=1.0$

ZD	=	Zero Dependence (independent)
LD	=	Low Dependence
MD	=	Medium Dependence
HD	=	High Dependence
CD	=	Complete Dependence
$Pr[F_{"N"}S_{"N-1"}ZD]$	=	Probability of failure in subtask N given success in subtask N-1, with a zero dependence between the two subtasks.
Q	=	Is the HEP modified by the PSF, other than dependence.

TABLE 3.3-5

SUMMARY OF HUMAN ERROR PROBABILITIES

Operator Action of Event Tree Node	Time Available	Used in Event Tree	HEP (without hardware failures)	Dependencies/ Notes
Manual Valve Restoration After T&M	N/A	--	2.1E-05	--
Air or Motor-Operated Valve Restoration After T&M	N/A	--	4.2E-07	--
Containment Isolation	N/A	--	6.50E-04	--
Energize H_2 Igniters	N/A	--	1.32E-04	--
Align Battery Charger	N/A	--	3.34E-05	--
Supply Additional Water to AFW	N/A	--	5.20E-05	--
Misposition TDAFWP Fan Switch	N/A	--	3.04E-06	--
Re-open AFW Valves to S/G	N/A	--	5.20E-05	--
Isolate MCM-221	N/A	--	4.23E-06	--
Isolate S/G	N/A	--	5.23E-06	--
Switchover to HP Recirculation	30 min	--	6.31E-06	--
Switchover to CS or LP Recirculation	20 min	--	5.31E-05	--
Align ESW to AFW	45 min	--	6.31E-06	--
Close CBs for Alternate 69kV Bus	N/A	--	6.54E-04	--

TABLE 3.3-5 (Cont'd.)

SUMMARY OF HUMAN ERROR PROBABILITIES

Operator Action of Event Tree Node	Time Available	Used in Event Tree	HEP (without hardware failures)	Dependencies/ Notes
Close Output CB/Switch in Switchyard	N/A	--	6.50E-03	--
Open Output CB K & K1	N/A	--	6.54E-04	--
Strip Emergency Buses	N/A	--	6.51E-03	--
Restore Control Air in LOOP	10 min	--	1.08E-03	--
Place Standby Plant Air Compressor into Service	10 min	--	1.08E-03	--
Assure Containment Drain Operability	N/A	--	3.38E-07	--
Restore CCW Within One Hour	60 min	CCW	1.30E-03	--
Restore CCW Within Eight Hours	8 hrs	CCW	6.62E-03	--
Restore ESW Within One Hour	60 min	ESW	1.30E-02	--
Restore ESW Within Eight Hours	8 hrs	ESW	6.52E-03	--
Trip RCPs During Loss of CCW/ESW	N/A	CCW,ESW	1.30E-02	--
TDAFWP Room Door Left Closed	N/A	--	1.05E-04	--
OLI-Depressurization to Allow Low Pressure Injection	20 min	MLO	1.51E-01	LPI,PORVP, SGPORV,AF4
PBF- Primary Bleed and Feed	30 min	SLO, SLB, LSP, TRA, TRS, SGR, 25A, SBO	3.37E-05	HP2,PORVP

TABLE 3.3-5 (Cont'd.)

SUMMARY OF HUMAN ERROR PROBABILITIES

Operator Action of Event Tree Node	Time Available	Used in Event Tree	HEP (without hardware failures)	Dependencies/ Notes
OA5- Steam Generator Depressurization and Condensate Feed	30 min	TRA	3.06E-05	13COND
OA1- Cooldown and Depressurize RCS and Terminate Safety Injection Before the Ruptured S/G Fills	30 min	SGR	1.05E-04	13PORV
OA2- Cooldown and Depressurize the RCS and Terminate Safety Injection After the Ruptured S/G Fills	60 min	SGR	3.55E-02	13PORV
OA3- Provide Long-Term Cooldown and RCS Depressurization	10-20 hrs	SGR	2.11E-04	LPR,13PORV
XHR- Recovery of AC Power for Station Blackout	1-8 hrs	SBO	3.05E-02	--
RCD- RCS Cooldown for Station Blackout	60 min	SBO,CCW,ESW	6.99E-04	SGPORV
RRI- Restoration of RCS Inventory	60 min	SBO,CCW,ESW	6.80E-04	PORVP,HP4, HPI
MF1- Initiation of Main Feedwater	30 min	TRA	1.63E-04	12FEED
MRI- Manual Rod Insertion	1 min	ATW	1.37E-03	--
PP1,PP2- Primary Pressure Relief	N/A	ATW	1.46E-03	PZRSAFE
LTS- Long-Term Shutdown	N/A	ATW	1.35E-03	HP5

TABLE 3.3-5 (Cont'd.)

SUMMARY OF HUMAN ERROR PROBABILITIES

Operator Action of Event Tree Node	Time Available	Used in Event Tree	HEP (without hardware failures)	Dependencies/ Notes
OIB- Isolation of RHR Pump Seal LOCA	N/A	ISL	5.40E-02	--
OA6- Post-LOCA Secondary-Side Cooldown	2 hrs	SLO,MLO	6.23E-04	SGPORV,AF4
2AV- Unit 2 AFW Crosstie	30 min	VDC	2.08E-04	2AF
RCE- RCS Cooldown with RWST Conservation	10 min	ISL	1.08E-03	SGPORV, 13PORV

3.3.4 Common Cause Failure Data

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

To assure that the effects of common cause were properly addressed in the Cook Nuclear Plant analysis, common cause failures were modelled at the system level in the fault trees. The Multiple Greek Letter (MGL) method was used for quantifying common cause failures. The Cook Nuclear Plant IPE used the MGL method and parametric factors (beta, gamma, and delta) as defined in Reference 30 as follows:

- BETA conditional probability ($Pr(x \geq 2 | x \geq 1)$) that the common cause of a component failure will be shared by one or more additional components

- GAMMA conditional probability ($Pr(x \geq 3 | x \geq 2)$) that the common cause of a component failure that is shared by one or more components will be shared by two or more components additional to the first

- DELTA conditional probability ($Pr(x \geq 4 | x \geq 3)$) that the common cause of a component failure that is shared by two or more components will be shared by three or more components additional to the first

The CCF analysis took the following approach for each modelled system:

1. Identify CCF groups
2. Place CCFs in the system fault trees
3. Calculate CCF values
4. Quantify CCFs with the system fault tree failures

Note: Human error was addressed separately in the IPE process and was not included in common cause failure calculations.

The evaluation of the Cook Nuclear Plant component failure data indicated that there had been no common cause events at the Cook Nuclear Plant applicable to current maintenance and operation practices. As a result, a generic CCF database of common cause events was developed using Reference 7. The generation of the generic CCF database was prepared using Reference 35. The specific MGL factors resulting from this analysis are included in Table 3.3-6.

Table 3.3-6 is not a comprehensive listing of MGL factors for all components. Thus, an average common cause component group, the ALL row in Table 3.3-6, was quantified from a composite of all the common cause failures for all components in the database. The ALL MGL factors were applied to components which have no history of common cause failure, but were judged by the system analyst to have a potential for common cause failure. As an example, air operated valves (AOVs) do not appear in Table 3.3-6, therefore, the ALL MGL factors would apply to the AOVs in a system where the AOVs are grouped into a common cause failure group. In general, the components from the database were judged to be more complex and more susceptible to common cause failures than those components which would employ the ALL MGL factors. The ALL MGL factors are, therefore, conservative.

Use of the generic CCF database in the Cook Nuclear Plant IPE was judged to be conservative. Further examination was required only for those CCF events which appeared to be dominating the core damage

frequency. For those CCF events which dominated core damage, expert judgment was used to verify that the common cause event's applicability to Cook Nuclear Plant was either appropriate or conservative.

TABLE 3.3-6

COOK NUCLEAR PLANT IPE MGL COMMON CAUSE FACTORS

Component	Beta	Gamma	Delta
Reactor Trip Breakers	0.16	0.40	0.61
Diesel Generators	0.025	0.15	0.25
Motor Operator Valves	0.038	0.23	0.69
Safety Relief Valves	0.094	0.66	0.66
Check Valves	0.06	0.33	0.52
Pumps			
High Head	0.10	0.28	0.19
Residual Heat Removal	0.077	0.15	0.43
Containment Building Spray	0.057	0.24	--a--
Auxiliary Feedwater	0.021	0.20	0.52
Service Water and Component Cooling Water	0.032	0.63	0.84
Chillers	0.11	0.33	0.52
Fans	0.13	0.33	0.52
All*	0.08	0.33	0.52

* Average of all component failures.

a - Value of factor is not calculated. A value equal to the value for the average of all component failures (All) is used for the generic MGL screening method, that is, delta = .52

3.3.5 Quantification of Unavailability of Systems and Functions

The fault trees which modelled plant systems and operator actions (such as primary plant bleed and feed) were constructed and quantified using the GRAFTER code system (Reference 10). The Fault Tree Analysis Guidelines for Cook Nuclear Plant (Reference 48) provided guidance for fault tree construction. Plant systems are described in Section 3.2. Supporting descriptions for the operator actions fault trees are found in Section 3.3.3 and in the Human Reliability Analysis Notebook (Reference 15). Table 3.3-7 is a list of fault tree/event tree heading probabilities which includes the system and function unavailabilities. These values include common cause failures that were discussed in Section 3.3.4.

Note that the unavailabilities listed in Table 3.3-7 were quantified assuming that all support systems were available (i.e. - support system failure values were not included within Table 3.3-7 values.) Support system unavailabilities were accounted for in the quantification of the initiating event accident sequences (Section 3.3.6).

TABLE 3.3-7

UNAVAILABILITY OF SYSTEMS AND FUNCTIONS

AF1	6.69E-05	HPI	3.56E-05
AFS	1.22E-03	HP2	1.20E-03
AFT	5.79E-02	HP3	5.16E-04
AFH	7.60E-04	HP5	9.83E-04
AF2	6.88E-05	HPR	1.26E-03
AF3	4.60E-04	T11A/T11D	2.34E-05
AFC	5.72E-03	T11B/T11C	2.34E-05
AF4	6.69E-05	T11AP/T11DP	1.33E-04
2AF	2.43E-04	T11BP/T11CP	1.32E-04
CSI	1.48E-04	11A/11D	3.20E-06
CSR	7.75E-04	11B/11C	2.97E-06
CCWE	8.89E-05	11AZ/11DZ	6.61E-05
CCWW	3.12E-03	11BZ/11CZ	4.21E-05
CCWEL	1.35E-04	1AB/1CD	1.82E-02
CCWWL	3.17E-03	DCB	6.22E-05
ESWE	2.18E-05	DCA	3.29E-04
ESWW	6.89E-03	DCN	3.42E-05
ESWEL	9.79E-05	DCNSBO	1.61E-05
ESWWL	8.10E-03	CRID1/2/3/4	2.81E-06
NESW1	1.18E-04	120AFW	1.23E-03
NESW2	2.79E-04	ELSC	1.23E-03
CONAIR1	6.27E-04	12FEED	1.17E-02
CONAIRLS	1.55E-02	13COND	5.43E-04
CONAIR2	2.40E-03	13CONDMF	4.54E-04
CONAIR2L	3.96E-03	MSI	3.68E-04
CONAIR3	2.23E-04	SSV	9.68E-02
CONAIR3L	2.23E-04	SGI	2.48E-03
CONAIR4	2.23E-04	SGPORV	5.62E-05
CONAIR4L	2.23E-04	PORVP	8.05E-04
CONAIR5	2.23E-04	PZRSAFE	1.79E-02
CONAIR5L	2.23E-04	PZRSAFE1	1.11E-03
ACC	6.00E-04	13PORV	1.93E-04
LPI	3.09E-04	CF	5.45E-04
LPR	8.47E-04	HI	1.37E-04

3.3.6 Quantification of Sequence Frequencies

The IPE Project used a PC based (IBM PS/2) PRA software code (WLINK code system, Reference 46) to quantify the initiating event accident sequences. The event trees described in Section 3.1 provided the structure for propagation of the initiating event sequences. Using a fault tree linking process, the WLINK code (Reference 46) linked all necessary support system fault trees into each event tree top event node. Afterwards, the WLINK code (Reference 46) quantified each initiating event accident sequence. This yielded accident sequence frequencies and dominant accident event cutsets. These outputs were then used in yet another quantification which produced an overall core damage frequency value (internal events) and overall dominant contributors to core damage frequency. These results are explained in Section 3.4.

3.3.7 Internal Flooding Analysis

This section summarizes the methodology and results of the internal flooding evaluation.

3.3.7.1 Information Assembly

The following information was assembled and tabulated in the Cook Nuclear Plant Internal Flooding Analysis Notebook:

1. Flood Zones

   The flood zones were chosen to correspond to the existing fire zones developed for analysis of compliance with 10CFR50 Appendix R requirements. In general, barriers separating Appendix R zones were found to be applicable to the internal flooding analysis. Those Appendix R zones bounded by anything other than actual barriers were analyzed on a case-by-case basis for flooding applicability.

2. Sources of Flooding and Spraying

   Sources of flooding and spraying and the location of the piping associated with these sources were identified.

3. Location of Components Critical to Operation or Safe Shutdown

   Components and their locations that were considered critical to operation or safe shutdown were identified.

4. Description of Flood and Spray Events

   For each flood zone, flooding and water spray effects on components and equipment were described.

3.3.7.2 Major Assumptions

The following major assumptions were made during the analysis:

1. Equipment within a zone that is subject to a flooding or spray event is disabled.

2. Containment flooding due to a LOCA or high energy line breaks was determined not to be a concern since this is a consideration in the plant's design basis. As a result, containment flooding was not reviewed in this flooding analysis.

3. Low- and medium-energy insulated pipes were assumed to drip if a leak developed (i.e., they are

not a spray source). Bare pipes were assumed to be spray sources for a 10 foot line-of-sight radius.

4. Lines which were not normally filled with liquid (i.e., drain lines and dry fire protection piping) were not considered to be credible flooding or spray sources.

3.3.7.3 Methodology

The methodology used for the flooding analysis can be summarized as follows:

1. Possible flood-induced initiating events were identified and, for the purposes of this analysis, include only a transient without power conversion systems available.

2. A qualitative screening was performed using internal AEPSC calculations to identify the significant flooding events. The capability of a flooding event leading to a transient event and the possibility of disabling safe shutdown components as a result of the flood were the two factors considered as significant during the screening analysis. If the total loss of components within a flood zone would not initiate a transient, or if no safe shutdown equipment would be affected, then the flood zone was removed from further analysis.

3. Quantitative analyses were performed for those areas identified as significant. For quantitative analysis, the following calculation of initiating event frequency was made:

   For flooding in the turbine hall, a worst case flood would result from an ESW discharge line break. The frequency of this event was conservatively calculated using the probability of a main condenser expansion joint failure. The initiating event frequency was calculated as follows:

   Cond = Condenser
   EJ = Expansion Joint

$$= 2.5E{-}4 \ \frac{\text{failures}}{\text{EJ-yr}} \times 6 \text{ cond} \times 2 \ \frac{\text{EJ}}{\text{cond}}$$

$$= 3.0E{-}3/\text{yr}$$

   *Note: The failure rate for the condenser expansion joint is from Reference 67.

3.3.7.4 Results

The only significant flooding scenario found was an ESW discharge line break causing a flood of the turbine building sub-basement. This flood is postulated to disable the NESW pumps, which in turn cause failure of the plant and control air compressors. The transient with steam conversion systems unavailable event tree was quantified for this scenario and yielded a core melt contribution of 2.00E-07/year.

3.4 Results and Screening Process

3.4.1 Application of Generic Letter Screening Criteria

Following the guidance provided in NUREG-1335, the following screening criteria were utilized to determine potentially important systemic sequences and system failures.

1. Any systemic sequence that contributes 1.00E-07 or more per reactor year to core damage.

2. All systemic sequences within the upper 95% of the total core damage frequency.

3. All systemic sequences within the upper 95% of the total containment failure frequency.

4. Systemic sequences that contribute to a containment bypass frequency in excess of 1.00E-08 per reactor year.

Table 3.4-1 provides a listing of each initiating accident event, the CDF value, the initiating event frequency and percent contribution to CDF. The total core damage frequency at Cook Nuclear Plant is 6.26E-5. Table 3.4-2 provides the important systemic sequences, descriptions of sequence progression, major contributors to sequence failure and major operator actions. The sequences in this table are grouped by accident event, starting from the most dominant accident (small LOCA) to the least (interfacing systems LOCA or V-sequence LOCA).

As noted in the Section 3.1, containment systems (containment spray, hydrogen igniters and containment air recirculation and hydrogen skimmer systems) were modelled within the accident sequences. To account for core damage, therefore, each sequence within the event trees was traced backwards to the point were actual core damage occurred. The beginning of the sequence to this point was defined as a "systemic sequence". Summing the failure values that branched out from this point gave the sequence failure values shown in Table 3.4-2. The first 100 sequences from the accident sequence quantification, which provided input to Table 3.4-2, are shown in Table 3.4-3. Each sequence cutset shown in Table 3.4-3 lists the sequence failure probability value, the accident event that initiated the sequence (IEV-xxx), the sequence progression (SUC-xxx indicates that top event xxx was successful) and the component/event identifier which identifies the component/event that failed the sequence. An importance ranking of the dominant contributors to CDF is found in Table 3.4-4. Each contributor in Table 3.4-4 is listed by its' component/event identifier, its' overall importance to CDF, number of sequence cutsets the identifier appeared in, its' overall contribution to CDF (derived from its importance to CDF and identifier placement within the logic trees) and the failure probability value of the identifier by itself. Contributors that begin with "SUC-" indicate success of these particular event tree top nodes (success that occurred most often) and do not indicate failure of any kind. Contributors that begin with "IEV-" are directly linked to accident events (e.g. IEV-SLO indicates that the small LOCA event was the most significant accident with a 47.3% contribution to CDF, an accident CDF value of 2.963E-5 and a small LOCA initiating event frequency of 6.80E-3.)

TABLE 3.4-1

ACCIDENT EVENT SUMMARY

Accident	Core Damage Value	Initiating Event Frequency	% Contribution
SLO	2.96E-05	6.80E-03	47.3
CCW	1.38E-05	8.71E-04	22.1
SGR	7.07E-06	7.20E-03	11.3
MLO	4.31E-06	9.17E-04	6.9
ATW	2.85E-06	4.67E-05	4.6
SBO	1.13E-06	1.40E-05	1.8
LLO	9.52E-07	3.00E-04	1.5
SWS	6.04E-07	3.73E-05	0.96
VDC	6.04E-07	1.16E-02	0.96
SLB	5.69E-07	3.30E-04	0.91
VEF	3.00E-07	3.00E-07	0.48
TRA	2.86E-07	3.80	0.46
TRS	2.83E-07	1.20E-01	0.45
LSP	1.73E-07	4.00E-02	0.28
ISL	5.38E-08	6.70E-07	0.08

TABLE 3.4-2

SUMMARY OF SIGNIFICANT SEQUENCES

KEY

ACC - Accumulators

AF1, AF4 - Auxiliary Feedwater Actuation

AF2 - Auxiliary Feedwater to intact Steam Generator(s)

AF3 - Auxiliary Feedwater to faulted Steam Generator

AFC - Auxiliary Feedwater Continues

AFH - Half flow - 900 GPM Aux. Feedwater flow

AFS - Auxiliary Feedwater Actuation with Steam Line Break

AFT - Turbine - Driven Auxiliary Feedwater flow

2AV - Op. Action to provide Unit 2 AFW flow

AMS - ATWS Mitigation System Actuation Circuitry

BRH - Residual Heat Removal System Breach

CF - Containment Recirculation Fans

CH1 - Component Cooling Water System restored w/in 1 hour

CH8 - Component Cooling Water System restored w/in 8 hours

CNU - Reactor core not uncovered

CSI - Containment Spray Injection

CSR - Containment Spray Recirculation

EH1 - Essential Service Water System restored w/in 1 hour

TABLE 3.4-2 (Cont'd.)

SUMMARY OF SIGNIFICANT SEQUENCES

KEY (cont'd.)

EH8 - Essential Service Water System restored w/in 8 hours

HI - Hydrogen Igniters

HPI, HP2, HP3 - Emergency Core Cooling System (High Pressure) Injection

HPR - Emergency Core Cooling System (High Pressure) Recirculation

LPI - Residual Heat Removal (Low Pressure) Injection

LPR - Residual Heat Removal (Low Pressure) Injection

LTS - Long Term Shutdown

MF1 - Main Feedwater

MS1 - Secondary Side Isolation

OA1 - Depressurize and Cooldown Primary Side Before Filling

OA2 - Depressurize and Cooldown Primary Side After Filling

OA3 - Cooldown and Depressurize Primary Side Per ECA-3.1/3.2 (Subcooled/Saturated)

OA5 - Op. Action to Depressurize a Steam Generator

OA6 - Primary Cooldown Using AFW and Steam Dump

OIB - Op. Action to Isolate Residual Heat Removal LOCA

OLI - Depressurize Primary and Initiate Low Pressure Injection

SGI - Steam Generator Isolation by Main Steam Isolation Valve Closure

SSV - Integrity Maintained or Restored in Steam Generator

TABLE 3.4-2 (Cont'd.)

SUMMARY OF SIGNIFICANT SEQUENCES

KEY (cont'd.)

PBF - Primary Side Bleed and Feed

PPR - Primary Side Pressure Reduction

RCD - Reactor Coolant System (Primary Side) Cooldown

RCE - Reactor Coolant System Cooldown and Refuel Water Storage Tank Conservation

RCP - Reactor Coolant Pumps Tripped

RPL - Reduced Power Level (less than 40%)

RRI - Restore Reactor Coolant System Inventory

RVC - Residual Heat Removal Relief Valve Closure

XHR - AC Power Restoration In X Hours

Major Contributors

AFW = Auxiliary Feedwater

CC = Common Cause (Common Mode) Failures

CST = Condensate Storage Tank

CTS = Containment Spray System

ECCS = Emergency Core Cooling System (High Pressure)

ESF = Engineered Safety Features (Reactor Protection)

MOV = Motor Operated Valve

TABLE 3.4-2 (Cont'd.)

SUMMARY OF SIGNIFICANT SEQUENCES

KEY (cont'd.)

Major Contributors (cont'd.)

MSIV = Main Steam Isolation Valve

OE = Operator Error

PORV = Power Operated Relief Valve

RHR = Residual Heat Removal

SI = Safety Injection (part of ECCS)

Note: An asterisk (*) by a sequence indicates that the sequence is within the upper 95% of CDF

The accident events are presented in order of importance. The values within the parenthesis beside each EVENT are as follows: event core damage value, initiating event frequency, event importance contribution to CDF.

The Table 3.4-2 columns are described as follows:

Sequence Failure Probability - sequence failure value (Section 3.4.1 describes how value was derived).

Sequence Progression - accident sequence progression up through core damage

Failure Area(s) - Point(s) in sequence where system/operator failure occurred

Major Contributors - major items that failed the Failure Area(s)

Operator Actions - event tree top event operator actions (ex. OA3) and operator actions found within the plant system top events (ex. HPR)

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Small LOCA-SLO (2.96E-05 - 6.80E-03 - 47.3%)				
Top Events: SLO-HP2-OA6-PBF-CSI-HPR-CSR-HI-CF				
* 1.350E-05	SLO-HP2-OA6-CSI-HPR	CSI, HPR	- CC ECCS recirculation-SI pumps fail - CC ESF relays - SI pump failure - RHR pump failure - valve maintenance - ESW pump maintenance - ECCS/RHR pump train maintenance	OA6, OE switch to cold leg recirculation (HPR)
* 1.173E-05	SLO-HP2	HP2	- CC ECCS Injection-SI pumps fail - CC ESF signal fail - CC ESF relays - CC both ESF trains - check valve failure	None
* 4.379E-06	SLO-HP2-OA6-PBF	OA6, PBF	- plant air compressors fail - CC compressed Air System - valve failure - compressed air safety valve failure	OA6, PBF

TABLE 3.4-2 (Cont'd.) SUMMARY OF SIGNIFICANT SEQUENCES				
Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Loss Component Cooling Water-CCW (1.38E-05--8.71E-04--22.1%)				
Top Events: CCW-RCP-AF1-MF1-CH1-RCD-CH8-CNU-RRI-CSI-HPR-CSR-HI-CF				
* 1.13E-05	CCW - RCP	RCP	RCP	RCP
1.626E-06	CCW-RCP-AF1-CH1-RRI-CSI-HPR	HPR	- CC ECCS recirculation-SI pumps fail	RCP, CH1, RRI, OE align CST backup water supply (AF1), OE switch to cold leg recirculation (HPR)
8.643E-07	CCW-RCP-AF1-CH1-RRI	RRI	- OE diagnose RCS cooling needs - CC ESF signal - CC both ESF trains	RCP, CH1, RRI, OE align CST backup water supply (AF1)

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Steam Generator Tube Rupture - SGR (7.07E-06--7.20E-03--11.3%)				
Top Event: SGR-AF2-AF3-HPI-SGI-OA1-SSV-OA2-OA3-PBF-CSI-HPR-CSR-HI-CF				
5.995E-06	SGR-AF2-HPI-SGI-OA1-SSV-OA2	OA1, OA2	- Plant air compressors fail - CC compressed air (85 psi reduces) - CC compressed Air System-valve failure - CC stm Gen. PORVs	OA1, OA2, OE align CST backup water supply (AF2), OE recognize event and isolate stm Gen. (SGI)
6.598E-07	SGR-AF2-HPI-SGI-OA1-SSV-OA3	OA1, SSV, OA3	- Plant air compressors fail (OA1, OA3) - faulted steam generator safety valves fail to reseat	OA1, OA3, OE recognize event and isolate stm gen (SGI)
2.503E-07	SGR-AF2-AF3-HPI	AF2, AF3, HPI	- Turb. driven AFW pump failure (AF2, AF3) - ESF signal failure (HPI)	OE align CST backup water supply (AF2)
1.194E-07	SGR-AF2-HPI-SGI-OA3	SGI, OA3	- CC 250 VDC fuses - MSIV failure to isolate (SGI) - regulator failure to Pressurizer PORVs (OA3) - Plant air compressor failure (OA3) - CC RHR pump failure	OA3, OE align CST backup water supply(AF2), OE recognize event and isolate stm. gen. (SGI)
3.440E-08	SGR-AF2-HPI-SGI	HPI, SGI	- CC 250 VDC fuses - CC ESF signal (HPI) - MSIV failure to isolate (SGI)	OE align CST backup water supply, OE recognize event and isolate stm gen (SGI)

TABLE 3.4-2 (Cont'd.) SUMMARY OF SIGNIFICANT SEQUENCES				
Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Medium LOCA-MLO (4.31E-06--9.17E-04--6.9%)				
Top Event: MLO-ACC-HP2-OA6-OLI-CSI-HPR-LRP-CSR-HI-CF				
1.802E-06	MLO-ACC-HP2-OA6-CSI-HPR	CSI, HPR	- CC ECCS recirculation-SI pumps fail - CC ESF relays (CSI, HPR)	OA6, OE switch to cold leg recirculation (HPR)
1.493E-06	MLO-ACC-HP2-OA6	OA6	- OE diagnose cooldown need - Plant air compressors fail - CC compressed air system - compressed air safety valve failure	OA6
5.490E-07	MLO-ACC	ACC	- Accumulator check valve failure	None
2.541E-07	MLO-ACC-HP2-OLI	HP2, OLI	- CC ECCS injection - SI pumps fail (HP2) - OE failure to diagnose de pressurization need (OLI) - CC ESF signal (HP2) - check valve failure (HP2)	OLI
2.192E-07	MLO-ACC-HP2-OLI-CSI-LPR	HP2, LPR	- CC ESF signal (HP2, LPR) - ESF signal failure - both trains (HP2, LPR)	OLI, OE switch to cold leg recirculation (LPR)

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Anticipated Transient Without Scram - ATW (2.851E-06--4.67E-05--4.6%)				
Top Event: ATW-RPL-MRI-AMS-AF1-AFH-PPR-LTS-CSI-CSR-HI-CF				
2.308E-06	ATW-RPL-MRI-AMS-AFH-PPR	RPL, PPR	- Reactor Power >40% (RPL) - Pressure relief inadequate early in fuel cycle	MRI, PPR, OE align CST backup water supply (AFH)
3.910E-07	ATW-RPL-MRI-AMS	RPL, AMS	- Reactor Power >40% (RPL) - ATWS mitigation system failed	MRI
EVENT: Station Blackout-SBO (1.132E-06--1.40E-05--1.8%)				
TOP EVENTS: SB0-AFT-RCD-AFC-XHR-CNU-RRI-AF1-PBF-CSI-HPR-CSR-HI-CF				
4.050E-07	SBO-AFT-RCD-AFC-XH1-CNU	CNU	CNU-core uncovered	RCD, XH1
5.779E-07	SBO-AFT-RCD-AFC-XH1	XH1	- Failure to restore electrical power (XH1)	RCD, XH1

TABLE 3.4-2 (Cont'd.) SUMMARY OF SIGNIFICANT SEQUENCES				
Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors'	Operator Actions
EVENT: Large LOCA-LLO (9.523E-07--3.00E-04--1.5%)				
TOP EVENTS: LLO-ACC-LPI-CSI-LPR-CSR-HI-CF				
3.031E-07	LLO-ACC-LPI-CSI-LPR-CSR	CSR	- CC-CST system (CTS pump failure) - CC ESF relays - OE-switch CTS to recirculation	OE switch to cold leg recirculation (LPR); OE switch CTS to recirculation (CSR)
2.941E-07	LLO-ACC-LPI-CSI-LPR	LPR	- CC-RHR recirculation (RHR pump failure) - RHR pump failure (injection and recirculation) - CC ESF relays - OE switch RHR to recirculation	OE switch to cold leg recirculation (LPR)
1.794E-07	LLO-ACC	ACC	- Accumulator check valve failure	None
1.754E-07	LLO-ACC-LPI	LPI	- CC ESF relays - CC ESF signal - ESF signal unavailable - both trains - RHR check valve failure - CC RHR injection (RHR pump failure)	None

TABLE 3.4-2 (Cont'd.) SUMMARY OF SIGNIFICANT SEQUENCES				
Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Loss of Essential Service Water-SWS (6.04E-07--3.73E-05--0.96%)				
TOP EVENTS: SWS-RCP-AF1-MF1-EH1-RCD-EH8-CNU-RRI-CSI-HPR-CSR-HI-CF				
4.843E-07	SWS-RCP	RCP	RCP	RCP
EVENT: Loss of 250 VAC Power - VDC (6.04E-07--1.16E-02--0.96%)				
TOP EVENT: VAC-AF1-2AV-CSI-CSR-HI-CF				
6.037E-07	VDC-AF1-2AV	AF1, 2AV	- OE align CST backup water supply	2AV, OE align CST backup water supply (AF1)

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Steam Line Break-SLB (5.69E-07--3.30E-04--0.91%)				
TOP EVENT: SLB-HP3-MS1-AFS-PBF-CSI-HPR-CSR-HI-CF				
3.146E-07	SLB-HP3	HP3	- CC ECCS injection (MOV failure) - CC ESF signal - CC ESF relays - ESF signal unavailable-both trains - ECCS check valve failure	None
2.399E-07	SLB-HP3-MS1	MS1	- CC Main Steam (MSIVs fail to shut) - CC ESF relays - CC ESF signal - ESF signal unavailable-both trains	None
EVENT: Reactor Vessel Failure-VEF (3.00E-07--3.00E-07--0.48%)				
TOP EVENT: VEF-CSI-CSR-HI-CF				
2.995E-07	VEF	VEF	VEF	None

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Transients with Steam Conversion Available-TRA (2.86E-07--3.80--0.46%)				
TOP EVENT: TRA-AF1-OA5-MF1-PBF-CSI-HPR-CSR-HI-CF				
2.857E-07	TRA-AF1-OA5-MF1-PBF	AF1, OA5, MF1, PBF	- OE-align CST backup water supply (AF1, OA5) - Plant air compressors fail (MF1, PBF) - CC compressed air system (MF1, PBF) - Compressed air safety valve failure (MF1, PBF) - DC power vent fan failure - DC power fuse failure	OA5, MF1, PBF, OE align CST backup water supply (AF1)
EVENT: Transients Without Steam Conversion Available-TRS (2.83E-07--1.20E-01--0.45%)				
TOP EVENTS: TRS-AF1-PBF-CSI-HPR-CSR-HI-CF				
2.647E-07	TRS-AF1-PBF	AF1, PBF	- AFW turb. driven pump failure (AF1) - CC 4KV power ventilation failure - CC 250 VDC power - OE-align CST backup water supply - Air pressure regulator failure (PBF)	PBF, OE align CST backup water supply (AF1)

TABLE 3.4-2 (Cont'd.)
SUMMARY OF SIGNIFICANT SEQUENCES

Sequence Failure Probability	Sequence Progression	Failure Area(s)	Major Contributors	Operator Actions
EVENT: Loss Offsite Power-LSP (1.73E-07--4.00E-02--0.28%)				
TOP EVENT: LSP-AF1-PBF-CSI-HPR-CSR-HI-CF				
1.677E-07	LSP-AF1-PBF	AF1, PBF	- AFW turb. driven pump failure (AF1) - CC 4 KV power ventilation failure - CC 250 VDC power - OE-align CST backup water supply - Plant air compressors fail - CC-Compressed Air System (valve failure)	PBF, OE align CST backup water supply (AF1)
EVENT: Interfacing Systems LOCA (V-Sequence) - ISL (5.38E-08--6.70E-07--0.08%)				
TOP EVENT: ISL-BRH-HP2-OIB-AF4-RCE-RVC				
3.680E-08	ISL-BRH-HP2-OIB	OIB	OIB	OIB
1.260E-08	ISL-BRH-HP2-OIB-AF4-RCE-RVC	RVC	RVC	OIB, RCE, OE align CST backup water supply (AF4)

TABLE 3.4-3

CDF.OUT

1.	1.13E-05	5	IEV-CCW	SUC-CSI	SUC-CSR	SUC-HI	RCP					
2.	7.78E-06	7	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	G-HHRS------CM			
3.	4.60E-06	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	F-HP2-------CM					
4.	2.74E-06	7	IEV-SLO	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41PS			
5.	2.62E-06	10	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41PS
6.	2.12E-06	9	IEV-ATW	SUC-MRI	SUC-AMS	SUC-AFH	SUC-CSI	SUC-CSR	SUC-HI	RPL	UE1	
7.	1.41E-06	9	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	X-CC-CONAIR3	
8.	1.36E-06	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-SI-SIGNAL					
9.	1.05E-06	8	IEV-HLO	SUC-ACC	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	G-HHRS------CM		
10.	9.84E-07	9	IEV-CCW	SUC-RCP	SUC-AF1	SUC-CH1	SUC-RRI	SUC-CSI	SUC-CSR	SUC-HI	G-HHRS------CM	
11.	8.47E-07	6	IEV-SLO	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CC-CONAIR				
12.	8.09E-07	9	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	X-CC-CONAIR	
13.	6.78E-07	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FBCV---SI101FO					
14.	6.78E-07	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	F-CV---SI185FO					
15.	6.05E-07	7	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-HPR-RELAY			
16.	5.83E-07	5	IEV-VDC	SUC-CSI	SUC-CSR	SUC-HI	Z-TK-SLOWCSTHE					
17.	5.60E-07	7	IEV-HLO	SUC-ACC	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	OA6-DIAG-MN-HE			
18.	5.53E-07	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-HP2-RELAY					
19.	5.25E-07	8	IEV-CCW	SUC-RCP	SUC-AF1	SUC-CH1	SUC-CSI	SUC-CSR	SUC-HI	RRI-DIAG-MN-HE		
20.	4.88E-07	6	IEV-SLO	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-AM----SV44CO				
21.	4.81E-07	6	IEV-SWS	SUC-EH8	SUC-CSI	SUC-CSR	SUC-HI	RCP				
22.	4.66E-07	9	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	X-AM----SV44CO	
23.	4.05E-07	9	IEV-SBO	SUC-AFT	SUC-RCD	SUC-AFC	SUC-XH1	SUC-CSI	SUC-CSR	SUC-HI	SN1	
24.	3.91E-07	7	IEV-ATW	SUC-MRI	SUC-CSI	SUC-CSR	SUC-HI	RPL	AMS			
25.	3.87E-07	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	QXSI					
26.	3.70E-07	8	IEV-HLO	SUC-ACC	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41PS		
27.	3.59E-07	6	IEV-SBO	SUC-AFT	SUC-RCD	SUC-AFC	SBO-I-TR-CK-HE	SBO-V-TR-CK-HE				
28.	3.53E-07	9	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	E-CC-SGPORV	
29.	2.99E-07	4	IEV-VEF	SUC-CSI	SUC-CSR	SUC-HI						
30.	1.99E-07	7	IEV-HLO	SUC-ACC	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CC-CONAIR3			
31.	1.83E-07	5	IEV-SBO	SUC-AFT	SUC-RCD	SUC-AFC	SBO--------XH1 0					
32.	1.71E-07	8	IEV-CCW	SUC-RCP	SUC-AF1	SUC-CH1	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-SI-SIGNAL		
33.	1.58E-07	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAPM---PP35EPS-R	IB--RH-VALVETM		
34.	1.58E-07	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IBPM---PP35WPS-R	IA--RH-VALVETM		
35.	1.58E-07	6	IEV-SGR	SUC-CSI	SUC-CSR	SUC-HI	DMPT-----PP4PS	Q-CC-SI-SIGNAL				

TABLE 3.4-3 (Cont'd.)

CDF.OUT

36.	1.55E-07	7	IEV-HLO	SUC-ACC	SUC-OLI	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-SI-SIGNAL		
37.	1.42E-07	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FAPM---PP26NPS	FB--SI-VALVETM			
38.	1.42E-07	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FBPM---PP26SPS	FA--SI-VALVETM			
39.	1.41E-07	7	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--SI-VALVETM	BPM---1PP7WTM	ESW-MULT1-7W		
40.	1.41E-07	9	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-VALVETM	BBPM---1PP7WTM	ESW-MULT1-7W
41.	1.29E-07	7	IEV-LLO	SUC-ACC	SUC-LPI	SUC-CSI	SUC-CSR	SUC-HI	I-LPR-------CM		
42.	1.24E-07	7	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--CC-TRAINTM	BBPM---1PP7WTM	ESW-MULT1-7W		
43.	1.14E-07	7	IEV-HLO	SUC-ACC	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CC-CONAIR		
44.	1.13E-07	6	IEV-SLB	SUC-HP3	SUC-CSI	SUC-CSR	SUC-HI	E-CC-MS1			
45.	1.10E-07	7	IEV-LLO	SUC-ACC	SUC-LPI	SUC-CSI	SUC-LPR	SUC-HI	L-CC--CTSSYS--		
46.	1.07E-07	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IBPM---PP35WPS-R	IA--RH-TRAINTM	
47.	9.55E-08	5	IEV-SLB	SUC-CSI	SUC-CSR	SUC-HI	F-HP3-------CM				
48.	9.49E-08	9	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-TRAINTM	BBPM---1PP7WTM	ESW-MULT1-7W
49.	9.26E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	GAPM---PP26NPS	GBPM---PP26SPS	
50.	9.24E-08	7	IEV-HLO	SUC-ACC	SUC-CSI	SUC-CSR	SUC-HI	F-HP2-------CM	OLI-DIAG-SC-HE		
51.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NACV-SI170L3FO				
52.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NACV-SI166L3FO				
53.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NBCV-SI170L2FO				
54.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NBCV-SI166L2FO				
55.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NACV-SI170L1FO				
56.	9.15E-08	5	IEV-HLO	SUC-CSI	SUC-CSR	SUC-HI	NACV-SI166L1FO				
57.	8.15E-08	8	IEV-HLO	SUC-ACC	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-HPR-RELAY	
58.	8.14E-08	7	IEV-TRA	SUC-CSI	SUC-CSR	SUC-HI	Z-TK-SLOWCSTHE	X-CM--1OME41PR	X-CM--2OME41PS		
59.	7.73E-08	9	IEV-ATW	SUC-MRI	SUC-AMS	SUC-AFH	SUC-CSI	SUC-CSR	SUC-HI	RPL	X-PC--XRV186FO
60.	7.65E-08	9	IEV-CCW	SUC-RCP	SUC-AF1	SUC-CH1	SUC-RRI	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-HPR-RELAY
61.	7.50E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAPM---PP35EPS-R	IBPM---PP35WPS-R	
62.	7.41E-08	9	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	GAPM---PP26NPS	BBPM---1PP7WTM	ESW-MULT1-7W
63.	7.40E-08	5	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	C-CC-CCW----CM				
64.	7.19E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--SI-VALVETM	CBMV--CH0420FO			
65.	7.19E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--SI-VALVETM	BBMV--WM0737FO			
66.	7.19E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--SI-VALVETM	BBMV-1WM0702FO			
67.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAMV--IM0340CC	IB--RH-VALVETM	
68.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	CAMV--CH0419CC	IB--RH-VALVETM	
69.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAMV--ICM305CC	IB--RH-VALVETM	
70.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAMV--IM0310FC	IB--RH-VALVETM	

TABLE 3.4-3 (Cont'd.)

CDF.OUT

71.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	LAMV--IM0215FC	IB--RH-VALVETM		
72.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IBMV--IM0350CC	IA--RH-VALVETM		
73.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	CBMV--CM0429CC	IA--RH-VALVETM		
74.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IBMV--ICM306CC	IA--RH-VALVETM		
75.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IBMV--IM0320FC	IA--RH-VALVETM		
76.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	LBMV--IM0225FC	IA--RH-VALVETM		
77.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-VALVETM	CBMV--CM0420FO		
78.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-VALVETM	BBMV--WM0737FO		
79.	7.17E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-VALVETM	BBMV-1WM0702FO		
80.	7.06E-08	3	IEV-TRS	DNPT-----PP4PS	A-FN----FANSCM							
81.	6.79E-08	4	IEV-SLO	SUC-CSI	SUC-HI	B-CC-ESW						
82.	6.67E-08	9	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IAPM---PP35EPS-R	BBPM---1PP7WTM	ESW-MULT1-7W	
83.	6.59E-08	4	IEV-SLB	SUC-HP3	SUC-HI	Q-CC-CP-SIGNAL						
84.	6.59E-08	5	IEV-SLB	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-SI-SIGNAL					
85.	6.57E-08	7	IEV-MLO	SUC-ACC	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-AM----SV44CO			
86.	6.50E-08	7	IEV-SLO	SUC-HP2	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41TM			
87.	6.30E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FBMV--QM022600	FA--CC-TRAINTM				
88.	6.30E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--CC-TRAINTM	CBMV--CM0420FO				
89.	6.30E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--CC-TRAINTM	BBMV--WM0737FO				
90.	6.30E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--CC-TRAINTM	BBMV-1WM0702FO				
91.	6.21E-08	10	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-SSV	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41TM
92.	6.19E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FAPM---PP26NPS	FB--SI-TRAINTM				
93.	6.13E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FAMV--QM022500	FB--CC-TRAINTM				
94.	6.07E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FAPM---PP26NPS	FBPM---PP26SPS				
95.	6.02E-08	9	IEV-LLO	SUC-ACC	SUC-LPI	SUC-CSI	SUC-CSR	SUC-HI	I-1PUMPINJ--FA	I-1PUMPRECIRFA	TWO	
96.	6.00E-08	7	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FAPM---PP26NPS	BBPM---1PP7WTM	ESW-MULT1-7W			
97.	5.98E-08	6	IEV-LLO	SUC-ACC	SUC-CSI	SUC-CSR	SUC-HI	Q-CC-SI-SIGNAL				
98.	5.86E-08	6	IEV-SLO	SUC-CSI	SUC-CSR	SUC-HI	FA--SI-VALVETM	BBPM---1PP7WPS				
99.	5.84E-08	8	IEV-SLO	SUC-HP2	SUC-OA6	SUC-CSI	SUC-CSR	SUC-HI	IA--RH-VALVETM	BBPM---1PP7WPS		
100.	5.80E-08	10	IEV-SGR	SUC-AF2	SUC-HPI	SUC-SGI	SUC-CSI	SUC-CSR	SUC-HI	X-CM--1OME41PR	X-CM--2OME41PS	E-AM----SV32FC

TABLE 3.4-4

CDF.IMP

COMPLNK Version 2.20 11/ 2/1991 20:36:31

File created by linking CDF.WLK WLINK Ver. 2.31

SYSTEM UNAVAILABILITY (Q) = 6.26E-05
NUMBER OF BASIC EVENTS = 520
NUMBER OF CUTSETS = 3228

	BASIC EVENT	IMPORTANCE	# CS	CONT. TO Q	FAILURE PROBABILITY
1	SUC-HI	98.13	3114	6.148E-05	9.998E-01
2	SUC-CSI	97.95	3065	6.137E-05	9.994E-01
3	SUC-CSR	96.48	2658	6.044E-05	9.987E-01
4	IEV-SLO	47.29	1290	2.963E-05	6.800E-03
5	SUC-HP2	33.90	1279	2.124E-05	9.983E-01
6	SUC-OA6	24.43	1110	1.531E-05	9.983E-01
7	IEV-CCW	22.09	162	1.384E-05	8.710E-04
8	RCP	18.83	9	1.179E-05	1.300E-02
9	G-HHRS------CM	15.82	52	9.913E-06	1.150E-03
10	IEV-SGR	11.28	303	7.067E-06	7.200E-03
11	SUC-AF2	10.88	281	6.814E-06	9.999E-01
12	SUC-HPI	10.82	274	6.777E-06	9.997E-01
13	SUC-SGI	10.63	231	6.660E-06	9.975E-01
14	X-CM--2OME41PS	10.40	185	6.517E-06	8.440E-02
15	X-CM--1OME41PR	10.24	97	6.416E-06	4.800E-03
16	SUC-SSV	9.57	133	5.995E-06	9.032E-01
17	F-HP2-------CM	7.53	46	4.717E-06	6.780E-04
18	SUC-ACC	7.23	802	4.532E-06	9.994E-01
19	IEV-MLO	6.89	587	4.317E-06	9.170E-04
20	IEV-ATW	4.55	115	2.853E-06	4.670E-05
21	RPL	4.52	99	2.834E-06	8.400E-01
22	SUC-MRI	4.44	97	2.781E-06	9.986E-01
23	SUC-AF1	4.28	256	2.683E-06	9.999E-01
24	SUC-RCP	4.23	242	2.651E-06	9.870E-01
25	SUC-CH1	3.98	153	2.491E-06	9.987E-01
26	SUC-AMS	3.90	96	2.443E-06	9.900E-01
27	SUC-AFH	3.83	66	2.401E-06	9.989E-01
28	UE1	3.39	10	2.123E-06	5.480E-02
29	Q-CC-SI-SIGNAL	3.28	43	2.056E-06	2.000E-04
30	X-CC-CONAIR	3.03	25	1.896E-06	1.250E-04
31	X-CC-CONAIR3	2.84	27	1.779E-06	2.180E-04
32	SUC-RRI	2.72	212	1.706E-06	9.990E-01
33	IA--RH-VALVETM	2.22	140	1.389E-06	7.020E-03
34	BBPM---1PP7WTM	2.03	164	1.271E-06	1.480E-03
35	ESW-MULT1-7W	2.03	164	1.271E-06	2.000E+00
36	IEV-SBO	1.81	26	1.132E-06	1.400E-05
37	X-AM----SV44CO	1.74	20	1.092E-06	7.200E-05
38	SUC-RCD	1.71	18	1.073E-06	9.984E-01
39	Z-TK-SLOWCSTHE	1.64	111	1.030E-06	5.300E-05
40	SUC-AFT	1.64	17	1.028E-06	9.411E-01
41	SUC-AFC	1.63	14	1.022E-06	9.943E-01
42	IEV-LLO	1.51	227	9.437E-07	3.000E-04

TABLE 3.4-4 (con't)

CDF.IMP

43	IA--RH-TRAINTM	1.49	114	9.357E-07	4.740E-03
44	IAPM---PP35EPS-R	1.47	118	9.185E-07	3.330E-03
45	FA--SI-VALVETM	1.25	65	7.848E-07	7.020E-03
46	Q-CC-HPR-RELAY	1.23	14	7.695E-07	8.940E-05
47	IB--RH-VALVETM	1.18	73	7.383E-07	7.020E-03
48	F-CV---SI185FO	1.17	14	7.312E-07	1.000E-04
49	FBCV---SI101FO	1.11	11	6.939E-07	1.000E-04
50	FA--CC-TRAINTM	1.09	95	6.848E-07	6.150E-03
51	IBPM---PP35WPS-R	1.09	58	6.837E-07	3.330E-03
52	BBMV-1WM0702FO	1.03	119	6.429E-07	1.510E-03
53	BBMV--WM0737FO	1.00	92	6.256E-07	1.510E-03
54	CBMV--CM0420FO	1.00	92	6.256E-07	1.510E-03
55	IEV-SWS	.96	89	6.041E-07	3.730E-05
56	IEV-VDC	.96	26	6.037E-07	1.160E-02
57	SUC-LPI	.94	196	5.899E-07	9.994E-01
58	OA6-DIAG-MN-HE	.92	11	5.794E-07	6.130E-04
59	QXSI	.92	18	5.752E-07	5.700E-05
60	IEV-SLB	.91	152	5.688E-07	3.300E-04
61	Q-CC-HP2-RELAY	.90	11	5.660E-07	8.150E-05
62	DNPT-----PP4PS	.89	137	5.589E-07	1.100E-01
63	RRI-DIAG-MN-HE	.89	4	5.548E-07	6.130E-04
64	FAPM---PP26NPS	.86	49	5.393E-07	2.990E-03
65	BBPM---1PP7WPS	.83	105	5.227E-07	1.230E-03
66	SUC-EH8	.79	5	4.978E-07	9.935E-01
67	GAPM---PP26NPS	.78	77	4.898E-07	3.700E-03
68	E-CC-SGPORV	.71	13	4.445E-07	5.450E-05
69	IAMV--IM0310FC	.69	112	4.329E-07	1.510E-03
70	LAMV--IM0215FC	.69	112	4.329E-07	1.510E-03
71	SUC-XH1	.69	5	4.293E-07	9.557E-01
72	IAMV--ICM305CC	.68	103	4.284E-07	1.510E-03
73	SBO-V-TR-CK-HE	.68	8	4.267E-07	5.100E-01
74	CAMV--CM0419CC	.66	83	4.117E-07	1.510E-03
75	IAMV--IM0340CC	.66	83	4.117E-07	1.510E-03
76	SN1	.65	1	4.050E-07	3.300E-02
77	AMS	.63	3	3.917E-07	1.000E-02
78	SBO-I-TR-CK-HE	.62	6	3.886E-07	5.460E-02
79	IBMV--IM0320FC	.52	66	3.247E-07	1.510E-03
80	LBMV--IM0225FC	.52	66	3.247E-07	1.510E-03
81	IBMV--ICM306CC	.51	61	3.221E-07	1.510E-03
82	CBMV--CM0429CC	.49	47	3.083E-07	1.510E-03
83	IBMV--IM0350CC	.49	47	3.083E-07	1.510E-03
84	SUC-LPR	.48	98	3.031E-07	9.987E-01
85	FBPM---PP26SPS	.48	19	3.013E-07	2.990E-03
86	FA--SI-TRAINTM	.48	44	3.003E-07	2.710E-03
87	IEV-VEF	.48	2	2.991E-07	3.000E-07
88	IEV-TRA	.46	34	2.852E-07	3.800E+00
89	IEV-TRS	.45	104	2.825E-07	1.200E-01
90	SUC-HP3	.41	27	2.542E-07	9.990E-01
91	FB--SI-VALVETM	.38	23	2.398E-07	7.020E-03
92	QBREK610X1K	.37	69	2.338E-07	5.700E-04
93	FAMV--QM0225OO	.37	56	2.299E-07	1.510E-03
94	IB--RH-TRAINTM	.37	49	2.291E-07	2.190E-03
95	QBREK604K	.37	59	2.287E-07	5.700E-04

3.4.2 Vulnerability Screening

The information summarized in Tables 3.4-1 and 3.4-2 form the bases for sequence screening and identification of vulnerabilities. Table 3.4-2 shows the sequences for each accident starting with the most dominant accident (small LOCA) and proceeding in descending order of contribution to core damage frequency (CDF). As seen in Table 3.4-1, small LOCA (SLO), Loss of Component Cooling Water (CCW) and Steam Generator Tube Rupture (SGR) are the top three contributing accidents to CDF with contributions of 47.3% (SLO), 22.1% (CCW) and 11.3% (SGR).

For the SLO event, failure of the Emergency Core Cooling System (ECCS) during either the cold leg injection or recirculation phases produced the two leading sequences within the SLO event (Table 3.4-2.) Common mode failure of the safety injection (SI) pumps (part of ECCS) and failure of the Engineered Safety Features (ESF) system to actuate the ECCS dominated these two sequences. The third leading sequence resulted from functional failures to cool down the reactor coolant system followed by failure to initiate primary feed and bleed cooling. Hardware and common mode failures in the compressed air system, which supplies air to the pressurizer and steam generator PORVs, contributed mostly to these failures. Also note that all three sequences are within the upper 95% of CDF.

The Loss of CCW event, like SLO, was dominated by three sequences (see Table 3.4-2). Operator failure to trip the reactor coolant pumps (RCPs) after losing seal cooling from CCW, which leads to gross seal failure, solely controlled this sequence. Within the second sequence, cold leg recirculation dominated by a common mode failure of the SI pumps again turned up, as in SLO. The third sequence was controlled by the functional failure to restore reactor inventory (RRI) after CCW was restored (which then allows use of the ECCS charging pumps). Here, operator error and ESF signal failures dominated. Only the first sequence (trip RCPs) is within the upper 95% of CDF.

The SGR event had multiple sequences contributing to it. Rather than discuss each sequence (which is shown in Table 3.4-2), it is noted that compressed air system failures (hardware and common mode) and ESF signal trouble again appeared as dominant. Additionally, the top sequence was influenced by steam generator PORV common mode failures.

The SGR event, in addition to the interfacing systems LOCA (ISL) event, is also significant since the containment is bypassed. The ISL event is also shown in Table 3.4-2. Per NUREG-1335, sequences from these accidents greater than 1.0E-08 were reviewed. The ISL event was the least contributor to CDF. Even so, these events are important since fission product source terms are directly released to the environment.

Table 3.4-4 lists an overall importance ranking for the CDF quantification. As expected, those individual items that contributed to the SLO, CCW and SGR events are most important. The failure to trip the reactor coolant pumps after losing seal cooling and the plant air compressor failures are significant, next to the ECCS related common mode failures. The reactor coolant pumps are important since gross seal failure was assumed upon a loss of seal cooling. The plant air compressors are important because the pressurizer and steam generator PORVs need control air to operate to allow secondary side cooling and primary feed and bleed. The Unit 1 control air compressor and backup PORV air supplies were not modeled due to either capacity or availability concerns. Following further sensitivity analysis on the control air system and the SLO event tree, it was found that had a more refined model been used within the SLO event tree, it would have noticeably lowered the SLO contribution to CDF and reduced the significance of the control air system failures.

3.4.3 Decay Heat Removal Evaluation (USI A-45)

This section provides a brief evaluation of the decay heat removal functions at the Cook Nuclear Plant. The purpose of this section is to examine whether or not the risks associated with a loss of decay heat

removal can be lowered in a cost effective manner. As stated in NUREG-1289 (Reference 64), this issue is concerned with small break (less than 6" equivalent diameter) LOCA, transient, and loss of offsite power events.

Decay heat removal during the first 24 hours following a plant trip is accomplished by the following key functions at Cook Nuclear Plant:

- During a medium (2" to 6" diameter) LOCA event, decay heat is removed directly by the Emergency Core Cooling Systems (ECCS). This includes the charging, safety injection (SI), and residual heat removal (RHR) injection and recirculation systems, and associated operator actions.

- During a small (3/8"to 2") break LOCA (SBLOCA) event, there will not be enough flow from the break to directly remove adequate decay heat. For SBLOCAs, decay heat is removed through the secondary side of the steam generators by the auxiliary feedwater (AFW) or main feedwater (MFW) system while reactor coolant system (RCS) inventory is maintained by ECCS injection and recirculation. If the decay heat removal through the secondary side systems fails, bleed and feed operations are needed on the primary side. The bleed and feed operation requires a high pressure injection system, the pressurizer PORVs, and associated operator actions.

- During a transient event (including a loss of offsite power event), decay heat is removed through the secondary side of the steam generators by the AFW and MFW systems. Unlike the SBLOCA event, however, RCS inventory make up is not required. Should the secondary systems fail, then primary bleed and feed operations would be initiated.

Evaluation of USI A-45 was addressed for both internal and external initiating events. These evaluations were based on the results of the Cook Nuclear Plant IPE and IPEEE and are summarized below.

3.4.3.1 Internal Events

Event trees were developed by the Cook Nuclear Plant IPE to model the most important events and systems necessary to mitigate the above listed accidents. As stated in Section 3.1.2 of this submittal, the success criteria for the top events modelled within the event trees also considered the effect of the top event on both core heat removal and containment response. In some cases, the success criteria specified in the event trees were more restrictive than required to solely remove decay heat because of the consideration of the containment response.

Given that successful decay heat removal depends on the above systems and operations, the following discussion of these functions and their respective features is provided.

- The AFW system consists of three redundant trains: two trains contain a motor driven pump and the third contains a turbine driven pump. Each of the motor-driven AFW pumps feed two steam generators each. The turbine-driven AFW pump is capable of feeding all four steam generators. The normal water supply to the AFW is from the condensate storage tank (CST) and can be supplemented with essential service water or with the opposite unit's CST. These backup supplies are modeled in the Cook Nuclear Plant IPE AFW system fault trees since the CST may not contain enough water to remove decay heat for a full 24 hours.

  The failure probability of the AFW system was calculated to be 6.69E-05 with all support systems available. Therefore, the failure of the AFW system was found to be an insignificant contributor to the core damage frequency.

- If the affected unit's AFW system is not available, the operators are directed to attempt to crosstie to the opposite unit's motor-driven AFW pumps. It is possible for each of the opposite unit's motor driven AFW pumps to supply two of the affected unit's steam generators. The failure probability of the AFW crosstie was calculated to be 2.43E-04. Considering that the crosstie is only used if the

affected unit's AFW system fails, this failure probability is considered sufficiently small to have an insignificant impact on the core damage frequency.

- o If it is not possible to crosstie the AFW systems, the operator then attempts to establish an alternate feedwater flow to the steam generators via the main feedwater pumps.

  The MFW system includes the following components: two turbine-driven main feedwater pumps, two main feedwater heater strings, and two main feedwater pump condensers. In addition, the MFW system requires cooling from the condensate system and the circulating water system.

  The MFW system is required to operate when both units' AFW systems fail. As described above, this is very improbable. The failure probability of the MF1 event, which includes the failure of the MFW and condensate systems, was calculated to be a relatively low value of 1.66E-04.

- o With no feedwater flow to the steam generators, the operators are instructed per EOP OHP-4023.FR-H.1 (Reference 4) to initiate primary side bleed and feed cooling. The cooling path is feeding from the high head ECCS (charging and SI) system and bleeding from the pressurizer PORVs.

  Two of the three pressurizer PORVs are required for the bleed operation in order to prevent overpressurization of the RCS during various accident conditions. The high head ECCS was modelled in the level I quantification as requiring one of two SI pumps and one of two charging pumps to provide flow to the RCS. The requirement for two pumps was due to consideration of the containment response. For decay heat removal purposes, however, only one of four ECCS pumps would be sufficient (Reference 4).

  The bleed and feed operation is used only when both the AFW and MFW fails, which is very improbable. The operator diagnosis and action was calculated to have a failure probability of 1.86E-04. The failure probability of the entire bleed and feed operation as modelled in the IPE is 2.19E-03.

- o The high head injection system of the ECCS consists of 2 SI and 2 charging pumps. The failure probability of this system (HP2) was calculated as 1.2E-03.

- o The low pressure injection system of the ECCS consists of 2 RHR pumps which can be used when the RCS pressure is below the shutoff head of the RHR pumps. The failure probability of this system (LPI) was calculated to be a relatively low value of 3.1E-04.

- o When the low level RWST alarm setpoint is reached, the operators transfer from the injection to the recirculation phase based upon EOP OHP-4023.ES-1.3 (Reference 4). The operator failure to perform a transfer to ECCS recirculation is included in the high head and low head recirculation models. The failure probability of this operator action for high head ECCS recirculation is 6.31E-06 and the failure probability for the low head recirculation is 5.31E-05.

- o The high head recirculation system (HPR) consists of the 2 SI and 2 charging pumps which can be used to maintain the plant in a long-term stable condition for sequences with the RCS pressure above the RHR pump shutoff head. This mode of operation requires suction from the RHR pumps. The failure probability of this system was calculated as 1.3E-03.

- o The low head recirculation system (LPR) consists of 2 RHR pumps which can be used to maintain the plant in long-term stable condition for sequences with the RCS pressure less than the RHR pump shutoff head. The failure probability of this system was calculated to be a relatively low value of 8.5E-04.

The dominant accident sequences for plant core damage are discussed in detail in Section 3.4. These accident sequences were the product of the level I quantification, which linked support systems into the

event tree top event nodes. Thus, failure values associated with these sequences account for support system (component cooling water, essential service water and safety related electrical systems) contributions to failure.

Of the items in Section 3.4, those which are applicable to failure of the decay heat removal function and not considered just for the effect on containment response and whose frequency is greater than 1E-6/yr are presented below. Note that the listed core damage frequency corresponds with the information presented in Section 3.4.

Core Damage Frequency	Initiating Event	System Failure(s)
1.35E-05	SLO	HPR fails due to common cause failures of the HP system during recirculation phase
1.17E-05	SLO	HP2 fails due to common cause failures of the HP system during injection phase
1.63E-06	CCW	HPR fails due to common cause failures
1.80E-06	MLO	HPR fails due to common cause failures

As presented above, the ECCS system is the dominant contributor to the failure of the decay heat removal function at the Cook Nuclear Plant. The table below presents the results of an importance analysis of the dominant contributors to the Cook Nuclear Plant core damage frequency which are relevant to the failure of the decay heat removal function.

Importance Contribution	Event Description
16%	HPR - Common Cause Failures
8%	HP2 - Common Cause Failures
3%	ESFAS Signals - Common Cause Failures

As discussed in Section 3.3.4, the common cause failure probabilities for the ECCS system are considered to be conservative due to the application of generic multiple greek letter (MGL) factors and the method in which all common cause failures are lumped into one failure event. It is expected that the HP2 and HPR contributions to core damage would decrease if Cook Nuclear Plant specific MGL factors were analyzed and separated out by similar components instead of grouping the failure probabilities of all components within the system into one common cause failure event.

To summarize the information presented in this section, there are several redundant means for decay heat removal at the Cook Nuclear Plant. All of the decay heat removal functions have relatively low failure probabilities and those which do contribute significantly to the plant core damage frequency are either conservatively calculated (high pressure common cause) or, as with the case of the control air system, are being reviewed.

3.4.3.2 Internal Flooding

Internal flooding vulnerabilities at Cook Nuclear Plant were analyzed using an internal flooding PRA. Consistent with the internal initiating events previously discussed, plant systems and the expected plant responses to various flooding events were analyzed using quantitative techniques. AEPSC calculations and

detailed plant walkdowns were used to determine those areas in the plant that were vulnerable to internal flooding. All areas inside the plant were analyzed to determine the likelihood of accident initiation. The only event of significant probability found to require further analysis was a transient without steam conversion systems available caused by a flood in the turbine building sub-basement.

Consistent with the internal events quantification, the initiating event was calculated and the support systems were linked into the accident events. The event tree used to quantify the effects of internal flooding was based on the internal events event/fault trees. Therefore, only the internal flooding dominant contributors impacting decay heat removal are discussed here.

The internal flooding core damage frequency for Cook Nuclear Plant is 2.00E-07/year. The dominant contributors are failure of the NESW pumps due to the flood, and the subsequent failure of the compressed air system due to loss of NESW cooling to the compressors. Failure of the compressed air system will result in failure of the ability to provide primary bleed and feed cooling, one of the methods of decay heat removal. As discussed in section 3.4.3.1, the AFW system and the crosstie of AFW flow to the opposite unit's AFW system are the primary means of decay heat removal for transients. The internal flooding PRA determined that these are not affected by flooding at Cook Nuclear Plant. Additionally, availability of the ECCS is not affected by internal flooding. It is, therefore, concluded that flooding effects on decay heat removal systems at Cook Nuclear Plant are not a significant concern.

3.4.3.3 External Events

Seismic Contributors:

The seismic portion of this project was addressed by modelling plant systems and seismically initiated accident events in a seismic PRA (SPRA) that evaluated seismic levels beyond the 0.2g design basis earthquake (DBE) for Cook Nuclear Plant. This process was very similar to the internal events sequence quantification described in Section 3.3.5. Transients (feedwater failure), LOCAs (small, medium and large) and Loss of Offsite Power events were among the accident events analyzed in the SPRA. Plant walkdowns, which reviewed spatial interactions and equipment mountings, and a component failure (fragility) analysis provided input to the SPRA.

As with the internal events quantification, support system contributions were linked into the accident events. The SPRA event trees and plant system fault trees were based on the internal events event/fault trees (modified for seismic events). Therefore, only the seismic dominant contributors impacting decay heat removal are discussed here.

The seismic core damage frequency based on a site specific seismic hazard curve is 1.83E-05. The dominant seismic contributors affecting decay heat removal are Auxiliary Building failure, 4kV/600V AC transformer failure and emergency diesel generator (EDG) fuel oil day tank failures. These dominant contributors are currently designed to withstand a 0.2g DBE at Cook Nuclear Plant and failed at levels higher than 0.2g in the SPRA.

Auxiliary Building failure was conservatively assumed to destroy all components inside, which included the ECCS pumps, component cooling water system and much of the safety related electrical power system. Additionally, fuel oil day tank and electrical transformer failures were governed by failures of adjacent walls, walls which meet DBE criteria. The very conservative assumption of auxiliary building failure coupled with electrical failures became dominant beyond the 0.2g DBE. Thus, it is concluded that seismic effects on decay heat removal systems at Cook Nuclear Plant are not a significant concern.

Internal Fire Contributors:

Internal fire vulnerabilities at Cook Nuclear Plant were analyzed using an internal fire PRA. As for the internal initiating events previously discussed, plant systems and the expected plant responses to various fires were analyzed using quantitative techniques. Internal AEPSC documentation supporting the work

performed in response to 10CFR50 Appendix R and detailed plant walkdowns were used to determine those areas in the plant that were vulnerable to internal fires. All areas inside the plant were analyzed to determine the likelihood of accident initiation. The only event of significant probability found to require further analysis was a Loss of a Single Train of 250 V DC Power. The only zones requiring extensive quantification were the zones housing the Engineered Safety Features switchgear and the cable vaults.

As with the internal events quantification, the initiating event was calculated and the support systems were linked into the accident events. The event tree used to quantify the effects of internal fire was based on the internal events event/fault trees. Therefore, only the internal fire dominant contributors impacting decay heat removal are discussed here.

The internal fire core damage frequency for Cook Nuclear Plant is 1.65E-07/year. The dominant contributors are failures of electric power buses that are destroyed by the fire. The contribution to core damage frequency is low and, therefore, it is concluded that fire effects on decay heat removal systems at Cook Nuclear Plant are not a significant concern.

Other External Events Contributors:

This section discusses those vulnerabilities attributable to any external event other than seismic events, internal floods, and internal fires. Specifically examined in the other external events analysis are external flooding, aircraft accidents, severe winds, ship impact accidents, off-site and on-site hazardous materials accidents, turbine missiles, and external fires. An approach which meets the intent of that displayed in Figure 1 of Generic Letter 88-20, Supplement 4 (Reference 9) was used in the evaluation of other external events. Existing information and analyses were utilized as much as possible to analyze the subject events. No vulnerabilities were identified that required detailed quantification of any accident events. It is, therefore, concluded that the effects on decay heat removal systems from any of the other external events described here are not a significant concern at Cook Nuclear Plant.

3.4.4 USI and GSI Screening

The IPE and IPEEE (External Events) addressed the following issues:

1. USI A-45, Shutdown Decay Heat Removal Requirements.

2. USI A-17, System Interactions in Nuclear Power Plants.

3. NUREG/CR-5088, "Fire Risk Scoping Study".

4. GI 131, Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants.

5. The Eastern U.S. Seismicity (The Charleston Earthquake) Issue.

USI A-45 is addressed in Section 3.4.3. USI A-17 was addressed through both IPE and IPEEE plant walkdowns. The remaining issues are addressed in the IPEEE submittal for Cook Nuclear Plant.

4.0 BACK END ANALYSIS

The following sections describe the back end analysis of the Cook Nuclear Plant IPE. The arrangement of this portion of the submittal differs slightly from the order specified in Reference 20. Because of the way in which the Cook Nuclear Plant Level 2 analysis was conducted, this rearrangement of sections was made to make the submittal follow a clear, logical pattern.

4.1 Containment Analysis

4.1.1 Containment Description

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

4.1.1.1 Containment Structure

The Cook Nuclear Plant containment is a Westinghouse ice condenser design. The plant containment is sectioned into several compartments consisting of a total free volume of approximately 1,300,000 cubic feet. Figure 4.1-1 illustrates a vertical section of the Cook Nuclear Plant containment. The large containment volume above the operating deck is referred to as the "Upper Compartment." The "Lower Compartment" is that portion of containment which is inside the crane wall but outside the biological shield wall and is between the containment floor and the operating deck. The "Annular Compartment" is that part of containment below the operating deck but outside the crane wall. The "Cavity" includes the reactor cavity and the instrument tunnel. Separate compartments are modeled for the "Ice Condenser" and "Ice Condenser Upper Plenum."

Figure 4.1-2 is a schematic of the Cook Nuclear Plant containment compartments, showing their free volumes as modelled in the level 2 analysis and the cross sectional area associated with the flow paths between these sections. The floor of the lower compartment is at elevation 598'-9". Water begins to spill from the lower compartment into the reactor cavity at elevation 610'-0". As shown in Figure 4.4-1, the lower and annular compartments are separated by the crane wall with large ventilation openings allowing communication between the two compartments. The top of these ventilation openings is at elevation 612'-0". In addition, there are smaller openings through piping sleeves which allow system piping to pass from the annular compartment to the lower containment. While the flow area of each of these sleeves is small, there are numerous sleeves and it is believed thet these openings will allow water level in the lower and annular compartments to equalize. Floodup level in the Cook Nuclear Plant containment is to approximately 614' if the entire contents of the RWST, accumulators, reactor coolant system (RCS), and ice condenser are considered. Because of the piping sleeves mentioned above, injection of the RWST alone will probably not allow water to spill from the lower compartment into the cavity until sufficient ice has melted to raise the water level in both the lower and annular compartments to the 610'-0" elevation.

The ice condenser containment is designed as a vapor suppression system. Ice is a one-time, non-renewable vapor suppressant. Ice mass maintained during normal operation is typically greater than 2.37×10^6 lbm. The physical design of the ice condenser containment is that all high energy piping is contained in the portion of containment below the operating deck. This ensures that all energy released from any pipe breaks is released to the lower compartment. Energy released from accidents in the lower compartment forces a mixture of air, water and steam into the ice condenser. Here the steam is condensed through contact with the ice surface, limiting containment peak pressures. Air continues to flow through the ice condenser and into the upper compartment. Recirculation fans return the cooled air to the annular compartment where it flows back to the lower compartment.

The reactor containment structure is a reinforced vertical concrete cylinder with a slab base and a hemispherical dome. A welded steel liner is provided as a membrane to prevent leakage. The ice condenser runs circumferentially (300°) between the crane wall and the outer containment wall, extending above and below the operating deck. Door panels are located at the upper portion of the ice condenser above the operating deck and at the bottom of the ice condenser. These doors open when an accident creates sufficient differential pressure between the lower and upper compartments. A detailed description of the Cook Nuclear Plant containment can be found in chapter 5 of the UFSAR.

The Cook Nuclear Plant containment structure acts as a fission product barrier. However releases from the containment may occur during severe core melt accidents due to either preexisting leakage pathways or breaches caused by structural failures. Containment structural failures may result due to containment overpressurization following depletion of the ice condenser inventory. Containment failure modes are summarized in Section 4.3 of this submittal and are based on phenomenological evaluation summaries completed as part of the Source Term Notebook. The latter are discussed in the following section. As discussed in Section 4.3, containment failure in the annular compartment at the basemat/cylinder junction is assumed for use in the source term analysis and some of the water on the floor of the containment is assumed to leave containment through this break. The containment would continue to pressurize while water is discharged through the break, but would subsequently depressurize once the water on the floor was depleted.

4.1.1.2 Containment Systems

The Cook Nuclear Plant containment design relies on containment spray recirculation for long term containment heat removal. Containment spray injection is only a short term containment pressure reduction system. The containment spray system can be supplemented by the RHR sprays by diverting the recirculation flow of the RHR system from the core to the RHR spray headers. The hydrogen igniter system and the containment recirculation fans help prevent the accumulation and stratification of hydrogen in containment. Significant design aspects of these systems are discussed briefly below.

<u>Containment Spray Injection and Recirculation</u>

The containment spray system provides spray cooling water to the containment atmosphere via spray ring headers located on the containment dome and in the lower and annular compartments. The system consists of two independent 100% capacity flow trains, each of which is designed to provide 3200 gpm to the ring headers. The containment spray system is illustrated schematically in Figure 4.1-3. A single refueling water storage tank (RWST) and spray additive tank serve the two trains. Additional independent ring headers in the upper compartment are supplied by the residual heat removal (RHR) system and can be used to supplement the containment spray system. In order to allow spray water in the upper compartment to return to the lower compartment, there are three drain holes in the refueling canal with a combined flow area of about 2.2 ft^2. This direct flow from the lower to the upper compartments is the only direct communication between these compartments.

Operation of the containment spray system is automatically initiated when containment pressure increases above 2.9 psig. Both containment spray pumps are started and water is pumped from the RWST to the spray nozzles. When the RWST low level alarm is received, transfer to recirculation will be initiated by the control room operators. The operators must stop the operating containment spray and RHR pumps, close the valves in the suction line to the RWST, verify cooling water to the containment spray heat exchangers, open the isolation valves from the recirculation sump, then restart the previously operating pumps. In the recirculation phase, water that is injected by containment spray and spilled from the break collects in the lower containment and flows to coarse and fine mesh strainers into the recirculation sump. This allows the water to be cooled during spray recirculation. Each containment spray system heat exchanger is designed to remove 107.8 MBtu/hr.

The RHR system can be used during the recirculation phase to supplement the containment spray system. This operation would be initiated by the operators in the event containment pressure increases to 8 psig after the initial blowdown. An off-take from each of the two RHR trains, downstream of the RHR heat exchangers, supplies 2,000 gpm to two RHR spray ring headers in the upper compartment. Flow from a train of the RHR system to the core must be terminated before the RHR spray is initiated. Allowing an RHR pump to simultaneously supply the core, centrifugal charging and safety injection pump suctions, and the RHR spray headers is prohibited as this could result in RHR pump runout.

Distributed Ignition System (DIS)

The containment is provided with a distributed ignition system (i.e., hydrogen igniters) for post accident hydrogen control in order to minimize hydrogen buildup in the containment atmosphere. There are two trains of igniters employing a total of seventy igniter assemblies (glow plugs) located throughout the containment building. Once energized by the control room operators, the igniters cause any combustible mixtures near the igniter to ignite, thereby helping to support continuous burns and eliminate any large hydrogen accumulations.

Air Recirculation/Hydrogen Skimmer System

The Air Recirculation/Hydrogen Skimmer System is the only safety related ventilation system in the containment. This system is activated by a containment pressure high-high signal (i.e., when containment pressure reaches 2.9 psig). The system consists of two independent systems which include fans, back draft dampers, valves, piping and ductwork.

The primary function of this system is to prevent hydrogen pocketing and stratification. In addition, the system aids in maintaining containment pressure within the limits of accident analyses. The system operates by continuously blowing air from the upper containment through the annular compartment into the lower containment. Hydrogen pocketing is prevented by drawing air out of potential pocketing areas into the fan suction.

Both air recirculation/hydrogen skimmer trains have an air recirculation fan located at El. 629' in the upper containment. Each system has a total capacity of 41,800 cfm. The fans discharge via the annular space between the crane wall and the containment liner into the lower containment. Both trains are automatically actuated by the high-high containment pressure signal after a ten minute delay. This delay is to allow the transient of the initial blowdown to complete.

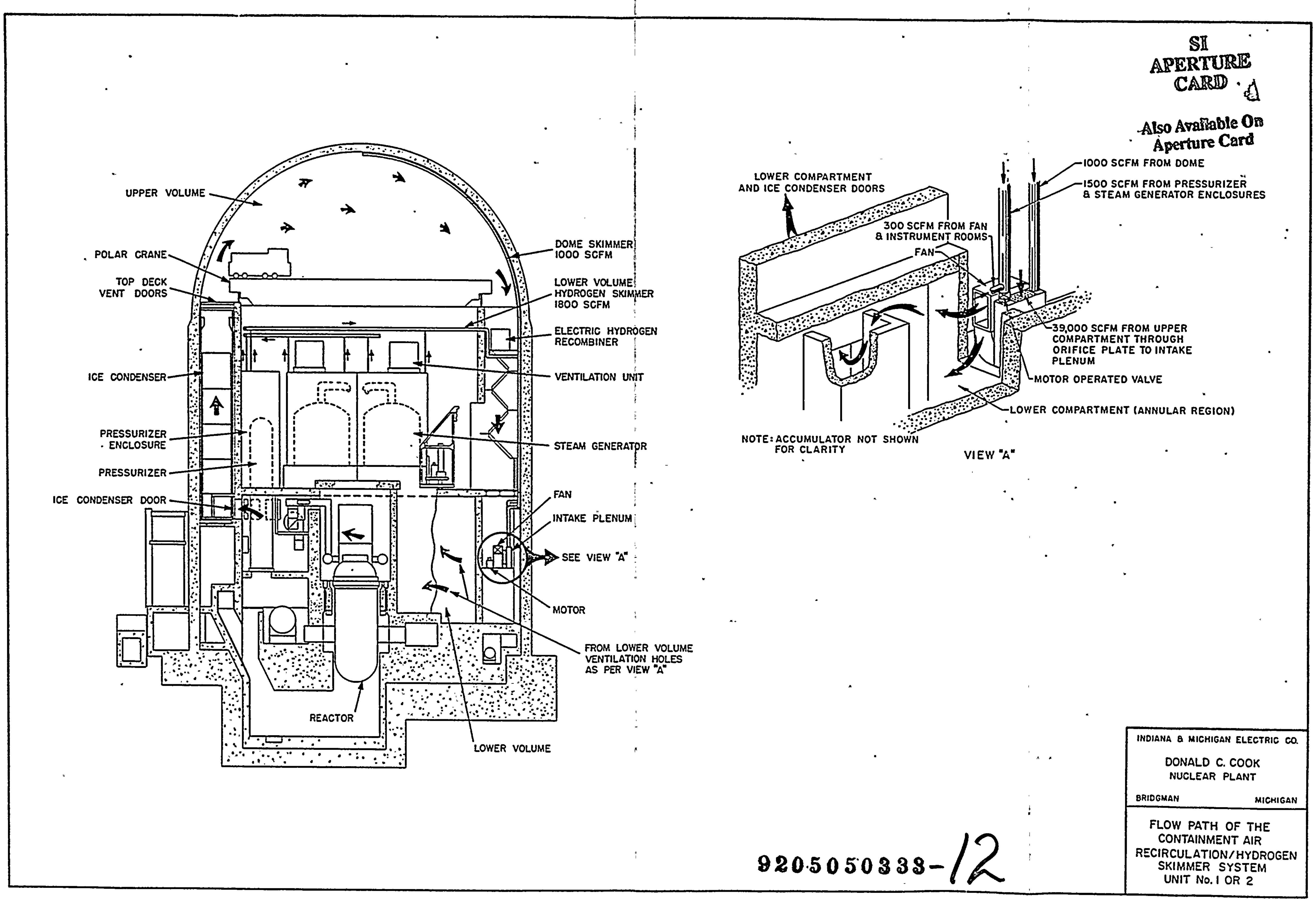


Figure 4.1-1 Air recirculation flow path through D. C. Cook ice condenser. (Ref. FSAR Fig. 5.5.3).

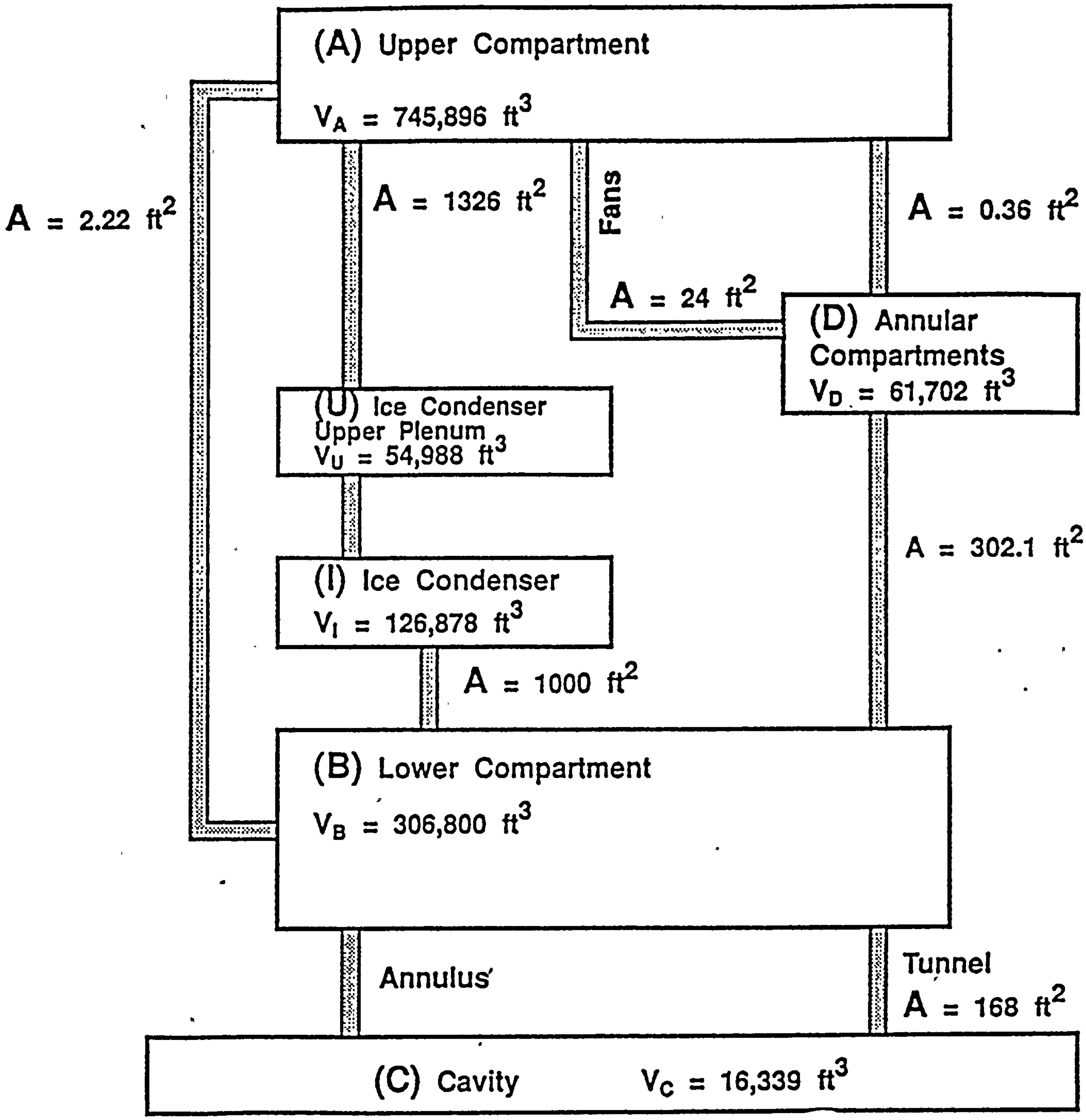


Figure 4.1-2 D. C. Cook containment volumes and flow paths.

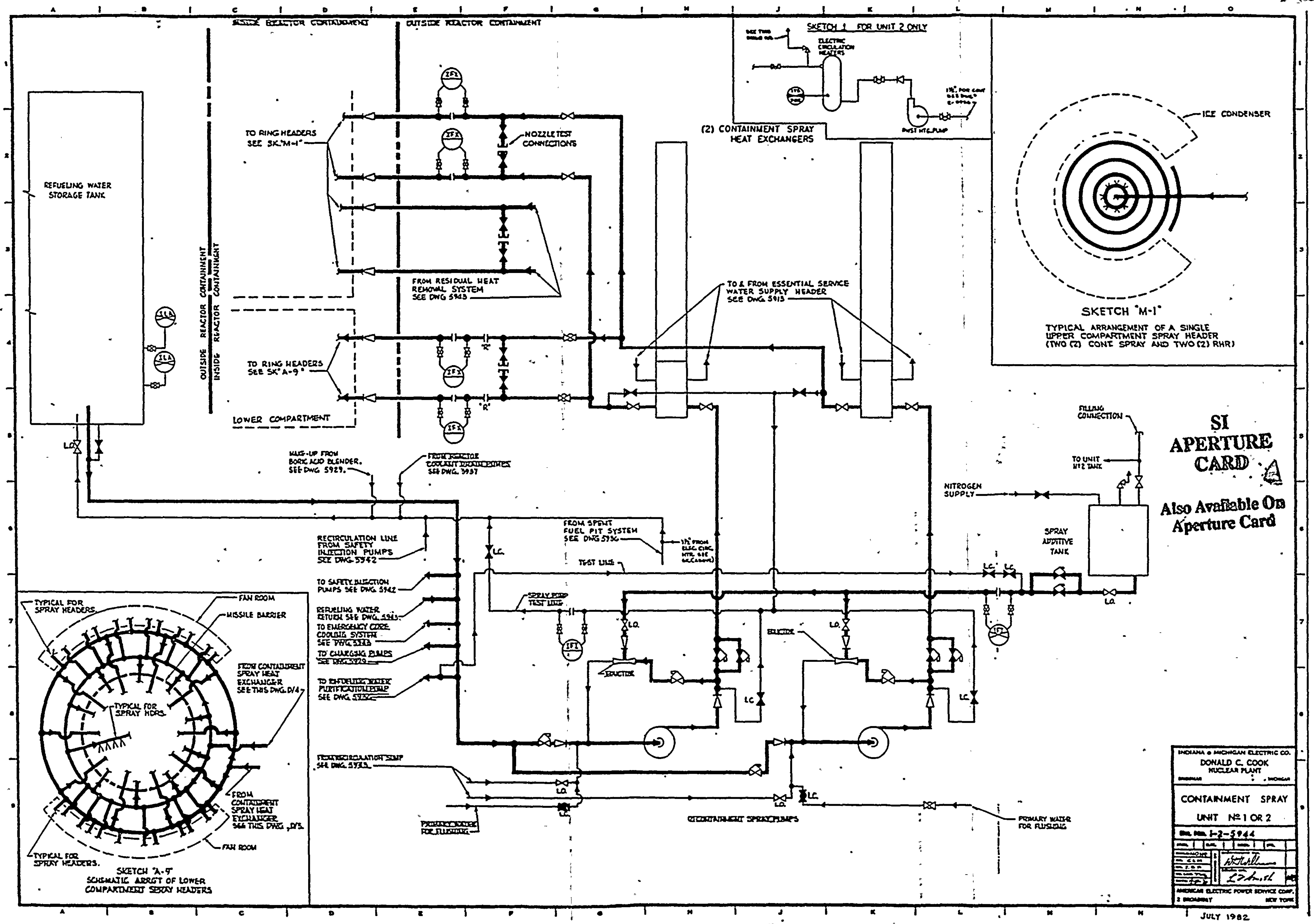


Figure 4.1-3 D. C. Cook containment spray system, taken from FSAR Figure 6.3-1.

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4.2 Plant Models and Methods for Physical Processes

The Cook Nuclear Plant containment and source term analyses are part of a traditional level 2 PRA analysis. The analysis couples a probabilistic assessment of containment response to postulated initiating events with a physical model which examines the plant response and addresses the impact of phenomenological uncertainties.

Since assumptions regarding key severe accident phenomena may dictate the analysis outcome, due consideration of phenomenological issues is a cornerstone of the present approach to the containment and source term analysis. The IPE addresses the phenomenological issues through plant specific phenomenological evaluations and sensitivity studies. This two-prong approach provides a bounding assessment of source term release timing and magnitude.

The phenomenological evaluation studies are the principle means of addressing the impact of phenomenological issues on plant response. The resultant papers address a wide range of phenomenological issues and provide an in-depth review of plant-specific features which influence the consequences of such phenomena. The phenomenological evaluation summaries are supported by information and experimental results available from open literature. These papers investigate both the likelihood of occurrence and the probable consequences of key severe accident phenomena. The summary papers provide a technical basis for development of the containment event trees. The results of the phenomenological evaluation summaries are described in the subsections below.

The Modular Accident Analysis Program (MAAP) Revision 3.0 version 17.02 (Reference 13) is used in the Cook Nuclear Plant IPE to provide an integrated approach to the modelling of plant and containment thermal hydraulic response and fission product behavior during severe accidents. The plant physical model is defined in a MAAP parameter file which provides MAAP with information required by the code to perform calculations of plant-specific fission product transport and thermal hydraulic response to postulated accident sequences. The parameter file developed for the Cook Nuclear Plant provides a complete, realistic description of the plant for MAAP simulation. The data within the parameter file describing the containment and primary system configuration remains unchanged for all accident sequences.

Sensitivity studies are used to determine which phenomenological issues have a significant impact on the likelihood or timing of containment failure and the magnitude of the source term release. These sensitivity studies are conducted both within the phenomenolgical evaluation papers and by MAAP calculations. In performing MAAP calculations, several model parameters are investigated with respect to the influences of modeling uncertainties on the radionuclide source terms. In particular, uncertainties in the various physical processes are considered as documented in the IDCOR/NRC issue resolution process. The various phenomena and the uncertainties are described in letters from T. Speis of the NRC (References 49, 50, 51). References 9 and 20 also provide summaries of those parameters that have been judged to have a significant effect on containment failure and source terms. Additional guidance on sensitivity studies is provided by EPRI (Reference 52).

The probabilistic models are embodied in containment event tree (CET) described in Section 4.4 of this submittal and the level 1 accident sequence event trees described in Section 3.1 above.

Results obtained with the probabilistic and physical plant models are closely linked. For instance, the CET structure depends on MAAP analyses to 1) define event tree success criteria, 2) establish timing of key events for understanding of sequence progression, and 3) determine the accident sequence outcomes. Furthermore, sequences demonstrated by the quantification task to be dominant contributors to the overall core damage frequency become the basis for MAAP calculations in support of the source term analysis. Finally, MAAP analyses and phenomenological evaluation summaries are used to investigate the effect of phenomenological

uncertainties on the source term assessment. The use of MAAP as described above provides the necessary deterministic complement to the probabilistic analysis. A detailed discussion of the containment event tree models is provided in Section 4.4 of this submittal.

Plant specific phenomenological evaluations have been performed in support of the Cook Nuclear Plant IPE in order to determine the likelihood of all postulated containment failure modes and mechanisms identified in Reference 20. These detailed evaluations were performed to systematically address the controlling physical processes specific to the Cook Nuclear Plant configuration. The results of these evaluations are summarized below.

4.2.1 Containment Overpressurization

A plant-specific structural analysis has been performed for the Cook Nuclear Plant containment (Reference 60) to determine its ultimate internal pressure capacity and its most likely failure locations. The dominant failure mode was identified as bending shear failure in the reinforced concrete containment basemat adjacent to the reinforced concrete cylinder wall. The median pressure capacity at this location was calculated to be 57.8 psig, with the total variability from randomness and uncertainty of 0.14 and 0.14 respectively. In 1991, the potential failure of the containment basemat was reassessed using "as is" material properties. Because the uncertainties in material properties were reduced, the median capacity for this location increased to 66.5 psig. The median failure pressure for the personnel access hatch, equipment hatch, and concrete cylinder are 80.2 psig, 78.4 psig, and 84.0 psig, respectively (Reference 60).

Figure 4.2-1 illustrates the containment fragility curve, or cumulative probability distribution function. The basemat/cylinder junction is the dominant failure location for the Cook Nuclear Plant containment. The pressure capacity that is exceeded with 95% frequency at 95% confidence, considering both inherent randomness about the mean and the uncertainty in the mean itself, is 36 psig (50.7 psia). This value was used for computing containment failure in the level 2 analysis. The reanalysis of containment ultimate capacity completed in 1991 calculated a HCLPF failure of 45.8 psig. While the use of either the median failure pressure or revised HCLPF failure pressure would provide additional time between core melt and containment failure, this additional time would have only a minor impact on the source term analysis. The value of 36 psig was used because the analysis has been previously submitted and reviewed by the NRC staff.

Uncertainties surrounding the exact containment failure size and location are discussed and accounted for in the sensitivity analysis of Section 4.7.

4.2.2 Containment Isolation Failure

Containment isolation failure is a possible containment failure mode at the Cook Nuclear Plant. Containment isolation failure refers to mechanical or operational failure to close containment fluid system penetrations, which communicate directly with the containment or the primary system. This failure may occur prior to or shortly following, the initiation of core damage and would impair the ability of the containment to limit fission product release to the auxiliary building or the environment. Containment isolation would fail on one or more of the following conditions:

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

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

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

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

Failure of containment isolation was considered for lines which meet each of the following screening criteria:

1) the line penetrating containment directly communicates with either the containment atmosphere or the reactor coolant system and it is not part of a closed system outside of containment capable of withstanding severe accident conditions.

   and

2) the line penetrating containment is greater than 2 inches in diameter.

4.2.3 Containment Bypass

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

4.2.4 Direct Containment Heating (DCH)

The relevant experiments for DCH have been reviewed and have produced one specific conclusion: given the necessary RCS conditions for high pressure melt ejection, containment structures (geometry) have a first order (dominant) mitigating influence on the potential for DCH. Mechanistic models for debris dispersal, which take into account entrainment from the cavity and de-entrainment at the tunnel exit were used to evaluate the containment response to a high pressure melt ejection. The evaluation of direct containment heating shows the resulting pressurization expected due to this phenomena would be well below a value that would challenge containment integrity. DCH is not considered as a containment failure mode for the Cook Nuclear Plant.

4.2.5 Steam Explosions

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

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

4.2.6 Molten Core-Concrete Attack

Molten core-concrete attack within the Cook Nuclear Plant reactor cavity is evaluated for the most severe accident sequence (i.e., a large LOCA without RWST injection) using a simple, bounding analysis model which assumes that the concrete ablation rate is proportional to the total heat generation rate due to decay heat and chemical reactions. The model uses empirical parameters determined from available experimental data. The evaluation indicates that melt-through of the containment basemat would not occur until well beyond the 24 hour mission time. Furthermore, containment failure due to overpressurization would occur long before basemat meltthrough. Therefore, molten core-concrete attack is not a likely containment failure mode for the Cook Nuclear Plant.

4.2.7 Thermal Attack of Containment Penetrations

The physical configuration of the Cook Nuclear Plant's ice condenser containment is that there are very distinct compartments within the containment. This compartmentalization results in all mechanical and electrical penetrations being located within the upper and annular compartments. An evaluation of debris dispersal reveals that the majority of entrained debris would be de-entrained at the turn within the instrumentation tunnel exit, while the rest would be confined within the lower compartment region even for a high pressure melt ejection. This is postulated due to the physical configuration of the reactor cavity and because the instrument tunnel has a large opening to the lower compartment but is sealed to the annular compartment. There are, therefore, no direct paths by which corium could contact any containment penetrations. The operational limit of the non-metallic materials are shown not to be exceeded by the maximum gas temperatures predicted for containment compartment regions during severe accident sequences and there were no locations identified during the containment walkdowns where standing hydrogen flames would be expected near penetrations. Hence, thermal loading of penetration non-metallic materials would not cause degradation and leakage from the containment under conditions expected in the containment during a severe accident.

4.2.8 Vessel Thrust Force

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

4.2.9 Hydrogen Combustion

Potential detonability and flammability of the Cook Nuclear Plant containment atmosphere are evaluated as part of the IPE. Detonation was evaluated based on both geometric configuration and detonation cell width scaling. Both of these methods concluded that the likelihood of deflagration to detonation transition (DDT) is very low and was not considered a failure mode in the analysis. It is far more likely that combustible gas would be consumed within containment by deflagration rather than detonation.

Deflagration of hydrogen was evaluated to determine whether the resulting pressure rise would be sufficient to challenge containment integrity. The first step in this evaluation was to determine the mass of hydrogen required to produce a pressure rise sufficient to cause containment failure. Then the conditions required to achieve this amount of hydrogen were examined to determine if such a scenario was probable. The results of the evaluation showed, that even for a total station blackout at the Cook Nuclear Plant, a worst case sequence with respect to hydrogen combustion, it is unlikely that enough hydrogen would accumulate to produce a deflagration that could challenge the containment ultimate pressure capacity. Furthermore, if a scenario did exist which could produce a hydrogen burn of sufficient magnitude to fail the containment, it

is much more likely that the containment would have failed due to slow overpressurization well before such a large amount of hydrogen could accumulate. None of the sequences addressed in the containment and source term analysis could realistically threaten containment due to hydrogen combustion. Hydrogen combustion, therefore, is not considered a failure mode of the Cook Nuclear Plant containment. Because hydrogen combustion is not a likely failure mechanism for the Cook Nuclear Plant, it was concluded that a backup power supply for the hydrogen igniters would provide no noticeable benefit in reducing the frequency or consequences of containment failure.

4.2.10 Summary

In summary, the approach to the assessment of containment response adopted in the Cook Nuclear Plant IPE program links together probabilistic models in the CET with physical plant models contained in MAAP. These models are supplemented through the use of plant specific phenomenological evaluations to provide in-depth technical arguments which reduce phenomenological uncertainties and examine realistic plant response to severe accident phenomena. The evaluations of the phenomenological issues and their potential impact on the source term analysis are summarized in Table 4.2-1.

Table 4.2-1
PHENOMENOLOGICAL EVALUATION SUMMARIES
ON POSTULATED CONTAINMENT FAILURE MODES

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

Table 4.2-1
(Continued)

PHENOMENOLOGICAL EVALUATION SUMMARIES
ON POSTULATED CONTAINMENT FAILURE MODES

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

Table 4.2-1
(Continued)

PHENOMENOLOGICAL EVALUATION SUMMARIES
ON POSTULATED CONTAINMENT FAILURE MODES

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

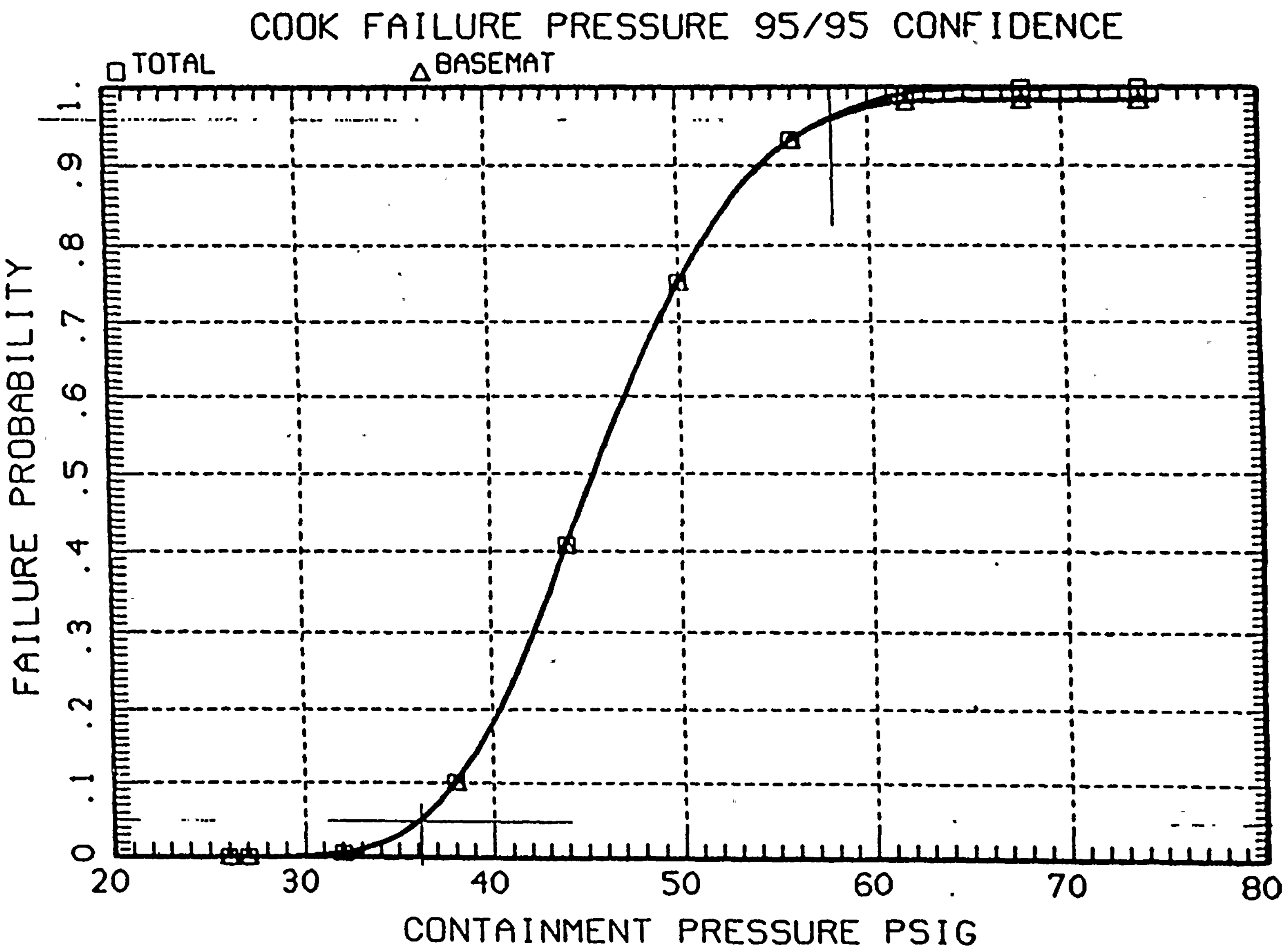


Figure 4.2-1 D. C. Cook containment fragility curve for basemat failure.

4.3 Containment Failure Characterization

Plant-specific phenomenological evaluations, summarized in the above section, have been performed in support of the Cook Nuclear Plant IPE to determine the likelihood of all postulated containment failure modes and mechanisms identified in Reference 20. These detailed evaluations were performed systematically to address the controlling physical processes or events specific to the Cook Nuclear Plant configuration and they are incorporated into the source term notebook.

Through the phenomenological evaluations summarized in the above section, several of the severe accident phenomena were demonstrated to be inconsequential for the Cook Nuclear Plant containment since the since the predicted pressures resulting from a realistic assessment of these phenomena are far less than the containment ultimate strength. The phenomena considered unlikely to result in failure of the containment are: hydrogen combustion, direct containment heating, steam explosions, molten-core concrete attack, thermal attack of containment penetrations, and vessel thrust forces.

The more likely containment failure modes are containment overpressure, containment isolation failure, and containment bypass. These failure modes are discussed in more detail below.

4.3.1 Containment Overpressure

Containment overpressure, defined as a failure mode caused by steaming and/or generation of non-condensible gases, is a potential containment failure mode for the Cook Nuclear Plant. Depending on the specific accident sequence characteristics, overpressure failures may be observed across a wide range of event times. The potential for containment overpressure failure exists in severe accident scenarios where sufficient containment heat removal through spray recirculation is not available.

Overpressure failure is expected to be a slow mechanism such that containment pressurization would be approached gradually. From the evaluation discussed in Section 4.2.1 above, overpressure failure is assumed to occur at 36 psig at the bottom of the annular compartment as a result of a shear failure at the basemat-cylinder junction. Because the resulting stresses in the containment structures are approached slowly, it is assumed that a relatively small rupture area (i.e., "Leak-before-break" behavior) would be induced. This is supported by the published experimental evidence for steel lined concrete containment structures.

At the time of containment failure, there is water accumulated in the lower and annular compartments as a result of ice melt and RCS blowdown. If the RWST is injected, more water would be in the containment. The presence of water in the lower and annular compartments not only prevents immediate airborne release from the break, but also allows containment pressurization to continue until sufficient water is discharged from the containment to allow noncondensible gases and steam to be released from containment. It has been assumed that the continued pressurization of containment will be at a rate which will not result in an additional failure location prior to expulsion of the water.

In order to estimate a delay time between containment overpressure failure and airborne fission product release, a time delay must be calculated and assumptions must be made regarding the amount of water present and the size of the containment failure. A one hour delay time from containment failure was assumed to account for the time required to expel the water in the lower and annular compartments. Assuming a Bernoulli flow, pushing a $2x10^6$ lb, 11 ft tall water pool, the approximate mass and level of water expected in containment for sequences with RWST injection, through a 5.6" hole under pressure of 50 psia would take about 1 hour. This one hour or less of water draining time is accounted for in the source term analysis calculations. The amount of water present in each sequence will vary the time period after containment failure when airborne release would begin. In the source term analysis summarized in Section 4.7. the following assumptions were used. For sequences with RWST injection, a one hour time delay following containment failure is assumed prior to any airborne release. For sequences without RWST injection, 35 minutes is assumed because the only water present in containment would be due to ice melt and RCS

blowdown. Hence, in the calculations, total loss of containment water from the annular and lower compartments is modeled according to the above time delay following the basemat failure.

In summary, overpressure failure of the Cook Nuclear Plant containment is assumed to result in a 5.6-inch equivalent diameter hole in the containment basemat as a result of shear failure of the basemat-cylinder junction. The 0.17 ft^2 hole allows water which has accumulated in the lower and annular compartments to be released from the containment. After initial containment failure, a time delay, which varies depending on the amount of water present in the containment, is assumed prior to any airborne release of fission products to the outside environment.

4.3.2 Containment Isolation Failure

For the Cook Nuclear Plant, if a line penetrating the containment boundary is not isolated, the most likely cause of the isolation failure has been identified as a failure to close an administratively controlled manually isolated line. This implies that the conditions for isolation failure exist prior to the accident initiation. Containment isolation is assumed to result in a ten-inch equivalent diameter pipe which allows direct communication between the air space in the annular compartment and the outside environment. This location was chosen because most of the containment penetrations are located in the annular compartment. The location above the containment water level was chosen to demonstrate the different effects of a containment failure in the air space.

4.3.3 Containment Bypass

As discussed in Section 4.2.3, two potential containment bypass accidents were evaluated as part of the level 1 analysis, steam generator tube ruptures and interfacing systems LOCAs. Each of these initiators is evaluated separately in the source term analysis.

For the SGTR accident scenario, a tube rupture involving a double ended rupture of a single tube (0.943 sq. in break area) is assumed. This rupture may be followed by a release to the environment through the steam generator safety or relief valves.

The interfacing systems LOCA accident scenario was assumed to be initiated by simultaneous failure of two motor operated isolation valves which are the pressure interface between the reactor coolant system and the RHR pump suction piping. These valves are located in the cooldown piping from the loop two hot leg to the RHR pumps. In this sequence, primary system fluid at full RCS pressure is postulated to pass into a low pressure segment of the RHR system where it causes both RHR pump seals to fail. An evaluation of the RHR pump seals identified an upper bound leakage area of 0.18 ft^2 if both pump seals fail (Reference 47). This upper bound break area is in the range of a medium LOCA. Hence a rapid depressurization of the RCS due to the loss of inventory from the break would be expected for this accident scenario. This break area also constitutes a relatively large area through which fission products are transported out of the containment and into the auxiliary building.

4.4 Containment Event Trees (CETs)

The primary function of the containment event tree is to provide a systematic method for reorganizing the results of the level 1 accident sequence quantification into a form more suitable for source term analysis. It is assumed prior to entry into the CET, that core damage and vessel failure has occurred. The CET describes the containment response to a core melt accident and accounts for system interactions, human actions, and key phenomenological issues by defining a functional set of top events and their failure and success states. Each combination of top event success and failure states then leads to a unique CET end state which provides information about ex-vessel sequence progression, containment status, and source term release. Thus, the CET describes the accident sequence beyond core melt and serves as a directory for binning of sequences in the source term analysis.

The structure of the CET is arranged to first determine whether the containment is impaired due to isolation failure. Bypass sequences, which release directly to the auxiliary building or the environment via ECCS piping or through the steam generator secondary side without the benefit of containment fission product retention capabilities, were treated as a level 1 initiating event and, therefore, are not a top node on the CET.

The CET then describes RCS conditions at vessel failure. Specifically the CET determines whether the vessel fails due to a high or low pressure core melt. The distinction between high and low pressure core melt accident sequences is based on RCS pressure at the time of vessel failure. Elevated RCS pressure at vessel failure would drive the gas flow through the cavity region with sufficient velocity to cause debris entrainment to other regions of the containment.

The next decision point considers the status of the reactor cavity, i.e., whether the accident sequence has resulted in a wet cavity at the time of vessel failure. This decision can be determined primarily on the basis of RWST injection to the containment, however, as discussed in Section 4.1.1.1, the volume of the RWST alone may not be sufficient to flood containment to the level where spillover to the cavity would occur. The implication of a wet cavity, following vessel failure, involves the potential for source term scrubbing and debris coolability.

Finally, containment protection was addressed through nodal decisions regarding containment sprays and hydrogen control systems. The status of these systems was embedded within the level 1 event trees and was maintained within the CET so that the severity of the source term release could be distinguished between CET end states. The CET as described above remains applicable to those cases where containment failure occurred prior to vessel failure.

Finally, to adequately bridge the Level 1 and 2 efforts, the following assumptions were applied:

- Core damage under level 1 leads to vessel failure.
- Those systems successful under level 1 were considered functional under level 2.
- Systems which failed in the level 1 analysis were considered failed throughout the level 2 analysis.
- Justification for the basis of the timing of important events and operator actions was determined in the level 1 effort.

4.4.1 CET Top Events and Success Criteria

The containment event tree developed for the Cook Nuclear Plant IPE is displayed in Figure 4.4-1. The CET top events and their success criteria are described in detail in the subsections that follow.

4.4.1.1 Containment Isolation

Containment isolation refers to the closure of containment penetrations to limit the release of radioactive fluids following an accident. Failure to isolate does not imply containment failure but rather that its function has been impaired. As summarized in Section 4.3.2 of this submittal, the most likely single event which will cause containment isolation failure has been identified as a failure to close an administratively controlled manually isolated line. This implies that the conditions for isolation failure existed prior to the accident initiation. A scalar value for the failure probability of this event was calculated in the containment isolation analysis. This value, 1.2×10^{-4}, was assigned to this node during CET quantification.

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

4.4.1.2 High Pressure Melt

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

The gas velocity required to entrain molten debris can be characterized by the value of the superficial gas velocity required for supporting liquid films (Reference 13). Following RPV failure, the gas velocity and its likelihood of exceeding the "critical" velocity for entrainment increase with increasing RCS pressure. Thus, sequences which result in a high pressure melt ejection following RPV failure exhibit varying degrees of debris displacement and entrainment from the cavity to the containment lower compartment. Typical low pressure sequences, such as a large LOCA, however, result in all of the debris remaining in the cavity region (i.e., no entrainment). In addition to the RCS pressure, the degree of entrainment is influenced by the cavity/instrument tunnel geometry and the amount of molten debris present at the time of RPV failure.

The plant damage state definition for each dominant core melt sequence from the level 1 analysis included an indication of the RCS pressure at the time of core damage. Coupled with additional knowledge of the level 1 sequence progression, this defined a priori the likelihood of a high pressure melt ejection (HPME) following RPV failure. Thus, the split fraction for this CET node was specified as 0 or 1 where 0 indicates no HPME occurs (i.e., all debris remained in the cavity) and 1 indicated HPME could occur (i.e., displacement and entrainment of debris to the lower compartment was possible).

The occurrence of a high pressure melt ejection affected containment response by influencing long term sequence progression. It also impacted the source term by increasing the airborne fission product concentration. Postulated early containment failure modes resulting from uncertainties surrounding vessel blowdown thrust forces, direct containment heating, and steam explosions were discussed and discounted in the phenomenological evaluation summaries (Section 4.2). The impact of HPME on the long term containment response was due to the resulting debris distribution. Debris distribution affected the requirements for maintaining debris coolability, the degree of molten core-concrete interactions, and the steaming rate of containment water pools.

4.4.1.3 RWST Injection

The intent of this node was to determine if injection of the RWST to the containment was successful prior to vessel failure. RWST injection could have been either through the emergency core cooling systems or containment sprays. RWST injection influenced the timing of ice depletion and containment failure, and therefore, directly impacted the source term release.

Systems available for use in the level 2 analysis were already accounted for in the level 1 quantification and a review of the level 1 plant damage state would reveal the state of the RWST injection on the CET. Thus, for a particular sequence, an appropriate split fraction of 0 or 1 was assigned to this node.

Success of the RWST injection node delayed core-concrete attack and the accompanying generation of aerosols and airborne fission products. Thus, the RCS injection node would have a strong influence on the potential

radionuclide release from containment. Success of this node was predicated on the level 1 sequence identifying injection into the containment.

4.4.1.4 Containment Spray Injection and Recirculation

The next two nodes deal with the status of the containment spray system. This system provides for short term pressure control and long term containment heat removal.

Operations of the sprays would cool the atmosphere, condense steam, and enhance steam flow through the ice condenser. This process would also tend to de-inert the containment thus increasing the relative hydrogen and oxygen molar concentrations. Following vessel failure, spray operation would provide a pool of water to cool the debris and scrub fission product releases. Airborne fission products would also be scrubbed by sprays thereby minimizing any potential source term release.

Success of containment spray recirculation is important because of its long term containment heat removal role. The interaction of core debris with containment water pools, mechanical structures, and atmosphere can result in a heatup and pressurization of the containment. The pressurization is a function of the rate of gas production (condensible and non-condensible), and the rate of increase of the containment gas temperature. Since the gas production and temperature rise can be characterized by the levels of decay heat generation within the core debris, it is necessary to establish some form of containment heat removal which meets or exceeds the decay heat generation rates. Failure to establish any form of long term containment heat removal would result in sustained containment pressurization and an eventual failure at the containment boundary.

The containment spray recirculation node accounts for the containment response due to the success or failure of containment heat removal systems. Failure of this node will result in a failure of the containment due to overpressure while success of this node results in maintaining pressure within the capacity of the containment. During the time period when containment integrity can be challenged due to overpressure, the debris decay heat levels are sufficiently low so that the operation of one containment spray pump plus heat exchanger can adequately remove decay heat. Thus, operation of one containment spray train in recirculation was required for success of the containment spray recirculation node.

The split fractions for the containment spray injection or recirculation nodes for a particular sequence were 0 or 1 depending on the system availability as defined in the level 1 plant damage state.

4.4.1.5 Hydrogen Igniters

The Cook Nuclear Plant containment design incorporates 70 thermal igniters throughout various compartments. The intent of the igniters is to provide ignition of lean hydrogen-air mixtures throughout the containment, thus releasing moderate amounts of energy to the containment over an extended period of time. Operation of the igniters will minimize any rapid containment pressurization caused by hydrogen burns.

Failure of the hydrogen igniter system as in the case of a station blackout was assessed as an uncertainty in addressing the impact of hydrogen within the containment. The amount of hydrogen and other non-condensible gases generated during a sequence was considered as part of the uncertainty analysis. Success of this node assumes one functional train of hydrogen igniters operating for 48 hours. An appropriate split fraction of 0 or 1 was assigned to this node based on level 1 sequence definition.

4.4.1.6 Containment Recirculation Fans

The containment recirculation fans serve a primary function of containment air mixing. This is achieved by circulating air from the upper compartment to the lower compartment following an accident. By virtue of air recirculation, this process also minimizes any local hydrogen concentrations by uniformly distributing the

hydrogen throughout the containment air space. Operation of the recirculation fans may enhance hydrogen burning by circulating oxygen to previously inerted areas of the containment. The failure probability of this system was quantified in the level 1 accident analysis and, therefore, an appropriate split fraction of 0 or 1 was assigned to this node based on level 1 sequence definition.

4.4.2 CET Structure and End States

The CET top events, as described above, were arranged in a manner which depicts the accident progression and provides appropriate grouping for source term evaluation. Failure of containment spray injection was assumed to result in failure of spray recirculation. In addition, if the hydrogen igniters operated successfully, operation of the recirculation fans would have only a minimal impact on the overall containment response and, therefore, was bypassed.

The level 1 plant damage state descriptor consists of a series of alphabetic characters. The first depicts the category of the initiating event, the second the RCS pressure, and the third is a series of characters addressing the status of primary protection systems. These characters were defined in Section 3.1.5 of this submittal and are reflected in Figure 4.4-1.

The combination of top events resulted in the 25 CET end states shown in Figure 4.4-1. Utilizing the definitions from section 3.1.5, each CET end state was labeled with an appropriate plant damage state designator. To address level 2 results, a designator defined and assigned as summarized in Section 4.7 of this submittal, was added which depicts the status of the containment and source term release.

Based on the analysis summarized in Section 4.4.1, it was concluded that end states identified as HR (RCS at high pressure, containment spray injection and recirculation successful), HRI (HR and hydrogen igniters fail), LR (RCS at low pressure, containment spray injection and recirculation successful), and LRI (LR and hydrogen igniters fail) would be successful in that containment failure will not occur. States HRIF and LRIF are sequence dependent and required further analysis before concluding success or failure of the containment. The balance of the CET end states constitute containment overpressure failures or isolation impairments.

Containment isolation is shown as branching directly to a transfer tree. Failure to isolate primarily impacted source term release by providing a potential alternate release path following core damage or vessel failure. For the Cook Nuclear Plant, containment isolation failure was treated as an event which was independent of any initiator or status of top nodes. Therefore, every CET end state could also include containment isolation failure. Following quantification of CET events with successful isolation, dominant sequences were reviewed for evaluation of isolation failure.

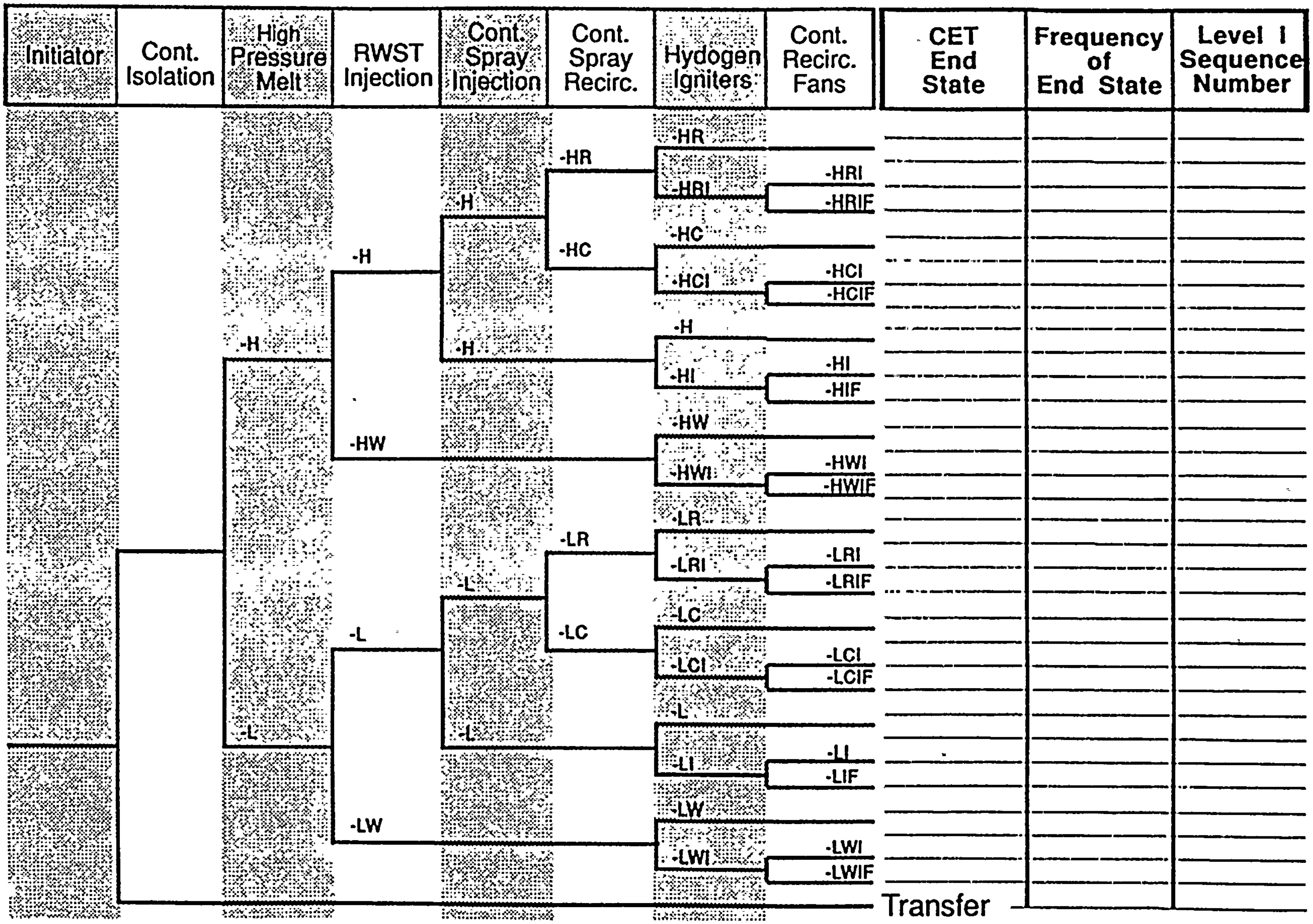


Figure 4.4-1 D. C. Cook containment event tree.

4.5 CET Quantification

CET quantification was performed for the core damage accident sequences from the level 1 analysis. The CET quantification, summarized in Table 4.6-1, assigned each core damage sequence, along with its frequency, to a particular CET end state. These end states and the cumulative frequencies formed the basis for grouping similar sequences into accident sequence bins. Quantification of the CET was performed in two steps. The first step of the quantification assumed that containment isolation was successful. The second step of the quantification accounted for containment isolation failure.

The first step of the CET quantification was performed individually for each initiating event category. In this step, the probabilities of CET top event success and failure states were determined by the level 1 accident sequence quantification. The split fractions for each CET branch were assigned as 0's and 1's based on a review of the level 1 accident sequence definition. CET quantification simply involved following the CET branching logic for each level 1 accident sequence to arrive at a particular CET end state. The frequency of each CET end state is then the sum of the frequencies of the level 1 accident sequences assigned to that end state.

The failure probability of containment isolation was not calculated as part of the level 1 accident sequence quantification but, as summarized in Section 4.4.1.1, was calculated as part of a separate analysis to be 1.2E-04. The containment isolation top event in the CET is assigned this value, 1.2E-04, as a scalar. In the second step of CET quantification, the CET end states from the above step were reviewed by initiating event. The CET end state with the highest frequency and, in some cases the CET end state with the second highest frequency, was selected. The frequency of each of these end states was multiplied by the failure probability of containment isolation. The sequence was then worked through the CET as above.

These resulting CET end states were ranked by frequency and screened for use in the source term analysis. The effects on containment were determined by MAAP analysis. The ranking and screening process and MAAP analyses are described in the following sections.

4.6 Bins and Plant Damage States

The results of the level 1 analysis must be processed into a form more suitable for source term analysis. The first step in this processing is the CET quantification described in the section above. The remaining processing involved grouping similar sequences into accident sequence "bins" to reduce the total number of sequences analyzed. Source term quantification could then be performed by analyzing a single, representative accident sequence from each bin.

The binning process used for source term analysis can be summarized as follows:

(A) Screening process: From the results of the CET quantification, a certain number of CET end states was selected. These CET end states were selected based on the sequence screening criteria guidelines given in Reference 20.

(B) Sequence binning: The CET end states identified above were grouped such that each group would result in similar source terms.

(C) Selection of representative sequences: At least one sequence was selected from those included within each group in step B. The selected sequence(s) were analyzed for the source term analysis.

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

These steps are described in more detail in the subsections that follow.

4.6.1 Screening Process

The CET results were reviewed, tabulated and organized according to core damage frequency with the results listed in Table 4.6-1. The sequence numbers in the third column correspond to the sequence number from the level 1 event trees. When several sequences appear on the same line in Table 4.6-1, the sequences have the same initiator (column 1) and CET end state (column 2). The core damage frequency (column 4) is the sum of all the frequencies for all the sequences listed on the line. It is from this listing that screening was performed.

NUREG 1335 (Reference 20), Section 2.1.6 describes the screening process for front-end results. The following compares these requirements to the results tabulated in Table 4.6-1.

NUREG-1335	Table 4.6-1
Any systemic sequence that contributes 1E-7 or more per reactor year	Sequences through line 22 of Table 4.6-1
All systemic sequences within the upper 95% of the core damage frequency	Sequences through line 12 of Table 4.6-1
All systemic sequences within the upper 95% of the total containment failure frequency	Sequences through line 23 of Table 4.6-1
Systemic sequences which contribute 1E-8 to containment bypass	Sequences on line 30 need to be included
In any case, should not exceed 100 most significant sequences	Sequences through line 40 of Table 4.6-1

Reported in this submittal are the top 100 accident sequences of the level 1 analysis as screened through the CET. Lines one through 40 of Table 4.6-1 included 100 accident sequences. In addition, reporting these sequences exceeds the minimum reporting requirements of Reference 20.

4.6.2 Sequence Binning and Selection of Representative Sequences

The binning of the 40 CET end states was based on a series of decisions that resulted in grouping together like sequences. The first grouping of the screened end states was based on the status of containment for the sequence. Thus all sequences fall initially into the following status:

- Containment isolation failure
- Containment bypassed
- Containment overpressure/success.

Very few of the 40 CET end states fell under the impairment or bypassed categories. Each of these containment status categories, therefore, was treated as a separate bin.

The majority of sequences fell into the containment overpressure/success status. Thus, several bins were necessary to address the sequences within this containment status category. Source term releases for these categories depend on the timing of such events as vessel failure, ice depletion, and containment failure. The availability of water to cool the debris through vessel or containment injection would impact the timing of these events.

To address the availability of water for debris cooling, the CET and the top 40 CET end states were reviewed. This review indicated the following functional groupings for maintaining water within containment and the vessel:

I. Successful containment spray injection and recirculation

II. Successful containment spray injection but failed spray recirculation

III. Failed containment spray injection and recirculation.

The first group encompasses those sequences with an "R" designator as the third letter in their CET end state description. The second group would is identified by a "C" designator as the third letter in the end state description. The third group's designator contains a "W" as the third letter in the designator. A fourth group was identified by the review of the CET. This group had successful RWST injection through the ECCS but failed all containment spray. None of the sequences in the top 40 CET end states fell into this group, therefore, this group was not considered for further analysis.

Sequences in Group I would result in containment success. Small releases through prexisting normal leakage leakage paths may occur. This leakage would be within the limits permitted by technical specifications. Group II sequences would have a water pool covering the debris ejected from the vessel. This water pool would cause continuous steaming which would cause the quickest depletion of the ice mass and, therefore, the earliest containment failure. Group III would have resulted in a dry reactor cavity, no steaming of a water pool covering the core debris and, therefore, the latest containment failure.

The three groups discussed above are further divided into two categories based on reactor coolant system pressure at the time of reactor pressure vessel failure. The split of the groups into high and low pressure categories was made by review of the second letter in the CET end state designators. This split results in six bins into which all sequences other than bypass or isolation failure were placed.

The initiators for the level 1 work were categorized as A (large LOCA), S (small LOCA), T (transient) and G (tube rupture). Each of the six bins, therefore, could have four initiators. Large LOCAs, those end states with an A desiginator, are only low pressure events by definition and therefore will not require a split based on pressure. Vessel failure for transients and steam generator tube ruptures typically occurs at high pressure. CET end states for these events, therefore, were not split into two pressure categories in order to simplify the analysis.

With the assumptions described above, the groups become:

GROUP I(H)	GROUP II(H)	GROUP III(H)
SHR	SHC	SHW
THR	THC	THW
GHR	GHC	GHW

GROUP I(L)	GROUP II(L)	GROUP III(L)
ALR	ALC	ALW
SLR	SLC	SLW

The six groups above represent the bins under the containment status category of overpressure/success. All accident sequences within the top 40 CET end states which were not containment bypass or containment isolation failure sequences fell into one of these bins. The designators represent initiator (first letter), RCS pressure (second letter) and availability of water (third letter). One or multiple sequences are selected from each group for further evaluation in the source term analysis.

Within each initiator class included in the groups above (A, S, T, G), different initiators occur. For example, transient (T) will include as initiating events loss of offsite power, steam line break, etc. Thus, in selecting a THW sequence, more than one could be selected based on the actual initiator. Furthermore, in selecting sequences from class G (SGTR), special consideration must be given to whether the integrity of the faulted steam generator is maintained. Containment bypass to the environment would be established if a steam generator PORV, safety valve, or MSIV on the faulted steam generator fails to close or remain closed. SGTR sequences were selected to represent situations with and without faulted steam generator integrity.

A representative sequence from at least one of the CET end states in each of the above bins was selected for source term analysis. The sequence selected for analysis was one of the top 100 sequences which were grouped into the top 40 CET end states of Table 4.6-1. The sequences were selected for analysis in order to bound the other sequences which appeared in that group. The sequences selected for source term analysis (called analyzed sequences) are summarized in Table 4.6-2. The remaining sequences within the top 40 CET end states of Table 4.6-1 are called bounded sequences and are summarized in Table 4.6-3.

In addition to selection of representative sequences from the top 40 CET end states listed in Table 4.6-1, three additional sequences were selected for source term analysis, one each from initiators A, S, and T. These three additional sequences, indicated in Table 4.6-1, were analyzed with failure of containment isolation. None of these sequences appear in the top 100 list.

Release categories, the last letter in the CET end state designator of Table 4.6-1 are defined in Table 4.6-4. Fission product release categories were calculated as part of the source term analysis which is summarized in the Section 4.7 of this submittal.

4.6.3 Conclusions

The frequency of containment overpressure failure, based on the quantification summarized above, was calculated to be 2.08×10^{-6} per year. Given that the core damage frequency calculated by the level 1 analysis is 6.26×10^{-5} per year, the conditional probability of containment failure on overpressure given core damage is 3.3×10^{-2}. The frequency of containment failure, based on initiator category, is summarized in Figure 4.6-1.

This figure shows transient events are the major contributor to containment failure.

The CET quantification indicates that there are three major paths through the CET to containment overpressure failure. These are failure to inject the RWST, failure to inject through containment sprays, and failure to align for recirculation. Figure 4.6-2 illustrates the contribution to containment failure for each of these sequences. As shown, containment failure is dominated by failure to inject and failure to align for recirculation.

The frequency of containment bypass events, as calculated by the CET quantification above, is 7.2×10^{-6}. This gives a conditional probability of containment failure given core damage of 0.113. The major contributors to containment bypass events are steam generator tube ruptures. Interfacing system LOCAs (V-sequences) contribute less than one percent to this value. Failure to isolate containment contributes only 0.01% to the conditional probability of containment failure with a frequency of 6.52×10^{-9}.

TABLE 4.6-1
DONALD C. COOK NUCLEAR PLANT
DOMINANT SEQUENCE LIST

INITIATOR	END STATE	SEQUENCE NO.	FREQUENCY	ANALYZED SEQUENCES
1. SLO	SHR-S	13,23,32	$1.59x10^{-5}$	
2. CCW	THR-S	2,12,22,41,50,187	$1.38x10^{-5}$	
3. SLO	SLR-S	2	$1.31x10^{-5}$	2
4. SGR	GHR-C	3,33,72,82	$5.99x10^{-6}$	3
5. ATWS	SHR-S	30,39,48,67,85	$2.79x10^{-6}$	
6. MLO	SHR-S	12,32,41	$2.29x10^{-6}$	
7. MLO	SLR-S	2,22	$1.97x10^{-6}$	
8. SGR	GHR-T	13,23,42,52,91	$9.98x10^{-7}$	13
9. SBO	THWIF-M	50,80,130,190	$6.64x10^{-7}$	50,190
10. LLO	ALR-S	5,14,23	$6.33x10^{-7}$	5
11. ESW	THR-S	2,12,22,41,32,186	$5.96x10^{-7}$	
12. VDC*	THR-S	3	$5.85x10^{-7}$	
13. SLB	THR-S	13,22,31	$4.75x10^{-7}$	
14. SBO	THR-S	2,32,41,71,121, 172,181	$4.68x10^{-7}$	181
15. SLO	SLC-J	5	$3.56x10^{-7}$	5
16. LLO	ALC-J	2,8,17,26	$3.16x10^{-7}$	8
17. VEF	ALR-S	1	$2.99x10^{-7}$	
18. TRA	THR-S	5,15	$2.83x10^{-7}$	
19. TRS	THWIF-M	21	$1.77x10^{-7}$	
20. SLO	SHC-J	26,35	$1.14x10^{-7}$	35
21. LSP+	THR-S	3,13	$1.1x10^{-7}$	
22. TRS	THR-S	3,13	$1.03x10^{-7}$	
23. SLO	SHWIF-M	40	$9.91x10^{-8}$	
24. SLB	THW-G	28,19	$8.48x10^{-8}$	
25. SGR	GHWIF-T	50,99	$6.73x10^{-8}$	50
26. LSP	THWIF-M	21	$6.28x10^{-8}$	21

TABLE 4.6-1 (Cont'd)
DONALD C. COOK NUCLEAR PLANT
DOMINANT SEQUENCE LIST

INITIATOR	END STATE	SEQUENCE NO.	FREQUENCY	ANALYZED SEQUENCES
27. ATWS	THR-S.	2,11,21	$5.82x10^{-8}$	
28 CCW	THC-M	5,15,189	$5.5x10^{-8}$	
29. MLO	SLC-J	5,25	$5.47x10^{-8}$	5
30. ISL	V-T	2,3,4,5,6	$5.38x10^{-8}$	5
31. MLO	SHWIF-M	40	$1.33x10^{-8}$	40
32. VDC	THW-M	9	$1.18x10^{-8}$	
33. CCW	THWIF-M	20	$1.12x10^{-8}$	
34. ESW	THC-M	5,15,50,189,195	$7.41x10^{-9}$	
35. VDC	THC-M	6	$6.23x10^{-9}$	
36. SLO	SHW-M	38	$4.94x10^{-9}$	
37. CCW	THW-M	191	$4.79x10^{-9}$	
38. SLB	THIF-M	39	$4.69x10^{-9}$	
39. TRA	THWIF-M	23	$4.32x10^{-9}$	
40. SLB	THC-M	34	$4.31x10^{-9}$	
41. SLO	SL-J	9	$4.31x10^{-9}$	
42. SGR	GHWIF-C	11,41	$4.09x10^{-9}$	
43. LLO	ALWIF-W	22	$3.98x10^{-9}$	
44. SLO	SHRI-S	3,24,33	$3.37x10^{-9}$	
45. SGR	GHW-T	9,97	$2.68x10^{-9}$	
46. SGR	GHC-C	6	$2.68X10^{-9}$	
47. SGR	GHC-T	45,94	$2.59X10^{-9}$	
48. TRS	THW-M	19	$2.53x10^{-9}$	
49. MLO	SHC-M	15,35	$2.12x10^{-9}$	35
50. SLO	SH-M	29	$2.02x10^{-9}$	
51. SLO	SHR'-E	13,23,32	$1.91x10^{-9}$	
52. CCW	THR'-E	2,12,22,41,32,186	$1.66x10^{-9}$	

TABLE 4.6-1 (Cont'd)
DONALD C. COOK NUCLEAR PLANT
DOMINANT SEQUENCE LIST

INITIATOR	END STATE	SEQUENCE NO.	FREQUENCY	ANALYZED SEQUENCES
53. SLO	SLR'-E	2	1.58×10^{-9}	2
54. CCW	THRIF-S	188	1.49×10^{-9}	
55. ATWS	SHC-M	33,51	1.22×10^{-9}	
56. TRA	THW-M	21	1.18×10^{-9}	
57. ATWS	SHW-M	36	1.16×10^{-9}	
58. SGR	GHRI-C	4	9.27×10^{-10}	
59. LSP	THW-M	19	8.45×10^{-10}	
60. SGR	GHR'-E	3	7.18×10^{-10}	
61. VDC	THRI-S	4	6.22×10^{-10}	
62. MLO	SH-M	9,18	5.56×10^{-10}	
63. ATWS	SHWIF-M	47	5.05×10^{-10}	
64. MLO	SHR'-E	12,32,41	4.85×10^{-10}	
65. ESW	THWIF-M	20	4.73×10^{-10}	
66. ATWS	SHRI-S	31	4.3×10^{-10}	
67. LLO	ALIF-M	13	4.03×10^{-10}	
68. ATWS	THC-M	24	3.64×10^{-10}	
69. ATWS	SHR'-E	30,39,48,67,85	3.35×10^{-10}	
70. VEF	ALC-J	4	2.7×10^{-10}	
71. MLO	SLR'-E	2,22	2.36×10^{-10}	
72. VEF	ALW-M	7	1.78×10^{-10}	
73. MLO	SL-M	29	1.49×10^{-10}	
74. VDC	THWIF-M	11	1.48×10^{-10}	
75. MLO	SHRI-S	3	1.38×10^{-10}	
76. SBO	THWIF'-G	50,80,130,190	7.97×10^{-11}	190
77. LLO	ALR'-E	5,14,23	7.6×10^{-11}	
78. ESW	THR'-E	2,12,22,32,41,186	7.15×10^{-11}	

TABLE 4.6-1 (Cont'd)
DONALD C. COOK NUCLEAR PLANT
DOMINANT SEQUENCE LIST

INITIATOR	END STATE	SEQUENCE NO.	FREQUENCY	ANALYZED SEQUENCES
79. VDC	THR'-E	3	$7.02x10^{-11}$	
80. LLO	ALW-M	20	$6.96x10^{-11}$	
81. LLO	AL-M	11	$6.67x10^{-11}$	
82. VEF	ALRI-S	2	$6.25x10^{-11}$	
83. SLB	THR'-E	13,22,31	$5.71x10^{-11}$	
84. SBO	THR'-E	2,32,41,71,121, 172,181	$5.62x10^{-11}$	
85. LLO	ALC'-U	2,8,17,26	$3.79x10^{-11}$	8
86. VEF	ALR'-E	1	$3.59x10^{-11}$	
87. TRA	THR'-E	5.15	$3.4x10^{-11}$	
88. TRS	THWIF'-G	21	$2.12x10^{-11}$	
89. LLO	ALRI-S	6	$1.7x10^{-11}$	
90. LLO	ALCI-V	3	$1.45x10^{-11}$	
91. LSP	THR'-E	3,13	$1.32x10^{-11}$	
92. TRS	THR'-E	3,13	$1.24x10^{-11}$	
93. VEF	ALWIF-W	9	$3.91x10^{-12}$	

NOTE:

1. Ref Table 4.6-2 for definition of end state release category.
2. Core damage frequency = 6.26×10^{-5}.
3. Primed sequence end states denote a failure to isolate.
4. * Denotes upper 95% of core damage frequency.
5. + Denotes upper 99% of core damage frequency

LEGEND

SLO - Small LOCA
MLO - Medium LOCA
ESW - Loss of essential service water
SBO - Station blackout
TRA - Transient with steam conversion
VEF - Vessel failure
SLB - Steam line break
LLO - Large LOCA
SGR - Steam generator tube rupture
CCW - Loss of component cooling water
ATWS - Ant. trip without scram
LSP - Loss of offsite power
TRS - Transient witout steam conversion
VDC - Loss of single 250 V DC train

Table 4.6-2

END STATES SELECTED FOR ANALYSIS

GROUP	DESCRIPTION	TYPE	TABLE 4.6-1 LINE	END STATE	INITIATOR
I(H)	Successful injection & recirculation	SHR	Not selected but bounded by THR		
		THR	2	THR	CCW
			14	THR	SBO
			21	THR	LSP
		GHR	4	GHR	SGR
			8	GHR	SGR
I(L)	Successful injection & recirculation	ALR	10	ALR	LLO
		SLR	3	SLR	SLO
II(H)	Successful injection, failed recirculation	SHC	20	SHC	SLO
		THC	Not selected		
		GHC	Not selected		
II(L)	Successful injection failed recirculation	ALC	16	ALC	LLO
		SLC	15	SLC	SLO
			29	SLC	MLO
III(H)	Failed injection and recirculation	SHW	31	SHWIF	MLO
		THW	9	THWIF	SBO
		GHW	25	GHWIF	SGR
III(L)	No sequences in the top 100				

Table 4.6-3

D. C. COOK NUCLEAR PLANT
BINNING OF BOUNDED SEQUENCES

BOUNDED			ANALYZED		
INITIATOR	END STATE	RELEASE CATEGORY	INITIATOR	END STATE	RELEASE CATEGORY
LSP CCW SLO MLO ATWS ESW VDC SLB TRA TRS ATWS	THR THR SHR SHR SHR THR THR THR THR THR THR	S S S S S S S S S S S	SBO	THR	S
TRA TRS LSP CCW SLO CCW VDC SLO SLB	THWIF THWIF THWIF THWIF SHWIF THW THW SHW THIF	M M M M M M M M M	SBO LSP	THWIF THWIF	M M
SLB CCW ESW VDC	THC THC THC THC	J J J J	SLO MLO	SHC SHC	J J
SLO	SL	J	SLO	SLC	J
SLB	THW	G	SBO	THWIF'	G
VEF	ALR	S	LLO	ALR	S
SGR FAULTED S/G INTEGRITY MAINTAINED	GHC	C	SGR	GHR	C
SGR FAULTED S/G INTEGRITY NOT MAINTAINED	GHC	T	SGR	GHWIF	T

NOTES:

1. Ref. Table 4.6-1 for clarification of bounded and analyzed sequence numbers.

2. The prime in THWIF' indicates containment isolation failure.

Table 4.6-4

RELEASE CATEGORY DEFINITION

Release Category	Definition
A	No containment failure within 24 hour mission time but failure could eventually occur without accident management action; noble gases and less than 1/10% volatiles released.
B	Containment bypassed with noble gases plus less than 1/10% of the volatiles released.
C	Containment bypassed with noble gases plus up to 1% of the volatiles released.
D	Containment bypassed with noble gases and up to 10% of the volatiles released.
E	Containment failure prior to vessel failure with noble gases and less than 1/10% of the volatiles released (containment isolation impaired).
F	Containment failure prior to vessel failure with noble gases and up to 1% of the volatiles released (containment isolation impaired).
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired).
H	Early containment failure with the noble gases and less than 1/10% volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful).
I	Early containment failure with noble gases and up to 1% of the volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful).
J	Early containment failure with noble gases and up to 10% of the volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful).
K	Late containment failure with noble gases and less than 1/10% volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful).

Table 4.6-4 (Continued)

RELEASE CATEGORY DEFINITION

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

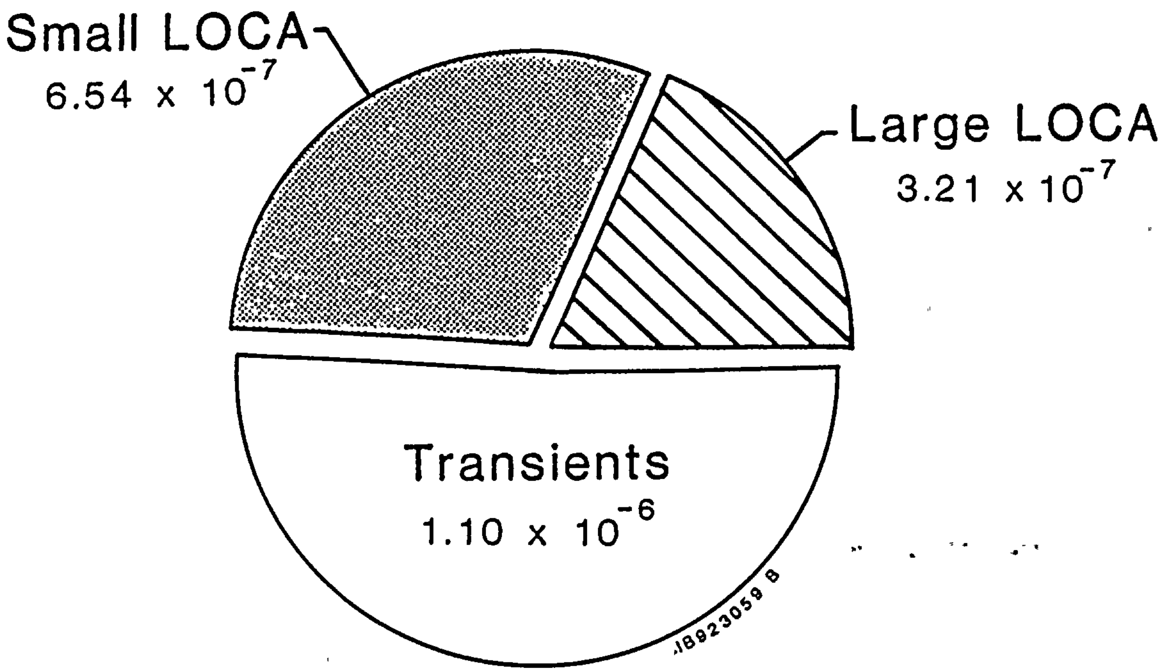


Figure 4.6-1 Frequency of containment failure (per reactor year).

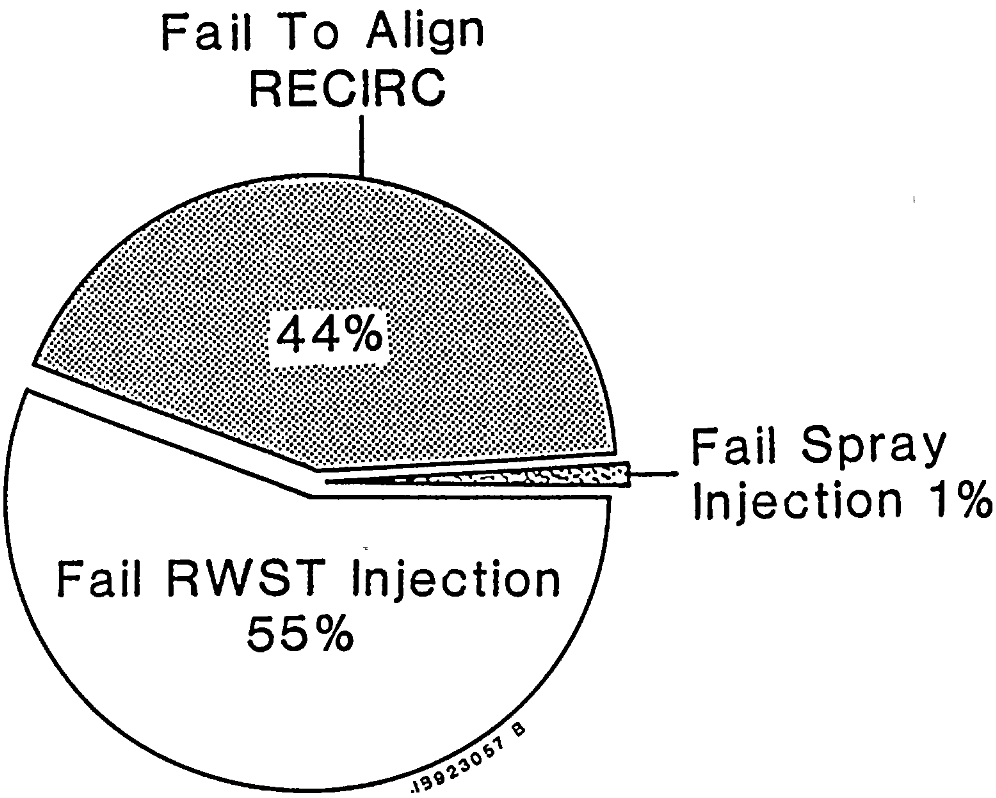


Figure 4.6-2 Contribution to containment failure.

4.7 Accident Progression and Radionuclide Release Categorization

4.7.1 Accident Progression

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

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

In the absence of external cooling of the RPV, relocation of the molten core into the lower head is assumed to lead directly to failure of the reactor vessel. The Cook Nuclear Plant IPE took no credit for potential in-vessel recovery. If the RCS is at high pressure at the time of vessel failure, then high pressure melt ejection (HPME) could possibly displace or entrain some core debris into the lower compartment. Most of this debris is de-entrained by lower compartment structures. If the RCS is at low pressure at the time of vessel failure, then low pressure melt ejection results in only a small amount of core debris escaping the cavity.

In either case, if sufficient water is already available in containment prior to RPV failure, then the core debris is quickly quenched upon expulsion from the reactor vessel into the cavity region. Steam generated due to boiling this water melts the containment ice. When the ice is depleted, the steam begins to pressurize the containment. If containment heat removal (spray recirculation) is available, the containment water pools can be recirculated to cool core debris.

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

Complete (or nearly complete) melting of the ice supply, in combination with failure of spray recirculation, eventually results in failure of the Cook Nuclear Plant containment due to overpressurization. The time required to fail the containment by overpressurization depends upon the steaming rate, which in turn depends on the amount of water injected into the containment during the accident sequence progression and on the distribution of core debris following core relocation. The failure mechanism associated with containment overpressure is due to exceeding the ultimate strength of certain key structural components or attachments. This limit is most likely to be approached gradually, so that the energy delivered is only sufficient to induce a relatively small rupture area.

The severity of the source term depends strongly on the containment failure timing. Failure in the immediate time period of vessel failure is clearly the most serious, as the overall airborne fission product mass produced during a severe accident is never greater than it is in the small span of time directly after vessel failure. Whenever containment pressurization lags considerably behind vessel failure, substantial fission product

retention through naturally occurring deposition mechanisms (e.g., sedimentation, impaction, etc.) is facilitated.

Containment failure at the Cook Nuclear Plant is most likely to occur at the basemat/cylinder junction. The floor in the annular compartment is expected to be flooded at the time of containment failure, due to melting of the ice and/or spray injection. Water would also occupy the lower compartment floor. These two compartments (annular and lower) are separated by the crane wall but are able to communicate through pipe sleeves and other openings. The level of communication between the annular and lower compartments and the size of the break in containment determines the subsequent containment response and accident progression.

Open communication between the annular and lower compartments would lead to expulsion of all water in the annular and lower compartments upon failure at the basemat. This leaves only water in the cavity, which will not be replenished by recirculation once it has boiled off. Expulsion of water pools from containment sweeps out some portion of the fission products which had settled into them. The containment pressure most likely continues to go up during this water discharge, but it may not increase enough to fail the containment elsewhere (i.e., in the containment gas space) before all the water on the floor had been ejected. Once the water has all been discharged, containment depressurization would begin as the containment atmosphere vents through the break at the basemat junction.

4.7.2 Source Term Analysis

The purpose of the source term analysis was to quantitatively describe the magnitude and composition of fission product release to the environment resulting from the severe core damage accidents defined in the level 1 study.

To adequately address the complexities associated with fission product transport and release and to account for the specific level 1 sequence definitions including operator actions, the Donald C. Cook source term analysis relied on the integrated severe accident analysis code, MAAP. This code couples the plant thermal hydraulic response and fission product behavior to properly model feedback between the two. Furthermore, MAAP can analyze all phases of severe accident progression accounting for the impact of the primary system, containment, engineered safety features, and operator actions. In regard to fission product transport, MAAP begins tracking the fission products as they exist in the normally intact fuel matrix. This initial fission product inventory was organized by chemical properties into 12 groups within MAAP. The initial inventory of each of the 12 fission product groups specific to Cook Nuclear Plant as derived from the MAAP parameter file are as follows (Reference 61):

Fission Product Group	Initial Inventory (lb)
1) Noble Gases (Xe, Kr)	1047
2) CsI (volatile)	81
3) TeO_2	0
4) SrO	212
5) MoO_2	823
6) CsOH (volatile)	646
7) BaO	299
8) La_2O_3 (& Pr_2O_3 + Nd_2O_3, Sm_2O_3 + Y_2O_3)	1566
9) CeO_2	661
10) Sb	3
11) Te_2 (volatile)	84
12) UO_2 (& NpO_2 + PuO_2)	196280

The fission products are specific to Cook Nuclear Plant and based on ORIGEN calculations (Reference 62). This total inventory of fission products is generally characterized as noble gases, volatile

fission products (groups 2, 6, 11) and non-volatile fission products (groups 3, 4, 5, 7, 8, 9, 10, 12). The source term analysis summarized in this section reports the mass fraction released for each of these three categories.

4.7.2.1 MAAP Analyses

Nineteen sequences (15 from the top 40 CET end states of Table 4.6-1, three sequences addressing failure to isolate, and one other sequence of interest) were selected for source term analyses using MAAP 3.0B Revision 17.02. This section describes these sequences. Several assumptions made for the MAAP calculations are outlined here since they significantly affect the calculated source term results.

(1) The level 2 analysis assumed a 48 hour mission time, while the level 1 analysis used a mission time of 24 hours. Hence, accident progression was studied for a period of time beyond which Accident Management activities would be implemented to alter the course of the accident.

(2) A crane wall separates the lower compartment from the annular compartment. It is assumed that holes in the crane wall at various elevations would allow an equalized water level in both compartment at any time. For further details see Section 4.1.1.1.

(3) Airborne fission product release to the environment was assumed to occur through a small break of about 0.17 ft^2 located at the bottom of the annular compartment near the basemat-cylinder junction due to shear failure of the basemat concrete. This failure was assumed to occur due to overpressure at 36 psig (Reference 53). At the time of containment failure, there is water accumulated in the lower and annular compartments, regardless of whether there is injection of the RWST. This water is the result of ice melting. However, if the RWST is injected, more water would be in the containment. The presence of water in the lower containment not only prevents immediate airborne release from the break, but also allows containment pressurization to continue until all water is discharged from the containment. Assuming a Bernoulli flow, pushing a $2x10^6$ lb, 11 ft tall water pool out of the containment through a 5" hole under pressure of 50 psia would take about 1 hour, without considering soil resistance. This one hour or less of water draining time is accounted for in the calculations, depending on the amount of water present in each sequence. For sequences with RWST injection, one hour is assumed. For sequences without RWST injection, 35 minutes is assumed. Hence, in the calculations, total loss of containment water from the annular and lower compartments is modeled according to the above time delay following the basemat failure.

(4) For the purposes of computing the containment response and long term source term response, fission products deposited in the annular and lower compartments were assumed to be swept out of the containment by water. It is uncertain, however, how much of the fission product inventory would be transported with the water. Hence, all fission products deposited in the annular and lower compartments prior to containment failure should be considered as available for aqueous release, and not a measure of the actual water release. The handling of aqueous fission product release is addressed in the level 3 analysis.

Accident progression parameters for all analyzed sequences are summarized in Table 4.7-1 This table contains information such as accident timing and conditions, hydrogen burns and source term.

Table 4.7-2 summarizes environmental release in terms of "airborne release" and "available for aqueous release." Based on the MAAP source term results, a release category was assigned and a frequency of each release category was calculated for each of the indicated sequences. These results are shown in Table 4.7-3.

Each analyzed sequence is described below.

Large LOCA - Sequence 5 (LLO-5)

Sequence Description:

This accident scenario is initiated by a 4.0 sq. ft. (27 inch equivalent diameter) large break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) ACC - 3 of 3 accumulators inject to the cold legs

(2) LPI - 1 of 2 RHR pumps inject to 1 of 3 intact cold legs

(3) CSI - 1 of 2 trains of containment spray injection operational

(4) CSR - 1 of 2 containment spray trains switched from the RWST to the recirculation sump

(5) HI - 1 of 2 trains of hydrogen igniters in both upper and lower compartments operational.

This sequence assumed an operational failure to establish low pressure recirculation (LPR).

Automatically actuated RHR injection and containment spray injection pumps were assumed operational until the water level in the RWST reached the 32% low level setpoint. From this point, the RHR injection was terminated and the containment spray recirculation was assumed to operate through the entire accident time frame of 48 hours.

Sequence Quantification:

This accident scenario resulted in a rapid blowdown of primary system inventory into the lower compartment. Containment spray injection was activated immediately following RCS blowdown. Within a half a minute after the break initiation, the blowdown was complete and the primary system pressure dropped below 400 psia. Twenty four minutes into the accident, the RHR injection was terminated and containment spray was switched to recirculation mode when the RWST low level setpoint was reached. Without continued injection into the primary system, due to failure to establish low pressure recirculation, the core uncovered at 37 minutes. This was followed by core relocation to the lower head at 1.5 hours and failure of the vessel one minute later.

The cavity was dry when core debris was discharged onto the cavity floor, and remained dry for the whole accident time frame. Water in the lower compartment reached its highest level (10.6 ft) after ice depletion at 27 hrs. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity. These MAAP results assumed that only 68% of the RWST inventory was injected into the containment due to the unsuccessful switch to recirculation mode at 32% low level setpoint.

Substantial concrete floor erosion due to core-concrete attack began 30 minutes after vessel failure. At the end of the 48 hr time frame, concrete erosion depth totaled 6 ft, which yielded an average erosion depth rate of 0.13 ft/hr. Non-condensible gases generated from the core-concrete attack caused containment pressure to increase at the rate of 0.4 psi/hr. The containment pressure reached 30.5 psi at 48 hrs. To reach the containment ultimate pressure at this rate, it would take 100 hrs (4 days).

This accident sequence resulted in 28.7% oxidation of in-core Zircaloy cladding. About 504 lbs of hydrogen burned in the lower compartment, in the ice condenser upper plenum and in the upper compartment.

Intermittent autoignition burns, due to high gas temperature, occurred inside the cavity for 10.5 hrs totaling 513 lbs of hydrogen.

The maximum pressure and temperature in the upper compartment were 37.2 psia and 1369°F, respectively. These maximum conditions occurred for a very short time during the burn in the upper compartment at 3.7 hrs.

The release of fission products to the environment was assumed to occur through normal leakage (2×10^{-5} ft^2) from the annular compartment since the containment did not fail during this accident scenario. Using CsI an indicator for volatile fission products and CeO_2 as an indicator for the non-volatile fission products, environmental release for this accident scenario is calculated to be:

	Airborne Release @ 48 hrs.
Noble gases (%)	0.19
Volatile fission product (CsI) (%)	1.4×10^{-5}
Non-volatile fission product (CeO_2) (%)	1.6×10^{-6}

Large LOCA - Sequence 8 (LLO-8)

Sequence Description:

This accident scenario is also initiated by a 4.0 sq. ft. (27 inch equivalent diameter) large break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes for this sequence were modeled for the source term analysis based on a 48 hour mission time:

(1) ACC - 3 of 3 accumulators inject to the cold legs

(2) LPI - 1 of 2 RHR pumps to 1 of 3 intact cold legs

(3) CSI - 1 of 2 trains of containment spray injection

(4) HI - 1 of 2 trains of hydrogen igniters in both upper and lower compartment were operational.

This sequence assumed an operational failure to establish low pressure recirculation (LPR) and containment spray recirculation (CSR). Automatically actuated RHR injection and containment spray injection pumps were assumed operational until the water in the RWST was depleted.

Sequence Quantification:

This accident scenario results in a very rapid blowdown of primary system inventory into the lower compartment and rapid depressurization of the primary system similar to sequence LLO-5. Within half a minute after the break initiation, the primary system pressure dropped below 400 psia. Approximately 40 minutes into the accident, the RHR injection was terminated due to the depletion of the RWST inventory. Without continued injection into the primary system due to the failure of low pressure recirculation, the core was uncovered about 20 minutes later. This was followed by core relocation into the lower head at 1.9 hours

and the failure of the vessel one minute later. The timing of vessel failure was delayed by 23 minutes compared to sequence LLO-5. This was because this sequence allowed the ECCS pumps to completely drain the RWST.

There was no water in the cavity when core debris was discharged onto the cavity floor. However, water in the lower compartment reached the cavity curb height (11.2 ft) and spilled into the cavity almost immediately after vessel failure. This spill-over of water into the cavity was a result of injecting all RWST inventory into the containment. Steaming from within the cavity eventually caused the depletion of ice at 4.7 hrs. Following the ice depletion, the containment pressure increased from 19 psia at the rate of 12 psi/hr. The containment failed on overpressure at 7.4 hours.

Cavity dryout occurred at 21 hrs into the accident. Concrete floor erosion due to core-concrete attack began about 2 hours after the cavity dried out. At the end of the 48 hr time frame, concrete erosion depth totaled to 3.4 ft, which yielded an average erosion rate of 0.14 ft/hr and the temperature of core debris was 3000°F. About 1000 lbs of hydrogen was produced by concrete erosion.

This accident sequence resulted in in-vessel generation of 660 lbs of hydrogen due to 28.9% oxidation of in-core Zircaloy cladding. About 392 lbs of hydrogen were burned in the lower compartment and in the ice condenser upper plenum.

The maximum temperature in the upper compartment prior to containment failure was 212°F. After containment basemat junction failure, the containment pressure continued to increase until water in the annular and lower compartments was completely discharged out through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that one hour would be required to force the water out of containment. The containment reached 64 psia before gradually depressurizing through the break in the basemat junction. Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, environmental release for this accident scenario is calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ Containment failure
Noble gases (%)	100	---
Volatile fission product (CsI) (%)	7.7	28.0
Non-volatile fission product (CeO_2) (%)	0.12	$3.3x10^{-3}$

Large LOCA - Sequence 8' (LLO-8')

Sequence Description:

This accident scenario is the previous large LOCA sequence (LLO-8), with a failure to isolate in the annular compartment. An area equivalent to a 10" diameter pipe was assumed to be open to the outside of the containment.

Sequence Quantification:

This sequence assumed containment failure at the beginning of an accident as a result of failure to isolate the containment. The unisolated pipe precluded any substantial containment pressurization after ice depletion as was seen in sequence LLO-8. Maximum pressure attained during the sequence was less than 18 psia.

Hence, containment pressure and temperature were quite different from LLO-8. However, accident timing was very similar to sequence LLO-8 including time of core uncovery, time of core relocation, time of vessel failure, time of ice depletion, and steaming rate as well as cavity spill-over and dryout time. Similar basemat erosion was also observed.

Steaming from the cavity during a period of 17 hrs between ice depletion and cavity dryout resulted in containment pressure 2 to 3 psi above the ambient level. This slightly high pressure, compared to ambient pressure for sequence LLO-8, helped suppress revaporization of fission products. The resulting source term was similar to sequence LLO-8.

	Airborne release @ 48 hrs.
Noble gases (%)	100
Volatile fission product (%)	11.5
Non-volatile fission product (%)	0.22

Medium LOCA - Sequence 5 (MLO-5)

Sequence Description:

This accident scenario is initiated by a 0.049 sq. ft. (3 inch equivalent diameter) medium break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) ACC - 3 of 3 accumulators inject to the cold legs

(2) HP2 - one SI pump and one charging pump inject to intact cold legs

(3) OA6 - RCS cooldown using 450 gpm of AFW to SGs and 2 out of 4 SG PORVs open for steam dump

(4) CSI - 1 of 2 trains of containment spray injection operational

(5) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed an operational failure to establish high pressure recirculation (HPR) and containment spray recirculation (CSR). All ECCS pumps were assumed operational until depletion of the RWST. Auxiliary feedwater was assumed available from the time of accident initiation. A two hour delay was assumed before operators initiated RCS cooldown.

Sequence Quantification:

This accident scenario resulted in a blowdown of primary system inventory into the containment lower compartment. About one hour after the break initiation, the primary system pressure had dropped to nearly 600 psia. High pressure injection was terminated due to depletion of the RWST, 1.2 hrs into the accident. Without continued injection into the primary system due to the failure of high pressure recirculation, the core started to uncover at 1.5 hrs. This was followed by core relocation to the lower head at 5.2 hours and the failure of the vessel.

There were 9 ft of water in the cavity when core debris was discharged onto the cavity floor. Water in the lower compartment reached the cavity curb height (11.2 ft) and spilled into the cavity about 2 hours prior to vessel failure. The spill-over of water into the cavity was a result of injecting all RWST inventory into the containment. Steaming from within the cavity eventually caused the depletion of ice at 6.3 hrs. Following ice depletion, the containment pressure increased from 19 psia at the rate of 10.7 psi/hr. The containment basemat failed on overpressure at 9.3 hours.

Cavity dryout occurred at 24 hrs into the accident. Concrete floor erosion due to core-concrete attack began 3 hours after the cavity dried out. After 48 hrs, the concrete erosion depth totaled 2.8 ft which corresponded an average erosion rate of 0.14 ft/hr, and the temperature of core debris was 3000°F. About 850 lbs of hydrogen were produced during core-concrete interaction.

This accident sequence resulted in generation of 840 lbs of hydrogen due to 35.9% oxidation of in-core Zircaloy cladding. About 499 lbs of hydrogen produced during the accident were burned in the lower compartment, upper compartment, and upper plenum of the ice condenser.

The maximum temperature in the upper compartment (804°F) occurred during a hydrogen burn, immediately after vessel failure. Following the containment basemat failure, water in the lower containment was assumed to be completely discharged through the 5.6" (0.17 ft^2) break within an hour. During this time, the containment pressure continued to rise (to 62 psia) before gradual depressurization of the containment began.

Using CsI as an indicater for volatile fission products and CeO_2 as an indicator for the non-volatile fission products, environmental release for this accident scenario is calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	100	---
Volatile fission product (CsI) (%)	2.4	14.0
Non-volatile fission product (CeO_2) (%)	0.10	0.0

Medium LOCA - Sequence 35 (MLO-35)

Sequence Description:

This accident scenario is initiated by a 0.049 sq. ft. (3 inch equivalent diameter) medium break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) ACC - 3 of 3 accumulators inject to the cold legs

(2) CSI - 1 of 2 trains of containment spray injection operational

(3) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed an operational failure to successfully establish containment spray recirculation (CSR). Automatic actuation of the containment spray injection was assumed until water in the RWST was depleted.

Sequence Quantification:

This accident scenario resulted in a blowdown of primary system inventory into the lower compartment and depressurization of the primary system. About 40 minutes after the break initiation, the primary system pressure dropped to nearly 600 psia. Containment spray injection was actuated and then was terminated due to the depletion of the RWST, 1.3 hours into the accident. The core uncovered at 29 minutes. This was followed by core relocation to the lower head at 2.44 hours and failure of the vessel a minute later. The primary system pressure prior to vessel failure was about 250 psia.

There was no water in the cavity when all core debris was discharged onto the cavity floor. Water in the lower compartment reached the cavity curb height (11.2 ft.) and spilled into the cavity about 4.5 hours after vessel failure. The spill of water into the cavity was a result of injecting all RWST inventory into the containment. Steaming from within the cavity eventually caused the depletion of ice at 8.65 hrs. Following the ice depletion, the containment pressure increased from 20 psia at the rate of 12.5 psi/hr. The containment failed on overpressure at 11.1 hours.

Cavity dryout occurred 35 hrs into the accident. Concrete floor erosion due to core-concrete attack began shortly after vessel failure. The erosion was temporarily terminated for 27 hrs due to water spill-over from the lower compartment until the cavity dried out. After 48 hr the concrete erosion depth totaled 2.5 ft, which yielded an average erosion rate of 0.12 ft./hr, and the temperature of core debris was 3000°F. About 580 lbs of hydrogen were produced prior to containment failure by concrete erosion, of which 447 lbs burned in the cavity.

This accident sequence resulted in in-vessel generation of 950 lbs of hydrogen due to 35.3% oxidation of in-core Zircaloy cladding. About 403 lbs of hydrogen were burned in the lower compartment, the ice condenser upper plenum, and the upper compartment.

The maximum temperature in the upper compartment (665°F) occurred during a burn in the upper compartment about 1.5 hrs before vessel failure. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2)in the basemat junction. It was assumed that one hour would be required to push water out of the containment. The containment pressurized to 64 psia before gradually depressurizing through the break in the basemat junction.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this accident scenario is calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	94	---
Volatile fission product	2.4	23.0
Non-volatile fission product (CeO_2) (%)	4.1×10^{-4}	16.0

Medium LOCA - Sequence 40 (MLO-40)

Sequence Description:

This accident scenario is initiated by a 0.049 sq. ft. (3 inch equivalent diameter) medium break LOCA at the bottom of a horizontal run of the cold leg. Only one event tree node for this sequence (Reference 47) was modeled for the source term analysis based on a 48 hour accident time frame:

(1) ACC - 3 of 3 accumulators inject to the cold legs

Sequence Quantification:

This accident scenario resulted in a blowdown of primary system inventory into the lower compartment. The core uncovered at 28 minutes into the accident. This was followed by core relocation to the lower head at 2.44 hrs and the failure of the reactor vessel one minute later. Prior to vessel failure, the primary system was at approximately 260 psia. Due to low primary system pressure at vessel failure, transport of core debris to the lower compartment did not occur. All core debris remained in the cavity.

The cavity was dry when core debris was discharged onto the cavity floor, and remained dry for the whole accident time frame. Water in the lower compartment reached its highest level (7 ft) after ice depletion at 14.4 hrs. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity.

Substantial concrete floor erosion due to core-concrete attack began about 30 minutes after vessel failure. At the end of the 48 hr time frame, concrete erosion depth totaled 5.2 ft, which yielded an average erosion rate of 0.12 ft/hr. About 750 lbs of hydrogen were produced by concrete erosion; all 750 lbs burned in the cavity. Non-condensible gases generated from the core-concrete attack caused containment pressure to rise at the rate of 1.3 psi/hr. The containment failed on overpressure at 37.9 hrs.

This accident sequence resulted in in-vessel generation of 890 lbs of hydrogen due to 38.6% of in-core Zircaloy clad oxidation. About 504 lbs of hydrogen were burned in the lower compartment, and the ice condenser upper plenum.

The maximum temperature in the upper compartment (480°F) occurred at the time of containment failure as a result of gradual heating of containment gas by debris in the dry cavity. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 4" diameter break (0.09 ft^2) in the basemat junction. (Note that the computer analysis for this sequence used a 4" vice a 5.6" equivalent diameter failure size as discussed in Section 4.3.1. This difference was due to a typographical error in the computer input. Since the hole size assumed would have only a minor effect on the overall results, no new analysis was performed.) It was assumed that one hour would be required to push water out of the containment. The containment pressurized 53 psia before gradually depressurizing through the break in the basemat junction. Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this sequence was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	76	—
Volatile fission product	7.7	34.0
Non-volatile fission product (CeO_2) (%)	$8.0x10^{-4}$	20.0

Small LOCA - Sequence 2 (SLO-2)

Sequence Description:

This accident scenario is initiated by a 3.14 sq. inch (2 inch equivalent diameter) small break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes for this sequence (Reference 47) were modeled for source term analysis based on a 48 hours accident time frame:

(1) HP2 - 1 SI pump and 1 charging pump inject to intact cold legs

(2) OA6 - RCS cooldown using 450 gpm of AFW to SGs and 2 out of 4 SG PORVs open for steam dump

(3) CSI - 1 of 2 trains of containment spray injection operational

(4) CSR - 1 of 2 containment sprays trains switched suction from the RWST to the recirculation sump

(5) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed an operational failure to successfully establish high pressure recirculation (HPR).

Automatically actuated ECCS pumps were assumed operational until the water level in the RWST reached the 32% low level setpoint. From this point on, only containment spray recirculation was assumed to operate through the entire 48 hours. Auxiliary feedwater was assumed available from the time of accident initiation. A 2 hour delay time was assumed before operators performed the steam dump for RCS cooldown.

Sequence Quantification:

This accident scenario resulted in a blowdown of primary system inventory into the lower compartment and depressurization of the primary system. All HP2 injection was terminated when the RWST low level setpoint was reached, approximately 1 hour into the accident. Without continued injection into the primary system due to the failure of high pressure recirculation, the core uncovered at 1.94 hours. Within 3 hours after the break initiation, the primary system pressure dropped to about 200 psia. This was followed by core relocation to the lower head at 7.77 hours and the failure of the vessel one minute later.

The cavity was dry when core debris was discharged onto the cavity floor and remained dry for the whole accident time frame. Water in the lower compartments reached its highest level (10.6 ft) at the end of the 48 hr time frame. At this time, ice depletion was nearly complete. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the

cavity. This MAAP result assumed that only 68% of RWST inventory was injected into the containment due to switching to recirculation mode at the 32% low RWST level setpoint.

Substantial concrete floor erosion due to core-concrete attack began 30 minutes after vessel failure. At the end of the 48 hour time frame, concrete erosion depth totaled 5.2 ft, which yielded an average erosion rate of 0.13 ft/hr. Non-condensible gases generated from the core-concrete attack caused containment pressure to increase at the rate of 0.17 psi/hr. The containment pressure reached 26 psia at 48 hrs. To reach the containment ultimate pressure at this rate, it would take 8 days.

This accident sequence resulted in generation of 720 lbs of hydrogen gas due to 31.5% oxidation of in-core Zircaloy cladding by steam and 980 lbs of hydrogen gas due to core-concrete attack. About 498 lbs of hydrogen produced during the accident were burned in the lower compartment, the ice condenser upper plenum, and the upper compartment. Intermittent auto-ignition burns of 544 lbs of hydrogen, due to high gas temperature, occurred inside the cavity for 12 hrs following vessel failure.

The maximum pressure and temperature in the upper compartment were 28.6 psia and 500°F, respectively. These maximum conditions occurred for a very short time and were due to a hydrogen burn in the upper compartment shortly after vessel failure.

The release of fission products to the environment was assumed to occur through normal leakage (2×10^{-5} ft^2) from the annular compartment since the containment does not fail during this accident scenario. Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this accident scenario is calculated to be:

	Airborne release @ 48 hrs.
Noble gases (%)	0.16
Volatile fission product (CsI) (%)	6.9×10^{-6}
Non-volatile fission product (CeO_2) (%)	1.2×10^{-6}

Small LOCA - Sequence 2' (SLO-2')

Sequence Description:

This accident scenario is the same as sequence SLO-2 but with a failure to isolate the containment in the annular compartment. Hence, in addition to the successful event tree nodes which are modeled in sequence SLO-2, an area equivalent to a 10" diameter pipe open to outside the containment was assumed.

Sequence Quantification:

The timing of major core and containment events were very similar to sequence SLO-2, except that timing for the events in this sequence was delayed 1 hour. This was due to a delay in the automatic actuation of the containment spray injection as a result of isolation failure. This delay in spray actuation helped conserve the RWST inventory for vessel injection use.

Fission product release to the environment increased several orders of magnitude from sequence SLO-2. They are calculated to be:

	Airborne release @ 48 hrs.
Noble gases (%)	90.5
Volatile fission products (CsI) (%)	$4.1x10^{-2}$
Non-volatile fission products (CeO_2) (%)	$2.9x10^{-3}$

Small LOCA - Sequence 5 (SLO-5)

Sequence Description:

This accident scenario is initiated by a 3.14 sq. inch (2 inch equivalent diameter) small break LOCA at the bottom of a horizontal run of the cold leg. The following event tree nodes of this sequence (Reference 47) were modeled for source the term analysis based on a 48 hour accident time frame:

(1) HP2 - 1 SI pump and 1 charging pump inject to intact cold legs

(2) OA6 - RCS cooldown using 450 gpm of AFW to SGs and 2 out of 4 SG PORVs open for steam dump

(3) CSI - 1 of 2 trains of containment spray injection operational

(4) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed an operational failure to establish high pressure recirculation (HPR) and containment spray recirculation. Automatic operation of high pressure injection and containment spray injection pumps was assumed until the depletion of the RWST. Auxiliary feedwater was assumed available from the time of accident initiation. A two hour delay was assumed before operators performed the steam dump for RCS cooldown.

Sequence Quantification:

This accident scenario resulted in a blowdown of primary system inventory into the lower compartment and depressurization of the primary system. For two hours after the break initiation, primary system pressure remained in the neighborhood of 1000 psia. All injection was terminated due to depletion of the RWST, 1.3 hours into the accident. Without continued injection into the primary system due to the failure of high pressure recirculation, the core uncovered at 5.6 hours. This was followed by core relocation to the lower head at 7.7 hours and the failure of the vessel one minute later.

There was no water in the cavity when core debris was discharged onto the cavity floor. Water in the lower compartment reached the cavity curb height (11.2 ft) and spilled into the cavity about 15 seconds after vessel failure. Steaming from within the cavity eventually caused the depletion of ice at 10 hrs. Following the ice depletion, the containment pressure increased from 18.5 psia at the rate of 9.2 psi/hr. The containment failed on overpressure at 13.5 hours.

Cavity dryout occurred at 30 hrs into the accident. Concrete floor erosion due to core-concrete attack began 3 hours after the cavity dried out. After 48 hrs, the concrete erosion depth totaled 2.1 ft which yielded an

average erosion rate of 0.14 ft/hr, and the temperature of core debris was 3000°F. About 820 lbs of hydrogen were produced by concrete erosion.

This accident sequence results in in-vessel generation of 760 lbs of hydrogen due to 33.2% oxidation of in-core Zircaloy cladding. About 365 lbs of hydrogen produced during the accident were burned in the lower compartment, the ice condenser upper plenum and in the upper compartment.

The maximum 630°F temperature in the upper compartment occurred during hydrogen burn in this compartment about 50 minutes prior to vessel failure. After containment failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" break (0.17 ft^2) in the basemat junction. It was assumed that one hour would be required to push water out of the containment. The containment pressurized to 61 psia before depressurizing gradually through the break.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this accident scenario is calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	100	---
Volatile fission product	1.5	9.0
Non-volatile fission product (CeO_2) (%)	9.5×10^{-3}	0.0

Small LOCA - Sequence 35 (SLO-35)

Sequence Description:

This accident scenario is initiated by a 3.14 sq. inch (2 inch equivalent diameter) small break LOCA at the bottom of a horizontal run of the cold leg. Only two event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) CSI - 1 of 2 trains of containment spray injection operational

(2) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment operational.

Automatic operation of containment spray injection was assumed until the depletion of the RWST.

Sequence Quantification:

This accident scenario results in relatively high primary system pressure until vessel failure. Containment spray injection was actuated and then was terminated due to the depletion of the RWST, 1.5 hours into the accident. Due to the failure of high pressure injection, the core uncovered at about 1 hr. This was followed by core relocation to the lower head at 2.63 hours and the failure of the vessel one minute later. At the time of vessel failure, the primary system pressure was about 400 psia. High pressure melt ejection caused 80% of the core debris to be transported to the lower compartment.

There was no water in the cavity when core debris was discharged onto the cavity floor. However, water in the lower compartment reached the cavity curb height (11.2 ft) and spilled into the cavity almost immediately after vessel failure, preventing cavity floor core-concrete attack. Steaming (mostly from the lower compartment) eventually caused the depletion of ice at 10.6 hrs. Following the ice depletion, the containment pressure increased from 18 psia at the rate of 6.0 psi/hr. The containment failed on overpressure at 15.9 hours.

Dryout in the cavity and lower compartment did not occur during the accident time frame. Thus, no concrete erosion occurred except for a small initial erosion of lower compartment floor concrete of (0.008 ft depth) following high pressure melt ejection. The erosion was immediately terminated due to the presence of a large amount of water in the lower compartment.

This accident sequence results in in-vessel generation of 740 lbs of hydrogen due to 32.5% oxidation of in-core Zircaloy cladding. About 387 lbs of hydrogen produced during the accident were burned in the lower compartment, the ice condenser upper plenum, and the upper compartment.

The maximum 580°F temperature in the upper compartment occurred for a very short time during hydrogen burn in the upper compartment slightly, prior to vessel failure. After containment failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that one hour would be required to push water out of the containment. The containment pressurized to 62 psia before depressurizing gradually through the break.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this accident scenario was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	100	---
Volatile fission product (CsI) (%)	5.4	24.0
Non-volatile fission product (CeO_2) (%)	1.8×10^{-4}	2.8

Steam Generator Tube Rupture - Sequence 3 (SGR-3)

Sequence Description:

This accident scenario is initiated by a steam generator tube rupture involving a double ended rupture of a single tube (0.943 sq. in break area). The following event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident frame:

(1) AF2 - 1 of 3 auxiliary feedwater pumps to intact steam generators (a total 200,000 lb/hr aux. feed flow was assumed based on minimum total AFW flow specified in the EOP E-0 (Reference 4).

(2) HPI - 1 charging pump injecting to 1 of 4 cold legs

(3) SGI - closure of MSIV on faulted steam generator

(4) SSV - all secondary relief valves on faulted S/G not stuck open

(5) CSI - 1 of 2 trains of containment spray operational

(6) CSR - 1 of 2 containment spray trains switched suction from RWST to recirculation sump

(7) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed failure to establish RCS cooldown and depressurization.

A charging pump was assumed operational until the water level in the RWST reached the 32% low level setpoint. After this point, containment spray would be in recirculation mode.

Sequence Quantification:

This accident scenario resulted in a pressure decrease in the primary system to the level of the steam generator safety valve setpoint. The primary system discharged coolant to the secondary side of the faulted steam generator through the tube rupture area until equilibrium was attained. Within 20 minutes after the accident initiation, the primary system pressure dropped to about 1300 psia and remained at this level until high pressure injection using one charging pump was terminated (7.8 hrs) into the accident due to the RWST low level. Following charging pump termination, both the RCS and the faulted steam generator stabilized at 1080 psia until vessel failure, which occurred at 29.7 hrs.

Up to the time of vessel failure, containment conditions remained essentially undisturbed since there was no direct discharge of primary system fluid into the containment. The ice mass remained unchanged. Containment spray (in recirculation mode) was actuated only at vessel failure. High pressure melt ejection during vessel failure transported about 70% of core debris to the lower compartment; the rest was left in the cavity.

This accident sequence resulted in 940 lbs of hydrogen generation due to 40% oxidation of in-core Zircaloy cladding and 42 lbs of hydrogen due to minimal core-concrete attack. About 107 lbs of hydrogen burned in the lower compartment, the upper compartment, and the ice condenser upper plenum, and 51 lbs burned in the cavity. A fraction of in-core hydrogen escaped to the environment through the S/G safety valves.

The maximum pressure and temperature in the upper compartment were 22.2 psia and 170°F, respectively. These maximum conditions occurred as a spike due to hydrogen burn in the upper compartment shortly after vessel failure.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this sequence is calculated to be:

	Airborne release @ 48 hours
Noble gas (%)	13.5
Volatile fission products (CsI) (%)	0.4
Non-volatile fission products (CeO_2) (%)	3.7×10^{-6}

Steam Generator Tube Rupture - Sequence 13 (SGR-13)

Sequence Description:

This accident scenario is initiated by a steam generator tube rupture involving a double ended rupture of a single tube (0.943 sq. in. break area). The following event tree nodes for this sequence were modeled for the source term analysis based on a 48 hr accident time frame:

(1) AF2 -	1 of 3 auxiliary feedwater pumps to intact steam generators (total 200,000 lb/hr aux. feed flow was assumed)
(2) HPI -	1 charging pump injecting to 1 of 4 cold legs
(3) SGI -	closure of MSIV on faulted steam generator
(4) SSV (failure) -	secondary relief valves on faulted S/G stuck open
(5) CSI -	1 of 2 trains of containment spray operational
(6) CSR -	1 of 2 containment spray trains switched suction from RWST to recirculation sump
(7) HI -	1 of 2 trains of hydrogen igniters in both upper and lower containment is operational.

This sequence assumed loss of the faulted steam generator integrity when the S/G PORV stuck open at 30 minutes into the accident. The successful nodes of this sequence are similar to sequence SGR-3 except that the faulted steam generator was overfilled leading to the stuck-open S/G PORV. Therefore, the containment bypass to the environment was established through the tube rupture and the stuck-open S/G PORV.

A charging pump was assumed operational until the water level in the RWST reached the 32% low level setpoint. Containment spray, if actuated, would be in recirculation mode after this point.

Sequence Quantification:

This accident scenario resulted in a primary system pressure decrease to the level of the steam generator safety valve setpoint. Further depressurization occurred when the S/G PORV stuck open. The primary system pressure dropped to ~ 550 psia prior to termination of high pressure injection at 6.7 hrs into the accident. After the termination of high pressure injection, the primary system pressure dropped quickly to 330 psia. Core uncovery started at 13.8 hrs. This was followed by core relocation to the lower head at 15.7 hrs and failure of the reactor vessel one minute later. At the time of vessel failure, the RCS pressure was about 370 psia. All discharged core material remained in the cavity.

Blowdown following vessel failure caused ice melting and containment spray actuation in recirculation mode. Water from ice melting was available for spray recirculation. The cavity was dry when core debris was discharged onto the cavity floor, and remained dry for the whole accident sequence. Water in the lower compartments reached its highest level (2.8 ft) after ice depletion at 48 hrs. At this time there was more than a million pounds of ice left unmelted. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity.

Substantial concrete floor erosion due to core-concrete attack began 2 hours after vessel failure. At the end of the 48 hr time frame, concrete erosion depth totaled 4 ft, which yielded an average erosion depth rate of

0.13 ft/hr. Non-condensible gases generated from the core-concrete attack caused containment pressure to increase at the rate of 0.1 psi/hr. The containment pressure reached 19.5 psi at 48 hrs.

The maximum pressure and temperature in the upper compartment were 20.4 psi and 154°F, respectively. These conditions occurred during the primary system blowdown following vessel failure.

This accident resulted in 980 lbs of hydrogen generation due to 37.4% oxidation of in-core Zircaloy cladding and 850 lbs of hydrogen due to core-concrete attack in the cavity. About 612 lbs of hydrogen burned in the cavity, and 95 lbs burned outside the cavity. A fraction of in-core hydrogen escaped through the stuck open S/G PORV.

The release of fission products to the environment mostly occurred through the stuck open S/G PORV. Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, release to the environment for this sequence was calculated to be:

	Airborne release @ 48 hrs.
Noble gases (%)	90.0
Volatile fission products (CsI) (%)	25.0
Non-volatile fission products (CeO_2) (%)	8.2×10^{-3}

Steam Generator Tube Rupture - Sequence 50 (SGR-50)

This accident scenario is initiated by a steam generator tube rupture involving one steam generator. A double ended rupture of a single tube (0.943 sq. in. break area) was assumed for this analysis. This sequence only models one event tree node, i.e., AF2 - a supply of auxiliary feedwater to the intact SGs (Reference 47). A total 200,000 lb/hr of auxiliary feedwater flow distributed equally to the 3 intact SGs was assumed. The MSIV on the faulted steam generator was not closed, due to the failure to isolate the faulted steam generator. This resulted in containment bypass to the environment. The bypass path would be from the primary system to the secondary system and eventually to the environment.

Sequence Quantification:

This accident scenario resulted in a pressure decrease in the primary system through the leak at the tube rupture, due to the pressure difference between the primary system and the secondary side of the faulted steam generator. Within 20 minutes after the accident initiation, the primary system pressure has decreased to about 1100 psia, and it remained at this level for 3 hrs. Pressure in the secondary side of intact steam generators increased rapidly from 692 psia (at accident initiation time) to about 1080 - 1090 psia and remained in this range until vessel failure. Pressure in the faulted SG decreased to a level below 200 psia and stabilized at this level for about an hour before dropping further.

The core uncovered at 2.9 hrs. This was followed by core relocation to the lower head at 4.3 hrs. and failure of the reactor vessel one minute later. Primary system pressure was 530 psia prior to vessel failure, and high pressure melt ejection resulted in the transport of 95% of core materials discharged from the RPV to the lower compartment.

Ice depletion occurred at 14 hrs. About 4.5 hrs later the containment failed on overpressure. This accident sequence resulted in 35.6% oxidation of Zircaloy cladding. No hydrogen burn was observed during the

sequence since most in-vessel hydrogen leaked out to the environment prior to vessel failure. No severe core-concrete attack occurred during the 48 hrs time frame.

The maximum temperature of 260°F in the upper compartment prior to containment failure occurred during the blowdown following vessel failure. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that 35 minutes would be required to push water out of the containment. The containment pressurized to 56 psia before gradually depressurizing through the break in the basemat junction.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, release to the environment for this sequence was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	100	---
Volatile fission products (CsI) (%)	47.3	19.0
Non-volatile fission products (CeO_2) (%)	9.6×10^{-3}	6.2×10^{-3}

Interfacing System LOCA - Sequence 5 (ISL-5)

This accident scenario is initiated by a RHR cooldown suction LOCA due to the failure of two motor operated valves which are the pressure interface between the primary system and the RHR pump suction piping. In this sequence, primary system fluid at full reactor pressure is postulated to pass into a low pressure segment of the RHR system where it causes the RHR pump seals to fail. An evaluation of the RHR pump seals identified an upper bound leakage area of 0.1 ft^2 if both pump seals fail (Reference 47). This upper bound break area is in the range of a medium LOCA. Hence a rapid depressurization of the RCS due to the loss of inventory from the break would be expected for this accident scenario. This break area also constitutes a relatively large area through which fission products are transported out of the containment and into the auxiliary building. This quantification assumed the upper bound leakage area of 0.1 ft^2. In addition to the loss of RCS inventory to the auxiliary building, RCS depressurization is accelerated by the fluid loss through RHR system relief valves. These three relief valves are connected to the RHR piping to provide overpressure protection and relieve, to the containment, 900 gpm each at 450 psig. Two event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) BRH - RHR piping does not fail but pump seal leakage in both of the RHR trains results

(2) HP2 - 1 of 2 charging pumps and 1 of 2 SI pumps inject to 1 intact cold leg

Since emergency core cooling recirculation is lost for this type of accident, high pressure ECCS injection (HP2) are assumed to operate until the depletion of the RWST inventory. Operator actions to isolate the RHR seal LOCA is assumed unsuccessful and, since this sequence fails to isolate the break, no mitigation actions for RCS cooldown and depressurization are considered further.

Sequence Quantification:

This accident scenario resulted in a rapid blowdown of primary system inventory directly into the auxiliary building without passing through the containment volume. There was a slight containment pressure increase due to a small discharge of steam into the containment through the RHR relief valves. However, substantial pressurization of the containment did not occur until vessel failure. Within an hour after the RWST inventory depletion, which occurred at 4.9 hours, the core uncovery began. Vessel failure occurred at 6.7 hrs from the time of accident initiation.

The cavity was dry when core debris was discharged onto the cavity floor, and remained dry for the whole accident time frame. Water in the lower compartment reached its highest level (5.3 ft) after ice depletion at 45 hrs. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity.

Substantial concrete floor erosion due to core-concrete attack began about 1 hour after vessel failure. At the end of the 48 hr time frame, concrete erosion depth totaled 5.3 ft, which yielded an average erosion rate of 0.13 ft/hr. Containment pressure reached its maximum value of 18.8 psia during the RCS blowdown following vessel failure.

This accident resulted in 830 lbs hydrogen generation due to 36.2% oxidation of in-core Zircaloy cladding and 970 lbs hydrogen due to core-concrete attack in the cavity. About 658 lbs of hydrogen burned in the cavity, and 73 lbs burned outside the cavity. A fraction of in-core hydrogen escaped to the auxiliary building.

The release of fission products from the primary system into the auxiliary building during the sequence is calculated to be:

	Airborne release @ 48 hrs.
Noble gases (%)	99.9
Volatile fission products (CsI) (%)	93
Non-volatile fission products (CeO_2) (%)	0.25

A large part of the fission product release from the primary system would plate out in the auxiliary building. If the results of an EPRI evaluation of containment bypass through a typical auxiliary building (Ref. EPRI NP-6586-L Volume 1) is used as a basis, the decontamination factors expected for volatile and non-volatile fission products are between 7.5 and 20 and between 11.4 and 67.5, respectively. Based upon these bounds, the estimated environmental release is:

	Airborne release @ 48 hrs.
Noble gases (%)	99.9
Volatile fission products (CsI) (%)	4.7 - 12.5
Non-volatile fission products (CeO_2) (%)	3.8×10^{-3} - 2.3×10^{-2}

Station Blackout - Sequence 50 (SBO-50)

This accident scenario is initiated by a loss of offsite AC power accompanied by the loss of the onsite emergency AC power distribution system; therefore, critical safety and support systems which rely on AC power were not available.

Since RCP seal support systems were unavailable, leakage of RCS fluid through the RCP seals occurred. Seal leakage flow rate was determined based upon the success or failure of the first three event tree nodes, i.e., (1) turbine-driven AFW, (2) reactor cooldown, and (3) continuation of AFW. This determined the time limit by which power must be restored to prevent core uncovery as discussed in the event tree notebook (Reference 47). The RCP seal leakage flow rate was then determined from a Westinghouse curve representing the relationship between core uncovery time and RCP seal leakage (Reference 39). The leakage flow rate for this sequence was 96 gpm per pump. The following event tree nodes for this sequence (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame.

(1) AFT -	Turbine-driven pump delivers flow to steam generators for 4 hrs.
(2) RCD -	RCS cooled down by 2 of 4 steam generator PORVs. One hour delay was assumed for this action.
(3) AFC -	Turbine-driven pump continues to deliver flow to steam generators for additional 2 hours past AFT 4 hour mission time.

Sequence Quantification:

This accident scenario resulted in high primary system pressure until vessel failure. With the availability of turbine-driven auxiliary feedwater for 6 hours and the success of RCS cooldown, the primary system pressure dropped to 300 $\sim$ 400 psi for about 8 hours, causing the seal leakage flow to be mostly liquid. As a result, the containment pressure increased slowly to 44 psia without substantially melting the ice. Core uncovery was delayed until 12.23 hrs into the accident. This was followed by core relocation to the lower head at 14.03 hrs, and the failure of the reactor vessel a minute later. Prior to vessel failure, the primary system was at approximately 1700 psia. High pressure melt ejection during vessel failure caused about 87% of the core debris to be transported to the lower compartment. The remainder accumulated in the reactor cavity. Substantial ice melting did not occur until vessel failure.

This accident sequence results in in-vessel generation of 960 lbs of hydrogen due to 42.1% oxidation of in-core Zircaloy cladding. No concrete erosion occurred. The temperature of core debris at 48 hrs was 1900°F in the lower compartment and 1100°F in the cavity.

Water in the lower compartment reached its highest level (7.0 ft) after ice depletion at 24.6 hrs. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity. However, because only 13% of core debris was in the cavity, water expelled from the vessel during vessel failure kept the cavity wet for 7 hours. Steaming from both the lower compartment and from the cavity eventually caused the depletion of ice at 24.6 hrs. Following the ice depletion, the containment pressure increased from 21 psia at the rate of 8.8 psi/hr. The containment failed on overpressure at 27.9 hours. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that 35 minutes would be required to push water out of the containment. The containment pressurized to 56 psia before gradually depressurizing through the break in the basemat junction.

Using CsI as an upper bound indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this sequence was calculated to be

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	82.0	---
Volatile fission products (CsI) (%)	1.6	30.0
Non-volatile fission products (CeO_2) (%)	3.3×10^{-9}	1.6×10^{-3}

Station Blackout - Sequence 190 (SBO-190)

This accident scenario is initiated by a loss of offsite AC power accompanied by the loss of the onsite emergency AC power distribution system; therefore, critical safety and support systems which relied on AC power were not available.

Since RCP seal support systems were unavailable, leakage of RCS fluid through the RCP seals occurred. The RCP seal leakage flow rate depended upon the success or failure of the first three event tree nodes, which determined the time limit by which power must be restored to prevent core uncovery, as discussed in the event tree notebook (Reference 47). The RCP seal leakage flow was then determined from a Westinghouse curve representing the relation between core uncovery time and RCP seal leakage [Westinghouse, 1986]. Since this sequence has no successful nodes, the leakage flow for this sequence was determined to be equivalent to the maximum rate expected for the seal leakage (Reference 47). This was 480 gpm per pump.

Sequence Quantification:

This accident scenario resulted in high primary system pressure until vessel failure. Without the availability of turbine-driven auxiliary feedwater, the core uncovered at 1.7 hrs into the accident. This was followed by core relocation to the lower head at 3.3 hrs and the failure of the reactor vessel a minute later. Prior to vessel failure, the primary system pressure was about 450 psia. High pressure melt ejection during vessel failure caused about 10% of core debris to be transported to the lower compartment. The rest was left in the cavity.

Water in the lower compartment reached its highest level (7.0 ft) after ice depletion at 14 hrs. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity. Steaming from the lower compartment eventually caused the depletion of ice at 14 hrs. Following the ice depletion, the containment pressure increased from 20 psia at the rate of 2.76 psi/hr. The containment failed on overpressure at 25.2 hours.

Concrete floor erosion due to core-concrete attack began 1.5 hours after vessel failure. At the end of 48 hr time frame, concrete erosion depth totaled 5.2 ft which yielded an average erosion rate of 0.12 ft/hr. The temperature of core debris in the cavity was 2800°F. About 960 lbs of hydrogen were produced by concrete erosion, of which 532 lbs burned in the cavity.

This accident sequence resulted in in-vessel generation of 760 lbs of hydrogen due to 33.4% oxidation of in-core Zircaloy cladding. About 251 lbs of hydrogen were burned in the lower compartment, the ice condenser upper plenum, and the upper compartment.

The maximum temperature in the upper compartment (520°F) occurred during a burn in the upper compartment about 2 hrs after vessel failure. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that 35 minutes would be required to push water out of the containment. The containment reached 53 psia before gradually depressurizing through the break in the basemat junction.

Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this sequence was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	89	---
Volatile fission products (CsI) (%)	0.4	48.0
Non-volatile fission products (CeO_2) (%)	1×10^{-3}	7.5

Station Blackout - Sequence 190′ (SBO-190′)

Sequence Description:

This accident scenario consisted of the addition of containment isolation failure to sequence SBO-190. A failure area equivalent to a 10" diameter pipe, open to outside the containment from the annular compartment, was assumed.

Sequence Quantification:

This sequence assumed containment failure from the beginning of an accident as a result of failure to isolate the containment. The unisolated pipe precluded any substantial containment pressurization following ice depletion, as was seen in sequence SBO-190. Hence, containment pressure and temperature were quite different from SBO-190. However, accident timings were very similar to sequence SBO-190 including the time of core uncovery, the time of core relocation, and the time of vessel failure. Ice depletion occurred 3.5 hrs earlier than for the base case. No hydrogen burn was observed.

Using CsI and CeO_2 as indicators for release of volatile and non-volatile fission products, respectively, the environmental release for this accident scenario was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	100	---
Volatile fission products (CsI) (%)	5.32	---
Non-volatile fission products (CeO_2) (%)	5.65×10^{-2}	---

Station Blackout - Sequence 181 (SBO-181)

Sequence Description:

As described in sequence SBO-50, RCP seal leakage was determined based upon the success or failure of the first three event tree nodes. This station blackout sequence was not successful in any of thr event tree nodes, resulting in maximum seal leakage of 480 gpm per pump. The following event tree nodes (Reference 47) were modeled for the source term analysis based on a 48 hour accident time frame:

(1) XH5 - power restored within 1 hour with additional 1/2 hour for equipment recovery

(2) CSI - 1 of 2 trains of containment spray injection after 1.5 hour

(3) CSR - 1 of 2 containment spray trains switched suction from the RWST to the recirculation sump

(4) HI - 1 of 2 trains of hydrogen igniters in both upper and lower containment is operational after 1.5 hours

Sequence Quantification:

This accident scenario resulted in high (1100 ~ 1200 psia) primary system pressure until core uncovery at 1.73 hrs into the accident. This was followed by core relocation to the lower head at 3.33 hrs and the failure of the reactor vessel a minute later. Prior to vessel failure, the primary system pressure was slightly above 400 psia. High pressure melt ejection during vessel failure caused about 38% of core debris to be transported to the lower compartment. The rest was left in the cavity.

The cavity was dry when core debris was discharged onto the cavity floor, and it remained dry for the whole accident time frame. Water in the lower compartment reached its highest level.(10.3 ft) at the end of 48 hr time frame. At this time, there was still 200,000 lbs of ice in the ice condenser. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity. This MAAP analysis assumed that only 68% of RWST inventory was injected into the containment, due to the switch to recirculation mode at the 32% low level setpoint.

Substantial concrete floor erosion due to core-concrete attack began about 2 hours after vessel failure. At the end of the 48 hr time frame, concrete erosion depth totaled 3.7 ft, which yielded an average erosion rate of 0.09 ft/hr. Non-condensible gases generated from the core-concrete attack caused containment pressure to increase at the rate of 0.1 psi/hr. The containment pressure reached 20.4 psi at 48 hrs. To reach the containment ultimate pressure at this rate, it would take 14.6 days.

This accident sequence resulted in generation of 745 lbs hydrogen due to 32.5% oxidation of in-core Zircaloy cladding and 660 lbs. hydrogen due to core-concrete attack. About 489 lbs of hydrogen produced during the accident were burned in the lower compartment, the ice condenser upper plenum, the upper compartment, and the annular compartment. Intermittent autoignition burns of 513 lbs hydrogen, due to high gas temperature occurred inside the cavity for 6 hrs. No other burns were observed after 12 hrs into an accident.

The maximum pressure and temperature in the upper compartment were 25 psia and 524°F, respectively. These maximum conditions occurred for a very short time during a hydrogen burn in the upper compartment just prior to vessel failure.

The release of fission products to the environment was assumed to occur through normal leakage (2 x 10^{-5} ft^2) from the annular compartment since the containment did not fail during this accident scenario. Using CsI

as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this accident scenario was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	0.15	---
Volatile fission products (CsI) (%)	1.1×10^{-5}	---
Non-volatile fission products (CeO_2) (%)	3.4×10^{-7}	---

Loss of Offsite Power - Sequence 21 (LSP-21)

Sequence Description:

This accident scenario is initiated by a loss of offsite AC power to the normal distribution lines serving the station and is followed by immediate restoration of an alternate or emergency AC power source to at least one safety bus. This sequence does not model any event tree nodes (Reference 47).

Sequence Quantification:

This accident scenario resulted in high (~2300 psia) primary system pressure until vessel failure at 4 hours into the accident. High pressure melt ejection during vessel failure caused about 60% of core debris to be transported to the lower compartment. The rest was left in the cavity.

Water in the lower compartment reached its maximum level (7 ft) at the time of ice depletion. This water level was lower than the cavity curb height (11.2 ft). Hence, water accumulated on the lower compartment floor but never spilled into the cavity. Steaming from the lower compartment eventually caused the depletion of ice at 9.7 hrs. Following the ice depletion, the containment pressure increased from 18 psia at the rate of 6.9 psi/hr. The containment failed on overpressure at 14.3 hours.

No substantial core-concrete attack was observed for the entire 48 hr time frame due to the proportional distribution of core debris in the cavity and in the lower compartment, which resulted in temperature of 2500°F for debris in the cavity and of 1600°F for debris in the lower compartment.

This sequence resulted in in-vessel generation of 880 lbs of hydrogen due to 38.3% oxidation of in-core Zircaloy cladding. About 310 lbs of hydrogen were burned in the lower compartment, the ice condenser upper plenum, and the upper compartment.

The maximum temperature of 750°F in the upper compartment occurred during burn in the upper compartment about one hour after vessel failure. After containment basemat junction failure, the containment pressure continued to rise until water in the annular and lower compartments was completely discharged through the 5.6" diameter break (0.17 ft^2) in the basemat junction. It was assumed that 35 minutes would be required to push water out of the containment. The containment pressurized to 56 psia before gradually depressurizing through the break in the basemat junction. Using CsI as an indicator for volatile fission products and CeO_2 as an indicator for non-volatile fission products, the environmental release for this sequence was calculated to be:

	Airborne release @ 48 hrs.	Available for aqueous release @ containment failure
Noble gases (%)	88	---
Volatile fission products (CsI) (%)	6.2	29.0
Non-volatile fission products (CeO_2) (%)	5.0×10^{-7}	2.6×10^{-4}

4.7.3 Sensitivity Analysis

Reference 20 identified in-vessel and ex-vessel phenomena which might have significant effects on containment failure timing and the associated source term release. These phenomena are listed in Table 4.7-4. The purpose of this section is to summarize how these phenomenological uncertainties were addressed in the Cook Nuclear Plant IPE. In addition to addressing the uncertainties identified in NUREG-1335, sensitivity analyses were also performed to determine the range of fission product release due to the uncertainties in containment failure size and location.

4.7.3.1 Approach

In order to address the uncertainties in in-vessel and ex-vessel phenomena outlined in Reference 20 a two step approach was taken. The first step was to address the phenomena in detailed Phenomenological Evaluation Summary papers (References 53-59). These phenomenological evaluations were concerned with the unlikelihood or likelihood of containment failure mechanisms, as well as containment fragility. If the phenomena were conservatively evaluated as not contributing to containment failure, uncertainties associated with these phenomena were given no further consideration.

The second step addressed, through the variation of relevant MAAP model parameters, phenomenological uncertainties not addressed in the Phenomenological Evaluation Summaries. The ranges of MAAP model parameters for sensitivity analyses were recommended in the EPRI document EPRI TR-100167 (Reference 52). With these two steps, all phenomena relevant to the Cook Nuclear Plant are addressed.

4.7.3.2 Scope

Table 4.7-4 lists (1) phenomena identified in Reference 20 for sensitivity analyses, (2) analyses performed to address the corresponding phenomena, and (3) related MAAP parameters for sensitivity runs. Table 4.7-5 lists the suggested values of MAAP parameters in consideration of uncertainties in the parameters themselves or uncertainties in the use of those parameters.

4.7.3.3 MAAP Analyses

Through the two step approach described in the previous section, several uncertainties, as indicated in Table 4.7-4 would have no impact on containment failure and, consequently, no impact on source term. Therefore, no further analyses are required.
Phenomena that may have had an impact on source term and were studied as sensitivity cases are as follows:

- Hydrogen burn completeness

- In-vessel hydrogen production/core relocation
- Hot leg creep rupture failure in a high pressure sequence
- Reduced debris coolability
- Large basemat failure size
- Containment failure location at the equipment hatch

MAAP analyses of these phenomena are described below.

Hydrogen Burn Completeness in LLO-5

Purpose: This analysis assesses the effects of the flame flux multiplier on hydrogen burn completeness for sequence LLO-5. The base case LLO-5 showed high temperature/pressure caused by burn in the upper compartment.

Conditions: The MAAP parameter FLPHI, which represents burn completeness was increased from a value of 2 to 10 to enhance burn completeness in turbulent well mixed atmospheres caused by containment sprays.

Results: Several key timings were identical to the base case (sequence LLO-5) including time of core uncovery, vessel failure and basemat erosion. Burn characteristics differed slightly from the base case. The most notable one was an early occurrence of a burn in the upper compartment. In the sensitivity run, a burn in the upper compartment occurred 40 minutes after RPV failure (compared to 2 hours after RPV failure for the base case). As a result, ice depleted 20 minutes earlier than the base case.

Besides the effects on pressure and temperature spikes which lasted for a very short time period, the overall containment pressure and temperature behaviors were approximately identical to the base case. There was no change in fission product release to the environment from the base case. Therefore, the effects of burn completeness on the overall containment performance and source term were concluded as not important.

In-vessel Hydrogen Production in LLO-5

Purpose: This sequence assesses the effects of the core blockage model on clad oxidation and, hence, in-vessel hydrogen production.

The use of this model is generally known to reduce the in-core hydrogen production. The core blockage model in MAAP does not allow for oxidation and gas flow through core nodes after the onset of melting in that node. MAAP's core blockage model accounts for the effects of channel blockage phenomena such as surface area to volume reduction after melting, geometric deformation, hydraulic diameter reductions, and movement of unreacted Zircaloy to lower part of the core.

The core blockage model was not used in the source term calculations of the base cases.

Conditions: The core blockage model is implemented by setting FCRBLK equal to 1.

Results: The overall accident timing and containment response are similar to the base case. The clad oxidation was reduced to 20.8% compared to 28.7% for the base case. Hydrogen production was reduced by 185 lbs due to core blockage. No burn occurred in the upper compartment. Hence, the analyses of the base cases were conservative with respect to hydrogen production during an accident. There was no effect on source term as a result of implementing core blockage model.

Hot Leg Creep Rupture Failure

Purpose: In the base sequence (LSP-21), an elevated hot leg/surge line temperature up to 1230°F occurred prior to vessel failure when the RCS pressure was at an elevated pressure of 2350 psia. These elevated RCS conditions approached the low-end range of potential creep rupture failure of the hot leg. This sequence assesses the effects on source term should creep rupture of the hot leg occur.

Conditions: Hot leg failure based on a 7" break size was assumed to occur just prior to vessel failure as the hot leg temperature approached its maximum value of 1230°F.

Results: Failure of the hot leg prior to RPV failure resulted in converting a high pressure sequence into a low pressure sequence due to rapid RCS depressurization through the break area. Hence, the vessel failed at low pressure (~ 200 psia). All discharged core debris accumulated within the cavity. The cavity remained dry and the debris remained uncooled for the whole accident time frame causing more non-volatile fission product releases and severe concrete erosion. There was no direct steaming by debris from the water pool in the lower compartment. Compared to the base case, ice depletion for the sensitivity case was delayed by 2 hours and containment failure was delayed by 13 hours.

Gas temperatures inside the vessel after vessel failure remained much higher (1500°F ~ 2000°F) than the base case (< 1000°F) causing more volatile fission products to come out of the primary system and deposit in the relatively cold annular and lower compartments (since all debris was in the cavity). As a result, fission products available for aqueous release through the break in the basemat increased tremendously from the base case while the volatile airborne release reduced from 6.2% to 0.6%.

	Airborne release (%)		Available for aqueous release (%)	
	Sensitivity run	Base Case	Sensitivity run	Base Case
Noble	93	88	---	---
Volatile	0.58	6.2	64.6	28.4
Non-volatile	3.2×10^{-3}	5×10^{-7}	8.3	2.6×10^{-4}

It is noted that the overall containment responses including source term were similar to an analyzed station blackout sequence (SBO-190).

Reduced Debris Coolability in SBO-50

Purpose: This sequence assesses the effects of reduced debris coolability on the source term release.

Conditions: The critical heat flux for the debris, water interface is controlled by MAAP parameter file variable FCHF. This variable was reduced from .1 to .02.

Results: The reduced debris coolability showed no effect on vessel failure timing and debris distribution (92% transported to the lower compartment). Containment pressure and temperature characteristics, ice depletion time and containment failure time were similar to the base case. However, fission product transport during the time from vessel failure to containment failure was affected. 5% more volatile fission products were released to the containment from the RPV. Slightly more fission products deposited in the annular and lower compartments were observed in this run. Hence, the amount of volatile and non-volatile fission products available for aqueous release at the time of containment failure was higher than the base case. Airborne volatile fission products slightly decreased from 1.6% for the base case to 1.5%. The comparison of source term is shown below:

	Airborne release (%)		Available for aqueous release (%)	
	Sensitivity run	Base Case	Sensitivity run	Base Case
Noble	82	82	---	---
Volatile	1.5	1.6	35	30.0
Non-volatile	$7x10^{-9}$	$3.3x10^{-9}$	$2x10^{-3}$	$1.6x10^{-3}$

Large Basemat Failure Size

Purpose: To assess the effects of basemat junction failure size on source term and to establish a range for airborne fission product release due to uncertainties in the containment failure size. The aqueous release is a function of the deposited fission products on the lower and annular compartment floors. Break size does not impact these deposits thus the aqueous release is not impacted by this sensitivity.

Sequences: All base sequences that result in containment failure due to overpressure.

Conditions: The basemat failure size is increased from a small value (0.17 ft^2) to a very large value of 40 ft^2 such that maximum source terms would be calculated. The calculated source terms represent the upper limit for the airborne while the base source terms represent the lower limit of airborne source term range defined by uncertainties in failure size. Total water loss from the containment was assumed to occur almost instantaneously following containment failure.

Results: Accident event timings and containment response characteristics were identical up to the containment failure time. The source terms are compared with the base cases in Table 4.7-6. The results show that in most cases a very large failure size leads to an increase, by a factor of 2 or less, in volatile and non-volatile airborne source term releases except for the non-volatile source term releases in seq. LSP-21 which increase by 5 orders of magnitude (from $5x10^{-7}$% to 0.01%). Another exception is SLO-35 where volatile fission product release instead of increasing, decreased slightly from 5.4% to 3.4%.

Equipment Hatch Failure

Purpose: Failure at the basemat may be small in break size resulting in continued containment pressurization and a failure at the equipment hatch prior to the expulsion of water from the lower compartment.

Sequences: All base sequences that result in containment failure due to overpressure.

Conditions: To perform MAAP analysis of equipment hatch failure, the containment ultimate pressure of 58 psig is assumed. The failure size is assumed the same as the base cases. No containment water loss is allowed to occur through the basemat junction failure (which occurs at 36 psig) to simulate a situation that water could not be pushed out of the containment due to uncertainties in failure size and location, and, possibly, soil resistance, for example. Consequently, no aqueous release of fission products is automatically assumed.

Results: Accident timings and containment response characteristics were identical to the base sequences up to the basemat failure time. For MLO-40, containment overpressure occurred at a much slow rate (~ 1.6 psi/hr); and the equipment hatch failed about 9 hours after the basemat failure.

The airborne source terms are compared with the airborne source terms of the base case in Table 4.7-6. Substantial reduction by several factors in volatile release and a reduction by a factor of 2 or less in non-volatile release would result with a failure at the equipment hatch. The exceptions to this conclusion are LLO-8 and MLO-35 where an increase in volatile release was observed.

It should be noted that the airborne source terms for cases that end in failing the equipment hatch with a partial loss of containment water would lie between the values calculated here and the base values; and the aqueous release would be smaller than the base values.

4.7.4 Conclusion

The level 2 analysis discussed above reviewed containment failure modes and the resulting source term release. Given core damage, there is an 85.4% probability containment will be successful. There is, therefore, a 14.6% probability that containment will be successful. The conditional probability of containment failure by failure mode is shown in Figure 4.7-1. The ranking of source term release categories is summarized in Table 4.7-3.

Of the bypass sequences, SGTR sequences are the major contributors to the overall source term. Emergency operating procedures instruct the operators to terminate all feedwater flow to the faulted steam generator. Should the integrity of the faulted generator not remain intact, then fission products released to the steam generator through the tube rupture can pass directly to the environment without the benefit of scrubbing by the secondary side water. Maintaining water level in the secondary side of the faulted steam generator could substantially reduce the source term.

Failure of the Cook Nuclear Plant containment will most probably be due to shear of the basemat concrete at the cylinder basemat junction. Because of the many uncertainties surrounding this failure mechanism, it was difficult to predict the characteristics of the actual breach of containment. The Cook Nuclear Plant IPE assumed that containment overpressurization would result in a 5.6" equivalent diameter hole through the basemat to the atmosphere. Another key assumption in the level 2 analysis resulting from the containment failure location, in the basemat below any expected water level in containment, was that water in the lower and annular compartments would be lost through the basemat failure and that the water must be lost before any airborne release could occur.

The size of the failure was found through sensitivity studies to have only a small effect (i.e. only a factor of 2 or less) on the overall airborne and aqueous fission product releases. The location of the containment failure, however, could dramatically lower the overall fission product release. Should containment failure occur above the water level in containment, then aqueous fission product release would be virtually eliminated. Containment failure location would not have a large effect on the overall airborne release.

The assumption that all water in the lower and annular compartments would be lost following containment failure has a great impact on the source term analysis. This assumption was predicated on the physical configuration of the containment observed during walkdowns. The lower and annular compartments of the Cook Nuclear Plant containment are separated by the crane wall which has sleeves to allow piping to pass between the compartments. While there is only a small amount of clearance between the pipe and the wall, it was assumed that the large number of these penetrations can allow water level in both the lower and annular compartments to equalize. When containment failure occurs, therefore, water inventory will be lost from both compartments.

Another significant impact of the assumption that water level equalizes in the annular and lower compartments is that the reactor cavity is dry for most containment failure sequences. The inventory of the RWST is not enough to fill both the lower and annular compartments to the level where spillover to the reactor cavity would occur. Ice melt, however, combined with RWST injection will cause spillover from the lower compartment to the reactor cavity. Because of the accident sequence timing, insufficient ice has melted prior to vessel failure to allow spillover to the reactor cavity for most accident sequences. The dry cavity results in greater fission product entrainment and airborne concentrations.

Table 4.7-1

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	L L O		
Sequence No.	5	8	8'(3)
Sequence Frequency	2.831×10^{-7}	1.124×10^{-8}	1.349×10^{-12}
Sequence Designator	ALR-S	ALC-J	ALC-U
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.62	0.91	0.90
Time of Core Relocation (hr)	1.50	1.89	1.86
Time of Vessel Failure (hr)	1.52	1.90	1.88
Time of Ice Depletion (hr)	26.95	4.72	4.52
Time of Containment Failure (hr)	None(2)	7.36	0.0
Maximum Upper Compartment Pressure (psia)	37.2	>50.7	17.9
Maximum Upper Compartment Temperature (°F)	1369	212	345
Cavity Water Level @ Vessel Failure (ft)	0.0	0.0	0.0
Fraction of Clad Reacted in Vessel (%)	28.7	28.9	28.3
Hydrogen Burn Location(1)	L,U,IU,C	L,IU	L,IU,C
Total Hydrogen Burned (lb) Outside/Inside Cavity	504/513	392/0	405/35
ENVIRONMENTAL RELEASE @ 48 hr			
Noble Release (%)	0.2	100	100
Volatile FP Release (%)(4)	1.4×10^{-5}	35.7	11.5
Non-Volatile FP Release (%)(4)	1.6×10^{-6}	0.12	0.22

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Event did not occur during 48 hrs. time frame.
3) The prime indicates containment isolation failure.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-1 (Continued)

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	M L O		
Sequence No.	5	35	40
Sequence Frequency	4.457×10^{-8}	1.777×10^{-9}	1.334×10^{-8}
Sequence Designator	SLC-J	SHC-M	SHWIF-M
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	1.48	0.47	0.47
Time of Core Relocation (hr)	5.24	2.44	2.44
Time of Vessel Failure (hr)	5.26	2.46	2.46
Time of Ice Depletion (hr)	6.30	8.65	14.39
Time of Containment Failure (hr)	9.26	11.10	37.91
Maximum Upper Compartment Pressure (psia)	>50.7	>50.7	>50.7
Maximum Upper Compartment Temperature (°F)	840	665	480
Cavity Water Level @ Vessel Failure (ft)	9.0	0.0	0.0
Fraction of Clad Reacted in Vessel (%)	35.9	35.3	38.6
Hydrogen Burn Location(1)	L,U,IU,C	L,U,IU,C	L,IU,C
Total Hydrogen Burned (lb) Outside/Inside Cavity	499/0	403/447	504/750
ENVIRONMENTAL RELEASE @ 48 hr			
Noble Release (%)	100	94	76
Volatile FP Release (%)(4)	16.4	25.4	41.7
Non-Volatile FP Release (%)(4)	0.1	16.0	20.0

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Event did not occur during 48 hrs. time frame.
3) The prime indicates containment isolation failure.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-1 (Continued)

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	S L O			
Sequence No.	2	5	35	2'(3)
Sequence Frequency	1.315×10^{-5}	3.558×10^{-7}	1.122×10^{-7}	1.578×10^{-9}
Sequence Designator	SLR-S	SLC-J	SHC-J	SLR'-E
CORE/CONTAINMENT RESPONSE				
Time of Core Uncovery (hr)	1.94	5.56	1.02	5.58
Time of Core Relocation (hr)	7.77	7.66	2.63	7.64
Time of Vessel Failure (hr)	7.79	7.68	2.65	7.66
Time of Ice Depletion (hr)	None(2)	10.0	10.6	47.3
Time of Containment Failure (hr)	None(2)	13.5	15.9	0.0
Maximum Upper Compartment Pressure (psia)	28.6	>50.7	>50.7	20.8
Maximum Upper Compartment Temperature (°F)	500	630	580	460
Cavity Water Level @ Vessel Failure (ft)	0.0	0.0	0.0	0.0
Fraction of Clad Reacted in Vessel (%)	31.5	33.2	32.5	34.9
Hydrogen Burn Location(1)	L,U,IU,C	L,U,IU,C	L,U,IU	L,U,IU,C
Total Hydrogen Burned (lb) Outside/Inside Cavity	498/544	365/0	387/0	517/915
ENVIRONMENTAL RELEASE @ 48 hr				
Noble Release (%)	0.16	100	100	90.5
Volatile FP Release (%)(4)	6.9×10^{-6}	10.5	29.4	0.04
Non-Volatile FP Release (%)(4)	1.2×10^{-6}	9.5×10^{-3}	2.8	2.9×10^{-3}

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Event did not occur during 48 hrs. time frame.
3) The prime indicates containment isolation failure.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-1 (Continued)

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	S G R		
Sequence No.	3	13	50
Sequence Frequency	5.989×10^{-6}	6.600×10^{-7}	2.900×10^{-8}
Sequence Designator	GHR-C	GHR-T	GHWIF-T
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	24.71	13.78	2.89
Time of Core Relocation (hr)	29.66	15.66	4.34
Time of Vessel Failure (hr)	29.68	15.68	4.36
Time of Ice Depletion (hr)	None(3)	None(3)	14.0
Time of Containment Failure (hr)	None(3)	None(3)	18.5
Maximum Upper Compartment Pressure (psia)	22.2	20.4	>50.7
Maximum Upper Compartment Temperature (°F)	170	154	260
Cavity Water Level @ Vessel Failure (ft)	0.0	0.0	0.0
Fraction of Clad Reacted in Vessel (%)	40.0	37.4	35.6
Hydrogen Burn Location(1)	L,U,IU,C	L,IU,C	none
Total Hydrogen Burned (lb) Outside/Inside Cavity	107/51	95/612	0/0
ENVIRONMENTAL RELEASE @ 48 hr			
Noble Release (%)	13.5	90.0	100
Volatile FP Release (%)(4)	0.4	25.0	66.3
Non-Volatile FP Release (%)(4)	3.7×10^{-6}	8.2×10^{-3}	0.02

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Decontamination factor = 7.5 - 20 (for volatile), 11.4 - 67.5 (for non-volatile).
3) Event did not occur during 48 hrs. time frame.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-1 (Continued)

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	ISL
Sequence No.	5
Sequence Frequency	3.680×10^{-8}
Sequence Designator	V-T
CORE/CONTAINMENT RESPONSE	
Time of Core Uncovery (hr)	5.68
Time of Core Relocation (hr)	6.71
Time of Vessel Failure (hr)	6.73
Time of Ice Depletion (hr)	44.9
Time of Containment Failure (hr)	None(3)
Maximum Upper Compartment Pressure (psia)	18.8
Maximum Upper Compartment Temperature (°F)	278
Cavity Water Level @ Vessel Failure (ft)	0.0
Fraction of Clad Reacted in Vessel (%)	36.2
Hydrogen Burn Location(1)	L,C
Total Hydrogen Burned (lb) Outside/Inside Cavity	73/658
ENVIRONMENTAL RELEASE @ 48 hr	
Noble Release (%)	100
Volatile FP Release (%)(4)	4.7-12.5(2)
Non-Volatile FP Release (%)(4)	3.8×10^{-3} - 2.3×10^{-2}(2)

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Decontamination factor = 7.5 - 20 (for volatile), 11.4 - 67.5 (for non-volatile).
3) Event did not occur during 48 hrs. time frame.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-1 (Continued)

CONTAINMENT STATUS AND LEVEL II SOURCE TERM SUMMARY OF DOMINANT SEQUENCES

SEQUENCE TYPE	S B O				LSP
Sequence No.	50	190	181	190'[3]	21
Sequence Frequency	5.779×10^{-7}	7.419×10^{-8}	1.970×10^{-8}	8.903×10^{-12}	6.27×10^{-8}
Sequence Designator	THWIF-M	THWIF-M	THR-S	THWIF-G	THWIF-M
CORE/CONTAINMENT RESPONSE					
Time of Core Uncovery (hr)	12.23	1.73	1.73	1.73	2.76
Time of Core Relocation (hr)	14.03	3.29	3.33	3.32	3.96
Time of Vessel Failure (hr)	14.05	3.31	3.35	3.34	3.98
Time of Ice Depletion (hr)	24.57	14.06	None[2]	10.40	9.73
Time of Containment Failure (hr)	27.93	25.17	None[2]	0.0	14.31
Maximum Upper Compartment Pressure (psia)	>50.7	>50.7	25	18.6	>50.7
Maximum Upper Compartment Temperature (°F)	265	520	524	240	750
Cavity Water Level @ Vessel Failure (ft)	0.0	0.0	0.0	0.0	0.0
Fraction of Clad Reacted in Vessel (%)	42.1	33.4	32.5	32.9	38.3
Hydrogen Burn Location[1]	None	L,U,IU,C	L,U,A,IU	None	L,U,IU,C
Total Hydrogen Burned (lb) Outside/Inside Cavity	0/0	251/532	489/513	0/0	310/77
ENVIRONMENTAL RELEASE @ 48 hr					
Noble Release (%)	82	89	0.15	100	88
Volatile FP Release (%)[4]	31.6	48.4	1.1×10^{-5}	5.32	35.2
Non-Volatile FP Release (%)[4]	1.6×10^{-3}	7.5	3.4×10^{-7}	5.65×10^{-2}	2.6×10^{-4}

NOTES:
1) Compartments: U = upper, L = lower, A = annular, C = cavity, IU = I/C upper plenum.
2) Event did not occur during 48 hrs. time frame.
3) The prime indicates containment isolation failure.
4) See Table 4.7-2 for a split between airborne and aqueous release.

Table 4.7-2

ENVIRONMENTAL RELEASE

SEQUENCES	Vessel to CMT Failure Time (hr)	Volatile Fission Products			Non-Volatile Fission Products		
		Total (%)	Airborne (%) @ 48 hours	Aqueous (%) @ CMT Failure	Total (%)	Airborne (%) @ 48 hours	Aqueous (%) @ CMT Failure
LLO-8	5.5	35.7	7.7	28.0	0.12	0.12	3.3×10^{-3}
MLO-5	4.0	16.4	2.4	14.0	0.10	0.1	0.0
MLO-35	8.6	25.4	2.4	23.0	16.0	4.1×10^{-4}	16.0
MLO-40	35.4	41.7	7.7	34.0	20.0	8×10^{-4}	20.0
SLO-5	5.8	10.5	1.5	9.0	9.5×10^{-3}	9.5×10^{-3}	0.0
SLO-35	13.2	29.4	5.4	24.0	2.8	1.8×10^{-4}	2.8
SGR-50	14.1	66.3	47.3	19.0	0.02	9.6×10^{-3}	6.2×10^{-3}
SBO-50	13.9	31.6	1.6	30.0	1.6×10^{-3}	3.3×10^{-9}	1.6×10^{-3}
SBO-190	21.9	48.4	0.4	48.0	7.5	1.0×10^{-3}	7.5
LSP-21	10.3	35.2	6.2	29.0	2.6×10^{-4}	5×10^{-7}	2.6×10^{-4}

Table 4.7-3

RELEASE CATEGORY AND PROBABILITY

RELEASE CATEGORY	DEFINITION	FREQUENCY	P(RC|CD)[1]
S	Success	5.40E-05	.854
C	Cont. bypassed - <1% volatiles released	5.99E-06	0.096
M	Late containment failure - >10% volatiles released	1.13E-06	0.018
T	Containment bypassed - >10% volatiles released	1.12E-06	0.018
J	Early containment failure - >10% volatiles released	8.41E-07	0.013
G	Containment failure prior to vessel failure - >10% volatiles released	8.48E-08	0.0013

NOTE:

1. Conditional probability of release category given core damage.

Table 4.7-4

ANALYSES TO ADDRESS UNCERTAINTIES DISCUSSED IN NUREG-1335

Phenomena	Analyses Performed	Related MAAP Parameter
• Performance of containment heat removal systems	• All source term analyses conservatively assumed only 1 train of containment spray with heat exchanger in MAAP analyses. No further sensitivity analysis is required.	
• In-vessel phenomena - H_2 production and combustion in containment	• MAAP analysis (sequence LLO-5) using a higher value of "flame flux multiplier" which promotes H_2 burn completeness	FLPHI
	• MAAP analysis (LLO-5) with and w/o core blockage model which results in different in-vessel hydrogen production	FCRBLK
	• Discussed in summary paper on hydrogen combustion; sensitivity analysis is required	
- Core relocation characteristics	• MAAP analysis (LLO-5) with and w/o core blockage model	FCRBLK
- Fuel/coolant interactions	• MAAP analysis (SBO-50) using reduced critical heat flux	FCHF
- Mode or RV meltthrough	• Discussed in summary paper on thrust forces at RPV failure. Thrust forces would not cause containment failure; sensitivity analysis on thrust forces not required	
	• All analyses assumed a 60 sec. melt-through time	
- Induced failure of RCS pressure boundary at high RCS pressure/ temperature	• SBOs assumed range of induced pump seal LOCAs from 96 gpm to 480 gpm per pump. LSP-21 assumed no seal leakage. Maximum surge line/hot leg temperature was calculated to be 1230°F of potential creep rupture failure of hot leg. Analysis of LSP-21 with hot leg failure was performed.	ABB

Table 4.7-4 (Continued)

ANALYSES TO ADDRESS UNCERTAINTIES DISCUSSED IN NUREG-1335

Phenomena	Analyses Performed	Related MAAP Parameter
• Ex-vessel Phenomena		
- Direct containment heating (at high RCS pressure)	• Addressed in summary paper on DCH as not causing containment failure; no further analysis is required	
- Potential for early containment failure due to pressure load	• Early containment failure modes due to ex-vessel steam explosions and hydrogen detonation addressed in summary papers as not causing containment failure; no further analysis is required	
	• Early containment failure (defined as less than 6 hrs after vessel failure) were analyzed in base cases including sequences LLO-8, MLO-5, SLO-5, SLO-35	
- Early failure via debris attack of containment penetrations	• Uncertainties addressed in summary paper on Penetration Thermal Attack as not causing containment failure; no further analysis is required	
- Long-term core-concrete interaction	• Discussed in MCCI summary paper as not causing basemat failure prior to containment overpressure	
	• Concrete erosion to several degrees of severity is a common phenomenon in base case sequences	
-- Water availability	• It was a common situation in most base cases that core debris remained uncooled in the dry cavity, and for high pressure sequences in the lower compartment after containment failure	
-- Debris coolability	• MAAP analysis (SBO-50) assumed reduced debris coolability by reducing FCHF	FCHF

Table 4.7-5

RANGE OF MAAP MODEL PARAMETERS
ACCORDING TO EPRI TR-100167

MAAP Parameter Range	Value Used in the Analyses
• TTRX-RV failure delay time	• All sequence used best estimate value, TTRX = 60 seconds
• FCMDCH - Fraction of debris mass that contributes to DCH	• All sequences used FCMDCH = 0.03
• FCHF - reduce from 0.1 to .02	• Sequence SBO-50 performed with FCHF=0.02. Best-estimate value for base cases was 0.1
• TJBURN - gas jet auto burn temp; increase to 3000 K	• All sequences used best estimate value of 1060K
• FLPHI - flame flux multiplier controlling H_2 burn completeness	• Sensitivity performed with LLO-5 used FLPHI = 10. Base cases used FLPHI = 2
• FCRDR; increase core "dump" fraction to 0.8 to decrease re-vaporization rate	• Revaporization was a common phenomenon after containment failure in most base cases. All analyses conservatively used low value of 0.1
• FCRBLK - core blockage	• LLO-5 performed with and w/o blockage model; base cases conservatively not used blockage model to allow more H_2 production
• TAUTO - raise auto ignition temp to 3000 K	• All base cases used best estimate value of 983K. Autoignition was common in most sequences with dry cavity.

Table 4.7-5 (Continued)

RANGE OF MAAP MODEL PARAMETERS
ACCORDING TO EPR TR-100167

MAAP Parameter	Analyses Performed
• DXHIG - offset H_2 mole fraction	• Station blackout with no power restoration used DXHIG = 1.0 to prevent burn without ignition source • Most non-station blackout base sequences burned H_2 at about 6% with igniters and at 8% without ignitors
• Set vapor pressure multiplier for CsI, CsOH revaporization to 0.1	• Revaporization was common in most base cases
• ABB - induces RCS boundary failures for high pressure sequence	• Sensitivity analysis for LSP-21 with hot leg creep rupture failure
• PCF - containment failure pressure	• Fragility curve developed in containment overpressure summary paper, a conservative low-end failure of the basemat junction was used for base cases. Equipment hatch failure was performed as sensitivity.
• ACFPR - recommend using large failure area	• Base case analyses performed with small break (leak-before-break) • Sensitivity analyses for all sequences were performed with a large containment failure size

Table 4.7-6

Range of Airborne Fission Product Release
Under Different Containment Failure Scenarios

Sequences	Small Containment Break Size (0.17 ft^2)				Large Containment Break Size (40 ft^2)	
	Equipment Hatch Failure Containment Water Not Lost[1)]		Basemat Failure Containment Water Lost[2)]		Basemat Failure Containment Water Lost[2)]	
	Volatile	Non-Volatile	Volatile	Non-Volatile	Volatile	Non-Volatile
LL0-8	12.5	0.2	7.7	0.10	10	0.3
MLO-40	2	$4x10^{-4}$	7.7	$8x10^{-4}$	12	$9x10^{-4}$
LSP-21	2.9	$4x10^{-7}$	6.2	$5x10^{-7}$	13	0.01
SLO-35	0.13	$1x10^{-4}$	5.4	$2x10^{-4}$	3.4	$4.6x10^{-4}$
MLO-5	1.7	0.12	2.4	0.1	5.6	0.26
MLO-35	10	$4x10^{-4}$	2.4	$4x10^{-4}$	7.1	$8x10^{-4}$
SLO-5	1.0	0.1	1.5	0.1	2.7	0.18
SBO-50	1.3	$3x10^{-9}$	1.6	$3x10^{-9}$	2.0	$3x10^{-9}$
SBO-190	0.2	$6x10^{-4}$	0.4	$1x10^{-3}$	0.6	$1x10^{-3}$
SGR-50	30.0	$1x10^{-2}$	47.3	$1x10^{-2}$	47.3	0.01

1) No aqueous release
2) See available fission products for aqueous release in Table 4.7-2

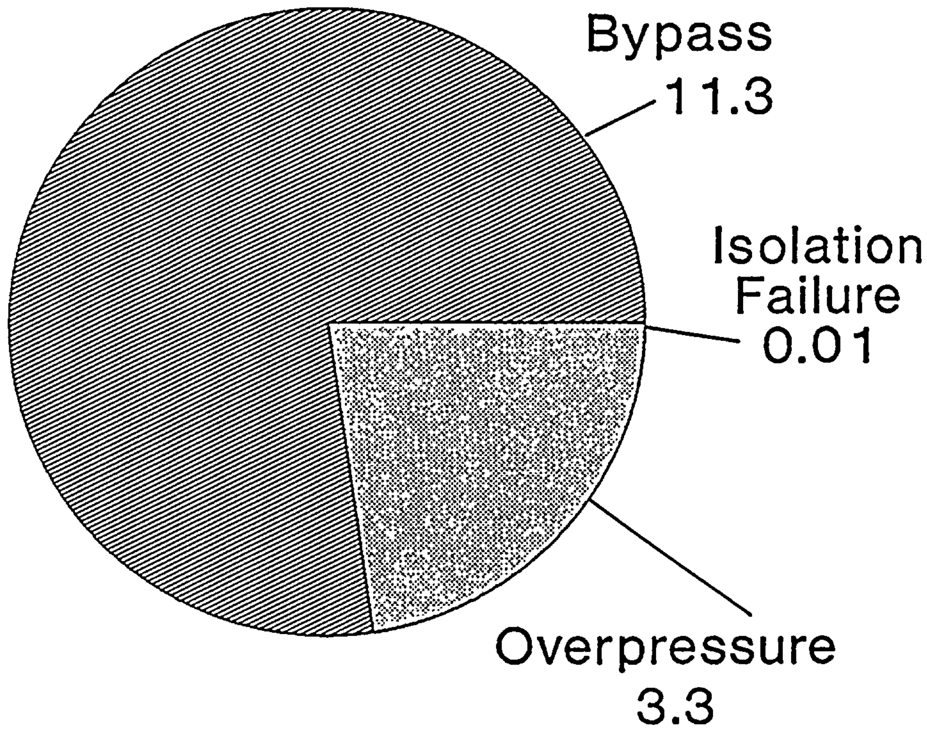


Figure 4.7-1 Conditional Probability of Containment Failure by Failure Mode

5.0 UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

AEPSC has committed substantial personnel and financial resources to its IPE program. Due to the magnitude of the Cook Nuclear Plant IPE Program, AEPSC engaged the Individual Plant Evaluation Partnership (IPEP) to support and direct the analysis efforts on the full-scope Level III PRA as well as the the IPE External Events Analysis. The IPEP companies include Westinghouse, Fauske and Associates, and TENERA. AEPSC created a Cook Nuclear Plant PRA Team which effectively utilizes its personnel resources and provides AEPSC with complete control and involvement in the IPE analyses. In the organization structure, IPEP personnel provided the overall task leadership while both the IPEP team and the AEPSC team jointly performed all the analyses. Interactions between AEPSC personnel and the IPEP team were conducted on a continual basis and intensively at each step to resolve issues and incorporate plant specific knowledge. In addition to the IPE personnel, other AEPSC engineering and support staff provided design and operational information at the direction of the program coordinator.

Figure 5.1-1 depicts the overall organizational structure for the AEPSC Cook Nuclear Plant IPE program. As shown, AEPSC established a PRA Project Coordinator who was responsible for the overall performance of the IPE project and served as the primary point of contact for the Cook Nuclear Plant PRA. For the Cook Nuclear Plant IPE, an Independent Review Team of AEPSC middle level management actively reviewed all results and insights.

The AEPSC team members were trained and involved in all aspects of the IPE project. This included performing or reviewing the Level I, and external events, reviewing the Level II analysis, and performing the Level III analysis.

The IPEP organization supported AEPSC in the Cook Nuclear Plant IPE project with a core of experienced PRA personnel, led by a Technical Project Manager. The technical project manager was responsible to the AEPSC Project Coordinator for directing and coordinating project activities and maintaining the project schedule and budget. The project manager was the primary interface between the IPEP team and the AEPSC PRA Project Coordinator.

In summary form, the following describes the task-by-task participation of the AEPSC IPE team engineers in the development of the Cook Nuclear Plant PRA:

Data Collection and Analysis - AEPSC engineers collected plant-specific data and developed the data base used for the systems analysis.

Initiating Event Analysis - Following training by Westinghouse, AEPSC engineers developed the initiating event frequencies.

Accident Sequence (Event Tree) Analysis - AEPSC co-developed this analysis with Westinghouse. AEPSC engineers also performed the "special" initiator analysis.

Systems Analysis - AEPSC engineers developed 14 of the 19 system notebooks which included the creation of the system fault tree models. The remainder, developed by Westinghouse, were extensively reviewed and approved by AEPSC.

Human Reliability Analysis (HRA) - The HRA and recovery actions were co-developed by AEPSC and Westinghouse.

Internal Flooding Analysis - Westinghouse performed this analysis. Following training, AEPSC participated in the plant walkdown and reviewed the analysis.

Quantification & Sensitivity Analyses - The quantification and sensitivity analyses were developed by AEPSC with training and continued support provided by Westinghouse.

Containment Performance Analysis - FAI was the lead for this task. FAI performed all the Level II analysis. AEPSC engineers received training on the MAAP code and reviewed and approved the Level II analysis.

Consequence Analysis - Following training by Westinghouse, the Level III analysis was developed by AEPSC with continued support from Westinghouse.

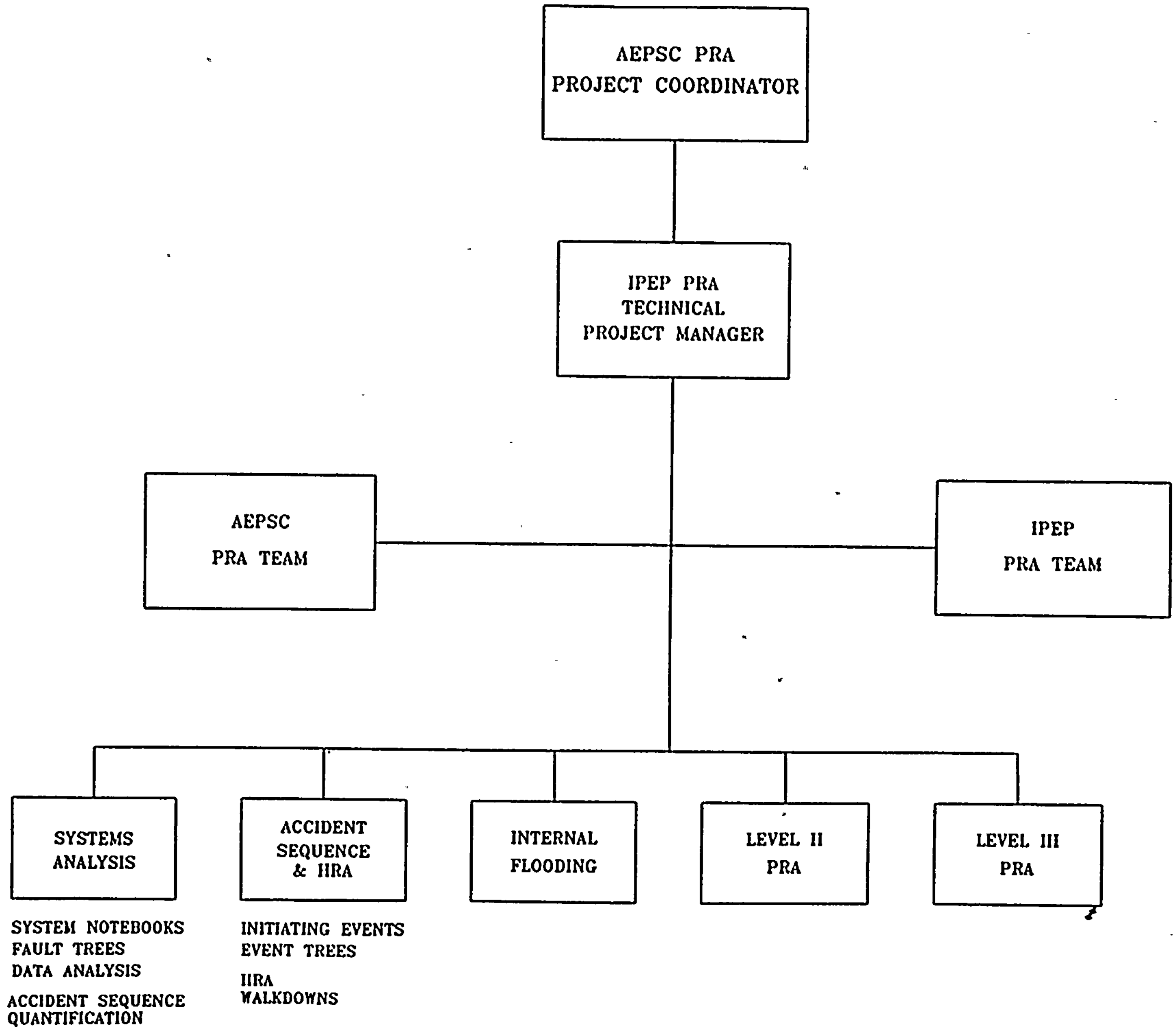


Figure 5.1-1
Cook Nuclear Plant IPE Project Organization

5.2 Composition of Independent Review Team

Although the Cook Nuclear Plant IPE program was performed to satisfy the requirements of 10CFR50 Appendix B, an additional Independent Review Team (IRT) was organized to review the IPE analysis. This team generally consisted of middle level managers from applicable engineering and operations organizations as indicated in Table 5.2-1. The IRT conducted formal meetings to review, comment on and approve all aspects of the IPE analysis. In addition to the IRT review and the independent 10CFR50 Appendix B review, the IPE analysis was reviewed internally by appropriate system engineers and operations staff members to ensure that the system model and assumptions were correct and the findings reasonable.

TABLE 5.2-1

INDEPENDENT REVIEW TEAM REPRESENTATION

Department or Division	Title
Cook Nuclear Plant Operations	Supervisor
Design Division	Manager, Structural & Analytical Design, Nuclear Section
Nuclear Engineering Division	
-Electrical	Manager, Instrumentation & Control
-Mechanical	Manager, Technical Support
Nuclear Operations Division	Group Manager, Safety, Licensing, & Assessment
	Consulting Engineer
Quality Assurance	Manager, QA Engineering

5.3 Areas of Review, Major Comments, and Resolution of Comments

All areas of the Cook Nuclear Plant PRA were subject to independent review through either the 10 CFR 50 Appendix B process, the IRT, or a support organization internal to AEPSC. While outside consultants were involved in both analysis tasks and review tasks, no single consultant performed both tasks for any part of the PRA. This approach assured AEPSC involvement in all aspects of the Cook Nuclear Plant PRA.

All comments and their respective resolutions were formally documented. Resolutions were disposed through immediate changes to the PRA models if the effects were anticipated to be significant to the results. Those resolutions anticipated to have an insignificant effect on the PRA results typically became action items to be performed in the next revision of the PRA.

Major comments and their resolution are summarized below.

Comment No. 1

"The conclusions [of the Level II analysis] did not properly reflect the culmination of the accident."

Resolution

This concern relates to the IPE team's judgement that it is appropriate to analyze only the more rapid accident progression during the first two days, and not to address the survivability of equipment needed for the significantly slower "post-accident" phase. The analysis time considered for the Cook Nuclear Plant level II PRA, 48 hours, is consistent with the standard methods used by other utilities in PRAs and, we believe, identifies the most significant vulnerabilities. The commentor's concerns with the longer accident times are that the hydrogen recombiners and air recirculation fans may be needed for the period of the accident when hydrogen is being generated due to very slow decomposition of the containment sump water. This phase of the accident is late in the accident, beyond 72 hours, and there are many actions which could be taken to mitigate the effects of this phenomena. Although it would be preferable that this equipment survive, it was judged that there would be sufficient time and manpower to maintain reasonable control of the containment in this phase of post accident cleanup. As a result, no change to the analysis was made. In addition, it is noted that the NRC has stated in Appendix C of NUREG-1335 (item 6.2) that "Because of limitations in modeling scenarios that extend over long periods of time, the nominal assumption of 24 hours is sufficient..." for evaluating accident sequences.

Comment No. 2

The viability of the MAAP code for use is analyzing hydrogen phenomena was questioned.

Response

The MAAP code is a very broad based, widely accepted code used to analyze the complete accident sequence, provide the framework for understanding the important features and systems interaction of severe accidents, and to calculate the radionuclide releases. For this analysis, the MAAP code is not used to determine the conditions which cause containment failure. The conditions which cause containment failure are determined based on phenomenological evaluations completed using hand calculations and other methods. The calculation supporting the position that hydrogen would not be likely to cause containment failure is based on manual evaluations and very limited use of the MAAP code for simple, easily verifiable cases. It was not believed the concerns regarding the use of the MAAP code were justified in this instance since the code was only used for its more widely accepted use, where the accuracy of the hydrogen modelling is of secondary importance.

Comment No. 3

"The analyses underpredict the impact of severe accident generated hydrogen on the containment integrity."

Response

This concern primarily involves the possibility that containment could fail due to a hydrogen pressure spike either due to the loss of the containment air recirculation fans or due to detonations. The containment failure modes were determined by phenomenological evaluations. Containment failure due to hydrogen combustion was shown to be improbable because the conditions necessary to generate sufficient hydrogen and to accumulate it in the proper location to cause containment failure would not likely exist. Should those conditions exist, it is much more likely that containment would have previously failed due to overpressurization caused by steaming. The phenomenological evaluation supporting this position is valid whether or not hydrogen recombiners and/or air recirculation fans are operating. The possibility of detonation was found negligible by a manual evaluation.

Comment No. 4

"The analysis fails to identify vulnerabilities that are unique to Cook Plant."

Response

This concern is related to the commentor's position on comments 2 and 3. The commentor believes that, since there is at least uncertainty in whether hydrogen can directly cause containment failure, the survivability of equipment used to control hydrogen should be addressed as a vulnerability, particularly the containment air recirculation fans. Since the IPE team's position is based on an evaluation which shows that it is unlikely that sufficient hydrogen can be generated and accumulated in the proper location to cause a hydrogen burn of sufficient magnitude to challenge containment, without some other mechanism causing containment failure, it was not appropriate to consider the air recirculation fan's survivability as a vulnerability from a severe accident perspective.

Comment No. 5

"The PRA should consider high energy line breaks outside of containment..."

Response

This concern is due to the commentor's belief that a high energy line break outside containment will result in a different accident progression than a steam line break inside of containment, the accident modelled in the IPE to bound all high energy line break accidents. The different accident progression would occur because the high energy line break could destroy equipment in the immediate vicinity of the break or equipment needed to mitigate the accident if it is not qualified for the environmental conditions caused by the high energy line break. This concern was addressed by the IPE team in three areas: 1) initiating event frequency, 2) accident progression, and 3) equipment survivability. These are discussed in more detail below.

The initiating event frequency for the IPE for large steam line/feed line break events includes consideration of failures both inside and outside containment. If high energy line breaks outside containment were to be modelled separately from high energy line breaks inside of containment, then the initiating event frequency for each event would be less than the value used when the events are considered together. Further, when considering only those break locations outside of containment that are in the vicinity of equipment modelled

in the steam line/feed line break analysis, the initiating event frequency would be significantly lower than that used in the IPE analysis.

With respect to the accident progression, a high energy line break inside containment models all the equipment which would be required to mitigate a high energy line break event outside containment. Because of train separation of most of the modelled equipment it was considered unlikely that a postulated break outside of containment would be in close proximity to more than one train of equipment. The impact on accident progression, therefore, was not considered to be significant.

Finally, the IPE team reviewed the list of environmentally qualified equipment and found that all major equipment modelled in the accident was qualified for the conditions expected to result from a high energy line break outside containment.

5.4 Living PRA Program

The Cook Nuclear Plant PRA is designed to be periodically updated over the remaining lifetime of the plant by utility risk assessment and engineering personnel. The living PRA program will address appropriate plant design and operating changes that occur.

The Cook Nuclear Plant PRA may be used to varying degrees for nuclear plant support in many different areas such as:

- Operator training in risk dominant sequences.
- Safety evaluations.
- Establishment of equipment surveillance test intervals.
- Prioritization of important equipment and systems.
- Establishment of allowable outage times (AOT) in Technical Specifications for safety related equipment.

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

For Cook Nuclear Plant, the results of the PRA indicate that there are no major severe accident vulnerabilities that require immediate corrective action. AEPSC is currently investigating possible modifications to procedures and components which were dominant contributors to core damage frequency. These modifications include:

- added emphasis on RCP seal temperature in the emergency operating procedures. Failure to trip the RCPs following the initiation of the accident was found to be a dominant contributor in the Loss of CCW and a Loss of ESW events.

- instruction to open the CVCS cross-tie valve to the opposite unit early in the accident response. Failure of RCP seal cooling was found to be a significant contributor to core damage frequency in the Loss of CCW and Loss of ESW events. The initiation of charging flow from the opposite unit should provide sufficient RCP seal cooling to prevent seal damage.

- modifications to the compressed air system (Unit 1 control air compressor) to increase the capacity of the system. Failure of the compressed air system was found to be a significant contributor to core damage frequency. Even though acceptable event tree modeling modifications would lower compressed air contributions and virtually eliminate this vulnerability, AEPSC is currently evaluating cost beneficial upgrades to the capacity of the Unit 1 control air compressor.

- operator training on the impact of primary and secondary system heat removal on containment pressure response and the possibility of containment failure preceding core melt. In addition, procedural upgrades will be considered to minimize the possibility of such situations arising.

- modifications to the EOPs to instruct the operators to maintain feedwater flow to the faulted steam generator during a steam generator tube rupture event when secondary side integrity can not be maintained. This has a significant affect on reducing offsite releases during a SGTR with a stuck open steam generator safety valve or PORV.

- operator training on the importance of a wet reactor cavity on potential fission product releases. This training will emphasize injecting the maximum amount of water possible from the RWST to the containment prior to switchover to recirculation.

The major factors affecting the safety of Cook Nuclear Plant were identified as :

- Two motor-driven and one turbine-driven auxiliary feedwater pumps, which for most accidents, provide three redundant headers.

- The ability to cross-tie auxiliary feedwater flow from the opposite unit, thus affording additional redundancy in that system.

- A very reliable offsite electric power grid that reduces the likelihood of loss of offsite power and station blackout events at Cook Nuclear Plant.

- The ability to cross-tie component cooling water flow from the opposite unit, thus affording additional redundancy in that system.

- The containment ultimate failure pressure over three times design pressure with a 95/95 confidence level. Containment ultimate failure pressure is almost five times design pressure as compared to the median failure value.

- Operation of containment sprays in the recirculation can be expected to prevent containment failure for practically all accidents.

- An essential service water (ESW) system, shared between both Units, that operates with the Unit-to-Unit cross-tie valves open. This allows one Unit to readily rely on ESW flow from the opposite Unit.

7.0 SUMMARY AND CONCLUSIONS

AEPSC has performed a Level III Probabilistic Risk Assessment of internally initiated events for the Donald C. Cook Nuclear Plant. This study was performed using a fault tree linking methodology and meets the requirements of 10CFR50 Appendix B. The Cook Nuclear Plant PRA documents the computer models and the results of the analysis. While the Individual Plant Evaluation Partnership (IPEP) was contracted for the Cook Nuclear Plant PRA, AEPSC personnel were involved to a great degree in every aspect of this analysis through either detailed review of contract work or actual performance of the analysis. The contract with IPEP includes a complete transfer of technology. This technology transfer allows AEPSC to biannually update the Cook Nuclear Plant PRA in-house without additional contract work.

Results of the PRA indicated that the core damage frequency for internal events is 6.26E-5 per year. The examination has identified that the initiating event with the largest contribution to core damage frequency is Small LOCA (SLO) with a 47% contribution. Failure of the Emergency Core Cooling System (ECCS) during either the cold leg injection or recirculation phases produced the two most dominant sequences for this event. Common mode failure of the safety (SI) pumps (part of the ECCS) and failure of the Engineered Safety Features (ESF) system to actuate the ECCS dominated these two sequences. The third leading sequence resulted from functional failures to cool the reactor coolant system (RCS) followed by failure of primary bleed and feed cooling. Hardware and common mode failures in the compressed air system, which supplies air to the pressurizer and steam generator PORVs, contributed mostly to these failures.

The Loss of Component Cooling Water (CCW) event, which contributed 22% to the core damage frequency, was dominated by three sequences. The first sequence was solely dominated by the failure of the operator to trip the running reactor coolant pumps (RCPs) after seal cooling from CCW is lost, thus leading to gross seal failure. The second sequence was dominated by failure of ECCS cold leg recirculation due to common mode failure of the SI pumps. The third sequence was involved the functional failure to restore reactor inventory after CCW was restored (which then restores the ECCS charging pumps). This failure was dominated by operator error and ESF signal failure.

The Steam Generator Tube Rupture (SGTR), comprising 11% of the core damage contribution, involved multiple sequences of significant contribution. The dominant failures associated with these sequences were hardware and common mode failures of the compressed air system and failures of ESF signals. Failure of the compressed air system prevented remote cooldown and depressurization of the RCS thereby allowing reactor coolant inventory to be lost to the secondary side of the steam generators until there is not sufficient reactor coolant inventory remaining to transfer heat from the core. The significance of the SGTR event is that containment may be directly bypassed and fission products released directly to the environment following core melt.

All other internal initiating events were very small contributors and displayed no significant vulnerabilities in addition to those previously discussed in this section. The redundancy afforded by the auxiliary feedwater system (two motor-driven pumps and one turbine-driven pump) is significantly beneficial for the transient events. The extremely reliable electric power grid, of which Cook Nuclear Plant is a part, greatly influenced the initiating event frequencies for the Loss of Offsite Power and Station Blackout events, thus directly influencing their small contributions to core damage frequency.

Failures due to common cause were dominant contributors in almost every initiator modeled. The reason that common cause so often appears, however, is due to the modeling approach. The system fault trees typically modeled common cause failures for the entire system in a single gate. This treated all common cause failures, whether for pumps, valves, or any other component, as a single failure. If the individual component common cause contributions had been modeled in individual gates, the overall common cause contribution to core damage frequency would have been greatly reduced. It was concluded, therefore, that common cause failures are not a dominant contributor to core melt frequency.

The first step in the level II analysis was to determine the possible failure modes for the Cook Nuclear Plant containment. This analysis showed that the overpressure failure for the containment would most likely occur

due to shear of the concrete basemat at the cylinder basemat junction. This failure mode causes water in the containment to be lost and a subsequent loss of long term recirculation cooling. In addition, this failure mode eliminates many post containment failure accident management strategies. The conditional probability of containment failure on overpressure given core melt is only 0.033. Even though this failure probability is small, the failure mechanism may preclude post containment failure recovery and accident management actions because the containment failure may cause failure of piping and electrical connections to the containment. The overall frequency of containment failure was found to be small at 9.1E-6 per year. Given that a core melt accident has occurred, there is approximately an 85% chance that containment integrity will be maintained. As indicated above, containment failure on overpressurization is expected 3.3% of the time following core melt. The dominant containment failure mode following core melt was found to be containment bypass following a steam generator tube rupture.

The probability of containment failure due to hydrogen combustion was also evaluated as part of the level II PRA. This analysis showed that, for any credible scenario, the chance of the containment failing due to hydrogen combustion was unlikely. It was determined that it was improbable that sufficient hydrogen could be generated during the time period when hydrogen igniters were unavailable so that a hydrogen burn could fail containment. For those sequences where sufficient hydrogen could to be generated so that a hydrogen burn could fail containment, it is much more likely that containment would have already failed due to overpressurization.

Supplement 3 to Generic Letter 88-20 (Reference 9) requested an evaluation of the vulnerability of ice condenser containments to the interruption of power to hydrogen igniters. It is concluded, as summarized in the above discussion, that the interruption of power to the hydrogen igniters does not constitute a vulnerability of the Cook Nuclear Plant Containment. As a result, providing a backup DC power supply to the hydrogen igniters would not provide any noticeable decrease in the frequency or offsite consequences of containment failure of the Cook Nuclear Plant.

Using containment atmospheric release source terms from the Level II analysis, offsite consequences were calculated using the MELCOR Accident Consequence Code System (MACCS), Reference 14. Table A-1 presents the results for the overall combined effects of the three evacuation schemes (evacuation, no evacuation and a combination of evacuation and sheltering) and the long term effects. As shown in Table A-1, the SGR50 sequence (steam generator tube rupture) source terms dominate early and cancer fatalities and whole body doses. SGR50 is a containment bypass sequence for which the containment also fails on overpressure. Virtually all of the noble gases and > 10% of the volatile fission products are released from containment. SGR50 containment basemat failure yielded the most cancer fatalities and SGR50 upper containment failure (equipment hatch failure - Level II sensitivity) yielded the most early fatalities. Figures A-1 through A-4 present early fatalities, cancer fatalities and whole body doses at 50 miles and 100 miles for SGR50. Figure A-3, population dose up to 50 miles (mean value curve), is comparable to population doses up to 50 miles for the Surry, Sequoyah and Zion plants in NUREG-1150 (Reference 66), which also used the MACCS code. Figures A-1, A-2 and A-4 could be compared to NUREG-1150, however, NUREG-1150 evaluated site regions up to 1000 miles whereas this project evaluated up to 100 miles. To address the expected consequences of SGR50, a procedure change is under investigation to maintain steam generator level in the event of this accident. This modification is expected to greatly reduce the offsite dose consequences of this sequence.

The Cook Nuclear Plant PRA was developed to meet the requirements of 10CFR50 Appendix B. This approach forced detailed review beyond that addressed in Generic Letter 88-20 (Reference 9). AEPSC is confident that this analysis adequately meets the intent of Generic Letter 88-20, including Supplements 1 and 3. No major plant vulnerabilities have been identified which require immediate action or significant hardware changes. Changes to both procedures and hardware, however, are being considered to address minor vulnerabilities. These are identified in Section 6.0.

It is the intention of AEPSC to use the Cook Nuclear Plant PRA as a decision-making tool in many aspects of engineering support and plant operations. Since the PRA is a highly technical document and uncertainties

do exist in the analysis, the use and interpretation of PRA results and conclusions is currently limited to those individuals who have been intimately involved with its development. This approach avoids the problems that might arise from misinterpretation of the study.

Also, AEPSC's internal commitment to regularly update the PRA ensures that the document will be a "living" document that adequately reflects the current plant configuration and operating requirements. This "living" PRA concept coupled with the review requirements of 10CFR50 Appendix B makes the Cook Nuclear Plant PRA a valuable decision-making tool for most safety related questions.

8.0 LIST OF REFERENCES

1) Donald C. Cook Nuclear Plant Updated Final Safety Analysis Report, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, July 1989

2) Donald C. Cook Nuclear Plant Unit 1 Technical Specifications, American Electric Power Service Corporation

3) Donald C. Cook Nuclear Plant Unit 2 Technical Specifications, American Electric Power Service Corporation

4) Emergency Operating Procedures, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, in conjunction with "Emergency Response Guidelines," Background, High Pressure Version, Westinghouse Owners Group, Revision 1, September 1983

5) EPRI NP-3583, "Systematic Human Reliability Procedure (SHARP)," June 1984.

6) EPRI NP-3878, "Large Scale Hydrogen Combustion Experiments," October 1988.

7) EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," June 1985.

8) "Fault Tree Analysis Guidelines for Donald C. Cook Nuclear Plant," Westinghouse Electric Corporation, Revision 0, December 1989

9) Generic Letter No. 88-20 "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.44(f)" and its supplements. Supplement No. 1 "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f); Supplement No. 2 "Accident Management Strategies for Consideration in the Individual Plant Examination Process"; and Supplement No. 3 "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities".

10) WCAP 11693, Revision 3, "GRAFTER Code Sytem User Manual for Version 1.6", Westinghouse Electric Corporation, 1990.

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

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

13) Fauske & Associates, "Modular Accident Analysis Program", 1990; also "MAAP 3.0B Complete Code Manual," EPRI NP-777-1 CCML.

14) NUREG/CR-4691, "MELCOR Accident Consequences Code System (MACCS)", Sandia National Laboratories, 1990.

15) Notebook for Quantification of Human Error Probabilities, Revision 0, 1991

16) NSAC-108, The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants.

17) NSAC-144, "Loss of Offsite Power at U.S. Nuclear Power Plants," April 1989.

18) NUREG-0909, "January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," April 1982.

19) NUREG-1032, Evaluation of Station Blackout Accidents at Nuclear Power Plants, June 1988.

20) NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989.

21) NUREG/CR-1174, "Evaluation of System Interactions in Nuclear Power Plants," August 1989.

22) NUREG/CR-1278, "Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983.

23) NUREG/CR-2678, "Flood Risk Analysis Methodology Development Project Final Report," June 1982.

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

25) NUREG/CR-2815, "Probabilistic Safety Analysis Procedure Guide," Rev. 1, August 1985.

26) NUREG/CR-2300, "PRA Procedures Guide," January 1983.

27) NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," EG&G Idaho, Inc., May 1985.

28) NUREG/CR-4142, "A Review of the Millstone 3 Probabilistic Safety Study," April 1986.

29) NUREG/CR-4550, "Analysis of Core Damgage Frequency from Internal Events: Methodology Guidelines

30) NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Volume 1, February 1988 and Volume 2, January 1989.

31) NUREG/CR-5536, "Mitigation of Direct Containment Heating and Hydrogen Combustion Events in Ice Condenser Plants," October 1990.

32) Severe Accident Policy Statement (50FR43621)

33) Success Criteria Verification Notebook, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, Revision 0

34) Technical Report 23.5, "Evaluations of Containment Bypass and Failure to Isolate Sequences for the IDCOR Reference Plants," IDCOR Program Report, August 1986.

35) "Treatment of Common Cause in Fault Tree Models," RMOI Guidebook 2, Westinghouse Electric Corporation, Revision 0, August, 1990

36) WASH-1400, "Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants," October 1975.

37) WCAP-9600, "Report on Small Break Analysis for Westinghouse NSSS Systems," Volume 1, June 1979.

38) WCAP-9914, "PORV Sensitivity Study for LOFW-LOCA Analyses," July 1981.

39) WCAP-10541, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Westinghouse Electric Corporation, Revision 2, November 1986.

40) WCAP-10858-PA, "AMSAC Generic Design Package," Revision 1, July 1987.

41) WCAP-11902, "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," October 1988.

42) WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs," Westinghouse Electric Corporation, December, 1988.

43) WCAP-12078, "Input and Output Parameters for the Accident Analyses Performed for Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1," December 1988.

44) WCAP-12135, "Donald C. Cook Nuclear Plant Units 1 and 2, Rerating Engineering Report," September 1989.

45) WCAP-12922, WCGS PRA Event Tree Analysis Notebook, Westinghouse Electric Corporation, May, 1991.

46) WCAP-12075, Revision 2, "WLINK Code System User Manual for Version 2.3.1," Westinghouse Electric Corporation, 1990.

47) Event Tree Notebook, Donald C. Cook Nuclear Plant, American Electric Power Service Corporation, Revision 0, 1991.

48) "Fault Tree Analysis Guidelines for Donald C. Cook," Westinghouse Electric Corporation, Revision 0, December 1989.

49) T. Speis, USNRC, letter to A. Buhl, IT Corporation, "Transmittal of Final NRC Technical Issue Positions for Issues 1, 2, 3, 5, 6, 11, 13A, and 17," September 22, 1986.

50) T. Speis, USNRC, letter to A. Buhl, IT Corporation, "Transmittal of Final NRC Technical Issue Positions for Issues 4, 9, 10, 12, 13B, 15 and 16" November 26, 1986.

51) T. Speis, USNRC, Letter to A. Buhl, IT Corporation, "Position Papers for the NRC/IDCOR Technical Issues" March 11, 1987.

52) Gabor, Kenton and Associates, "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI TR-100167 (191).

53) Fauske & Associates, "A Phenomenological Evaluation Summary on Containment Overpressurization in Support of the D. C. Cook Nuclear Plant Individual Plant Evaluation," FAI/91-150, 1992.

54) Fauske & Associates, "Donald C. Cook Nuclear Plant Phenomenological Evaluation Summary on Direct Containment Heating in Support of Individual Plant Examination," FAI/91-87, 1992.

55) Fauske & Associates, "Donald C. Cook Nuclear Plant Phenomenological Evaluation Summary on Steam Explosions in Support of Individual Plant Examination," FAI/91-25, 1992.

56) Fauske & Associates, "Phenomenological Evaluation Summary on Molten Core-Concrete Interaction in Support of the Donald C. Cook Nuclear Plant Individual Plant Examination," FAI/91-29, 1992.

57) Fauske & Associates, "A Phenomenological Evaluation Summary on Thermal Loading of Containment Penetrations in Support of the Cook Nuclear Plant Individual Plant Evaluation," FAI/91-64, 1992.

58) Fauske & Associates, "Phenomenological Evaluation Summary on Thrust Forces at Reactor Vessel Failure in Support of the Cook Nuclear Plant Individual Plant Examination," FAI/91-85, 1992.

59) Fauske & Associates, "A Position Paper on the Probability and Consequences of the Deflagration and Detonation of Hydrogen in the Cook Nuclear Plant Individual Plant Examination," FAI/91-175, 1992.

60) Ravindra, M. K., 1981, "Containment Internal Pressure Capacity D. C. Cook Nuclear Generating Station Phase II - Probabilistic Evaluation," Report No. 80C129-2.

61) Fauske & Associates, 1991, MAAP Parameter File for D. C. Cook Unit 1, prepared for the American Electric Power Service Company.

62) "Radiation Analysis Manual, D. C. Cook Units 1 and 2", Westinghouse Electric Corporation, attachment to letter AEP-88-331, July 26, 1988; H. C. Walls (Westinghouse) to R. B. Bennett (AEPSC).

63) Fauske & Associates and Stone & Webster, "Evaluation of the Consequences of Containment Bypass Scenarios," EPRI NP-6568-L, Vol. 1, 1989.

64) NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," 1988.

65) "Technical Report 18.1, Evaluate Atmospheric and Liquid Pathway Dose," IDCOR Program Report, 1984.

66) NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", Volume 1 December 1990

67) "Internal Flooding Analysis Methodology," RMOI Guidebook 7, Westinghouse Electric Corporation, Revision 0, March 1990

68) Calculation CN PORI-90-230-R0, "Quantification of Frequencies of LOCAs and Steamline/ Feedline Break Initiating Event Frequencies," Westinghouse Electric Corporation, May 13, 1991.

69) Letter J.C. Hoebel, Westinghouse Electric Corpororation, to J.B. Kingseed, AEPSC, March 19,1990, AEP-90-117/NS-RMOI-PRA-90-077.

APPENDIX A

OFFSITE CONSEQUENCES ANALYSIS

(LEVEL III PRA)

[illegible]

[illegible]

[illegible]

APPENDIX A

Using containment atmospheric release source terms from the Level II analysis, offsite consequences were calculated using the MELCOR Accident Consequence Code System (MACCS), Reference 14. Table A-1 presents the results for the overall combined effects of the three evacuation schemes (evacuation, no evacuation and a combination of evacuation and sheltering) and the long term effects. As shown in Table A-1, the SGR50 sequence (steam generator tube rupture) source terms dominate early and cancer fatalities and whole body doses. SGR50 is a containment bypass sequence for which the containment also fails on overpressure. Virtually all of the noble gases and > 10% of the volatile fission products are released from containment. SGR50 containment basemat failure yielded the most cancer fatalities and SGR50 upper containment failure (equipment hatch failure - Level II sensitivity) yielded the most early fatalities. Plant conditions represented by SGR50 are: reactor coolant system at high pressure, refueling water storage tank not emptied (containment spray and high/low head charging draw from it), hydrogen igniters and containment air recirculation fans failed. Figures A-1 through A-4 present early fatalities, cancer fatalities and whole body doses at 50 miles and 100 miles for SGR50. The results in these figures account for core damage and containment failure probabilities (from Table 4.7-3) associated with SGR50. The results in Table A-1 do not account for core and containment failures, but only present results of source term releases. Figure A-3, population dose up to 50 miles (mean value curve), is comparable to population doses up to 50 miles for the Surry, Sequoyah and Zion plants in NUREG-1150 (Reference 66), which also used the MACCS code. Figures A-1, A-2 and A-4 could be compared to NUREG-1150, however, NUREG-1150 evaluated site regions up to 1000 miles whereas this project evaluated up to 100 miles.

The containment release source terms selected as input into the Level III analysis are representative of the containment release categories found in Table 4.7-3. Of the containment bypass categories ("C" and "T"), it was decided that category "T" would be analyzed for offsite consequences. Source term SGR50 represents category "T" and released virtually all of the noble gases and approximately 47% of the volatile nuclides. This volatile release is significantly greater than the < 1% volatile release in category "C". Also, even though category "C" has a higher failure probability, the failure value is in the same order-of-magnitude range as category "T".

For the containment basemat failure source terms, all of the noble gases, most of the volatile and a noticeable amount of the non-volatile fission products escaped the containment structure. However, as shown in the Cook Nuclear Plant UFSAR (Reference 1), the containment basemat (annulus) is 21 feet below grade (grade is 608 ft. and the annulus is at 587 ft.). Within IDCOR Technical Report 18.1 (Reference 65), which dealt with atmospheric and liquid pathway doses, it was concluded that for basemat failures due to reactor fuel melt-through, only small population doses would be expected. Interdictive measures would be needed in the worst cases. These conclusions were reached using the MAAP code (which Level II used) and the CRAC2 code. Therefore, liquid pathway doses are not considered in this project.

TABLE A-1
Level III Offsite Consequence Results

ACCIDENT SEQUENCE	FATALITIES (MEAN VALUES) EARLY 0 - 100 MILES	CANCER 0 - 100 MILES	WHOLE BODY DOSES (REM - MEAN VALUES) 0 - 10 MILES	0 - 50 MILES	0 - 100 MILES
SBO181 $	0.00E+00	1.41E-01	4.04E+02	5.03E+02	8.18E+02
MLO40 *	0.00E+00	2.68E+03	2.74E+05	2.43E+06	1.56E+07
LLO8 **	0.00E+00	2.05E+03	3.05E+05	1.92E+06	1.21E+07
SGR50 **	1.77E+00	4.59E+03	8.19E+05	4.84E+06	2.65E+07
MLO40 **	0.00E+00	1.13E+03	1.87E+05	1.20E+06	6.55E+06
SLO35	0.00E+00	3.32E+02	1.35E+05	4.29E+05	1.92E+06
SGR50 *	1.13E+02	3.58E+03	2.89E+06	5.49E+06	1.94E+07
ISL5	1.82E-03	2.09E+03	3.72E+05	2.12E+06	1.21E+07

$ No containment failure.
* MLO40 and SGR50 Level II sensitivity studies with failure in upper containment.
** Containment basemat failure source terms.

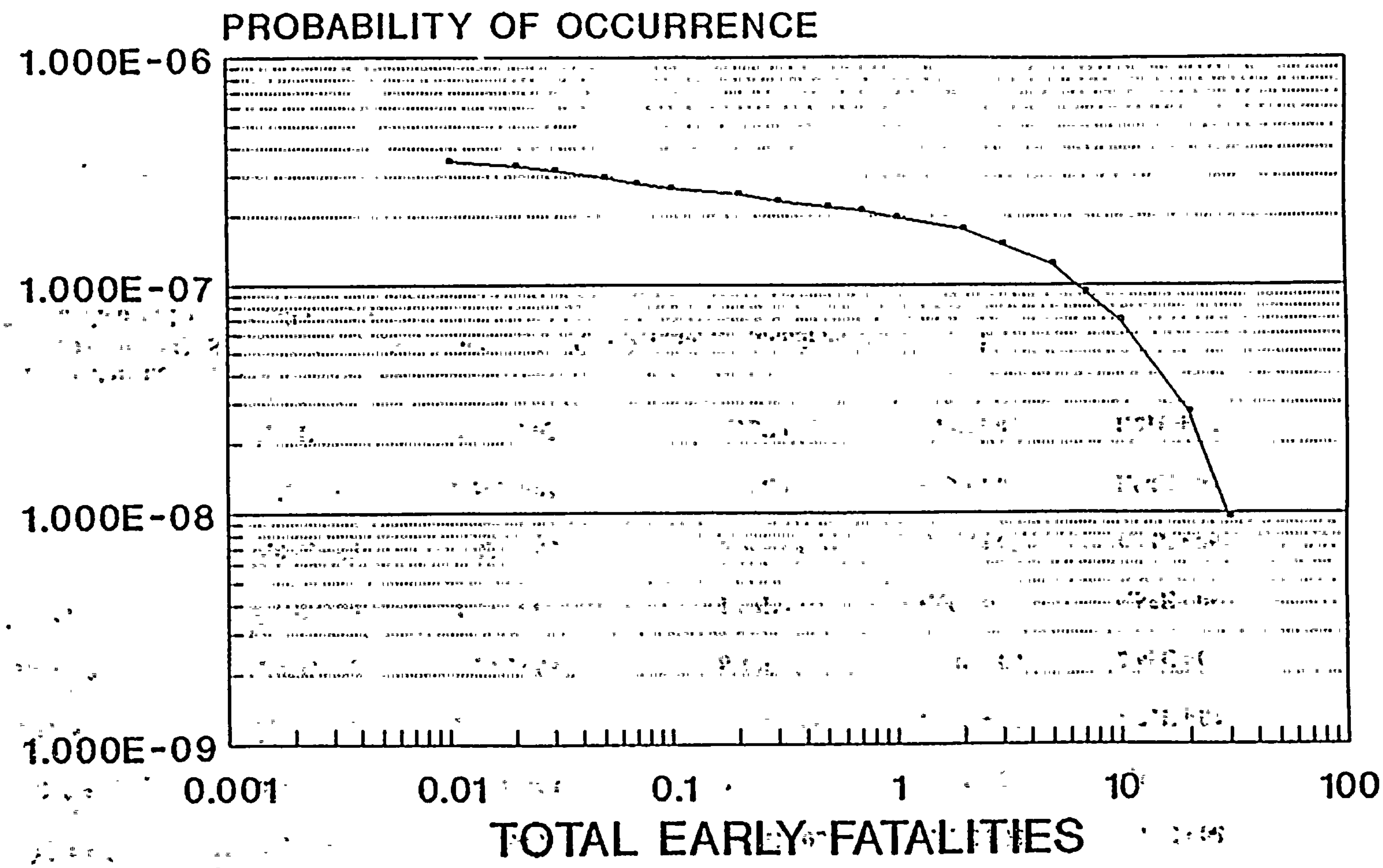
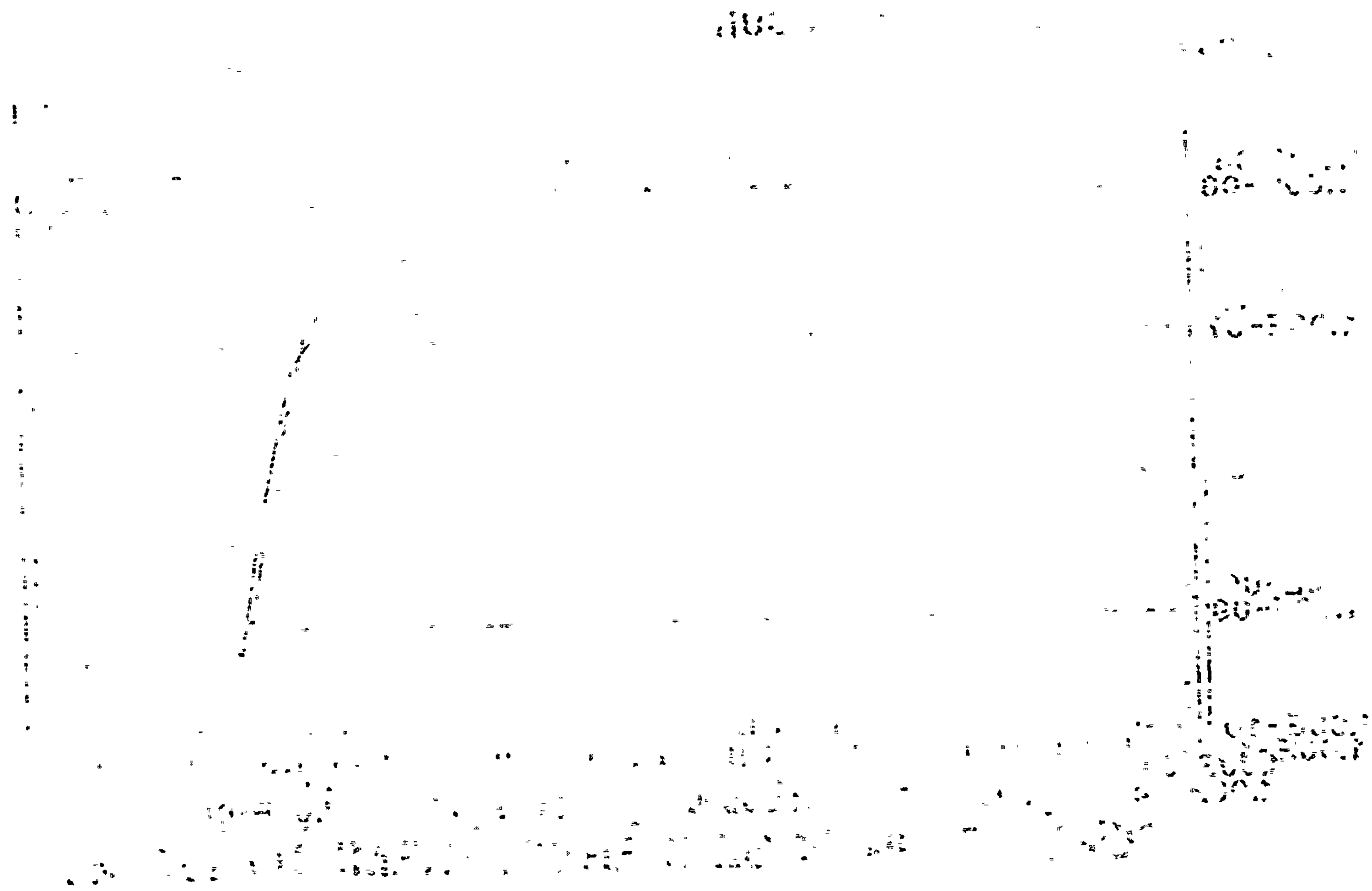


Figure A-1
SGR50 Early Fatalities 0-100 Miles



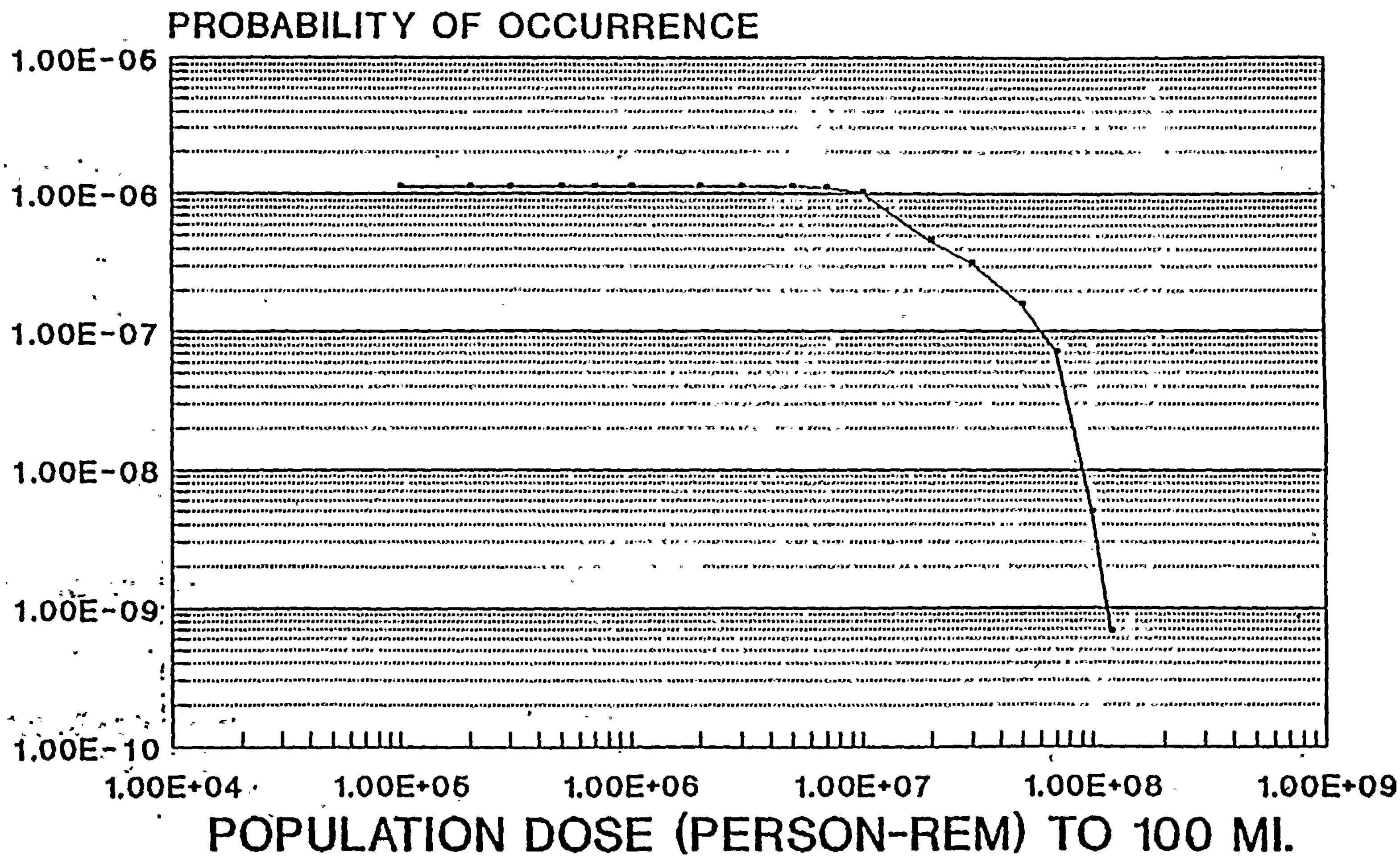
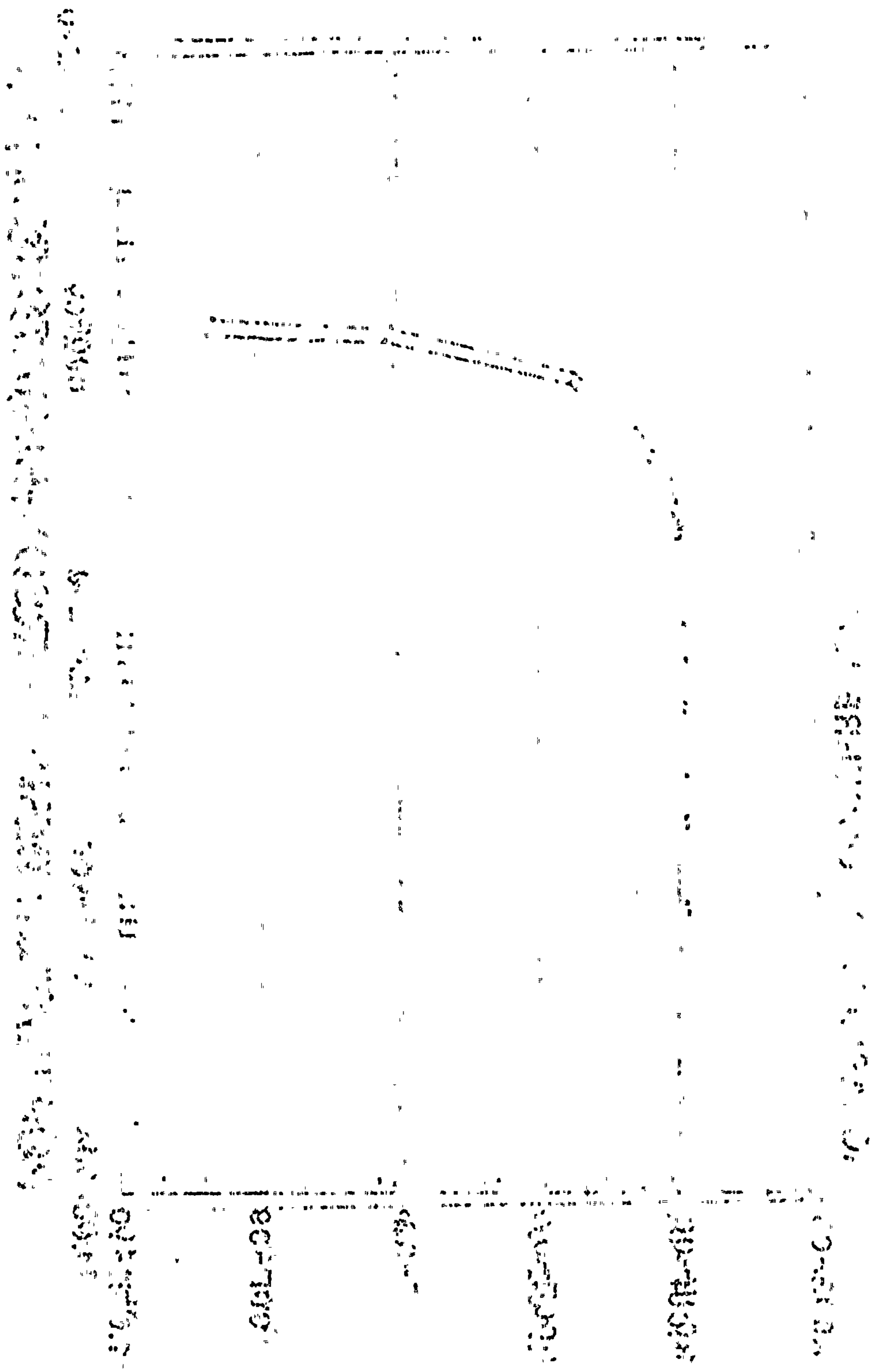


Figure A-4
SGR50 Population Dose (Person-REM) to 100 Miles



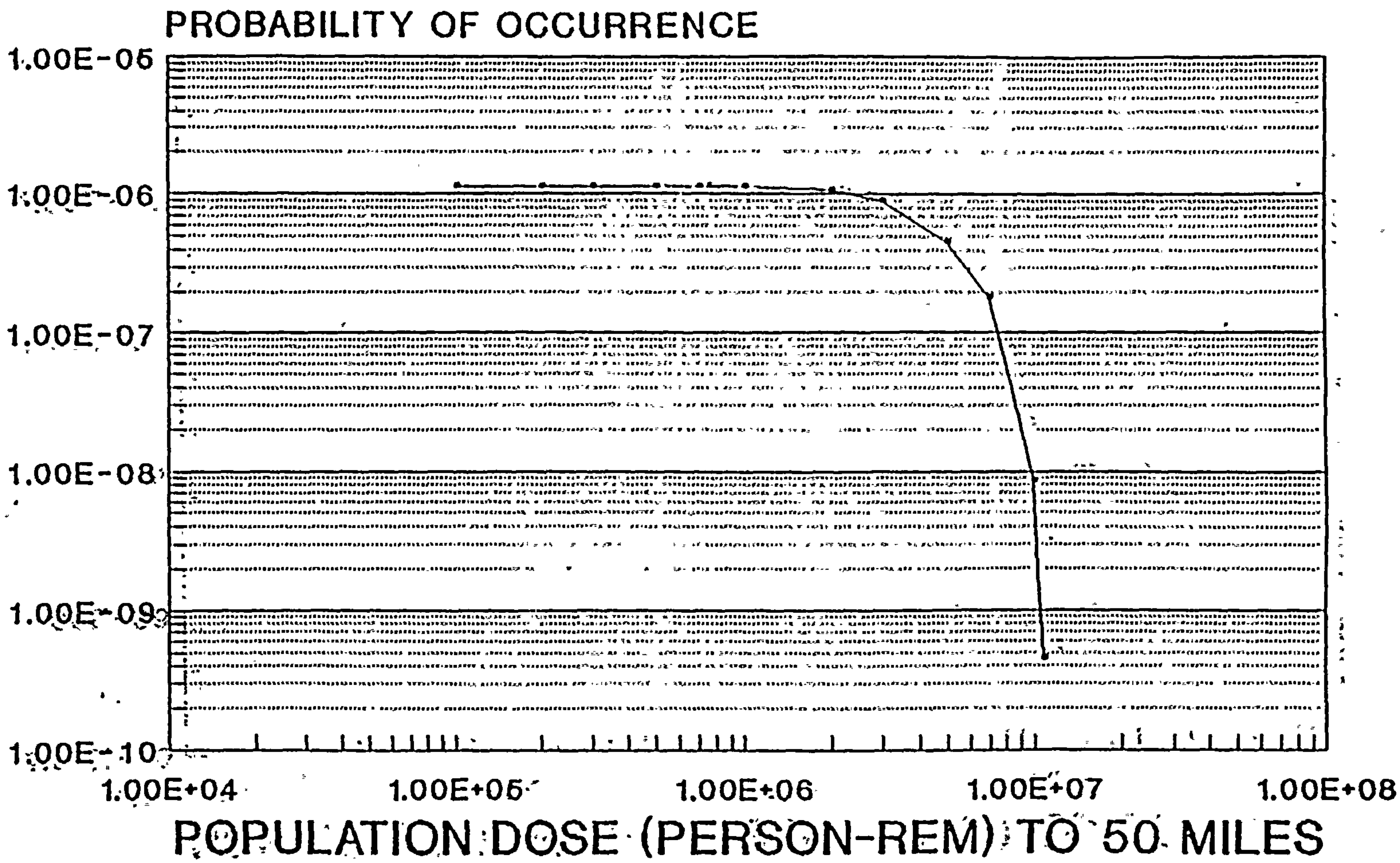
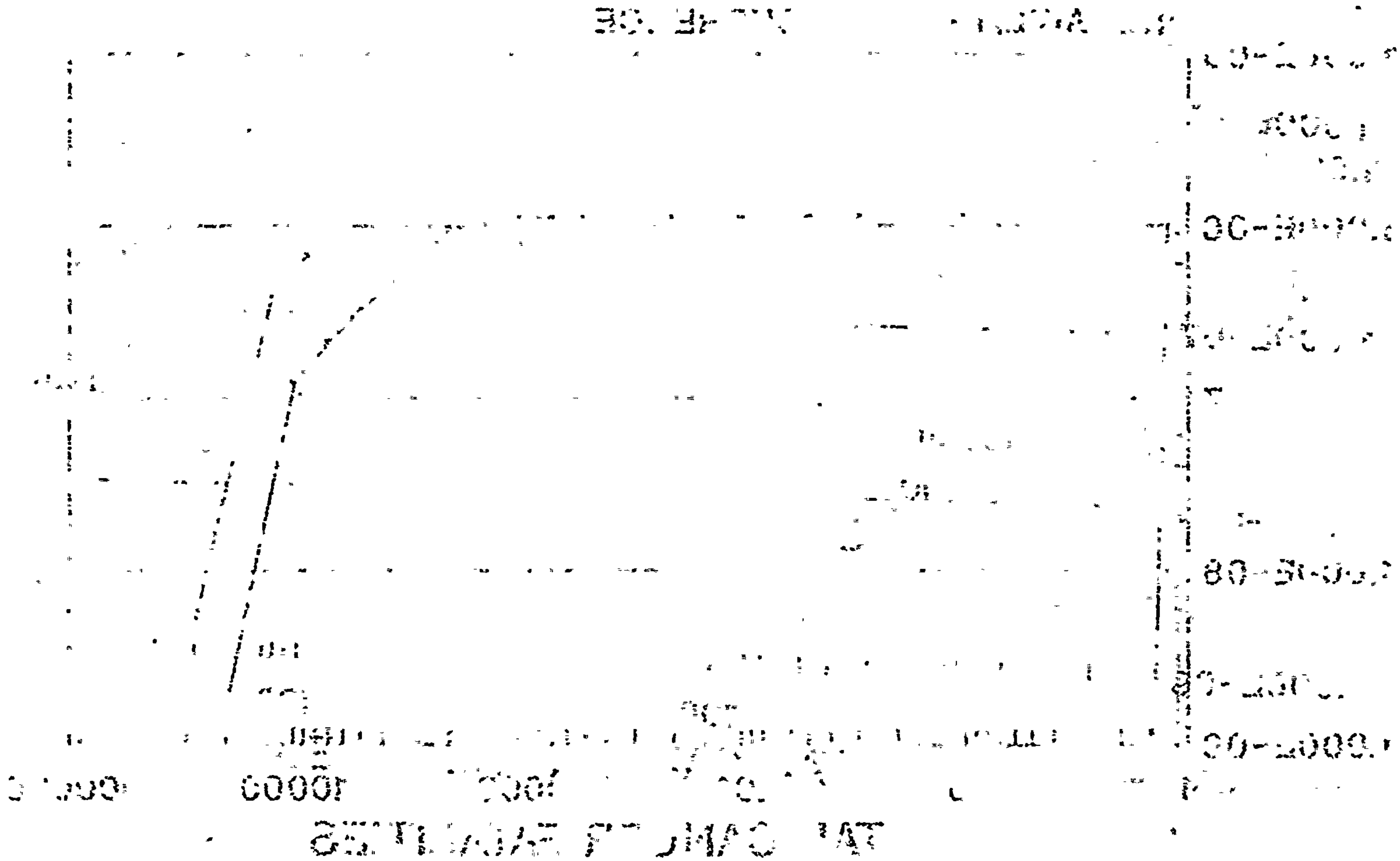


Figure A-3
SGR50 Population Dose (Person-REM) to 50 Miles



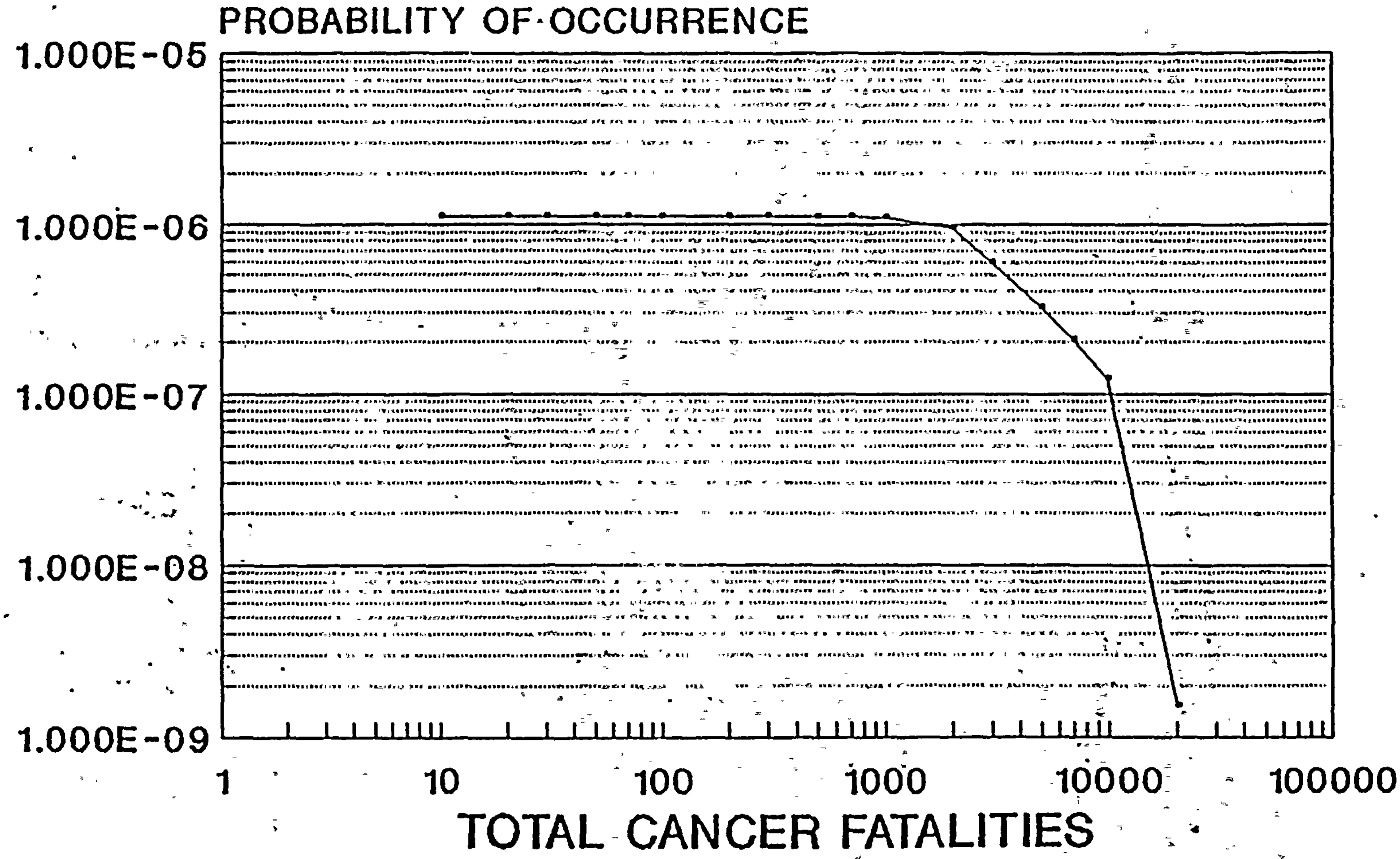


Figure A-2
SGR50 Cancer Fatalities 0-100 Miles

- Identification [illegible] consideration of clues that may help the operator [illegible] (including [illegible] lights and alarms).

- Identification [illegible] consideration of conditions that may impede the operator in [illegible] task.

- Identification and consideration of dependencies between tasks. (Table 3.3-[illegible] lists [illegible] used to calculate probabilities conditional on success or failure of previous task.)

d. Quantification [illegible] Defined in PRA

This task involved the cumulative quantification of the HEP for each human action, [illegible] Bank, the models generated for the operator actions, the performance shaping factors [illegible] the THERP method.

3.3.3.2 General Assumptions

It was generally assumed that there were [illegible] within a particular action. However, if a strong dep[illegible] existed, such [illegible] were [illegible] individually. A deta[illegible] appropriately addressed.

For many operator actions, step-by-[illegible] to guide the operator [illegible] Credit for the availability of these [illegible] appropriate [illegible] HEP Data Bank. For other operator [illegible] procedures are available [illegible] direction is provided to the operator [illegible] operator action depends [illegible] the training and memory of the [illegible] may be reasonably [illegible] with the operators were conducted [illegible] operators had [illegible] knowledge and [illegible] with these situations. Estimates of HEP [illegible] with operators [illegible] availability of written procedures.

Significant improvements have occurred within [illegible] training to emergency response procedures. The procedures in place today for most commercial nuclear power plants (including the [illegible] Nuclear Plant) are more symptomatic in nature. Unless specific diagnostic difficulties were noted, it was assumed that the EOPs currently in place for the Cook Nuclear Plant are adequate to [illegible] transient symptoms and ensure that the operator provides the correct functional response.

Due to [illegible]